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ANTICIPATING POTENTIAL WASTE ACCEPTANCE CRITERIA FOR DEFENSE SPENT NUCLEAR FUEL*

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ABSTRACT

The Office of Environmental Management of the U.S. Department of Energy (designated here as DOE/EM) is responsible for the safe management and disposal of DOE-owned defense spent nuclear fuel and high level waste (DSNF/DHLW). A desirable option, direct disposal of the waste in the potential repository at Yucca Mountain, depends on the final waste acceptance criteria, which will be set by DOE's Office of Civilian Radioactive Waste Management (OCRWM). However, evolving regulations make it difficult to determine what the final acceptance criteria will be. A method of anticipating waste acceptance criteria is to gain an understanding of the DOE-owned waste types and their behavior in a disposal system through a performance assessment and contrast such behavior with characteristics of commercial spent fuel. Preliminary results from such an analysis indicate that releases of 99Tc and 237Np from commercial spent fuel exceed those of the DSNF/DHLW; thus, if commercial spent fuel can meet the waste acceptance criteria, then DSNF can also meet the criteria. In large part, these results are caused by the small percentage of total activity of the DSNF in the repository (1.5%) and regulatory mass (4%), and also because commercial fuel cladding was assumed to provide no protection.

GENERAL PURPOSE OF THE ANALYSIS

The Office of Environmental Management of the U.S. Department of Energy (designated here as DOE/EM) is responsible for the safe management and disposal of DOE-owned defense spent nuclear fuel and high level waste (DSNF/DHLW). A desirable option is direct disposal of the waste in the potential repository at Yucca Mountain, which is being designed and characterized by DOE's Office of Civilian Radioactive Waste Management (OCRWM). The potential repository is designed primarily for disposal of commercial spent fuel, although a portion is reserved for DSNF/DHLW. Thus the final waste acceptance criteria for the potential repository, which will be set by the OCRWM, are expected to accommodate characteristics of commercial spent fuel.

The viability of direct disposal of DSNF/DHLW, then, depends on the final waste acceptance criteria. However, evolving regulations make it difficult to determine what these criteria will be. Currently, the technical criteria as promulgated by the U.S. Nuclear Regulatory Commission (NRC) in 10 CFR 60 (which is under revision per the

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Energy Policy Act of 1992 (Pub. L. 102-486)) establish stringent minimum requirements for disposal subsystems. Because these criteria are already spelled out, their requirements would appear to be the logical starting point for establishing waste acceptance criteria. However, recent proposed amendments to the Nuclear Waste Policy Act (NWPAA, 1997, §206.f.5.A), along with guidance from the NAS (NAS/NRC, 1992; 1995), emphasize an evaluation of the whole disposal system, rather than regulations per system component.

At the same time, DOE/EM contractors that manage DSNF storage sites and are responsible for preparing the DSNF for disposal would prefer definite procedures for characterizing and treating the DOE-owned waste—procedures that could be followed now prior to completion of the lengthy licensing procedures and even before knowledge of the final waste acceptance criteria. Although such arbitrary limits are appealing, because they would permit a straightforward course of action, they have the potential to be extremely costly by overcompensating for the unknown, and would themselves be subject to analysis in the later stages of regulatory licensing to ensure that they provide safe conditions.

An alternative to arbitrary limits is to gain an understanding of the DOE-owned waste types and their behavior in a disposal system through a performance assessment and contrast such behavior with characteristics of commercial spent nuclear fuel.* Given that the final waste acceptance criteria will accommodate the behavior of commercial spent fuel, a preliminary study that indicates whether the DSNF/DHLW performs better (or worse) than commercial fuel is a strong indicator of the level of waste characterization and treatment that will be necessary before disposal.

The purpose of the study reported here, referred to as the 1997 DSNF/DHLW performance assessment (or 1997 PA), is to assist the DOE/EM by providing information about its waste so that it is well acquainted with the issues and options available as site-specific regulations for the potential Yucca Mountain repository emerge. In the 1997 PA, the behavior of the DOE-owned waste was compared to commercial spent fuel after disposal in the potential Yucca Mountain repository. The radioisotope inventory—72,960 Mg of heavy metal—included a large proportion of commercial spent nuclear fuel (86.6%), with only 9.5% of the mass of heavy metal from DHLW, and 3.9% from DSNF. Also, the DSNF represents only 1.5% of the activity (curies), because the fuel was usually "burned" for only a short time. Thus, the DSNF's small activity (1.5%) and mass of heavy metal (3.9%) is very likely within the error band of the commercial spent fuel's characteristics, and so the possibility of the DSNF influencing the overall performance of a geologic system is remote. However, some types of DSNF have characteristics that might seem to differ substantially from commercial fuel. Of particular interest are characteristics related to 10 CFR 60 criteria, specifically, the prohibition of combustibility, as in the graphite blocks of Fort St. Vrain fuel, and pyrophoric material, such as metallic uranium spent fuel. In the 1997 PA, these characteristics were analyzed to confirm the supposition that DSNF would not influence disposal system performance. Detailed knowledge of the post-disposal behavior of its waste can place the DOE/EM in a strong position for negotiating acceptance criteria that are prudent, cost-effective, and time-saving.

An example can help to clarify this purpose. The pyrophoric nature of N-Reactor spent fuel, because of its metallic uranium, might be expected to adversely affect conditions in the repository. However, O_2 must be present for the pyrophoric fuel to pose a danger. In its study, Sandia has proposed that as corrosion progresses on the massive amounts of steel in the waste containers, the amount of O_2 in the repository will deplete; thus, the repository environment may greatly diminish the need for concern about the metallic uranium. Consequently, this type of fuel might be accepted without restrictions, which would also eliminate the need for the DOE/EM to reprocess or otherwise condition the fuel, at additional expense. The OCRWM may also benefit from this detailed analysis (O_2 transport is not currently considered in overall repository performance) because the corrosion results are relevant to the life of waste containers and so could aid the OCRWM in decisions about further modeling refinements for licensing of the repository.

* It would also be an advantage to use performance assessment for defining waste acceptance criteria for commercial spent fuel, given the current regulatory environment which encourages a total system analysis rather than compliance by subsystem. However, exploration of this topic is outside the scope of this paper.

Analysis Goals

The first goal of this analysis was to demonstrate the minor influence of the DSNF in a predominately commercial repository using modeling conditions similar to those currently assumed by the Yucca Mountain Project's (YMP) Managing and Operating (M&O) Contractor, TRW, when they conduct their Total System Performance Assessment (TSPA) for the OCRWM.

A second goal, closely related to the first, was to identify the most sensitive parameters through a preliminary analysis of the results to determine which DSNF characteristics should be carefully estimated or measured and which could be neglected, after demonstrating their minor influence.

A third goal was to continue development and implementation of modeling features to examine the availability of O_2 in the repository. This study is an important step in examining the unique characteristics of DSNF. The specific characteristics of concern are (1) the potential for pyrophoric behavior of the uranium metallic fuels in up to 89% of the heavy metal mass (e.g., Category 1, N-Reactor), and (2) the potential to go critical in the near or far field in up to 19% of heavy metal mass. Studying these specific characteristics requires developing detailed process models for the degradation of the containers and fuel matrix within the repository. An important phenomenon with regard to degradation is the availability of oxygen. Thus evaluating oxygen transport in the repository became the third goal.

Analysis Approach

In general, this analysis used the methodology of two related studies (1993 and 1994 PAs [Rechard, ed., 1993, 1995]). The methodology was originally established by Sandia National Laboratories for assessing the long-term performance of the Waste Isolation Pilot Plant (WIPP) geologic disposal system (DOE, 1996; Rechard, 1995). Modeling conditions for the analysis were coordinated with those to be used in TRW's model for the potential Yucca Mountain repository. As a means of comparison and benchmarking, Sandia's method of analysis provided the complex model to TRW's more simplified model (M&O, 1997), i.e.,

- The TRW performance assessment team used simple analytic expressions developed from abstractions of complicated analyses performed earlier in the program.
- Sandia simulated the complex details of contaminant transport in heated, unsaturated tuff directly in its analysis to complement the development of abstractions used by the TRW performance assessment team.
- Sandia improved the unsaturated zone modeling by including transport of O_2 in order to evaluate whether reactants such as O_2 and H_2O are limited significantly, thus reducing the rate of oxidation of container or waste.
- Sandia adjusted its modeling assumptions to match current TSPAs, such as
  - Artificially increasing the corrosion rate of carbon steel so that containers fail quickly.
  - Neglecting credit for cladding of spent nuclear fuel.
  - Using the same radioisotope inventory.

Because the already influential source term was made even more important by the use of dose performance criteria as recommended by the NAS (NAS/NRC, 1995), Sandia

- used a simple process source term model but tightly coupled it with the two-phase flow model to more adequately model radioisotope releases.
- chose a more accurate representation of the behavior of DSNF to more definitively identify DSNF fuel characteristics that have the most influence on performance.
- included in the source-term model both oxic and anoxic corrosion rates and solubilities (however this feature was not important in this preliminary work because carbon steel corrosion rates were so high).

RESULTS

The main performance criteria used in the 1997 PA are based on 40 CFR 191 (EPA, 1993), guidance from the NAS in 1995 regarding a proposed future standard for repository licensing (NAS/NRC, 1995), and a proposed 1997 amendment to the Nuclear Waste Policy Act of 1982 (NWPAA, 1997). The criteria examined are the probabilistic maximum doses received by an individual; the time period for analysis is generally 50,000 yr (the deterministic simulations with sampled parameters set at their mean or median value were run out to 100,000 yr).

Although the absolute position of the results with relation to the total system criteria of 40 CFR 191 (or replacement regulations) is crucial when a site applies for a license, the more important results here are those that can be used for project guidance with regard to the potential repository at Yucca Mountain. These are (1) the relative position of the various wastes in relation to the commercial spent nuclear fuel and (2) the relative importance of model parameters in explaining the value of a particular performance measure.

Final results from the 1997 PA should be available in spring of 1998. *Preliminary* findings from this performance assessment are presented here and are grouped into four areas: specific findings about the waste form, the waste containers and repository, and the geologic barrier, and then general findings about the system.

Findings for Waste Form

Background on Waste

The waste form is the type of waste (e.g., uranium metal) in combination with a specific treatment option. In the 1997 PA, only a minimal treatment option was considered, which means that the waste form and waste category are essentially the same. The waste under study includes ~2831 Mg heavy metal of DSNF, codisposed with ~6960 Mg heavy metal of DHLW, for a total of 9791 Mg heavy metal. As noted earlier, the radioisotope inventory used in this study—72,960 Mg of heavy metal—includes a large proportion of commercial spent nuclear fuel (86.6%), with only 9.5% of the mass of heavy metal from DHLW, and 3.9% from DSNF. (Note that the DSNF does not include the Navy propulsion reactor spent nuclear fuel). Within the DSNF, 2.9% (75.3% of the 3.9%) of the heavy metal mass is N-Reactor spent nuclear fuel, which was grouped as Category 1 for the analysis. The remaining 1.0% (24.7% of the 3.9%) of DSNF represents over 200 types of experimental fuel, and was grouped into 12 categories, for a total of 13 DSNF categories (see Table I). In the 1997 PA, the 12 DSNF categories representing 1.0% of the heavy metal mass in the repository were emplaced in large disposal containers together with between three and five handling containers of DHLW in borosilicate glass (the codisposal option).

[place Table I here]

Two categories of commercial spent fuel were modeled: 21-PWR (Pressurized Water Reactor) assemblies and 44-BWR (Boiling Water Reactor) assemblies (Categories 14 and 15) (Table I). Their combined mass of heavy metal was 63,170 MTHM. The commercial spent nuclear fuel will potentially be sent from 110 different reactors; however, the radioisotope inventory supplied for this analysis was an average recently estimated by the OCRWM for its performance assessment, TSPA-VA, to be completed in 1998.

Summary of Information Gathered about Waste Form

When using the source term code, CST*, the preliminary comparison of commercial spent fuel with DSNF/DHLW showed that releases of ^{99}Tc and ^{237}Np** from commercial spent nuclear fuel exceed those of DSNF/DHLW both per MTHM and per package (Fig. 1). Thus, if commercial spent fuel can meet regulatory performance criteria, then the DSNF/DHLW can also meet the criteria. In large part, this result is caused by the lower burnup values for DSNF and by its small percentage of total activity in the repository (1.5%) and regulatory mass (4%). These results also assume that cladding of the commercial spent fuel provides no protection (i.e., has failed) and so, in contrast to an earlier study (1994 PA), the type and integrity of cladding were not important. However, temperature-sensitive alteration rates of the fuel matrix appeared to be especially important because the corrosion rates of the containers were assumed to be so rapid that failures occurred when the repository was still hot. Releases of ^{99}Tc were limited only by the amount of waste that was exposed. Releases of ^{237}Np were limited by the amount of waste exposed and by its solubility since the ^{237}Np solubility was reduced two orders of magnitude from that used in the previous DSNF PA (Rechard, ed. 1995) and the YMP TSPA (M&O, 1995).

[place Fig. 1 here]

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* CST is a model developed specifically for the DSNF PA that is used as a subroutine in the flow and transport codes.

** Neptunium, technetium, and iodine radioisotopes are the primary contributors to human dose at the accessible environment since they are poorly retarded by the volcanic tuff. Because iodine behaves similarly to technetium, only neptunium and technetium are discussed in this section.

Findings for Waste Container and Repository Design

In the 1997 PA, the current YMP plans for the disposal containers and codisposal configurations for the DSNF, DHLW, and commercial fuel were incorporated, although the variety of dimensions and configurations were reduced to ease the modeling burden. In general, handling containers (canisters) of DSNF and DHLW were placed together in a disposal container (overpack); disposing of DSNF/DHLW in one container is referred to as the codisposal option. The commercial spent fuel was packaged with either 21-PWR assemblies or 44-BWR assemblies.

For the 1997 PA, the disposal region of the repository was sized to accommodate ~73,000 MTHM of spent fuel or equivalent high-level waste, with defense and commercial spent fuel comingled. The mine design for center in-drift emplacement is a tunnel and pillar configuration, with the entire repository consisting of 87 tunnels, 1236-m-long and bored by machine. The tunnels are surrounded and connected by access drifts, 7.62 m in diameter, which lead to ramps that connect to the surface.

Container Types

The DSNF and DHLW were modeled as being placed in both handling and disposal containers. Category 1 (N-Reactor fuel) was modeled in a multi-canister overpack (MCO), which consisted of a 0.95-cm-thick, 61-cm outer diameter stainless steel shell. DSNF Categories 2 through 13 were modeled in a 6.35-mm 304L stainless steel handling container. The DHLW handling container was modeled as the 0.95-cm-thick, 61-cm-diameter standard DOE canister.

The disposal container for all waste packages included a 10-cm-thick outer carbon steel layer and a 2-cm-thick inner Inconel 625 layer. For Waste Package 1, four MCOs were placed in a disposal container; for Waste Packages 2 through 13, DSNF/DHLW were codisposed in various configurations (Table II); for Waste Packages 14 and 15, 21-PWR and 44-BWR assemblies were placed in disposal containers, respectively.

[place Table II here]

Orientation of Waste in Repository

The orientation of waste packages was center in-drift on pedestal emplacement (M&O, 1995). The design goal in the 1997 PA was to use the hot repository concept. A center-to-center spacing of 28.6 m between the 5.3-m-wide tunnels was designed to maintain a heat load equivalent of 85 MTHM/acre for commercial fuel. The tunnels were assumed to be lined with a 0.2-m-thick precast concrete liner. A thick segment on the base of the tunnel forms an invert into which a minimum of 3 piers can be set for container support. Only the ramps to the disposal area were assumed to be backfilled and effectively sealed.

Container Conditions

The repository tunnels in which the waste packages are to be emplaced include an air gap between the waste package and the intact host rock. During the regulatory period, rubble from the back (top) and ribs (sides) of the tunnels was assumed to eliminate the air gap, to some extent. If rubble was present (absence or presence of rubble was a sampled parameter) and if the H_2O content of the tuff were high enough, movement of groundwater from the partially saturated tuff through the rubble to the waste packages was assumed. For the 1997 PA, four container groups were defined that specified (a) whether the inner Incoloy layer of the waste disposal container failed at the time of emplacement (yes/no) and (b) whether there was an air gap between the waste package and the intact host rock (yes/no). The corrosion submodel used in the 1997 PA included (a) generalized and localized corrosion and (b) oxic (humid and wet) and anoxic corrosion reactions.

Also, the effect of infiltration in the repository from climate change was modeled by means of a sinusoidal function to simulate potential variation of precipitation. Minimum infiltration, maximum infiltration, and time period were sampled. The resulting average infiltration over the model grid was varied and, in addition to this regional variation, a sample factor focused infiltration at one cell over the repository near the crest of Yucca Mountain between 1 and 10 times that over the rest of the repository.

Summary of Information Gathered about Repository Design

Results from BRAGFLO_T* showed that the thermal loading from combining the commercial spent fuel with the codisposed DSNF/DHLW was sufficient to boil away all of the water near the repository center. Maximum temperatures near the repository center were between 125°C and 145°C. Peak temperatures occurred in the first 30 to 300 yr (Fig. 2). At the repository edge, some runs (about one third) produced superheated steam, while others maintained the water in the liquid phase; hence protection from drying out the tuff was not consistent at the edge (Fig. 2).

[place Fig. 2 here]

Water vaporization during early times displaced the noncondensible gases (e.g., O_2) within the repository. The oxygen mass fraction was at or near zero from approximately 40 to 200 yr. At later times, the water vapor recondensed and the oxygen migrated back to the repository (Fig. 3).

[place Fig. 3 here]

Summary of Information Gathered about Waste Container

When the anoxic corrosion rate of the carbon steel was set high, then failure of 72% of the containers occurred in the first 3000 yr. Hence, the alteration of the fuel matrix also was rapid because it occurred when the repository was still hot.

Containers at the edges of the repository breached earliest, followed by the containers under the high infiltration zone artificially specified near the crest of Yucca Mountain. Containers near the edge breached earliest because (1) they were below the boiling point longer than other containers at early times and (2) the oxygen mass fraction at the edge was not suppressed as low or as long as at the center of the repository. Under the lower infiltration zone and away from the edges, containers with an intact air gap and with a good Inconel layer never breached.

Technetium is very soluble so its releases were dominated by advective releases. Neptunium is much more insoluble, and so its advective and diffusive releases were more similar. The time histories of releases were quite different across the repository because of differences in temperature and water flux, which affected the time of radionuclide exposure and the rate at which they were removed from the containers.

Findings for Geologic Barrier

The geologic barrier of the tuff disposal system comprises the sequence of tuff that isolates the repository from the accessible environment, i.e., the surface and any location elsewhere that is beyond 5 km from the repository (EPA, 1993). Only information from the unsaturated zone is presented here.

Unsaturated Zone

The stratigraphy of the tuff disposal system was idealized as a series of constant thickness hydrologic modeling units ("pancakes") with a dip of 4.6°. The thirteen units of the unsaturated zone consisted of consecutive tuff layers with degrees of welding and porosity that were similar. The repository was assumed to be located in the unsaturated zone about 260 m below the surface and 380 m above an aquifer. Two faults, Solitario Canyon and Ghost Dance, were modeled as having an effect on flow, with Solitario Canyon assumed as a lower permeability zone and Ghost Dance as a higher permeability zone. Welded lithologies were assumed to be more densely fractured than nonwelded.

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* BRAGFLO_T is a code modified specifically for the DSNF PA from software developed for the WIPP Project.

In the unsaturated zone, the composite porosity simplification (also called equivalent continuum) of the dual permeability model was used for fluid flow, i.e., the fractures and matrix were assumed to be in local thermodynamic equilibrium. Thus a single saturation and permeability function was used to represent the fracture and matrix media. For transport in the unsaturated zone, the model used a similar formulation. Two-phase flow (liquid and gas) with heat conduction, convection, and phase changes were included in the two-dimensional model. The cross-section modeled was perpendiçular to the 4.6° dip. Model layers were offset across the Solitario Fault but not across the Ghost Dance Fault.

Summary of Information Gathered about Geologic Barrier

The gas phase flow in the repository was driven by the heating of the repository at early times and subsequent cooling at later times. Gas flow was circulatory, with gas flowing along the sides of the repository, especially down from the surface through the more permeable Ghost Dance fault. Gas then moved vertically up through the repository and returned to the surface. As the repository cooled in time, the gas flow weakened.

The nuclide transport showed that the highly soluble technetium demonstrated a high release at early time with a resulting slug of high concentration that was transported via water flow to the vicinity of the water table with maximum concentrations at approximately 22000 yr (Fig. 4a). The transport of the less soluble neptunium from the failed canisters (mean solubility = 5.818 x 10^6 kg/m^3) was more gradual because precipitation occurred within the waste packages (Fig. 4b). Advection and diffusion of ^{237}Np from the packages brought the fluids in the repository floor up to the solubility limit soon after container breach. Continued release of neptunium maintained the grid blocks below the repository at the solubility limit well past the calculation times. At 100,000 yr, the contours around the concentration at the solubility limit approached the water table, with the 2 x 10^6 kg/m^3 contour reaching the water table. Thus the concentration profile was spatially distributed between the repository and the water table (Fig. 4b).

[place Fig. 4 here]

Overall Performance (Doses from Water Extracted from Unsaturated Zone)

For the preliminary results, the maximum annual effective dose equivalent was evaluated for a maximally exposed individual who received the peak dose by taking all drinking water from the *unsaturated* zone, consuming two liters of drinking water per day, 365 days per year.

Radionuclide Source Locations

For the calculations from consumption of water contaminated with radionuclides located at (a) the center and edge of the repository, (b) the region in the repository beneath the area of increased infiltration, and (c) a maximum concentration region just above the water table. The pathway to humans was from a well, located at one of the sources listed above (with no dilution), to a farm family that drank the water directly and subsisted on crops and animal products grown on the farm using the same contaminated water. A biosphere transport code, GENII_A, was used to calculate a 50-yr committed dose equivalent based on a 1-yr exposure, i.e., the 50-yr committed dose from contaminated water and food ingestion over a 1-yr period.

Summary of General Information Gathered on Total System Performance

Time histories of the committed dose were generated for a 100,000-yr period for the deterministic runs with parameters set at mean and median values. The doses from water taken at the repository location beneath the area of increased infiltration showed the highest peaks of the three repository locations. Early spikes that occurred for sources inside the repository (at times much earlier than 1000 yr) were due to ^{99}Tc and ^{129}I, whose concentrations then fell off rapidly. Although ^{237}Np concentrations also rose early, they did not rise above their plateau, at about 17 rem, and there was little change out to 100,000 yr.

As noted above, the doses from the locations inside the repository for this analysis were based on unrealistic assumptions. However, these calculations show the attenuation of peak concentration (and dose) as the radioisotopes move through the geologic barrier. Very significant attenuation of concentrations of ^{129}I and ^{99}Tc occurred during the transport to the water table from the repository through the unsaturated zone, for the doses were substantially reduced (by several orders of magnitude) at early times. However, at later times and out to 100,000 yr, at which points the dose was due almost entirely to ^{237}Np, the attenuation was not nearly so marked.

POTENTIAL WASTE ACCEPTANCE CRITERIA

Guidance Based on This Analysis

The overall dose calculations confirm results from past performance assessments (e.g., Rechard, ed., 1995; M&O, 1995) that the unretarded radioisotopes of technetium, iodine and neptunium (i.e., ^{99}Tc, ^{129}I, and ^{237}Np) are the most important radioisotopes for evaluating potential doses. In turn, the specific results about the waste form show that releases of ^{99}Tc, ^{129}I, and ^{237}Np from DSNF are less than those from commercial spent nuclear fuel on both a per package and per mass of heavy metal basis. Hence, if the commercial spent fuel can meet future regulatory criteria, then DSNF can also meet the criteria*.

In general, the overall results about the waste form, container, and geologic repository show that the source term model (which includes the waste form and container, and its interaction with the geologic barrier) is very important for evaluating the efficacy of the potential Yucca Mountain repository. (The source term model appears to be much more important here than for other types of repositories such as the Waste Isolation Pilot Plan [WIPP] repository in bedded salt for transuranic nuclear waste.) Only a few properties of the DSNF significantly influence the source term however, and thus only these properties should become part of the waste acceptance criteria. First, the inventory of ^{99}Tc, ^{129}I, and ^{237}Np is of primary importance. These radioisotopes are a function of the consumption of fissile uranium (^{235}U) ("burnup"); thus, burnup is an important DSNF characteristic to determine. (Because DSNF typically has lower burnup than commercial fuel, the DSNF typically contains fewer of these radioisotopes.) It is also important to note that Category 1, consisting predominantly of N-Reactor Fuel, makes up ~80% of the DSNF; hence, the N-Reactor fuel should garner the most attention by DOE/EM. Second, the hot repository concept makes temperature-dependent alteration rates of the fuel matrix (and possibly cladding) important information to acquire. Third, the specific results for the geologic barrier show that oxygen is likely to be depleted in the repository when the container fails, and thus pyrophoricity and combustibility of DSNF are unlikely to affect overall performance of the repository. Oxygen and water will eventually become available, but very slowly (i.e., the length of time required to oxidize or otherwise alter the fuel matrix deep in the geologic repository would be much shorter than the time it would take in a surface waste processing facility specifically designed to oxidize the waste for eventual disposal).

In summary, because the mass, volume, and activity of the DSNF are modest in relation to the commercial fuel, analysis of the waste forms should continue to demonstrate that DSNF is acceptable in the potential Yucca Mountain repository. Any negative characteristics of DSNF discovered over time are not likely to adversely influence the entire disposal system. Therefore, Sandia will likely recommend, after completing the 1997 PA, that direct disposal of DSNF remain the primary option considered by the DOE/EM, a recommendation that was supported by its previous performance assessments. This recommendation is predicated on the assumption that analysis to enhance DOE/EM's understanding of its waste will continue so that convincing arguments with regard to waste acceptance criteria can be presented to the NRC regulator. It is Sandia's belief that preliminary analysis that generates understanding of the real issues is less expensive in the long run than adopting expensive treatment and packaging options in the hope of reducing later analysis costs and gaining faster acceptance during licensing of the repository. Treatment and packaging plans, based on information gained through the waste analysis, can be developed as a secondary option but not necessarily implemented. The advantage of continued preliminary analysis is that it not only informs the decisionmakers about significant issues but can also contribute to safe, reasonable, and cost-efficient criteria for acceptance of DOE-owned waste.

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* An important caveat to this finding is that the modeled performance of the DSNF and commercial fuel is based on the YMP's current Total System Performance Assessment. Certain assumptions could change as the YMP TSPA evolves, such as accounting for the protection from the cladding of the commercial nuclear fuel, which would produce results more similar to those reported by Rechard, ed. (1995).

Value of Sensitivity Analysis

Comparison of the behavior of DSNF with commercial spent fuel is an excellent method of gauging the level of waste characterization and treatment necessary before disposal of DSNF. However, it is not without uncertainty, because the results are dependent on the features used in the model. Thus, the addition of features such as credit for cladding, which is currently not included in OCRWM's analysis, could affect the ranking of DSNF performance after disposal. Therefore, it may be in the DOE/EM's best interests not only to analyze its waste under disposal conditions currently envisioned by OCRWM but also to focus on significant parameters, such as the effect of cladding, that are revealed as important through sensitivity analysis.

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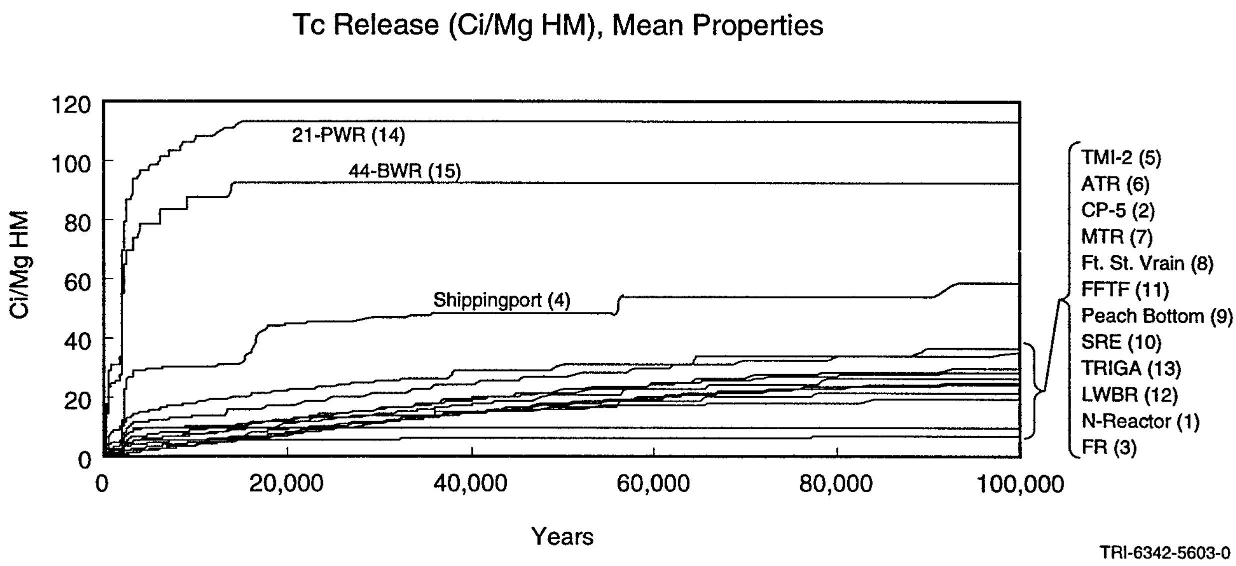
Table I. Spent Nuclear Fuel Categories

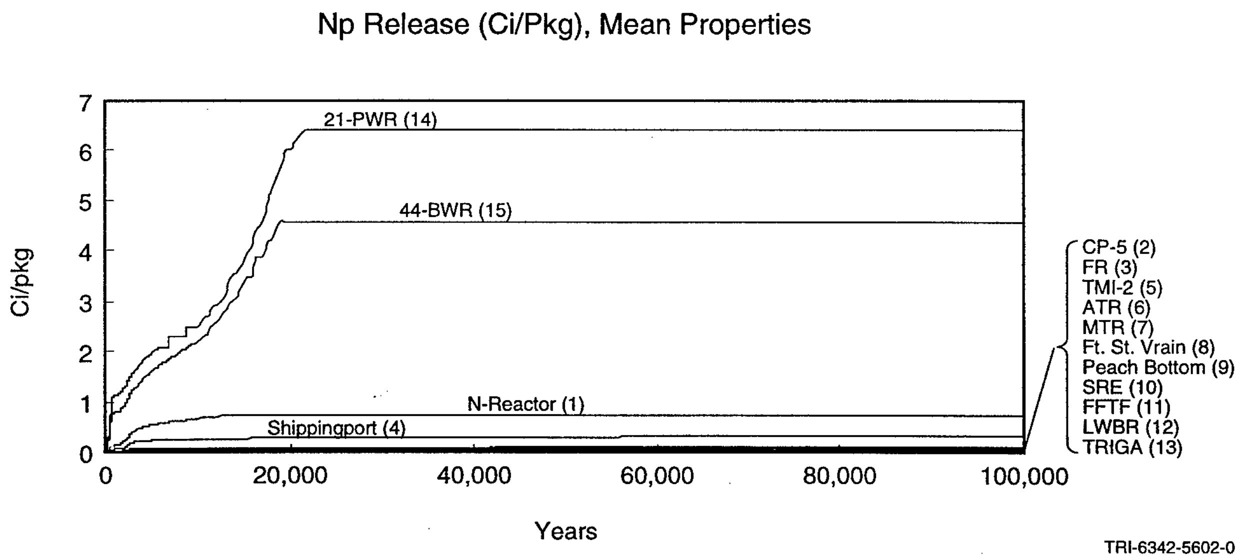
No.	Waste Type	Represented by	Representative Cladding	Curies (2030 yr) (Total/ Category)	DSNF Total Mass of Heavy Metal (Total Mg/ Category)
1	Uranium metal	N-Reactor	Failed	1.35×10^7	2132
2	Uranium-zirconium alloy	Chicago Pile 5 (CP-5)	Zircaloy	5.05×10^5	0.040
3	Uranium-molybdenum alloy	Fermi Reactor	Zircaloy	1.35×10^6	420.9
4	Uranium oxide-intact clad	Shippingport	Zircaloy	2.14×10^7	79.8
5	Uranium oxide-failed clad	Three Mile Island (TMI-2)	Zircaloy	2.91×10^7	88.3
6	Uranium aluminum alloy	Advanced Test Reactor (ATR)	Aluminum	5.01×10^7	9.42
7	Uranium silicate	Materials Testing Reactor (MTR)	Aluminum	1.27×10^7	12.7
8	Uranium-thorium carbide-intact clad	Ft. St. Vrain	Graphite	2.59×10^7	24.7
9	Uranium-thorium carbide-failed clad	Peach Bottom	Graphite	4.65×10^6	1.61
10	Uranium/plutonium carbide	Sodium Reactor Experiment (SRE)	Stainless steel	2.45×10^5	0.057
11	Mixed oxide fuel	Fast Flux Test Facility (FFTF)	Stainless steel	1.58×10^7	3.96
12	Uranium-thorium oxide	Shippingport Light Water Breeder Reactor (LWBR)	Zircaloy	2.16×10^6	55.5
13	Uranium/zirconium hydride	Training, Research, and Isotope production-General Atomic (TRIGA)	Stainless steel	2.96×10^6	2.22
14	Commercial fuel, PWR	21-PWR (Pressurized Water Reactor)	Zircaloy	8.37×10^9	41,530
15	Commercial fuel, BWR	44-BWR (Boiling Water Reactor)	Zircaloy	3.70×10^9	21,640

Table II. Codisposal Configuration Options for DOE-Owned Waste

Waste Cate-gories	Disposal Container				Handling Container			
					DSNF		DHLW	
	Length (m)	Diameter (m)	Mass (Mg)	Number	Number	Diameter (cm)	Number	Diameter (cm)
1	5.30	1.725	16.73	118	4	61	0	—
12	5.30	1.725 (std)	15.12	69	1	61	3	61
3, 5	3.79	1.725 (std)	15.12	650	1	25.4	4	61
10, 11	5.30	1.725 (std)	21.20	357	1	25.4	4	61
4, 13	3.79	2.0 (super)	19.02	305	1	43.2	5	61
2, 6, 7, 8, 9	5.30	2.0 (super)	26.38	1632	1	43.2	5	61



(a)



(b)

Fig. 1. Fraction from the waste package of (a) ^{99}Tc normalized by mass of heavy metal radionuclides in inventory and (b) ^{237}Np normalized by number of waste packages in each category. (Preliminary results from 1997 DSNF PA.)

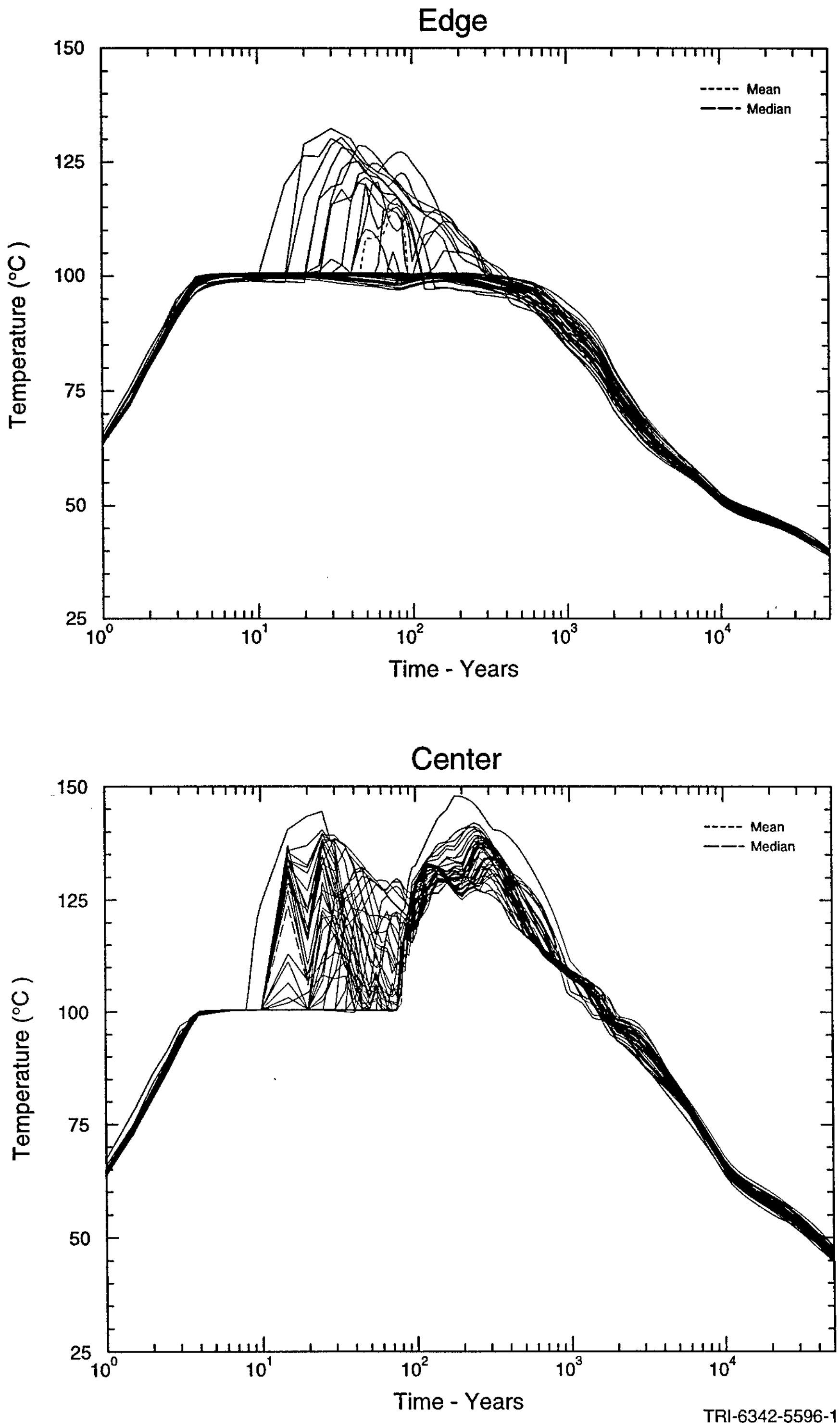


Fig. 2. Temperature at edge and center of potential repository over 50,000 yr. (Preliminary results from 1997 DSNF PA.)

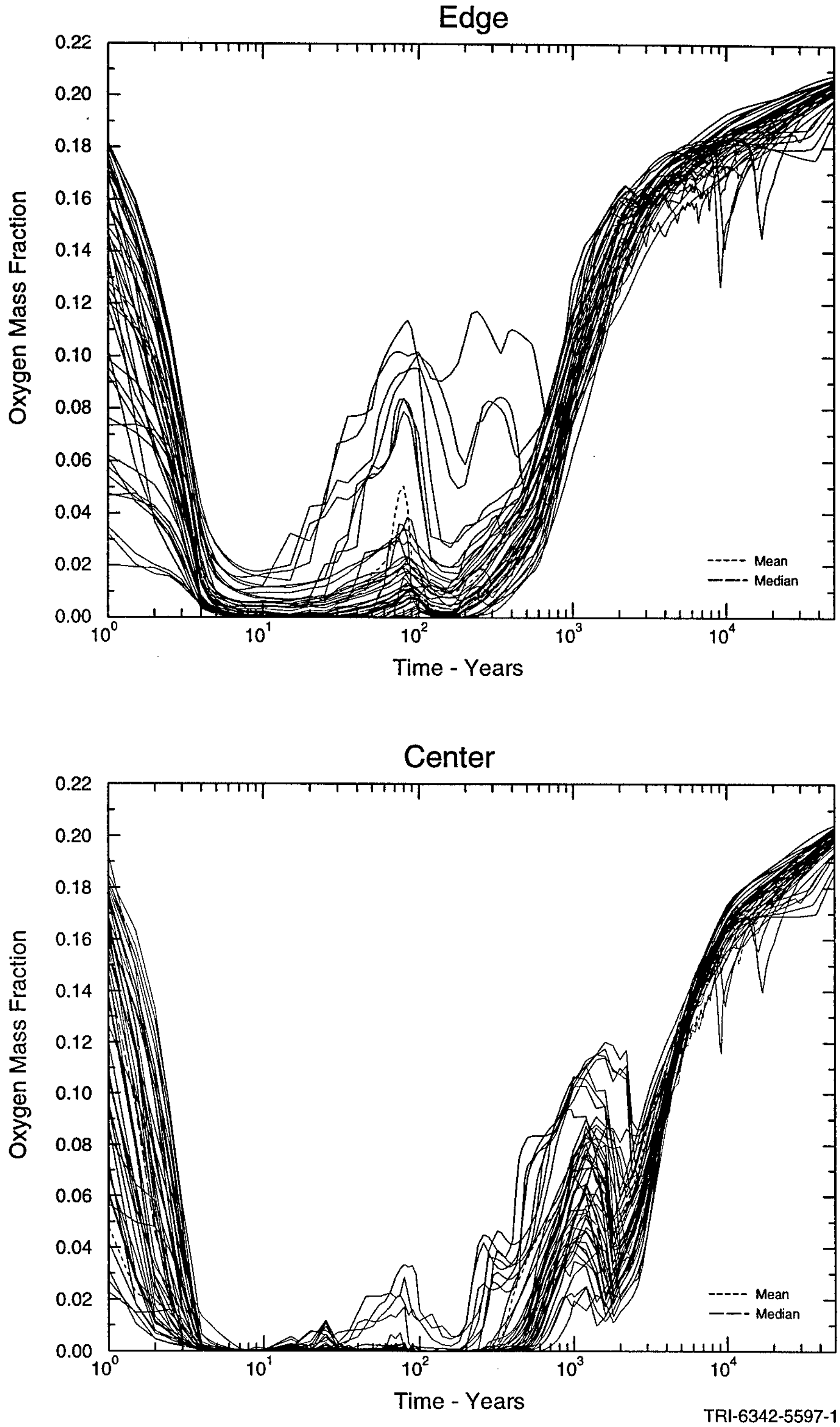


Fig. 3. Mass fraction of O_2 available for oxic alteration of fuel matrix at edge and center of potential repository over 50,000 yr, assuming very rapid failure of containers. (Preliminary results from 1997 ÐSNF PA.)

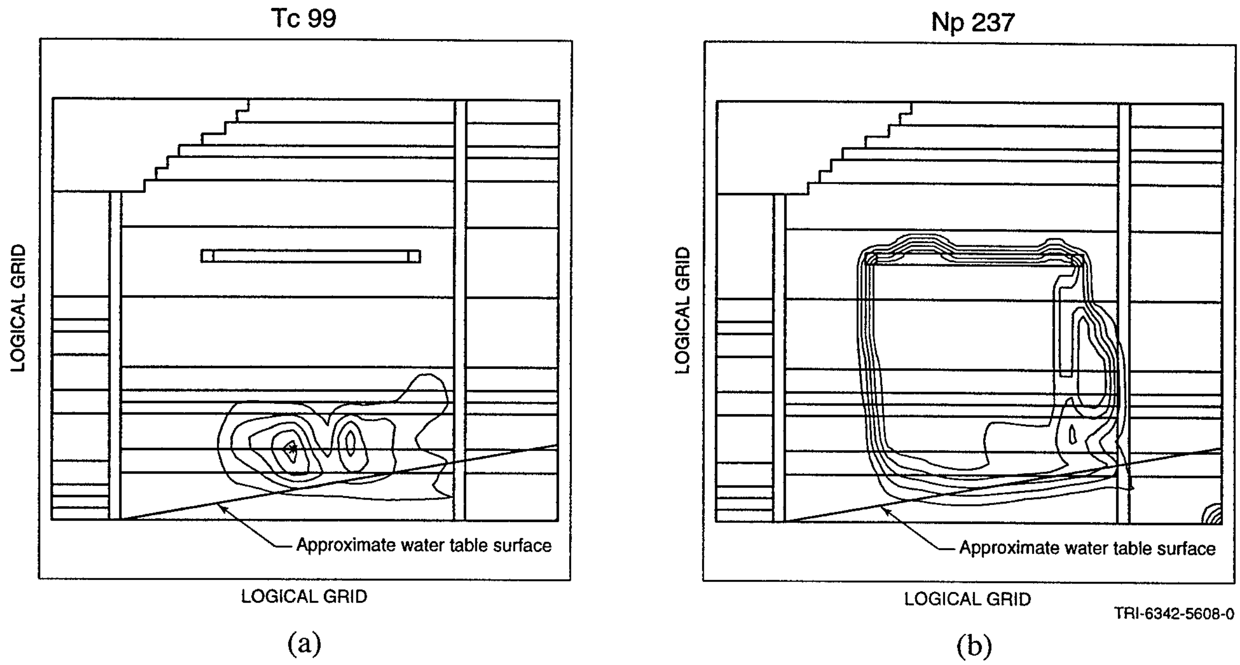


Fig. 4. Contours of radionuclide tracer concentrations (kg/m^3) using mean values for (a) ^{99}Tc at 22,000 yr and (b) ^{237}Np at 100,000 yr. The time period represents the time at which each of these radioisotopes reaches peak concentration at the water table surface. (Preliminary results from 1997 DSNF PA.)

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