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Abstract

This work develops the tools to investigate the hypothesis that the savings in repository space associated with the implementation of advanced nuclear fuel cycles can result in sufficient cost savings to offset the higher costs of those fuel cycles. Since the largest costs for building and operating a repository are variable costs that scale with the length of tunnel to be drilled in the underground system, the cost of emplacing waste in those repositories should be connected to the amount of space used by that waste and NOT to the amount of energy produced by that waste. When such an accounting scheme is used, the differential in space required for reprocessing HLW vs spent fuel amounts to a fuel cycle costs savings that can potentially pay for the separations process itself.

This report summarizes the development of approximate numerical models to calculate the thermal loading limit of spent nuclear fuel and other high-level waste in a Yucca Mountain-like geologic repository. Such models are necessary to determine the loading density for a generic waste form/package, and hence the amount of space required for a given quantity of material. By implementing this model in the VISION fuel cycle simulation code, we can easily examine the impact of separations efficiency and other fuel cycle parameters on the total repository space required by a nuclear enterprise over century time scales.

Moreover, by deriving the cost per unit space in the repository from the current nuclear waste fee, it is possible to determine the degree of cost savings available in each alternative fuel cycle. Since this cost savings scales linearly with the nuclear waste fee, it is possible to find an increased nuclear waste fee that is the break-even point for the advanced fuel cycle. Any waste fee above that point suggests that the advanced fuel cycle would cost less than the typical “once-through” fuel cycle.

We do find that there are potential space savings of at least 8x that is necessary for the modest growth scenarios used here to be accommodated by a single Yucca-Mountain-like repository. Similarly, we find that an increase of the nuclear waste fee by only a few mills/kWh may be sufficient to pay for MOX reprocessing and by about 8 mills/kWh may be sufficient to pay for an advanced TRU recycling fuel cycle.

All results reported here useful for comparison purposes, but are less useful for determining absolute quantities.

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Project Objective:

This project will develop models to study the impact of a policy that retains a single spent fuel storage repository (Yucca Mountain). An increasing amount of technology-driven analysis is being done to determine how reprocessing and separation technologies can improve the loading of Yucca Mountain while maintaining the same general design concept and licensing basis. However, little policy-driven analysis has been conducted to study the adoption of new reprocessing and separation technologies assuming only a single repository is constructed. Given the current mismatch between the legislative limit for capacity of Yucca Mountain and the expected spent fuel inventory from the existing reactor fleet, a single-repository policy would likely improve the economic attractiveness of advanced fuel cycle technologies. In addition, this project will study the relationship between fuel cycle advances driven primarily by repository performance and those advances necessary for a transition to a sustainable fuel cycle that is both economical and feasible. This project supports the Department of Energy's evaluation of additional geologic repositories for spent nuclear fuel (beyond Yucca Mountain), which is one of the near-term goals of the Advanced Fuel Cycle Initiative.

Background

This project complements a number of other investigations of potential repository benefits due a potential advanced fuel cycle by aiming to define strategies that optimize the cost and schedule of introducing advanced fuel cycle capabilities, rather than simply determining the costs of different scenarios and schedules.

This work uses the design and license basis for a geologic repository at Yucca Mountain as the reference for determining the parameters for the thermal model that determines the packing density of high level waste. The benefit of this approach is that it allows specific parameters to be calculated for a specific design. The disadvantage is that its applicability is limited by the degree to which the physics of an alternate repository will be similar to that of Yucca Mountain.

Defining Metrics for Repository Performance

Ultimately, the most important measure of repository performance, and the basis for licensing, is the dose to an individual at the boundary of the site. However, a rigorous assessment of dose at the boundary is not possible in the context of a high-level systems analysis, so a simplified model is necessary, based on more easily calculable limits.

Using the documents available on the licensing support network for the Yucca Mountain project [1], a literature review was performed to investigate any existing specifications on waste packages and the analysis methods by which they are determined. This review identified performance measure for the repository as a whole and a number of specific criteria for the waste

packages, however most are based on the fact that these packages contain once-through light water reactor fuel.

Previous work by Argonne National Laboratory (ANL) has developed some simple specifications for the allowable heat load for packages emplaced in Yucca Mountain [2]. In order to still meet the currently anticipated licensing basis, there are three simple limits: a limit on the instantaneous decay power at the time the forced convection ceases, a limit on the total released decay energy over a time period following the end of forced convection, and a limit on the instantaneous decay power at time of emplacement in the repository. These limits are being explored by ANL for use in fuel cycle studies and will be considered in this project also.

In addition to the thermal limits for repository loading, ANL has studied radiological limits. Early indications are that repository loading will be limited by thermal considerations and not radiological considerations. That is, any loading which meets the repository thermal performance limits will necessarily have radiological loadings that result in exposures below the stated limits. Changes in radiological exposure limits and regulations during the last quarter may have an impact on this conclusion.

Thermal Limits

The previous work on repository benefit was reviewed to determine the correct value of those thermal limits for implementation in these models. There are two temperature limits on waste disposal in Yucca Mountain: the temperature of the rock at any position may not exceed 200°C and the rock temperature midway between drifts may not exceed 96°C. The origin of the first limit is driven by limitations in the database for the geologic materials near the repository. The behavior of fluid and chemical species transport in the host rock is better established for temperatures below 200°C. If the rock were allowed to exceed this temperature, substantial uncertainties would be introduced into the overall safety assessment of the repository. The second limit is established by the design philosophy of the repository. Keeping a point between tunnels will remain below the boiling point of water (96°C at this altitude) ensures that there is always a pathway for water, as rare as it may be, to move through the system and not accumulate above the repository plane. If this temperature limit were not observed, water could be suspended above the repository on a vapor barrier in the rock, only to drain into the repository all at once when the temperature drops.

Previous work by Wigeland, et al, has shown that there are three times at which these temperature limits translate into loading limits: the drift wall can exceed 200°C at emplacement or at closure and the between drift temperature may reach 96°C many centuries after closure. The most challenging limit to determine is the long term mid-drift temperature because the time over which the integration should take place is a function of the isotopic composition itself. More precisely, the time at which the maximum long term mid-drift temperature is reached is a non-trivial function of the thermal time constant for the repository host rock and the time constant of the radioactive decay of the waste resident in that host rock.

Dose Limits

In terms of improved repository loading constraints for VISION, a literature review to establish an appropriate source term model was conducted. This review indicated that a relevant definition of a limiting source term metric will reflect the magnitude of radiotoxicity reaching the reasonably exposed individual described by the legislation. Source term then will need to be modeled as radiotoxicity that has escaped from failed waste containers and traveled through the rock to a location 18 km downstream from the disposal site. It is clear that a source term model incorporated into VISION should combine the already implemented radiotoxicity data with a canister failure model and data reflecting preferential isotopic transport through the granite tuff. Studies from the literature have determined that while the relationship between repository loading and total source term from a once through fuel cycle can be approximately modeled as linear, other fuel cycles, particularly those involving various waste forms, will require a more intricate model. So, while the preferential hydrogeologic transport of isotopes can be simulated in a straightforward manner, incorporation of waste form/container failure presents a deeper challenge to the model if it is to allow for the myriad waste forms anticipated in advanced fuel cycles. A high fidelity source term model that incorporates waste package failure rate, nuclide dissolution rate, and advective transfer rate has the potential for sensitivity to nuclide concentration in waste package loading. Such sensitivity complicates the viability of alternate loading schemes as well as waste separation scenarios.

Another complication is the concentration dependence of nuclide release: once a waste container is breached, the dissolution rate and the advective transfer rate through the granite tuff depend directly upon the diffusion coefficient of that nuclide, which for some nuclides is heavily dependent upon the concentration. Also, treatment of waste package failure rate varies between models. While some employ EBSFAIL, a part of the EBSPAC module used in the TSPA code, other models incorporate their own hydrogeologic approximations of canister degradation, and still others assume immediate waste canister failure in order to focus on dissolution and transfer. The key difference in models of nuclide release from its waste matrix into the saturation water involves the mode of water flow through the packages. While some model water moving through the waste packages at a constant volumetric rate ('flowthrough model'), others adopt less conservative assumptions relying on weather-based predictions of hydrogeologic activity. A reasonable model for fuel cycle simulations will be most similar to the flowthrough model.

Repository Performance Models for Systems Analysis

The primary capability to be delivered by this work is the comparison of the repository capacity requirements for alternative fuel cycles and the economic feedbacks that such capacity requirements will create. This section describes the choice of systems analysis code, and introduced components of the model that are necessary to deliver this capability.

Systems Analysis Codes

Both DANESS [3] and VISION [5] were identified as possible targets for development of the repository models of this project. This work originally targeted DANESS v2.x for implementation of the repository benefit analysis. However, the emergence of VISION as a primary decision analysis tool for the DOE at the same time that we identified software limits of the iThink/Stella platform for DANESS [4], caused a change in direction to develop models for VISION.

Later in the project, version 3.x of DANESS was also targeted for this development. In preparation for implementation of a capacity-based model based on isotopic compositions, inconsistencies in the isotopic decay model were discovered. Furthermore, the software infrastructure still had severe limitations that would require removing other models to make room for the repository model. The ISEE systems framework has been studied to develop an approach for implementing the capacity-based model. At the close of this project, the iThink infrastructure had advanced sufficiently to allow this implementation in DANESS v4.x, although it was never attempted.

Beyond these fundamental limitations, other characteristics of fuel cycle simulation codes became important in the development of this work.

Feedback and Decision Making Strategies

There was an early focus on the deployment of nuclear fuel cycle facilities and the assumption that economic feedback loops would allow downstream costs (e.g. HLW disposal costs) to influence upstream decision making on the deployment of facilities. While neither system had a direct feedback algorithm, DANESS did have a mechanism for rolling back time to correct deployment decisions that proved unwise (generally due to cost). However, it was not clear that such a mechanism would be able to force broad changes in fuel cycle concept, rather than smaller changes in deployment within a given concept. When it was introduced, VISION had no mechanism to make deployment selections based on downstream results. Due to the lack of such feedback and concerns about the ability to robustly implement such feedback, optimization was considered as a more robust mechanism for finding the appropriate fuel cycle solutions. Although they can be contrasted in many ways, feedback or decision making algorithms within the model provide a temporally local optimization and not a global optimization. In the search for optimal solutions, or trade-offs between optimal solutions, the global optimizations found by a true optimization mechanism are more robust and valid than the solutions found by a successive application of local optimizations in the time domain. The approach changed from relying on decision making algorithms within a given simulation to that of performing separate simulations/scenarios that could then be compared *a posteriori*, possible as part of a broader optimization approach.

In concert with the SINEMA/GENIUS effort at INL, an assessment of optimization strategies has been completed, with particular attention to the contrast between “local” optimization that is common among current tools and “global” optimization that can be expected to find idealized policy options [Publication 1]. It may be necessary for fuel cycle simulation tools to reduce the degree and complexity of their decision making and feedback algorithms to permit a global optimization approach to be successful. These algorithms are intended to simulate the real decisions made by, and constraints faced by, actors in the development of a nuclear enterprise. The result is a set of successive optimizations based on local information that may not lead to a global policy optimum. On the other hand, global optimization evaluates the overall objectives of the system in the absence of these algorithms to identify an ideal development scenario. An interesting combination of these two types of optimization can be used to design policy instruments: algorithms can be added to the local optimization scheme in attempt to force the local optimization to achieve the same performance as the global optimum.

In the transition from local to global optimization, the decision variable space may increase in size substantially. As such, some careful consideration must be given to the selection of decision variables and the degree to which some decision-making/feedback algorithms may be determined to always contribute to a global optimum and permit a reduction in the size of the decision variable space. An example has been developed for the case of reactor construction schedules, with a substantial savings in the decision variable space arising from an algorithm that maximizes reactor capacity factors at all points in time.

In addition to the general literature of optimization methods and techniques, some attention was paid to the concept of meta-models that can be used to accelerate the optimization process. Since the scenario tools, given their infrastructure, are likely to be used as “black-box” tools that provide results as a function of some broad set of input parameters, it may be necessary to functionalize the relationship between the input parameters and the output values in some approximate form to determine the neighborhood of the optimum parameter settings, returning to the scenario simulation itself for the final determination of the optimum input parameters [1].

Repository Capacity Model

In addition, neither DANESS nor VISION had a sophisticated representation of a geologic repository – in fact neither was able to treat a repository of limited size and both simply accumulated material in the single repository with no deployment considerations. DANESS, in particular, will overfill storage and disposal facilities without perturbing the fuel cycle scenario parameters. In particular, in an over-constrained scenario with limited storage and repository space and a required energy demand, the solution from DANESS placed more value in the energy demand constraint, ensuring sufficient energy supply and filling the repository to 250% capacity. In VISION v1.x/v2.x, the repository was not even treated as a facility with a fixed/stated capacity.

Conceptual development of a repository capacity model resulted in a proposal for inclusion of such a model in both DANESS and VISION [2]. It is useful to formulate a repository loading and costing model based on the available space in the repository. However, the total cost of waste disposal is best expressed in units of cost per unit energy. A model that permits both of these goals is proposed in equation (1) where C is the disposal cost per unit of energy generated [\$/MWh], U is the unit cost of space in the repository [\$/m], ρ_{limit} is the limiting waste loading density [kg_{HLW}/m], α is the fraction of the HLW that is composed of fission productions [kg_{FP}/kg_{HLW}] and E is the amount of generated energy per unit mass of fission products [MWh/kg_{FP}].

$$C = \frac{U}{\rho_{limit} \alpha E} . \quad (1)$$

The limiting waste loading density is presumed to be calculable for a given repository design and licensing basis from the isotopic composition of a given HLW stream. The intermediate result of C/U [m/MWh] allows a system studies tool to determine when a repository will be full based on the amount of tunnel space available rather than the mass of material being loaded into the tunnel. The inverse of this quantity can be defined as the energy loading (EL) of the repository,

$$EL \equiv \frac{U}{C} = \rho_{limit} \alpha E . \quad (2)$$

One important feature of this model is that the economic quantities are related to physical characteristics of the repository and the waste disposed therein. Another important feature is that these quantities are entirely independent of assumptions about the form or history of the waste. Moreover, these measures must capture the fundamental concept of the repository capacity model introduced in this work: the *value* of a repository is separate from its *cost*. The value of the repository represents the economic value of the electricity/energy produced by the material being stored there, while the cost represents the economic expense of making that capacity available. The quantity developed to represent the value, αE , is the *embodied delivered energy* and the quantity developed to represent the cost is simply the length of the repository.

The review of technical limits for the repository found that a thermal limit model would be possible for implementation in a fuel cycle systems analysis tool, but that a source term/dose limit model would be more challenging. As such, this work focuses on developing a thermal limit model to calculate the limiting waste loading density, ρ_{limit} .

Embodied Energy for Fuel Cycle Comparison

The concept of embodied energy is best related directly to the quantity of fission products in any material stream since they are the only indicator of actual energy generation. This simplifies the modeling of embodied energy necessary to determine the cost per unit energy of different fuel

cycle options. However to best partition the costs of different fuel cycle processes, it may also be necessary to estimate the potential energy remaining in the material.

In general, the term *embodied energy* is used in the energy analysis field to account for the process input energy used to produce a material or item. For example, in a nuclear fuel cycle, the mining, conversion and enrichment processes would increase the embodied energy of a unit of nuclear reactor fuel due to the energy used in performing these processes. While this quantity is certainly important in a comprehensive fuel cycle analysis, this work will concentrate more on the theoretically available fission energy (*embodied potential energy* [EPE]) and expended fission energy (*embodied delivered energy* [EDE]). Although many economic quantities are normalized by the delivered value (often converted to electrical energy via a thermodynamic conversion efficiency), most fuel cycle study tools track masses/mass flows of materials within a fuel cycle scenario. Understanding both the potential and delivered energy of those materials in economic terms is necessary when performing economic comparisons of different scenarios. The simplest approach is to ascribe roughly 200 MeV of EPE to every remaining actinide atom and roughly 100 MeV of EDE to each fission product atom. This results in approximately 1 MW_d/g of EPE for actinides and a similar quantity of EDE for fission products.

Associating the available energy exclusively with the actinides and the expended energy exclusively with the fission products preserves economic concepts that will be important for advanced fuel cycles. In particular, recycling of spent nuclear fuel results in a (primarily) fission product stream with a very high specific expended energy and an actinide stream with a very low specific expended energy and a high specific available energy. This concept is directly related to the already demonstrated repository benefit of recycling spent nuclear fuel. The variable cost of constructing a deep geologic repository is dependent on the length of the tunnels and independent of the EDE being disposed in those tunnels. By increasing the EDE of the material stored per unit length of tunnel, the economic utilization of the repository resource is improved.

Since different combinations of processes and reactor systems will have a wide array of total costs and also result in a wide array of EDE and EPE values for different material streams in the system, it is important to track the economic value of the streams through the use of the specific EDE and EPE quantities. Since this the EDE is most often related to the electrical energy and not the thermal energy, it is necessary to track this quantity as a separate characteristic of a material due to the variety of thermal efficiencies that may exist in a fuel cycle scenario. On the other hand, the thermal efficiency that will be applied to the utilization of the EPE is generally not known (or may even be varied). Therefore, this quantity need not be tracked as a separate characteristic of a material since it is directly related to the quantity of actinides in the material (a quantity which is generally tracked already).

Thermal Loading Limit Model

The energy loading is a product of this embodied delivered energy and the limiting loading density. With the above strategy for determining the embodied energy, a model for the limiting loading density was developed, based only on the thermal limits.

Collaboration with Argonne National Laboratory allowed us to take advantage of their thermal model of the Yucca Mountain repository developed in SINDA/G. Given the three distinct limiting cases described above, the model developed here determines a limiting repository loading density, ρ_{lim} [kg/m], for each case and then uses the most limiting loading density. All three models are based on some form of linear superposition, combining the thermal response of the system to each isotope weighted by the relative mass of that isotope in the waste stream. For each of the 60 isotopes tracked by VISION, ORIGEN-ARP¹⁰ was used to determine the long term decay power resulting from 1 kg of that isotope, including the impact of its decay products. The reference model was then used with each isotope alone to determine the thermal response of the system to a single isotope.

Drift Wall @ Emplacement

The proximity of the heat source to the drift wall results in a thermal response that is directly proportional to the instantaneous decay power, D [W/kg], in the waste packages:

$$\Delta T_1 \propto D \rho_{lim,1}. \quad (3)$$

However, at these early times, the time constant for the reduction in the decay power is much shorter than the thermal time constant for the host rock. Therefore, an effective mean decay power, D_{eff} , is used:

$$\begin{aligned} \Delta T_1 &\propto D_{eff} \rho_{lim,1} \\ D_{eff} &= (1-a) D_o + a D_{0.3y} \end{aligned} \quad (4)$$

Where D_o is the decay power at emplacement and $D_{0.3y}$ is the decay power 0.3 years after emplacement. A number of relevant isotopic combinations were chosen from VISION a decay power transient for each combination was determined using linear superposition. This set of results was used to fit the following model for the first loading density limit:

$$\rho_{lim,1} = \frac{C_1}{D_o + a(D_{0.3y} - D_o)}, \quad (5)$$

where $C_1=7085.9$ W/m and $a=0.9023$. The average relative difference between this model and the reference results is 1.8% over eight different isotopic recipes, with a maximum relative difference of 6.7% for a very small loading (11 kg/m).

Drift Wall at Closure

Because the time constant of the decay power is much longer at closure than at emplacement, the form of equation (3) is appropriate for this limit. Again, a set of results from the reference model was used to fit the following model for the second loading density limit:

$$\rho_{lim,2} = \frac{C_2}{D_{closure}}, \quad (6)$$

where $C_2=662$ W/m. When comparing the results of this model to the reference model, the root-mean-squared discrepancy is 16 W/m (2.4%) over 12 different isotopic recipes. The maximum relative difference was an under prediction of the loading density limit by 6.4%.

Mid-Drift at Long Times

Because of the 41m of rock that separate the heat source and the mid-drift point, the long term transient response of the system requires a more complex approximate model. The time at which the limiting temperature is reached is strong function of the transient decay power behavior and thus a function of the specific isotopic mixture.

The temperatures at the mid-drift, as a function of time for 10,000 years, were determined for each isotope on its own, using the loading limit for that single isotope. Under the assumption of superposition, the change in temperature at the mid-drift should be a linear combination of the individual isotope results, weighted by their relative mass fractions. In this problem, however, superposition is not strictly true, so a variety of studies were carried out to determine under which conditions superposition is a reasonable approximation.

First the linearity of the temperature response with respect to the material loading was tested. Because of the underlying temperature gradient between the surface and the water table, this was found to be highly non-linear. When the material loading is low, the underlying gradient dominates. As the material loading increases, this underlying gradient becomes less significant and the results approach a linear response. Another source of non-linearity is the inclusion of boiling/condensation of water in the host rock. This, in effect, makes the material properties a function of the temperature disturbing the inherent linearity of the governing equation. In practice, the primary effect is to delay the onset of the limiting temperature as energy is deposited in the boiling of water rather than being free to conduct to the mid-drift. For both of these effects, we determined that using single isotope results that are derived from the limiting material loading for that isotope will mitigate the non-linearity and allow for a model based on linear superposition.

When this was tested by adding the responses from two isotopes together, it was found that the largest differences between the superposition model and the SINDA/G results occur when isotopes reach their peak temperatures at very different times. This is due to the above-mentioned effects and the manner in which they compound for multi-isotope problems. The

final round of testing was based on the thermal response from mixtures of isotopes characteristic of spent fuel and high-level waste recipes used in VISION 1.4. When the model results for the material loading is compared to the results found by SINDA/G, the model is able to approximate the loading of material into the drifts within 8% in all cases, and within 3% on average.

In the final model, the time-dependent temperature change at the mid-drift point was determined for the limiting loading density of each isotope, i , and normalized to a single kg of that isotope creating a so-called specific temperature change (STC_i) vector. Superposition was invoked with these STC_i vectors to determine an STC vector for a given isotopic recipe. The maximum value of this vector, STC_{max} , indicates the time at which this mixture of isotopes will cause the temperature limit to be reached and hence determines the loading limit:

$$\rho_{lim,3} = \frac{\Delta T_{lim,3}}{STC_{max}}. \quad (7)$$

Results from this model were compared to the reference model results for 21 different isotopic recipes. The average relative difference was 2.7% with a maximum relative difference of 7.6% (under prediction).

Implementation in VISION

At each time step in a VISION simulation, all three of these limits are calculated for each of the flows of material into (or out of) the repositories. The minimum loading density limit is used for each stream and the total required repository space [m] is accumulated. Both the short term decay powers and long term STC vectors for each isotope were added to the spreadsheet-based data input for use in these models. Because of a strong sensitivity to the time between emplacement and closure, a user-defined closure time was implemented, requiring isotopic decay powers up to 160 years.

Repository loading metrics were added to the model including the overall mass loading factor, measured in tonnes/m, and, more importantly, an energy loading (EL) measured in MWd_{th}/m. For the latter, the fission product mass was used as an indicator of the total number of fissions and hence the embodied delivered energy (EDE) from that material. The ELF is a meaningful metric for the comparison of the repository impact of different fuel cycle choices because it effectively provides normalization for the total energy generated in the nuclear system.

Scenario Analysis

Several VISION v1.x base case scenarios were run to investigate their potential benefit with respect to repository loading. Table I summarizes results from these test cases for new spent fuel (SF) and high level waste (HLW). In all cases involving UOX, 43.2 kt of legacy spent fuel (LSF) was sent to the repository consuming 15.7 km and delivering 1617 TWd of energy. In the FR Breeder-IMF case all LSF was reprocessed with none left in the repository in the year 2100.

A simple separations model was used in which 99.9% of actinides (and 0.1% of other isotopes) were returned to the fuel cycle and the remaining isotopes were sent to the repository as HLW.

The packing density in all test cases was limited by the temperature at closure. When the same cases were run using immediate emplacement and 80 year decay time before closure, the temperature at emplacement was always limiting. At some intermediate decay time, the LSF and SF would still be limited by the closure temperature while the HLW would be limited by the emplacement temperature. Further study would be needed to determine the optimal decay time before waste is loaded into the repository.

The results described here are should be used for comparing the relative advantages and disadvantages of each cycle option, rather than for determining absolute repository loading factors.

Table I. Summary of VISION 1.4 Base Case Results after Implementation of the 200°C Temperature Limits.

Case	HLW Mass (Kt)	HLW Length (km)	HLW MPF (Mt/m)	HLW EL (GWd/m)	SF Mass (Kt)	SF Length (km)	SF MPF (Mt/m)	SF EL (GWd/m)	Overall l MPF (Mt/m)	Overall EL (MWd/m)
UOX-UOX	0	0	---	---	200	122	1.64	79.6	1.77	82.3
UOX-MOX	18.4	49.8	0.37	345	86.5	59.7	1.45	70.3	1.18	184
UOX-IMF	19.7	53.0	0.37	348	20.9	15.2	1.38	70.7	1.00	252
UOX-FR Burner	20.7	55.8	0.37	347	43.0	24.0	1.79	86.8	1.12	241
FRBreeder- IMF	22.6	60.1	0.38	352	0	0	--	--	0.38	352

Using the repository loading model described above, a detailed analysis of the impact of separation efficiency on repository size was performed across a set of characteristic fuel cycle options selected from the VISION base cases. Ten (10) cases were selected to provide a number of illustrative comparisons. Table II summarizes the cases that were selected with comments on the specific comparison that it supports. All cases use the same total nuclear energy demand curve as their input, but supply that demand with different combinations of reactor technology. Of particular interest are comparisons of fuel cycle choice, including specific transuranic isotopes in thermal recycling schemes, and fuel cycle transition timing. Case 1 represents a standard business-as-usual once-through fuel cycle.

Although currently an active area of model improvement in VISION, only a very simple separations model was implemented in the version used here. In the separations module, each isotope is separated into two streams, one that goes to the repository labeled as high-level waste (HLW) and one that returns to the fuel cycle. A single separations efficiency, e , was defined to represent the partitioning of all elements. Depending on the fuel cycle scenario, a fraction e of the recycled transuranic elements were returned to the fuel cycle and $(1-e)$ were sent to the repository. Conversely, $(1-e)$ of the non-recycled transuranic elements and fission products were

returned to the fuel cycle and the remaining e were sent to the repository. For each scenario, only transuranics that would eventually be used in reactor fuel were recycled and they were recycled at all opportunities. That is, even if a given element was not to be used in fuel during Phase 1, it would be recycled if it was eventually going to be used in Phase 2. Three different separation efficiencies were compared: 0.9, 0.99, 0.999.

In some cases, an additional waste stream was added to the VISION model for the separation of Cs and Sr. These elements are known to have an important impact on the loading density limit at emplacement and/or closure and deserve special consideration. In these cases, a separate loss fraction was defined, l_{CsSr} , and that fraction of the Cs and Sr was sent the repository and the same fraction was returned to the fuel cycle, with the remainder being placed in a separate storage location. In addition to the cases where Cs and Sr were treated like other fission products using the separation efficiencies above, three different loss fractions were compared: 0.001, 0.01, 0.1.

Table II. Summary of scenario selection

Case #	Phase 1 (2025)	Phase 2 (2040)	Comments	Comparison
1	UOX	UOX	Base case representing once-through fuel cycle	
2	UOX	MOX(PN)	Late transition to PuNp MOX fuel	
3	UOX	MOX(PNA)	Late transition to PuNpAm MOX fuel	Compare to case 2 to consider impact of Am
4	UOX	FR Burner	Late transition to fast reactor burner (CR=0.25)	Compare to cases 1-3 to consider impact of transition type
5	MOX(PN)	MOX(PN)	Early transition to PuNp MOX fuel	Compare to case 2 to consider impact of transition timing
6	MOX(PNA)	MOX(PNA)	Early transition to PuNpAm MOX fuel	Compare to 5 to consider impact of Am Compare to 3 to consider impact of timing
7	IMF(PNA)	IMP(PNA)	Early transition to PuNpAm IMF fuel	Compare to 6 to consider impact of fuel matrix
8	IMP(PNAC)	IMF(PNAC)	Early transition to PuNpAmCm IMF fuel	Compare to 7 to consider impact of Cm in fuel cycle
9	MOX(PN)	FR Burner	2-Tier transition to fast burner reactor (CR=0.25)	Compare to 4 to consider impact of transition timing and type Compare to 5 to consider impact of fast reactors
10	FR Burner	FR Burner	Early transition to fast burner reactors (CR=0.25)	Compare to 4 to consider impact of transition timing Compare to 5-8 to consider impact of transition type

The results reported here represent the state of the system at the end of the simulation, in the year 2100. All results are normalized to the standard once through fuel cycle scenario, Case #1. The standard value of comparison is the ELF. The mass loading and ELF values from these analyses are more useful in a comparative sense across different scenarios than in an absolute sense. Nevertheless, it is illustrative to note the mass loading and ELF values for Case #1 (UOX-UOX) as a reference point. This scenario places 482 kt of spent fuel in 441 km of repository drift tunnel, with an ELF of 49 $\text{GWd}_{\text{th}}/\text{m}$. As such, any improvement in ELF that is greater than about 8 indicates that the total repository space required within this century is less than a single Yucca Mountain equivalent.

MOX Transitions

Case #2 represents the least aggressive change in the fuel cycle, with a late transition to a modest MOX fuel cycle, recycling only Pu and Np. The maximum improvement in ELF is only 1.55. Case #5 accelerates that transition by 15 years with almost no improvement in the maximum ELF (1.61). The impact of both actinide separation efficiency and Cs/Sr loss fraction is small in both cases. (Figure 1)

When Am is also recycled in these MOX scenarios, the ELF for a late transition (Case #3) shows another small improvement (1.77) while the ELF for an early transition (Case #6) shows a substantial improvement (8.72). There is little benefit to the ELF realized by reducing the separations efficiency above 0.99 or reducing the Cs/Sr loss fraction below 0.01. (Figure 2)

Both late transition cases result in more spent fuel being placed in the repository before recycling begins and thus the repository still contains 43 kt on spent fuel that is yet to be retrieved for use in the advanced fuel cycle. Because the improvement realized in the MOX(PN) fuel cycle is so small, the impact of this once-through spent fuel in the repository is not seen in the late transition to MOX(PN) (Case #2), but penalizes the late transition to MOX(PNA) (Case #3).

Figure 1. Improvement in ELF for Case #5.

Figure 2. Improvement in ELF for Case #6.

IMF vs. MOX

Case #7 can be compared directly to Case #6, both representing early transitions to thermal recycling of Pu, Np and Am. Based on the IMF fuel recipes included in VISION, the MOX improves the ELF (8.72) more than IMF (4.00) when only recycling these three actinides. The impact of also recycling Cm (Case #8) is substantial however, increasing the maximum improvement in the ELF to 41.6. The benefit from increasing the actinide separation efficiency to 0.999 is greater than that realized by reducing the Cs/Sr loss fraction below 0.01. (Figure 3)

Fast Reactors

Three of the scenarios included fast burner reactor deployment. For a late transition directly to fast reactors (Case #4), the maximum improvement in ELF is 5.35, again penalized for the 43 kt remaining once-through spent fuel in the repository. By accelerating that transition (Case #10), the maximum improvement in ELF is increased to 351, the best improvement of all the scenarios examined. A two-tiered transition to fast reactors, including MOX(PN) in the first phase (Case #9) realizes much of this benefit, however, with a maximum improvement in the ELF of 290.

With an actinide separation efficiency of 99% and Cs/Sr loss fraction of 1%, Case #10 will already realize an improvement in the ELF of 44.3, and Case #9 will realize an improvement in the ELF of 34.1. (Figure 4 & Figure 5)

Figure 3. Improvement in ELF for Case #8.

Figure 4. Improvement in ELF for Case #9.

Figure 5. Improvement in ELF for Case #10

Table III. Maximum & Final Repository Space Requirements [km] Under Different Fuel Cycle Scenarios

Case #	$e=0.99, l_{CsSr}=0.01$		$e=0.999, l_{CsSr}=0.001$	
	Max Length	Final Length	Max Length	Final Length
1	441	441	441	441
2	245	245	244	244
3	292	219	219	219
4	126	80	126	75
5	222	222	221	221
6	233	43	80	42
7	106	103	106	106
8	79	15	79	10
9	78	11	78	1.3
10	72	8.8	72	1.1

Maximum Repository Capacity

For many of the scenarios, once-through spent fuel is placed in a repository prior to being retrieved for use in the advanced fuel cycle. In some cases, the maximum repository space required during the middle of the century is much larger than the space required at the end of the century, as shown in Table III. In Case #6, the maximum repository length is also a strong function of the separation efficiency and/or Cs/Sr loss fraction. In all cases, the maximum repository length is greater than the space estimated for 63 kt of spent fuel in Yucca Mountain, according to this model.

Based only on the thermal loading limits for the 10 cases considered here, a number of fuel cycle choices can realize substantial repository capacity benefits (ELF improvement > 8), even with modest separation efficiencies. In any scenario that includes an early transition to an advanced fuel cycle and the recycling of Am in the fuel cycle, a factor of 8 improvement in the ELF is realized with an actinide separation efficiency of 0.99 and a Cs/Sr loss fraction of 0.01. In all cases that include Cm recycling, there is additional benefit realized by improving the actinide separation efficiency. However, given other limitations on repository loading (e.g. dose limits, stability of waste forms, etc.) it is not clear that this additional loading benefit can be realized.

Economic Impact of Repository Benefit

VISION version 2.0.6 included the economic sub-model VISION.econ [13] as well as a more detailed separations scheme. The new separations calculation included more streams from reprocessing and more accurately defined a final resting place for these streams. The separations module tracks the isotopes and can define where the isotopes exit to, based on an efficiency of each isotope entering each waste stream. The calculation of the length used by the waste streams entering a repository was to be updated to reflect the new separation scheme. The calculation had to include the isotopes entering the repository as LSF, SF, or HLW. This newly updated repository length calculation was tested versus the previous version (1.5) with equivalent input regarding the waste streams entering the repository as well as the separation efficiency.

VISION.econ was analyzed to identify key areas of interest to be able to change from a mass based waste fee to a length based fee. VISION.econ breaks down the cost of electricity into modules based on the cost basis report (CBR), *Advanced Fuel Cycle Cost Basis* [12]. The cost is determined by randomly sampling a price range giving a probability distribution of cost for each module and separated into a front end, back end, and separations cost. Table IV lists the economics modules and describes which material flows provide the basis for calculating the costs at that module.

The most important issue identified is that the VISION.econ calculation of separation costs is based on the mass entering separations in the main VISION model. This does not account for the actual mass/length sent to the different waste streams based on the separations model in the VISION calculation. In particular, the geologic repository costs in VISION.econ are based on the current policy and derive a price for disposal based on the mass of the SNF or HLW (\$/kg). They are unaffected by increasing energy produced from a given amount of fuel, separations efficiencies, or by separating the fuel into streams for more efficient disposal. The length based fee will account for increasing burnup, changing separation efficiencies, and more efficient disposal options which will affect the capacity of the repository.

As an initial step to implementing the length based cost, the flow into and out of the repository expressed as length was connected to VISION.econ. This will be used to determine a cost of disposal based on length the spent fuel or HLW used in the repository instead of mass of the waste. This connection will allow for a VISION simulation to simultaneously calculate and compare the cost of a fuel cycle based on the mass of the waste and the length used by the waste in a repository.

Table IV. Breakdown of cost in VISION.econ, with notes on areas of interest which may need to be addressed to accurately implement length based costing scheme.

Module	Description	Material Flows	Notes
Before separations			
A	Mining and Milling	U ore consumption	
B	Conversion	U conversion rate	
C	Enrichment	SWU per year	
D	Fuel Fabrication	Fabricated fuel (mass)	
E2	Dry Storage	SF to reactor dry storage (mass)	
L1	Geologic Repository	All fuel discharged from dry storage before separations begins (mass)	SF sent to repository and later removed currently receives no benefit
K1	Disposition of depleted U	U tails from enrichment (mass)	
I	Monitored Retrievable Storage	Fuel not able to enter the repository nor reprocessing (mass)	
Separations			
F1 & F2	Aqueous and Pyro separations	All isotopes entering separations (mass)	All calculations derived from separations are based on mass of reprocessed fuel entering separations not on actual mass/length sent to waste streams
Streams from separations before final storage			
G1-G4 HLW FP	Conditioning and packaging of HLW	All isotopes entering separations except U, Np, Pu, Am, and Cm (mass)	
G3	Conditioning and packaging of LLW	Waste stream from separations considered LLW-ABC (mass)	Conditioning counted twice (G1-G4 HLW FP)
G1-G4 Cladding	Conditioning and packaging	Flow into separations multiplied a parameter 'cladding percentage of waste' (mass)	
Storage of separations streams			
J	Near Surface Disposal	LLW-ABC (currently ^3H and Kr) (mass)	
K2 and K3	Burned U disposition	U entering separations (mass)	
E3	Actinide Storage	Actinides ^{237}Np - ^{250}Cm entering separations (mass)	
L2	Geologic Repository HLW	All isotopes w/o final disposal location (mass)	

The currently legislated waste fee of \$1/MWh_e (1 mill/kWh_e) was used to as the reference repository cost and converted to length based costs based on both the energy loading factor and the mass loading factor of the repository for standard UOX spent nuclear fuel. Since the current waste fee has been deemed adequate for the current once through plan, it makes a useful reference.

The length based fee can be calculated as

$$LF = 24 * B_f * MLF * \eta_{th} * WF = 0.458 \frac{\$M}{m}, \quad (8)$$

based on the Mass Loading Factor (*MLF*) of current UOX SNF (1.1 MTIHM/m), thermal efficiency, η_{th} , (GW_d_e/GW_d_{th}), assumed uniform burnup, *B_f*, (51 GWd/MTIHM), and nominal energy-based waste fee, *WF*, (1 mill/kWh_e). A similar calculation can be performed based on the energy loading factor. The total repository costs for a once through UOX fuel cycle were calculated using three methods: a cost based on the energy-based waste fee, a cost proportional to the mass of spent fuel, and a cost proportional to the repository space consumed. All three methods gave answers within 1% of the mean. A similar verification was performed for high-level waste, this time comparing a method proportional to the mass of spent fuel entering the separations plant to a method proportional to the quantity of repository space used by the HLW. The Cost Basis Report prepared by Idaho National Laboratory quotes a unit cost for the mass-based method that is based on a series of assumptions that are equivalent to a mass loading factor of 2.5. The mass- and length-based results found here are within 4% for waste streams with a mass loading factor near 2.5.

This capability now allows users of VISION to translate the technical repository, benefit based on heat load calculations, to an economic benefit. In particular, the financial savings in disposal costs can be compared for different fuel recycling schemes. Table V shows that a fuel cycle with only Pu and Np separation actually costs more using the length-based method than the mass-based method. This is because the mass loading factor for this scheme is less than 2.5, achieving less repository benefit than assumed in the Cost Basis Report.

Table V. Mass and length-based calculation of total HLW repository cost from 2000 to 2100 with varying separation efficiencies of Pu and Np.

TRU Separation Efficiency	Repository Cost (\$B)		
	<u>Length</u>	<u>Mass</u>	<u>% Difference</u>
90.00	120.7	93.4	29.23%
99.00	108.1	93.2	15.99%
99.90	107.1	93.2	14.91%
99.99	106.8	93.1	14.72%
No separation (UOX)	242.7	242.6	-

On the other hand, Table VI shows that if Pu, Np, Am and Cm are all removed from the waste, about 25% of the disposal costs can be avoided if the overall separations efficiency is 99% or higher.

Table VI. Mass and length-based calculation of total HLW repository cost from 2000 to 2100 with varying separation efficiencies of Pu, Np, Am, and Cm.

TRU Separation Efficiency	Repository Cost (\$B)		
	<u>Length</u>	<u>Mass</u>	<u>% Difference</u>
90.00	102.7	93.4	9.96%
99.00	71.1	91.9	-22.63%
99.90	68.9	92.0	-25.11%
99.99	68.4	91.5	-25.25%
No separation (UOX)	242.7	242.6	-

Finally, Table VII shows that when Cs & Sr are removed in addition to 99.9% of Pu, Np and Am, the savings can be nearly 70%.

Table VII. Comparison of total repository cost calculated by VISION for mass and length-based approaches with separation of Pu, Np and Am at 99.9% efficiency and varying separation efficiency of Cs and Sr. Total repository cost for the no separation is included for comparison.

Cs/Sr Separation Efficiency	Repository Cost (\$B)		
	<u>Length</u>	<u>Mass</u>	<u>% Difference</u>
0.00	92.9	91.6	1.42%
90.00	34.1	91.7	-62.81%
99.00	28.9	91.7	-68.48%
99.90	28.2	91.5	-69.18%
99.99	27.9	92.2	-69.74%
No separation (UOX)	242.7	242.6	-

This economic measure of the repository benefit can be incorporated into the costs of the whole nuclear fuel cycle to examine the extent to which the repository benefit offsets the other fuel cycle costs. While effect is small for the nominal nuclear waste disposal fee of \$1/MWh_e, the magnitude of this economic repository benefit is linearly related to the nuclear waste disposal fee.

Figure 6 shows the estimated levelized cost of electricity for a MOX Pu-Np-Am fuel cycle, showing the effect of different methods for assessing the repository costs as the nuclear waste fee changes. Since the mass loading factor for this fuel cycle is approximately 2.5, both the mass-based and length-based methods have the same response to a changing waste fee, and the MOX

fuel cycle becomes less expensive if the waste fee is approximately 2.5 mills/kWh_e. Since the mass loading factor is much higher when Cs and Sr are removed, the length based economic measure shows this benefit and this fuel cycle is less expensive than the once-through UOX cycle with a waste fee of approximately 1.75 mills/kWh_e.

Figure 7 shows a similar comparison for a full actinide fast recycle fuel cycle. Since the mass loading factor is larger than 2.5, the mass-based method shows a larger response to the nuclear waste fee than the length-based method. However, the impact of this methodology on the economic cross-over point is small, requiring a waste fee between 8.5 and 9 mills/kWh_e. When Cs and Sr removal is included, the cross-over point is reduced to about 6 mills/kWh_e.

Figure 6. The levelized cost of electricity for a UOX direct disposal fuel cycle versus a MOX Pu-Np-Am thermal recycle, using different methods to assess the repository costs.

It is important to note that the bulk of the fuel cycle costs are calculated using the standard methodology in VISION, based in turn on the Cost Basis Report. As such, none of these costs vary parametrically with the recycling scheme or with the waste fee. A more realistic model might need to account for such variations. Thus, the quantitative results given here are only indicative of the effect that would be seen in a more rigorous analysis.

Figure 7. The levelized cost of electricity for a UOX direct disposal fuel cycle versus a fast reactor actinide recycle, using different methods to assess the repository costs.

The final analysis considers a variable nuclear waste fee that escalates as space in the repository is consumed. A simple functional form was chosen to linearly increase the length-based fee after the first 50 km of tunnel space (enough to emplace 55,000 metric tonnes of once through UOX). This analysis demonstrates the expected result that the back end costs for the fast reactor fuel cycle (85 km) is least affected by the increasing fee, with only 85 km of tunnel required through 2100, than either the MOX PNA (156 km) or UOX (363 km) fuel cycles. While the once through fuel cycle sees an increase from 1.32 mills/kWh_e to 4.89 mills/kWh_e, the fast reactor fuel cycle changes from 0.31 mills/kWh_e to 0.35 mills/kWh_e. Although the functional form of this particular escalating waste fee is not representative of a real economic response, this indicates the impact of an escalating waste fee on the relative costs of different fuel cycle schemes.

Related Efforts

VISION is also being used to develop code-comparison benchmark problems for use in the development of the GENIUS fuel cycle simulation tool. This new tool is focused on modeling discrete facilities and discrete flows of materials in multi-region international simulations. For testing purposes, we are able to construct GENIUS scenarios that mimic the single region continuous flow, facility fleet models that are inherent in VISION. Results from simulations mimicking basic VISION once through fuel cycles have yielded agreement with GENIUS mass flows. This achievement will allow for the further pursuit of comparison benchmarking of the GENIUS tool with more intricate VISION base cases.

Comparison of mass flows in simulations involving multi-region scenarios with a combination of light and heavy water reactors as well as run-time trade affinity constraints have produced results in agreement with equivalent combinations of VISION base cases (More results can be found in the upcoming Global 2009 paper, Oliver and Wilson). While the exploration of run time constraints on the GENIUS repository is ongoing, the demonstrated capability for GENIUS to incorporate these dynamic trade affinity constraints is integral to development of a repository constraint paradigm. Furthermore, the flexibility of the GENIUS code will allow analysis of multiple repository scenarios, so a near term benchmarking goal for GENIUS may include modeling such scenarios.

Finally, a decay module has been incorporated into GENIUS that models decay of all material flows of interest dynamically during run time. This will further improve the potential for GENIUS to advance toward repository-constrained optimization scenarios as well as a higher fidelity analysis of the isotopics of the fuel cycle.

Conclusions

The work did provide evidence for the hypothesis that an increase in the nuclear waste fee would ultimately create an economic incentive to pursue advanced nuclear fuel cycles purely due to the savings on the ultimate disposal costs. In particular, there are indications that only modest increases are necessary for some fuel cycles to recover the increased costs of those fuel cycles from the savings in HLW disposal costs.

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