Proposed Waste Management Metrics for the 2013 Evaluation and Screening of Fuel Cycle Options

Fuel Cycle Research & Development

Prepared for
U.S. Department of Energy
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Appendix E
FCT Document Cover Sheet

Name/Title of Deliverable/Milestone	Fiscal Year 2011 Perspectives in Nuclear Waste Management – Argonne National Laboratory Activities / M3FT-12AN0814017
Work Package Title and Number	Perspectives in Nuclear Waste Management / FT-12AN081401
Work Package WBS Number	**1.02.08.14**
Responsible Work Package Manager	W. Mark Nutt (electronic signature) (Name/Signature)
Date Submitted	

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This deliverable was prepared in accordance with Argonne National Laboratory

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QA program which meets the requirements of

☒ DOE Order 414.1 ☐ NQA-1-2000

This Deliverable was subjected to:

☒ Technical Review	☐ Peer Review
Technical Review (TR)	**Peer Review (PR)**
Review Documentation Provided	**Review Documentation Provided**
☐ Signed TR Report or,	☐ Signed PR Report or,
☒ Signed TR Concurrence Sheet or,	☐ Signed PR Concurrence Sheet or,
☐ Signature of TR Reviewer(s) below	☐ Signature of PR Reviewer(s) below

Name and Signature of Reviewers

S.D. Sevougian (see e-mail concurrence attached following FCT Document Coversheet)

*Note: In some cases there may be a milestone where an item is being fabricated, maintenance is being performed on a facility, or a document is being issued through a formal document control process where it specifically calls out a formal review of the document. In these cases, documentation (e.g., inspection report, maintenance request, work planning package documentation or the documented review of the issued document through the document control process) of the completion of the activity along with the Document Cover Sheet is sufficient to demonstrate achieving the milestone. QRL for such milestones may be also be marked N/A in the work package provided the work package clearly specifies the requirement to use the Document Cover Sheet and provide supporting documentation.

Appendix E
FCT Document Cover Sheet

Name/Title of Deliverable/Milestone	M4FT-12PN0305141-Issue draft set of waste management performance metrics
Work Package Title and Number	Systems / Knowledge Management Support - PNNL
Work Package WBS Number	**1.02.03.05**
Responsible Work Package Manager	J.D. Vienna (electronic signature) (Name/Signature)
Date Submitted	

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This deliverable was prepared in accordance with Pacific Northwest National Laboratory

(Participant/National Laboratory Name)

QA program which meets the requirements of

☒ DOE Order 414.1 ☐ NQA-1-2000

This Deliverable was subjected to:

☒ Technical Review

☐ Peer Review

Technical Review (TR)

Review Documentation Provided

☐ Signed TR Report or,

☒ Signed TR Concurrence Sheet or,

☐ Signature of TR Reviewer(s) below

Name and Signature of Reviewers

S.D. Sevougian (see e-mail concurrence attached following FCT Document Coversheet)

Peer Review (PR)

Review Documentation Provided

☐ Signed PR Report or,

☐ Signed PR Concurrence Sheet or,

☐ Signature of PR Reviewer(s) below

*Note: In some cases there may be a milestone where an item is being fabricated, maintenance is being performed on a facility, or a document is being issued through a formal document control process where it specifically calls out a formal review of the document. In these cases, documentation (e.g., inspection report, maintenance request, work planning package documentation or the documented review of the issued document through the document control process) of the completion of the activity along with the Document Cover Sheet is sufficient to demonstrate achieving the milestone. QRL for such milestones may be also be marked N/A in the work package provided the work package clearly specifies the requirement to use the Document Cover Sheet and provide supporting documentation.

From: "Sevougian, S David" <sdsevou@sandia.gov>
Date: January 31, 2012 6:29:17 PM CST
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Subject: **RE: [EXTERNAL] RE: Waste Management Metrics**

Mark,

Based on your indicated final changes to the document, I have completed my independent technical review of Proposed Waste Management Metrics for the 2013 Evaluation and Screening of Fuel Cycle Options, FCRD-UFD-2012-000061, REV 0 and concur with the report.

Please cc me on the transmittal e-mail for this deliverable.

Thank you,
Dave

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CONTENTS

PROPOSED WASTE MANAGEMENT METRICS FOR THE FY13 FUEL CYCLE OPTIONS EVALUATION

1. INTRODUCTION

The Fuel Cycle Options (FCO) Campaign is preparing for the evaluation and screening of fuel cycles in 2013 to identify those that would significantly benefit the sustainable use of nuclear power in the U.S. The FCO Campaign requested support from the FCT campaigns to contribute to the development of the evaluation and screening process, especially in areas where reviewers suggested improvements to the pilot demonstration of the process in FY11. The fuel cycles will be evaluated with respect to a number of criteria, such as nuclear waste management and resources.

A set of performance metrics that address these criteria is planned to be completed during FY12. To support this development, early in FY12, each campaign was tasked with developing appropriate performance metrics; explanation of the applicability and relevance of each metric; and the performance basis for evaluation. Subsequently in FY12, discussions and iterations with the FCO Campaign will result in the consensus set of metrics that will be used for the 2013 screening.

This report provides an initial draft set of waste management metrics for use in the evaluation and screening of fuel cycle options that will be conducted in 2013. This report was developed jointly by the Used Fuel Disposition (UFD) and Separations and Waste Forms (SWF) Campaigns.

2. Evaluation Criteria

Nine high-level criteria have been identified for Fuel Cycle Option screening [Ref. 1]. Nuclear Waste Management is one part of every nuclear fuel cycle and encompasses the safe and economic deployment of storage, waste treatment, waste packaging, transportation, and disposal systems. All of the other criteria listed below, with the exception of resource utilization apply to Nuclear Waste Management. Therefore, the proposed metrics for the Nuclear Waste Management criterion discussed herein coincide with four of the fuel cycle criteria (***italicized*** below): Safety, Financial Risk and Economics, Environmental Impact, and Development and Deployment Risk. Further discussion on each of these criteria, specific metrics, and inputs required to quantify the metrics, are provided in Section 3.

- Nuclear Waste Management
- Proliferation Risk
- Nuclear Material Security Risk
- ***Safety***
- Financial Risk and ***Economics***
- ***Environmental Impact***
- Resource Utilization
- ***Development and Deployment Risk*** (including technical maturity, development time, cost, and licensing)
- Institutional Issues

Institutional Issues, which include public and stakeholder acceptance, are also important for siting, licensing, and development of nuclear waste management facilities. Institutional Issues are not limited only to nuclear waste management, but apply to the entire nuclear fuel cycle. Stakeholder acceptance is inherently subjective and difficult to quantify. Accordingly, an alternative approach is often used in studies of this type, whereby stakeholder input to the study is elicited and factored into the assessment of other metrics [Ref. 2]. Such an approach could be considered in the 2013 screening of fuel/evaluation cycle options and the UFD Campaign has experience in such elicitations and could provide support to the FCO Campaign in conducting such evaluations. However, no specific metrics for stakeholder acceptance are proposed at this time for nuclear waste management.

The Resources criterion would not be discriminating for nuclear waste management because the resource utilization advantages are realized in other, upstream parts of the fuel cycle. The only potential area where waste management metrics may be warranted relates to the management and disposal of uranium mill tailings and depleted uranium (DU)/recycled uranium (RU) if they become wastes in particular fuel cycles. Quantifying such a metric for uranium mill tailings is within the scope of the Fuel Resources activities within the SWF Campaign and no attempt to quantify such a metric is provided herein. The quantities of DU and RU that would be generated and the decision as to whether DU and RU are deemed as waste for the different fuel cycle options is the scope of the FCO campaign. However, safety metrics are provided herein should these materials be deemed as waste for a given fuel cycle option.

For the Proliferation Risk criterion, the discriminators are related to the availability and ease of access to fissile materials including their separation from highly radioactive fission products. These aspects are predominantly determined upstream from Nuclear Waste Management. The Security of nuclear materials is critically important, except that it is accomplished by applying security measures that ensure a comparable level of risk from theft or sabotage for all potentially viable fuel cycles. Hence, security would be non-discriminating with respect to nuclear waste management, except possibly on cost, and the cost of security measures in waste management would be incidental to other waste management costs, and therefore non-discriminating. An exception may be for fuel cycle options that require extended periods of storage when more significant costs associated with security could be incurred (considered here as a cost impact) or if the UNF storage option is inherently prone to diversion of materials (which are believed to be excluded from fuel cycle options being considered as “no-go” decisions).

The technical maturity of concepts, facilities, and systems for managing wastes from future fuel cycles is an important aspect of the 2013 screening/evaluation of fuel cycle options. A discussion of the technical maturity of different concepts and technologies associated with nuclear waste management is provided in Section 4.

2.1 Safety, Economics, and Environmental Impact Criterion

This section summarizes the overall rationale behind the proposed metrics for the Nuclear Waste Management criterion that coincide with three additional fuel cycle criteria: Safety, Economics, and Environmental Impact.

2.1.1 Safety Criterion

Compliance with applicable regulations is demonstrated through safety analyses. These safety analyses are specific to the design of the facility, the hazard and characteristics of the material being handled, and the site where the facilities would be located. Detailed models, design data, site properties/characteristics would be used to determine worker and public health effects (radiological and non-radiological) associated with operating the nuclear facility. This is common for all nuclear facilities and systems, not just those associated with waste management.

The safety of disposing nuclear waste must also be demonstrated after the waste is emplaced and the disposal facility is closed. This is typically termed as post-closure safety analysis and is used to demonstrate that the risk to the public will remain below regulatory limits. As with demonstrating "operational" safety, post-closure safety analyses are specific to the design of the disposal facility, the hazard and characteristics of the waste being disposed, and the characteristics of the site where it is located. Detailed models, design data, site properties/characteristics would be used to determine public health effects associated with disposing of nuclear waste.

There is a fundamental difference between operational and post-closure safety regarding when risks to the public and workers occurs. A properly sited and designed disposal facility will effectively isolate wastes from the environment for a long period of time and risk to the public would be essentially negligible. Protection is ensured through the use of both natural and engineered barriers that work together to contain the emplaced waste and to limit the release of radionuclides should any of the barriers fail. Only when the engineered barriers fail is there any potential risk to the public and this is not expected to occur until well in the future: hundreds to thousands of years for low-level radioactive waste (LLW) disposal facilities, and tens of thousands of years and longer for high-level radioactive waste (HLW) disposal facilities. This differs from operational safety, which represent more near-term risks, during the operational period of the facility: on the order of a few to several decades.

It is not currently possible to conduct detailed operational and post-closure safety analyses for each of the fuel cycle options under consideration for the following reasons:

- Only limited information is available regarding the characteristics of the material that would be handled in the back end of the fuel cycle options under consideration and the wastes that would be disposed of.

- Storage and transportation system designs are well developed for used nuclear fuel (UNF) from light water reactors (LWRs) and borosilicate glass HLW forms, but none have been developed for other UNF types and HLW forms.

- Only generic facility design concepts in various geologic media for the disposal of UNF and HLW are being developed and evaluated in the UFD Campaign. Generic facility design concept development efforts are not currently underway in the UFD Campaign for the disposal of LLW (Class, A, B, C, and Greater than Class C (GTCC) or other materials that may be deemed waste (i.e., DU and RU).

- System-level models have been developed by the UFD Campaign to evaluate the long-term performance of generic geologic disposal systems. However, their output cannot be used as an absolute measure of risk associated with geologic disposal due to the generic nature of the systems they are representing.

- System-level models are not being developed by the UFD Campaign to evaluate other disposal concepts (i.e., the near-surface disposal of LLW or the greater confinement disposal of GTCC).

While it is not possible to conduct operational and post-closure safety analyses to support the 2013 evaluation and screening of fuel cycle options, it is possible to quantify "indicators" of safety that can serve as nuclear waste management metrics. In general, there are a few key parameters that affect operational and post-closure safety. Specific parameters themselves or combinations of them can serve as safety indicators and are proposed as waste management metrics. These are discussed in detail below.

2.1.2 Economics Criterion

Safety and economics, as it pertains to nuclear waste management, are tightly linked, primarily through the hazard of the material that must be treated, packaged, stored, transported, and disposed of. A fundamental tenet is that all of these steps will be done safely, in accordance with applicable regulations, or they would not be allowed. However, there is a cost associated with achieving the required level of safety, given the hazard of the material being handled. This can easily be seen in the differences in technologies used and the cost of disposing of LLW and HLW.

While the hazard-cost linkage is clear, there are other clear cost drivers associated with the management of a given classification of waste. Two of the key drivers are the quantity of material that must be managed (i.e., volume or mass per GWe-yr of nuclear generation) and the rate at which material must be processed through the facility. For example, in the case of a disposal facility, the volume of material that must be disposed has a direct influence on the size of the disposal facility, influencing the disposal space that would need to be excavated and the amount of material that would be needed to construct the facility. In addition, the rate that waste would be emplaced would also impact the size and/or number of the operational facilities; a higher processing rate could require larger facilities and/or additional facilities to achieve the needed throughput.

The management of UNF and HLW is also affected by the thermal output of the waste. The design complexity and size of storage and disposal systems and facilities are directly affected by the thermal output of the material being managed. Storage canister size, waste package size, storage configuration, the use of passive or active ventilation, storage duration, etc. are all influenced by the thermal output of the waste. The size and loading of storage systems, in particular dry storage systems, are affected by the thermal output of the waste and have thermal limits imposed on them.

All geologic disposal systems have thermal limits imposed to: 1) maintain the isolation properties of the engineered and/or natural barriers, 2) limit the effects of coupled thermal-hydrologic-chemical-mechanical processes and the need to model them over long periods of time, and 3) preclude coupled effects that could hinder emplacement operations (e.g., accelerated salt creep). The combination of waste thermal output and the media-specific thermal limits affects the placement of waste in a geologic repository and has a direct impact on the size of a repository and the amount of rock that must be excavated. Cooler waste can typically be emplaced closer together, decreasing the size of a repository as compared to disposing of higher heat generating waste.

The waste thermal output may also result in a need for decay storage for a period of time prior to emplacement to allow the thermal output to decrease sufficiently prior to emplacement. There is a cost trade-off associated with intermediate-term decay storage of wastes and repository disposal efficiency. In addition to decay storage, the thermal output of a waste form can be reduced by decreasing the loading of heat-generating isotopes within a waste form. This could be achieved for a given waste stream by simply reducing the loading density of the waste form. However, this would increase the volume of waste that would need to be processed, stored, transported, and ultimately be disposed of. A second approach, which is fuel cycle dependent, is to reduce the inventory of heat generating isotopes through management within the fuel cycle. For example, heat generating fission products (Cs, Sr) could be separated to allow for decay storage and transuranic isotopes (Pu, Am, Cm) could be recovered for subsequent transmutation.

In any case, thermal management is primarily an exercise in engineering to optimize the waste management system within thermal constraints. While there are safety-related implications, they are managed through the thermal constraints. Thus, thermal management and thermal effects should be considered as affecting economic-related metrics.

It is not possible to conduct detailed engineering or optimization analyses for each of the fuel cycle options being considered in the 2013 evaluation and screening of fuel cycle options for the same reasons as discussed above. However, it is possible to quantify the key cost drivers and develop rough order-of-magnitude estimates of waste management costs as a function of these drivers. This is discussed further below.

2.1.3 Environmental Impact Criterion

The environmental impact associated with any nuclear facility is assessed through a detailed environmental impact analysis or environmental assessment. Items evaluated include impact to environmental aesthetics, impact to cultural resources, impact to biological resources, worker occupational health risk, and public environmental risk. The last two items are safety-related criterion and are discussed above (although they may be treated differently and more broadly in an environmental assessment than in a compliance assessment).

It is not possible to conduct detailed environmental assessments for each of the fuel cycle options being considered in the 2013 evaluation and screening of fuel cycle options for the same reasons as discussed above. However, the key drivers that would likely influence an environmental assessment are the same as those discussed in Sections 2.1.1 and 2.1.2 and relate to the hazard and quantity of the material being managed, the size of the facilities, the transportation of fuel and wastes, and the duration that they are in operation.

In addition, it is expected that the environmental impacts associated with waste management are likely to be smaller than other aspects of the fuel cycle, in particular resource recovery. The size (footprint) of other fuel cycle facilities is also likely to be significantly larger than waste management facilities and would have a larger influence on the environment. Potentially the largest environmental impact of a nuclear fuel cycle is related to the rate at which fossil fuel consumption can be off-set by generation of electricity using nuclear power. Those cycles that are capable of generating more power will inherently be more environmentally effective. In comparison, waste management environmental impacts are likely to be relatively insignificant in comparison.

As such, it is recommended that environmental management metrics not be explicitly considered within the Nuclear Waste Management criterion. An exception to this is the wastes generated from fuel resource recovery and processing (i.e., uranium mill tailings). Appropriate metrics should be developed as part of the Environmental Impact criterion.

2.2 Linkages with Other Aspects of the 2013 Evaluation and Screening of Fuel Cycle Options

Transportation is a key component of the waste management system, but is also a fundamental aspect of an entire fuel cycle. This report does not contain any metrics related to the transportation of nuclear materials. Rather, it is recommended that a single set of metrics be developed that covers transportation for all aspects of the fuel cycle, including the nuclear waste management component. Factors that may have the most influence on transportation include 1) the number of transportation steps involved, 2) the hazard of the material being transported at each step, and 3) the number of shipments needed at each step. The UFD Campaign has transportation-related expertise and is available to support the FCO Campaign in the development of transportation-related metrics.

Safety related metrics are proposed in this report for radioactive material storage, disposal operations, and post-closure performance of disposal systems. Economic metrics for radioactive material storage and disposal are also proposed. Similar metrics are needed for all aspects of the nuclear fuel cycle in order to make meaningful comparisons between fuel cycle options. As an example, fuel cycle options that improve safety and economics associated with waste management may have trade-offs with respect to safety and economics in other aspects of the fuel cycle. As an example, recycling facilities used to recover key radionuclides important to waste management for subsequent transmutation should also be evaluated considering safety and economic metrics (and perhaps others) associated with those facilities so as to compare the trade-offs and impacts. Such comparative impacts cannot be explored unless there are similar metrics for other aspects of the nuclear fuel cycles under consideration in the 2013 screening and evaluation of fuel cycle options.

Safety and economic metrics associated with waste treatment and packaging are not provided herein. Waste treatment and packaging would be a component of a separations/recycling facility and, as such, the metrics associated with these processes cannot be established until a specific technology, or set of technologies, is assumed for a given fuel cycle option. A key example of this is the capture and immobilization of volatile isotopes from separations. Accordingly, waste treatment and packaging metrics (safety and economics) should be developed in conjunction with the entire metric set for separations/ recycling.

The waste management metrics proposed in this report are "independent" of the fuel cycle options in that they can be developed without specific assumptions and knowledge regarding the technologies used within a specific fuel cycle option. The metrics recommended are indicators based on the "quantities" (i.e., volumes, isotopic inventories) of waste that must be managed for a given fuel cycle option, or estimated costs based on the "quantities." These "quantities" must be obtained for each fuel cycle option under consideration and once defined for a given fuel cycle option, the waste management metrics can be calculated. The UFD and SWF Campaigns are currently developing these quantities, as discussed in Section 5, and will continue to support the FCO Campaign in estimating these "quantities." However, it is important to note that for a large set of fuel cycle options, the data required for these input "quantities" is extensive and may not be gathered in time for the 2013 evaluation and screening. If that is the case, the "indicator" metrics proposed in the following sections will have to be simplified further and estimated subjectively by a panel of experts, such as the Evaluation and Screening team suggested in the Screening Charter [Ref. 1].

3. Nuclear Waste Management Metrics

This section describes each of the proposed Nuclear Waste Management metrics. The method for calculating the metric and a justification for each metric are provided. Several of the recommended metrics are fully developed herein. However, additional effort is required to fully develop the metrics for use in the 2013 screening and evaluation of fuel cycle options. In such cases, the overall method for calculating the metric is described; however the specific algorithm for computing the metric will be developed during fiscal years 2012 and 2013 and provided to the FCO Campaign.

3.1 Storage and Disposal Operational Safety

As discussed above, while it is not possible to conduct operational safety analyses to support the 2013 evaluation and screening of fuel cycle options, it is possible to quantify "indicators" of safety that can serve as nuclear waste management metrics. In general, there are a few key parameters that affect operational safety. Specific combinations of these parameters serve as safety indicators and are proposed as waste management metrics.

Storage Operations Safety Indicator

Storage operations includes loading the UNF or radioactive waste (LLW, GTCC, HLW) into dry storage canisters, the draining, drying, and sealing of the canister, transport to the dry storage facility, placement of the canisters in the dry storage facility, monitoring, remedial activities (as necessary, including re-packaging), and removal of the canister from the dry storage facility for subsequent transport. The operational safety risk triplet is the hazard of the materials being handled, the probability of an accident occurring at each of the steps summarized above, and the consequences of an accident. The safety indicator for storage operations, given in Equation 1 below, addresses each of these components. This safety indicator is applicable for storage operations (except transportation) for all classes of radioactive materials (UNF, HLW, LLW, and GTCC).

$$SI_{S-O} = \sum_{i=1}^{N} \log\left(\frac{R_{inh}}{2000\,mrem/yr}\right) \bullet DF_{SF} \bullet C_S \bullet \left[RP + (T_S / 60\,years)\right] \qquad \text{Eq. 1}$$

Where:

	SI_{S-O}	=	Safety indicator for storage operations [GWe-yr^{-1}]
	R_{inh}	=	Inhalation radiotoxicity of material being handled [summed over all important radionuclides in the storage form; mrem /yr]
	DF_{SF}	=	Dispersibility factor of the storage form of the material being handled [1-10; 1 for monolithic solid, 10 for readily dispersible]
	C_S	=	Number of packages/canisters that are loaded and stored [(GWe-yr)$^{-1}$]
	T_S	=	Duration of the storage period (years)
	RP	=	Number of packaging and re-packaging steps (if required) during the storage period (≥ 1, since there will always be at least one packaging step)
	N	=	Number of material categories i being stored for a particular fuel cycle (i.e, different UNF types, HLW forms, etc).

A fuel cycle option may have multiple storage forms or used nuclear fuel that would be stored. Each of these storage forms may have different radiologic hazards, dispersibility factors, numbers of canisters, and storage durations. The storage operations safety indicator therefore considers all storage forms for a given fuel cycle option.

The inhalation radiotoxity is a measure of the hazard of the material being handled. Inhalation radiotoxity is recommended for use because accident scenarios that result in the dispersion of material typically result in larger impacts. Both the concentration of radionuclides in the storage forms and radionuclide dose conversion factors are needed to calculate the radiotoxicity.

This metric could be construed as a measure of long-term risk associated with storage of highly radioactive materials, which it is not. Thus, in order to avoid a comparison of computed radiation exposures, a logarithm is taken for the radiotoxicity. In addition, in order to avoid a comparison of computed radiation exposures, all results are normalized to an annual dose of 2000 mrem. The 2000 mrem/year normalization factor is recommended by the IAEA as the occupational exposure limit for practices involving radioactive materials [Ref. 3, Appendix II-5].

The dispersibility factor addresses the consequences of an accident. Less dispersible solid storage forms will have a lower release fraction under accident scenarios than more dispersible storage forms. A range of 1-10 is assumed with 1 being for robust monolithic solid forms and 10 for readily dispersible forms. The dispersibility factor for the waste form is an input to be determined by the SWF Campaign for the storage forms that would be generated for each fuel cycle option or by the UFD Campaign for used nuclear fuels.

The likelihood of an accident occurring that could potentially result in releases and routine operational exposure throughout all of the dry storage steps summarized above is directly proportional to the number of canisters or packages that are would be stored. The likelihood of an accident occurring is also proportional to any re-packaging that would be required over the storage period (potentially important over very long storage periods). The likelihood of an accident occurring during the storage period is proportional to the time that the material would be stored. Note that the time component in the indicator is normalized to 60 years, corresponding to the recent NRC waste confidence ruling [10 CFR 51.23] that states:

> The commission has made a generic determination that, if necessary, spent fuel generated in any reactor can be stored safely and without significant environmental impacts for at least 60 years beyond the licensed life for operation (which may include the term of a revised or renewed license) of that reactor in a combination of storage in its spent fuel storage basin at either on-site or off-site independent storage installations.

The storage operational safety indicator will be sensitive to the inhalation radiotoxicity, which is a function of the isotopic inventory in the stored material. Different isotopic concentrations in the stored material will have significantly different inhalation radiotoxicity values. Radioactive waste from different fuel cycle options could have significantly different isotopic inventories (i.e. recovery of actinides for subsequent transmutation), potentially resulting in order-of-magnitude differences in the inhalation radiotoxicity. In addition, decay storage can also change the isotopic inventory in a waste form, also causing order-of-magnitude differences in the inhalation radiotoxicity. It is recommended that the inhalation radiotoxicity be determined when the material is placed into storage as that would be when it is most hazardous.

The likelihood of an industrial accident occurring during the construction and operation of the storage facility is also directly proportional the number of canisters that would be loaded and stored. Worker exposures due to routine operations would also be directly proportional to the number of canisters that would be handled and the time period over which the material would be stored. However, routine worker exposure would be limited through administrative controls at a storage facility.

<u>Disposal Operations Safety Indicator</u>

Disposal operations includes waste receipt, re-packaging (if necessary), waste emplacement, facility monitoring, and facility closure activities. The operational safety risk triplet is the hazard of the materials being handled, the probability of an accident occurring, and the consequences of an accident. The safety indicator for waste disposal operations, given below, addresses each of these components. This safety indicator is applicable for disposal operations for all classes of radioactive waste (UNF, HLW, LLW, and GTCC).

$$SI_{D-O} = \sum_{i=1}^{n} \log\left(\frac{R_{inh}}{2000\ mrem/yr}\right) \bullet DF_{WF} \bullet C_w \bullet (RP+1) \qquad \text{Eq. 2}$$

Where: $SI_{D\text{-}O}$ = Safety indicator for disposal operations [$(GWe\text{-}yr)^{-1}$]

R_{inh} = Inhalation radiotoxicity of material being handled [summed over all important radionuclides; mrem / yr]

DF_{WF} = Dispersibility factor of the waste form material being handled [1 for solid; 10 for dispersible]

C_w = Canisters of waste disposed [$(GWe\text{-}yr)^{-1}$]

RP = Number of repackaging steps required per unit handled (note that this is different than in Equation 1 and accounts for the possibility that material could be re-packaged for disposal elsewhere)

n = Number of waste categories i to be disposed of for a particular fuel cycle (i.e, different UNF types, HLW forms, etc)

A fuel cycle option may have multiple waste forms or used nuclear fuel that would be disposed of. Each of these storage forms may have different radiologic hazards, dispersibility factors, numbers of canisters, and re-packaging steps. The disposal operations safety indicator therefore considers all waste forms for a given fuel cycle option.

The inhalation radiotoxity is a measure of the hazard of the material being handled. Inhalation radiotoxity is used because accident scenarios that result in the dispersion of material typically result in larger impacts. Both the concentration of radionuclides in the waste forms and radionuclide dose conversion factors are needed to calculate the radiotoxicity.

This metric could be construed as a measure of long-term risk associated with storage of highly radioactive materials, which it is not. Thus, in order to avoid a comparison of computed radiation exposures, a logarithm is taken for the radiotoxicity. In addition, in order to avoid a comparison of computed radiation exposures, all results are normalized to an annual dose of 2000 mrem. The 2000 mrem/year normalization factor is recommended by the IAEA as the occupational exposure limit for practices involving radioactive materials [Ref. 3, Appendix II-5].

The dispersibility factor addresses the consequences of an accident. Less dispersible solid waste forms will have a lower release fraction under accident scenarios than more dispersible waste forms. A range of 1-10 is assumed with 1 being for robust monolithic solid forms and 10 for readily dispersible forms. The dispersibility factor for the waste form is an input to be determined by the SWF Campaign for the waste forms that would be generated for each fuel cycle option or by the UFD Campaign for used nuclear fuels.

The likelihood of an accident occurring that could potentially result in releases during repository operation is directly proportional to the number of waste packages or canisters handled, and the number of repackaging steps needed prior to emplacement. The likelihood of an industrial accident occurring during the construction and operation of the underground facility is also directly proportional number of canisters/waste packages that would be disposed of.

The disposal operational safety indicator will be sensitive to the inhalation radiotoxicity, which is a function of the isotopic inventory of the waste at the time of disposal. Different classes of radioactive waste (i.e., HLW vs. LLW) will have significantly different inhalation radiotoxicity values. Radioactive waste from different fuel cycle options could have significantly different isotopic inventories (i.e. recovery of actinides for subsequent transmutation), potentially resulting in order-of-magnitude differences in the inhalation radiotoxicity. In addition, decay storage can also change the isotopic inventory in a waste form, also causing order-of-magnitude differences in the inhalation radiotoxicity.

The disposal operational safety indicator will also be sensitive to the number of canisters/waste packages that must be disposed. As above, different numbers of canisters/waste packages for the various classes of radioactive waste (i.e., LLW vs. HLW) will be generated for a given fuel cycle option and these will vary for the different fuel cycle options under consideration.

3.2 Disposal Post-Closure Safety

As discussed above, while it is not possible to conduct post-closure safety analyses to support the 2013 evaluation and screening of fuel cycle options with the resources available, it is possible to quantify "indicators" of safety that can serve as nuclear waste management metrics. In general, there are a few key parameters that affect post-closure safety. Specific parameters themselves or functional relationships based on these parameters serve as safety indicators and are proposed as post-closure safety waste management metrics.

Disposal Post-Closure Safety Indicator – UNF & HLW

The post-closure performance of a geologic disposal facility depends on the performance of all the barriers, engineered and natural. Allocating performance requirements to individual barriers is difficult because the performance of each of the barriers plays a role in the overall performance of the disposal system and the coupled performance of the barriers depends on the disposal system environment and the occurrence of any disruptive events that may impact the integrity of the system. The waste form is one of the engineered barriers and is integral to the design and performance of a geologic disposal facility.

The purpose of a waste form is to immobilize the radionuclides in a form that isolates them from the environment. It must be recognized that the waste form would be only one of the multiple barriers that comprise a disposal system. Additional engineered and natural barriers also serve to isolate radionuclides from the environment; no one single barrier is typically relied on to solely isolate radionuclides over the long periods of time that disposal systems are protective of public health and safety. The entire disposal system itself is protective of public health and safety.

As discussed in Section 2.1.1, it is not currently possible to conduct detailed post-closure safety analyses for each of the fuel cycle options under consideration, for a variety of reasons, including:

- Only limited information is available regarding the characteristics of the material that would be handled in the back end of the fuel cycle options under consideration and the wastes that would be disposed of.
- Only generic facility design concepts in various geologic media (salt, clay/shale, granite, and deep borehole) for the disposal of UNF and HLW are being developed and evaluated in the UFD Campaign [Ref. 4].
- Generic disposal system-level models (GDSMs) have been developed by the UFD Campaign to evaluate the long-term performance of generic geologic disposal systems [Ref. 5]. However, their output cannot be used as an absolute measure of risk associated with geologic disposal due to the generic nature of the systems they are representing.

However, these tools can be used to develop media-specific post-closure safety indicators for the 2013 evaluation and screening of fuel cycle options. The proposed safety indicator for the geologic disposal of UNF and/or HLW is given in the functional relationship below.

$$SI_{GD-PC_G} = \sum_{j=1}^{n} \log\left(\frac{\sum_{i=1}^{N} D_{G,i}(I_i, F_d)}{100\ mrem/yr}\right) \qquad \text{Eq. 3}$$

Where:

Where:	SI_{GD-PCG}	=	Safety indicator for post-closure geologic disposal for media type G [GWe-yr^{-1}]
	$D_{G,i}$	=	Parametric relation that captures disposal system performance for media G for a combination of key radionuclide i inventory and waste form fractional degradation rate (mrem/yr)
	I_i	=	Inventory of key radionuclide i to be disposed of (kg/GWe-yr)
	N	=	Number of key radionuclides important to post-closure safety
	F_d	=	Fractional degradation rate of the waste form [yr^{-1}]
	n	=	Number of waste categories j to be disposed of for a particular fuel cycle (i.e, different UNF types, HLW forms, etc)

The GDSMs [Ref. 5] will be used to develop the $D_{G,i}(I_i, F_d)$ response surfaces for key radionuclides for each disposal media. This will be done by simulating the release of radionuclides from hypothetical waste forms with different combinations of key radionuclide inventory (I_i) and waste form fractional degradation rate (F_d) and the subsequent radionuclide transport through the engineered and natural barriers for the generic disposal environments under consideration. Simulations will be conducted for generic geologic disposal systems in salt, clay/shale, and granite media and for deep borehole disposal in crystalline rock.

The resulting peak annual dose for each key radionuclide will be collected for the different radionuclide inventory / waste form fractional degradation rates to build a numerical "response surface" to represent the functional form shown in Equation 3, the $D_{G,i}(I_i, F_d)$ response surfaces. These response surfaces can then be used to determine $D_{G,i}(I_i, F_d)$ for each radionuclide once the waste form radionuclide inventory and fractional degradation rate is defined and used in Equation 3 to determine the overall disposal post-closure safety indicator.

$D_{G,i}(I_i, F_d)$ response surfaces will be developed for two broad post-closure scenario classes, undisturbed and disturbed repository conditions, for each of the four generic media under consideration. Previous simulation results [Ref. 6] will be used to develop the $D_{G,i}(I_i, F_d)$ response surfaces as requested by the FCO Campaign.

Radiation exposure, or dose, is used as a metric of GDSM performance and is determine using biosphere dose conversion factors developed in the International Atomic Energy Agency's (IAEA) BIOMASS project for a simple drinking water well pathway. However, in order to avoid a comparison of computed radiation exposures, all results are normalized to an annual dose of 100 mrem. The 100 mrem/year normalization factor is recommended by the IAEA as the limit to "relevant critical members of the public" for practices involving radioactive materials [Ref. 3, Appendix II-8].

This approach is identical to that previously used to support the establishment of waste form performance criteria [Ref. 6] and is described in more detail in Appendix A. It inherently captures the isolation capabilities of the natural and engineered barriers that are, for the most part, independent of fuel cycle options and includes the two primary independent variables that would be affected by different fuel cycle options: radionuclide inventory and waste form fractional degradation rate.

Representative designs for geologic disposal systems in salt, clay/shale, and granite media and for deep borehole disposal in crystalline rock will be used to define engineered and natural barrier system conceptualizations within the GDSMs. The current UFD baseline properties of both the natural and engineered barriers will also be used.

The radionuclide inventory – waste form fractional degradation rate response surfaces, $D_{G,i}(I_i, F_d)$ from Equation 3, for generic unsaturated tuff and saturated clay environments for undisturbed and disturbed scenarios cases are provided in Appendix A as examples. Note that research on generic unsaturated tuff environments is not being conducted and therefore there are no GDSMs that can specifically be used to develop a post-closure safety metric. However, as discussed above, similar calculations were performed to support the development of waste form performance criteria. The results of these calculations can be used to develop the post-closure safety metrics for a generic unsaturated tuff environment. Thus, the data provided in Appendix A for unsaturated tuff is considered complete.

Disposal Post-Closure Safety Indicator – LLW (Class A, B, C)

It is expected that Class A, B, and C LLW generated by advanced fuel cycles would be disposed of in "traditional" near surface disposal facilities. As will be shown below, the disposal post-closure safety indicator for LLW is a function of both the volume of the different classes of LLW generated and the radionuclide inventory (concentration) in each LLW waste class. In addition, a variety of LLW waste types may be generated by advanced nuclear fuel cycles (as is the case with the current once-through fuel cycle). Quantifying the different LLW types, their volumes, and their radionuclide inventories for advanced fuel cycles to support the 2013 screening and evaluation of fuel cycle options is expected to be challenging and uncertain. The level of uncertainty in the LLW waste types, estimated volumes, and radionuclide inventories need to be considered, propagated through the post-closure safety indicator for LLW, and factored into the overall screening and evaluation of fuel cycle options.

As discussed above, simulation tools have not been developed to evaluate the post-closure performance of near-surface disposal systems for Class, A, B, C LLW. Efforts to develop such models, initiated in FY2011, are not being pursued in FY2012 and a disposal post-closure safety indicator of the form shown in Equation 3 cannot presently be developed.

In lieu of this, a safety indicator for the disposal of Class, A, B, and C LLW of the form shown in Equation 4 can be used for the disposal of LLW in near-surface disposal facilities. This safety indicator considers the inventory in the different classes of LLW accounts for the mobility of each radionuclide by dividing the concentration by the retardation coefficient (which is a function of the distribution coefficient, the media density, and the media porosity).

The resultant "mobile concentration" of each radionuclide is then converted to a measure of annual radiation exposure (ingestion dose, ICRP 72), then weighted by the volume of each class of LLW. This metric could be construed as a measure of long-term risk associated with LLW disposal, which it is not. Thus, in order to avoid a comparison of computed radiation exposures, a logarithm is taken. In addition, in order to avoid a comparison of computed radiation exposures, all results are normalized to an annual dose of 100 mrem. The 100 mrem/year normalization factor is recommended by the IAEA as the limit to "relevant critical members of the public" for practices involving radioactive materials [Ref. 3, Appendix II-8].

Advanced fuel cycles may also generate mixed hazardous/radioactive LLW. The proposed safety indicator shown in Equation 4 does not consider mixed hazardous/radioactive LLW. While the disposal of these wastes could potentially prove challenging, the overall risk associated with mixed hazardous/ radioactive LLW is likely to be relatively small in comparison to LLW that does not contain mixed hazardous materials due to the relatively lower volume of mixed waste that would be generated. For example, estimates of LLW generation indicate that of the total volume that would be generated from aqueous separations processes, less than one percent of the volume generated would be mixed hazardous/radioactive LLW [Ref. 7]. Therefore, for the purpose of the 2013 screening and evaluation of fuel cycle options, the contribution of mixed hazardous/ radioactive LLW to the overall LLW disposal post-closer safety indicator can be neglected.

$$SI_{LLW-PC} = \log\left[\frac{\sum_{j=1}^{N-A} V_{A_j} \sum_{i=1}^{N}\left(\frac{CA_{i,j}}{R_i} DCF_i\right) + \sum_{j=1}^{N-B} V_{B_j} \sum_{i=1}^{N}\left(\frac{CB_{i,j}}{R_i} DCF_i\right) + \sum_{j=1}^{N-C} V_{C_j} \sum_{i=1}^{N}\left(\frac{CC_{i,j}}{R_i} DCF_i\right)}{100\, mrem/yr}\right]$$

$$R_i = 1 + \frac{\rho\, Kd_i}{\phi}$$

Eq. 4

Where:	SI_{LLW-PC}	=	Safety indicator for post-closure disposal: Class A, B, and C LLW [GWe-yr^{-1}]
	$V_{\#j}$	=	Volume generation rate of LLW Class # (Class A, B, and C LLW) and waste type j (different types of waste in each class) [m^3/GWe-yr]
	$C\#_i$	=	Concentration of key radionuclide i in LLW Class # (Class A, B, and C LLW) and waste type j [g/m^3]
	R_i	=	Retardation coefficient of radionuclide i (element) in a hypothetical natural system [unitless]
	Kd_i	=	Distribution coefficient of radionuclide i (element) in a hypothetical natural system [cm^3 liquid / g solid]
	ρ	=	Density of hypothetical natural system material [g/cm^3]
	ϕ	=	Porosity of hypothetical natural system material [unitless]
	DCF_i	=	Ingestion dose conversion factor for radionuclide i [mrem/yr/g]
	N	=	Number of key radionuclides important to post-closure safety
	N-#	=	Number of LLW waste types in each Class # (Class A, B, and C LLW)

The mobility, measured through the retardation factor, is a function of the distribution coefficient and the density and porosity of the media where a disposal facility would be placed. The distribution coefficient is also media dependent. Given that the media where LLW from advanced fuel cycles would be disposed is not known, it is appropriate to use a consistent and common set of parameters across the different fuel cycle options. Distribution coefficients for geologic media used in performance assessments for the draft DOE *Greater than Class C Environmental Impact Statement* [Ref. 8, Table 4-19] can be used and are shown in Table 1. The density and porosity for these geologic media are 1.6 g/cm^3 and 0.4, respectively [from Ref. 8, Table 4-18]. The resultant retardation factors are also shown in Table 1.

Ingestion dose conversion factors developed by the International Commission on Radiologic Protection (ICRP 72) are used and are shown in Table 2. Concentrations from the draft DOE *Greater than Class C Environmental Impact Statement* [Ref. 9] are used and are shown in Table 3.

Given the difficulties associated with estimating LLW types, volumes, and radionuclide inventories, an approach that assumes that the radionuclide concentrations in Class A, B, and C LLW are at the levels in the waste classification tables in 10 CFR 61.55 could be used. This approach will result in conservative estimates of radionuclide concentrations. In reality, waste generators would apply the sum of fractions approach per 10 CFR 61.55 and actual concentrations would be lower. In addition, it must be recognized that this approach may not reflect the actual radionuclide inventories/concentrations that could potentially be present in LLW generated by advanced nuclear fuel cycles.

The limiting radionuclide concentrations for Class A, B, and C LLW from 10 CFR 61.55 are shown in Table 3. The 10 CFR 61.55 classification tables are divided into short-lived and long-lived radionuclides. As shown in Table 1, of the short-lived radionuclides, Ni-63 has the longest half-life, 100 years. It is expected that the engineered barriers (packaging, etc.) of a near surface facility would effectively isolate the waste for multiple half-lives and the short-lived radionuclides would decay soon and not contribute to any potential post-closure exposure.

Using the data and assumptions discussed above, Equation 4 becomes the parametric relation shown in Equation 5.

$$SI_{LLW-PC} = \log\left[\frac{V_A \bullet 2.6\times10^{10} + V_C \bullet 1.6\times10^{11}}{100}\right] \qquad \text{Eq. 5}$$

Where: SI_{LLW-PC} = Safety indicator for post-closure disposal: Class A, B, and C LLW [GWe-yr^{-1}]

$V_\#$ = Volume: Class A and C LLW [m^3/GWe-yr]

Table 1. Distribution and Retardation Coefficients for use in Post-Closure Safety Indicators

Element	Kd (cm^3/g)	Retardation Coefficient
C	0	1
Tc	3	13
I	0	1
H	0	1
Co	2	9
Nb	50	201
Ni	12	49
Sr	24	97
Cs	51	205
U	0	1
Pu	10	41

Source: Ref. 8, Table 4-19

Table 2. Ingestion Dose Conversion Factors (ICRP 72)

Class A, B, C and GTCC LLW

Isotope	Effective DCF_{ing} (Sv/Bq)	Effective DCF_{ing} (mrem/g)
C-14	5.8E-10	9.6E+06
Co-60	3.4E-09	1.4E+10
Cs-137	1.3E-08	4.2E+09
H-3	4.2E-11	1.5E+09
I-129	1.1E-07	7.2E+04
Nb-94	1.7E-09	1.2E+06
Ni-59	6.3E-11	1.9E+04
Ni-63	1.5E-10	3.1E+07
Pu-239	2.5E-07	5.8E+07
Pu-240	2.5E-07	2.1E+08
Sr-90	2.8E-08	1.4E+10
Tc-99	6.4E-10	4.1E+04
U-233	5.1E-08	1.8E+08
U-234	4.9E-08	1.1E+06
U-235	4.7E-08	3.8E+02
U-238	4.5E-08	5.6E+01

Uranium and Daughters

Isotope	Effective DCF_{ing} (Sv/Bq)	Isotope	Effective DCF_{ing} (Sv/Bq)
Ac-227	1.1E-06	Po-216	0
Ac-228	4.3E-10	Po-218	0
Bi-210	1.3E-09	Ra-223	1.0E-07
Bi-211	0	Ra-224	6.5E-08
Bi212	0	Ra-226	2.8E-07
Bi-214	1.1E-10	Ra-228	6.9E-07
Fr-223	2.4E-09	Rn-219	0
Pa-231	7.1E-07	Rn-220	0
Pa-234m	0	Rn-222	0
Pb-206	0	Th-207	0
Pb-207	0	Th-227	8.8E-09
Pb-208	0	Th-228	7.2E-08
Pb-210	6.9E-07	Th-230	2.1E-07
Pb-211	1.8E-10	Th-231	3.4E-10
Pb-212	6.0E-09	Th-232	2.3E-07
Pb-214	1.4E-10	Th-234	3.4E-09
Po-210	1.2E-06	U-234	4.9E-08
Po-212	0	U-235	4.7E-08
Po-214	0	U-236	4.7E-08
Po-215	0	U-238	4.5E-08

Disposal Post-Closure Safety Indicator – Greater than Class C (GTCC) LLW

GTCC LLW cannot be disposed of in "traditional" near surface disposal facilities. Two disposal approaches are under consideration in the draft DOE *Greater than Class C Environmental Impact Statement* [Ref. 9] for the disposal of existing and projected GTCC (projections based on present-day practices). These are deep geologic disposal and disposal in enhanced confinement type facilities. The preferred alternative has not yet been selected. As such, it is not possible to project the disposal method that would be used for GTCC wastes generated by future fuel cycles. The FCO Campaign would ultimately need to decide the disposal method for GTCC waste and may want to conduct sensitivity analyses considering both deep geologic and enhanced confinement type disposal pathways should the disposal post-closure safety indicator for GTCC prove to be significant for different fuel cycle options.

Table 3. Concentrations in Class A, B, and C LLW from 10 CFR 61.55 for use in Post-Closure LLW Disposal Safety Indicator

Isotope	Half-Life (yr)	Class A		Class B		Class C	
		(Ci/m^3)	(g/m^3)	(Ci/m^3)	(g/m^3)	(Ci/m^3)	(g/m^3)
Long-Lived[A]							
C-14	5730	0.8	1.79E-01	NO CLASS B		8	1.79E+00
Ni-59	76000	22	2.76E+02			220	2.76E+03
Nb-94	20300	0.02	1.07E-01			0.2	1.07E+00
Tc-99	210000	0.3	1.74E+01			3	1.74E+02
I-129	1.57E+07	0.008	4.53E+01			0.08	4.53E+02
TRU[C]	24065	10	2.63E-01			100	2.63E+00
Short-Lived[B]							
H-3	12.32	40	4.13E-03	> Class B			
Co-60	5.3	700	6.19E-01	> Class B			
Ni-63	100.1	3.5	6.17E-02	70	1.23E+00	700	1.23E+01
Sr-90	28.8	0.04	2.90E-04	150	1.09E+00	7000	5.07E+01
Cs-137	30.2	1	1.16E-02	44	5.08E-01	4600	5.31E+01

[A]From 10 CFR61.55, Table 1 (Ci/m^3 converted to g/m^3)

[B]From 10 CFR61.55, Table 2 (Ci/m^3 converted to g/m^3)

[C]Concentration in nCi/m3. Assumes Pu-239 for activity, Pu-239 for molecular mass, and a density of 1.6 g/cm^3

As will be shown below, the disposal post-closure safety indicator for GTCC is a function of both the volume of GTCC generated and the radionuclide inventory (concentration) in the GTCC waste forms. In addition, a variety of GTCC wastes may be generated by advanced nuclear fuel cycles. Quantifying the different GTCC types, their volumes, and their radionuclide inventories for advanced fuel cycles to support the 2013 screening and evaluation of fuel cycle options is expected to be challenging and uncertain. The level of uncertainty in the GTCC waste types, estimated volumes, and radionuclide inventories need to be considered, propagated through the post-closure safety indicator for GTCC, and factored into the overall screening and evaluation of fuel cycle options.

The post-closure safety indicators for geologic disposal discussed of UNF and HLW above (Equation 3) could be used for cases when GTCC wastes are assumed to be disposed of in deep geologic disposal facilities. Again, the GTCC types, their volumes, and their radionuclide inventories would all have to be estimated.

As discussed above, simulation tools have not been developed to evaluate the post-closure performance of enhanced confinement type disposal systems for GTCC wastes. Efforts to develop such models, initiated in FY2011, are not being pursued in FY2012 and a disposal post-closure safety indicator of the form shown in Equation 3 cannot presently be developed.

In lieu of this, a safety indicator for the disposal of GTCC of the form shown in Equation 6 can be used for the disposal of GTCC in enhanced confinement type facilities. This safety indicator considers the different GTCC waste types, the volume of each waste type that would be generated, the radionuclide inventory in each waste type (concentration), and accounts for the mobility of each radionuclide by dividing the concentration by the retardation coefficient. The resultant "mobile concentration" of each radionuclide is then converted to a measure of annual radiation exposure (ingestion dose, ICRP 72), then weighted by the volume of each type of GTCC.

This metric could be construed as a measure of long-term risk associated with enhanced confinement GTCC disposal, which it is not. Thus, in order to avoid an absolute comparison of computed radiation exposures, a logarithm is taken. In addition, in order to avoid a comparison of computed radiation exposures, all results are normalized to an annual dose of 100 mrem. The 100 mrem/year normalization factor is recommended by the IAEA as the limit to "relevant critical members of the public" for practices involving radioactive materials [Ref. 3, Appendix II-8].

$$SI_{GTCC-PC} = \log\left[\frac{\sum_{j=1}^{n} V_{GTCC_j} \sum_{i=1}^{N}\left(\frac{C_{GTCC_{i,j}}}{R_i} DCF_i\right)}{100\ mrem/yr}\right]$$

$$R_i = 1 + \frac{\rho\, Kd_i}{\phi}$$

Eq. 6

Where:	$SI_{GTCC\text{-}PC}$	=	Safety indicator for post-closure disposal GTCC [GWe-yr^{-1}]
	V_{GTCCj}	=	Volume generation rate of GTCC waste type j [m^3/GWe-yr]
	$C\#_i$	=	Concentration of key radionuclide i in GTCC waste type j [g/m^3]
	R_i	=	Retardation coefficient of radionuclide i (element) in a hypothetical natural system [unitless]
	Kd_i	=	Distribution coefficient of radionuclide i (element) [cm^3 liquid / g solid]
	ρ	=	Density of hypothetical natural system material [g/cm^3]
	ϕ	=	Porosity of hypothetical natural system material [unitless]
	DCF_i	=	Ingestion dose conversion factor for radionuclide i [mrem/yr/g]
	N	=	Number of key radionuclides important to post-closure safety
	n	=	Number of GTCC waste types

The mobility, measured through the retardation factor, is a function of the distribution coefficient and the density and porosity of the media where a disposal facility would be placed. The distribution coefficient is also media dependent. Given that the media where GTCC from advanced fuel cycles would be disposed is not known, it is appropriate to use a consistent and common set of parameters across the different fuel cycle options. Distribution coefficients for geologic media used in performance assessments for the draft DOE *Greater than Class C Environmental Impact Statement* [Ref. 8, Table 4-19] can be used and are shown in Table 1. The density and porosity for these geologic media are 1.6 g/cm^3 and 0.4, respectively [from Ref. 8, Table 4-18]. The resultant retardation factors are also shown in Table 1.

Ingestion dose conversion factors developed by the International Commission on Radiologic Protection (ICRP 72) are used and are shown in Table 2. Concentrations from the draft DOE *Greater than Class C Environmental Impact Statement* [Ref. 9] are used and are shown in Table 3.

Given the difficulties associated with estimating GTCC types, volumes, and radionuclide inventories, an alternative approach could be applied that uses the average concentrations of key radionuclide shown to be important to long-term post-closure performance in the draft DOE *Greater than Class C Environmental Impact Statement* [Ref. 9]. It must be recognized that this approach may not reflect the actual radionuclide inventories/concentrations that could potentially be present in GTCC generated by advanced nuclear fuel cycles.

A number of different sites with differing properties were evaluated to determine their potential for disposing of GTCC waste in the draft DOE *Greater than Class C Environmental Impact Statement* [Ref. 9]. A limited set of radionuclides were shown to be important across this broad set of sites and properties: C-14, Tc-99, I-129, U-233, U-234, U-235, U-238, Pu-239, and Pu-240. The inventory of these radionuclides in the different GTCC waste streams (existing and projected) and the total volume of these waste streams were obtained from the supporting documents to the draft DOE *Greater than Class C Environmental Impact Statement* [Ref. 8] and are also shown in Table 4. These inventories and waste stream volumes were used to compute an average radionuclide concentration for GTCC.

Ingestion dose conversion factors developed by the International Commission on Radiologic Protection (ICRP 72) are used and are shown in Table 2. Concentrations from the draft DOE *Greater than Class C Environmental Impact Statement* [Ref. 8] are used and are shown in Table 3.

Using the data and assumptions discussed above, Equation 6 becomes the parametric relation shown in Equation 7.

$$SI_{GTCC-PC} = \log \frac{V_{GTCC} \bullet 5.2\times10^{7}}{100\,mrem/yr}$$

Eq. 7

Where: $SI_{GTCC\text{-}PC}$ = Safety indicator for post-closure disposal: Class A, B, and C LLW, GTCC, DU, RU [GWe-yr^{-1}]

V_{GTCC} = Volume: GTCC [m^3/GWe-yr]

Disposal Post-Closure Safety Indicator – Depleted/Recycled Uranium

Fuel cycle options under consideration in the 2013 screening and evaluation of fuel cycle options may involve uranium enrichment and the separation of uranium during recycling. These processes would generate DU and/or RU. These materials may either be considered as a resource within a given fuel cycle option or a waste for ultimate disposal. As discussed above, the quantities of DU and RU that would be generated and the decision as to whether DU and RU are deemed as waste for the different fuel cycle options is the scope of the FCO campaign. Disposal post-closure safety indicators are provided in this section should these materials be deemed as waste for a given fuel cycle option.

Table 4. Radionuclide Inventory in GTCC

	Group 1[2]					Group 2[3]						
Radionuclide	**GTCC Activated Metals (Ci)**	**GTCC Other Waste – RH (Ci)**	**GTCC-like Activated Metals (Ci)**	**GTCC-like Other Waste – CH (Ci)**	**GTCC-like Other Waste – RH (Ci)**	**GTCC Activated Metals (Ci)**	**GTCC Other Waste – CH (Ci)**	**GTCC Other Waste – RH (Ci)**	**GTCC-like Other Waste – CH (Ci)**	**GTCC-like Other Waste – RH (Ci)**	**Average Concentration (Ci/m^3)**	**Average Concentration (g/m^3)**
Volume → (m^3)[1]	879	34	12.8	740	1220	2000	1600	2400	1200	1600		
C-14	2.30E+04	5.80E-03	6.80E+02	1.30E+01	1.00E+02	1.00E+04	4.40E+00	1.50E+02	5.90E+00	9.00E+00	2.91E+00	6.52E-01
Tc-99	4.50E+03		3.20E-01			1.90E+03	1.00E-03	1.70E+01	1.30E-01	3.20E+00	5.49E-01	3.19E+01
I-129	1.90E+00					2.10E+00	2.90E-03	5.40E-02		3.80E-03	3.47E-04	1.97E+00
U-233		6.00E-01		9.40E+00	7.90E+02	3.80E+00		7.40E+00	4.10E+00	6.40E+00	7.03E-02	7.26E-02
U-234				4.40E+01	1.60E+00	2.00E-01	9.70E-03	3.90E-01	1.90E+01	2.90E+01	8.06E-03	1.29E+00
U-235		5.20E-03		1.60E-01	3.50E-01	7.20E-02	4.80E-04	3.70E+00	8.00E-03	1.40E-02	3.69E-04	1.71E+02
U-238				9.10E-02	1.10E+01	8.40E-01	1.00E-02	3.10E+00	3.90E-02	7.30E-02	1.30E-03	3.85E+03
Pu-239	4.50E+03	2.50E+01		9.00E+02	2.90E+03	2.10E+03	4.90E+01	4.50E+02	4.00E+02	6.40E+02	1.02E+00	1.65E+01
Pu-240		7.50E+00		7.10E+02	1.80E+03	1.60E+02	4.50E+01	2.40E+02	3.20E+02	5.10E+02	3.25E-01	1.42E+00

Source: [Ref. 8], Argonne National Laboratory, Post-Closure Performance Analysis of the Conceptual Disposal Facilities at the Sites Considered for the Greater-than-Class-C Environmental Impact Statement, ANL/EVS/R-10/8, October 2010 (www.gtcceis.anl.gov/documents/docs/ANL_EVS_R-10_8.pdf)

[1]Table 2-1

[2]Table 2-2

[3]Table 2-5

The disposal method for DU and RU is currently not known. However, based on current activities within the U.S. Nuclear Regulatory Commission (NRC), it can be assumed that they would be disposed of in near-surface disposal facilities.

The NRC has directed the staff to pursue a limited rulemaking to specify a requirement for a site-specific analysis and associated technical requirements for unique waste streams including, but not limited to, the near-surface disposal of significant quantities of depleted uranium [Ref. 10]. In this limited rulemaking, the NRC is not proposing to alter the waste classification scheme. However, for unique waste streams including, but not limited to, significant quantities of depleted uranium, the NRC recognizes that there may be a need to impose additional criteria on its disposal at a specific facility or deny such disposal based on unique site characteristics. Those restrictions would be determined through a site-specific analysis, which would satisfy the requirements developed through the rulemaking process.

As discussed above, simulation tools have not been developed to evaluate the post-closure performance of near-surface disposal systems for DU and RU. Efforts to develop such models, initiated in FY2011, are not being pursued in FY2012.

In lieu of this, a safety indicator for the disposal of DU and RU of the form shown in Equation 8 can be used for the disposal of GTCC in enhanced confinement type facilities. This safety indicator considers the inventory in DU, and RU and accounts for the mobility of each radionuclide by dividing the concentration by the retardation coefficient. The resultant “mobile concentration” of each radionuclide is then converted to a measure of radiation exposure (ingestion dose, ICRP 72), then weighted by the volume of the material.

This metric could be construed as a measure of long-term risk associated with disposal, which it is not. Thus, in order to avoid a comparison of computed radiation exposures, a logarithm is taken. In addition, in order to avoid a comparison of computed radiation exposures, all results are normalized to an annual dose of 100 mrem. The 100 mrem/year normalization factor is recommended by the IAEA as the limit to “relevant critical members of the public” for practices involving radioactive materials [Ref. 3, Appendix II-8].

$$SI_{DU-PC} = \log\left[\frac{V_{DU} \bullet \sum_{i=1}^{N}\left(\frac{C_{DU_i}}{R_U} DCF_i\right)}{100\ mrem/yr}\right]$$

$$SI_{RU-PC} = \log\left[\frac{V_{RU} \bullet \sum_{i=1}^{N}\left(\frac{C_{RU_i}}{R_U} DCF_i\right)}{100\ mrem/yr}\right] \qquad \text{Eq. 8}$$

$$R_i = 1 + \frac{\rho\, Kd_U}{\phi}$$

Where: $SI_{\#\text{-}PC}$ = Safety indicator for post-closure disposal: DU, RU [GWe-yr^{-1}]

$V_{\#}$ = Volume: DU, RU [m^3/GWe-yr]

$C\#_i$ = Concentration of key uranium radionuclide i: DU, RU [g/m^3]

R_i = Retardation coefficient of radionuclide i (element) in a hypothetical natural system [unitless]

Kd_U = Distribution coefficient of uranium [cm^3 liquid / g solid]

ρ = Density of hypothetical natural system material [g/cm^3]

ϕ = Porosity of hypothetical natural system material [unitless]

DCF_i = Ingestion dose conversion factor for uranium radionuclide i [mrem/yr/g]

N = Number of key uranium isotopes in DU and RU important to post-closure safety

The mobility, measured through the retardation factor, is a function of the distribution coefficient and the density and porosity of the media where a disposal facility would be placed. The distribution coefficient is also media dependent. Given that the media where DU and/or RU from advanced fuel cycles would be disposed is not known, it is appropriate to use a consistent and common set of parameters across the different fuel cycle options. Distribution coefficients for geologic media used in performance assessments for the draft DOE *Greater than Class C Environmental Impact Statement* [Ref. 8, Table 4-19] can be used and are shown in Table 1. The density and porosity for these geologic media are 1.6 g/cm^3 and 0.4, respectively [from Ref. 8, Table 4-18]. The resultant retardation factors are also shown in Table 1.

Ingestion dose conversion factors developed by the International Commission on Radiologic Protection (ICRP 72) are used and are shown in Table 2. Concentrations from the draft DOE *Greater than Class C Environmental Impact Statement* [Ref. 9] are used and are shown in Table 3.

The concentration of uranium isotopes in either DU or RU for use in Equation 8 is determined using Equation 9. The fraction of U-235 in DU depends on the enrichment of the fuel for a given fuel cycle scenario and is approximately 0.3 weight percent (w/o), with the remainder being U-238. The mass fractions of uranium isotopes in RU depends on the burn-up of the fuel.

$$\begin{aligned}
C_{U-234} &= \rho_U \bullet f_{U-234} \quad \text{(RU Only)}\\
C_{U-235} &= \rho_U \bullet f_{U-235}\\
C_{U-236} &= \rho_U \bullet f_{U-236} \quad \text{(RU Only)}\\
C_{U-238} &= \rho_U \bullet f_{U-238}
\end{aligned} \qquad \text{Eq. 9}$$

Where: $C_{U\text{-}\#}$ = Concentration of uranium radionuclide in DU or RU [g/m^3]

ρ_U = Density of DU or RU [g/m^3] (density of UO_2 = 10.97 g/m^3)

$f_{U\text{-}\#}$ = Mass fraction of uranium radionuclide in DU or RU

The hazard associated with the disposal of DU and/or RU increases with time due to daughter product build-up, peaking after about one-million years. As such, the contribution of daughter products in the U-234, U-235, and U-238 decay chains to the overall safety indicator should be considered. Figure 1 shows the total ingestion dose conversion factor (mrem per gram initial uranium) for each uranium isotope and its daughter products, calculated using the ICRP-72 ingestion dose conversion factors. The peak total DCF for each uranium isotope, factoring in daughter product contribution is shown below.

- U-234: 3.33E+07 mrem/g
- U-235: 1.58E+04 mrem/g
- U-236: 1.13E+04 mrem/g
- U-238: 3.06E+03 mrem/g

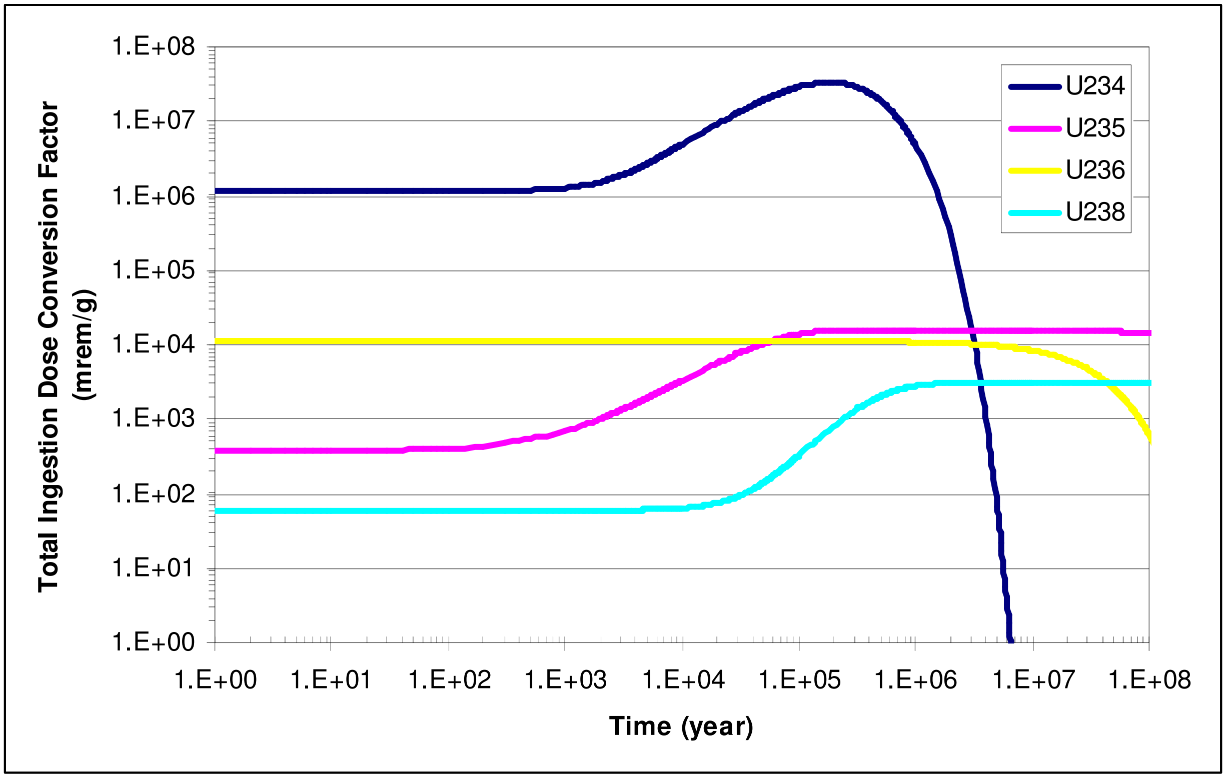

Figure 1. Uranium Ingestion Dose Conversion Factor

3.3 Economics

The UFD and SWF campaigns understand that an update to the *Advanced Fuel Cycle Cost Basis* [Ref. 11] and that information in this updated report will be used to develop the economic metrics for the 2013 screening and evaluation of fuel cycle options (WBS No., 1.02.12.03, Work Package FT-12IN120303, Milestone M2FT-12IN1203031, due August 13th, 2012). The UFD and SWF campaigns can support the FCO in revising the *Advanced Fuel Cycle Cost Basis* report.

This section provides recommended economic metrics under the Nuclear Waste Management criterion that should be considered for both the storage and disposal of UNF and high level nuclear waste that should be considered in the update of the *Advanced Fuel Cycle Cost Basis* report. This section also provides information pertaining to each of the recommended economic metrics that can be used in the update to the *Advanced Fuel Cycle Cost Basis* report. This section also describes cost estimates that are currently being developed by the UFD and SWF campaigns that could also be used in the update to the *Advanced Fuel Cycle Cost Basis* report.

The complexity of making cost estimates for relatively undefined or pre-conceptual systems must be recognized. Detailed bottom-up cost analyses would ultimately be required to fully estimate the cost of a given fuel cycle option. However, simple estimates using existing information are expected to be adequate to support the 2013 screening/evaluation of fuel cycle options. The economic metrics recommended in this section cover the range of potential components that could be utilized within a fuel cycle option, but recognizes that not all aspects will necessarily be used for each fuel cycle option under consideration.

All economic metrics are provided herein on cost per mass or volume of material basis (i.e., dollars / kg HM). In the case of UNF, the economic metrics can be converted to an energy-generated basis once the discharge burn-up of the UNF for a given fuel cycle scenario is defined.

3.3.1 Used Nuclear Fuel and High-Level Waste Storage

This section provides economic information for the on-site and off-site storage of UNF and HLW. It must be recognized that these cost estimates are uncertain, however they can be used to comparatively evaluate disposal costs for the different fuel cycle options under consideration in the 2013 screening and evaluation of fuel cycle options.

3.3.1.1 Routine At-Reactor Storage of Used Nuclear Fuel

At-reactor storage of UNF is a routine part of reactor operations. Different reactor concepts utilize different storage concepts (i.e., wet storage pools, basket storage, dry storage pits). Such storage facilities will be included in the overall design, and cost, of a nuclear power plant. However, no information regarding cost associated with routine storage is available, except for the storage of used LWR fuel. The cost of on-site wet storage of used LWR fuel has been estimated in *Advanced Fuel Cycle Cost Basis* [Ref. 11, Table E1-4] to range from $100/kg HM per year to $500/kg HM per year with a nominal cost of $300/kg HM per year. These estimates are based on very limited data as reflected in the following statement provided with the estimate in the *Advanced Fuel Cycle Cost Basis* report.: "*Cost data are not considered to be of high quality because there has been no common basis or consistent approach.*"

The low estimate is intended to reflect a case of well planned capacity that may be anticipated for new reactors, the nominal case is intended to reflect current practice in the nuclear industry, and the high estimate is intended to reflect a condition where significant amount of fuel management is needed and/or the construction of a new wet storage facility.

Since no additional cost estimate information regarding the cost of storing used LWR fuel are available, the estimates provided above were used to develop costs for the routine storage of UNF at-reactor for use in the 2013 screening and evaluation of fuel cycle options. While those cost estimates were developed for used LWR fuel, they are also used to develop metric estimates for advanced nuclear power plants that may be included in different fuel cycle options. The following two estimates are recommended for the 2013 screening and evaluation of fuel cycle options:

- Current LWR Fleet: $300/kg HM per year – This assumes that the current fleet of nuclear power plants in the U.S. will follow current practice.

- Future Reactors: $100/kg HM per year – This assumes that well planned UNF systems will be included as part of the design of future reactors and UNF management costs will be less than that of the current U.S. fleet. This estimate is intended to be applied to all aspects of at-reactor UNF that are associated with the "routine" management of UNF. It does not include the cost of developing an Independent Spent Nuclear Fuel (SNF) Storage Installation for managing UNF, which is discussed in below.

3.3.1.2 Independent On-Site Storage of Used Nuclear Fuel

The on-site storage of UNF, beyond the routine management of UNF associated with the reactor, is part of the fuel management strategy for the existing U.S. nuclear fleet. Independent Spent Fuel Storage Installations (ISFSIs) have been and continue to be deployed at U.S. reactor sites. An ISFSI was also constructed to store UNF discharged from the Fort St. Vrain gas-cooled reactor. Common, independent (not directly linked to an operating reactor) wet storage pools have been constructed at nuclear power plants internationally.

The fuel cycle options under consideration in the 2013 screening and evaluation of fuel cycle options may consider the use of additional, independent on-site storage as an integral part of UNF management. The cost of such ISFSIs would depend on the size of the facility (capacity), the type of facility (i.e., wet type storage, vault, dry storage canisters), the thermal output of the material being stored, and how long it is operated. The recommended cost of an on-site wet, independent used LWR fuel storage facility is based on the high estimate provided in *Advanced Fuel Cycle Cost Basis* [Ref. 11, Table E1-4] and is $500/kg HM per year. As discussed in Section 3.3.1.1, this estimate is intended to reflect a condition where a new wet storage facility is constructed for off-site wet storage.

The total life cycle cost of on-site dry storage of used LWR fuel over a 40 year period has been estimated in *Advanced Fuel Cycle Cost Basis* [Ref. 11, Section E2-6] at $120/kg HM per year if incurred while a reactor is still operating and $250/kg HM per year if the reactor shutdown.

This estimate reflects the cost of current practice for the existing LWR fleet. A high-end estimate of $300/kg HM is also provided, which is intended to reflect a condition where significant regulatory change, increased design, and more robust canisters are required.

The General Accounting Office (GAO) estimated the cost of on-site dry storage for three different scenarios: 70,000 MTHM for a 100 year period, 153,000 MTHM for a 100 year period, and 153,000 MTHM or a 500 year period with re-packaging occurring every 100 years [Ref. 12]. The GAO discounted the costs and reported them in 2009 present value dollars, shown in Table 5 for the 100 year storage scenarios.

Table 5. GAO Estimate of the Cost of On-Site Dry Storage (2009 present value dollars)

	Minimum	Mean	Maximum
	70,000 MTHM, 100 years		
Total Cost (billion $)	10	18	26
Unit Cost ($/Kg HM)	143	257	371
	153,000 MTHM, 100 years		
Total Cost (billion $)	13	22	34
Unit Cost ($/Kg HM)	85	144	222

Source: Ref. 12, Table 8.

The GAO estimates show the effects of both economy of scale and discounting when comparing the total and unit costs between the 70,000 and 153,000 MTHM cases. The majority of the cost of on-site dry storage is associated with the cost of licensing the facility, the capital cost of constructing the facility, and the cost of procuring and loading the canister/cask systems. Annual operating costs are estimated at $200,000 per year for on-site dry storage facilities at operating reactor sites and $3-4.5 million per year for shutdown and decommissioned reactors [Ref. 12, Table 7]. Additional licensing and facility construction would not be required to accommodate an increased quantity of fuel being transferred to dry storage. The most significant additional cost would be in the procurement and loading of additional dry storage casks and such costs would not be incurred until well into the future. Discounting to 2009 present value dollars leads to a decrease in "apparent costs." In addition, the GAO cost estimates also incurred annual operating and maintenance costs both during reactor operations and following reactor shutdown.

For the 2013 screening and evaluation of fuel cycle options, it is recommended that the cost of on-site storage for those fuel cycle options that have discharge fuel with characteristics similar to the existing LWR fuel be $120/kg HM per year, based on the estimate provided in *Advanced Fuel Cycle Cost Basis* for costs incurred while a reactor is still operating. This reflects that minor changes to existing technologies and the associated regulatory framework would be required, as discussed in *Advanced Fuel Cycle Cost Basis*. The GAO estimated costs shown in Table 5 were not considered as representative since they assume a 100 year on-site storage period.

The on-site storage of advanced nuclear fuels would likely require significant regulatory change, increased design, and possibly more robust storage canisters. This would indicate that a higher cost of storage should result. It is possible that once these are accomplished costs would decrease, possibly approaching the current cost of on-site storage. However, to reflect such challenges and uncertainties, it is recommended that the cost of on-site storage for those fuel cycle options involving advanced reactor concepts be $300/kg HM per year.

3.3.1.3 Off-Site Storage of Used Nuclear Fuel and High Level Waste

The fuel cycle options under consideration in the 2013 screening and evaluation of fuel cycle options may consider the use of independent off-site storage as an integral part of UNF and/or HLW management. The cost of such a facility would depend on the size of the facility (capacity), the type of facility (i.e., wet type storage, vault, dry storage canisters), the thermal output of the material being stored, and how long it is operated.

<u>Off-Site Storage of Used Nuclear Fuel</u>

The UFD Campaign is estimating the cost of off-site facilities for the storage of UNF. A range of facility types (dry horizontal configurations, dry vertical configurations, dry vaults, and wet pools) are being considered and costs will be estimated for different facility capacities and processing rates. These estimates will be used to develop economic indicators for off-site dry storage of UNF for use in the 2013 screening/evaluation of fuel cycle options.

<u>Storage of High-Level Nuclear Waste</u>

Storage costs for HLW are dependent on waste form characteristics such as thermal output, dispersibility of the waste form, and physical size of the packages. These characteristics influence the design and hence the cost of storage facilities for HLW. For instance high heat waste forms typically require forced ventilation to remove the high heat loads. Low heat waste forms can be cooled by passive, natural draft ventilation systems. Furthermore, dispersible waste forms require filtration of cooling air to mitigate the release of radionuclides in the event of an accident and the physical size of the waste container has an influence on facility size.

Many different storage scenarios can be developed depending on the waste form characteristics being considered. Cost estimates for many of the potential scenarios for storing HLW have been previously developed in engineering alternative studies. These studies provided estimates for storage of a variety of waste forms representing a variety of waste form characteristics and storage facility configurations. The estimates addressed storage of low and high heat vitrified waste forms, high heat dispersible alumina silicate waste forms and high heat bentonite clay waste forms. The facilities that were estimated ranged from passively ventilated, unfiltered facilities to facilities with forced air cooling and filtration. Both high efficiency particulate air (HEPA) and deep-bed sand filtration systems were estimated.

Information from the EAS cost estimates is being used to develop cost estimates for other potential HLW storage scenarios. These estimates along with the original EAS estimates will be used to define a metric that can then be used to estimate storage costs for other waste forms to be developed in the future.

3.3.2 Radioactive Waste Disposal

This section provides estimates for the cost of near-surface disposal of Class A, B, and C LLW, the disposal of GTCC (either deep geologic or enhanced confinement-type), and the deep geologic disposal of UNF and HLW. It must be recognized that these cost estimates are uncertain, however they can be used to comparatively evaluate disposal costs for the different fuel cycle options under consideration in the 2013 screening and evaluation of fuel cycle options.

3.3.2.1 Near Surface Disposal

Class A, B, and C LLW generated in the U.S. is disposed of either in U.S. DOE owned or commercially owned facilities, depending on origin. All LLW generated by the commercial U.S. nuclear industry is disposed of in commercial facilities. The cost of commercial LLW disposal varies for different sites and different producers, somewhat affected by fees levied by individual states where sites are located, but in general is market driven. Disposal costs are proprietary and not readily available. LLW generated by the U.S. DOE is disposed of either in U.S. DOE owned or commercial facilities with the pathway influenced by market conditions.

The cost of near surface LLW has been estimated in *Advanced Fuel Cycle Cost Basis* [Ref. 11, Table J-7] to range from \$450/m^3 to \$2500/ m^3 with a nominal cost of \$1,200/ m^3. The nominal cost reflects estimated cost of disposal at the U.S. DOE facility located at the Nevada Nuclear Security Site. The high and low estimates cover the range of costs for U.S. DOE generated LLW in different facilities.

In 2004, the General Accounting Office (GAO) [Ref. 13] reported that commercial LLW disposal costs are approximately \$400 per cubic foot (\$14,000 per cubic meter), with projections of well over \$1,000 per cubic foot in the future (\$35,000 per cubic meter). GAO further reported that in 2004 the Barnwell disposal facility was charging \$1,625 per cubic foot for some LLW (\$57,000 per cubic meter).

EPRI also provided "round number" estimates at \$4,000 – 5,000 per cubic foot (approximately \$140,000 - \$176,000 per cubic meter) for the disposal of commercial Class B and C waste, \$500 per cubic foot (approximately \$18,000 per cubic meter) for Class A waste that require shielded transportation, and \$3-\$4 per pound for dry active waste (DAW) that does not require shielded transportation (Class A – paper, wood, rags, tools, sheet metal, etc.) [personal communication, J. Kessler, EPRI Nov. 11, 2011]. Assuming a DAW density of 0.25 g/cm^3 [Ref. 14] results in an estimated cost of disposing DAW at \$1,650 - \$2,200 per cubic meter.

For the 2013 screening and evaluation of fuel cycle options, it is recommended that the cost of near surface LLW disposal be $1,650/m^3 for Class A LLW (assuming the majority will be DAW) and $150,000/m^3 for Class B and C LLW.

3.3.3 GTCC Disposal

The Draft *Greater than Class C Environmental Impact Statement* [Ref. 9] considered the disposal of approximately 120,000 cubic meters of GTCC waste in three different enhanced confinement type near surface concepts (borehole, trench, and vault) and for the deep geologic disposal in the Waste Isolation Pilot Plant (WIPP). The estimated total cost for each disposal concept and the normalized cost (120,000 m^3 of GTCC) are shown in Table 6. The costs of disposal in the WIPP reflect costs that would be incurred by placing GTCC into an already operating deep geologic facility. The costs of disposal in the borehole, trench, and vault concepts reflect the construction and operation of new facilities.

Table 6. Cost Estimates for GTCC disposal

GTCC Disposal Alternative	Construction Cost ($M)	Operations Cost ($M)	Total Cost ($M)	Normalized Cost ($/m^3)
WIPP	14	560	574	4783
Borehole	210	120	330	2750
Trench	88	160	248	2067
Vault	360	160	520	4333

Source: Ref 9, Table S-5

For the 2013 screening and evaluation of fuel cycle options, it is recommended that the cost of disposing GTCC in enhanced confinement facilities be $3,050/m^3 (average of borehole, trench, and vault cost estimates). It is recommended that the cost of disposing GTCC in a deep geologic repository, assuming it is co-located with disposed fuel or HLW, be $4,800/m^3. However, it is noted that these estimates for the disposal of GTCC waste are significantly lower than those above for the disposal of Class B and C LLW. This difference should be pursued in the update to the *Advanced Fuel Cycle Cost Basis* [Ref. 11]

3.3.4 Deep Geologic Disposal

The development of a deep geologic facility for the disposal of SNF and/or high level nuclear waste will proceed through a series of steps, each having associated costs. These steps are

- Site selection (scientific studies, evaluation/volunteer process, incentive package)
- Site characterization
- Design (conceptual → preliminary → detailed)
- Licensing
- Construction
- Operations
- Monitoring / Closure

The overall cost of deep geologic disposal of UNF and/or HLW is uncertain as it has yet to be demonstrated anywhere in the world. Much of this uncertainty is associated with the first four steps in the development process shown above. Challenges associated with siting, design, and licensing world-wide have resulted in delays and increased costs. However, once a site is ultimately selected and a detailed design developed, estimates of construction, operations, and monitoring/closure costs would be much less uncertain. Of all the steps shown above, the cost of facility construction and operations would likely be the largest contributor to the total life cycle cost of a geologic disposal facility.

The cost of facility construction and operations would depend on several factors. Surface facility construction and operation costs would depend on the size of the facility, the required throughput, and the types/characteristics of the wastes that would be processed. Sub-surface facility construction and operation costs would depend on the type of rock that would be mined, the subsurface design (tunnels, emplacement boreholes, alcoves/rooms), and the amount of rock that would have to be excavated. The latter would depend on the total capacity of the facility and the waste emplacement approach used to meet thermal limits. The design of the engineered barrier system that would have to be procured, fabricated, and emplaced would also influence overall costs.

The UFD campaign is investigating mined geologic disposal in salt, clay/shale, and crystalline rock environments and deep borehole disposal in crystalline basement rock [Ref. 15]. Generic design concepts have been developed in Fiscal Year 2011 [Ref. 4] and continue to be developed. Efforts are underway in Fiscal Year 2012 to determine the factors that influence cost for these generic facilities.

4. Development and Deployment Risk

This section discusses the Development and Deployment Risk criterion with respect to Nuclear Waste Management. The challenges associated with deploying UNF, LLW, and HLW storage and disposal facilities are well known and mostly relate to social-political and institutional aspects. As discussed above, Institutional Issues, which include public and stakeholder acceptance, are also important for siting, licensing, and development of nuclear waste management facilities. Again, Institutional Issues are not limited only to nuclear waste management, but apply to the entire nuclear fuel cycle. Stakeholder acceptance is inherently subjective and difficult to quantify. Accordingly, an alternative approach is often used in studies of this type, whereby stakeholder input to the study is elicited and factored into the assessment of other metrics [Ref. 2]. Such an approach could be considered in the 2013 screening of fuel/evaluation cycle options and the UFD Campaign has experience in such elicitations and could provide support to the FCO Campaign in conducting such evaluations. However, no specific metrics for stakeholder acceptance are proposed at this time for nuclear waste management.

This section discusses the technology readiness level (TRL) associated with the different waste management facilities. A TRL indicates the maturity level of a given technology, as defined by the DOE in U.S. Department of Energy Technology Readiness Assessment Guide [Ref. 18]. As such, the TRL is an indication of the technical readiness to deploy, as opposed to the socio-political readiness discussed above. The TRL assessment provided in Table 7 is based on precedence established both in the U.S. and internationally in storing and disposing radioactive materials. The assessment shows that the storage and disposal of nuclear materials is very mature, in particular for the current waste streams generated. The primary deployment hurdle for storage and disposal facilities is not one of technology development, but rather siting and licensing.

An evaluation of TRLs for waste forms was initiated in Fiscal Year 2008, but was subsequently terminated in Fiscal Year 2009. An initial assessment conducted in Fiscal Year 2008 following the DOE-EM guidelines for conducting technology readiness assessments, indicated that certain factors for each waste form under consideration would result in low TRLs. However, these assessments were based on qualitative judgments a more thorough effort is still required to assess wasteform TRLs.

Table 7. Nuclear Waste Management Technology Readiness Level Assessment

Facility	TRL	Discussion
Used light water reactor fuel storage	8-9	Used light water reactor (LWR) fuel is routinely stored in wet and dry facilities at existing power plants in the U.S. Similar facilities could be deployed at independent locations. Potential gaps have been identified in the technical basis for licensing dry storage of high-burnup fuels and extended dry storage of all LWR fuel.
Used advanced reactor fuel storage	7 – 8	Storage systems for a variety of different reactor types have been designed and deployed (i.e., sodium fast reactors, gas cooled reactors). Fuel storage would be an integral part of advanced reactor designs. Specific storage facility designs would have to be designed, licensed, and constructed.
Low-level waste storage	9	LLW storage is done routinely
GTCC storage	9	GTCC storage is done routinely
High level waste storage	8-9	HLW generated from the U.S. defense mission is routinely stored. HLW is also routinely stored in countries that reprocess used LWR fuel (i.e., France, the United Kingdom, Japan).
LLW Disposal	9	LLW is routinely disposed of in near-surface facilities
GTCC Disposal	8 – 9	The Waste Isolation Pilot Plant (WIPP) and the DOE-EM Greater Confinement disposal facility are used to dispose of defense transuranic waste. These materials are very similar to commercially generated GTCC. Internationally, intermediate level waste, which is similar to the expected GTCC generated by U.S. commercial fuel reprocessing, is also disposed of routinely.
UNF/HLW deep geologic disposal	7 5 for deep borehole disposal	The WIPP is an operating deep geologic repository for transuranic wastes. The submittal of the Yucca Mountain License Application as well as licensing actions underway in Sweden and Finland also demonstrate the technical maturity of deep geologic disposal. Detailed designs of repositories were completed in support of license applications. Underground research laboratories have also been constructed in granite and clay/shale environments that demonstrate viability. The WIPP is an operating deep geologic repository for low heat defense wastes. Demonstrations of emplacement systems have also been tested (e.g., by ANDRA in France) The TRL for deep borehole disposal concepts has yet to be demonstrated in an engineering/pilot scale environment (needed to achieve a TRL 6).
Waste form fabrication facility	9 for HLW in borosilicate glass 2 for advanced waste forms	Conversion of HLW into borosilicate glass is commonly practiced world-wide using a number of technological approaches. However the immobilization of gaseous fission products and waste forms for unique waste streams from the most advanced fuel cycles have not been demonstrated past the TRL 2 stage. Also, the generation of advanced waste forms for HLW immobilization with very high durability or improved economics is unproven.

5. Inputs Required

As discussed previously, the Nuclear Waste Management metrics recommended in Section 3 are indicators based on the "quantities" (i.e., volumes, isotopic inventories) of waste that must be managed for a given fuel cycle option, or estimated costs based on the "quantities." These "quantities" must be obtained for each fuel cycle option under consideration and once defined for a given fuel cycle option, the waste management metrics can be calculated. Once the "quantities" are defined, they can be used to calculate intermediate parameters such as inhalation radiotoxicity and thermal output. The combination of the "quantity" and intermediate parameters can then be used to compute the safety and economic Nuclear Waste Management metrics recommended in Section 3. The UFD and SWF Campaigns are available to support the FCO Campaign in estimating these "quantities."

The UFD campaign has estimated and continues to estimate UNF and HLW inventories and waste form characteristics that can be used to evaluate different fuel cycle options. In addition to UNF projections, estimates of waste canister count, waste volume, isotopic inventory, and thermal output have been developed for different recycling technologies with different source inputs. Estimates for several fuel cycle technologies are complete [Ref. 16]. The UFD campaign also has estimated and continues to estimate LLW inventories, focusing on the volume of Class A, B, and C LLW, and GTCC. Estimates for several fuel cycle technologies are complete [Ref. 7 and 17].

The UFD and SWF Campaigns will continue to work with the FCO Campaign to estimate additional waste inventories (UNF, HLW, and LLW) and characteristics as the fuel cycle options and associated technologies are developed for the 2013 screening and evaluation of fuel cycle options.

Both safety and economic Nuclear Waste Management metrics for UNF and HLW have the time that material is stored as a one of the input parameters. All geologic disposal facilities had thermal constraints and analyses completed in Fiscal Year 2010 [Ref. 4], which showed that decay storage would be required in many cases to reduce the waste thermal output before it can be efficiently emplaced in geologic disposal facilities. As such, the thermal constraints for a disposal environment and the thermal output of the waste can have a direct input on both the size of storage facilities and the amount of time the waste must be stored, both of which affect safety and economic metrics.

The UFD Campaign is developing generic design concepts for the disposal media being investigated (granite, clay/shale, salt, and deep boreholes) and is performing thermal load management studies for these concepts. The UFD campaign will develop a high-level response surface of the required decay storage time as a function of waste thermal output for each of the generic design concepts in the different disposal media.

The post-closure geologic disposal safety indicator discussed in Section 3.2 has the waste form fractional degradation rate as an input. The SWF Campaign will develop estimates for the waste form fractional degradation rate for the waste forms associated with the different fuel cycle options and technologies being considered in the 2013 screening and evaluation of fuel cycle options.

It is important to note that for a large set of fuel cycle options, the data required for the foregoing input "quantities" is extensive and may not be gathered in time for the 2013 evaluation and screening. If that is the case, the safety "indicator" metrics proposed above will have to be simplified further and estimated subjectively by a panel of experts, such as the Evaluation and Screening team suggested in the Screening Charter [Ref. 1].

6. References

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APPENDIX A: Methodology for Computing the Post-Closure Safety Indicator

A-1. Modeling Approach

The UFD GDSMs will be used to simulate the release of radionuclides from hypothetical waste forms with different combinations of key radionuclide inventories (I_i) and waste form fractional degradation rate (F_d) and the subsequent radionuclide transport through the engineered and natural barriers for the generic geologic disposal environments under consideration by the UFD. These are geologic disposal systems in salt, clay/shale, and granite media and for deep borehole disposal in crystalline rock. The resulting peak annual dose for each key radionuclide will be collected for the different radionuclide inventory / waste form fractional degradation rates to build a numerical “response surface” to represent the functional form shown in Equation 3.

This approach is identical to that previously used to support the establishment of waste form performance criteria [Ref. A-1] and is described in more detail in Appendix A. It inherently captures the isolation capabilities of the natural and engineered barriers that are, for the most part, independent of fuel cycle options and includes the two primary independent variables that would be affected by different fuel cycle options.

Figure A-1 is a conceptual representation of the UFD GDSMs that will be used to model the disposal system in salt, clay/shale, crystalline, and deep borehole media to determine the functional relationship for the post-closure safety indicator shown in Equation 3 above. Representative designs for geologic disposal systems in salt, clay/shale, and granite media and for deep borehole disposal in crystalline rock [Ref. A-2] will be used to define engineered barrier system conceptualizations within the UFD Generic Disposal System Models (GDSMs) [Ref. A-3]. The current UFD baseline properties of both the natural and engineered barriers will also used [Ref. A-3]. These designs and property sets will be used to determine the safety indicators for post-closure geologic disposal using the methodology described below.

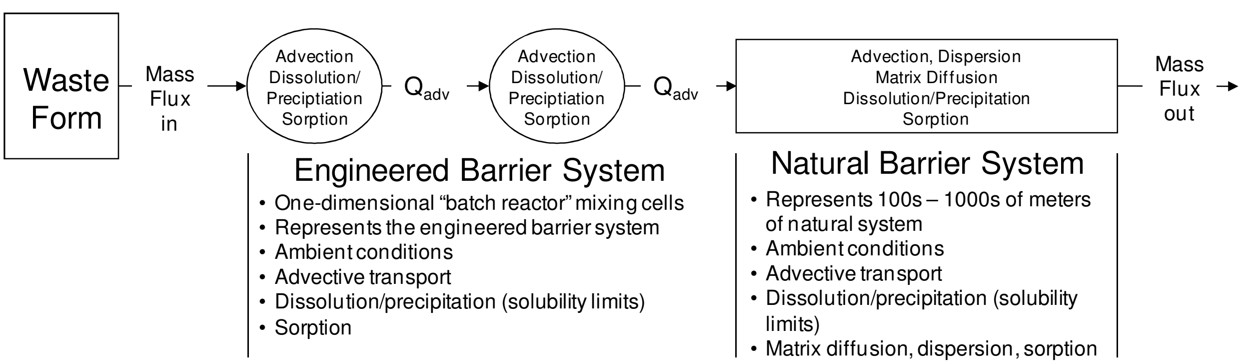

Figure A-1: Conceptual Description of UFD GDSM Applications for Developing Post-Closure Safety Metrics

The release rate from a waste form is related to waste form performance (fractional degradation rate) and inventory contained in the waste form through Equation A-1.

$$\dot{m}_j(t) = F_{wf} \bullet I_{WF,j}(t)$$
$$I_{WF,j}(t) = I_{WF0,j}(t)\exp^{-(F_{WF}+\lambda_j)t} \quad \text{Eq. A-1}$$

Where: $\dot{m}_j(t)$ = the mass flux of radionuclide j from the EBS at time t (g/yr)

F_{wf} = the fractional degradation rate of the waste form (yr^{-1})

λ_j = the decay constant of radionuclide j (yr^{-1})

$I_{wf,j}$ = the inventory of radionuclide j in the waste form (g)

$I_{wfo,j}$ = the initial inventory of radionuclide j in the waste form at time t (g)

The approach utilizes Equation 3 to model waste form performance by considering different combinations of initial radionuclide inventory ($I_{wfo,j}$) and waste form fractional degradation rate (F_{wf}) for the representative designs and properties of each media under consideration.

The initial inventory of each radionuclide will be varied from 0.001 to 10 times the inventory in commercial LWR UNF with a burn-up of 40,000 MWd/MTHM, 30 years after reactor discharge (from Carter et.al., 2011). This range was chosen so as to capture the range of potential radionuclide inventories, on a per MTHM basis, that could be isolated in a waste form.

The waste form fractional degradation rate will be varied from 10^{-9} yr^{-1} to 10^{-2} yr^{-1}. This range is extremely broad (100 year to 1 billion year waste form) and is expected to capture fractional degradation rates of advanced waste forms. Note that Equation 3 essentially becomes a steady-state release rate at low waste form degradation rates.

In order to account for daughter product in-growth and the isotopic mix among radionuclides, the following approach will be used.

- The key <u>fission product radionuclides</u> will be modeled independently
- The <u>actinide elements</u> (U, Pu, Am, and Np) will be modeled individually. This was done because several of the actinide element decay chains intersect. For example, $^{241}Am \rightarrow {}^{237}Np \rightarrow {}^{233}U$ and $^{242}Pu \rightarrow {}^{238}U$.
- The inventory of <u>actinide radionuclides</u> will be varied by changing the element inventory (0.001 – 10 times that of LWR SNF) while maintaining the isotopic mix constant at that for LWR UNF (50,000 MWd/MTHM, 20 year cooled).

An actual waste form could contain a variety of the actinide radionuclides and the initial elemental and isotopic mix of radionuclides would likely vary from that of LWR UNF. For example, some radionuclides may be concentrated while others reduced within an actual waste form.

The fundamental reason for treating fission product radionuclides and actinide elements independently is in the application of dissolved concentration limits in the near-field. Dissolved concentration limits apply on an element-basis, rather than on a radionuclide basis. There will be "competition" among radionuclides of a given element when calculating near-field concentrations while invoking solubility constraints. Since individual fission product radionuclides are being evaluated rather than multiple radionuclides within a given element, they can each be treated independently.

Several radionuclides are important for actinide elements (uranium, neptunium, and plutonium) and there will be solubility "competition" among them. Treating actinide radionuclides individually would not allow for this "competition" and would result in increased dissolved concentrations, leading to higher near-field release rates and subsequent annual doses as compared to treating actinide elements. The consideration of actinide elements and varying the inventory on an element-basis appropriately allows for this solubility "competition" when calculating dissolved concentration limits. Varying the actinide inventory on an element-basis (0.001 to 10 times that of LWR UNF) is also consistent with potential used fuel recycling processes which could separate actinides on an element-basis, rather than a radionuclide basis (i.e., separations will recover X% of the uranium and Y% of the transuranic elements).
Thus, the approach utilized in this effort captures variations in the elemental mix between fission products, between fission products and actinide elements, and between actinide elements by treating them independently. The approach also captures the changing actinide element and isotopic mix within a single decay chain, but does not consider any effects across decay chains. The individual treatment of actinide elements neglects the cross-decay-chain "competition" of radionuclides when evaluating dissolved concentration limits. For example, the uranium daughter products from the decay of plutonium radionuclides do not "compete" with any uranium initially present. This leads to increased dissolved concentrations in the near-field and corresponding annual doses for radionuclides that are initially present and those that build-in due to radioactive decay of parent radionuclides.

These simplified GDSMs do not include any containment features of any other engineered barriers, such as waste packaging, and releases from waste form into the engineered barrier system begin immediately at a rate dictated by the waste form fractional degradation rate. The modeling approach used herein is conservative with respect to fission product and actinide-parent radionuclides in that release rates into the engineered barrier and natural systems and corresponding annual doses are maximized.

However, isolation of the waste by additional engineered barriers would allow actinide-daughter in-growth with the magnitude of the in-growth depending on the longevity of the additional engineered barriers. The modeling approach used herein accounts for actinide-daughter in-growth within the waste form as it degrades, but does not consider the effects of additional in-growth that would occur due to waste form isolation in other engineered barriers. Thus, actinide daughter product releases into the near field and corresponding annual doses would be increased.

Radiation exposure, or dose, is used as a metric of GDSM performance. Biosphere dose conversion factors developed in the International Atomic Energy Agency's (IAEA) BIOMASS project for a simple drinking water well pathway (ERB 1) were used [Ref. A-4]. The IAEA BIOMASS biosphere dose conversion factors were used for all GDSMs. The results presented in this report should not be construed as being indicative of the true performance of a disposal system or compared to any regulatory performance objectives regarding repository performance for the following reasons:

- The GDSMs are very simplistic and do not include many of the features, events, and processes that need to be considered in an assessment of disposal system performance.
- The determination of biosphere dose conversion factors does not depend on the generic disposal system environment, but rather on the biosphere beyond the generic disposal system environment, the habits of the population in that biosphere, and potentially the regulatory framework. A variety of biospheres and local populations could be present over a given generic disposal system environment and the resulting dose conversion factors may vary significantly.

Nevertheless, in lieu of a specific site, the reference biosphere allows for the assessment of post-closure safety indicators for a realistic biosphere for different generic disposal system environments.

A-2. Scenarios Considered

Post-closure safety indicator metrics are developed for two scenario classes. The first scenario class is undisturbed repository conditions. The engineered and natural barriers are assumed to not be affected by any disruptive events. Note that the waste isolation capabilities of waste packaging materials, a key isolation barrier, were not considered in the development of the post-closure safety indicators for the undisturbed scenario class.

The second scenario class is disturbed repository conditions. Under this scenario class, the disposal system can be disrupted by either be natural (i.e., seismic) or human induced events. However, defining the specific events that could potentially disrupt a geologic disposal system is both site- and design-specific. As such, it is not possible to define specific disruptive conditions to assess in the UFD GDSMs and any number of such events could be speculated.

The primary objective of evaluating disruptive scenarios is to assess the resilience of a disposal system to remain protective of public health and safety after the disruptive event occurs. However, disruptive events could potentially degrade the performance of both engineered and natural waste isolation barriers. Thus, in order to avoid undue speculation regarding specific disruptive events and scenarios and to test the resilience of a geologic repository, stylized representations that circumvent key isolation barriers were used to develop the safety indicators for disturbed scenario classes. These are described below. Again, note that the waste isolation capabilities of waste packaging materials, a key isolation barrier typically affected by disruptive scenarios, were not considered in the development of the post-closure safety indicators for the disturbed scenario class.

- Unsaturated Tuff – The primary consequence of disruptive events in a generic unsaturated tuff disposal system is the degradation of the waste packaging materials. However, since the development of the post-closure safety indicators does not consider the waste isolation capabilities of waste packaging materials, the safety indicator for the disruptive scenario class is assumed to be identical to that for the undisturbed scenario class. It is recognized that disruptive scenarios could alter the natural system of an unsaturated tuff environment.
- Salt – A stylized disruptive scenario has yet to be developed.
- Granite – A stylized disruptive scenario has yet to be developed.
- Clay – A stylized disruptive scenario has yet to be developed.
- Deep Borehole – A stylized disruptive scenario has yet to be developed.

A-2.1 Modeling Results

This section provides the data for each key radionuclide in each of the disposal environments needed to develop the post-closure safety metric shown in Equation 3 above, repeated as Equation A-2.

$$SI_{GD-PC_G} = \sum_{j=1}^{n} \log \left(\frac{\sum_{i=1}^{N} D_{G,i}(I_i, F_d)}{100\ mrem/yr} \right) \qquad \text{Eq. A-2}$$

Where:

$SI_{GD\text{-}PCG}$	=	Safety indicator for post-closure geologic disposal for media type G [GWe-yr^{-1}]
$D_{G,i}$	=	Parametric relation that captures disposal system performance for media G for a combination for radionuclide inventory and waste form fractional degradation rate (mrem/yr)
I_i	=	Inventory of key radionuclide i to be disposed of (kg/GWe-yr)
N	=	Number of key radionuclides important to post-closure safety
F_d	=	Fractional degradation rate of the waste form [yr^{-1}]
N	=	Number of waste categories i to be disposed of for a particular fuel cycle (i.e, different UNF types, HLW forms, etc)

The GDSMs [Ref. 12] will be used to develop the $D_{G,i}(I_i, F_d)$ response surfaces in Equation A-2 for key radionuclides for each geologic disposal media. This will be done by simulating the release of radionuclides from hypothetical waste forms with different combinations of key radionuclide inventory (I_i) and waste form fractional degradation rate (F_d) and the subsequent radionuclide transport through the engineered and natural barriers for the geologic disposal environments under consideration. Simulations will be conducted for generic geologic disposal systems in salt, clay/shale, and granite media and for deep borehole disposal in crystalline rock. The tables provided in this section for each generic disposal environment are the $f(I_i,F_d)$ relationship shown in Equation A-2.

A-2.1.1 Unsaturated Tuff

<u>Undisturbed Performance</u>

Disposal system models for generic tuff environments are not under development by the UFD Campaign. However, a model of a generic tuff environment was developed under the SWF Campaign in Fiscal Year 2009 [Ref. A-1]. That model was intended to support the development of waste form performance and utilized the same approach as described above. The results generated by that modeling effort are used to develop the post-closure safety indicators for a generic unsaturated tuff environment. The indicators for each key isotope for the undisturbed performance scenario class are shown in Table A-1.

Table A-1. Undisturbed Post-Closure Safety Indicators for Key Isotopes – Unsaturated Tuff

		Inventory Factor (IF)				
		0.001	0.01	0.1	1	10
		Tc-99 (NF=1.2 x 10^3 g)				
Degradation Rate (yr^{-1})	1.E-09	5.E-11	5.E-10	5.E-09	5.E-08	5.E-07
	1.E-08	5.E-10	5.E-09	5.E-08	5.E-07	5.E-06
	1.E-07	5.E-09	5.E-08	5.E-07	5.E-06	5.E-05
	1.E-06	5.E-08	5.E-07	5.E-06	5.E-05	5.E-04
	1.E-05	5.E-07	5.E-06	5.E-05	5.E-04	5.E-03
	1.E-04	4.E-06	4.E-05	4.E-04	4.E-03	4.E-02
	1.E-03	3.E-05	3.E-04	3.E-03	3.E-02	3.E-01
	1.E-02	2.E-04	2.E-03	2.E-02	2.E-01	2.E+00
		I-129 (NF=2.9 x 10^2 g)				
Degradation Rate (yr^{-1})	1.E-09	2.E-11	2.E-10	2.E-09	2.E-08	2.E-07
	1.E-08	2.E-10	2.E-09	2.E-08	2.E-07	2.E-06
	1.E-07	2.E-09	2.E-08	2.E-07	2.E-06	2.E-05
	1.E-06	2.E-08	2.E-07	2.E-06	2.E-05	2.E-04
	1.E-05	2.E-07	2.E-06	2.E-05	2.E-04	2.E-03
	1.E-04	2.E-06	2.E-05	2.E-04	2.E-03	2.E-02
	1.E-03	1.E-05	1.E-04	1.E-03	1.E-02	1.E-01
	1.E-02	8.E-05	8.E-04	8.E-03	8.E-02	8.E-01
		Cs-135 (NF=6.9 x 10^2 g)				
Degradation Rate (yr^{-1})	1.E-09	6.E-14	6.E-13	6.E-12	6.E-11	6.E-10
	1.E-08	6.E-13	6.E-12	6.E-11	6.E-10	6.E-09
	1.E-07	6.E-12	6.E-11	6.E-10	6.E-09	6.E-08
	1.E-06	5.E-11	5.E-10	5.E-09	5.E-08	5.E-07
	1.E-05	1.E-10	1.E-09	1.E-08	1.E-07	1.E-06
	1.E-04	2.E-10	2.E-09	2.E-08	2.E-07	2.E-06
	1.E-03	2.E-10	2.E-09	2.E-08	2.E-07	2.E-06
	1.E-02	2.E-10	2.E-09	2.E-08	2.E-07	2.E-06
		Se-79 (NF=4.5 x 10^1 g)				
Degradation Rate (yr^{-1})	1.E-09	2.E-12	2.E-11	2.E-10	2.E-09	2.E-08
	1.E-08	2.E-11	2.E-10	2.E-09	2.E-08	2.E-07
	1.E-07	2.E-10	2.E-09	2.E-08	2.E-07	2.E-06
	1.E-06	2.E-09	2.E-08	2.E-07	2.E-06	2.E-05
	1.E-05	2.E-08	2.E-07	2.E-06	2.E-05	2.E-04
	1.E-04	8.E-08	8.E-07	8.E-06	8.E-05	8.E-04
	1.E-03	2.E-07	2.E-06	2.E-05	2.E-04	2.E-03
	1.E-02	2.E-07	2.E-06	2.E-05	2.E-04	2.E-03

Determine Inventory Factor (IF) as: IF = M (g/GW-yr_e) / NF, where M is the mass of the radionuclide (per GW-yr_e) in the waste form and the normalization factor NF is given in the table. Logarithmic interpolation for calculated IF and waste form fractional degradation rate

Table A-1. Undisturbed Post-Closure Safety Indicators for Key Isotopes – Unsaturated Tuff (cont.)

		Inventory Factor (IF)				
		0.001	0.01	0.1	1	10
		Pd-107 (NF=3.8 x 10^2 g)				
Degradation Rate (yr^{-1})	1.E-09	3.E-14	3.E-13	3.E-12	3.E-11	3.E-10
	1.E-08	3.E-13	3.E-12	3.E-11	3.E-10	3.E-09
	1.E-07	3.E-12	3.E-11	3.E-10	3.E-09	3.E-08
	1.E-06	3.E-11	3.E-10	3.E-09	3.E-08	3.E-07
	1.E-05	3.E-10	3.E-09	3.E-08	3.E-07	3.E-06
	1.E-04	2.E-09	2.E-08	2.E-07	2.E-06	2.E-05
	1.E-03	2.E-08	2.E-07	2.E-06	2.E-05	2.E-04
	1.E-02	1.E-07	1.E-06	1.E-05	1.E-04	1.E-03
		Sn-126 (NF=4.5 x 10^1 g)				
Degradation Rate (yr^{-1})	1.E-09	3.E-15	3.E-14	3.E-13	3.E-12	3.E-11
	1.E-08	3.E-14	3.E-13	3.E-12	3.E-11	3.E-10
	1.E-07	3.E-13	3.E-12	3.E-11	3.E-10	3.E-09
	1.E-06	3.E-12	3.E-11	3.E-10	3.E-09	3.E-08
	1.E-05	2.E-11	2.E-10	2.E-09	2.E-08	1.E-07
	1.E-04	3.E-11	3.E-10	3.E-09	3.E-08	1.E-07
	1.E-03	4.E-11	4.E-10	4.E-09	3.E-08	1.E-07
	1.E-02	4.E-11	4.E-10	4.E-09	3.E-08	1.E-07
		Zr-93 and Daughter (NF=1.1 x 10^3 g)				
Degradation Rate (yr^{-1})	1.E-09	1.E-11	1.E-10	1.E-09	1.E-08	1.E-07
	1.E-08	1.E-10	1.E-09	1.E-08	1.E-07	1.E-06
	1.E-07	1.E-09	1.E-08	1.E-07	1.E-06	1.E-05
	1.E-06	1.E-08	1.E-07	1.E-06	1.E-05	1.E-04
	1.E-05	1.E-07	1.E-06	1.E-05	1.E-04	1.E-03
	1.E-04	1.E-06	1.E-05	1.E-04	1.E-03	1.E-02
	1.E-03	9.E-06	9.E-05	9.E-04	9.E-03	9.E-02
	1.E-02	6.E-05	6.E-04	6.E-03	6.E-02	6.E-01

Determine Inventory Factor (IF) as: IF = M (g/GW-yr_e) / NF, where M is the mass of the radionuclide (per GW-yr_e) in the waste form and the normalization factor NF is given in the table. Logarithmic interpolation for calculated IF and waste form fractional degradation rate

Table A-1. Undisturbed Post-Closure Safety Indicators for Key Isotopes – Unsaturated Tuff (cont.)

		Inventory Factor (IF)				
		0.001	0.01	0.1	1	10
		Np-237 and Daughters (NF=6.5 x 10^2 g)				
Degradation Rate (yr^{-1})	1.E-09	3.E-10	3.E-09	3.E-08	3.E-07	3.E-06
	1.E-08	3.E-09	3.E-08	3.E-07	3.E-06	3.E-05
	1.E-07	3.E-08	3.E-07	3.E-06	3.E-05	3.E-04
	1.E-06	2.E-07	2.E-06	2.E-05	2.E-04	2.E-03
	1.E-05	2.E-06	2.E-05	2.E-04	2.E-03	1.E-02
	1.E-04	1.E-05	1.E-04	1.E-03	9.E-03	3.E-02
	1.E-03	4.E-05	4.E-04	3.E-03	1.E-02	3.E-02
	1.E-02	5.E-05	5.E-04	4.E-03	2.E-02	3.E-02
		Pu and Daughters (NF=4.9 x 10^3 g)				
Degradation Rate (yr^{-1})	1.E-09	3.E-09	3.E-08	3.E-07	3.E-06	3.E-05
	1.E-08	3.E-08	3.E-07	3.E-06	3.E-05	3.E-04
	1.E-07	3.E-07	3.E-06	3.E-05	3.E-04	3.E-03
	1.E-06	3.E-06	3.E-05	3.E-04	3.E-03	1.E-02
	1.E-05	3.E-05	3.E-04	3.E-03	1.E-02	5.E-02
	1.E-04	2.E-04	2.E-03	6.E-03	2.E-02	1.E-01
	1.E-03	4.E-04	2.E-03	6.E-03	3.E-02	2.E-01
	1.E-02	5.E-04	2.E-03	6.E-03	3.E-02	2.E-01
		Am and Daughters (NF=2.1 x 10^3 g)				
Degradation Rate (yr^{-1})	1.E-09	8.E-10	8.E-09	8.E-08	8.E-07	8.E-06
	1.E-08	8.E-09	8.E-08	8.E-07	8.E-06	8.E-05
	1.E-07	8.E-08	8.E-07	8.E-06	8.E-05	8.E-04
	1.E-06	7.E-07	7.E-06	7.E-05	7.E-04	7.E-03
	1.E-05	5.E-06	5.E-05	5.E-04	5.E-03	3.E-02
	1.E-04	3.E-05	3.E-04	3.E-03	2.E-02	6.E-02
	1.E-03	1.E-04	1.E-03	8.E-03	3.E-02	7.E-02
	1.E-02	1.E-04	1.E-03	9.E-03	3.E-02	7.E-02
		U and Daughters (NF=9.8 x 10^5 g)				
Degradation Rate (yr^{-1})	1.E-09	5.E-10	5.E-09	5.E-08	5.E-07	5.E-06
	1.E-08	5.E-09	5.E-08	5.E-07	5.E-06	5.E-05
	1.E-07	5.E-08	5.E-07	5.E-06	5.E-05	5.E-04
	1.E-06	5.E-07	5.E-06	5.E-05	5.E-04	4.E-03
	1.E-05	3.E-06	3.E-05	3.E-04	3.E-03	2.E-02
	1.E-04	1.E-05	1.E-04	1.E-03	9.E-03	2.E-02
	1.E-03	3.E-05	3.E-04	2.E-03	1.E-02	2.E-02
	1.E-02	3.E-05	3.E-04	2.E-03	1.E-02	2.E-02

Determine Inventory Factor (IF) as: IF = M (g/GW-yr_e) / NF, where M is the mass of the radionuclide (per GW-yr_e) in the waste form and the normalization factor NF is given in the table. Logarithmic interpolation for calculated IF and waste form fractional degradation rate

Disturbed Performance

As discussed above, the indicators for each key isotope for the disturbed performance scenario class are identical to those as undisturbed performance scenario class and are shown in Table A-1.

A-3. References

[A-1] W. Nutt, E. Morris, Y. Wang, J. Lee, C. Jove-Colon, S. Chu, Generic Repository Concept Analyses to Support the Establishment of Waste Form Performance Requirements – Generic Tuff and Salt Model Development and Results, GNEP-WAST-PMO-MI-DV-2008-000146, July 2009.

[A-2] E. Hardin, J. Blink, H. Greenberg, M. Sutton, M. Fratoni, J. Carter, M. Dupont, R. Howard, Generic Repository Design Concepts and Thermal Analysis (FY11), FCRD-USED-2011-000143, Rev. 0, August 2011.

[A-3] D. Clayton, G. Freeze, T. Hadgu, E. Hardin, J. Lee, J. Prouty, R. Rogers, W. Nutt, J. Birkholzer, H. Liu, L. Zheng, and S. Chu, Generic Disposal System Modeling – Fiscal Year 2011 Progress Report, FCRD-USED-2011-000184, August 2011.

[A-4] IAEA (International Atomic Energy Agency), 2003, “Reference Biospheres” for Solid Radioactive Waste Disposal, IAEA-BIOMASS-6, July 2003.