Global Nuclear Energy Partnership Integrated Waste Management Strategy

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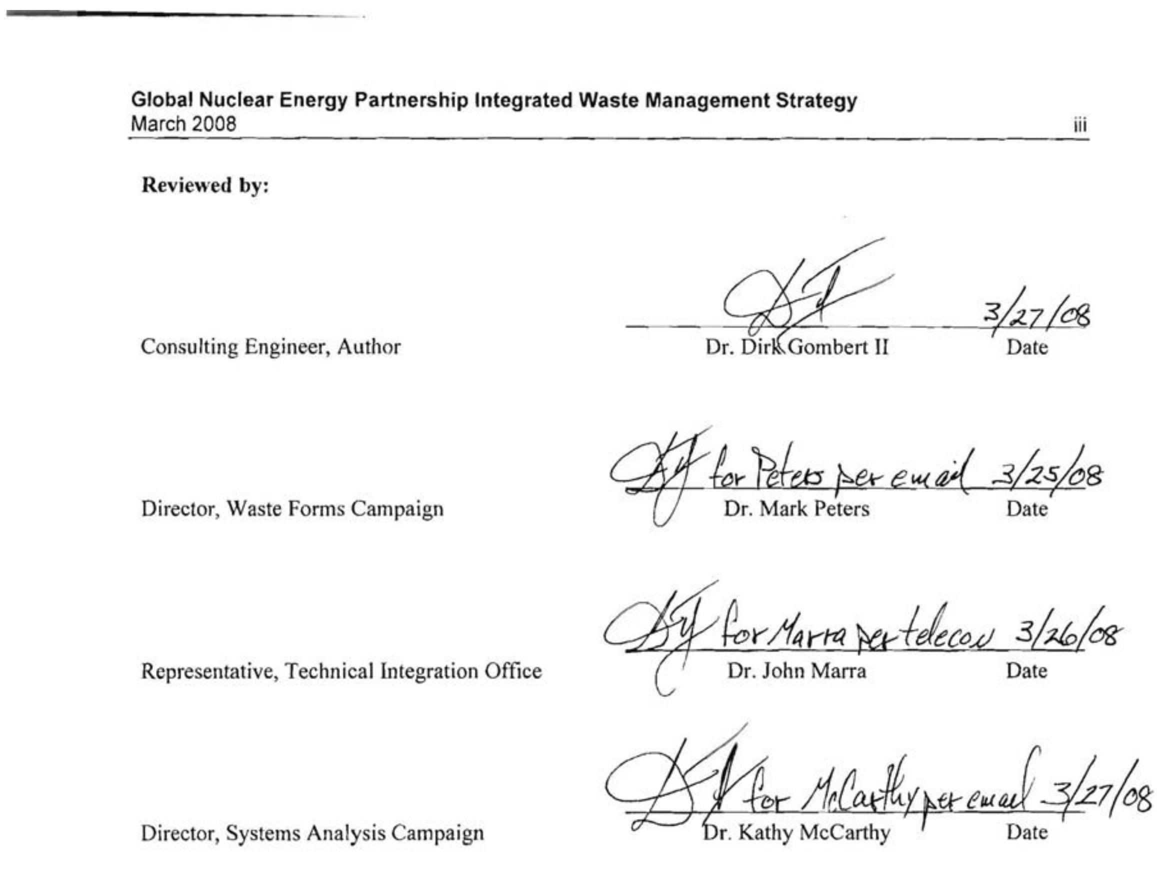

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EXECUTIVE SUMMARY

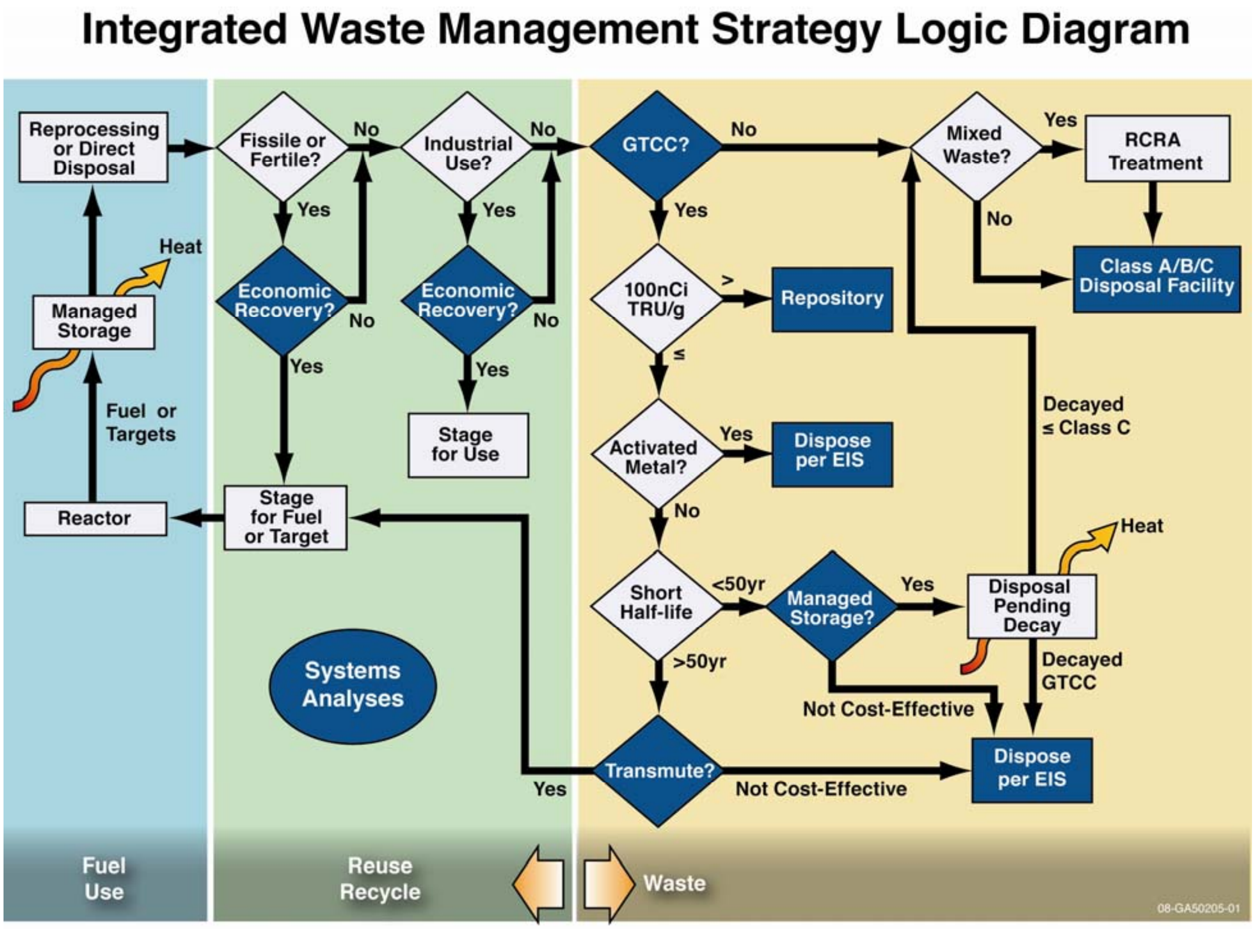
Key to successful implementation and expansion of nuclear energy in the United States is establishing an effective waste management system to allow routine disposition of spent fuel and all byproduct and waste streams. While it would be possible to manage and dispose wastes that would be generated from the advanced separations proposed in the Global Nuclear Energy Partnership (GNEP) program under a policy and regulatory framework that is essentially identical to what is currently in place, a more optimal approach exists that, if implemented, could result in a more efficient system for managing and disposing radioactive materials. An Integrated Waste Management Strategy (IWMS) is described here to manage and disposition all wastes from any radioactive operations, including, but not limited to nuclear fuel reprocessing. The IWMS was developed considering the need for a sustainable nuclear fuel cycle effectively integrating waste management in a manner that can be commercialized and eventually adapted to be used internationally.

The GNEP Program proposes a combination of thermal and fast reactors, supported by fuel recycling and the use of recovered transuranic materials in nuclear fuel. This comes close to closing the fuel cycle and improves the system for managing and disposing of radioactive wastes derived from electricity generation. As an example, removing actinides and key heat-generating fission products from the waste could permit increased waste loading densities within a geologic repository, allowing for increased disposal capacity in a facility of the same size. As long as the primary function of the repository is not impacted, i.e. controlling the potential radiation dose to the public, the same facility could hold the waste from generating more electricity. This is desirable to support expanded use of nuclear power and sharing the benefits of nuclear energy with the international community while maintaining proliferation resistance and safe waste disposal.

This IWMS does not establish a program or propose policy. Rather, it provides a logical basis for radioactive waste disposal on which potential changes to current programs and policies can be formulated and evaluated. The current version of the Nuclear Waste Policy Act (NWPA) could be interpreted to include any and all wastes from fuel reprocessing that are highly radioactive as high level wastes requiring repository disposal regardless of the magnitude or duration of the risks they pose. All wastes generated by the processes being developed under GNEP can be disposed under the current regulations, however, significant efficiencies can be realized if the partitioned wastes can be dispositioned based on their characteristics rather than their origin. This IWMS is based on the premise that the NWPA can be clarified or amended to use a graded scale of waste management considering the actual risks of specific waste streams similar to the structure of chemical waste regulations currently accepted by the public. Such an alternative approach would allow for more efficient, and cost-effective, management and disposal of nuclear waste while safely protecting the public from unacceptable risk.

A new generation of fuel reprocessing, including both the UREX+ family of aqueous processes and advanced electrochemical (Echem) separations now in development under the auspices of the DOE Office of Nuclear Energy (NE), separates fuel into several fractions, thereby partitioning the wastes into groups of common chemistry and to a great extent, risk. Advanced separations: 1) make possible recycling of long-lived hazardous elements as nuclear fuel so they can be transmuted into shorter-lived wastes, 2) allow greater flexibility in managing the individual waste streams based on the duration, type, and magnitude of risk, and 3) enable development of specialized waste forms to more effectively immobilize groups of radionuclides per this waste management strategy. Partitioning the wastes could allow more effective management for these materials under provisions similar to those of 10 CFR 61, which establishes limits for disposal based on the concentrations of long and short half-life species: essentially a risk management based approach.

The simplest way to summarize the IWMS is by reference to the following logic schematic showing how any radioactive material could be dispositioned.



Though designed with GNEP advanced separations in mind, this strategy could also be used to effectively disposition any radioactive materials from current reprocessing technologies similar to those currently used in France, Japan, and the United Kingdom and could still be used with future transition to the GNEP systems.

Discard radioactive materials (represented here as radioactive fuel reprocessing streams) enter the schematic from the left in the first panel. In the middle panel, the materials are considered for potential recycle or other beneficial use. The economic recovery must consider not only the value of the materials in the market, but the value to society of avoiding the liabilities of waste disposal. If recovery cannot be justified, then the material is designated waste (right panel) and classified based on risk similar to the provisions of 10 CFR 61 Licensing Requirements for Land Disposal of Radioactive Waste which is already ingrained in the U.S. market. With some expansion of the lists of long and short-lived radionuclides considered in 10 CFR 61 to make them more comprehensive, all wastes can be assigned disposition paths pertinent to the risks they pose.

The IWMS institutionalizes and exploits the unique nature of radioactive wastes; they decay to stable elements. Unlike hazardous chemical wastes such as toxic metals that pose a health risk indefinitely, radioisotopes decay, and the more intensely radioactive they are, the sooner they decay to stable elements. The IWMS integrates three waste management concepts proposed under GNEP:

- Recycling useful fissile elements as fuel,

- Destroying long-lived actinides by transmutation, and
- Storing short-lived species in engineered facilities to allow them to decay to innocuous levels while dissipating heat in a safe and secure manner.

Thus, the IWMS proposes an approach that is consistent with the current regulatory framework established in 10 CFR 61 and considers characteristics intrinsic to each radioactive waste to address risk in the most efficient way. Wastes would no longer be classified by point of origin, but would be dispositioned based on potential health and environmental risk. Emphasis is also placed on energy recovery, recycle, and beneficial reuse, concepts that are clearly important to a sustainable energy future. In addition, this regulatory development would build on the best aspects of U.S. and international (for example the International Atomic Energy Agency) guidance waste regulation, and bring both into closer congruence in support of a global nuclear energy program with cooperative waste management.

Collaborative efforts by the Department of Energy Offices of Nuclear Energy, Environmental Management, and Civilian Radioactive Waste Management are needed to implement this Integrated Waste Management Strategy. It is also essential to promote close collaboration amongst the DOE, the Nuclear Regulatory Agency and the Environmental Protection Agency to ensure that waste management policies and regulations develop in a complementary fashion to support efficient routine waste management. This will allow commercial operations to design facilities with known waste liabilities and plan investments accordingly. Finally, this type of regulatory basis, that covers all radioactive materials from any sources, serves as a useful foundation for globalizing the benefits of nuclear energy while ensuring a safe management structure for ultimate disposition of wastes.

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ACRONYMS

ANL	Argonne National Laboratory
BNFP	Barnwell Nuclear Fuel Plant
BSG	borosilicate glass
BWR	boiling water reactor
CCD/PEG	chlorinated cobalt dicarbollide-polyethylene glycol
CFR	code of federal regulations
CO_2	carbon dioxide
CWF	ceramic waste form
D&D	decommissioning and demolition
DOE	United States Department of Energy
DU	depleted uranium
Echem	Electrochemical processing also called pyrochemical processing
EIS	Environmental Impact Statement
EM	Office of Environmental Management (a Program Office in the United States Department of Energy)
ENS	Enhanced Near Surface
EPA	United States Environmental Protection Agency
ERL	economic recovery limit
FP	fission products
FPEX	fission product extraction
GNEP	Global Nuclear Energy Partnership
GTCC	greater than Class C
GWd	gigawatt-days
HEU	highly enriched uranium
HLW	high-level waste
HTO	Tritiated water (hydrogen-tritium oxide, or water containing tritium)
INL	Idaho National Laboratory
IWMS	Integrated Waste Management Strategy
LLW	low-level waste (also LLW-SLB or LLW in Class A, B, or C that is acceptable for SLB)
LWR	light water reactor
MOX	mixed U-Pu oxide (fuel)
MTHM	metric tons of heavy metal
MTIHM	metric tons of initial heavy metal (actinide metals prior to irradiation)

MWF	metal waste form
nCi	nano-curie, one billionth of a curie
NE	Office of Nuclear Energy (a Program Office in the United States Department of Energy)
NRC	United States Nuclear Regulatory Commission
PPE	personal protective equipment
PWR	pressurized water reactor
R&D	research and development
RCRA	Resource Conservation and Recovery Act
RW	Office of Civilian Radioactive Waste Management (Program Office in the United States Department of Energy)
SLB	shallow land burial
SNF	spent nuclear fuel
SNL	Sandia National Laboratory
SRNL	Savannah River National Laboratory
SS	stainless steel
TALSPEAK	Trivalent Actinide Lanthanide Separations by Phosphorus-reagent Extraction from Aqueous Komplexes
TRU	transuranic
TRUEX	transuranic extraction
UDS	undissolved solids
UREX	uranium extraction (an aqueous separations process or in the case of variations UREX+, UREX+1a, etc., a family of aqueous separations processes)
WIPP	Waste Isolation Pilot Plant
YMF	Yucca Mountain Facility

CAMPAIGN/PROGRAM
GLOBAL NUCLEAR ENERGY PARTNERSHIP INTEGRATED WASTE MANAGEMENT STRATEGY

1. INTRODUCTION

The GNEP fuel-cycle concept using a combination of aqueous (UREX+) and electrochemical (Echem) separations makes it possible to tailor waste forms to specific wastes and to expand the capacity of a single repository to last for generations. A credible strategy for managing radioactive wastes from any future nuclear fuel-cycle must provide acceptable disposition paths for all wastes regardless of reactor technology, fuel reprocessing scheme(s), and/or the degree of fuel-cycle closure. As shown in the fuel cycle depicted in Figure 1, Fuel Processing (separations) and Fuel Fabrication have important interfaces with Material Recycle/Disposition and they act together in closing the fuel cycle. Integrating the strategy into the fuel cycle depends on continued analyses of the interfaces to optimize fuel design, separations, and reuse-recycle options with consideration of treatment, storage, and disposal systems for all wastes.

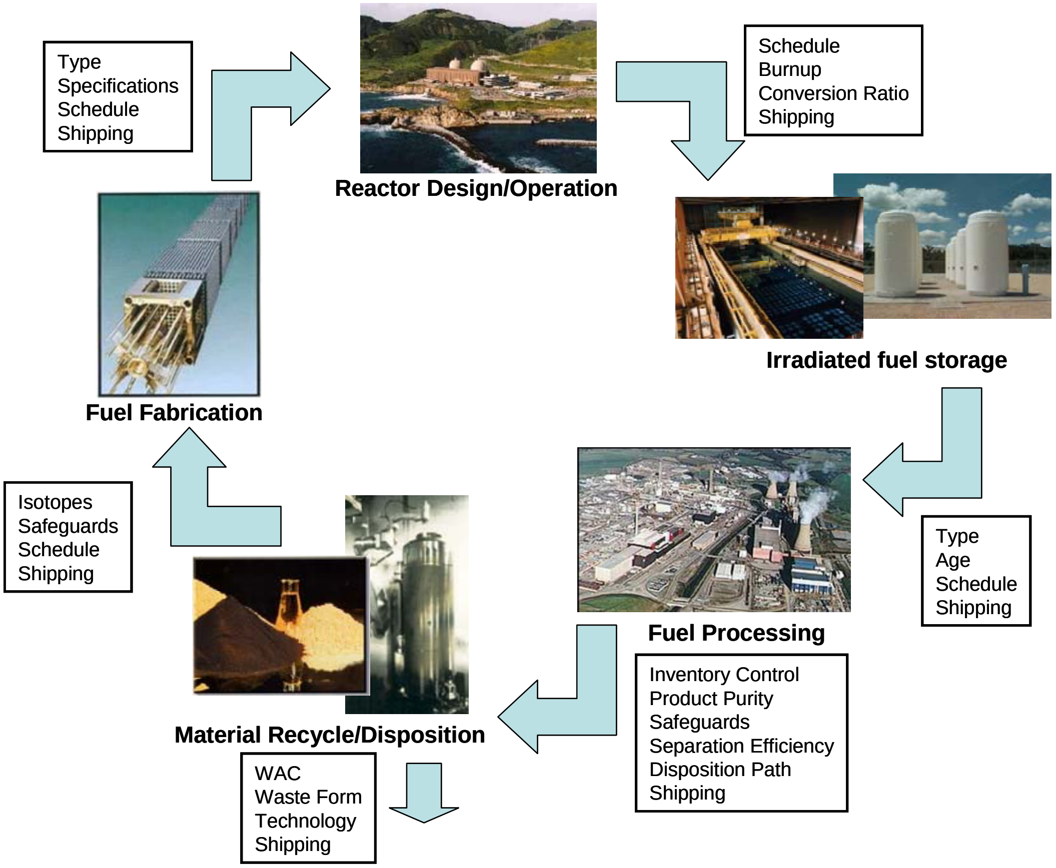

Figure 1. A closed nuclear fuel cycle showing integration of material recycle and disposition.

Thus, an integrated waste management strategy (IWMS) is critical to the success of the entire fuel cycle. The IWMS should provide guidance to optimize waste management aspects of the fuel-cycle to provide a safe, secure, and cost-effective practicable system to support advanced nuclear fuel fabrication and waste/storage form(s). The strategy proposed here is based on U. S. regulations. Application to another country or more broadly internationally will require adaptation potentially based on some negotiation. The strategy identifies waste management limitations in current U.S. regulations and describes the potential advantages of change.

No regulatory changes are necessary to dispose of wastes from a closed fuel cycle, allow fuel recycling or to implement the advanced separations proposed under GNEP. Advanced waste forms can also be used to reduce the volume of material that must be transported and disposed. With fuel recycling, waste management would be at least as effective as it is today, and quite possibly more effective regardless of any regulatory changes affecting waste disposition. However, the benefits of closing the fuel cycle and using advanced separations to partition what has been historically classified as high level waste (HLW) were not completely envisioned when current regulations were promulgated and the regulations may limit the benefits of these new technologies unless some changes are considered. The IWMS described here is purely to show how regulations and policy could be adapted to make the system more efficient while being equally protective. Cost and liability of responsible radioactive waste disposition to ensure the safety of the public are important considerations in future use of nuclear power. Key factors in optimizing the system that are discussed here as background to any potential future strategy include: 1) legislated limits on repository capacity, 2) 2) regulatory definition of what is considered HLW, and 3) the technical limits on capacity of a geologic repository.

2. BACKGROUND

2.1 Nuclear Waste Policy Act[1]

The management and disposal of spent nuclear fuel and high level nuclear waste is governed by the Nuclear Waste Policy Act (NWPA), as amended. Under the NWPA, the term HLW is defined as:

> “(A) the highly radioactive material resulting from the reprocessing of spent nuclear fuel, including liquid waste produced directly in reprocessing and any solid material derived from such liquid waste that contains fission products in sufficient concentrations; and (B) other highly radioactive material that the Commission, consistent with existing law, determines by rule requires permanent isolation.”

Further, the NWPA goes on to define a capacity limit on the permanent isolation (repository):

> “The Commission decision approving the first such application shall prohibit the emplacement in the first repository of a quantity of spent fuel containing in excess of 70,000 metric tons of heavy metal or a quantity of solidified high-level radioactive waste resulting from the reprocessing of such a quantity of spent fuel until such time as a second repository is in operation.”

Considering the capacity limit first, it is clear that technology and events have overcome our best intentions that established the legislative limit of 70,000 metric tons of heavy metal (MTHM) for the first repository. Although it was well known that the U.S. would eventually need greater repository capacity than 70,000 MTHM, it was likely presumed that a first repository would be in operation before the material in storage and use exceeded the capacity and a second repository was needed. The first generation of nuclear power plants in the U.S. are now achieving 90% capacity factors and most of them are expected to receive license extensions allowing continued operation well into this century. Disposal of the fuel currently in storage and that already charged to operating reactors will exceed the legislative limit and necessitates a second repository. Thus action is needed regardless of what fuel-cycle is in effect.

The current repository capacity is based on the initial fuel charged to the reactor and is independent of the type, mass, or characteristics of the waste. Thus, even if fuel reprocessing is implemented in this country and all of the uranium and plutonium is recycled, the remaining material “resulting from the reprocessing," even though it represents only a few per cent of the initial fuel mass, will still exceed the repository capacity. This is due to the wording of the Act that describes the capacity as 70,000 MTHM or "a quantity of solidified high-level radioactive waste resulting from the reprocessing of such a quantity of spent fuel." Amendments to the Act to modify the capacity limit have been proposed for consideration in Congress,[2,3,4] but are not currently under consideration. Thus, the first repository is limited to disposal of material already in the system, and any future energy scenario including a significant contribution from nuclear power will likely necessitate several additional repositories unless a strategy is adopted to make more efficient use of geologic disposal. If the legislated capacity limit is increased this strategy can be used to maximize use of the first repository. If the capacity limit is not changed, this strategy would still be appropriate to maximize use of the next repository.

This IWMS suggests approaches to increase the allowed technical design disposal capacity by incorporating:

- Recycling long-lived actinides in reactor fuel to eliminate the long-term potential dose consequences and the heat of their decay,
- Developing highly durable tailored waste forms based on waste chemistry to increase waste loading to reduce volume while maintaining or reducing release rates over geologic time
- Integrating SNF storage, recycling, and waste staging prior to disposition to control the heat input from relatively short-lived fission products.

While evaluating the current regulatory structure in light of current technology, perhaps it would be wise to also consider the definition of HLW. As defined in the NWPA, HLW is source based. Regulations enacted by the U.S. Environmental Protection Agency (EPA) and the U.S. Nuclear Regulatory Commission (NRC) for the geologic disposal of SNF and HLW include either the same or similar definitions, and all share the concept that HLW is derived from reprocessing SNF. In effect, the current NWPA could be interpreted to classify any and all "highly radioactive" materials derived from fuel reprocessing to be HLW regardless of its radioactivity. As written, HLW could equally include:

- ^{3}H and ^{129}I, both biologically active elements, gases in their pure forms, with half-lives at both extremes at 12.3 years and 16 million years respectively
- ^{239}Pu, and ^{241}Am alpha emitters with half-lives of 24 thousand and 430 years respectively
- ^{137}Cs, a readily dispersed soluble ion and ^{90}Sr, a sparingly soluble alkaline earth element that deposits in bone if ingested, with half-lives of 30 and 29 years respectively
- Any other process solutions and residues from fuel reprocessing that can be described as "highly radioactive" materials.

When signed, this legislation was conceived to safeguard the public from the wastes derived from fuel reprocessing as practiced at that time and directed all HLW to geologic disposal. The PUREX process, designed in the 1940s for plutonium and uranium extraction primarily for military purposes, left all of the other radioactive species behind with processing chemicals as waste. Faced with developing protective constraints for a mixture of species representing a broad range of risks, legislators had no real alternative than to lump everything together assigning protection for the greatest risk, hence, the all encompassing definition of HLW to be disposed in a repository. PUREX and similar approaches still used world-wide to separate U and Pu and potentially other actinides will continue to generate HLW that still results in a complex mixture of safety and health risks. In addition, fuel cladding and activated metal fuel hardware, which are highly radioactive and derived from fuel reprocessing, could also be considered as HLW, as they are in France.

While it would be possible to continue to manage and dispose wastes that would be generated under the GNEP under a policy and regulatory framework that is essentially identical to what is currently in place, a more optimized approach exists that would allow similar protection while making more efficient use of repository space. For example, in 1994 the IAEA revised their proposed waste classifications and HLW was defined based on radionuclide content and heat generation rather than source;[5] no reference to reprocessing was included in the definition. Such an alternative approach may allow for more efficient and cost-effective management and disposal of nuclear waste while safely protecting the public from unacceptable risk. Bringing the U.S. waste classifications into closer congruence with IAEA policy is attractive in itself to developing international markets in nuclear power. Taking this one step further, this IWMS suggests building on the knowledge gained over the last 30 years and designing waste management to allow radionuclides to decay and to dissipate the heat prior to geologic disposal, to optimize use of repository space.

A new generation of fuel reprocessing, as represented by the UREX+ family of processes now in development under the DOE Office of Nuclear Energy (NE), separates fuel into several fractions, thereby partitioning the wastes into groups of common chemistry and to a great extent, similar risk. Advanced separations allow greater flexibility in managing the individual waste streams based on the duration, type, and magnitude of the risk they pose, and an opportunity to develop specialized waste forms to more effectively immobilize radionuclides effectively as described in this waste management strategy.

Partitioning the wastes allows for more effective management of these materials under provisions similar to those of 10 CFR 61, which establishes limits for disposal based on the concentrations of long and short half-life species, essentially a risk management based approach. Class A/B/C wastes can be interred in surface disposal sites with graduated levels of protection applied to waste form, packaging and placement. Disposal requirements for wastes contaminated with concentrations of certain species at limits greater than Class C (GTCC) is defined in 10 CFR 61 as follows:

> "Waste that is not generally acceptable for near-surface disposal is waste for which form and disposal methods must be different, and in general more stringent, than those specified for Class C waste. In the absence of specific requirements in this part, such waste must be disposed of in a geologic repository as defined in part 60 or 63 of this chapter unless proposals for disposal of such waste in a disposal site licensed pursuant to this part are approved by the Commission."

Basically, this provision for GTCC wastes could obviate the need for a dedicated source-based definition for HLW. Instead, a risk-based approach would be used for all nuclear waste regardless of source. A graded approach to disposal requirements for Class A, B, C and GTCC type waste consistent with that described in 10 CFR 61 could be applied, the top level of which will be repository disposal. In 1981 the International Atomic Energy Agency proposed classifying wastes into one of three classes: high, intermediate and low level wastes,[6] and current U.S. waste classifications/definitions match very closely with the classification proposed by the IAEA. In particular, the definition of GTCC LLW matches well with the IAEA classification of intermediate level waste. The U.S. LLRW Class A, B, and C wastes also match classifications according to acceptable levels of radioactivity during the time which administrative controls can be expected to last. Updating this methodology with the 1994 definition suggested by IAEA would eliminate the source-based HLW definition. Taking this a step further, one can see that the classification of HLW serves little purpose in light of advanced separations that partition the historic fuel reprocessing wastes into streams of similar chemistry and radiological risk that can be managed on a graded scale.

Eliminating the source-based HLW classification is not essential to implementing GNEP or this IWMS, but it enables a more flexible management approach with no loss of safety considerations for managing the fractionated wastes separately as dictated by their respective characteristics rather than their point of origin. It would also allow more practicable options to be considered for management of fuel reprocessing wastes rather than the prescriptive regulations for HLW.

This one change would also allow the NRC/DOE to exploit the advantages of new technical developments in waste processing and disposal. Currently, the baseline waste form for HLW in the U. S. is borosilicate glass containing waste elements as oxides. Unfortunately several fission product elements have limited solubility in glass, which results in low waste loading and requires production of more glass. Matching the waste form to the target waste stream(s) chemistry allows the disposal system to achieve more optimum waste loading with comparable or improved performance. Not only can the waste form be matched to the waste, but the disposal environment could also be matched. Some elements are more stable in a low oxygen reducing environment, while others are more stable as oxides. The simplest efficient combination of waste forms may be an oxidized form such as glass or ceramic for readily oxidized elements and a metallic form for readily reduced species. Thus, a more efficient waste management system that can make the most effective use of advanced waste forms and disposal design for each waste is enabled by this proposed change in legislation without sacrificing any of the protection provided through the current law.

2.2 GTCC Environmental Impact Statement[7]

The Greater-Than-Class C Low-Level Radioactive Waste Environmental Impact Statement (GTCC LLW EIS) is being prepared by the DOE Office of Environmental Management (EM) pursuant to the Low-Level Radioactive Waste Policy Amendments Act of 1985 (LLRWPAA) and the National Environmental Policy Act of 1969. Section 3(b)(1)(D) of the LLRWPAA assigned the Federal Government (DOE) responsibility for the disposal of GTCC LLW that results from NRC and Agreement State licensed activities. In the EIS, currently being drafted, several types of GTCC waste are being considered that are similar to the wastes that would be produced from fuel reprocessing using advanced separations:

1. Sealed sources containing concentrated ^{137}Cs and ^{241}Am
2. Activated metals primarily from decommissioning nuclear reactors
3. Other wastes including debris and contaminated equipment.

The EIS is considering a range of disposal options including:

1. Geologic disposal similar to the defense transuranic (TRU) waste repository, the Waste Isolation Pilot Plant (WIPP) in New Mexico and the proposed HLW Yucca Mountain Facility (YMF) in Nevada
2. Enhanced Near Surface (ENS) disposal within 30 meters of the surface using barriers and special waste forms and packaging
3. Intermediate depth boreholes deeper than 30 meters, filled with waste up to 30 meters of the surface and backfilled.

The GTCC EIS is currently considering an inventory of 5,600 cubic meters of commercial GTCC wastes and DOE “GTCC-like wastes”, about 18% of which is currently stored, with the balance to be generated by 2062 with the decommissioning of commercial power plants. The EIS also considers the potential wastes estimated for future fuel reprocessing. This forward looking effort could provide an excellent basis for safe disposition of materials from fuel reprocessing. Some combination of the disposal options could be used to dispose of any of the fuel reprocessing wastes.

2.3 Repository Capacity

The overarching factor that must be considered in developing a repository for disposing radioactive waste is public safety. A repository must effectively contain radioactive wastes and be sited and designed such that any potential releases of radioactive materials would not lead to the public being exposed to harmful levels of radiation. Estimates of potential dose to a receptor are based on radiation exposure scenarios theorized for a repository during operation and after closure. Under this primary functional requirement are various design technical requirements that are affected by site-specific conditions.

The size of an available host rock formation is the most obvious constraint in optimizing a repository. A small site could limit the capacity for waste based purely on waste volume. Advanced waste forms designed to maximize waste loading would then be critical to maximizing the waste that can be emplaced. Temperature is also one of several design constraints evaluated by the DOE Office of Civilian Radioactive Waste Management (RW) to optimize the repository design while meeting the primary goal of limiting potential dose.[8] Operating temperature limits are placed on the repository to confine performance modeling within known hydrothermal conditions to increase confidence in the predicted long term performance. Design issues including groundwater drainage, fuel cladding and package effects, and host-rock degradation lead to constraints on the heat output of waste packages. Thermal modeling using parameters such as fuel burnup and age, natural characteristics of the host rock formation and forced ventilation is used to predict performance. These models can then be used to optimize a loading configuration of waste packages and emplacement drift spacing, or a “baseline capacity” of the repository. Modeling to date indicates it is likely that thermal limits will be more constraining than volume requirements. The availability of a large area would allow wastes to be placed as far apart as needed to control temperature regardless of heat generation. For example, thermal constraints could be met by placing the wastes very far apart. The optimization of a design of a repository with a host rock having a very large area would likely be constrained only by the costs of site characterization, mining, ground support.

Additional modeling done outside the design effort sponsored by RW for licensing the YMF has considered the potential for increasing the baseline capacity by taking advantage of advanced separations to partition wastes having different rates of heat generation.[9,10] The modeling would vary with a different site, different geologic formation type, and different design specifications, but similar trends are expected using a design approach similar to the current YMF baseline. This modeling indicates that heat produced by the long-term decay of the transuranic actinides, primarily plutonium and americium, is the primary thermal load limiting repository capacity. Hypothetically, if the Pu/Am could be removed from the waste, the next most limiting thermal load is caused by the relatively rapid decay of Cs/Sr over the first few hundred years. The analyses take this an additional step, and if the Pu/Am and the Cs/Sr are removed from the repository to a sufficient extent, the next limiting heat source is another minor actinide (MA), Cm. Based on these thermal analyses, highly efficient (99.9%) separation and removal of these elements from the waste placed in the repository could potentially increase the capacity of a repository at Yucca Mountain over 100 fold when considering thermal aspects alone. This would allow continued use of the one repository well beyond the 21st century should a decision be made to utilize that repository for disposing wastes from the GNEP. Again, these results are expected to be similar for a repository located in any geologic media and very high repository loading could be achieved. This establishes the basis for a sustainable waste management component of a closed fuel cycle that involves transuranic recycle and management of heat producing fission products.

Thus, in addition to the primary design requirement of controlling potential dose to the public, repository design capacity for waste disposal may be limited by:

1. Legislative constraints

2. Radionuclide release which is controlled by the inventory, durability of waste forms supplemented by engineered barriers and migration through the geologic media
3. Volume of the waste and available size of the host rock
4. Heat generation and temperature limits due to the geology.

To maximize the potential benefits to repository capacity made possible by advanced separations, an IWMS must identify new ways to disposition separated materials to optimize within the potential limits on the repository design. The legislative constraint and the potential for revision were discussed above. Advanced waste forms are being tailored to waste chemistry to minimize potential for releases and specifically developed to reduce volume. Also, recycling useful elements and managing wastes to allow decay is a significant factor in the IWMS to control heat.

As wastes, the actinides represent a very long-term dose risk due to alpha decay and spontaneous fission which releases neutrons. They also release heat that gradually builds up in a repository over very long times. Fortunately, the actinides are all either fissile or fertile, having different characteristics depending on the energy of an incident neutron. This makes them potentially useful for fuel in thermal or fast reactors. If not economically useful for electricity generation, they can at least be destroyed in targets placed in a reactor for that purpose. Key to this management strategy is cost-effective separation of the actinides from the spent fuel and fuel/target fabrication for destruction. To maximize repository benefits, efficient separation and recovery during fuel reprocessing is obviously essential, but recovery from other process wastes and equipment must also be maximized and any losses during recycling must be minimized. The other key to implementing this strategy is fabricating and qualifying a family of fuels and/or targets containing the range of actinide mixtures to be expected from repeated recycling. Fabrication of U/TRU fuel and/or targets differs from UO_2 and MOX fuel fabrication because remote operations are required due to significant gamma and neutron radiation.

Relatively short lived radionuclides including ^{137}Cs and ^{90}Sr represent a much shorter term hazard. While designing a repository for many thousands of years of performance has had varying degrees of acceptance and is challenging to prove, engineering design for hundreds of years is represented by examples well known in the international community. Cathedrals and monuments lasting hundreds to thousands of years showcase the engineering knowledge available at the time they were designed. If stabilized in a suitable waste form, radioactive elements could be packaged and stored in a secure manner designed to dissipate the decay heat until such time that heat did not limit disposal. The Cs/Sr could be separated and stabilized in a dedicated waste form or managed with the other fission products in a combined waste form to effectively manage the short-term heat. The Cs/Sr bearing waste could be disposed in a sacrificial part of the repository, or if separated in a pure enough form (<100nCi/g TRU) the Cs/Sr waste could potentially be disposed as Class A/B/C waste after sufficient decay. No disposal limit has been established for ^{135}Cs, so the risk posed by this nuclide at the concentration in the waste form would have to be considered in the ultimate disposal analysis.

Separating the long lived actinides from the shorter lived fission products allows destruction and elimination of the former and management of the latter within the range of engineering design experience. This strategy depends on advanced separations and discipline to implement and sustain a policy of actinide recycle and fission product management in the future. In the event that either commitment becomes unacceptable in the future, the benefits to the repository would have to be evaluated against curtailing the strategy and implementing an alternative engineering solution. One option is to immediately dispose wastes in the repository, which would of course reduce the overall repository capacity benefits of reprocessing.

In summary, the IWMS: 1) demonstrates that wastes can be managed under current law/policy, 2) suggests approaches for more effective waste management, and 3) discusses the activities necessary (including technology development, separations efficiencies, and policy and regulatory changes) to implement more efficient waste management. The IWMS offers potential benefits supporting long term sustainable radioactive waste management from a closed fuel cycle by:

1. Summarizing how modifying or eliminating the legislated repository capacity limit in the NWPA could enable significant benefits to a closed fuel cycle. This change is only necessary to facilitate use of the first repository to its technical capacity limit. This strategy could also be applied to a second repository.
2. Reconsidering the utility of the definition of HLW based on developing separation science, and the practicality of regulating the separated wastes under provisions similar to those of 10 CFR 61.
3. Expanding on the list of potential wastes in the GTCC EIS currently being developed by DOE-EM, and using the results as a basis for disposing fuel reprocessing wastes.
4. Recycling actinides as fuel and/or to be destroyed as reactor targets.
5. Managing short-lived fission products in secure storage to allow heat dissipation.

The IWMS involves long term commitments to fully appreciate the potential benefits over current SNF management, and can benefit significantly from collaborative interaction amongst the DOE Offices NE/EM/RW and the NRC and EPA.

3. REPROCESSING WASTES

3.1 PUREX and Variants Including Co-extraction Concepts

As described above, PUREX or some close variant is used worldwide for fuel reprocessing. It is based on a single solvent extraction process. The waste solution from the first cycle of extraction, technically first-cycle raffinate, has historically been classified as HLW. Other process waste solutions including second and third cycle raffinates (if used to purify the product), solvent scrub and decontamination liquids can also be combined in the HLW as was done in U.S. with defense wastes. The solution is acidic unless neutralized, nitrate based, and contains all of the discarded elements including fission products (FP) made up of alkaline and alkaline earth, transition metal, and lanthanide (Ln) elements, minor actinides (MA) such as americium, neptunium, and curium, process additives such as ferrous sulfamate ($Fe[NH_2SO_3]_2$), and trace amounts of solvent. This process can be varied slightly to co-extract Np with U/Pu, but the wastes are similar other than depletion in Np. Note that the repository capacity expansion described above is limited with a single step extraction process because heat-producing americium (Am), cesium (Cs), and strontium (Sr) are disposed in the HLW.

Activated metals including fuel hulls (Zircaloy or stainless steel (SS)) and hardware (Zircaloy, SS, and Inconel) are also a waste stream that may be considered GTCC or HLW. Residual fuel undissolved solids (UDS) not affected by the extraction processing are captured from solution and have been generally vitrified with the liquid HLW.

3.2 UREX+/Echem

Under GNEP, advanced separation flowsheets proposed include the aqueous UREX+ and Electrochemical (Echem) family processes. The UREX+ process uses four main extraction steps (UREX, CCD-PEG or FPEX, TRUEX, and TALSPEAK) to separate U, Tc, Cs/Sr, FP, Ln, and Pu/MA into six separate fractions. (Note that UREX refers to a specific extraction to recover U and Tc, versus the UREX+ family of processes that includes UREX with several subsequent extraction steps.) Additional extractions could possibly be used to partition Pu/Np, Am, Cm. Fuel cladding and hardware are combined in a separate stream. Residual fuel undissolved solids (UDS) not participating in the extraction processing are left in the dissolver or clarifier and can be treated separately or combined with the FP stream that contain the same elements. Aqueous processing also releases fission and activation product gases including ^{3}H, I, ^{14}C, Kr, Xe.

Echem is a non-aqueous process conducted in molten chloride salts that makes use of electrochemical dissolution, selective reduction (plating), and adsorption to partition groups of elements. Processes are being developed to yield U, Pu/MA, Ln, Cs/Sr, and cladding/hardware/transition metals in five separate streams. Echem also releases gaseous FP, though substantially more of the iodine and carbon are retained than in aqueous processing; these elements report to the molten salts.

3.2.1 Waste Stream/Form Descriptions

3.2.1.1 Technetium (Tc)

In the aqueous process, most of the technetium dissolves in the acid-dissolution step, but the remaining fraction (~20%) remains with the undissolved solids. The soluble technetium and the uranium are separated from the rest of the dissolved fuel components in the UREX separation step. The technetium is then removed from solution by capture on an organic-based anion exchange resin, such as Reillex-HP, and the uranium is denitrated and stored as an oxide powder for reuse. Process development is still in progress, but two schemes being developed in parallel for stabilization of Tc. In either scheme Tc is eluted from the resin. The recovered Tc is then chemically precipitated and reduced to a metal by steam reforming or reduced directly on an iron substrate. Both processes result in Tc being incorporated into a metallic alloy waste form. Both processes ultimately result in the Tc metal incorporated in a metallic alloy. Technetium (half-life 2.13×10^{5} years) produces a low-energy beta radiation and by itself can be handled safely in a glove box. If alloyed with activated metals and/or the undissolved solids as described below, the operation would require remote operation. Dispersal of Tc immobilized from a metallic alloy requires it be oxidized to the soluble and mobile pertechnetate anion.

In the Echem process fuel is dissolved electrolytically in molten chloride salts, leaving those elements that are more noble than uranium as an undissolved metal residue (i.e., Tc, Zr, Mo, Ru, Rh, Pd, and Te). When processing metal fuels, the SS cladding remains containing the other (noble) transition metals. With oxide fuels, the Zircaloy cladding can be removed if the fuels are broken down first by voloxidation (a roasting step); otherwise the cladding remains with the metal residue. This metallic waste stream also includes a stainless steel anode basket that holds the fuel in the molten salts. A basket can be reused for processing up to about ten batches of unclad fuel and then is disposed with the accumulated metallic wastes. The metals wastes must be processed in a manner that keeps Tc from oxidizing to pertechnetate (TcO_4^-).

3.2.1.2 Cesium/Strontium (Cs/Sr)

Cesium and strontium are separated together from UREX raffinate in a stream also containing barium and rubidium (alkaline/alkaline earth elements). If the FPEX separation scheme is used, these elements will exist in dilute nitric acid. Otherwise, the CCD-PEG strip solution is weak nitric acid also containing approximately 100 grams per liter of organic chemicals as well, which may complicate processing to an oxide waste form.

In Echem processing, the Cs/Sr waste stream differs depending on whether oxide or metallic fuel is being recycled. The first step in processing oxide fuel is to break down the fuel using voloxidation as described above followed by electrochemically reducing the fuel to metal in a molten salt bath. The alkaline and alkaline earth elements are not reduced, and the Cs/Sr with Ba/Rb remains with the lithium used in the process as molten chloride salts. When processing metal fuel this separation does not occur; the Cs/Sr/Ba/Rb are left in the salt after the U/TRU/Ln have all been removed, thus leaving the Cs/Sr waste stream mixed with K/Li as chloride salts. Iodine also remains in this stream as iodide salts. In either case, the anhydrous molten Cs/Sr chlorides (whether they are mixed with Li or Li/K) are fed to the stabilization process. Iron is also used to capture chlorine liberated in the process. The waste $FeCl_3$ can be disposed in the Cs/Sr waste form as considered in the waste from amounts shown in the tables later in this document. An evaluation to determine if it is better to dispose of these materials separately has not yet been done.

Particular issues in developing a process, waste form, and management approach for Cs/Sr include:

- Due to the relative abundance of ^{137}Cs/^{90}Sr, their relatively short half-lives (30.17 and 28.9 years respectively), and the high-energy gamma radiation and heat produced by their decay, significant shielding and heat dissipation will be prime considerations in developing a treatment process for remote use in a hot-cell. The waste form/package/storage/transportation/disposal system must have sufficient heat transfer to maintain temperatures within the constraints of waste form/package/system/transportation/disposal thermal stability. Heat generation rate per volume of waste form is a function of waste loading, waste form density, and time.
- After decay storage, a significant amount of radioactivity will remain due to ^{135}Cs with a half-life of 2.3×10^6 years. This isotope is not currently used to determine the class of LLW in 10 CFR 61.55, however, if significant quantities are to be disposed then total β-γ restrictions to shallow land burial (SLB) systems should be considered.
- The Cs/Sr stream from Echem will likely contain too much iodine to be considered for SLB.
- Waste form must be stable to radiolytic reactions, and/or the package/storage/ transportation/disposal system must be designed to mitigate any radiation effects such as gas generation.
- Waste form must maintain adequate performance while ^{137}Cs/^{90}Sr decay to other elements with different valence and size (Cs^+ to Ba^{++}, and Sr^{++} to Y^{+3} to Zr^{+4}).
- The stream contains barium initially, and roughly 50% of the Cs decays to more Ba. Barium is regulated under the Resource Conservation and Recovery Act (RCRA) as a hazardous constituent, and the leaching potential of any waste form containing barium must be less than the maximum leachability limit of 100 mg Ba/liter when measured with the Toxicity Characteristic Leaching Procedure. Generally, the time between stabilization and disposal is relatively short. Within the extended decay storage concept, the time for which RCRA compliance must be confirmed is an open issue, whether at the time of waste form production or immediately prior to disposal up to 300 years later.
- Waste loading will be a compromise between minimizing the waste volume (which affects processing, handling, storage, and transportation costs), adequate waste form durability and mitigating heat and dose issues.
- Waste form/package/storage system must mitigate the risk of accidental or deliberate dispersal.
- Packaging to allow transportation will likely require secondary containment; this must be designed to allow adequate heat dissipation (this may complicate overland transport) while maintaining adequate shielding.
- Design must be based on a storage duration of roughly 100 to 300 years after fuel discharge. Actually, storage is up to 300 years for a rotating inventory in a last-in last-out management scheme. Thus, the storage/disposal facilities are needed for more than 300 years or could be designed and operated in staged cells.
- Heat rate from decay decreases exponentially with time. Cost effectiveness may drive wet storage initially, then active convective cooling, and then a passive cooling system.

- In the Echem process, iodine may be carried over into the Cs/Sr waste stream. Inclusion of ^{129}I in the waste could eliminate the potential to decay the waste to Class C levels or less to allow shallow land disposal of the waste. This issue may be resolved pending the development of a Cs/Sr separation process.

3.2.1.3 Undissolved Solids (UDS) from Fuel Dissolution

In the UREX+ process, the UDS make up the sludge filtered from the dissolver product after dissolution of the fuel. These solids are primarily transition metal particles and oxides including the undissolved balance of Tc, Zr, Mo, Ru, Rh, Pd, and Te. The UDS may also contain small amounts of TRU contamination, but work is underway to minimize TRU losses. This sludge is the waste stream that must be stabilized.

In Echem, the undissolved elements are similar elements to those listed for UREX+. When processing metal fuels, the stainless steel or Zircaloy cladding is also included with the metallic residues. The undissolved metals (similar elements to the UDS in aqueous processing) may also contain small amounts of Zircaloy cladding particles and TRU contamination. These metallic wastes must be stripped of carryover salt prior to final stabilization.

3.2.1.4 Lanthanides (Ln)

Lanthanides are separated in UREX+ in the TALSPEAK process. The Ln-bearing solution is nitric acid-based (about 6M nitric acid) with only trace organic content. This aqueous solution is the feed to Ln waste stabilization process. Yttrium is the only non-Ln to be carried into this waste in significant quantities, but the primary route for any trace actinides not recovered for recycle is also in this lanthanide stream.

Echem process development may allow the lanthanides to be separated and stripped of carryover salts by distillation. This would allow stabilization in a high waste loading glass. The baseline used in this document is that the lanthanides will be combined with the Cs/Sr waste form in a glass-bonded mineral. This stream will contain the lanthanides, Y, C, Br, and I.

3.2.1.5 Balance of Fission Products (FP)

In the UREX+ process the balance of dissolved transition metal FP are segregated in the TRUEX extraction. The elements Se, Zr, Mo, Ru, Rh, Pd, Ag, Cd, Sn, Sb, and Te, as well as added Fe, sulfate, acetic acid, and possibly fluoride in aqueous nitrate solution are included in the feed to the FP waste stabilization process. Pretreatment steps may be required, e.g., removal of sulfate, prior to stabilization.

The Echem process does not generate this stream. Fission products are distributed between the metallic and salt wastes in the electrorefiner depending on their oxidation potential.

3.2.1.6 Volatile Radionuclides (Including Tritium, Iodine (I), Carbon-14 (^{14}C), Krypton (Kr), and Xenon (Xe))

Tritium is evolved during a voloxidation step for oxide fuels prior to treatment by either UREX+ or Echem. Metal fuels processed through Echem will evolve volatile radionuclides including tritium during dissolution in molten salt. In any case, tritium is oxidized to form tritiated water, which is then captured in a molecular sieve as the first stage in off-gas treatment. It is likely that the molecular sieve will be regenerated and the tritiated water will be condensed to yield a relatively high concentration aqueous waste stream. The total flow of water will depend on the humidity of the gas flow into the voloxidizer as well as any in-leakage into the process. Concentration of the tritium in the final waste form will dictate if the waste can be disposed as Class A; there are no limits in 10 CFR 61 for tritium in Class B/C. Due to the relatively short half-life of tritium (12.28 years) this waste could be a candidate for decay storage, which would allow disposal as Class A depending on concentrations of other radionuclides captured in the stream. Disposal options for LLW above Class A are currently limited, and this may factor into how these wastes are blended or concentrated prior to stabilization and disposal.

Iodine may be evolved to some degree from both UREX+ and Echem during voloxidation when processing oxide fuels. However, most of the iodine is expected to be released from the UREX+ process during acid dissolution. Iodine carried into the molten salt processing of oxide or metal fuels in Echem is expected to be retained in the molten salt and not released. State-of-the-art gaseous iodine capture is believed to be on silver-zeolite, and the loaded zeolite is feed to the waste stabilization process. The iodine entering the molten salt in the Echem process is purged from the system in the Cs/Sr waste salt. Concentration of the iodine in the final waste forms will dictate if the waste can be disposed as Class A/B/C or GTCC. It is likely that fully loaded zeolites will be GTCC and disposition for these wastes are currently assumed to be disposal in a geologic repository.

Similar to iodine, ^{14}C may be released to some degree from UREX+ and Echem during voloxidation when processing oxide fuels. However, most of the ^{14}C is expected to be released from UREX+ during acid dissolution of the fuel as carbon dioxide (CO_2). Residual ^{14}C carried into the molten salt processing of oxide or metal fuels in Echem is expected to volatilize during processing or be retained in the molten salt. State-of-the-art ^{14}C capture as CO_2 is believed to be in a caustic scrub, and the alkaline $Na/CaOH/CO_3$ solution is feed to the waste stabilization process. The total ^{14}C concentration in the waste will depend on the ambient CO_2 of the gas flow into the voloxidizer, dissolver, and any in-leakage into the aqueous or Echem processes. Any ^{14}C retained in the molten salt in the Echem process is purged from the system in the Cs/Sr waste salt. The concentration of the ^{14}C and any other contaminating radionuclides in the final waste forms will dictate if the waste can be disposed as Class A/B/C or GTCC. Available disposal options for the stabilized scrub may affect how the waste is blended or concentrated prior to stabilization and disposal.

Similar to iodine, the noble gases Kr and Xe may be released to some degree from UREX+ and Echem during voloxidation when processing oxide fuels. The balance of the noble gases in either UREX+ or Echem with oxide or metal fuel is expected to be released during acidic or electrochemical dissolution. With even one year of aging of the fuel after removal from the reactor, essentially all of the radioactive Xe isotopes are decayed and the remaining xenon is stable. However, if Kr must be captured (due to ^{85}Kr content), the xenon may still affect how the krypton is captured. Capture is likely to be based on sorption on a solid matrix, and Xe may compete with Kr for active sorption sites. The loaded sorbent could be stabilized as the ultimate waste form (large volume), or the sorbent could be regenerated to release the gases for storage in pressurized cylinders. Due to the short half-life of ^{85}Kr (10.72 years), this waste could be a candidate for decay storage, with potential reuse or release after decay. There are no concentration limits listed in 10 CFR 61 for ^{85}Kr in LLW, so the only concentration drivers are to minimize waste volume while managing decay heat.

3.2.1.7 Waste Metals Including Cladding Hulls and Non-fuel Bearing Hardware (Metals)

Non-fuel bearing hardware includes tie plates, springs, water rods, spacer grids, guide tubes, spacers, etc., made of various stainless steels, Inconels, and Zircaloys. Depending on where these metals are located in the assembly, they will have differing degrees of activation, but will be contaminated sufficiently to be Class A/B/C or GTCC. Disposal options for LLW above Class A are currently limited,[11] and this will factor into if and how these wastes are blended prior to stabilization and disposal. This metal scrap will have a surface oxide coating that may contain radioactive contamination as well as the internal activation products.

The UREX+ process is most likely to be used for Zircaloy-clad LWR oxide fuels; however, it may also be used on SS-clad oxide or metal fuels. After the fuel is chopped and voloxidized, the hulls may be rinsed and possibly leached to minimize any residual fuel or other surface contamination.

The metallic waste from Echem is described in the section above on Tc. Voloxidation of oxide fuels would allow the separation of Zircaloy cladding, otherwise the metallic wastes include cladding, Tc, and transition-metal FP and the anode basket. The metals may also be contaminated by small amounts of TRU.

3.3 Other Reprocessing Wastes

In addition to the design basis wastes partitioned during fuel reprocessing are the two categories Balance of Plant and Job Control Wastes. The first category is from all of the ancillary processes that support a reprocessing facility including analytical laboratories, waste water treatment (WWT), heating, ventilation and air conditioning (HVAC) and decontamination activities. The second category encompasses the consumables used to minimize exposure during maintenance activities, including containment tents, personal protective equipment (PPE), tools, and general combustibles such as rags, bags, and tags (RBT) wastes. These LLW can be Class A/B/C or GTCC, depending on the environments from which they arise. Accurate classification is a responsibility of the generator and may require routine or periodic assay to verify. Each balance of plant waste (from waste water treatment or a particular filter bank) remains fairly constant unless there is an upset condition, but job control wastes can be extremely variable depending on the maintenance being done and the level of containment that must be breached. One estimate conducted under GNEP made the practical assumption that wastes coming from operations handling actinides would likely be GTCC and wastes coming from other operations would be Class A/B/C.[12] Using this type of approximation there is ~5-20× as much GTCC waste as that historically called HLW, and ~3-5× as much Class A/B/C waste as GTCC. These are very rough approximations, but the point is that while much of the emphasis is on the primary separations wastes, there will be significant waste volumes requiring disposal as Class B/C and GTCC. Waste minimization efforts, and technologies to reduce volumes further such as compaction, steam-reforming and incineration, will likely be used where cost-effective, but disposal options for the residuals are still likely to be limited in the U.S. Regulations are in place for these wastes, but just as DOE-EM is evaluating options GTCC disposal facilities, there are few currently licensed facilities for routinely disposing Class B/C materials, and new facilities are needed. The Barnwell facility in South Carolina can take Class A/B/C wastes but will limit customers to the Atlantic Compact states in 2008. Similarly, the Hanford site in Washington only accepts waste from the Northwest and Rocky Mountain Compacts. The Energy Solutions Clive, Utah site accepts wastes from all states, but is limited to Class A. The IWMS does not address state specific rules for disposal site licensing, but acknowledges that additional capacity is needed for Class A/B/C wastes for GNEP to be effective in the U.S.

3.4 Summary on Reprocessing Wastes

A summary of the GNEP UREX+ and Echem waste forms concepts is shown in Table 1. Using 50 GWd/MTHM fuel aged 20 years as a basis, a summary of the expected amounts of each type of waste per MTHM from aqueous reprocessing is shown in Table 2.[13] Table 3 lists similar waste statistics for processing metal fast reactor 107 GWd/MTHM fuel aged 5 years. Waste forms for gaseous radionuclides would be similar and are not listed. Both tables list the amount of wastes expected from processing one MTHM, but note that the fast reactor fuel has generated 107/51~2× the energy. Thus, when comparing the waste production of the two processes the volumes from Echem should be halved to compare on an equal energy basis. Mass of the elements in the waste (plus and chemicals added in separations) divided by percent waste loading gives mass of waste form expected. Mass of waste form divided by density gives unpackaged waste form volume. Waste loading and density ranges were provided by leading

Table 1. Waste form concepts.

Waste Stream	Waste Form(s)	
	UREX+	Echem
Tc	Metal Alloy, possibly containing UDS and transition metal FP. Alloy may require Zr/Fe, which could come from cladding and hardware.	Metal Alloy containing UDS and transition metal FP. Alloy may contain cladding, and may require supplemental Zr or Fe, which could come from additional cladding and hardware. Will also contain used metal fuel baskets used in electro-reduction and used niobium crucibles.
Cs/Sr	Glass or Ceramic, process design should consider ramifications of high heat, high radioactivity, powder handling should be avoided. May be combined with lanthanide stream if both are to be disposed of in the same location.	Glass-bonded sodalite. May contain ^{129}I and possible ^{14}C carried over from electro-reduction. May also contain lanthanide fission products if the two streams are combined.
Ln	Glass— borosilicate glass if segregated as separate Ln stream. May be combined with Cs/Sr and/or transition metal fission products.	Borosilicate glass if segregated. Glass-bonded sodalite if combined with Cs/Sr.
FP	Metal alloy potentially combined with Tc and UDS. Borosilicate glass if combined with lanthanides (potentially in combination with Cs/Sr).	Included in metal waste form with Tc described above.
UDS	Metal alloy potentially combined with Tc and FP.	Included in metal waste form with Tc described above.
Metals—Cladding/Hardware	Compacted metal. Metal ingot if cost effective or required for disposal.	Compacted metal. Metal ingot if cost effective or required for disposal.
Tritium	Grouted tritiated water (HTO).	Grouted tritiated water (HTO).
Iodine	Encapsulated silver zeolite.	Glass-bonded sodalite w/Cs/Sr and/or lanthanide FP.
Carbon-14	Grouted $Na_2/CaCO_3$.	Grouted $Na_2/CaCO_3$ with residual in glass-bonded sodalite w/Cs/Sr and/or lanthanide FP.
Kr	Decayed in fuel then released. Pressurized gas cylinder w/wo Xe if fuel storage is insufficient.	Decayed in fuel then released. Pressurized gas cylinder w/wo Xe if fuel storage is insufficient.

Table 2. Estimated volumes and disposition of recommended waste forms for UREX+ process wastes per MTHM.

	Waste	Waste Form	Density MT/m^3		Waste Loading wt%		Waste Form Mass MT		Waste Volume m^3		Disposal
Stream	g/MTIHM	Type	Lo	Hi	Lo	Hi	Lo	Hi	Lo	Hi	Options
Hydrogen	5.90E+00	Grouted HTO	2.1	2.5	25	30	1.8E-04	2.1E-04	7.1E-05	1.0E-04	Decay/LLW
Iodine	3.92E+02	Encapsulated Zeolite	2.1	2.5	2	6.9	5.7E-03	2.0E-02	2.3E-03	9.3E-03	GTCC
Krypton	5.31E+02	Gas 1–50 atm	0.004	0.185	9	100	5.3E-04	5.7E-03	2.9E-03	1.5E+00	Decay
Carbon	1.31E+03	Grouted $CaCO_3$	2.1	2.5	6	10	4.8E-02	8.0E-02	1.9E-02	3.8E-02	GTCC
If Cs/Sr are separated, and the balance of transition metal FP can be reduced to metals the following three waste forms are made possible. The FP stream includes iron added during separations, and this iron replaces the iron needed to make the Tc metal alloy, thus there is essentially no increase in volume. The lanthanides can be stabilized in high waste loading glass.											
Cs/Sr	7.99E+03	Glass/Ceramic	1.5	4.0	20	50	1.6E-02	4.0E-02	4.0E-03	2.7E-02	Decay/HLW/ LLW
Tc/UDS/FP*	4.33E+04	Metal Alloy	7.6	8.2	40	85	5.1E-02	1.1E-01	6.2E-03	1.4E-02	HLW/GTCC
Ln	1.59E+04	Glass	3.0	4.0	30	60	2.6E-02	5.3E-02	6.6E-03	1.8E-02	HLW/GTCC
If the metallic FP are not reduced to metal they will be immobilized with the lanthanides in glass, thereby increasing the glass volume 6-8×. The Tc metal alloy will require addition of iron, possibly from the activated hardware stream. The hardware stream would be reduced accordingly.											
Ln/FP*	6.20E+04	Glass	2.5	3.2	20	30	2.1E-01	3.1E-01	6.4E-02	1.2E-01	HLW/GTCC
Hulls	2.51E+05	Metal	4.6	6.6	93	100	2.5E-01	2.7E-01	3.8E-02	5.9E-02	HLW/GTCC
Hardware	5.65E+04	Metal	4.6	6.6	93	100	5.7E-02	6.1E-02	8.6E-03	1.3E-02	GTCC

Notes:
Hydrogen (HTO) stream includes estimate of humidity in dried voloxidation sweep gas, tritium disposed as cemented tritiated water without sorbent
Iodine stream also captures bromine, halogens captured on silver zeolite and encapsulated (potentially grouted)
Krypton waste loading reflects low end contaminated with Xe, high-end captured pure, stored for decay as compressed gas
Carbon stream also captures natural CO_2 from dissolver aeration, disposed as grouted carbonate
Cs/Sr stream includes Ba and Rb stabilized as ceramic or glass for decay storage

*Technetium reduced to metal and alloyed with UDS/transition metal FP/portion of Zr cladding/portion of SS hardware, high waste loading dominated by Fe from $FeSO_4$
Lanthanides vitrified as borosilicate glass
*Ln+FP stream includes iron added as TRUEX reagent. Assumes sulfur can be volatilized during vitrification, captured in Offgas and disposed as LLW.

UDS combined in Tc alloy waste form, Tc/UDS/FP waste loading could approach 85% because this waste is dominated by Fe added in separations
Hulls and hardware based on PWR fuel, compacted as low density or melted to yield high density, 93% reflects metals added to lower melting point if metals are melted.

Table 3. Estimated volumes and disposition of recommended waste forms for Echem process wastes.

All mass in MT/MTHM	Mass from Fuel	Mass from Separations	Waste Form	Density MT/m^3	Waste Loading mass%	Waste Form Mass MT	Waste Volume m^3	Disposal
Stream								
Metals[1]	8.79E-01	1.10E-01	Metal Alloy	7.75	100%	9.88E-01	1.27E-01	HLW/GTCC
Cs/Sr[2]	1.88E-02	4.54E-01	Glass-bonded Mineral	2.4	33%	1.43E+00	5.97E-01	HLW/GTCC
Ln	2.47E-02	0.00E+00	Glass	4	50%	4.95E-02	1.24E-02	HLW/GTCC
Cs/Sr and Ln forms can be considered separately as above or as one form as shown below, but not both.								
Ln/Cs/Sr[3]	4.36E-02	4.72E-01	Glass-bonded Mineral	2.4	33%	1.56E+00	6.51E-01	HLW/GTCC
SS Hardware[4]	1.75E+00	0.00E+00	Compacted Metal	5.5	100%	1.75E+00	3.18E-01	GTCC

Notes:

Waste/MTHM cannot be compared directly to aqueous wastes, fast reactor fuel used was for 107 GWd/MTHM or roughly 2x the electrical generation as the 51 GWd/MTHM fuel used in Table 2.

Waste forms for gaseous radionuclides are similar to those shown in Table 2 for aqueous reprocessing (doubled based on higher burnup) and are not duplicated here.

Waste loading calculated as mass% of radionuclides and separation process chemicals in final waste form.

[1] Metal waste form also incorporates SS Echem processing basket.

[2] Cs/Sr waste form contains iodine and is likely >100nCi/g TRU, thus not potentially LLW after decay, waste from separations also includes waste K/LiCl and FeCl$_3$

[3] Lanthanides combined with Cs/Sr in glass bonded sodalite if they can not be separated and put into glass.

[4] The SS hardware stream is activated metal from the fuel element from above/below the reactor core.

authorities in the DOE complex with decades of experience developing waste forms. Minimum waste loadings are supported by data, and maximum waste loadings are projections of what is believed possible based on waste form chemistry. Much more detail on the technical basis for these waste forms is provided in the reference. Note that two entries are shown in Table 2 for the Ln and FP streams. Taken together, as has been done historically with PUREX, these streams are vitrified in HLW glass. However, when separated the Ln stream can be loaded to a greater percentage of the glass, and the FP stream can be chemically reduced and added to the Tc/UDS metallic waste form. In this manner, the glass volume is reduced, and essentially no volume is added to the metal waste form because the FP stream provides most of the iron that would otherwise have to be added to make the alloy waste form. This concept requires two waste forms (metallic and glass) and two processing systems, but the Tc is more stable as a metal and is difficult to incorporate in glass. The overall economic value of this concept must still be evaluated. Note also in Table 2 that each waste stream has two possible disposal pathways: HLW as dictated by the NWPA, or GTCC or Decay (temporary storage) pathways based on risk which are described in the waste disposition schematic below.

Table 3 also shows two options. If the lanthanides can be separated effectively with Echem, they could be stabilized in a high waste loading glass similar to the concept in Table 2. If not, the Ln and Cs/Sr stream are immobilized in a lower waste loading glass bonded mineral.

3.5 Fuel Cycle Wastes

Before proceeding to the IWMS logic schematic, one other fuel cycle waste should be addressed. Fuel fabrication generates some waste and this will continue regardless of a fuel reprocessing strategy. Due to the value of enriched uranium, fabricators design and operate their facilities to minimize losses, but there are still finite losses of fissile material as well as generation of consumables and maintenance wastes. If fuel is reprocessed, and all actinide elements are recycled, several operations that are now done in glove boxes will require remote handling. Fuel/target fabrication will be more complex and will generate GTCC wastes. Losses of actinides must be minimized to meet GNEP goals for maximizing benefit to the repository. Similar to the economic discard limit for plutonium losses during the cold war, an economic recovery limit for evaluating how aggressively to pursue recovery of actinide losses during reprocessing and fuel/target fabrication must be considered in design of future facilities.

3.6 Waste Disposition Schematic

Proposed logic for future waste disposition is illustrated in Figure 2. Note that materials coming into the schematic from the left panel are shown leaving reactor operation, but the logic would apply equally well for any radioactive wastes including medical, diagnostic, defense, or space program byproducts or wastes.

3.7 Potential Fuel Recycle

The first decision diamond in the recycling/reuse center panel of the diagram is an evaluation for recovery of fissile or fertile isotopes. There are at least two key reasons to consider actinide elements for recycle as fuel. The first is to increase the capacity of a repository to make the waste management portion of the fuel cycle sustainable. The second is to make maximum use of uranium and the transuranic elements to make the fuel portion of the cycle sustainable. Irradiated uranium and recovered plutonium are obvious candidates for recycle, but off-specification fuel pellets and scraps from making fuel also come from the fuel cycle. In addition, the investment in GNEP technologies could be leveraged to process other actinide sources; enrichment tails, defense SNF, surplus highly enriched uranium (HEU) and weapons materials could possibly be considered. While uranium recycle may not be cost-effective at this time, long term sustainability suggests considering all fissile and fertile elements for their energy potential. Wastes containing actinides should be considered for potential processing to recover these elements for fuel or

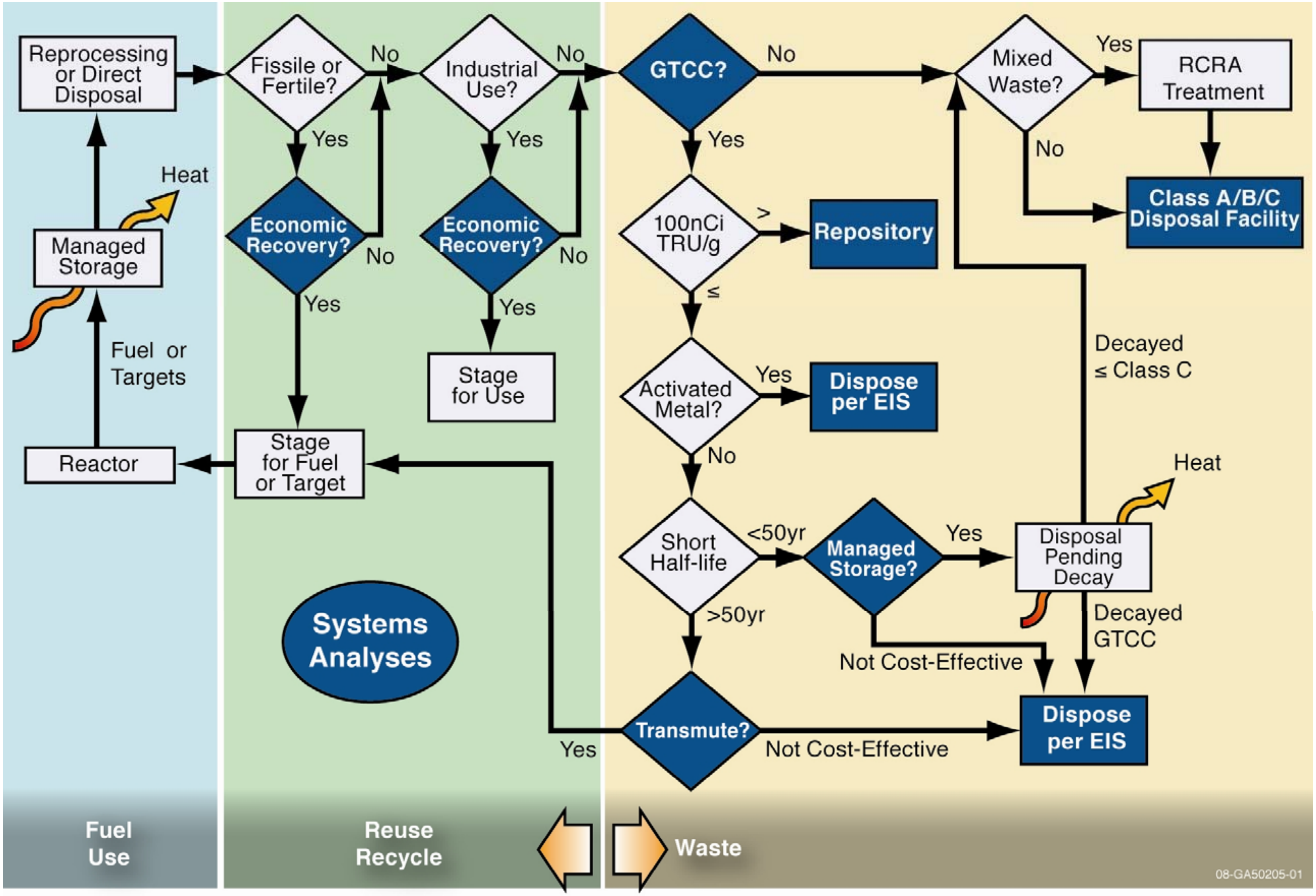

Figure 2. Proposed logic schematic for future waste disposition.

target production. However, there will always be some de minimus concentration, below which recycle is not efficient, which may be imposed by a difficult to extract matrix or recovery step, or isotope mixtures that make recycle impractical. If the cost of recovery cannot be justified, disposition of the wastes continues to the next evaluation. While it is not within the scope of the IWMS to analyze the cost-effectiveness of actinide recovery and the economic limit for when material is considered waste, the framework for such an analysis is described below.

3.8 Economic Recovery Limit

During the cold war, the value of plutonium was based on the cost of reactor time to irradiate uranium and separation operations to recover and purify weapons grade material. The cost to recover plutonium from scrap and process wastes was compared to this value. Materials that were more costly to recycle than "new" plutonium were said to be below the "economic discard limit" and they were disposed.[14] Unfortunately, the priorities of the time did not include consideration of the long-term costs of waste management or material safeguards, and the federal government through DOE-EM now bears those liabilities.

In the future vision of GNEP, the fuel-cycle employs thermal and fast reactors designed to deliver energy as efficiently as possible while integrating a waste management strategy that considers all fissile and fertile isotopes as potential fuel. Losses are unavoidable, but will be minimized. However, as is the case with all recycled materials, there will be some combination of matrices and concentrations at which the cost of recovery for recycle is not justified. This economic recovery limit (ERL) will be dynamic and will depend on many factors.As viewed by the LWR based utility industry in the U.S. today, the cost of fuel fabricated with recovered uranium cannot compete with virgin uranium-oxide fuel, because even with the recent increase in value of uranium, fuel reprocessing and re-enrichment of irradiated uranium is still more costly. From the standpoint of a sustainable energy supply, with a goal of maximizing the return on investment of the entire fuel cycle, additional considerations must be balanced to avoid future costs similar to those now born by DOE-EM. Assuming that a fuel reprocessing facility exists, the considerations for recovering U/TRU from a waste will include some combination of the examples listed in Table 4.

All of the cost elements shown can be estimated with some preliminary design work, but some of the values may not be readily quantifiable. Similar to the policy of as low as reasonably achievable (ALARA) for radiation exposure, the value of reducing dose from the repository may be intangible. While increasing the repository capacity to avoid building an additional facility or facilities may produce direct capital savings, the benefits of not having to site another repository may be of far greater importance. Also, reducing the duration of hazard and waste management liability may have intrinsic value in gaining public support, but establishing a monetary value may be difficult to estimate. In addition to the direct energy benefits to the commercial fuel cycle, leveraging GNEP technologies to create a fissile/fertile recycling capability provides the potential for beneficial use/disposal of defense and orphan materials for which society now bares the costs of management. It is essential that the potential value of recycling these materials in cost-avoidance of continued management is credibly quantified to establish equitable recovery limits and charges.

Any time the commercial value of recovered material is clearly greater than the cost of recovery there will be an economic incentive to do so. When trying to establish an economic recovery limit for any particular waste, however, the delta between cost and value may depend on a social value that warrants government intervention. If there is an increased cost to implement that is offset by an overall benefit to society, the balance could possibly be made up by an incremental increase in the fee paid into the Nuclear Waste Fund. Key to acceptance of such a strategy is to assess the true life-cycle costs of disposing of wastes and allocating them fairly to the beneficiaries of the fuel-cycle through the electrical rate base. Establishing a true ERL for primary fuel-cycle wastes will require additional evaluation, and will require revaluation as the fuel-cycle and recovery processes develop. To maximize the benefits of actinide recovery and stay competitive in the energy market, comparable life-cycle costs for other energy sources should similarly be considered in the rate base.

Table 4. Balancing factors in evaluating economic recovery limits for fissile and fertile materials.

Values		Costs
Energy value as fuel		Advanced separations capital and operating
Reduced source-term and dose from repository		Advanced recovery from variable fuel/target matrices
Increased repository capacity due to reduction in heat-load		Re-enrichment of contaminated feedstock
Shortened duration of hazard and management liability		Purification/blending of fuel and target recipes as isotopic mixture changes
IWMS with advanced separations provides technical basis for reclassifying HLW as GTCC		Interim storage/safeguards and transportation of SNM
Pathway for beneficial disposal of other surplus or orphan materials currently managed by the federal government		Increased LLW/GTCC waste treatment/transportation/disposal
Impact on repository thermal limits and resulting impact on capacity		Remote vs. glove-box fabrication of fuel
		Qualification of variable fuels/targets

3.8.1 Values

Energy value as fuel: this is the economic value of the electricity generated per mass of recovered material. In effect, the recovered material offsets the value of "new" uranium and extends the natural uranium resource.

Reduced source-term and dose from repository: recycling actinides reduces the very long-term radionuclide inventory. Also, segregating iodine and technetium and developing dedicated waste forms could reduce migration of these radionuclides. Reducing the source-term in the repository will likely reduce that estimated dose. Reducing the dose of waste in the repository can allow for an increased capacity of waste without increasing the dose to the maximum receptor.

Increased repository capacity due to reduction in heat-load: reducing the inventory of primary heat-generating radionuclides in wastes has been shown to potentially allow placement of the residuals from more fuel in the repository without exceeding design temperature limits.[15] This heat effect is likely to be beneficial for any geology.

Shortened duration of hazard and management liability: separating very long-lived radionuclides from spent fuel and segregating them from wastes disposed in a repository has been shown to reduce the radiotoxicity of the repository inventory to less than that of naturally occurring uranium ore in several hundred years.[16] A key benefit of the GNEP envisioned fuel-cycle is to reduce the required performance duration of the repository from unprecedented times exceeding thousands of years to times over which many man-made structures have existed.

Technical basis for reclassifying HLW as GTCC allowing risk-informed waste management: this value only pertains to spent fuel and reprocessing. Segregating actinides with advanced separations provides the technical impetus for considering the residuals as GTCC wastes rather than HLW. Additional separations to segregate iodine, technetium, cesium and strontium envisioned under the GNEP program make an even more compelling argument that the definition of HLW may be made obsolete, because what was once lumped together as HLW (both in physical form and in regulations) will now be fractionated into specific streams for beneficial reuse, engineered decay storage, and disposal. The logic for this is based on the fact that the radio-hazard of separated waste streams, if regulated on their specific characteristics under existing regulations, would not be HLW.

Pathway for beneficial disposal of other materials; the federal government manages and safeguards many materials that could be recycled as nuclear fuel. Though outside the scope of the GNEP strategy for fuel reprocessing, this value captures the potential benefit of technology that is developed to recycle commercial materials that could also be used to dispose of other fissile materials as fuel or targets.

Reducing actinide inventory in the repository reduces the long-term heat source, and makes it possible to put more waste in a repository while maintaining the thermal constraints.

3.8.2 Costs

Advanced separations and complexity: the cost of building and operating a commercial fuel reprocessing facility are assumed to be amortized over the total mass of fissile/fertile material to be recycled from design basis fuel. This cost is due to modifications and additional processing to specifically separate minor actinides from waste.

Advanced recovery from variable matrices: this cost captures all modifications and additional processing to extract/recover U/Pu/MA from anything other than the commercial fuel for which the reprocessing facility was designed.

Re-enrichment of contaminated feedstock: uranium recovered from spent fuel will be more enriched than natural ore but less enriched than necessary for fuel charged to a reactor. Thus it has value in requiring less enrichment, but it presents some additional costs to handle due to contaminants. This cost captures any additional expense to enrich contaminated materials.

Purification/blending of fuel recipes: recycling U/Pu/MA from variable sources will result in materials of different isotopic ratios and trace contaminants. As is the case for recycling of essentially any material, purification and blending costs to make the material usable must be accounted for in the recycling life-cycle. This is a significant extension of the current blending ability in the enrichment and fuel fabrication industries.

Interim storage/safeguards and transportation of SNM: the reprocessing facility and the fuel fabrication facility will be as closely linked as practical, but some secure surge storage will be necessary. These costs may be offset by similar costs avoided and savings produced by recycling defense and orphan SNM that are currently stored.

Increased LLW/GTCC waste treatment/transportation/disposal: processing will generate more LLW/GTCC than direct disposing SNF/SNM or essentially any other recycled material. The overall mass of U/TRU disposed will be reduced, but additional processing will generate more waste. Due to the very low levels of TRU contamination required to render waste GTCC (~2ppm), much of this waste will be GTCC.

Remote fabrication of fuel: recipes including significant amounts of americium and curium will necessitate remote fabrication due to gamma and neutron radiation, thus incurring a cost premium.

Qualification of variable fuel: the mixture of actinides to be recycled will vary by element and isotope depending on the source material. As recycled actinides are repeatedly recycled, the mixture and contaminants will continue to evolve. It is unknown at this time how wide a range will be acceptable for efficient use in a recycling burner reactor, but it is expected that certification of fuel(s) will represent an additional cost.

3.9 Potential Reuse

There are potential uses for radioisotopes for heat, radiolysis, lighting, imaging and sterilization in many industries: medicine, diagnostics, and the space program to name a few. However, medical isotopes typically require high elemental and isotopic purity, which means stringent separations requirements for mixes of isotopes. Current medical applications focus on isotopes made through irradiation of high purity starting materials designed to yield higher purity isotopes of interest. Non-medical applications have a higher potential for use, but these applications generally require only small amounts of material so that the cost of providing this material from waste is probably not cost efficient in most cases. Tritium is an exception in that it currently is recovered from heavy water and used commercially. Two other potential exceptions are ^{90}Sr for long term power supplies in remote locations and Xe (after decay storage to remove the radioactive portion) for lights and plasma televisions.

The UREX+ combination of extractions, coupled with pretreatment of fuel by voloxidation, and supported by dedicated offgas treatment technologies has the potential to yield fairly pure ^{3}H, I, Kr/Xe, Tc and Cs/Sr/Ba/Rb. Additional treatment could potentially separate the Cs and/or Sr from Ba/Rb, Kr from Xe, and zirconium from Zircaloy cladding. Having a continuous supply of these materials may provide the basis for an emerging market. The following provides a brief summary of the uses for each isotope considered for recycle.

3.9.1.1 Carbon-14

Carbon-14 has a half-life of 5730 years and is a beta emitter. Cryogenic distillation is used to produce ^{14}C for commercial use, where it is used for labeling drug and other chemical compounds.

3.9.1.2 Cesium-137

Because ^{137}Cs decays to ^{137m}Ba by a beta discharge, then ^{137m}Ba decays to ^{137}Ba by a strong gamma, it has received much less interest in commercial applications than some radioisotopes. However, the strong gamma emitter that makes ^{137}Cs problematic in many applications makes it a good candidate for sterilizing certain food products as well as medical devices and materials. Cesium-137 is used in a very similar manner to ^{60}Co as a gamma source. Cesium-137 has a half-life about six times greater than ^{60}Co, which has advantages and disadvantages. With a longer half-life, a ^{137}Cs source needs to be replaced less frequently and maintains a more constant dose. However, that longer half-life also makes disposal more problematic. Historically, cesium chloride has been used in ^{137}Cs sources because of its high decay energy density. However, because cesium chloride is highly soluble and readily transportable if released from the sealed source, newer sources are using a ceramic form.

3.9.1.3 Iodine-129

Iodine-129 has a half-life of 1.7 X 10^7 years and is a beta emitter. A limited amount of work has been done exploring the potential for ^{129}I as a medical tracer and as an in vivo treatment for cancer. No commercial use is expected to be identified soon.

3.9.1.4 Krypton-81 and -85

Several krypton isotopes are released in small amounts from spent fuel as a gas during separations. Krypton-81 has a half-life of 2.1X10^5 years and is an X-ray emitter, while ^{85}Kr has a half-life of 10.76 years and is a beta emitter with gamma emission from ^{85m}Rb to ^{85}Rb. In the past, ^{85}Kr has been explored as an indicator for equipment wear and as a medical tracer, but no uses were identified in recent literature.

3.9.1.5 Xenon

Several radioactive Xe isotopes are present in spent fuel; of these, ^{126}Xe has the longest half-life at 36.41 days. However, since spent fuel is stored for a minimum of one year (for on-site separations of metal fuel) before separations is done, spent fuel is not a practical source for radioactive Xe. Stable (non-radioactive) Xe has several current and potential commercial uses including lights, plasma televisions, ion thrusters in spacecraft, lasers, and anesthesia. Typically, stable Xe is produced as a byproduct of air separation for oxygen and nitrogen. Depending on the cost of separation and decay storage and the end-use for Xe; it may be possible to recycle Xe from spent fuel.

3.9.1.6 Strontium-90

Strontium-90 has been identified as having a significant potential benefit for medicine. In 1996, Pacific Northwest National Laboratory patented a process for the extraction of ultra-high purity ^{90}Y daughter product from ^{90}Sr. Although it has not yet received FDA approval, ^{90}Y shows great promise as a medical isotope in the treatment of several cancers, including non-Hodgkins lymphoma.

A second medical application of ^{90}Sr is the treatment of pterygium, a benign progressive fibrovascular growth originating in the eye. Surgery is the traditional treatment for pterygium. However in cases of recurrent and symptomatic pterygia, beta irradiation using ^{90}Sr has shown excellent results. Strontium is used in the form of a circular applicator, which is placed directly on the eye. The pure beta radiation released in the decay to ^{90}Y is the active process.

Strontium-90 has also been used to generate electricity in a radioisotope thermoelectric generator (RTG). Sr-90 is a beta emitter that can supply 0.46 kW_{th}/kg when new. RTGs based on ^{90}Sr have been used in Russia to supply electricity in remote locations such as light houses. There seems to be relatively little interest in this application in the US.

3.9.1.7 Technetium-99

Technetium-99m with a six hour half-life is the most widespread radioisotope used in the medical industry. It decays via a low energy gamma, so can be used as a tracer in a variety of applications. It is produced as a decay product from ^{99}Mo. Mo-99 with a half-life of 66 hours is placed in a lead crucible and then when needed, the soluble ^{99m}Tc product is washed out of the crucible using a saline solution. Tc-99 cannot be used as a medical tracer.

Although ^{99}Tc has been shown to offer excellent corrosion protection for steel, its radiotoxicity has prevented its use to date.

3.9.1.8 Tritium

Most commercially available tritium is generated as a by-product of nuclear power. The main non-military uses for tritium are as a component in “self-powered” lights, as a tracer/label for medical research, and as a material in fusion research. Tritium is also used in nuclear weapons. The global commercial use of tritium is estimated at about 400g per year. Ontario Power operates a tritium removal facility that produces about 2.5 kg of tritium per year and tritium removal facilities also exist in France

3.9.1.9 Depleted Uranium

Due to its high density depleted uranium has been used for shielding for both transportation casks and medical and industrial radiography equipment. A significant, and controversial, use has been in military armament, both weapons and armor. However, military uses have become controversial, because of the residual DU, usually in the form of oxides, can be finely dispersed, exposing military and civilian personnel.

3.10 GTCC Waste

If surplus or rejected material cannot be recycled as fuel or reused for another application, its disposition path crosses into the right panel in the diagram, and it is considered a waste. The first evaluation is to determine if the waste is GTCC. This is where the strength of this risk-based approach is best illustrated.

3.10.1 Long-Lived Alpha Contamination

The first decision point is for materials that are over 100 nCi transuranic (TRU) elements per gram of waste, (TRU waste in DOE nomenclature). Similar wastes from defense activities are disposed in WIPP. (Note that less than two parts per million of defense Pu, or only two grams in a metric ton, yields 100 nCi/g, and many of the materials coming from a reprocessing plant are expected to have trace TRU contamination.) One cost of the GNEP approach is a significant increase in the amount of GTCC process, fuel fabrication, balance of plant, and job control wastes. Disposal of this category of waste from defense activities in WIPP is a strong precedent. While it is possible that the GTCC EIS could come to a different conclusion for disposal of this category of wastes (such as a borehole), the diagram shows this waste directed to a repository based on the logic that future disposal will follow the WIPP precedent. It may be prudent to revaluate whether 100 nCi/g contamination warrants deep geologic disposal for all future commercial wastes, but again disposal at WIPP is a significant precedent. It is also assumed that the barriers for migration of TRU would be satisfactory for chemical hazards as well, so similar to the WIPP, a GTCC repository disposal is assumed to meet or be exempt from RCRA land disposal restrictions. This assumption is not necessary to deploy GNEP or this strategy, but it builds logically on historic precedent, and simplifies the IWMS.

A brief discussion of the geologic repository concept is pertinent at this point. The U.S. is currently operating a well run and safe repository in salt near Carlsbad, New Mexico that serves as a significant precedent for permanent isolation of radioactive wastes. The repository option referred to for disposition of several wastes in the flowsheet could be any of the options considered previously including salt, unsaturated tuff, and saturated basalt, shale, granite, argillite, and clay. If the actinides are efficiently removed from waste, the potential for wanting to retrieve the waste is essentially eliminated, and the repository design concept could be significantly simplified. Also, a repository need not necessarily be the very deep concept historically considered for HLW. In Japan, different repository designs are considered for waste depending on their activity and heat output.[17] It would also be logical to consider different design concepts depending on other waste characteristics such as half-life, chemistry, and solubility.

As GNEP matures and is deployed internationally, it will likely become apparent that having a geologic repository in each country no matter how small its nuclear program is not practical. Regional or multinational facilities either based on treaties between governments and/or completely commercially operated promoted by open markets will probably be necessary. Commercial operation may be an attractive option, but will require careful regulatory design and costs will likely be internalized in the electricity market using a fee based system similar to that in the NWPA.

3.10.2 Activated Metals

The next decision point under GTCC waste is for activated metal; this category is broken out separately because internal contamination in corrosion-resistant alloys is already considered stabilized in a relatively durable waste form and it is considered separately in the GTCC-EIS currently under development by DOE-EM. For GNEP fuel reprocessing this category would be primarily fuel hardware and hulls, but could also possibly include a metal alloy containing other metallic elements from reprocessing. Data to date indicates that cladding will likely be contaminated to >100nCi TRU/g and in this strategy would be classified as GTCC (by exceeding the TRU limit). If not GTCC for TRU contamination, it would be considered here. Under a broader application this grouping would include reactor and pressure vessel components from decommissioned reactors. There is some precedent for disposal of large activated pressure vessels in shallow land burial (SLB) at the DOE Hanford site, but the diagram defers the destination to the results of the GTCC EIS, which could include borehole, repository or some other system.

3.10.3 Short Half-Life and Decay Storage

After activated metals, relatively short half-life isotopes are considered. A cutoff of a 50 year half-life is fairly arbitrary, but it is long enough to include the primary FP of importance to the repository, namely Cs/Sr. Using ten half lives as a goal, which reduces the radionuclide concentration by a factor of ~1000, a 50 year half-life would call for a 500 year decay storage facility. This is within the practical range of engineering solutions with dedicated management. This category would also include ^{3}H and ^{85}Kr, the lanthanides, and Cm if separated. Many of the potential issues for decay storage were identified in an earlier report. If managed decay storage is not practical or society wishes to cease this strategy in the future, the result is the same as before decay storage was considered; the material is routed to a repository which truncates the benefits. If the isotopes are allowed to decay to Class C levels or lower, and they meet other requirements to be disposed in the near surface, they would enter the commercial LLW system.

In 2006 the Organization for Economic Cooperation and Development (OECD) Nuclear Energy Agency published a report entitled *The Roles of Storage in the Management of Long-lived Radioactive Waste*[18]. This report examined the role that storage plays, or might play, in OECD member countries and draws conclusions on these roles. The findings and conclusions of this report are summarized below.

- Storage of radioactive waste is valuable for:
  - Decay storage – allowing levels of radioactivity and heat to decline before the next step or process in the waste management strategy.
  - Buffer storage – to provide stock for an ongoing process, transportation step, or disposal.
  - Interim storage – waiting for a waste management step to be deployed or while waiting for a decision to be made on the next step.
  - Strategic storage – for materials that may have a potential future use.
- Storage has been done safely and securely for the past several decades. Storage could continue for decades given proper controls, supervision, and maintenance

Extensive experience and technical knowledge related to storage exists. Storage is firmly regulated. Whether the decay interval between the reactor and ultimate disposal should occur prior to reprocessing in fuel storage, in surge storage during reprocessing and waste stabilization, or some sort of interim storage prior to closing a disposal site is the subject of a trade study to consider design options, security, and cost. Key to decay storage is designing a cost-effective system to allow heat to dissipate prior to placement in a disposal facility. Cost-effectiveness is defined as purely that the life-cycle costs of reprocessing and waste management with decay storage are lower than without decay storage. If this is not possible then the wastes are disposed without a decay interval.

Several studies have been suggested to determine how processing spent nuclear fuel to separate certain elements, followed by managing the wastes using a decay interval would benefit a geologic repository by altering the decay profile of the emplaced waste. They include:

Case 1: Direct disposal of SNF 25 years after discharge from the reactor

Case 2: Direct disposal after extended interim storage of SNF for decay

Case 3: Reprocess SNF early to remove plutonium and limit in-growth of Am

Case 4: Limited reprocessing recovering U/Pu with delayed emplacement of waste

Case 5: Reprocessing with 99.9% Pu/Am recovery with immediate emplacement of waste

Case 6: Reprocessing with 99.9% Pu/Am recovery and interim storage of waste for decay

Case 7: Reprocessing with 99.9% Pu, Am, Cs and Sr recovery and interim storage of Cs/Sr waste (At reprocessing site)

Case 8: Reprocessing with 99.9% Pu, Am, Cs and Sr recovery and interim storage of Cs/Sr waste (At centralized site)

Case 9: Reprocessing with 99.9% TRU, Cs and Sr recovery and interim storage of Cs/Sr waste

Removal of the minor actinides (Am, Np and Cm) and the short lived fission products (Cs and Sr) can significantly benefit a geologic repository; however, achieving these benefits will require increasingly more capital investment in process development, equipment development, infrastructure and facilities. The additional processing required to achieve the repository benefits will increase operating and maintenance costs over the life of the facilities. The additional interim storage of the wastes prior to emplacement in a geologic repository is a relatively under-evaluated concept and will result in additional operating and maintenance costs for the duration of the interim storage period. Trade studies are needed to determine if this concept can be implemented cost-effectively as described above. Cost factors to consider include:

- Procurement or lease of a fleet of aging casks in sufficient quantity to temporarily store the SNF for the duration of the aging period.
- Number of shipments required and any additional constraints imposed by short-cooled fuel.
- Design, construction, and licensing of an away-from-reactor interim storage facility. D&D of these facilities at end of life needs to be considered.
- Equipment and infrastructure to safely store the SNF aging casks for the duration of the aging period.
- Possible repackaging facility and infrastructure to repackage SNF after a sufficient aging period from sealed containers into new sealed containers to consolidate SNF into a fewer number of containers prior to disposal and to place them into the final approved waste package for disposal.
- Operating and maintenance costs to support the aging operations for the duration of the aging period.
- Process development costs for a separations process to extract Am along with the Pu.
- Process development costs for a separations process to extract Cs and Sr from the fission product waste stream.

- A reprocessing plant comparable or greater in scope to existing PUREX reprocessing plants.
- Process, equipment development and facility costs associated with the solidification and packaging of the U/Pu/Am product and the solidification and packaging of the Cs/Sr waste (with a sufficient quantity of solidification and packaging lines to meet throughput requirements).
- Packaging in a shipping package designed, manufactured, utilized and maintained in accordance with 10 CFR 71 to accommodate the higher radiation and decay heat of the Am contained in the product.
- Equipment and facilities to remotely load the U/Pu/Am product containers into the shipping packages.
- Process, equipment development and facility costs associated with the separations process and solidification and packaging of the products.
- Solidification and packaging operations for the fission products will require remote operations in a shielded cell facility.
- Waste solidification and packaging operations must produce a waste form that meets the waste acceptance criteria of a disposal site including a waste form qualification program.
- Interim storage of the solidified waste in a shielded facility is required to provide a sufficient buffer between the solidification operations and subsequent transportation to the repository. Forced air cooling (as opposed to natural draft) is likely to be required due to the high thermal load of waste from short-cooled fuel.
- Design and construction of a transfer system to transport Cs/Sr waste containing containers from the reprocessing plant to the interim storage facility.
- Design and construction of a shielded storage facility of sufficient capacity to temporarily and safely store the wastes for the duration of the aging period. Include forced ventilation if required for cooling. Consider future D&D cost of the facility.
- Equipment and infrastructure to support placement of the Cs/Sr waste containing containers in a shielded storage facility.
- Operating and maintenance costs to support the aging operations for the duration of the aging period such as radiological surveys, facility maintenance, inspections, non-destructive tests, etc.

3.10.4 Long Half-Life and Transmutation

The last GTCC category is for long lived isotopes such as ^{99}Tc and ^{129}I. These elements have been evaluated as candidates for transmutation targets for destruction in a reactor. Analyses to date are not definitive, but suggest that creation of targets may be problematic, reactor operating efficiency could be reduced, and waste generation in separations and target fabrication may be unacceptable. It is clear that the elements would not be destroyed in a single irradiation pass through a reactor, which means the targets would have to be reprocessed and recycled. It is also likely that several recycles would be necessary to reach a steady-state inventory in the fuel cycle, similar to actinide destruction. If this is not found to be practical, then these elements would be stabilized in waste forms for placement in a repository. As above, the GTCC EIS could recommend a different option such as borehole disposal that could be just as acceptable as the repository shown in the diagram.

3.10.5 Class A/B/C LLW

If the waste is not GTCC or is allowed to decay to within Class C levels, it enters the commercial LLW management system. As noted above, the Cs/Sr stream from GNEP reprocessing either from UREX+ or Echem would contain barium, which could render this stream a mixed waste. This category could also include decontamination solutions, solvents, contaminated equipment, etc. Commercial power generation has all but eliminated generation of mixed waste, but when reactors are decommissioned there is expected to be more wastes containing PCBs and asbestos. After treatment for the RCRA contaminant or condition, these wastes can be disposed. Currently, there are limited disposal facilities for routine disposition of Class B/C wastes, and they are commonly blended or load averaged with Class A wastes as permits allow. Balance of plant operations for fuel reprocessing will significantly add to the nominal rate of Class A/B/C waste generation today. These wastes include protective clothing, failed equipment, filters, water and offgas treatment wastes, containers, etc. This is one penalty that cannot be avoided with fuel reprocessing, but it can be mitigated by judicious design and management. Waste minimization programs and technologies including compaction, steam-reforming, and possibly incineration will likely be used to reduce volumes. In the future, additional licensed facilities to dispose Class A/B/C wastes will be needed.

The Low-Level Radioactive Waste Policy Act (LLRWPA) establishes responsibilities for the disposal of low-level radioactive wastes for both the States and the Federal Government.[19] Each State, either by itself or in cooperation with other States, is responsible for the disposal of:

- Class A, B, or C radioactive wastes generated within the state;
- Low-level radioactive waste that is generated by the Federal Government except for waste that is owned or generated by the Department of Energy; and
- Class A, B, or C radioactive waste generated outside the State and accepted for disposal.

The Federal Government is responsible for the disposal of:

- Low-level radioactive waste owned or generated by the Department of Energy; and
- Any other low-level radioactive waste with concentrations of radionuclides that exceed the limits for class C radioactive waste.

The issue of adequate commercial LLRW disposal capacity has been recognized by several organizations. In 2004 the GAO evaluated LLRW disposal capacity in the United States and concluded that[20]:

> “Although no shortfall in disposal availability appears imminent, uncertainties remain about future access to disposal facilities. Even with the prospect of new disposal options, there is no guarantee that they will be developed or be available to meet national needs for class B and C wastes disposal. While LLRW generators have options available to mitigate any future disposal shortfall, including storing waste, storage is costly and it can lead to increased safety and security risks. Therefore, continued federal oversight of disposal availability and the conditions of stored waste is warranted.”

The GAO recommended that:

> “The Congress may wish to consider directing NRC to report to it if LLRW disposal and storage conditions should change enough to warrant congressional evaluation of alternatives to ensure safe, reliable and cost effectiveness of disposal availability.”

In commenting on the GAO report, the NRC stated that the GAO provided an accurate summary of the current LLRW disposal activities at that time, of which there has been no significant change and potential issues that may arise in the future. The NRC stated that given the failure to develop any new sites under the LLRWPA, the GAO should explore alternatives that “would potentially provide a better legal and policy framework for new disposal options for commercial generators of LLRW.” In response, the GAO stated that such an evaluation by them was not required “as long as the NRC places no time limits on storage and provides assurance that it is safe and secure, and any shortfalls in disposal capacity would be managed in the short-term.” The GAO believes it is the NRC’s responsibility to report to Congress on when such an evaluation is needed.

Thus, there has been no effort to evaluate alternatives for increasing the commercial capacity for commercially generated LLRW. This is further evident by the NRC staff determination in SECY-07-1080 that such activities are of low priority. The issue continues as is evident in an article published in the May/June 2007 issue of Radwaste Solutions.[21] That article concludes that there is a crisis in regard to commercial disposal capacity.

Individual states and compacts would be responsible for the disposal of LLRW generated by commercial nuclear reactors as established in the LLRWPA. However, the responsibility (commercial or Federal) for disposing LLRW that would be generated from recycling facilities is unclear. The responsibility for disposing LLRW generated by uranium enrichment facilities established in the USEC Privatization Act[22] may establish precedent for the responsibility of disposing LLRW generated from recycling facilities. The USEC Privatization Act states that:

- the DOE, at the request of the generator, shall accept for disposal LLRW generated by either United States Enrichment Corporation or any person licensed by the NRC to operate a uranium enrichment facility under applicable sections of the Atomic Energy Act;
- no State or interstate compact shall be liable for the treatment, storage, or disposal of any LLRW attributable to the operation, decontamination, and decommissioning of any uranium enrichment facility; and
- a generator may enter into agreements for the disposal of LLRW with any other person other than the DOE that is authorized by applicable laws and regulations to dispose of such wastes.

LLRW generated by uranium enrichment facilities can be disposed either in Federal or commercial disposal facilities. However, LLRW disposal is expected to be in commercial disposal facilities. Revision 2 of the Low-Level Waste Disposal Capacity Report[23] indicates that the LLRW generated at the Portsmouth and Paducah gaseous diffusion plants will be disposed in both Federal and commercial disposal facilities. The environmental impact statement for the National Enrichment Facility states that all LLRW (Class A only) would be disposed in a commercial facility.[24]

The issue of LLRW disposal capacity will ultimately need to be solved independently of GNEP. The existing 104 nuclear power plants, other generators of LLRW, and any new plants that are constructed will ultimately need capacity to dispose of their LLRW. However, the GNEP is predicated on the deployment of new reactors and recycling facilities. Capacity for disposing LLRW will be needed to support a growing nuclear enterprise as envisioned by the GNEP. Market forces may solve this issue without intervention, but it may be prudent for the program to be supportive of and involved in efforts to evaluate alternatives for assuring LLRW disposal capacity. It may also be necessary to work to assure that a disposal pathway exists for LLRW generated by new reactors deployed as part of the GNEP program and generated at recycling facilities should there be a desire to dispose of the waste in a commercial facility.

3.10.6 Transportation Issues

GNEP will use existing technologies and packages to the fullest extent possible. The system for transporting radioactive materials is mature, with a robust regulatory infrastructure and over 50 years experience in package design, manufacture, testing, certification, use and maintenance. During the early developmental phases, it is likely that existing packages can be used to meet GNEP needs. However, GNEP technology is likely to result in waste streams that have not been accommodated under the current transportation infrastructure. Therefore, four types of actions will likely be required:

- acquire additional casks conforming to existing, certified designs;
- amend some certifications of existing casks to address specific new payloads;

- design new cask interior structures to stabilize and customize the fit of the payload; and
- develop new casks for specific types of wastes, such as Cs/Sr. As the GNEP program develops and moves to commercial operations, there will be a need to optimize cask designs. This will create a need for the development and qualification of casks specific to the GNEP waste forms.

3.10.7 Path Forward

This strategy is defined in terms of a path forward that identifies areas to be addressed in support of an overall systems perspective to accommodate the various different waste stream scenarios that are being considered.

- Evaluate GNEP waste shipments with respect to their attractiveness to sabotage and resultant physical protection plans will need to be developed. Consider physical protection systems to guard against radiological sabotage. 10 CFR 73.37 provides criteria for these systems. For example, the physical protection system must; (1) provide for early detection and assessment of attempts to gain unauthorized access to, or control over, spent fuel shipments, (2) provide for notification to the appropriate response forces of any spent fuel shipment sabotage attempts, and (3) impede attempts at radiological sabotage or spent fuel shipments within heavily populated areas, or attempts to illicitly move such shipments into heavily populated areas, until response forces arrive. Demonstration of meeting these criteria are provided in a Security Plan, which will need to be developed if it is determined that specific GNEP shipments are an attractive sabotage target.
- Evaluate mass flows of the different waste streams. Some of these waste streams are relatively small. Truck-sized shipments will probably be sufficient. Some of the waste steams are large, thereby requiring the need to evaluate the advantages and disadvantages of truck v. rail shipments. Based on these analyses, estimate numbers and types of casks required for each waste stream based on waste form, waste form production, and throughput. Within this activity, deliberately segregate those wastes streams that can be shipped in Excepted Packages, IPs and Type A packages v. those that will need to be shipped in Type B packages.
- Assess the capability of the existing certified cask inventory in meeting the identified needs for shipping GNEP wastes. Identify gaps in the inventory and recommend cask development that will be needed in order to support the GNEP program.
- Identify waste streams that will need specific evaluation relative to security planning.
- Develop schedule and budget estimates for development of the transportation infrastructure.
- Identify regulations that will need revisions, or regulatory gaps that need coverage.

3.10.8 Disposition Logic Summary

The waste disposition logic described here offers the potential to make the most of advances in separation science and makes a truly sustainable radioactive fuel cycle including waste management a possibility without the necessity of repeated repository permitting. This flexible waste treatment, storage, and disposal capability is based on highly efficient separations and a national commitment to recycling fuel resources and secure storage of wastes. The disposition logic can also be applied to phased deployment including the current policy of direct disposal of SNF, return to the historic HLW generation from PUREX or a similar process, partial implementation of UREX+, all four UREX+ extractions, or an advanced UREX+ variant with separations of specific actinides. These options range from no repository benefit to long term sustainability of the fuel cycle, but disposal paths for all of the wastes from any of these options can be determined using the disposition schematic.

3.11 Research Needs

To fully accomplish the GNEP goals a systems approach to developing the new fuel cycle must include reevaluation of waste forms and disposition plans for radioactive waste and byproducts. A comprehensive waste management strategy is absolutely essential to achieving the GNEP goals, and to be accepted this strategy must be both practical and support a cost-effective life-cycle. To this point this document has described key aspects of how waste could be effectively managed to make efficient use of resources. This section is focused on technical studies and research needed to support decisions that must be made to implement any strategy, and identifies key data gaps and alternatives to investigate.

Demonstrating a commercially-viable fuel cycle will necessarily drive the GNEP program to demonstrate an optimized waste management strategy that considers the scale and dynamics of complex systems, including fuel fabrication, reprocessing, storage, disposal, and the associated ancillary infrastructure (e.g., transportation) and material flow through the system. Feedback amongst fuel fabrication and recycling and waste and byproduct management is essential to optimize the fuel cycle. The GNEP program must effectively coordinate and integrate research and development in all of these areas. Systems Analyses are essential to evaluate alternatives balancing technology maturity, environmental impact, social values, and of course cost. An example described earlier in this document is evaluating the many ways to mitigate thermal effects in a repository. Some combination of fuel storage and separations, and waste stabilization and storage, followed by ultimate disposition of all materials will be the most practical and cost-effective. Simply because technology is available to make an option possible does not necessarily infer that the option is the most feasible. Partitioning used fuel into many streams and stabilizing the wastes into many forms maximizes the options for waste management, but at a cost of more complex operations, more supporting systems and facilities, and likely more secondary wastes. Equally credible is the scenario described above, combining the waste streams based on target element chemistry, with all of the easily oxidized elements stabilized as oxides in glass or ceramics, and all readily reducible elements combined in a metallic alloy. If these waste forms can be practically made, the systems required could be simpler and less costly. Appropriate systems analyses must be done to determine the most practical commercial approach considering all of the GNEP goals for a sustainable fuel cycle that can be implemented internationally.

For all of the waste streams expected to result from aqueous and electrochemical reprocessing, an initial waste form, treatment technology, and disposal/storage path have been identified.[13, 25] In most cases, the waste form chemistry and performance and the process technology efficiency have been demonstrated and validated on an engineering-scale, and in some cases two or more options have been suggested. However, some of these waste forms are only conceptual and substantial R&D is necessary to determine if they can be made in a remote environment and if they perform as well as expected. Data is needed to evaluate waste treatment technology options that will be considered to produce acceptable waste forms to meet the likely disposal strategy. The following is not a comprehensive list of required R&D; rather it is a list of topics identified in the course of developing this IWMS.

3.11.1 General

1. Evaluate all waste/byproducts for beneficial reuse.
2. Conduct a comprehensive technology readiness assessment on potential processes to manufacture waste forms to prioritize R&D.
3. Characterize recommended waste forms to support eventual acceptance to planned or conceptual disposal facilities.
4. Evaluate potential for cross-contamination in waste streams and potential ramifications to disposition strategy to prioritize research.
5. Determine the optimal waste stream combinations for co-immobilization and disposal.

6. Perform trade studies for LLW treatment to reduce volume, recover radionuclides, and dispose.
7. Develop detailed mass balances to show radionuclide inventories in various waste streams and amounts and categories of LLW.

3.11.2 Technetium

1. A method is needed to effectively capture the soluble Tc and transition metal elements from their respective aqueous solutions in forms amenable to alloying. Studies of the potential methods should address incorporation of the capturing substrate into waste form alloys, including Fe-based and Zr-based systems.
2. Optimize alloy formulations for waste loading to immobilize transition metal fission products including Zr, Mo, Ru, Rh, Pd, and Tc. Niobium from spent crucibles from Echem processing should also be considered.
3. Evaluate alternative reductants to ferrous sulfamate in the separations process to minimize the concentrations of Fe and S in the FP stream.
4. The capacity of alloys to accommodate non-metallic feed materials must be determined. These should include $Zr(MoO_4)_2$ and TcO_2. Slag formation and effects on overall durability should be characterized.
5. A mechanistically-based model for alloy degradation and the release of radionuclides must be developed. This is needed to calculate the long-term performance of the alloy waste form in a disposal system.

3.11.3 Cs/Sr

1. Determine the probable effects of transmutation on the solids and the fate of the stable decay daughters.
2. Optimize glass and/or ceramic formulations for waste loading, considering the effects of radiolysis, transmutation, thermal output, and durability during decay storage.
3. Develop strategy and technical data supporting disposal of waste form following decay storage considering the RCRA requirements for Ba, and ^{135}Cs content.
4. Optimize process for removal of Cs and Sr from Echem salts and optimize waste form for waste loading, considering the affects of radiolysis, transmutation, thermal output, and durability during decay storage.
5. Develop preliminary flowsheets that consider the effects of processing high decay-heat waste streams.
6. Evaluate benefits of heat management concepts including incorporating waste form in a metal matrix, using canisters having internal or external fins, wet vs. dry storage, etc. to determine if the added thermal conductivity and heat transfer is cost-effective for decay storage.
7. Develop a new concept for immobilization of Cs/Sr bearing salts from Echem that allow higher waste loading. Consider processes to separate chlorides prior to or during waste treatment. Consider effects of iodine in Cs/Sr waste from the E-chem process.

3.11.4 Ln/FP

1. Optimize glass/ceramic formulations for waste loading of the combined Ln/FP waste stream.
2. Optimize glass formulations for waste loading of the separate Ln stream.

3. Develop operating envelope for processing glasses/ceramics in an induction melter, including operating techniques to maximize tolerance for transition metals.
4. Evaluate alternative reductants to ferrous sulfamate in the separations process to minimize the concentrations of Fe and S in the FP stream.
5. Optimize ceramic formulations for waste loading.
6. Optimize process for partitioning of Ln from Echem salts and optimize waste form for waste loading.

3.11.5 Volatile Radionuclides

1. Characterize iodine contamination on tritium beds and develop methods to minimize or selectively desorb.
2. Select baseline I capture and immobilization materials/processes.
3. Evaluate methods to maximize long-term retention of iodine on silver sorbent, including encapsulation techniques.
4. Evaluate need for Kr capture.
5. Optimization studies for Xe/Kr separation using solid sorbents.
6. Determine Tritium DF on 3A molecular sieve from very low dew point gas streams.
7. Quantification of Rb corrosion issues in Kr storage cylinders.
8. Develop of non-grout-based ^{14}C waste form, if required for repository.

3.11.6 Waste Metals

1. Characterize activation of the hardware components to refine technical basis for co-disposal versus segregation of hardware and TRU contaminated cladding.
2. The conditions required to melt the collected hardware, including the needed additives, melting temperature, and cost, should be determined to provide a technical basis supporting the decision to compact or melt the hulls and hardware.

4. CONCLUSIONS

The fuel cycle proposed under GNEP offers the first maximally closed, sustainable radioactive fuel cycle with an integrated waste management strategy. It would certainly be possible to manage and dispose all wastes that would be generated under the Global Nuclear Energy Partnership (GNEP) program under a policy and regulatory framework that is essentially identical to what is currently in place, but a more optimal approach is proposed that could be the basis of a more efficient system for managing and disposing radioactive materials. Emphasis is also placed on energy recovery, recycle, and beneficial use, concepts that are clearly important to a sustainable energy future. In addition, this regulatory development would build on the best aspects of U.S. and international (IAEA) waste regulation, and bring both into closer congruence in support of a global nuclear energy program with cooperative waste management.

Efficient separations coupled with a national commitment to actinide recycle and fission product management can reduce the very long-term repository designed performance requirement to a shorter engineering design time frame. By operating within precedented engineering models, risk and uncertainty can be reduced to standards common in current designs. In addition, one repository can be used for many generations into the future, allowing time for learning and development prior to any need for next generation facilities. Technically, GNEP separations make possible simplified WM regulations based on risk with the disposal environment and waste form tailored to waste characteristics. The IWMS is a basis for collaboration amongst DOE-EM/NE/RW, NRC, and EPA.

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