FUEL CYCLE INTEGRATION ISSUES ASSOCIATED WITH P/T TECHNOLOGY

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ABSTRACT

The three primary interfaces between a generic partitioning and transmutation (P/T) technology and the existing United States fuel cycle are the light-water reactor (LWR) spent fuel inventory, the reprocessed uranium (RU) stream, and the high-level waste stream. The features and implications of these three interfaces are reviewed, and the implications for P/T system design and for waste management are assessed. The variability of transuranic nuclide composition in the LWR spent fuel is calculated, and potential implications for transmutation system core design are discussed.

The radiological characteristics of the RU stream are presented, and options for disposition of the stream are reviewed. Most P/T scenarios assume that RU will be recycled to LWRs. This study demonstrates, however, that LWR recycle cannot totally consume the reprocessed stream, and disposal of a waste uranium stream with high levels of radiologically-significant isotopes will still be necessary. The radioactivity of the tails stream for enrichment plants resulting from a dedicated RU campaign is calculated. The tendency of gaseous diffusion plant enrichment technology to deplete the tails stream of minor uranium isotopes is seen as a benefit and an advantage over Atomic Vapor Laser Isotope Separation-type technology. Finally, the implications of P/T on LWR-origin wastes reporting to the repository is discussed, and several significant differences between LWR-origin wastes and high-level wastes originating from transmutation systems are assessed.

INTRODUCTION

Several different concepts for partitioning and transmutation (P/T) of high-level radioactive waste are currently being proposed or under investigation in the United States (U.S.). Partitioning technologies for light-water reactor (LWR) spent fuel include both aqueous processes (1) and pyrochemical techniques (2). Transmutation technologies include traditional reactor concepts, such as LWRs employing mixed-oxide (MOX) fuel (3), and fast reactors such as the Advanced Liquid Metal Reactor (ALMR) (4), as well as subcritical devices driven by accelerator production of neutrons, such as the Phoenix Concept (5), and the Accelerator Transmutation of Wastes (ATW) System (6). Most of the assessment and research performed to date by technology advocates has focused on the fundamental processes or the system design of the transmutation device or partitioning process rather than on fuel cycle issues.

Activity at Oak Ridge National Laboratory has been directed at defining and describing the integration issues associated with a nuclear fuel cycle flow sheet employing P/T technology. Independent of the specifics of a technology, all P/T systems would have common interface points with the U.S. nuclear fuel cycle. Embodied in these interface points are both the requirements on the P/T system and the information for characterizing incentives and disincentives for deploying P/T technology.

This paper reviews the features and implications of the three primary interfaces between a generic P/T technology and the existing U.S. fuel cycle: (1) the LWR spent fuel inventory, (2) the reprocessed uranium stream, and (3) the high-level waste streams.

Each of these subjects is large and complex and involves issues that are currently unresolved and under study. The objective of this paper is simply to identify and discuss some of the issues that appear to be important to an overall evaluation of P/T systems concepts.

LWR SPENT FUEL INVENTORY INTERFACE

An important distinction exists between fuel reprocessing and P/T. The fuel reprocessing enterprise has the freedom to decide not to reprocess and recycle spent fuel that is unattractive from an economic resource-recovery point of view, such as low-burnup fuel. The P/T concept, however, has as a mission the reduction, by many orders of magnitude, of certain undesirable nuclides in the waste streams. Given that only a very small fraction of spent fuel can be "rejected" by a P/T enterprise, a P/T system must therefore be capable of accommodating a wide range of spent fuel characteristics.

Variability of nuclide composition is a spent fuel characteristic that may be important because virtually all transmutation systems propose to configure the TRU nuclides recovered from discharged LWR fuel in critical or near-critical cores. To date, all transmutation system core analyses assume nonvariable distributions of nuclide concentrations, that is, the analyses implicitly assume that the as-loaded composition and reactivity of a fuel pin, assembly, and/or fuel batch can be tightly specified and well controlled. However, the U.S. spent fuel inventory is neither homogeneous nor well-blended, and nuclide variability will need to be accommodated in the transmutation systems just as it is currently accommodated in the proposed waste canisters of the repository.

The characteristics of the U.S. inventory of spent fuel, as generated between the years 1970 and 2016, have been calculated for the year 2018 (a nominal date for the large-scale deployment of P/T technology). This approach made use of the Department of Energy (DOE) Characteristics Data Base (7) and its description of past and expected U.S. power reactor campaigns. The ensemble of fuel assemblies projected

to be in the U.S. spent fuel inventory in the year 2018 was divided into 1200 bins, with each bin representing a particular set of values of fuel burnup, initial fuel enrichment, fuel age or cooling time, and reactor type [pressurized-water reactor (PWR) vs boiling-water reactor (BWR)]. Using the ORIGEN2 code (8), the nuclide compositions of all 1200 bins were calculated. Results were configured in a data base with flexibility to graphically portray the variability of desired characteristics.

An example of a parameter of interest to system designers is the relative quantity or mass of minor actinides (defined as Np, Am and Cm) to the total quantity or mass of fissile plutonium nuclides (^{239}Pu and ^{241}Pu) in the transuranic (TRU) portion of LWR spent fuel. The quantity of minor actinides, which are net neutron absorbers, is significant for several reasons, such as its effect on the neutron economy and on the swing in reactivity during the period of time that the fuel is irradiated (the "burnup reactivity swing"). Figure 1 shows the variability of the mass ratio of the minor actinides to the fissile plutonium component in the LWR spent fuel inventory in the year 2018. Also shown in Fig. 1 are the values that correspond to the average of the current U.S. inventory and to the standard PWR case (33,000 MWd/MTU, 3.2% enrichment, 10 years cooling).

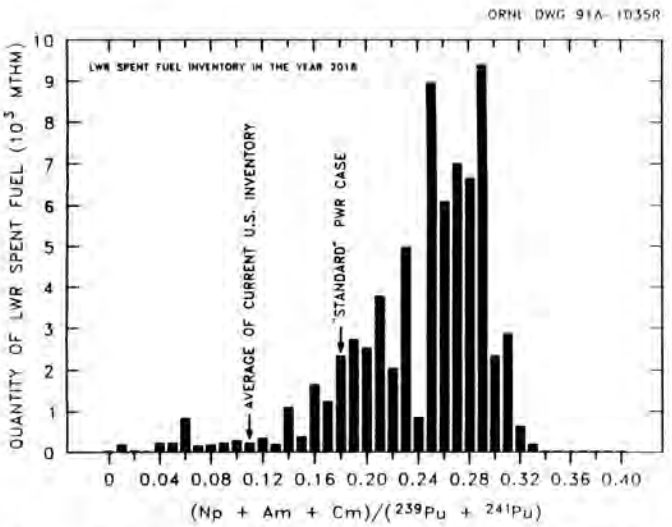

Fig. 1. Variability of LWR fuel: ratio of minor actinides to fissile plutonium. (The "standards" PWR case refers to 33,000 MWd/MTU, 3.2% enrichment, 10-year cooling time.)

The variability in the minor actinide to fissile plutonium ratio originates from several sources. The quantity of Am and Cm in a discharged spent fuel assembly depends strongly, and in a nonlinear way, on the level of fuel burnup sustained by the assembly. As discharged assemblies age, the fissile nuclide ^{241}Pu, with a half-life of 14.4 years, decays into ^{241}Am, which is a difficult nuclide to "burn." Given a value of fuel burnup, the initial uranium enrichment also has a significant effect on Am and Cm generation. Thus, variability in these quantities (i.e., burnup, enrichment, age) causes the variability in minor actinide composition as shown in Fig. 1.

The variability in LWR fuel shown in Fig. 1 appears to be large. The question is: will it be difficult for a P/T system to deal with this level of variability? No quantitative assessment of this question has been attempted, but several qualitative observations are possible.

- Accelerator-driven transmutation systems are expected to operate at neutron multiplication values of between 0.90 to 0.95 (5,6) and be comprised of "cores" that are nearly 100% TRU elements. As the range of variability in fissile content in LWR-origin TRUs appears to be larger than the proposed criticality margin in the accelerator driven cores, the variability in TRU content will warrant further attention.
- Devices that utilize fast neutrons for actinide transmutation, such as the ALMR and Phoenix concepts, will have somewhat less sensitivity to TRU nuclide composition variability than thermal neutron devices, such as MOX-fuel LWRs and the ATW System.
- If the LWR reprocessing plant is based on aqueous technology, significant blending of discharged LWR fuel will occur due to the large quantity of fuel that is dissolved and in an aqueous plant. However, for a pyroprocessing plant that operates in a batch mode and utilizes batches that are roughly the same size as an LWR fuel assembly, little blending of disparate fuel characteristics would be expected to occur. Thus, aqueous processing has an advantage over pyroprocessing in regard to its ability to minimize variability of recovered batches of TRU elements.

Another outcome of the calculation depicted in Fig. 1 is that analyses of the transmuter system design and performance should not be based on the characteristics of the current spent fuel inventory or the standard PWR case. The spent fuel inventory available to the future transmutation enterprise will have a higher concentration of minor actinides (i.e., Np, Am, and Cm) than current fuel inventories due to the trend towards higher burnups, as well as the increased age of low burnup fuels. Current P/T assessments (3-6) do not adequately address this fact.

REPROCESSED URANIUM STREAM INTERFACE

All partitioning schemes that are currently envisioned produce a stream of recovered uranium. This uranium may be highly contaminated in some schemes and of high purity in others. Reprocessed uranium (RU) that has been decontaminated of fission products and TRU elements still presents a handling and disposition problem because of the minor isotopes of uranium. The significance of minor uranium isotopes in the nuclear fuel cycle is briefly reviewed below.

Minor Uranium Isotopes

There are three minor uranium isotopes in RU that have a significant impact in the nuclear fuel cycle, and a fourth (^{233}U) that is long-lived but has negligible impact.

The isotope ^{236}U has a larger neutron absorption cross section than ^{238}U, and, thus, its presence in thermal reactor fuel causes a net reduction in reactivity. LWR fuel containing ^{236}U must be enriched to a higher ^{235}U assay, which requires more separative work to achieve the same burnup as fuel without ^{236}U. When RU is enriched in gaseous diffusion plants (GDPs) or gas centrifuge plants, the ^{236}U nuclide is partially enriched and concentrated in the product stream.

The higher levels of ^{236}U in fuel comprised of RU will also lead to higher concentrations of ^{232}U in the discharged fuel.

The isotope ^{232}U is present in quantities of < 1 ppm in most RU but is significant because of its short half-life (70 years). This gamma radiation from ^{232}U daughter nuclides, primarily ^{208}Tl, ^{212}Bi, and ^{212}Po poses a radiological health risk to handlers of recovered uranium and interferes with assay control in both the enrichment and fuel fabrication plants.

The naturally occurring isotope ^{234}U is found in elevated levels in RU because it is a decay daughter of ^{238}Pu. Uranium-234 dominates the long-term risk and radiological toxicity of recovered uranium because of its half-life (2.46×10^5 years), which is short relative to the major uranium isotopes, and the fact that its decay chain leads to the production of ^{226}Ra and ^{222}Rn. If the Environmental Protection Agency (EPA) release limits for containment requirements, as specified in 40 CFR 191, subpart B, Appendix A (9), are used as a relative measure of radiotoxicity or hazard, then the inventory of ^{234}U present in spent fuel represents a greater radiotoxicity source term or hazard, by a factor of 2, than the analogous inventory of ^{237}Np in spent fuel. This comparison is shown in Table I. In terms of long term risk, this comparison suggests that there may be benefits to burning ^{234}U in addition to ^{237}Np and other TRUs or, alternately, that the benefits of burning ^{237}Np may be limited if ^{234}U is not burned as well.

In P/T scenarios, three disposition options are typically discussed for recovered uranium: long-term storage, disposal in either an HLW or a LLW repository, or reenrichment and recycle as LWR fuel. Complex institutional issues are involved in all these options, and it is unclear which of these options may turn out to be optimal.

Long-Term Retrievable Storage

This option does not represent an ultimate disposition scenario, but it is usually characterized as an option that postpones decisions about RU disposition until better information is available. The design of an RU storage facility and the chemical and physical form of the stored RU have not yet been defined. One characteristic that might weigh decisively against this option is the costs associated with the potential requirement of storing RU in a shielded facility. Even if the RU has been cleanly separated from fission products and transuranics, this characteristic must still be considered. Figure 2 shows calculated values of dose rate at 1 m from the surface of RU storage drums, assuming UO_2 as a storage form. The results were obtained by using ORIGEN2 (8) to calculate PWR spent fuel gamma source terms, QAD-CG (10) to calculate gamma-shielding effects and ANSI/ANS-6.1.1 (11) conversion factors to translate photon fluxes to dose rate. The growth in the dose rate, as shown in Fig. 1, is caused by the build-up of ^{232}U decay daughters. These dose rate values are similar to values calculated by Forsey and Gresley (12) for RU assuming a nominal concentration of 10 ppm ^{232}U. Forsey and Gresley conclude that long-term storage of RU will involve a specialized facility with shielding and automated positioning and retrieval systems.

Regardless of the outcome of economic trade-offs, formidable policy and institutional issues exist regarding long-term retrievable storage of RU.

Recycle as LWR Fuel

There is a large literature base on the recycle of RU in LWRs, and reprocessing and recovery of uranium is underway in France and Great Britain. The P/T scenario for recycle of

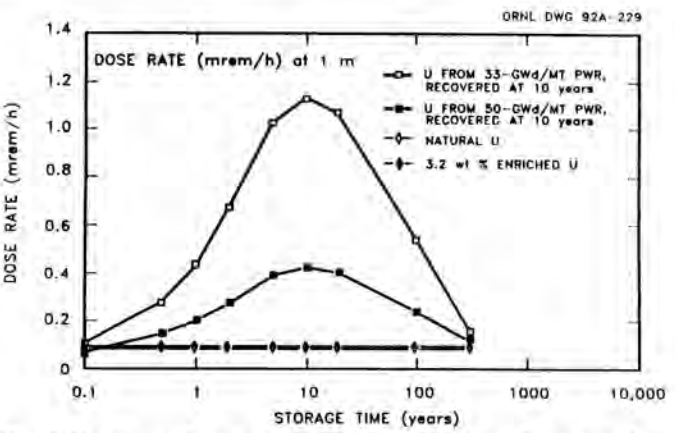

Fig. 2. Dose rate from stored RU as a function of storage time. (Uranium is assumed to be stored as UO_2 in quantities of 560 kg U, and the storage cask is assumed to be constructed of stainless steel with thickness of 0.635 cm.)

TABLE I

Relative Toxicity of Selected Long-lived Actinide Nuclides in Spent PWR Fuel

Nuclide	EPA 10 CFR 191 Concentration limits for containment requirements (Ci/1000 MTHM)	Nuclide inventory in spent PWR fuel[a] (Ci/1000 MTHM)	Relative Hazard
^{232}U	1,000	30	<0.1
^{234}U	100	1,310	13.1
^{235}U	100	10	0.1
^{236}U	100	360	3.6
^{238}U	100	310	3.1
^{237}Np	100	660	6.6
Total Pu	100	7,800,000	78,000

[a]Spent fuel is from PWR, 50,000 MWd/MTHM burnup, initial enrichment of 4.2% ^{235}U, 10-years cooling time (7).

RU has several features that distinguish it from most past analyses:

- The time frame for P/T systems deployment is 2015 or later, and, as discussed earlier, high fuel burnup is expected to be typical of LWR operations. In these analyses, a value of 50,000 MWd/MTU for a PWR fuel with an average as-loaded enrichment of 4.2% ^{235}U has been assumed. This assumption results in higher concentrations of minor uranium isotopes in spent fuel.
- P/T scenarios place an emphasis on minimizing the amount of actinides, including uranium, that is designated as waste reporting to the HLW repository.

The first interface issue that must be faced is the purity specification for UF_6 feed to a gaseous diffusion plant (GDP). The ASTM specification on UF_6 purity (13) for reprocessed uranium feed to a gaseous diffusion plant requires that the total alpha activity level attributable to Pu and Np nuclides be no higher than 25 Bq/g U, or 0.67 nCi/g U. This requirement implies minimum decontamination factors (DFs) of about 10^3 for Np and 10^7 for Pu for high-burnup PWR fuel. These high DF values have been achieved by multistage fluoride volatility separations during the conversion to UF_6 of the uranium stream from a reprocessing plant (e.g., PUREX-based technology). These purity requirements may pose more of an issue for reprocessing plants based on low DF pyrometallurgical separations technology. The processing steps required to decontaminate and convert the metal uranium stream from pyrometallurgical reprocessing plants have not yet been defined.

Assuming that adequate decontamination of RU is feasible and economical, there are still several interface issues associated with the P/T recycle scenario that have not been addressed. Table II shows the calculated characteristics of uranium streams at various points of an RU recycle system employing gaseous diffusion plants (GDPs) for enrichment steps. These calculations assumed a GDP tails assay of 0.2% ^{235}U and that a dedicated RU enrichment campaign at the GDP had not diluted the RU with natural uranium (NU). This latter assumption is obviously an ideal limit and should be improved upon in subsequent analyses.

The results shown in Table II can be used to address several questions. One such question is the status of the tails stream from a dedicated RU enrichment campaign. The tails from an RU GDP campaign results in uranium of approximately the same alpha activity level as NU, with a ^{232}U concentration that is lower than the original RU stream by about a factor of 5. Thus, the mass-selective nature of gaseous diffusion separations essentially decontaminates the tails of the minor uranium isotopes and should yield a tails stream that can be handled in the same way as the usual GDP tails from NU operations.

Dedicated RU campaigns in gas centrifuge enrichment plants would be expected to produce tails streams very similar to the GDP tails streams. However, enrichment plants based on Atomic Vapor Laser Isotope Separation (AVLIS) technology would not be expected to decontaminate the tails stream of minor uranium isotopes, which might result in the requirement that the enrichment tails stream be designated for disposal in a HLW repository. (This option is discussed in the next section of this paper.) Thus, for P/T scenarios that have an objective of reducing the quantity of actinides that report to a HLW repository, AVLIS-type enrichment separations may be undesirable.

A second question addressed by Table II is the level of radioactivity associated with the enriched RU. Because it is assumed that the RU is reenriched to 5.0%, the ^{232}U concentration is calculated to increase by a factor of 7.5. The alpha activity is calculated to increase by a factor of 6 relative to unenriched RU. The occupational radiological aspects associated with managing these radiation levels during GDP product operations and the fuel fabrication step may be significant and should be evaluated. Dilution of enriched RU with enriched NU that reduces the specific activity (but not the total activity) of RU streams in these fuel cycle facilities is an option that should also be considered. AVLIS-type enrichment processes would result in an enriched RU stream with ^{232}U levels that are identical to the initial RU stream. However, as mentioned earlier, AVLIS-processing of RU would have the potentially undesirable effect of leaving the enrichment tails stream contaminated.

A third question with the recycle option is the disposition of the irradiated RU fuel after it has been discharged from the LWR. Given the high values of fuel burnup assumed in this

TABLE II

Calculated Isotope Concentrations of Uranium Streams in a Uranium Recycle Scenario[a]

Stream	Uranium Isotope Assays, %				Alpha Activity (nCi/g)
	^{236}U	^{235}U	^{234}U	^{232}U	
NU	0.0	0.711	0.005	0.0	690
GDP-enriched NU	0.0	4.2	0.033	0.0	2490
RU (PWR, 10-year decay)	0.589	0.713	0.02	3.6E-7	2060
GDP-enriched RU fuel	3.85	5.0	0.15	2.7E-6	12700
GDP tails, RU campaign	0.20	0.20	0.005	7.2E-8	790
Discharged RU fuel (PWR, 10-year decay)	3.79	1.11	0.108	3.8E-6	10300

[a]All PWR calculations assume fuel burnup of 50 GWd/MTHM.

study, a second pass of the RU through LWR reactors does not appear to be an attractive option because of the high concentration of ^{236}U isotope relative to the ^{235}U isotope. A second-pass reenrichment and irradiation can be achieved by diluting the discharged first-pass RU with NU, but the neutronics and cost penalties for this need to be evaluated.

Multiple cycles of RU through LWRs many not be possible even if AVLIS-type enrichment processes are used. Use of an AVLIS-process could indeed eliminate the enrichment of the ^{236}U isotope during the re-enrichment step. However, multicycle buildup of ^{236}U to undesirable levels will still occur due to the continuing production of ^{236}U during the reactor step in the cycle.

After some finite number of irradiation cycles, it seems inevitable that RU will be so unattractive that further recycle is not practical, and disposal in a repository must be undertaken. It is significant that LWR recycle of RU cannot consume all of the RU stream and will ultimately produce a waste uranium stream with relatively high levels of minor uranium isotopes. Unless this waste uranium stream can be accommodated as core material by the transmutation system, as may be the case with ALMRs, the option of recycling RU through LWRs yields a uranium waste stream that must be disposed of in a repository.

Disposal in an HLW or LLW Repository

Scenarios involving disposal of large quantities of recovered uranium are not treated explicitly in U.S. regulations. In its Final Environmental Impact Statement on 10 CFR Part 61 "Licensing Requirements for Land Disposal of Radioactive Waste" the Nuclear Regulatory Commission (NRC) stated:

> Uranium has been removed as a radionuclide that must be considered for waste classification. The Commission's analysis shows that the types of uranium-bearing wastes being disposed of do not present a sufficient hazard to warrant limitation on the concentration of this naturally occurring material. Both depleted and enriched uranium do not contain daughter products in any quantity because of the relatively short time since the uranium was refined from the ore, compared to the half-lives of the uranium isotopes. The daughter products are disposed of primarily as uranium mill tailings. Primarily for these reasons, the uranium limits were dropped (14).

Thus, current regulations place no limits or constraints on the disposal of uranium of any origin in LLW repositories. However, it is unrealistic to expect that quantities in excess of 10,000 MT of RU would be proposed for disposition as LLW without a reexamination of the relevant regulations. It is difficult to predict the outcome of such a reexamination. As seen in Fig. 2, the presence of ^{232}U in reprocessed uranium does significantly increase the gamma-radiation hazard level as compared to natural and enriched uranium for a period of several hundred years. On the other hand, the hazard level of RU (in units of nCi/g) as inferred from the EPA release limits for containment requirements (9) is below that of enriched uranium (Table II) which, according to the above NRC statement, does not present a sufficient hazard to warrant limitation. Although the regulatory status of RU as LLW is unclear, it is reasonable to conclude that the disposition of RU in a LLW repository is not assured and other disposition options should be considered.

Disposition of RU in an HLW repository would be expected to be allowable, but such disposition may reduce some of the benefits to the HLW repository that are claimed for the P/T fuel cycle. A chemical and physical waste form for RU would need to be selected and qualified. Uranium metal may be unacceptable as a waste form due to its pyrophoricity. Uranium oxides (UO_2, U_3O_8, UO_3) tend to be more stable that other uranium forms and may be suitable for repository emplacement. This subject requires a considerable amount of additional study.

As a waste form, RU would have low levels of specific decay heat, and emplacement of RU in boreholes may not be necessary. Although alternate emplacement strategies for RU have not yet been defined, it seems reasonable to expect that emplacement of RU in a repository may not reduce the areal capacity of the repository by a significant amount.

The major disadvantage with disposal of RU in a HLW repository is that it may potentially reduce some of the incentives that are presently claimed for P/T systems. The reduction of repository health risk by reducing the toxicity of emplaced wastes in the repository is a major part of the rationale for research on P/T systems. By allowing RU to report to the repository, some of the underlying justification for P/T is weakened. Figure 3 quantifies this concern by showing how the addition of the RU stream increases the calculated toxicity of the repository waste inventory in the P/T scenario. The slight upturn in repository waste inventory toxicity curve at decay times of 10^5 years is due to the buildup of ^{234}U daughter products.

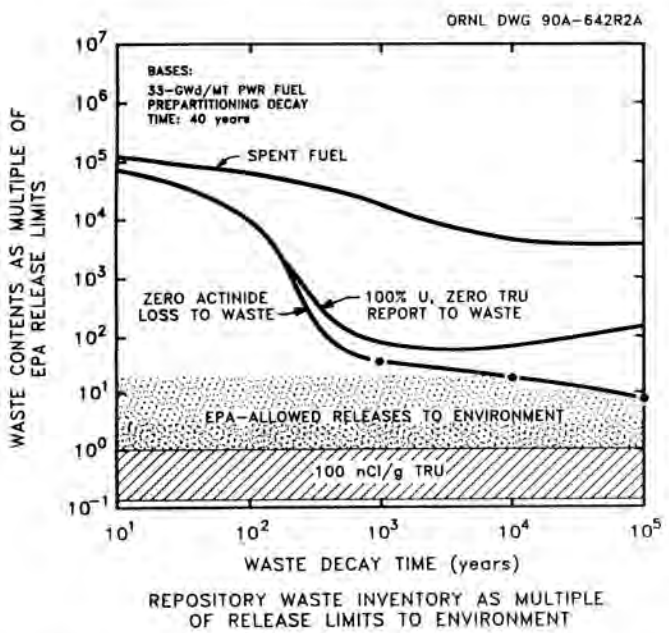

Fig. 3. Effect of reprocessed uranium on HLW toxicity in P/T scenarios.

HLW STREAMS INTERFACE

Large-scale deployment of P/T would certainly have a profound impact on HLW disposal. Both the composition and chemical form of wastes to be emplaced in an HLW repository would be changed. A comprehensive discussion of this topic

is beyond the scope of this paper, but several summary comments can be made.

Potentially, P/T systems could impact an HLW repository by reducing both the magnitude and the uncertainty of repository health risks and enabling strategies for increasing the areal capacity of a repository. These impacts are due to the following:

1. The composition of LWR-origin wastes will be altered. The most obvious effect is to eliminate the vast majority of actinides from the LWR-origin wastes that are to be emplaced in the repository. Actinide removal has the potential to reduce risks from expulsive release and human intrusion scenarios, eliminate some of the incentives for deliberate human intrusion, and reduce uncertainties in repository performance in the slow leach and migration (SLAM) scenario (15). Additionally, work by Croff (16,17) has discussed how removal of the actinides from LWR-origin wastes results in a lower decay heat and shorter thermal decay half-life as compared to spent fuel, thus creating an opportunity to increase the areal capacity of a repository by a factor of 4 to 5.
2. The chemical form of LWR-origin wastes will be altered. P/T concepts require processing of spent fuel that will result in an alternative waste form for LWR-origin wastes. As pointed out by Forsberg (15), selection of borosilicate glass (or equivalent) may result in a waste form that does not degrade under oxidizing conditions such as expected at the candidate Yucca Mountain site. The release rate from glass is expected to be much lower for most radionuclides than from spent fuel. As a result, the risk from the SLAM scenario and all other scenarios due to all radionuclides should be reduced.
3. The projected composition of waste streams originating in the transmutation device will be different than the waste streams originating in LWRs. Because the predominant fissioning nuclides in most transmutation systems are ^{239}Pu and ^{241}Pu, the fission yields of low-mass fission products are different than LWR systems, in which ^{235}U is the dominate fissioning species. Additionally, transmuter systems employing fast neutron fluxes can be expected to yield lower quantities of activation products. Based on values calculated by General Electric Co. in a recent EPRI report (18), Table III shows the quantities of selected fission products and activation products reporting to waste for a standard LWR and the proposed metal fuel ALMR actinide burner system. Notable composition differences shown in Table III are:
   - In the ALMR, the quantity of the activation product ^{14}C is lower than the analogous LWR value by more than a factor of 10^3. This lower quantity has the potential to reduce the repository source term for ^{14}C releases. The ALMR-origin source term for Ni activation products is also lower, but by a smaller factor of 3 to 6.
   - The ALMR yield of ^{90}Sr will be reduced relative to LWR systems by about 30% but the ALMR ^{137}Cs yield is increased by a similar amount. Thus, ALMR fission product decay heat characteristics can be expected to be similar to those of the LWR.
   - The ALMR yields of ^{135}Cs and ^{151}Sm are higher than the equivalent LWR yields by factors of about 7 and 13, respectively. The effect of these higher yields on repository risk is not expected to be major, but further work is needed.

TABLE III

Comparison of Selected Fission Product and Activation Product Quantities in PWR Spent Fuel and ALMR HLW

Nuclide	Nuclide quantities (Ci/33,000 MW(th)d)	
	PWR	ALMR
Fission products		
^{90}Sr	56,200	39,000
^{93}Zr[a]	1.9	2.4
^{99}Tc	12.9	17.7
^{107}Pd	0.1	0.3
^{126}Sn	0.8	2.2
^{129}I	0.03	0.06
^{135}Cs	0.3	2.2
^{137}Cs	80,800	103,000
^{151}Sm[b]	298	3890
Activation products		
^{14}C	2.3	0.0006
^{59}Ni	6.1	1.1
^{63}Ni	76.8	27.3

[a] Includes ^{93}Zr activation product.
[b] The disparity of values for this nuclide is due to burnout in PWR. The ^{151}Sm isotope has a large thermal neutron absorption cross section.

A complete systems analysis of waste disposal in P/T scenarios is advocated and is viewed as being central to the evaluation of both the mission and the research and development (R&D) goals of proposed P/T technologies.

CONCLUSIONS

Several fuel cycle interface issues have been identified for P/T scenarios, including spent fuel variability, disposal of reprocessed uranium, and the nature of P/T waste streams and HLW repository impacts.

- Transmutation devices will need to accommodate fuel batches that have variable compositions. This variability will be more pronounced if the partitioning process is itself a batch process, such as is the case for proposed pyroprocesses. In any case, the methodology of transmutation reactor core design, fuel qualification, and safety analyses may be somewhat different than for analogous LWR activities, because of the projected inability to specify, within a tight range of values, the as-loaded actinide composition of the fuel.
- The spent fuel inventory will have a relatively higher concentration of minor actinides, that is, Np, Am, and Cm, than current fuel inventories due to the trend towards higher burnups, as well as the in-

creased age of low burnup fuels. Current P/T assessments do not adequately reflect this fact.

- Most P/T scenarios assume that RU will be recycled to LWRs. Multicycle buildup of ^{236}U, and perhaps ^{232}U, will limit the number of times that RU can economically be cycled. For high burnup LWR operation, the limit may be a single pass of RU before ^{236}U is unacceptably high. The remainder of the uranium stream will have high levels of the minor uranium isotopes that will need to be burned in the transmutation device or disposed in the HLW repository.
- A potential problem with the use of AVLIS technology for enriching RU is the fact that it does not decontaminate the enrichment plant tails of the minor uranium isotopes, possibly resulting in the need for the AVLIS tails to report to waste. Most P/T scenarios have the objective of minimizing the amount of uranium reporting to waste.
- Disposal of RU in a HLW repository appears to be feasible, but it has the undesirable effect of reducing the incentives for P/T. Waste forms and emplacement strategies for RU need to be defined.
- There are some important differences between LWR-origin wastes and wastes that originate from a P/T system. The ALMR system was seen to have lower ^{14}C production by a factor of 4000 than LWRs, with potential benefits to repository performance.
- A complete systems analysis of waste disposal in P/T scenarios is advocated and is viewed as being central to the evaluation of both the mission and the R&D goals of proposed P/T technologies.

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