CoRWM

Committee on Radioactive Waste Management

Radioactive Waste Burning by Nuclear Transmutation: Technical Report

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Contents

Executive Summary

Some of the radioactive waste that nuclear reactors produce, if left untreated, will be radioactive for more than 100,000 years. It is UK Government policy that this radioactive waste will be disposed of in a GDF, which will contain the waste safely over that long timescale.

Some vendors/developers of AMRs are proposing ‘waste burning reactors’. Radioactive waste *burning* or *transmutation* are shared terms that describe treatment of long-lived radioactive waste to shorten the timescale over which it is radioactive. This involves reconfiguring the nuclear make-up of individual isotopic species to render them shorter-lived by destroying the longer-lived isotopes. This is attractive because, if achievable at scale, it would reduce the period over which waste might be judged to present a concern and hence the time needed to isolate it, thereby reducing the cost and complexity needed to make a safety case for the GDF.

Long-lived, high-level radioactive waste begins life as spent nuclear fuel discharged from nuclear reactors. It comprises residual uranium and plutonium, long-lived fission products and the minor actinides. Its radiotoxicity decays with time, with uranium, plutonium and the minor actinides (i.e., isotopes of americium and neptunium) being its longest-lived components.

If these long-lived components can be transmuted by nuclear reactions contrived by man, only the relatively short-lived fission product component would remain. Most of this will have decayed away by ~300 years, with its major thermal contribution having dissipated within 60 years. Therefore, this component will not need facilities designed for very long-term storage and disposal > 100,000 years. Even if only partial transmutation is possible, this might still reduce the design burden on a repository, i.e., by reducing the inherent need for resilience or capacity. This might be, for example, by reducing the quantity of long-lived species responsible for the *radiotoxic* risk associated with the environmental safety case, and of those responsible for *heat* and the implication this has for repository capacity, its design and that of the containers within it.

Although the radiotoxic/heat-generating risk associated with the long-lived fission product component (principally caesium-137 and strontium-90 and via their daughters barium-137 and yttrium-90, respectively) is shorter-lived than the actinide component, these species might still be ‘burned’, at least in principle. However, this must be balanced with the potential for other species present to be converted inadvertently to yield additional long-lived, problem species, and for neutron losses in such reactions to quench the process. Separating the problem species for dedicated treatment might reduce these risks but this will be complicated by the high levels of radioactivity involved and can be difficult to achieve for isotopes with similar chemistry. Consequently, many waste burning schemes involve setting the separated, long-lived fission product component aside for the few hundred years needed for its decay, focusing on converting the minor actinides instead.

Recycle of residual uranium and plutonium from spent fuel is often assumed in most waste burning schemes, requiring that facilities for conventional nuclear reprocessing and refabrication into fresh fuel are already in place. However, whilst their use as fuels implies recycle (i.e., a ‘closed fuel cycle’), this does not necessarily imply minor actinide consumption via waste burning as well.

Of the minor actinides, americium-241 and neptunium-237 are the prominent long-lived sources of heat and radiotoxicity, respectively. Curium is a lesser concern because its dominant isotopes are not long-lived, but it is difficult to separate chemically from Am, notwithstanding recent developments in separations chemistry. If curium is not removed, its radioactivity will complicate fuel manufacturing necessary to integrate americium and neptunium for burning. This might be avoided by waiting for the curium isotopes to decay but would delay waste burning by at least several decades. This notwithstanding, not *all* uranium and plutonium can be transformed via recycle into minor actinide species suitable for burning.

The accepted route for americium-241 and neptunium-237 burning is to destroy them via neutron induced fission. To achieve this, an important trade-off must be struck between fission and neutron capture processes; whilst both reactions destroy these isotopes, only fission yields the supply of neutrons to sustain the process as well. The balance of fission over capture improves with increasing neutron energy, with the probability of capture declining rapidly relative to fission. In many schemes, this mandates the use of neutrons with energies > 1 MeV for actinide burning.

Two physical approaches to actinide burning have been proposed: *homogeneous* and *heterogeneous*. In the former, the separated minor actinide material is reintroduced in fuel designed for, say, a fast reactor with the fissile content adjusted to cater for the consumption of neutrons in the minor actinides. Several repeat cycles are usually considered necessary to achieve target reductions in radiotoxicity, as otherwise much longer periods of isolation or disposal will be necessary to cater for the higher, residual minor actinide fraction. Heterogeneous approaches use individual targets containing minor actinide material in a fast neutron environment that are withdrawn for processing when the waste has been consumed.

Similarly, two alternative fast neutron environments have been suggested: *fast reactors* or *accelerator-driven systems*. Fast reactors are perhaps better suited to the long-term, cycle-based requirements. It is important at this stage to note that most fast reactors require an amount of plutonium as fuel. The UK Government has recently taken the decision to immobilise the UK inventory of civil separated plutonium[1]. Industrial-scale reprocessing ended in the UK in 2022. In the absence of proposals from industry to reprocess spent fuel, vendors proposing a waste burning scheme with fast reactors would have to source their own plutonium fuel. This paper's focus is on the effectiveness of waste burning schemes and does not comment on UK new nuclear strategy or fuel procurement more broadly.

Accelerator-driven systems are likely to be cheaper but their operation potentially more difficult to sustain for the long periods of time anticipated to be necessary for effective radioactive waste consumption.

In summary, the following considerations are important concerning radioactive waste burning:

- If feasible, radioactive waste burning is only worthwhile for the minor actinides, i.e., isotopes of americium and neptunium, and usually assumes conventional reprocessing and recycle of residual uranium and plutonium, and either storage of the fission product component for > 300 years or its disposal in a separate zone in a GDF.
- Chemical separation of species targeted for burning is necessary to avoid the inadvertent regeneration of problem isotopes, and the potential for neutron absorption on non-target isotopes to undermine the process. Achieving this at scale is not yet proven, and much research and pilot plant development activity is required to do so. Such development and any subsequent implementation at scale would also give rise to additional radioactive waste.

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[1] 'Plutonium Disposition Strategy', 24 January 2025, UIN HCWS388, https://questions-statements.parliament.uk/written-statements/detail/2025-01-24/hcws388

- Radioactive waste burning is only feasible at high neutron energies and/or high fluxes, as otherwise, too many neutrons are absorbed to sustain the process. This mandates the use of either fast reactors or accelerator-driven, high-flux neutron spallation sources.
- Significant reductions in radiotoxicity are necessary if waste burning is to make a worthwhile reduction in the time that long-lived radioactive waste must be isolated. Several cycles of burning and separation processes are likely to be necessary to achieve this, requiring long periods in between to allow for the decay of fission products and curium (i.e., much longer than for RepU or MOx fuel manufacture).
- The probabilities of capture and fission for the actinides are not known experimentally to a sufficient degree of accuracy to enable waste burning plant design with the fidelity required. Their measurement involves difficult experiments and the skills necessary are in short supply, despite recent advances in radioactive ion beam systems.
- Secondary and end-of-life waste (e.g., contaminated plant infrastructure arising as intermediate level waste and reprocessing losses etc.) from waste burning activities might have to be disposed of in a GDF and there is much uncertainty about the nature, hazard potential and radioactivity of this waste. Furthermore, much of the extant HLW stockpile is not in a form compatible with waste burning, necessitating geological disposal in any case.
- The schedule necessary for waste burning to influence existing GDF plans is extremely ambitious, with the benefits unlikely to justify the cost of dedicated reprocessing.

1 Introduction: overall objective and motivation

The ultimate objective of nuclear *waste burning* is to shorten the time over which radioactive waste poses a risk to health by *transmuting* the isotopes responsible for its long-lived radioactivity to shorter-lived alternatives.

This premise is supported by the illustration in Figure 1 (Schmidt, 1985), (WPPT, 2006) which is a plot of radiotoxicity versus time after fuel discharge for high level waste (‘**HLW**’) arising as spent nuclear fuel from a reactor, depicted relative to the radiotoxicity of the uranium ore from which the fuel was made. The latter (shown in green) constitutes the baseline for the material having returned to its natural state. Spent fuel contains all the radioactivity generated by the action of neutrons in the fuel, and comprises residual uranium (‘**U**’), plutonium (‘**Pu**’), other actinides and long-lived fission products (‘**LLFPs**’); long-lived activation products can also be present, for example chlorine-36 and carbon-14 present (half-lives 301,000 years and 5,730 years, respectively) arising from chlorine and nitrogen impurities in the uranium. HLW arises as the raffinate from a first stage of reprocessing of the spent fuel and is the primary form required by most transmutation schemes. However, more widely, HLW can refer to the spent fuel form prior to this first stage of separation, particularly where reprocessing is not accepted practice; further consideration of the isotopic components within HLW follows later in this paper.

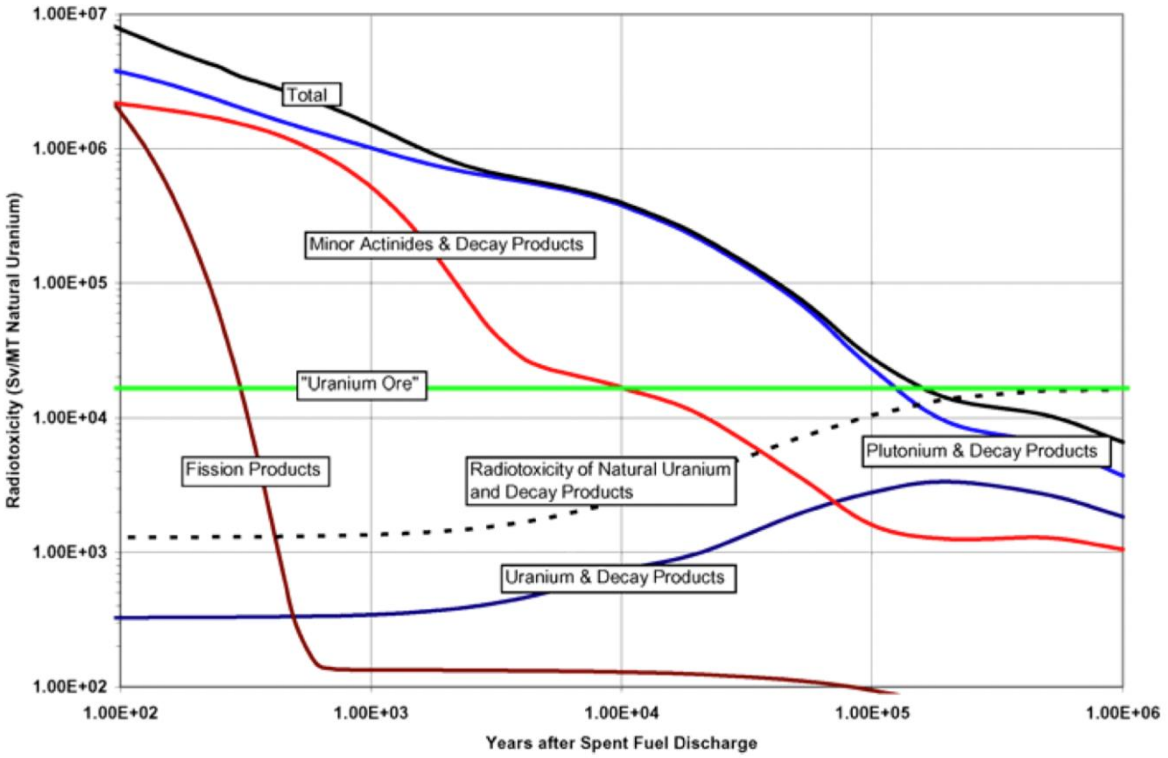

Figure 1. Relative radiotoxic risk by ingestion for spent fuel from a uranium-fed light water reactor after 100 years (Schmidt, 1985), *(WPPT, 2006) (Salvatores, 2011).*

Figure 1 illustrates the potential benefits of the conversion defined above: untreated, the total radiotoxicity of the spent fuel (denoted by the black line) decays with time due to the natural radioactive decay of its constituent radioactive isotopes. This constitutes the maximum time required for spent fuel to reach the radiotoxicity of the corresponding uranium ore (shown in green) at approximately 130,000 years, with the HLW derived from it (red line) reaching this point at 10,000 years. The concern is that the long-lived, isotopic constituents responsible for this very gradual decline could pose a threat to human health and the natural environment, long after the benefit of the power generated which was responsible for the waste arising has been enjoyed, and in an era when the nature of this threat cannot be forecasted accurately today. For completeness, the blue line indicates the contribution to this trend by the constituent Pu and its decay products, the red line the contribution by the minor actinide (‘**MA**’) component and the brown line the trend in natural decay of the fission product (‘**FP**’) content.

Figure 1 implies that, if the plutonium and actinide components can be destroyed, i.e., transmuted as per the fundamental objective defined above, then the timescales over which HLW poses a radiotoxic threat might be reduced from ~130,000 years to, ultimately, just the several hundred years required for the FPs to decay naturally (notwithstanding a small number of very long-lived FPs we shall consider later). If achievable, this could reduce the risk to future generations and the period over which there is confidence that a waste repository continues to isolate the waste in it from causing harm. It is emphasised, however, that the plutonium and MA content need to be *destroyed*, not merely removed, as otherwise they would still need to be disposed of and this would undermine the fundamental objective of waste burning. In some works, recycle of plutonium being implicit is heralded as a further advantage in waste burning, as it avoids the potential for plutonium to be recovered from a repository at some point for nuclear weapons (Beller, 2001).

Fission product	Half-life / a	Thermal ^{235}U yield/%	σ / b		Actinide	Half-life / a	σ_f / b		σ_c / b	
			n,γ (thermal)	n,2n (10 MeV)			Thermal	0.5 MeV	Thermal	0.5 MeV
^{79}Se	3.3×10^{5}	0.05	-	-	^{237}Np	2.1×10^{6}	0.02	0.5	150	0.30
^{85}Kr	10.8	0.22	2	-	^{241}Am	432	3	0.1	700	0.60
^{90}Sr	28.9	4.51	0.01	-	^{242m}Am	141	6500	2.5	1500	0.25
^{93}Zr	1.5×10^{6}	5.46	5	-	^{243}Am	7370	0.08	0.06	80	0.40
^{99}Tc	2.1×10^{5}	6.14	25	0.05	^{242}Cm	0.45	5	1.0	20	0.20
^{107}Pd	6.5×10^{6}	1.25	1.5	-	^{244}Cm	18.1	1	0.5	15	0.25
^{113m}Cd	13.9	0.0008	20,000	-	^{238}Pu	87.7	15	1.5	410	0.20
^{121m}Sn	55	0.00005	-	-	^{239}Pu	24,110	735	2.0	270	0.10
^{126}Sn	2.3×10^{5}	0.11	0.6	-	^{240}Pu	6561	3	0.5	290	0.15
^{129}I	1.6×10^{7}	0.84	30	-	^{235}U	7.04×10^{8}	575	1.0	100	0.18
^{135}Cs	2.3×10^{6}	6.9	8.5	-	^{238}U	4.47×10^{9}	$< 10^{-5}$	$<10^{-3}$	2.5	0.10
^{137}Cs	30.2	6.3	0.3	-						
^{151}Sm	88.8	0.53	15,000	-						
^{155}Eu	4.8	0.08	4,000	-						

Table 1: The typical range of isotopes present in HLW arising from uranium-based spent nuclear fuel discharged from a fission reactor, in terms of (left-to-right): the corresponding FP isotope, its half-

life, thermal fission yield, microscopic cross-section for neutron capture and neutron multiplication (n,γ and n,2n, respectively, for thermal and 10 MeV neutron energies), the corresponding MA isotope and relevant plutonium and uranium isotopes with their fission (σ_f) and capture (σ_c) cross-sections for thermal and 0.5 MeV neutron energies (all cross-section data: Nuclear Energy Agency).

To enable waste burning, it is usually anticipated that the waste under scrutiny is firstly refined to isolate the problematic long-lived isotopic species to be targeted for transmutation. This step is often referred to as *partitioning* with the approach overall known as *partitioning and transmutation* ('**P&T**'), *waste burning* or, when dedicated only to the MA component, as *actinide burning*. Such terms are relatively interchangeable. P&T will be used hereafter in this work.

In an ideal scenario, removal of the MAs, uranium and plutonium (and implicitly their decay products) for treatment would leave just the FPs. Those that remain after the immediate period of storage of the spent nuclear fuel ('**SNF**') when it is discharged from a reactor are usually denoted long-lived, i.e., LLFPs. Most of the nuclides comprising the LLFP component (i.e., strontium-90 ('**^{90}Sr**') and caesium-137 ('**^{137}Cs**') which are heat generating) return to the relative radiotoxic risk ('**RRR**') baseline (expressed in terms of Sievert per mega-tonne of natural uranium, i.e., the green line in Figure 1) in less than approximately 300 years after the SNF is discharged; beyond that strontium-90 and caesium-137 are not anticipated to constitute a significant contribution to the post-closure radiotoxicity of a disposal facility. However, several LLFP nuclides, e.g., technetium-99 ('**^{99}Tc**'), iodine-129 ('**^{129}I**') etc., are longer-lived and, albeit with the latter significantly lower in yield, present long-term risks via their chemical potential for transport with ground-water movement. In the UK, the convention is slightly different, with the long-lived definition being associated with isotopes with a half-life of > 100 years. Hence, ^{137}Cs, ^{90}Sr and ^{85}Kr are considered medium-lived and thus to constitute challenges primarily for storage rather than disposal.

Some scenarios consider P&T of the LLFPs alongside that of the longer-lived MA component, but these works can neglect their influence on neutron economy, the need to avoid the inadvertent regeneration of long-lived isotopes and challenging isotope separation requirements by either hydrometallurgical or pyro-metallurgical means. In general, it is important to consider the specific isotopic composition of both the LLFP and MA contributions to HLW when assessing the viability of P&T. For maximum effectiveness (Bowman, 1992), both the LLFP and MA ought to be converted to target a reduction in radioactivity to a level sufficiently low to eliminate the need for geological disposal, and the cost should be a modest addition to the cost of the generated power responsible for the waste. Conversely, it has been recognised that LLFP transmutation is no longer addressed by any international research programme (Salvatores, 2011).

2 Typical composition of spent fuel and the role of recycled uranium and plutonium

Nuclear fuel discharged from low-enriched, uranium-based, thermal-spectrum fission power reactors typically comprises (in % mass proportion): residual uranium (95.5%), plutonium (0.8%), LLFPs (0.4%) and the MAs (0.1%). Since, aside from uranium, the MAs and plutonium are the longest-lived, the most highly radiotoxic and are also heat generating (dominating heat generation >60 years after waste emplacement), they are the main reason untreated radioactive waste constitutes a risk to health and the environment on very long timescales. Assuming the normal period of cooling following discharge from a reactor over which the relatively short-lived FP components decay away naturally, the typical isotopic composition of HLW arising from reprocessing of spent fuel is as given in Table 1 above.

Aside from the more abundant, long-lived species listed in Table 1, some isotopes are only produced in fission reactors in relatively small quantities and so are not major contributors to the overall risk in the disposal post-closure safety case. Hence, they might be ignored in terms of transmutation, especially since the benefit of converting them is unlikely to outweigh the partitioning effort necessary to sequester them to the high levels necessary to achieve comprehensive levels of transmutation.

Likewise, some species may require too much energy to convert relative to that required for alternative waste management strategies (usually storage) or their presence might act to undermine the transmutation process itself (usually by consuming too many neutrons and hence compromising the neutron economy necessary to sustain conversion). Further, some problem species might be regenerated inadvertently via attempts to destroy others (see below). For these reasons, partitioning of the waste is usually considered necessary to focus transmutation on those isotopes bringing the greatest advantage.

Of particular importance to P&T is the role of uranium and plutonium. Both elements usually constitute *fuel* rather than *waste* by way of the potential for them to be ‘recycled’, that is, refabricated into new fuel following their extraction from spent fuel via conventional reprocessing. When destined for re-use in this way, recovered uranium is often termed *RepU* and plutonium has been the fissile constituent in mixed-oxide fuel (*MOx*), alongside uranium (although not in the UK).

The designation of recycled uranium and plutonium as fuels can imply that they are considered separated from the host waste medium in most P&T flowsheets, i.e., ‘The first requirement of P&T strategies is the deployment of aqueous or dry spent fuel reprocessing techniques’ (Salvatores, 2011). This assumes that reprocessing, i.e., the separation of uranium and plutonium from the LLFP and MA fraction by processes such as PUREX (Plutonium-Uranium Recovery by Extraction) previously used at Sellafield, is feasible. Whilst true for ‘conventional’ metal and oxide fuels, for experimental (often termed *exotic*) fuel forms, significant research still needs to be performed to render this feasible and potential candidate processes have yet to be scaled up.

In the UK, industrial-scale reprocessing ended in 2022, and it is UK Government policy that in the absence of any proposals from industry for reprocessing, new nuclear power stations should proceed on the basis that spent fuel will not be reprocessed. It is therefore worth noting that any P&T scheme proposed could not make use of existing reprocessing infrastructure in the UK.

Relevant to the exotics mentioned above, and to fuel forms manufactured from partitioned waste for the purposes of transmutation that are insoluble in nitric acid, are variants of pyro-metallurgical processing (Laidler, 1997). Here, the SNF is either dissolved in a molten salt or is already in this form so that the metal components within it can be separated across electrodes immersed in the salt by the application of a strong electrical field, as per electrorefining. A typical cycle might involve separating the uranium in dissolved, declad fuel from the MA and LLFPs, with the latter separated from each other via a further stage of electro-refinement. Pyroprocessing offers the potential advantages of implicit safeguarding because plutonium is not evolved separately from the MA component and thus does not arise independently of the high radiation levels of the latter. It is also compatible with the liquid fuel forms implicit in molten salt reactor designs and more tolerant of the heat and radiation evolved from waste modules irradiated to achieve transmutation. However, this has not yet been demonstrated at industrial (or nuclear) scale.

The major isotopic uranium and plutonium components in spent fuel, ^{238}U and ^{239}Pu, have long half-lives and hence they present a radiotoxic burden to disposal at very long timescales. However, to date reprocessing has been performed not for the purposes of transmutation but because these isotopes are refabricated relatively easily into fuel (RepU and MOx) for recycle in LWRs due to their relatively low specific radioactivity. Such reprocessing is tried and tested technology and has been achieved at scale in the UK, USA, France (Warin, 2007) and Japan.

However, the use of RepU and MOx as fuel has not been widespread to date primarily because virgin uranium has remained attractive in terms of cost, especially given that the major cost of nuclear power is usually associated with the construction of the generating plant, rather than the fuel. Further, it should be emphasised that consumption of the uranium and plutonium will never be 100% by recycle, because the fissile content in the fuel and its mechanical integrity falls to unviable levels first, necessitating its discharge, and because there are limits to the amount of ^{239}Pu that can be introduced to compensate for nuclear reactivity in depleted uranium cycles.

There can also be limits on the number of recycles of uranium and plutonium that are possible because of the buildup of problem isotopes, such as ^{236}U, that undermine neutron economy (in the case of ^{236}U due to neutron absorption in a significant resonance in the 1-10 eV neutron energy region). In some schemes, the use of plutonium as fuel can precipitate waste with a higher proportion of MA than that derived from uranium-based fuels, somewhat undermining the fundamental objective of P&T, at least in the short term, and complicating the ease with which reactivity is managed alongside the contribution from MAs provoked to fission. However, conventional reprocessing to liberate RepU and plutonium for MOx *does* afford the LLFP/MA raffinate stream, i.e., a nitric acid based high-activity liquor, in a form that might then be partitioned for transmutation purposes.

3 General concepts

3.1 Long-lived fission products

As defined above, P&T concerns the separation (i.e., *partitioning*) of specific, longer-lived radioactive species in a radioactive waste stream and their subsequent transformation (as per *transmutation*). The supporting nuclear physics suggests that, for example, prominent LLFPs such as caesium-137 (with a half-life of 30 years defining the figurative threshold for ‘long-lived’) might be converted to caesium-138 via neutron capture which then decays to the stable barium-137 with a half-life of 33 minutes, as below,

$$^{137}\text{Cs} + n \rightarrow {}^{138}\text{Cs} \xrightarrow{\beta^-} {}^{138}\text{Ba}$$

and similarly, for strontium-90,

$$^{90}\text{Sr} + n \rightarrow {}^{91}\text{Sr} \xrightarrow{\beta^-} {}^{91}\text{Y} \xrightarrow{\beta^-} {}^{91}\text{Zr}$$

where the half-lives of strontium-91 and yttrium-91 are 9.7 hr and 61 days, respectively, and zirconium-91 is stable.

The percentage yields of caesium-137 and strontium-90 as fission products from thermal fission of uranium-235 at 6.3% and 4.5%, respectively, are amongst the largest and they will remain prominent in SNF or HLW derived from it for the period < 300 years after discharge from a reactor, as per the brown line in Figure 1.

The yields of technetium-99 (‘^{99}Tc’) and caesium-135 are similar at 6.1% and 6.9%, respectively, notwithstanding their much lower radiotoxicities compared to strontium-90 and caesium-137 reflected by their much longer half-lives. The relatively short half-lives of strontium-90 and caesium-

137 and the low radiotoxicity of technetium-99 and caesium-135 has often caused P&T to be focused on the MA component, with the anticipation being that the LLFP component is stored instead, notwithstanding influence of the long half-life, mobility and biological uptake of these species (and iodine-129) on the long-term performance requirements of a GDF. The justification of P&T concerning LLFPs tends to concern the balance of costs borne by the current generation and that which follows potentially almost immediately, in relative terms (in say a couple of hundred years' time). Nonetheless, some examples in what follows do refer to the LLFP component as they illustrate some of the generic challenges concerning P&T, cf., *regeneration* and *neutron economy*.

3.2 Minor actinides

The objective in transmuting the MAs, which include isotopes of curium, americium and neptunium, is to induce them to fission (given it is assumed that for P&T the more abundant plutonium and uranium have been recycled for fuel). In this scenario, the LLFPs might either be left in the transmutation medium to decay or removed from the raffinate by a subsequent cycle of partitioning, with the objective being a significant reduction in the long-lived MA component to reduce the radiological risk if a GDF barrier fails earlier than designed, or is compromised by inadvertent re-discovery by a future society (e.g., as a result of drilling, mining, tunnelling etc.).

To illustrate, the untreated contribution to radiotoxicity by curium and americium falls to one tenth after 1,000 years with americium-243 and curium-245 being the dominant, residual isotopic contributions thereafter. Neptunium-237 is the prominent remaining contribution to radiotoxicity at very long cooling times through to >100,000 years.

P&T of the MA component, if achieved on a sufficiently comprehensive scale, constitutes a means to reduce these timescales to levels where much greater confidence in both the coherence of society and integrity of GDF containment is possible, ultimately targeting the limit set by the LLFPs (~300 years) rather than that set by the MAs (~10,000 years), as per Figure 1.

The reaction probabilities, quantified in terms of the microscopic cross-sections for the corresponding nuclear reactions for these processes, are critical in gauging whether the *time* required for such transformation is advantageous, relative to that required for natural decay. They also enable the implications for the key parameters of irradiation *flux* and nuclide *enrichment* to be estimated, and for the potential for competing pathways to be considered too. The issue of competing pathways is important in identifying if *more* radiologically problematic isotopes are at risk of being produced than might be destroyed and similarly by avoiding the generation of neutron poisons from lanthanide elements which might otherwise undermine the transmutation process.

3.3 General challenges

The challenges of P&T have long been those of *industrial scale-up* rather than *scientific feasibility*. Regarding the former, the treatment of relatively large quantities of HAW (in radiological terms) is necessary if P&T is to realise the reduction in radiotoxicity justifying its use. Otherwise, the burden requiring a repository of a similar scale and design would likely remain, especially given that a significant portion of the associated cost of such a facility can be due to access and infrastructure, rather than its scale. Although not insurmountable, constraints on industrialisation of P&T might be expected concerning, for example, the need for: reprocessing, intense sources of irradiation, fuel manufacture involving radioactive materials and additional secondary and end-of-life wastes arising from such processing. Additional wastes might easily contain sufficient radioactivity to justify disposal in a repository themselves if not anticipated, albeit as ILW, again undermining the fundamental premise for P&T. Moreover, these additional waste streams, in terms of their material composition and processing route (i.e., immobilisation in, for example, a glass or ceramic, or encapsulation in a cement) are yet to be designed, even at the laboratory R&D scale. Some of the LLFPs, technetium-99 and iodine-129, in particular, present challenges in waste management due their specific chemical characteristics (Saslow, 2020) (Bailey, 2023).

4 Fundamental requirements

4.1 Complete burning

One of the earliest reviews of P&T is that of Schmidt (Schmidt, 1985). This summarises the advantage in terms of lowering RRR. Similarly, Doke (Doke, 1974) refers to a *hazard index* for this purpose defined in terms of, for example, the dilution by water necessary to return waste to the radiological status of uranium ore. Schmidt (1985) also considered the modifications needed to a reactor (i.e., a uranium-fed LWR) for it to consume MAs advantageously including, for example, a slight (i.e., 3.3% to 3.4%) modification to the uranium-235 enrichment to offset the neutrons consumed by transmutation processes (notwithstanding the curium issue that undermines LWR use for this purpose, as below).

These works also highlighted the need for much better remote handling capabilities to fabricate fuel incorporating residual HAW because of the inherent radioactivity beyond what is needed for MOx manufacture. An example of a ‘complete burner’ is the BREST-OD-300 which is a lead-cooled, fast reactor currently under construction in Siberia. However, even for ‘complete burners’ in which the scenario can be fast neutron consumption of both MA and plutonium content in the same reactor, some degree of partitioning is often implied, if not to refine the waste, to derive the plutonium etc. with which to fuel such designs. Confusion is possible when the implication is one of waste burning but plutonium consumption is the objective rather than waste reduction, especially where conceptual flowsheets lack detail (Anon., 2025) (Oliver, 2025).

4.2 Partitioning

Burning partitioned material once is different to a process involving multiple recycle steps because the former utilises the relatively expensive processes of MA separation and refabrication of MA-containing fuel only once and is likely to achieve only a limited degree of radiotoxicity reduction. By contrast, multiple recycle steps utilise a plant for longer, potentially bringing better return on investment, and most importantly will achieve greater radiotoxicity reduction given that even a small quantity of residual untreated, MA-containing waste material is still likely to require permanent disposal in a repository. The scale and hence the cost, design and the need for public acceptance concerning such an installation for P&T may not be reduced significantly by operating only a single cycle, especially if only a partial reduction in radiotoxicity is possible.

On the other hand, multiple recycle entails the extra cost of several separation cycle processes of either bulk fuel (*homogeneous* recycling) or of separate targets used in MA burning (*heterogeneous* recycling). Both homogeneous and heterogeneous routes are likely to require long timescales (for the requirements of initial cooling, reprocessing, separation, fuel fabrication, treatment, cooling and separation) and therefore also pose greater levels of risk if repeated, and the need for maintenance. In practice, the term ‘multiple’ recycle in studies published to date has implied a practical limit of only two or three recycles. The overall quantitative reductions possible in such schemes are key to the corresponding business case but, conversely, these have rarely been known with sufficient plant-scale detail thus far.

The PUREX process cited earlier has been used extensively at scale to extract Pu and U from SNF. Tributyl phosphate (TBP, the extractant) is more stable, less flammable and gives better separation than hexone, as was used in the REDOX process. For the purposes of partitioning for transmutation, extraction of the residual MA is required subsequent to PUREX, which might be achieved by either the TRUEX (transuranium extraction) or DIAMEX (DIAMide EXtraction) processes. TRUEX has the advantage of being like PUREX and hence might be potentially compatible across facilities, but it has the disadvantage that very aggressive conditions are required to back-extract the transuranic species from the organic phase. Back-extraction is easier in the DIAMEX process, but higher acidity conditions are necessary to extract americium. This could limit its potential to scale to industrial levels (Choppin, 2000).

Significant effort has been invested, particularly in Europe, to overcome these challenges albeit at laboratory scale and with simulated PUREX raffinate, see for example (Modolo, 2014). This has showed recovery potential for trivalent actinides via, for example, the SANEX (Separation of ActiNide(III) elements by Extraction) process. However, challenges remain concerning low flow rates and loading of the extractant, the need for this to be incinerable leaving no mineral ashes to ensure no additional solid waste arises from solvent use, i.e., the CHON concept (Madic, 1998), and the need for a better understanding of a diverse range of safety issues. The two main partitioning schemes under consideration for spent LWR fuel are reproduced in Figure 2 from (Christiansen, 2009). This depicts the multistage extraction of the LLFPs following uranium and plutonium separation via PUREX, separation of the MA and lanthanide components and, subsequently, separation of americium and curium for transmutation. An allied, single-step process (dotted line) is also depicted leading to the same result. This is anticipated in future but is currently only conceptual because of the difficulty of MA and lanthanide separation at high acidities.

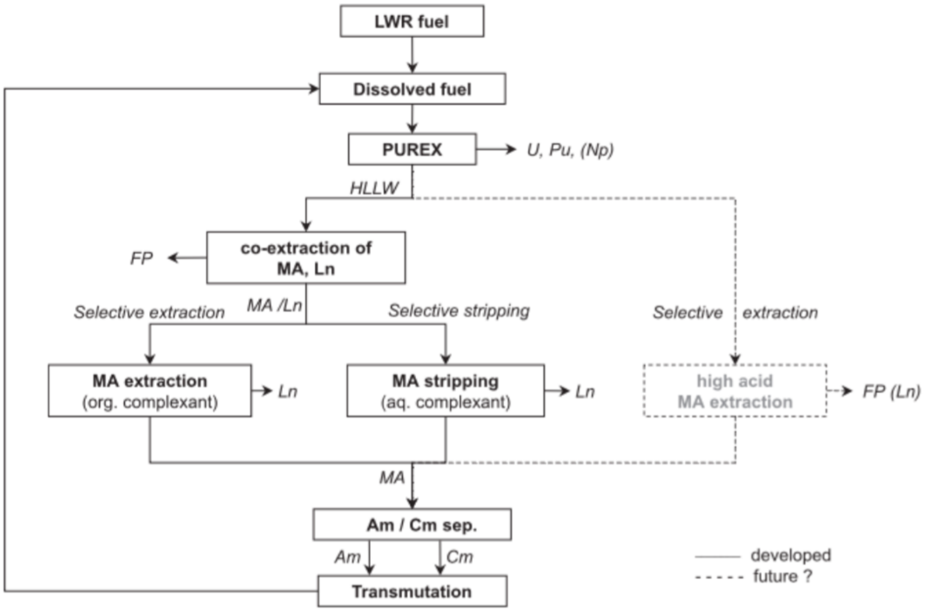

Figure 2. Strategies for the separation of the minor actinides from spent LWR fuel (Christiansen, 2009).

The influence of radiotoxicity reduction targets by P&T is illustrated in more detail in Figure 3. This is a plot (Various, 2010) of ingestion radiotoxicity (dose per tonne spent fuel) versus time for the total, actinides (MAs and plutonium are shown combined in this case), FPs and uranium in equilibrium. Figure 3 is qualitatively as for Figure 1, but includes the influence of P&T cycles for three different curium/plutonium removal levels. It shows that simply removing plutonium and uranium as per 'conventional' reprocessing does not reduce radiotoxicity to a sufficient degree because of the residual MA component. To reduce radiotoxicity significantly, as per earlier definition of P&T to shorten the time needed in a repository, or to avoid the need for a repository entirely, requires that the MAs are recovered from the reprocessing stream ('partitioned' as above) and transmuted, prior to disposal, with the range in time horizons for the contrasting levels of removal being between 500 and 1,500 years.

The dominant isotopes of neptunium, americium, and curium in HLW, long-term, aside from the much more prevalent uranium and plutonium species, are neptunium-237, americium-241 and curium-245, respectively. Their radiological properties that motivate the attention of P&T are as follows: neptunium-237 is the major contributor to long-lived *radiotoxicity* in HLW due to its very long half-life through α decay whereas the americium and curium isotopes are the major contributors to *decay heat*.

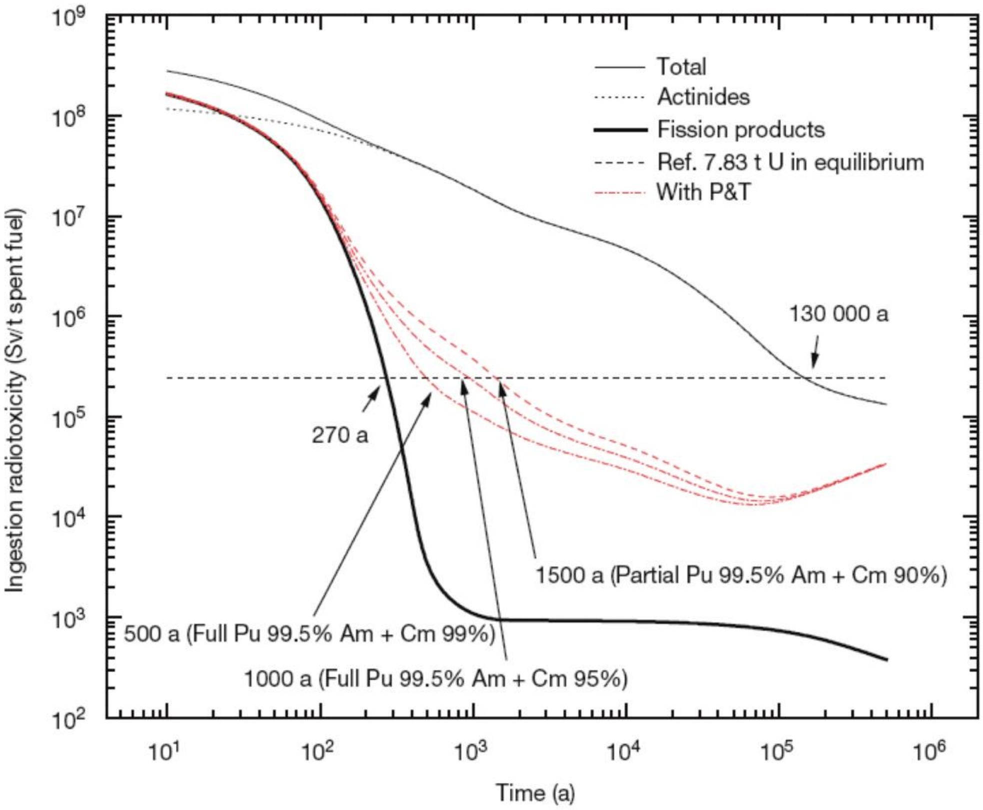

Figure 3. Ingestion radiotoxicity (Sv per tonne spent fuel) versus year for the total, actinide, fission product and following P&T, all in reference to uranium in equilibrium (Various, 2010). *Note the three scenarios associated with plutonium, americium and curium removal and their influence on ingestion radioactivity constituting different degrees of transmutation benefit. The influence of the form of the material in question, e.g., spent fuel, vitrified product etc. is not known but is likely to be significant at plant scale.*

Americium-241 dominates decay heat beyond 300 years, once the prominent LLFPs in this respect (^{90}Sr and ^{137}Cs) and short-lived curium isotopes ^{244}Cm and ^{242}Cm have decayed. Arising predominantly from the decay of ^{241}Pu and to a lesser degree that of ^{245}Cm, ^{241}Am is the prominent limiting factor concerning the space needed in a repository to manage decay heat after disposal.

Consequently, a significant reduction in the area required in a repository for the waste it is designed to hold has been shown to be feasible having removed plutonium and americium, by a factor in the range of 4.3 to 5.4, albeit dependent on removal efficiency and recycle losses being minimised (Wigeland, 2006), with a threshold for losses of <0.2% having been quoted (Christiansen, 2009). However, heat management is not a dominant factor in the footprint of the UK GDF illustrative designs because emplacement positioning is determined by engineering and safety constraints associated with construction and backfilling with interim storage periods then determined to ensure compliance with the applicable thermal limits of the illustrative design. Most existing UK spent fuel stocks do not challenge these thermal limits based on assumed emplacement dates of 2075-2110 with even new build LWR fuel (e.g., Hinkley Point C etc.) only anticipated to require 60-85 years cooling post irradiation; MOx and some advanced reactor fuels might be exceptions in this regard, but then neither are they necessarily compatible with P&T. Moreover, perceived capacity benefits need to be balanced against the secondary, additional and potentially novel wastes likely to arise from P&T itself as any gains in GDF footprint associated with spent fuel would be offset partially by new operational and decommissioning ILW arisings.

Curium contributes to both neutron and γ-ray emissions and, whilst comprised of the relatively short-lived isotopes ^{242}Cm and ^{244}Cm, it poses a long-term issue because the former is generated continuously from the decay of ^{242}Am (half-life 16 hours), as per,

$$^{242}\text{Am} \rightarrow {}^{242}\text{Cm} + \beta^{-}\ (2.8\ \text{MeV})$$

with ^{242}Am generated via neutron capture on ^{241}Am ($\sigma_{\gamma}\sim$700 b (thermal), ~1 b (fast)). This is important, for example, were attempts made to destroy ^{241}Am in a thermal spectrum reactor because its fission cross-section is orders of magnitude lower ($\sigma_f\sim$3 b) with the situation with respect to capture only reversed for fast neutron energies, > 1 MeV; hence ^{241}Am transmutation rates might be comparatively low whilst risking curium regeneration; curium is also the precursor to californium-252 (‘^{252}Cf’) formation via successive thermal neutron capture, which is a further contributor to neutron and γ-ray emissions, as discussed below.

Interim storage, of say 50 to 200 years of the unpartitioned HLW might be preferable to earlier attempts to transmute, especially if these are to involve thermal neutron fields, as this would allow for ^{244}Cm to decay, given its longer half-life (18.1 years) relative to ^{242}Cm. Otherwise, ^{244}Cm activity (primarily α but also spontaneous fission) will complicate the manufacture of fuel required for fast-neutron destruction of neptunium and americium (and more so than for MOx concerning the analogous species ^{240}Pu, for example, because of the higher specific activity of ^{244}Cm and hence higher curium neutron and γ-ray emission rates). However, it is worthy of note that this is longer than the cooling time needed for spent fuel to be considered for a GDF. Further, americium and curium are difficult to separate from one another being elementally similar which would be aided by a period of storage allowing the curium to decline; ^{244}Cm removal by decay also avoids the pathway by which it is activated ($\sigma_{\gamma}\sim$15 b (thermal), 0.3 b (fast)) to form ^{245}Cm in-reactor during recycling (^{245}Cm has a 8,500-year half-life, $\sigma_{\gamma}\sim$350 b (thermal), 0.2 b (fast), cf. ^{241}Am ($\sigma_{\gamma}\sim$700 b (thermal), 0.6 b (fast)) since ^{245}Cm is an α emitter that effectively regenerates ^{241}Am and subsequently ^{237}Np via natural decay. The thermal fission cross-section of ^{245}Cm ($\sigma_f\sim$2500 b) is significantly greater than that of ^{241}Am (σ_f~3 b) suggesting that, if formed, ^{245}Cm could be consumed by a period of conventional reactor operation if needed, which is not feasible for americium.

4.3 Neutron economy

Doke (Doke, 1974) presents a clear argument for why partitioning is necessary for transmutation as opposed to complete burning on the basis that the cross-sections for the destruction of LLFP isotopes tend to be small (c.f., σ_γ~300 mb for thermal neutron capture on caesium-137 as opposed to σ_f~575 b for thermal fission in uranium-235). In the absence of partitioning, of say ^{137}Cs aside from the much longer-lived variant ^{135}Cs (σ_γ~8.5 b)[2], recycle would risk neutrons being wasted inadvertently converting such LLFPs that are neither easy to transmute nor a long-term priority to convert in the sense that MAs are. It is much better for neutron economy (hypothetically at least) to extract (enrich) nuclides with a specific susceptibility to transmutation and then treat them in relative isolation, assuming feasible elemental partitioning schemes can be identified and noting complicated examples such as americium and curium cited above.

Aside from neutron economy, the waste problem might be exacerbated if isotopes susceptible to neutron capture are left in that are not priorities for transmutation, as per ^{242}Am and ^{244}Cm discussed above. For example, successful transmutation of krypton-85 (σ_γ~2 b, with a half-life ~ 11 years, therefore requiring ~ 50-100 years to reach residual levels by decay alone) risks being undone if the stable isotope krypton-84 (σ_γ~100 mb) is present because the latter constitutes the seed by which ^{85}Kr is regenerated if not removed prior to transmutation of ^{85}Kr. Similarly, if the transmutation target is ^{137}Cs but it cannot be isolated from ^{135}Cs (half-life 2.3×10^6 years) then regeneration of ^{137}Cs is feasible, and similarly for ^{90}Sr, albeit with a lower cross-section from ^{89}Sr (σ_γ~425 mb). As Doke (Doke, 1974) concludes, even assuming 100% separation, the flux in conventional LWRs is too low to transmute these isotopes on an effective basis.

It is worthy of note that, by definition, accurate measurement of some cross-sections as referred to above will be necessary long *before* P&T plant are designed, let alone constructed and operated. Otherwise, neutron flux densities and enrichment levels might be implied that are not yet achievable. However, cross-section measurements of radioactive nuclides can pose difficult and, in some cases, yet unresolved experimental challenges if required levels of uncertainty are to be achieved. From a positive perspective, progress on some key measurements in ^{239}Pu has been made recently (Wright, 2025) (Tovesson, 2010) and there has been international investment in radioactive ion beam facilities offering greater potential for such measurements.

Where multiple recycle campaigns are forecast (which could take a long time to complete for the reasons of storage and decay discussed earlier), it might be advantageous to store the shorter-lived LLFPs separately, principally to allow for the decay of ^{137}Cs and ^{90}Sr (cf., ^{135}Cs is less of a concern given its much longer half-life at 2.3 million years and relatively low-energy β^- emission, E_{Max}= 269 keV).

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[2] Caesium-133 (σ_γ~30 b), although stable, has a significantly lower yield from thermal fission in ^{235}U compared to both ^{137}Cs and ^{135}Cs.

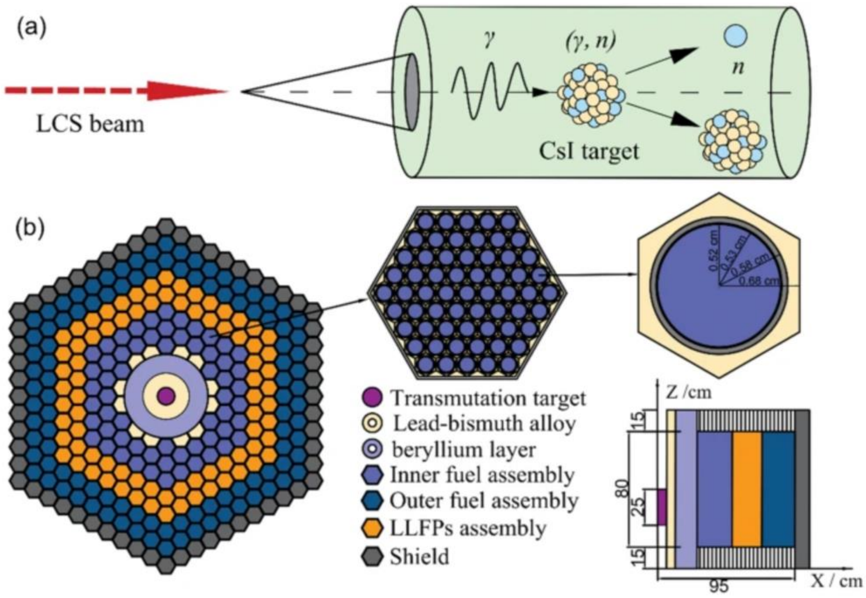


Figure 4. A typical schematic of the transmutation protocol for a) caesium (top) and b) the core/blanket design (Sun, 2023).

4.4 Avoiding regeneration

Further to the cross-section discussion above, ^{137}Cs is a case in point given its ubiquity as a relatively high-yield, high-radiotoxicity and heat generating LLFP because its neutron capture (n,γ) cross-section is challengingly low for the purposes of transmutation. However, this has only been measured experimentally four times, by two distinct authors (Wada, 2000) (Maidana, 1998), giving ~ (275±75) mb, hence with ~30% error and not revisited since 2000. In terms of regeneration, the (n,γ) cross-section for the intermediate nuclide ^{136}Cs has not been measured experimentally at all (it is challenging because of its 13-day half-life). By contrast, its role in understanding the hypothetical need for partitioning in the LLFP component is potentially important, as it is thought to be the 99% pathway (Sun, 2023), and as per Figure 4 above, from ^{135}Cs,

$$^{135}\text{Cs} + n \rightarrow {}^{136}\text{Cs} + n \rightarrow {}^{137}\text{Cs} + \gamma$$

The lack of accurate cross-section measurements cited above (Aliberti, 2004) could undermine the technological viability of transmutation because they underpin estimates of the neutron flux requirements; as Bowman et al. (Bowman, 1992) remark '*accurate prediction of the performance of the system requires more and better nuclear data*'. In this case, the ^{137}Cs (n,γ) cross-section is potentially too low and the risk of its regeneration too high, with there being limited prospect for it to be enriched in preference to ^{135}Cs. Consequently, near-surface storage of ^{137}Cs-contaminated wastes (LLFPs) is preferred, or alternatively placement in a separate area of a repository, having been separated from MA-containing wastes assuming P&T is viable for this separated stream.

Aside from HAW, much ^{137}Cs also arises as contamination in complicated environments, such as ponds and silos etc., from which its recovery and refinement for P&T might constitute an intractable remediation challenge of a scale and cost much more demanding than routine decommissioning itself, whereas the potential for *in-situ* P&T has not been explored to date because of significant deployment constraints. Consequently, some ^{137}Cs-containing wastes will need to be stored prior to near surface disposal, irrespective of the potential to transmute LLFP wastes.

5 Concepts specific to the minor actinides

5.1 General principles

Given that the transmutation of the LLFP component is not yet technologically feasible at scale for the reasons described above and the advantages in doing so are limited given its relatively short-lived state, notwithstanding ^{99}Tc and ^{129}I, this leaves the MAs. These have the longest-lived hazard index at ~10^6 (1 million) years and offer the greatest long-term benefit if transmuted, particularly concerning the engineered barrier systems and environmental safety case, as these are simplified if the time for which the MA hazard exists can be shortened. A reduction in the peak radiological environmental dose is not anticipated via P&T because, for example, ^{237}Np is relatively immobile compared to shorter-lived, volatile elements.

A cycle to reduce MA content might comprise:

1. *Reprocessing* SNF to separate the residual uranium and plutonium component from the MA/LLFP-bearing liquor (tried and tested practice and has been demonstrated at scale in the UK, USA, France and Japan, albeit not for the purposes of transmutation).
2. *Re-introduction* of the MA content into new fuel incorporating enriched uranium (the latter necessary to offset the neutron fraction consumed in transmuting MA nuclides) to contrive of a feasible, *in-reactor* scheme. This is not tried and tested practice and has not been demonstrated commercially, at scale. The LLFP component could be set aside having been separated (i.e., being potentially suitable for near-surface disposal after a period of approximately 300 years, as per the illustration in Figure 3).

A reduction in radiotoxicity of at least 99.9% is desirable, if approaches such as that outlined above are to be worthwhile. Otherwise, some waste will prevail with levels of radiotoxicity sufficient to justify permanent disposal and hence continue to pose an intrusion risk. This is considered potentially achievable (Doke, 1974) but is not without known technological challenges. It is these that have prevented it being achieved at scale thus far.

Further, in support of a circular economy, the energy expended by transmutation 'devices' to achieve this (some examples of which are described below) should be lower than was generated originally in making the waste now being treated. Conversely, the energy that might be saved by civilisation not needing to be long-term custodians of the waste, e.g., needing only simplified disposal facilities, etc. is also relevant. However, these factors have rarely been addressed in the literature to date because they are difficult to quantify without a firm commitment to one specific disposal scheme or another. Similarly, the waste generated by the transmutation process must be less (primarily in radiological terms one might assume) than that which is removed by it, as per the comment above. For example, greater end-of-life wastes are anticipated from reprocessing, separation and radioactive fuel manufacture etc. but of a lower radiotoxicity than for direct disposal of SNF or vitrified HLW. The trade-off in this respect is one of catering for a greater volume of lower-radiotoxicity end-of-life wastes compared to significantly small volumes of material sustaining high levels of radiotoxicity out to very distant time horizons.

5.2 Quantitative aspects

Schmidt (Schmidt, 1985) identifies the equation describing the quantity ΔN nuclei transmuted of the total N of a given isotope,

$$\Delta N = (\lambda + \sigma\phi) N \, \Delta T \tag{1}$$

where λ is the decay constant of the corresponding isotope, σ the sum of the capture and fission cross-sections, ϕ the neutron flux and ΔT the duration of irradiation. Equation 1 illustrates that the most effective devices will be those with as high a flux, that operate at a sustained level of operation and at neutron energies selected to optimise σ.

In addition to the MAs, transmutation has also been suggested for carbon-14, zirconium-93, technetium-99 and iodine-129 to reduce their impact beyond 1,000 years and, conversely, for tritium, carbon-14, krypton-85, strontium-90 and caesium-137 to reduce their impact in the first few hundred years of storage. Strontium-90 and caesium-137 are of additional interest due to their prominent decay heat and hence the potential for them to undermine safe geological disposal in the first few hundred years of disposal, notwithstanding their low cross-sections that might better favour storage for the purpose of decay, and the specific approach to accommodate decay heat of a given repository design, as discussed earlier.

Consideration of P&T must also be made with reference to the arguments against its use. These include geological disposal being regarded as safe with the long-term risk associated with it being negligible, thus discounting the need for P&T. Moreover, an argument could be made that P&T is unnecessary since a large inventory of legacy waste, that is not compatible with it and therefore requires a GDF, already exists and will continue to be generated during the time that P&T technologies and their associated fuel cycles become feasible, wherein additional contributions to the inventory in the form of secondary operational and decommissioning wastes from the P&T activity itself appear unavoidable.

Further, long-term research concerning alternatives such as P&T and the prospect of heralding unrealistic utopias could undermine public debate concerning geological disposal and the equitable identification of sites for such facilities. The perception of a solution being possible via P&T might also divert attention from the need to ensure HAW is in a form that can be consigned to a repository or detract from political decisions needed to commit to significant disposal plans and costs, whereas '*the incremental costs of introducing P&T appear to be unduly high in relation to the prospective benefits*' (Ramspott, 1992). Meanwhile, unsuccessful attempts at P&T might burden the environment and the current generation with the additional significant risk of yet more problem legacies and environmental impact rather than less, e.g., untransmutable waste streams comprising partitioned and concentrated americium species could be more challenging to dispose of than the original spent fuel etc.

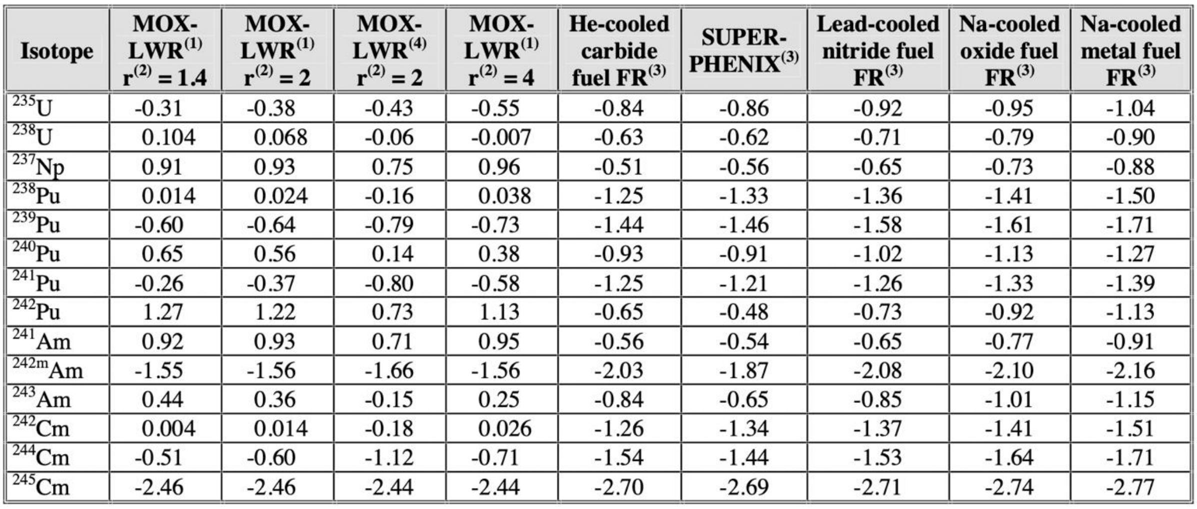
5.3 Neutron energy spectra and flux benefits

As noted above, a key requirement for successful transmutation is sufficient neutron economy. Consequently, adjustments might be necessary for transmutation schemes to accommodate waste-containing fuels to cater for neutrons consumed in transmuting problem isotopes whilst (depending on the method) sustaining other priorities (primarily the chain reaction in an associated reactor). This is usually quantified by the ratio of capture-to-fission cross-section, α, where,

$$\alpha = \frac{\sigma_c}{\sigma_f} \tag{2}$$

The dependence of α with increasing neutron energy illustrates why fast neutron spectra are often favoured for the transmutation of MAs by fission because the probability of neutron capture declines more quickly than that of fission. Increasing neutron energy avoids the higher rates of capture at thermal neutron energies and hence the potential for this to poison the process. This is important because, whilst problem isotopes are consumed by either process, fission maintains neutron production which is critical to sustaining the potential to transmute subsequent isotopes in a chain. Further, neutron capture risks an escalation in problematic MA content rather than its destruction, given the latter is the objective. This is also a reflection of how poor a fuel MAs are for thermal-spectrum reactors (Systems, 2013), transmutation aside, and hence why they are removed along with the SNF to be replaced by fresh fuel. It also highlights why MAs might be consumed in fast spectrum reactors buoyed by lower neutron capture losses.

This principle is extended (WPPT, 2006) by the so-called D_J-value, which is the number of neutrons needed to transform nucleus J and its reaction products into fission products. These are shown in Table 2 as a function of reactor/fuel type, reproduced from (WPPT, 2006). D_J is an arithmetic series comprised of the products of the effect on neutron economy and the associated cross-sections. A positive D_J implies that MA transmutation consumes neutrons (bad) and a negative D_J the converse, i.e., neutron production (good).



Isotope	MOX-LWR$^{(1)}$ $r^{(2)} = 1.4$	MOX-LWR$^{(1)}$ $r^{(2)} = 2$	MOX-LWR$^{(4)}$ $r^{(2)} = 2$	MOX-LWR$^{(1)}$ $r^{(2)} = 4$	He-cooled carbide fuel FR$^{(3)}$	SUPER-PHENIX$^{(3)}$	Lead-cooled nitride fuel FR$^{(3)}$	Na-cooled oxide fuel FR$^{(3)}$	Na-cooled metal fuel FR$^{(3)}$
^{235}U	-0.31	-0.38	-0.43	-0.55	-0.84	-0.86	-0.92	-0.95	-1.04
^{238}U	0.104	0.068	-0.06	-0.007	-0.63	-0.62	-0.71	-0.79	-0.90
^{237}Np	0.91	0.93	0.75	0.96	-0.51	-0.56	-0.65	-0.73	-0.88
^{238}Pu	0.014	0.024	-0.16	0.038	-1.25	-1.33	-1.36	-1.41	-1.50
^{239}Pu	-0.60	-0.64	-0.79	-0.73	-1.44	-1.46	-1.58	-1.61	-1.71
^{240}Pu	0.65	0.56	0.14	0.38	-0.93	-0.91	-1.02	-1.13	-1.27
^{241}Pu	-0.26	-0.37	-0.80	-0.58	-1.25	-1.21	-1.26	-1.33	-1.39
^{242}Pu	1.27	1.22	0.73	1.13	-0.65	-0.48	-0.73	-0.92	-1.13
^{241}Am	0.92	0.93	0.71	0.95	-0.56	-0.54	-0.65	-0.77	-0.91
^{242m}Am	-1.55	-1.56	-1.66	-1.56	-2.03	-1.87	-2.08	-2.10	-2.16
^{243}Am	0.44	0.36	-0.15	0.25	-0.84	-0.65	-0.85	-1.01	-1.15
^{242}Cm	0.004	0.014	-0.18	0.026	-1.26	-1.34	-1.37	-1.41	-1.51
^{244}Cm	-0.51	-0.60	-1.12	-0.71	-1.54	-1.44	-1.53	-1.64	-1.71
^{245}Cm	-2.46	-2.46	-2.44	-2.44	-2.70	-2.69	-2.71	-2.74	-2.77

Table 2: $\mathrm{D_J}$*-value for a variety of MA isotopes and as a function of fuel and reactor type [4]. Superscripts denote the following: (1)* $\phi=10^{14}$ *n / cm*^{2}*.s, (2)* r*=moderator-to-fuel ratio, (3)* $\phi=10^{15}$ *n / cm*^{2}*.s, (4)* $\phi=2.5\times10^{15}$ *n / cm*^{2}*.s.*

The data in Table 2 highlight the dependence of the D_J-value on neutron flux ϕ due to the competition between absorption rates (capture and fission) which are also dependent on decay, see below. Considering the key ‘problem’ isotopes of americium and ^{237}Np, for example, the data in Table 2 suggest that transmutation of ^{241}Am in all LWR designs *consumes* neutrons and that hence higher moderator-to-fuel ratios and fast spectra are needed to overcome this. Whilst these are less easy to construe in a LWR, compared to the fast reactor variants on the right of the table, the latter enjoy much less operational experience; likewise concerning neutron spectra for ^{237}Np.

The data in Table 2 also indicate that whilst raising the neutron flux helps (see MO$_X$ LWR @ ϕ *=2.5×10*^{15} *n / cm*^{2}*.s)*, a negative D_J cannot be effected by this action alone. Using a significantly higher *thermal* neutron flux (i.e., ϕ >10^{16} *n / cm*^{2}.s) has been postulated (Bowman, 1992) because this might realise the additional benefit of triggering two-stage conversion pathways. For example, the favourable thermal neutron *capture* cross-sections in ^{237}Np and ^{241}Am (σ_γ~150 b and 700 b, respectively) can be exploited to yield intermediary isotopic species (^{238}Np and ^{242}Am). The latter have more favourable thermal *fission* cross-sections (σ_f~1,500 – 5,000 b) compared to ^{237}Np and

^{241}Am, thereby affording a route to MA destruction via induced fission at thermal neutron energies, rather than fast, as per,

$$^{237}\text{Np} + n \rightarrow {}^{238}\text{Np} + n \rightarrow \text{fission}$$

$$^{241}\text{Am} + n \rightarrow {}^{242}\text{Am} + n \rightarrow \text{fission}$$

assuming the thermal neutron flux is high enough to enable the short half-lives of the intermediate species (^{238}Np, 2.12 days; ^{242}Am, 16 hr) to be skipped. However, contriving of such fluxes, thermalising these fields, isolating off-gasses from the accelerator, exhausting heat from the system, achieving chemical separation and maintaining a flowing lead spallation target results in a complicated, and as yet, only hypothetical design.

Fast neutrons not only yield favourable D_J values but are also necessary to breed ^{239}Pu from ^{238}U because more neutrons are produced in the fission of ^{239}Pu induced by neutrons at higher energies than thermal and hence, there are sufficient for the neutron capture on ^{238}U necessary for breeding, to sustain the chain reaction and to allow for unavoidable losses due to capture and leakage. At thermal neutron energies the neutron yield from thermal fission of ^{239}Pu and, for that matter ^{235}U, is too low. By contrast, the fission neutron yield in uranium-233 (^{233}U) *is* sufficient for thermal-spectrum breeding and hence it has been suggested that thermal waste burning might also be feasible. However, this is complicated because ^{233}U is not sufficiently abundant naturally and has to be derived from thorium-232 (^{232}Th) in the thorium fuel cycle. Also, because there is insufficient ^{233}U primordially, its breeding would have to be primed by ^{235}U before enough ^{233}U has been made to drive waste burning itself, and this must be recovered by reprocessing of the thorium fuel. Further, and as discussed below, ^{241}Am consumption and minimising ^{252}Cf production are both favoured at fast neutron energies. This notwithstanding, a potential benefit of this scheme is that the yield of uranium-232 (^{232}U) as a by-product of n/2n reactions in the thorium cycle and the complication of the high-energy γ-ray emissions associated with its thallium-208 (^{208}Tl) decay product is quenched at thermal neutron energies (Wojciechowski, 2018), and plutonium is not produced in the thorium fuel cycle.

6 Devices and concepts

Whilst non-neutron induced methods of transmutation (primarily using protons) are feasible, they do not induce fission as effectively and are thus more applicable to LLFPs than MAs. However, a large fraction of the energy generated in producing LLFPs is consumed if their conversion is attempted with protons. Consequently, most research focuses on neutron-based methods that target MAs; the use of nuclear explosives, thermonuclear reactors, spallation reactors and fission reactors has been postulated. For the purposes of this work, nuclear explosives are not considered further, given the Comprehensive Test Ban Treaty etc. and because of the potential for underground detonations to add to the subterranean anthropogenic radionuclide inventory whilst not affording designed-in imperatives of isolation and confinement akin to a GDF. In the main, techniques focus on the use of fast fission reactors and accelerator-driven systems, with fusion systems being a longer-term possibility.

6.1 Fission reactors

The neutron environment for transmutation afforded by fission reactors stems from, primarily, fission in uranium and plutonium. These elements have the advantage that relatively little development work has been necessary to postulate their use in transmutation. Further, their self-sufficiency is also attractive, i.e., the disposal of waste employs the same technology responsible for generating it via what might be termed the 'ultimate' closed fuel cycle. However, this does not remove the requirements for separation and manufacture of fuel that involves highly radioactive materials, and which mandates significant development effort having not been attempted before.

Various fission reactor technologies have been explored (albeit largely theoretically) for transmutation: the light water reactor (LWR), fast breeder, thorium high-temperature reactor, the CANada-Deuterium-Uranium (CANDU) reactor and the molten salt reactor. Technical requirements usually include reactivity, neutron flux, cost, risk, and compatibility with the waste to be transmuted, such as isotopic composition, radioactivity, and thermal output.

By way of conclusions, the conventional LWR is not considered appropriate for MA transmutation because, for example, when plutonium recycling is attempted simultaneously, the yield of curium and americium is significantly higher (max. ×25) than, say, than for fast breeders. This suggests that radioactivity and thermal output could be problems associated with the waste from LWRs used in this way; hence the emphasis earlier concerning reprocessing to remove residual quantities of uranium and plutonium for consumption, as per conventional MOx flowsheets, where assumed plutonium/americium removal efficiencies are typically ~99.9%.

Conversely, if a homogeneous uranium-fuelled scheme is followed, i.e., with the MA content entrained *in* reactor fuel with the plutonium removed, then the cost of fuel fabrication is likely to escalate prohibitively. On the other hand, in a heterogeneous scheme where there are separate elements or assemblies containing the MAs used as 'targets' in a reactor environment, self-shielding and power peaking might arise that are likely to result in low transmutation rates. To combat these effects, reintroducing maximum MA dopant levels of ~2% plutonium-equivalent (this means the MA induced fission contribution as per what would be afforded if it were plutonium), as per the self-generated equilibrium inventory in LWRs, has been suggested. To accommodate the MA contribution, in some cases, it has also been suggested that it might be possible to reduce the initial uranium-235 enrichment because this is supplanted by the fission in the MA, implying ^{235}U fuel savings.

Transmutation rates in heterogeneous fast breeder reactors (FBRs) (see Figure 3) are ~34.5% of the MA present which is ~25% higher than for the homogeneous equivalent with, say, 10% maximum MA content feasible, thereby replacing the necessary plutonium content. The pebble bed high temperature reactor (HTR) has also been cited as having potential, notwithstanding the need for a better understanding of fuel fabrication and reprocessing of a suitable fuel type (TRISO), for which this has yet to be achieved at scale, and bearing in mind TRISO has not been conceived with reprocessing in mind. On the other hand, the lead-cooled fast reactor (LFR) has also been studied (Sun, 2023) but, as a case in point, this work does not substantiate its use in terms of the implied need for reprocessing, fuel manufacture, regeneration, partitioning, and thermal consequences (particularly of the implied formation of ^{238}Pu); neither does this work specify whether (n,γ) or (n,fission) is the preferred scheme (it is likely to be neutron-induced fission given the fast neutron spectrum); this work is not alone concerning the sparse detail on these important factors.

Heterogeneous approaches can favour transmutation of curium and americium as these elements are consumed more quickly than uranium or plutonium, and hence the target assemblies might be swapped out more regularly than the host fuel. Further, target manufacture with its inherent handling issues can be done independent of conventional fuel manufacture. By contrast, the burnout rate for neptunium is like that of conventional, uranium-based fuels and may not favour this approach.

6.2 Spallation sources and accelerator driven systems

Unlike reactors, spallation neutron sources exploit the interaction of protons or deuterons on nuclei to yield the high surplus of neutrons necessary for transmutation purposes. High-current, high-energy proton accelerators are typically considered best in this regard. Depending on the target arrangement, target material, and incident proton energy, between 10 and 60 neutrons per interaction are anticipated which, with moderation, say 10^{16} n cm^{-2} s^{-1} thermal being needed for efficient FP transmutation, whereas a fast spectrum is advantageous for MAs, as described earlier.

A typical spallation scheme might also involve a reactor powering an accelerator electrically which is used to supply a beam to a spallation reactor. The latter might comprise a blanket into which FP material is fed having been separated from HLW, targeting typically ^{90}Sr and ^{137}Cs, notwithstanding the issues described above. This approach was found to be effective *theoretically* for FP reduction, but significant materials challenges are anticipated.

Accelerator-driven systems (see for example (Wu Y. B., 2016)) are often premised on the benefit that they combine the approaches described above with an accelerator driving a subcritical core which therefore cannot suffer a criticality accident because the reaction is halted by stopping the accelerator. However, particle accelerators are not known for their reliability when compared to a conventional fission reactor, recognising that higher levels of availability are essential to forecast operational costs. Further, the need to dissipate reactor decay heat remains, which has arguably been a more significant cause of reactor accidents than criticality. Energy balance is also important concerning how much energy the accelerator will require and related advancements in accelerator technology is likely to be necessary. Perhaps unsurprisingly, spallation sources are considered most relevant for the long-term radioactive inventory for ^{99}Tc and ^{129}I, and for MA transmutation because of the larger cross-sections for these nuclides, and for small amounts of exotic wastes; achieving the optimum beam energy and current is important.

6.3 Fusion reactors

More neutrons are available per unit energy per interaction in a fusion reactor (cf., ~0.06 neutrons MeV^{-1} assuming a deuterium-tritium process) than in a fission reactor (~0.014 MeV^{-1}). Although fusion has not yet been sustained on the industrial scale necessary for transmutation, the first proposal for waste burning in a fusion reactor stems back more than 50 years to Gore and Leonard (Gore, 1974). A wide variety of related studies have followed that have considered deuterium–tritium and deuterium–deuterium sources and applications for LLFP and MA transmutation. More recently (Wu Y. Z., 2006), the fission-fusion concept has been reported combining a subcritical fission core driven by a fusion source.

The works cited above suggest that a neutron wall loading of 10 MW m^2 would reduce long-term ingestion hazards associated with fission-borne wastes by three orders of magnitude but also specified that such a wall loading was beyond technical capabilities then (1985), as it is now. For example, ~0.1 MW m^{-2} was achieved in 1997 at JET (DTE1) with 10 MW m^{-2} being more akin to a fusion power plant device such as the planned DEMO; the ITER neutron wall load at 500 MW is anticipated to be ~0.7 MW m^{-2} (Packer, 2024). To consume MA content at the rate necessary to make a difference over what are likely to be considerable timescales, one might also argue that the supporting technology needs to be scalable now, which is not the case for fusion reactors. Moreover, fusion reactors are usually premised on the basis that they do not produce radioactive waste, rather than being combined in systems targeting its consumption, and the radioactivity involved may compromise the ease with which they are decommissioned relative to fusion systems limited to tritium etc.

6.4 Plutonium and/or MA recycling

Previous policy drivers for reprocessing of nuclear fuel in the UK included the recycling of plutonium, whilst MAs and LLFPs were separated for disposal. Plutonium recycling usually involves the re-introduction of plutonium in fresh fuel, removal of MAs/LLFPs for disposal, along with the discharge (rather than recycle) of the residual uranium to avoid the build-up of ^{236}U. Optimised accordingly, calculations for LWRs (thermal spectra) (WPPT, 2006) suggest a factor of 3 increase in partial plutonium consumption with increasing moderator-to-fuel ratio.

However, this is achieved with an increased build-up of MAs, as mentioned earlier, and consequently whilst plutonium *is* consumed there is a risk of exacerbating the MA problem as a by-product of plutonium consumption. The greater neutron economy achieved by increasing ^{235}U enrichment might favour both plutonium and MA consumption but also risks a curium-dominated MA inventory with its inherent fuel handling complications, also mentioned earlier, due to the relatively short half-life neutron and γ-ray emissions (principally from ^{244}Cm). On the other hand, fast neutron spectra result in increased MA production from plutonium-fuelled fission, particularly of curium, as per the ^{252}Cf issue described below.

Consequently, concepts suggesting the combination of plutonium consumption and MA ‘burning’ need to be treated with caution irrespective of the neutron spectrum involved. A further complication (Gwin, 1976) (Tovesson, 2010) is that the neutron cross-sections for ^{239}Pu have been measured experimentally only occasionally (Wright, 2025). Important features such as the s-wave resonance at 0.3 eV are not entirely understood theoretically, particularly in terms of the uncertainty in energy, despite reassessment having been mandated (De Saint Jean, 2014).

For homogeneous recycling, where the separation of plutonium and MA is not implied, the impact on fuel fabrication needs to be considered if acceptable core performance is to be achieved. Multiple recycle in a PWR, whilst consuming plutonium and neptunium, generates high levels of ^{252}Cf from ^{244}Cm, along with the decay heat, γ and neutron exposure issues associated with the former. The intermediate curium isotopes from ^{244}Cm through to ^{249}Cm are sufficiently long-lived to sustain successive activation, as is the ^{249}Bk daughter of ^{249}Cm which activates to ^{250}Bk which then decays to ^{250}Cf. Californium-250 (along with the more massive californium isotopes) has the large n,γ thermal cross-section needed for ^{252}Cf to form; the same is not true at fast neutron energies.

The ^{252}Cf issue described above influences the choice of MA to be targeted for P&T and whether waiting for this and the curium-244 to decline for say 30 years is worthwhile, prior to further recycle attempts and ultimate disposal of what remains (since ^{244}Cm is the majority curium isotope with a half-life of 18 years with ^{242}Cm replenished by the decay of the much longer-lived but usually less abundant ^{242}Am). Curium removal and allowing for its decay could increase capacity in a repository significantly beyond that made possible by plutonium and americium removal alone (Wigeland, 2006), given its neutron emission and thermal load are avoided; conversely, the curium separation to enable this would be difficult to achieve at scale. The californium issue has caused MA transmutation in thermal-spectrum PWRs to be disregarded (Warin, 2007).

Heterogenous recycling is potentially much slower because of the limited flux possible, the requirement to separate americium and curium from plutonium, the hypothetical use of a large fraction of a fast reactor ‘park’ to achieve the required reductions, and the need to control complex in-reactor power distributions.

Although, as described, most P&T schemes assume plutonium and uranium separation for reuse in a reactor, as otherwise plutonium becomes the principal radiotoxic component after only a few decades in-repository, plutonium is made and consumed unavoidably in all uranium-fuelled fission reactors (i.e., not that derived via thorium), with the isotopic composition and neutron spectrum being controlling factors; the capture-to-fission cross-section α for ^{239}Pu and ^{241}Pu at thermal energies is relatively low, i.e., implying more fission than capture, highlighting their tendency to contribute neutrons via induced fission, in-reactor. Conversely, even-numbered plutonium isotopes (predominantly ^{240}Pu in terms of abundance) exhibit high values of α indicating a tendency to absorb neutrons.

Attempts to recycle plutonium (predominantly ^{239}Pu, being the majority isotope) lead to even-numbered plutonium accumulation and hence more neutron absorption and thus fewer neutrons; increasing the fissile content in response to sustain recycling (that is, the proportion of ^{239}Pu and ^{241}Pu) leads to more even-numbered content and, consequently, increased MA production. Consequently, the potential for widespread plutonium recycling via MOx in conventional thermal-spectrum reactors such as PWRs is limited. Fast neutron spectra (c.f., sodium fast reactors) bring into play fission from the wider range of susceptible isotopes, reduced MA production via capture relative to its consumption by fission, greater neutron balance and hence the potential for better stabilised, safer core performance. However, this has only been perfected at scale in a few plant worldwide (e.g., Dounreay, Monju, Superphenix etc.) and never specifically for actinide burning.

6.5 Accelerator-driven systems

Overall, dedicated ‘cores’ for transmutation have a reduced margin for reliance on key passive safety functions because of a much lower delayed neutron fraction, reduced Doppler effect (and hence fuel-temperature feedback) due to a lower isotopic fertile abundance (^{238}U) and a reduction in the void reactivity coefficient (concerning plant using liquid metal coolants). Accelerator-driven systems (ADS) overcome these issues by operating at sub-critical levels. However, assuring the subcritical state via continuous and effective monitoring would be crucial, as would be stringent requirements of accelerator reliability. In comparison with burning in conventional reactors, relatively little experience exists in the operation of the powerful accelerators that would be needed to drive fast subcritical reactor cores.

7 Context in respect of fuel cycle

Another early acknowledgment of the perceived potential of P&T is that of Strauss (Strauss, 1973). This highlights the industrialisation log jam, i.e., ‘…it will be quite some time before any single concept is fully developed and instrumented to provide a working disposal facility’ but also the importance of fuel cycle policy in the context of P&T. This work describes how in 1973 reprocessing was in full flow as a private enterprise at several sites in the US, with forecasts of both the growth in nuclear electricity generation and the need for reprocessing being 4-fold by 1980, and the anticipation that this would ‘continue on up thereafter’.

By 1977, however, commercial reprocessing in the US had been deferred indefinitely by presidential decree (Carter, 1977) on the basis that ‘…a viable and economic nuclear power program can be sustained without…’ and to curb the potential for nuclear weapons to proliferate. Interest in nuclear power slowed after the Three Mile Island accident in 1979, and due to increased competition from cheap gas and the significant monies lost associated with unfinished reprocessing plant.

Not now having a reprocessing capability and, for example, the generic design assessment (GDA) for the EPR (S. Tromans, 2024) that rule it out in the future, suggests P&T as being suitable either only for legacy HLW (notwithstanding much of this in the UK has been put beyond further partitioning by vitrification), or that distinct partitioning capabilities would need to constitute part of any new P&T infrastructure. The latter would incur very significant cost and would need to be compatible with fuel variants aside from LWR and have ready access to a separated raffinate stream from an existing reprocessing facility. It is worthy of note that the US recently passed a cost-sharing program of the order of $10M to help nuclear start-ups afford the federal licencing process for nuclear fuel recycling (Kaufman, 2024). Consequently, this situation might be subject to change, albeit remotely.

It is important to note, that across the several P&T scenarios such as integrated (Gen-IV fast reactors) or separate ('double strata') waste minimisation, partial management of MA (c.f., just americium and neptunium) from existing reactors or a targeted reduction of legacy transuranic stockpiles (ADS) etc., all mandate reprocessing and are incompatible with an open fuel cycle. If partial management is implied of a more limited number of isotopes, then separation of curium from americium is still likely to be necessary which, as stated earlier, is not trivial and would likely incur extra cost and environmental impact.

Existing spent fuel streams in the UK can vary widely, and the systems designed to separate and manufacture fuel from them would need to be likewise compatible. This will require significant research with, as above, some separations (c.f., Am and Cm) being challenging and as has been remarked (Bowman, 1992), the 'chemistry costs are not well understood'. The inadvertent production of secondary wastes would also need to be monitored if the justification for the approach is not to be undermined. As highlighted above, fuel production is also challenging because of significant decay heat and neutron emission requiring remote handling, with americium also likely to cause helium production *in* the fuel. It is worthy of note that, whilst MOx fuel manufacture has been achieved at scale, integrating MA content into these processes has not and nor has this been sustained.

In January 2025 the UK Government decided to immobilise the UK's civil separated plutonium inventory. The concepts of waste burning and the use of plutonium in reactors are often conflated. Consuming plutonium in mature (thermal) reactor technologies in the UK on a conventional basis, i.e., MOx, would require dedicated fuel manufacturing facilities for which the existing Springfields capability is incompatible whilst attempts at Sellafield have been mothballed at significant cost, as above, amidst there being no commercial appetite. More complicated fuel cycles, such as molten salt liquid fuels have not been explored in this regard in the UK beyond the laboratory scale and, if attempted, would mandate primary pyro-metallurgical plutonium/MA extraction not yet attempted at scale in the UK; integrating 'waste' for consumption such as an MA stream would encounter the challenges described above concerning handling of significantly radioactive materials, not to mention uncharted requirements in terms of reactor control with an enriched, MA-containing fuel component. As above, significant adoption of plutonium-based fuel cycles would mandate a great deal of nuclear data assessment with which to inform simulations, reactor design and fuel composition. Moreover, a portion of the UK's plutonium inventory is contaminated, i.e., by Cl from storage in PVC, carbon from incomplete calcination and by Am due to age. This is a challenge for reusability, particularly of older stocks, implying the requirement for disposal in any event.

8 Summary of the academic literature

Over 1200 academic papers have been published concerning transmutation and radioactive waste, spanning the period of nearly 60 years since Slanksky and Buckham's pioneering work in 1969

(Slanksky, 1969). The field escalated significantly after ~1990, partly due to the re-emergence of ADS, peaking at nearly 60 papers per year through to ~2015, with interest declining since.

Amongst the most highly cited works are those of Bowman et al. (Bowman, 1992) concerning high-flux, accelerator-driven transmutation with thermal neutrons but which does not appear to have been revisited of late; the 2011 review by Salvatores and Palmiotti (Salvatores, 2011) primarily concerning the benefit of more efficient utilisation of repository space, and combatting the potential for long-term increases in actinide mobility and intrusion. The most highly cited paper relating to ADS is that of (Wu Y. B., 2016) closely followed by that concerning the Belgian MYRRHA project (it Abderrahim, 2001). The former focuses on the fuel design, heralding the benefits of a subcritical core. It offers a design which can operate in both critical and subcritical modes but does not expand on the challenges of remote handling and chemical separation. The latter focuses on an ADS designed to cater for both MA and LLFP transmutation by affording fast and thermal neutron environments, respectively. (Wu Y. Z., 2006) concerns an ambitious, subcritical, fusion-driven hypothetical design concept aiming to transmute both MA and LLFP wastes whilst breeding new fissile fuel.

In terms of important and substantial papers, that quantifying the benefits of P&T in terms of avoided heat generation to repository capacity is that of (Wigeland, 2006) followed by the comprehensive assessment of the field of (National Research Council, 1996), albeit focused on Yucca Mountain (since superseded) with a similarly extensive summary being that of (Ramspott, 1992) focused on the US geological disposal system. A relevant summary of the chemical separations priorities concerning P&T is that of (Choppin, 2000) and the work of (Modolo, 2014) and the references therein in terms of advanced separations. A summary of the French P&T research program through to 2007 is that of Warin (Warin, 2007), and that of the US accelerator-driven waste transmutation program is that of Beller et al. (Beller, 2001).

9. Conclusion and Recommendations

This review of radioactive waste burning by nuclear transmutation indicates that realising practical benefits on an industrial scale might be achievable, but which would involve considerable investment to overcome various technical challenges. It also indicates that the options that might be available for development are heavily reliant on the nuclear material inventory and on the availability of nuclear fuel cycle facilities.

In the UK, the nature of the nuclear material inventory is such that not all the separated uranium and plutonium can be recycled in reactors. This determines that geological disposal will remain necessary, not least for other legacy intermediate level waste and for vitrified HLW containing LLFPs and MAs, neither of which are suitable for P&T. In some circumstances it might be envisaged that reducing the size of the inventory for disposal may have benefits, however, such arguments require clarity on the available geology and the costs of construction of a GDF. When such clarity is established, arguments for reducing the amount of waste for on disposal are only likely to be compelling if a significant proportion of both the plutonium and the minor actinides are destroyed.

Further clarity on the period of prolonged storage and associated costs to render P&T feasible would be needed, and compared to the cooling time and additional footprint associated with disposal, and there would need to be underpinning evidence that waste burning of the MAs only, with LLFPs such as ^{129}I, ^{135}Cs and ^{99}Tc untreated, has a safety case benefit for the GDF. There would also need to be sufficient commercial appetite for separated U and Pu derived from spent fuel to justify reprocessing, and proposals to do so would be subject to public consultation.

Government should ensure that proposals for AMRs or fuel cycle facilities with the potential to treat radioactive waste are subject to scrutiny at an early stage, for example, during the application stage of the Justification of Practices Regulations[3].

Proposals from vendors for waste burning in reactors or fuel cycle facilities should:

- Specify the *fuel cycle* with which they are concerned, and the *component of radioactive waste* that is targeted for burning;
- Include quantitative estimates of the target *reduction in radiotoxicity*, i.e., the % removed; the estimated increase in *GDF capacity*; the *duration* required for treatment; estimates of *secondary waste arisings* and *recycle/partitioning losses*; and *value for money*;
- Provide clarity on the absolute extent of destruction possible via waste burning because *plutonium burning* might imply the production of *additional minor actinide waste* whereas *minor actinide burning* in isolation does not imply the plutonium is destroyed; and
- Provide clarity on the isotope and chemical nature of any secondary radioactive wastes that may be generated through P&T and associated fuel cycles, including a proposed method for radionuclide immobilisation or encapsulation that is compatible with long-term storage and / or disposal.

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[3] https://www.gov.uk/government/publications/the-justification-of-practices-involving-ionising-radiation-regulations-2004-guidance-on-their-application-and-administration

Glossary

ADS	Accelerator-driven system
CANDU	Canada deuterium uranium reactor
DEMO	Demonstration power plant
DTE	Deuterium-tritium experiments
EPR	Evolutionary pressurised water reactor
FP	Fission product
GDF	Geological disposal facility
HAW	High-active waste
HLW	High level waste
ILW	Intermediate level waste
ITER	International thermonuclear experimental reactor
JET	Joint European torus
LFR	Lead fast reactor
LLFP	Long-lived fission product
LWR	Light-water reactor
MA	Minor actinide
MOx	Mixed-oxide
NEA	Nuclear energy agency
OECD	Organisation for Economic Cooperation and Development
RepU	Reprocessed uranium
RRR	Relative radiotoxic risk
P&T	Partitioning and transmutation
SNF	Spent nuclear fuel
TRISO	TRi-structural ISOtropic
UKAEA	United Kingdom Atomic Energy Authority

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Feedback

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