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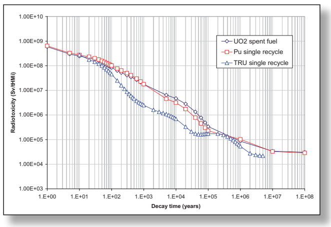
Minor Actinide Burning in Thermal Reactors

A Report by the Working Party on Scientific Issues of Reactor Systems



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NEA No. 6997

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Cover photo: The Minor Actinide Laboratory at the EC Institute for Transuranium Elements in Germany (European Communities).

Foreword

The actinides (or actinoids) are those elements in the periodic table from actinium upwards. Uranium (U) and plutonium (Pu) are two of the principal elements in nuclear fuel that could be classed as major actinides. The minor actinides are normally taken to be the triad of neptunium (Np), americium (Am) and curium (Cm). The combined masses of the remaining actinides (i.e. actinium, thorium, protactinium, berkelium, californium, einsteinium and fermium) are small enough to be regarded as very minor trace contaminants in nuclear fuel. Those elements above uranium in the periodic table are known collectively as the transuranics (TRUs).

The operation of a nuclear reactor produces large quantities of irradiated fuel (sometimes referred to as spent fuel), which is either stored prior to eventual deep geological disposal or reprocessed to enable actinide recycling. A modern light water reactor (LWR) of 1 GWe capacity will typically discharge about 20-25 tonnes of irradiated fuel per year of operation. About 93-94% of the mass of uranium oxide irradiated fuel is comprised of uranium (mostly ^{238}U), with about 4-5% fission products and ~1% plutonium. About 0.1-0.2% of the mass is comprised of neptunium, americium and curium. These latter elements accumulate in nuclear fuel because of neutron captures, and they contribute significantly to decay heat loading and neutron output, as well as to the overall radiotoxic hazard of spent fuel. Although the total minor actinide mass is relatively small – approximately 20-25 kg per year from a 1 GWe LWR – it has a disproportionate impact on spent fuel disposal, and thus the longstanding interest in transmuting these actinides either by fission (to fission products) or neutron capture in order to reduce their impact on the back end of the fuel cycle. The combined masses of the trace actinides actinium, thorium, protactinium, berkelium and californium in irradiated LWR fuel are only about 2 parts per billion, which is far too low for them to be considered candidates for transmutation.

Fast reactors are needed to transmute TRUs because fast neutron cross sections are generally more effective in the fissioning of TRUs. However, studies have demonstrated that TRU transmutation rates can also be achieved in thermal reactors, although with serious limitations due to their accumulation through recycling and their impact on the safety of the plants.

The transmutation of TRUs could potentially be carried out in many thermal reactors operating today, while waiting for a similar programme in fast reactors to allow commercial-scale operations in 20 to 30 years or more. Investment in fuel cycle plants could lead to even more efficient transmutation in fast reactors towards the end of the century. In the interim, the potential contribution of thermal reactors should not be overlooked.

A considerable amount of research has already been conducted on minor actinide transmutation in thermal reactors, and the purpose of this report is to summarise the findings of this research. The report concentrates on general conclusions related to thermal reactors and foregoes a lengthy examination of the more technical details. While a commercial-scale implementation programme for a specific reactor type will need to address a multitude of very specific questions, the objective of this report is to provide the broad understanding necessary to inform high-level strategy and decision making.

Chapter 1 of the report provides an introduction to minor actinide nuclear properties and discusses some of the arguments in favour of minor actinide recycling. The introduction is not specific to thermal reactors but could apply to any nuclear system in general. Chapter 2 discusses the potential role of thermal reactors in minor actinide recycling; Chapter 3 looks at the various technical issues and challenges presented by minor actinide recycling; Chapter 4 examines fuel cycle issues; Chapter 5 presents implications for thermal reactor operations, fuel design, core management, and safety/dynamics responses; Chapter 6 explores operations and safety issues; Chapter 7 considers the economics of minor actinide recycling and attempts to make some quantitative estimates insofar as current knowledge permits; Chapter 8 discusses research and development needs and Chapter 9 summarises findings, making recommendations for the direction of future R&D efforts.

Acknowledgements

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1. Introduction

1.1 Neptunium, americium and curium nuclear properties

It would be helpful to start by reviewing the basic nuclear decay and cross-section properties of Np, Am and Cm, because this will highlight the reasons why these elements are considered a challenge in the nuclear fuel cycle. Table 1.1 lists the nuclear decay properties of the main neptunium, americium and curium isotopes, including their half-lives, decay modes, branching ratios, gamma emissions, effective dose coefficients, total neutron emission rates and decay heat outputs.

Of the TRUs in irradiated nuclear fuel, neptunium, americium and curium are the three that are usually considered as candidates for partitioning and transmutation (P&T). Although there are other TRUs present, these three are dominant in terms of mass. Irradiated LWR fuel typically contains, about 600 g/tHM of neptunium; 850 g/tHM of americium and about 50 g/tHM of curium at 10 years after discharge. Because the ^{241}Am arises from ^{241}Pu decay and curium isotopes are relatively short lived, these proportions vary depending on the cooling time after discharge and the discharge burn-up.

Neptunium, which is predominantly represented by the single isotope ^{237}Np, is a significant contributor to long-term radiotoxicity, because of its very long half-life. However, ^{237}Np does not contribute significantly to decay heat output.

Americium is generally considered as a prime candidate for transmutation because it is present in relatively large amounts and is a significant contributor to gamma activity and radiotoxicity, especially after about 500 years cooling time when the contribution of fission products has decreased by several orders of magnitude. Americium has reasonably large nuclear cross-sections and is amenable to destruction in an intense neutron flux by a combination of neutron captures and fissions. In irradiated nuclear fuel, ^{241}Am is the dominant nuclide, though there are small but significant quantities of ^{242}Am, ^{242M}Am and ^{243}Am.

Curium makes a significant contribution to gamma activity and radiotoxicity and is also a major contributor to neutron emissions. Curium is not well suited to transmutation, because the fission and capture cross-sections of the principal isotopes (^{242}Cm and ^{244}Cm) are quite low and it is not usually possible to transmute them effectively. Although ^{242}Cm has a very short half-life (163 days), it is continually generated in irradiated fuel from the decay of ^{242}Am (141 year half-life).

The relatively short half-life of ^{244}Cm (18.1 years) suggests that interim storage may perhaps be a better strategy for this isotope, allowing time for it to decay naturally before being incorporated in recycling targets. Such a decay storage strategy would help reduce the strong heat source from ^{244}Cm in fabricating the target fuels. Reducing the mass of ^{244}Cm in the target fuel would also reduce the production rate of ^{245}Cm. Although much of the ^{245}Cm produced during irradiation of target fuels in a thermal reactor undergoes fission, that which does remain will be problematic because of its 8 500-year half-life. One rationale for irradiating Cm and not separating it, is that for some of the proposed separation processes it is not possible to separate Am and Cm individually and therefore if Am is to be recycled, then Cm must be as well.

The decay modes are included principally because many of these nuclides undergo spontaneous fissions that are accompanied by neutron emissions, which determine the shielding requirements for storage, transport and handling. The spontaneous fission rate for ^{237}Np is zero. The spontaneous fission rates of the americium isotopes are all very low and although the typical Am-Cm mix from irradiated LWR fuel contains about 90% Am, the 10% or so curium content dominates neutron emissions. This is mainly due to the contribution of ^{244}Cm, which is the predominant curium isotope and has a short half-life and a relatively high spontaneous fission branching ratio. In addition to spontaneous fission neutrons, there is also a neutron source from (α,n) reactions.

Gamma emissions are difficult to characterise, because in some cases there are multiple lines (e.g. there are 129 lines for ^{241}Am). Table 1.1 identifies the gamma emissions with the highest intensity (which usually tends to be at low energy) and also the line (or lines) with the highest intensities at high energies. These high-energy gamma emissions are the most difficult to shield and are important in defining the shielding requirements.

Direct gamma emissions from ^{237}Np are relatively mild because of the low decay rate and the low energy of the highest intensity emissions. However, its short-lived daughter ^{233}Pa is in secular equilibrium with ^{237}Np and has an energetic gamma of 312 keV at intensity 0.386 that dominates overall gamma emissions from the ^{237}Np decay chain.

^{241}Am has a 60 keV line at 0.359 intensity. ^{243}Am is a strong gamma emitter, with a75 keV line at 0.674 intensity. Moreover, ^{239}Np, which is a short-lived daughter of ^{243}Am decays, has a very energetic gamma line at 278 keV of 0.144 intensity. The shorter half-life of ^{241}Am and its higher abundance means that ^{241}Am is the dominant contributor. Gamma emissions from curium are dominated by ^{244}Cm.

The last column in Table 1.1 indicates total neutron emissions from the oxide, which combines spontaneous neutron emissions with α-n emissions. For ^{241}Am and ^{243}Am, α-n emissions dominate over spontaneous fissions (the figures quoted in Table 1.1 give the total neutron emissions for the oxide), but for ^{242}Cm and ^{244}Cm the spontaneous neutron source dominates. Since Cm is the dominant neutron emitter in the Am-Cm mix, the form of the material (whether metal, oxide, carbide or nitride) does not significantly affect neutron emissions.

An important consideration for neptunium, americium and curium is radiotoxicity. The ingestion effective dose coefficients as listed in ICRP-72 [1] are the same for all the isotopes listed in Table 1.1 (5.0E-4 Sv/Bq) so that their direct contributions to radiotoxic potential are proportional to the decay rates. Because of its very long half-life, ^{237}Np only contributes significantly to total radiotoxicity at very long cooling times. ^{241}Am is the dominant contributor of the americium isotopes, due to its higher abundance and its half-life being shorter than ^{243}Am. ^{244}Cm is the highest contributor of the Cm isotopes, due to its abundance and short half-life. The total radiotoxicity depends on the ingestion effective dose coefficients of the respective decay chains (i.e. contributions both from the decay of the parent nuclide and all the daughters in its decay chain), but the various decay chains are not overly dissimilar in this respect.

The contribution of americium and curium to radiotoxicity decays by a factor of about 2 after 100 years, 10 after 1 000 years, and then falls to about 0.5% of its initial value after 10 000 years. At cooling times of over 1 000 years, the long-lived nuclides, ^{243}Am and ^{245}Cm are the dominant contributors. This decay profile is much slower than the decay profile of fission products and with full recycling of uranium and plutonium would become the limiting factor controlling the radiotoxic content of a repository at 1 000 years.

Table 1.1: Decay properties of neptunium americium and curium

Nuclide	Half-life	Decay modes	Branching ratios	Principal gamma emissions	Neutron emissions (spontaneous + alpha-n from oxide) n/sec/gHM	Decay heat (W/kgHM)
^{237}Np	2.14E6 years	Alpha	1.0 α	29 keV intensity 0.153; 86 keV intensity 0.123	351.4	2.07E-2
^{239}Np	2.35 days			278 keV intensity 0.144		
^{241}Am	432 years	Alpha and spontaneous fission (+neutron emissions)	1.0 α 3.77E-12 SF	60 keV intensity 0.359; 662 keV intensity 3.6E-6	2752	114.7
^{242}Am	16.02 hours	Not relevant because of short half-life				
^{242M}Am	141 years	Internal decay; alpha and spontaneous fission (+neutron emissions)	0.9955 internal decay; 0.0045 α; 1.6E-10 SF	49 keV intensity 1.9E-3; 163 keV intensity 2.3E-4	186.6	4.5
^{243}Am	7370 years	Alpha and spontaneous fission (+neutron emissions)	1.0 α 3.7E-11 SF	75 keV intensity 0.674; 662 keV 1.1E-5	138.6	6.4
^{242}Cm	163 days	Alpha and spontaneous fission (+neutron emissions)	1.0 α 6.33E-8 SF	44 keV at 3.25E-4 intensity; two at up to 942 keV combined intensity 6E-7	2.360E7	121228.0
^{243}Cm	29.1 years	Alpha and positron emission (β+)	0.9976 α 0.0024 β+	278 keV at 0.14 intensity; five up to 755 keV combined intensity 8.5E-5	4.869E4	1860.7
^{244}Cm	18.1 years	Alpha and spontaneous fission (+neutron emissions)	1.0 α 1.35E-6 SF	43 keV at 2.5E-4 intensity; 818 keV at 7.2E-7 intensity	1.092E7	2841.8
^{245}Cm	8 500 years	Alpha	1.0 α	175 keV at 0.095 intensity	125.3	5.8
^{246}Cm	4 730 years	Alpha and spontaneous fission (+neutron emissions)	1.0 α 2.61E-4 SF	45 keV at 2.8E-4 intensity	8.759E6	10.2

Source: JEF-PC nuclear data program, OECD-NEA; neutron emissions from FISPIN inventory programme.

Because of the gamma and neutron emissions, handling Am and Cm demands substantially more shielding than is required for MOX fuel fabrication. For example, at the Institute for Transuranium Elements (ITU) in Karlsruhe, Germany, the Minor Actinide Laboratory used to fabricate small quantities of Am and Cm into targets has 50 cm of water shielding and 5 cm of lead shielding at working level. This is less than the shielding requirement for irradiated fuel, but substantial nonetheless. At the ITU Minor Actinide Laboratory, normal operations involving Am and Cm samples are carried out using remote manipulators, supported with remote control and robotics. The cells are, however, standard gloveboxes that allow manual intervention in the absence of radioactive sources, as required for maintenance and repair.

Table 1.2 shows a sample of neutron cross-sections for the various Np, Am and Cm isotopes. It shows 3-group fission and capture cross-sections weighted with an LWR neutron spectrum, in the fast, epithermal and thermal energy ranges. These cross-sections are only intended for illustration and in practice will vary depending on the initial enrichment and on the burn-up, as well as the specific LWR type. The data do not include minor cross-sections such as multiple neutron events (n, 2n) or (n, 3n).

The epithermal cross-sections include resonance self-shielding, which is highly dependent on the specific reactor conditions. In particular, the resonance region cross-sections depend on the physical form of the minor actinide or transuranic (MA-TRU) fuel, whether inhomogeneous or homogeneous, for which different resonance self-shielding factors would be expected to apply.

The ratios of fission to capture are indicative of whether the nuclides will primarily undergo fission or neutron captures as shown in Table 1.2. This table shows the fast, epithermal and thermal cross-sections for fission and capture when weighted by LWR neutron energy spectrum. ^{241}Am is more complicated because neutron captures may lead to either the ground state of ^{242}Am or its metastable state. In order to clearly demonstrate this complexity, the data for ^{241}Am have been split over three rows, one for ^{241}Am(n,γ)^{242}Am, one for ^{241}Am(n,γ)^{242M}Am and one for ^{241}Am fission.

In the case of ^{237}Np, the dominant nuclear reaction over the three neutron groups is capture leading to ^{238}Np. However, ^{238}Np is predominantly removed by fissions. The end-point is the destruction of ^{237}Np by fissions via two neutron interactions, with the first neutron capture as the rate-controlling step.

^{241}Am also predominantly undergoes neutron capture, transmuting either to ^{242}Am or ^{242M}Am (the branching ratio favours the ground state ^{242}Am). The ^{242}Am can then be removed by fission or capture, with fission largely dominating for the ^{242M}Am branch, but slightly less so for the ground state branch. In either case, removal of ^{241}Am is via two neutron reactions, the first again being rate limiting.

Thermal captures also dominate over fissions for ^{242}Cm and ^{244}Cm, though for ^{243}Cm and ^{245}Cm the thermal fissions dominate. The overall trend is that neutron captures tend to predominate in a thermal neutron spectrum, building up higher nuclides in the process. Fast neutron interactions make only a minor contribution to the overall reaction rate, because the cross-sections are much lower.

In terms of the core design behaviour, the cross-sections in Table 1.2 suggest that although neptunium, americium and curium contribute limited useful energy output from fissions in a thermal reactor, captures are nevertheless significant. Therefore, their behaviour in a thermal reactor is to some extent analogous to that of burnable poisons.

Table 1.2: Neutron interaction cross-sections for Np, Am and Cm weighted with LWR neutron spectrum (barns)

Nuclide	Fast fission	Fast capture	Epithermal fission	Epithermal capture	Thermal fission	Thermal capture	Dominant reaction
^{237}Np	0.86	0.62	0.02	50.49	0.01	164.50	Capture
^{238}Np	2.34	0.07	74.43	8.46	934.9	93.74	Fission
^{241}Am to ^{242}Am	-	0.73	-	75.37	-	654.70	Capture
241Am to ^{242M}Am	-	0.14	-	11.26	-	72.75	
^{241}Am fission	0.86	-	0.60	-	4.98	-	
^{242}Am	2.19	0.33	99.49	29.97	1087.00	2423.00	Fission
^{242M}Am	2.24	0.23	118.00	17.44	4713.00	1106.00	Fission
^{243}Am	0.71	0.72	0.09	143.31	0.03	43.71	Capture
^{242}Cm	1.18	0.17	0.33	9.48	2.81	7.60	Capture
^{243}Cm	2.48	0.07	150.00	24.63	233.40	48.44	Fission
^{244}Cm	1.07	0.28	0.67	36.89	0.49	7.16	Capture
^{245}Cm	2.12	0.13	54.93	10.00	784.90	119.80	Fission
^{246}Cm	0.00	0.10	0.00	7.31	0.00	0.67	Capture

1.2 Rationale for minor actinide transmutation

1.2.1 Radiotoxicity reduction

The black points in Figure 1.1 plot the total ingestion radiotoxicity (in Sieverts) of 1 tonne of spent LWR UO_2 fuel, irradiated to a burn-up 45 GWd/t as a function of cooling time after discharge. The radiotoxicity decay profile can be divided crudely into three segments:

The first segment, which extends up to a cooling time of ~500 years, is dominated by the radiotoxicity of the fission products. These are mostly relatively short-lived, except for a small contribution from some very long–lived fission products such as ^{129}I. The short-lived fission products dominate the total radiotoxicity up until about 500 years, after which time all the main contributors have decayed. The second segment, which covers decay times between 1 000 years and 10^5 years is dominated by the contribution of plutonium and the minor actinides, which by the end of this period have essentially all decayed. Finally, in the segment after 10^5 years, the radiotoxicity is determined by the very long-lived fission products and ^{237}Np.

The red points in Figure 1.1 plot the equivalent radiotoxicity decay profile, but in this case assuming a reprocessing cycle in which the plutonium has been recycled a single time as MOX fuel. The MOX recycling case data have been generated in such a way that the radiotoxicities per unit of useful electrical output are comparable so as to provide a fair comparison. The radiotoxicity of this single recycling MOX scenario is initially slightly higher than the equivalent UO_2 fuel, because the MOX fuel inventory has a higher content of minor actinides. For longer periods up to ~10^5 years, however, the MOX case is slightly lower because some of the plutonium has been destroyed by fissions. Overall, the radiotoxicities of the UO_2 and single recycling MOX scenarios are comparable.

It may be technically possible to design geological repositories that will isolate the bulk of the radionuclides from the biosphere for hundreds or even thousands of years. The design objective would be initially to contain the radionuclides for perhaps several hundred years and then rely on engineered and natural barriers to delay the subsequent transport of mobile radionuclides out of the repository and into the biosphere until perhaps several thousand years later. But no conceivable repository design could be guaranteed to isolate the radionuclides over the geological time scale of 1 million years or more, which would be needed for complete decay. Therefore, repository designers start with the premise that on geological timescales there will be non-containment of the radionuclides, but that the design should aim to delay the breakdown of containment as long as possible.

When discussing the design of a repository, there are two distinct (but linked) aspects. One is the purely technical assessment and the second is the presentation to the public. The technical assessment will necessarily involve complex analysis such as the migration rate of radionuclides and peak dose uptake over very long timescales (up to 1 million years), which are not readily understandable to the general public. For periods up to about 1 000 years, the general public might reasonably be expected to understand that engineered solutions might be effective, because historic structures have survived for such periods and longer. But for longer periods, abstract technical arguments are likely to be much less persuasive.

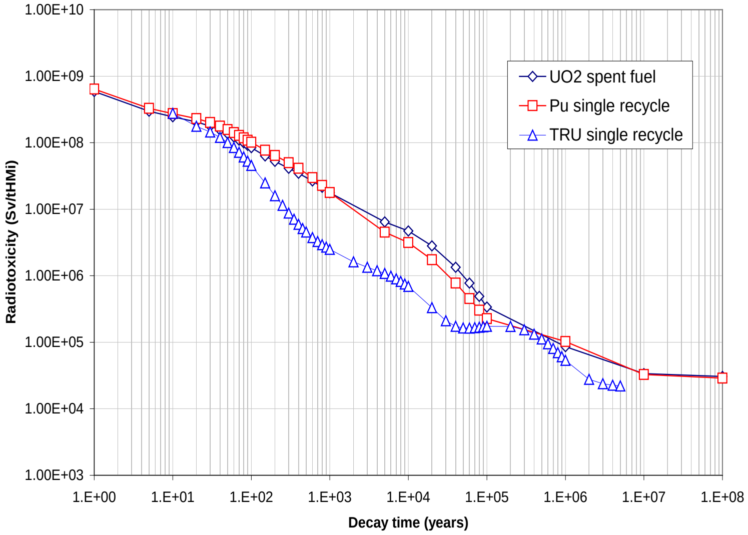
It would be more persuasive if the total radiotoxicity could be brought down to the level of the third segment on the more understandable 1 000-year timescale. The blue points in Figure 1.1 plot how this might be achieved, by separating out the minor actinides and transmuting them. This has been done in a somewhat artificial way here, by separating all the TRU from the 1 tonne of spent UO_2 fuel and assuming they are irradiated to almost complete destruction in a thermal reactor spectrum. Although this is not a rigorous approach, it serves here to indicate what might be achievable in an ideal scenario and therefore indicates the maximum extent to which minor actinide burning could reduce the radiotoxicity. In this case, the total radiotoxicity reaches a much lower plateau in the second segment, approaching that of the third (long-term) segment. This is an important part of the rationale for minor actinide partitioning (separation) and transmutation, because it brings forward the long-term radiotoxicity plateau towards the 1 000 year timescale, which the general public could better relate to.

This same argument has often been expressed in a different way by regarding the radiotoxicity of the original uranium ore prior to extraction as a target threshold for the repository to meet. The argument is that once the radiotoxicity of the spent fuel has decayed to below that of the uranium ore, then this is a convenient level at which future concern can be regarded as no longer significant. While this argument has some merit, especially when applied for comparative purposes, it is nevertheless somewhat arbitrary because although the radiotoxicity matches that of the original ore, it is difficult to argue that the geological environment of the repository exactly matches that of the original ore body. For this reason, this report does not to use this threshold definition, but rather regards minor actinide transmutation as a possible means of accelerating the timescale during which the radiotoxicity decreases.

Although radiotoxicity is a very important parameter, its impact on the performance of a geological repository is limited. A recent OECD/NEA study [2] has highlighted that the removal of minor actinides from the inventory of a geological repository has very little impact on peak radiological environmental dose. This is because, in the repository environments, the minor actinides tend to have low mobility and environmental dose is controlled instead by volatile elements. However, the study does note that removal of minor actinides does have a major beneficial effect in intrusion scenarios. Intrusion scenarios involve penetration of waste packages by deep drilling activities, where there is the potential of the drilling operators receiving significant radiological doses from the minor actinides, depending on the specific scenario postulated.

All scenarios for minor actinide transmutation are dependent on the development of effective chemical separation methods for separating minor actinides from fission products, uranium and plutonium, with high separation factors being achievable. The presence of significant amounts of minor actinides in fission product streams has the potential to undermine the justification for minor actinide transmutation strategies.

Figure 1.1: Radiotoxicity profiles for LWR UO$_2$, LWR MOX and LWR TRU recycling



1.2.2 *Decay heat reduction*

The principal contributors to decay heat from irradiated fuel at short cooling times (up to 50-70 years) are the medium half-life fission products ^{137}Cs and ^{90}Sr. Decay heat from the transuranics such as plutonium, americium and curium only contributes to about 10% of the total at 5 years cooling. The transuranic contribution varies only slightly up to cooling times of 100 years and represents about 50% of the total at a cooling time of about 70 years, at which time the short and medium half-life contributors have largely decayed.

Since the capacity of any geological repository is likely to be limited primarily by the decay heat output, it is beneficial to remove the transuranics by recycling them and also to separate out and store the ^{137}Cs and ^{90}Sr until they have decayed. Such a strategy of TRU recycling and Cs/Sr managed storage allows for a considerably increased capacity in the geological repository and defers the need for follow-on repositories.

1.3 Minor actinide transmutation strategies

1.3.1 *Homogeneous/heterogeneous recycling*

There are two options for recycling minor actinides, homogeneous and heterogeneous recycling.

Homogeneous recycling involves the incorporation of minor actinides as an integral component of the fuel. In an LWR, this might entail a concentration of a few per cent of minor actinides dispersed in the fuel rods. Fuel assemblies containing minor actinides might constitute a fraction or all of the fuel assemblies in the core. Homogeneous recycling is normally considered best suited to neptunium recycling and less suitable for americium and curium. This is partly because the burn-out rate of neptunium fits relatively well with that of conventional uranium or plutonium fuel. Another advantage is that the radiological shielding requirements for neptunium are not as onerous as those for americium and curium. Homogeneous recycling has the benefit of allowing the same fuel design to be used across the core. The disadvantage, however, is that all of the fuel must be manufactured within a fabrication facility that can meet the shielding and containment requirements, which are more demanding than those of conventional uranium or plutonium fuel fabrication. Neutron emissions and decay heat output from the minor actinides are major complicating factors in fuel fabrication. Homogeneous recycling therefore demands a fuel fabrication plant that has a capacity sufficient to deal with large fuel throughputs, potentially sufficient to meet the entire refuelling requirements of the reactor. This could have an adverse impact on the economics of homogeneous recycling.

A recent study of the homogeneous recycling of neptunium in PWR MOX fuel has evaluated the impact on neutron source, gamma source and decay heat [3] and has well illustrated this impact. The neutron and γ-source characteristics of the spent Pu+Np MOX fuel (with 11.5 w/o Pu+Np) and the decay heat behaviour of the fuel were compared against the reference MOX case (9.0 w/o Pu). The study demonstrated a slightly reduced neutron for the Pu+Np case compared with the reference MOX case, while the γ-sources were very similar. However, decay heat was slightly elevated in the Pu+Np case.

Heterogeneous recycling is generally considered a better option for americium and curium. It involves loading the minor actinides in separate, distinct fuel assemblies, often referred to as “target assemblies” or “target elements”. These can be positioned in the core in such a way as to optimise burn-out of the minor actinide content, and they may also be retained for a longer dwell time than the conventional fuel to maximise burn-out. This increased flexibility in loading is more suitable for americium and curium. In addition, heterogeneous recycling allows the manufacture of the target assemblies to be carried out independently of the manufacture of the bulk fuel, and thus the stringent shielding and containment measures only need to be applied to a relatively small throughput. The need to ensure compatibility between the bulk fuel and the target fuel, however, is a disadvantage.

1.3.2 Single/multiple recycling

Complete transmutation of the minor actinides is not possible in any reactor system, meaning that a residual mass of minor actinides will remain at the end of the irradiation of MA-TRU fuel. With a single recycling scheme this residual mass will eventually contribute to the inventory of the geological repository. With homogenous recycling, the residual minor actinide content would be emplaced in geological disposal as an integral part of the spent fuel. With heterogeneous recycling, the irradiated target assemblies would be encapsulated for geological disposal, with no further reprocessing required. Separating minor actinides from fission products and uranium/plutonium is a costly step, as is incorporating them in homogeneous fuel or heterogeneous targets. With a single recycling strategy, these very significant costs are only incurred once, though with only a limited reduction of the minor actinide inventory in the repository.

Multiple recycling of homogeneous MA-TRU fuel implies the reprocessing of the bulk fuel and the separation of the minor actinides for incorporation in a new generation of fuels. For heterogeneous minor actinide targets, multiple recycling implies the reprocessing of the target assemblies and separation of the minor actinides for incorporation in a new generation of target assemblies. This might be accomplished best by blending the recycled irradiated targets with LWR spent fuel at the beginning of reprocessing operations. Such blending dilutes the heavier actinides and ensures that the fissile concentration remains high enough for continued cycles of transmutation.

Multiple recycling has the potential to achieve very low residual levels of radiotoxicity and decay heat, but suffers from a number of disadvantages:

- The total mass of minor actinides that needs to be separated and processed into target assemblies is increased, implying higher plant throughputs and higher overall costs.
- The total mass of minor actinides that are circulating in the active fuel cycle (i.e. separation, fabrication storage, transport, in-reactor irradiation) is higher. This increases the exposure of nuclear workers to radiological dose and increases the risk posed both to workers and the public in the event of a core accident leading to uncontained release of radionuclides.
- The timescales required to actually implement multiple recycling are potentially very long, bearing in mind that each cycle of separation, target fabrication, in-reactor irradiation, cooling and transport may take 10 years or more. A requirement to recycle 10 times, for example, would take at least 100 years, which is long relative to the lifetimes of nuclear facilities and difficult for decision makers to commit to.

For these reasons, if it is necessary to implement multiple recycling of MA-TRU fuels, it is important that the number of recycling processes should be kept to a manageable level, for example, no more than 2 or 3.

1.3.3 Storage

An alternative strategy to the recycling of curium is to store the irradiated fuel for >30 years prior to reprocessing. Although the half-life of ^{242}Cm is very short (163 days), it is continually replenished because it is from decays of ^{242}Am (^{242}Cm is in secular equilibrium with ^{242}Am, which undergoes β^- decay with a 141 year half-life). ^{244}Cm is the dominant curium isotope and has an 18.1 year half-life, so that extended storage of >30 years will allow most of it to decay prior to reprocessing.

After reprocessing, the remaining curium could be separated and stored. After a sufficiently long period of interim surface storage, the curium and associated decay products (mainly ^{240}Pu) could be encapsulated and packaged for geological disposal. The advantage of storage is that the geological repository does not need to accommodate the neutron emissions and heat load of the curium, potentially allowing for an increase in capacity. The disadvantage is that it may be difficult to design a reprocessing plant to separate curium, and there would of course be a cost associated with the dedicated fuel cycle plants needed to treat and store the curium stream.

1.3.4 Minor actinide recycling in thermal reactors versus fast reactors

Concerns arising from the existence of plutonium and minor actinide stockpiles accumulated over the years have led to the emergence of systems potentially capable of reducing these stockpiles through burning and transmutation processes.

A used UOX fuel (UO_2 enriched in ^{235}U), originating from a PWR (pressurised water reactor), contains approximately 1% plutonium (Pu) with 60 to 70% fissile isotopes. After a few cooling decades, this plutonium represents more than 90% of the radiotoxicity of the fuel. Plutonium multi-recycling allows a significant reduction of the physical quantity of plutonium while benefitting from the natural energy available.

In a PWR where the thermal neutron flux is enhanced, the production of neutrons is primarily controlled by the contributions of ^{239}Pu and ^{241}Pu. The even isotopes of plutonium behave like poisons ($\sigma f/\sigma c<<1$). With plutonium recycling in a PWR core, two effects dominate, each of them reinforcing the other. The deterioration of the isotopic quality, with the accumulation of the even isotopes, requires an increase in the total mass of plutonium (via the increase in the plutonium content). The increase in ^{239}Pu and

^{241}Pu masses contributes to softening the neutron flux towards the thermal zone where reduced production rates will decrease the plutonium quality, further increasing the plutonium content needed. In addition, accumulation of minor actinides, in particular of ^{242}Pu, further contributes to the production of higher actinides (^{243}Am and then curium).

In a SFR (sodium fast reactor), the dominant contribution to the neutron balance is provided by the odd isotopes, although all the isotopes contribute to neutron production via nuclear fission. Consequently, the production of higher actinides decreases in particular for minor actinides like americium and curium. In addition, the neutron balance is more favourable in a SFR spectrum with a surplus of neutrons, which offers the possibility of ^{238}U conversion and the transmutation of minor actinides.

The deterioration of the plutonium quality in a thermal spectrum makes it possible to consider limiting MOX (Mixed Oxide fuel PuO_2-UO_2) multi-recycling in the standard PWR. The solutions to ease the multi-recycling will come with innovative fuels (MOX-UE, CORAL, APA) and require, in all cases, a limitation of the initial plutonium content to respect the safety criteria; a contribution from enriched uranium is required to maintain the burn-up rates objective. These options would make it possible to save, at best, between 10% and 20% of natural uranium in a nuclear PWR park. Studies to increase the conversion factor (PWR HFC) have been conducted by reducing the moderation ratio and by introducing fertile pins (depleted U) into the core. These changes are limited in terms of natural uranium savings for a PWR nuclear park (at best 30%, i.e. using only 1% of Uranium ore).

The favourable neutron spectrum in fast neutrons makes it possible to stabilise or even improve the initial content of plutonium and its isotope content during multi-recycling in an SFR core. In contrast to what occurs in a PWR, the improvement and the stabilisation of the plutonium characteristics during SFR multi-recycling will not deteriorate the core safety coefficients and would thus allow unlimited multi-recycling with respect to safety. The effectiveness of a fissile core for the reaction chain is characterised by the number of neutrons emitted for a core neutron absorption (the η parameter). The values of this parameter for the ^{235}U (1.9) and ^{239}Pu (2.3) isotopes show that plutonium is the best choice in SFR cores. The physics of the fast spectrum also allow the possibility of adapting the SFR core to the production or the burning of plutonium with appropriate changes in the fertile zones (depleted U) and in the core design. The production of plutonium (breeding) can be increased by the use of blankets (Breeding Gain: BG = 0.2 for SUPER-PHENIX). This has been demonstrated in the PHENIX plant where reprocessing has been done since 1980 for the whole core around 5 times, hence allowing the plant to burn the fuel it has been producing itself (breeding ratio: 1.16). The consumption of plutonium in SFR can be adjusted so that the plutonium inventory can either be stabilised or even decreased. Studies show that one can change within the same plant, the breeding core into a burner core. The CAPRA fuel concept in an EFR core for instance can burn approximately 75 kg/TWhe of plutonium. A demonstration has been carried out in PHENIX for some sub-assemblies. Furthermore, the number of neutrons in excess for achieving the neutron balance can be used to efficiently burn minor actinides being produced by PWR operation and hence reduce to very low levels the radiotoxicity of nuclear waste accumulated over the years.

1.3.5 Minor actinide recycling in thermal reactors as a technology demonstrator

TRU recycling in thermal reactors could have a role to play as a technology demonstrator for more advanced minor actinide recycling schemes that might follow later when, for example, Generation IV systems are operational.

The major benefit of beginning minor actinide recycling in existing thermal reactors is that it would remove the long-lived radiotoxic nuclides from the fission product waste. The investment cost for minor actinide recycling will be significant in the fuel cycle plants in particular. Investment in TRU recycling in LWRs could, however, prepare for TRU recycling in Generation IV reactors.

If it is possible to achieve sufficient throughputs of minor actinides in LWRs and sufficient burn-out in thermal reactors, then it would be self-justifying.

Investment in the fuel cycle plants accommodating TRU will ease future commitments to Generation IV systems, which will be able to fully benefit from such an investment. This, in turn, will reassure decision makers, who may otherwise be reluctant to commit to investment in both fuel cycle plants and Generation IV systems. A relatively small-scale demonstration programme (perhaps in a limited number of thermal reactors) would be sufficient to demonstrate the technology on an industrial or sub-industrial scale before a full deployment, which would have to become economically viable.

2. Potential role and objectives of minor actinide recycling in thermal reactors

This section surveys the various theoretical studies that have been carried out which examined minor actinide burning in thermal reactors.

2.1 Light water reactor (LWR) studies

In preparation for this Expert Group study, OECD/NEA compiled a list of recent publications (see Bibliography) relevant to minor actinide recycling in LWRs. Table 2.1 summarises these publications, identifying the country in which the work was carried out, the study topic and highlighting the main conclusions.

The situation in France is that the nuclear fuel cycle has always been viewed as a very long-term commitment with a very ambitious target of eventually achieving total self-sufficiency with fast reactors. France is also very keen to recycle minor actinides to reduce long-term radiotoxicity and to this end regards fast reactors as the best means to achieve this end. Although it is accepted in France that minor actinide recycling in LWRs is technically feasible, any such programme would merely be an intermediate technology demonstration step towards the eventual goal of minor actinide recycling in fast reactors.

The dominance of the US in the bibliography reflects the particular interest there in more short/medium term goals which are oriented towards reducing demand in the high level waste repository. At the time these studies were carried, the US was interested in analysing the role of the recycling of irradiated fuel as an alternative to direct disposal. This strategy would have decreased the capacity utilised in the repository, especially if the minor actinides are recycled and some heat producing fission products (Sr and Cs) are removed and stored. The driver in the US at the time was focused on the short/medium term gains that could be made with TRU recycling in LWRs and this is the reason for the interest shown there. More recently, the US position has been evolving and at the time of writing waste management approaches in the US are being re-examined.

Although the various studies Table 2.1 investigated different scenarios, a number of common themes emerge from the survey of the conclusions:

- There is a strong bias towards PWRs rather than BWRs, the reason for which is not clear. BWR fuel assembly design is more heterogeneous and in particular relies more heavily on uranium/gadolinium burnable poison rods and for this reason it is possible that minor actinide recycling in PWRs is considered technically more feasible. Despite the bias, the fact remains that PWR and BWR neutron spectra are not so dissimilar and it is reasonable to assume that what is feasible in PWRs is also feasible in BWRs (though detailed confirmatory analysis would be needed).

- All of the studies confirm that worthwhile quantities of minor actinides can be recycled in LWRs, with useful quantities destroyed in the irradiation lifetime of the fuel assemblies containing the minor actinides.

- Many of the studies specifically analysed the impact of minor actinide recycling on the 3-D core behaviour and were not limited to just lattice calculations of single assemblies. Many of these studies also analysed the impact on thermal-hydraulic characteristics as well. All of the studies that examined these aspects confirmed satisfactory core and thermal-hydraulic behaviour. The fact that so many independent studies have reached the same conclusion is very strong evidence of feasibility.

2.2 Heavy water reactor (HWR) studies

Atomic Energy of Canada Ltd (AECL) have carried out very extensive studies which demonstrate the flexibility of CANDU heavy water reactors (HWRs) to burn fuels with different materials compositions. Two recent studies have investigated the potential of CANDU HWRs for MA-TRU burning [4] [5].

Reference [4] modelled the depletion behaviour of two different types of fuel rods containing TRU in the CANFLEX advanced fuel assembly bundle. The first fuel type was MOX fuel comprised of TRU in a conventional UO_2 matrix. The second rod type was an Inert Matrix Fuel (IMF) comprised of TRU in an inert matrix. The MOX fuel assembly was assumed to reach a very high burn-up of 45 GWd/tHM, while the IMF assembly was taken to the equivalent energy output. The MOX fuel showed a worthwhile decrease in TRU inventory of 40% at discharge, while the IMF fuel achieved a reduction of 71%. The long term heat load for the MOX assembly showed a reduction in the range 30-60%, while that for the IMF was 70-80%.

Reference [5] investigated the potential of CANDU HWRs and LWRs for burning americium. The HWR analysis modelled the irradiation of CANFLEX bundles in a CANDU reactor, with fuel rods containing various initial fuel compositions of uranium and americium. The discharge burn-ups of the fuel bundles ranged from 7.5 and 21 GWd/tHM. In some of the cases examined the americium was assumed to be homogeneously distributed in the fuel, while in two cases a centre pin consisting of a 3.7 or 7.0 w/o americium in zirconia was used. The various cases showed transmutation fractions for the americium ranging from 46% to 79%. These are slightly higher than the transmutation fractions achieved in the LWR variants examined, the highest of which was 71%. This indicates that HWRs have the potential to burn minor MA-TRU slightly more efficiently than PWRs.

Another observation from the HWR study was that HWRs burning americium recycled from PWR fuel reprocessing in a synergistic fuel cycle can achieve very high support ratios. In the synergistic fuel cycle, the americium from reprocessing the spent fuel from several LWRs is fed to a single HWR. The support ratio is the electrical output of these LWRs divided by the electrical output of the americium burning HWR and values as high as 20 were obtained: so that 1 GWye of americium burning HWR capacity is needed to burn the americium from 20 GWye of LWR output. It is important to minimise the capacity of MA-TRU burning reactors relative to conventional reactors, especially if MA-TRU burning is less economic, as it minimises the overall economic penalty on the entire reactor fleet. The equivalent support ratios with MA-TRU burning in PWRs were smaller, the best ratio being just 3.

The study in [5] shows that a synergistic mix of LWRs and HWRs could be effective in reducing the inventory of ^{241}Am produced per GWye of useful energy output. One of the main benefits is a reduction in the decay heat output of spent nuclear fuel, which may ultimately lead to a reduction in the footprint of the geological repository.

A limitation of the studies performed to date on minor actinide burning in HWRs is that they have been confined to lattice code studies. These are adequate to give an indication of the transmutation rates that can be achieved with different fuel compositions, taking into account of the local neutron spectrum. However, additional analysis will be required to develop the studies further, specifically whole-core nuclear design assessments. Whole-core analyses are needed to determine the impact on nuclear safety related parameters such as power peaking factors, reactivity coefficients and shutdown margins, which would be used to confirm safe behaviour of the core under normal operation and in fault conditions. During this next level of analysis, it is possible that further optimisation of minor actinide loading strategies may be needed to ensure satisfactory core safety performance.

2.3 High-temperature reactor (HTR) studies

High-temperature reactors (HTRs) have a long history dating to the 1960s. HTRs are graphite moderated (thermal neutron flux) helium cooled reactors with a ceramic fuel form, capable of very high coolant outlet temperatures (theoretically up to ~1 000°C) and therefore very high thermal efficiencies. HTR technology demonstrators and prototypes were built in the 1960s in the United Kingdom (DRAGON), Germany (Jülich HTR, THTR-300) and US (Peach Bottom, Fort St Vrain), but were abandoned in the mid 1980s partly because the commercial environment was not yet suitable and partly as a reaction to the Chernobyl accident. More recently there has been a revival of interest in HTRs firstly in South Africa (where plans to build a prototype pebble bed modular reactor (PBMR) have recently been abandoned) and in China, where a twin unit PBMR (2x250 MWe) is planned.

HTR cores are known to be very flexible in that they can accommodate a wide range of fuel types and the ceramic fuel is capable of very high burn-ups. These characteristics make HTRs potentially very suited to TRU burning. This was highlighted in a study of HTRs carried out in the EURATOM 5th Framework HTR-N Project [6]. This examined several different aspects of HTRs, starting with a study of the accuracy of nuclear design modelling methods and nuclear data libraries when applied to the HTTR-10 and PROTEUS-HTR research reactor cores. The study also included a cell burn-up benchmark exercise for an HTR containing plutonium fuel and an assessment of the characteristics of HTRs as plutonium burners. The HTR-N Project, however, did not carry out any detailed analysis of HTRs with MA fuels and was intended as a preliminary analysis to prepare for a later MA study.

The follow-up study was eventually carried out under EURATOM 6th Framework. This was the PUMA Project, which was specifically designed to investigate the potential role of HTRs in burning minor actinides. PUMA - the acronym stands for "Plutonium and Minor Actinide Management in Thermal High-Temperature Gas-Cooled Reactors" – was a Specific Targeted Research Project (STREP) within the EURATOM 6th Framework. The PUMA project ran from September 1, 2006, until August 31, 2009 and involved a consortium of 14 European partner organisations and one from the US [7].

PUMA's main objective was to investigate the possibilities for the utilisation and transmutation of TRU, particularly minor actinides in contemporary and future (high temperature) gas-cooled reactor designs. The PUMA project assessed the impact of the introduction of Pu/MA-burning HTRs at three levels: fuel and fuel performance (modelling), reactor (transmutation performance and safety) and reactor/fuel cycle facility park.

PUMA investigated the core physics of Pu/MA fuel cycles in HTRs and demonstrated nuclear stability of a Pu/MA HTR core, under both normal and abnormal operating conditions. The starting point of this investigation comprised the two main contemporary HTR designs, namely the pebble-bed type HTR, represented by the South-African PBMR, and hexagonal block type HTR, represented by the GT-MHR. The results confirm the flexibility of contemporary (and near term future) HTR designs and their ability to accept a variety of Pu- and Pu/MA-based fuels (possibly in combination with thorium). The results show a significant capability for Pu/MA burning, while maintaining the favourable safety characteristics of conventional uranium-fuelled HTRs.

Pu/MA transmuters are envisaged to operate in a global system with a mix of different reactor types and fuel cycle facilities. Fuel cycle studies were therefore performed in PUMA to study synergistic fuel cycles involving HTRs and other reactor types (LWRs) and fast reactors and to quantify waste streams and radiotoxic inventories.

PUMA also assessed the technical, economic, environmental and socio-political impacts. PUMA showed that a Pu/MA-loaded HTR would give worthwhile reductions on the quantities of TRU in spent fuel and high level waste for geological disposal.

2.4 Survey of experimental studies

A recent IAEA report on minor actinide fuel development [8] summarises the experimental work that has been carried out worldwide on minor actinide fuels and they are listed here in Table 2.2. To date, all the experiments carried out on minor actinide fuels have been at laboratory scale. The irradiations have mostly been in fast reactors, though many of the results are also relevant to thermal reactors.

Table 2.1: Summary of historical studies of minor actinide recycling in thermal reactors

Study	Year	Country	Scope	Conclusions
Preliminary multi-cycle transuranic actinide partitioning and transmutation Studies, ORNL/TM-2007/24	2007	US	Minor actinide (Np-Am-Cm) recycling in thermal (LWR) and fast spectrum (ABR) systems	LWRs more efficient at burning MA than ABRs; optimum performance by irradiating Pu or Pu/Np in fast spectrum systems and Am/Cm in thermal spectrum.
Can thermal reactor recycling eliminate the need for multiple repositories? C.W. Forsberg, E.D. Collins, J.P. Renier, C.W. Alexander	2004	US	Minor actinide (Np-Am-Cm) recycling in LWRs	Recycling of TRU increases Yucca Mountain capacity x 5. Cs/Sr storage + TRU by factor 42. TRU recycling in LWRs and Cs/Sr storage effective at increasing effective capacity of repository.
Multiple tier fuel cycle studies for waste transmutation, R.N. Hill *et al.* ICONE-10-22575	2002	US	Transuranic recycling in LWRs, LWR transmutation core, HTR, ADS and fast spectrum reactors	LWRs capable of reducing TRU waste radiotoxicity to <0.6% of once-through cycle. Limiting factor is separation plant losses.
Advanced LWR Multi-Recycling Concepts, E.A. Hoffman, R.N. Hill, T.A. Taiwo, ANS_Trans_Vol_93_Nov05_363-364	2005	US	Multiple recycling of transuranics in LWR in inert matrix fuel (IMF)	Demonstrates that it is neutronically feasible to recycle transuranics in inert matrix fuel (IMF) in LWRs
Assessment of transuranics stabilisation in PWRs T.K. Kim, J.A. Stillman, T.A. Taiwo, R.N. Hill, P.J. Finck, M. Salvatores, PHYSOR 2002	2002	US	Transuranic stabilisation in CORAIL PWR fuel assembly	Confirms that the neutronics of TRU multiple recycling in PWR is feasible with the CORAIL assembly design and variations thereof.
Feasibility of multi-recyling of Pu and MA in PWRs using combined non-fertile and UO_2 (CONFU) fuel, E. Shwageraus, P. Hejzlar, M.S. Kazimi, MIT_CONFU_0164	2003	US	Transuranic burning in PWR using fertile free fuel rods	3-D core models used to confirm feasibility of MA and Pu burning in PWR.

Table 2.1: Summary of historical studies of minor actinide recycling in thermal reactors (*continued*)

French R&D on the Partitioning and Transmutation of Long-Lived Radionuclides – An International Peer Review of the 2005 CEA Report. OECD/NEA. ISBN 92-64-02296-1.	2006	France	Comprehensive review of entire transuranic fuel cycle by OECD/NEA International Review Team (IRT). Five thermal reactor recycling strategies considered.	Concludes that although PWRs can minimise continuing production of minor actinides, they do not have the potential to completely burn their own MA production (therefore not good for phase-out scenarios), Curium accumulates. Incorporating even small amounts of MAs has a large impact on the fuel cycle. IRT views thermal reactors as useful for an interim plutonium management but not worthwhile for MA. Needs to be viewed in context of unified French programme.
Feasibility of Multi-recycling of Pu and MA in PWRs using combined non-fertile and UO_2 (CONFU) fuel. E. Shwageraus, P. Hejzlar, M.S. Kazimi, Center for Advanced Nuclear Energy Systems, Nuclear Engineering Department, Massachusetts Institute of Technology, Cambridge, MA, US	2003	US	Whole-core analysis of combined non-fertile and fertile free fuel pins (CONFU) used to recycle transuranics.	Demonstrates neutronic and T/H feasibility. However, does not consider larger implications for the fuel cycle.
Effect of Thermal-Spectrum Transmuter Deep Burn-up of Transuranics on Fast-Spectrum Transmuter Performance J.A. Stillman, R.N. Hill	2003	US	Assessment of transuranic deep burn concepts in PWR core. MOX 2x recycling and several CORAIL variants.	Deep burn-up of TRUs feasible in PWRs.
Fuel Cycle Impacts of Uranium-Plutonium Co-extraction, Temitope Taiwo, Frank Szakaly, Taek-Kyum Kim, Robert Hill	2008	US	Analysis of PWR and ALMR core performance with different transuranic fuels including co-extraction CX MOX.	Confirms feasibility of re-use of co-extraction (CX) recycling material in PWR.
Assessment of a heterogeneous PWR assembly for plutonium and minor actinide recycling, T.A. Taiwo,* T.K. Kim, J.A. Stillman, R.N. Hill, M. Salvatores, P.J. Finck	2005	US	Multiple recycling of transuranics in PWRs with CORAIL fuel. Calculates material attractiveness.	Confirms feasibility of TRU multi- recycling material in PWR.
Actinide transmutation using pressurised water reactors, M. Visosky, P. Hejzlar, M. Kazim, Proceedings of ICAPP '06 Reno, NV US, June 4-8, 2006, Paper 6197	2006	US	Analysis of CONFU-B (CONFU-burn-down) fuel in PWR, designed for deep burn of transuranics.	CONFU-B fuel assemblies demonstrated multi-recycling of TRU.

Table 2.1: Summary of historical studies of minor actinide recycling in thermal reactors (*continued*)

Repository impact of limited actinide recycling, Roald A. Wigeland, Theodore H. Bauer, Robert N. Hill, John A. Stillman, Proceedings of GLOBAL 2005,Tsukuba, Japan, 9-13 October 2005, Paper No. 496	2005	US	Analysis of limited recycling of transuranics in PWRs. MOX, CORAIL and inert matrix fuel (IMF) examined.	Confirms benefits to repository loading are possible with limited TRU recycling in PWRs, just by keeping the TRUs in the fuel cycle. Increase in repository drift loading is worthwhile but modest with limited PWR recycling. Fast reactors needed ultimately to deal with TRU to obtain maximum benefit in drift loading capacity.
Preliminary multi-cycle transuranic actinide partitioning-transmutation Studies, Emory D Collins *et al.* Global Nuclear Energy Partnership Systems Analysis Working Group, February 2007	2007	US	System analyses report on preliminary multicycle transuranic actinide partitioning-transmutation studies	Detailed feasibility study of multiple partitioning/transmutation cycles in LWRs.

Table 2.2: Summary of minor actinide fuel experiments

Experiment	Reactor	Fuel materials tested
SUPERFACT-1	Phénix	U-Pu, U-Pu-Np, U-Pu-Am, U-Np and U-Np-Am oxides
EFFTRA-T4	HFR Petten	Am in $MgAl_2O_4$
X501	EBR-II	Am and Np in metal fuel pins
METAPHIX	Phénix	U-Pu-Zr, U-Pu-Zr-MA-Rare Earth, U-Pu-MA metal fuel rods
ECRIX-B and ECRIX-H	Phénix	U-Pu-Zr, U-Pu-Zr-MA-Rare Earth, U-Pu-MA metal fuel rods
CAMIX-COCHIX	Phénix	Am-Zr-Y oxide, Am-Zr-Y oxide in MgO matrix
FUTURIX-FTA	Phénix	U-Pu-Am-Np-Zr metal Pu-Am-Zr metal U-Pu-Am-Np nitride Pu-Am-Zr-N Pu-Am oxide in Mo cermet matrix Pu-Am-Zr oxide in Mo cermet matrix Pu-Am oxide in MgO matrix
FUJI	HFR Petten	Np-Pu-U oxide sphere-pac, Pu-U oxide sphere-pac, Pu-U oxide vibro-pac, Pu-U oxide pellet
AFC-1	ATR	Pu-Am-Zr, Pu-Am-Np-Zr l, Pu-Zr, U-Pu-Am-Np-Zr, U-Pu-Am metal
DOVITA	BOR-60	U-Pu-Np oxide
HELIOS	HFR Petten	Am cermet/cercer
HELIOS AM-1	Joyo	Am MOX and Np-Am MOX
AMBOINE	BOR-60	U-Am oxide, U-Am-Mg oxide
CONFIRM	HFR Petten	Pu-Zr nitride and Am-Zr nitride
GACID	Monju	U-Pu-Am-Np-Cm oxide

3. Issues associated with utilisation of MA-TRU fuels

3.1 Recycling modes – homogeneous and heterogeneous

In any type of reactor, there are two possible modes for recycling TRUs and these are known as homogeneous and heterogeneous recycling:

In the homogeneous recycling mode the TRUs are incorporated as an integral component of the nuclear fuel. For example, the normal fuel material such as low enriched UO_2 or PuO_2/UO_2 mixed oxide (MOX) might also contain a minor actinide component such as NpO or AmO_2, smeared homogeneously. Practical considerations usually demand that the TRU component in homogeneous fuel should be at a low concentration, representing no more than a few percent of the total heavy metal mass. Keeping the TRU concentration low limits the radiological dose and decay heat outputs that need to be managed in manufacturing, transport and handling. It also limits the extent to which the minor actinide loading affects the nuclear design performance of the fuel and core.

Homogeneous recycling has some beneficial features:

- The basic characteristics of the fuel are only modified to a small extent if the minor actinide loading is kept low. This implies that the existing knowledge base for uranium and MOX fuels is largely still applicable. In particular, if the minor actinide content is kept low, the underlying fuel performance characteristics remain close to those of the dominant component of the fuel matrix (either UO_2 or MOX), thereby allowing continuity with the vast knowledge base of mainstream LWR fuel behaviour.
- Ensuring that all the fuel conforms to the same mechanical, thermal-hydraulic and nuclear design simplifies the design and licensing in all these areas.

On the other hand there are some disadvantages:

- The higher radiological dose and decay heat outputs in manufacturing affect the entire fuel load. The extra costs involved in engineering the fuel manufacturing plant are imposed on a fuel manufacturing plant that must have sufficient capacity to accommodate the entire fuel load. Any problems affecting the MA-TRU fuel would compromise availability of the entire fuel load.
- The concentration of MA-TRU in the homogeneous fuel is typically quite low and this may restrict their overall utilisation, if the overall mass of minor actinides incorporated in the heterogeneous fuel is low.
- The MA-TRU is constrained to the same core residence time and the same thermal and fast neutron fluences that normal fuel would be exposed to. Because the fission and capture cross-sections of the minor actinides are relatively low, this implies that the minor actinides may not be fully transmuted before the fuel is discharged.
- Handling, interim storage, transport and disposal of the irradiated fuel is complicated by the higher radiological doses and heat outputs associated with the MA-TRU.

- If the irradiated fuel is to be recycled, the mainstream recycling plant must be capable of handling the elevated TRU content.

The homogeneous option can be regarded as technologically not far removed from MOX recycling, because much of the existing LWR knowledge base would remain applicable. It could even be argued that normal MOX fuels already constitute a type of homogeneous MA-TRU fuel because of the presence of ^{241}Am from ^{241}Pu decay. Therefore, it might be possible to regard homogeneous fuels as an extension to existing technology. This might simplify its qualification, though qualification would still be needed.

Heterogeneous recycling entails loading the MA-TRU in special target fuel rods that are distinct from the main driver fuel. The MA-TRU target fuel rods will not contribute fully to neutron multiplication and will therefore need to be supplied with excess neutrons from the main fuel type. They constitute targets for the excess neutrons and this is why they are called target rods. The advantages are:

- The design of the MA-TRU target rods, their positioning in the core and their residence times can be optimised specifically to maximise TRU utilisation.
- The manufacturing of the target rods would be carried out in a dedicated plant that is separate from the normal fuel fabrication plant. The throughput of the MA-TRU fabrication plant would only need to be a small fraction of the throughput of the mainstream fuel type, thereby simplifying its design and operation. The mainstream fuel supply is independent of MA-TRU fuel supply.

The disadvantages are:

- Because the major component of minor actinide target rods comprises Np, Am or Cm, there is very little knowledge base on which to demonstrate satisfactory fuel behaviour. The knowledge that exists derives from small scale experiments and is insufficient to guarantee satisfactory performance at commercial scale.
- Having separate designs for the driver fuel and the target fuel complicates the nuclear and thermal-hydraulic design.
- Heterogeneous target fuels demand the provision of a second fuel cycle that runs parallel to the mainstream fuel cycle. All the costs of this second fuel cycle and its associated facilities add to the costs of the mainstream fuel cycle.

There is no reason why homogeneous and heterogeneous recycling should not be carried out in parallel, with homogeneous recycling used for Np and heterogeneous for Am and Cm and there may be benefits from the used of shared facilities.

3.2 Reprocessing

In conventional reprocessing facilities, the minor actinides follow the fission products into the high level waste stream, eventually being incorporated in vitrified high-level waste (VHLW). Recycling minor actinides will require modifications to the normal reprocessing flow sheet to separate them from the fission products and route them separately or grouped together in some way. While it is relatively straightforward to keep Np with Pu, it is more difficult to separate trivalent Am-Cm from trivalent lanthanide fission products and even more difficult to separate Am from Cm.

While conventional reprocessing plants produce pure PuO_2 as a recycling output stream, modified flow sheets are now of interest in which the Pu is recovered as a co-product with U, Np or Am. With >60-70% uranium in the co-product, the proliferation materials' attractiveness is reduced compared with pure PuO_2, with higher uranium fractions giving greater decreases in attractiveness.

Irrespective of the flow sheet chosen, the reprocessing plant and associated facilities will need to provide for the following steps which are additional to those found at current reprocessing plants:

- provision for chemical separation of the minor actinides from fission products;
- a route for chemically converting the minor actinides to oxide powder that is subsequently usable in recycled fuel or targets;
- provision for shielded handling and storage of the minor actinide powder or co-product;
- to avoid the need to transport minor actinides or co-products, fuel fabrication should be co-located and integrated with the separations facilities at the same site.

While the mass of minor actinides is small (about 0.1% of the initial heavy metal mass of the fuel), the facilities required will nevertheless be substantial and their cost can be expected to be very significant, both in terms of initial investment and operational costs. Unit costs in terms of Monetary Units per kg of minor actinide are likely to be high. These minor actinide facilities will need to meet the same radiological protection, safety and environmental impact standards as the conventional reprocessing facilities. A major goal of fuel cycle R&D will be to maximise the simplicity, flexibility and cost effectiveness of the recycling route.

3.3 Separation technologies

There has been extensive research and development performed on advanced separation methods over the past decade. The research has been performed across the globe, including: North America (US, Canada), Europe (EU, France, UK), and Asia (the Russian Federation, Japan, the Republic of Korea, China, India). This research can be categorised into a few broad categories, the third one of which is most relevant here:

- modifications to the PUREX process;
- alternative aqueous processes to recover uranium;
- processes to recover the minor actinides (Am, Cm);
- processes to remove fission products for waste management purposes;
- pyrochemical methods of recovering uranium and transuranics.

The PUREX process is being used for commercial spent fuel separations in France, the Russian Federation, Japan and UK and other non-commercial PUREX plants have been built and operated in other countries such as the US. The PUREX process separates uranium and plutonium from the fission products and other actinides. Both the uranium and plutonium streams are very pure suitable for recycling, with only traces of cross-contamination and fission products. However, in recent years sensitivities over the production of pure plutonium have led to the study of alternative separation schemes where a co-product of uranium and plutonium and possibly other radionuclides would be recycled instead. Some of these alternative schemes could also be used to separate the minor actinides.

Two primary approaches to this have been developed. In one approach, only uranium is extracted, and plutonium is left with the other fission products and actinides. This process is called the Uranium Extraction (UREX) process and was developed in the United States within the last decade. The other approach is to co-extract uranium and plutonium (as is done in the PUREX process) but to co-strip a fraction of uranium with the plutonium so that pure plutonium is never separated in the process. This process has been studied as a variation of the UREX process in the US, developed as the COEX™ process in France, and as the NUEX process in the UK.

Separation of the trans-plutonium actinides from the lanthanides and remaining fission products for possible transmutation has been a major component of several national spent nuclear fuel strategies for the middle of the 21st century. This has been the subject of a substantial amount of research over the past decade in nearly every fuel cycle state.

The UREX approach would require the least initial capital investment and technology development to implement. The raffinate from the uranium separation process would be solidified into a form that would provide adequate heat transfer properties. Ideally, this storage form would be readily dissolved in nitric acid, for future treatment to recover the TRUs for future transmutation in reactors. The separation of trivalent TRUs from lanthanides is a challenging technological problem that has been under intense study for the past several years. While much progress has been made in the development of extractants, there does not appear to be a technology currently ready for industrial-scale use. The approach of placing the TRUs and lanthanides together in interim storage would allow for a few additional decades of research and development of new separation technologies before such technology would need to be implemented.

Two processes that have been extensively developed and tested are the Transuranium Extraction (TRUEX) process and the Diamide Extraction (DIAMEX) process for separation of the actinides and lanthanides together. Both processes would produce a relatively pure TRU/lanthanide fraction and the raffinate would contain transition and noble metals. The primary difference between the processes is the composition of the extractants, with the diamide extractants following the C, H, O, N principle of containing only carbon, hydrogen, oxygen and nitrogen atoms to allow ease of incinerating spent solvent and the TRUEX extractant containing phosphorous. Another extractant that exhibits high separation efficiencies for TRUs and lanthanides is N,N,N',N'-tetraoctyldiglycolamide or TODGA. The development of this extractant is relatively new, but is under investigation by a number of research laboratories around the world.

The TRUEX process uses a solvent comprised of octyl (phenyl)-N,N-diisobutylcarbamoylmethylphosphine oxide (CMPO), and tri-n-butyl phosphate in a paraffin hydrocarbon diluent. The TRUEX process is very effective in extracting 3, 4 and 6 valent metals from nitric acid solutions. Complexants, such as oxalic acid, can be added to reduce the extraction of transistion metals, such as zirconium and molybdenum.

The DIAMEX process originally utilised dimethyl-dibutyl-tetradecylmalonamide (DMDBTDMA) as the extractant in a hydrocarbon diluent. Recently, a new extractant, dimethyl-dioctyl-hexaethoxymalonamide (DMDOHEMA) has been developed that appears to have better extraction properties than DMDBTDMA. This new extractant has been tested in a counter-current flowsheet test with actual concentrated high activity PUREX process raffinates.

Following the separation of the TRU/Ln fraction, an additional process after the TRUEX or DIAMEX process would be required to separate trivalent actinides from lanthanides. The state of development of An/Ln partitioning technologies is less than for the other technologies described above.

Trivalent actinide/lanthanide separation is difficult to accomplish due to the similarities in the chemical properties of the trivalent actinides and lanthanides. Various solvent extraction processes have been studied including: the extraction of the lanthanides from the trivalent actinides with the TALSPEAK process, the coextraction of the trivalent actinides and lanthanides, with selective stripping of the actinides from the lanthanides with the reverse TALSPEAK process, the di-isodecylphosphoric acid (DIDPA) process, the SETFICS process, the PALADIN process, selective actinide extraction (SANEX) processes using Cyanex 301, the SANEX-III and SANEX-IV processes (24), as well as processes utilising bis-triazinyl-1,2,4 pyridines (SANEX-BTP). Additional research is being conducted on the GANEX process (group actinide extraction), which is similar to the combined DIAMEX-SANEX approach, but attempting to combine them into a single process.

3.4 Fuel manufacturing

The manufacture of types of nuclear fuel has to be able to meet very stringent requirements on radiological dose to operational staff. Meeting the accepted international standards on personnel dose uptake is relatively simple for uranium fuels, but gets more difficult when progressing to U-Pu, U-Pu-Np, U-Pu-Np-Am and TRU fuels respectively, where TRU denotes the transuranics Np, Pu, Am, Cm and small quantities of Bk, Cf and Es.

With uranium manufacture it is necessary to control internal dose uptake by limiting exposure of operating personnel to uranium in air. There is also a requirement to control surface dose exposure, but the gamma source in uranium fuel is weak and exposure can be controlled without shielding (usually through control of occupancy time and in some fabrication plants by using automated fabrication and inspection techniques). Neutron dose rates and radioactive decay heat output are both negligible and do not require any control measures with uranium fuel. Uranium fuel manufacture remains a largely hands-on process with the operators being able to handle the fuel pellets and fuel rods without physical protection.

At a MOX fuel manufacturing plant, the higher internal dose uptake associated with PuO_2 demands that all stages of the manufacturing process leading to the production of pressure tight fuel rods need to be carried out in a glovebox environment. Moreover, MOX fuel has higher gamma activity that contributes to external dose and neutron production and heat output are no longer negligible. The gamma and neutron sources from MOX fuel are dominated by the contribution of ^{241}Am, which builds up following ^{241}Pu decay. The ^{241}Am present in irradiated fuel is removed when the fuel is reprocessed, but builds up subsequently as the ^{241}Pu decays. The combination of high ^{241}Pu at the time the fuel is reprocessed and long storage time for the recovered plutonium leads to high ^{241}Am content and higher gamma and neutron sources. Co-location of separation and fuel fabrication facilities, with close coupling and coordination of separation and fuel fabrication, would minimise the impact of ^{241}Am build-up. In some instances (e.g. US), long-aged spent fuel, in which most of the ^{241}Pu has already decayed, could be selectively reprocessed and the subsequent build-up of ^{241}Am almost eliminated.

Stepping up from U-Pu fuel to U-Pu-Np fuel causes a modest increase in gamma and neutron activity of the order of a factor 2 depending on the Np content [9]. This is mainly caused by increased production of ^{238}Pu from ^{237}Np neutron captures, although there is also a contribution from the adaptations needed to the plutonium content to maintain reactivity equivalence in the presence of the Np. U-Pu-Np homogeneous fuel assembly manufacture may therefore be possible at fabrication plants which are similar to those used for MOX fuel currently, but with increasing shielding provision for higher Np contents. Again, close coupling of separations and fuel fabrication facilities would be effective in reducing the effect of ^{237}Np gamma emissions. It is the in-growth of ^{233}Pa that is the main contributor to gamma emissions and if fuel fabrication is completed within about 6 months of separation, gamma emissions from ^{233}Pa can be kept below problematic levels.

Adding technologically relevant quantities of Am to nuclear fuel (for example in U-Pu-Np-Am) increases gamma activity, neutron source and decay heat significantly, approaching a factor of 10 or more [9]. This demands a higher degree of shielding provision and perhaps the use of remote fabrication and inspection methods to avoid unacceptable operator doses.

At the top of the hierarchy is TRU fuel, which, in addition to Np, Pu and Am contains small, but very significant quantities of higher actinides, most importantly Cm. The gamma activity of TRU fuel can be three times or more higher than U-Pu-Np-Am fuel; the neutron activity is potentially orders of magnitude higher and decay heat two or more times higher. With these gamma and neutron activities, TRU fuel manufacture would demand shielding measures comparable with those needed to handle irradiated fuel in reprocessing plants. All the fuel fabrication and inspection steps would need to be carried out remotely or automatically. The technological challenges of TRU fuel manufacture should not be underestimated. Fuel designs that are amenable to remote fabrication, such as vibro-compaction, are clearly well suited to TRU fuel if they can be demonstrated to perform satisfactorily.

3.5 Fresh fuel transport and handling

UO_2 and MOX fuel assemblies for LWRs are transported in unshielded containers. With the increased gamma and neutron doses from U-Pu-Np, U-Pu-Np-Am and TRU fuels, there will be a requirement to install adequate shielding in or around the transport packages and also to ensure the handling of fuel assemblies during export from the fabrication plant and upon receipt at the LWR plant is protected with sufficient shielding.

A potential issue with fuel handling is that of transferring heated assemblies into wet storage. Since the heat output from minor actinide fuel is likely to be much higher than UO_2 and MOX fuels, a cooling phase may be required before minor actinide fuel can be transferred. There is also a potential question as to whether dry storage would be feasible.

3.6 Irradiation testing

Prior to commercial scale deployment of homogeneous or heterogeneous MA-TRU fuel there will need to be a programme of materials properties measurements and extensive irradiation testing. Materials properties measurements would initially be made on un-irradiated samples.

Irradiation testing would start with small-scale irradiation tests of short length fuel rods typically done in facilities such as the OECD/NEA Halden Reactor Project (HRP). Irradiated samples would typically be subjected to post-irradiation examination (PIE) to confirm satisfactory irradiation performance and provide irradiated fuel properties measurements.

Following small scale irradiation tests there would be a requirement to carry out demonstration irradiations in a commercial reactor, which might perhaps involve the irradiation of small numbers of fuel rods or a small number of complete assemblies. Typically, fuel rods might be discharged after 1, 2 and 3 irradiation cycles to confirm the irradiation behaviour is as expected. The objective would be to demonstrate satisfactory irradiation endurance prior to loading minor actinide assemblies at commercial scale.

The entire process from initial design to start of commercial scale loading might be expected to take at least 10 years and therefore irradiation testing is a rate limiting step. Commercial scale loading also pre-supposes the availability of a suitable fuel fabrication plant, of which none exist at present. The overall timescale required is therefore rather longer, at least 15 to 20 years, so that MA-TRU fuels need to be seen as a medium-term option that will definitely not be available in the immediate future.

The degree of rigour required of the irradiation testing and demonstration programme depends on the type of fuel selected. It could be argued that for U-Pu-Np fuel much of the existing fuel performance database for MOX fuel will remain valid and therefore a less demanding test programme would be required, facilitating early introduction. However, for TRU fuel, particularly a novel fuel type such as TRU fuel in non-uranic matrix [i.e. inert matrix fuel (IMF)], a more rigorous and presumably longer testing programme would be required.

With any new fuel design there is a degree of technical risk and the possibility of poor in-core behaviour becoming manifest only after the start of commercial-scale loading. Utilities would be understandably reluctant to bear this risk alone and some degree of governmental underpinning may be needed to encourage utilities to adopt minor actinide burning. In many respects, the nature of the benefits of minor actinide burning would be societal and not specific to the utility itself. In this context, governmental underwriting of technical risks would be at the very least desirable and might even be essential.

3.7 Licensing

One of the advantages of burning minor actinides in LWRs is that it would be possible to benefit from the comprehensive licensing regimes that are already in place. Even with a very radical TRU fuel, it would be a sensible objective to try to retain intact as much of the existing licensing basis as possible. With a small loading of minor actinides in the core it might be possible to keep within the existing nuclear design and thermal-hydraulic limits so that the overall safety behaviour of the fuel and core fits within current limitations. In this case, the only alterations to the safety case required would be those which specifically relate to the fuel performance of the fuel assemblies that contain the minor actinides. This might have a carry-over to the rest of the safety case, for example, if fuel failure source terms are affected, but the overall expectation is one where the safety case is largely unaffected.

For higher minor actinide loadings it is probable that the existing nuclear design and thermal-hydraulic limits will be exceeded, in which case there will be an impact on the broader safety case. There may be a need for core modifications such as control rod and burnable poison design changes or other core modifications to adjust the reactivity feedback coefficients.

3.8 Irradiated fuel

With homogeneous MA-TRU fuel, there will be relatively minor implications for irradiated fuel arisings. The presence of minor actinides may reduce the mean discharge burn-ups if it is not possible to compensate with higher initial fissile loading, which will give a corresponding increase in irradiated fuel discharges. The inventory of the irradiated fuel will be altered by the minor actinides, with higher specific heat loadings and neutron outputs the likely outcome.

With the inhomogeneous strategy there would be several major effects on irradiated fuel arisings: the first is that there would be two distinct types of fuel discharged, the conventional driver fuel assemblies and the minor actinide target assemblies, possibly necessitating some duplication of back-end fuel cycle facilities. The driver fuel would be entirely conventional, though there would possibly be differences arising from the higher initial fissile loading required and also possibly a higher mass discharged if the mean discharge burn-up cannot be kept constant. The minor actinide target fuel would be completely different in nature to the driver fuel, possibly with a higher decay heat output and higher neutron activities. The target fuel might require a different regime for in-pond cooling and different arrangements for transport. Whether the target fuel is intended for geological disposal or reprocessing and recycling, there will be completely different requirements to those of the driver fuel.

3.9 Waste

With homogeneous minor actinide recycling the waste arisings will hardly be affected compared with a conventional fuel cycle. If the irradiated fuel is reprocessed, there will be a higher inventory of TRUs compared with more conventional fuels. If the minor actinides are recycled further, then there would be little or no impact on the incorporation rate of fission products in the VHLW glass. However, if the TRUs are not recycled further, there is the potential for the higher neutron source to deleteriously affect the vitrification incorporation rate. The effect of this might be to increase the number of VHLW glass containers per GWye. To determine whether this actually turns out to be the case would require a detailed mass-flow scenario analysis of the entire fuel cycle, accounting for the lower mass flow of TRUs in the first generation of fuel reprocessed (i.e. the minor actinides from this fuel are recycled). Although recycling of Np and Am destroys a high fraction, the Cm continues to build up and the neutron source from ^{244}Cm is potentially a limiting factor.

With heterogeneous recycling, the situation regarding VHLW waste volumes is changed significantly, in that there are now two fuel types that may potentially be reprocessed: the driver fuel and the minor actinide target fuel. If only the driver fuel is reprocessed, then the VHLW waste volume is likely to be reduced, because the transuranic mass in the VHLW will be lower if some is recycled in target assemblies. However, if the target assemblies are also reprocessed, these will have a very high neutron activities and decay heat outputs and unless some of the transuranics are recycled further, there is the possibility that the vitrification incorporation rate may be adversely affected. Again, detailed mass-flow scenario analysis is needed to provide definitive answers.

3.10 Impact on geological disposal

Reference [4] shows that americium burning in both LWRs and HWRs leads to significant reductions in the decay heat load in the geological disposal facility. This is beneficial because the decay heat load generally limits the capacity of a geological disposal site.

3.11 Overall timescales

Bearing in mind, as discussed in Section 3.5, the need for extensive radiation testing, the earliest possible date for commercial scale minor actinide recycling in LWRs would be about 2020, with perhaps 2025 or 2030 being more realistic. If minor actinide recycling becomes established, the subsequent timescales for worthwhile recycling programmes will be measured in decades and it is quite probable that minor actinide recycling instigated in new build LWRs between 2020 and 2030 would still be active when these plants reach the ends of their lifetimes in 2080 or 2090.

While the overall timescale will depend on the specific scenario and in particular whether there is multi-recycling of the minor actinides, the fact is that any minor actinide recycling programme is likely to require a very long-term commitment of many decades. This does not fit well with the time horizons of commercial utilities and is another reason why governmental involvement will be necessary to provide the necessary incentives.

3.12 Non-technical considerations

As already hinted at in this section, non-technical issues that will need to be addressed include economics and societal: as will be discussed in Chapter 7, there will be significant costs associated with minor actinide recycling and the benefits are mostly in favour of broader society and not for the utilities. Therefore the question of economics is not just an issue for utilities and fuel cycle companies, but also for society and governments. Mechanisms will need to be found by which the wider societal benefits will be recognised and utilities compensated for the additional costs they will incur.

It is possible that in some countries societal concerns may preclude the continued use and further expansion of nuclear energy unless there is a definite resolution of spent fuel and waste disposal strategies. In some cases, such strategies may demand some form of minor actinide recycling.

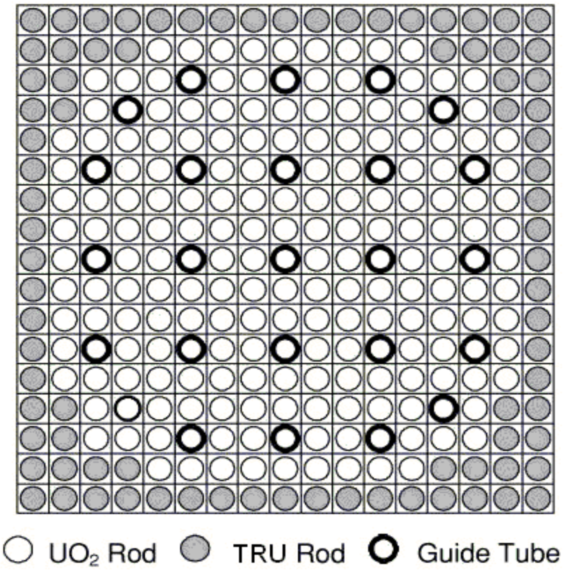
4. Fuel cycle issues

4.1 Source terms

This section discusses the source terms from MA-TRU fuels for heat output, gamma emissions, neutron emissions and radiological doses in the context of a specific example. A publication from the bibliography [9] provides source term data that is useful here for illustration. It was chosen as an example because it represents a very detailed and comprehensive study and includes all the elements needed here to illustrate the practicalities on transuranics recycling in LWRs. The Taiwo paper describes an analysis of multi-recycling scenarios in a PWR using the CORAIL assembly concept that originated in France [15-17]. This was one of a number of assembly designs that were developed in France initially in the context of plutonium recycling, but which were also investigated in the context of minor actinide recycling [18-21].

CORAIL (Figure 4.1) is a PWR fuel assembly that is dimensionally compatible with conventional assembly designs, but differs in that it has a heterogeneous distribution of fuel rods. The central part contains conventional low-enriched UO_2 fuel rods which act as the driver. Around the periphery is a zone with fuel rods containing Pu. Pu-Np, Pu-Np-Am or TRU. The CORAIL fuel assembly can be categorised as a homogeneous recycling option, with the transuranics concentrated in the peripheral fuel rods. The CORAIL assembly has 84 MOX fuel rods out of the 264 total, so that the TRU rods represent one third of the fuel mass.

Figure 4.1: CORAIL PWR fuel assembly



The main objective of recycling the transuranics in the scenario studies in is not so much to transmute them as to store them in active circulation in the core and thereby delay them going to the geological repository (this is referred to as the "delay line" concept). Recognising that the transuranic inventory in active circulation increases in each successive recycling, a possible end solution could be transmutation in a fast reactor system. This is the same end-point as envisaged in France, but the delay line strategy allows the fast reactors to be introduced at a much later stage.

Table 4.1 shows equilibrium isotopic composition data extracted from the Taiwo paper [9] for four scenarios:

- equilibrium recycling of Pu only;
- equilibrium recycling of Pu+Np;
- equilibrium recycling of Pu+Np+Am;
- 3rd recycling of all transuranics.

Table 4.1: Equilibrium isotopic compositions for multi-recycling scenarios from [9]

Target material	Pu Equil	Pu-Np Equil	Pu-Np-Am Equil	TRU 3rd recycling
Cooling time (y)	5	5	5	5
Uranium enrichment (weight %)	4.62	4.9	5.5	4.98
TRU content (%)	8.45	9.93	16.17	8.39
Fissile content (%)	44.67	43.22	36.23	45.23
^{237}Np (%)	1.66E-3	2.84	2.23	3.00
^{239}Np(%)	0	0	0	0
^{238}Pu (%)	3.55	6.13	10.95	5.34
^{239}Pu (%)	34.2	33.25	28.27	34.17
^{240}Pu (%)	23.33	22.38	20.55	24.37
^{241}Pu (%)	10.48	9.98	7.89	9.84
^{242}Pu (%)	27.4	24.44	18.99	11.50
^{241}Am (%)	1.04	0.99	6.01	5.63
^{242m}Am (%)	-	-	0.06	0.03
^{243}Am (%)	-	-	5.05	3.28
^{242}Cm (%)	-	-	-	0
^{243}Cm (%)	-	-	-	0.03
^{244}Cm (%)	-	-	-	2.22
^{245}Cm (%)	-	-	-	0.43
^{246}Cm (%)	-	-	-	0.07

Equilibrium means that the materials are recycled many times until the inventories no longer change between successive recycling. In the case of full transuranic recycling, equilibrium is only reached after a very large number of recycling that has no practical relevance and therefore data only for the 3rd recycling step is shown here, which is point judged to be the practical limit on account of the high neutron source.

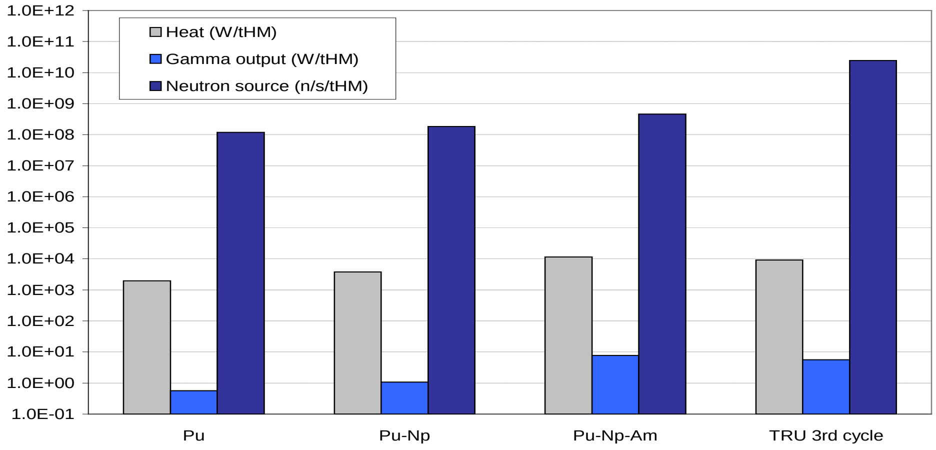
In each case the cooling time between discharge and the start of the next irradiation cycle is 5 years. Although the compositions are only valid in the specific scenario considered in the Taiwo paper, they are nevertheless useful here as representative compositions which yield illustrative source terms. The Taiwo paper assumes an equilibrium discharge burn-up of 45 GWd/t in a 3-batch fuel cycle. The ^{235}U enrichment is adjusted as necessary to maintain reactivity over the required discharge burn-up.

Table 4.2 and Figure 4.2 show the heat, gamma and neutron source terms for the four isotopic compositions at the time of fuel fabrication. Decay heat ranges over a factor of ~5 and gamma output ranges over a factor ~14. However, the neutron source increases by more than two orders of magnitude for full TRU cycle (note the logarithmic vertical axis in Figure 4.1). While the step from conventional Pu fuel to Pu-Np is only a factor of ~2 for heat and gamma and smaller for neutrons, there is a much larger increase in the neutron source for Pu-Np-Am and full TRU recycling, because of the strong neutron source from ^{244}Cm. Continuing full TRU recycling beyond the 3rd recycling step causes the neutron source to increase even further and one of the reasons cited for stopping at 3rd recycling is that at this point the neutron source is considered as strong as might be practically manageable.

Table 4.2: Decay heat, gamma and neutron sources during fuel fabrication for different TRU isotopic compositions

Target material	Pu Equil	Pu-Np Equil	Pu-Np-Am Equil	TRU 3rd recycling
Decay heat (W/tHM)	1960	3766	11600	9227
Gamma output (W/tHM)	0.57	1.08	7.79	5.58
Neutron output (n/s/tHM)	1.20E+08	1.84E+08	4.60E+08	2.43E10

Figure 4.2: Decay heat, gamma and neutron sources during fuel fabrication for equilibrium isotopic compositions



4.2 Fuel fabrication

The Taiwo paper evaluates surface and 1 m doses for the CORAIL-Pu, CORAIL- Pu+Np and CORAIL-TRU fuel assemblies and these are listed in Table 4.3. In each instance, gamma activity dominates the dose rate, with the contribution from neutrons only significant for CORAIL-TRU (the neutron doses from 2nd recycling TRU are orders of magnitude higher).

Table 4.3: Dose rates (mSv/h)

Dose rates (mSv/h)	CORAIL-Pu at equilibrium	CORAIL-Pu+Np at equilibrium	CORAIL-TRU 2nd recycling
Total dose at pellet surface	49.0	88.0	151.1
Total dose at fuel clad surface	1.3	1.9	44.6
Total dose at 1 m	4.2E-3	5.3E-3	0.2

Compared with CORAIL-Pu, there is only a modest increase in dose rates for CORAIL-Pu+Np, suggesting that with suitable design and operation modifications, existing MOX fuel fabrication plants may be adapted for CORAIL-Pu+Np. However, there is a large increase in dose rates for CORAIL TRU, especially at the fuel clad surface and the 1 m dose, even in the 2nd recycling step.

With a low throughput of Pu-Np, the increase in occupational exposure with CORAIL-Pu+Np might be manageable because the increased exposure would apply only for a small proportion of the time. However, with a high throughput of Pu-Np, it might be necessary to install additional gamma shielding and/or increase the level of automation in fabrication and inspection operations. For CORAIL Pu-Np-Am fuel the gamma source still dominates dose rates, but is now substantially higher and this would demand a different type of fuel fabrication plant, with significantly more gamma shielding and/or a higher degree of automation of the fabrication and inspection operations.

Conventional MOX fuel fabrication operations can generate significant quantities of recycled material from fuel pellet grinding operations and from defective pellets or defective fuel rods. Recycling operations will be more complicated for the more challenging fuel types. Co-location of the fuel fabrication plant with the reprocessing plant would simplify the recovery and recycling of such material, especially for Pu-Np-Am and TRU fuels and might avoid the need to construct a separate recycling facility alongside the fabrication plant.

4.3 Transport

To date, minor actinide fuel rods have only been manufactured on very small scales suitable for laboratory and irradiation testing and there has been no requirement to transport minor actinide fuels in commercially relevant quantities. Strategies will need to be developed for transporting fresh minor actinide fuel assemblies to power plants and for transporting the spent fuel assemblies. The unshielded fresh fuel transport containers used for UO_2 and MOX will not be suitable for minor actinide fuels and shielded transport containers will be required. One approach may be to develop specifically designed shielded transport packages. Another possibility might be to adapt existing spent fuel transport packages for this purpose. Both cases will have implications for at-reactor fuel receipt and storage operations, with implications for handling requirements (e.g. crane/elevator load requirements), radiological dose uptake and accident scenarios (e.g. dropped fuel accidents).

It is likely that existing spent fuel transport packages may be useable as-is or adaptable for transporting irradiated minor actinide fuels. However, the higher heat output and neutron output may require special strategies to be developed for loading the minor actinide fuels, perhaps as part of shipment of normal UO_2 fuel assemblies, which could be used to partially shield the elevated neutron emissions and dilute the heat load.

For both fresh and spent minor actinide fuel transport there is no immediate prospect of a commercial requirement developing and until this happens there will be no incentive for fuel transport companies to make the necessary investments.

4.4 Utilisation rates

Utilisation rates for the minor actinides will depend on the intended mission of minor actinide burning. A minimum goal might be to ensure that the future equilibrium inventory of minor actinides should be stabilised, implying that the utilisation rate should match the rate and which the minor actinides accumulate in irradiated fuel. A more ambitious goal might be to recycle minor actinides from historic irradiated fuel stocks, which would require a higher utilisation rate. The minor actinide core loadings required for both scenarios would depend also on whether the MA-TRU fuels are irradiated just once or multiple recycled to equilibrium. In the latter case, the minor actinide mass flows from equilibrium and historic irradiated fuel will be supplemented by the mass flows from recycled minor actinide targets.

France has already decided that LWRs are not the most suited to the more ambitious goal of reducing the historic minor actinide inventory and prefer to rely on fast reactors for this task. Indeed, in France, only part of the spent fuel has been reprocessed (around 10%). The plutonium from the reprocessed fuel has been used to fabricate mixed oxide fuel (MOX) for use in LWRs. This action did enable a reduction in the rate of build-up of plutonium and also the number of spent fuel assemblies in repositories (almost 1/5 UOX spent fuel sub-assemblies). Some MOX spent fuel sub-assemblies have been reprocessed but only for demonstrating the ability of current reprocessing technology. The minor actinide inventory of the reprocessed fuel has been incorporated in VHLW glass and is practically irrecoverable for further transmutation. In the US the situation is different with the historic irradiated fuel still in interim storage and therefore the historic minor actinide inventory remains accessible. The US position was different in that the main goal was the extension of the effective lifetimes of the geological repository and a range of different utilisation rates could potentially have satisfied this requirement. As noted earlier, however, the US position is now evolving and the goal may be modified.

It is helpful to quantify the minor actinide inventories as a fraction of the total plutonium content:For PWR UOX fuel irradiated to 45 GWd/t the ratios are:

- Np/Pu ~3%;
- Am/Pu ~7%;
- Cm/Pu ~0.4%.

With the CORAIL assemblies, the maximum possible TRU fuel fabrication requirement is about 6-7 tHM/GWye at a burn-up of 45 GWd/t (ie 30% of the total fuel requirement of ~22 tHM/GWye). Reference [9], which assumes all the TRU is continuously recycled, indicates a total TRU content of ~7.5% for 2nd recycling CORAIL, of which 1.76% is ^{237}Np, so the maximum mass of Np recycled per year is ~9 kg/GWye. The Am content is 7.7% of the total, equating to ~38 kg/GWye.

This study does not take into account the impact of such recycling on the safety of the plant. In practice, after 2 or 3 recycling processes, the plutonium content needs to be increased to recover the necessary criticality margins for operating the plant and hits the 12% equivalent plutonium content limit that is acceptable for safety reasons. The impact of minor actinides on the safety criteria is even larger than the plutonium with roughly 1% of minor actinides being equivalent to 3% of plutonium.

4.5 Irradiated fuel inventories

Assuming that the MA-TRU fuel has the same mean discharge burn-up as conventional fuel, then the total inventory of irradiated fuel will be unaffected. However, the activity of the minor actinide component will be much higher and this implies modifications to the subsequent storage, transport and reprocessing operations.

4.6 Radiotoxicity and environmental impact

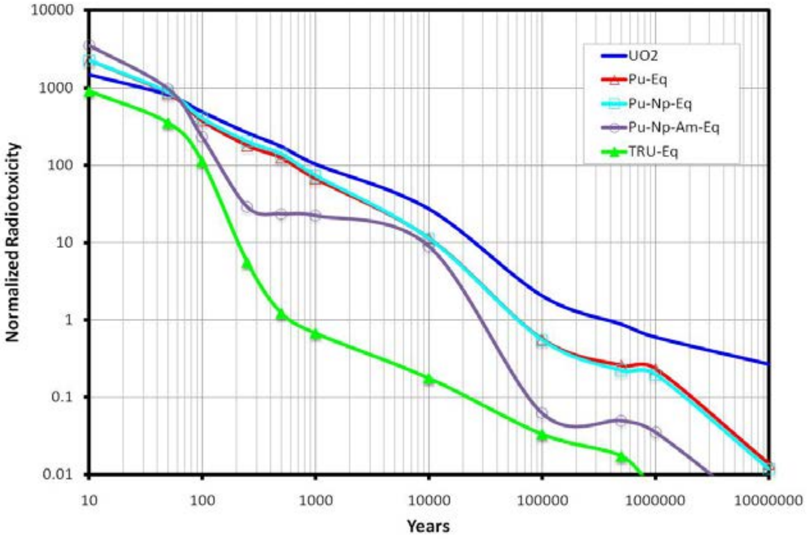
Figure 4.3, which is reproduced from [9], shows the evolution of radiotoxicity with decay time for seven different fuel types, ranging from conventional UO_2 fuel, to six different variants of plutonium and transuranic recycling in the CORAIL fuel assembly (two of which do not recycle plutonium have not been considered here).

Compared with the once-through UO_2 fuel cycle, all the CORAIL recycling options are effective in reducing the radiotoxicity of spent nuclear fuel at cooling times of >100 years. Although all the CORAIL recycling options result in an increase in radiotoxicity at shorter cooling times, this is not a concern because the spent fuel emplacement would not take place then.

The CORAIL-Pu and CORAIL-Pu+Np options have virtually identical equilibrium radiotoxicity profiles, a factor 2-3 lower for cooling times between 100 and 1000 years and somewhat higher for very long cooling times. A much larger benefit is obtained with the equilibrium CORAIL-Pu+Am fuel cycle, especially for cooling times of a few hundred to 1000 years.

The largest benefit is obtained in the equilibrium CORAIL-TRU fuel cycle, which reduces the radiotoxicity of spent fuel by about two orders of magnitude compared with the once-through LWR fuel cycle at cooling times >500 years. However, the neutron and gamma emissions are so strong that such a fuel cycle would probably be impractical to implement. Limiting CORAIL-TRU to 2nd or 3rd recycling would be more realistic, but will yield a smaller benefit in radiotoxicity.

Figure 4.3: Radiotoxicity versus decay time for CORAIL fuel variants (based on [9])



5. Fuel and core design

5.1 Fuel design

The impact of minor actinide recycling on fuel design depends on whether homogeneous or heterogeneous recycling is adopted:

With homogeneous recycling it is feasible for the fuel assembly design to remain essentially unchanged from that used for the conventional UO_2 or MOX fuel. Any modifications that are required to the fuel rod design are likely to be relatively minor ones to accommodate higher internal gas pressures from helium and fission gas release, such as decreased stack height to accommodate an increase in plenum volume. Maintaining the standard fuel assembly mechanical and thermal-hydraulic designs ensures full compatibility with existing fuel. Because the presence of minor actinides in the fuel tends to decrease reactivity, it may be necessary to increase the initial fissile loading of ^{235}U or plutonium. If LEU is the main fissile driver material, there is the issue of the 5.0 w/o fabrication limit to consider. Currently, commercial LWR fuel fabrication plants are only permitted to manufacture LEU fuel with a ^{235}U enrichment below 5.0 w/o, for criticality safety reasons. Since many LWRs are already approaching this limit, the requirement to maintain reactivity may cause it to become the limiting factor even earlier. However, fuel fabricators are beginning to consider how they may be able to extend the criticality safety limit and it is possible that the limit might be relaxed by the time commercial LWR utilities are ready to recycle minor actinides.

With heterogeneous recycling, the minor actinide bearing fuel rods will necessarily be different to those of the driver fuel. For example, in the CORAIL assembly the TRUs are loaded in the peripheral fuel rods and these are distinct to the conventional LEU rods of the driver fuel. The LEU and TRU rods are identical in terms of external dimensions, but the TRU rods may have different internal designs to accommodate additional helium and fission gas release. Other approaches may go further and use a different mechanical design for the TRU rods, introducing the complication of demonstrating mechanical and thermal-hydraulic compatibility. It is even conceivable that the entire fuel assembly design may be non-standard, but this would further exacerbate the compatibility issues.

5.2 Core design

A recent study has determined the impact of loading minor actinide target rods on the core design of a 17x17 PWR core [10]. The objective was to determine the impact of loading minor actinide target rods containing americium and curium in roughly the proportion they arise from irradiated PWR fuel after a cooling time of 6 years. To this end, the exact design of the target rods was not of interest and the entire focus was on the impact they would have on the nuclear design parameters. The starting point was to generate an equilibrium UO_2 core in order to define a reference case for comparison:

5.2.1 Equilibrium UO_2 core

A 4-loop PWR with 193 17x17 fuel assemblies was used for the reference core. The targeted energy output was an industry standard of 18 months. The reference equilibrium core design, calculated using a 3-D nodal core design code, has a cycle burn-up of 18.3 GWd/t, which corresponds to 477 effective fuel power days (efpd) at the nominal full power rating of 38.4 MW/tHM. The load factor corresponds to 87%. The 84 feed assemblies have initial enrichments of either 4.38 or 3.98w/o, with an overall mean of about 4.2 w/o.

Gadolinia (Gd_2O_3) admixed with the UO_2 was used as a burnable absorber to (i) reduce the excess reactivity early on in the fuel cycle and (ii) to keep the power distribution within the design limits. As is standard practice, the Gd_2O_3-doped rods were enriched to a lower concentration (2.0 w/o ^{235}U) than the bulk of the assemblies to counteract the lower thermal conductivity of Gd_2O_3-doped fuel. The remainder of the fuel was enriched to a slightly higher than nominal value to give the required mean ^{235}U enrichment.

5.2.2 Equilibrium Am-Cm core

For the purposes of this study, it was decided to adopt the simple approach of loading the Am-Cm target fuel rods in the control rod guide tubes of the standard PWR fuel assembly. It is acknowledged that such an approach may cause practical engineering concerns, such as maintaining adequate coolant flow in the restricted flow channel between the target rods and the control rod guide tubes. However, for the purpose of determining the impact on the nuclear design, such concerns were set to one side in the expectation that a more considered engineering approach would give similar results for the core behaviour.

Recognising the need to restrict the total Am-Cm loading, only one third of fuel assemblies were assumed to contain Am-Cm rods and each of those assemblies that was assumed to contain 24 rods in the control rod guide tubes. Obviously, the Am-Cm bearing fuel assemblies must avoid the control rod locations and this helps to minimise the impact on control rod reactivity worths.

The target rods were assumed to consist of $AmCmO_2/CmO_2$ in an inert matrix and clad in Zircaloy-4. The inert matrix was of no interest to this study and assumed to be neutronically transparent. The remainder of the fuel rods in the Am-Cm assemblies was assumed to contain 4.2 w/o $^{235}UO_2$ (with no Gd_2O_3 dopant). The Am-Cm concentration was adjusted so that the variation of reactivity with burn-up was similar to that of a fuel assembly containing discrete B-10 poison rods. The objective here was to take advantage of the Am-Cm to perform as a burnable poison and help control overall reactivity.

To match approximately the reactivity of a 4.2 w/o UO_2 fuel assembly doped B-10 burnable absorbers it was found that a loading of 4.2 kg of Am-Cm was required, distributed equally among the 24 guide tubes of the assembly. In every 84 assembly reload, 24 of the assemblies were assumed to contain 24 Am-Cm rods. The total mass of Am-Cm loaded in every 18 month cycle is therefore close to 100 kgHM.

Table 5.1 shows the isotopic composition of the Am-Cm feed to the minor actinide target rods. This corresponds to the isotopic composition obtained from PWR fuel discharged with a burn-up of ~50 GWd/t and cooled for 6 years.

Table 5.1: Isotopic composition of Am-Cm feed mix

Nuclide	w/o
Am-241	63.85
Am-242	<0.01
Am-243	25.35
Cm-242	<0.01
Cm-243	0.09
Cm-244	9.78
Cm-245	0.82
Cm-246	0.11

The Am-Cm rods were assumed to reside in the core for three 18 month cycles, to maximise their overall burn-up (which is higher than that of the UO_2 fuel, roughly equivalent to 60 GWd/t, though in the inert matrix the actual burn-up expressed per tonne of heavy metal is much higher). The cycle length of the equilibrium Am-Cm-doped core was 17.1 GWd/t, which is a reduction of 1.2 GWd/t on the reference UO_2 core, even though the ^{235}U loadings are very nearly equivalent. This represents loss of approximately 31 efpd, due to the negative neutron balance of Am-Cm, which has important economic implications for an utility. This could be avoided by increasing the initial ^{235}U enrichment, but this still means that the utility incurs an unavoidable cost penalty.

The balance between minor Am-Cm destruction and production in the core is quite favourable. In the Am-Cm rods the equilibrium core destroys approximately 70 kg of Am-Cm in every 18 month cycle, from the 100 kg loaded initially. In the remainder of the core, which contains UO_2 fuel rods, approximately 35 kg of Am-Cm are produced. The net Am-Cm destruction is therefore ~-35 kg Am-Cm per 18 month fuel cycle. A single reactor loaded with Am-Cm in this way is able to destroy all the Am-Cm it generates itself and that of a matching reactor loaded entirely with UO_2 fuel. Although Am is destroyed very effectively, in common with other reactor systems, the Cm tends to accumulate through neutron captures.

5.2.3 Nuclear design parameters

Table 5.2 indicates how the nuclear design limits are modified in the equilibrium core with Am-Cm target rods compared with the reference UO_2 equilibrium core.

Table 5.2: Impact of Am-Cm target loading on equilibrium nuclear design parameters

Parameter	Am-Cm impact	Significance
Radial power peaking factor $F_{\Delta H}$	Marginal violation of nuclear design limit	Probable mitigation with core loading pattern optimisation and/or relaxation of limits
Axial power peaking factor F_Z	Small	None
Overall power peaking factor F_Q	Small increase	Probable mitigation with core loading pattern optimisation and/or relaxation of limits
Moderator temperature coefficients	Marginal violation of nuclear design limit	Probable mitigation with core loading pattern optimisation and/or relaxation of limits
Doppler coefficient	Small increase	None
Boron coefficients	Slight reduction in magnitude (less negative), marginal violation of nuclear design limits	Probable mitigation with core loading pattern optimisation and/or relaxation of limits
Control rod reactivity worths	Slight increase in magnitude (more negative) within nuclear design limits due to power density re-distribution effects	None
Shutdown margins	Small increase in end-of-cycle shut-down margin	None
Delayed neutron fraction	Small	None
Boration limits	Moderate impact on boron concentrations within normal nuclear design limits envelope	None

The main observations are:

- The radial power peaking factor $F_{\Delta H}$ is increased (see Table 5.3), slightly above the nuclear design limit of 1.528, but the study just used the equilibrium UO_2 assembly loading pattern without re-optimisation. It is judged that the slight violation could be avoided with loading pattern optimisation.
- The overall power peaking factor F_Q is increased slightly, in line with the higher $F_{\Delta H}$, though within the nuclear design limits.
- The moderator temperature coefficients are more negative with Am-Cm loading. The precise effect varies depending on the core condition, but the incremental changes are all modest, with the magnitude of the moderator coefficient changing by no more than 8 pcm/°C. This may be partly caused by the reduction in overall moderator/fuel ratio as a consequence of loading the Am-Cm target rods in the control rod tube locations. Tables 5.4 and 5.5 indicate the Hot Full Power (HFP) and Hot Zero Power (HZP) moderator temperature coefficients for the two equilibrium

cores. The calculations were carried out at Near Beginning of Cycle (NBOC), Middle of Cycle (MOC) and End of Cycle (EOC) core conditions.

- The fuel temperature (Doppler) coefficients are only slightly more positive in the Am-Cm core. Hence the Doppler reactivity defect from full to zero power (i.e. the positive reactivity insertion associated with cool-down from HFP to HZP is slightly increased, as indicated in Table 5.6.
- The boron coefficients are slightly less negative in the Am-Cm core, consistent with higher thermal neutron absorption, as indicated in Tables 5.7 and 5.8 for HFP and HZP conditions. Some of the calculated values are close to or marginally outside the nuclear design limits. Again, the violations are so marginal that loading pattern optimisation would expect to be effective in mitigation.
- The control rod reactivity worths are slightly modified with the particular distribution of Am-Cm loaded fuel assemblies. Table 5.9 compares control rod reactivity worths at HZP conditions for the two equilibrium cores. The reactivity worths quoted correspond to all control rods inserted except for the highest reactivity worth control rod, which is assumed stuck out of core.
- Table 5.10 shows the impact of Am-Cm loading on shutdown margins, which include allowances for uncertainties. The NBOC shutdown margin is more than 800 pcm smaller in the equilibrium Am-Cm core. However, the EOC shutdown margin is actually slightly higher in the Am-Core than it is at NBOC. This contrasts with the equilibrium UO_2 core, which shows the behaviour whereby the shutdown margin becomes more limiting at EOC conditions. The much smaller impact of Am-Cm at EOC is presumably because of the much lower inventory of Am remaining at EOC.
- Table 5.11 shows that the delayed neutron fractions are hardly changed in the Am-Cm equilibrium core.
- Table 5.12 shows the beginning of cycle (BOC) boration limits at HFP, HZP and also Cold Zero Power (CZP) core conditions. The relevant boron concentrations tend to be higher in the equilibrium Am-Cm core, but all the nuclear design limits are satisfied. RIL indicates that the control rods are inserted to their insertion limits. ARO denotes all control rods withdrawn and ARI denotes all control rods inserted.

Table 5.3: HFP radial peaking factor $F_{\Delta H}$

Cycle burn-up (GWd/t)	Eq UO_2 core	Eq Am-Cm core
0	1.463	1.541
0.15	1.465	1.537
2	1.383	1.495
4	1.369	1.502
6	1.384	1.517
8	1.455	1.537
10	1.477	1.511
12	1.449	1.484
14	1.422	1.468
16	1.400	1.454
17.1	-	1.445
18.3	1.369	

Table 5.4: HFP moderator temperature coefficient (pcm/°C)

GWd/t	EQUO2	AMCM
NBOC	-33.7	-38.1
MOC	-42.3	-48.6
EOC	-70.8	-74.4

Table 5.5: HZP moderator temperature coefficient (pcm/°C)

GWd/t	EQUO2	AMCM
NBOC	-10.8	-11.0
MOC	-11.7	-20.1

Table 5.6: Doppler defects from HFP to HZP in pcm

GWd/t	EQUO2	AMCM
0.15	985	1025
EOC	855	886

Table 5.7: HFP boron reactivity coefficients and nuclear design limits (pcm/ppm)

GWd/t	EQUO2	Limits	AMCM	Limits
NBOC	-6.9	-11.5,-6.3	-6.4	-11.5,-6.2
MOC	-7.3	-11.8,-6.6	-6.6	-11.8,-6.7
EOC	-8.3	-12.5,-7.5	-7.4	-12.5,-7.5

Table 5.8: HZP boron reactivity coefficients and nuclear design limits (pcm/ppm)

GWd/t	EQUO2	Limits	AMCM	Limits
NBOC	-7.1	-10.9,-5.6	-6.6	-10.8,-5.5
MOC	-7.4	-11.2,-5.9	-6.8	-11.2,-5.9
EOC	-8.4	-11.8,-6.7	-7.4	-11.8,-6.6

Table 5.9: HZP control rod worths in pcm (minus the highest worth rod)

GWd/t	EQUO2	AMCM
0.15	6116	5389
EOC	6247	6545

Table 5.10: Best-estimate shut-down margins (pcm)

GWd/t	EQUO2	AMCM
0.15	4520	3625
EOC	3483	3784

Table 5.11: HZP delayed neutron fraction

GWd/t	EQUO2	AMCM
BOC	0.006237	0.006220
EOC	0.005286	0.005263

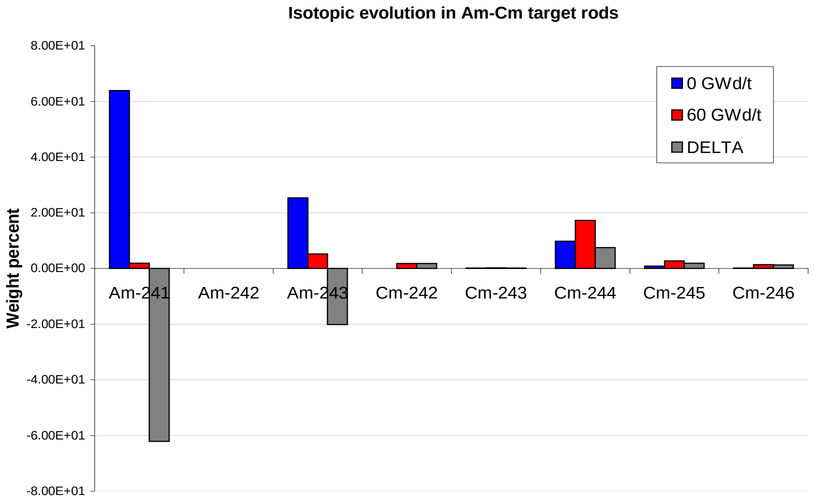
Table 5.12: Boration limits

Core Condition	Design limit	EQUO2	AMCM
HFP BOC, no Xe, rods at RIL	2 000	1 361	1 469
HZP BOC, ARO, No Xe, k=0.99	2 250	1 499	1 654
CZP, BOC, No Xe, ARO, k_{eff} = 0.99	2 500	1 764	2 014
CZP, BOC, No Xe, ARI, k_{eff} = 1.0	1 600	1 075	1 396
CZP, BOC, No Xe, ARI, k_{eff} = 0.95	2 500	1 608	1 999

5.3 Material balance

Figure 5.1 shows the material balance for americium and curium for the target rods. The blue bars show the initial weight percentage of the various isotopes in the Am-Cm target rods, while the red bars show the weight percentages after the fuel assembly in which the Am-Cm rods are inserted has been irradiated to its 60 GWd/t mean discharge burn-up. The grey bars show the difference between end-of irradiation and beginning of irradiation. The americium isotopes all show net destruction by fission or captures (this also applies to ^{242}Am, which is at too low a concentration to show on this scale. In contrast, all of the curium isotopes show net in-growth. Overall destruction of americium and curium amounts to approximately 70 w/o of the initial Am-Cm mass.

Figure 5.1: Material balance for americium and curium target rods



5.4 Fuel matrices

It is usually considered beneficial to use an alternative diluent matrix to UO_2 for minor actinide target fuel. It is important to disperse any transuranic materials in a matrix, because the power density would otherwise be excessive. For example, because of its high fission cross-section, plutonium needs to be diluted by a factor of 10 or so, to obtain a manageable fission density and normally this is achieved by dispersing it in a depleted UO_2 matrix. Even though neptunium and americium do not fission as readily, the heat production from fissions and neutron captures combined can be excessive unless there is a matrix.

Inert matrix fuel (IMF) is beneficial to use as the matrix because the materials used are intended to avoid neutron interactions and in particular fertile captures from ^{238}U to ^{239}Pu are avoided. Since the objective is to destroy minor actinides, generating ^{239}Pu from fertile captures is unhelpful as the freshly produced ^{239}Pu provides the basis for generating new minor actinides. IMF avoids this counterproductive step. Candidate materials include cerium oxide (or ceria CeO), zirconium oxide (or zirconia ZrO) or yttrium stabilised zirconia (YSZ). The IMF matrix needs to be resistant to high temperatures, corrosion and irradiation, compatible with the fissile material and fission products and preferably have a low neutron capture cross-section. There are many materials that are being researched and many are promising, but any selected for commercial application will need to be proven with extensive irradiation testing.

Proving the suitability of IMF for commercial reactor application is one of its main drawbacks. Another potential concern is that the absence of ^{238}U resonance captures significantly lowers the effectiveness of the Doppler feedback mechanism. However, the deleterious effects of IMF are thought to be manageable if the core fraction of IMF fuels containing minor actinides is limited, as has been demonstrated in recent studies [10].

Another candidate material is thorium oxide (or thoria ThO_2), which is not an inert matrix, but nevertheless is useful in this context. Natural thorium is comprised entirely of ^{232}Th and undergoes neutron capture to ^{233}U, which is fissile. As a matrix material for MA-TRU fuel, ^{233}U may make a useful contribution towards the neutron balance, compensating for the negative neutron balance of the minor actinides. Using thoria as the matrix may therefore help reduce the cycle length/initial enrichment penalty noted earlier in Section 5.2.2. Despite this fertile production mechanism, thoria is still compatible with efficient minor actinide burning, because ^{233}U has a very low capture cross-section and in the majority of neutron interactions it fissions. Only a very small proportion undergoes neutron capture to ^{234}U and, in turn, this has a relatively low capture cross-section. Compared with ^{238}U as the seed for transuranic production, ^{233}U produces orders of magnitude less partly because the ^{234}U cross-section is low and partly because the capture chain starts of 5 mass units smaller. For example, to produce ^{241}Am from ^{238}U requires three neutron capture events, while from ^{233}U eight neutron capture events are needed. With the low capture cross-sections for ^{233}U and ^{234}U, needed to start the process, the probability of accumulating eight captures is vanishingly small.

6. Operations, safety and licensing

6.1 Initial fissile loading

Referring to the fission and capture cross-sections of the minor actinides in Table 1.2, it is very clear that all the majority of the minor actinides nuclides make a negative contribution to neutron balance. When a fission event occurs ν new neutrons are born and the neutron that initiated the fission is lost. Therefore the contribution to neutron balance from fission events is proportional to $(\nu\text{-}1)\sigma_f$ (σ_f is the fission cross-section) and the contribution of neutron captures is $-\sigma_c$ (σ_c is the capture cross-section), giving a total contribution $(\nu\text{-}1)\sigma_f - \sigma_c$. Although there are some compensation mechanisms, such neutron captures on ^{238}Pu and ^{244}Cm to produce fissile ^{239}Pu and ^{245}Cm respectively, the overall neutron balance is still negative.

A negative neutron balance implies that the minor actinides must be supplied with neutrons in a thermal reactor and the source of these neutrons is the driver fuel. In the case of LEU driver fuel, this means that the initial ^{235}U enrichment must be increased. If plutonium fuel is used as the driver, a higher initial plutonium loading is required. Failure to increase the initial fissile loading implies that in a batch loaded reactor the cycle length will be shorter and the discharge burn-up reduced.

With LEU fuel, increasing the initial fissile loading is a cost penalty because of the additional uranium ore, conversion and enrichment requirements. Alternatively, a utility might choose to accept a shorter cycle length, but this involves an operational cost penalty, as will be discussed in Chapter 7.

With plutonium fuel, increasing the initial fissile content incurs no cost penalty, but the total plutonium content is usually limited by the need to maintain a negative void coefficient. In conventional MOX fuel the void coefficient normally becomes positive when the total plutonium content is ~12 w/o. Increasing the initial plutonium content to compensate for the negative neutron balance of minor actinides risks encroaching on the void coefficient limit.

The study discussed in Section 5.2 illustrates just how much of an impact the negative neutron balance of minor actinide core loading has on a PWR core. Although the results are specific to the particular reactor and core loading pattern, they still provide a useful reference point because the spectral characteristics of different PWRs (and even BWRs) are not radically different and as an indicator of the order of magnitude effect, the results are transferable.

Recall from Section 5.2 that the equilibrium 18 month PWR fuel cycle was loaded with slightly over 100 kgHM of Am-Cm target rods. The initial enrichment of the UO_2 driver fuel was kept constant and the 3-D nodal whole core model indicated a reduction in equilibrium cycle length from 18.3 GWd/t to 17.1 GWd/t. It is estimated that the 18.3 GWd/t cycle length could be restored by increasing the mean initial ^{235}U enrichment of the entire reload batch by 0.26 w/o. This is a modest penalty that could easily be accommodated technically (provided the 5.0 w/o fabrication limit is not exceeded), though at a cost to the utility. Chapter 7 shows how this translates into an economic penalty.

6.2 Impact on plant operations

This section reviews the impact of MA-TRU fuels on operations for an LWR utility. Much of the discussion, however, will also be valid for other thermal reactor types such as CANDU and HTR where minor actinide loading might be adopted.

6.2.1 Fresh fuel receipt

The first area affected is fresh fuel receipt and handling. Conventional LEU or MOX fuel is normally delivered to the reactor site in unshielded transport containers and the fresh fuel can be handled safely while it is transferred to the fuel ponds. No radiological shielding measures are required for LEU or MOX fuel, though there may be measures taken to minimise exposure to the gamma emissions from MOX fuel. The fresh fuel may be subject to inspection as part of the process before it is loaded in the fuel pond.

MA-TRU fuels would be transported in shielded transport containers and the utility would need to put arrangements in place so that it is able to receive shielded containers and open them safely. It may be necessary to build a new container receipt area and this would need to be equipped with a crane that can handle the additional weight. The utility would also need to install any physical shielding that may be necessary and put into place radiological operating procedures to protect workers during fresh fuel receipt and transfer operations. The modified fuel receipt facility and the operational procedures will need to be approved by the licensing authorities, taking account of any limitations due to heat load during transfer to wet storage, as noted earlier in Section 3.5.

The utility will incur an initial investment cost on account of the changes needed and possibly may incur an additional operational cost every time a delivery of MA-TRU fuels is made. Another potential cost element for the utility would be if it is necessary to upgrade the security arrangements at the reactor site. The security requirements needed for MOX fuel are already more stringent than those for LEU, because of the increased risk posed by plutonium-bearing fuel. With MA-TRU fuels the source term following a terrorist attack is likely to increase further and a review of security arrangements would be necessary to ensure the appropriate level of protection is implemented. This represents another potential operational cost increase to the utility. Bearing in mind that LWRs are refuelled off-line, it is possible that more stringent security arrangements would only need to be in place for the period when MA-TRU fuels are first received and stored in the fuel ponds. Once the MA-TRU fuel is transferred to the core and irradiated, it poses no additional threat.

6.2.2 Core loading

Core loading pattern optimisation will be performed prior to each operating cycle. As noted in Section 5.2, the studies carried to date indicate that MA-TRU fuel assemblies can be accommodated mostly within the normal operating envelopes and acceptable loading patterns are therefore considered feasible.

Other issues that a utility would need to address prior to irradiating MA-TRU fuel assemblies would include:

- review of the core loading strategies that ensure sub-criticality during the reloading procedure;
- review and analysis of fresh fuel handling accident scenarios, including misloading events;
- review of procedures for approaching criticality, including neutron detector source range settings. It may be necessary to determine whether MA-TRU fuels with high intrinsic neutron sources will have any impact on source range detector sensitivities.

In addressing these areas, the utility would incur an initial investment and licensing cost, followed by (presumed) smaller on-going operational costs during each refuelling operation.

6.2.3 Core operation

During the actual operation of the core, the core characteristics would be affected by the presence of the MA-TRU fuel, but if the nuclear design limits are satisfied there is no reason to suppose the core would behave in any untoward manner. There would, however, be an increased risk of the core-follow data not conforming with expectation, especially for the first cores to be loaded with minor actinides. There is no operational experience of LWRs burning MA-TRU fuels and nuclear data libraries cannot be relied upon to the same extent with MA-TRU fuels as they can for conventional fuels. These uncertainties might manifest themselves in the core departing from the expected behaviour and this is a risk that the utility would be exposed to. This would be addressed with a strategy of initially loading small numbers of minor actinide rods in a core, followed by a progressive increase of minor actinide loading with an associated monitoring programme to ensure acceptable behaviour and provide code validation data.

With any new fuel design, there is always an elevated risk of unexpected fuel behaviour, perhaps leading to premature fuel failures. This will be an issue for MA-TRU fuels, especially so for the first commercial-scale loadings. Again, it is the utility that will be exposed to the risk of fuel failures and the costs associated with investigating fuel failures and possibly premature discharge of the fuel assemblies containing failed rods. Conventional LEU and MOX fuels benefit from a very expansive operational experience base that will initially be lacking for MA-TRU fuels. Even as experience accumulates, it is likely that MA-TRU fuel irradiation experience will always lag behind LEU and MOX, so that there will remain an elevated risk of fuel failures.

6.3 Code validation

The core design analysis to date has been carried out using the standard nuclear design codes validated for LEU and MOX cores. In LEU fuel the inventories of the minor actinides are only of the order of 0.1% by mass and any nuclear data uncertainties for the minor actinides would only be expected to have a minor impact on in-core performance. For MOX fuels the minor actinide inventories are considerably higher, but so far there has been no indication that minor actinide cross-sections are affecting the predictive power of the core design codes. However, it is always possible that unknown cancellation effects may be occurring that are masking any biases in minor actinide data and that the underlying biases may become evident at higher loadings.

Minor actinide nuclear data has already been identified as a priority area for nuclear data R&D [11]. New cross-section evaluations for minor actinides will need to be followed up by sensitivity analyses to determine the effect on integral core parameters. The new evaluations will eventually need to be tested against the first commercial-scale minor actinide irradiations to establish the validation base for the nuclear data libraries and nuclear design codes. This can be expected to be a prolonged activity before the nuclear design methods can be considered mature for minor actinide applications.

Unexpected nuclear design behaviour is another risk that will fall on the utilities. At worst, the utility would risk losing output if there was, for example, a need to temporarily down rate or possibly suffer a shortfall cycle length due to insufficient reactivity.

6.4 Licensing timescales

The industrial utilisation of new types of fuel requires detailed preparation from the technical as well as from the licensing viewpoint. In particular, the licensing situation of all concerned areas has to be reviewed: transportation, manufacturing, design, and irradiation in the reactor. Acquiring new licenses or the adaptation of existing licenses may be a tedious and time consuming process and should start early in the project.

A general sequence of steps for licensing of new types of fuel after specification of the manufacturing process is described in Table 6.1.

Dependent on the manufacturing, design and irradiation experience with similar types of fuel Phase I and II may be overlapping. A reduction of the time needed for industrial implementation of MA-TRU fuel is dependent on the fuel design. If the admixture of minor actinides to the new fuel expected to have only minor impact on the thermal-mechanical behaviour, then the required tests can partly be performed in parallel. In any case, the time for commercial implementation of new types of fuel is considerable and dependent on expert opinion on the necessity for test irradiations it is in the range of minimum 10-15 years.

Table 6.1: Licensing of MA/TRU fuel

	Characterisation of licensing phases	Manufacturing	Design	Transportation, handling, reprocessing, long-term storage	Irradiation
Phase I: (1-2 years; >5 years if irradiation tests are required)	Definition of boundary conditions for intended use of MAThR fuel (type of reactor, operation mode, etc.). Evaluation of licensing requirements	Determination of manufacturing facility and check for required adaptations of specifications, guidelines, extensions of licenses and of manufacturing equipment; Manufacturing of test pellets	Determination of - design parameters (e.g. fuel thermal conductivity, radial power profile) from ▪ literature ▪ manufacturing tests ▪ test irradiations - applicability of design codes	Identification of possible shipping casks. Identification of boundary conditions for reprocessing and/or long-term storage	If deemed necessary: Irradiation of test fuel rods for qualification of thermal-mechanical and nuclear design parameters; continuous surveillance and post-irradiation examinations
Phase II: (> 7 years including manufacturing, post irradiation examinations)	Preparation of licensing	Adaptation of manufacturing equipment, specifications and guidelines; Initiation of the licensing process for manufacturing; Qualification of the manufacturing process; Manufacturing of fuel pellets and rods in LTA quantities	Qualification of thermal-mechanical and nuclear design parameters (e.g. fuel dimensional stability, thermal conductivity, power profile, nuclear cross-sections) and design codes; Adaptation of specifications	Update of - licensing status of the shipping/storage casks for requirements arising from the new fuel - existing guidelines for shipping and handling (manufacturing site, plant site, storage site)	Insertion of Lead Test Assemblies (LTAs) for generic demonstration of irradiation behaviour; continuous surveillance and post-irradiation examinations
Phase III: (1-2 years)	Licensing		Thermal mechanical and neutronic design of pellet, fuel rod and fuel assembly	If required: licensing of the shipping cask	Licensing of insertion of fuel assemblies in reload batch size
Phase IV:	Commercial implementation	Manufacturing of MAThR fuel rods/assemblies	Update of generic design		Insertion of final design MAThR fuel assemblies in batch size

7. Economics

7.1 Costs and benefits

The economics of minor actinide recycling in thermal reactors is a complex subject for which there are, at present, no definitive answers. Nevertheless, it is important to consider the economic issues here, even if only qualitatively, to highlight what questions will need to be addressed before minor actinide recycling can be implemented in thermal reactors.

When introducing a new fuel design or other fuel cycle innovation, a utility would normally expect to bear an additional cost in implementing the change, but would expect to demonstrate a tangible benefit in return. For example, an increase in discharge burn-up would incur extra initial cost for the fuel because of the higher initial enrichment required. In return, the utility would reduce its fuel throughput and possibly make other operational savings that would compensate.

There are clear direct costs to the utility associated with minor actinide recycling that will similarly need to be balanced against the benefits. However, a complicating factor for minor actinide recycling is that the direct costs will need to be met by the utility, while the benefits may not necessarily accrue to the utility, but rather to society as a whole. Another complicating factor is that the benefits may be in the form of reduced future radiological dose commitments and a mechanism is needed to equate the dose reduction to an economic saving. This raises the possibility that mechanisms may need to be found whereby the utility is compensated for its direct costs through subsidisation mechanisms from governments.

The additional costs that a utility would incur if it decided to recycle minor actinides in a thermal reactor would be:

1. The cost of separating the minor actinides during reprocessing and incorporating them in the nuclear fuel, in either homogeneous or heterogeneous MA-TRU fuels.
2. The additional costs associated with transport of MA-TRU fuels to the reactor site.
3. The additional costs associated with the safe handling and transfer of the MA-TRU fuel to pond storage.
4. Any direct cost of the additional fissile material needed to compensate for the minor actinide loading. For a uranium fuelled thermal reactor, a higher initial enrichment would be incurred, which implies a higher cost for uranium procurement and enrichment services. For a plutonium-fuelled thermal reactor, it is possible that a compensatory increase in initial plutonium loading will be required and that no additional direct cost will be incurred if the plutonium is owned by the utility.
5. The additional costs associated with licensing the reactor for operation with MA-TRU fuels.
6. Any additional costs associated with discharge, storage and transport of MA-TRU fuels.
7. Any additional costs associated with management of spent MA-TRU fuels, which might be stored and conditioned for disposal or recycled further. This would

include the management and disposal of any additional waste streams generated by management of spent MA-TRU fuels.

Balancing these costs are the benefits of minor actinide recycling:

(a) The avoided future cost of repository construction and operation that can be viewed principally as an economic benefit (through there may be other beneficial aspects, such as increased political acceptance). This might be evident as a direct economic benefit to the utility if its spent fuel costs were reduced. However, with a levelised cost methodology, the spent fuel costs will incur a discounting effect over many years such that such a reduction may not have a noticeable effect on the levelised fuel cycle cost.

(b) Minor actinide recycling will involve an increase in radiological doses to workers during separation, fuel fabrication and all the subsequent operations involving MA-TRU fuels. On the other hand, recycling of minor actinides may reduce the future radiological dose commitment from geological repositories. If it can be demonstrated that the future avoided dose outweighs the additional operational dose commitment, then this would constitute a benefit. Any avoided future radiological dose uptake represents a societal benefit that could be regarded as an economic gain.

It is important here to try to estimate at least the order of magnitude of the costs associated with items 1) to 7) above would be and to balance these against the economic benefits (a) and (b). This is an important input to the debate that will need to take place between utilities and government before minor actinide recycling can be realised. It should be understood that minor actinide recycling is currently very far from commercial realisation and that any economic analysis will inevitably be very imprecise at this stage. Definitive economic estimates will not be available until minor actinide recycling has actually been implemented at commercial scale, at which point construction and operating costs for the associated fuel cycle plants will be better known. Nevertheless, Section 7.2 attempts to provide some indicative estimates of the order of magnitude costs that the utility will incur.

7.2 Cost impact

Each of the seven areas where minor actinide recycling would incur additional costs is considered here in turn. In some cases it is premature to attempt to make a quantitative estimate and all that can be done here is to make some qualitative observations. The overall cost analysis will therefore be incomplete, but the overall analysis is still worthwhile because it quantifies at least some of the costs and identifies where the gaps are:

7.2.1 Separation

It is not possible to estimate the cost of a minor actinide separation plant at this time. No commercial scale plant designs have been developed for a reprocessing plant with a minor actinide separation capability and there are no engineering estimates of the capital and operating and maintenance (O&M) costs. Moreover, it might be difficult in practice to disentangle the cost of the minor actinide separation facilities from the overall costs of the reprocessing plant. A more valid approach might be to consider the entire plant as a single entity and concentrate instead on demonstrating that the overall back-end costs are competitive with conventional reprocessing plants. This suggests that the problem should perhaps be inverted and the need for economic competitiveness to be imposed as a requirement on the design of the integrated plant. While it may be acceptable for a reprocessing plant capable of separating minor actinides to have a higher unit cost than a conventional reprocessing plant, any such cost penalty should be kept within reasonable bounds. Reprocessing unit costs indicated in the 1994 OECD/NEA study of back-end costs are of order of 1 000 USD per kgHM. It would perhaps be sensible to aim for any cost penalty associated with minor actinide recycling to be no more than a few hundred USD/kgHM, or justifying it to utilities will be very difficult.

7.2.2 Transport

No transport packages have been licensed for MA-TRU fuels and it is not possible at this time to estimate the effect on transport costs. It is possible that existing spent fuel transport packages might be adaptable for carrying MA-TRU fuels, which would avoid the need to develop and license new package designs. It would be very beneficial if the same packages used for transporting spent fuel assemblies to the reprocessing plant could also be used to ship MA-TRU fuel assemblies on the return trip, rather than being shipped empty. Whether this is possible and what loadings of MA-TRU fuels could be permitted depends on the specific details of the transport package designs and in particular the neutron source strength of the MA-TRU fuels. It will have to be sufficient for now to note that spent fuel transport, although expensive in absolute terms, only represents a very small fraction of the fuel cycle cost and even if a substantial multiplier was necessary for MA-TRU fuels, it is unlikely to have more than a minor overall impact.

7.2.3 Fresh fuel receipt and storage

The utility would need to install any additional equipment needed for receipt, inspection, handling and storage of un-irradiated MA-TRU fuels and to license such operations with the licensing authority. For heterogeneous minor actinide target fuels, the utility would need to provide equipment and personnel for receipt and storage, loading and un-loading the targets in-core during refuelling operations. The initial equipment installation is likely to represent a substantial cost perhaps a few million dollars and it seems reasonable to allow for an order of magnitude allocation of USD 1-2 m per reload for heterogeneous targets and perhaps less for homogeneous MA-TRU fuels.

7.2.4 Core reactivity effect

The presence of MA-TRU fuels in the core will have an impact on the nuclear design behaviour of the core. A recent study of americium-curium targets loaded in a PWR core [10] suggests that worthwhile loading of 100 kg of minor actinides could be accommodated in a PWR while satisfying the normal nuclear design limits. This will ensure that the nuclear design performance of the core will not be a factor that will limit core output. However, the analysis in [10] does confirm that the presence of minor actinides in the core depresses core reactivity, requiring a compensatory increase in initial fissile loading. For a uranium fuelled core, this implies increased uranium ore and enrichment procurement costs which can be evaluated with reasonable confidence.

The utility could compensate for the reduction in core reactivity by increasing the initial enrichment of the fuel. This appears as a direct cost penalty through the increase in uranium ore conversion and enrichment costs. In terms of minimising the operational penalty, this is the preferred option, because it preserves the nominal cycle length and the utility does not lose out on generating revenue. The enrichment penalty equates to an increase in the initial ^{235}U enrichment of 0.26 w/o averaged across the entire 84 reload batch, as indicated in [10]. Assuming a nominal mean enrichment of 4.2 w/o, a uranium ore price of 130 USD/kg, conversion at 10 USD/kg, enrichment at 90 USD/kgSWU, a 5% discount rate and 0.5% losses this equates to a direct cost penalty to the utility of USD 5.6 m per 18 month reload. This would be very significant to a utility.

7.2.5 Licensing

The utility would need to submit a safety case to the licensing authority to demonstrate prior to loading MA-TRU fuels. This would need to cover all aspects of operations involving MA-TRU fuels from receipt and storage, core loading irradiation, discharge, storage and transport. The safety case would need to show that the presence of MA-TRU fuels would not impact safe behaviour of the fuel and core in normal operation and in fault conditions. The cost of generating the initial safety case is likely to be significant, perhaps several million US dollars. Spread over 10 years and allowing for other costs associated with licensing such as fuel monitoring programmes to demonstrate satisfactory performance of the MA-TRU fuels, it would be reasonable to assign an order of magnitude allocation of USD 1-2m per reload.

7.2.6 Discharge, storage and transport of spent MA-TRU fuels

There is the possibility that spent fuel handling operations at the reactor may be impacted by MA-TRU fuels. Spent fuel is discharged into the fuel cooling ponds, where it will remain until such time as it is transferred to dry storage or transported to a separate spent fuel facility for interim storage/disposal or reprocessing. There is the possibility that spent fuel management cost at the reactor site may be impacted by the different characteristics of MA-TRU fuels. In particular, transport costs may be impacted if the MA-TRU fuels have a higher heat output or neutron output than conventional fuel assemblies. The possible cost implications cannot be estimated at present, because the necessary R&D work has not been carried out.

7.2.7 Spent fuel management

Spent fuel management costs may be affected by MA-TRU fuels. Spent MA-TRU fuel that is handled along with conventional fuels in a fuel conditioning plant or reprocessing plant may have a cost impact. It may be necessary to make plant modifications to equipment and possibly to operational procedures, both of which could affect unit costs. It is possible that plant throughputs would be limited while handling MA-TRU fuels, which could have a major impact on unit costs. Alternatively, where MA-TRU fuels needed to be handled in separate facility, the entire investment and operating cost of such a facility would need to be supported either by the utilities using it or by the government. Since no designs for spent fuel conditioning or reprocessing plants capable of handling minor actinides have been developed, it is premature to attempt to make a quantitative estimate at this time.

7.2.8 Overall cost impact

These are just very crude and incomplete estimates, but they amount to a quantifiable economic cost to the utility of the order of USD 10 m per reload, which is substantial. This estimate excludes several other cost components which are also likely to be substantial and so should be regarded as a lower bound figure that is intended only to inform the debate at this early stage of development.

7.3 Economic benefits

7.3.1 Repository cost savings

It is possible that as a direct result of minor actinide recycling, the future capacity of the geological repository may be reduced or that capacity may be deferred for a number of years. In such a scenario, there would be an economic benefit to the organisation; usually the government that is responsible for funding the construction and operation of the repository. It is not possible to attempt a meaningful estimate of how much of a saving this might represent, not least because any such saving is likely to be very dependent on the specific scenario that might apply to a given country. In theory, a utility might benefit by seeing a reduction in its spent fuel disposal levy, but given the long lead times involved, discounting effects are likely to make the benefit very marginal.

It is possible that the socio-political benefits might actually outweigh any direct economic effect. It is conceivable that minor actinide recycling might be viewed as removing a potential barrier to sustainability if it reduces the repository capacity needed or defers capacity until later.

7.3.2 Avoidance of radiological doses

The economic equivalent of avoided future radiological doses can in principle be calculated given knowledge of:

- The reduction in radiotoxicity results from minor actinide recycling. Once the fuel cycle scenario has been defined, this can easily be evaluated using fuel cycle scenario analysis codes ORION [12] and NFCSS [13].
- The transfer coefficients of radionuclides from the geological repository to biosphere as a function of time. This is much more difficult to evaluate, as it depends on the specific geological and hydrological characteristics of a particular repository.

Calculating the economic benefit of avoided future radiological doses will require a better understanding of the transfer coefficients and also requires agreement as to how to assign an economic value to avoided dose i.e. what is the equivalent economic cost of 1 man-Sievert of exposure. There have been various attempts to estimate the Sv-equivalent cost in the literature and different valuations have been arrived at. Reference [14] makes an important distinction between 1 man-Sievert of exposure to radiation workers, who choose to accept the small risk associated with occupational exposure and members of the public exposed to aerial and water-borne exposure without being able to choose otherwise. The work reported in [14] arrived at a weighting factor to apply to public exposure of between 2 and 6 times that which should apply to radiation workers. This was based on an attitude survey from an opinion poll and although the precise values to use are debatable, the principle appears sound. The weighting in favour of the cost of public exposure will benefit future avoided dose uptake over operational dose commitment.

At the present time, the lack of understanding of radionuclide migration from repositories prevents quantitative estimates of the economic benefit from avoided future dose. Any such assessment is likely to be different for each specific country's scenarios and requires a consensus to be established as the economic equivalent value of avoided dose.

7.4 Subsidisation mechanisms

Section 7.2 has quantified some direct costs that a thermal reactor operator would incur in recycling minor actinides in a thermal reactor. Although incomplete, those costs which have been quantified (however approximately) are quite significant. On the other hand, Section 7.3 has identified potential benefits, which though not quantified, may potentially be sufficient to counter balance the costs. However, the important point to bear in mind, is that while the utility will incur additional costs, the benefits are likely to be in the form of indirect societal returns. In these circumstances, it may be necessary for governments to put in place subsidisation mechanisms to encourage thermal reactor operators to recycling minor actinides.

For a private company, agreement of the level of subsidisation would be a matter for commercial negotiation. At the very least, a utility might expect its immediate direct costs to be covered and other factors might also influence the negotiations. For example, there would be an element of risk associated with minor actinide recycling and a utility might demand compensatory payment to cover it.

The overall conclusion is that subsidisation mechanisms are likely to be necessary to encourage thermal reactor operators to recycle minor actinides. At present, it is not possible to fully quantify the competing effects and further R&D is required to quantify on the one hand the additional costs and on the other hand, the economic equivalent benefits.

8. Research and development needs

This chapter discusses the research and development requirements for minor actinide recycling in thermal reactors. It follows the flow of minor actinides through each fuel cycle stage, starting with separation of the minor actinides from spent nuclear fuels during reprocessing:

8.1 Separations

Flowsheet development will be needed for the selected minor actinide separations technology. R&D is needed to take separations technology from its present immature state to commercial readiness. Facilities work would initially be carried out at laboratory scale and this would be followed by a sub-commercial scale development/demonstration facility, leading eventually to the construction and operation of the full scale plant. The minimum timescale for commercial implementation is of the order of 20 years.

It would be useful if at an early stage conceptual separation plant designs could be developed which would be useful for informing the justification process and provide initial indications of capabilities, limitations. Of particular interest would be to characterise the waste streams from the minor actinide separation process and check whether they would be compatible with existing waste streams and disposal requirements.

Most R&D has been focused on the initial separation of minor actinides from spent fuel. However, another important aspect will be the subsequent management of irradiated minor actinide target fuels, which may involve conditioning for disposal or further separation and recycling steps. It is important that R&D on target management is not neglected.

8.2 Fuel fabrication

R&D is needed to develop the selected fuel fabrication technology to commercial readiness. A small number of laboratory scale fuel fabrication facilities are already available that could carry out small scale development work. Further progress would require a demonstration scale fuel fabrication facility, leading to the design and construction of a commercial scale plant. The timescale to commercial readiness is estimated at 10 to 15 years, as noted earlier in Section 6.4.

As, with separations technology, it would be useful for justification if conceptual fuel fabrication plant designs were available at an early stage.

8.3 Fuel transport

The technical requirements for transport of un-irradiated and irradiated MA-TRU fuels will need to be developed. It would be very helpful if existing transport container designs could be adapted for transport of MA-TRU fuels. An early evaluation of the feasibility of using existing transport container designs for MA-TRU fuels would be helpful to establish any limitations and provide initial indications of costs. Theoretical studies could be completed on a short timescale, perhaps 2 to 3 years and provide an useful indication of limitations and potential costs. Development and licensing of new or modified transport containers would require perhaps 5 to 10 years, but is unlikely to proceed until there is a clear commercial requirement.

8.4 Fuel design

R&D is required to progress MA-TRU fuel designs from the conceptual stage, where they are at present, to commercial readiness. Further research is needed into materials properties for MA-TRU fuels to provide the information needed for the fuel performance codes which are used to demonstrate satisfactory thermo-mechanical fuel behaviour. The fuel behaviour codes will require validation against experimental measurements and from post irradiation examination (PIE).

Experimental work will be required in the following areas:

- Fuel materials properties measurements.
- Small scale irradiation tests of MA-TRU fuels in test reactors, which will require suitable fuel fabrication facilities to be available and qualified to handle minor actinides in small quantities (up to a few kg).
- Small scale irradiation trials in power reactors (small numbers of individual minor actinide rods) leading to intermediate scale irradiation trials (small numbers of minor actinide assemblies) and eventually to full scale commercial loadings.

The timescale for irradiation testing of MA-TRU fuels and validation of fuel performance codes is of the order of 10 to 15 years, with a further lead time of perhaps 5 years for test fuel fabrication, giving an overall programme of 15 to 20 years.

8.5 Core design

Core design R&D is required to develop viable core loading patterns and fuel management schemes for MA-TRU fuels and to validate nuclear design methods. Minor actinide nuclear cross-section data will need to be improved and this may require new experimental measurements and nuclear data evaluation effort. Validation of nuclear design methods may necessitate integral measurements in zero power facilities, followed by small-scale irradiation trials in commercial reactors. The overall programme would be similar to that for fuel design, 5 years fabrication followed by 10 to 15 years to progress to commercial scale loading.

8.6 Spent fuel characterisation

Spent fuel characterisation data will need to be generated for MA-TRU fuels. This includes decay heat and neutron source data, which is important for spent fuel storage and transport, reprocessing and disposal. The inventory codes used to generate decay heat and neutron source data will need to have their validation base extended to cover the higher minor actinide inventory. The improved cross-section data needed for core design will also be applicable to the inventory codes, but improved nuclear data decay data may also be required. PIE of MA-TRU fuels irradiated in test reactors may be needed to validate the inventory codes. The timescale matches that of core design R&D, 15 to 20 years to progress to commercial core loadings.

8.7 Fuel cycle assessment

Fuel cycle R&D is needed to provide a full understanding of minor actinide recycling to inform the justification process using the full range of assessment tools that are becoming increasingly prevalent worldwide. Useful preparatory work could begin immediately, but improved assessments will be possible once the other R&D work above starts to produce data which will better characterise minor actinide plants. The justification of minor actinide recycling is not as straightforward as justifying new build plants, whose main benefit is the value generated by electricity production. It will be essential to be able to support clear justification statements for minor actinide recycling, which requires a very good understanding of the benefits and detriments and the balance between them. Such assessment work will be required throughout the period leading up to the commercial application of minor actinide recycling.

The fuel cycle assessment will need to encompass technical aspects to do with materials flows, radiological dose uptake to workers and the public, long term radiotoxicity, repository heat load and economics. Economics assessment is a major part of the justification and will need to be developed as the conceptual plant designs evolve and the available estimates of plant construction and operational costs improve.

8.8 Overall timescales

From the discussion above, it is clear that the minimum timescale for development of minor actinide recycling in thermal reactors is of the order of 15 to 20 years to commercial readiness. The required R&D will involve several parallel R&D streams that will be closely linked to one another. It is only likely to happen if an early consensus can be established as to the benefits of minor actinide recycling in thermal reactors and a clear need agreed on. The R&D programme will need to be well funded to achieve the goal in the 15 to 20 years and any lack of commitment will extend the R&D period for longer.

9. Summary and recommendations

The purpose of this study was to review the literature to determine whether thermal reactors are a viable alternative to fast reactors as minor actinide burners. The clear outcome of the review is that thermal reactors can indeed burn minor actinides effectively and could play a valuable role in reducing the radiotoxicity of nuclear waste. Although thermal reactors may in some respects be more limited in what they can achieve, they may nevertheless be valuable in this role, either as an intermediate step towards a fleet of minor actinide burning fast reactors or as an end point in their own right.

As an intermediate step, minor actinide burning in thermal reactors could be used to demonstrate the technologies required, and possibly expedite the deployment of minor actinide burning fast reactors. These include minor actinide separation, MA-TRU fuel fabrication and transport and spent fuel management. Large numbers of thermal reactors are operational today and only a few would be needed to demonstrate the technology. In principle, there is no reason why an R&D programme on minor actinide burning in thermal reactors could not begin immediately, perhaps demonstrating the technology within a period of 10 to 15 years. Even the most optimistic forecasts for fast reactor deployment envisage only a few demonstration reactors being operational within such a timescale. This is one possible role of minor actinide burning in thermal reactors. A successful outcome would demonstrate the required technology on a commercial scale and help inform the debate as to whether minor actinide burning is necessary over the long term.

The objective of minor actinide burning being adopted on a commercial scale in current and new build thermal reactors within a much shorter timescale is an end-point in its own right. The reactors are already in operation or being built and large-scale minor actinide burning sufficient to stabilise the minor actinide inventory could possibly be achieved within 20 to 30 years, much sooner than could be reasonably expected in fast reactors. The capacity to demonstrate stabilisation of long-lived radionuclides would be helpful in justifying new nuclear build, especially if many countries were to adopt aggressive new build construction programmes.

Minor actinide burning R&D is currently still at the conceptual stage and has yet to progress beyond theoretical studies and small-scale laboratory sample sizes. To progress to a commercial-scale demonstration and ultimately commercial deployment will require considerable commitment. The necessary steps would be:

- to develop separation flowsheets at laboratory scale and then progress to commercial-scale separation facilities, demonstrating that these can meet the requisite radiological and environmental standards;
- to develop MA-TRU fuel designs, demonstrating these in small scale irradiation programmes and then in commercial scale irradiation programmes. Utilities will need to be convinced that MA-TRU fuels can be reliably irradiated;
- to set up a fuel performance and nuclear design code validation programme;
- to develop MA-TRU fuel fabrication plant designs and demonstrate their viability at commercial scale, again meeting the requisite radiological and environmental standards.

- to develop strategies for managing irradiated MA-TRU fuels safely, either in preparation for eventual direct disposal or recycling, again meeting the requisite radiological and environmental standards;
- to demonstrate that the radionuclide waste streams from minor actinide separation, fuel fabrication and recycling plants can be integrated with existing radioactive waste;
- to ensure that LWRs can achieve relative efficiency in minor actinide burning if limited in the number of recycling and hence in scope. Transmuting in these reactors could hence prepare a more effective minor actinide burning in fast reactors.

The principal obstacle to the implementation of these programmes will be securing the necessary funding from governments, since it is unlikely to come from utilities because the benefits, as discussed in the report, will be mostly societal and not of direct value to the utilities. Before such large scale funding can be justified, clear demonstrations of the benefits will be required. This requires an international effort to develop improved models of future world scenarios for nuclear growth and in particular to indicate how the spent fuel and/or high-level waste arisings will be managed. If it can be demonstrated that current strategies for managing high-level waste from nuclear plants will ultimately limit the deployment of nuclear reactors, then the result will be a strong justification for minor actinide burning.

References

[1] ICRP (1995), *Age-Dependent Doses to Members of the Public from Intake of Radionuclides, Part 5 Compilation of Ingestion and Inhalation Coefficients*, ICRP Publication 72, Ann. ICRP 26 (1).

[2] NEA (2011), *Potential Benefits and Impacts of Advanced Fuel Cycles with Actinide Partitioning and Transmutation*, OECD, Paris.

[3] Ronald J. Ellis (2009), "Reactor physics assessment of the inclusion of unseparated neptunium in MOX reactor fuel", *Advances in Nuclear Fuel Management IV* (ANFM 2009), Hilton Head Island, South Carolina, USA, 12-15 April 2009, American Nuclear Society, LaGrange Park, IL.

[4] B Hyland, G.R. Dyck (2007), "Actinide burning in CANDU reactors", Global 2007 Conference on Advanced Nuclear Fuels Cycles and Systems, Boise, Idaho.

[5] B. Hyland, R.J. Ellis, G.R. Dyck, G.I. Maldonado, J.C. Gehin, G.W.R. Edwards (2009), "Transmutation of americium in light and heavy water reactors", Proceedings of Global 2009, Paris.

[6] J.C. Kuijper *et al.* (2004), "HTR-N reactor physics and fuel cycle studies", 2nd International Topical Meeting on HTR technology, Beijing, China, held on 22-24 September 2004.

[7] J.C. Kuijper *et al.* (2010), "Plutonium and minor actinide management in thermal high-temperature gas-cooled reactors", Publishable Final Activity Report, European Commission, PUMA Project document PUMA-1006-D411, NRG Document 21944/10.104869, Nov 2010, www.puma-project.eu.

[8] IAEA (2009), *Status of Minor Actinide Fuel Development*, *IAEA Nuclear Energy Series*, No. NF-T-4.6, Vienna.

[9] T.A. Taiwo, T.K. Kim, J.A. Stillman, R.N. Hill, M. Salvatores, P.J. Finck (2006), "Assessment of a heterogeneous PWR assembly for plutonium and minor actinide recycle", *Nuclear Technology*, Vol 155, No. 1.

[10] G.M. Thomas, K.W. Hesketh, S. Arm (2010), "The potential of pressurised water reactors for recycle of americium-curium", Physor, Pittsburgh, Pennsylvania, US, 9-14 May 2010.

[11] NEA Nuclear Data High Priority Request List, www.oecd-nea.org/dbdata/hprl/.

[12] A. Worrall, R. Gregg (2007), "Scenario analyses of future UK fuel cycle options", *Journal of Nuclear Science and Technology*, Vol. 44, p.249.

[13] IAEA Nuclear Fuel Cycle Simulation System (NFCSS), www-nfcis.iaea.org/.

[14] C. Schieber, T. Schneider, L. Eeckhoudt, P. Godfroid (2000), "The monetary value of the man-sievert: A proposal for public exposure", *Bulletin of the Nuclear Protection Evaluation Centre*.

[15] G. Youinou, A. Zaetta, A. Vasile, M. Delpech, M. Rohart, JL. Guillet (2001), "Heterogeneous assembly for plutonium multi-recycling in PWRs: The CORAIL concept", *GLOBAL 2001,* Paris, France.

[16] H. Golfier, J. Bergeron, A. Puill, M. Rohart (2000), "Multi-recycling of plutonium and incineration of americium, curium and technetium in PWRs", *Proc. Int. Conf. Safewaste'2000,* Montpellier, France.

[17] J. Bergeron, A. Puill, S. Aniel, M. Rohart, G. Rouvière (2000), "Plutonium multi-recycling in PWR: The CORAIL concept"*, ICONE 8*, USA.

[18] G. Youinou, F. Varaine, A. Vasile (2003), "Plutonium and americium multirecycling in the European pressurized reactor (EPR) using slightly over-moderated U-235 enriched MOX fuel assemblies", presented at the International Conference on Advanced Nuclear Energy and Fuel Cycle Systems (GLOBAL 2003), New Orleans, US.

[19] H. Golfier, M. Rohart, S. Aniel, J. Bergeron, J-P. Deffain (2001), "Plutonium and minor actinides recycling in PWRs with new APA concepts"*, GLOBAL 2001*, Paris, France.

[20] F. Varaine, C. Chabert, G. Youinou, D. Warin (2003), "Comparative review of the long-lived wastes transmutation performances in nuclear reactor", *Global 2003* New Orleans, LA.

[21] F. Varaine, C. Chabert, G. Youinou, D. Warin (2005), "Transmutation In Industrial Thermal Reactors (PWR)", ENC 2005 Paper no.265, Versailles, 11-14 December 2005.

Further reading

Canada

- Bronwyn Hyland, Gary Dyck, Andrew Morreale and Ragnar Dworschak: Transmutation of Actinides in CANDU Reactors, 10th Information Exchange Meeting on Partitioning and Transmutation in Mito, Japan on, 6-10 October 2008.
- B. Hyland and G.R. Dyck: Actinide Burning in CANDU Reactors, Global-2007, Advanced Nuclear Fuel Cycles and Systems, Boise, ID, September 9 - Friday, September 14, 2007.
- A.R. Dastur, A.S. Gray, N. Gagnon D.B. Buss, R.A. Verrall: The Role of CANDU in Reducing the Radiotoxicity of Spent Fuel, Global-93, Proceedings: Future of Emerging Nuclear Systems, Seattle, 1993.
- J.H. Bultman, N.Gagnon, A.R. Dastur: Statics and Dynamics of CANDU Transuranics Burners, International Conference on Evaluation of Emerging Nuclear Fuelcycle Systems, Global'95, 11-14 September 1995, Versailles, France.
- P.S.W. Chan, M.J.N. Gagnon, P.G. Boczar, R.J. Ellis, R.A. Verrall: CANDU® – A Versatile Reactor for Plutonium Disposition or Actinide Burning, Global '97 International Conference on Future Nuclear Systems, held in Yokohama, Japan, 5-10 October 1997, AECL-11853.
- T. A. Taiwo, T. K. Kim, F. J. Szakaly, R. N. Hill, and W. S. Yang, G. R. Dyck, B. Hyland, G.W.R. Edwards: Comparative Study of Plutonium Burning in Heavy and Light Water Reactors, Proceedings of ICAPP 2007 CW-123700-CONF-002, Nice, France, 13-18 May 2007.
- P.G. Boczar, M.J.N. Gagnon, P.S.W. Chan, R.J. Ellis, R.A. Verrall, A.R. Dastur: Advanced CANDU Systems for Plutonium Destruction, NATO Advanced Research Workshop (ARW) on *"Advanced Nuclear Systems Consuming Excess Plutonium"*, Moscow, Russian Federation.

France

- Updates on major recent developments in France, 39th RWMC Meeting 15-17 March 2006.
- Commission Nationale d'Évaluation Relative aux recherches sur la gestion des dechets Radioactifs, RÉSUMÉ ET CONCLUSIONS, Rapport d'Evaluation No 11, Juin 2005.
- Andra research on the geological disposal of high-level long-lived radioactive waste Results and perspectives, Agence nationale pour la gestion des déchets radioactifs (ANDRA) June 2005.
- French R&D on Partitioning and Transmutation of Long-lived Radionuclides, An International Peer review of the 2005 CEA Report, ISBN: 978-92-64-03256-9, ISBN: 92-64-02296-1, OECD/NEA 2006.
- J. Journet, G. Vambenepe, J. Vergnes (EdF), F. Biagi, H. Sztark, Minor Actinides Transmutation in Oxide Fuelled Fast Reactors, Global-93, Proceedings, Future of Emerging Nuclear Systems, Seattle, 1993.
- Transmutation of Minor Actinides in PWRs: Preparation of the "ACTINEAU" Experiment, Y. Guerin, N. Cocuaud, T. Duverneix, J.L. Faugere, C. Morin (CEA), Global-93, Proceedings: Future of Emerging Nuclear Systems, Seattle, 1993.

- A. Puill, J. Bergeron: Incineration of Actinide Targets in a Pressurized Water Reactor Spin Project, Global-93, Proceedings:Future of Emerging Nuclear Systems, Seattle, 1993.
- M. Delpech, J. Tommasi, A. Zaetta, M. Salvatores, H. Mouney, G. Vambenepe: Transmutation of Americium and Curium: Review of Solutions and Impacts, Fifth OECD/NEA Information Exchange Meeting on Actinide and Fission Product Partitioning and Transmutation , SCK-CEN, Mol, Belgium, 25-27 November 1998
- F Varaine, C Chabert, G Youinou, D Warin: Transmutation in industrial thermal reactors (PWR), ENC 2005, Versailles, France, 11-14 December 2005
- A. Zaetta, F. Varaine, R. Jacqmin, D. Warin, Neutronic aspects for reactor systems for transmutation, ENC 2005, Versailles, France, 11-14 December 2005 .
- G. Youinou, F. Varaine, A. Vasile, Plutonium and Americium Multi-recycling in the European Pressurized Reactor (EPR) Using Slightly Over-Moderated U-235 Enriched MOX Fuel Assemblies, Global 2003, New Orleans, US, 16-20 November 2003.
- F. Varaine *et al.* Comparative review of the long lived wastes transmutation performances in nuclear reactor, Global 2003, New Orleans , US, 16-20 November 2003.
- H. Golfier, M. Rohart, S. Aniel, J. Bergeron, J.P. Deffain, “Plutonium and Minor Actinides Recycling in PWRs with New APA Concepts”, Proceedings of GLOBAL 2001 Conference, Paris, France, 2001.
- J.P. Grouiller, L. Boucher, P. Bourdot, F. Varaine, M. Delpech, D. Warin: A synthesis of possible separation and transmutation scenarios studied in the frame of the French law for waste management, Global 2005, Tsukuba, Japan, 9-13 October 2005.
- A. Vasile, F. Varaine, M. Delpech, Minor actinides recycling in current pressurized water reactors, ICAPP’06, Reno, US, 4-8 June 2006.
- L. Boucher, J.P. Grouiller, C. Courtois, S. David, M. Maurin, Impact on nuclear waste production of different transmutation scenarios, ICONE14, Miami, US, 17-20 July 2006.
- D. Warin: Status of the French Research Program on Partitioning and Transmutation, Journal of Nuclear Science and Technology, Vol. 44, No. 3, p. 410–414 (2007).

Japan

- Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management, National Report of Japan for the Second Review Meeting, October 2005.
- S. Hirusawa: Overview on the Nuclear Energy Policy and on the High-Level Waste Management in Japan, Expert Group on Timing of High-Level Waste Geological Disposal, OECD/NEA, 27-29 March 2006.
- Overview on the Nuclear Energy Policy and on the High-Level Waste Management in Japan (1st Draft) 28 February 2006.
- M. Yui, S. Kawakami, H. Makino: Cost Analysis of Direct Disposal of Spent Fuel in Japan.
- The Challenges and Directions for Nuclear Energy Policy in Japan – Japan’s Nuclear Energy National Plan – Nuclear Energy Policy Planning Division, Ministry of Economy, Trade and Industry (METI), December 2006.
- S. Okajima: Physics and Data Challenge of Minor Actinide Recycle - Integral Experimental Facility for Next Generation - In depth discussion, 19th OECD/NEA/NSC meeting, 26 June 2008.
- M. Osaka, H. Serizawa, M. Kato, K. Nakajima, Y. Tachi, R. Kitamura, S. Miwa, T. Iwai, K. Tanaka, M. Inoue, Y. Arai: “Research and Development of Minor Actinide-containing Fuel and Target in a Future Integrated Closed Cycle System”, Journal of NUCLEAR SCIENCE and TECHNOLOGY, Vol. 44, No. 3, p. 309–316 (2007).

- H. Takano, H. Akie, K. Kaneko *et al.* “Higher Actinide Confinement/Transmutation Fuel Cycle in Fission Reactors”, Proc. of International Conf. on the Physics of Reactor, Operation, Design and Computation, 23-27 April 1990, Marseille, France, Vol.4, pp. III. 145-155, 1990.
- S. Fujine, H. Takano, T. Yamada *et al.* “Fuel Cycle System for Plutonium Recycling in Very High Burnup LWRs”, Proc. of GLOBAL’93 Inter. Conf. on Future Nuclear Systems: Emerging Fuel Cycles & Waste Disposal Options, Vol.2, pp. 1157-1162, Sep. 12-17, Seattle, 1993.
- H. Takano, K. Nishihara, K. Tsujimoto *el al.* “Transmutation of Long-Lived Radioactive wastes Based on Double-Strata Concept”, Progress in Nuclear Energy, Vol. 37, No. 1-4, pp. 371-376, 2000.
- Shelley, H. Akie, H. Takano *et al.* “Radiotoxicity hazard of inert matrix fuels after burning minor actinides in light water reactors”, Progress in Nuclear Energy, Vol. 38, No. 3-4, pp. 439-442, 2001.
- Y. Ando, K. Nishihara, H. Takano, “Estimation of Spent Fuel Compositions from Light Water Reactors”, J. Nucl. Sci. Technol. Vol. 37, No. 10, pp. 924-933, 2000.
- T. Okubo, Y. Nakano, Y. Fukaya, S. Uchikawa (JAEA), “Advanced LWR Concept of FLWR for TRU Recycling With MOX Fuel”, ARWIF-2008 Proceedings.
- H. Takano, H. Akie, M. Handa *et al.* “A Concept of Self-Completed Fuel Cycle Based on Lead-Cooled Nitride Fuel Fast Reactors”, ICENES’93 Seventh Inter. Conf. on Emergency Nuclear Energy Systems, p. 308, 1993, Tokyo.
- T. Mukaiyama, H. Takano, T. Ogawa *et al.* “Partitioning and Transmutation Studies at JAERI Both Under OMEGA Program and High Intensity Proton Accelerator Project”, Progress in Nuclear Energy, Vol. 40, No. 3-4, pp. 403-413, 2002.
- H. Takano, T. Ikegami, “Activities on R&D of Partitioning and Transmutation in Japan”, Inter. Conf. Information Exchange Meeting for Actinide and Fission Products Partitioning and Transmutation,” CD-ROM, Oct.14-16, 2002, Jeju, Republic of Korea.
- H. Takano, H. Oigawa, T. Sasa *et al.* “Activity on R&D for P&T in JAERI”, Proc. of International Workshop on P&T and ADS Development 2003.

Mexico

- J.L. François, R. Guzmán, C. Martín del Campo: Design of a BWR Fuel with Plutonium and Minor Actinides Using Tabu Search, ANS Spring meeting, Transactions Anaheim, Ca, 8-12 June 2008.

Russian Federation

- Kotchetkov, I. Krivitskiy, N. Rabotnov, A. Tsiboulia, S. Iougai: Calculation and Experimental Studies on Minor Actinides Reactor Transmutation, Fifth OECD/NEA Information Exchange Meeting on Actinide and Fission Product Partitioning and Transmutation, SCK•CEN, Mol, Belgium, 25-27 November 1998.

United States

- T. Greifenkamp, G. Ivan Maldonado, J. Gehin, M. DeHart: Analysis of Minor Actinide Target Depletion in a Pressurized Water Reactor Lattice, ANS Spring meeting, Transactions Anaheim, Ca, 8-12 June 2008.
- R. A. Wigeland, T.H. Bauer, R.N. Hill, J.A. Stillman: Impact on Geologic Repository Usage from Limited Actinide Recycle in Pressurized Light Water Reactors, Journal of Nuclear Science and Technology, Vol. 44, No. 3, p. 415–422 (2007).
- B. Bradley, J. Ragusa: Plutonium and Minor Actinides Transmutation in Standard PWRs Fuel Assemblies, ANS Spring meeting, Transactions Anaheim, Ca, 8-12 June 2008.

- R. Sagdeev, P. Tsvetkov, W. Sadowski, G. Maldonado, A. Kritz, J. Gehin, Y. Gohar, C. Liu, J. Ragusa, W. Charlton, G. Bateman, M. DeHart: Development of a Joint US-RF Educational Program in Advanced Energy Technologies, ANS Spring meeting, Transactions Anaheim, Ca, 8-12 June 2008.
- M. K. Meyer, S.L.Hayes, W. J. Carmack, H. Tsai: THE EBR-II X501 Minor Actinide Burning Experiment, ANS Spring meeting, Transactions Anaheim, Ca, 8-12 June 2008.
- Dale Lancaster: Actinide Burning In A Standard Pressurized Water Reactor, Global-93, Proceedings: Future of Emerging Nuclear Systems, Seattle, 1993.
- K. D. Dobbin, J.V. Nelson, S.F. Kessler, R.P. Romberg, D.W. Wootan: Evaluating the Efficacy of a Minor Actinide Burner: Global-93, Proceedings: Future of Emerging Nuclear Systems, Seattle, 1993.
- R.N. Hill, T.A. Taiwo, J.A. Stillman, D.J. Graziano, D.R. Bennett, H. Trellue, M. Todosow, W.G. Halsey, A. Baxter: "Multiple Tier Fuel Cycle Studies for Waste Transmutation", Proceedings of ICONE10, 10th International Conference on Nuclear Engineering, Arlington, VA, 14-18 April 2002.
- T. Taiwo, F. Szakaly, T-K Kim, R. Hill: "Fuel Cycle Impacts of Uranium-Plutonium Co-extraction"Physor-2008, International Conference on the Physics of Reactors "Nuclear Power: A Sustainable Resource" Interlaken, Switzerland, 14-19 September, 2008.
- R.A. Wigeland, T.H. Bauer, R.N. Hill, J.A. Stillman: "Repository Impact of Limited Actinide Recycle", Proceedings of GLOBAL 2005 Tsukuba, Japan, 9-13 October 2005 Paper No. 496 R.
- E.A. Hoffman, R.N. Hill, T.A. Taiwo: "Advanced LWR Multi-Recycle Concepts", Trans. ANS, November 2005.
- E. Shwageraus, P. Hejzlar, M.S. Kazimi: "Feasibility of Multirecyling of Pu and MA in PWRs Using Combined Non-Fertile and UO_2 (CONFU) Fuel", Global 2003 New Orleans, LA, 16-20 November 2003.
- J.A. Stillman, R.N. Hill: "Effect of Thermal-Spectrum Transmuter Deep Burnup of Transuranics on Fast-Spectrum Transmuter Performance", Global 2003 New Orleans, LA, 16-20 November 2003.
- T. A. Taiwo, T. K. Kim, J. A. Stillman, R. N. Hill, M. Salvatores, P. J. Finck: "Assessment of a Heterogeneous PWR Assembly for Plutonium and Minor Actinide Recycle", Nuclear Technology Vol. 155 July 2006.
- M. Visosky, P. Hejzlar, M. Kazimi, Actinide Transmutation Using Pressurized Water Reactors, Proceedings of ICAPP '06, Reno, NV US, 4-8 June 2006.
- R.A. Wigeland, T.H. Bauer, T.H. Fanning, E.E. Morris "Repository Impact of LWR MOX and Fast Reactor Recycling Options".
- R.A. Wigeland,* Theodore H. Bauer, Thomas H. Fanning,and Edgar E. Morris: "Separations and Transmutation Criteria to Improve Utilization of a Geologic Repository", Nuclear Technology Vol. 154 April 2006.
- T.K. Kim, J.A. Stillman, T.A. Taiwo, R.N. Hill, P.J. Finck, M. Salvatores: "Assessment of Transuranics Stabilization In PWRs", PHYSOR 2002, Seoul, Republic of Korea, 7-10 October 2002.
- John A. Stillman: Proliferation Resistance Metrics for Separated Streams of Reactor-Grade Transuranics, Trans. ANS June 2005.
- E.D. Collins, G.D. DelCul, J.P. Renier, B.B. Spencer: "Preliminary Multi-cycle Transuranic Actinide Partitioning-Transmutation Studies, February 2007, ORNL/TM-2007/24".
- C.W. Forsberg, E.D. Collins, J.P. Renier, C.W. Alexander: "Can Thermal Reactor Recycle Eliminate the Need for Multiple Repositories?"

International

- Nuclear Energy Today, OECD 2003 OECD No. 53079 2003, ISBN 92-64-10328-7.
- Advanced Nuclear Fuel Cycles and Radioactive Waste Management OECD/NEA 2006 ISBN: 92-64-02485-9.
- Risks and Benefits of Nuclear Energy, © OECD 2007, NEA No. 6242, ISBN: 978-92-64-03554-6.
- Innovation in Nuclear Energy Technology, © OECD 2007, NEA No. 6103 Nuclear Energy, ISBN: 978-92-64-00662-1.
- -Management of Recyclable Fissile and Fertile Materials, © OECD 2007, NEA No. 6107, Nuclear Energy Agency, ISBN: 978-92-64-03256-9.

European Commission (ITU)

Fabrication:

- A. Fernández, D. Haas, R.J.M. Konings, J. Somers, Transmutation of actinides: Qualification of an advanced fabrication process based on the infiltration of actinide solutions, J. Am. Ceram. Soc. 85 (2002) 694.
- R.J.M. Konings, D. Haas, Fuels and targets for transmutation. C. R. Physique 3 (2002) 1013.
- Fernández, R.J.M. Konings, J. Somers, D.Haas, Fabrication of CERamic-CERamic composites pellets for the transmutation of actinides, J. Mat. Sci. Lett. 22 (2003) 119.
- Y. Croixmarie, E. Abonneau, A. Fernández, R.J.M. Konings, F. Desmoulière, L. Donnet, Fabrication of transmutation fuels and targets: the ECRIX and CAMIX-COCHIX experience. J. Nucl. Mater. 320 (2003) 11.

EFTTRA-T4

- R.J.M. Konings, R. Conrad, G. Dassel, B. Pijlgroms, J. Somers, E. Toscano, The EFTTRA-T4 experiment on americium transmutation. J. Nucl. Mater. 282 (2000) 159.
- F.C. Klaassen, K. Bakker, R.P.C. Schram, R. Klein Meulekamp, R. Conrad, J. Somers, R.J.M. Konings, Post-irradiation examination of the EFTTRA T4bis/ter irradiation on AmOx and UO_2 in MgAl2O4 spinel. J. Nucl. Mater. 319 (2003) 108.
- T. Wiss, R.J.M. Konings, C.T. Walker, H. Thiele, Microstructure characterisation of irradiated Am-containing MgAl2O4. J. Nucl. Mater. 320 (2003) 85.

General

- T. Ogawa, D. Crawford, R.J.M. Konings, S. Pillon, R.P.C. Schram, M. Verwerft, J. Wallenius, State-of-the-art Technology of Fuels for Burning Minor Actinides – An OECD/NEA Study, Proceedings of the GLOBAL 2005, No. 316, Tsukuba, Japan, 9-13 October 2005.
- D. Haas, D.J. Hamilton, Fuel cycle strategies and plutonium management in Europe, Progress in Nuclear Energy 49 (2007), pp. 574-582.

ADS

- Advanced fuel cycles for accelerator-driven systems: fuel fabrication and reprocessing The Fuel Fabrication and Processing Subgroup of the technical Working Group on ADS, May 2001.
- See also: www.nea.fr/html/trw/docs/.
- Minor Actinide Property Database (MADB) IAEA , www-nfcis.iaea.org/MADB/MADB-Main.asp.

Minor actinides Physor-2008

- V. Maslov (Joint Inst for Nuclear and Energy Research, Belarus): Minor actinides neutron data: advanced analysis, log053.
- F. Ganda, E. Greenspan (University of California, US): Plutonium and minor actinides multi-recycling in PWR using hydride fuels, log590
- Letourneau, O. Bringer (CEA, France), I. Al Mahamid (New York State Dep of Health, US), F. Chartier, E. Dupont (CEA, France), P. Mutti (ILL, France), L. Oriol, S. Panebianco, Ch. Veyssière (CEA, France): Detailed investigation of neutron emitters in the transmutation of minor actinides, log581.
- Mengoni, R. Capote, M.A. Kellett, A.L. Nichols, (IAEA, Austria) International efforts to measure, model and evaluate nuclear data for the minor actinides, log290.

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Minor Actinide Burning in Thermal Reactors

A modern light water reactor (LWR) of 1 GWe capacity will typically discharge about 20-25 tonnes of irradiated fuel (spent fuel) per year of operation. Despite the low content of about 0.1-0.2% of minor actinides in spent fuel, these actinides can nonetheless contribute significantly to decay heat loading and neutron output, as well as to the overall radiotoxic hazard of spent fuel. For this reason, there has long been an interest in transmuting minor actinides to reduce their impact on the back end of the fuel cycle.

Fast reactors are needed to effectively transmute transuranics (TRUs), including minor actinides. However, recent studies have demonstrated that TRU transmutation rates can also be achieved in thermal reactors, although with certain limitations due to the accumulation of transuranics through recycling and their impact on the safety of power plants. The transmutation of TRUs could potentially be implemented in a substantial number of thermal reactors operating today, while waiting for a similar programme in fast reactors to allow for commercial-scale operations in 20 to 30 years or more.

This publication provides an introduction to minor actinide nuclear properties and discusses some of the arguments in favour of minor actinide recycling, as well as the potential role of thermal reactors in this regard. Various technical issues and challenges are examined from the fuel cycle, operations, fuel designs, core management and safety/dynamics responses to safety and economics. The focus of this report is on the general conclusions of recent research that could be applied to thermal reactors. Further research and development needs are also considered, with summaries of findings and recommendations for the direction of future R&D efforts.

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