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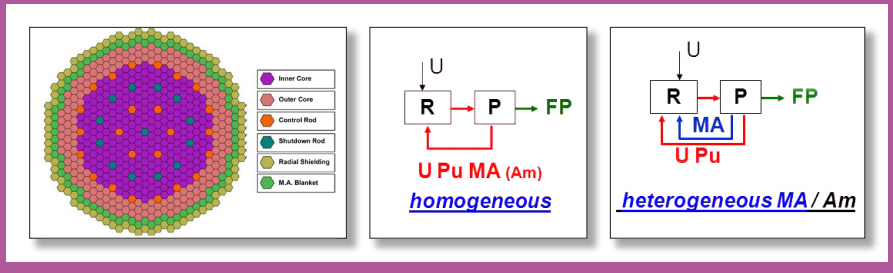
Homogeneous versus Heterogeneous Recycling of Transuranics in Fast Nuclear Reactors



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Homogeneous versus Heterogeneous Recycling of Transuranics in Fast Nuclear Reactors

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Foreword

Under the guidance of the Nuclear Science Committee (NSC) and the mandate of the Working Party on Scientific Issues of the Fuel Cycle (WPFC), the objective of the Task Force on Homogeneous versus Heterogeneous Recycling of Minor Actinides in Fast Reactors (TFHH) is to compare the different studies on spent nuclear fuel recycling methods that have been performed in several international laboratories. In particular, the comparisons will examine the criteria for choosing between homogeneous and heterogeneous recycling modes, the specific scenarios for implementation, potential non-proliferation issues and strategies for Cm management. Moreover, the study will point out the potential impact, both on the reactor core and on the power plant (i.e. lay-out, maintenance, availability, etc.).

Fuel and target related issues will be summarised with respect to potential limitations on maximum allowable minor actinide (MA) content, residence time, helium production and management, remote fabrication implications, etc. These evaluations will reflect previous and ongoing or planned irradiation programmes. Specific scenario studies will also be suggested in order to highlight specific needs and requirements, both for the short and long term. This study provides a better understanding of key issues and potential limitations, and allows the experts to make recommendations for further analytical and/or experimental demonstrations if needed.

Acknowledgements

The NEA Secretariat expresses its sincere gratitude to the members of the WPFC Expert Group on Innovative Fuels (EGIF) for providing a consistent assessment of fuel issues. The collaboration of K. Pasamehmetoglu, Chair of the EGIF, and in particular that of N. Chauvin, are gratefully acknowledged.

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Executive summary

There has been a progressive build-up of an international consensus on the preferred technologies and strategies to be deployed in order to minimise wastes, within a sustainable development of nuclear energy. There is also a wide consensus (based on the understanding of basic physics features) on the advantages of the approach based on critical fast reactors since they offer unique characteristics.

Fuel transuranics (TRU) multi-recycle is a mandatory feature if both the resource sustainability and the waste minimisation objectives are pursued. The resulting TRU transmutation can be implemented in fast neutron spectrum reactors according to two main options, i.e. the so-called homogeneous and heterogeneous modes.

In this study, the homogeneous mode implies the multi-recycle of grouped TRU fuels (i.e. with not-separated TRU) in a fast reactor. An alternative to homogeneous recycle in fast reactors could separate out the less radioactive component of the LWR spent nuclear fuel (SNF) transuranics (e.g. Pu and Np) in order to make driver fuels and use the remaining minor actinides, MA (primarily Am and Cm) in target fuels/assemblies. Consequently, the driver and target fuels can be managed separately in the fuel cycle. This separate management and recycle of the Pu (with or without Np) driver and MA target fuels constitutes the heterogeneous recycle.

The objective of the Task Force on Homogeneous versus Heterogeneous Recycle of Minor Actinides in Fast Reactors (TFHH) has been to compare the different studies that have been performed in several international laboratories. In particular, the criteria for a choice between homogeneous and heterogeneous recycle modes have been defined and compared. The specific scenarios for implementation, potential non-proliferation issues, strategies for Cm management, etc. have also been considered. Moreover, the study has shown the potential impact, both on the reactor core and on the power plant (i.e. lay-out, maintenance, availability etc). Fuel and target related issues have been summarised with respect to potential limitations on maximum allowable MA content, residence time, selium production and management, remote fabrication implications, etc.

In Chapter 2, the rationale and objectives for transmutation have been summarised and criteria for the comparison of different transmutation modes have been defined.

Reactor issues have been analysed in detail in Chapter 3. The key issues of impact on reactivity coefficients, reactor operation and fuel cycle features, have been critically described both for the homogeneous and heterogeneous recycle modes. In the latter case most of the innovative proposals made in different laboratories around the world have been considered. Even if both modes appear to be feasible from the reactor point of view and no real show-stopper has been indicated in either case, the two modes offer different advantages and different potential implementation difficulties.

As expected, fuel cycle issues play a key role. One specific result is that the footprint of the target utilisation could be much higher than what is anticipated in the fuel cycle, due to the difficulty of handling the target assemblies. Associated economics evaluations are difficult at this point and have not been approached systematically up to now. This is certainly a crucial field that should be tackled in future studies within detailed dynamic system analysis studies.

The information in Chapter 4 on fuel related issues is mostly based on preliminary results of a review report of the OECD/NEA Expert Group on Innovative Fuels (EGIF). The analysis performed within that Expert Group covers all the fuel types envisaged in homogeneous recycle (oxide, metal, nitride) and the target type fuels envisaged for heterogeneous recycle; for each fuel type, information is given on issues related to a) fuel design; b) fuel performances; and c) fuel cycle impacts.

If, from the fuel point of view, the homogeneous recycle mode appears to be a rather robust route, consolidated by recent complementary experimental programmes with respect to the pioneering SUPERFACT experiment for oxide fuels, demonstrations on the scale of a fuel pin bundle (in capsules) or on that of an assembly are still missing. The objective of such demonstrations should prove that ~5% MA loading in the fuel does not affect safety and economic performance. Today, no experiments exist that would allow researchers to assert that the presence of minor actinides in the fuel does not reduce the operational life time of the fuel. Furthermore, the limited access to fast reactors in the world will not allow such a demonstration to be done for a few more years.

Regarding the heterogeneous recycle mode, development of Am/Cm targets for a fast reactor is expected to be a complex endeavour. In this respect, the choice of a UO_2 matrix looks very promising. Moreover, the option of limiting the recycle to Am presents an obvious interest in terms of fuel development and handling, but also drawbacks, related to Cm chemical separation (at large scale) and its management in order to avoid a negative impact on the repository performances.

Moreover, in the case of the heterogeneous recycle mode, besides He production, decay heat problems become more significant than with the homogeneous recycle mode since the MA content is high. Therefore, one disadvantage for fuel fabrication is significant with respect to single sub-assembly design. On the other hand, since the number of fabricated sub-assemblies with certain decay heat is lower in the heterogeneous concept, comparison of combined disadvantages from decay heat level and the number of sub-assemblies with decay heat will give appropriate information on the relative advantage of the concept with respect to the homogeneous recycle mode, which by definition, implies the presence of MA-bearing fuel in the whole core.

In summary, the multi-criteria analysis performed in Chapter 5 shows no large discrepancies between the two modes of MA transmutation. Both of them are feasible. Most of the criteria are equivalent. Some of them indicate some advantages versus drawbacks. The final choice will depend on the importance given by decision-makers to the different criteria. More precisely, the analysis shows:

- With the homogeneous mode:
  - Fuel fabrication is easier but impacts all fuel fabrication streams.
  - Fuel behaviour under irradiation is quite close to the typical standard one.
  - No flexibility is possible between production of electricity and long-lived wastes mission.
  - Some impact on core reactivity coefficients limits content of minor actinides to about a few percents of total heavy isotopes within the fuel. Higher content will entail redefining and optimising the core design compared to the standard one.
  - A higher mass flux (number) of sub-assemblies with MA has to be managed within the fuel cycle compared to heterogeneous mode.
- With the heterogeneous mode:
  - Fuel fabrication is more complex but is concentrated on a limited mass flux.
  - Fuel behaviour under irradiation is quite different from the standard one

- Due to a large production of curium and helium released. Targets design must be validated.
- No impact on standard fuel cycle. A high degree of flexibility is given to adapting operation as a function of the different strategic policies.
- No impact on main reactor core parameters but higher thermal load of spent fuels compared to the standard one limits minor actinide content within targets and/or define new handling process at reactor stage,
- Due to lower neutron fluxes, the minor actinides inventories are quite higher than in the homogeneous mode.

As for proliferation resistance, it does not appear possible to unambiguously determine that one method of minor actinide recycle presents a greater or lesser proliferation risk than the other, or that minor actinide recycle causes a greater or lesser proliferation risk than plutonium-recycle alone, especially since one aspect of such a determination would include the host state intentions and motivations. In addition, it should be recognised that the separation technologies for obtaining plutonium from spent fuel appear to be so widely known.

1. Comparative study on homogeneous versus heterogeneous recycling of TRU in fast reactors

1.1 Background

There has been a progressive build-up of an international consensus on the preferred technologies and strategies to be deployed in order to minimise wastes, within a sustainable development of nuclear energy. A posteriori, one can say that three major approaches have been more or less consistently compared by the international community:

- use of existing thermal neutron reactors;
- use of accelerator-driven systems (ADS) or other external source-driven systems (e.g. fusion fission hybrids);
- use of critical fast reactors.

Practically, for all approaches, the multi-recycle of the TRU-loaded fuel is mandatory. A few potential alternatives to the multi-recycle have also been considered.

The use of thermal neutron reactors induces potential economic penalties [e.g. the need for a significant fuel enrichment increase to cope with the presence of minor actinides (MA) in the core] and of potential severe penalties in the fuel cycle (e.g. at fuel fabrication during TRU multi-recycle). However, a deep burn option has also been investigated, making use of HTRs.

The use of ADS is still considered (in particular in Europe and Japan) an option (to be validated) with respect to the use of burner critical fast reactors with low conversion ratio, or, as we will see, with respect to the heterogeneous recycle in fast reactors.

In contrast, there is a wide consensus (based on the understanding of basic physics features) on the advantages of the approach based on critical fast reactors. In fact, this type of reactor offers unique characteristics:

- A favorable neutron balance, which allows to introduce MA of any type and in significant amounts, without perturbing the reference performances of the corresponding core without MA.
- A neutron spectrum which allows fissions to dominate captures for all TRUs. This feature allows limiting with respect to thermal reactors the build-up of higher mass nuclei, e.g. the build-up of ^{252}Cf during TRU multi-recycle.
- The flexibility to burn or breed fuel, or to be iso-generator (a system that has a zero net production of TRU constituents in the fuel).
- The possibility to benefit from the favorable characteristics indicated above, whatever the Pu vector, the type of fuel (oxide, metal, nitride, carbide) and the type of coolant (sodium, heavy liquid metal, gas).

Fuel multi-recycle is a mandatory feature if both the sustainability and the waste minimisation objectives are pursued. Once-through, deep-burn options can only be considered temporary options in order to reduce the TRU inventories waiting for the

introduction of advanced fuel cycles based on multi-recycle. If the further recycle of the fuel leftover from the deep-burn pass is not envisaged, it could result in the disposal of still large amounts of TRUs, potentially with a high content of high mass isotopes like Cm, Bk and Cf isotopes (in particular if the deep burn has been envisaged in a thermalised neutron spectrum).

As will be described in Chapter 2, there are two main options to implement TRU transmutation in fast neutron spectrum reactors, i.e. the so-called homogeneous and heterogeneous modes. In this study, the homogeneous mode implies the multi-recycle of grouped TRU fuels (i.e. with not-separated TRUs) in a fast reactor. An alternative to the homogeneous recycle in fast reactors could separate out the less radioactive component of the LWR SNF transuranics (e.g. Pu and Np) in order to make driver fuels and use the remaining minor actinides (primarily Am and Cm) in target fuels/assemblies or stored as waste or use in homogeneous fuels in the future. Consequently, the driver and target fuels can be managed separately in the fuel cycle. This separate management and recycle of the Pu-Np driver and MA target fuels constitutes the heterogeneous recycle (see scheme below in Chapter 2).

1.2 Ongoing studies and motivation for an international comparative study

The ongoing studies, in particular in France, in the United States (initiated under the Advanced Fuel Cycle Initiative) and in Japan, motivated the launching of a comparative evaluation of the performance of homogeneous and heterogeneous recycle in a fast spectrum reactor and its associated fuel cycle.

Potential advantages and disadvantages of both recycle modes have been identified and investigated in different institutions. Most of the crucial issues are related to the fuel cycle characteristics (impact on the fuel fabrication installations, etc.) to the fuel forms (fabrication, characterisation, performance evaluation under irradiation, etc.) and to the reactor characteristics (reactivity coefficients, decay heat, etc.).

In fact, the different international studies often offer different perspectives and are based on different objectives, hypotheses and different experimental results. For example, studies and assessments on MA targets have identified a variety of challenges, and indicated that target design and implementation strategies still require development, despite the different perspectives, objectives and backgrounds of the different studies.

This report summarises the findings of a task force, set up under the guidance of the Nuclear Science Committee (NSC) and the mandate of the Working Party on Scientific Issues of the Fuel Cycle (WPFC). The objective of the Task Force on Homogeneous versus Heterogeneous Recycle of Minor Actinides in Fast Reactors (TFHH) is to compare the different studies that have been performed in several international laboratories. In particular, this new effort could provide the criteria for a choice between homogeneous and heterogeneous recycle modes, the specific scenarios for implementation, potential non-proliferation issues, strategies for Cm management, etc. Moreover, the study will indicate the potential impacts, both on the reactor core and on the power plant (i.e. lay-out, maintenance, availability, etc.). Fuel and target related issues will be summarised with respect to potential limitations on maximum allowable MA content, residence time, helium production and management, remote fabrication implications, etc. These evaluations will reflect previous and ongoing or planned irradiation programmes. Specific scenario studies could also be suggested, in order to underline specific needs and requirements, both for the short and long term.

The study should make it possible to reach a better understanding and consensus on key issues and potential limitations, and allow us to make recommendations for further analytical and/or experimental demonstrations if needed.

2. Aims for minor actinides transmutation

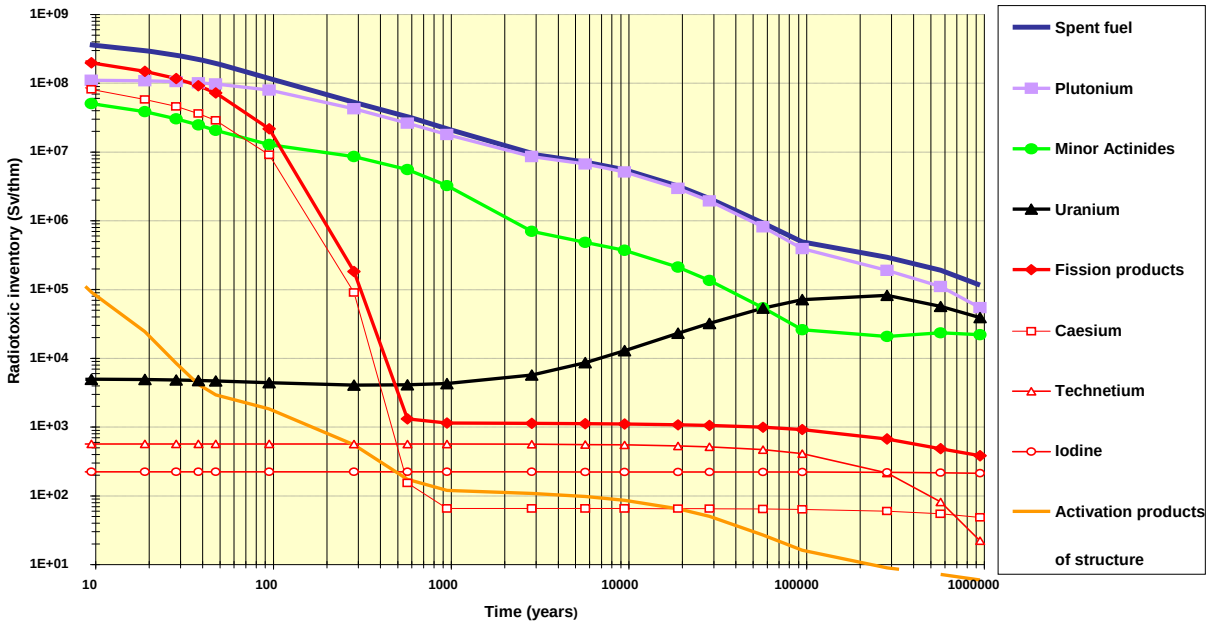
The main purpose of minor actinides transmutation is to reduce the mass and the radiotoxicity inventory of the long-lived nuclear waste. Reducing the amount of actinides to be disposed also reduces the heat load of the wastes. This reduction may have some favourable impacts on a spent nuclear fuel repository.

On the other hand, minor actinides recycling increases heat load, neutron and gamma sources at all the different steps of the fuel cycle: fabrication, reactor, and reprocessing. Consequently, a global assessment of minor actinides recycling needs to evaluate gains for long term compared to impacts on short term for the different modes of transmutation envisaged.

2.1 Mass and radiotoxicity

Spent uranium dioxide (UOX) arising from a current pressurised water reactor (PWR) contains in mass about: 95% of uranium, 4% of fission products, 1% of plutonium and 0.1% of minor actinides as neptunium, americium and curium. Development over a million years of the radiotoxic inventory of a ton of UOX spent fuel discharged at 60 GWd/t is indicated in Figure 1.

Figure 1: Radiotoxic inventory of UOX spent fuel in Sv/ton-heavy metal



The radiotoxic inventory of the fuel decreases over time as its activity drops through natural decay. The spent fuel radiotoxicity level mainly comes from:

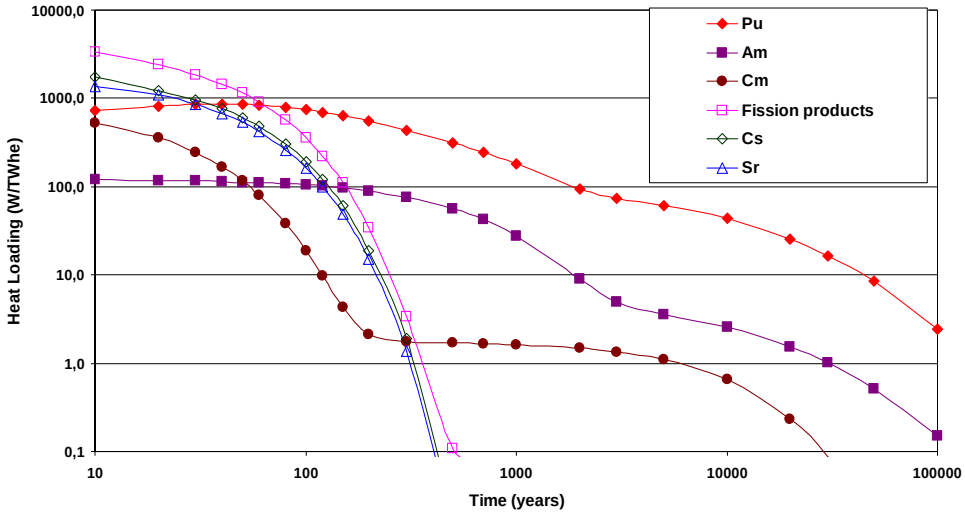
- Caesium-137, strontium-90 and plutonium for 200 years after fuel discharge from the reactor.
- Plutonium and minor actinides (mainly americium) for one hundred thousand years and more.
- Uranium, plutonium and minor actinides (mainly americium and neptunium) in the very long term.

The trend is quite similar when considering the heat load of the spent fuel as indicated in Figure 2.

2.2 Heat load

The heat load of the spent fuel is dominated in the short term by fission products (mainly caesium-137 and strontium-90). For the medium- and long-terms, the main contributors are plutonium, americium and curium.

Figure 2: Heat load of UOX spent fuel in W/TWhe in function of time



Minor actinides recycling will lead to reduce contribution of plutonium and increase americium and curium ones.

2.3 Neutron source

The main contributor of neutron source at short term until 200 years is ^{244}Cm, which has a specific neutron source of about 10^7 n/s by gram. Minor actinides recycling will have great impact on this parameter.

Recycling all actinides decreases the masses to be sent to the waste disposal. Plutonium recycling is the first step to reduce the long lived radioactive waste impacts. An enhanced reduction of toxicity, activity and heat load of the wastes can be achieved

by recycling minor actinides. Each minor actinide having its own characteristics will impact the waste properties at different time periods:

- Recycling neptunium reduces long term toxicity and activity (million years).
- Recycling americium reduces heat load, activity and toxicity at some thousands of years.
- Recycling curium reduces activity and toxicity at some tens of thousands of years.

As transmutation will not change the waste classes, high-level waste (HLW) should be managed irrespective of the options. But partitioning and transmutation would reduce the waste burden even if the reduction is limited by the toxicity release of the long-lived fission products, which will have a higher mobility in the biosphere than actinides for the major disposal sites.

2.4 Transmutation and the operation of the reactor

The introduction of elements to be transmuted in a reactor affects core performance characteristics. The transmutation of minor actinides both consumes (through capture) and generates (through fission) neutrons. Their impact depends on the type of reactor considered. In a fast spectrum, the production of higher isotopes is quite limited in comparison to a thermal spectrum. This has two main consequences:

- On the neutron balance, in a fast neutron reactor, transmutation of all of the minor actinides contributes to a surplus of neutrons whilst it consumes neutrons in a thermal neutron reactor.
- On the fuel cycle, even if the production by mass is quite small, the very high spontaneous fission yields of some higher isotopes such as ^{246}Cm and ^{248}Cm, as well as those of higher elements produced, such as berkelium and californium, represent a major drawback for fuel cycle operations.

Another aspect to be considered is actinides fission and transmutation rates achieved under irradiation. Analysis indicates the following main points:

- Under standard conditions of reactor irradiation, it is not possible to achieve complete elimination of the relevant elements. It is necessary to carry out multiple recycling.
- The fission rates obtained in a fast reactor (FR) are higher, by a factor of 2 to 6, than those obtained in a PWR.
- In a PWR, since the transmutation of the elements considered is primarily achieved by successive captures, greater numbers of higher-mass isotopes will be produced.

Adding long-lived minor actinides to a reactor core alters its neutronic behaviour and could therefore require new sub-assembly designs. There are two main different transmutation modes:

- The homogeneous one in which the radionuclides to be transmuted are mixed with the standard fuel (i.e. containing Pu and no minor actinides) and therefore impact all of the fuel cycle facilities (see Figure 3).
- The heterogeneous mode for which the radionuclides to be transmuted are separated from the standard fuel, and placed in dedicated targets (see Figure 4).

Also, different options have to be evaluated: only separated minor actinides as neptunium or americium, either alone or with curium.

Figure 3: Schematic diagram of homogeneous recycling mode

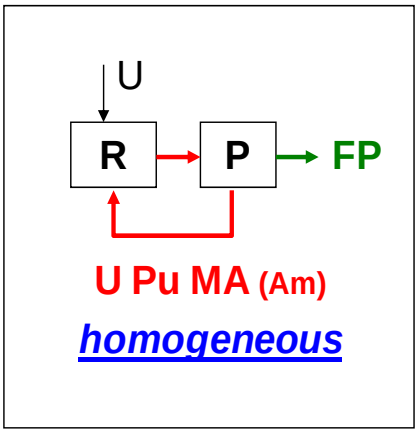
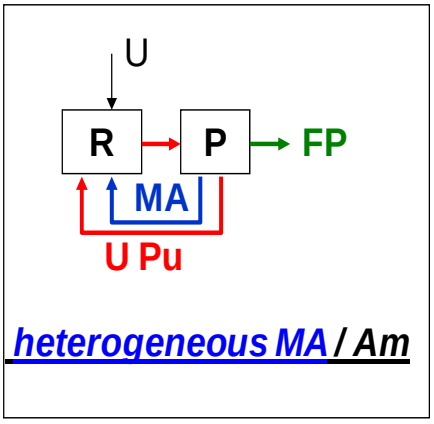


Figure 4: Schematic diagram of heterogeneous recycling mode



From the core neutronic standpoint alone, the main restriction to introducing minor actinides into critical reactors is linked to their impact on the core reactivity coefficients and kinetics parameters. Indeed, this would produce:

- a drop in the fuel temperature coefficients (doppler effect);
- an increase in the coolant void reactivity effect;
- a reduction in the delayed neutron fraction;
- a reduction in the prompt neutron lifetime;
- a reduction in the burn-up swing;
- a drop in the axial expansion coefficient;
- a drop in the radial expansion coefficient;
- a drop in the fuel density coefficient;
- a drop in the structure density coefficient.

If the actinides are recycled using a homogeneous method (distributed throughout the fuel), the initial load in the core could be limited by criteria based on the kinetics and reactivity coefficient. These restrictions limit the maximum acceptable minor actinide content in the fuel to a few percent.

If the actinides are recycled using the heterogeneous method, in the form of targets introduced into outlying regions of the core, the penalties in terms of core reactivity coefficient are reduced. The maximum acceptable actinide content in the targets depends more on considerations related to the fuel cycle.

2.5 Consequences on fuel cycle parameters

The impact on the whole fuel cycle of minor actinides recycling depends on the transmutation mode. Main variables as power decay heat, neutron and gamma sources may influence greatly the manufacturing, handling and reprocessing methods and possibilities for fuel assemblies or targets containing the elements to be transmuted.

The largest variations appear during the manufacturing phase. Indeed, after irradiation, the contribution of the heavy isotopes is masked by the predominant contribution of the fission products. Significant variations are observed, which increase as a function of actinides such as neptunium, americium and curium.

Issues are quite different when considering each actinide:

- Neptunium

Transmutation of ^{237}Np leads to the lowest impact due to its low activity (radioactive period equal to 2.2 million years). Two impacts on the fuel cycle have to be noted:

  - An increase of the gamma source, due to contribution of the daughter ^{233}Pa, which emits some hard γ.
  - A production of ^{238}Pu, with a higher activity (radioactive period of 88 years), which has quite a high neutron emission yield.

- Americium

Impact is larger mainly on gamma source at fabrication stage due to ^{241}Am, which emits some γ at 60 keV and even more with the 200 keV γ emission by ^{239}Np (period 2.4 days) produced by alpha reaction of ^{234}Am (radioactive period of 7 400 years).

Another impact is linked to the production of ^{238}Pu and ^{244}Cm on neutron source and heat load.

- Curium

Transmutation of curium is the most questionable due to its very important impact on the fabrication stage: large increase of heat load and neutron source due to ^{244}Cm and on gamma source due to ^{243}Cm.

2.6 Global impacts and performances

Assessing a long-lived waste transmutation mission can only be achieved through an overall view of the global cycle (fuel, reactors, and cycle plants) and not focusing on the performance of just one of the links in the chain. The scenario studies which consider the cycle's developments over time are essential for assessing the interest of and the possibilities to implement minor actinides partitioning and transmutation.

3. Reactor issues

3.1 Background

This introductory section provides background information on the heterogeneous and homogeneous recycle approaches, with primary focus on the former. It provides an elementary discussion of the production of plutonium and minor actinides from reactor operations, definitions of the two recycle approaches, and additionally the strategies that have been proposed for the heterogeneous recycle approach in fast reactors. The recycle approaches being considered by the United States, France, and Japan are also briefly discussed.

3.1.1 Background on plutonium and minor actinide production

The commercial water-cooled reactors (LWRs and CANDUs) utilised for electricity production use uranium-based fuels. The nuclear energy released by these reactors (and used for making electricity) is from the fission process in which the initial nuclide is split into fission products and other nuclear particles with a tremendous release of energy. Plutonium (Pu) and the minor actinides [neptunium (Np), americium (Am), curium (Cm) and higher actinides (e.g. Cf)], collectively called the transuranic elements (TRU), are formed through neutron capture by nuclides in the fuel. The capture of a neutron in ^{238}U without a fission reaction results primarily in the creation of ^{239}Pu. ^{237}Np is produced primarily from the neutron capture in ^{235}U and subsequent nuclear reaction and decay. From ^{239}Pu, neutron captures lead to the creation of the heavier isotopes. Coupled with subsequent decay, neutron interactions with key gateway isotopes (e.g. ^{241}Pu, ^{242}Pu, ^{243}Am) result in the formation of higher actinides (i.e. Am+Cm+Bk+Cf). Hence, removing uranium from the fuel is one strategy that has been proposed for minimising the formation of new transuranic isotopes by avoiding ^{239}Pu creation.

^{241}Pu is formed primarily by successive neutron capture in ^{239}Pu. Long post-irradiation cooling times, typical of much of the LWR commercial industry, would allow ^{241}Pu to decay into ^{241}Am, allowing for a gateway into the higher actinides, particularly ^{242}Cm, which subsequently decays to ^{238}Pu. The decay of ^{241}Am results in the production of ^{237}Np. Further neutron absorptions, not leading to fission, transmute ^{241}Pu into the non-fissile ^{242}Pu, which, in turn, is the gateway into ^{243}Am and ultimately ^{244}Cm.

Compared to thermal reactors, in a fast reactor the relative creation of the higher actinides is reduced because the probability of creation of the gateway isotopes is reduced as a result of the higher fission to absorption ratio. In the plutonium breeding mode in fast reactors, the relative content of the higher actinides in the TRU material is even lower. The probability of capture per neutron absorption is high for the gateway nuclides in a thermal reactor spectrum like in light water reactors. So, if this system is used for plutonium recycling, relatively more minor actinides are produced compared to fast reactors.

With regards to solutions proposed for minimising or consuming the production of these nuclides, it is noted that the presence of a moderating material in the reactor decreases the neutron energies, typically increasing the neutron reaction rates. This tends to increase the transmutation rate (including the fission rate) of the initially loaded nuclides. It is noted that transmutation might take place through the destruction of the TRU by the fission process or the creation of a higher or lower mass nuclide by one of the

paths discussed above. It has been observed that neutron moderation can greatly increase the transmutation rate of americium isotopes, in a relatively short time (e.g. greater than 95% of initial mass reduced in less than a decade). Most of this reduction takes place through the capture process. However, the enhanced americium transmutation comes at the expense of an increase in the ^{244}Cm generation rate and a potentially power peaking problem in a fast reactor (can be alleviated by design).

Neutron capture in ^{243}Am leads to the production of ^{244}Cm. The ^{244}Cm builds up until the rate of transmutation from its creation from ^{243}Am is matched by the rate of ^{244}Cm conversion into ^{245}Cm. The ^{244}Cm decays into ^{240}Pu, which is more long-lived. Subsequent neutron captures in ^{244}Cm and decays ultimately lead to generation of ^{252}Cf. Both ^{244}Cm and ^{252}Cf have a primary decay mode by alpha emission. However, their secondary decay mode is by spontaneous fission. ^{244}Cm has a spontaneous fission yield of 1.3×10^{-4}%, whereas ^{252}Cf is much higher at 3.09%. However, the ^{244}Cm concentration is much higher than that of ^{252}Cf in a discharged moderated target. Due to the short 2.6 years half-life of ^{252}Cf, only a few years of post-reactor decay time, such as in a cooling pond, is needed to allow the ^{252}Cf decay away in order to significantly reduce the spontaneous fission neutron emissions in used and recycled fuel.

These transuranic elements (plutonium and minor actinides) constitute the major contributors of radioactivity (radiation and heating) in used nuclear fuels after a few decades. Consequently, they have to be buried in a secure disposal site, or utilised or managed in the fuel cycle to minimise their potential environmental effects. The possible utilisation and management paths for these transuranics using potential recycle approaches in fast reactor are the focus of this study.

Reactor systems, including fast and thermal reactors, have been considered for the transmutation (particularly burning) of TRU. This necessitates the recovery of the pertinent TRU from used nuclear fuel by reprocessing and separation processes and the fabrication of the material into fuel forms that could be used in the reactor system. Alternatively, the material or portion of it could be stored for future utilisation or burial in a repository. The focus here is on transmutation in fast reactors. Compared to thermal reactor systems, fast reactor systems are considered more efficient for TRU management because of their higher fission to absorption rate ratios which favours fission (burning), higher flux level, and higher surplus neutron which provides flexibility for the transmutation of the minor actinides (MA), e.g. using leakage neutrons in core periphery for transmutation, and that also allows a very wide range of different TRU compositions, e.g. different MA/Pu ratios, to be transmuted.

3.1.2 Rationale for homogeneous or heterogeneous recycling

The transuranic elements are major contributors to the radiotoxicity and decay heating of used nuclear fuel. The bulk of the TRU in light water reactor used nuclear fuel (UNF) is plutonium (Pu), which constitutes about 85-90% of TRU, with the balance being mainly neptunium (Np), americium (Am) and curium (Cm). The plutonium isotopes are the leading contributors to radiological hazard to the geological repository. The ^{237}Np (and its highly radioactive daughters) is a long-term hazard in the repository because of its long half-life (over 2 million years), and the curium and americium are short-to intermediate-term hazards, though they decay into long-lived hazardous nuclides in the repository. When a solution is found to the hazard posed by plutonium, it will become necessary to consider approaches for hazards posed by the other minor actinides for the reduction of long-term hazards.

Approaches have been considered for reducing the Pu and MA impacts in a geological repository as a means of reducing the space required for their disposal for a given unit of energy production. Transmutation or burning of the material has been considered a potential approach for this purpose.

The primary utilisation of plutonium in fast reactors has centered on its use for power generation and in particular to support the breeding of additional fissile materials (^{239}Pu or ^{233}U), in order to provide an effective use of natural resources. Waste minimisation is also an envisaged mission for fast reactors, according to the sustainability criteria of advanced nuclear systems, e.g. Generation IV systems. For these purposes, the multi-recycle of all the transuranics TRU has been considered. This is the so-called *homogeneous recycle* approach which allows reduction or stabilisation of the MA inventory in the fuel cycle. If the primary intent is MA burning (i.e. a drastic reduction of an existing MA stockpile from irradiation of uranium-based fuels in commercial LWRs), or if it is found necessary to separate MA completely or partially from Pu in the standard reactor fuel, it is pertinent to explore the possibility to concentrate MA in special pins (targets) in dedicated sub-assemblies at specific positions in the core. This is the so-called *heterogeneous recycle* approach. The schematics for the homogeneous and heterogeneous recycle approaches are presented in Figure 5. The potential reasons why the heterogeneous recycle approach could be used are summarised in Table 1.

Previous or existing fast reactor systems have traditionally utilised fuel not initially containing the minor actinides. The addition of minor actinides would make fuel handling more difficult, due to the intense radiation emissions and heat from them. The core safety characteristics might also be modified and the potential degradation should be examined in principle or eliminated by design. For this reason, it has been proposed that the minor actinides should be confined to a fraction of the existing fuel cycle. The expectation is that with the concentration of these materials, only a small fraction of the nuclear infrastructure will be needed for their handling and utilisation.

Figure 5: Schematics for homogeneous and heterogeneous recycle approaches

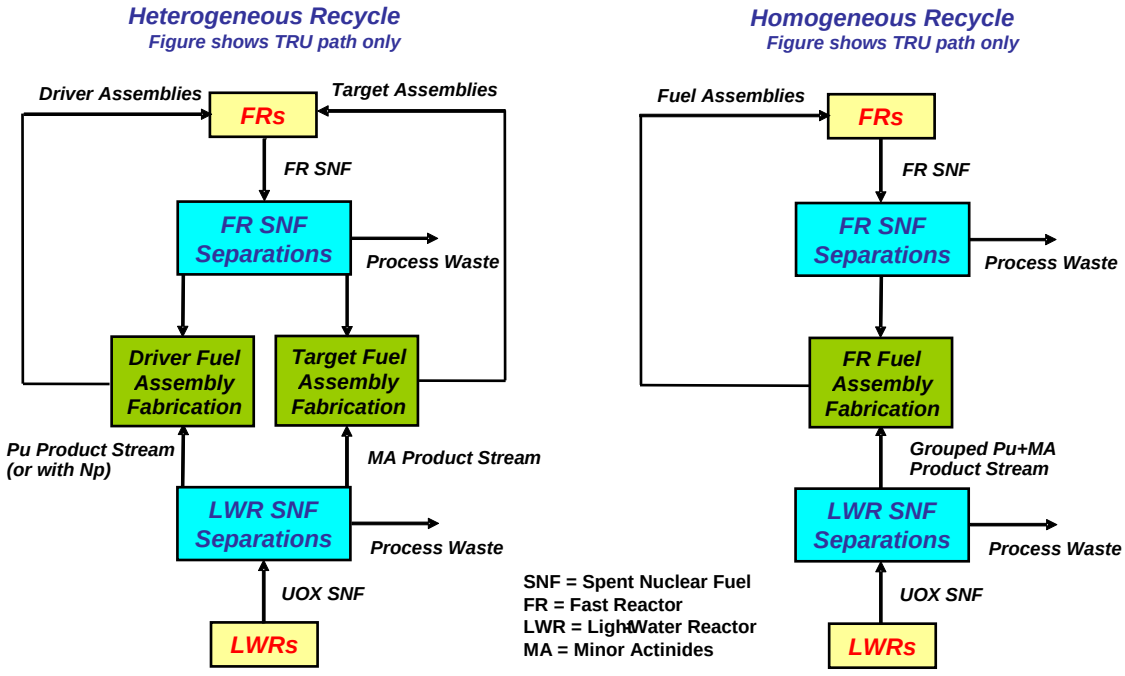


Table 1: Reasons for using the heterogeneous recycle approach in fast reactor fuel cycles

Reason	Comment
Existing fuel recycle infrastructure (in particular fuel fabrication) could be affected by higher doses from minor actinide utilisation, with potential economic consequences.	Not an issue in some countries such as the United States (no facilities yet), but could be in Europe and Japan, which have existing MOX fuel facilities. Incentive to minimise MA handling equipment and assemblies containing MA in the fuel cycle.
Rapid initiation of transmutation campaign with fast reactors as part of an advanced fuel cycle strategy or in order to preserve the economic competitiveness of fast reactors and optimise the transmutation in targets.	Fast reactors can be used for material burning and fissile material production to support nuclear sustainability. Intent is to delay utilisation of MA in initial fuel to allow additional R&D, qualification and regulatory acceptance of innovative transmutation fuels. Very active fuel to be handled. No large experimental database for innovative fuel and its behaviour under irradiation.

3.1.3 Heterogeneous recycling strategies

In recent times the recycle of minor actinides in fast reactors has been associated with their transmutation. Transmutation in nuclear reactors is primarily by the fission process or the capture process. Fission results in the production of fission products that generally tend to be short-lived (there are some exceptions), and as such is regarded as burning of the material into stable and less radiotoxic nuclides. Conversely, the capture process typically tends to result in the creation of other TRU, though subsequent reactions of the daughter nuclide could lead to fission of TRU. As noted above, fast reactors are efficient burners. Their relatively high surplus neutrons also allow concepts in which the fast reactor core zone could be moderated in order to enhance the neutron cross-section for transmutation.

Based on the considerations above, heterogeneous recycle strategies in fast reactors have been suggested for *burning* and/or *stabilisation* (management) of MA derived from LWR used nuclear fuels. The MA burning strategy could be used when MA recycle has been delayed relative to Pu recycle in order to get the MA content from an above steady-state level to the steady-state level that would exist in a stabilised system. In the strategy, it has been envisaged to perform the complete burning of the target material in a single-recycle through the reactor system. However, this deep burning of the target fuel in a single-recycle has been found to necessitate the use of advanced fuels and materials that can withstand the high neutron doses (displacement per atom levels >200 dpa), and harsh temperature and irradiation environment in a reactor for a much longer period of time than required for typical fast reactor assemblies. Furthermore, the target fuel composition, towards the end of irradiation, would be dominated by fission products and the fuels behaviour is largely unknown.

The stabilisation strategy alleviates some of the problems associated with the deep-burn, single-recycle strategy. In this case, the target material could be multi-recycled through the fast reactor core on the same or slightly different time scale as typical fast reactor assemblies. This would, however, require the reprocessing of the used target fuel material. In this stabilisation strategy, however, traditional fuels such as uranium oxide can be used.

Not having to recycle is a potential reason why the single-recycle strategy has been considered attractive. However, unless the material is nearly 100% consumed (including

the daughter products, which could be highly radioactive), then a solution must be found for the management of the used target fuel.

3.1.4 Recycling approaches by countries

The advanced nuclear fuel cycle programme of the United States has not yet made a selection between the homogeneous and heterogeneous recycle approaches. Nevertheless, most of the fast reactor studies under the advanced fuel cycle programmes have been performed for the homogeneous recycle approach. This has been predicated on the need to reduce the proliferation risk associated with the use of transuranics in the fuel cycle. This intent has been supported by the use of the intrinsic barrier that exists from the use of the highly radioactive TRU-containing fuel. The use of Pu-only fuel, particularly if the plutonium is separated on its own, has not been supported by the programme. This is an evolution from the USDOE Integral Fast Reactor (IFR) Programme, in which all the transuranic elements are group separated by the pyrochemical process to minimise the proliferation risk. It should be noted that in this connection, the Generation IV International Forum (GIF) has supported Generation IV fast reactor systems using TRU-containing fuels for the purpose of TRU management and nuclear system sustainability. Very limited evaluation of the heterogeneous recycle approach has been conducted under the US advanced fuel cycle programme.

The French CEA has considered the heterogeneous recycle approach since the day of its CAPRA/CADRA programmes. It is one of the approaches that have been evaluated, in addition to the homogeneous recycle one. The primary motivation for the heterogeneous recycle approach is the need to leave unperturbed the existing facilities that have been developed for the LWR MOX fuel cycle, with the intention of using the same facilities for the fabrication of MOX fuels for the fast reactor. In the current French recycle path, the minor actinides are recovered with the fission products and other nuclear wastes during reprocessing with the intent of final burial in a repository. Should the need to recover the minor actinides be ascertained in ongoing technical studies, then in the short-term at least, the CEA has considered an option that extracts the minor actinides in the separations steps and fabricates minor-actinide-containing target assemblies in a sub-facility of the MELOX plant. These target fuels will then be utilised in advanced fast reactors which are planned to be deployed around 2040 (the ASTRID prototype is planned to be started in 2020). Significant efforts have been devoted to the evaluation of fast reactor target fuels in the CEA programmes. Within the context of the French law on long-lived wastes management, a statement on minor actinides recycling option and the selection of the mode of transmutation between homogeneous and heterogeneous will be made in 2012.

The current position of Japan on homogeneous versus heterogeneous recycle approach is well formulated. The Japan Atomic Energy Agency (JAEA) currently considers the homogeneous recycle approach using MOX fuel in fast reactors as the reference one. The JAEA assumptions and calculations show that in the homogeneous recycle fast reactor fuel, the minor actinide content would be less than 5% under normal assumptions, with this fraction decreasing to less than 1% as the continuous recycling of the fuel reaches an equilibrium (see Section 2.4). Research in the heterogeneous recycle approach is, however, being pursued as an alternative to the homogeneous recycle approach, as a technical risk mitigation approach should problems arise with the homogeneous recycle approach.

3.2 Fast reactor transmutation issues associated with homogeneous recycling

This section provides an overview of the reactor types that have been considered for homogeneous recycle and an assessment of the impacts of minor actinide heavy metal content on the homogeneous recycle reactor safety. It also includes information comparing the homogeneous and heterogeneous recycle approaches as they affect core

performance and transmutation, and the sensitivity of those results to system parameters such as the conversion ratio, neutron moderation, cycle length, target location, target loading, etc. The implications of the fuel forms and types that have been considered for homogenous and heterogeneous recycle reactors will be discussed in Chapter 4. Issues associated with reactor fuel performance under high helium production in the target fuel and fuel handling difficulties arising from the highly radioactive target fuel will also be presented in Chapter 4. Discussions of some overall systems considerations are provided, including evaluations of the impact of the number of target assemblies in the core and the time it takes to stabilise MA following separation in the fuel cycle.

3.2.1 Fast reactor types for homogeneous recycling

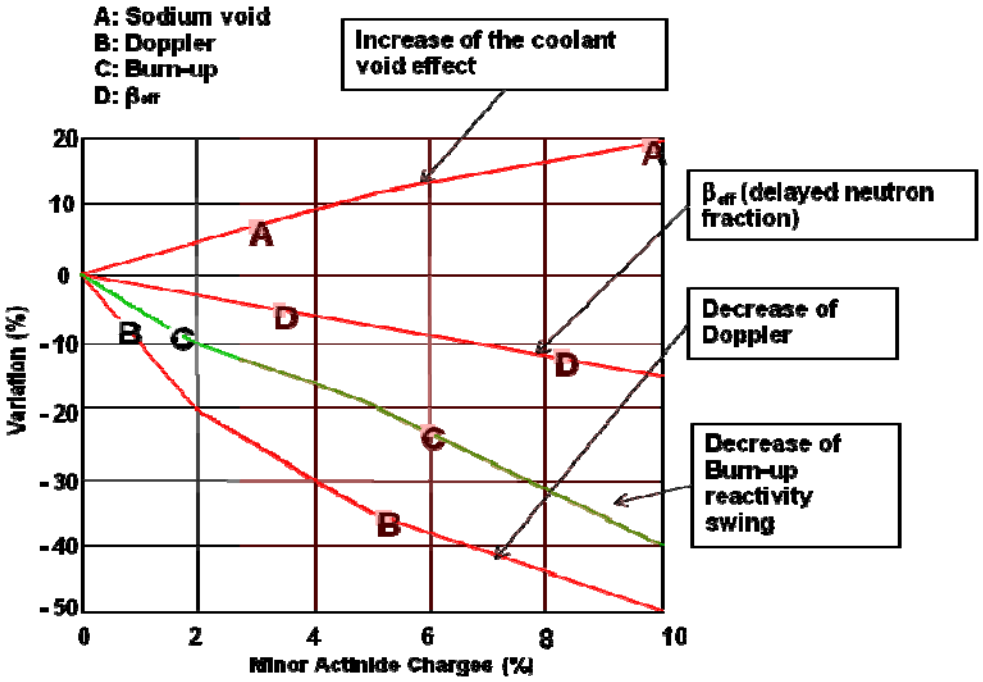
The most extensive studies on homogeneous recycle of MA in fast reactors have been done with designs using sodium as the coolant. Other studies considering Generation IV gas-cooled and lead-cooled fast reactor systems have, however, been performed [19, 29]. As part of the current OECD/NEA effort on the study of homogeneous versus heterogeneous recycle in fast reactors, ENEA has also evaluated the use of the European Lead-cooled Reactor System (ELSY) for the transmutation of transuranic elements (see Section 3.4.6). These studies have complemented those that have considered thermal reactors as the transmutation vehicle, which is not the subject of this report. From a transmutation point of view, the behaviour of different fast reactor systems is expected to be fairly similar.

There are, however, differences in core physics and safety considerations that might affect how a fast reactor core is configured and the maximum loading of MA in the core. For example, with gas-coolant the impact of coolant voiding would not be as high as in a sodium-cooled system. Consequently, if the philosophy for core safety is to minimise the void worth to Doppler coefficient ratio, then the gas-cooled system might allow a higher loading of the minor actinides in the core. Additionally, from a system cost point of view, the larger sizes of the gas-cooled and lead-cooled systems (necessitated by design) compared to a sodium-cooled system, could provide advantages to the sodium-cooled system. The maturity of the sodium-cooled system compared to that of the other ones could make it attractive for any transmutation mission in the short-term.

3.2.2 Assessment of variations in reactivity feedback coefficients for homogeneous recycling in fast reactors

Some previous papers have indicated the changes in reactivity coefficients and kinetics parameters when legacy minor actinides are added to pre-existing qualified fast reactor fuel form (e.g. when transitioning from Pu-U based fuel to TRU-U based fuel). The French CEA has done evaluations to investigate the variations in reactivity feedback coefficients [10]. The results of that study are summarised in Figure 6 [10]. In the study for a large 1 500-MWe Na-cooled fast reactor, the MA content in the homogeneous recycle fuel was increased from 0 to 10%. It is observed that the coolant void reactivity effect increases by up to 20%, the Doppler constant decreases by up to 50%, the delay neutron fraction (beta effective) decreases by up to ~15%, and the burn-up reactivity swing by up to 40%.

Figure 6: Variations in reactivity and safety parameters as function of minor actinide content in homogenous recycle fuel



In the CEA safety considerations, the ratio of the coolant void effect to the Doppler constant is a limiting parameter with the intent to minimise its value.

As shown in Figure 6, this ratio will decrease with the increase of the MA content. This undesired trend has been the basis for the CEA to suggest limits for the minor actinides content in very large size fast reactor cores. In fact, smaller cores with higher leakage or other specific core features (e.g. the presence of an upper Na plenum, etc.) would modify the picture. Currently, the CEA has suggested that the minor actinides (MA) content in the heavy metal could be limited to 2.5 to 3% for a large-size sodium-cooled fast reactor (e.g. the European fast reactor design), 5% in a medium to small size fast reactor (e.g. Phenix design), and 5% in gas-cooled fast reactor fuel. These are very preliminary indications that should be confirmed with further studies and analysis. The higher limit for the gas-cooled fast reactor, relative to the sodium-cooled fast reactor system, is envisaged if the lower coolant void effect in the system plays an important role in the safety case. However, a more detailed safety analysis should be performed to determine the limit of the MA content in the fuel.

A recent study has reviewed the issue of whether there is a limit for MA in homogeneous fast reactor fuels or not [24]. Two sodium-cooled fast reactors (FRs) with different core size and MA content were analysed in that study: (1) a burner sodium-cooled fast reactor (SFR) with minor actinide to plutonium (MA/Pu) ratio of ~0.1 and conversion ratio (CR) of ~0.25 and (2) a standard large size European fast reactor (EFR) with a limited content of MA in the fuel, equivalent to the equilibrium value. The design characteristics are summarised in Table 2.

Table 2: SFR and EFR characteristics

	SFR	EFR
Power	840 MWth	3 600 MWth
Fuel type	U-TRU-Zr metal fuel	U-TRU oxide fuel
Outer core	Stainless still reflector	U blanket
TRU content	56%	22%
MA content	10.8%	1.22%
Doppler reactivity	850-300K	1 520-300K
Na void effect	Only core	Core and blankets

Figure 7 and Figure 8 show the impact of variable MA content, calculated using direct relative composition perturbations, on the sodium void reactivity, Doppler reactivity coefficients, and effective delayed neutron fraction (β_{eff}), for both the EFR and SFR systems.

Figure 7: SFR: Sodium void and Doppler reactivity coefficients and β_{eff} at various MA%

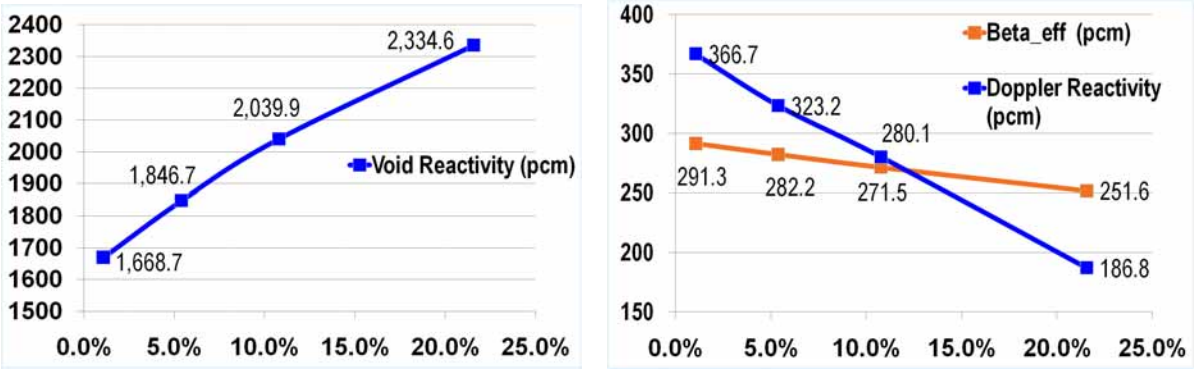
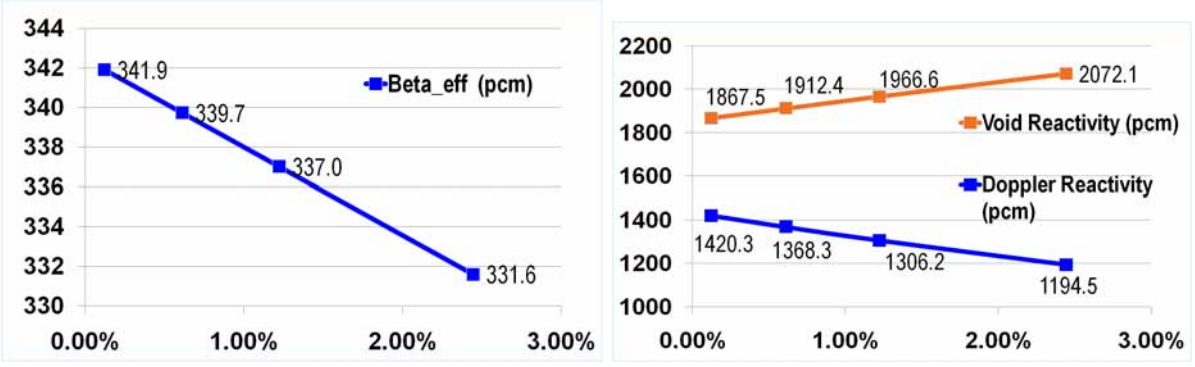


Figure 8: EFR: Sodium void and Doppler reactivity coefficients and β_{eff} at various MA%



These results show that doubling the amount of MA in the EFR (i.e. from 1.2 to 2.4%); the void reactivity will only increase by roughly 100 pcm. In the most extreme case for the SFR, doubling the MA content from 10.8 to 21.6% corresponds to an increase of the positive sodium void effect by ~300 pcm. This value is significant, but still relatively modest. In any case, the impact of MA increase on reactivity coefficients is very much related to the approach taken in core and fuel design, and it can be optimised according to pre-defined objectives and constraints.

The sensitivity analysis performed in that study allows us to understand the observed trends. The analysis shows that the global effect of each isotope can be interpreted as the combination of competing effects:

The increase of capture at high energy (e.g. E>100 keV) results in a flattening of the high energy slope of the adjoint flux (responsible for the positive reactivity Na void effect). By contrast, the capture increase at low energy (e.g. E<100 keV) results in a flattening of the low energy slope of the adjoint flux (responsible for a negative contribution to the same coefficient). The sum of the two effects results in a decrease in the Na reactivity coefficient, due to the predominance of low energy effects.

The opposite effects are observed for the high energy range and the increase at high energy of the fission cross-section will result in an increase in the energy gradient of the adjoint flux at high energy (and then of the associated positive reactivity effect), that is counterbalanced, in the case of fissile isotopes, by a corresponding effect of opposite sign at low energy. This means that "fertile" isotopes with threshold fission cross-sections will induce only the increase in the positive reactivity effect at high energy, while the predominance of low energy effects for fissile isotopes give a global decrease of the positive reactivity.

In summary, the sensitivity analysis shows that any increase of the amount of "fertile" isotopes results in an increase of the positive Na-void reactivity coefficient. Inversely, any increase of fissile isotopes results in a decrease of the Na void reactivity. This feature clearly explains the effects of MA isotopes increase, i.e. positive for ^{241}Am, ^{243}Am, ^{237}Np, and ^{244}Cm and negative for ^{242m}Am and ^{245}Cm.

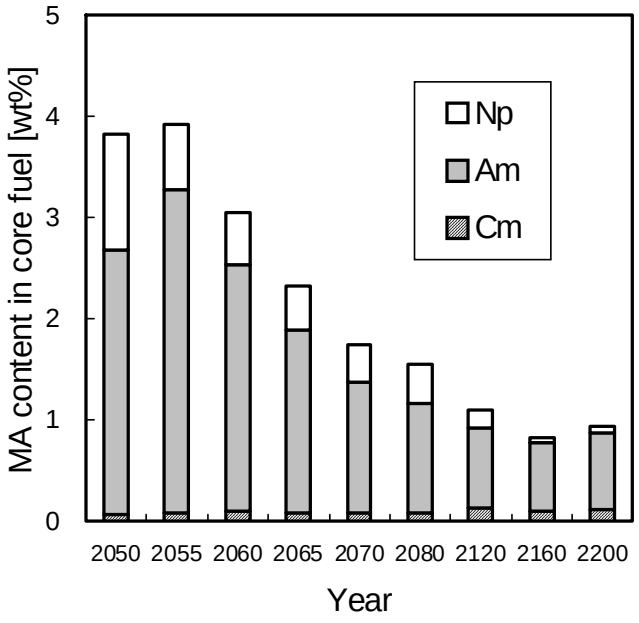
The paper concluded that it is not possible to define rather theoretical "upper limits" for the MA content in the fuel of a fast reactor that would recycle TRU in a homogeneous mode. It noted, however, that a careful physics analysis of the reactivity coefficients (e.g. void reactivity and Doppler reactivity effects) performed for each system can suggest appropriate design measures that would limit the potential safety impact of the MA content increase in the core. This point is further illustrated by the results of the JAEA approach, see Section 3.2.4.

3.2.3 Minor actinide content evolution for homogeneous recycling in fast reactors

Before discussing results obtained from a more systematic evaluation of core safety, it is important to present the types of minor actinide contents that have been projected for fast reactor fuel as a result of continuous homogeneous recycle of fuel material in a SFR. The results of a JAEA study evaluating this trend during a transition from a LWR fuel cycle to a SFR fuel cycle is provided in Figure 9 [23]. It is observed that the minor actinide content varies from 1 to 4 wt%. This is the natural evolution of the minor actinide content and is possible because of the breeding core design and the high fission-to-absorption ratio in fast reactors. It is possible that the actual content could be higher if the fuel cycle deployment is first initiated with fast reactors using fuels containing only plutonium with the minor actinides allowed to accumulate. There might subsequently be the need to initially load minor actinides in quantities higher than their natural equilibrium content with recycle in fast reactors. This is one advantage of the partitioning of the Pu and MA as it allows the option to utilise and manage the two materials in any favourable ratio, even in the homogeneous recycle approach. Such flexibility could also be used to alleviate any potential issues with radiation dose during

fabrication by initially using a minor actinide content lower than that from UNF TRU and to reduce the MA content of the homogeneous recycle fuel towards the equilibrium value in recycle stages.

Figure 9: Variation in minor actinide content in homogeneous recycle fuel resulting from fast reactor deployment in 2050, JAEA study



3.2.4 JAEA approach and results of JSFR related studies

JAEA has been investigating the JAEA sodium-cooled fast reactor (JSFR) core with minor actinide bearing oxide fuel under the homogeneous recycling concept as reference for future fuel recycle in Japan: the FaCT (Fast Reactor Cycle Technology Development Project) programme. The approach includes minor actinide recycle from LWR spent fuel as well as equilibrium fast reactor recycle.

Table 3 shows the characteristics of the FaCT reference core design [14]. The design parameters have been selected to be representative of future commercial fast reactor systems. The core outlet temperature is 550°C, which leads to 700°C of cladding maximum mid-wall temperature with design engineering uncertainties. The core fuel average burn-up is 150 GWd/t, which corresponds to 230-250 GWd/t peak burn-up and 250 dpa fast neutron dose. To achieve high cladding temperature and high burn-up with high neutron dose ODS ferritic steel with excellent high temperature strength and irradiation swelling resistance has been proposed as cladding material. Figure 10 shows the FAIDUS (fuel sub-assembly with inner duct structure) concept of the JAEA reference core concept, which is based on core safety considerations in the case of core disruptive accidents [17]. Typical oxide fuel core configurations are indicated in Figure 11 [14] and Figure 12 [23]. These concepts are examples of the so-called homogeneous radial two-region core.

Table 3: Design conditions of FaCT reference core

Design constraints and assumptions	
Plant conditions	
Power output [MWe/MWth]	750 / 1765
Coolent outlet/inlet temp. [°C]	550 / 395
Core performance targets	
Breeding ratio	1.1
Discharge burn-up [GWd/t]	150 / >80
core/core + blanket	
Operation cycle length [month]	>18
Safety requirements	
Sodium void reactivity [USD]	<6
Core height [cm]	<100
Average core specific	>40
heat [kW/kg-MOX]	
Sub-assembly (SA) concept	FAIDUS
Fuel specifications	
Fuel composition	Multi-recycled FBR TRU with
Fuel smear density [%TD]	residual FPs
Cladding material	82
	ODS steel
Thermal hydraulics condision	
Max. cladding temp. [°C]	700
Bundle pressure drop [MPa]	<0.2
Fuel integrity limits	
Max. linear hear rate [W/cm]	430
Max. fast neutron fluence [n/cm^2]	5×10^{23}

Figure 10: FAIDUS concept of JSFR

This SA has inner duct installed at a corner and a part of upper shielding element is removed.

At CDA (core disruptive accident), molten fuel enters the inner duct channel and goes out into the outside through the upper shielding.

This SA has superior performance for discharge of molten fuel to prevent compaction of it.

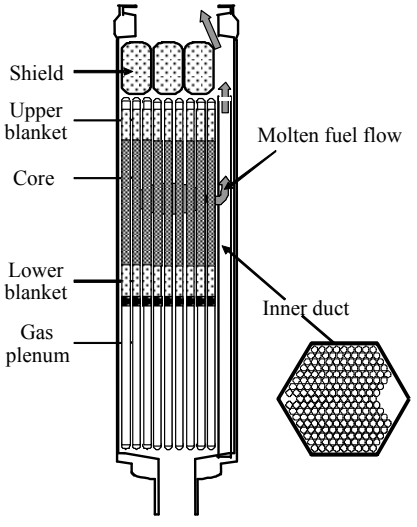
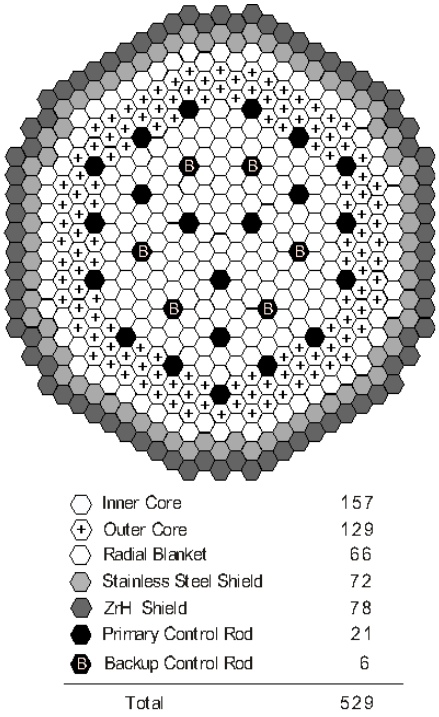
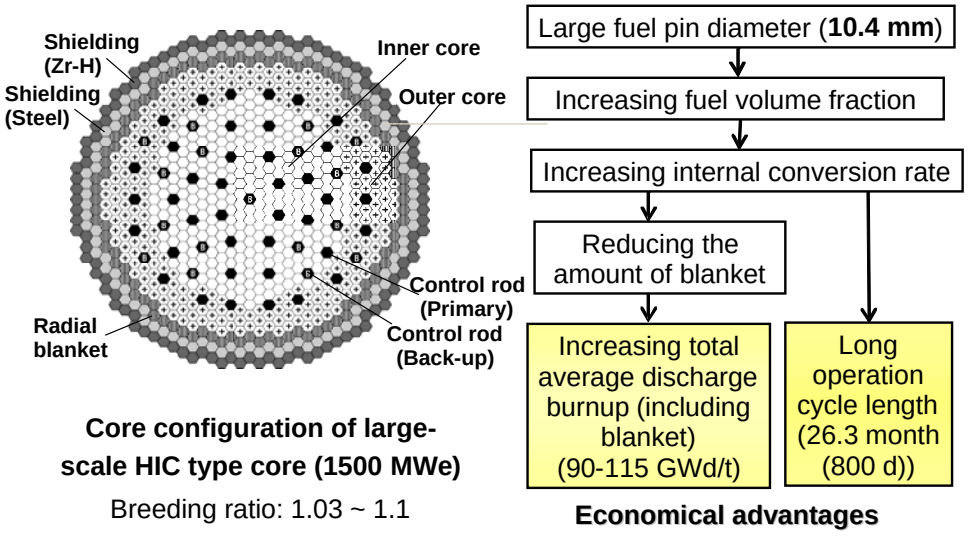


Figure 11: JSFR 750 MWe core configuration



Source: Reference [14].

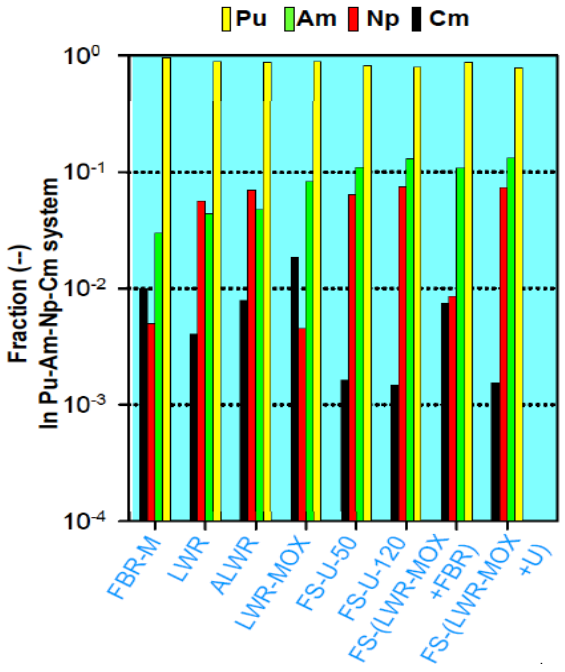
Figure 12: JSFR 1 500 MWe core configuration



Various TRU compositions have been applied to the core design studies. Most previous studies have concentrated on the equilibrium core with continuous loading of the same composition of driver fuels. This means that the TRU compositions are

considered to be core average conditions. Examples of TRU compositions are illustrated in Table 4, in Figure 13 [21] and Figure 14 [18]. The right four cases of Figure 13 are model cases from a Japanese domestic fast reactor deployment scenario analysis, which simulate the mass balance to supply TRU from LWR spent fuel to fast reactors in the transition phase from LWR use to fully fast reactor use. Some of them have around 20-25% minor actinide contents in TRU, which corresponds to 5% minor actinide content in heavy metal in the fuel with 20% of plutonium enrichment, a typical enrichment of JSFR core. This means that about 5% of minor actinide content could be a typical maximum value of a Japanese domestic scenario of JSFR deployment. TRU from LWR spent fuel contains more minor actinides than the TRU in fast reactor equilibrium cycle. Other TRU compositions considered include fast reactor recycle fuel starting with TRU from LWR spent fuel and (U, Pu) without intended americium and curium. Examples of fast reactor recycle fuel composition with minor actinides are indicated in Figure 14. As regards (U, Pu) without intended americium and curium, such fuel is expected in the early stage of fast reactor deployment. In the early stage of the deployment, fast reactor cores may start with (U, Pu) fuel which contains unintended americium from the decay of ^{241}Pu.

Figure 13: TRU compositions considered in core design studies (2005)

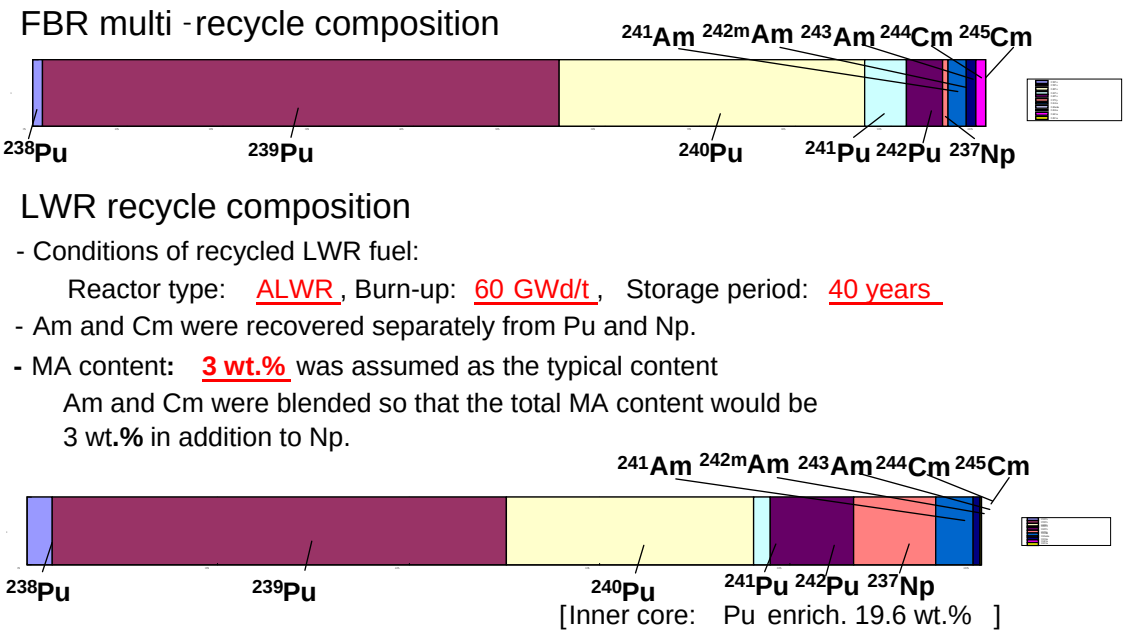


Source: Reference [21].

Table 4: TRU compositions considered in FR core design studies (2005)

FBR-M	Multi-recycle of FBR
LWR	Spent LWR fuel after 4 years cooling (45-49 GWd/t)
ALWR	Spent ALWR fuel after 4 years cooling (60 GWd/t)
LWR-MOX	Spent LWR-MOX fuel after 5 years cooling (45-49 GWd/t)
FS-U-50	Spent LWR (30 yrs cooling) and ALWR fuel (20 yrs cooling) = 9:1
FS-U-120	Spent LWR (70 yrs cooling) and ALWR fuel (40 yrs cooling) = 3:7
FS-(LWR-MOX+FBR)	Spent LWR-MOX (40 yrs cooling) and FBR fuel = 5:5
FS-(LWR-MOX+U)	Spent LWR-MOX (110 yrs cooling), LWR (90 yrs cooling) and ALWR (40 yrs cooling) fuel = 0.5:1.9:7.6

Figure 14: TRU compositions considered in core design studies (2007)



Source: Reference [14].

Table 5 shows the characteristics of a 1 500-MWe core with fast reactor equilibrium TRU composition [14]. Reactor core operation cycle length is 26 months with four refuelling batches, thus fuel lifetime is about nine years in this design. The burn-up reactivity swing is below 3% $\Delta\rho$ for 26-month operation. The sodium void reactivity worth is below six dollars, which is the tentative limiting value set from core safety considerations. This design achieves break-even breeding without radial blanket. Table 6 shows the core characteristics with TRU composition in Figure 9 in comparison with those of fast reactor equilibrium TRU composition [18]. The minor actinide content of the

LWR recycle composition is 3% in heavy metal and the core characteristics satisfy the design conditions. The cores with both compositions achieve a breeding ratio of ~1.10; the LWR recycle composition with 3% MA requires fewer of axial blankets, since higher MA content gives higher core internal conversion.

Fuel composition changes with fuel recycle have also been investigated. The TRU composition changes with fuel recycle in a fast reactor are indicated in Figure 15 [15].Core characteristics of fast reactor recycle fuel compositions were evaluated assuming that the equilibrium core is established with certain recycle fuel compositions. Among the evaluated core characteristics, sodium void reactivity, burn-up reactivity swing and the Doppler coefficient are indicated in Figures 16 and 17 [14]. Since plutonium isotopic composition and MA content changed during recycle, the data shown in the Figure became scattered due to the changes of the core characteristics. They show good correlations. Based on this analysis, possible core average compositions were tentatively selected by JAEA. Table 8 shows the characteristics of a 750-MWe core with such TRU compositions [16]. The maximum MA content of this study reaches 6% in heavy metal and its core characteristics satisfy the design condition. As described, TRU composition applied in this study is core average composition. TRU composition of each fuel sub-assembly can vary to both higher MA content and lower MA content.

Table 5: 1 500-MWe core characteristics
(fast reactor equilibrium cycle TRU composition)

Items		Breeding core	Break even core
Nominal full power [MWth]		3 570	←
Operational cycle length [months]		26	←
Refueling batch [core/RB]		4 / 4	4 / -
Pu enrichment (Pu/HM) [wt%]	Inner core	18.3	18.3
	Outer core	20.9	21.1
Burn-up reactivity [%Δk/kk']		2.3	2.5
Breeding ratio		1.10	1.03
Discharge burn-up [GWd/th]	Core	147	150
	Core+blanket	90	115
Maximum liner power [W/cm]		398	411
Core specific power [kW/kg-MOX]		41	41
Maximum neutron dose* [n/cm^2]		5.0×10^{23}	4.9×10^{23}
Pu fissile inventory [t/GWe]		5.7	5.8
Doppler coefficient** [Tdk/dT]		-5.7×10^{-3}	-5.8×10^{-3}
Sodium void reactivity** [USD]		5.3	5.3

* E>0.1MeV, ** EOEC.

Core transient calculations have been performed for typical cores. Transient events include design-base events and design extended conditions. In the design basis events, the core scram system and heat removal system protect the core and fuel from unacceptable damage. In the design-extended conditions, passive shut-down systems protect the core without sodium boiling and significant fuel centreline melting. The JSFR plant protection systems protect the core with minor actinide bearing oxide fuel. Further

analyses are in progress to show the feasibility of recent design of the cores such as (U, Pu) cores and TRU cores.

Regarding the core disruptive accident, JAEA performed core behaviour calculations. The results indicated that six dollars of sodium void reactivity was feasible and that value was selected as one of the design conditions. Further calculations will be made to check the feasibility of cores with MA bearing fuel in core disruptive accidents.

Table 6: Core characteristics comparison -1/3-
(FBR multi-recycle TRU and LWR-SF TRU*)

Items	FBR multi-recycle comp. core**	LWR recycle comp.core**
Core height [cm]	100	100
Ax. blank. height (upper/lower) [cm]	20/20	15/20
Operation cycle length [EFPM]	26.3	26.3
Fuel reloaded batch (core /rad. blank.) [-]	4/4	4/4
Pu enrichment (IC/OC) [wt.%]	18.3/20.9	19.6/22.1
Ave. discharge burn-up (core/core+blanket) [GWd /t]	147/90	146/93
Burn-up reactivity [%Δk/kk]	2.3	1.8
Breeding ratio (core/total) [-]	0.84/1.10	0.84/1.10

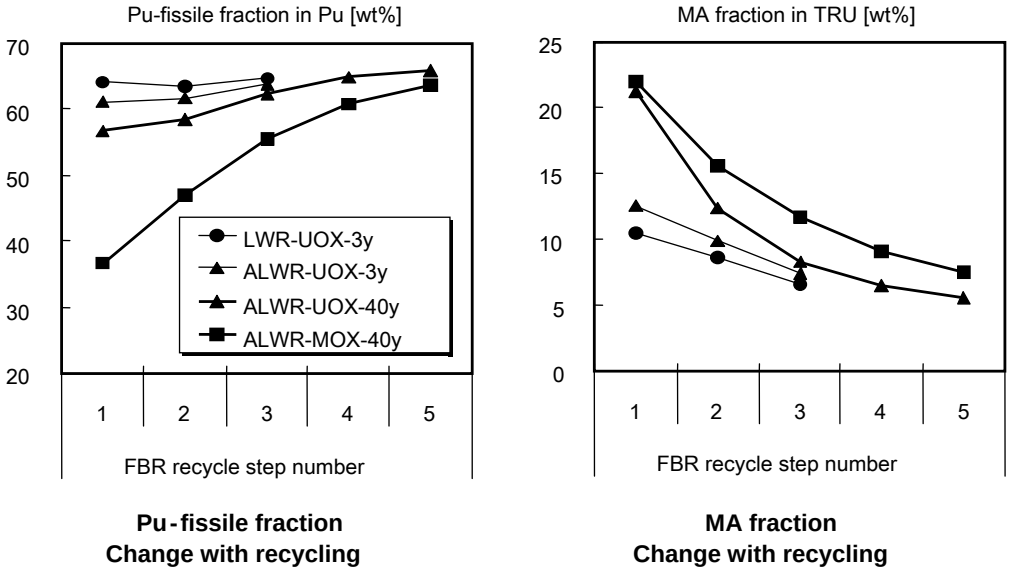
* See Figure 9.
** Core type: breeding core.

Table 7: Core characteristics comparison -2/3-
(FBR multi-recycle TRU and LWR-SF TRU)

Items	FBR multi-recycle comp. core	LWR recycle comp. core
Ave. core specific heat [kW/kg-MOX]	41	40
Max. linear power (IC/OC) [W/cm]	398/396	411/395
Max. fast neutron fluence [$x10^{23}n/cm^2$(E>0.1 MeV)]	5.0	4.9
Sodium void reactivity (EOEC) [$]	5.3	5.9
Doppler coefficient (EOEC) [$x10^{-3}$Tdk /dT]	-5.7	-4.5

Note: LWR recycle comp. core can attain the core performance targets. The core reactivity characteristics degrades by applying the LWR recycle composition, but the safety condition is satisfied.

Figure 15: TRU composition change in the fast reactor



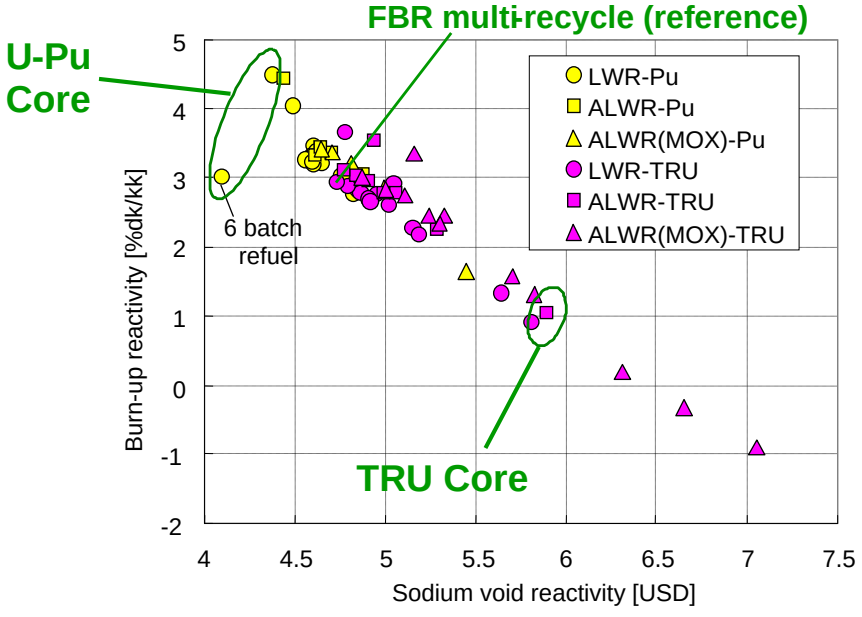
Pu-fissile fraction Change with recycling

MA fraction Change with recycling

1 recycle step = about 15 years

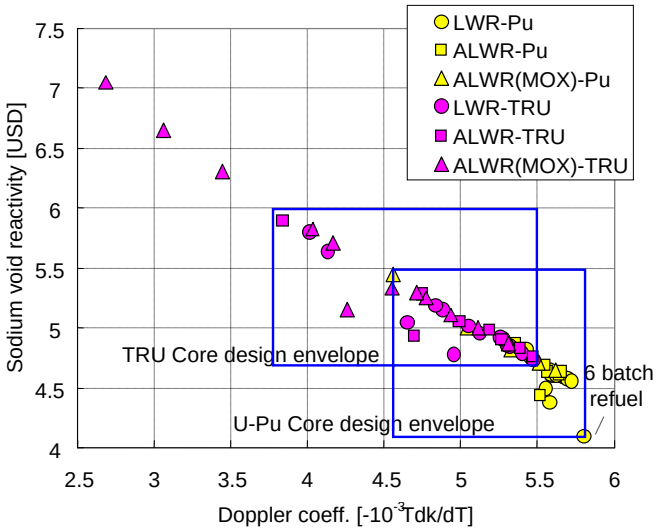
Source: Reference [15].

Figure 16: Burn-up reactivity and sodium void reactivity of fast reactor cores with TRU recycle



Source: Reference [14].

Figure 17: Sodium void reactivity and Doppler coefficient of fast reactor with TRU recycle



Source: Reference [14].

Table 8: Core characteristics comparison -3/3-
(U-Pu core, reference core and TRU core*)

Item	U-Pu core	Reference core	TRU core
TRU composition	LWR 3-year spent fuel (MA: 0.1 wt%)	FBR multi-recycle (MA: 1 wt%)	ALWR 40-year spent fuel (MA: 6 wt%)
Core height [cm]	100	100	100
Axial blanket thickness (upper/lower) [cm]	25/30	20/20	20/20
Operation cycle length [month]	18.2 (6 batch)	27.4 (4 batch)	27.4 (4 batch)
Pu enrichment (IC/OC) [wt%]	17.4/22.6	17.8/22.3	18.4/23.9
Burn-up reactivity [%dk /kk]	3.0	2.9	1.0
Breeding ratio	1.1	1.1	1.1
Sodium void reactivity (EOEC) [USD]	4.1	4.7	5.9

* See Figure 12.

3.2.5 Systematic evaluation of the impact of minor actinide loading for homogeneous recycling in fast reactors

A systematic evaluation of the impact of different minor actinide loadings on homogeneous recycle fast reactor cores can be deduced from a study performed at Argonne National Laboratory (ANL) [33]. In that work, oxide and metallic fuel equilibrium cycle cores with minor actinides content of 3.2% to 42.9% in the heavy metal of a fresh homogeneous recycle fuel were considered along with various conversion ratios (see Table 9). The ranges of minor actinide contents were obtained by considering two sources of fuel: one in which the MA/Pu ratio in the used nuclear fuel is 0.1 (similar to LWR UNF discharge fuel), and the other in which the MA/PU ratio is ~1, similar to fuel that has been continuously recycled in a thermal spectrum (or potentially for a situation where the minor actinides have accumulated and their fraction is being artificially increased for consumption in the fuel cycle).

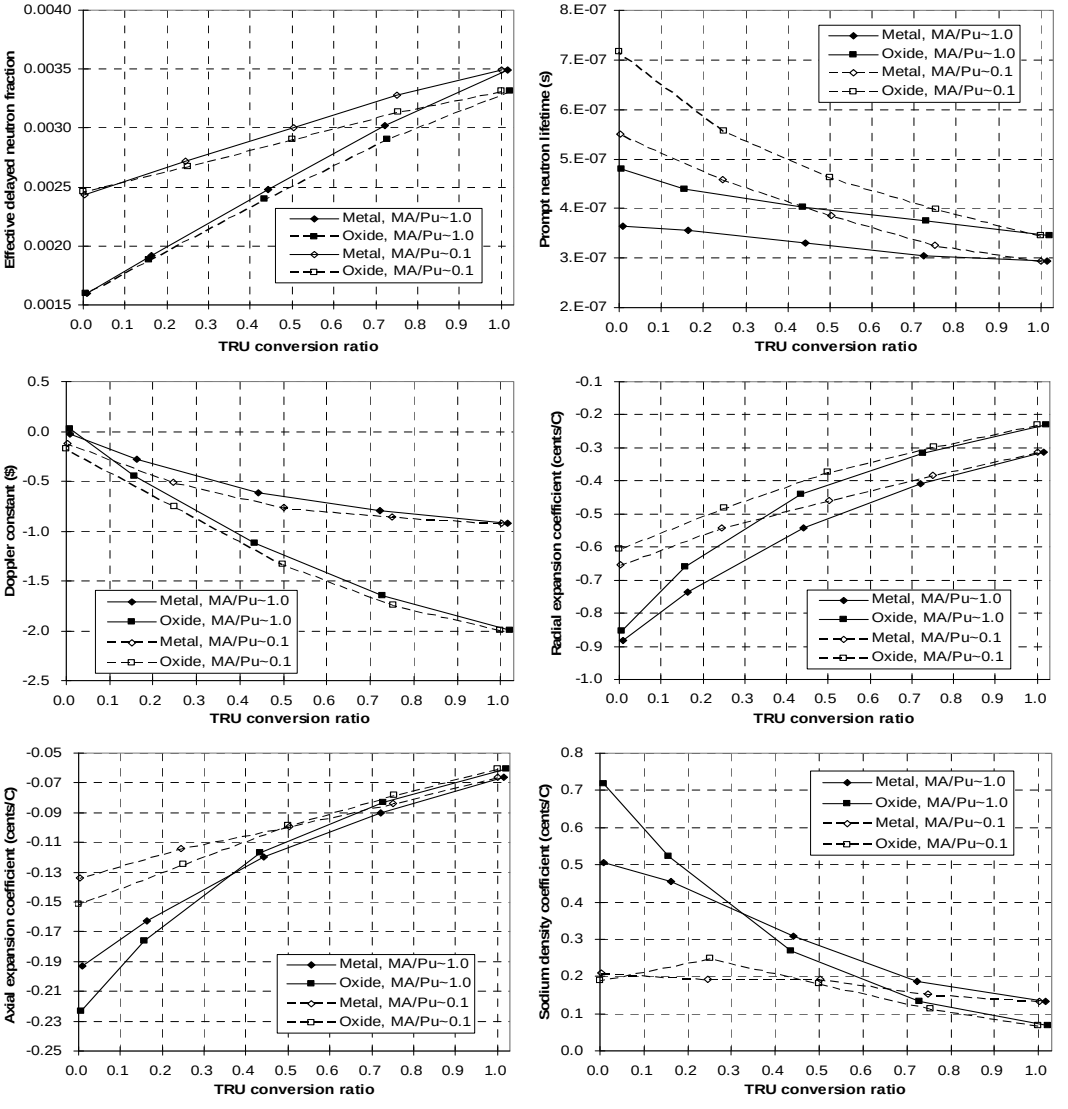
Table 9: Minor actinide content (%) of equilibrium cycle charge heavy metal (HM) as a function of conversion ratio and TRU feeds

Target TRU conversion ratio		1.00	0.75	0.50	0.25	0.00
TRU feed of MA/Pu~0.1	Metallic fuel	0.5	1.7	3.7	7.6	15.4
	Oxide fuel	0.8	2.3	4.5	8.2	15.6
TRU feed of MA/Pu~1.0	Metallic fuel	0.5	5.3	13.0	28.2	42.1
	Oxide fuel	0.8	6.2	14.7	29.8	42.0

The impacts of the MA loading on reactivity coefficients are presented in Figure 18. These data come from the end of cycle (EOC). The sodium void worth plots are not available, but the sodium density coefficient plots are presented. The voidance of the progression could be modeled by detailed safety analysis tools rather than by an arbitrary postulation method. The MA/Pu=0.1 and 1.0 cases in Figure 18 correspond to charge fuel MA/HM values of 1.7% and 5.3%, respectively, at a TRU conversion ratio of 0.75 (see Table 9). The results indicate differences between some of the reactivity coefficients and kinetics parameters over the TRU conversion ratio range (e.g. effective delayed neutron fraction, prompt neutron lifetime, radial and axial expansion coefficients and the sodium density coefficient). These differences are significantly larger at low conversion ratio values (typically below CR <0.6) where the MA/HM ratio is higher.

As indication of reactor safety, the study also evaluated integral reactivity parameters for the quasi-static reactivity balance analysis of the passive regulation of power [32]. The results indicated that except for the oxide case with high TRU conversion ratio, the cores satisfy the sufficient conditions for passive safety, irrespective of the MA/HM ratio. Even for the oxide case, the core could be made safer by using engineered systems, as has been done in oxide fuel fast reactors that have been operated or continue to be operated. The additional cost of the engineered-system would have to be evaluated in order to see their impact on the overall reactor cost. The detail of safety evaluations should be also performed [33].

Figure 18: Kinetics parameters and reactivity coefficients at EOC versus TRU conversion ratio



3.3 Issues related to heterogeneous recycling

3.3.1 TRU composition of driver and target assemblies

Various combinations of actinides have been considered for loading into homogeneous and heterogeneous recycle cores. For the homogeneous recycle core, if it is assumed that all minor actinides in used fuels of advanced fast reactors and operating LWRs could be sent to a repository, then only plutonium would need to be utilised in the fuel for the homogeneous recycle fast reactor. In most cases, a uranium-based fuel is assumed ([U,Pu] fuel). Contrarily, if minor actinides are to be transmuted, then they would have to be recovered with the plutonium in the separation processes in a grouped separation technology for subsequent utilisation in the fabrication of a [U,Pu,MA] fuel for use in the homogeneous recycle fast reactor.

The concept of heterogeneous recycle implies that that the plutonium and minor actinides are managed separately in the core and fuel cycle. In that case, core driver and target zones are defined. In the driver fuel, the traditional [U,Pu] fuel is typically assumed. In some heterogeneous recycle studies, Np has also been included in the driver fuel to minimise any potential adverse impacts it might have in certain repository settings. It also reduces the mass of the MA to be irradiated in the target. If the Np is not introduced in the driver fuel, then it must be recovered with other minor actinides (primarily Am and Cm) for the fabrication of the target fuel. However, for the target pin, various combinations of actinides have been considered. These include:

- [U,Np,Am,Cm];
- [U,Np,Am];
- [U,Am,Cm];
- [U, Am];
- Am+Cm only;
- Am only.

With the utilisation of thorium in the fuel cycle, additional combinations could be envisaged. The non-inclusion of Cm in the target fuel is predicated in the need to make fuel handling in the fuel cycle easier (Cm comes with high neutron radiation and heating). The Cm could be stored for about 100 years and the daughter products from its decay (plutonium nuclides primarily) could be re-introduced in a fast reactor core or disposed of in a repository. A problem with long cooling time is that this time could be longer than the reactor lifetime, which implies that the material being cooled becomes the legacy of another reactor core. It is important to note that, even without Cm initially in the target fuel, the element will be created by the transmutation of Am, and its quantity in the used fuel could be more than when Cm is recycled in the target.

3.3.2 Fuel types according to strategy

3.3.2.1 Fuels for heterogeneous recycle cores

Various types of fuels have been considered for homogeneous and heterogeneous recycle cores, depending on the goal of the transmutation system. For such systems, considerations must be given to fuels for the driver and target regions. For the driver region, traditional fuel types have been considered, including oxide, metallic, nitride, hydride, and carbide fuels, which are uranium based. A special case is the zero conversion ratio cores, which have typically been designed, based on physics consideration, with no uranium in the fuel pin.

Similar fuels have also been considered for the target region. In addition to these fuels, inert matrix fuels containing no uranium initially have also been considered. Other concepts have used a moderator with the inert matrix fuel to enhance transmutation of the minor actinides in the target fuel. These fuels for heterogeneous recycle strategies of MA stabilisation or burning are discussed in the following sections.

3.3.2.2 Fuels for MA stabilisation

As discussed in Section 3.1, conceptually, a heterogeneous recycle system could be considered for the management of the minor actinides in two primarily ways. These are either the burning or the stabilisation of the minor actinides in the system. If the intent is MA stabilisation, it is possible to envision a system that could continuously recycle the TRU in the heterogeneous recycle core, without complete destruction of the minor actinides in a given recycle stage through the core. In this case, it is pertinent to find a fuel that is readily reprocessed and compatible with recycle in the core. For this reason, the recent trend is that the heterogeneous recycle cores use uranium-based fuel for both

the driver and target assembly fuels [30], [22], [13], [23]. The use of uranium-based fuel is typically attractive because nearly all the test and commercial power reactors have used this fuel. Some of these early reactors have also used blanket fuels that have been managed separately from the driver fuels and it is expected that this basic experience can be extended to the target pins.

3.3.2.3 Fuels for burning MA

Non-uranium fuels become more attractive if the goal of the heterogeneous recycle system is minor actinides burning in a single recycle through the core. By not using uranium in the fuel, the additional production of plutonium and the minor actinides can be nearly eliminated. The challenge is to ensure that these fuels can stay in the core for many years to attain targeted irradiation levels before fuel performance limits are reached. Various fuels have been considered for MA burning. For the most part non-uranium fuels have been proposed in order to minimise the production of additional transuranics. The matrix for the MA is a ceramic or a metal that is inert from the neutron interaction viewpoint. The goal is that such an inert matrix fuel (IMF) should allow for a transmutation rate in the 90% or higher range and a good level of safety in the event of an incidence or accident. A specific issue is the very high helium production (due to neutron capture in ^{241}Am and successive decay of the ^{242}Cm created from fuel irradiation), which has an impact on both the target behaviour under irradiation and on the target assembly conceptual design.

The fissile atoms and the support matrix can either form a solid solution or be integrated in a composite fuel like ceramic inclusion in ceramic matrices (Cercer) or ceramic inclusions in metal (Cermet). The respective composition and concentration of the different parts of these composites are adjusted in order to take into account the different effects induced by irradiation. The main requirements are good thermal and mechanical properties for the matrix and chemical stability in the course of its evolution for the actinide phase. Possible ceramic or metal candidate materials have been selected with the criteria concerning their basic properties (thermal and mechanical properties, activation with neutrons, chemical compatibility with neighboring materials, etc.) and their behaviour under irradiation [4].

Under irradiation in the reactor, three main sources of damage have to be considered for the IMF: fast neutrons interaction, effects of fission fragments and alpha decay products. These energetic particles produce damage through nuclear interactions and the consequences, which depend on the material, may significantly affect the bulk properties: changes of lattice parameter, phase changes, amorphisation, swelling, and evolution of thermal and mechanical properties [4].

Examples of materials that have been considered include:

- Inert matrices: $MgAl_2O_4$, MgO, Al_2O_3, ZrO_2, CeO_2, $Y_3Al_5O_{12}$, Y_2O_3, TiN, NiO, TiO_2, CeO_2, W, Nb, V, Cr. MgO is considered a good candidate due to observed performance in irradiation test. MgO has a high melting point (~3 000^{0}C), high thermal conductivity, and a relatively high resistance to corrosive medium; however, it softens the neutron spectra. ZrO_2 is also considered to be a good candidate due to its good structure stability under irradiation.
- Actinide compounds: oxide or mixed oxide solid solutions.
- Target concepts: the first targets were based on a concept of a micro dispersion of the actinide phase in the host matrix e.g. (MgO + AmO_x), or $MgAl_2O_4$ + AmO_x).
- Moderators: ZrH_2, CaH_2, $^{11}B_4C$, BeO could be used to soften the neutron spectrum in the target zone.
- Hydride fuels: the presence of hydrogen results in a softened neutron spectrum in the target zone.
- Metallic fuels, e.g. U-MA-40Zr or MA-40Zr.

3.3.3 Core residence times for targets

Different core residence times of target assemblies have been calculated in various transmutation studies. The residence time has been typically limited by the neutron fluence or irradiation induced damage limit of the cladding and fuel performance issues. The latter (fuel performance) is typically alleviated by design changes such as reduction in loading or dilution over additional assemblies. So the former is typically most limiting. Since two primary types of target loading have been considered in fast reactors (core interior and core periphery), those are the focus of the ensuing discussion.

If the targets are loaded in the core interior, the trend is that the targets would have to be managed on the same fuel management scheme as the driver fuel assemblies. Conversely, if the targets are loaded in the core periphery, then these could be managed separately on a different core residence time interval than the driver fuel. This is because the fluence accumulated per unit time is typically lower. So for some designs, it has been observed that target assemblies stay two to three times longer than the driver assemblies. The softer spectrum in the core periphery also helps to prolong core residence (though not the primary contribution). By using moderated targets that are being promoted for near complete destruction of the MA in a single recycle, the target assembly has to reside longer in the core, and as such the assemblies are introduced principally in the core periphery. In any case, the residual TRU composition should be carefully assessed in terms of decay heat, radiotoxicity and dose.

3.3.4 Core conversion ratio impacts

The overall conversion ratio of the heterogeneous recycle core has no direct bearing on the ability of target fuel assemblies to burn or stabilised minor actinides (MA). Even in a breeder reactor core, the target fuel can be configured for the burning of the MA by increasing the minor actinides to heavy metal ratio (minor actinide content). Conversely, in a low conversion ratio core, it is conceptually possible to design a target fuel for MA stabilisation. There are, however, potential external limitations to the loading of minor actinides arising from limits imposed by fuel performance and handling considerations. Recent studies considering the decay heat allowable for cooled and fresh fuel assemblies suggest that MA content of about 10% is all that could be tolerated in the target assemblies. This content of course depends on the number of target pins and total mass of the minor actinides in the fuel assembly.

While the conversion ratio might have no direct impact on the function of the target assemblies, its secondary impact on the function should, however, be considered in the overall strategy for MA transmutation. Specifically, the balance of MA production in the driver fuels and MA stabilisation or burning in the target should be considered. A low conversion ratio system, for example, would typically have higher minor actinide content in the heavy metal to be reprocessed than a high conversion (breeder) ratio system. Consequently, if the overall system strategy is for MA destruction, it is important that the target fuel have an MA burning rate that accommodates the overall burning of MA in the system (for the MA stabilisation and burning cases). This in turn implies high minor actinide content in the target fuel.

3.3.5 Impact of neutron moderation in target fuel assemblies

Due to the absence of uranium in an inert matrix fuel (IMF) target, it is expected that a high net consumption of the MA would be obtained as no new MA is produced from neutron capture. Similarly, moderating the assembly should result in higher transmutation rate due to the softening of the neutron spectrum (lower damage rate) and thus the potential for longer residence of the target in the core, and the higher neutron interaction rates for the same flux level. These trends were evaluated in an Idaho National Laboratory (INL) study, which compared the transmutation performance of different target assemblies for MA in sodium-cooled fast reactors [3]. In the study, Pu+Np

are assumed to be initially in the driver fuel assemblies and the other minor actinides in the target fuel assemblies. A heterogeneous recycling strategy is investigated, whereby after each reactor pass, unburned MA from the targets is blended with MA produced by the driver fuel and additional MA from used nuclear fuel (UNF). The study evaluated cases with the targets in the core periphery with the intent of minimising overall safety implications by keeping the assemblies out of the high-importance central region of the core.

The main difference in the target assembly designs investigated is the selection of a homogeneous lattice versus a heterogeneous lattice of target pins. A homogeneous target pin lattice was considered with MA-O_2/UO_2, MA-O_2/MgO or MA-ZrH_x matrix compositions. Two heterogeneous lattices were investigated, one with 55 moderator pins, the other with 150 moderator pins (see Figure 19). These two configurations represent two possible scenarios of introducing increasing amounts of moderation to the target irradiation vehicles. The heterogeneous pin lattice offers the advantage of mechanical separation of the MA-carrying pins from the moderator pins. An alternative approach is to select a matrix material that is hydrogenous.

Using a transmutation efficiency indicator (ratio of the final to initial minor actinide or transuranic masses) as a basis, it was found that the fertile versus inert matrix fuel option makes a significant difference to the transmutation performance of the target design. See Figure 20, where (1-E_x), the transmutation efficiency indicator is the ratio of the final mass to the initial mass; E_x is the fraction of the initial mass that is removed by irradiation of x. There is a clear delineation between the transmutation efficiency of the moderated and unmoderated assemblies, with the unmoderated assemblies showing significantly lower transmutation efficiency for minor actinides. The UO_2 and MgO target designs have the lowest values. Additionally, it was observed that the targets can be a net producer of transuranics even though they are a net destroyer of MA. This would clearly depend on the initial loading of the minor actinides in the target assembly and the conversion ratio of the core. The results also suggest that the hydrogen stoichiometry does not greatly improve transmutation efficiency. Because of the softer spectrum brought about by the local moderation, the target assemblies could be irradiated for much longer than 10 cycles. The number of cycles had to be increased to 18 in order to irradiate the targets to the fluence limit and thus higher transmutation efficiency.

A significant concern with the utilisation of moderator materials is the possibility for localised power peaking, leading to excessive fuel and clad temperatures, either inside the target or in immediately neighboring core fuel sub-assemblies [20]. There are potential design solutions such as using a strong absorber within the target [20] to reduce this power peaking effect as the large capture cross section at low energy would reduce the number of low-energy neutrons that leak from the target sub-assembly into the neighboring core fuel. Another approach is to use a transition row of non-moderated target assembly between the driver fuel assemblies and the moderated target assemblies.

Another issue with the use of moderated target assemblies is the production of higher mass nuclides. These nuclides are a burden in the fuel cycle due to the associated radiation sources, whatever the strategy used for the moderated targets (multi-recycle or once-through).

Recent work has found that the issue of localised power peaking in adjacent driver assemblies due to in-flux of thermal neutrons from moderated targets was relatively minor [1]. This work explored the pin-by-pin localised power peaking effects caused by radial target in both ex-core and in-core locations, using [3] as a starting reference. The study found that power peaking in the target itself was an issue due to the eventual accumulation of ^{239}Pu as a result of successive transmutations of minor actinides (e.g. ^{241}Am to ^{242}Am, beta decay to ^{242}Cm, alpha decay to ^{238}Pu, to ^{239}Pu). The eventual high ^{239}Pu concentration in the absence of neutron capturing ^{238}U in a thermal spectrum created unacceptably high assembly power peaking within the fertile-free moderated

targets, especially when in in-core positions. In order to take advantage of high minor actinide destruction while minimising the power peaking effects, a hybrid scenario was devised where the targets resided ex-core in a moderated assembly for the first 506.9 effective full power days (EFPDs) to destroy minor actinides and then were moved to an in-core arrangement with the moderating rods removed for the remainder of the life cycle. The concept of removable rods midway through a core life cycle is based on a removable burnable poison design originally devised for PWRs. The hybrid model had an assembly and pin power peaking of less than 2.0 and a higher heavy metal and minor actinide destruction rate than the un-moderated heterogeneous targets either in-core or ex-core [1]. The thermal-then-fast approach decreased the amount of ^{239}Pu accumulation in the target assembly compared to the fully moderated case. The shift to the fast spectrum greatly reduced the reactivity worth of the ^{239}Pu that had already accumulated while in the thermal spectrum. This mid-term spectral shift had the added benefit of burning some of the curium and higher mass actinide isotopes that resulted from americium transmutation during the thermal leg of the irradiation.

Figure 19: Homogeneous and heterogeneous lattice geometries

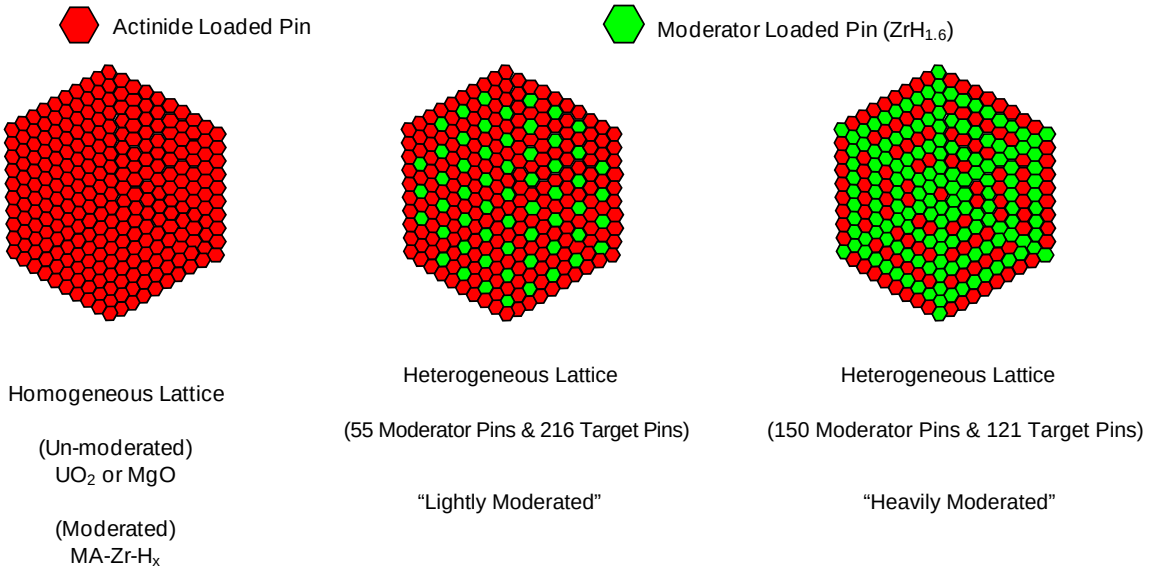
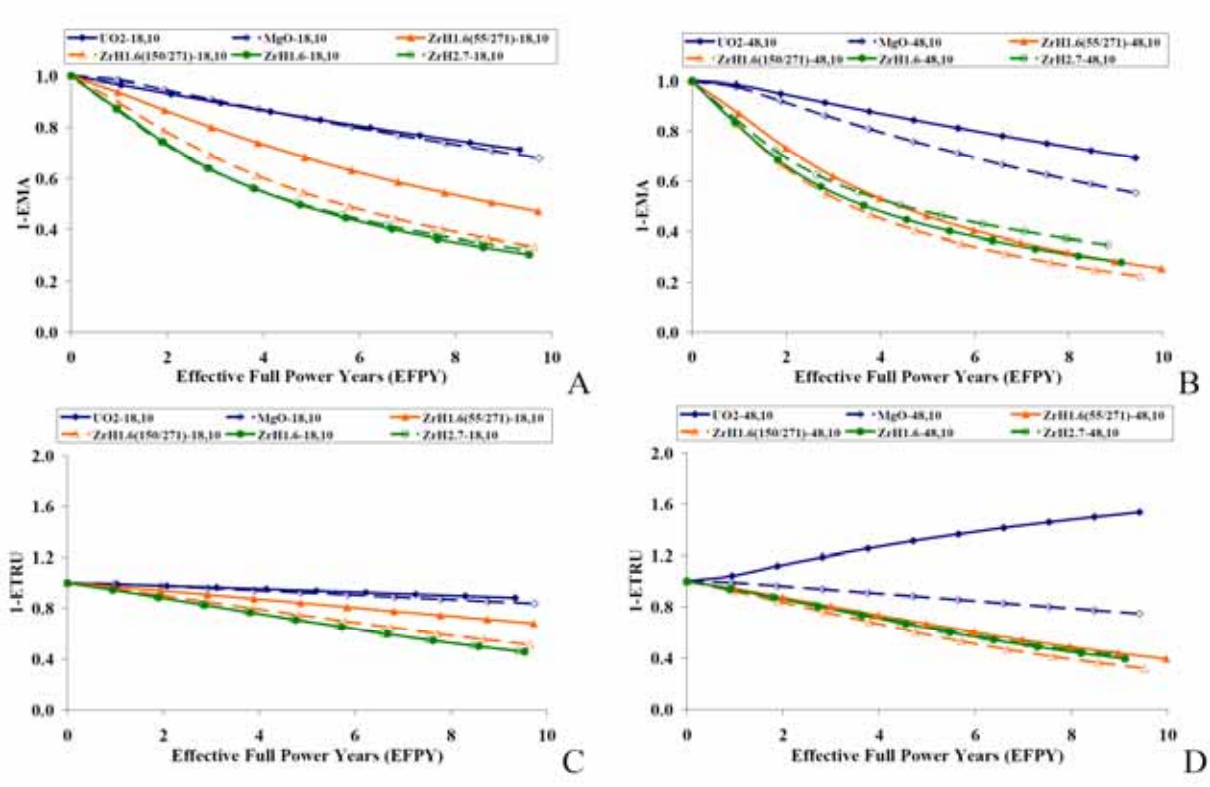


Figure 20: Depletion behaviour normalised to minor actinides (MA) or transuranics (TRU) transmutation efficiency

a) E_{MA} for 18-targets, b) E_{MA} for 48-targets, c) E_{TRU} for 18-target,
d) E_{TRU} for 48-targets; 10-cycle irradiation (no fluence optimisation)



3.3.6 Helium generation in metallic target fuel

An evaluation of the impact of helium (He) generation in metallic target fuel has been performed. Typically, xenon (Xe) and krypton (Kr) are the main insoluble gas types generated during nuclear fission as fission products. However, irradiation of fuel containing Am and Cm leads to the production of helium (He) gas atoms as a result of neutron capture in ^{241}Am which leads to the production of ^{242}Cm; ^{242}Cm alpha decays into ^{238}Pu, producing He. Depending on the amounts of Am and Cm in the fuel, He build-up in the fuel can be significant and can exceed the amounts of Xe and Kr generated as fission products. This is particularly the case for heterogeneous target fuel that is expected to contain significant quantities of those minor actinide elements.

Depending on a number of factors, helium is either retained in the fuel slug or released outside of the fuel slug. Helium is released outside from the fuel slug to plenum volume and then the plenum pressure will increase. If it is retained within fuel slug material it will increase fuel volumetric swelling (ultimately enhancing the effects of fuel cladding mechanical interaction, FCMI, especially at high burn-ups). Increase in plenum pressure will increase stresses on the cladding. Those stresses, combined with possible decrease in cladding thickness because of fuel cladding chemical interaction (FCCI), can be large enough to cause cladding failure. Particularly, a long residence time in target fuel could increase the effects of He generation, which might be required to achieve specific target burn-up levels, since target fuel is expected to be located in the reactor core outer region where neutron flux is lower. This increase in residence time can lead to a

significant effect on cladding thermal creep, especially if the cladding temperature is high, affecting fuel integrity.

The major fuel irradiation performance issues of the TRU containing fuel are dimensional stability, microstructural evolution, fuel-cladding compatibility and He generation. Limited experimental data are available regarding the behaviour of this type of fuel, which might be fabricated in different forms including oxide, metal, inert matrix dispersion, hydride, carbide and other forms. This lack of experimental data makes it difficult to evaluate quantitatively its effect on fuel performance. Meanwhile, the amount of He generation can be evaluated quantitatively from reactor neutronics calculations.

Generally, no significant amount of He is generated in nuclear fission of typical commercial reactor fuels. Instead, it can be generated indirectly, as mentioned above, in fuel that contains Am and Cm. ^{241}Am is transmuted through n–γ reaction into ^{242}Am which decays into ^{242}Cm, which subsequently α –decays into ^{238}Pu. This He generation scheme is similar for both thermal and fast neutron irradiation with the only differences being the probabilities of neutron interaction and decay (e.g. branching ratio), which results in a difference in transmutation rates.

The amount of He retained in transmutation fuel during irradiation depends on different factors including fuel type (oxide, metal, dispersion, etc.) and the amount of TRU in the fuel in addition to fuel operating conditions (temperature and burn-up). For a given mass of fuel the helium generation rate in the target fuel that initially contains the minor actinides will be significantly higher than in the driver fuel with no minor actinides.

In general, He is insoluble in fuel material and its diffusion is expected to be larger than Xe and Kr, especially at low temperatures. This facilitates He transport within fuel material to grain boundaries affecting bubble nucleation and growth and possibly allowing for a possibility of full release of He to the outside of the fuel slug. This might be the case for certain types of fast reactor fuel, in particular metallic fuel with appropriate smeared density where interconnection of fuel porosities after short periods of irradiation allow for a large fission gas release fraction to the plenum region.

In order to evaluate the performance of the homogenous and heterogeneous fuel designs a set of evaluation criteria should be defined. Those criteria are based on the design criteria that were developed for the EBR-II Mark-V metallic fuel pins in 1994 as part of the safety case for the introduction of this fuel into EBR-II. They are based on the Clinch River Breeder Reactor (CRBR) and General Electric PRISM criteria, combined with general experience of the behaviour of metallic fuel cladding material with HT-9 steel under irradiation. Establishment of those criteria ensures that the designs meet general requirements that the fuel pins will satisfy their functional requirements and performance objectives in a safe and reliable manner based on current technology. The evaluation criteria and analysis methods address the geometry, temperature regimes and loading mechanisms that are expected to influence fuel performance during normal operations. Stress loading on the cladding from FCMI is not an issue for the metallic fuel design considered here due to the relatively low discharge burn-up (about 10 at %). Only steady state criteria applicable to normal operation of the fuel are considered in the evaluation. Fuel melting and eutectic liquefaction possibilities have not been considered. The evaluation is based on limiting inner cladding temperature for designs of interest to temperatures less than 650°C and typical linear powers that do not lead to fuel melting.

Evaluation of the expected performance of heterogeneous target fuel design shows that He generation in this type of fuel can have a significant impact on its performance. Pre-specified evaluation criteria can be violated at different inner cladding temperatures, especially if the effects of FCCI are taken into account. This degradation in fuel performance is not expected in heterogeneous and homogenous driver fuels. In order to improve the performance of target fuel designs a number of possible design changes can be introduced. These changes include longer plenum region, lower inner fuel cladding

temperatures, and improved cladding alloy including the introduction of cladding liner or thicker cladding. Any of those changes will impact the expected economics of the reactor, though increasing cladding thickness might be the most appropriate change.

3.3.7 Location of target assemblies

The preference in the heterogeneous recycle approach is to have the target pins and the driver fuel pins in separate assemblies (i.e. different homogeneous assemblies). This ensures that the pins are not mixed and allows different approaches for fabrication and management of the minor actinide containing assemblies, consistent with the intent of minimising their footprint in the fuel cycle. This separation leads to different options for placement of the target assemblies in the fast reactor core. The target assemblies could be placed in the core interior or the core periphery. The rationale for such placements and the associated performance results are provided for the two types in the following sections.

3.3.7.1 Target assemblies in core periphery zones

The advantages of loading the target assemblies in the core periphery include the use of leakage neutrons for the transmutation of the minor actinides and a relatively softer spectrum that could enhance transmutation due to relatively higher absorption cross-sections. This softer spectrum could be additionally enhanced by using a moderated zone at the core periphery. A disadvantage of periphery loading is that the flux level in this region is relatively lower than in the core interior and the flux gradient is steeper. The lower flux level necessitates that the target assembly reside in the core longer than the driver fuel assembly if the design goal is to achieve the highest burn-up attainable within the displacement per atom (dpa) or fluence limit in the target assemblies. Consequently, the driver and target assemblies would have different in-core fuel management strategies. The longer residence time of the target assemblies implies that the MA throughput in the core would be lower than for inner core loading case, for the same MA loading and number of MA assemblies. Since leakage neutrons are used for transmutation, additional row(s) of target assemblies could be loaded, to increase the throughput, with little effect on core performance.

The steep flux gradient in target assemblies loaded at the core periphery would have to be managed to ensure that fuel pin performance is not degraded. There would be significant variations in the spatial profile of both dpa (factor ~3) and power density (factor of ~2) in blanket assemblies due to strong flux and spectral gradients in the core periphery. The power density also varies significantly over time in the blanket due to rise to power resulting from the creation of plutonium in the uranium-based target. The target assemblies could be rotated or shuffled into different core sectors to ensure uniform burn-up of the pins. The implementation and design is further complicated by the temperature dependence of fission gas release behaviour of UO_2 based fuel. While the hot side of the assembly may experience normal fission gas generation and release from the UO_2-based fuel, the back side of the fuel assembly is cold enough for fission gas generation and release behaviour to remain questionable. Normal operation would be to rotate these assemblies in order to flatten the burn-up distribution across the assembly. It is suspected that this would be very detrimental to the fuel matrix stability and in fact may cause the fuel matrix to disintegrate as fission gas bubbles would be rapidly generated and forced out of the matrix. Short-term research should be focused on examining this behaviour [6].

3.3.7.2 Target assemblies in active core radial zone

By loading the target assemblies in the core interior, the steep flux gradients associated with loading at the core periphery are eliminated and the flux level is higher. The flux spectrum is, however, harder. The higher flux level in the core interior reduces the residence time required to reach a certain burn-up within the dpa limit. This target assembly residence time is similar to that of the driver fuel assemblies. However, because

of the harder flux spectrum, the burn-up to dpa-limit ratio is decreased. Consequently, the overall burn-up of the target assemblies could decrease. An advantage of inner core loading is that the throughput of the MA in the core is increased.

An ANL study evaluated the impact of swapping twelve target assemblies from the core periphery with driver assemblies in the core interior [13]. According to the study, uranium-based target pins with 40% MA loading in the heavy metal were considered. It was found that this variation in target assembly location reduces the target assembly average discharge burn-up by ~20% (that of the driver assembly remains the same). The residence time of the target assemblies was reduced from ~8 years (when on the periphery) to 4.5 years (when in the interior).

A recent JAEA heterogeneous recycle design study has expressed preference for loading the target assemblies in the core interior, in the region between the inner core and outer core zones [22]. This was based on sensitivity studies comparing the results of heterogeneous recycle cases with inner and periphery loadings with those of the homogeneous recycle core. Different positions in the inner core were considered before this boundary position between the inner and outer core zones was found the most favorable based on considerations for the maximum linear heat rate, MA transmutation, Doppler coefficient, burn-up reactivity swing, and sodium void worth.

In the JAEA study, uranium-based oxide fuel was considered. In order to ensure a relatively flat power distribution, the inner target assemblies in the heterogeneous cores employed fuel pins containing the minor actinides mixed with a fissile material (Pu), hence resulting in the use of MOX target pins. Np is assumed to be co-processed with the Pu, which are fed together to the MOX fuel fabrication facility (consequently, Pu and Np are in both the driver and target pins of the heterogeneous recycle core). Am and Cm are assumed to be accompanied by rare earth nuclides (^{150}Sm in the study) in the extraction process, but at a low proportion (1% as advanced partitioning technology is assumed). In the heterogeneous recycle core, the Am and Cm are initially in the target pins. A possible advantage of this concept is to reduce the number of fuel pins and assemblies with minor actinides, which may reduce the fuel fabrication load.

The results of the heterogeneous recycle approaches are compared with those of the homogeneous recycle approach in Table 10. The core layout for the heterogeneous recycle core with 20% MA loading in the heavy metal is displayed in Figure 21. Table 10 also contains the typical loading constituents of the MOX target pins. In the study, target MA loading of 10 to 20% in heavy metal was considered and the results for them are provided in the Table. For these two heterogeneous recycle cases, the overall core loading of Am+Cm is ~2.8%, which is the same MA loading content of the reference homogeneous recycle case. The overall MA loading (including Np) of the homogeneous and heterogeneous recycle cores is ~4.2%. The core burn-up including the blanket assemblies is 90 GWd/t. The burn-up of the heterogeneous recycle target is about 173 GWd/t and that of the homogeneous and heterogeneous recycle driver fuels is about 145 GWd/t. The cores employ blankets to obtain a breeding ratio of about 1.13.

With the Am target in-core loading, MA transmutation rate and sodium void reactivity are equivalent to that of the reference core. However, due to the significant power mismatch during irradiation and limited performance of high MA content fuel, fuel design and thermal hydraulic design are required to resolve the issues. With Am target in ex-core loading, since the driver fuel contains little amount of minor actinide, there are no significant drawbacks in core characteristics, but the MA transmutation rate in the Am target is about 1/3 that of the homogeneous recycle/loading concept. If the lifetime of Am target of ex-core loading is extended, this drawback may be reduced. However, another drawback of thermal hydraulic design may become significant. In the case of long-life Am target, power increase with irradiation becomes significant, which results in an increase in the required flow rate of Am target and a decrease in the flow

rate of driver fuel. This leads to the drawback of core performance such as a decrease in core outlet temperature.

In the JAEA point design for the heterogeneous recycle core, the potential increase in the fuel pin cumulative damage function from helium gas production was alleviated by a target design modification. This helium production is primarily from the neutron capture in ^{241}Am, the products of which ultimately decay to ^{238}Pu, by alpha decay. The proposed approaches for alleviating the potential damage from helium production included increasing the plenum length, reducing the power in the target pins (i.e. the Pu content), shortening the Am+Cm target height, and increasing the target pin cladding thickness. The JAEA proposed increasing the cladding thickness as the most feasible measure for coping with the helium production issue. This led to the refinement of the core design and results different from those in Table 10.

Table 10: Core characteristics of the homogeneous and heterogeneous recycle cases with MA loading of 10% and 20% around 10th assembly layer

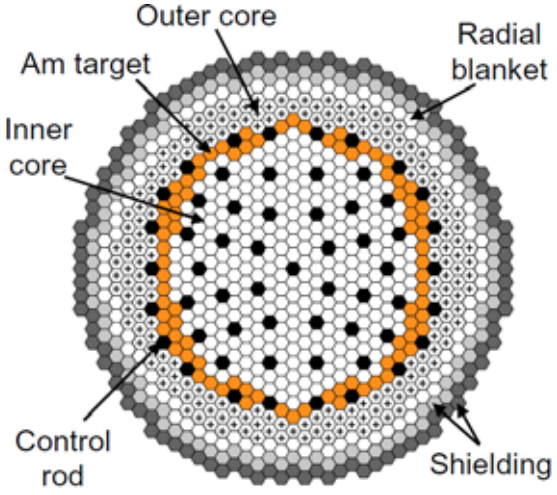
Core power is 1 500 MWe/3570 MWt, cycle length is 26.3 months in homogeneous case, core breeding ratio is ~1.13

Item	Homogeneous loading method	Target loading method	
		Am/HM: 20%	Am/HM: 10%
Number of fuel assembly*	287 / 274 / -	288 / 192 82	206 / 192 / 164
Pu enrichment (Pu/HM)* [wt%]	18.2 / 21.4 / -	18.5 / 23.2 / 17.0	18.5 / 23.5 /17.4
Burn-up reactivity [%Δk/kk']	1.55	1.53	1.38
Maximum linear heat rate*, **	357 / 356 / -	332 / 332 / 332	336 / 336 / 336
Sodium void reactivity [USD]	5.8	5.6	5.8
MA transmutation rate [%fuel life]	34	33	36

* Inner core fuel/outer core fuel/Am target.
** By two-dimensional calculation.

Figure 21: Core configuration of Am-target loading case

Am content: 20 wt%; loading position: around the 10th assembly layer



3.3.7.3 MA targets in assembly axial locations

An alternative to using radial target assemblies is the introduction of the MA target material into the axial blanket location in assemblies. In this regard, the leakage neutrons from the core could be used for transmutation of the minor actinides [2], [29]. A disadvantage of this approach over the radial target assembly approach is that the target and driver fuel materials will have to be produced and managed together (including transfer between fuel cycle facilities). This does not allow the reduced footprint of target materials in the fuel cycle, which is one of the primary reasons for considering the heterogeneous recycle approach. Introducing MA into the axial blanket might not be attractive in such a fuel cycle scenario, which intends to reduce the footprint of the MA in the fabrication facilities. However, there are advantages if the intent is to deploy a few dedicated MA burning reactors synergistically to support many Pu-only burning reactors, for a large stockpile of legacy MA (a different mission).

3.3.8 Minor actinide core loading impacts

Three kinds of MA loading information are relevant to the heterogeneous recycle approach. The first is the loading of MA in the target pin (i.e. the MA content in the heavy metal). The second concerns the loading of MA containing assemblies in the core. The third is the fraction of cores in a national nuclear infrastructure (park) that are loaded with minor actinide-containing assemblies.

Clearly, the higher the MA content in the fuel pin, the lower the number of assemblies/cores that contain the minor actinide containing pins/assemblies. For non-uranium target fuel, the MA content in the heavy metal is 100%, unless thorium is used as the fuel base. For uranium based fuels, studies have been performed for MA content loading fractions of 10% to 40%. The upper limit is the order of the content of TRU (Pu primarily) that has been used for oxide fuel. It is noted that if the TRU fraction in heavy metal of a target falls below 10% to <5%, then the MA content approaches that of the natural evolution of MA is fast reactor homogeneous recycle fuel. In that case, the number of target assemblies will approach that of the assemblies in the homogeneous recycle approach and thus the original intent of limiting the footprint of MA in the fuel cycle becomes unattainable.

3.3.8.1 Impacts of decay heat and radiation sources on loading limit

The fresh and used fuel decay heat and radiation levels might be a basis for limiting the MA content in targets. Very high decay heat and radiation source levels could make target handling and fabrication burdensome or expensive. A CEA study evaluated core designs with MA contents of 40% and 10% in a uranium-based oxide targets in a 3 600-MWt sodium fast reactor [30]. The ambitious 40% case was selected to be close to what is expected for accelerator-driven system fuel. This loading enables maximising the MA loaded mass in the blankets and may lead to a small part of the power plant fleet using blankets to achieve MA equilibrium (production of the whole SFR fleet equals destruction in the radial blankets). The lower 10% case was considered easily achievable by short-term technology and is likely a more realistic view of a situation in which all SFR cores have radial blankets.

In the CEA core design, the MA target assemblies are confined to the core periphery and the continuous recycle of the material is assumed; the minor actinides in the spent driver and target assemblies are used to make the target fuel. The target assemblies are assumed to stay in the core twice as long as the driver fuel assembly. The schematic of the heterogeneous recycle approach is provided in Figure 22. The figure shows that the core contains a core (driver zone) and blanket (target zone). The original MA composition is 76% Am, 17% Np, and 7% Cm. The core layout is presented in Figure 23 [31]. About 15% of the overall assemblies are target assemblies.

The CEA results indicated that the transmutation rate is close to 40% for both cases. The fraction of cores with blankets in the power plant fleet to ensure MA equilibrium is 23% for

the 40% MA case and 88% for the 10% MA case (equilibrium evaluation). Assuming a 40% thermal efficiency, the MA consumptions are -12 kg/TWeh and -3.5 kg/TWeh respectively. The equilibrium evaluation also shows that for the 40% MA case, about 40 target assemblies have to be fabricated per year, while 200 sub-assemblies would have to be fabricated for the 10% MA case, for the French fleet (400 TWh/year).

Table 11: Thermal power and neutron source for fresh assembly

Parameter	SFR homogeneous (UPu + 0.7% MA)	Radial blanket 40% MA	Radial blanket 10% MA	Driver fuel Pu
Thermal power (kW)	0.7	21.6	5.4	0.5
Neutron emission (n/s)	1.7×10^{9}	8.0×10^{10}	1.9×10^{10}	2.0×10^{7}

The thermal power and neutron source have been identified as fresh target assembly parameters that could be used to characterise the handling challenges for the front end of the fuel cycle. Results are summarised in Table 11. The thermal power and neutron sources are observed to be significantly higher for the radial blanket (target) in both the 40% and 10% MA cases compared to the homogeneous recycle fuel assembly or to the standard Pu driver fuel.

The post-irradiation decay heat levels for the cases are displayed in Figure 24 [5]. The decay heat of the homogeneous recycle fuel assembly is seen to be quite similar compared to standard Pu fuel driver. The current limit of 40 kW to handling assemblies (sodium surrounding) is reached after 3 days compared to 2 days for the standard fuel. For the target with 20% MA content, the limit is reached after 10 days. Considering the second limit of 7.5 kW to clean-up assemblies (gas surrounding), this limit is reached after less than 200 days for homogeneous recycle and standard driver fuel. For 20% MA target, it is necessary to wait about 1 000 days in order to respect the current limit.

To reduce this waiting time, specific equipment would have to be developed to deal with a high heat level to handle these assemblies. CEA is evaluating the necessary equipment. The study by CEA concluded that the range 10% to 20% MA content seems a good balance between transmutation performances and back/front end impact (neutrons source, decay heat, etc.) compared to the 40% MA case [30]. Similarly, difficulty with high MA content has been observed in ANL and JAEA studies, for example. To avoid costly redundancies of the storage facilities (particularly long-term in-vessel storage), a research and development effort is needed, both for the transport and the management of sub-assemblies with a high decay heat.

Concerning the neutronic power increase during irradiation for minor actinides bearing sub-assembly a factor ranging from 2 to 3 can be reached between beginning and end of life in large SFR cores, depending on the initial minor actinides (MA) composition. It is observed that the lower the MA content, the larger the power swing reached. For low MA content such as 10% the power factor is quite similar to what is expected for traditional depleted UO_2 blanket for the same residence time. The main difference between these configurations comes from the initial blanket absolute power, which is rather low for depleted UO_2 in comparison with 10% MA content, which exhibits higher fission rate. As the main impact relies on the outlet temperature of those sub-assemblies, the only significant differences may be on absolute flow, which would be twice as high at maximum for MA bearing sub-assemblies. Some relevant thermohydraulic studies of flow rate in these core regions could indicate the real impact of thermal stripping on structure zone, such as the core cover.

Figure 22: Heterogeneous recycle approach of CEA study

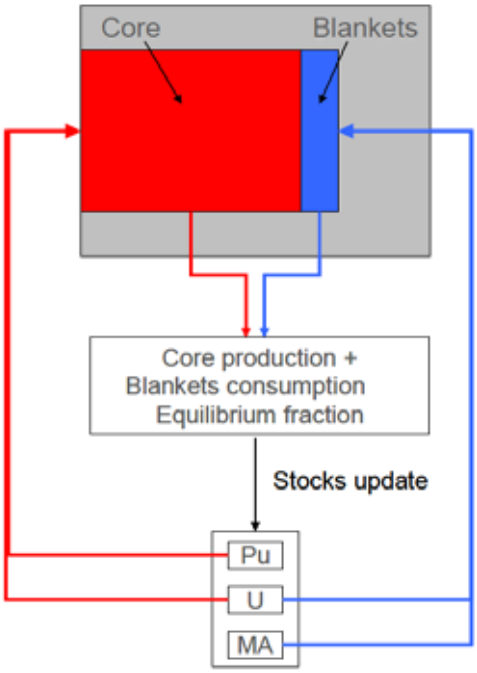


Figure 23: Heterogeneous recycle core layout in CEA study

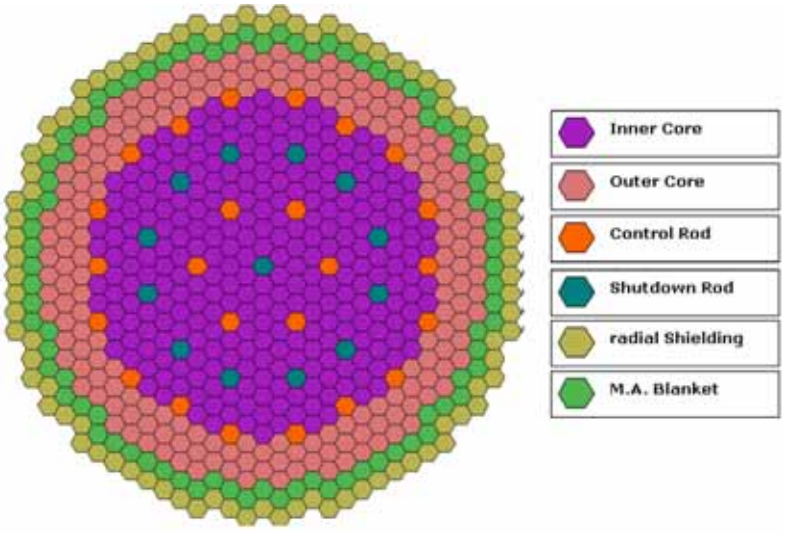
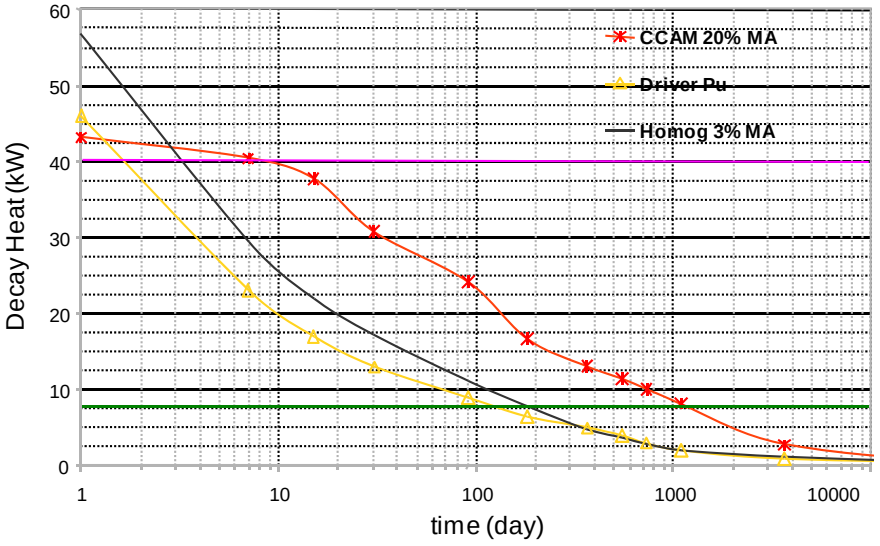


Figure 24: Time dependence of decay heat after irradiation



Source: Reference [5].

3.3.8.2 Spent and fresh fuel handling issues

Given the higher decay heat when the minor actinides are present in the fresh fuel, relative to the conventional fast reactor fuels containing mixtures of only uranium and plutonium, the issues associated with managing this heat during fuel handling must be adequately evaluated. A recent study has noted that the decay heat from a fresh fuel sub-assembly depends on fuel assembly size and inventory and minor actinide contents [7 and 8]. In the case of a sodium-cooled, small modular fast reactor, the decay heat level of the homogeneous TRU-bearing fresh fuel is estimated to be 1.2 kW for a sub-assembly with metallic fuel of 2.1% minor actinide per heavy metal. This decay heat level is more than ten times higher than that of the conventional uranium-plutonium mixed fuel. Because of the higher decay heat and radiation levels, TRU-bearing fresh fuel will most likely require special shipping casks and storage which can accommodate these features without degrading the fuel characteristics.

The study indicated that the TRU shipping cask and storage facilities are expected to be of the dry type to avoid any damage from the interaction of residual moisture on the fuel with sodium coolant. The cladding temperature during shipping and storage must be kept below the cladding creep temperature to avoid creep damage before core loading. In this case, passive decay heat removal from the shipping cask is critical, since any active cooling device is not suitable for an economical shipping cask. Since the only experience with fresh transuranic-bearing fast reactor fuel handling has been at the experimental level, a detailed feasibility study that most likely includes fundamental experiments will be required.

These considerations have to be put in perspective, comparing heat loads from different fuels and targets, as the ones considered in the CEA study [30], discussed above. Table 12 shows decay heat values for different target compositions of interest.

Table 12: Decay heat values for two minor actinide contents

Target MA content in HM (%)	Fresh fuel power (kW)	Decay heat (kW) of irradiated target assembly after		
		3 months	12 months	36 months
40	21	52	31	21
10	5.4	15	9	6

All steps of the fuel handling system will be affected, as can be deduced from the data shown in the table. In particular, the approach with 40% of MA/HM in the target, which strongly reduces the number of assemblies to be handled, is very penalising. In fact, transport and maintenance under gas are impossible, and a solution with presence of Na is needed. Moreover, after irradiation and to comply with current regulation, the target sub-assemblies should cool down for approximately 12 months. A long decay delay should also be planned in a separate cell, to comply with the criteria for cleaning the assembly (<7.5 kW).

In the case of 10% MA/HM targets (which implies a much larger number of target subassemblies to be handled), the transport under gas could be possible. After irradiation, a relatively quicker time for removal could be envisaged, if the solution of handling under Na is chosen. If handling under gas is chosen, however, a decay period of ~12 months is necessary. In both cases, it will be necessary to cool down the subassemblies before cleaning.

The CEA study indicated that, without a revision of the regulation developed for Phenix and Superphenix, the handling of target assemblies could be a show-stopper. Consequently, the costs of the storage facilities could be reduced by the research and development of transport and management assemblies with a high decay heat and for the cleaning (washing) installation. The objective, namely to wash at higher temperature and quicker, could improve the sub-assembly design and avoid Na retention.

These handling difficulties with the highly radioactive *used* and *fresh* target fuels could affect reactor availability and could be a major issue for a nuclear utility. It is therefore important that additional evaluation of this issue be performed.

3.4 Systematic study of characteristics of heterogeneous versus homogeneous recycle

3.4.1 Homogeneous and heterogeneous recycle core designs

A previous ANL study has evaluated the impacts of heterogeneous recycle using Pu+Np driver and minor actinide target fuel assemblies in metallic fuel fast reactor cores by comparing results with those obtained for a reference homogeneous recycle core using driver assemblies containing grouped transuranic (TRU) fuel [13]. Parametric studies were then performed on the reference heterogeneous recycle core to evaluate the impacts of variations in the pre- and post-separation cooling times, target material type (uranium and non-uranium based), target amount and location, and other parameters on the system performance. The study focused on startup, single-recycle cores for the purpose of quantifying impacts. The core cycle length was estimated such that the burn-up reactivity swing is within the reactivity control capability of the primary control system. The discharge burn-up was determined by adjusting the fuel residence time such that the peak fast fluence is within the fast fluence limit imposed on HT-9 cladding and duct material (4×10^{23} n/cm^2).

The reference single homogeneous recycle startup core model was derived from an existing equilibrium cycle model for a 1 000-MWt sodium-cooled, advanced fast reactor core, which uses ternary U-TRU-Zr metallic fuel with a conversion ratio (CR) of 0.75. The

TRU material for the reference design was assumed to be recovered from LWR UNF that was irradiated to 50 GWd/t and stored for five years prior to reprocessing. The model was the basis for the development of the different heterogeneous recycle cores evaluated in the study. These heterogeneous recycle cores use both driver and target fuel assemblies.

For all cores, the heterogeneous recycle driver fuel is assumed U-(Pu+Np)-10Zr, while the target is assumed U-MA-40Zr fuel (MA is Am, Cm, and higher actinides); LWR-UNF TRU vector: ^{238}Pu =0.024, ^{239}Pu=0.50, ^{240}Pu=0.24, ^{241}Pu=0.11, ^{242}Pu=0.07, ^{241}Am=0.03, ^{243}Am=0.016, ^{244}Cm=0.006, ^{245}Cm=0.0004. The inclusion of Np in the driver fuel is based on the assumption that it simplifies the used fuel aqueous separations process as Np has a tendency to go with plutonium, and thus this option reduces the number of separations steps. It also decreases the quantity of the higher actinides arising from LWR UNF for which additional waste management solution would be required. Uranium-based target fuel is considered because European studies have indicated the difficulties associated with the use of non-uranium (e.g. inert matrix fuels). However, non-uranium targets were also evaluated to maximise the MA destruction rate (and avoid plutonium production in targets).

The Pu+Np content in the heavy metal of the driver fuel is determined to satisfy the condition, that is the core should criticality be maintained for the duration of the cycle. For the target, the MA content in the heavy metal (HM) is fixed; generally at 40% (depleted uranium, makes up the remainder) for the fertile fuel targets and 100% for the non-uranium targets. Recent CEA studies indicate that the 40% MA content in the heavy metal is probably not realistic, since post-irradiation and fresh-target assembly handling will definitely necessitate a lower content.

All the designs used three enrichment zones for power flattening, with relative enrichment ratios of 1.0, 1.25, 1.50, from the inner to the outer zone. All configurations consisted of 144 fuel assemblies (total driver and target assemblies). Figures 25 and 26 show the radial core layouts for the reference homogeneous recycle core and the heterogeneous recycle core with 12 targets assemblies on the core periphery. Other cores with 24 and 36 target assemblies on the periphery were also evaluated, along with a core with the 12 target assemblies distributed in the core interior. For all the cases evaluated, the cladding and duct material is HT-9, the duct flat-to-flat distance is 15.71 cm, the duct thickness is 0.394 cm, the active (fuel) height is 101.6 cm, and the metal fuel smeared density is 75%TD. The pin diameter, pin-to-diameter ratio and wire-wrap dimension were, however, allowed to change depending on the design goal.

Figure 25: Reference homogeneous layout

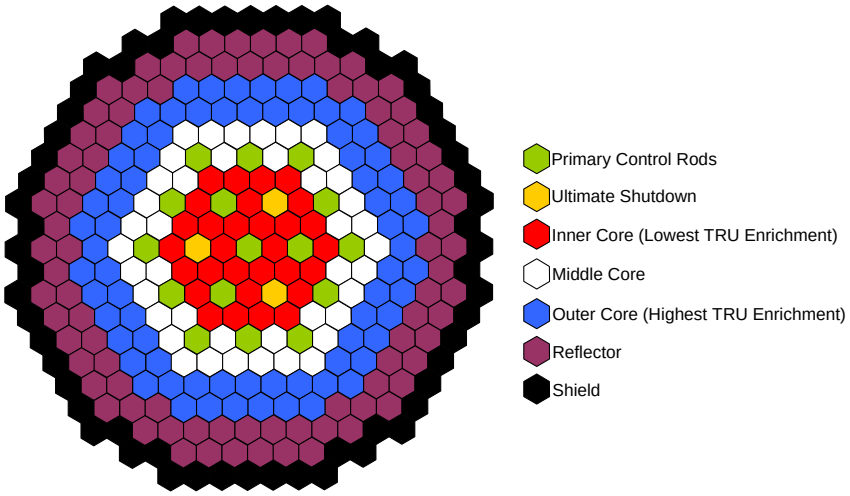
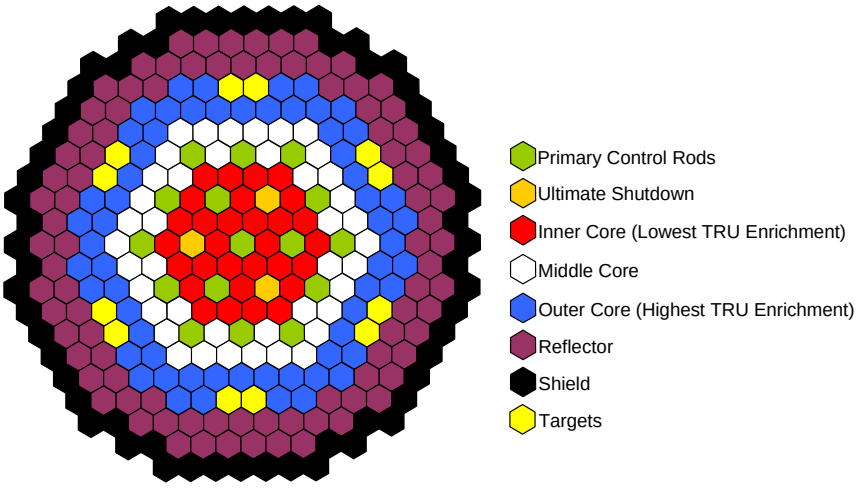


Figure 26: Reference heterogeneous layout (12 target assemblies)



3.4.2 Comparison of performance results

The performance results for the reference startup homogeneous and heterogeneous recycle cores are summarised in Table 13. The reference heterogeneous recycle core has 12 target assemblies, which contain higher TRU (all MA) loading, on the low reactivity-worth core periphery. Due to the higher zirconium content in the metallic alloy target fuel, the heavy metal loading of the target assembly is lower than that of the driver assembly. Additionally, the target assemblies are assumed to be shuffled to maximise their core residence time and remain in the core for more cycles (11 versus 8) than the driver fuel assemblies that they replace.

Both the heterogeneous and homogeneous recycle cores are seen to generally destroy transuranics. For the heterogeneous recycle approach, it has been found that a significant amount of the minor actinides is destroyed in the target assemblies. On the other hand, a significant quantity of minor actinides is produced in the driver assemblies, which are used for recycling the plutonium and neptunium (Pu+Np).

A significant difference arises from the potential coolant orificing requirements for the target assemblies in the heterogeneous recycle core. The results show a significant increase in the average power of the target assembly (a factor of ~2) from charge to discharge state because of the breeding of fissile material. The peak to average power within the targets is quite similar to that of the average core, though this gradient is developed over a shorter distance due to the positioning of the targets in the core periphery. Detailed thermohydraulic analysis would need to ensure adequate cooling of the target assemblies.

The results also indicate a significant accumulation of helium in the target assemblies because of the very high alpha-particle decay of the higher actinides. The helium concentration in the discharged targets is 1.586 gram per initial ton of HM (g/MTIHM) in target material compared with 12 g/MTIHM in the driver and 40 g/MTIHM in the homogeneous recycle fuel (driver) assemblies. The high mobility of helium will add substantially to the noble gas concentration in the fission gas plenum of the targets. This will present a significant design challenge to ensure the integrity of the fuel pin without an excessively large fission gas plenum. (See Section 3.3.6 on the impact of helium on target fuel performance and design.)

Table 13: Performance summary for startup homogeneous (homo) and 12-target heterogeneous (hetero) recycle cores

Parameters		Homo	Hetero
Average discharge burn-up, MWd/kg	Reactor	102	106
	Driver	N/A	106
	Target	N/A	109
TRU/HM in driver fuel, %		20.4	20.3
TRU conversion ratio	Reactor	0.750	0.737
	Driver	N/A	0.740
	Target	N/A	0.631
Fissile breeding ratio	Reactor	0.823	0.770
	Driver	N/A	0.529
	Target	N/A	2.108
Net minor actinide production, g-MA/g-TRU charged	Reactor	-0.007	-0.007
	Driver	N/A	0.013
	Target	N/A	-0.453
Net TRU actinide production g-TRU/g-TRU charged	Reactor	-0.115	-0.124
	Driver	N/A	-0.125
	Target	N/A	-0.106
Average discharge to charge power	Target	N/A	1.90
Peak to average power density	Reactor	1.69	1.75
	Target	N/A	1.79
Helium concentration at discharge, g/MTIHM	Driver	38	12
	Target	N/A	1.586

Note: The target is uranium based with 40% TRU in HM.

3.4.3 Reactivity parameters of heterogeneous recycle cores

For completeness of this discussion it is also instructive to consistently compare the core reactivity coefficients and kinetics parameters of heterogeneous recycle and homogeneous recycle cores that have been designed. The results of an ANL study are summarised in Table 14 [13]. The study was performed for a 1 000-MWt metallic fuel core with a TRU conversion ratio of about 0.75 and MA/HM value of 40% in the target of the heterogeneous target case. In the table, BOC and EOC mean beginning of cycle and end of cycle, respectively. Data values are provided for the reference homogeneous recycle core and heterogeneous recycle core using uranium (reference case) and unmoderated non-uranium based targets.

Table 14: Reactivity coefficients and kinetics parameters

Parameters		Start. homo	Ref. hetero	Non-uranium targets
Effective delayed neutron fraction	BOC	0.00354	0.00356	0.00348
	EOC	0.00349	0.00351	0.00342
Prompt neutron lifetime, μs	BOC	0.333	0.340	0.331
	EOC	0.335	0.341	0.332
Radial expansion coefficient, ¢/ºC	BOC	-0.321	-0.322	-0.328
	EOC	-0.334	-0.335	-0.341
Axial expansion coefficient, ¢/ºC	BOC	-0.069	-0.070	-0.071
	EOC	-0.072	-0.072	-0.074
Fuel density coefficient, ¢/ºC	BOC	-0.638	-0.640	-0.654
	EOC	-0.676	-0.680	-0.694
Structure density coefficient, ¢/ºC	BOC	0.066	0.062	0.066
	EOC	0.071	0.067	0.071
Sodium void worth, USD	BOC	4.87	4.51	4.93
	EOC	5.36	5.02	5.45
Doppler coefficient, ¢/ºC	BOC	-0.093	-0.098	-0.097
	EOC	-0.095	-0.100	-0.098
Reactivity swing, %Δk		2.00	2.22	2.04
Transient initiator (USD/rod)		0.35	0.39	0.37

Differences in the reactivity parameters are evident from Table 14, but none of these differences suggest large effects relative to the startup or equilibrium (not shown) homogeneous recycle cases. Most, but not all, of the cases have reactivity coefficients bounded by those of the startup and equilibrium homogeneous recycle cases. Consequently, it appears that heterogeneous recycle would not have a significant impact on the performance of the fast reactor. It is noted that this similarity in the reactivity coefficients and kinetics parameters between equivalent homogeneous and heterogeneous recycle cores have been observed in most studies reviewed [19], [22], but this trend depends on the TRU contents of the core designs.

However, greater operational margins might be required since power in the targets is from MA primarily and consequently, there is a larger uncertainty in the results arising from nuclear data uncertainty. Additionally, detailed safety analysis is required for the heterogeneous recycle option. This is necessary to evaluate the dynamics of transients that are initiated locally in the target regions, where the local reactivity coefficients and kinetics parameters are degraded relative to those of the driver fuel zone. The issue is whether there would be incoherent propagation of the transient scenario that could adversely affect core safety trends.

3.4.4 Comparison of radiological data for homogeneous and heterogeneous recycle fuels

Radiological characteristics of the used nuclear fuels from the two cores have been compared using the spontaneous neutron source, heavy metal gamma source and decay heat as bases. Results are summarised in Table 15. Comparisons were done for the fresh and discharge fuels after five years cooling.

The results in Table 15 show that the fresh (TRU) fuel for the homogeneous recycle core has ~150 times the neutron source strength and 4 times the gamma energy emission rate of the driver (Pu+Np) fuel for the heterogeneous recycle core. The decay heat is reduced by a factor of ~2 in the driver assemblies relative to that of the homogeneous recycle core. These parameters indicate a significant benefit for manufacture of the heterogeneous recycle driver fuel (Pu+Np) relative to the homogeneous fuel (TRU).

It is also observed that the neutron source strength in the targets is ~36 times that of the homogeneous fuel and over 5 000 times that of the heterogeneous recycle driver fuel. The gamma energy source is ~28 times the homogeneous fuel and 130 times the driver fuel. The decay heat in the target assemblies is also very high; nearly 140 kW/MT for the fresh target (over 20 times higher than homogeneous recycle fuel). A meaningful quantification of these differences would require detailed design analysis of the fuel fabrication facilities for the different feed streams and the relative throughputs of these streams. For the discharge state, the radiological properties of the driver and target fuels have been compared at five-year post-irradiation in the fast reactor. The results show that the discharge heterogeneous recycle driver fuel has higher neutron and gamma emission rates compared to the fresh fuel values, while the target has a slight decrease at this time point. The large increase in the driver quantities is due to the build-up of the minor actinides with irradiation.

The higher radiological indicators of the heterogeneous recycle target fuel relative to the heterogeneous recycle driver fuel or homogeneous recycle fuel suggest potential difficulties with handling the target fuel assemblies. These issues are discussed in subsequent sub-sections below.

Table 15: Radiological properties of fresh and discharge driver and target fuels

	Homogeneous	Heterogeneous	
Parameter	Driver	Driver	Target
Fresh fuel			
Spontaneous neutron source (n/s/MTIHM)	1.2E+10	8.1E+07	4.3E+11
Total gamma source (W/MTIHM)	6.4	1.6	174
Total decay heat (kW/MTIHM)	7.1	3.4	140
Discharge fuel			
Spontaneous neutron source (n/s/MTIHM)	1.4E+10	2.4E+09	4.1E+11
Total gamma source (kW/MTIHM)	1.29	1.48	1.67
Total decay heat (kW/MTIHM)	11.8	8.0	153.6

Notes: The target-to-driver assembly heavy metal mass ratio is 45%.
The initial heavy metal mass of the homogeneous recycle fuel assembly is 96 kg.The heterogeneous recycle driver and target assemblies initially have heavy metal masses of 96 kg and 43 kg, respectively.

3.4.5 Effect of LWR UNF cooling time

The impact of the LWR UNF cooling time prior to separation has been evaluated by using the driver and target isotopics resulting from three post-irradiation cooling periods: 5 (reference value), 15, and 30 years in the reference heterogeneous recycle core with 12 target assemblies. All other core design parameters of the reference design were retained.

A short pre-separation cooling period minimises the build-up of ^{241}Am from the decay of ^{241}Pu, but would not allow significant decay time for some of the higher actinides (e.g. ^{244}Cm), which are highly radioactive. The results indicated slight changes in the TRU conversion ratio,

decreasing with increasing cooling time. This resulted in a slightly higher TRU or MA destruction in the target assemblies of the core using the 30-year cooled fuel. For this fuel, the helium concentration at discharge is also ~21% higher than that of the reference core.

The results showed that allowing the LWR UNF to cool longer prior to irradiation in a heterogeneous recycle core would substantially reduce the technical challenges of producing the fresh target assemblies, with the neutron source and decay heat levels decreased by factors of 5.7 and 2.4 (difference between reference and 30 years cooled fuel cases). However, increased cooling period for the LWR UNF would lead to an increased fraction of target assemblies in the heterogeneous recycle core, for the recycling of the same quantity of LWR UNF in a given period of time.

Based on the study assumptions, twice as many target assemblies are required in a core using 30 years cooled fuel versus that using 5 years cooled fuel, if the intent is to use all MA in a single-recycle in the fast reactor. Subsequent passes would require significant increase in the number of target assemblies because the LWRs would continue to produce new TRU, which must be recycled along with the large fraction of TRU that remains in the discharged fast reactor fuel.

The long cooling time of the LWR UNF will build Am, and it might be preferable to separate the material into the Pu+Np (driver) and MA (target) streams. This separation would allow the target stream to cool longer prior to introduction into the heterogeneous recycle core. A case for which the target material was cooled for an additional 25 years post-separation was found to provide the advantage of a smaller target feed stream, while allowing significant reduction in the neutron emission rates and decay heat in the targets, compared to the reference heterogeneous recycle case (with zero year extended cooling time).

The extended cooling would require significant storage of minor actinides and a suitable storage form. Even though the isotopic mix in the targets is dependent on the extended cooling time, the performance of the targets in the heterogeneous core was found similar to that of the reference case. The results indicated that the neutron source in the fresh target assemblies will be reduced by nearly a factor of three by cooling the targets an additional 25 years primarily due to the decay of ^{244}Cm (18.1-year half-life). The heavy metal decay heat is half that of the reference case.

3.4.6 Study of homogeneous versus heterogeneous recycle with European Lead-cooled Reactor System (ELSY)

All of the results presented so far in this chapter have been for sodium-cooled fast reactor systems. A preliminary set of evaluations have been done by ENEA, as part of the activities of the current OECD/NEA expert group, to characterise homogeneous and heterogeneous TRU recycle approaches with the European Lead-cooled Reactor System (ELSY). ELSY is a 1 500-MWth MOX (U,Pu) fueled and lead-cooled pool-type fast reactor, developed within the 6th European Union Framework Programme, for the purpose of electric energy production in a competitive and safe design.

The wrapper-less (PWR-like open–square) assembly design has been used in the study. The reference core configuration (with MA in the initial fuel) is characterised by an equilibrium "adiabatic" core configuration that stabilises the minor actinide content during the equilibrium cycle. Homogeneous recycle cores with Pu-only (reference case) and TRU bearing fuels and heterogeneous recycle cores containing Pu-only (driver) and Pu+MA ("target") bearing fuels were considered with simplifying assumptions, since no optimisation of TRU transmutation in those cores was done. It was assumed that the ELSY reactor fuel uses TRU derived from the spent fuel of a PWR/UO_2 system initially fueled with uranium containing 4.5% ^{235}U, burned to 45 GWd/t and cooled for 15 years. In the homogeneous recycle cases using Pu-only or TRU bearing fuels, all the fuel assemblies in the core used the fuel type.

Table 16: ELSY core characteristics with 1% MA in homogeneous and 1%, 10% and 20% MA in heterogeneous recycle fuels and comparison to reference core without MA**

Parameter	Hom 1.0% MA	Δ(%)	Het 1.0% MA	Δ(%)	Het 10% MA	Δ(%)	Het 20% MA	Δ(%)
Pu Frac $_{aver.}$	18.853	+7.1	17.735	+0.74	17.891	+1.63	17.829	+1.28
MA/MA+Pu	5.335	/	5.347	/	16.267	/	28.129	/
MA_Mass	350.79	/	351.35	/	1221.24	/	2457.48	/
Pu Frac in periphery asssemblies	/	/	20.99	/	24.7	/	28.8	/
k_{eff}	1.02971	-0.04	1.02973	-0.03	1.03025	+0.02	1.03018	+0.01
aver. $\Phi_{v/cm2s}$	1.891E+15	-0.74	1.889E+15	-1.36	1.839E+15	-3.97	1.783E+15	-6.89
n-Source $_{n/s}$	1.312E+20	+0.23	1.312E+20	+0.23	1.320E+20	+0.84	1.330E+20	+1.60
P $_{peak\ FA\ (MW)}$	12.556	+1.02	12.850	+3.39	13.215	+6.32	13.919	+12.0
P* $_{Fract/Zone}$	0.264/0.339/ 0.397	+3(1)*	0.261/0.335/ 0.404	+ 2(1)*	0.253/0.322/ 0.425	-5(2)*	0.248/0.315/ 0.436	+8(3)*
BU* $_{EoL/Zone}$	76.7 /82.4 /75.5	+0.9(3)*	73.6 /81.6 /79.3	+4(3)*	67.5 /89.9 /86.6	+14(3)*	62.7 /77.8 /92.5	+21(3)*
Δk/kBU Equil	-0.00900	|21.7|	-0.01112	|3.22|	-0.00994	|13.5|	-0.00627	|45.4|
ΔM $_{MA\ Equil}$	- 4.8	- 115	- 0.6	-102	- 78.9	- 362	- 179.2	- 695
ΔM $_{MA\ EoL}$	- 0.9	- 101	+13.5	- 89	- 274.5	- 325	- 647.4	- 630
aver. $E_{n,(MeV)}$	0.3998	+ 1.2	0.3992	+ 1.0	0.4099	+ 3.7	0.4218	+ 6.7
β_{eff}	0.00314	-7.65	0.00322	-5.29	0.00296	- 12.9	0.00300	- 11.8
$\Lambda_{r\dot{i}s\_lifespan\ sec}$	1.0955 E-6	-6.30	1.0878 E-6	-6.96	0.9595 E-6	- 17.9	0.8521 E-6	- 27.1
$A_{D\ 1200->1\ 800\ K}$	-0.00429	|44.2|	-0.00592	|23.0|	-0.00456	|40.7|	-0.00393	|48.9|
$A_{D\ 1\ 200->600\ K}$	-0.00815	|9.69|	-0.00740	|0.40|	-0.00737	|0.81|	-0.00452	|39.2|
Δk/k $_{AC-Void}$	+0.04034	+ 4.26	+0.04006	+ 3.541	+0.04152	+ 7.32	+0.04767	+ 23.2

* (1), (2), (3): ELSY core fuel zones with the maximum % variation.

** Calculations were done with MCNPX / JEFF 3.1 NEA cross section libraries. The standard deviations quoted for the k_{eff}, reactivity worth's, and β_{eff} are of the order of ± 0.25E-3 to ±1E-3.

For the heterogeneous recycle cores, only the assemblies on the core periphery (third of three core zones) contain the minor actinides. Heterogeneous recycle cases using initial fuel having MA contents of 1%, 10%, and 20% were considered. Homogeneous recycle cases with no MA and with 1% MA in the initial fuel were evaluated. The loading for the cases were adjusted to ensure the same BOL criticality state. The performance results for the cases are summarised in Table 16. These results show small differences between the homogeneous recycle cores with no MA (not shown in Table 16, since reference for comparison) and that with 1% MA content in the homogeneous fuel, except for the Doppler effect, which shows a 44% difference. As expected, a significant MA transmutation rate is observed if the loaded amount is significantly higher than its

"equilibrium concentration" (adiabatic core configuration, about 1%). Increasing the MA loading in the core degrades ELSY core safety parameters.

Preliminary safety analysis also shows a worsening of the ELSY core safety performances, with an increase in MA loading. In particular, the degradation of the safety characteristics becomes critical as the transmutation rate becomes significant. The study, however, concluded that in order to meet the requirement of MA efficient transmutation in ELSY, without jeopardising the safety characteristics, some modifications of the core design would be necessary.

3.5 System and deployment issues

3.5.1 Fraction of cores in nuclear park

Using the core results for the 10% MA content heterogeneous recycle case, the CEA has evaluated the fraction of the fast reactors in the French nuclear park that would have to be loaded with target assemblies [30]. The scenario study considered the French nuclear park with a constant nuclear energy demand of 430 TWhe/year. The current nuclear park is assumed to be replaced between 2020 and 2050 by a mixed nuclear park: 66% of Generation III European pressurised reactors (EPRs) and 33% of Generation IV sodium fast reactors. From 2080 to 2100, the EPRs will be replaced by SFRs. The plutonium is recycled in the fissile part of the SFR core. The separation of the minor actinides at the reprocessing step will start in 2038. The minor actinides will be recycled in the radial blankets of the SFR from 2040 (10% content of MA). The results indicate that the minor actinides inventory can be stabilised with the heterogeneous mode of transmutation using minor actinides in the radial blankets of the SFR. It was concluded that having a minor actinide content of about 10% in the radial blankets in the target necessitates involving 100 % of the SFR in the park in the transmutation process.

A more recent study [9] indicates that the use of 2 rows of SFR radial blankets with a 20% content is necessary from 2040 to 2080 in order to limit at around 10 tons the interim minor actinides storage. After 2080, a single row will be enough to transmute minor actinides without an important storage.

3.5.2 Fraction of target assemblies in heterogeneous recycle core

A fundamental question for the use of the heterogeneous recycle approach is the fraction of the reactor assemblies that must be target assemblies in order to consume the amount of MA equivalent to its proportion in the LWR used nuclear fuel (UNF). This presumes that the equivalent amount of Pu or Pu+Np in that the same UNF would be utilised for the driver fuel. This question is readily answered by running systems dynamics tools with well-defined scenarios. However, the results to be reported here are based on simple ratios of masses. For the calculation, the data derived from the systematic study for the metallic-fuel heterogeneous recycle core of Section 3.4.2 have been used. That core has 144 driver and target assemblies. The reference heterogeneous recycle core shown in Figure 26 used 12 target assemblies, but was not optimised for mass ratio, due to symmetric core considerations for the design study. The actual number of target assemblies would have been ~9-10 assemblies.

The results with the simple mass ratio evaluations are presented in Table 17. Several data are presented for two sets of target contents. The first case assumes Pu+Np are in the driver and Am+Cm is in the target. In the second case, Pu only is in the driver and Np+Am+Cm is in the blanket. In the Table, data for the two cases are presented as function of the LWR UNF cooling time (5, 15 and 30 years). For the mass ratio evaluations, it is noted that the mass of a driver assembly is 96 kg (U-TRU-10Zr fuel assumed) while that of the target assembly is 43 kg (U-MA-40Zr fuel assumed), based on data from the reference heterogeneous recycle core calculations. The lower heavy metal loading in the target assembly is needed to compensate for the degradation in the thermo-physical

properties of the fuel with high TRU content. It is additionally assumed that these assembly masses will be the same for the second case containing Np+Am+Cm in the target assemblies. Other assumptions pertain to the required fissile enrichment and core conversion ratios which are presumed fairly constant (even though, they would be slightly different due to the change in neutron importance from relocating the Np to the target or from the change in composition due to the cooling time). It is also assumed that the core could be made critical using 132 driver assemblies as in the reference configuration. There would be slight differences since each of the new cores should be optimised in order to ensure core criticality and to meet burn-up requirements and controllability and thermal safety limits.

Table 17: Fraction target assemblies in heterogeneous recycle reactor

Case	Am+Cm in target			Np+Am+Cm in target		
	Number of assemblies		Fraction of target assemblies	Number of assemblies		Fraction of target assemblies
Cooling time (yr)	Driver	Target		Driver	Target	
5	132	9	0.06	132	17	0.11
15	132	15	0.10	132	24	0.16
30	132	21	0.14	132	31	0.19

Note: For the evaluation, the driver assembly has 96 kg of heavy metal and the target assembly has 43 kg of heavy metal. A TRU content of 20.4% is assumed in the driver assembly, and a MA content of 40% in the target assembly.

This simple analysis shows some very interesting trends. First, in order to use the proportions of Pu and MA in LWR UNF, the core must have target assemblies comprising 5% to 20% of the total reactor core assemblies. The higher end being for target fuel that has been cooled for 30 years and for which Np+Am+Cm is in the target. If Am+Cm only is in the target, then 14% of total assemblies are target assemblies for 30 years cooled fuel case. In any event, these are significant numbers of target assemblies.

Now consider a scenario in which the target assemblies are in the core periphery and thus have to stay longer in the core in order to better approach the fluence limit, to effectively burn the target fuel. Consequently, prior to the next recycle pass of the target (in the recycle case) or prior to the introduction of the next batch of target assemblies (in the burn-down case), additional MA would have accumulated from the processing of the LWR UNF. Thus, the mass of MA in the target assemblies would have to increase, or the number of target assemblies would have to increase to match MA production at the separations plant. This would increase the MA fraction in the reactor or the MA mass would accumulate in storage. Both options are unattractive.

Another point to note is that the reference study assumed a MA content of 40% in the heavy metal of the U-MA-Zr target fuel. The previous US, CEA and JAEA studies have indicated that this is problematic for fuel handling in the fuel cycle, and that MA contents as low as 10% to 20% might be reasonable. In this case, the required number of target assemblies would increase significantly and could more than double (even with a reduced Zr content giving the same heavy metal loading as the driver fuel). Care must be taken when comparing values of the MA content that is practical in the heavy metal of the target fuel pin. Due to the thermo-physical properties issue mentioned above, the driver and target pins of the metallic-fuel fast reactor have different masses, while this is not necessarily the case for the oxide fuel pin. The specific mass loadings in the oxide case would depend on the pin design, accounting for helium generation to ensure adequate fuel performance. If equal masses were used for the metallic driver and target fuel assemblies, the target assembly fractions shown in Table 17 would be reduced by roughly a factor of 2.

The final case study is for 8 core rows of assemblies, which contain the 144 driver and target assemblies of the reference heterogeneous recycle reactor design. If additional core space is required for 19 (=31-12) additional assemblies (e.g. for the case with Np+Am+Cm in the target and 30 years cooling of LWR UNF), then the assembly location in row 9 would be required; there are 6*(9-1) = 48 locations in row 9 and 54 assembly positions in row 10. Consequently, an additional row of assemblies is required. Conversely, if 112 (=4*31-12) target assemblies are required (for a 10% MA content in the heavy metal), then row 10 positions and some locations in row 11 would be needed to accommodate the assemblies. For these two scenarios, the core radial diameter should be increased by one row or two rows or advanced materials should be developed in order to minimise the number of reflector- and shielding-assembly rows. In any case, this would increase the cost of the reactor, though design modifications might be used to alleviate this cost increase. It is also possible that such an increase in the diameter would not be possible if the original design was constrained by diametric size to allow shipping of the vessel.

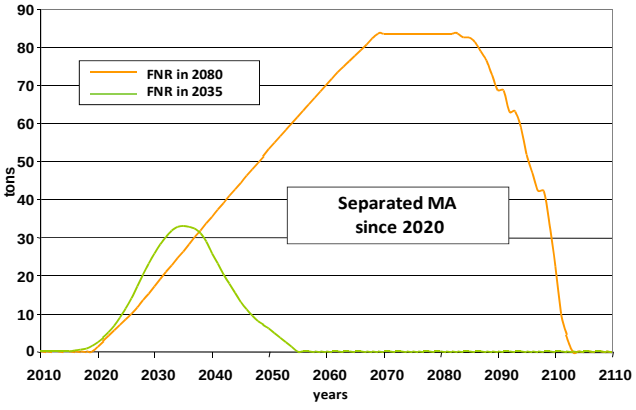
The foregoing discussion indicates that the use of the heterogeneous recycle might require target fraction that is high (within practical constraints) and penalise capital cost. This must be compared to fuel fabrication improvement to ascertain any fuel cycle benefit of the approach. These results show that the footprint of the target utilisation could be much higher than what is anticipated in the fuel cycle due to the difficulty of handling the target assemblies. Relaxing this target loading in the core would result in the accumulation of minor actinides in temporary storage, which is undesirable. The confirmation of these results using a systems dynamics code is recommended for any follow-on study.

3.5.3 Fast reactor deployment time

Some existing international nuclear fuel cycles utilise U-Pu mixed oxide fuel with the plutonium recovered from LWR used nuclear fuel. Minor actinides are included with the waste products of the LWR UNF separation process, with current intentions of entombing the waste in a nuclear waste repository. Alternatively, if the minor actinides were recovered from the used nuclear fuel for the purpose of being transmuted in a fast reactor, it is important to take into consideration the time between their recovery and their utilisation in the fast reactor.

A CEA study has evaluated the impact of this time on the time it takes to burn the initial material [10]. Figure 27 summarises the results. Two cases are shown in the figure for the purposes of illustration of trends. For both cases, it is assumed that the separation of the minor actinides starts in 2020. In the first case, it is additionally assumed that fast reactors are deployed at 2035, consistent with available fuel for powering the reactors. In the second case, the fast reactors are assumed deployed after 2070. The results show that the time required to consume the minor actinides is comparable to the time needed to deploy the fast reactors.

Figure 27: Time for minor actinides consumption



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4. Fuel studies related to homogeneous and heterogeneous recycle

4.1 Introduction

The homogeneous recycle approach can directly address the incorporation of all of the TRU elements in the transmutation fuel. The relative merits of the homogeneous and heterogeneous recycle approaches from the fuel perspective must be compared based on an assessment of the issues dealing with all the TRU elements together as in the homogeneous recycle approach (a new challenge); and separating the task into driver elements which are conceptually not a major extension of existing technology and fabrication and performance of targets containing Am-Cm. These issues require considerations of both separations technologies and approaches and reactor design issues.

In the homogeneous recycle approach, the TRU elements are separated from the irradiated fuel (LWR or SFR), in their natural relative concentrations. In the heterogeneous recycle approach the Np and Pu would typically be separated together, as would the Am and Cm. Therefore, in principle, the content of Np-Pu in the driver and Am-Cm in target elements are totally decoupled and determined by reactor performance and other considerations. Since a potential advantage of the heterogeneous recycle approach is the ability to relegate the problematic Am-Cm in the TRU to a limited number of fuel elements, the loading in the heterogeneous targets needs to be relatively high (10 to 40%) in order to limit the number of targets that need to be fabricated and handled and their impact on reactor performance.

The composition of the fuel (homogeneous/driver/target) is determined by the details of the spent fuel (type of reactor, discharge burn-up, cooling time, fuel form, etc), separations process and reactor design. The extraction process will determine which TRU elements will be removed from the irradiated fuel, whether they are co-extracted with some uranium and the order in which they will be extracted. These behaviours will affect the detailed composition of the feed material going into the fuel fabrication process. The specifics of the reactor design govern the composition of the fuel by setting the fissile loading required to achieve a desired cycle length and manage the spatial power distribution in order to remain within performance and safety parameters. These characteristics depend on the reactor power level, dimensions, conversion ratio, etc.

Regarding the homogeneous recycle mode, this chapter is mostly based on preliminary results of a review report of the OECD/NEA Expert Group on Innovative Fuels [1]. The analysis performed within that expert group covers all the fuel types envisaged in homogeneous recycle (oxide, metal, nitride) and the target type fuels envisaged for heterogeneous recycle. As for references, this report should be taken as a major reference that gives all detailed, specific references.

4.2 Summary on transmutation fuels development status

The issues related to fuel design, fuel performance and fuel cycle impacts of various innovative fuels are evaluated and compared in Table 18 [1]. Regarding fuel performance, R&D issues and open questions are summarised. Moreover, the level of qualification [in terms of a technological readiness levels (TRL)] and the validation data base are also specified. The detail of TRL is shown in the report of OECD/NEA Expert Group on Innovative Fuels [1].

4.3 Discussion

4.3.1 Homogeneous recycle

4.3.1.1 Oxide fuels

In the case of oxide fuel, the inclusion of minor actinides results in changes in some fuel properties and irradiation behaviour. The typical concerns of fuel properties are melting temperature, thermal conductivity and oxygen potential. For irradiation behaviour, the typical concerns are helium production, fuel-cladding chemical interaction related to fuel oxygen potential and fuel melting limit related to fuel thermal properties such as melting temperature and thermal conductivity. Minor actinide elements redistribution during the irradiation is also an issue of fuel melting limit.

As for fuel melting temperature, the presence of americium decreases the melting temperature of oxide fuel. Curium is considered to decrease the melting temperature, too. However, since its content is less than americium, the contribution of curium is limited. Americium also decreases the thermal conductivity of oxide fuel. These two kinds of contribution decrease the power-to-melt of oxide fuel, thus the maximum linear power is limited in lower value than oxide fuel without americium. The americium redistribution behaviour, which may increase the americium content and may result in the decrease of melting temperature at the centre of the fuel, should be taken into account in the fuel linear power limit evaluation.

Americium increases oxide fuel oxygen potential and may increase the cladding inner surface wastage due to fuel-cladding chemical interaction during the irradiation. Moreover, oxygen potential of oxide fuel decreases with O/M ratio. Therefore, fuel-cladding chemical interaction might be limited by reducing the fuel O/M ratio. It is important to understand the quantitative change of oxygen potential due to changes of americium content and O/M ratio of the fuel.

Helium production by alpha decay of minor actinide nuclide increases fuel pin inner pressure and may affect fuel swelling behaviour. The predominant alpha decay nuclide is ^{242}Cm built-up from the irradiation of ^{241}Am. The increase of the fuel pin inner pressure should be considered in the fuel pin design by means of fuel pin gas plenum volume increase, cladding thickness increase, etc.

Decay heat during the fabrication is also an issue of minor actinide-bearing fuel. Its major heat source is alpha decay of ^{244}Cm. ^{241}Am contributes to fresh fuel decay heat source through the nuclide chain as ^{241}Am (n,γ), ^{242}Am, beta decay to ^{242}Cm and alpha decay to ^{238}Pu.

Table 18: Comparison on design and performance of difference fuel types

Items \ Fuel for transmutation		Oxide fuel			Metal fuel		Nitride fuel	Target	
		Homogeneous MOX fuel		Minor actinide bearing blanket			Homogeneous nitride fuel	Dispersion fuel	
Design	AM total content	1-3%	3-8%	10-20%	1-3%	>10%	30-70 wt%	35 wt%	10 wt%*
	Pu%	20-30%	20-30%	0	0-30%	20-60%	30-40 wt%	9	10 wt%
	Am%	1-2%	3-5%	10-20%	1-3%	10-12%	20-30 wt%	17	5
	Cm%	0-0.2%	0-0.4%	0	0-0.3%	0-1%(?)	3-9 wt%	1.5	0.5
	Np%	0-2%	0-3%	0	0-3%	0-10%	25-35 wt%	17	5
	Other						(to AM total)	MgO:30%	Mo:80%
							ZrN: 30-70 wt%		
	Pu vector						51/13/23/2/11 (238/239/240/241/242)	Target	Target
	Support	$UPuO_X$	$UPuO_X$	UO_2	U-Pu-Zr	U-Zr or Zr	ZrN	MgO	Mo
	Fuel pin design	Std** with possible larger plenum	Larger plenum	*To be designed*	Std / Std with larger plenum	Std / Std with larger plenum	Std with larger plenum	Std with larger plenum	Std with larger plenum
	Fuel sub-assembly design	Std	Std	*To be designed*	Std	Std	Std	Std (to be confirmed)	Std (to be confirmed)
Fuel performances	Power rate	Std (450W/cm)	reduced	50 to 150 W/cm	Std (300-500W/cm)	Std or lower (300-500W/cm)	300-520 W/cm	Reduced in absence of Pu	Reduced in absence of Pu
	Burn-up	Std (~14at%) if FCCI is acceptable	reduced	>2000EFPD	Std (<20at.%) if FCCI is acceptable	Std or higher (>20at.%) if FCCI is acceptable	600 EFPD	10 (if He can be managed)	10 (if He can be managed)
	Safety criteria	1/ Effect of MA on properties → margin to melt 2/ Effect of FCCI → margin to clad failure	Idem + inner pressure increase due to He + Need of transients tests to check FCMI	*To be analysed (clad damage and wastage, transients,etc)*	FCCI (Margin to cladding failure)	FCCI (Margin to cladding failure)	FCMI at high burn-up or transient Center line temperature should be kept <2 000K at normal operation	Centre line temperature should not exceed 1 500 ^{0}C due to possible MgO vaporisation	Centre line temperature determined by Mo melting point. Very little known on He behaviour

* As taken from FUTURIX FTA FX 6. It is likely due to the high conductivity of Mo, that a far higher TRU content could be achieved. The absence of reliable data resulted in conservative margin to melt safety criteria for the FUTURIX irradiation of CERMET fuels.

** Std: Standard.

Table 18: Comparison on design and performance of difference fuel types *(continued)*

Items \ Fuel for transmutation		Oxide Fuel			Metal fuel		Nitride Fuel	Target	
		Homogeneous MOX fuel		Minor actinide bearing blanket			Homogeneous nitride fuel	Dispersion fuel	
Fuel performances	R&D issues open questions	Clad wastage at high burn-up?	Clad wastage at high burn-up?		Effect of Am and lanthanides on FCCI	FCCI behaviour of U-free and low-U fuel	FCMI behaviour transient scenario N-15 enrichment	FCCI Clad interaction at high BU	FCCI He and gas behaviour in a transient FCCI, FCMI
	Level of qualification (TRL)	4-5	3-4		4-5	4	3	3-4	3-4
	Validation data base	Superfact AM1 TRABANT AFC 2C and 2D	AM1 AFC 2C 2D	Superfact, In the future: MARIOS, DIAMINO.	X501 METAPHIX AFC-2A, -2B (under irradiation)	AFC-1B, -1F, -1H FUTURIX-FTA (irradiation completed)	AFC-1AE, -1F FUTURIX-FTA (irradiation completed)	ECRIX* FUTURIX** FTA	FUTURIX LWR DEPUTY HELIOS
Fuel Cycle Impact	Fabrication: process	Std	Std	Under development	Injection casting or gravity casting	Injection casting or gravity casting	Std	Std	Std
	Fabrication: facility	Main question: AM% limit to keep in glove box	Hot cells?	Hot cells?	Hot cell or glove box with inert atmosphere	Hot cell with inert atmosphere	Hot cell with inert atmosphere	Shielded or hot cell	Shielded or hot cell
	Reprocessing: process	Std test to be panned	Std test to be panned	To be tested	Electrometallurgical process	Electro-metallurgical process	Electro-metallurgical process	Possible, but residues may need more than 4M HNO3	Possible questions about Mo and indeed ^{92}Mo recycling
	Reprocessing: facility	Std	Std	Std	Hot cell with inert atmosphere	Hot cell with inert atmosphere	Hot cell with inert atmosphere	Std	Std but modified for ^{92}Mo recovery
	Cooling	Depend on MA%	Depend on MA%	Depend on MA%	Depend on burn-up (<1 year?)	Depend on burn-up (<1 year?)	~3 year	Depend on MA%	Depend on MA%
	Transport	Depend on MA%	Depend on MA%	Depend on MA%	Std or on-site fuel cycle	Std or on-site fuel cycle	On-site fuel cycle	Depend on MA%	Depend on MA%
	Level of qualification (TRL)	4-5	3-4		4-5	4	3	3-4	3-4
	Validation data base	Superfact reprocessing tests in ITU and CEA		Superfact reprocessing tests in ITU and CEA	EBR-II driver treatment, METAPHIX fuel pyro-processing test at ITU	None	None	Dissolution of ECRIX at CEA	None

* ECRIX fuel had a lower AmO2-x content.

** FUTURIX FTA had Pu in fuel two Am:Pu ratios were tested (50:50 and 70:30).

4.3.1.2 Metal fuels

MA addition to fuel alloys is considered to mainly affect the following thermal properties and irradiation phenomena of metal fuel:

- solidus temperature of fuel alloys;
- fuel-cladding chemical (or metallurgical) interaction (FCCI);
- liquefaction at the fuel/cladding interface;
- release of He gas generated by alpha-decay and transmutation of MA.

On the fuel performance, the technical issues in MA-bearing metal fuel are to understand and quantify the effect of MA addition on these thermal properties and irradiation phenomena by out-of-pile tests and/or irradiation tests.

The solidus temperature can be measured by thermal analysis of the fuel alloy samples. Some solidus data have been measured to date, but are not sufficient. More measured data will be necessary to quantify the solidus temperatures of MA-bearing fuel alloys in a wide compositional range.

The FCCI for MA-bearing metal fuel will be understood by out-of-pile tests with diffusion couples consisting of MA-RE alloys and Fe-base cladding materials. Irradiation tests of MA-bearing metal fuel with an appropriate cladding material are desired for quantification of FCCI.

Liquefaction at the fuel/cladding interface will be understood by out-of-pile tests with diffusion couples consisting of MA-bearing fuel alloys and Fe-base cladding materials. Heating tests of irradiated MA-bearing metal fuel with an appropriate cladding material are essential to quantify the rate of liquid phase attack.

Thermodynamic assessments of relevant alloy systems such as U-Pu-Zr-MA, MA-RE-Fe and U-Pu-MA-Fe will effectively assist understanding of the solidus temperature, FCCI and the liquefaction behaviour. He gas release behaviour in MA-bearing metal fuel is understood through the X501 and METAPHIX tests: He gas release is much higher than fission gas release. Further progress in post-irradiation examination of the METAPHIX fuel pins will be helpful for a better understanding of He gas behaviour. The nuclear data library of neutron reaction cross-sections of MA should be established as a basis of the estimation of He gas generation as well as MA transmutation rate.

As for fuel fabrication, the major technical issues in the fabrication of MA-bearing fuel are suppression of Am evaporation in the casting process. When a casting technology other than injection casting is applied, feasibility of remote and automatic operation should be confirmed to minimise radiation exposure from MA.

4.3.1.3 Nitride fuels

The advantages of nitrides are high actinide density, good thermal conductivity and high melting point, which harden the neutron flux and improve the safety margins. This results in a lower central fuel temperature. Alternatively, it allows a higher actinide content in the fuel. Furthermore, actinide nitrides show a mutual miscibility in a wide range of compositions, allowing great flexibility. A major drawback is the production of ^{14}C through (n, p) reactions on ^{14}N imposing the use of nitrogen enriched in the isotope ^{15}N. Even if it was shown that a relatively low enrichment level was needed in the second stratum, that level did not exceed the level produced in the first stratum.

The main concern is the limited thermal stability of AmN and the high vapour pressure of Am. Am redistribution in the fuel can be expected during irradiation, especially in the case of high linear power and high radial or axial thermal gradient. ZrN was early suggested as the best adjuvant to improve nitride stability, considering neutronic and thermo-physical properties. In addition, it crystallises in a cubic lattice,

which corresponds with the lattice of plutonium nitride so that solid solution is formed with minor actinide nitrides. Heating AmN and (Am,Zr)N in He atmosphere revealed that the dissociation of AmN starts at ~1 300 °C and selective vapourisation of Am in (Am,Zr)N occurred at higher temperatures.

The use of liquid metal as bond in the pin could be an interesting design solution to limit the fuel operation temperature and therefore the volatilisation and redistribution risk, but this solution makes more complex the fabrication and reprocessing processes. Anyway, fuel behaviour at high temperature, during power transient conditions, remains an issue to be addressed in depth.

The fabrication process is also significantly affected by Am and AmN vapourisation. Difficulties may arise because of the necessary temperature limitation of the heat treatments during nitride synthesis or sintering under the vapourisation temperature to avoid Am losses.

R&D on MA nitride fuel is still at a basic level though an important progress has been recognised. More progress of the laboratory scale experimental study is necessary by using MA, including Cm, which should contribute to the design and operation of the engineering plant. Development of economical 15N-enrichment process and measures against heavy decay heat is also essential. On the other hand, the fundamental study on He behaviour in MA-bearing fuel is important not only for MA nitride fuel but also for the other fuel concepts for transmutation of MA.

4.3.2 Target fuel for the heterogeneous recycle

Development of Am/Cm targets for a fast reactor is expected to be a more complex endeavour. Currently no fully proven technology exists for fast spectrum Am/Cm targets.

4.3.2.1 Oxide fuel targets

In the initial studies of the heterogeneous recycle the fertile component (^{238}U) used as matrix was removed from the fuel and replaced by a neutronically inert matrix, and the resulting fuels were named inert matrix fuels (IMF). The matrix itself serves to dilute the transuranium material, so that acceptable power levels and fuel operating temperatures can be reached. A number of materials were proposed. Most were oxides [e.g. magnesium aluminate spinel ($MgAl_2O_4$), yttrium stabilised zirconia ($(Zr,Y)O_2$)]. Other materials considered were zirconium and titanium nitrides, SiC and various metals (Mo, Cr, and V). Apart from their neutronic inertness, these materials should possess specific properties, consistent with in pile reactor performance. Swelling due to neutron, irradiation should be low, their thermal conductivities should be at least as high as UO_2 and their mechanical properties should not induce unacceptable strains when in contact with cladding materials. Furthermore, the inert matrix materials should not react either with the fuel, the cladding material or the reactor coolant (e.g. Na, Pb, or Pb/Bi). They should be readily available and pose no economic penalty.

Concerning their interaction with the fissile phase, the inert matrix should not form a low temperature eutectic, compound a low melting point, or a non cubic crystallographic structure. Indeed, their interaction with the actinide phase is a design criterion. Yttrium stabilised zirconia (YSZ) has the same crystallographic structure as the actinide dioxides, enabling their incorporation into the crystal lattice to form a solid solution, i.e. a single ceramic (CER) phase is obtained. In contrast, the fuel or target can be designed as a composite, in order to tailor its properties. Dispersion of the (mixed) actinide oxide in a second material with a higher thermal conductivity is a convenient means to increase the overall thermal conductivity of the sample. Ceramic-ceramic (Cercer) and Ceramic-metal (Cermet) configurations have been studied.

For composite fuels, the size of the fissile-bearing particles can be chosen to optimise the material behaviour during irradiation. While the fissile component will be subjected to significant neutron, fission product α-recoil damage, the matrix itself will only be affected by

neutron irradiation. A thin shell (~12µm) around the fissile particle will be subjected to damage. For a given volume fraction (%vol), the proportion of the matrix represented by this shell varies dramatically with particle size. For large particle sizes (e.g. macrodispersion), the majority of the matrix is only affected by the less harmful neutron damage. In contrast, composites with small particles (i.e. microdispersions) result in near complete matrix damage approaching that of the most extreme microdispersion, i.e. that of the solid solution.

Early attempts, such as the EFTTRA-T4 experiment, where the selected matrix was magnesium aluminate spinel ($MgAl_2O_4$), resulted in unacceptable fuel irradiation behaviour.

Testing of several inert matrix fuels currently under study will provide additional fuel performance information (MgO-based, molybdenum matrix, nitride, and metal).

In fact, within the inert matrix fuel (IMF) concept, the choice of fuel matrix is not only dependent on the irradiation feasibility, but also on the transmutation strategy. Highly insoluble matrices, such as magnesium aluminate spinel or zirconia, are ideally suited to "once through and then out (OTTO)" strategies, whereby a single irradiation to ultra high burn-up is followed by direct geological disposal. Achievement of such high burn-ups is problematic, due to the long irradiation times and concomitant irradiation damage of fuel and its cladding material. Thus multi-recycling of the fuel must also be considered, in the case where soluble matrices are required. In general, aqueous routes based on the PUREX process are advantageous. MgO is a good candidate as it is readily soluble under such conditions. It also has a higher thermal conductivity than UO_2 and a reasonably high melting point (2 850 °C).

Table 19 [1] gives a summary of the oxide matrix irradiations and their status.

4.3.2.2 Targets with UO_2 matrix

In view of the reduced number of options in the case of an inert matrix support for the target and also in view of a number of not yet fully satisfactory results for the remaining inert matrix concepts (i.e. MgO and Mo based), the use of a UO_2 matrix has been advocated and further studies have been initiated.

The preliminary thermo-mechanical and thermal-hydraulic designs of the minor actinide bearing blanket (MABB) and (UO_2 matrix) concept positioned on the periphery of the SFR core provided favourable indications towards the feasibility of such assemblies containing about 7% to 20% MA with 0.50 mm thick ODS cladding based on a series of strong assumptions, which will require careful experimental validation. A minor actinide concentration above 20% is currently not considered feasible for manufacturing and handling reasons (very high thermal power). It should nevertheless be mentioned that integrating these assemblies into the reactor may affect the flattening of the core outlet temperature range. The (Ux MAy) Oz fuel is supposed to be characterised by an optimised microstructure designed to promote the release of significant quantities of helium. This microstructure is presumed to limit helium-induced swelling under irradiation, though this has not been modeled in this preliminary design phase due to the lack of experimental data. This optimised microstructure is based on a fuel with stable open porosity under irradiation, though its manufacturability remains to be demonstrated.

Table 19: Inert matrix oxide fuels irradiation programmes and their status

Fuel form	Composition	Reactor	Programme	Status
				(may 2010)
CER	$(Zr, Y, Am)O_2$	Phénix	CAMIX	Irradiated
	$(Zr, Y, Am)O_2$	HFR Petten	HELIOS 2	Irradiated
	$(Zr, Y, Pu, Am)O_2$	HFR Petten	HELIOS 3	Irradiated
Cercer	$MgAl_2O_4$ - $AmAlO_3$	HFR Petten	EFTTRA T4	PIE complete
	MgO - AmO_2	Phénix	ECRIX- B	Irradiated
	MgO - AmO_2	Phénix	ECRIX- H	Irradiated
	MgO - $(Zr, Y, Am)\ O_2$	Phénix	COCHIX	Irradiated
	MgO - $(Pu, Am)\ O_2$	Phénix	FUTURIX 7	Irradiated
	MgO - $(Pu, Am)\ O_2$	Phénix	FUTURIX 8	Irradiated
	MgO - $Zr_2Am_2O_7$	HFR Petten	HELIOS 1	Irradiated
Cermet	Mo - $(Pu, Am)\ O_2$	Phénix	FUTURIX 5	Irradiated
	Mo - $(Zr,Y,Pu,Am)O_2$	Phénix	FUTURIX 6	Irradiated
	Mo - $(Pu, Am)\ O_2$	HFR Petten	HELIOS	Irradiated
	Mo - $(Zr,Y,Pu,Am)O_2$	HFR Petten	HELIOS	Irradiated

Based on target values for the composition, temperature and neutron flux of this concept, information on the behaviour of these fuels under irradiation (and helium production) is very limited and remains to be characterised within the scope of an experimental programme. The following aspects need to be explored in particular: i) fuel properties and their variations under irradiation, ii) free swelling and stress, and iii) gaseous releases during steady-state and transient conditions, including the mid-cycle rotation transient.

4.3.2.3 Metallic target fuels

Metallic target fuels can be fabricated by powder metallurgical techniques using material extracted from the pyrochemical processes. The metallic products from the pyrochemical processes are hydrided and then dehydrided to produce metal powders. Powder metallurgy is a common, well-developed industrial technique for making metal shapes in order to exact standards of density (>95% theoretical density) and dimensions (+0.001 in.). However, the use of powder metal techniques to fabricate the U-MA-Zr target should be investigated. This has the advantage that no melting of the metals is required. All components can be hydrided and dehydrided under easily attained temperatures and pressures in order to produce metal powders that can be blended, cold pressed and then sintered to produce a strong metal form with sufficient porosity to release helium and suitable for long-term storage or neutron irradiation.

Because americium has a very low solubility in uranium, melting of these components will produce a two-phase alloy with an americium-rich phase unless americium is soluble in Zr-U alloys. The powder metal technique can overcome this

problem by producing a homogeneous structure with an Am-Zr phase well dispersed in a U-Zr phase.

4.3.3 General issues associated with minor actinide-bearing fuels development

Minor actinide bearing fuels have characteristics such as high dose/decay heat, high production of He and degraded thermodynamic properties which are more or less different from (U,Pu) fuels. These characteristics are the source of disadvantages and issues of minor actinide-bearing fuels in terms of certain R&D requirements and fuel less effective processes/performance themselves. In terms of fuel recycle scheme, such fuel issues are fuel fabrication matter, fuel design/irradiation performance matter and fuel transportation/handling matter. It is pertinent to discuss fuel issues from these aspects in order to have a common understanding of the issues.

In the fuel fabrication process, the main disadvantage comes from high dose and decay heat, and fuel thermodynamic properties. The main disadvantage of fuel thermodynamic properties is the lack of expertise concerning research activity requirements. On the other hand, the major disadvantages come from high dose and decay heat, which will lead to the modification of the fuel fabrication process and systems such as requirements of full remote fuel fabrication in heavy shielding facility, for example hot cells, the small fuel sub-assemblies size to limit the sub-assembly temperature during fabrication and reduced scale of mass production. With the heterogeneous concept, decay heat problem becomes more significant than that of the homogeneous concept since MA content is high. Drawbacks of fuel fabrication are important from the viewpoint of single sub-assembly. On the other hand, since the number of fabricated sub-assemblies with certain decay heat is smaller in the heterogeneous concept, a combined drawback comparison of decay heat level and number of sub-assemblies with decay heat will give appropriate information of relative advantage of the concept.

The high vapour pressure of Am and Cm related gas species could be of additional concerns for the heterogeneous concept (in the case of high Am and Cm contents). In the case of the homogeneous concept, low chemical activity of Am and Cm due to low content in oxide may limit the vapour pressure of those gas species during the fabrication. However, oxides with high Am and Cm content may present some problems with of vapourisation losses during fabrication.

In fuel design and irradiation performance, the main parameters to be considered are fuel thermal performance degradation due to degraded thermodynamic properties, fuel pin internal pressure increase due to He production and possible increase of cladding inner surface wastage due to high oxygen potential. Change in fuel design is more significant due to fuel pin internal pressure increase than due to fuel thermal performance degradation and irradiation data are needed to identify the significance of MA contribution to fuel cladding inner surface wastage. Regarding fuel thermal performance and fuel pin internal pressure increase, the contribution of MA becomes significant with MA content of the fuel. Therefore, the heterogeneous concept has more drawbacks in these matters. A fuel pin design study showed that fuel pin design change is limited in the homogeneous concept and it is rather significant in the heterogeneous concept using driver fuel region. This difference is further enhanced by the difference of power histories of homogeneous concept fuel and heterogeneous concept fuel. In the case of the heterogeneous concept fuel, since fissile nuclides generation during irradiation is significant, fuel pin power tends to increase with irradiation. This results in the high fuel temperature and high cladding temperature at the end of fuel pin life, which gives a disadvantage to fuel pin design. High fuel cladding temperature at the end of fuel life also presents a problem for core thermal hydraulics. High MA content fuel in heterogeneous concept requires high coolant flow rate to limit the cladding temperature at the end of fuel life. This is a common concern for both in-core loading and core peripheral loading of the heterogeneous concept.

In fuel transportation and handling, sub-assembly temperature due to the decay heat of MA nuclide is the major concern. This is similar to the decay heat problem in fuel fabrication. Thus the heterogeneous concept has a more significant disadvantage than the homogeneous concept from the viewpoint of single sub-assembly heat removal problem. On the other hand, since the number of fabricated sub-assemblies with certain decay heat is smaller in the heterogeneous concept, combined drawback comparison of decay heat level and number of sub-assemblies with decay heat will give appropriate information on the relative advantage of the concept.

Performance of Am-bearing fuels and targets is driven by the production of large amounts of helium gas, approximately 50 cm^3 per gram of Am transmuted, from the americium neutron capture and decay sequence. This high gas production rate should be carefully accounted for, in particular in the fuel/target design.

Fabrication of Am/Cm targets would occur in a hot cell, presenting the same set of issues as fabrication of U-Pu-Np-Am-Cm fuel. Development of Am/Cm targets would require several years of scoping irradiations to reach the current state of development of U-Pu-Np and U-Pu-Np-Am fuels.

It should be noted that the level of difficulty in the fabrication and performance of Am/Cm targets is directly related to the loading. However, if sufficient high loadings are not attainable, the potential benefits of the heterogeneous recycle will be decreased. A large number of targets will be needed in order to avoid negative impacts on reactor performance and the size of the target production facility should be optimised in order to satisfy the goal of the transmutation.

In this respect, high specific heat generation associated with the target of the heterogeneous recycle presents a specific difficulty, which should be solved in order to make the target fabrication attainable.

Considering the MA target, the minor actinide target should be cooled for a longer time or the MA content of the target pin be reduced significantly to reach an equivalent heating rate. These two approaches could make the heterogeneous recycle less attractive as they effectively reduce the loading of the MA in the core or increase the number of target assemblies in the core.

4.3.4 Summary

In summary, a number of general statements can be made in order to provide some guidance in the homogeneous versus heterogeneous recycle mode comparison.

In fact, from the fuel point of view, the homogeneous way is a rather robust route, consolidated by recent complementary experimental programmes with respect to the pioneering SUPERFACT experiment for oxide fuels.

Mainly, demonstrations on the scale of a fuel pin bundle (in capsules) or on that of an assembly are missing. The objective of such demonstrations should prove that ~5% MA loading in the fuel does not affect safety and economic performance.

Today, no experiment exists yet that could allow researchers to assert that the presence of minor actinides in the fuel does not reduce the end of life of the fuel. Furthermore, difficult access to fast reactors in the world will not allow such a demonstration to be carried out for at least another ten years. It is within this context that the Idaho National Laboratory in the United States is currently conducting a complementary programme to SUPERFACT and AM1 in the ATR thermal reactor. Its objective is to reach a higher burn-up of 15 to 20 at%. Irradiation is designed to approach the conditions of an irradiation in rapid flux. Normally, this experiment should provide the lacking elements pertaining to cladding resistance to internal corrosion and the behaviour of transmutation fuel with high burn-up.

Furthermore, the international GACID project (Global Actinide Cycle International Demonstration), to be carried out within the framework of the Generation IV International Forum and aimed at the demonstration of transmutation in the homogeneous mode on the scale of an assembly in the Japanese MONJU reactor, could provide important demonstrations. Development of Am/Cm targets for a fast reactor is expected to be a more complex endeavour.

It should be noted that the level of difficulty in the fabrication and performance of Am/Cm targets is directly related to the loading. However, if sufficient high loadings are not attainable, the potential benefits of the heterogeneous recycle will be decreased. A large number of targets will be needed in order to avoid negative impacts on reactor performance and the size of the target production facility should be optimised in order to satisfy the goal of the transmutation. The state of development of Am/Cm target technology is low relative to both U-Pu-Np and U-Pu-Np-Am fuels due to the availability of data from previous successful fast reactor irradiation tests. Am/Cm target technology for utilisation in a sodium-cooled fast reactor thus currently lags several years behind U-Pu-Np and U-Pu-Np-Am fuel development.

The risk associated with the development of U-Pu-Np-Am-Cm fuels is certainly higher than for U-Pu-Np fuel. Moreover, high risks can also be expected for deployment of a transmutation system that requires separate U-Pu-Np driver fuels and Am/Cm targets. Their relative merits should be confirmed by the ongoing irradiation programme and the few supplementary irradiations foreseen.

At any rate, it can be stated that the development of both U-Pu-Np-Am-Cm fuel and Am/Cm targets will require the use of remote R&D facilities to produce fuel for testing and to develop and validate remote fabrication processes. The physical infrastructure requirements for development of a fuel/target system are thus roughly the same as for a homogeneous U-TRU system. Overall fuel development resource requirements are probably larger for the driver fuel/target system due to the need for two distinct fuel development programmes – one for driver fuel (equivalent to that required for U-Pu-Np-Am-Cm) and one for Am/Cm targets.

In balance, the development of a transmutation system based on U-Pu-Np driver fuel and Am/Cm could arguably present a higher development risk and could require more resources than the development of a system that uses a single U-TRU fuel. Am/Cm target technology lags behind U-TRU fuel by several years, which may equate to a schedule delay for qualification of Am/Cm target technology relative to U-Pu-Np and U-Pu-Np-Am-Cm fuels.

However, in view of the reduced number of options in the case of an inert matrix support for the target, and also in view of a number of not yet fully satisfactory results for the remaining inert matrix concepts (i.e. MgO and Mo based), the use of a UO_2 matrix has been advocated as a more promising option.

At any rate, in the case of the heterogeneous concept, decay heat problems become more significant than with the homogeneous concept since MA content is high. Therefore, the drawback of fuel fabrication is significant from the aspect of single sub-assembly. On the other hand, since the number of fabricated sub-assemblies with certain decay heat is lower in the heterogeneous concept, comparison of combined disadvantages from decay heat level and the number of sub-assemblies with decay heat will give appropriate information on the relative advantage of the concept with respect to the homogeneous recycle mode, which by definition, implies the presence of MA-bearing fuel in the whole core.

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5. Parameters and criteria for homogeneous and heterogeneous modes comparison

Several parameters have to be analysed in order to establish a global assessment of the advantages and drawbacks due to minor actinides recycling (Np, Am or Cm) modes. Recent publications present multi-criteria analysis. From these analyses it can be concluded that multi-criteria approaches weighting different parameters are difficult to implement and conclusions are not drawn clearly due to a too large scope of parameters and criteria, quantitative or qualitative.

The purpose of this chapter is to focus on some criteria and on main issues in reactor and fuel cycle linked to homogeneous or heterogeneous options based on the results presented in the previous chapters. As stated in the Generation IV Initiative, global criteria are sustainability, preservation of natural resources, safety, waste minimisation, and proliferation resistance. From them, we have considered the following classes for our analysis:

- technical criteria:
  - for the fuel cycle;
  - for the reactors;
  - for the waste disposal;
- radiological and health impacts;
- knowledge and R&D;
- non-proliferation and safeguards.

The comparison of the technical criteria between the homogeneous and the heterogeneous options has been performed in terms of plutonium recycling. Table 20 shows nuclear fuel cycle parameters including fabrication and spent fuel treatment issues. The main conclusions are:

- Demonstrations were performed for fabrication, irradiation and treatment for both homogenous and heterogeneous cases.
- Fuels are not fully validated for homogeneous and heterogeneous. Homogeneous fuel behaviour is closer to the current knowledge on Pu fuel. Management of helium production is a key issue for the heterogeneous mode. Dedicated irradiation tests with curium have to be performed.
- Processes (fabrication – spent fuel treatment) have been only validated.
- Material management flexibility due to a separated plutonium management in the case of the heterogeneous case.
- Higher thermal load of fresh and irradiated fuels in the case of the heterogeneous case compared to the homogeneous case → Impacts on fuel management and transportation.
- Higher mass fluxes of sub-assemblies with MA with the homogeneous mode compared to the heterogeneous mode.

Table 20: Comparison on nuclear fuel cycle issues

	Homogeneous (all MA)	Heterogeneous	Homogeneous/heterogeneous comparison
Technical complexity for fuel fabrication	One dedicated facility for fuel fabrication Hot cell for fuel (neutron source is about 10^9 n/s per sub-assembly)	Two dedicated facilities: one for standard fuel; one for MA target fabrication Glove box for fuel & hot cell for MA target (MA target neutron source is 5.10^9 to 10^{10} n/s per sub-assembly)	Two facilities for heterogeneous mode but standard Pu fuel fabrication is not modified. Give some flexibility in case of difficulties on MA fabrication Glove box or hot cell for standard Pu fuel depends on isotopic composition of Pu (neutron source is about 10^7 n/s per subassembly for Pu isotopic composition as 3% ^{238}Pu, 55% ^{239}Pu, 30% ^{240}Pu, 5% ^{241}Pu, 6% ^{242}Pu and 1% ^{241}Am)
Maturity of the processes	Preliminary test: performed Laboratory scale: done (fabrication – treatment) Development scale: not planned Industrial scale: not planned	Preliminary test: performed but not for curium Laboratory scale: done (fabrication – treatment) but not for curium Development scale: not planned Industrial scale: not planned	Same level of maturity except for Cm in heterogeneous mode
Irradiations performances behaviour	Some irradiations and PIE done (w/o Cm)	Some irradiations and PIE done (w/o Cm)	Preliminary tests are performed for both homogeneous and heterogeneous: MA content objectives are reached but not for burn-up objectives. Dedicated irradiations with curium have to be performed.
Thermal load of fresh fuel	1.5 kW/ass. (First load with 4% MA) (Equ with 1% Am 0.2% Cm)	2.5 kW/ass. (20% Am) to 7.5 kW/ass. (20% MA) 4 kW/ass. (10% MA)	Larger values for heterogeneous mode compared to homogeneous mode. To respect current limit of 6 kW/ass, some limitation on MA content (lower than 20%)
Thermal load of irradiated fuel (after 5 years cooling)	3.0 kW/ass. (First load with 4% A.M) (Equ with 1% AM 0.2% Cm)	5 kW/ass. (10 %) to 10 kW/ass. (20 %)	Higher values for heterogeneous (compared to Pu recycling alone = 1.5 kW/ass.)
MA total inventories in the fuel cycle Reactor + Cycle (7 years)	1.2 t/GWe	1.8 t/GWe	Higher MA inventories for heterogeneous mode
Sub-assemblies with MA mass fluxes	7.5 t/GWe.year	1.0 t/GWe.year (20%MA) 2.0 t/GWe.year (10%MA)	Larger mass fluxes for homogeneous mode

Table 21: Comparison on reactor issues

	Homogeneous	Heterogeneous	Homogeneous/ heterogeneous comparison
Impact on reactivity coefficients (versus sole Pu fuel) Change in: Doppler - void effects, burn-up swing	Impact is about 10 to 20% for MA content < 5%	- No impact, no fuel driver zone modification for targets in breeder zones - Power in target zone has to be managed by a coolant flow optimisation	- For heterogeneous mode no impact on core reactivity coefficients - For homogeneous impact limited for MA content < 5%. Upper level requires to re-design the core
Safety (versus only Pu fuel)	- Probably no additional severe accident initiators - Detailed safety analysis done but not fully completed - Safety validation experiments needs	- Some sequences must be take into account: local void effect due to large helium source for example - No detailed analysis performed - Safety validation experiments needs	More advanced safety studies performed on homogeneous mode compared to heterogeneous
Thermal load of irradiated fuel: time to reach limits with current process: (Handling 40 kW/ass.) Clean-up 7.5 kW/ass.)	- Time to handling: 5 days - Time for clean-up: 1 year	-Time to handling: 10% Am: 1 day 20% Am: 50 days 10% MA: 3 days 20% MA: 50 days - Time for clean-up: 10% Am: 1.5 years 20% Am: 5 years 10% MA: 2 years 20% MA: 15 years	Higher values for heterogeneous mode compared to homogeneous mode. To respect limits with current technologies needs some limitation on MA content (lower than 20%)
Transmutation performances	5 kg/TWhe at equilibrium	5 kg/TWhe at equilibrium	Similar
MA in core inventory	First load: 2.0 t/GWe Equil.: 0.5 t/GWe	For 20%MA = 2.4 t/GWe For 10% MA = 1.2 t/GWe	Higher MA inventory in reactor for heterogeneous mode.

Table 21 shows reactor issues. The main conclusions are:

- No impact on core parameters for the heterogeneous case compared to plutonium recycling. Impact on shielding design must be checked.
- Design optimisation is required for the homogeneous mode.
- New experiment safety validation is required.
- Equivalent transmutation performance.

- Higher MA mass inventories in core for the heterogeneous mode.
- Handling limitations for high MA content for the heterogeneous mode.

Table 22 shows radioactive waste disposal issues. Globally, no differences between the two modes, homogeneous and heterogeneous, are noted about impact on disposal compared to Pu recycling alone. Some gains are real on the number of wastes packages, on thermal load, on activity and radiotoxicity of wastes and on disposal size but at the same level for both homogeneous and heterogeneous modes. No gain on radioactivity release into the biosphere is expected because it is dominated by long-lived fission products for all different types of storage. Only for tuff type site is a contribution of Np noted.

Table 23 shows radiological and health impacts on normal fuel and reactor operations issues and Table 24 shows R&D needs on ractor and fuel development.

Table 22: Comparison on radioactive waste disposal issues

	Homogeneous	Heterogeneous	Homogeneous/ heterogeneous comparison
Number of canister to be stored (versus sole Pu recycling = 12 canisters/GWe.year)	≈ 11 waste canisters/GWe/year	≈ 11 waste canisters/GWe/year	Similar
Thermal load (versus sole Pu recycling) at 100 and 1 000 years	Gain about factor 6 at 100 years to factor 150 at 1 000 years. Gain are reduced to factor 4 at 100 years and factor 15 at 1 000 years if considered Am recycling alone	Gain about factor 6 at 100 years to factor 150 at 1 000 years. Gain are reduced to factor 4 at 100 years and factor 15 at 1 000 years if considered Am recycling alone	Similar (Thermal load at short term are dominated by fission products)
Radiotoxicity (versus sole Pu recycling) at 300 years to 10^6 years	Gain about 10 to 100 in function of time	Gain about 10 to 100 in function of time	Similar
Activity (versus sole Pu recycling) at 300 years to 10^6 years	Gain about 10 to 100 in function of time	Gain about 10 to 100 in function of time	Similar
Storage size (versus sole Pu recycling)	Gain of 10% on number of canisters to be stored. Gain on heat load depends on type of storage	Gain of 10% on number of canisters to be stored. Gain on heat load depends on type of storage	Similar
Radioactivity releases to biosphere (versus sole Pu recycling)	Depends on type of deep storage: gain on reduction of Np release for tuff site as Yucca Mountain	Depends on type of deep storage: gain on reduction of Np release for tuff site as Yucca Mountain	Similar

Table 23: Radiological and health impacts issues

	Homogeneous	Heterogeneous	Homogeneous/ heterogeneous comparison
Normal operation doses (storage – fuel cycle facilities – transportation)	Neutron source at fabrication stage increased by factor 100 compared to Pu recycling	Neutron sources at fabrication stage increased by a factor 100 to 1 000 compared to Pu recycling No constraints for 4/5th of total fuel assemblies (Pu driver zone)	Higher intrinsic constraints for heterogeneous but concentred on 1/5th of total fuel assemblies. Fabrication technical processes have to respected limits doses

Table 24: R&D needs on reactor and fuel development

	Homogeneous	Heterogeneous	Homogeneous/ heterogeneous comparison
Reactor	- Improved nuclear data - Reactivity coefficients variation must be validated	- Improved nuclear data - Targets power evolution and cooling must be validated - Handling process must be optimised	Some specific and different R&D points must be studied
Fuel	- Tests with Cm content - Fuel behaviour under transients accidents	- Tests with Cm content - Targets behaviour under transients accidents	Irradiation tests are incomplete and have to be extend to curium cases
Maturity status	At laboratory scale	At laboratory scale	More R&D needs for heterogeneous mode to reach pilot and industrial stages

5.1 Non-proliferation

Proliferation risk is considered to be a "host state" issue, where the adversary is considered to be an independent nation, and considers civilian facility misuse, material diversion from civilian nuclear facilities, or the development of clandestine facilities, with the goal of developing and obtaining practical nuclear weapons. It is important to note that proliferation risk is distinct from the security issue of theft by a sub-national or terrorist group, where the assumptions about capabilities and goals are likely to be quite different, and specific attributes of the fuel cycle may have different levels of importance.

The evaluation of proliferation risk has been the subject of efforts to develop assessment methodologies such as the recent activities in the Generation IV programme and in the IAEA through the INPRO project [1 and 2]. These methodologies emphasise that the evaluation of proliferation risk is largely a subjective process, such as the following statement from reference [2], p. 19:

> *"INPRO has produced one basic principle that requires that proliferation resistance features and measures be implemented throughout the full life cycle for INS (innovative nuclear energy systems) and that both intrinsic features and extrinsic measures be utilised. To comply with this basic principle requires …; the commitment and obligations of States*

be adequate; … Country profiles would be prepared to evaluate the commitments, obligations and policies of states, both technology developer states and technology user states, regarding non-proliferation."

As a consequence, the results of any evaluation of proliferation risk depend on a great extent on assumptions that are made about adversary capabilities and intent [3 and 4]. The material attractiveness can also be included as a factor in such evaluations [5]. In addition, country capabilities and intentions can and do evolve over time, such that an evaluation completed at one point in time may no longer be valid later, especially under a different set of circumstances with respect to underlying technical capabilities, security, or diplomatic motives and situations, etc. Lastly, it must be kept in mind that the transfer of certain technologies conveys know-how to a state that could subsequently form the basis for clandestine facilities, even if a declared nuclear energy system is not misused. These considerations greatly complicate proliferation risk assessments and any conclusions that may be reached.

The effectiveness of international safeguards, and the degree to which a particular facility, technology, or process allows effective implementation of safeguards is important to proliferation risks. When comparing the homogeneous and heterogeneous recycle of minor actinides, the use of plutonium/uranium fuel in the heterogeneous case may facilitate implementation of existing safeguards technologies and best practices, the same as for plutonium-only recycle, while with the homogeneous recycle, the combination of plutonium with the minor actinides complicates measurement and detection of plutonium, an issue which is the subject of ongoing development of instrumentation technologies. The estimation of any change in proliferation risk with respect to plutonium-only recycle or in the potential to misuse the technologies will depend on the outcome of such developments.

As a result, at this time it is not possible to unambiguously determine that one method of minor actinide recycle presents a greater or lesser proliferation risk than the other, or that minor actinide recycle causes a greater or lesser proliferation risk than plutonium-recycle alone, especially since one aspect of such a determination would include the host state intentions and motivations, and recognising that the separations technologies for obtaining plutonium from spent fuel appear to be so widely known.

Material attractiveness is also related to the security issues posed by the sub-national threat and their desire to obtain a nuclear explosive device. In this sense, the use of material attractiveness as an indicator of proliferation risk must be considered very carefully in that the relevance can depend entirely on the viewpoint of the people performing the assessment and the assumptions made about adversary capabilities.

5.2 Summary

The multi-criteria analysis performed shows no large discrepancies between the two modes of minor actinides transmutation. Both of them are feasible. Most of the criteria are equivalent. Some of them indicate some advantages versus drawbacks. The final choice will depend on the importance given by decision-makers to the different criteria. More precisely, the analysis shows:

- With the homogeneous mode:
  - Fuel fabrication is easier but impacts all the fuel fabrication streams.
  - Fuel behaviour under irradiation is quite close to the typical standard one.
  - No flexibility is possible between production of electricity and long-lived wastes mission.
  - Some impact on core reactivity coefficients limits content of minor actinides to about a few percentages of total heavy isotopes within the fuel. Higher content

will entail redefining and optimising the core design compared to the standard one.

  - A higher mass fluxes (number) of sub-assemblies with MA has to be managed within the fuel cycle compared to the heterogeneous mode.

- With the heterogeneous mode
  - Fuel fabrication is more complex but is concentrated on a limited mass flux.
  - Fuel behaviour under irradiation is quite different from the standard one due to a large production of curium and helium released. Targets design must be validated.
  - No impact on standard fuel cycle. A high degree of flexibility is given to adapting operation as a function of the different strategic policies.
  - No impact on main reactor core parameters but higher thermal load of spent fuels compared to the standard one limits minor actinide content within targets and/or define new handling process at reactor stage.
  - Due to lower neutron fluxes, the minor actinides inventories are quite higher compared to the homogeneous mode.

No real discrepancies are shown on the others items.

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Acronyms

ABR	advanced burner reactor
ADS	accelerator-driven system
AFCI	Advanced Fuel Cycle Initiative
ANL	Argonne National Laboratory
ATR	advanced test reactor
BNL	Brookhaven National Laboratory
BOC	beginning of cycle
CEA	Commissariat à l'énergie atomique et aux énergies alternatives
Cercer	ceramic-ceramic
Cermet	ceramic-metal
DPA	displacements per atom
ENEA	Italian National Agency for New Technologies, Energy and Sustainable Economic Development
EOC	end of cycle
EPR	European pressurised reactor
FCCI	fuel cladding chemical interaction
FCMI	fuel cladding mechanical interaction
GFR	gas-cooled fast reactor
GNEP	Global Nuclear Energy Partnership
HM	heavy metal
INL	Idaho National Laboratory
JAEA	Japan Atomic Energy Agency
JSFR	JAEA sodium-cooled fast reactor
LFR	lead-cooled fast reactor
LWR	light-water-cooled reactor
MA	minor actinide(s), e.g. neptunium (Np), americium (Am), curium (Cm)
MABB	minor actinide bearing blanket
MOX	mixed oxide
OECD/NEA	Organisation for Economic Co-operation and Development/Nuclear Energy Agency
O/M	oxide to metal atom ratio
ORNL	Oak Ridge National Laboratory

PIE	post-irradiation examination
P&T	partitioning and transmutation
SFR	sodium-cooled fast reactor
TRL	technological readiness level
TRU	transuranic elements, e.g. plutonium (Pu), Np, Am, Cm
ULOF	unprotected loss of flow
ULOHS	unprotected loss of heat sink
UNF	used nuclear fuel
USDOE	United States Department of Energy
UTOP	unprotected transient overpower

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Homogeneous versus Heterogeneous Recycling of Transuranics in Fast Nuclear Reactors

Fuel transuranics (TRU) multi-recycling is a mandatory feature if both the resource sustainability and the waste minimisation objectives for future fuel cycles are to be pursued. The resulting TRU transmutation can be implemented in fast neutron spectrum reactors according to two main options commonly referred to as the homogeneous and heterogeneous modes.

In this study, the two alternatives have been compared in terms of reactor core feasibility, fuel development and impact on the fuel cycle. The multi-criteria analysis indicates that there are major challenges in minor actinide-loaded fuel development, its experimental validation and possibly in its reprocessing. Both modes of recycling have an impact on the overall fuel cycle, even if at different stages, for example complex target fabrication and handling in the case of heterogeneous recycling and full core fuel fabrication in the case of homogeneous recycling. The study finds that an economic evaluation according to specific implementation scenarios should still be undertaken.

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