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Title:	Alternatives to the direct disposal of spent fuel in a geological disposal facility: routes derived from spent fuel reprocessing
Summary:	The Strategic Environmental Assessment report supporting the decision to implement geological disposal of radioactive waste must contain an evaluation of the radiological impact on the environment and the population; a specific section also puts the performance of the selected concept in the perspective of the performance of alternative solutions. The present note reviews the performance of alternative routes to geological disposal of waste associated with the spent fuel that would involve its reprocessing. These routes are sometimes perceived as offering alternative solutions for the long-term management of radioactive waste. This review note reveals areas of possible benefits in the framework of the fuel cycle strategy, through the reduction of the final inventory, better use of primary resources and conditioning in waste forms optimized for disposal. These benefits are to be put in the perspectives of several concerns in relation to robustness and societal implications, in particular the timeframe associated with multiple reprocessing steps and the risks associated to higher amounts of minor actinides at all steps of the fuel cycle. The need for a disposal facility for the ultimate, long-lived, waste is, however, not relieved in view of the inevitable losses in the reprocessing process, of the limited applicability of separation and transmutation technologies to treat long-lived fission products, and the absence of incentives to treat waste other than spent fuel. Hence, despite a possible contribution to the spent fuel management strategy, these routes cannot be considered as alternative solutions for the long-term management of high-level waste or (low- and intermediate-level) long-lived waste.
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Alternatives to the direct disposal of spent fuel in a geological disposal facility: routes derived from spent fuel reprocessing

Extended abstract

ONDRAF has issued a proposal for a national policy for the long-term management of high level radioactive waste and long-lived waste in Belgium [1], which is supported by a Strategic Environmental Assessment (SEA) report [2]. That policy (or plan) identifies geological disposal as the way forward for Belgium. The plan is submitted to a public consultation that includes the advice from relevant authorities and institutions, among which, FANC. In preparation of its advice [3], the nuclear Regulatory Body FANC/Bel-V has reviewed, independently from ONDRAF, the performance of the reference solution and of identified possible alternative options. In line with the justification principle of § 20.1.1.1 of RGPRI [4], a comparison of the options is conducted for a series of relevant indicators in relation to operational and long-term safety, robustness and societal concerns. Overarching safeguards and nuclear security considerations have also been taken into account though to a much lesser extent.

The present note reviews the performance of different fuel cycle scenarios for these indicators. Although the different stages of the fuel cycle have been considered, the emphasis is given to the back end of the cycle, mostly through the influence of ultimate waste characteristics on the performance of a (geological) disposal facility. The review also addresses whether the need of geological disposal, as a long-term solution for those waste, is relieved. In view of the broad scope retained, the document may also serve as basis for assessing other policies related to the fuel cycle.

The following fuel cycles that involve reprocessing have been analysed and compared to the **direct disposal of spent fuel elements**:

i. **The MOX (mixed uranium-plutonium oxide) route**, which is a conventional reprocessing route today applied at industrial level. The route can further be declined into:
   a. *Plutonium mono-recycling:* spent uranium oxide fuel is reprocessed; spent MOX fuel goes to disposal. This route fits with the existing light water reactor technologies.
   b. *Plutonium multi-recycling*: this scenario involves the reprocessing of MOX fuel for at least one additional reprocessing cycle and typically necessitates fast reactor systems.

ii. **Advanced partitioning schemes**, where one distinguishes:
   a. *Partitioning and transmutation*, where next to uranium and plutonium, other actinides are separated from fission products for subsequent use as reactor fuel. These scenarios are both considered in the perspective of uranium resource preservation and to reduce the actinide inventory in the ultimate waste.
   b. *Advanced partitioning and conditioning*, which shows similarities with *a.* in the sense that nuclides are separated according to their chemical behaviour. The focus is not only on actinide separation, but also on fission product separation for specific conditioning.

iii. **TOP-MOX (MOX Transfer of Property) and regional synergy approaches** are variants of the former options, where the plutonium or transuranic elements are considered as resources by other parties and are transferred to those.

Routes based on reprocessing are often presented as an option to reduce the radiotoxicity of the waste to a reference level in much shorter timeframes, spanning only a few generations. This may lead to the perception that geological disposal is not the only way forward for high level and long-lived radioactive waste. As it appears in this review note, these routes offer some benefits in the framework of the fuel cycle strategy, through the reduction of the final inventory (both in terms of volume and radiotoxicity), better use of primary resources and conditioning in waste forms optimized for disposal. These benefits should be put in the perspectives of several concerns in relation to robustness and societal implications, in particular with the timeframe associated to multi-reprocessing and the risks associated to higher amounts of minor actinides at all steps of the fuel cycle. With respect to the long-term management of radioactive waste, the inevitable losses in the reprocessing process, the limited applicability of separation and transmutation technologies to treat long-lived fission products, and the absence of incentives to treat waste other than spent fuel do not relieve the need for a disposal facility for the ultimate waste. Hence, despite a possible contribution to the spent fuel management strategy, these routes cannot be considered as *alternative* solutions for the long-term management of high-level or long-lived waste.

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List of abbreviations

Acronym	Definition
ADS	Accelerator Driven System
AFI	Accessible Fraction of Inventory (equiv. to “Instant Release Fraction”, IRF)
ALI	Annual Limit on Intake
ASN	Autorité de Sûreté Nucléaire
BWR	Boiling Water Reactor
CANDU	Canadian Deuterium-Uranium (reactor type)
CEA	Commissariat à l’énergie atomique et aux énergies alternatives
CSD-C	Colis standard de déchets compactés
CSD-V	Colis standard de déchets vitrifiés
EBS	Engineered Barrier System
EFP6	European Framework Programme 6
FANC	Federal Agency for Nuclear Control
FP	Fission Product
FR	Fast Reactor
HWR	Heavy Water Reactor
HLW	High Level Waste
IRF	Instant Release Fraction (equiv. to “Accessible Fraction of Inventory”, AFI)
IRSN	Institut de Radioprotection et de Sûreté Nucléaire
KQ	Key Question
LWR	Light Water Reactor
MA	Minor Actinides
MOX	Mixed (uranium – plutonium) Oxide
MSR	Molten Salt Reactor
NIRAS	Nationale instelling voor radioactief afval en verrijkte splijtstoffen (EN: National organization for radioactive waste and enriched fissile material)
NPP	Nuclear Power Plant
ONDRAF	Organisme national des déchets radioactifs et des matières fissiles enrichies (EN : National organization for radioactive waste and enriched fissile material)
OTC	Once-Through Cycle
P&C	Partitioning and Conditioning
P&T	Partitioning and Transmutation
PNGMDR	“Politique nationale pour la gestion durable des matières et déchets radioactifs”
PUREX	Plutonium Uranium Redox Extraction
PWR	Pressurized Water Reactor
RepU	Reprocessed Uranium
SEA	Strategic Environmental Assessment
TOP-MOX	MOX Transfer to Other Parties
TTC	Twice-Through Cycle
UOX	Uranium Oxide

1 Context

Within the framework of the PR1104 FANC project on geological disposal, a collaboration between FANC and Bel V, forming together the regulatory body, has been set up [5]. The key objective of the collaboration is to develop and maintain expertise and synergies inside the regulatory body in order to ensure a high quality of the regulatory body guidance and review performed in the framework of the decisional process related to the Belgian geological disposal programme. The achievement of this objective involves the deployment of a R&D programme addressing the key questions identified in the document "Strategic Research Needs" (SRN) [6].

The SRN note is structured into key issues considering the needs to be addressed by the regulatory body and the current safety concept defined by ONDRAF/NIRAS. The present version of the SRN note addresses the following key issues:

- Key issue 1: Characterisation
- Key issue 2: Processes important for safety
- Key issue 3: Long-term stability
- Key issue 4: Feasibility
- Key issue 5: Assessment

For each key issue, key questions (KQ) that should be investigated by the regulatory body are identified. The research and development (R&D) programme is elaborated according to the deployment plan [7] (currently defined for the period 2017-2021) for investigating the key questions identified in the SRN.

2 SRN deployment action and key questions

Table 1: Key Issue #5 – Safety Assessment: Key Questions related to alternative options to geological disposal

Key questions related to source term
What are the safety issues to be considered for the comparison of options for the long-term management of B&C wastes?
What are the alternative options to geological disposal?
What are the pros & cons of each identified options?

3 Scope | Topic description

In the Belgian context of radioactive waste management, one distinguishes between category A waste consisting of short-lived, low and medium level activity waste, and categories B and C consisting of (low and intermediate level) long-lived and high-level wastes, respectively. In order to set up a national policy (or plan) for category B and C waste, a proposal has been issued by ONDRAF/NIRAS, under the form of a Royal Decree proposal [1]. At this stage, the policy is purely conceptual and generic and solely defines **a geological disposal facility on the Belgian territory as the long-term radioactive waste management solution for Belgium**; it does not address neither a specific host rock, neither a specific site, neither a specific technology or design, neither does it propose a calendar for the implementation of this solution.

This national policy is to be seen as a plan or programme in the sense of the Belgian Law of February 13, 2006 [8], transposing in Belgian regulation the EC Directive 2001/42/EC of 27 June 2001 [9]. It foresees for national plans and programmes for which an environmental assessment is required*: [T]the draft plan or programme and the environmental report [...] shall be made available to the authorities [...] and the public. The authorities [...]and the public [...] shall be given an early and effective opportunity within appropriate time frames to express their opinion on the draft plan or programme and the accompanying environmental report before the adoption of the plan or programme or its submission to the legislative procedure.* ONDRAF has therefore prepared, in support to the proposal for a national policy on the long-term management of the radioactive waste and of the spent fuel, a Strategic Environmental Assessment (SEA) report [2] that is also part of the public consultation and submitted to the advice of relevant authorities. A SEA that addresses radiological considerations is generally made up of two parts, one for the more "classical" concerns in relation to the environment, and one for the aspects specifically related to the radiological risk. In Belgium, the evaluation of the "classical" part falls within, for example, the competences of the regional and/or national environmental authorities; the radiological part, in turn, is central to the evaluation by the Nuclear Regulatory Body – FANC and Bel V. A specific section of the

SEA also puts the performance of the selected concept in the perspective of the performance of alternative options.

In the framework of the assessment of alternative options, the question arises whether **fuel cycles scenarios that are based on reprocessing may offer an alternative to geological disposal**. Several major fuel cycle scenarios that involve fuel reprocessing are investigated in this note:

i. **MOX routes**: in this reprocessing route that is today applied at industrial level, uranium and plutonium are considered as resources and are recovered. The route is further declined into:
   a. *Plutonium mono-recycling or twice-through cycle (TTC)*
      This is a Light Water Reactor (LWR)-only scenario, with reprocessing of spent UO_2 fuel (UOX fuel) into MOX fuel and disposal of MOX fuel elements – it is often also called "twice through cycle".
   b. *Plutonium multi-recycling (FR-MOX)*
      Although the reprocessing of MOX fuel and its further use for a third cycle in LWR seems possible with some performance loss, the approach generally considered consists of successive U and Pu reprocessing cycles combined with irradiation in *fast reactor systems, although studies are today being conducted, investigating multiple cycles in LWR*. Fission products and minor actinides belong to the waste stream in this approach and are vitrified.

ii. **Advanced partitioning schemes**, where one can distinguish two approaches
   a. *Partitioning and transmutation (P&T)*
      Next to uranium and plutonium, other actinides are separated for subsequent use as reactor fuel. These scenarios are both considered in the perspective of uranium resource optimisation and to reduce the actinide inventory in the ultimate waste.
   b. *Partitioning and conditioning (P&C)*
      It shows similarities with *ii.a.* (P&T) in the sense that nuclides are separated according to their chemical behaviour. The focus is not only on actinide separation, but also on fission product separation in order to apply specific conditioning for the different waste streams. All conditioned fractions (including transuranic elements) are considered as waste and are to be disposed of.

iii. **TOP-MOX and regional synergy scenarios**, where the uranium, plutonium and possibly the other transuranic elements recovered in routes *i.* or *ii.* are considered as resources for other parties. Countries with a stagnant or phase-out nuclear energy policy would feed other parties investing in fast neutron spectrum facilities (fast reactors or accelerator-driven systems) with exploitable resources. In turn, they would recover conditioned waste with reduced radiotoxicity: minor actinides (MA) and fission products (FP) in MOX routes; or fission products and the non-recovered fraction of the actinides if advanced partitioning schemes are applied in the partner countries. This option is less present in the literature, but is for example considered today in The Netherlands [10] and is discussed as a possible way forward in Belgium [2], [11]; it should be emphasized that it offers some kind of solution on a local scale – e.g. a country – but returns, at a global level, to one of the former options (*i.* or *ii.*).

These routes are compared here to a reference scenario, which is the **direct disposal of UOX spent nuclear fuel**. This reference scenario is not to be confused with the reference Belgian scenario proposed by ONDRAF in the framework of the SEA supporting the decision on geological disposal of category B&C waste [2]. The Belgian context is rather a mix of the Once-Through-Cycle and of the Twice-Through-Cycle and implies direct disposal of spent UOX and possibly MOX fuel assemblies, as well as the disposal of vitrified waste.

Reprocessing, in particular the "partitioning and transmutation" route, was originally perceived as – and developed for – offering alternative options to geological disposal for the long-term management of the spent nuclear fuel [12], [13]. The inventory and the waste forms are indeed heavily modified to the extent that the possibility of surface or near-surface disposal of the ultimate waste is sometimes investigated [14], [15]. These advanced technologies were, for example, extensively studied in the French approach to the long-term management of high level radioactive waste, where it was identified as a priority research topic in the so-called "*loi Bataille*" [16], as well as in the framework of the "*Politique nationale*

pour la gestion durable des matières et déchets radioactifs" (PNGMDR) [17]. CEA, IRSN and ASN have, however, expressed doubts about the possibility to use separation and transmutation (the most innovative reprocessing route) as a long-term management solution for high level or long lived radioactive waste [18]–[21]. Moreover, the applicability of those technologies to waste other than spent fuel are generally not considered according to this survey.

While, in general, the search for alternative options is driven by the waste production mode and start from a given situation – a pre-existing radiological inventory –, one seeks, in fuel cycle scenarios, a change of paradigm by optimizing, in an integrated approach, the entire chain of the fuel cycle from the feed material until disposal ("from cradle to grave"). The performance assessment of the various fuel cycles therefore must be appreciated over the whole cycle rather than for the final disposal solution solely. The present survey therefore addresses several safety indicators, and to a much lesser extent certain security and safeguards considerations, such as operational and long-term safety, hazard and risk reduction, robustness and societal concerns. Those concerns are not discussed from a disposal perspective solely, but are addressed in a holistic approach, covering the entire fuel cycle. The survey may evoke, for contextual purposes solely, other aspects of these fuel cycle scenarios, such as financial aspects or technology choices for energy supply. On the same line of thinking: robustness against policy changes is evaluated. The latter is taken as a purely external constraint, without any judgment on the appropriateness of these policy changes.

With respect to the disposal concerns, a consensus emerges today that the different reprocessing routes offer alternative options for the spent fuel management strategy and for the waste forms – the physical, chemical and radiological characteristics of the waste –, but do not fully levy the need for a disposal facility for the ultimate high-level or long-lived waste as a long-term management solution.

The present note therefore addresses the performance of the different routes from two perspectives for the backend of the fuel cycle:

- **Does the inventory reduction enable for alternative disposal options to geological disposal as long-term management solutions?** This concern mostly relates to the long-term hazard reduction (§ 5.4.2).
- **What is the impact of the fuel scenario on a geological disposal solution, assuming that such solution is required?** This concern is reflected in most safety indicators addressed in the survey.

4 Importance for safety

An educated evaluation of alternatives to the proposed national policy for the long-term management of radioactive waste and spent fuel is required to guarantee that it offers the best option for the present and for future generations. Among those options, the possibility offered by spent fuel reprocessing to effectively reduce the radiotoxicity of the ultimate waste to that of uranium ore in a timeframe spanning only several generations (in comparison to several hundred thousand year) may lead to the perception that geological disposal is not the only way forward for the long-term management of radioactive waste.

Contrarily to other alternatives to geological disposal that have been addressed purely from a back-end perspective in the framework of the Belgian Regulatory Body's advice on the policy proposal for the long-term management of high-level or long-lived radioactive waste [3], an holistic approach to the fuel cycle seems more appropriate in the present case to reflect the level of safety and the societal constraints for the present and future generations. Various safety indicators have therefore been addressed in the literature study, considering the impact on the different steps of the fuel cycle.

The focus has generally been given to the dominant contributors for a given theme, for example the spent fuel and the vitrified waste when it relates to the activity / radiotoxicity of high-level waste, or the compacted fraction (hulls and end-fittings) when it relates to long-lived, intermediate-level waste. Less focus is given to secondary waste from the reprocessing route, as they typically feature the same radionuclides at a lower concentration. Still, their contribution in terms of volumes remains appreciable. The present work reflects a high-level analysis based on a generic description of facilities that may exist or are still to come. It would benefit from periodical re-examination in the light of new developments in the front-end (in particular the extraction activities) and the back-end of the cycle (storage, conditioning, disposal, reprocessing), as well as in terms of reactor technologies. The same applies to security and safeguards, where conclusions are also driven by facility- and site-specific considerations.

5 Synthesis of the literature study

5.1 Framework and boundaries of the present review

In a will to address alternatives to geological disposal offered by various fuel cycle scenarios, we limited ourselves to five scenarios that are the most documented in the open literature. Other fuel cycle scenarios are also found that are variants or combinations of the main scenarios addressed here. Whenever possible we also kept the description as generic as possible, without promoting a specific technology; several considerations, however, find their roots in the present context in Belgium, Europe or on the international scene, and address more specifically the reprocessing of existing Light Water Reactor (LWR) fuels, although the most advanced fuel cycles may require synergies between different fleet of nuclear reactors to optimise the global cost and the fuel cycle efficiency.

Considering the predominance of uranium oxide-based fuel (UOX) or mixed uranium-plutonium oxide-based fuel (MOX) worldwide, the focus of this review is also given on the uranium reprocessing cycle. Alternative fuel cycles have been proposed for various reactor designs, based on the breeding of thorium, see e.g. [22]. Those cycles, however, show, from front-end and back-end perspectives, similarities with advanced reprocessing cycles considered today for uranium-based fuel.

The focus has generally been given to the dominant contributors for a given theme, (e.g. spent fuel and vitrified waste for radiotoxicity), to a lesser extent to secondary waste. Those typically feature the same radionuclides at a lower concentration, but may dominate the volumes to be disposed of.

5.2 Description of the fuel cycle options considered

In the early days of peaceful applications of nuclear energy, the expected development of the technology targeted reprocessing of the nuclear fuel and a transition towards a closed fuel cycle [12]. Worldwide, important efforts and achievements were observed from the mid-50's to mid-80's in the field of reprocessing – let us just remember the Eurochemic plant in Belgium – and for the development of fast – or fast breeder – reactors to burn in a more 'sustainable' way the nuclear fuel. In parallel, the possibility to re-use part of the plutonium as MOX fuel in LWRs was demonstrated at an industrial scale, the so-called 'MOX route'. These euphoric developments were promptly tempered at the end of the 80's as a combination of several factors, ranging from societal concerns on safety and proliferation to economic aspects [12]. Only few countries – USA, France, Russia, Japan, South Korea – maintained an active fast reactor programme, however, often at a slower pace. This work reviews a series of major fuel cycle scenarios in order to investigate whether they could offer alternatives to geological disposal, or to assess the impact they may have on geological disposal performance. As underlined formerly, the performance of variants of these cycles for different indicators is also investigated in the literature [12], [23]–[25]; conclusions from the major fuel cycles addressed here should remain transferable to those, pending an educated transposition.

5.2.1 Direct disposal route

- *OTC: open cycle route (or "once-through cycle")*

In the direct disposal route, also referred to as "once-through cycle" (OTC) or open cycle, fuel elements[1] are not reprocessed following to cool-down and interim storage. They are rather considered as waste and are conditioned as it, in a package suitable for geological disposal. This route is often considered as the reference case for performance assessment of routes involving reprocessing. Incentives to follow this route are multiple and derive from economic or societal considerations. From economic perspectives, the main incentive today is the low uranium price that makes re-processing a more expensive alternative. Societal aspects cover a large spectrum of concerns, ranging from non-proliferation arguments, nuclear safety and nuclear security arguments, changing perceptions on the nuclear energy policy (e.g. phase-out), or concerns to reduce the burden of waste management for the next generations.

[1] We consider OTC for current reactor designs – LWR, HWR or GCR – where the fuel consists of 'assemblies' (PWR, BWR), or 'bundles' of fuel rods (CANDU reactors). Those are generically referred here as "fuel elements" for the sake of simplicity.

Two main arguments are often quoted to oppose this route: the poor usage of natural resources – the residual content still features an appreciable amount of fissile isotopes, notably ^{235}U, ^{239}Pu and ^{241}Pu, and fertile[2] or fissionable[3] isotopes (most actinides) – and the long-term radiotoxicity of the material [12], [24], [26]. Regarding the latter concern, the difficulty lies not only in the societal acceptance of long-term disposal (the difficulty to appreciate the meaningfulness of safety demonstration over periods of time far beyond common perception) but requires, from a technical side, the development of a safety case for radioactive waste disposal facilities, which is a far more complex exercise than for the safety assessment of conventional nuclear installations. A side observation is that the selection process of materials for nuclear applications, particularly the fuel materials (fuel + cladding), were historically conducted based on their good performance in reactor but gave little attention to the backend of the fuel cycle. As evoked previously, efficient reprocessing of the fuel – although not yet fully demonstrated – was, historically, the expected way forward.

5.2.2 Conventional reprocessing | MOX route

The MOX route, where uranium and plutonium are considered as resources, is today applied at industrial level. The reprocessing usually follows the well-established PUREX method, where U and Pu are recovered as separate fractions [27]–[30]. Fission products and minor actinides belong to the waste stream in this approach and get vitrified. In view of the separation efficiency of the process (~99.9% extraction efficiency), a very limited amount of U and Pu also joins the waste stream and ends up as vitrified waste. Still, this residual fraction has an important effect on the radiotoxicity reduction of the waste, considering the dominant role of Pu in the long-term radiotoxic hazard [28].

Reprocessed uranium has a residual ^{235}U fraction typically close to that of natural uranium (between 0.5 and 1 wt.%) depending on the actual burnup of the feed UO_2 fuel; it can therefore be further valorised as nuclear fuel, following an enrichment process. However, it also features the presence of ^{236}U (virtually absent in natural uranium) and a larger fraction of ^{234}U, which roughly increases in the same proportion as ^{235}U upon enrichment. These isotopes have shorter half-lives than ^{235}U or ^{238}U, and they feature an appreciable neutron capture cross-section, yet a negligible fission cross-section in the thermal energy range. This affect the radiotoxicity of the reprocessed enriched uranium (REU) fuel and lets them act as neutron poisons. Regarding plutonium isotopes, the Pu isotopic vector highly depends on the burnup of the feed uranium fuel and on decay time. Indeed, only isotopes with an odd mass number are fissile (^{239}Pu and ^{241}Pu) and their concentration progressively saturates as burnup increase – when removal by neutron capture or fission balances their production in view of large neutron reaction cross-sections. In the case of ^{241}Pu, the decay time also plays an important role, in view of its relatively short half-life (14.3 y). Oppositely, even isotopes of plutonium generally do not reach saturation because of lower capture cross-sections, so that their poisoning effect keeps increasing with increasing burnup. Next to those isotopes, the minor actinide content (dominated by ^{237}Np, $^{241\text{-}243}$Am, $^{242\text{-}246}$Cm) also increases with burnup.

❖ *LWR-MOX: plutonium mono-recycling route*

The MOX route is further declined into two options, depending on the reactor fleet available to utilize MOX fuel [12], [24]. The LWR-MOX option or mono-recycling, consists of using MOX fuel assemblies in LWR, thereby saving uranium resources. Spent MOX fuel is then often considered as a waste – one also refers to the route as “twice through cycle” (TTC) or “once-through reprocessing cycle” –, considering the progressive loss of fissile isotopes and the large increase of MA associated with multiple reprocessing steps. Although several authors quote the possibility of a second reprocessing step (reprocessing MOX fuel once) by dilution with LWR-UO_2 fuel, the incentives to reprocess MOX fuel and to re-use it in LWR were often quoted as uncertain [12], [31]. Recently, however, new studies have been performed in France, in view of the postponement of the construction of a fast reactor fleet [32], for possible implementation of plutonium multi-recycling in the EPR fleet instead, by using mixed enriched uranium (~3 wt.% enrichment) and plutonium oxide fuel (~8 wt.% Pu) in a so-called “MIX” fuel concept [33]–[35].

The advantages of the mono-recycling route are limited in terms of the impact on the need for a geological disposal facility. One quotes limited reduction (from -5 to -25%) of the inventory and of its

[2] Fertile: yields a fissile isotope after neutron capture

[3] Fissionable: can undergo fission by high energy (fast) neutrons or thermal neutrons

radiotoxicity [12], [31], [36], as well as in terms of other relevant factors [31]; this is translated, from front-end perspectives, in a reduction of the feed uranium ores and, from back-end perspectives, in a reduction of the overall footprint of the facility for an identical electricity production. It should, however, be noted that the definition of the mono-recycling route varies according to the source, in particular for what concerns the definition of waste and resources: for example, Poinssot et al. [31] still consider the spent MOX fuel as a resource, which is then stored rather than disposed of.

❖ *FR-MOX: plutonium multi-recycling route*

The second option consists of multiple U and Pu recycling and irradiation, generally assumed to occur in fast reactor (FR) systems, although recent studies in France also investigate the possibility of multiple reprocessing cycles in a conventional LWR fleet. FR-MOX may directly originate from the reprocessing of LWR-UO_2 fuel and the successive FR-MOX reprocessing; the variant that involves a first stage as LWR-MOX is, however, preferred today in France in view of the economics of the cycle, but also because of the unavailability of fast reactor systems at industrial level as of today. In order to keep a sufficient quality of the Pu vector – ratio of fissile to fertile isotopes –, dilution with spent LWR-UO_2 fuel might be needed during reprocessing; a ratio LWR-MOX/LWR-UO_2 of 1:2 is quoted [12], [31]. The dilution is needed to avoid a too important degradation of the organic molecules used to separate U and Pu due to the increased alpha activity of MOX fuel (particularly: ^{238}Pu and ^{244}Cm). Slightly better performance is obtained in a mixed approach making use of both LWR, including a first cycle with MOX irradiation in LWR, and FR [12], [31], [36]. Extrapolating values quoted in [12], [36], it seems that a closed cycle with FR only, where plutonium is reprocessed and minor actinides directed to waste streams, is about twice more penalizing in terms of MA waste but preserves more uranium resources because of its feed with depleted or natural uranium solely.

5.2.3 Advanced partitioning schemes

❖ *P&T: partitioning and transmutation route*

The incentive behind partitioning and transmutation (P&T) is to reduce the inventory of plutonium and minor actinide isotopes, in the waste in order to reduce its intrinsic radiotoxicity over the long-term. Pu and MA are indeed dominant contributors to radiotoxicity over times between 10^3 and 10^6 y. In view of the low mobility level of these species in appropriately selected host rocks, the largest impact of the P&T route does not relate to anticipated evolution scenarios, rather to alternative evolution scenarios, such as human intrusion, where direct pathways to the radioactive inventory are created [37], [38].

The FR-MOX route, where U and Pu are recovered, was shown to drastically reduce uranium feeds, the waste volumes and the plutonium inventory in the ultimate waste, at equivalent electrical output. Still, the inventory of minor actinides remains considerable within this approach. Considering that most actinide nuclei are unstable and can undergo fission when reacting with fast neutrons, the idea rapidly emerged to 'burn' actinides in fast reactors, similarly as for plutonium. The reactor design should be chosen such as to efficiently burn those actinides, i.e. burning them faster than they are produced. Minor actinides containing fuels, however, pose great challenges to control criticality in reactor operations, because the fission of minor actinides generates a much lower fraction of delayed neutrons compared to prompt neutrons than, for example, uranium or plutonium. The fraction of minor actinides in fast reactors is therefore limited to a few percent. To overcome this issue, the Accelerator Driven System, ADS, concept was developed, which uses the reactor in subcritical mode, coupled to a spallation neutron source, itself driven by a particle accelerator. The reactor then acts as an amplifier with respect to the neutron source, so that when the neutron source vanishes (e.g., by turning off the accelerator), the chain reaction in the reactor also comes to an end. Such design is much less dependent on the fraction of delayed neutrons, so that larger minor actinide contents may be considered. ADS are therefore often presented as waste burners.

While the PUREX reprocessing method for U and Pu is already developed at an industrial scale, the recovery of minor actinides does not reach the same technological maturity; that level also varies according to the elements which should be recovered. Another concern relates to non-proliferation, so that several research groups investigate the possibility to recover actinides together (co-extraction), rather than in separate fractions. The approaches to partitioning that are today investigated to separate actinides from fission products can be grouped into processes derived from an aqueous (hydrometallurgical) route, or from a pyrometallurgical route. Next to actinide separation, research is

ongoing to separate several fission product elements: iodine – the concern being the isotope of mass 129, ^{129}I, with a half-life of 1.6 10^7 y –; technetium (^{99}Tc, with a half-life of 2.1 10^5 y); caesium and strontium in view of their large contribution to decay heat over the first few hundred years – ^{137}Cs and ^{90}Sr feature a half-life of about 30 y; another Cs isotope, ^{135}Cs, also features a very long half-life of 2.3 10^6 y. The possibility of transmuting those long-lived fission products is sometimes evoked, although the technical feasibility seems compromised as of today, in view of small neutron cross-sections and the need, for example for ^{135}Cs, to proceed to isotopic separation prior to irradiation.

❖ *P&C: Partitioning and conditioning route*

The "partitioning and conditioning" (P&C) route lies midway between the "direct disposal" and the "partitioning and transmutation" routes. Here, the aim is to provide adequate and specific conditioning to radioactive nuclides according to their chemical behaviour, both in terms of separation and of their long-term behaviour in geological disposal conditions. The underlying consideration is that spent fuel assemblies and individual fuel rods were designed and optimised for in-reactor performance, which is different from a geological disposal environment. From a technical point of view, the required surface facilities do not differ from those envisaged for the partitioning and transmutation route, except for the ultimate treatment of actinides – fuel fabrication on the one hand; conditioning as a separated waste fraction on the other hand.

5.2.4 Regional synergies

❖ *TOP-MOX: Transfer of property of reprocessed U, Pu or MA*

Another route, which offers a solution on a local scale – for example for a country –, is also discussed here. It consists of yielding the actinides recovered from reprocessing to another stakeholder that considers it as a valuable resource: as fuel for its reactor fleet. From a global perspective, the approach, however, falls back on either the MOX, either P&T routes. Two strategies are considered:

- MOX Transfer to Other Parties (TOP-MOX) is proposed today in France in the wake of conventional reprocessing. This approach was, for example, retained in part by nuclear power plant (NPP) operators in The Netherlands [10].
- Regional Synergies are evoked in several long-term strategy studies, such as in reference [39]. Countries with a stagnant or phase-out nuclear energy policy feed with exploitable resources other parties investing in fast spectrum facilities (fast reactors or accelerator-driven systems) and in turn recover conditioned waste with reduced radiotoxicity – minor actinides and fission products, or in the most optimistic case only fission products, plus the non-recovered fraction of U, Pu, MA.

5.3 Nuclear security and safeguards

While nuclear security address the risk of sabotage and theft of nuclear material and other radioactive material, safeguards concerns relates to the risk of proliferation of nuclear material among states. Fissile material are of primary interest when dealing with the risk of theft of nuclear material and non-proliferation; in turn, the radiological inventory relating to the material is the most important factor when dealing with the risk of sabotage. Hence, in terms of strategy and from nuclear security and non-proliferation points of view, reducing the possibilities for diversion of fissile material at the different steps of the fuel cycle by a combination of intrinsic and extrinsic features is of primary importance. Those intrinsic features primarily relate to the material composition and the possibilities to divert technological processes. Extrinsic measures then relate to institutional measures, such as safeguards (fissile inventory accountancy), the reduction of fissile material transfers and physical protection measures.

In terms of theft and safeguards, the actual concerns relates to uranium and plutonium[4]. However, the development of advanced partitioning techniques could bring additional concerns for neptunium and americium compounds, since the bare-sphere critical mass for several of their isotopes is appreciably small[5], close to that of ^{235}U [13].

[4] Especially high enriched uranium (in ^{235}U) and weapon grade plutonium (high content of fissile ^{239}Pu).

[5] ^{239}Pu and ^{241}Pu feature a bare-sphere critical mass of the order of 10 kg. The even plutonium isotopes ^{240}Pu and ^{242}Pu, as well as ^{235}U, ^{237}Np and ^{241}Am feature slightly larger values, in the range 40 – 100 kg [13].

❖ *Open fuel cycle*

In the open fuel cycle, depending of the industrial capabilities in a State, the largest proliferation risks may relate to the front-end, considering that uranium appears in a separated form and that the enrichment technology could potentially be diverted for military purposes. In terms of international control, measures are then required and applied, notably international safeguards. Regarding the back-end side, spent nuclear fuel keeps a certain attractiveness in view of its plutonium content. Still, proliferation risks are somewhat inhibited by the necessity to separate the elements present in the spent nuclear fuel; the decay heat and particle emissions are additional intrinsic features that complicates the handling of the material. In terms of nuclear security, the radiological source term is such that the material remains attractive in terms of sabotage.

❖ *Pu mono- and multi-recycling*

The conventional MOX route is based on the PUREX process, where uranium and plutonium are recovered as separate fractions. The possibility exists to separate neptunium as well from the fission products and other minor actinides; however, this is not applied today for the LWR-MOX route [27]. In terms of safeguards, the reprocessed uranium fraction is not particularly more attractive from proliferation point of view than natural uranium; to the contrary, the presence of ^{234}U and ^{236}U rather diminishes its attractiveness. The largest perceived threat then relates to the separated plutonium fraction obtained after conventional reprocessing. Some proliferation resistance is recovered at the next stage (MOX fuel fabrication) when uranium and plutonium oxides powders are mixed, considering that separation of the heavy metal elements would be required again to produce nuclear explosive devices. The progressive decay of ^{241}Pu (half-life of 14.3 a) into ^{241}Am also adds proliferation resistance as well as the radiological characteristics of this radioisotope.

From physical protection perspective, MOX fuel represents a higher risk in terms of theft. The need to use reprocessing facilities results in supplementary diversion paths, in particular as plutonium will be separated from uranium before being mixed to form MOX powder. The fact the nucleal material is subject to particular supplementary operations (including the dissolution in Nitric acid) may also increase its attractiveness for sabotage purposes during its processing, or during the transport between facilities.

❖ *Advanced partitioning schemes*

Advanced fuel cycles in the P&T approach address proliferation aspects from the early development of the reprocessing method: the aim of various approaches under development today is to co-extract actinides to avoid process streams featuring a separated element. Due to the larger content of several alpha-emitters (^{238}Pu, ^{241}Am, ^{243}Am) and neutron emitters (curium isotopes, particularly ^{244}Cm), the separated material is also often much more active and emits more heat than in the PUREX process. This limits the possibilities for the handling of the material, as it requires even more complex installations. In addition, the presence of neutron emitters in quantity complicates the detonation process of nuclear explosive devices [13]. Co-extraction of actinides therefore provides intrinsic proliferation resistance features. In terms of nuclear security, at this stage the risks of sabotage may be considered as comparable to the Pu mono- and multi-recycling option. For the risk of theft, this scheme can benefit from the co-extraction as the nuclear material cannot be directly used to build a nuclear explosive device.

The co-location, or the integration within a single facility, of spent fuel storage, reprocessing and fuel fabrication is also sometimes evoked [28]. On the same line, some advanced fuel cycle concepts, based on the molten salt technology (Molten Salt Reactors, MSR), would feature on-line separation integrated into the reactor building. Those schemes limit the amount of material separated at a time, relies on the physical protection of a single site and avoids transport of fuel material between facilities, except for the uranium (or thorium) feed material [13].

❖ *TOP-MOX and regional synergies*

Strategies involving the transfer of resources to other countries share the characteristics with the underlying reprocessing route excepted the fact that the material in different forms will have to be shipped from regions to others, which involve supplementary risks to deal with and specific safeguards provisions. Also, considering the fact that Belgium is party to the Non-Proliferation Treaty and a member of the Nuclear Supplier Group, export and shipment of the material will be subject to careful attention and controls that may involve limitations in the scope of this scheme.

5.4 Impact on safety

5.4.1 Operational safety

❖ *Radioactive releases to environment*

Radioactive releases to the environment occur at the different stage of the fuel cycle and occur in gaseous or liquid form. Release of solid radioactive waste is generally avoided in regulated practices, also in the case of suspension or dust, by mean of filtration of gaseous or liquid radioactive waste prior to their release; the exception relates to naturally occurring radioactive materials (NORM) in the tailings from mining operations. The figures quoted in this section mostly originate from two recent comprehensive studies from CEA [31], [40]. Applicable limits at the La Hague reprocessing facility are provided in [41].

In the open cycle, the major source of release occurs at the mining and milling stages for gaseous releases; it mainly relates to radon gas (mostly ^{222}Rn), which is naturally present in the ore in low concentration as a decay product of uranium. Estimated releases are 35 TBq/$t_{U\ nat.}$ or 800 kBq/kWh$_e$ when expressed per unit electricity produced [31]. In view of its chemical nature (noble gas) and short half-life (3.82 d for ^{222}Rn), dilution in the atmosphere is generally considered as an acceptable solution that leads to negligible dose to population and environment in comparison to the natural background radiation. In view of its α-emitter character and of the further radiotoxicity of its decay products, radon remains, however, a concern for the occupational dose to workers, particularly in limitedly vented atmosphere, such as underground mines.

Radioactive liquid effluents are dominated, from activity perspective, by tritium produced during reactor operations; estimated releases are of the order of a 1 – 3 kBq/kWh$_e$ [31], [42]. Solid waste volumes are then dominated by very low-level waste in tailings of mining operations; the inventory depends on the mining extraction technique and might be reduced when uranium is a by-product of extraction, or when in situ leaching is applied – under expansion over the last decades, this production mode today dominates [43]. This very low-level waste is not necessarily released into the environment (biosphere), depending on the site remediation strategy.

Reprocessing, although diminishing the feed material, is accompanied by additional releases to the environment. The fission gases (xenon, krypton) contained in the fuel feature get freed upon shearing and dissolution of the fuel. In the twice-through cycle (LWR-MOX route), they feature a similar level of activity (per unit electricity produced) as the radon release during mining operations, close to 600 kBq/kWh$_e$; this activity is, however, dominated by ^{85}Kr (β^--emitter), with a half-life of about 11 y. Considering the low radiotoxicity of ^{85}Kr and its inert chemical behaviour, dilution in the atmosphere is generally considered as an acceptable approach that leads to negligible dose in comparison to the natural background radiation. Still, one estimates that the concentration of ^{85}Kr in the atmosphere has increased by several orders of magnitude due to anthropogenic activities, in particular reprocessing of the spent fuel [44].

When reprocessing is conducted, liquid radioactive effluent releases occur predominantly during the reprocessing stage. The activity is dominated by the tritium contained in the fuel for the LWR-MOX route: the zircaloy cladding being an effective barrier to tritium migration, only 10% of the tritium produced by fission events indeed reach the primary coolant circuit [31]. The remaining 90% are then released, as radioactive liquid waste, upon fuel shearing and dissolution at the reprocessing plant, with estimated amount of 25 kBq/kWh$_e$. In advanced fuel cycles that use fast reactors, steel is anticipated as cladding material, which is much more transparent to tritium. The repartition of tritium between reactor and reprocessing waste may then deviate from the LWR-MOX figures.

Similar considerations apply to ^{14}C where about 90% of the inventory present, at trace level, in the fuel also gets discharged to the environment upon reprocessing [25]. The fraction present in the cladding or structure material is, however, retained in those materials and ends up in compacted waste forms.

In the once-through scenario and for most geological disposal concepts, iodine-129 (^{129}I) generally dominates the dose to environment and population upon engineered barrier failure and diffusion through the host rock to aquifers [12], [25]. Indeed, the retention of anions by adsorption is virtually absent in clay; this brings these species amongst the most mobile ones. With a half-life of 15.7 10^6 y, the activity remains low compared to previously quoted figures (rough estimate of 5 Bq/kWh$_e$); however, ^{129}I is one

of the most radiotoxic fission products in view of its high solubility in water and of its affinity for thyroid: it features a dose coefficient factor of 1.1 10^{-7} Sv/Bq.

In the context of reprocessing, iodine generally escapes as a gas during the dissolution process and is recovered in dedicated trapping systems. In view of the difficulty to find suitable waste forms that are stable (insoluble) from geological disposal perspective, the approach often retained today is discharge into the oceans [12], [24], [45], [46], which offers both volume and isotopic (by stable ^{127}I) dilution. Although the individual dose is small, the collective dose to present and future populations, and to the environment remains difficult to quantify [47]. The approach does neither follow the general approach for the long-term management of radioactive waste (i.e. confinement and containment) and has therefore become controversial [48]. Alternative solutions might be needed, for example disposal in salt domes, which offer confinement and retardation thanks to reduced mobility, as well as isotopic dilution by ^{127}I when migration occurs [12]. Discharge to oceans introduces a bias when comparing dose assessment for different fuel cycle options, as the ^{129}I inventory in the ultimate waste is reduced by a factor 50 – 100 in PUREX-like methods [13].

- *Occupational dose to workers and public exposure*

A comparison of the occupational dose to workers and public exposure for the OTC and a scenario involving 20% electricity production by mean of fast reactors is provided in [14]; it is derived from two former works [49], [50]. A more recent study addresses the exposure of workers in the nuclear sector over the years [51]. Based on these studies, no significant differences are expected for the various fuel cycle scenarios: all of them would provide acceptable doses, in line with current industry practices.

The relative importance of each step in the fuel cycle are, from exposure perspective, quite opposite in the two studies. The study [14] presents the front-end of the fuel cycle (mining, milling and conversion) as the dominant contributor to occupational dose to workers (>50%) in the open cycle. Advanced fuel cycles are then expected to reduce this figure thanks to reduced mining needs; an additional dose in relation to reprocessing is present but it does not outweigh the gain at the front-end side. Equivalent levels of exposure are expected for reactor operations in all fuel cycle scenarios, as dose limits are identical in all types of facilities. It is to be noted that the values provided suffer from large uncertainties, typically a factor 2, and are sensitive to technology in use and to the evolution of best practices.

The more recent study of [51], on workers solely, indicates a somewhat opposite trend, with reactor operations dominating exposure of workers in terms of collective exposure; the front-end having still a larger average dose per worker. Krahn et al. [51] discuss the apparent discrepancy on the basis of historical evolution of individual dose, worker population and worker efficiency with time. It turns out that both a stronger reduction of the workforce needed and of the average exposure are observed in the front-end stages, especially when uranium is recovered from in situ leaching, a technique that was not industrially developed at the time of the [14] study.

5.4.2 Hazard & risk reduction

- *Waste form (physical and chemical characteristics)*

High Level Waste (HLW)

Two main high-level waste (HLW) forms are expected, from the fuel cycle perspective, according to the chosen fuel cycle. In the OTC the spent nuclear fuel is directly disposed in the form of spent fuel elements, while other routes derived from reprocessing lead to vitrified HLW in the presently proposed schemes worldwide. The LWR-MOX has an intermediate character, in the sense that both spent nuclear MOX fuel and vitrified waste would get disposed of.

As of today, liquid waste streams from the reprocessing process, which contain all fission products (except some of the volatile ones) and minor actinides, are calcinated and mixed with borosilicate (B_2O_3, SiO_2) glass. The mixture is then poured in dedicated stainless-steel containers ("*colis standard de déchets vitrifiés*", CSD-V), where it solidifies. The volumes of HLW are then reduced to about a third of those generated in the OTC [31].

While vitrified waste may be considered as a homogeneous material, the nuclide inventory in spent fuel is not evenly distributed in the matrix and shows important variations according to the operating conditions, see for example the review conducted for the Belgian spent nuclear fuel by SCK CEN and

NIRAS/ONDRAF [52], or that of NAGRA for Swiss spent nuclear fuel [53]. Although many nuclides remain in solution in the uranium dioxide matrix, in particular actinide and rare-earth elements, others tend to precipitate during irradiation as separate oxide (e.g. Sr, Ba, Zr, Mo) or metal phases (e.g. Ru, Rh, Pd, Tc, Mo) , either within the grains, either at grain boundaries; several gaseous (Kr, Xe) and volatile (Cs, I, but also Br, Se, Te, Rb) elements also form gas bubbles or escape from the matrix to the rod free volumes.

These differences in the waste form is reflected in several phenomena and characteristics impacting the safety; they are further analysed from various perspectives in the remaining of this document.

Waste other than HLW

In parallel to the reduction of HLW, fuel cycles derived from reprocessing generate other waste streams that are not present in the OTC. The radionuclide vector (the radionuclides present and their concentration ratios) of those waste streams is not substantially different from that of the HLW: actinides, fission products and activated products. An important fraction, in terms of volume, then arise from technological waste of the reprocessing process. However, in terms of activity, the intermediate, long-lived, level waste (ILW-LL) made of the claddings and other assembly structures (hulls and end-fittings) dominates the source term. Those were cemented in the paste but are today compacted and conditioned in containers similar to the CSD-V containers. One generally refers to those as CSD-C ("*colis standard de déchets compactés*"). Other ILW-LL waste streams which were historically treated by bituminization are today cemented or simply compacted to reduce their volumes [41], so that the total (HLW + ILW-LL) volumes involved in the LWR-MOX (TTC) and the OTC routes remain qualitatively similar according to [31].

Reference [46] quotes, all in all, a waste volume reduction upon reprocessing by a factor of about two, split over HLW and ILW. Basically the same is reported for the TTC route. The volumes of waste in the OTC (~2m³ HLW per TWh_e) is, however, the double of that considered by [31]; similar figures are provided for the TTC route. In the latter one, HLW (vitrified waste and spent MOX fuel) still makes two third of the total waste, while the proportions become roughly inverse for multiple Pu reprocessing (FR-MOX route) or for the P&T route.

❖ *Inventory (radiological characteristics)*

Before addressing the inventory of the ultimate waste produced by the various routes, a comparison framework should be established. For simplicity, the comparison in this section will be limited to activity and radiotoxicity. The total inventory is, of course, heavily dependent on the extent of the domestic nuclear programme; for the purpose of the comparison, the comparison is conducted based on an equivalent amount of energy produced and equilibrium cycles. In closed cycles involving multi-recycling, the impact of the last, potentially non-reprocessed, cores on the inventory may lead to an additional ultimate waste inventory, for which a proper long-term management solution will be required. That complementary inventory can be expected to have a minor impact in terms of fission products but may lead to a pronounced increase of the actinide inventory depending on the extent of the nuclear program and on the handling of the phase-out.

It is then important to screen out which radionuclides are the most relevant in the comparison [54], [55]. Table 2 reflects the most critical radionuclides for the Belgian spent nuclear fuel context that consists of direct disposal for one part of spent UOX fuel – possibly also MOX fuel – and, for another part, of vitrified waste from past reprocessing activities, according to a recent collaboration between SCK CEN and ONDRAF/NIRAS [56]. The list is further subdivided according to the half-life – boundaries at 100 and 10 000 y were arbitrarily adopted in this work – and production mechanism of the nuclides. Nuclides with half-lives lower than 100 y are mostly relevant from operational or thermal phase (decay heat-driven phase) perspectives; those with intermediate (100 to 10 000 y) and long (> 10 000 y) also require proper consideration in the long-term safety assessment.

The list of nuclides in Table 2 remains globally relevant for the routes deriving from reprocessing, although additional activation products might need to get considered, depending on the material selection for future reactor types; one might for example think about ^{210}Po in the framework of lead-bismuth cooled fast reactors like the Myrrha project [57], [58].

The inventory is further discussed according to the production mode of nuclides in the next subsections.

Table 2 List of relevant radionuclides properties and their production process, for radiological disposal of spent nuclear fuel in the OTC or TTC routes, adapted from [52]. Nuclide selection: [59]. Source of the numerical data: JEFF-3.1 decay library [60], modified for ^{79}Se [61]. Control rod materials are not included in the survey.

	Radionuclide	Half-life	Main decay mode	Major source	Main location	Mostly present as
Half-life < 10^2 y	^{54}Mn	312 d	β^+, EC	^{54}Fe	Structures, cladding, fuel	Alloy. / impurity AP
	^{60}Co	5.27 y	β^-	^{59}Co	Structures, fuel, clad	Impurity AP
	^{3}H	12.3 y	β^-	FP	Fuel, cladding	FP
	^{85}Kr	10.8 y	β^-	FP	Fuel, plenum	FP
	^{90}Sr	28.8 y	β^-	FP	Fuel	FP
	^{106}Ru	1.02 y	β^-	FP	Fuel	FP
	^{110m}Ag	250 d	β^-	FP	Fuel	FP
	^{125}Sb	2.76 y	β^-	FP, ^{124}Sb	Fuel, clad	Alloying / impurity AP, FP
	^{134}Cs	2.07 y	β^-	FP	Fuel	FP
	^{137}Cs	30.0 y	β^-	FP	Fuel	FP
	^{144}Ce	285 d	β^-	FP	Fuel	FP
	^{147}Pm	2.62 y	β^-	FP	Fuel	FP
	^{151}Sm	90.0 y	β^-	FP	Fuel	FP
	^{154}Eu	8.59 y	β^-	FP	Fuel	FP
	^{155}Eu	4.75 y	β^-	FP	Fuel	FP
	^{238}Pu	87.7 y	α	^{237}Np	Fuel	Actinide
	^{241}Pu	14.3 y	β^-	^{240}Pu	Fuel	Actinide
	^{242}Cm	163 d	α	^{242m}Am	Fuel	Actinide
	^{243}Cm	30.0 y	α	^{242}Cm	Fuel	Actinide
	^{244}Cm	18 y	α	^{243}Cm, ^{244}Am	Fuel	Actinide
10^2 y < Half-life < 10^4 y	^{63}Ni	101 y	β^-	^{62}Ni	Structures, clad, fuel	Alloying / impurity AP
	^{14}C	5700 y	β^-	^{14}N, ^{13}C	Fuel, clad	Impurity AP
	^{93}Mo	4000 y	EC	^{92}Mo, ^{93}Nb	Structures, clad	Alloying / impurity AP
	^{108m}Ag	418 y	β^-	^{107}Ag	Fuel, clad	Impurity AP
	^{226}Ra	1600 y	β^-	(4n+2) DS	Fuel	Actinide
	^{229}Th	7340 y	α	(4n+1) DS	Fuel	Actinide
	^{240}Pu	6563 y	α	^{239}Pu	Fuel	Actinide
	^{241}Am	433 y	α	^{241}Pu	Fuel	Actinide
	^{242m}Am	141 y	IT	^{241}Pu	Fuel	Actinide
	^{243}Am	7365 y	α	^{242m}Am, ^{242}Pu	Fuel	Actinide
	^{245}Cm	8500 y	α	^{244}Cm	Fuel	Actinide
	^{246}Cm	4730 y	α	^{245}Cm	Fuel	Actinide
Half-life > 10^4 y	^{94}Nb	2.00 10^4 y	β^-	^{93}Nb	Cladding	Alloying AP
	^{59}Ni	7.6 10^4 y	β^+, EC	^{58}Ni	Structures, cladding, fuel	Alloying / impurity AP
	^{10}Be	1.6 10^6 y	β^-	^{9}Be	Fuel	Impurity AP
	^{36}Cl	3.01 10^5 y	β^-, β^+, EC	^{35}Cl	Fuel, clad	Impurity AP
	^{41}Ca	1.03 10^5 y	EC	^{40}Ca	Fuel	Impurity AP
	^{79}Se	3.56 10^6 y	β^-	FP	Fuel	FP
	^{93}Zr	1.53 10^6 y	β^-	^{92}Zr, FP	Fuel, cladding	FP, Alloying AP
	^{99}Tc	2.14 10^5 y	β^-	FP	Fuel	FP
	^{107}Pd	6.5 10^6 y	β^-	FP	Fuel	FP
	^{126}Sn	2.3 10^5 y	β^-	FP	Fuel	FP
	^{129}I	1.61 10^7 y	β^-	FP	Fuel	FP
	^{135}Cs	2.3 10^6 y	β^-	FP	Fuel	FP
	^{230}Th	7.54 10^4 y	EC, β^+, β^-, α	(4n+2) DS	Fuel	Actinide
	^{232}Th	1.41 10^{10} y	α	(4n) DS	Fuel	Actinide
	^{231}Pa	3.28 10^4 y	α	(4n+3) DS	Fuel	Actinide
	^{233}U	1.59 10^5 y	α	(4n+1) DS	Fuel	Actinide
	^{234}U	2.46 10^5 y	α	(4n+2) DS	Fuel	Actinide
	^{235}U	7.04 10^8 y	α	Fabrication	Fuel	Actinide
	^{236}U	2.37 10^7 y	α	^{235}U	Fuel	Actinide
	^{238}U	4.47 10^9 y	α	Fabrication	Fuel	Actinide
	^{237}Np	2.14 10^6 y	α	^{236}U, ^{241}Am	Fuel	Actinide
	^{239}Pu	2.41 10^4 y	α	^{238}U	Fuel	Actinide
	^{242}Pu	3.74 10^5 y	α	^{241}Pu	Fuel	Actinide
	^{244}Pu	8.00 10^7 y	α	^{242}Pu	Fuel	Actinide
	^{247}Cm	1.6 10^7 y	α	^{246}Cm	Fuel	Actinide
	^{248}Cm	3.4 10^5 y	α	^{247}Cm	Fuel	Actinide

Abbreviations: EC: Electron Capture | IT: Isomeric Transition | DS: Decay series | FP: Fission Product | AP: Activation Product

Fission products

The fission product inventory is limitedly affected by the retained fuel cycle option, considering that a fission event mostly yields two fission products and features a similar energy release, of about 200 MeV, for all the relevant actinide isotopes. For an equivalent amount of thermal energy produced (i.e. without consideration for energy conversion efficiency), one therefore expects a similar number of fission events and, hence, of generated fission products. However, the nature of the radionuclides features some dependency on the isotope undergoing fission, as the mass distributions of fission fragments differ.

Figure 1 reports the mass distribution of fission fragments as the fission yield per 100 fission events following fission (fast) neutrons or thermal neutrons reactions with several uranium and plutonium isotopes. It illustrates the asymmetric character of fission events, where a light fission fragment and a heavier one are generally produced. Symmetric fission (mass numbers in the range 110-120) is less often observed, even less for reactions with thermal neutrons, in particular for the fission of ^{235}U. Fission yield disparities according to the fission reaction are reported in Figure 2 for the radionuclides of highest interest (those identified in Table 2). Although the fission yield remains within the same order of magnitude for most of those nuclides, notable differences is observed for several noble metals – ^{106}Ru, ^{107}Pd and ^{110m}Ag – as well as for two isotopes of Europium ($^{154, 155}$Eu).

One should note that the values provided in Figure 1 and Figure 2 do not account for further activation of the fission products. As discussed in § 5.4.1, the disposed inventory of ^{85}Kr and ^{129}I is also much reduced when reprocessing is applied, because of discharge to atmosphere (~100% of ^{85}Kr) and to oceans (98 – 99% of ^{129}I); this has a strong impact on long-term dose assessments in the case of ^{129}I.

Advanced fuel cycle options also study the possibility to transmute some of the long-lived fission products (^{99}Tc, ^{129}I, ^{135}Cs). Still, the generally low reaction cross sections and the need for isotopic separation – to avoid activation of stable or short-lived isotopes of those elements – compromise the feasibility of transmutation of long-lived fission products. No separation would be needed for ^{99}Tc considering that it has a single long-lived isotope and no stable isotope; separation of Tc is also relatively straightforward with existing methods like PUREX, so that its transmutation could be envisaged. However, the absorption cross section remains relatively low, about 20 barn in the thermal energy range, so that high flux or long irradiation time would be required to substantially alter its inventory [12].

The P&C route would affect the waste form and the footprint of the geological disposal facility, but it has only indirect impact on the fission product inventory. Indeed, part of the waste (the heat-emitting fraction, for example Cs, Sr, Pu, Am) would be *stored* for a very long period, up to centuries, in surface, subsurface or geological *storage* facilities, and only later brought to final disposal. Although the inventory in the geological disposal facility is undeniably reduced during the extended storage phase, the process will at the same time require active safety systems and security measures. After disposal of all separated fractions, one falls back on quasi-identical radiological inventories, yet under different forms or volumes.

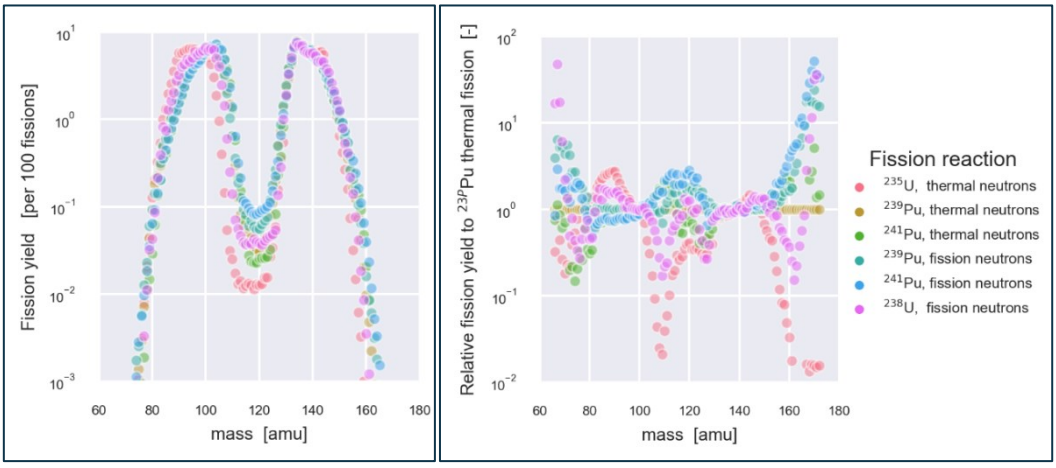

Figure 1 (Left) Fission yield and (Right) Relative fission yield to ^{239}Pu thermal fission following the fission of different nuclides in thermal and fast (fission) spectrum. Numerical data are derived from the ENDF/B-VI library [62]. The double peak shape reflects the trend for asymmetric fission events, where a light and a heavier fission fragment are generally emitted. ^{235}U thermal fission features a shift in fission yield peak for the lower mass fragment and a notably lower fission yield in the symmetric fission region (mass number 115 – 120). The data do not account for further activation of the fission products.

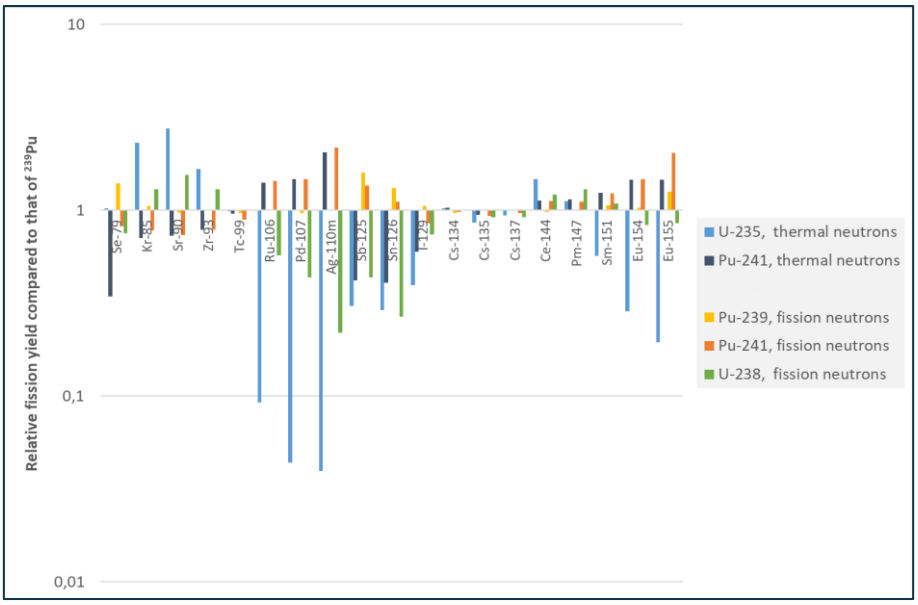

Figure 2 Relative cumulative fission yield for the mass series identified in Table 2, compared to ^{239}Pu thermal fission yield. For ease of interpretation, the corresponding critical radionuclides are identified rather than the mass series number. Data: [62].

Activation products

Several activation products are identified in Table 2, that are contained in structural parts of the assembly or in the cladding – as main alloying elements or as impurities – or in the fuel (impurities only); others may also be present in control rods (not covered in Table 2). One may anticipate the same isotopes to be of a concern in advanced fuel cycles, considering that, as of today, steel-based cladding or structure materials are anticipated.

Regarding the long-term safety, anions (e.g. $^{36}Cl^-$) are of uttermost concern in clay host rock and, to a lesser extent, granitic host rocks, together with organic materials (issue of ^{14}C). During reprocessing, the chlorine impurities present in the fuel may be expected to follow the same stream as ^{129}I – discharge to oceans, as of today–; hence a significant reduction could be expected. However, pyrometallurgical reprocessing processes may involve chlorine-based salts; in that case, chlorine impurities might still be present along the fuel cycle, get activated and end up in larger proportions in the waste streams compared to the OTC scenario.

For ^{14}C, the situation is more complex. ^{14}C generated by activation of impurities in claddings or structure materials will typically be disposed as such (OTC) or in compacted ILW waste following to reprocessing. The fraction of ^{14}C present in the fuel, also due to activation of impurities, is quoted to be discharged (>90%) upon reprocessing [25]. In advanced reprocessing schemes, the inventory of ^{14}C may however, significantly increase if carbide or nitride materials are considered as future fuel types. ^{14}C is indeed primarily generated by neutron capture in ^{13}C or ^{14}N (via (n, p) reaction then). Discharge of ^{14}C during reprocessing may then become a safety issue, to the extent that isotopic depletion / enrichment might be required for carbon (depletion of ^{13}C) or nitrogen, with enrichment required up to about 97% in ^{15}N.

Actinides

The main objectives of closing the fuel cycle are to improve the efficiency of natural resources usage and to diminish the long-term radiotoxicity of the waste, which is primarily due to actinides, in particular Pu and Am in the case of the OTC. No reduction of the inventory is expected in the P&C scheme, considering that only the waste form is affected, not the radiotoxic inventory.

TOP-MOX or regional synergy approaches would result in a much reduced actinide inventory, depending on the available reprocessing scheme: either solely the U and Pu fractions are reduced by about three orders of magnitude (conventional Pu reprocessing with the PUREX method), either a similar reduction is reached for minor actinides as well (advanced partitioning methods).

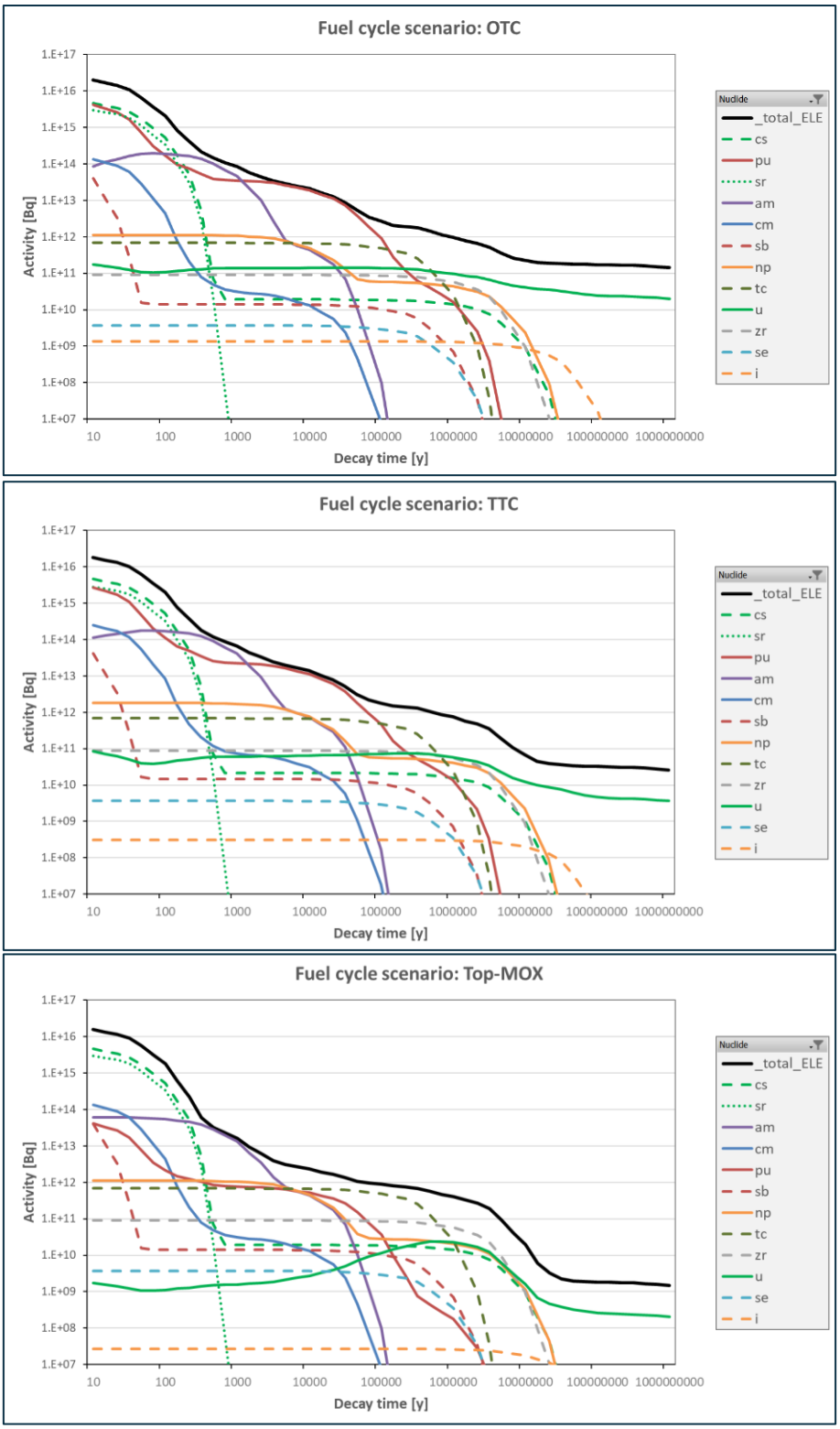

Figure 3 SCALE/ORIGEN [63] calculations of the activity evolution (including speciation), assuming an equivalent and simultaneous heat generation. The following fuel cycles are considered:

OTC: 1 t_{HM} of irradiated UOX fuel (4.25 wt.% enriched, 45 MWd/t_{HM} burnup), 10y cooling time.

TTC: 0.825 t_{HM} of UOX fuel (4.25 wt.% enriched, 45 MWd/t_{HM} burnup), reprocessed after 7y + 3y additional cooling time
+ 0.1 t_{HM} MOX fuel (7.5 wt.% Pu/(U+Pu), 45 MWd/t_{HM} burnup), 10y cooling time
+ 0.075 t_{HM} ERU fuel (4.6 wt.% enriched, 45 MWd/t_{HM} burnup), 10y cooling time.

TOP-MOX: vitrified waste from 1 t_{HM} reprocessed spent UOX fuel: 4.25 wt.% enriched, 45 MWd/t_{HM} burnup, 10y cooling time.

The LWR-MOX route (or TTC) is known to reduce the feed material by 15 to 20%. Although it results in a lower waste inventory in terms of mass per unit electricity produced, the gain in terms of radiotoxicity is limited because of the larger production of Am and Cm for that fuel cycle option, by a factor 2 to 3 (see reference [41], Table 26). The impact in terms of Pu inventory is often presented as beneficial compared to the OTC, but care has to be taken as the Pu inventory is sometimes considered as a resource [31], [41] and only a small fraction (0.1 – 0.2%) would follow the waste stream. If one hypothesizes disposal of that Pu inventory, the gain on Pu inventory becomes more limited, of the order of 30% [41].

Considering the variety of underlying hypotheses in the literature (e.g. regarding which waste are stored or disposed of, on the separation efficiency etc.), we report here an independent, small-scale, comparison of the OTC, TTC and TOP-MOX approaches; it was conducted for an equivalent energy output[6] by mean of the SCALE 6.2.3 code [63]. The calculations were performed with the ORIGEN module and pre-built ARP libraries. A comparison of the HLW activity of the OTC (1 t_{HM} spent UOX fuel), TTC (vitrified waste from 0.8 t_{HM} spent UOX fuel + 0.1 t_{HM} spent MOX fuel + 0.1 t_{HM} spent ERU fuel) and TOP-MOX (vitrified waste from 1 t_{HM} spent UOX fuel) approaches is illustrated in Figure 3. Figure 3 shows the appreciable gain for the TOP-MOX approach beyond 10 000 y decay time, where U and Pu activity dominates in spent UOX or MOX fuels. For illustration purposes, the activity evolution in the case 1 t_{HM} spent MOX fuel is reported in Figure 4; a one-to-one comparison of the activity with the previous cases can, however, be misleading, as one does not consider the performance of the first irradiation pass (UOX fuel irradiation) in the MOX-only irradiation case.

Advanced partitioning schemes, as reached in the FR-MOX and the P&T routes, have a much stronger impact on the actinide inventory as Pu (and other MA for the P&T route) are reprocessed as nuclear fuel at each cycle. Only a small fraction of the Pu is lost during reprocessing: a separation efficiency of 99.8 to 99.9% is often quoted; considering that several tens cycles are still required to consume the Pu inventory, a global reduction of the Pu inventory by a factor 100 compared to the OTC seems a fair order of magnitude. Regarding minor actinides, the FR-MOX option still results in a larger Am inventory – a factor of 5 is quoted in [41], Table 26 – but some reduction is expected for Np, by 70%, and 30% for Cm [41], Table 26. Based on similar partitioning efficiency, then applied to both Pu and MA, the P&T route would result in a reduction of their inventory by a factor of 100 compared to the OTC; even larger efficiency, up to a factor 1000 compared to the OTC, could be reached if one considers that larger fuel burnups could be achieved in the long term, reducing the number of reprocessing steps (and associated losses).

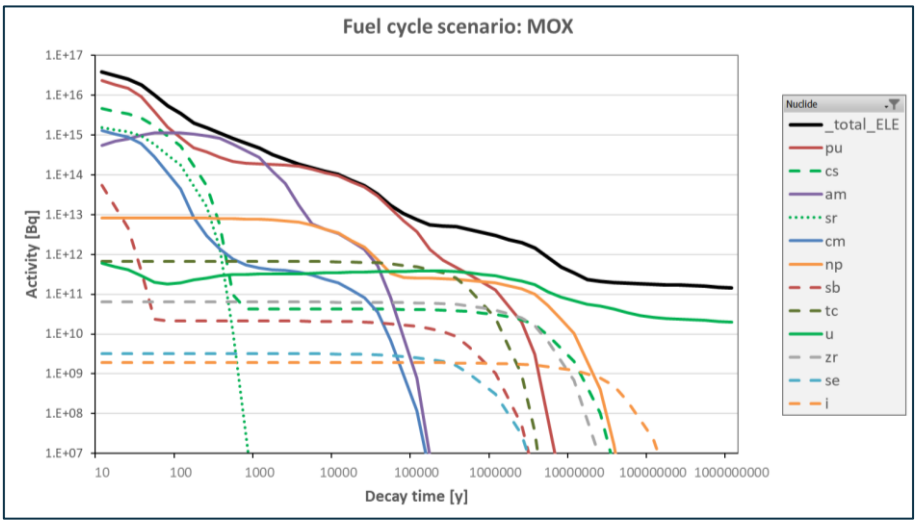

Figure 4 SCALE/ORIGEN [63] calculations of the repartition of the activity evolution for 1 t_{HM} spent MOX fuel (7.5 wt.% Pu/(U+Pu), 45 MWd/t_{HM} burnup, 10y cooling time). A one-to-one comparison with the graphs of Figure 3 is misleading, as the energy produced in the first irradiation pass (UOX fuel irradiation) is not accounted for in the MOX-only irradiation case.

[6] Reference case: 1 t_{HM} UOX spent fuel with a burnup of 45 MWd/t_{HM}

❖ *Radiotoxicity*

The spent fuel or vitrified waste inventory is often presented in terms of its total radiotoxicity rather than its activity, in order to account for the more harmful character of certain isotopes. The conversion is conducted, for each isotope, on the basis of dose coefficients such as those of ICRP [64], [65]. Care must be taken in interpreting fuel cycle comparison based on radiotoxicity solely. There is a trend in literature to reduce the comparison of different fuel cycles to the relative radiotoxicity, using as a reference, natural uranium or natural uranium ores – the comparison is conducted either on the basis of the equivalent amount of feed uranium material for the OTC cycle, either at equivalent mass between the spent fuel and a prototypical uranium ore. This reference level makes sense for (and takes its roots in) human intrusion scenario assessment, where the approach provides qualitative risk comparison with drilling through a uranium-containing rock. There are, however, several drawbacks in extrapolating such approaches beyond the human intrusion scenario context:

- Natural uranium still poses a safety hazard, so that it is misleading to interpret the time at which the reference level is reached as the required isolation timeframe.
- Natural uranium, in secular equilibrium, has a larger activity than the uranium feeding the fuel cycle as the secular decay products are removed upon refining.
- The depleted uranium produced during the enrichment step is often neglected in the assessment (it remains considered as a resource for advanced fuel cycles).
- A separation efficiency of 100% is sometimes assumed for advanced fuel cycles (i.e. no MA go to disposal), while more realistic figures are 99.8 to 99.9% per reprocessing cycle; it leads to an effective reduction of the disposed actinide inventory by a factor 100 to 1000.
- Radiotoxicity is not representative of the expected exposure pathways for future generations and environment, it is rather a straightforward evaluation, solely based on the inventory, that reflects exposure in some alternative evolution scenarios, such as human intrusion.

Some of these concerns are perfectly illustrated with Figure 5, where the intake hazard is expressed in terms of the US annual limit on intake (ALI) – corresponding to an annual dose of 50 mSv [66]. The TTC route never reaches the radiotoxicity of the same mass of uranium ore, as the concentration of uranium – dominating radiotoxicity in the long-term – in spent fuel and uranium ores differ by at least two orders of magnitudes. Multi-reprocessing (red curve), postulated with a 99.5% efficiency, leads to a reduction of the initial U and Pu but does not affect minor actinides; a reduction on the ALI index is observed at intermediate cooling times (around 10 000 y) when Pu decay dominates the radiotoxicity, and in the far long-term (> 10^7 y) when uranium radiotoxicity dominates. Advanced reprocessing schemes with 99% separation efficiency for MA (green curve) bring the spent fuel to the ALI index of natural uranium ore after about 10^7 y. Only with a fully efficient separation of actinides (only FP remaining), can one observe a huge reduction of the ALI index after a few hundred years. It should, however, be observed that long-lived fission products feature a radiotoxicity plateau beyond 1000 y, until decay in the 10^5 – 10^7 y range.

One may find in the literature (e.g. [41], [67]) similar graphs under slightly different hypotheses: full efficiency of the separation process, or a reference level taken as the radiotoxicity of uranium ore or that of natural uranium. An example arising from [67] is provided in Figure 6; misleading perceptions could be induced by considering that reference level as an acceptable risk or dose. Moreover, the plotting range selection for radiotoxicity ignores the radiotoxicity plateau of the fission product contribution – due to the long-lived fission products – and may suggest a smooth further decrease beyond the frame of the graph. Literature results have therefore to be taken with a critical thinking; in particular, the reduction of radiotoxicity with advanced fuel cycle appears not sufficient to relieve the need for disposal facilities performing over very long timeframes.

A comprehensive comparison of different options for the fuel cycle has been conducted in the frame of the EFP6 RED-IMPACT project [25], [68], [69]; the gain from a radiotoxicity perspective is illustrated in Figure 7. In the long term, where the contribution from actinide dominates, a small gain is observed for the TTC (about 20%); larger reductions by a factor 10 and 100 is observed for the FR-MOX and the P&T routes, respectively. Regarding the impact of a TOP-MOX or regional synergy approach on the waste radiotoxicity, a NEA study presents the gain on waste radiotoxicity for the exporting country (A); the gain by a factor of 1000 translates the postulated separation efficiency of 99.9%. The waste radiotoxicity for

the accepting country (B) is also reported; it is further subdivided in a component of waste produced before full P&T technology is available (postulated to occur in 2040 in [39]) and that for the waste produced after the technological implementation.

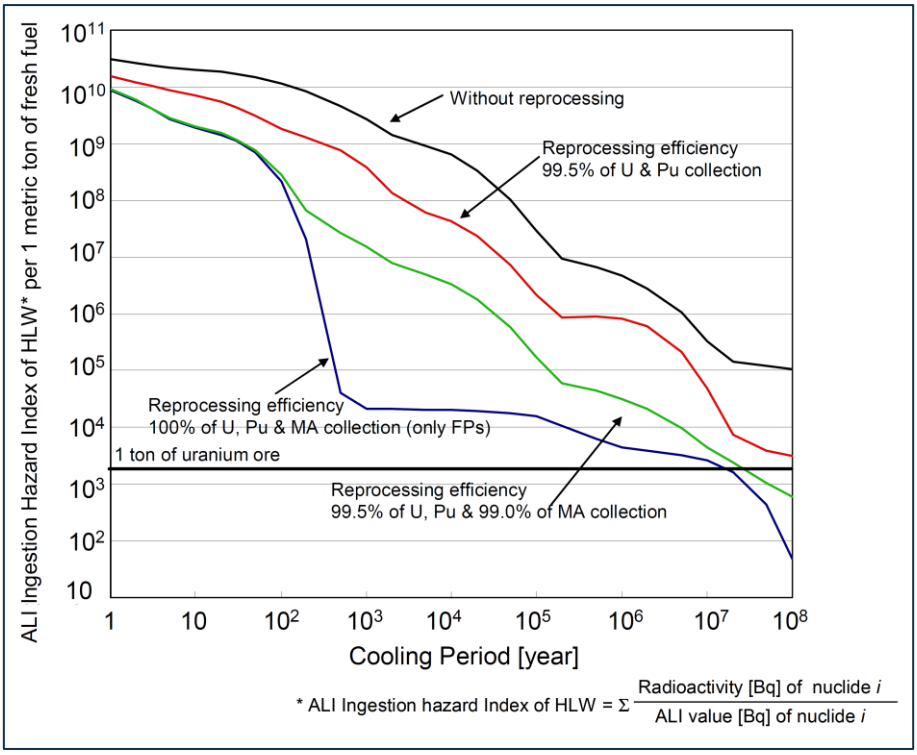

Figure 5 Radiotoxic hazard of 1 t_{HM} spent MOX fuel under different reprocessing assumptions. Hazard is expressed as the annual limit on intake (ALI), which corresponds to an annual dose of 50 mSv, considering the entire inventory would be ingested or inhaled. The hazard associated to 1 t uranium ore is also reported, with an ALI index around 1000. Source: [13] (© IAEA, 2010).

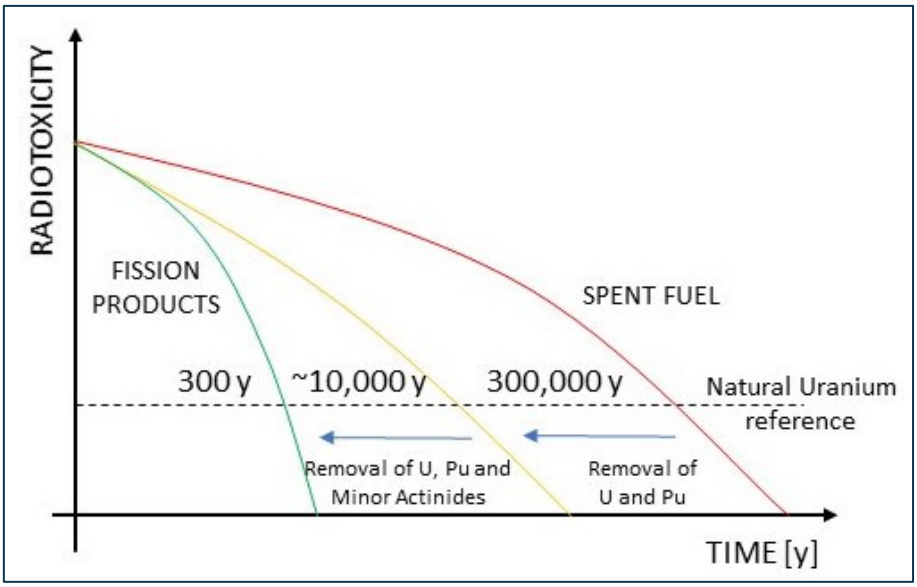

Figure 6 Radiotoxicity of spent nuclear fuel with natural uranium ore as a reference level, for different fuel cycle options. Source: Francesco Galluccio, available under CC BY-SA 4.0, via Wikimedia Commons, in [67].

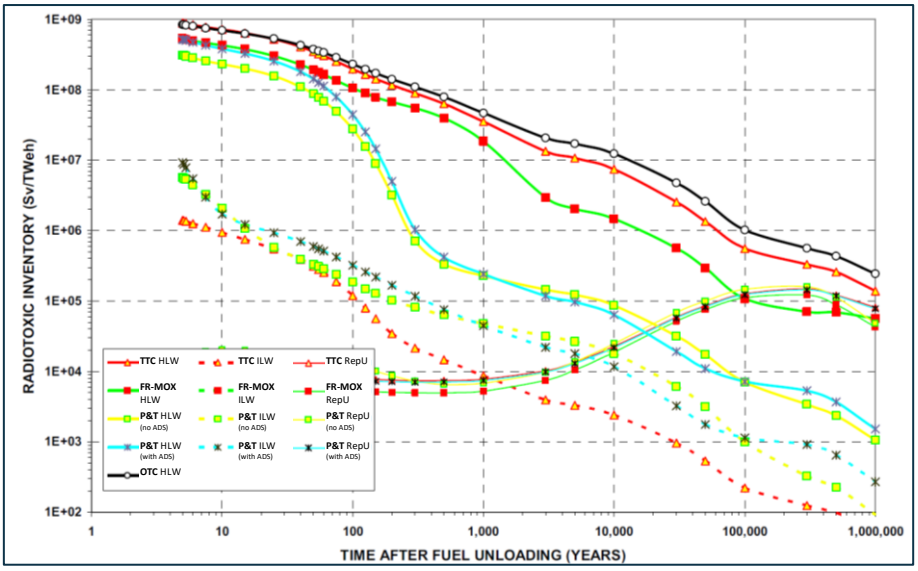

Figure 7 Radiotoxicity evolution in different fuel cycle scenarios – OTC, TTC, FR-MOX, P&T (2 variants: with or without ADS) – for the HLW, ILW and reprocessed uranium fractions. Source: [25] (© FZ Juelich, 2008; reproduced with permission); the legend of the original Figure was adapted to the context of the present work.

❖ *Decay heat*

The activity of a set of nuclides can be expressed in terms of decay heat by means of the energy release per decay event, which is specific to each individual decay process, following a similar approach as for the conversion to radiotoxicity. Decay heat evolution with time remains qualitatively similar to that already presented for waste activity and radiotoxicity; it is therefore only briefly described in this section. Over the first hundred years, decay heat is dominated by two couples of nuclides (β^-- followed by γ-decay): $^{137}Cs/^{137m}Ba$ and $^{90}Sr/^{90m}Y$; the small gap observed between the various routes at short cooling time mostly reflects the difference in energy conversion efficiency considered for light water reactors and fast reactors. The P&C route has been proposed to address the high decay heat fraction due to fission products, by separating Cs and Sr from other waste streams. Surface storage of Cs- and Sr-containing waste for at least one hundred year would enable to substantially reduce the decay heat of the remaining waste packages that could directly be disposed of.

Decay heat is an important parameter in designing an HLW disposal facility, since temperature may affect the performance – even the integrity – of the engineered or natural barriers. In particular, the temperature increase of clay is of paramount importance for geological disposal facilities in clay host rock, but also for other cases where clay is used as engineered barrier (filling material). This is further detailed in § 5.4.4.

❖ *Criticality safety*

The term criticality reflects the ability to sustain a nuclear reaction chain and is often characterized by an effective multiplication factor, k_{eff}, which, in simple terms, translates the average number of subsequent neutrons produced in the system by a source neutron. The chain reaction is maintained at $k_{eff} = 1$, when the number of neutrons produced equals that of neutrons lost (by capture or escape) by the system. While in a nuclear reactor, criticality is desired to produce heat in a controlled way, subcriticality must be guaranteed at the other stages of the fuel cycle, including disposal. Critical configurations require a proper combination of amounts of fissile, moderator and absorber materials and their geometrical arrangement; other parameter may also affect the level of criticality, such as material temperatures. Criticality in disposal facilities can be avoided by adjusting any of those parameters but improved robustness is achieved if one plays on a combination of those factors.

Criticality considerations also arise from safeguards perspective in the fuel cycle (cf. § 5.3). The focus then relates to the risk of diverting fissile material to obtain nuclear explosive devices, although the spread of radioactive materials by conventional explosives remains of a concern. One then mostly addresses the amount of fissile material needed to achieve bare-sphere fast criticality – the most

penalizing geometry in absence of moderator. The presence of gamma-emitting isotopes, leading to handling concerns in view of shielding requirements, and that of absorbing material or isotopes with a much larger critical mass – then requiring the diversion of more material – constitute additional proliferation resistance features.

In terms of criticality safety for a nuclear facility, in particular a storage or a disposal facility, the amount of fissile material present may still, by far, exceed the bare-sphere critical mass. The risk of fast criticality – i.e. criticality in absence of neutron moderation – remains, in general, very low thanks to the material composition of the waste (e.g. absence of pure materials) and the geometrical configuration (including the separation of the waste packages). The inventory of fissile material in a storage or disposal package is sometimes quoted to be insufficient to reach fast criticality without an additional separation step [70], but some sequences of events could still lead to the formation of a critical mass outside canisters by selective dissolution and precipitation.

The risk of criticality due to neutron thermalization remains the major concern from criticality safety perspectives; it is generally controlled by the geometrical configuration, the presence of absorbers and the removal of moderating materials but the safety assessment must also consider phenomena that would lead to changes in geometrical configuration or in material composition (e.g. water ingress in the system, removal of absorbers). In case burnup credit is applied, the evolution of the multiplication factor with time should be assessed, since the fissile content varies with time, as a consequence of the decay process. The largest k_{eff} values are expected at end of irradiation for spent UOX fuel (see Figure 8) and decreases rapidly due to ^{241}Pu decay (^{241}Am has a larger capture cross-section than fission cross-section in thermal spectrum). A minimum of k_{eff} is observed after about 100 y, when it increases again in view of the decay of ^{241}Am (to ^{237}Np), then of ^{240}Pu (to ^{236}U). At a few tens of thousands of years, a first peak is then observed that remains lower than the k_{eff} level at the end of irradiation. It is followed by a moderate k_{eff} reduction as ^{242}Pu decreases to ^{238}U; the plateau reached after 10 My corresponds to ^{237}Np decay.

The disposal of spent fuel elements should, hence, demonstrate sufficient safety margin in disposal conditions, assuming water ingress and other penalizing processes. For vitrified waste, the fissile inventory is reduced to U, Pu process losses and, possibly, minor actinides. That inventory is generally deemed insufficient to raise criticality issues [25]. Moreover, the glass matrix also offers both some 'dilution' of the remaining actinides in a larger volume and impedes water ingress within the waste form; these aspects also contribute to further reducing the risk of criticality events. Concerns remain for the P&C route, where it is not clear yet how uranium and plutonium would be conditioned if not considered as resources; it might be anticipated that this route offers some flexibility in managing the U and Pu inventory so as to reduce the risk of criticality compared to the direct disposal of spent fuel elements.

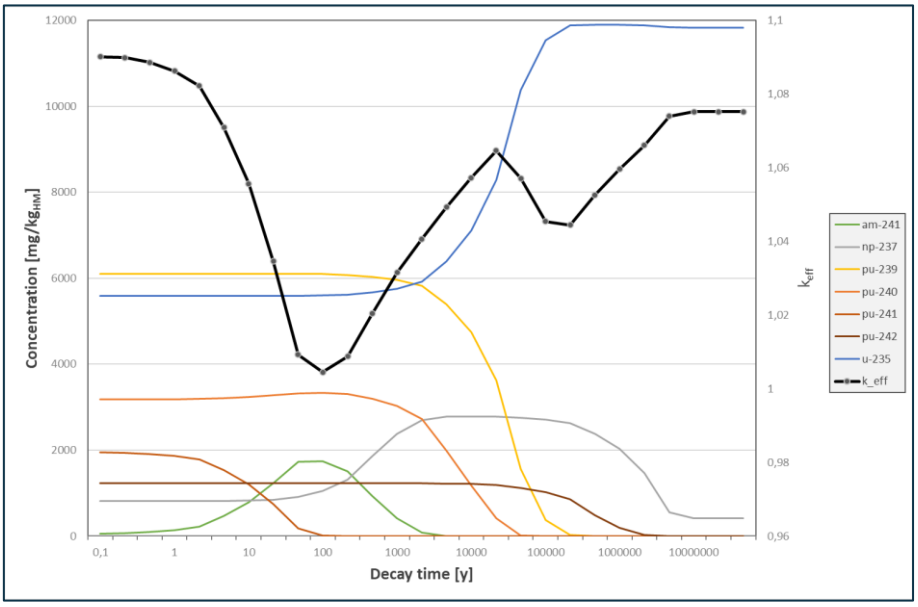

Figure 8 k_{eff} evolution with decay time for spent UOX fuel (average burnup of 55 MWd/kg$_{HM}$) for an infinite assembly lattice in water; the additional curves reflect the concentration of various actinides isotopes. Calculations performed with SCALE 6.2.3 [63].

In conclusion, from criticality safety perspectives, the preferred options are those involving multi-reprocessing (FR-MOX, P&T) and regional synergy strategies in view of the reduced U, Pu and minor actinide content, while equivalent performance may be attributed to the OTC and TTC routes. We anticipate the P&C route to offer an intermediate character, considering that the separation of fissile material from the spent fuel and the conditioning of those substances can be conducted with intrinsic counter-measures to avoid criticality – one may think for example about dilution with depleted uranium.

5.4.3 Long-term safety

With advanced partitioning and transmutation schemes, the radiological inventory of spent nuclear fuel and its waste forms could be heavily modified to an extent that the possibility of surface or near-surface disposal of the ultimate waste is sometimes investigated [14], [15]. Often, the comparison of different fuel cycle scenarios is conducted purely from radiotoxicity perspectives without consideration for the type of repository [12], [41], [47], [71], [72]. The repository characteristics are, however, known to result in speciation of the radionuclide releases to the environment; for example, actinide elements show a large increase of solubility as one moves from reducing to oxidizing conditions [46]; clay materials are also known to retard cation migration by sorption on clay mineral surfaces [25].

Comprehensive comparisons of fuel cycle options that account for both inventory and transport to the biosphere to derive the radiological impact to people and environment, are scarce in the literature. Many of the references found actually relate to the EFP6 RED-IMPACT project, where the dose to population was investigated for different repository concepts (in clay, granite or salt host rocks) and five different fuel cycles [25], [46], [68], [69], [73]. Some of the results obtained in that study are illustrated in Figure 9, for a repository in granite and clay host rocks. Both host rock feature similar trends: over the first million years dose is primarily due to fission and activation products – by order of importance in the OTC scenario: ^{129}I, ^{36}Cl, ^{14}C, ^{79}Se and ^{126}Sn; ^{135}Cs plays a role in granite in the range 1 – 10 million years but is expected to remain sorbed in the case of clay host rock. The iodine dose peak is lower in scenarios involving reprocessing; this does not reflect better performance of the vitrified HLW, rather the reduced expected inventory because a significant discharge of iodine to oceans took place at the reprocessing plant. ^{14}C is not predicted to play a major role in dose to the biosphere under the expected evolution scenarios, except in the case of P&T, where a larger inventory could be present due to activation of structure materials or the use of nitride or carbide fuel (cf. § 5.4.2).

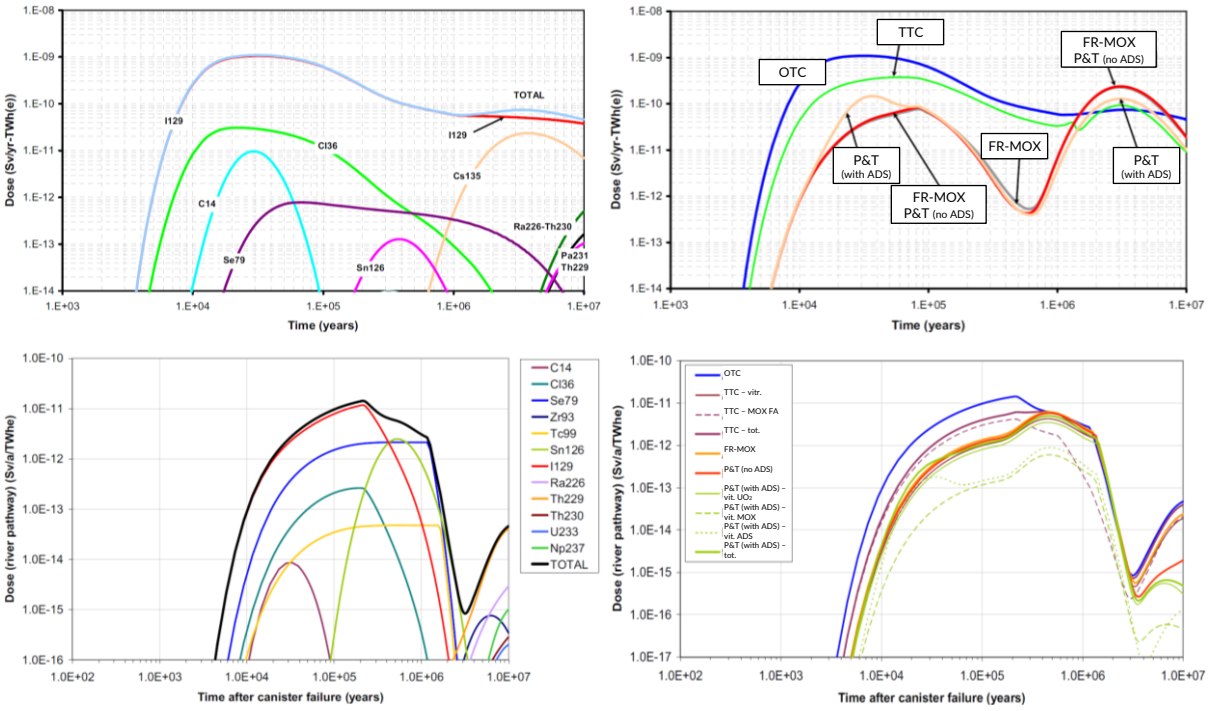

Figure 9 Dose to biosphere from a geological repository in granite (top graphs) and in clay (bottom graphs). The partition according to the radionuclides is illustrated for the OTC scenario (left column); a comparison of five fuel cycle scenarios is also reported (right column). Source: [25] (© FZ Juelich, 2008; reproduced with permission); the legend of the original Figure was adapted to the context of the present work.

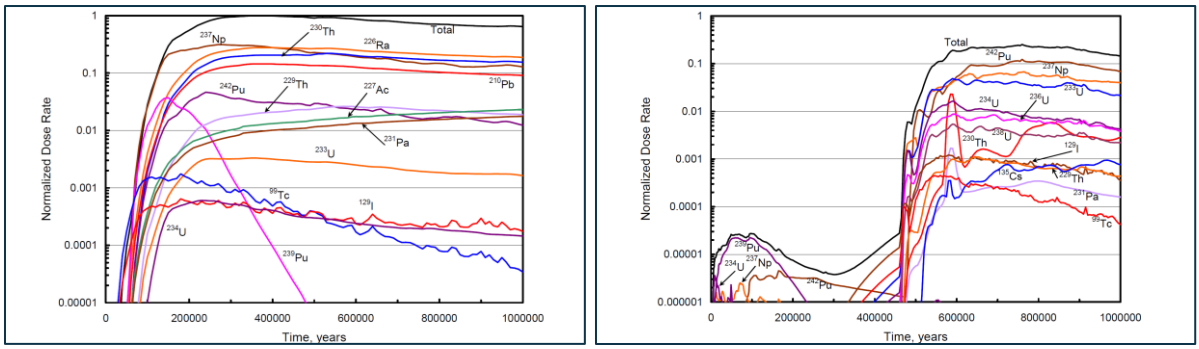

Figure 10 Relative total dose to environment and contribution of individual nuclides according to two models for direct disposal of spent fuel assemblies in the Yucca Mountain site. Contrarily to other repository designs, the dose is dominated by actinides. Source: [74] (Image courtesy of Argonne National Laboratory).

A second peak in the total dose to the biosphere is observed in the very long term (>1 – 10 million years), where the slow transport of actinides and their decay product play a dominant role. The impact of the fuel cycle scenario, with a large reduction of the actinide inventory in the FR-MOX and the P&T options, is primarily observed there. Those conclusions cannot be extrapolated to any design; the example of the Yucca Mountain site is provided in Figure 10, where dose over the first million year is dominated by the contribution of actinide isotopes, in view of the high solubility of these elements in water under oxidising environment. A larger impact of fuel cycle scenario on the dose to biosphere is expected in such conditions, although some studies point out that the dose reduction is not proportional to the inventory reduction, because the radionuclide flux to the biosphere is not limited by the waste matrix dissolution rate, rather by the solubility of minor actinides and fission products in underground waters [75].

5.4.4 Confinement / containment aspects

❖ *Repository design characteristics*

The fuel cycle scenario has an indirect impact on the repository design, considering that most of the design choices rather relate to the repository environment, in particular, the host rock in the case of geological disposal. By far the largest impact on the design arise from decay heat considerations: thermal aspects mostly drive the spreading required between waste packages in storage and in disposal facilities. In the case of geological disposal facilities, the temperature increase in the host rock or in some engineered barriers must be constrained to guarantee their long-term performance. This, in turn, affects the gallery length and spacing and, hence, the repository design and its cost. The footprint is obviously related to the extent of the nuclear programme; a fair comparison of the various options is therefore to consider an equivalent amount of energy produced.

For disposal in a clay host rock and a temperature limit of 100°C, [46] quote figures of about 350 W/m linear thermal load in the galleries; this correspond to an HLW gallery length of about 6m per TWh_e for the OTC; much compacter configurations are allowed in other geological formations, for example with gallery length of about 2 m/TWh_e in tuff host rock [23].

Slightly larger thermal loads seem possible for the P&T route, considering the faster decrease of the decay heat and the thermal inertia offered by the waste packages and the EBS [25]. The lower actinide inventory in vitrified waste also allow for a more compact repository design: from 40% gallery length reduction in MA-containing vitrified waste to 60 – 70% reduction when MA are reprocessed as fuel [13], [25], [46]. Even better performance, with a reduction of gallery length by an order of magnitude, is achieved in scenarios where Cs and Sr are separated and kept in storage for one hundred years [46].

❖ *Waste package and engineered barriers ageing*

Waste packages

In terms of waste packages, differences are to be expected between the different fuel cycle options, considering that spent fuel elements are disposed of in the direct disposal route (spent UOX fuel assemblies for LWR). Part of the high level waste also consist of spent fuel assemblies in the plutonium mono-recycling routes, then as spent MOX fuel assemblies, in combination with vitrified waste resulting from the reprocessing of UOX fuel assemblies. All routes based on reprocessing feature waste

conditioning that can be specifically designed for disposal. Today, waste vitrification in borosilicate glass is generally considered for high level waste, which show lifetime of several tens to hundreds thousands of years [25]. The lifetime of the matrix has a large effect on the estimated dose due to radioactive anions, while the effect is less sensitive for cations (cf. Figure 11). Cations are indeed less mobile; the peak dose induced by long-lived cation fission products occurs over million years timeframes, much beyond the waste package lifetime.

A promising candidate for future waste is Synrock, with lifetime larger than a million year [25], [76]; other matrices are also evoked for their stability, such as monazite, pyrochlore, zircon or zircolite [73]. One therefore anticipates that the waste packages developed for treating the waste streams of reprocessing routes may feature at least equivalent performance to directly disposed spent fuel.

Spent fuel elements were primarily designed and optimized for their irradiation performance; still, the dissolution of the uranium dioxide matrix seems to occur at a sufficiently low rate if appropriate conditions (reducing conditions) prevail [13]. In [13], one quotes a much lower solubility of the uranium dioxide matrix, by four orders of magnitude, compared to silica; it favours direct disposal of the spent fuel elements from this perspective. The (incomplete) segregation of several elements within the spent nuclear fuel rods, with fractions in the rod free volumes, at grain boundaries, in separate phases or as solute in the uranium dioxide matrix, leads to 'burst releases', from a geological timeframe perspective, that are not expected for the releases due to the progressive dissolution of vitrified waste [77]. One then generally distinguishes the "accessible fraction of the inventory" (AFI) – also called the "instant release fraction" (IRF) – which is the fraction that is rapidly released upon water ingress into the cladding, from the releases occurring in the wake of matrix dissolution. Hence, the degradation of spent fuel elements directly disposed in geological disposal conditions seems sufficiently good that the partitioning and conditioning is not required [14].

Engineered barrier system (EBS)

Scarce information has been found in literature, addressing the impact of reprocessing scenarios on engineered barriers or on their degradation mechanisms. From generic considerations, one expect no appreciable direct influence of the fuel cycle scheme on the main design features of the geological disposal facility (e.g. host rock, depth, engineered barriers etc.), considering most of the EBS are designed for their compatibility with the host rock formation [23]. Still, some characteristics of the engineered barriers seek to compensate or to delay waste package degradation.

Indirect effects may result from differences in the dimensions of the waste packages (e.g. allowing for smaller or shorter galleries), or from the differences in decay heat (e.g. allowing for reduced gallery spacing), resulting in a smaller geological disposal facility footprint.

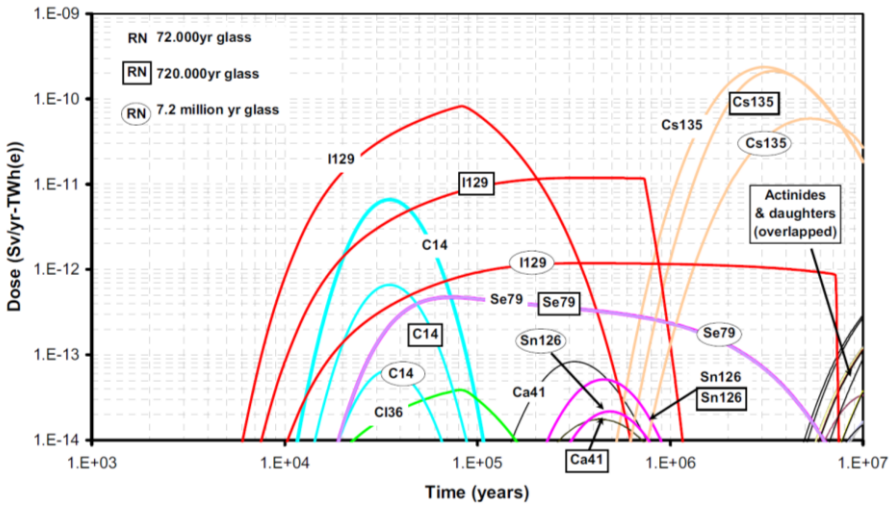

Figure 11 Effect of the glass matrix lifetime on the dose to population, for a repository in granite. The impact is most appreciable on anions (e.g. ^{129}I, ^{36}Cl), whose diffusion is limitedly delayed by sorption in clay or granite. Source: [25] (© FZ Juelich, 2008; reproduced with permission).

5.5 Impact on robustness

5.5.1 Altered evolution scenarios

Altered evolution scenarios (AES) are scenarios that diverge strongly from the expected evolution of the geological disposal facility, the host rock and their environment. Literature comparing the performance of different fuel cycles under altered evolution scenarios is scarce; the analysis will therefore rely on generic considerations and the synthesis report from an EC FP6 Project, RED-IMPACT [25], although only a repository in salt was considered from that perspective.

In general, altered evolution scenarios result in accelerated transport in the host rock but the qualitative aspects remain: the mobility of several long-lived fission and activation products are the largest concern under reducing conditions [25], [46]. For some of the scenarios[7], however, oxidizing conditions may prevail, which then results in high solubility of some of the actinides [73]. Actinide mobility is for example of a concern for the Yucca Mountain site in USA [46], [73], [78]; the host rock is a non-saturated volcanic tuff that features oxidising conditions and, hence, higher solubility of actinide elements. Fuel cycle scenarios with reduced actinide inventory then perform better from AES perspectives.

5.5.2 Human intrusion

Inadvertent human intrusion scenarios (HIS) are a particular kind of altered evolution scenarios that cannot be excluded for any disposal concept. Considering that a probability of occurrence is very difficult to estimate, such scenarios are systematically addressed in the safety case [79], [80]. The assessment is conducted from two perspectives, a qualitative one with design changes implemented to reduce the likelihood of intrusion and a quantitative one addressing the consequences of the intrusion, then with specific safety requirements for such scenarios.

On the one hand, the qualitative approach evaluates additional measures that are taken to reduce the likelihood of inadvertent human intrusion – depth, additional barriers, siting far from known geological resources… On the other hand, a conventional dose assessment is performed. Two types of exposition are then usually considered: that due to direct exposure to the excavated material – the intruders themselves, or local populations and environment when the excavated material is left at the surface – which generally relates to the radiotoxicity of the waste, and enhanced long-term contribution to the dose to local populations and environment, due to additional migration pathways created in the wake of the intrusion. Although the total inventory to be disposed of highly depends on the extent of the nuclear program – i.e. the total amount of energy produced –, human intrusion scenarios have a pronounced local character relative to the entire disposal facility. Only a small fraction of the inventory then gets affected, so that the impact of the different technologies on human intrusion scenarios is better appreciated on relative metrics; for the purpose of simplicity, we have limited ourselves to an identical amount of gross energy produced.

The fuel cycle technology has limited influence on the main design features of the geological disposal facility (e.g. host rock, depth, engineered barriers etc.), so that only indirect impact may be expected on human intrusion scenarios. Relative to the amount of energy produced, the reduced actinide inventory in closed fuel cycles may positively impacts the dose for both exposition pathways, considering that actinides dominate the long-term radiotoxicity of the waste. This is, however, somewhat balanced by the more compacted design of the repository with routes derived from reprocessing or by the more radiotoxic character of several minor actinide isotopes that are produced in larger amounts in the reprocessing routes [14]. Another aspect of human intrusion, already addressed with the non-proliferation concerns in § 5.3, relates to the attractiveness[8] of the waste, in particular the plutonium inventory. With respect to this criterion, risks of human intrusion is more largely reduced in advanced fuel cycle scenarios [71].

In terms of fission products, the inventory remains similar in all approaches considering that the released energy per fission limitedly varies according to the nuclide undergoing fission; for a given fission product,

[7] For example, penetration of oxidizing glacial melt water in bentonite buffers in the Swedish or Finnish concepts [46].

[8] One then considers intrusion motivated by access to the waste, rather than circumstantial or curiosity-driven intrusion. This is, however, not covered in IAEA SSR-5 [80], where only inadvertent intrusion is considered.

some variation is, however, observed in the fission yield of the different actinide isotopes. Only the waste package characteristics may then further affect dose and dose rates to an extent that is difficult to quantify from generic considerations. Indeed, one anticipates better degradation resistance of spent fuel compared to vitrified waste in presence of underground water and reducing conditions [13], although both should feature slow degradation. In the case of local perturbation, this gets balanced by concentration differences – and, hence, availability – of radionuclides in those vitrified waste.

In terms of human intrusion, the most penalizing scenarios therefore seem the MOX route when it involves the direct disposal of spent MOX fuel assemblies. It is followed by the open cycle in view of a lower concentration of plutonium and minor actinides in the UOX spent fuel assemblies to be disposed of. Vitrified waste produced in the FR-MOX, P&T and P&C routes, features a more pronounced reduction of the actinide inventory; this results in only slightly better performance, considering that the risk associated to fission products remains similar in all options.

5.5.3 Contextual uncertainties

- *Financial aspects*

Examination of financial aspects of the nuclear technology do not belong to FANC's attributions. However, a lack in financial resources could lead to impair the achievements of the safety objectives by or lead to a precipitated implementation of a waste management solution. Financial aspects are therefore briefly addressed to crudely assess the credibility of the different options.

Different studies have addressed a comparison of the total electricity generation cost from the various fuel cycle options [23], [25]. In general, one observes the lowest cost for the OTC cycle, closely followed by the TTC. The use of fast reactors for multiple recycling of plutonium results in an increase of 10 to 15% of the total, while the P&T option may lead to an increase up to 20% compared to the OTC. Cost indicators remain difficult to establish, particularly for technologies that are not yet at a level of industrial demonstration; they are also derived from a series of hypotheses that need to be assessed from a country's own perspective. They may, for example, result in more weight on marginal costs than fixed costs by assuming scaling factors associated to larger nuclear programme. All in all, quoted uncertainties are larger than the spread of cost estimate, so that financial aspects may not be decisive factors as such.

In terms of cost repartition, investment costs, and reactor operations and maintenance weight more than 80% of the total electricity generation cost; the fuel cycle itself, including waste disposal, only accounts for a small fraction (<20%) of the electricity generation cost [23], [25]. Costs associated to waste disposal weight a few percent (about 5% in the OTC) but they have become of societal concern, because they may be faced much beyond the closure of a national programme and be transferred to the next generations (such aspects are further discussed in "Intergenerational equity"). These concerns are partially addressed by accumulating provisions for dismantling and disposal, as observed in different countries.

- *Waste retrievability*

The decision-making process associated to geological disposal facilities is a step-by-step process that, coupled to the evaluation of alternative options, provides the flexibility to come back on previous decisions [80, Para. 1.19]; this generally covers the reversibility of waste emplacement operations (e.g. in French regulation [17]) but may also cover the possibility to retrieve[9] the waste after their emplacement [80, Para. 1.20]. The desirability to retrieve the waste packages indeed faces a growing societal debate [73] and is sometimes included in the prevailing circumstances for disposal of radioactive waste. In view of the very long stability of waste packages, both in case of direct disposal of spent fuel elements and of vitrified waste, reversibility of operations or retrievability of the waste packages in the post-closure phase is driven by the repository concept rather than by the waste package design. Limited

[9] A distinction is made between reversibility, retrievability and recoverability of the waste emplacement, according to time at which it takes place, the means needed to perform the operation and the integrity of the waste packages. Reversibility implies bringing back to surface, during the operational phase, the waste with the same equipment that was used for waste emplacement. Retrievability relates to an imposed constraint to maintain, during a certain period after emplacement, the ability to bring back integer waste packages to the surface by means that may differ from the original equipment used for emplacement. Recoverability relates to the same operations while integrity of the waste packages is not guaranteed anymore.

difference is then expected for all fuel cycle scenarios from this perspective; the only complications would arise from higher temperatures during the thermal phase (OTC, TTC, P&C) or from dose constraints, but these factors may also be present in the FR-MOX or P&T options in view of more compact layouts.

Another aspect of retrievability relates to the recovery of valuable resources present in the waste, in particular taking decisions that would compromise future fuel cycle scenarios [14]. Only in the case of direct disposal of spent fuel and in the partitioning and conditioning routes, one might identify the will to recover uranium, plutonium and, possibly, minor actinides. Vitrified waste contain only traces of uranium or plutonium, but may contain minor actinides in appreciable quantity; however, considering that minor actinide extraction from existing vitrified waste seems not considered economically viable [12], [71], [73], the recovery of those waste from a disposal facility will only result in additional cost and is therefore to be excluded. Recovery of fission or activation products, although present to various degree in spent fuel, neither seems economically viable.

❖ *Policy changes*

Various policy changes addressing the long-term management of radioactive waste can be imagined. One focuses here on bounding cases, taken as prevailing circumstances (i.e. without any judgement on their appropriateness), that have the largest expected impact on the existing waste and spent fuel:

- Technological breakthrough in the fuel cycle or waste management
- Phase-out of the nuclear energy programme over a relatively short timeframe

Technological breakthrough in relation to the fuel cycle or to waste management

Technological breakthrough, in terms of alternative solution to geological disposal for the long-term waste management, may be expected to be little dependent on the fuel cycle and on the waste form, so that this aspect can be excluded from the present analysis. The impact of technological breakthrough in relation to waste treatment depends on the type of conditioning – partitioning from vitrified waste being more difficult than from spent fuel – and the actual stage of the waste disposal process – waste packages in storage facilities being generally more easily accessible than in disposal facilities, in particular when geological disposal is considered as final solution.

One generally considers retrievability of spent fuel elements in geological disposal (i.e. for the OTC and TTC) to be feasible, although very expensive, until closure of the geological disposal facility, i.e. over the first few hundred years [12]. These aspects are central to the debate on the decision autonomy of future generations with respect to nuclear energy. Recovery of valuable resources from vitrified waste seems compromised from cost perspectives; one evokes an intermediate stabilization form for storage as a possible way forward, to keep the decision autonomy to the next generations [12], although this should somehow be balanced by proliferation risks considerations and the burden associated to the waste management transferred to the next generations.

From these arguments, the current forms of the OTC and TTC options seem to offer more flexibility in case of technological breakthrough, considering that "intact" spent fuel assemblies are disposed of. Regarding the P&C option, the performance will depend on the technical feasibility of re-conditioning the waste matrix, especially for the uranium and plutonium inventories. Technological breakthrough in the fuel cycle for the FR-MOX and the P&T options lead to more limited impact, considering that these are already advanced fuel cycles; if the technological breakthrough relates to waste treatment, the concerns relate to the possibilities of re-conditioning (even reprocessing) the existing waste packages.

Rapid phase-out

Economic considerations also apply but are beyond the scope of this reflexion (and may be secondary to the postulated phase-out decision). The rapid phase-out is evaluated here in term of its possible impact on the waste characteristics or on the performance of the disposal facility. A prompt, yet responsible, interruption of the fuel cycle would result in additional waste to be disposed: fresh fuel and its feed material, incompletely burned fuel and non-reprocessed spent fuel; one assumes here that none of these materials can still be considered as resources.

Although criticality issues must be properly assessed for waste packages containing fresh or partially burned fuel, the OTC and TTC routes seem quite robust against such policy changes, considering that the disposal facility would be designed to host conditioned fuel elements and the impact on the actinide inventory remains limited. For the TTC, reprocessing of UOX fuel assemblies is not excluded but is perhaps not required, as no larger constraints are expected to dispose of a series of UOX spent fuel assemblies in place of MOX ones. Regarding the feed material – natural or enriched uranium, or the available reprocessed material awaiting fuel fabrication –, it requires conditioning before entering the existing waste streams. A simple solution would be to process that feed material as fresh fuel, perhaps diluted with depleted uranium to relieve criticality risks, and treat it in a similar way as the spent fuel.

For the P&C route, the direct disposal of spent fuel is not foreseen. The existing assemblies and feed material should then undergo the partitioning process if it remains available abroad; by proper dilution with fully burned spent fuel, criticality risks may be managed during reprocessing. The impact on the final inventory is limited, as the repository design covers U and Pu disposal. A premature closing of the reprocessing facility – leading to the need to dispose of spent fuel assemblies directly – or the impossibility to rely on extended storage for heat-emitting waste may impose additional constraints on the disposal facility, to the extent that part of the waste could become incompatible with the retained concept.

The TOP-MOX, FR-MOX and P&T options require a more careful analysis. The repository design for these routes would address vitrified waste featuring a low content of uranium and plutonium, as well as low quantities of minor actinides in the case of P&T. A prompt interruption of the fuel cycle would result in large quantities of U, Pu and MA to be managed, to an extent that the waste become incompatible with the repository design. The most penalizing case is obtained when different technology fleet are at play, with a diversity of waste streams to be treated at once. An actinide inventory that becomes larger by orders of magnitude does not only pose an issue in terms of long-term safety (dose rate to the biosphere) but also in terms of the management of a larger decay heat, not considered upon initial repository design [12].

❖ *Societal changes*

In all fuel cycle options, decay heat is dominated in the short term (first decades) by the contribution from fission products. In general, a storage phase in dedicated facilities is foreseen to decrease the heat load in disposal facilities; this implies that the completion of the disposal of the waste produced in the last core cycles must be managed over at least two generations. Longer storage periods (centuries) are even required for the P&T or P&C options if separation of Cs and Sr is also conducted to diminish even further the constraints associated to decay heat removal on the disposal facility.

The availability of sufficient human resources and the knowledge transfer for waste management, as well as to construct and operate storage and disposal facilities therefore needs to be guaranteed over several tens of years to properly treat the different waste. This timeframe seems also sufficient to cover the decommissioning of all fuel cycle facilities. Competences will be required in various domains of the nuclear technology but also in R&D, material sciences, (radio)chemistry, civil engineering, geology and human sciences.

Obviously, similar concerns apply when the nuclear programme is fully operative, as well as during a progressive phase-out scenario; the latter period might be the most sensitive period from human resource perspectives. A progressive phase-out may be expected to spread over several decades in advanced fuel cycles that involve different and interdependent nuclear technology fleet. In these advanced fuel cycles, one also need to guarantee many additional specific competences for which back-up from conventional industries does not necessarily exist.

❖ *Intergenerational equity*

Intergenerational equity is central to societal and philosophical concerns in relation to nuclear waste; these concerns are, for example, addressed, from generic safety perspectives, in the IAEA Safety Fundamentals [81] – Principle 7: Protection of present and future generations. A specific paragraph addresses radioactive waste:

> *"Radioactive waste must be managed in such a way as to avoid imposing an undue burden on future generations; that is, the generations that produce the waste have to seek and apply safe, practicable and environmentally acceptable solutions for its long-term management. The generation of radioactive waste must be kept to the minimum practicable level by means of appropriate design measures and procedures, such as the recycling and reuse of material."*

We therefore postulate here that a final solution for the radioactive waste *is being* developed by the generation that benefits from the nuclear technology. Regarding the fuel cycle issues, the concerns are broader and address the ability of future generations to conduct their own choices, independently from the former generations. Several aspects have already been addressed in other parts of this note, for example safeguards, resource preservation, the financing of waste management, waste retrievability or robustness against policy changes. A comprehensive comparison of different fuel cycle scenarios was conducted in [37] for a series of indicators, that do not appreciably differ from the analyse conducted in this note. While a larger burden on the next generations and a tighter binding to the choices of the former generations is expected from fuel cycles involving reprocessing, these technologies also result in improved safeguards and a reduction of the natural resources needed that preserves those resources for potential use by future generations. Somewhat balanced performance of the various fuel cycles is therefore concluded.

❖ *Leave-out / abandonment scenario*

An abandonment scenario postulates that for an unforeseen reason (technological collapse, pandemic, war) the nuclear facilities are suddenly abandoned with a minimum of safety measures. The scenario shares similarities with the prompt phase-out scenario, except that the radioactive waste is neither processed neither disposed but *remains* in the various fuel cycle facilities. Comprehensive comparisons of fuel cycle options that address such an abandonment scenario are scarce in the literature and is therefore discussed here from generic perspectives.

An important fraction of the waste will require limited active means, for example those being either in passively cooled interim storage (waiting for conditioning or disposal) or already present in a disposal facility. Interim storage may offer passive safety features that allow for longer time before actions become required; classical technologies are also expected, so that competences could remain available in the population to address safety issues in the long term. For example, part of the spent fuel elements in the OTC could be stored in dry storage facilities that are passively cooled and provide shielding at the waste package level, offering as such some protection to the population until degradation of the waste packages take place. It remains, however, difficult to draw conclusions on the robustness of interim storage in absence of a detailed design. For the waste in disposal, abandoning an open repository shows similarities with human intrusion scenarios. In both cases, a correlation of the risk with the waste radiotoxicity may be hypothesized; this then favours the FR-MOX and the P&T schemes.

The situation is different for the fuel and radioactive materials remaining at fuel fabrication plants, reactors and reprocessing plants, which require extensive active safety means. A dominant parameter to compare the various fuel cycle scenarios then relates to the radiotoxic inventory of the fuel – in particular actinide element inventory –, integrated over all fuel cycle facilities. On this basis, the OTC would perform best [41]. A similar level of robustness may be expected for the TOP-MOX approach in exporting countries, considering that most of the spent fuel would be abroad; the situation is then opposite for accepting countries. Larger Pu, Am and Cm inventories are present at all stages of the fuel cycle in the TTC, and even larger amounts are to be foreseen in the FR-MOX or the P&T routes. The routes derived from reprocessing therefore result in a lower level of robustness against abandonment scenarios for the material present in reactor, and fabrication and reprocessing plants, although in all cases severe consequences are foreseeable.

5.5.4 Technological maturity

❖ *Fuel fabrication and irradiation*

A comprehensive review of fuel fabrication and irradiation Technical Readiness Levels (TRL) has recently been conducted by NEA in 2014 for advanced fuel cycles [82]. Schematic definitions of these levels are provided in Figure 12. The OTC, TTC and TOP-MOX approaches are today applied at an industrial scale and can be attributed a TRL of 9. For the FR-MOX option, plutonium multi-recycling still poses some industrialization challenges for the reprocessing and fuel fabrication phases, in view of the larger activity of the plutonium after several cycles (e.g. issue of ^{238}Pu); on reactor side, several liquid metal cooled fast reactors have been and are still operated worldwide, so that a TRL level of 7-8 could be attributed.

In the case of the P&T routes, TRLs up to 4 – 5, corresponding to a proof-of-principle level, are attributed in [82] for the fuel fabrication and irradiation aspects, depending on the type of fuel considered (metal, oxide, nitride or carbide forms).

TRL	Function	Definition
9	Proof of performance	Multiple years of operational experience established at industrial scale. Processing and recycle of minor actinide fuels/targets.
8		Full scale process demonstrated in a limited operational environment.
7		Prototype system demonstrated under conditions fully representative of operations.
6	Proof-of-principle	Engineering or pilot scale testing of technology component or process step. Process flowsheets proven through hot tests using spent fuel. Process models validated.
5		Technology component or process step validated at bench scale under relevant conditions. Process models developed. Proof-of-principle hot tests using spent fuel.
4		Technology component or process step validated under laboratory conditions. Tests performed using active materials in simulated feeds. Fundamental properties measured.
3	Proof of concept	Lab scale tests to prove concepts, fundamental data obtained.
2		Technology application developed and options investigated.
1		Initial concepts are proposed and basic principles established.

Figure 12 Definition of the Technical Readiness Levels (TRL). Used with permission of OECD-NEA, from [28] (© OECD-NEA, 2018); permission conveyed through Copyright Clearance Center, Inc.

❖ *Reactor technology*

The OTC, TTC and TOP-MOX approaches are today applied at an industrial scale and can be attributed a TRL of 9, as for the fuel fabrication and irradiation aspects. For the FR-MOX option, several liquid metal cooled fast reactors have been and are still operated worldwide, providing already some return of experience; a TRL level of 7 – 8 could then be attributed.

Regarding the P&T route, several reactor designs may be considered. Considering the similarities with the 'conventional' fast reactors of the FR-MOX route, but the lack of large-scale irradiation of fuel containing minor actinides in significant proportions, no TRL beyond the proof-of-principle level can be attributed. Other reactor designs (molten salt, accelerator-driven systems) feature a more limited return of experience than exists today for conventional fast reactors. A global TRL level could be set at around 6 – 7 for the reactor technology aspects, while acknowledging the significant efforts still needed to integrate reprocessing, fuel fabrication and irradiation in the picture for the P&T scheme.

❖ *Fuel reprocessing technology*

A comprehensive review of aqueous and pyrometallurgical fuel reprocessing methods and their TRL has recently been conducted by NEA in 2018 [28]. Industrial maturity is observed for uranium and plutonium reprocessing (PUREX process), and for Cs and Sr separation (cf. Figure 13). Full-scale demonstration is reached for the recovery of U, Pu, Np. The extraction of minor actinide elements only reaches today a proof-of-principle level, with some variation according to the exact process considered. None of the pyrochemical process (cf. Figure 14), which could assist in shortening the fuel cycle timeframe in advanced reprocessing schemes, attains today industrial demonstration scale.

5.5.5 Characteristic timeframes

While the OTC, TTC and P&C feature cycle timeframes of a few decades, most of it being storage to reduce the heat load before disposal, the fuel cycle options based on multi-reprocessing of the fuel require much longer timeframes. After irradiation during a few years (a typical figure would be 4 to 5 y), the fuel needs to cool down for several years – 3 to 7 y for conventional reprocessing, but 10 to 12 y could be required when multiple recycling steps of are considered [12] – before it can be handled in a reprocessing plant; aqueous reprocessing methods are indeed conducted with organic extractants, which are sensitive to the level of radiation, in particular α-decay. Reduced cooling times can only be considered when advanced reprocessing techniques, such as pyrometallurgical methods, become available [12]. Considering that irradiation in fast reactors would optimistically reach a burnup of 10 to 20% of the actinide inventory [12], [36], several reprocessing cycles are required before an appreciable reduction of the inventory is achieved; this gives a characteristic timeframe of at least a century for the FR-MOX and the P&T routes. Shorter timeframes can only be achieved with larger burnups, beyond 30% of the inventory or the industrialization of pyrometallurgical reprocessing techniques in centralized facilities, or with on-line reprocessing as proposed in some Molten Salt Reactor concepts. The timeframes derived from these considerations are coherent with detailed studies of the evolution of transuranic element inventory in advanced fuel cycles [12], [39], [83]. Interruption of the fuel cycle in advanced reprocessing schemes would result in very penalizing outcomes, particularly regarding the inventory of uranium, plutonium and minor actinides suddenly requiring disposal, as discussed in § 5.5.3.

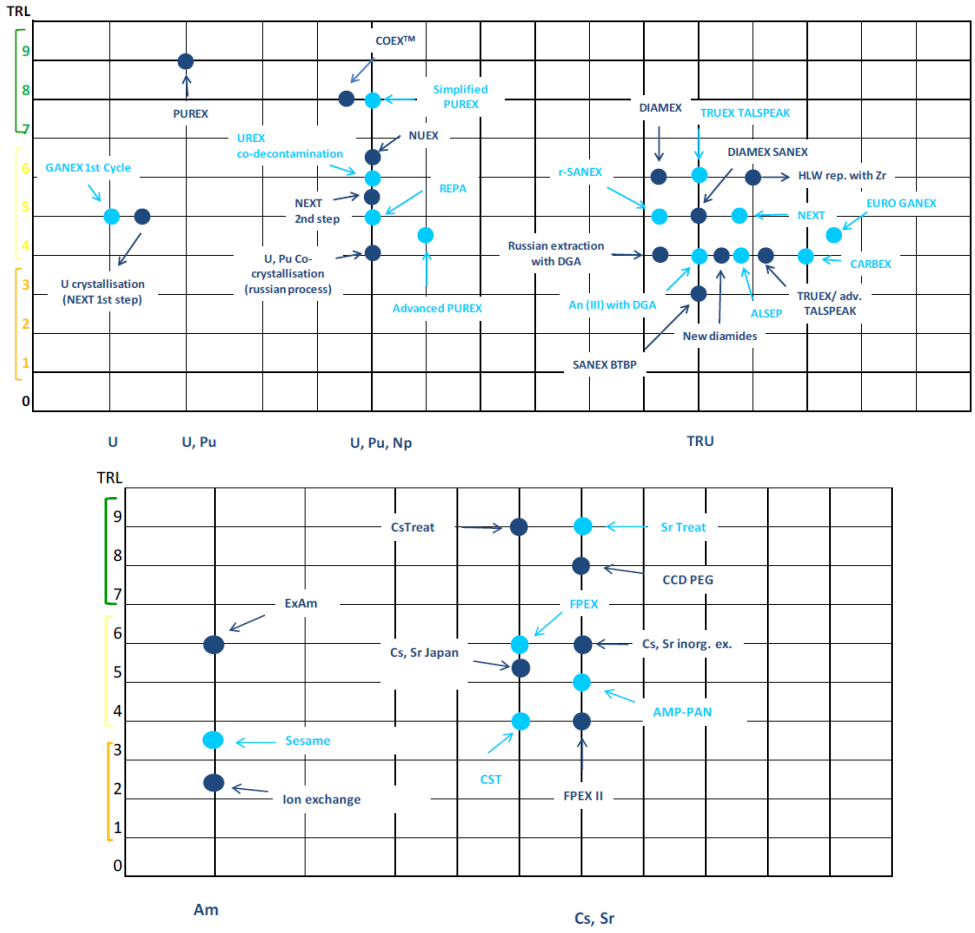

Figure 13 Technical Readiness Level (TRL) for the recovery of actinides or heat-emitting fission products from different aqueous separation methods. Used with permission of OECD-NEA, from [28] (© OECD-NEA, 2018); permission conveyed through Copyright Clearance Center, Inc.

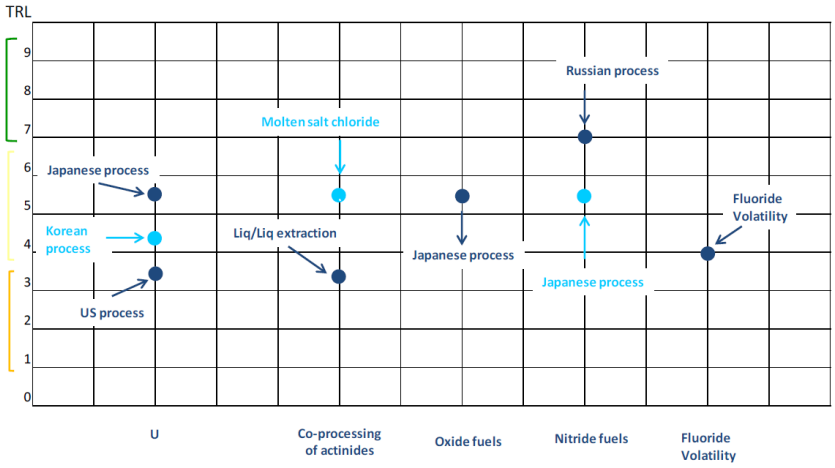

Figure 14 Technical Readiness Level (TRL) for different pyrochemical separation methods. Used with permission of OECD-NEA, from [28] (© OECD-NEA, 2018); permission conveyed through Copyright Clearance Center, Inc.

5.6 Environmental aspects (non-radiological)

Several non-radiological environmental indicators may be used, in parallel to radiological indicators already discussed in this note, to appreciate the consequences of the different fuel cycle scenarios from a broader framework. A comprehensive evaluation of various radiological and non-radiological indicators has been conducted at CEA, for the nuclear energy perspectives in France [31], [40]. The relative performance of the various fuel cycles is reported in Figure 15 for several environmental indicators. A distinction is made between the TTC as operated today in Gen-II reactors, and the equivalent scenario with EPR (Gen-III) reactors; notable differences are the heat conversion efficiency and the larger fraction of MOX fuel assemblies accepted in EPR reactors.

The reduction is modest for a few indicators; it then reflects operational releases which scale according to the energy conversion efficiency, which is higher in EPR and (sodium) fast reactors – the comparison is indeed conducted at equivalent *electric* energy production. Other indicators show a much larger decrease; these are often associated to the reduction of the feed uranium material required as the level of reprocessing increases.

Advanced reprocessing schemes generally seem to perform better for these non-radiological indicators.

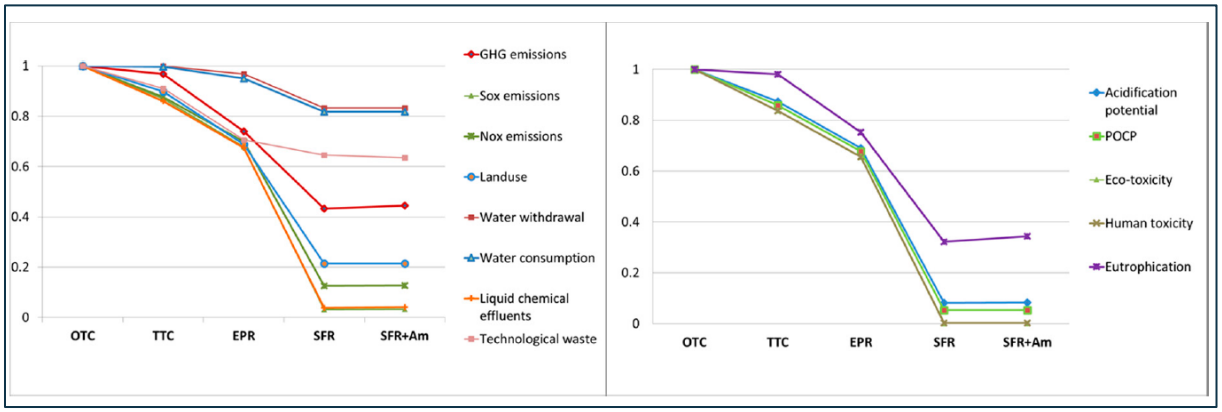

Figure 15 Relative performance of various fuel cycles, with respect to the OTC, for different environmental indicators. A distinction is made between the TTC as operated today in Gen-II reactors, and the equivalent scenario with EPR (Gen-III) reactors; notable differences are the heat conversion efficiency and the larger fraction of MOX fuel assemblies accepted in EPR reactors. Clear decreases are observed that either relate to improved energy conversion efficiency in EPR and fast reactors, either to the reduction of mining and milling activities in the front-end of the cycle, which is permitted in advanced reprocessing routes (Sodium Fast Reactors, SFR, are today considered by France, although the deployment is only foreseen for the second half of the century [84]) Source: [40] (licensed under Creative Commons — CC BY 4.0).

5.7 Advanced fuel cycle scenarios as alternative solutions for the long-term radioactive waste management

Several institutions have conducted extensive comparisons of the various fuel cycle scenarios, often through international efforts [12], [25], [39], [46], [74], [83]. Several studies, conducted on the basis of a state-of-the-art review on the fuel cycle options, also addressed the appropriateness of advanced fuel cycle scenarios as alternative solutions to geological disposal for the long-term management of radioactive waste. [14], [19]–[21], [85]–[87]. The main outcomes of such analyses are discussed in this paragraph.

In an analysis of the limitations of actinide recycling, Baetslé and De Raedt already concluded in 1997 to the inadequacy of P&T as an alternative to HLW disposal, while emphasizing the possibility to reduce, but not eliminate, the hazard associated to actinides, and the additional constraints that are imposed by MA-bearing fuel to the fuel cycle facilities [47].

During the EFP6 Red-Impact project, the impact of several fuel cycle scenarios on the inventory and on the performance of a geological disposal system was analyzed. The following main conclusions were drawn [25]:

“

- *A deep geological repository to host the remaining High-Level Waste (HLW) and possibly the long-lived Intermediate Level Waste (ILW) is unavoidable whatever procedure is implemented to manage waste streams from different fuel cycle scenarios including P&T of long-lived transuranic actinides.*
- *All European geological concepts and host formations (granite, clay, salt) feature excellent confinement properties for HLW and long lived ILW, in the long term. For the normal evolution of the geological repositories, dose levels at the surface are significantly lower than regulatory limits and natural radiological background. The very small long-term radiological impact and the differences between the considered scenarios are mainly due to the soluble long-lived fission or activation products (such as 129I or 14C) and the amount of long lived ILW in the different fuel cycles.*
- *Removing MA from ultimate waste to be disposed of reduces significantly the total long-lived radiotoxic inventory of the waste. In this way, the removal of MA can reduce the possible radiological impact in the very unlikely scenario of accidental human intrusion into a repository. However, it has nearly no effect on the long-term radiological impact under normal evolution of the repository, because MA (Am, Cm, Np) are almost insoluble in underground waters and they migrate extremely slowly in reducing conditions prevailing in European geological repositories.*
- *P&T of plutonium and MA can reduce the thermal load of HLW allowing a reduction of the emplacement galleries length up to a factor 3-6 after an interim storage cooling time (e.g. 50 years), for deep geological repositories in clay and hard rock formations. The necessary gallery length can be significantly reduced by using longer cooling times or by separation of Cs and Sr from the HLW for specific storage, conditioning and disposal.*
- *Improvements on the repository capacity by P&T and thermal load management could allow reducing the final size and number of repository sites. However, total cost of P&T deployment has to be compared to potential savings on the repository, in a full cost-benefit analysis.*
- *Particular attention should be paid to long lived ILW, separated Uranium and release/ confinement of volatile isotopes resulting from partitioning processes. Long lived ILW could become the dominant dose contribution if no further mitigation effort and/or low-activation material selection is made.*
- *Recycling of Pu is industrially implemented in some European countries providing the opportunity to Partition waste into classes and to Condition (P&C) each class in specific leach-resistant waste forms according to individual characteristics and potential radiological impacts.*
- *Scientific feasibility of P&T has been demonstrated. However, significant R&D efforts and commissioning of demonstration facilities at sufficient scale are still required to achieve viable industrial P&T and/or P&C processes and to improve the reliability of the estimations on ecological, social and economic impacts, from advanced fuel cycles.*

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In the framework of the French debate on fissile material and radioactive waste management (PNGMDR), CEA, IRSN and ASN have also expressed doubts about the implementation of transmutation as a long-term management solution [18]–[21]; different degrees of perception, however, exist in relation to the overall performance of these fuel cycle scenarios to reduce the inventory and the decay heat. IRSN indeed considers that partitioning and transmutation is not an alternative solution to geological disposal and rejects the applicability of the technique to existing waste. While an overall benefit over the entire fuel cycle is acknowledged for plutonium multi-recycling (FR-MOX route), IRSN sees little benefit, based on the state-of-the-art, for the P&T route in the management strategy for radioactive waste, considering that ultimate benefits in terms of long-term safety and constraints on the geological disposal concept are not balanced by the increased risks associated to MA-containing fuels nor by the operational constraints on the reactor, reprocessing and fuel fabrication facilities [19], [20]. CEA defends a more balanced view where benefits of the P&T route are highlighted but insists on the extent of the additional research needed to bring this route to industrial maturity.

In Belgium, based on similar arguments, the complementarity of the P&T route with the geological disposal solution has been promoted by SCK CEN and ONDRAF for reducing the disposal footprint and radiotoxicity of the ultimate waste [88].

5.8 Additional references not considered in this study

Additional documents have been identified and consulted in the framework of our survey, but not explicitly referenced: review works originating from the French debate on radioactive waste management (PNGMDR) [89]–[92], as well as from IAEA [93], [94], NEA [95], [96], [105], [97]–[104], or other comprehensive review efforts in relation to the fuel cycle [106]–[108].

6 Contradicting evidences and conflicting interpretations

Nuclear technology has always been controversial and at the centre of various technical, societal, philosophical and economic discussions. These controversies are exacerbated when it relates to the adoption of a fuel cycle and a waste disposal management scenario.

In the present note, various controversial or conflicting hypotheses and biases have been identified in the literature, that do not always enable to draw balanced conclusions from a single work; we do not expect that this note will be an exception to that rule. Still, the following items were identified to deserve critical attention when assessing the relative performance of different fuel cycle scenarios.

Comparison basis

Performance assessment of different fuel cycle scenarios is a very difficult exercise that requires a common basis for all options to be defined. Bias could be introduced if conclusions from such studies is blindly transposed to another framework; the validity of the underlying hypotheses should therefore be assessed with critical thinking. One of the most confusing aspects from the literature survey relates to differences in what is identified as resources or waste under a given approach. For example, in [31], [40], the TTC assumes MOX fuel assemblies to be reprocessed and the plutonium to be stored, awaiting future use; other studies consider disposal of spent MOX fuel assemblies in the TTC [25]. The different hypotheses have a large impact on the amount and type of waste to be disposed. Similar considerations apply to depleted uranium and reprocessed uranium stocks.

To give a few other examples, comparisons are often conducted for a given amount of electricity produced, assuming equilibrium material flows among all nuclear cycle facilities for a given production requirement. The underlying assumptions may hide scaling factors, for example on the fixed costs of a final repository or R&D costs to develop those advanced programmes. This may favour larger-scale nuclear programmes and cannot be extrapolated to smaller countries if regional cooperation is not present. The waste streams for the fuel of the last cores is also scarcely discussed in the literature, while its actinide inventory may exceed that generated in equilibrium cycles during decades.

One also generally observes higher efficiency attributed to advanced technologies, for example involving fast reactors, while the possibility of improving efficiency of the once-through-cycle, or that of moving to advanced nuclear technologies while disregarding reprocessing is scarcely addressed. The nuclear

programme may also aim at a different objective than electricity generation, in which case the figures of merit may strongly diverge.

Indicators

In comparing different fuel cycle scenarios, the attention is often given to the radiotoxicity and decay heat of the waste rather than other indicators. These quantities are indeed readily derived from the inventory evolution model and offer an easy comparison basis, while long-term dose assessment require a conceptual repository design and knowledge on transport properties through the host rock. Radiotoxicity is not necessarily representative of the dose to the biosphere in the expected evolution scenarios; the latter one is indeed dominated by the contribution from fission products in reducing conditions, as discussed in this note. Radiotoxicity remains, however, a representative parameter for human intrusion scenarios.

Another bias that is often introduced in radiotoxicity or dose comparison, relates to the contribution of ^{129}I. Iodine is indeed generally considered as fully discharged in the oceans upon reprocessing (<2% remains in the vitrified waste), while it remains one of the dominant contributors to dose in strategies involving direct disposal of spent fuel.

7 Gaps and research needed

- The diversity of sources is limited, even from international fora.
- Limited or self-sustained information is provided on several aspects, in particular: altered evolution scenarios and contextual uncertainties.

8 Conclusions

8.1 Answers to Key Questions

8.1.1 What are the safety issues to be considered for the comparison of options for the long-term management of B&C wastes?

A review of the advantages and disadvantages of various fuel cycle scenarios over the entire cycle has been conducted to assess whether they could offer alternatives to geological disposal as solutions for the long-term management of high level and long-lived radioactive waste. Several safety issues, as well as generic safeguards and security concerns, have been identified in this survey:

- Misuse of radioactive waste and spent nuclear fuel, where the impact on security and safeguards aspects needs to be evaluated.
- The impact on hazard and risk reduction, in relation to the waste form, the inventory to be disposed of, the radiotoxicity and decay heat evolution and criticality safety
- The impact on operational safety; in the case of the fuel cycle scenario, this also includes the impact on all stages of the fuel cycle, from front-end to back-end
- The impact on long-term safety, in particular the long-term dose assessment
- The impact on robustness, in particular:
    - Robustness against altered evolution scenarios and human intrusion
    - Robustness against contextual uncertainties (economics, retrievability, policy changes, societal aspects, abandonment scenario), put into perspective of the characteristic timeframe of the project
- The technological maturity of the techniques applied
- Environmental (non-nuclear) aspects

The performance of fuel cycle scenarios requires a holistic evaluation that address all security, safety and safeguards concerns over the entire cycle. This would reflect more fairly the level of safety and security, as well as the societal constraints for the present and future generations, compared to a situation where one focuses on the disposal aspects solely.

8.1.2 What are the alternative options to geological disposal?

Regarding geological disposal**, one cannot conclude that any of the options investigated would relieve the need of a disposal facility** performing over very long timeframes to guarantee the long-term management of long-lived waste, considering that:

- The fuel cycle scenarios derived from reprocessing offer a reduction of the actinide inventory, but not a complete elimination of that inventory.
- Transmutation of the long-lived fission product inventory is complex and difficult to achieve
- Waste other than spent fuel is not addressed by those technologies.

Even advanced partitioning schemes that involve transmutation of both actinides and fission products, would not allow for surface or near-surface disposal in view of the remaining long-term radiotoxic inventory; even less do they allow reaching clearance levels in a reasonable timeframe compatible with institutional and societal surveillance. However, it is recognized that advanced fuel cycle scenarios may have a very large impact on the siting, design and footprint of such facilities, as well as on the waste package forms.

8.1.3 What are the pros & cons of each identified options?

The survey conducted in the present note addresses various safety societal aspects, as well as in a much lesser extent nuclear security and safeguards concerns, in relation to the fuel cycle, from the front-end (mining) to the backend of the cycle (long-term safety assessment). Areas were identified where reprocessing and advanced partitioning offer advantages over direct disposal; these aspects relate to the reduction uranium feed material needs, of the radiotoxic inventory and of the decay heat load. Limited gain is expected with respect to long-term dose assessment perspectives, except for some specific altered evolution scenarios (including human intrusion scenarios) that lead to high actinide mobility or exposure to actinides. From societal concerns and robustness against policy changes perspectives, the long-term implications of advanced fuel cycles, however, bring penalizing aspects.

Table 3 roughly summarizes, by mean of a 5-level appreciation scale, the main relative performance of various fuel cycle scenarios, for a series of indicators. The interested reader should refer to the appropriate section of chapter 5 for the detailed arguments supporting the appreciation.

Table 3 Summary table: performance of various fuel cycle options, with respect to a given criteria. This appreciation should not be separated from its supporting basis, detailed throughout the manuscript. Legend: '◆': average; '+' or '-': slightly favourable or unfavourable; '++' or '- -': strongly favourable / unfavourable. The orange-highlighted lines are those for which the performance spread is the broadest; light grey text is used for indicators with little variation in the performance among the different fuel cycle options.

	Direct disposal	Pu reprocessing		Advanced reprocessing & fuel cycles		Regional synergies[2]
	OTC	LWR-MOX (TTC)	FR-MOX	P&T	P&C	TOP-MOX
Security & Safeguards • *Physical protection – risks of sabotage* • *Physical protection – risks of theft* • *Safeguards – risks of institutional diversion*	The considerations are often antagonist from one criteria to the other. They are, moreover, heavily dependent on siting and facility design, and, hence, difficult to summarize as single indicators. The reader is therefore invited to refer to the text for those aspects in relation to security and safeguards.					
Safety functions						
• *Operational safety*	-	◆	+	+	-	-
• *Hazard & risk reduction*						
○ Inventory	◆	-	+	++	◆	+
○ Decay heat	-	-	+	++	◆	+
○ Criticality safety	-	-	+	++	-	++
• *Confinement / containment performance*	◆	-	+	+	+	+
• *Ageing*						
○ EBS	◆	◆	◆	◆	◆	◆
○ Waste forms	◆	◆	+	+	+	+
• *Long-term safety*	◆	◆	+	+	-	+
Robustness						
• *Altered evolution*	◆	-	+	+	+	+
• *Human intrusion*	◆	-	+	+	+	+
• *Societal aspects*						
○ Geopolitics	-	◆	+	+	-	-
○ Policy change	+	◆	-	- -	◆	◆
○ Reversibility	◆	◆	◆	◆	◆	◆
○ Retrievability	◆	◆	◆	◆	◆	◆
○ Leave-out scenario	+	◆	-	- -	◆	-
○ Burden on next generations	+	◆	-	- -	-	+
• *Technological maturity*	++	++	◆	- -	-	+
• *Disposal timeframe*						
○ Fuel cycle timeframe	++	+	-	- -	+	++
○ Active / passive safety	+	◆	-	- -	◆	◆
○ Passive safety	+	◆	-	- -	◆	◆
Other figures of merit						
• *Use of resources*	-	◆	+	++	-	-
• *Financial aspects*[1]	+	◆	-	-	◆	◆
• *Environmental aspects*	-	◆	+	++	-	-

[1] Not formally the scope of this review, but these aspects are part of the broader framework for technology selection.
[2] From domestic perspectives; at a regional scale, one returns to the technology performance.

8.2 Considerations relative to advanced fuel cycle technologies for FANC's advice on the national policy for high-level and long-lived radioactive waste

Advanced nuclear technologies are currently developed worldwide. Partitioning and transmutation of long-lived radionuclides into nuclides with shorter half-lives could, in theory, be a long-term management solution for high level and long-lived radioactive waste. The feasibility of its industrial implementation remains, however, to be demonstrated.

Today's efforts on P&T focus on spent fuel reprocessing in order to close the fuel cycle and to reduce the long-term radiotoxicity of the ultimate waste. of the current developments in advanced partitioning remains do not enable to consider them as a long-term management solution:

- the technique does not eliminate all actinides from the waste streams because of imperfect separation of species; the residual actinides in the waste streams therefore still require long-term management means for the radioactive waste;
- extraction of minor actinides from existing vitrified waste seems a technological challenge and has scarcely been addressed;
- the applicability of P&T to waste other than spent fuel remains to be addressed ;
- the applicability of P&T to treat long-lived fission products, which are main contributors to the long-term dose to population and the environment, is sometimes evoked; the requirement to proceed to both elemental (chemical) and isotopic separation, as well as the low transmutation rate, result in a complex implementation of P&T techniques for those isotopes.

It results that **advanced nuclear technologies** do not enable to treat all waste types, neither to eliminate all nuclides that pose long-term radiological threats. They **should therefore be disregarded as alternative options for the long-term management of high level or long-lived radioactive waste. They may, however, be considered in the framework of the spent fuel management strategy**, for example to reduce its inventory, decay heat or long-term radiotoxicity. These reductions apply at equivalent energy generation and only become effective after a sufficient number of irradiation and reprocessing cycles, spreading over several decades.

The advantages of the various fuel cycle scenarios should be evaluated in the framework of a holistic approach that is optimized over the entire fuel cycle, beyond the long-term safety of the radioactive waste management process. The assessment should take the societal dimension into account and balance, for the different scenarios:

- proliferation and diversion risks in relation to fissile materials;
- nuclear security risks scenarios;
- the usage of natural resources and the impact on environment;
- operational safety, weighing safety benefits at the front-end and back-end stages with the constraints in relation to additional facilities needed to close the cycle;
- long-term safety, keeping in mind that actinide radiotoxicity is only one aspect of the radiological risk to population and environment;
- societal, ethical and philosophical concerns, notably regarding the burden imposed to the future generations and the freedom one leaves to those future generations to make their own choices in relation to energy policy and waste management;
- robustness against contextual uncertainties, for example in relation to political, institutional, societal, economical or technological evolutions. Those aspects should be put into the perspective of the timeframe associated to the fuel cycle. In case of an interruption of the process, the high radiotoxicity of the fuel at the various stages of the cycle may lead to exacerbated risks in relation to the fissile material, spent fuel and radioactive waste management, compared to the open cycle.

One may also observe that the closing of the fuel cycle implies new reactors that should respect the Belgian electro-nuclear energy phase-out policy [109].

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