RED-IMPACT

Impact of Partitioning, Transmutation and Waste Reduction Technologies on the Final Nuclear Waste Disposal

SYNTHESIS REPORT

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Editors

Werner von Lensa, Rahim Nabbi, Matthias Rossbach
Forschungszentrum Jülich GmbH, Germany

Authors

Dominique Greneche, Bertrand Quiniou
Areva NC, France

Lionel Boucher, Marc Delpech
CEA, France

Enrique Gonzalez, Francisco Alvarez
CIEMAT, Spain

Miguel A.Cuñado
ENRESA, Spain

Gonzalo Serrano, José Luis Cormenzana
Empresarios Agrupados, Spain

Wilhelm Kuckshinrichs, Reinhard Odoj, Werner von Lensa
Forschungszentrum Jülich GmbH, Germany

Janne Wallenius, Daniel Westlén
KTH, Sweden

Colin Zimmerman
Nexia Solutions Ltd, United Kingdom

Jan Marivoet
SCK•CEN, Belgium

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The results presented in this report are based on collaborative contributions by all twenty-three RED-IMPACT Partners (see Annex I).

Executive Summary

The impact of Partitioning and Transmutation (P&T) and waste reduction technologies on the nuclear waste management and particularly on the final disposal has been analysed within the EU-funded RED-IMPACT project. The partnership collects 23 organisations drawn from European nuclear industry, waste agencies, research centres and universities. Five representative scenarios, ranging from direct disposal of the spent fuel to fully closed cycles (including Minor Actinide (MA) recycling) with fast neutron reactors or Accelerator-Driven Systems (ADS), were chosen in the project to cover a wide range of representative waste streams, fuel cycle facilities and process performances. High and intermediate level waste streams have been evaluated for all of these scenarios with the aim of analysing the impact on geological disposal in different host formations such as granite, clay and salt. For each scenario and waste stream, specific waste package forms have been proposed and their main characteristics identified. Both equilibrium and transition analyses have been applied to those scenarios. The performed assessments have addressed parameters such as the total radioactive and radiotoxic inventory, discharges during reprocessing, thermal power and radiation emission of the waste packages, corrosion of matrices, transport of radioisotopes through the engineered and geological barriers or the resulting doses from the repository. The major conclusions of this study can be summarized as follows:

- 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 ^{129}I or ^{14}C) 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 economical impacts, from advanced fuel cycles.

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1. Introduction

It is the objective of the project to analyse the relative impacts of **Partitioning and Transmutation** strategies (P&T) on the whole nuclear fuel cycle especially with regard to waste management and geological disposal. The system studies focus on a realistic evolution of P&T technologies which can be deployed incrementally on an industrial scale as well as on future developments such as reactors of the third and fourth generation (Gen III & Gen IV) and Accelerator Driven Systems (ADS).

Due to the fact that the existing nuclear installations in Europe, both the reactors and the nuclear fuel cycle facilities, will still be in operation for several decades, it is essential to include them in the P&T and waste management scenarios and to make best use of upcoming innovations in fuel cycle technologies which may result in environmental, social and economic benefits and improved sustainability [1.1].

On the other hand, it has to be recognized that public perception of nuclear energy is strongly influenced by the waste issues of long-lived radiotoxides (actinides & fission products) and by the proliferation aspects [1.2] of the closed or open fuel cycle. As a result, European countries actually follow different policies on nuclear energy ranging from extended future deployment, via moratoria up to phase-out decisions. These national political facts have to be respected as options or restrictions of potential fuel cycle and waste management strategies. Nevertheless, even in a phase-out scenario benefits for waste management and geological disposal can be taken from advanced fuel cycles as current reactors may still be in operation for some decades allowing, for example, for improved burning i.e. transmutation of actinides stockpiles and more leach-resistant fuel matrices for direct disposal. Other national policies may allow for much extended solutions.

Any waste management – conventional or nuclear – has to be built on **three principal pillars**:

1. Reduction of waste generation (**R**)
2. Waste treatment and recycling of valuable material (i.e. **P&T**)
3. Conditioning (**C**) and disposal of non-recyclable residua

Reduction of waste generation is the most effective measure in any waste management strategy and has a strong influence on the build-up of stockpiles. Thus, a consistent system study must not only focus on the impact of P&T but has to include the possibilities for waste reduction in a prominent way.

This has to apply not only to actinides but also the long-lived fission products (LLFP) such as ^{129}I and ^{99}Tc which are more mobile in most geological repositories than actinides [1.3].

The present industrial practice of enhancing burn-up of LWR fuel, its reprocessing and subsequent Pu recycling (MOX) is already a good example for actual **'R/P&T'** resulting in much reduced High-Level Waste (HLW) disposal volumes. Development and deployment of advanced or innovative fuel cycles for LWR in response to national policies and industrial needs will allow more significant benefits for waste management and help to reduce the increase rates of plutonium stockpiles. Symbiotic fuel cycles with ultra-high burn-up (e.g. HTR) or applying fast neutron spectra will, in future, will lead to a more significant impact and reduction of radio toxicity. The scenarios on the impact of P&T can only be realistic if a logical sequence of technology evolution is taken into account.

Partitioning followed by Conditioning of MA and LLFP **(P&C)**, only, could be another pragmatic option for long-term storage and safe disposal or use in future transmuters. Ceramic immobilisation or storing Curium as UCm Oxide or ThCm Oxide where MOX is de facto the resulting decay product are other examples for improved waste handling.

With this background, the RED-IMPACT project consequently addresses three different time horizons, the related technological choices for R/P&T and their impact on waste management:

- **Short–term options** (using existing reactors & Advanced Fuel Cycles (AFC)): high performance fuel, higher burn-up, increased MOX loading, inert matrix fuel or Pu/Th mixed oxide fuel type for PWR, VVER, CANDU etc. plus the use of faster spectrum in the upper part of BWR core for improved transmutation potential, existing or evolving aqueous reprocessing technologies
- **Medium–term option**s (using Gen III+ reactors, more Advanced Fuel Cycles): symbiotic fuel cycle of LWR & HTR (Ultra-High Burn-Up), High Performance LWR (HPLWR), more advanced LWR fuel, HTR-transmuter fuel, advanced reprocessing / partitioning, leach-resistant fuel.
- **Long–term options** (ADS, Gen IV reactors, innovative Fuel Cycles, P&C etc.): ADS, SFR, LFR, GFR or MSR, innovative fuels and partitioning / conditioning techniques.

Emphasis is put on the industrial short - and medium-term options, whereas for the long-term, focus will be on those scenarios which have not yet been studied sufficiently (e.g. symbioses of Gen II/III with selected Gen IV reactors and ADS). Special attention is put on the transition phases between the existing and evolving technologies. Benefit can be taken from related European FP5 and FP6 projects [e.g. 1.4, 1.5], from former fuel cycle analyses [e.g. 1.6, 1.7, 1.8] and from direct collaboration with ongoing activities on the Gen IV, IAEA and NEA level to generate a consistent evaluation of relevant waste streams and operations in the nuclear fuel cycle.

Intermediate goals of the work programme are related to the following items:

- Overview and analysis of different **waste management strategies** including the type of geological disposal, irretrievability vs. retrievability, long-term vs. intermediate storage etc. and identification of relevant operations
- Elaboration of **industrial deployment scenarios** to identify the near-/medium-term technical options for R/P&T, their state of readiness, necessary R&D, realistic schedule till deployment and potential market shares. Eastern European reactors (VVER, CANDU) as well as BWR and their potentials are also addressed.
- **Performance indicators** for the different steps of the fuel cycle affected by R/P&T (e.g. radioactive releases from fuel cycle [1.9], dose on workers, long-term radiological impact, cost etc.): mining/milling, front-end of fuel cycle facilities, reactor/transmuter operation, back-end including intermediate storage, transport, reprocessing/partitioning, conditioning, disposal and dismantling & site rehabilitation. This WP will define industrial scenarios and input data for detailed analysis in the following Work Packages (WPs).
- **Quantification of waste streams** / waste reduction potentials based on simplified core calculations, elaboration of characteristics for proliferation, radioactivity, radio toxicity, heat generation etc.

- **Impact on waste management** including geochemical behaviour, retention capabilities and radiotoxic potentials as well as waste inventories / disposal requirements under different R/P&T or P&C strategies and implications on repository layout, repository operation and post-closure safety.
- **Fuel cycle modelling and (simplified) life cycle assessment** and consistent analysis of important operations in the fields of
  - fuel-related aspects (including ore demands and diverse fuel options),
  - reactor/transmuter operation of alternate/symbiotic systems,
  - waste management and partitioning
  - waste features / masses for storage and disposal
- **Synthesis and sustainability criteria** addressing sustainability issues and externalities in a quantified way by using simplified Multi-Criteria Decision Analysis techniques (MCDA).

In this report, the results are summarized and weighted for conclusions and recommendations including a quantified synthesis of benefits / disadvantages of the different options. Cross-cutting activities on the economic impact of R/P&T and P&C technologies in an industrial scale will also allow some orientation on their competitiveness.

It is expected that the RED-IMPACT project may contribute to an improved public and political perception of R/P&T as well as to industrial decision making by illustrating the impact of short - and medium-term fuel cycle options applied to the different existing LWR (PWR & BWR), GCR and HWR on waste management in Western and Eastern Europe and providing an assessment of medium- and long-term options. Deploying existing nuclear installations (reactors and fuel cycle facilities) for waste reduction, partitioning & transmutation is the most straight-forward tool to use plutonium stockpiles, followed by more advanced systems. It also represents an important basis for symbiosis with more innovative P&T technologies / systems as identified under Gen III+ and Gen IV.

2. Sustainability Challenges

2.1 Sustainability concept and challenges for energy systems

The guiding principle of 'Sustainable Development' is gaining increasing importance on different analysis levels. Considering the preparations by national governments, scientific advisory councils and other non-governmental organisations for the earth summit in Johannesburg in August 2002 and thereafter one notices that both developing and industrialised countries deal with the concept of Sustainable Development. Also taking into account numerous activities within Local Agenda 21 it is obvious that most activities are on political and administrative levels.

There exist three conceptual approaches (see figure 2.1):

1) Weak Sustainability – the paradigm of perfect substitutability: The concept of Weak Sustainability follows the utilitarian principle and can be called the paradigm of perfect substitutability. It is based on the assumption that the entire capital stock consists of natural capital, human capital and man-made capital, and it demands that the capital stock has to remain at least constant over time. According to the paradigm of perfect substitutability, there are no restrictions for the actual composition of the capital stock. For natural capital this means, that both non-renewable and renewable resources can be diminished since the stock of other capital goods increases, and that this is not in contradiction with sustainability. A prominent approach to put this concept into concrete terms is the (economic) Capital Approach, which aims to quantify all system aspects into one (monetary) indicator. Mainly, the lack of necessary information and data prevents to use this concept.

2) Strong Sustainability – the paradigm of complementarities: The concept of Strong Sustainability calls for keeping both the aggregate total value of man-made capital and natural capital and the total value of natural capital itself at least constant. In contrast to the first concept it restricts the substitution of capital components, because artificial capital goods cannot in any case replace the services and amenities of nature. It also implies barriers to the consumption of non renewable and renewable resources. Additionally, it assumes that services from natural and man-made capital goods are frequently complementary. A prominent approach of this concept is the Ecological Footprint which aims to assess the impacts of human activities by quantifying all system aspects into one (ecological) indicator. Mainly, inabilities to reflect important non-ecological system aspects prevent to use this concept.

3) Normative concept – indicator set: Sustainability, as defined in the Brundtland report and the Rio Declarations, is linked with ecological, social, economical, cultural and institutional development aspects of the world's societies. Viewing these different sections of sustainable development from a normative perspective for decision making the term "dimensions" or "pillars" has been widely established. Among the normative concepts made available so far, the Three-Pillar Model is most prominent, accounting aspects of the ecological, economical and social dimensions under equitable ranking conditions. The normative concept can be regarded as a mixed approach which contains elements of both Weak and Strong Sustainability. On the one hand it allows substituting capital goods, but on the other hand it enables, too, to define priorities for specific components of the indicator set.

   However, in practice the Three-Pillar Model (figure 2.2) turned out to be helpful for quantitative assessments within tight limits only. Nevertheless, stakeholders usually can accept the mixed approach because it allows reducing complex systems to a set of its main elements; they can communicate their individual preferences and identify their main indicators. Therefore, nevertheless, different stakeholders may support different sets of indicators.[1]

For energy systems, different stakeholders were involved to effectively incorporate Sustainable Development and took the challenge to define indicators and to create comprehensible indicator sets. Although considerable efforts were undertaken, there was only little success to agree upon common Sustainable Development energy indicator sets. Not surprisingly, this reflects different aggregation levels of energy systems, different framework conditions and assumptions, as well as different political viewpoints and different systems of values.

2.2 Concept for energy technologies, in particular for nuclear energy

Summarizing the above interpretations of sustainability concepts and its deductions the following preconditions as for sustainable energy supply systems are established. An energy supply system is sustainable, if

- the potential for provision of energy services increases or does not decrease for the next generation,
- the substances releases due to energy use are in line with the natural assimilation capacity,
- Energy services are provided with the least possible resource input, including the "environmental resource".

Although these preconditions are widely accepted, it is the details that are most difficult. As a consequence of the first precondition the known energy and resource base that is economically exploitable is only allowed to decrease in line with the system's ability to provide at least the same amount of energy services. This includes successful further explorations and development of advanced extraction technology, increasing primary and final energy productivities as well as making available new energy sources. The other two preconditions show similar complexities.

For nuclear energy generation, it has to be recognized that public perception and acceptance is strongly influenced by the potential for catastrophic accidents (e.g. Chernobyl), radioactivity releases from nuclear power plants and reprocessing facilities [2.1], the waste issues of radiotoxic long-lived radionuclides (minor actinides and fission products) and by the proliferation aspects of the closed or open fuel cycle. As a result, European countries actually follow different policies on nuclear energy ranging from extended future deployment, via moratoria up to phasing-out.

Although there are specific aspects for nuclear energy generation, as shown, there is no single energy generating technology, which might be regarded to be generally sustainable, if a life cycle perspective from cradle to grave is adopted. This perspective demands not only to look at the operation of an energy generating facility, but also to keep in mind the extraction of primary energy carriers, the construction and decommissioning phase as well as final waste treatment, all of this in its ecological, economic and social aspects. Therefore, comparative energy generating technology assessment is necessary.

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[1] As an example for energy indicator sets see Vera et al. (2005).

2.3 Sustainability indicators for comparative technology assessment

For a sustainability assessment of energy technologies solutions are to be formulated for the following critical aspects:

What are the technical systems to be assessed?

What indicators form a general set of Sustainable Development indicators which different actors and stakeholders can agree upon for technology assessment?

What are adequate measurement methods for both quantitative as well as qualitative indicators?

What are the relative weights of the indicators?

The assessment processes itself, should it be exclusively done by experts?

The generally very abstract normative goals and value concepts must be transformed into concrete "instructions for action". This requires indicators with which a development can be characterized as comprehensively as possible. The task of indicators is to provide information about the current system state. With the aid of indicators it should be possible to tell whether the system state observed is felt to be satisfactory or whether changes must be made. Furthermore, indicators should be quantifiable as far as possible. Time series are required for an assessment of the dynamics of the system.

Being aware of competing electricity generating technologies and keeping in mind international research activities RED-IMPACT supports the identification of sustainability indicators, which are appropriate for nuclear as well as non-nuclear technologies [2.2]. For the definition of the set of Sustainability indicators RED-IMPACT agreed upon a sequence of single steps to guarantee on the one hand to base its set on current publications and activities, not focusing only on nuclear energy, and to create on the other hand a set covering all relevant nuclear aspects for comparative Sustainability assessments of electricity generating technologies:

- Starting from a general definition of Sustainability and general sets of Sustainability indicators (UN-CSD, OECD, European Union, etc.),
- Avoidance of isolated 'nuclear interpretation' of Sustainability,
- Evaluation of different/common criteria & indicators for energy systems (e.g. German Study Commission Sustainable Energy Supply), including nuclear approaches (INPRO, IAEA [2.3] Gen IV, NEA, MIT, PSI etc.),
- Identifying consensual platform (including anti-nuclear approaches),
- Awareness of social dimension (e.g. specific risk perception and public acceptance mechanisms)

The project agreed upon the Three-Pillar-Approach and compiled a list of Sustainability indicators for comparative assessment of energy generating technologies (see Table 2.1).

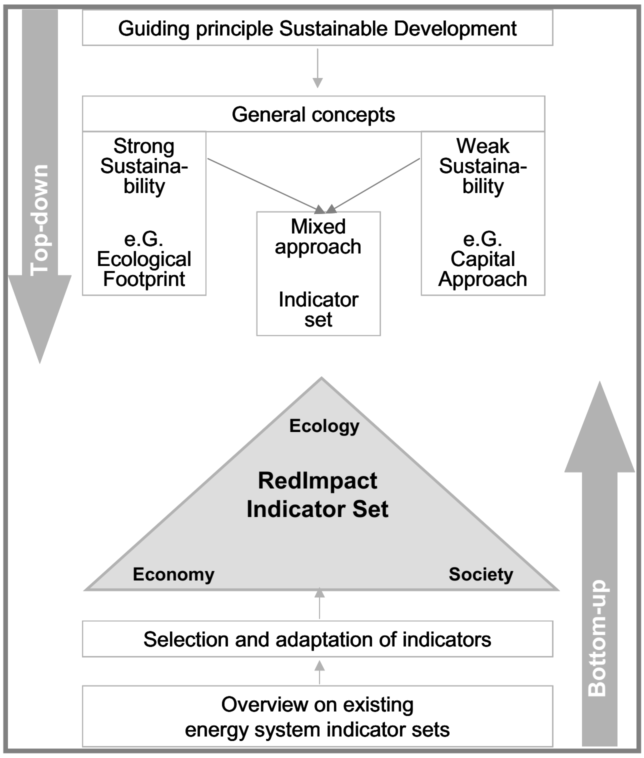

Figure 2.1: RED-IMPACT study approach on Sustainability indicator set for comparative technology assessment of energy (electricity) generating technologies.

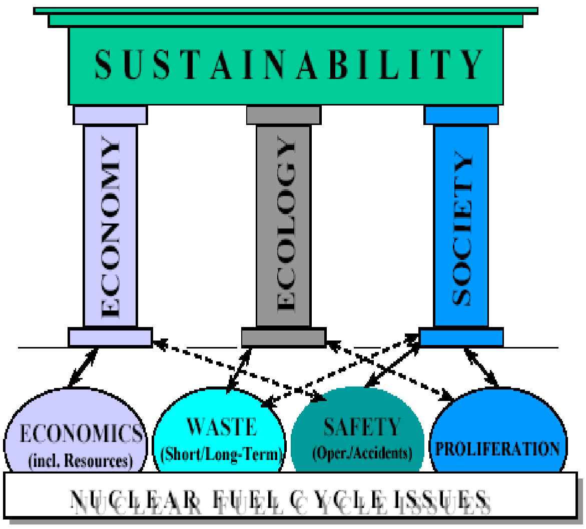

Figure 2.2: Sustainability and the canonical issues for the nuclear fuel cycle.

Table 2.1: RED-IMPACT Sustainable Development indicator set

Dimension	Criteria	Sustainable Development Indicators for Electricity Generating Technologies: General	Sustainable Development Indicators for Electricity Generating Technologies: Additional for Nuclear Systems
Economy	Private cost	▪ Production cost (total private life cycle cost)	
	Public cost	▪ External cost	
Social	Work opportunity	▪ Direct employment	
	Income generation	▪ Added value	
	Security of supply	▪ Fuel import dependency ▪ R/P ratio	
	Proliferation risk		1. Intrinsic proliferation resistance features 2. Extrinsic proliferation resistance measures
	Public acceptance	▪ NIMBY/BANANA	
	Risk aversion		▪ Kind of risk constraints ▪ Nature of risk Source ▪ Dimension of risk consequence
	Public security		▪ Measures against terrorist attacks
Ecology & Public Health	Non-renewable resource use	3. Use of energetic resources 4. Use of aluminium 5. Use of iron 6. Use of copper	
	Land use & occupation	7. Area for life cycle 8. Resettlement necessities 9. Noise pollution 10. Change of landscape	
	Climate change / air & water quality	11. GHG potential 12. Acidification potential 13. Eutrophication potential 14. Ozone 15. Toxic emissions 16. Radiological emissions 17. Discharged waste heat (e.g. cooling water)	
	Waste generation & management	18. Radioactive waste 19. Non-radioactive, toxic waste 20. Non-radioactive, non-toxic waste 21. Confinement times	
	Safety & health	22. Health impact of normal operation on workers 23. Health impact of accidents on workers 24. Health impact of normal operation on public 25. Health impact of accidents on public 26. Potential damage of severe accidents (range)	
	Ecosystem	27. Biodiversity/species	

3. Nuclear Fuel Cycle Options

3.1 Introduction

Whereas in the conventional waste management field, anybody is already acquainted to the main steps like:

- Avoidance / reduction of waste generation
- Separation of different waste streams
- Recycling of usable materials
- Waste combustion
- Conditioning of problematic wastes
- Deposition of residua

The analogous steps in the nuclear field are still subject of public disputes or controversy. But without applying the same logics, nuclear fuel cycles cannot approach to the sustainability requirements.

It has to be recognized that the *whole* fuel cycle has principally to undergo an evaluation for all its elements including:

- Mining / Conversion
- Enrichment
- Fuel Fabrication
- Nuclear Reactors for Power Generation or Transmutation
- Transport & Intermediate Storage
- Reprocessing / Partitioning / Conditioning
- Final Disposal in Geological Repositories

According to the general objectives of the RED-IMPACT project, an assessment of the industrial feasibility and needs of R&D for implementation of P&T in short, medium and long term has been undertaken. The main facilities of the nuclear fuel cycle which have an impact on the waste streams / quality like different reactor types / transmuters, reprocessing plants, conditioning processes have to be evaluated with regards to their performance and their influence on the final disposal requirements.

Within this frame, it was first needed to establish a comprehensive status of inventories, capabilities and potential lifetime of existing UE facilities with regard to material and waste flows and stockpiles as well as technologies used and performances achieved by these facilities. In addition, an overview of R&D studies on partitioning and transmutation world-wide was provided to identify industrial scenarios of fuel cycle options and their expected time scale for their industrial implementation.

3.2 The Principals of Partitioning and Transmutation

Partitioning and Transmutation (P&T) is considered as a way of reducing the burden on a geological disposal. Plutonium and the minor actinides are mainly responsible for the long-term radiotoxicity. If these nuclides are removed from the waste (partitioning) and fissioned (transmutation), the remaining waste loses most of its long-term radiotoxicity. This is important both in the case of a nuclear phase-out, as well as in the case of the continuous use of nuclear energy as contributor to a sustainable development. In the latter case, the main requirements are related to competitiveness, reduction of long-lived, highly active nuclear waste, saving of natural resources, improved safety characteristics etc. In order to assess the potential of transmutation, the following criteria usually are applied:

• The mass balance of transuranics (TRUs) including residual Pu and MAs;

• The radiotoxicity on diverse timescales.

In order to reach the goals of P&T the most effective way is to "burn" i.e. fission the actinides (Pu, Np, Am and Cm). The resulting fission products have, in general, much shorter half-lives and, after a few hundred years, are no longer hazardous. A few long-lived fission products (LLFP), such as ^{99}Tc, are sometimes taken into account, even if their contribution to the global radiotoxicity is rather limited – most of their contribution being to the so-called "residual risk". Numerous studies on transmutation have been performed worldwide using different types of reactors and different fuel cycle strategies. General conclusions can be drawn from the results of these studies.

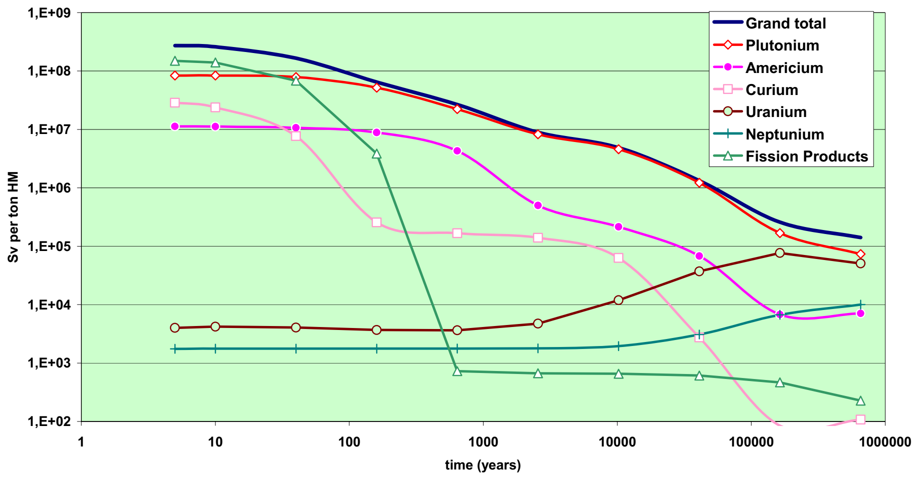

Figure 3.1: Spent Fuel Toxicity

It can be seen from figure 3.1 that the radio-toxicity inventory can be reduced up to a factor of 10 if all the Pu is recycled and fissioned. Reduction factors higher than 100 can be obtained if, in addition, the minor actinides (MAs) are burned. A prerequisite for these reduction figures is a nearly complete fission of the actinides, for which multi-recycling is a requirement. Losses during reprocessing and re-fabrication must be well below 1% and probably in the range of 0.1%.

In principle all types of reactors can be used to fission the actinides (thermal systems, fast systems, critical and sub-critical systems). Fast neutron systems however have significant advantages because of basic physical properties – all TRUs are neutron sources in a fast spectrum whereas most TRUs in a thermal spectrum act as neutron poisons. In addition, a thermal neutron energy spectrum increases the content of high-mass isotopes in the composition of the final waste discharged.

The addition of MAs to the fuel has adverse effects on safety parameters. This is true for both, thermal and fast neutron systems. The fraction of delayed neutrons and the Doppler coefficient are reduced. In liquid metal cooled reactors (LMRs) the positive void reactivity is increased. Because of the adverse effects on safety, the content of MAs in the fuel of critical reactors needs to be limited to a few percent. This has the consequence that large parts of the whole fuel cycle of a reactor park will be ”contaminated” by MAs. Depending on the particular strategy, up to 50% of all reprocessing and refabrication have to deal with fuel containing MAs.

Fuel containing MAs needs special handling precautions during fabrication and reprocessing. Significant economic penalties are to be expected. Even the feasibility of the PUREX process as it is applied today must be questioned (Pu-238 content, radiolysis, neutrons, etc.).

In order to overcome the adverse effects of MAs in the fuel, the so called ”Double Strata” strategy has been proposed. In the first (main) stratum, energy is produced using conventional reactors. Pu may or may not be recycled in this stratum depending on national policies. MAs are not recycled in the first stratum. MAs and some Pu, which no longer can be used in the first stratum, are transferred to the second stratum and will be burned there. Dedicated fuels, innovative reactor systems and new fuel cycle technologies will be applied in this stratum. As these probably are more expensive than the today’s commercial systems, the fraction of the second stratum to the whole reactor fleet must be kept small.

3.3 Technical Elements of P&T Scenarios

The most relevant technologies and facilities for P&T strategies are being addressed in the following for all stages of the nuclear fuel cycle:

- **Mining and conversion** will mainly be affected from the fuel consumption of a specific reactor type. Neutron economy and high thermal efficiency is an important asset for future reactors to reduce ore mining needs. The environmental impact of uranium or thorium mines cannot be neglected with regards to radioactive emissions and destruction of landscapes. Tailings from mining and conversion need to be included in a separate comprehensive environmental impact analysis. For RED-IMPACT, only the proportional fuel needs will be taken into account.

- **Nuclear reactors and associated fuels** strongly determine the generation of radioactive wastes as well as their capabilities for waste burning / transmutation. The following reactor systems have been regarded within the RED-IMPACT investigations in more detail:
  - **Light-Water Reactors** (LWR) as actually existing as well as their further evolution (Generation III). Different fuels can be applied in such a thermal system such as:
    - ***Uranium Oxide***
    - ***Plutonium / Uranium Mixed Oxide (MOX)***
    - ***Plutonium / Thorium Mixed Oxide (Pu/Th-MOX)***
    - ***Inert Matrix Fuels (IMF)***
    - ***Homogeneous or heterogeneous MA fuel designs***

  In principle, both types of LWR, i.e. Pressurised-Water Reactors (PWR) and Boiling-Water Reactors (BWR) need to be regarded separately because of the spectrum shift in dependence of the steam content of BWR.

- **Fast Reactors** (FR) as part of the Gen IV systems are offering fast neutron spectra for much better use of the fuel resources and MA burning. Gas-cooled (GFR) or liquid-metal-cooled fast reactors (LMR) can equally be applied for this purpose. Their fuel can be based on:
    - ***Plutonium / Uranium Mixed Oxide (MOX)***
    - ***Plutonium / Thorium Mixed Oxide (Pu/Th-MOX)***
    - ***Plutonium Nitride***
    - ***Transmutation Fuel containing MA***
- **Accelerator-Driven Systems** (ADS) which de facto represent under-critical fast reactors based on spallation neutron sources driven by high power particle accelerators. The spallation process has a high neutron yield but is also producing specific radioactive wastes (spallation products) with practically no long-lived MA. The fuels for ADS will be similar to those of fast reactor systems. Neutronic stability of under-critical ADS will allow to burn high MA containing wastes.
- **High-Temperature Reactors (HTR)** make use of coated-particle fuel allowing ultra high burn-up (Deep-Burn) in an epi-thermal spectrum reducing the number of recycling, reprocessing, refabrication and re-irradiation steps, significantly. HTR can accept a series of fuel compositions ranging from uranium, pure plutonium to thorium and mixtures with MA. HTR have not explicitly included into the selected scenarios but need to be reflected as another Gen IV system with high Pu / MA burning capabilities. One HTR module reactor per three large LWR would already stabilize or slowly reduce Pu inventories.
- **Other Reactor Systems** should also be regarded in a more extended study than provided by RED-IMPACT. This should address e.g.:
    - **Heavy Water Reactors (HWR)** due to the excellent neutron economy which allow e.g. to burn discharged LWR fuel for doubling energy harvest (DUPIC Cycle). They may also be effective for burning of Long-lived Fission Products (LLFP) which is currently not yet regarded in most P&T strategies.
    - **Molten Salt Reactors (MSR)** with extremely good neutron economy due to on-line reprocessing of the fuel. They represent a long-term option also for MA burning and fuel breeding.

- **Reprocessing** is a key process in a closed fuel cycle. Presently it only extracts residual uranium and useful plutonium for MOX fuel fabrication. The rest of the waste (fission products and MA) remain in a common waste fraction. Further evolution of extraction capabilities (Partitioning) may allow separating fission products from MA and specific waste treatment (**Conditioning**) according to the individual chemical behaviour of the different elements. Aqueous reprocessing represents the current state of the art whereas pyro-processing is a future option to treat high level waste from spent fuel. Secondary wastes from reprocessing/partitioning also need to be included in waste stream comparisons.
- **Intermediate storage** of spent fuel or vitrified waste from reprocessing is current practice as no geological repositories are under operation, yet. The time for storage will significantly affect the radiotoxicity levels of the waste either before reprocessing or before final storage. Long-term intermediate storage is another element of waste management strategies.

- **Geological Repositories** are indispensable for any kind of nuclear waste management approaches being based either on direct disposal of spent fuel or closed cycles. Different geological conditions result in different requirements for waste packages. Oxidizing or reducing chemical environments as well as wet or dry conditions have essential impact on the retention capabilities of a final repository.

The above mentioned technical elements can be combined into different P&T scenarios:

A-Scenarios are built on current industrial practice using LWR (Gen II & III) in an open cycle or with mono or multi-recycling of MOX. Different fuels are taken as variants. The reprocessing is based on hydro-metallurgical processes with some further technical evolution as expected for the next decades.

B-Scenarios assume more innovative technologies like using critical fast Gen IV reactors and ADS together with advanced reprocessing / partitioning techniques. Double strata approaches with more complex combinations of LWR, FR and ADS are also included.

It should be mentioned that the performance of the reprocessing/partitioning step may significantly govern the radiotoxicity discharged to the final repository. It may therefore represent a high priority as compared to the different reactor and transmuter developments. This also applies to conditioning techniques which will be decisive for the long-term retention and leaching of wastes in a final repository.

Finally, a balance has to be found between reduction of the most toxic long-lived wastes (MA & LLFP) and their packaging in long-term resistant matrices (e.g. ceramics). Both approaches lead to effective retention of radioactivity from the biosphere.

Another principal aspect for any nuclear waste transmutation is the need for remote fabrication & handling of transmutation fuels. Up-to-now, this is not yet necessary due to the limited alpha activity allowing hands-on management. Recycling radioactive waste for transmutation needs to handle highly radioactive materials. Thus remote techniques need to be developed and need to be accepted despite their economic draw-back if transmutation will be performed in future.

3.4. Status and inventory of nuclear facilities

3.4.1 Uranium mines

Only one mine is still in operation in Europe (apart from Russia). It is located in the Czech Republic. Others countries abandoned uranium ore processing due to a lack of profitability, for instance France (last mine shut down in 2001) or Bulgaria where more than 40 extraction sites and two hydrometallurgy plants had been operated until 1994.

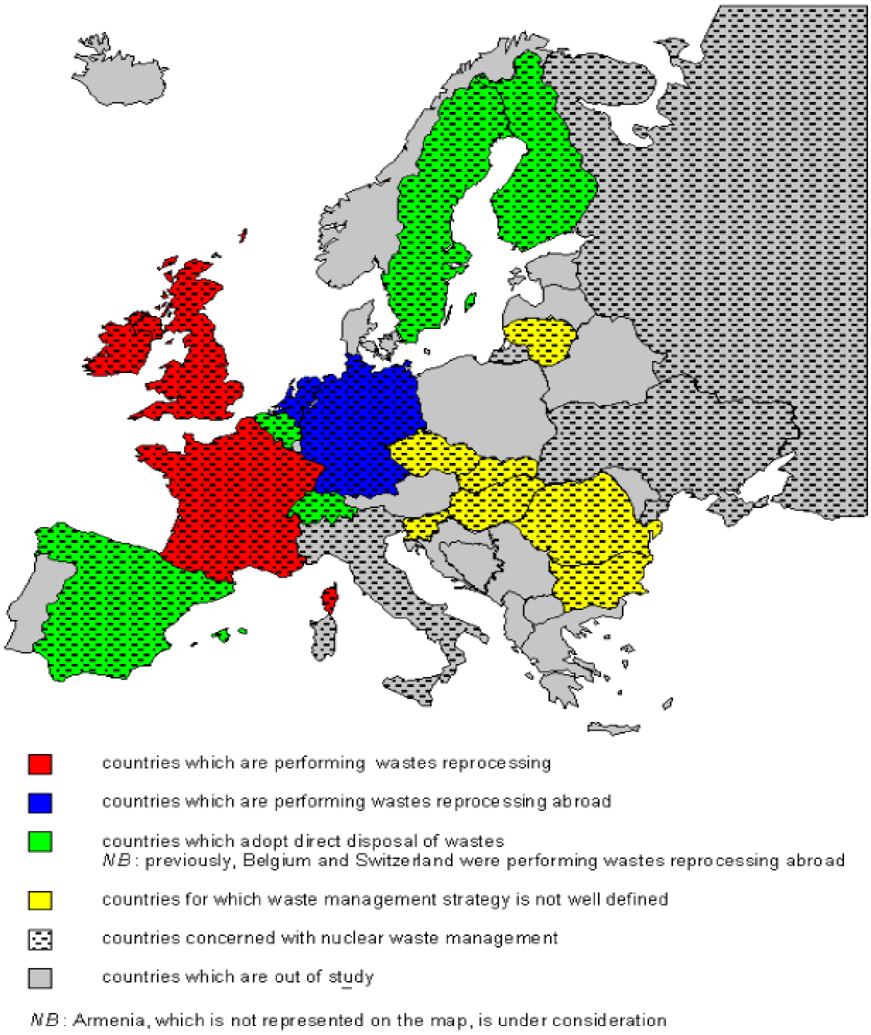

Figure 3.2: Status and inventory of nuclear facilities

3.4.2 Uranium conversion facilities

As far as uranium conversion is concerned, two main countries in Europe are concentrating all the industrial activities *i.e.* France and the United Kingdom. Uranium conversion activities can be split into:

- Front-end fuel cycle activities which correspond to the transformation of uranium concentrates extracted from mines to hexafluoride uranium (UF_6), with a possible intermediate stage under the tetra fluoride (UF_4) form. UF_6 is the uranium form required to the next step of uranium enrichment.
- Back-end fuel cycle activities which focus on the conversion of uranium from its form as a recovery product from reprocessing (uranyl nitrate) to its stable form for storage (U_3O_8).

Uranium conversion at the front-end of the fuel cycle represents a yearly uranium flow up to **20000 ton heavy metal** (tHM) for Europe, as derived from the design capacities of the facilities. Among the total UF_6 produced, 14.000 tHM are yielded in France by COMURHEX, and 6.000 tHM in United Kingdom by BNFL.

In 2003, in France, nearly 1800 tHM of uranium was converted from the reprocessed form to the storage form. Furthermore, the transformation of depleted UF_6 (waste from the uranium enrichment step) into depleted U_3O_8 was performed, in 2003, up to 11.570 tHM. It is mainly stored, only 125 tHM are being reused for the fabrication of MOX fuel. Consequently, France is currently stock-piling uranium; the total amount of U_3O_8 in interim storage was determined to be 97.920 tHM by the end of 2002.

3.4.3 Uranium enrichment facilities

Four countries are concerned with uranium enrichment plants; these are France, Germany, the Netherlands and the United Kingdom. To describe the effort required for the separation of ^{235}U from ^{238}U, the concept of separative work unit (SWU), which combines the values of product quantity and concentration of ^{235}U for each stream (feed stream, take out enriched uranium stream, take out depleted uranium stream), has been introduced.

The European capacity of production of enriched uranium is currently set around 16 million SWU each year. Considering that from eight kilograms of natural uranium (concentration of 0.71 % ^{235}U) the production of one kilogram of enriched uranium at 3.7 % (which is a classic enrichment for fuel loaded in European reactors) against seven kilogram of depleted uranium at 0.25 %, corresponds to five SWU, the total amount of enriched uranium in Europe which can be manufactured could be very roughly estimated at a value of 3200 tHM. In the same manner, approximately 20000 tHM of depleted uranium are generated which can be considered for the moment as a waste to be managed (interim storage) since only a little part of it is recycled in MOX fuel. A typical reactor of 900 MWe consumes yearly 100000 SWU.

3.4.4 Fuel fabrication

UO_2 fuel

UO_2 fuel elements fabrication facilities are located in various European countries. Operators are FBFC Romans in France, ANF GmbH Lingen in Germany, ENUSA Juzbado in Spain, WEC SE Vasteras in Sweden, and NOFC Springfields in the United Kingdom. The latter facility manufactures UO_2 fuel devoted to both the British advanced gas reactors (AGR) and the British light water reactors (LWR). By adding the design capacities of each facility, it is possible to assess the global amount of UO_2 fuel which is potentially manufactured in Europe. The value of **4070 tHM** is reached.

It has to be noticed that part of the UO_2 fuel elements produced by FBFC Romans in France (around 6 % in 2003) comes from recycling of reprocessed uranium and is used today in two 900 MW reactors (Cruas 1 and 2). It represents only 6 % of the manufactured UO_2 fuel but it may be expanded according to economic evaluation. This type of recycling is also implemented in Belgium.

MOX fuel

MOX fuel has been implemented to recycle valuable nuclear material (plutonium and uranium) as potential energy resource. This recycled fuel is used in various countries in Europe but only three facilities are currently fabricating it, since the AREVA NC ATPu plant (France) was shut down mid-2003. The Sellafield plant (United Kingdom) is considered even though it is still in a ramping up phase.

Fabrication of the MOX fuel burnt in European reactors is mostly ensured today by the BELGONUCLEAIRE plant (Belgium) and the MELOX plant (France) at a combined value of **183 tHM per year**. The AREVA NC ATPu plant shut down is compensated by the increase of the MELOX plant design capacity which has been authorized to be set at 145 tHM from 2004. It has to be noticed that European MOX fuel is partly consumed in Japan where the implementation of a MOX fuel fabrication facility is under preparation. The launch of the Sellafield plant production at its design capacity will increase the total European MOX fuel tonnage to 303 tHM per year.

The key figures of the MELOX production (France) in 2003 are detailed : to produce 113.8 tHM of MOX fuel elements, this plant was fed with 8.1 tHM of reprocessed plutonium and 108.6 tHM of depleted uranium, resulting from the defluorination of depleted UF_6 coming out from the uranium enrichment facility. Presently, the plutonium content in MOX fuel is on average 7 % but future progress are planned to improve efficiency of MOX fuel with 8.65 % plutonium content and to be implemented in 2004. Considering that 7 % plutonium content is a reliable average for all the MOX fuel currently fabricated, the total consumption of reprocessed plutonium and depleted uranium in Europe rises up respectively to **13 tHM per year and 175 tHM per year**.

Metallic U fuel

Springfields Magnox is the unique facility identified in Europe as producer of metallic uranium fuel. This fuel is obtained from natural uranium and is devoted to Magnox reactors. This facility, which yields 1300 tHM per year as design capacity, is planned to be shut down in 2009, simultaneously to the scheduled end of Magnox reactor utilization.

3.4.5 Nuclear power plants

At present, 162 reactors are in operation in Europe (against 59 already definitively shutdown). Three types of fuel are commonly burnt in European reactors: UO_2 fuel which represents the main part of fuel elements loaded in reactor cores (118 reactors), MOX + UO_2 fuel (36 reactors) and Magnox fuel (8 reactors), which is consumed in English gas cooled reactors (GCR) and will be abandoned around 2009.

Most of the UO_2 fuel elements loaded reactors are light water reactors (LWR). Among these LWR, several types are operated in Europe: pressurized water reactors (PWR), boiling water reactors (BWR), VVER (Russian technology reactors equivalent to PWR) and RBMK (Russian reactors moderated by graphite). Only the United Kingdom operates advanced gas cooled reactors (AGR) which are also running on UO_2 fuel, but it is noticeable that the last reactor commissioned in Great Britain belongs to the LWR class. Finally, it has to be noticed that there is only one reactor in Europe running on UO_2 fuel and heavy water and it is located in Romania.

Four countries are using a closed fuel cycle strategy by recycling plutonium in MOX fuel, these are France (20 reactors without including the experimental fast neutron reactor (FNR) Phénix), Germany (10 reactors, this number is not clearly established since some German reactors are licensed to burn MOX +UO_2 fuel but are not benefiting from this authorization), Switzerland (3 reactors) and Belgium (2 reactors).

3.4.6 Spent fuel storage facilities

One of the goals of this inventory is to provide relevant information to evaluate P&T scenario. For Europe, about **3450 tHM** of spent fuel are unloaded yearly from nuclear reactors. The amount of **uranium** contained in this discharged spent fuel can be evaluated at a rough value of **3300 tHM**. The mass of **plutonium** rise to **33 tHM** while the total content of **minor actinides** can be assessed to a value of **3.46 tHM**.
With this objective in mind, it appears of most interest to determine the amount of the main radionuclides potentially subjected to P&T; that is in a first round uranium and plutonium as high energetic materials, neptunium, americium and curium as minor actinides high contributors to radiotoxicity, the fission products strontium and caesium as high heat emitters, the fission products iodine and technetium as the most mobile radionuclides under disposal.

With regard to spent fuel storage (without considering pools from nuclear reactors), the situation of European countries is very changeable according to the fact that their unloaded spent fuel is reprocessed or not. Globally, countries doing reprocessing (France and the United Kingdom) own wet spent fuel storage facilities at their reprocessing plant sites. Countries which have chosen reprocessing strategy for their spent nuclear fuel and contracted foreign operators to perform this task (Germany, Belgium, and Switzerland) do not need so much important interim spent fuel storage.

In **Belgium** for instance, there is no centralized spent fuel storage facility. However, in Dessel, near Mol, the site of BELGOPROCESS includes several buildings for treatment and storage of low and high active waste. One of these is devoted to the storage during 75 years of the canisters of vitrified waste received from reprocessing in AREVA NC La Hague, plus other conditioned waste.

In **Germany**, there are two main spent fuel storage facilities but devoted to centralize high level activity waste from reprocessing and un-reprocessed spent LWR fuel (Gorleben) on one hand, un-reprocessed spent LWR and specific fuel (Ahaus) on other hand. Other interim storage facilities inventoried in Germany are not so significant (interim storage of research institutions, interim storage of industry, HLW storage from test Reprocessing Plant Karlsruhe).
The policy of nuclear waste management is presently changing, with regard to the « contract about the phase out of nuclear power generation » signed in 2001 and it has a direct impact on interim storage facilities need. Up to this decision, the spent fuel discharged from German reactor cores was reprocessed abroad and the returned HLW was stored only in the Gorleben interim storage facility. The new management of spent fuel of NPP, with end of reprocessing, will occur after June 30th, 2005 and will rely on the storage of spent fuel on site in decentralized interim storage facilities. Permits to build these facilities have already been granted for nearly all the following NPP sites and first buildings have been erected. Capacities of these facilities vary from 600 to 1900 tHM depending on the facility. To avoid transport of spent fuel for the time until the interim storage facilities will be commissioned, spent fuel can

be stored on-site in CASTOR casks inside simple concrete housings. Concerning both East German NPP sites (Greifwald and Rheinsberg), a common interim storage facility outside of NPP is planned at ZLN Greifswald.

The management of nuclear waste has also recently changed in **Eastern European countries** which possess nuclear fuels supplied by Russia, that is states where electricity is provided through RMBK (Lithuania) or VVER (other countries) reactors. At an earlier date, spent fuel unloaded from these reactors was generally, after a three-year period of cooling in pools, sent to Russia for reprocessing without any backs ending of the reprocessing waste. This situation has evolved during the 1990s notably as a consequence of the disintegration of the former Soviet Union. Today, part of the discharged spent fuel is still sent to Russia for storage and reprocessing but most of it is stored in storage ponds or in dry casks often at the reactor sites but sometimes away from them. Therefore, new interim spent fuel storage facilities are under consideration or construction.

In **Slovak Republic** one wet fuel storage is in operation on the NPP Bohunice site and the construction of another one is under preparation on the NPP Mochovce site.

In **Hungary** a modular vault dry storage facility is in operation and its expansion is performed following the amount of discharged spent fuel assemblies. It is located on the site of the unique NPP present in Hungary: Paks.

In **Sweden**, the centralized CLAB facility for wet interim storage of spent fuel in Oskarshamn was taken into operation in 1985, with an initial capacity of 5000 ton. Today 4000 tons of fuel is in storage and an expansion up to a capacity of 8000 ton is implemented in 2007.

In **Czech Republic**, the unloaded spent fuel is neither reprocessed nor exported. Presently, it is stored on the site of the main NPP in the country: Dukovany. This interim storage is expected to be full in 2005; another storage on the same site is planned for 2005-2006. This dry storage facility will meet spent fuel storage requirements for both (Dukovany and Temelin) nuclear power plants.

In **Bulgaria** a dry storage facility is planned around 2005-2006 at the NPP Kozloduy and will allow the reduction of the amount of spent fuel shipped to Russia which is an indispensable solution to face the Bulgarian spent fuel problem.

3.4.7 Waste repositories

Final storage sites for spent fuel in deep geological formations are planned by most European states. Some countries appear to be more advanced like Finland or Sweden while others still made no kind of final decision (Spain, Netherlands, the United Kingdom) and could be involved in the subject though international collaborations only. There is yet no geological repository in operation for HLW disposal, neither in Europe or anywhere else in the world. The only facilities for final disposal which are currently under way concern the management of low-level waste or very low-level waste and these facilities are surface ones. With regard to HLW, only laboratory stage has been reached for the moment and in the following a brief status of the situation in main European countries is reported.

By the end of 2000, the government of **Finland** has approved the choice of Olkiluoto site to set up an underground spent fuel disposal on condition that the in situ studies program

(ONKALO laboratory) validate the site. In this case, the final facility will be running in 2020.

In **Sweden**, the underground laboratory Aspo was implemented in 1994 to test and validate methods and models which have been previously investigated on characteristics and behaviour of crystalline rocks. Seven countries are working on this project (Germany, Canada, Finland, France, Japan, United Kingdom and Switzerland). Simultaneously, a selection process has started to define an appropriate granite site. Today, two sites are still under consideration: Osthammar and Oskarshamn and the final selection in 2008 should allow the delivery of an operation license in 2018.

In **Switzerland**, the technical feasibility of the final disposal of HLW is investigated into two media: in granite and in clay. Underground laboratory have been constructed for each case: Grimsel for granite media and Mont-Terri for clay media and these experimental facilities are the ground of fundamental international researches. No final disposal seems to be planned for the time being. **Belgium** and **France** are also at the same step with an underground facility in operation (Mol in Belgium) or in construction (Bure in France) but no date has been yet scheduled for the beginning of HLW disposal.

Contrary to most countries which plan to develop concepts with different repositories for different waste categories, **Germany** plans one general final repository for all kind of radioactive waste. Proof of applicability was launched in 2000 in Gorleben site but the program stopped for 3-10 years. However, there are two others underground facilities in Germany which are dedicated to long lived ILW disposal but none operates anymore: the loading of waste at the ERAM Morsleben site was suspended in September 1998 and in April 2001 a declaration issued that the loading will not be pursued. Concerning the Schacht Konrad, first licensed in 2002, law suits to prevent its operation have been solved.

The **Netherlands** consider the one general final repository option for all kind of radioactive wastes too, but, a definitive decision has not been made, since HLW storage is still a possible option. Currently all types of waste are centralized in Borssele, in a dry interim surface storage, which is planned to run for 100-150 years. In the meantime, various solutions for a common reversible geological disposal will be inventoried and compared.

3.5 Reprocessing techniques and facilities

Reprocessing allows the separation of energetic elements for recycling and a significant reduction in high level waste volume and radio-toxicity prior to final disposal.

Currently, two reprocessing plants are in operation in Europe, one in the **United Kingdom**, at the Sellafield site, and the other in **France**, at the La Hague site. In both plants, reprocessing operations involve the chemical separation of plutonium and uranium from the dissolved spent fuel. Plutonium and uranium are separated through solvent extraction steps using the proven industrial process PUREX. The highly radioactive remaining 3% waste, MA and FP, is calcified, incorporated into a borosilicate glass matrix and stored pending final disposal. Reprocessing one ton of spent nuclear fuel produces 0.15 m^3 of high level solid waste, which contains nearly 99 % of the radioactivity of the spent nuclear fuel. Plutonium and uranium may be recycled:

- Uranium may be sent to the conversion plant prior to a new enrichment,
- Plutonium may be sent to a dedicated mixed oxide (MOX) fuel fabrication plant.

Including both facilities, the European spent fuel reprocessing capacity rises to about 2550 tHM per year for uranium oxide fuel and 1500 tHM per year for Magnox fuel. The volume of solid wastes that are generated by these reprocessing activities was assessed to be of about 6600 m^3 per year for low level waste, 2500 m^3 per year for intermediate level waste and 230 m^3 per year for vitrified high level waste.

Two avenues are being explored to improve present reprocessing processes: hydro-metallurgical processes and pyro-chemistry.

3.5.1 Hydrometallurgical processes

Improvement of the existing PUREX process, including additional aqueous or solvent extraction steps, would enable further separation of radio-elements:

- Separation of neptunium: a significant fraction of neptunium is not extracted by the existing PUREX process. Modifications in the extraction / scrubbing flow sheet would increase the neptunium extraction yield,
- Separation of iodine: iodine is, for its main part, desorbed during spent fuel dissolution and recovered from the dissolution off-gas through soda washing. An alternative selective process allows the recovery and purification of iodine,
- Separation of technetium: an additional extraction step allows the recovery of more than 99% of the soluble fraction.

More advanced processes for waste management are under consideration which includes partitioning of long-lived radio-nuclides. Such processes would allow a reduction in the long-lived radio-toxicity of waste to be disposed of in geological repositories by recovering most of the heavy long-lived radioactive elements and thus would limit the required capacity of the repositories and enhance the safety of the final disposal.
The main radio-nuclides targeted for separation are actinides (Np, Am, Cm) along with uranium and plutonium, and some of the long-lived fission products (^{129}I, ^{99}Tc, ^{135}Cs).
Processes aiming at partitioning MA and FP are at different levels of development. For most of partitioning processes, the industrial feasibility has not yet been demonstrated.
New solvent extracting processes are under development and planned for demonstration (DIAMEX, SANEX, ...). These new extracting processes have been developed to recover long-lived radio-nuclides contained in extracts from the advanced PUREX process.
The scientific feasibility of the partitioning has been demonstrated by a massive research program supported by broad international cooperation. Its feasibility was demonstrated in 2001 following a series of tests conducted on dissolved spent fuel solutions, in the CEA's Atalante facility at Marcoule, France. The partitioning process performances were found very satisfactory: ~ 99.9 % of the minor actinides were recovered. The CEA has selected a route for the demonstration of the technical feasibility to be completed before 2006.

The following table lists the most interesting processes, their goal and their status of development.

Table 3.1: Objectives, processes and status of portioning of MA and FP.

Objective	Process	Status
Separation of U, Pu, FP+MA	PUREX	Industrial-scale process
Separation of U, Pu+FP+MA	UREX	Industrial feasibility
MA partitioning, one-extraction-cycle process	DIDPA process, SETFICS, PALADIN	Scientific feasibility
An+Ln co-extraction	TRUEX, DIAMEX, TRPO	Technical feasibility
An, Ln separation	TALSPEAK,CTH,SANEX, CYANEX, ALINA, BTP	Technical feasibility
Am, Cm separation	SESAME, Am precipitation	Technical feasibility
I, Np, Tc recovery	Advanced PUREX	Industrial feasibility
Cs and/or Sr recovery	Calixarenes, titanic acid	Technical feasibility

3.5.2 Pyrochemical processes

Alternative processes to aqueous extraction are also being investigated, notably those involving pyro-chemistry. These processes are more specifically dedicated to the reprocessing of transmutation targets, which are hardly soluble in aqueous nitric acid solution, and of some innovative fuels, such as thorium oxides, for future reactors.

Investigations are carried out within the framework of PYROREP, a CEA-coordinated European R&D program.

Theses processes which allow separation of nuclides from a radioactive waste stream involve several techniques such as: volatilization, liquid-liquid extraction using non-miscible metal-metal phases or metal-salt phases, electro-refining in molten salt, fractional crystallization, etc. They are generally based on the use of either salts with low melting points, such as chlorides or fluorides, or of molten metals, such as cadmium, bismuth or aluminum. Some pyro-chemical techniques have been tested on different types of fuel (oxide, metal), mainly for fast reactor applications.

For example, about 3 metric tons of fuel from EBR II has been treated by the Argonne National Laboratory and two additional tons will be treated. The process uses electro-refining in a molten salt. Separation of uranium from others actinides and fission products has been demonstrated at a semi-industrial scale. Other types of separation are only at the stage of scientific or technical feasibility.

Other pyro-chemical processes have been developed in this American laboratory:

- PYRO-A process: for the separation of transuranic elements and fission products contained in the oxide powder resulting from denitration of the UREX extract. This is performed in a $LiCl\text{-}LiO_2$ molten salt bath. An electro-refining step separates the transuranic elements on a solid cathode from the rest of the fission products which are immobilized in a ceramic composite waste form. The cathode deposit of transuranic elements is formed into ingots for subsequent fabrication of transmutation targets.
- PYRO-B process: for the processing and recycling of fuel from a transmuter reactor. An electro-refining step separates the residual transuranic elements from the fission products and recycles the transuranics as fuel to the reactor.

3.6 Definition of key parameters for processes and technologies

3.6.1 Inventory of processes and technologies under study

At present, policies of nuclear countries concerning waste management can be divided into three categories:

- countries which adopt direct disposal of spent nuclear fuel (called once through fuel cycle), like United States and in Europe: Finland, Spain and Sweden,
- countries which have own spent fuel reprocessing plants in order to recycle fissile uranium and plutonium formed during reactor operation, such as France, the United Kingdom and Russia,
- countries which are pursuing reprocessing abroad that is in Europe: Belgium, the Netherlands, Switzerland, and Germany until 2005 and out of Europe: Japan. It has to be noticed that Belgium and Switzerland are abandoning this strategy. In few years Japan will have its own spent fuel reprocessing plant.

It has to be noted that the situation in Eastern Europe states (Armenia, Bulgaria, Czech Republic, Hungary, Lithuania, Romania, Slovakia, and Slovenia) is quite unclear and has to be clarified. Generally speaking, these countries have contracts with Russia to reprocess part of their spent fuel but in the same time they often resort to interim storage for their spent fuel.

In order to ensure a sustainable development of nuclear energy, several countries are conducting research programs on implementation of a closed fuel cycle, which implies to recycle fissile elements in order to strongly minimize the radiotoxicity of the nuclear waste geologically disposed. Partitioning studies (to recover separately the main radiotoxic nuclides) and transmutation studies (to recycle or incinerate these separated radionuclides) are under investigation all over the world and are the subject of international collaborations particularly for the transmutation technologies which require tremendous investments. Among the countries working on P&T (Partitioning and Transmutation), France, Japan, Russia, South Korea and the U.S.A. are developing their own complete research programs.

The present status of nuclear waste management and R&D programs of these leading countries on P&T field is discussed in the following chapters.

France

PRESENT STATUS OF WASTE MANAGEMENT

France has a policy highly oriented towards nuclear energy for the production of electricity (about 80 %). France has early chosen to reprocess the spent fuel discharged from its nuclear power plants. It is realized at the La Hague site through the hydro-metallurgical PUREX process, which enables the co-extraction of uranium and plutonium by the complexing agent n-tributylphosphate (TBP) in the diluent hydrogenated tetra-propylene (TPH), and a selective stripping to recover separately these radionuclides. Isolated plutonium is recycled into a MOX fuel which is loaded into the core of 20 French reactors (among the 58 in operation).

The Waste Management Act of 1991 defined the legal framework for managing HLW. It organized the R&D along axis:

– Separation and transmutation (SPIN program)

- Disposal in deep geological layers
- Long-term (sub)-surface storage

A status report assessing the most promising R&D routes was submitted to French authorities in 2005. As a result, the legal framework for the waste management in France was updated with two important laws, both enacted in 2006:

- The Law on nuclear transparency and security established a new independent safety authority ASN (Autorité de Sûreté Nucléaire).

- The Programme Act on the sustainable management of radioactive materials and wastes is about all types of radioactive waste (not only long lived high level waste). The law defines financial arrangements for research, nuclear plants decommissioning charges, additional taxes on nuclear facilities to finance research programs. It provides also clear definitions for radioactive materials and waste and specifies that treatment is the way to reduce the volume and radiotoxicity of nuclear waste. A national management plan was created defining the solutions, the goals to achieve and the research actions.

In the frame of the program act, a major milestone was set in 2012 to select the Gen IV concept and transmutation-partitioning strategies:

- For the reactor, to select the innovations and the design of a prototype which will operate in 2020. To prepare this decision, the R&D is focused on the advance sodium fast system and an alternative one (gas or lead cooled, as a result of a comparative assessment of both reactor technologies by European stakeholders).
- For the fuel cycle strategy, to select or to assess the optimal option between :
    - The optimization of U and Pu recycling strategy using existing PWRs and EPR units to be implemented in France NPP as a first step, co-management of U, Pu and possibly Np in SFR as a second step (COEX^{TM} process).
    - The minor actinides (MA) homogeneous recycling in driver fuels of SFR reactors (with a low content of MA, typically 2.5 %).
    - The MA heterogeneous recycling with a higher content of MA (typically 10%-15% in mass) in blanket assemblies of SFR or as target in Accelerator Driven Systems (MA content of 40 to 50%).

European community

At a European level, the Institute for Transuranium Elements (ITU), one of seven institutes of the Joint Research Center, leads most of the research on P&T. Much of the activities are carried out on behalf of the European Commission and supported by the EURATOM budget.

PARTITIONING

The work carried out in pyrochemical reprocessing aims at the development of methods for the separation of actinides and lanthanides by electrolysis in molten salt media. The feasibility of this separation has been demonstrated.

In aqueous partitioning, ITU has demonstrated the feasibility of the recovery of minor actinides from concentrated highly radioactive extracts from the PUREX process, using the

DIAMEX process.

TRANSMUTATION AND CONDITIONING

Different concepts for fuels and targets for transmutation of minor actinides are being studied (EUROTRANS project). Emphasis is placed on the feasibility of fabrication and the physical and chemical properties of the candidate materials.
ITU has demonstrated the feasibility of the fabrication of (Y, Zr, An)O_{2-x} based magnesia composite pellets, by a combination of gel-supported process, infiltration and conventional blending techniques.

ITU is working in the elaboration of mineral-like ceramics for conditioning of actinides produced by the nuclear fuel cycle. Two groups of materials are studied: pyrochlores and monazite / brabantite type phosphate. Natural analogue studies have demonstrated their durability under geochemical conditions of a waste repository.

ITU is also working on LASER transmutation of ^{129}I, to produce ^{128}I though a (γ,n) reaction.

Japan

PRESENT STATUS OF WASTE MANAGEMENT

Currently, Japan has contracts to reprocess abroad (France, the United Kingdom) its discharged spent fuel. This closed fuel cycle strategy is going to be rapidly fully implemented on the Japanese territory, since a spent fuel reprocessing plant is expected to be licensed in few years, tests being, at present, performed on this new facility based on PUREX technology.

THE OMEGA PROGRAMME

The Japanese OMEGA (Options for Making Extra Gains in Actinides and Fission Products) R&D program on nuclide partitioning and transmutation technologies has been launched in October 1998. It is supported by the Japan Atomic Energy Research Institute (JAERI), the Japan Nuclear Cycle Development Institute (JNC) and the Central Research Institute of Electric Power Industry (CRIEPI), JAERI being the central institute devoted to nuclear energy researches.

The R&D areas covered by OMEGA program are:

- physical and chemical properties of minor actinides and fission products,
- partitioning of radionuclides from spent fuel dissolution liquors,
- transmutation: *i.e.* neutronic data, system design studies, reactor fuel and accelerator target development, high power accelerator development.

Japanese P&T research is divided into two distinct directions, each conducted by a specific Japanese nuclear research institute (despite interactions between them).

The first main line concerns the “double-strata fuel cycle” proposed by JAERI whose approach is close to the conventional closed fuel cycle. One stratum is dedicated to the commercial generation of electricity with a fuel cycle including the plutonium recycling. The second stratum is intended to the burning of high radiotoxic nuclides (minor actinides and long-lived fission products) in specific transmuters, notably in ADS.

The second line, led by JNC, involves a unique fuel cycle. Electricity is produced in fast neutrons reactors which are supposed to recycle the waste (i.e. long-lived radionuclides) they are generating within the fuel. This longer term nuclear power utilization is linked with the development of the fast reactor technology and the Japanese program should be synchronized with the international GEN-IV program, of which Japan is a member.

Korea

Up to now, spent nuclear fuel is stored at reactor sites, but Korean government has not decided on any definite policy for back-end fuel cycle, trying however to place the priority on the non-proliferation of nuclear fissile materials. The Korea Atomic Energy Research Institute (KAERI) have received strong support from the Government to conduct a R&D project of transmutation with the objective of key technology development by 2006 in the areas of partitioning and transmutation systems.

Korea intends to develop a self-reliant national energy supply with the end goal to be on an equal footing with advanced countries regarding nuclear power technology. In order the achieve so, Korea has launched a long-term R&D program, led by KAERI, which places the hybrid transmutation system and the pyrotechnical technology as the basis of the P&T cycle.

Russia

PRESENT STATUS OF WASTE MANAGEMENT

Russia accepts the concept of a closed nuclear fuel cycle that enables the reprocessing of spent nuclear fuel. Russia owns a reprocessing plant, the Mayak RT-l facility, which is currently being modernized and retrofitted. It is based on aqueous extraction process with TBP and is dedicated to the spent VVER-440 fuel. Another facility RT2 is planned (after 2010) for the reprocessing of spent VVER-1000 fuel in connection with the construction of facilities for the conditioning and (long-term) disposal of radioactive waste.

R&D PROGRAMME

The general strategy, which is oriented towards an increasing development of nuclear power, points out several key missions:

- construction of advanced thermal reactors which will run on enriched uranium until natural uranium reserves will be exhausted (next 20-40 years),
- reprocessing of spent fuel of thermal reactors with separation of Pu and long-lived nuclides,
- development of a new generation of fast reactors for large-scale electricity production which will meet economic efficiency, safety, waste minimization, proliferation resistance requirements,
- after 2030, deployment of a system of innovative fast reactors to incinerate plutonium and long-lived nuclides separated from spent fuel of thermal reactor.

U.S.A.

PRESENT STATUS OF WASTE MANAGEMENT

At present, the American policy concerning nuclear waste management is based on the Once-Through Fuel Cycle (open cycle) without any reprocessing of spent fuel but a direct storage. A geological disposal at Yucca Mountain has been recently approved by U.S. Congress and will be in operation around 2010. However, due to high volumes of spent fuel temporarily stored and the choice of an increasing development of nuclear power, the U.S. authorities will face rapidly the problem of saturation of Yucca Mountain site.

Therefore, the US DOE (Department of Energy) supports research programs to improve the current nuclear waste management and propose alternatives to the Once-Through Fuel Cycle policy. Three main objectives have also been highlighted:

- to reduce the spent fuel volume in order to optimize the utilization of the first geological repository at Yucca Mountain and to preclude need for a second disposal,
- to reduce the long-term radiotoxicity about a factor of 300. This factor would allow reduction of the radiotoxicity at level of the ore of natural uranium in less than 1000 years,
- to recover the energy value (e.g. Pu and U) from commercial spent fuel due to limited uranium resources (until mid 21st century if nuclear expansion).

With respect to these objectives, one of the priorities of US DOE is also to minimize the proliferation risk, which means that plutonium should not be recovered as an isolated product.

THE AFCI PROGRAMME

R&D programs involving P&T in USA are relatively recent and therefore the guiding lines are still changing. First programs ATW (Accelerator Transmutation Waste) and AAA (Advanced Accelerator Application), the latter started in 1999, were focusing on developing ADS to transmute long-lived radionuclides. The preliminary studies demonstrate that considering this unique option for transmutation was unrealistic (too expensive!) and that these hybrid systems would be useful only as a finishing treatment for limited streams.

Therefore, the current Advanced Fuel Cycle Initiative (AFCI), the substituting program for AAA, focuses on technologies which allow the transition from the present once-through cycle to an optimized closed fuel cycle. Also «more basic» transmuters such as thermal or conventional fast neutrons reactors are within the scope of this R&D program, which can be divided into two main phases:

- « AFCI series one » whose goal is, at mid-term, to minimize waste volume requiring geological disposal, first by extracting uranium and then by recycling plutonium as an impure material (e.g. mixed with neptunium).
- « AFCI series two » whose objective is more ambitious. It aims at reducing extensively waste radiotoxicity and minimizing the thern1al release of waste, which have been proven to be a limiting parameter for the total waste storage capacity.

Intensive research on advanced nuclear waste management are quite new in U.S., therefore, guidelines of programs appear yet to be still changing, testified by the recent introduction of the Integrated Program Approach in the AFCI program. However, goals to achieve, have been well established (need for «other Yucca Mountains» avoided, radiotoxicity reduction about a factor of 300, recovery of energetic fissile elements), as the priority given to limit the proliferation risk, which prevents for instance to resort to PUREX process.

More fundamental studies will be carried out in the next years, until 2007-2010, when a decision will be taken on the possible need for a second waste repository and on the linked nuclear waste management strategy. Until this deadline, researches will focus both on technologies for thermal and fast transmutation systems and on advanced aqueous and pyroprocessing separations technologies.

3.6.2 Definition of key parameters

In order to assess among the different strategies the most valuable way to produce electricity from nuclear energy and to manage spent fuel, different criteria have been identified. These indicators are expected to provide a basis for comparing the different considered fuel cycle scenarios. They could be useful to analyze the future of nuclear energy on different aspects and issues and consequently help clarifying this complex and open situation.

Criteria have been separated in the following subgroups:

Economics:

The cost of generated electricity (EUR/kWh) is all the more a key value since the European electricity market is open. It includes nuclear reactors running costs but also fuel fabrication and fuel treatment costs (separation and waste disposal). It depends on several parameters:

- **Resources** (cost evolution),
- **R&D** needs in term of costs, time (length for industrial deployment), risk (complexity, scientific and technologic maturity…),
- **Investments** required (costs, compatibility with present practice, use of existing facilities),
- **Production costs** (operating, maintenance, facilities dismantling costs…) for electricity production and waste management.

Sustainability:

- Electricity production (TWh (e) per ton fresh uranium (or thorium),
- Efficiency (net electric power / thermal power),
- Flexibility (possibility for the given strategy to be adapted according to technologic progresses or market conditions) and reversibility,
- Resources availability

Environment:

- Resources consumption (environmental impact of mines),
- Thermal release,
- Masses in fuel cycle (gaseous, liquid and solid).

General environment

- Releases (gaseous, liquid and solid) of radionuclides from mining of uranium (or thorium) until disposal of ultimate wastes,
- Radiotoxicity index (Sv),
- Release and toxicity of non-radiological effluents in the environment.

Radiological exposure for public and for workers

- Collective dose (man×Sv),
- Maximum individual dose (Sv/y),
- Maximum individual cumulative dose (Sv).

Safety of nuclear installations

- Criticality,
- Shielding devices (costs and operational management consequences).

Waste management

Short-term waste management aspects

- Radiotoxicity index (Sv) of waste (at the end of electricity production),
- Volume of HLW (m^3),
- Volume of long lived ILW (m^3),
- Volume of LLW (m^3).

Issues related to waste management prior to disposal

- Repository size HLW,
- Repository size long lived ILW,
- Suitability of existing/planned facilities to accommodate the new waste forms,
- Thermal loads
- Criticality.

Long-term consequences

- Evolution of radiotoxicity (Sv) with time,
- Maximum individual annual dose (Sv/y),
- Radiotoxic fluxes released in biosphere annual (Sv/y) and integrated over period of time (Sv).

Proliferation resistance

- Masses and isotopic composition of Pu,
- Masses of ^{233}U,
- Accessibility of Pu and ^{233}U.

Social

- Acceptability of fuel cycle strategy.

4. Back-End of the Fuel Cycle

Radioactive waste arises from the generation of electricity in nuclear power plants and from nuclear fuel cycle operations. As the waste represents a potential hazard to human health and the environment, it must be managed so as to reduce any associated risks to acceptable levels.

Radioactive waste occurs in a variety of forms with different physical and chemical characteristics. Radioactive waste may occur in gaseous, liquid and solid form. Waste may range from slightly radioactive to highly radioactive. Radioactive waste may be very small in volume or very large. Therefore, various facilities for the management of radioactive waste have been developed. Radioactive waste may also contain chemically and biologically hazardous substances and hazards associated with these substances have also to be considered in radioactive waste management.

4.1 Basic steps in radioactive waste management

Radioactive waste management has to consider the basic steps in the waste management process as parts of a total system [4.3].

Pre-treatment of waste is the first step in waste management that occurs after waste generation. It consists of, e.g., collection, segregation, chemical adjustment and decontamination and may include a period of interim storage. This step provides in many cases the opportunity to segregate waste streams for recycling or for disposal as non-radioactive waste when the radioactivity levels are sufficiently low to exempt them from regulatory control. It also provides the opportunity to segregate waste into short-lived and long-lived waste.

Treatment of radioactive waste includes those operations intended to improve safety or economy by changing characteristics of the radioactive waste. Basic treatment concepts are volume reduction, radionuclides removal and change of composition. Examples of treatment operations are: incineration and compaction of solid waste; evaporation, filtration and ion exchange of liquid waste. Treatment may lead to secondary radioactive waste.

Conditioning of radioactive waste involves those operations that transform radioactive waste into a form suitable for handling, transportation, storage and disposal. The operations may include immobilization of radioactive waste, placing the waste into containers and providing additional packaging. Examples of immobilization methods are solidification of low and intermediate level liquid waste in cement or bitumen, and vitrification of high level liquid radioactive waste in a glass matrix. Immobilised waste may be packaged in containers ranging from common steel drums to highly engineered thick-walled containers, depending on the nature of radionuclides and their concentrations.

Disposal is the final step in radioactive waste management. It consists of the emplacement of the radioactive waste in a disposal facility with reasonable assurance for safety. This safety is achieved by concentration and containment, which involves the isolation of the conditioned waste in a disposal facility. Isolation is attained by placing various barriers around the waste. The barriers can be either natural or engineered. A system of multiple barriers ensures that any release of radionuclides to the environment will occur at an acceptably low rate. Barriers can either provide absolute containment, such as thick-walled containers, or may retard the release of radionuclides to the environment, such as a buffer or host rock with high sorption capability. During the period of containment by the barrier system, the radionuclides in the waste will decay.

Waste management may also require complementary steps such as interim storage and transportation to and between the various waste management facilities.

4.2 Classification of radioactive waste

Important steps to unification of waste classification systems in use in various countries have been undertaken by IAEA (IAEA, 1994). It was concluded that a variety of properties of radioactive wastes could serve as criteria for classification of radioactive wastes (cf. Table 4.1) and that no unified system can therefore cover all possible issues of waste management.

Table 4.1: Important properties of radioactive wastes used as criteria for classification

• Origin
• Criticality
• Radiological properties o Half-life o Heat generation o Intensity of penetrating radiation o Activity and concentration of radionuclides o Surface contamination o Dose factors of relevant radionuclides
• Other physical properties o Physical state (solid, liquid or gaseous) o Size and weight o Compatibility o Dispersibility o Volatility o Solubility, miscibility
• Chemical properties o Potential chemical hazard o Corrosion resistance/corrosiveness o Organic content o Combustibility o Reactivity o Gas generation o Sorption of radionuclides
• Biological properties o Potential biological hazards

The main reason of the unified system proposed by the IAEA is therefore primarily limited to improving communication between stakeholders. The proposal focuses above all on safety aspects of radioactive waste disposal. Figure 4.1, taken from an IAEA publication [4.2], illustrates the proposed system.

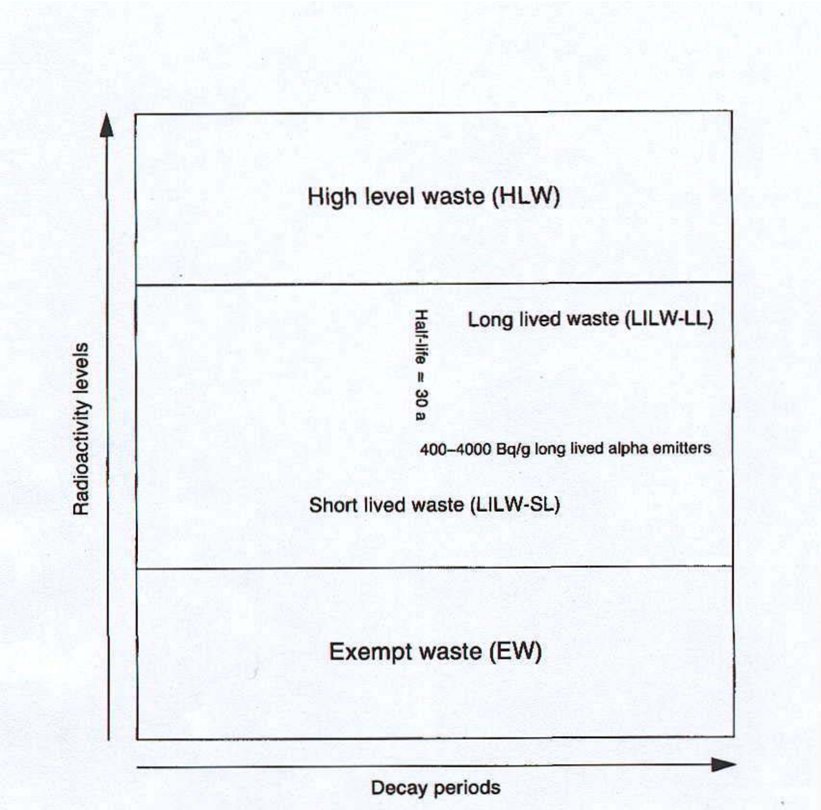

Figure 4.1: Proposed waste classification system (IAEA, 1994)

Along the vertical axis of Figure 4.1, radioactivity levels range from negligible to very high concentrations of radionuclides. As the level rises, there is an increased need to isolate the waste from the biosphere. Considering the horizontal axis, decay periods define the period during which the waste is radiologically dangerous. The radioactive waste is from this view divided into 3 main classes: 1) Exempt wastes, 2) Low and intermediate level waste (long lived ILW), divided further into short lived (SL) and long lived (LL) waste, and 3) High level waste. The typical characteristic of theses classes and proposed disposal options are given in Table 4.2.

Low level waste has been defined in the past as radioactive waste that does not require shielding during normal handling and transportation. A contact dose rate of 2 mSv/h was generally used to distinguish between low level and intermediate (medium) level waste. According to IAEA (1994) [4.2] recommendation this appears, however, of secondary importance in the present context.

Table 4.2: Typical characteristics of waste classes (IAEA, 1994)

Waste classes	Typical characteristics	Disposal options
1. Exempt waste (EW)	Activity levels at or below clearance levels which are based on an annual dose to members of the public of less than 0.01 mSv	No radiological restrictions
2. Low and intermediate level waste (long lived ILW)	Activity levels above clearance levels given in and thermal power below about 2 kW/m^3	Near surface or geological disposal facility
2.1 Short lived waste (long lived ILW-SL)	Restricted long lived radionuclide concentrations (limitation of long lived alpha emitting radionuclides to 4000 Bq/g in individual waste packages and to an overall average of 400 Bq/g per waste package)	Near surface or geological disposal facility
2.2 Long lived waste (long lived ILW-LL)	Long lived radionuclide concentrations exceeding limitations for short lived waste	Geological disposal facility
3. High level waste (HLW)	Thermal power above about 2 kW/m^3 and long lived radionuclide concentrations exceeding limitations for short lived waste	Geological disposal facility

Recognizing the diversity of classification systems in the Member States of the EU, the European Commission recommends [4.1] that the Member States and their nuclear industry adopt a common classification system of radioactive waste for national and international communication purposes as well as to facilitate information management in this field. This system would not, however, replace technical criteria where required for specific safety considerations such as licensing of facilities or other operations. The proposed classes (see Table 4.3) are very similar to the classes given in Table 4.2, but they are formulated in slightly different wording. The first class of the IAEA classification scheme (Exempt waste) can generally cover not only transitional wastes as the first class of EC classification scheme, but also waste that can include long-lived radionuclides with activity below clearance levels.

The EC recommendation emphasize that the main purpose of the classification system is to improve communication and to facilitate information management by providing a good descriptive tool enabling holdings of radioactive waste within the Community to be described to politicians and the public in a standard and easily understandable manner. The outlined system can be utilised in a qualitative assessment of possible disposal options for all types of waste arising from various fuel cycle facilities.

Table 4.3: EC Recommendation (1999) on a classification system for solid radioactive waste

1. **Transitional radioactive waste**
Type of radioactive waste (mainly from medical origin), which will decay within the period of temporary storage and may then be suitable for management outside the regulatory control system subject to compliance with clearance levels.
2. **Low and intermediate level waste (long lived ILW)**
In long lived ILW the concentration of radionuclides is such that generation of thermal power during its disposal is sufficiently low. These acceptable thermal power values are site-specific following safety assessments.
2.1 *Short-lived waste (long lived ILW-SL)*
This category includes radioactive wastes with nuclides half-live less than or equal to those of ^{137}Cs and ^{90}Sr (around 30 years) with restricted alpha long-lived radionuclide concentration (limitation of long-lived alpha emitting radionuclides to 4000 Bq/g in individual waste packages and to an overall average of 400 Bq/g in the total waste volume).
2.2 *Long-lived waste (long lived ILW-LL)*
Long-lived radionuclides and alpha emitters whose concentration exceeds the limits for short-lived waste.
3. **High level waste**
Waste with such a concentration of radionuclides that generation of thermal power shall be considered during its storage and disposal (the thermal power generation level is site-specific and this waste is mainly forthcoming from treatment/conditioning of spent nuclear fuel).

4.3 Waste disposal

The different waste types (cf. section 4.2) will eventually have to be disposed of in a final repository. Different types of repositories are considered for the disposal of low and intermediate level waste and of high level waste.

4.3.1 Disposal of short- lived long lived ILW

Short-lived long lived ILW can be disposed in a near-surface facility or in a mined repository at a depth of typically a few tens of metres (or in a deep geological repository in case of co-disposal with high level and/or long-lived wastes). Disposal facilities for short-lived long lived ILW constructed near the surface or at moderate depth are in operation in various EU countries, e.g. in France, Spain, UK, Czech Republic and Slovak republic. A scheme of a near surface disposal facility is shown in Figure 4.2.

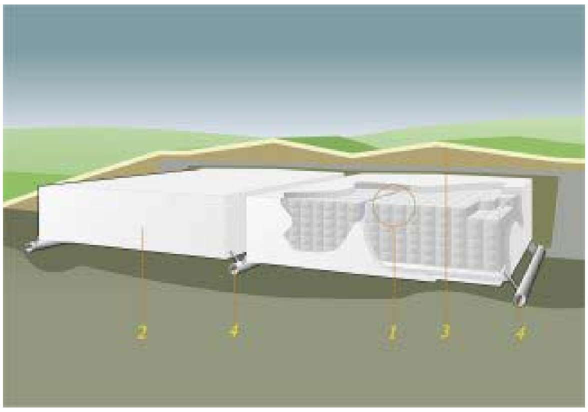

Figure 4.2: Scheme of a near surface disposal facility

Disposal facilities for short-lived long lived ILW constructed near the surface or at moderate depth are in operation in Sweden, Finland. A scheme of a disposal facility at moderate depth is shown in Figure 4.3.

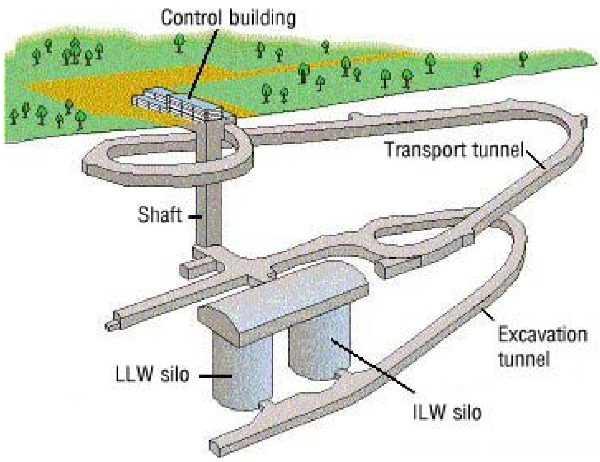

Figure 4.3: Scheme of a disposal facility at moderate depth

4.3.2 Disposal of long-lived ILW

Long lived ILW-LL has to be disposed of in a deep geological repository. Just two such facilities have been developed and were/are in operation: Morsleben (Germany, ended in September 1998) and WIPP (waste isolation pilot plant) in the US. A scheme of the German Konrad disposal facility is shown in Figure 4.4.

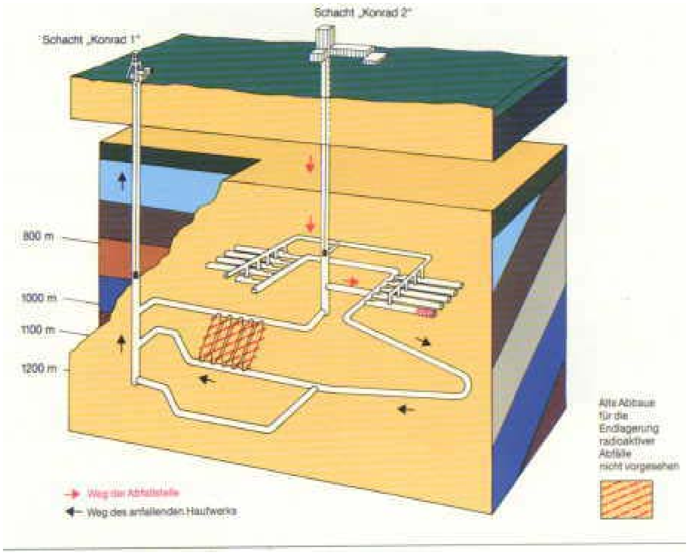

Figure 4.4: Scheme of the Konrad disposal facility

4.3.3 Geological disposal of HLW

Long-lived radioactive waste, such as spent nuclear fuel and high level waste from fuel reprocessing must be contained and isolated from humans and the environment for many thousands of years (NEA, 2004). Isolation means keeping the waste away from the biosphere making deliberate human intrusion to the waste difficult without special technical capabilities. The avoidance of locations that may attract inadvertent human intrusion is typically a factor in repository site selection. Since complete containment cannot be guaranteed for the whole of the period that waste represents a potential hazard, any eventual releases from the repository system should not present an unacceptable risk. The long-term safety of the repository is provided by the protective functions of the geological environment and the engineered barriers placed around the waste, as well as the stability of the waste form.

The disposal of long-lived waste in deep and stable geological formations is generally accepted as the internationally preferred option for the long-term management of high level radioactive waste.

Considered host formations

At present various types of geological formations are considered as possible host formations for deep disposal of high level radioactive waste. The main considered types are:

- hard rock formations: the considered formations are often granitic formations; this option is currently studied in, e.g., Canada, Finland, Japan, Spain, Sweden and Switzerland;

- argillaceous formations: the considered formations range from plastic clays, over endurated clays to mudstones; this option is currently studied in, e.g., Belgium, France, Japan, the Netherlands, Spain, Switzerland and recently in Germany;
- salt formations: as well salt layers as salt domes are being considered; disposal in salt domes is studied in Germany and the Netherlands;
- volcanic formations: examples of volcanic formations are tuff and basalt; the present US programme on high level radioactive waste disposal focuses on a welded tuff formation at Yucca Mountain.

Repository designs

In the case of geological disposal of high level radioactive waste, the geological barrier is complemented with a number of engineered barriers which depend on the characteristics of the host formation and of the disposed waste type. For a given repository site, disposal concepts are designed by taking into account waste characteristics such as radionuclide content, heat generation, criticality, radiation field, leaching rate, gas generation, etc.

Mainly, two main types of disposal configurations are considered for high level waste:
- disposal in galleries, where the waste is placed along the axis of a gallery;
- disposal in boreholes, where the waste is placed in horizontal or vertical boreholes that are drilled from a gallery.

A distinction has to be made between repositories located in water bearing formations on the one hand, and repositories located in salt or unsaturated formations on the other hand.

Repository designs developed for water bearing formations, i.e. hard rock and argillaceous formations typically consist of the following engineered barriers:
- a metallic container, often called canister or overpack, that has to remain intact during the initial gradient phase of the repository;
- a buffer surrounding the container, which consists in many designs of a swelling clayey material, such as bentonite; the buffer has to fill possible voids and fractures and prevent advective water flow around the container;
- a backfill is used to fill transport and access galleries;
- seals and stops are used to isolate the disposal gallery or borehole from the transport and access galleries and from the excavation disturbed zone, i.e. the first metres of the host formation that have been disturbed by the excavation of the gallery or borehole.

For repositories located in salt or unsaturated formations several above mentioned engineered barriers are also used, but other barriers may not be needed.

As an example, the repository concept, developed by ENRESA (the Spanish radioactive waste management agency), for disposal of high level radioactive waste in granite is given. It is based on the disposal of four spent fuel assemblies packed in carbon steel canisters in long horizontal galleries. The canisters are surrounded by high density bentonite. The disposal galleries are grouped into two symmetrically arranged disposal areas. Access is accomplished by means of "main drifts", which run perpendicular to the tunnels. The main drifts start from a central area, which includes the required underground infrastructure. Transport between the surface and the central underground area is accomplished by 4 access shafts. Figure 4.5 shows the underground installations.

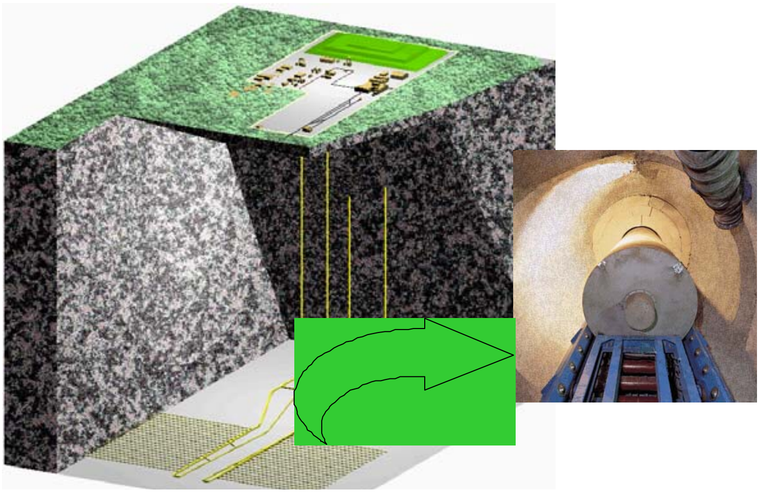

Figure 4.5: Underground installations for the ENRESA reference repository concept

5. Fuel Cycle Scenarios

5.1 Overview

One of the main tasks of this project has been the selection of a number of representative fuel cycle scenarios where the impact of Partitioning, Transmutation and Conditioning technologies on the final high level waste repository and the overall waste management will be studied.

The selection of these scenarios has been based on an overview of previous studies of P&T/C strategies performed in different EU framework programmes, NEA/OECD and other international organizations, taking into account the current national waste management policies of the different EU countries. The choice of scenarios is repository driven looking for a good representation of the different (primary and secondary) waste streams appearing in scenarios discussed in different EU countries. They are not necessarily the best choices for any particular country or situation and not a proposal of cycles to be implemented. They describe, however, coherent scenarios and should provide detailed information of the consequences of the key choices for the advanced fuel cycles and their waste management. In addition, the industrial scenarios have been chosen after the feedback of the whole collaboration, and have been consequently reevaluated and redefined, to make sure that they involve present practices or immediately deployable technologies.

The most important parameters for scenarios are the initial UO_2 enrichment, the LWR UO_2 fuel burn-up and the cooling time of irradiated fuel or high level wastes before storage in the final repository. All these parameters had been fixed to the same values for all the relevant scenarios (when possible).

Other parameters, like cooling times between different stages of the fuel cycles have some minimum requirements, but the precise value has only second order effects on the waste streams characteristics. For these parameters, whenever possible similar values have been proposed in the different scenarios, but they have not been forced to have exactly the same value.

Finally there are other parameters, like the share of installed power between different reactor types or the Pu and MA contents of the fuels of the different reactors, that are defined by the scenario constrains or optimization. No effort has been made to converge in the values of similar parameters for different scenarios. Although these differences imply that some care is needed to correctly interpret the comparison between scenarios, the differences must be accepted as scenarios characteristics (similar to the choices of components or connections of the scenario cycles).

The final scenarios selection includes three industrial scenarios, currently deployed in the nuclear power park or feasible at short-term, and two innovative scenarios, expected to by deployed at long-term. A brief summary of the main parameters of each selected scenario is shown in Table 5.1. Concerning the industrial scenarios, Scenario A1 using Thermal Neutron Reactors (TNR), the open cycle, has been chosen as a reference for all the advanced cycles. It represents the present situation on most countries with nuclear energy generation of electricity. Scenario A2 describes the only present commercial implementation of actinide recycling. It consists in the generation of energy in LWR and the recycling of Pu, only once, in the form of MOX also in LWR. The largest interest of Scenario A3 is the evaluation of the impact of the introduction of nuclear systems with fast neutron spectrum, but only for the Pu reutilization. Although no such reactor/cycle has been operated, the tests of sodium cooled reactors of significant power allow considering this kind of systems close to deployment.

Regarding innovative scenarios, Scenario B1 represents a Gen IV solution for sustainable nuclear energy production with waste minimization based on an Integral Fast Reactor. The fast reactors are configured with MOX fuel in the core and axial and radial blankets, where the MOX includes Pu and MA homogeneously. Scenario B2 is similar to the scenario A2 but with the introduction of a "second strata", based on a fast spectrum ADS, able to transmute the remaining Pu and MA into fission fragments. It concentrates the new technology (new reactors, reprocessing and fuel fabrication) in a relatively small part (in terms of mass flows) of the fuel cycle. This second strata operates with continuous recycling of the main actinides: U (only from ADS), Pu, Np, Am and Cm. Other scenario, considered for its study, was Scenario B3, which consisted in a double strata scenario with LWR with UO_2 and MOX, and fast reactors in the first stratum, and ADS with a MA burning strategy in the second one. This scenario is an alternative for the study of innovative strategies but no new calculations have been made within this project, and results are based on NEA studies.

With the aim of providing an assessment of the industrial feasibility and needs of R&D for implementation of P&T in short, medium and long term, the project has evaluated the date of availability for each major technology and the corresponding scenario. This work is summarized in Figure 5.1.

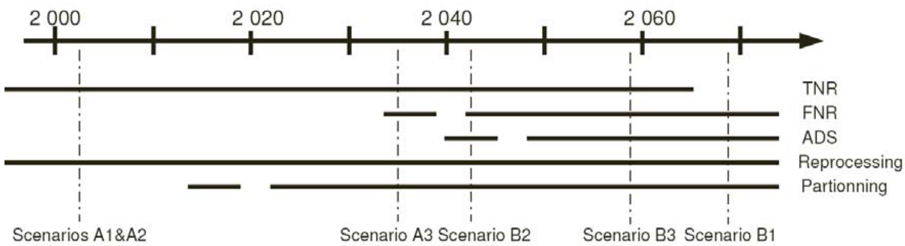

Figure 5.1. Date of availability for each major technology and corresponding scenario.

These estimations have allowed the collaboration to establish the background for the analysis of the transition between the beginning of the scenario, considering a current European nuclear power park, and a final equilibrium state. This approach, one of the firsts of its type, has been performed for several of the selected scenarios. However, these transition studies do not represent a proposal to be implemented or the best choices for a P&T strategy so the effort in its optimization has not been brought to the limit. On the contrary, the aim of this kind of studies is focused in learning about the wastes generation to provide basic information concerning strategic decisions with long-term effects. In addition, a methodology has been developed to precisely assess the feasibility of this type of advanced fuel cycle scenarios, taking into account different aspects as the existence of an initial inventory to eliminate and the large period of time necessary for the deployment for the new technologies.

Table 5.1. Main parameters of the selected scenarios.

Scenario	Type of reactors	Power (MWth)	Type of fuel	Burn-up (GWd/tHM)	Rate of actinide	Partitioned elements	Recycled elements
A1	LWR	2730	UO2	50	4.2% U-235	None	None
A2	LWR	2730	UO2	50	4.2% U-235	U Pu, only once	Pu, only once
			MOX	50	8.5% Pu		
A3	FR	3625	MOX	136	23.2% Pu	U, Pu	Pu
			Radial blankets	24	0.25% U-235		
			Axial blankets	15	0.25% U-235		
B1	FR	3625	MOX	136	23.2% Pu 2.7% MA	U + Pu + Np + Am + Cm	Pu + Np + Am + Cm
			Radial blankets	24	0.25% U-235		
			Axial blankets	15	0.25% U-235		
B2	LWR	2730	UO2	50	4.2% U-235	U + Pu + Np + Am + Cm	Pu + Np + Am + Cm
			MOX	50	10% Pu		
	ADS	850	(TRU:Zr)N	150	73.7% Pu 26.3% TRUs+U		

5.2 Basic scenarios

5.2.1 Scenario A1: Reference scenario - once through fuel cycle with Gen III reactors

This scenario is the simplest one. Nowadays, this scenario is the reference scenario in many countries (Sweden, USA…). Enriched uranium oxide fuel is charged in nuclear reactors. Then the spent nuclear fuel is discharged and, after cooling and conditioning, disposed of in a deep geological repository. This scenario is characterised by its maturity, as it is industrialised nowadays. The end of the fuel cycle (final repository) still needs development but is technically feasible.

Variants:
As there is no reprocessing in this scenario, the necessary hypotheses are very limited.

Three types of variants are proposed:

- sensitivity on the reactor type : LWR types (PWR, BWR, VVER),
- burn-up variation,
- fuel composition.

Final disposal of spent fuel are studied for A1 scenario in three types of geological media. These geological media depend on final storage sites planned by European states:

- Clay ,
- Granite ,
- Salt.

Human Intrusion scenarios are studied considering Spent Fuel assemblies from A1 scenario.

This scenario assumes the direct disposal of the present LWR spent fuel after an average burn up of 50GWd/tHM. The initial U-235 enrichment is set at 4.2% and the reference cooling time is 50 years. Spent fuel assemblies will first remain at the reactor pool and second at the interim storage facility in order to decay radiation and heat. A total cooling (or decay) time of 50 years has been established before emplacement inside the deep disposal repository.

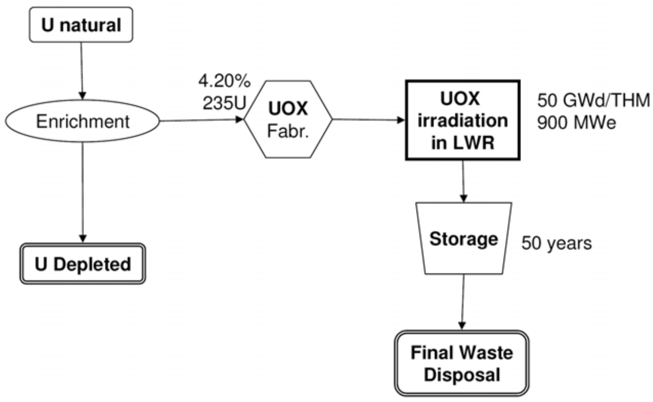

Figure 5.2: Scenario A1 - Reference scenario

Table 5.2: Characteristics of UOX Fuel Assemblies (scenario A1)

Lattice/reactor type	17 * 17	PWR
Physical dimensions:		
- Total length	4100	mm
- Section	217 * 217	mm^2
Mass of structure	145	kg
Mass of initial heavy metal	459.5	kg

Mass flow and waste streams:
The isotopic composition of spent fuel after 5 years of cooling time has been calculated [5.1] to provide the initial isotopic composition for the investigations on waste packages and their features [5.2] as reported in chapter 6.

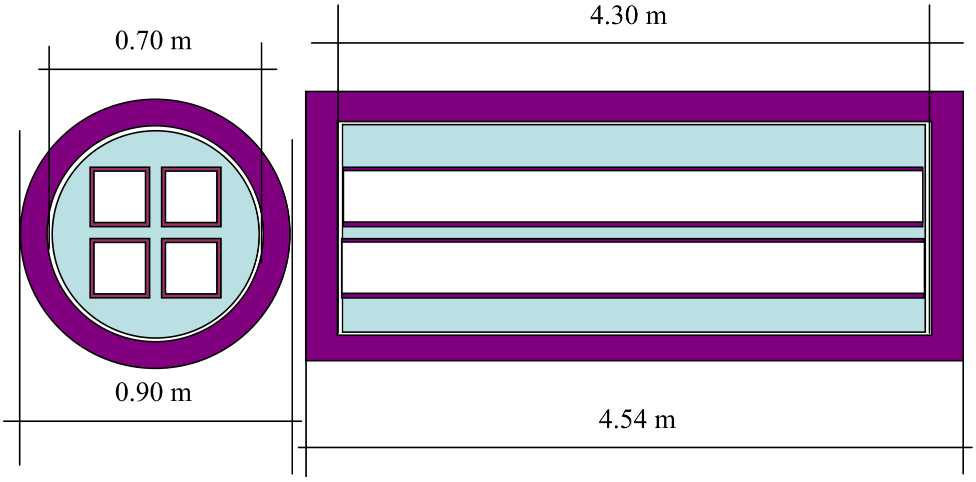

Figure5.3: Scenario A1: Waste Package for 4 UOX spent fuel assemblies

5.2.2 Scenario A2: Near term scenario - mono-recycling of Pu in Gen III reactors

This scenario is based upon single recycling of Pu in LWR.
The reprocessing of spent fuel allows for separation of uranium and plutonium from the fission products. The Pu is separated by reprocessing irradiated UO_2 fuel using aqueous partitioning system. Recovered Pu is mixed with depleted Uranium to prepare MOX fuel that will be burned in the same LWR of the reference scenario. The MA (Minor actinides) and FP (Fission Products) are vitrified and sent to a final disposal. The spent MOX as well as the HLW from the UO_2 reprocessing (including MA, FP and reprocessing losses) are cooled before being sent to the repository. The reprocessing loss rate is about 0.1% for U and Pu.

Nowadays, this strategy is used in France, Japan and some others European countries, except for the very end of the fuel cycle, i.e. the final repository which is under investigation. This scenario is also characterised by its maturity as it is industrialised today. The end of the fuel cycle (final repository) still needs development but is technically feasible.

Variants:
A2 a and b: Management of Pu: Double and Triple recycling of MOX.
The resulting HLW from the MOX reprocessing (including MA, FP and reprocessing losses) are cooled before being sent to the repository.
This scenario is technically feasible and has been demonstrated on a half-industrial scale.

A2 c: Management of Pu: Infinite recycling of MOX.
This strategy for Pu multiple recycling in LWR MOX fuels minimizes the cumulative mass of produced plutonium, while increasing the energy recovered from plutonium. It also paves the way for a multiple recycle in fast reactors to follow. However, there is a difficulty with the multiple recycle strategy in LWRs that does not apply to the same degree as in fast neutrons reactors. This is the issue of the build-up of the even mass plutonium isotopes, which will not fission directly in the thermal spectrum. Their build-up degrades the isotopic quality of the

plutonium leading to a requirement for impractically high initial plutonium contents as soon as the third recycle.
The build-up of the higher plutonium isotopes also contributes towards increased levels of the other minor actinides, particularly Am and Cm, with implications for radiotoxicity.
An additional consideration is that the reprocessing of MOX assemblies is technically more demanding than for UO_2 assemblies, which may result in a cost penalty.

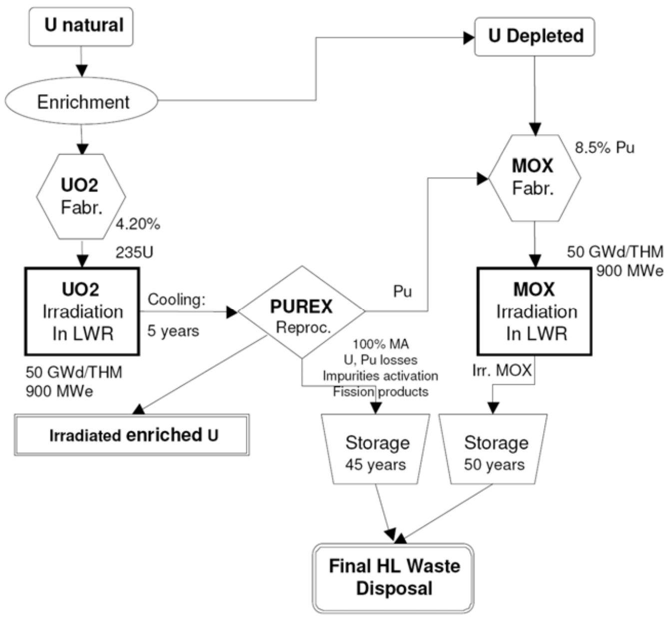

Figure 5.4: Scheme of Scenario A2.

A2d (or BN-1): Advanced fuel cycle: single Plutonium recycling in a standard PWR with Inert Matrix Fuel (IMF).
Such a scenario is a relevant one for a short term fuel cycle management which aims at finishing the Plutonium reprocessing scheme: decreasing the Pu stockpiles coming from the UO_2 irradiation, while allowing Uranium savings.
U is replaced by an inert component (oxides of zirconium, plutonium, yttrium, and erbium).
Such fuel is able to achieve roughly twice the plutonium destruction rate of conventional MOX.
IMF is used to manage plutonium inventories and to address the long term radiotoxicity of the spent fuel by minor actinide destruction.
Analytical studies in Paul Scherrer Institute (PSI) in Switzerland have indicated quite satisfactory behaviour for the IMF. Valuable data are considered to be crucial for the planning of a possible test for the IMF in an actual nuclear power plant environment.
They are studied in the context of medium-term plutonium utilization until 2020.

BN-2: Management of Pu: Multi-recycling of Pu as MIX fuel in LWR
Such a scenario implies the use of an advanced fuel often called MIX (or MOX/EUS) fuel, for which the fuel plutonium content remains fixed to 8 w% while allowing for an enriched uranium support to counter-balance the Pu isotopic degradation (reactivity aspect).

This scenario would decrease the total amount of actinides sent to final disposal by more than 35 %, compared to the base A2 scenario for the same energy production. The amount of plutonium sent to the waste would be about 10 to 5 times lower than for scenario A2, respectively at short and long time scales, while the amount of Am and Cm (present in lower quantities) would be increased by a factor 3 to 7.
An additional advantage is the distinct storage of reprocessed Uranium, whose radioactive features are easier to manage than the high level radioactive waste constituted by the FP and the minor actinides.
The effect on fission products is obviously weak, since no specific transmutation strategy is considered for them and those fission products quantities are just proportional to the amount of electricity generated. The slight differences arise from the different FP yields between uranium and plutonium isotopes.

The main disadvantages of such a scenario would be the increase of radiological source term with respect to the MIX fuel (fabrication, reprocessing, handling),

- the need of a long time operation (multirecycling) to get a significant benefit,
- the probable increase of fuel fabrication cost originating mostly from the need of enriched uranium to be mixed with PuO_2.

Other concepts have been developed:

- APA DUPLEX (Advanced Plutonium Assembly): it is a dual design incorporating both UO2 fuel rods and plutonium-bearing fuel rods within a single assembly. The innovative (Pu, Ce) O2 pellet fuel material will require extensive performance testing, including irradiation tests.
- CORAIL assembly combines UO2 and MOX rods in a conventional PWR mechanical design. It requires no new technology to be developed. This design is more tolerant of poor plutonium isotopic quality than conventional MOX and allows multiple recycle.

These scenarios require extended research and development programmes to establish on a commercial scale. They are studied in the context of medium-term plutonium utilization until 2020.

A2 e: Advanced fuel cycle: use of Pu Th MOX homogeneous (mixed oxide is manufactured and irradiated):
This scenario offers potential for improved resource utilization. Thorium is much more abundant in the earth than uranium. Thorium can be used as nuclear fuel as Th232 is fertile and transforms into U233 by neutron absorption which is fissile. A source of neutrons is needed to start the thorium fuel cycle, and this is usually done using enriched UO_2 or PuO_2. Reprocessing of spent (Th, U)O_2 fuel is in principle feasible using the THOREX process.
With this option, the production of MA would be reduced and no extra Pu would be generated. With regard to waste issue, the major advantage of the Thorium cycle is the significant reduction of MA production.
All thorium-based oxide fuels suffer from the disadvantage during reprocessing of the need to use hydrofluoric acid to obtain sufficient solubility. However, the chemical stability and leach resistance of thorium fuel is an advantage in case of direct disposal.
Because of radioactivity of thorium fuel, a new type of fuel manufacturing plant, using radiological containment and shielding arrangements, is required. This is common for all fuel production processes.
The technical feasibility of a closed cycle based on thorium must be demonstrated.

A2 f: Advanced fuel cycle: use of Pu Th MOX heterogeneous (special targets are fabricated and irradiated together with the MOX fuel)

A2 g: Advanced fuel cycle: MA recycling (Am + Cm): The separation of minor actinides from fission products constitutes an important step for the treatment of the waste. The scientific feasibility has been demonstrated on a genuine solution.
A number of processes for the fabrication of minor actinide targets are being developed. Actinide recycle and consumption appear promising as a waste management approach with significant gain on the waste radio toxicity inventory.
The separation of minor actinides is far from industrial maturity; the technical feasibility is not yet established. In order to design an economically viable plant, it is desirable to reduce the complexity of the separation processes.

A2 h: Advanced fuel cycle: Am recycling: This recycling could proceed either in a fuel form or as a target: the destruction of this isotope reduce the waste toxicity for storage times of about 1000 years. But there is no improvement, rather a slight deterioration for waste toxicities after 10000 years; this is due to the enhanced production of Cm from Am irradiation.
The technology must be developed and is technical feasibility must be demonstrated.

Final disposal of spent fuel:
Impact of Partition & Transmutation technologies on the final disposal of spent fuel assemblies is studied for “A2 a” scenario in three media: Clay, Granite and Salt:

- “A2 a” HLW,
- “A2 a” MOX

Human Intrusion scenarios are studied considering High Level Waste coming from “A2 a” scenario:

- “A2 a” vitrified HLW,
- “A2 a” MOX Spent Fuel assembly.

5.2.3. Scenario A3: Multirecycling of Pu in fast reactors

The scenario A3 will be used to evaluate the impact of the introduction of nuclear systems with fast neutron spectrum for the Pu utilization. The largest interest is in the time dependent case of progressive replacement of LWR by fast critical reactors, and the stationary limit has been assumed as a 100% park of relatively conventional sodium cooled European Fast Reactors, EFR.

This scenario produces 100% energy by means of Liquid Metal Fast Reactors (LMFR), recycling Pu and considering MA as a waste. The scenario simulation is based in the EFR CD9/91 configuration with MOX fuel (23.2% of Pu) in the central region of the core and depleted U in the axial and radial blankets. The irradiated fuels of all these regions are mixed together before reprocessing. The design of Scenario A3 (with details concerning storage, cooling and fabrication times) is shown in Figure 5.5.

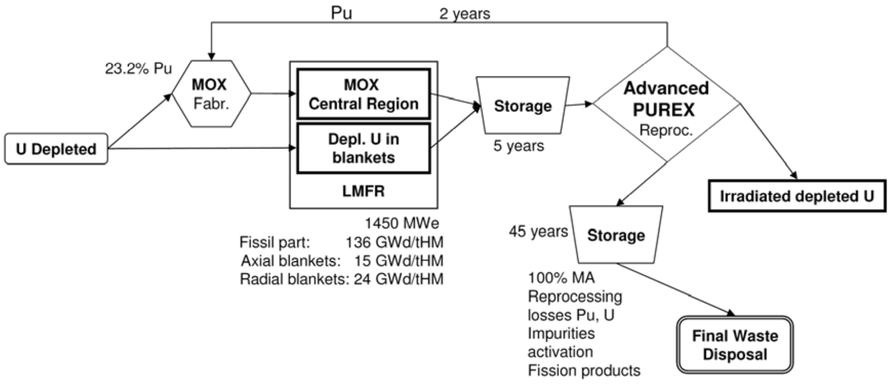

Figure 5.5: Scheme of Scenario A3.

From the wastes point of view there are only two main streams. Firstly, the irradiated depleted Uranium, which could be sent to a surface storage, although it could be used for future fuel fabrication to approach more of a sustainable nuclear energy utilization. Secondly, the HLW of the irradiated MOX after the Advanced Purex reprocessing: It contains most MA (100% of all actinides different from U and Pu is assumed), most fission fragments, the fuel impurities and their activation products, and the U and Pu reprocessing losses (0.1% assumed). As for the scenario A2, these HLW are assumed to be stabilized in a glass matrix, and finally sent to a geologically stable deep repository.

The canisters have to be stored in order to decay radiation and heat before emplacement inside the deep disposal repository. A decay time of 45 years has been established (the total cooling time for the wastes is 50 years, since the spent fuel was stored during 5 years before reprocessing).

The proposed waste package is designed to contain a single Universal Canister. The characteristics of the overpack (material, dimensions, etc.) are explained in detail in following sections. It has been assumed that the waste content of each Universal Canister is 40 kg of fission products and actinides. Starting from these assumptions, the following results have been calculated:

- the waste package production;
- the isotopic composition of a waste package containing 1 vitrified HLW separated in three components: Actinides, fission products, impurities and glass;
- the activity, thermal power and radiotoxic inventory of a waste package;
- the gamma and neutron emission spectra;
- the waste package radiation level.

No variants have been proposed to this scenario.

5.2.4. Scenario B1: Fast neutron GEN IV scenario

The scenario B1, very similar to A3, represents a Gen IV solution for sustainable nuclear energy production with waste minimization based on an Integral Fast Reactor. The reference design has been chosen as a sodium cooled European Fast Reactors, EFR, but loaded with advanced fuel including Pu (23.2%) and MA (Np, Am and Cm for a total of 2.7%)

homogeneously in the fuel of the central fissile region, and depleted uranium in the blankets. The design of Scenario B1 (with details concerning storage, cooling and fabrication times) is shown in Figure 5.6.

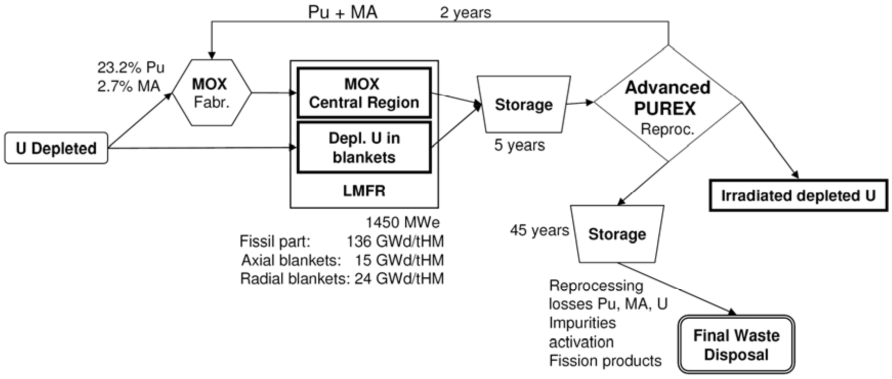

Figure 5.6. Scheme of Scenario B1: Long-term GEN-IV.

From the wastes point of view, this scenario is similar to A3 with the only difference that HLW contain only 0.1% (reprocessing losses) of MA.

Caesium and strontium are the main source of thermal power in the HLW during the first centuries, besides being important contributors to activity and radiotoxic inventory. The short and medium half-life of the majority of Cs and Sr isotopes allow that either the delay in the disposal of HLW or their partition from the HLW main waste stream could have a considerable impact in the first years of the geological repository functioning. Due to this, a series of variants minimizing the total amount of these elements as HLW have been studied:

a) Cs and Sr are partitioned from the HLW reducing their content to the 1% from reprocessing losses. After removal of Cs and Sr, the capacity of the canister is not driven by the heat load but still only 40 kg of HLW are loaded in each canister.
b) Taken into account that Cs has a long-lived isotope, only Sr is partitioned, so the whole amount of Cs and 1% of Sr from reprocessing losses are still present in the HLW. In this variant, the nominal 40 kg of the final HLW are loaded into the canister.
c) The normal management of Cs and Sr is maintained but the load of HLW per canister is increased up to 60 kg.
d) This variant is the same as a) but the heat load reduction benefit is assumed to increase the amount of HLW loaded per canister up to 60 kg.
e) This variant is the same as b) but the heat load reduction benefit is assumed to increase the amount of HLW loaded per canister up to 60 kg.

For variants scenarios a) and d), in the repository performance assessments, it has been taken into account that, after decay of ^{137}Cs, the remaining Cs would require geological disposal as it still contains the long-lived isotope ^{135}Cs.

5.2.5. Scenario B2: Simplified double-strata scenario with LWR and ADS

The scenario B2 is similar to the scenario A2 but with the introduction of a "second strata", based on a fast spectrum ADS, which transmutes the remaining Pu and MA into fission fragments. This second strata operates with continuous recycling of the main actinides (U, Pu, Np, Am and Cm), based on a pyro-metallurgical reprocessing with assumed losses of 0.1% for all these actinides. The scenario is the one with minimal introduction of new reactor technologies and would allow concentrating the new technology (new reactors, reprocessing and fuel fabrication) in a relatively small part in terms of mass flows of the fuel cycle. It should be noted that the used MOX contains a slightly different Pu fraction than in A2, in order to improve the global B2 cycle performance. Details about the design of Scenario B2 are shown in Figure 5.7.

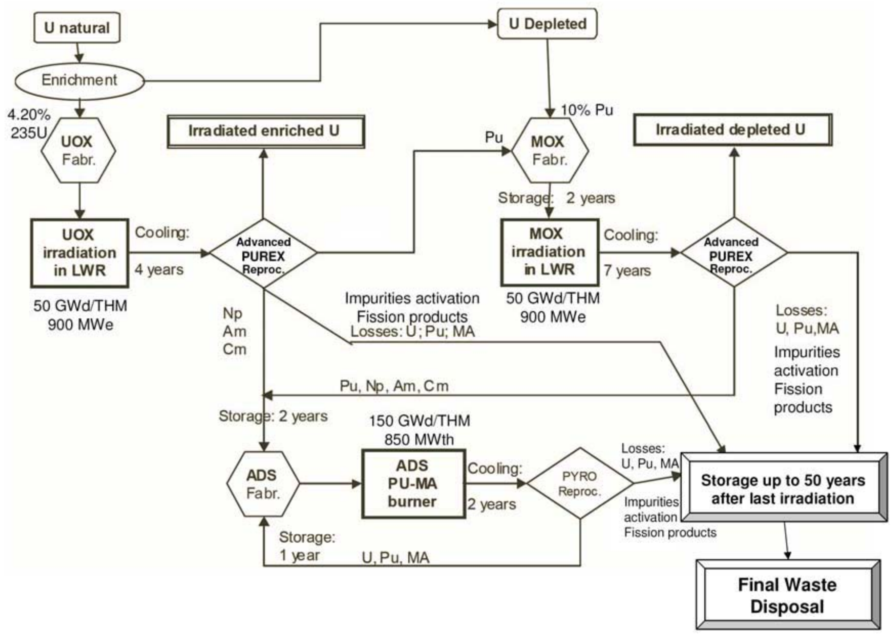

Figure 5.7: Scheme of Scenario B2.

From the wastes point of view there are several main streams. Firstly, there is the irradiated depleted and enriched Uranium, which could be sent to a surface storage, although they could be used for future fuel fabrication to approach more sustainable nuclear energy utilization. Secondly, the spallation target wastes, including most of the spallation products (100% assumed), most of the Pb/Bi activation products (100% assumed), the structural materials and their activation (100% of the structural materials in the Pb/Bi region assumed). The first two items are assumed to be collected by high efficiency filters in the Pb/Bi external circulation loop, the purified Pb/Bi is assumed to be fully recycled[2]. Finally, the HLW, coming from

[2] The hypotheses of this stream are very design dependent.

three different sources: irradiated and reprocessed UO2, MOX and ADS fuels. These HLW contain most fission fragments, the fuel impurities and their activation products and the U, Pu and MA (Np, Am and Cm) reprocessing losses (0.1% assumed) and most of all the other actinides (100% assumed). These HLW are assumed to be stabilized in different glass matrices, and finally sent to the geological repository. Special attention should be paid to the ZrN matrix of the ADS fuel. Although it is not considered in the present definition of Scenario B2, this matrix should be recycled if possible, due to the large amount of involved mass. The feasibility of the matrix recycling is presently being studied in several laboratories.

Under similar assumptions than previous scenarios, the following results have been calculated:

- the waste package production;
- the isotopic composition of a waste package containing 1 vitrified HLW separated in three components: actinides, fission products, impurities and glass;
- the activity, thermal power and radiotoxic inventory of a waste package;
- the gamma and neutron emission spectra;
- the waste package radiation level.

A variant of this scenario has been proposed. As the basic ADS fuel is composed of (TRU:Zr)N, nitrogen should be largely enriched in ^{15}N to avoid the creation of ^{14}C by irradiation. However, a certain amount of ^{14}C will be created, so a new design of ADS fuel has been proposed. In this variant, ADS fuel composition is $(TRU:Zr)O_2$. In addition, a new set of MOX parameters (Pu content of 8.5% instead of 10% and cooling time of five years for UO_2 spent fuel instead of four years) has been assumed for the first stratum to become more similar to Scenario A2.

In this variant, it has been proven that the ^{14}C creation when using oxides fuel is reduced compared with the basis scenario, reducing the maximum dose from the repository in the period from 30000 to 100000 years.

Partitioning of Cs and Sr is a management operation of the waste streams, but not of the fuel cycle scenario. As the main waste streams are very similar between scenarios with P&T of Pu and MA, reduction factors introduced by these variants must be also very similar. Hence, conclusions concerning B1 variants are easily applicable to Scenario B2.

5.2.6. Scenario B3: Simplified double-strata scenario with LWR and ADS

Finally, the scenario B3 is similar to the scenario B2 but with an additional stage of continuous Pu recycling in fast reactors within the first strata. The scenario will not be fully simulated, but the results from previous simulations in the NEA bibliography and some complementary calculations will be used for sensitivity studies and comparisons with B2. Figure 5.8 shows the design of Scenario B3.

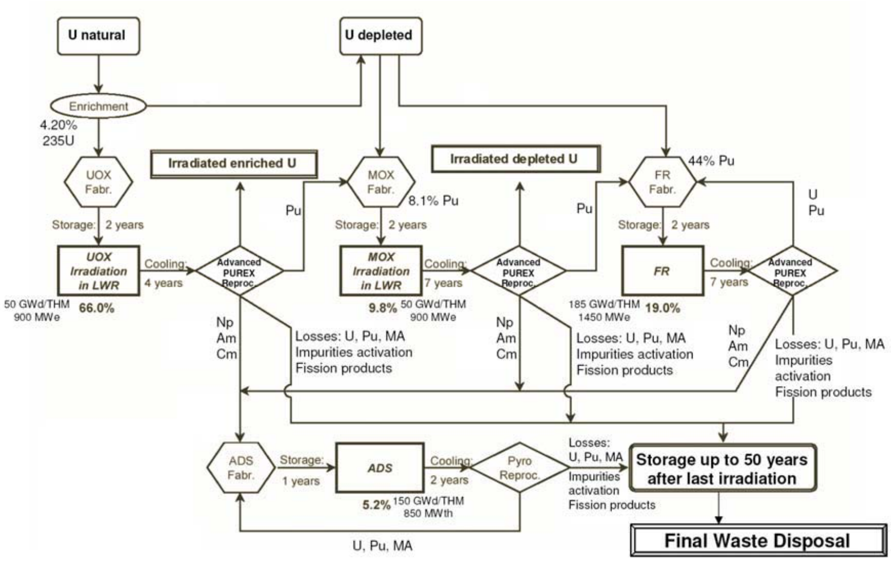

Figure 5.8: Scheme of Scenario B3.

5.3 Scenario calculations

5.3.1 Introduction

The scenarios selected in the RED-IMPACT project are divided in 3 categories:

1. *The equilibrium scenarios:*

These scenarios are considering an equilibrium situation corresponding to the final situation of the nuclear park evolution: for instance, a 100% GEN IV reactors park recycling Pu and minor actinides. The study of the equilibrium scenarios allows to simplify the waste package calculations and to make some comparisons between different strategies. However, this approach frequently considers asymptotic situations and does not take into account the following:

- the transition between the current nuclear park and the final situation, and
- the existing waste and their impact on the geological disposal.

The equilibrium scenarios studied in RED-IMPACT are described in chapter 5.1 and are reminded here:

- Scenario A1 : Reference scenario : continuation of the open cycle
- Scenario A2 : Continuation of the monorecycling of Plutonium in PWRs
- Scenario A3 : multirecycling of Plutonium in fast reactors
- Scenario B1 : Fast neutron Gen IV scenario
- Scenario B2 : simplified double strata with PWR + ADS

2. *The variant scenarios:*

The variant scenarios are equilibrium scenarios in which one parameter has been modified to study the effect of one change: burnup, reactor type, separation of Cs and Sr etc.

The variant scenarios considered in RED-IMPACT are the following:

- Variant on A1 and A2 scenarios with VVER reactors
- Variant with BWR
  Effect of a change in reactor type (BWR)
- Variant on A2 scenario involving PWR reactors:
  Multirecycling of Plutonium instead of single recycling
  Use of Inert matrix instead of Uranium matrix
- Variant on B1 scenario
  Effect of the separation of Cesium and Strontium
  Effect of a change in the incorporation rate of nuclides in the HLW
- Variant on B2 scenario
  Change in the ADS fuel type: oxides instead of nitrides in the fuel Matrix.

3. *The transition scenarios:*

The transition scenarios are studying the transition between the current nuclear park and the final equilibrium situation. These studies are more realistic than equilibrium scenarios but also more complicated. It allows to take into account the existing waste and the waste produced before the deployment of P&T in the case of advanced scenarios.

Three transition scenarios have been selected:

- scenario with a final situation corresponding to A3
- scenario with a final situation corresponding to B1
- scenario with a final situation corresponding to B2

The detailed description of theses scenarios is in the documents [5.1] and [5.2].

In the RED-IMPACT project, the existing waste has not been considered, because of the difficulty to obtain technical data from each country. However, the HLW produced from 2010 to 2200 have been calculated, and give a good idea of the impact of the transition.

It is important to remember that the complete calculation process has not been applied to all the scenarios. Only the equilibrium scenarios have followed the complete process.

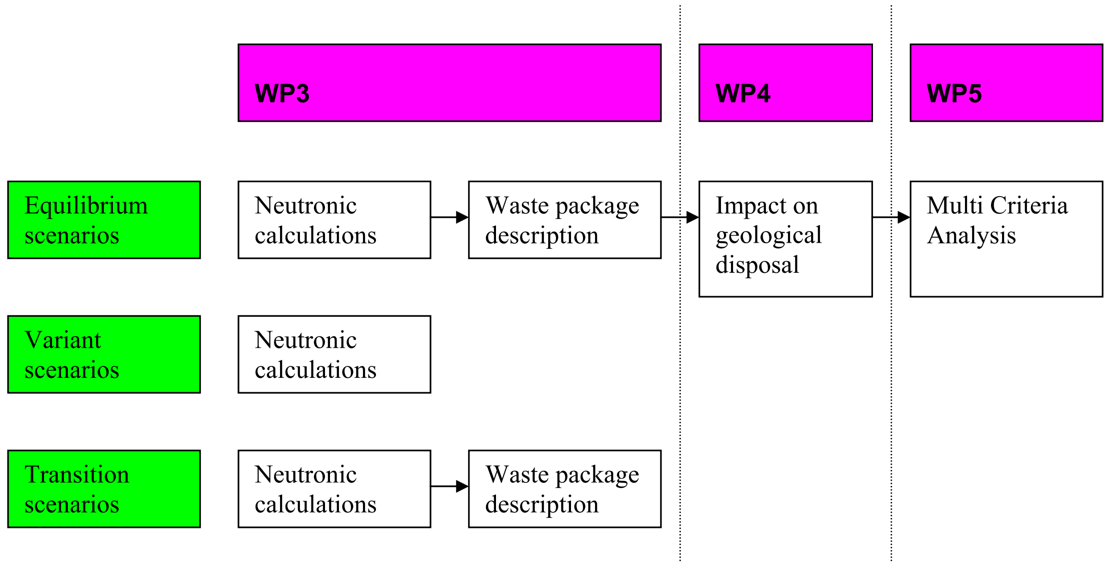

Figure 5.9: Calculations done for the three types of scenarios

5.3.2 Preliminary benchmark

Due to the high number of partners involved in the project, using different neutronic calculation tools, it was initially decided to perform a neutronic benchmark to assess the possibility of comparing the results obtained by different partners. The benchmark calculations have considered two very standard cases:

- PWR UOX fuel assembly at 4,2% U-235, 50 GWd/tHM
- PWR MOX fuel assembly at 8,5 % Pu, 50 GWd/tHM

The object of the benchmark was to compare the final isotopic composition given by the neutronic codes. The comparison was focused on U, Pu and minor actinides isotopes.

The benchmark assumptions and detailed results are reported in [5.3].

The main results are:

1) In the case of PWR UOX fuel assembly:
Acceptable discrepancies for U and Pu isotopes and Am241 (<10%)
Higher discrepancies for the other isotopes: Am 243, Np and Cm isotopes

2) In the case of PWR MOX fuel assembly:
Acceptable discrepancies for Pu239, 240 and 241
Higher discrepancies for the other Pu and minor actinides isotopes

Due to the high discrepancies obtained during the benchmark exercise, it was decided to reduce the number of neutronic tools used for the neutronic calculations of the project, so as to be able to make comparisons between the scenarios.

5.3.3 Equilibrium scenarios

Waste streams:

In the equilibrium scenarios, the following HLW and long lived ILW streams have been considered:

Table 5.3: HLW and long lived ILW waste streams

	A1	A2	A3 and B1	B2
HLW	LWR UOX spent fuel	- Glasses produced by the reprocessing of LWR UOX spent fuel - LWR MOX spent fuel	- Glasses produced by the reprocessing of MOX fast reactors spent fuel - Glasses produced by the reprocessing of radial blankets - Glasses produced by the reprocessing of axial blankets	- Glasses produced by the reprocessing of LWR UOX spent fuel - Glasses produced by the reprocessing of LWR MOX - Glasses produced by the reprocessing of ADS spent fuel
Long lived ILW	-	- Compacted wastes produced by the reprocessing of LWR UOX	- Compacted wastes produced by the reprocessing of MOX fast reactors spent fuel - Compacted wastes produced by the reprocessing of radial blankets - Compacted wastes produced by the reprocessing of axial blankets	- Compacted wastes produced by the reprocessing of LWR UOX spent fuel - Compacted wastes produced by the reprocessing of LWR MOX spent fuel - Compacted wastes produced by the reprocessing of ADS spent fuel + Zr of ADS fuel matrix

Fuel Irradiation Calculations:

For the scenarios A1, A2, A3 and B1, the neutronic code CESAR has been used. CESAR (Simplified Evolution Code for Reprocessing Applications) has been developed in the frame of a collaboration between CEA and AREVA NC [5.5]. The purpose of CESAR is to provide the required characterization data for burn up fuels from PWRs, BWRs, and FRs. The code calculates the evolution of material balances, activity, decay heat, and neutron source emitted by the irradiated fuel. The neutronic data libraries (cross sections sets) are supplied by the CEA reference calculation codes for neutron physics: APOLLO for thermal spectrum systems and ERANOS for fast spectrum systems. It uses a Runge-Kutta method for calculations during irradiation and a matrix type method for calculations between cycles and during cooling time. CESAR is validated against post irradiation experiment.

CESAR 4 handles the following types of fuel:

- PWR enriched uranium fuel produced from natural uranium or reprocessed uranium from PWR burn up fuel
- BWR enriched uranium fuel produced from natural uranium
- PWR MOX fuel
- UNGG fuel (natural uranium graphite gas type)

For the Scenario B2, the code EVOLCODE2 [5.6] has been used. EVOLCODE2 is a combined neutronic and burn-up isotopic evolution simulation system, combining a stochastic and a deterministic method. The stochastic method is realized by the general Monte Carlo N-Particle Transport Code MCNPX, [5.7] which is in charge of the neutronic calculations. MCNPX code provides EVOLCODE2 with the geometry data and the neutronic flux and power results. The Isotope Generation and Depletion Code ORIGEN2 [5.8] performs the isotopic burn-up evolution per geometry zones as requested by the user. A deterministic method, based on the Bateman equations, is applied. EVOLCODE2 links automatically all the information required by these codes in order to perform a detailed and precise simulation.

The results of the irradiation calculations are compiled in [5.9] and summarized in the table below:

Table 5.4: Equilibrium scenarios: material balance [kg/TWh(e)]

	A1	A2 UOX	A2 MOX	A3 core	B1 core	A3-B1 axial blankets (*)	A3-B1 radial blankets (*)	B2 UOX	B2 MOX	B2 ADS
Pu	29,3	29,3	-57,2	-26,5	-22,8	363,7	272,0	29,3	-57,2	-96,5
Am	1,7	1,7	14,7	4,7	-2,5	0,5	0,8	1,7	14,7	-19,0
Np	1,9	1,9	0,5	0,3	-1,8	1,0	0,8	1,9	0,5	-12,5
Cm	0,2	0,2	3,4	0,5	0,0	0,0	0,0	0,2	3,4	-1,3
Minor actinides	3,8	3,8	18,6	5,4	-4,4	1,5	1,6	3,8	18,6	-32,8
	total A1	**total A2**		**total A3**		**Total B1**		**total B2**		
Pu	29,3	20,6		1,0		4,4		1,5		
Am	1,7	3,0		4,4		-2,3		-0,3		
Np	1,9	1,7		0,3		-1,6		-0,5		
Cm	0,2	0,5		0,4		0,0		0,3		
Minor actinides	3,8	5,2		5,1		-3,9		-0,6		

(*) The values in these columns are normalized with the energy produced in the blanket alone, which is only 4% of the total energy produced in the reactor

Conclusions on the fuel irradiation calculations:
Concerning the Plutonium balance:
A positive Plutonium balance for all the scenarios. The reason is that for the scenario A1 and A2, PWR UOX fuel is dominating and MOX spent fuel is not reprocessed in A2. For the scenarios A3 and B1, the concept of reactor chosen is the EFR CD9/91 in its high breeding gain version; the breeding gain being close to 0,02.

Concerning the minor actinides balance:
A negative Americium and Neptunium balance for the advanced scenarios B1 and B2
A positive or null balance for Curium

The scenarios A3, B1 and B2 lead to a small increase of the total Plutonium inventory. From the minor actinides inventory point of view, the best scenarios are the transmutation scenarios B1 and B2.

The fuel impurities irradiation has also been calculated. The initial compositions of impurities and activation products in the fuel have been maximised and can be considered as conservative values.

Structural components activation calculations
The structural components activation has also been calculated in RED-IMPACT. The structural components taken into account are:

- The cladding
- The top and the bottom of the fuel assemblies
- The grids (case of PWR fuel assemblies)
- The hexagonal tube (case of Fast reactor fuel assemblies)
- ADS structural components : spallation target, beam, window, structure supporting the spallation target

For the scenarios A1 and A2 involving PWR reactors, the initial composition of impurities and activation products are deduced from experimental analysis.

For the scenarios A3 and B1 involving European Fast reactor (EFR), the problem of the lack of specification concerning the initial composition of impurities and activation products had to be solved. The values from the specification of fabrication of the Superphenix fuel assemblies have been used. These values can be considered as conservative. The mass of structural components have also been deduced from the Superphenix data.

5.3.4 Variant scenarios

The detailed variant scenario studies are presented in [5.10].

Variant with VVER reactors:
VVER-440 is pressure water reactor with hexagonal fuel assemblies (FA), triangular assembly and pin net and "black" control fuel assemblies. Five variants of basic scenarios A1 and A2 with reactor VVER-440 are listed in Tab.5.5. Special composed FA (CFA) containing 30 transmutation pins with all Pu and MA from 90 spent classical UOX pins in inert matrix is used in variant A2.V2 - see Fig 5.10.

Table 5.5: Variants with VVER reactors

Scenario	Scenario characteristics	Fuel	VVER-440 fleet
A1.V	Open cycle of VVER-440	classical **UOX** in VVER	178 UOX cores
A2.V1	Monorecycling of Pu in VVER-440	classical **MOX** in VVER	156 UOX, 22 MOX cores
A2.V2	Monorecycling of Pu and MA in IMF in VVER-440 (IMF)	Inert Matrix Fuel (**IMF**) in CFA	178 IMF cores
A2.V3	Monorecycling of Pu in PuThMOX in VVER-440	homogeneousu **PuThMOX**	158 UOX, 20 PuTh MOX cores
A2.V4	Monorecycling of Pu in thorium cycle in VVER-440	homogeneous **UPuThMOX**	137 UOX, 41 UPuTh MOX cores

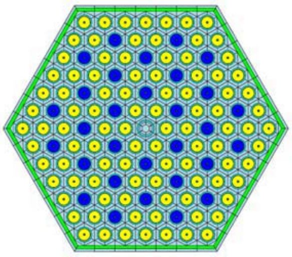

Figure 5.10: VVER Fuel assembly

Variant comparison:
The main results are summarized in the following tables:

Table 5.6: Variant VVER scenarios - Pu and minor actinides balance

	A1	**A1V**	*A2*	**A2V1**	**A2V2**	**A2V3**	**A2V4**
Pu	29,3	32,2	20,6	20,4	6,2	15,5	10,5
Am	1,7	1,8	3,0	1,6	1,3	1,5	1,3
Np	1,9	1,8	1,7	1,6	1,1	1,5	1,3
Cm	0,2	0,2	0,5	0,2	0,3	0,2	0,2
Minor actinides	3,8	3,8	5,2	3,3	2,7	3,3	2,9

Analysis:

Five scenarios with VVER-440 reactor (pressure water reactor with hexagonal fuel assemblies, triangular assembly and pin net and "black" control fuel assemblies) were analysed by spectral code HELIOS. Three advanced scenarios, enabling monorecycling of Pu in inert matrix fuel (composed FA), homogeneous PuThMOX and homogeneous UPuThMOX (VVER-440 Th-cycle) were compared with "classical" open cycle and MOX.

The main conclusions are:

- A1 scenario (open cycle in PWR) and A1V scenario (open cycle in VVER) are very close, both for Pu and minor actinides balances,
- A2 scenario (monorecycling of Pu in PWR) and A2V1 scenario (monorecycling of Pu in VVER) are also very close for the Pu balance. However, minor actinides production is higher in case of PWR,
- From the 4 variant scenarios with VVER based on single Pu recycling option, two scenarios can be selected as the best. Scenarios with Inert Matrix Fuel (A2.V2) and with UPuTh MOX (A2.V4) exhibit the smaller Pu and MA production, normalised on produced energy. IMF scenario performs also transmutation of Np and Am and exhibits even more favourable indicator values in comparison with UPuThMOX. Thorium cycle joins effectively Pu incineration with energy production from Th by some internal breeding of U-233.

Variant with BWR:
The studied variants deal with the continuation of the monorecycling of Plutonium in LWR and comprise the analysis of:

- ThPu MOX fuel in PWR,
- U MOX fuel in BWR and
- ThPu MOX fuel in BWR.

The following table shows the results for the variants of scenario A2.

Table 5.7: Material balance at end of decay (EOD)

	Δm/E kg/TWh$_e$				**Δm/m$_{BOC}$ wt.-%**			
	U PWR	ThPu MOX PWR	U MOX BWR	ThPu MOX BWR	U PWR	ThPu MOX PWR	U MOX BWR	ThPu MOX BWR
Th	0.0	-59.0	0.0	-65.5	-	-3.2	-	-3.0
Pa	0.0	2.0	0.0	1.6	-	-	-	-
U	-162.2	36.6	-81.1	26.4	-6.5	-	-3.45	16.3
Np	1.8	0	0.6	0.1	-	-	-	-
Pu	29.5	-115.9	-50.9	-94.1	-	-52.4	-34.6	-56.0
Am	1.7	4.9	3.97	3.8	-	309.7[3]	282.4	234.7
Cm	0.2	4.2	2.7	2.8	-	-	-	
TRU	33.3	-107.9	-43.6	-87.4	-	-45.4	-28.0	-50.1
Total	-128.9	-127.2	-124.7	-124.9	-5.2	-6.1	-5.0	-5.0

The consumption for Pu and TRU of ThPu MOX fuel is nearly twice larger as that of standard U MOX fuel independent of the reactor type (PWR or BWR). Comparing TRU/Pu transmutation in PWR and BWR with ThPu MOX fuel, the consumption per unit energy is maximal for PWR. But the maximum consumption for Pu and TRU per initial mass can be achieved with BWR.

Variant on A2 scenario involving PWR reactors:

A2 variant: single Plutonium recycling in PWR with Inert Matrix Fuel

A variant of the basis scenario A2 (continuation of the monorecycling in the PWRs) has been considered: a single Plutonium recycling in PWR IMF. Such a scenario is a relevant one for a short term fuel cycle management which aims at finishing the Plutonium reprocessing

3 The large values result from the very low initial content of Am-241 from 3 years Pu-241 decay.

scheme: decreasing the Pu stockpiles coming from the UO_2 irradiation, while allowing Uranium savings.

The detailed spent fuel results have been reported in a specific report [5.10]. Only the IMF assembly was compared to the MOX fuel assembly, as no change occurs for the reprocessed spent UOX fuel. Such results, expressed as ratios IMF/MOX for each nuclide and each time step, should help to assess the impact of this variant on the geological concepts.

As compared to irradiated MOX, spent IMF is characterized by:
- a global decrease of the activity and thermal power,
- a 20 % decrease of the activity after 100 years decay,
- a 30 % decrease of the thermal heat after 100 years decay.

Moreover the combined cycle UO_2 + IMF allows reducing much more Plutonium quantity and quality for underground disposal than for MOX. This fact consequently reduces the criticality risk and is in favour of non-proliferation criteria.

Although the source term of spent IMF is lower than the respective one for MOX, the choice of the inert matrix should also be driven by the safety reactivity coefficients, or the in-core behaviour, and also by the leaching properties and chemical stability when the spent fuel is disposed in geological media. A careful analysis should be done about the advantages and drawbacks of the two kinds of IMF: CERMET and CERCER.

A2 variant: Plutonium multirecycling in a standard PWR

This sensitivity study considers one particular variant of the RED-IMPACT A2 base scenario (single Pu recycling in LWR): the plutonium multirecycling in a standard PWR [5.11].

Such a scenario implies the use of an advanced fuel often called MIX fuel, for which the fuel plutonium content remains fixed to 8 w% while allowing for an enriched uranium support to counter-balance the Pu isotopic degradation (reactivity aspect).

It is shown that such a scenario would decrease the total amount of actinides sent to final disposal by more than 35 %, compared to the base scenario A2 for the same energy production. The amount of plutonium sent to the waste would be about 10 to 5 times lower than for scenario A2, respectively at short and long time scales, while the amount of Am and Cm (present in lower quantities) would be increased by a factor 3 to 7.

An additional advantage is the distinct storage of reprocessed Uranium, whose radioactive features are easier to manage than the high level radioactive waste constituted by the FP and the minor actinides.

The effect on fission products is obviously weak, since no specific transmutation strategy is considered for them and that fission product quantities are just proportional to the amount of electricity generated. The slight differences arise from the different FP yields between uranium and plutonium isotopes.

The main disadvantages of such a scenario would be (i) the increase of radiological source term with respect to the MIX fuel (fabrication, reprocessing, handling), (ii) the need of a long time operation (multirecycling) to get a significant benefit and (iii) the probable increase of fuel fabrication cost originating mostly from the need of an enriched uranium to be mixed with PuO_2.

Variant on B1 scenario:

Isotopes Cesium-137 and Strontium-90 are responsible of most of the short-term thermal power and radiation of the HLWs, thus affecting the waste management operations and the repository layout. Several variants of Scenario B1 have been analyzed regarding the possibility of partitioning and separate conditioning of Cesium and/or Strontium from the rest of the fission products. Separated Cs and Sr are assumed to be individually vitrified and conditioned in universal canisters loading 60 kg. The Scenario B1 variants that have been chosen are the following:

1. Variant B1.1 (40FP-60Cs-60Sr): Cesium and strontium are removed from the fission products stream. It is assumed that only 1% of Cs and 1% Sr go to the glass. A mass of 40 kg of non-volatile fission products plus actinides are loaded per Universal Canister.
2. Variant B1.2 (40FP-60Sr): Only strontium is removed from the fission products stream. It is assumed that only 1% Sr goes to the glass. A mass of 40 kg of non-volatile fission products plus actinides are loaded per Universal Canister.
3. Variant B1.3 (60FP): Scenario in which a mass of 60 kg of non-volatile fission products plus actinides are loaded per Universal Canister. Neither Cs nor Sr is partitioned from the fission products stream.
4. Variant B1.4 (60FP-60Cs-60Sr): Cesium and strontium are removed from the fission products stream. It is assumed that only 1% of Cs and 1% Sr go to the glass. A mass of 60 kg of non-volatile fission products plus actinides are loaded per Universal Canister.
5. Variant B1.5 (60FP-60Sr): Only strontium is removed from the fission products stream. It is assumed that only 1% Sr goes to the glass. A mass of 60 kg of non-volatile fission products plus actinides are loaded per Universal Canister.

The results of these variants are presented in the chapter 6.

Variant on B2 scenario:

The differences between the variant and the reference scenario are:

- The ADS fuel is composed by oxides in a Zr inert matrix, instead of nitrides. The equilibrium isotopic composition is also different.
- The cooling time after the irradiation in LWR-UO2 is now 5 years instead of 4 years.
- The Pu content in the MOX fuel is now 8.5%, as in the A2 Scenario, instead of 10%.

The isotopic compositions of the ADS wastes before reprocessing can be found in Annex 2, in addition to the initial ADS equilibrium composition. The wastes for the LWRs with the different fuels are supposed to be equal to the wastes coming from A2 Scenario.

The main advantage of this variant is to prevent the formation of ^{14}C due to the activation of Nitride in the ADS fuel matrix. The amount of ^{14}C is reduced in more than two orders of magnitude compared with the basis scenario. This reduces the maximum dose (due to only HLW) from the repository in the period from 30000 to 100000 years.

5.3.5 Transition scenarios

The transitions scenarios have been calculated with two different methods:

- A3 and B1: The calculations have been made with the code COSI. COSI is a neutronic code dedicated to the transition scenario calculations [4].
- B2: This transition scenario has been calculated performing simulation with EVOLCODE2 in a semiautomatic procedure. Different homogenised spent fuel pools have been considered to extract the required nuclear fuel for every year and reactor.

In the transition scenarios, the existing HLW have not been taken into account. Only the waste produced from 2010 are considered.

For each scenario, the annual reprocessing flux is calculated from 2010 to 2200. Using these values and the separation efficiency for each element, the amount of nuclides in the HLW is assessed:

Table 5.8: Transition scenarios – HLW nuclides content

	Nuclides in the HLW (tons)	% Fission products	Minor actinides in the HLW (tons)
A3 transition	19350	94,3	839
B1 transition	18343	97,9	128
B2 transition (*)	21696	98,3	131

(*) in this table, the assumption for B2 is a deployment of P&T in 2040, as for B1, so as to allow a fair comparison between scenarios.

This table brings the following comments:

- The nuclides in the HLW are mainly fission products
- The P&T allows to decrease the minor actinides inventory in the HLW and thus the long term (>100 years) radiotoxic inventory of the HLW.

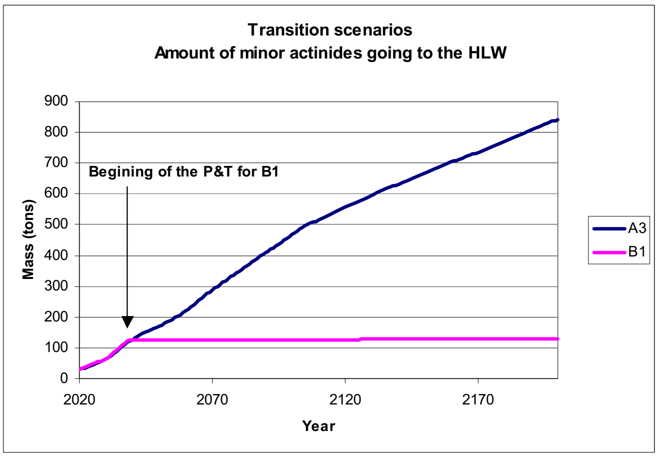

Figure 5.11: - Amount of minor actinides going to the HLW

With the Figure 5.11, we realize that the date of the deployment of the P&T has a major impact on the amount of MA in the HLW produced during the scenarios. In 2040, we already have 120 tons of MA in the HLW (in the European context, that is to say a 800tWh(e)/year nuclear park). Once the P&T is deployed, this amount remains almost constant.

Conclusions on the transition scenarios:
The transmutation of the minor actinides has not a significant effect on the total amount of nuclides going to the HLW. This value is highly dominated by the fission product contribution at 95 %. The difference between A3 and B1 or B2 is due to the separation of the minor actinides in the advanced scenarios.

However, the amount of minor actinides in the HLW is very much affected by P&T: this amount is almost constant from 2040 in B1: only the minor actinides losses (0,1%) are incorporated in the HLW from 2040. The results obtained with the B2 have been adapted so as to allow a fair comparison between transition scenarios: the date of P§T deployment assumed is 2040 whereas this date has been set at 2010 in the neutronic calculations.

5.3.6 Conclusions on scenario calculations

Many scenario calculations have been performed in the RED-IMPACT project. Most of these calculations concern equilibrium scenarios in which the situation corresponds to a final asymptotic situation of the nuclear park evolution.

The material balance for Plutonium and Minor actinides have been calculated for all the scenarios

The first type of scenarios is the equilibrium scenarios which have followed the complete process of study, including waste package description, impact on geological disposal and multi criteria analysis

The second type of scenarios is the variant scenarios based on equilibrium scenarios and the change of one parameter. The results show the possibility to reduce the production of Pu and minor actinides by the use of inert matrix or Thorium matrix in different types of reactors: PWR, BWR and VVER. The multirecycling of Pu in PWR have also been studied. In this case, the amount of plutonium sent to the waste would be about 10 to 5 times lower than for scenario A2, respectively at short and long time scales, while the amount of Am and Cm (present in lower quantities) would be increased by a factor 3 to 7. The technical feasibility of all these variant scenarios have not been analysed in the project.

The third type of scenarios is the transition scenarios, studying the transition between the current nuclear park and the final equilibrium situation. The results indicate that the radiotoxic inventory of the HLW produced before the deployment of P&T cannot be neglected. In this sense, P&T has to be deployed as soon as possible so as to reduce to total radiotoxic inventory of the HLW in the geological disposal.

6. Waste Composition and Waste Package Description

This chapter provides a description of possible final disposal waste packages for the conditioning of spent fuel assemblies and wastes (HLWs and long lived ILWs) generated during the reprocessing. The conditioning of these different wastes in suitable packages has to take into consideration the physical and chemical characteristics of the waste forms and the physical and chemical properties of the host rock and backfilling material in a final disposal facility. This has already been done within the different geological disposal programmes, from which the package concepts have been adopted. Nevertheless, no special matrices, other than borosilicate glass, have been used within the project for conditioning the reprocessing wastes generated in the different scenarios.

Five "Equilibrium Scenarios" have been analysed, representing ideal scenarios in which there are not evolutionary technologies. More, three "Transition Scenarios" starting with current technology, which evolve to advanced technologies for partitioning and transmutation, have been studied.

Isotopes Cesium-137 and Strontium-90 are responsible of most of the short-term thermal power of the HLWs, thus affecting the HLW geological repository design and dimensions. Several variants have been analyzed regarding the possibility of partitioning Cesium and/or Strontium from the rest of the fission products.

The key assumptions on the different reprocessing processes regarding the fraction of each isotope in the spent fuel inventory that goes to the vitrified HLW or to the compacted long lived ILW are summarized on Table 6.1 and Table 6.2.

Table 6.1: Fractions of main chemical elements incorporated into the vitrified HLW (%)

SCENARIO		A2	A3	B1	B2		
REPROCESSING TYPE		Standard PUREX	Standard PUREX	Extended PUREX	Extended PUREX	Extended PUREX	PYRO
REPROCESSED FUEL		UOX (PWR)	Blankets + Core (FR)	Blankets + Core (FR)	UOX (PWR)	MOX (PWR)	ZrN +TRUN (ADS)
ACTINIDES (An)	U and Pu	0.1	0.1	0.1	0.1	0.1	0.1
	MA	100	100	0.1	0.1	0.1	0.1
FISSION PRODUCTS (FP)	Noble Gases	0	0	0	0	0	0
	Iodine	1	1	1	1	1	1
	Noble Metals	100	100	100	100	100	0
	Others	100	100	100	100	100	100
LIGHT ELEMENTS (Fuel Impurities)	H	0	0	0	0	0	0
	Carbon	10	10	10	10	10	10
	Chlorine	1	1	1	1	1	1
	Others	100	100	100	100	100	100
Fuel Assembly Structurals (particles)		0	0	0	0	0	0
Zr of the ADS fuel matrix (also for the Zr-fission product)		-	-	-	-	-	1

Table 6.2: Fractions of main chemical elements incorporated into the compacted long lived ILW (%)

SCENARIO		A2	A3	B1	B2		
REPROCESSING TYPE		Standard PUREX	Standard PUREX	Extended PUREX	Extended PUREX	Extended PUREX	PYRO
REPROCESSED FUEL		UOX (PWR)	Blankets + Core (FR)	Blankets + Core (FR)	UOX (PWR)	MOX (PWR)	ZrN +TRUN (ADS)
ACTINIDES (An)	U and Pu	0.02	0.02	0.02	0.02	0.02	0.02
	MA	0.02	0.02	0.02	0.02	0.02	0.02
FISSION PRODUCTS (FP)	Noble Gases	0	0	0	0	0	0
	Iodine	1	1	1	1	1	1
	Noble Metals	0.2	0.2	0.2	0.2	0.2	100
	Others	0.2	0.2	0.2	0.2	0.2	0.2
LIGHT ELEMENTS (Fuel Impurities)	H	40	40	40	40	40	40
	Carbon	0	0	0	0	0	0
	Chlorine	0.2	0.2	0.2	0.2	0.2	0.2
	Others	0.2	0.2	0.2	0.2	0.2	0.2
Fuel Assembly Structurals		100	100	100	100	100	100
Zr of the ADS fuel matrix (also for the Zr-fission product)		-	-	-	-	-	100

6.1 Equilibrium Scenarios

The five studied equilibrium scenarios have been classified into two groups: the near-term scenarios, referred to as A-scenarios which are built on current industrial practice, and the second group of fuel cycle scenarios, referred to as B-scenarios, that involves more innovative technologies.

A: Industrial scenarios:
- Scenario A1: Once through fuel cycle with Gen III reactors
- Scenario A2: Mono-recycling of Pu in Gen III reactors
- Scenario A3: Multi-recycling of Pu in Sodium Fast Reactor

B: Innovative scenarios:
- Scenario B1: Multi- recycling of Pu and MA in Gen IV reactors
- Scenario B2: Mono-recycling of Pu in Gen III reactor and Burning of Pu and MA in ADS

6.1.1 Scenario A1

The A1 scenario is the reference scenario. Enriched uranium oxide (UOX) fuel is charged in LWR. After reaching an average burnup of 50 GWd/tHM the spent nuclear fuel is discharged and, after cooling (50 years) and conditioning, disposed of in a deep geological repository. The waste generated in this scenario is:
- HLW (UOX spent fuel)

6.1.2 Scenario A2

The A2-scenario is based on mono-recycling of Pu in PWR. Pu is separated from irradiated UOX-fuel through aqueous reprocessing (Standard Purex) and mixed with depleted uranium and manufactured into MOX fuel, which is burned in the same PWRs reaching also an average burnup of 50 GWd/tHM. Minor actinides (MA) and fission products (FP) are vitrified and sent to a final disposal. The spent MOX, as well as the waste from UOX reprocessing are cooled before being sent to the repository. The solid wastes generated in this scenario are:
- HLW (MOX spent fuel and vitrified wastes from UOX spent fuel reprocessing)
- long lived ILW (structural parts of the dismantled UOX spent fuel assemblies)

6.1.3 Scenario A3

In Scenario A3, energy is produced by fast reactors (sodium cooled FR) recycling Pu. The reactor consists of a fissile core of MOX fuel, and axial and radial blankets of depleted uranium oxide fuel. Both types of fuel are reprocessed (Standard Purex) together. The cooling time before reprocessing is five years. The minor actinides are not separated, but remain in the final HLW. Pu is recovered to fabricate new MOX fuel to be burnt in a fully closed cycle. The reprocessing losses, all the MA and most of the fission products will be vitrified and disposed of in a geological repository, as well as the long lived ILW. The MOX fuel in the core includes 23.2% Pu, reaching a burnup of 136 GWd/tHM, 15 GWd/tHM in the axial blankets and 24 GWd/tHM in the radial blankets. The solid wastes generated in this scenario are:
- HLW (vitrified wastes from FR spent fuel reprocessing)
- long lived ILW (structural parts of the dismantled FR spent fuel assemblies)

6.1.4 Scenario B1

The Scenario B1 is similar to A3. The only difference is the use of advanced technology for reprocessing (Extended Purex) to separate MA (Np, Am and Cm), and the fabrication of advanced MOX fuel including, in addition to Pu, 2.7% MA homogeneously dispersed, for the central fissile region of the FR core. The solid wastes generated in this scenario are:
- HLW (vitrified wastes from FR spent fuel reprocessing)
- long lived ILW (structural parts of the dismantled FR spent fuel assemblies)

6.1.5 Scenario B2

This scenario is similar to the Scenario A2, but with the introduction of a “second stratum” based on a fast hybrid reactor ADS. Most of the power production occurs in an LWR-park with UOX fuel as scenario A1. After 5 years of cooling the irradiated fuel is reprocessed (Extended Purex), Pu being separated in one stream and MA in another. All the recovered Pu is mono-recycled as MOX in the LWR park. The spent MOX fuel is then reprocessed (Extended Purex). In the second stratum, all the MA from the spent UOX fuel plus the Pu and MA from the spent MOX are recycled in fast neutron spectrum ADS in an inert matrix as nitride fuel, reaching a burnup of 150 GWd/tHM. This fuel is reprocessed (Pyro-reprocessing) two years after discharge from the core. The recovered Pu and MA are reused in ADS fuel fabrication together with Pu and MA from the LWR reprocessing in the first stratum. The HLW from the LWR (UOX and MOX) and ADS fuels partitioning, which include the FP and

reprocessing losses are vitrified and sent to the repository after 50 years total cooling time. The solid wastes generated in this scenario are:
- HLW (vitrified wastes from UOX, MOX and ADS spent fuel reprocessing)
- long lived ILW (structural parts of the dismantled spent fuel assemblies UOX, MOX and ADS, and the ADS operational parts that have to be periodically replaced)

6.2 Waste Forms

Two types of HLW forms will be managed and disposed of in geological repositories of the scenarios: spent fuel assemblies (SFA) and Universal Canisters loaded with the vitrified HLW (UC-V) from the different reprocessing operations (Scenarios A2, A3, B1 and B2). Only one type of long lived ILW form has been considered in the project: Universal Canisters loaded with compacted long lived ILW (UC-C). This consists of non-fuel materials of the spent fuel assemblies and technological wastes.

6.2.1 HLW forms

a) UOX SFA (Scenario A1)

The main characteristics of the UOX fuel assemblies in the Open Cycle (Scenario A1) are summarized in Table 6.3. The residual thermal power of one UOX SFA after 50 years of cooling will be in the order of 400 W. It is has been assumed that four SFA could be loaded in each waste package.

Table 6.3: Characteristics of UOX fuel assembly (Scenario A1)

Lattice/reactor type	17 * 17	PWR
Physical dimensions:		
- Total length	4100	mm
- Section	217 * 217	mm^2
Mass of structure	145	kg
Mass of initial heavy metal	459.5	kg

b) LWR-MOX SFA (Scenario A2)

The main characteristics of the MOX fuel assemblies used in the Closed Cycle with mono-recycling of plutonium (Scenario A2) are also summarized in Table 6.. The only difference with the UOX, before its irradiation in the reactor, is the mass of initial heavy metal (453 kg). The residual thermal power of one LWR-MOX fuel assembly after 50 years of cooling will be in the order of 1550 W. It has been assumed that only one SFA is loaded per waste package.

c) UC-V with vitrified wastes from the reprocessing (Scenarios A2, A3, B1 and B2)

Part of the liquid HLW from the Standard Purex reprocessing contains most of the fission products and minor actinides and some small amounts of U and Pu (reprocessing losses). This liquid waste is calcined, mixed up with molten borosilicate glass, and poured into canisters. The main companies offering spent fuel reprocessing services in the world (BNFL and Areva) make use of a standard canister, named Universal Canister CSD-V, whose main characteristics are summarized in Table 6.4.

Table 6.4: Characteristics of the Universal Canister for vitrified HLW (UC-V)

Material	Stainless Steel (C: 0.15%; Cr: 24%; Ni: 13%)
Physical dimensions:	
- Length	1 338 mm
- External Diameter	430 mm
- Wall thickness	5 mm
Mass:	
- Total	492 Kg
- Empty	80 Kg
Volume:	
- External	175 l
- Internal	170 l
- Vitrified Waste	150 l

Figure 6.1

This method of conditioning of wastes, that is, the vitrification and the use of UC-V has been assumed within the project also for the Extended Purex and Pyro-reprocessing. A load of 40 kg of fission products plus actinides per UC-V has been assumed.

6.2.2 Long lived ILW Forms

In order to standardise handling operations the long lived ILW forms are similar to UC-V for vitrified wastes. The main characteristics of the UC for compacted wastes (UC-C), named Universal Canister CSD-C, are summarized in Table 6.5.

Table 6.5: Characteristics of the Universal Canister for compacted long lived ILW (UC-C)

Material	Stainless Steel
Physical dimensions:	
- Length	1 335.5 mm
- External Diameter	430 mm
- Wall thickness	5 mm
Mass:	
- Total	725 Kg (if only structural material is loaded) 520 Kg (average)
- Empty	92.5 Kg
Volume:	
- External	175 l
- Internal	Approx. 160 l

Actually (in the reprocessing of LWR UOX fuel), the volume of long lived ILWs generated per ton of uranium reprocessed is about 0.16 m^3, that is, 1 UC-C/tU. Every UC-C contains an average mixture of the structural materials and technological waste. Most of the radioactivity corresponds to structural materials, being that of the technological wastes negligible.

Although all the UC-C is similar in dimensions, different long lived ILW forms can be envisaged depending on their content:

a) UC-C for SFA structural parts from the reprocessing (Scenarios A2, A3, B1 and B2)

The reprocessing of the spent fuel begins with the chopping of the end-fittings of the fuel assemblies and the breaching of the cladding of the fuel. Next, the irradiated nuclear fuel is dissolved in nitric acid and the remaining hulls, grids, springs and other structural elements removed. These parts arrive after several rinsing operations to the packaging station where they are mixed and super-compacted (to about 65% of the theoretical metal density) in disks and put inside UC-C. All these materials are contaminated with part of the reprocessing losses and small amounts of minor actinides and fission products. Liquid and gaseous effluents, as well as other possible streams have not been accounted for in this description.

b) UC-C for Zr and Noble Metals from ADS fuel pyro-reprocessing (Scenario B2)

All the Zirconium, mostly from the Zr-nitride of the ADS fuel matrix (also some amounts of Zr isotopes are generated by fission) and all the Noble Metals, would be separated from the fission products stream in the pyro-reprocessing, and would be conditioned in UC-C.

c) UC-C for ADS consumable parts (Scenario B2)

Some parts of the ADS core, in particular, the spallation source (eutectic Pb-Bi), its support structure, the beam window and the final fraction of the accelerator tube will have to be periodically replaced. These components are strongly activated and it has been assumed that they would be conditioned also in UC-C in order to standardise handling operations.

6.3 Waste Packages for Final Disposal

HLW forms have to be stored in order to decay radiation and heat before final disposal. Spent fuel assemblies will first remain at the interim storage (a total cooling time of 50 years has been established). For the canisters with vitrified HLW from the reprocessing a decay time of 45 years has been established, given that the spent fuel was previously stored for 5 years at the reactor pools before reprocessing. Long lived ILWs generated in the reprocessing do not need to be cooled before final disposal, but according to current international regulations they have to be also disposed of in a deep underground repository because of their content (long-lived nuclides). The final disposal of both types of wastes (HLWs and long lived ILWs) is foreseen in the same repository, but in separated emplacement areas.

6.3.1 Waste Packages Concepts

Three different waste packages for HLWs and one for long lived ILWs have been proposed for final disposal of the waste forms generated in the different scenarios. The main characteristics of the waste packages are summarized in the Table 6.6 and detailed below.

Table 6.6: Characteristics of the waste packages for final disposal

Waste category		HLW				LONG LIVED ILW
Scenarios		**A1**	**A2**		**A3/B1/B2**	**A2/A3/B1/B2**
Contents (waste forms)		4 SFA UOX	1 SFA MOX	1 UC-V	1 UC-V	4 UC-C
WP Shape		Cylinder	Cylinder	Cylinder	Cylinder	Cuboid
Wall Material		Carbon Steel	Carbon Steel	Carbon Steel	Carbon Steel	Concrete
Height (m)		4.54	4.54	1.6	1.6	2
Diameter or side (m)		0.90	0.65	0.65	0.65	1.50
Wall thickness (m)		0.10	0.10	0.10	0.10	About 0.15
WP External volume (m^3)		2.89	1.51	0.53	0.53	4.50
Approx. mass (t)	Waste form	2.7	0.7	0.4	0.4	2 to 3.5
	Container	15.5	10	2.5	2.5	10
	Total	18	11	3	3	12 to 13.5

<u>*HLW Packages for final disposal*</u>

The proposed packages for HLW contain waste forms such as spent fuel assemblies from LWR or Universal Canisters with vitrified HLW from the reprocessing processes:

a) Waste Package with 4 UOX SFA (Scenario A1)

The waste package (see Figure 6.2) consists of a carbon steel cylinder 4.54 m long with an external diameter of 0.9 m. The wall thickness is 0.1 m and the two lids are 0.12 m thick. There is inside a cast iron insert provided with four steel channels where the assemblies are placed. The mass of the package is 18170 kg, (9 650 kg correspond to the external canister, 5 860 kg to the insert and 2 670 kg to the four SFA).

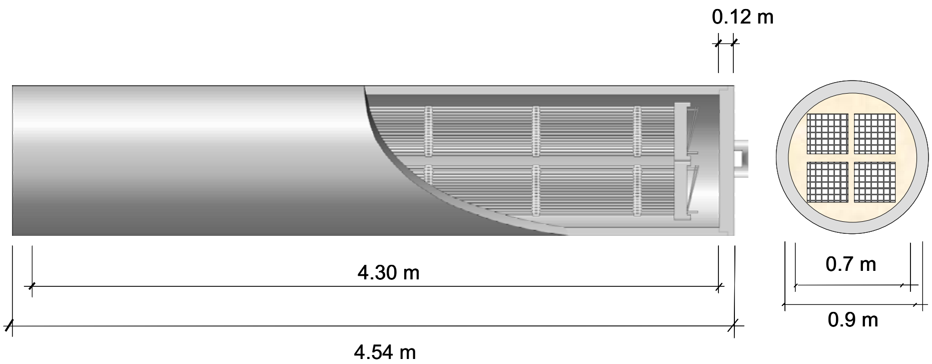

Figure 6.2: Scenario A1: HLW package for 4 UOX spent fuel assemblies

b) Waste Package with 1 LWR-MOX SFA (Scenario A2)

The waste package (see Figure 6.3) consists of a carbon steel cylinder 4.54 m long and 0.65 m external diameter. The wall thickness is 0.1 m and the two lids are 0.12 m thick. There is inside a cast iron insert provided with one steel channel where the assembly is placed. The mass of the package is 10 680 kg, (6 430 kg correspond to the external canister, 3 590 kg to the insert and 660 kg to the LWR-MOX SFA).

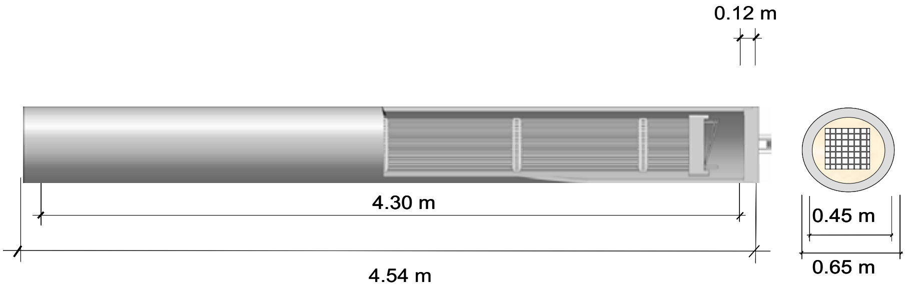

Figure 6.3: Scenario A2: HLW package for 1 MOX spent fuel assembly

c) Waste Package with 1 UC-V with vitrified wastes (Scenarios A2, A3, B1 and B2)

The package for vitrified HLW (see Figure 6.4) consists of a carbon steel cylinder 1.6 m long and 0.65 m external diameter containing one single UC-V. The wall thickness is 0.1 m and the two lids are 0.12 m thick. The mass package is 2 950 kg (2 460 kg correspond to the external container and 490 kg to the UC-V).

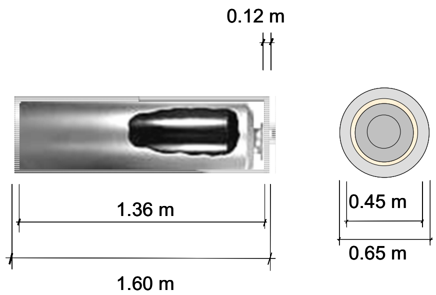

Figure 6.4: Scenario A2: HLW package for 1 Universal Canister with vitrified wastes

<u>Long lived ILW Packages for final disposal</u>

The long lived ILW package is a concrete box 2 m height and a square section of 1.5 m side provided with 4 holes and a cover lid. It contains four UC-C (see Figure 6.5). The mass of the waste package will be about 12 t from which 10 t corresponds to the external container and 2 t to the 4 UC-C with super-compacted wastes.

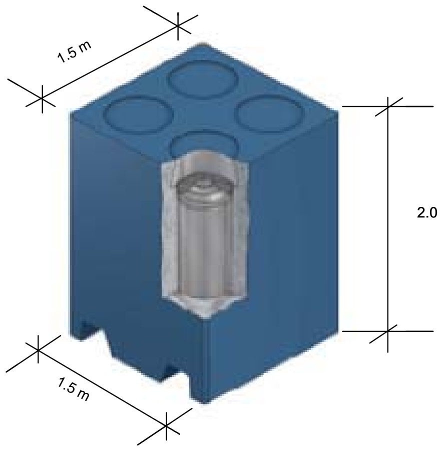

Figure 6.5: Scenario A2: Long lived ILW package for 4 UC with compacted wastes

6.3.2 Waste Packages Production and Volumes

Table 6.7 summarizes the production of HLW packages in each scenario.

The total volume of HLWs in the scenarios A3, B1 and B2 are very similar, about one third of scenario A1. The high thermal power of LWR-MOX spent fuel assemblies in scenario A2 limits the volume reduction in this scenario, because each waste package for SFA could load only one MOX fuel assembly.

Table 6.8 summarizes the long lived ILW packages production.

Table 6.7: HLW forms and packages production per scenario

Scenario	A1	A2		A3	B1	B2		
Fuel Type	LWR-UOX	LWR-MOX	LWR-UOX (reproc.)	FR-MOX (reproc.)	FR-MOX (reproc.)	LWR-UOX (reproc.)	LWR-MOX (reproc.)	ADS TRU-Nitride (reproc.)
Waste Forms per tHM	2.18	2.21	1.12	2.09	1.98	1.08	1.07	2.61
tHM / TWh(e)	2.46	0.25	2.21	1.15	1.15	1.91	0.24	0.12
Waste Forms per TWh(e)	5.35 SFA	0.54 SFA	2.48 UC-V	2.40 UC-V	2.27 UC-V	2.07 UC-V	0.26 UC-V	0.33 UC-V
Waste Forms per HLW Package	4	1	1	1	1	1	1	1
Waste Packages per TWh(e)	1.34	0.54	2.48	2.40	2.27	2.07	0.26	0.33
Package Volume (m^3)	2.89	1.51	0.53	0.53	0.53	0.53	0.53	0.53
HLWs Volume [m^3/TWh(e)]	3.87	0.82	1.32	1.27	1.21	1.10	0.14	0.17
HLWs TOTAL Volume [m^3/TWh(e)]	**3.87**	**2.14**		**1.27**	**1.21**	**1.41**		

The volume of long lived ILWs in Scenarios A3 and B1 is much greater than Scenarios A2 and B2. It is due to the important mass of structural parts in the fuel assembly of the MOX-FR

(3 t/tHM in the core assemblies and 2.3 t/tU in the blanket assemblies) compared with LWR (316 kg/tU). Most of the long lived ILW volume in Scenario B2 comes from the reprocessing of UOX fuels, although the number of long lived ILW generated in the reprocessing of ADS fuel is relevant due to great amount of Zr (mostly from the Zr-nitride of the fuel matrix) and the Noble Metals, which are separately obtained from the fission products stream in the pyro-reprocessing.

Table 6.8: long lived ILW forms and packages production per scenario

Scenario	A1	A2		A3	B1	B2			
Fuel Type	LWR - UOX	LWR-MOX	LWR-UOX (reproc.)	FR-MOX (reproc.)	FR-MOX (reproc.)	LWR-UOX (reproc.)	LWR-MOX (reproc.)	ADS-TRU Nitride (reproc.)	ADS (operation)
Waste Forms per tHM	-	-	1.00	4.10	4.10	1.00	1.01	6.22	0.27
tHM / TWeh	2.46	0.25	2.21	1.15	1.15	1.91	0.24	0.12	0.12
Waste Forms per TWe/h	-	-	2.21	4.71	4.71	1.91	0.24	0.78	0.03
Waste Forms per long lived ILW Package	-	-	4	4	4	4	4	4	4
Waste Packages per TWeh	-	-	0.55	1.18	1.18	0.48	0.06	0.19	0.01
Package Volume (m^3)	-	-	4.5	4.5	4.5	4.5	4.5	4.5	4.5
Long lived ILWs Volume (m^3/TWeh)	-	-	2.49	5.31	5.31	2.15	0.27	0.87	0.04
Long lived ILWs TOTAL Volume (m^3/TWeh)	**-**	**2.49**		**5.31**	**5.31**	**3.33**			

6.3.3 Mass Inventory of Relevant Radionuclides

Table 6.9 summarizes the amount of relevant radioactive fission and activation products to be disposed of in the geological repository. Next, a short discussion about the differences on the mass of these nuclides present in the solid waste of the different scenarios is given.

Fission products: In scenario A1 all the fission products, including the volatile, go to the repository. In the other scenarios part of the volatile fission and activation products are released to the environment as the fuel is reprocessed [2.1]. Scenarios A3 and B1 take advantage of the high electrical conversion efficiency of the FR (40%), because less fission reactions are required to produce the same electrical power in comparison with LWR (33%). In Scenario B2 it has been assumed (see Table 6.2) that all the Zirconium in the spent fuel (the irradiated Zr of the fuel matrix and the Zr isotopes produced by fission), together with the noble metals separated in the pyro-reprocessing are conditioned inside long lived ILW packages.

C-14: It is produced in Scenarios A3 and B1 due to the activation of nitrogen impurities in the SS parts of the fuel assemblies and fresh fuel of the FR. The big amount produced in the ADS (Scenario B2) is due to the nitride fuel used as fuel matrix. In order to minimize the production of C-14 in the ADS it has been assumed an enrichment of the N of 97% in N-15 isotope. Only a 10% of the C-14 present in the fuel goes to the vitrified HLW in the reprocessing (see Table 6.1). The remaining 90% is released into the environment during reprocessing.

Table 6.9: Mass of relevant fission and activation products in the solid wastes (HLWs + long lived ILWs)

Nuclides	Mass of main Fission and Activation Products in Solid Wastes [g/TWh(e)]									
	A1		A2		A3		B1		B2	
	HLW	LONG LIVED ILW	HLW	LONG LIVED ILW	HLW	LONG LIVED ILW	HLW	LONG LIVED ILW	HLW	LONG LIVED ILW
Fission Prod.	125810	-	106110	195	89723	189	89936	189	103868	191622(*)
C-14	0.3	-	0.03	0.01	0.2	2.3	0.2	2.3	3.1	0.1
Cl-36	6.0	-	0.3	0.5	0.002	0.0004	0.002	0.0004	0.04	0.6
Se-79	17	-	17	0.03	16	0.03	16	0.03	17	0.1
Sr-90	571	-	542	1.0	249	0.5	250	0.5	513	1.2
Nb-94	18	-	2.5	15	0	79	0	78.6	0.7	21
Tc-99	2931	-	2920	6.5	2298	209	2317	208	2969	10.9
Pd-107	837	-	937	1.5	1207	2.4	1217	2.4	828	296
Sn-126	80	-	85	0.1	108	0.2	109	0.2	93	0.3
I-129	634	-	82	5.7	6.7	6.6	6.7	6.7	6.9	9.3
Cs-135	1633	-	1750	2.9	4270	8.5	4270	8.5	2336	6.5
Cs-137	1405	-	1405	2.5	1137	2.3	1133	2.3	1455	3.8
(*) Zr of the ADS fuel matrix included.										

Cl-36: It is produced by activation of the chlorine fuel impurities. It has been assumed in Table 6.1 that only 1% of the Cl in the fuel matrix goes to the HLW stream in the reprocessing; the remaining 99% is released into the environment. As a consequence of this initial release, the amount of Cl-36 in scenarios A2, A3, B1 and B2 is lower than in scenario A1.

Sr-90: This radionuclide, together with Cs-137, produces most of the heat generated in the HLWs, which could influence the repository design. The production of Sr-90 in Scenarios A3 and B1 is lower than in the other scenarios by two reasons: (1) the higher electrical conversion efficiency of FR (40%) compared with LWR (33%), and (2) the fission of plutonium (mainly Pu-239) that yields less Sr-90 than the fission of U-235 in the UOX.

Nb-94: It is an activation product of natural niobium present in the structural parts of the fuel assemblies, mainly in the stainless steels of the non-fuel hardware. The fuel assemblies of Scenarios A3 and B1 have a much greater mass of structural parts per ton of heavy metal; this is the explanation for differences in the amount of Nb-94.

I-129: This very long-lived fission product is the main contributor to the radiological impact of the repository due to its high solubility, mobility and radiotoxicity. In the case of scenario A1 all the I-129 generated remains, obviously, in the nuclear fuel and thus, it is disposed of in the repository. In the other scenarios where spent fuels are reprocessed, 98% is released to the biosphere and only a small fraction (2%) goes to the repository (1% in the HLWs and 1% in the long lived ILWs, see Table 6.1 and Table 6.2, respectively).

Cs-135: This very long-lived fission product is an important contributor to the very long term radiological impact of the repository. The amount of Cs-135 in Scenarios A3 and B1 is due to the fission yield of Pu isotopes, which is higher than that of U-235.

Table 6.10 summarizes the amount of the relevant actinides to be disposed of in the repository. The mass of recovered uranium in the reprocessing (RepU) is also provided.

Table 6.10: Mass of relevant actinides in the solid wastes (HLWs+long lived ILWs) and RepU

Actinides	Mass of Actinides in solid Wastes [g/TWh(e)]									
	A1		A2		A3		B1		B2	
	HLW	LONG LIVED ILW	HLW	LONG LIVED ILW	HLW	LONG LIVED ILW	HLW	LONG LIVED ILW	HLW	LONG LIVED ILW
U	2302048	-	218730	414	880	176	865	173	2004	535
Np	2021	-	1890	0.4	547	0.2	2.5	0.5	6.0	1.3
Pu	25430	-	13688	4.6	485	31	162	32	107	30
Am	4417	-	4527	0.8	4102	1.8	13	2.6	25	6.9
Cm	52	-	149	0.01	107	0.02	1.3	0.3	4.9	0.9
TRU	31920	-	20254	5.7	5241	33	179	36	142	39
TOTAL An	2333968	-	238984	420	6122	209	1044	209	2146	573

	Mass of Reprocessed Uranium [g/TWh(e)]				
	A1	**A2**	**A3**	**B1**	**B2**
RepU	-	2 069 080	874 243	861 351	1 996 130

The differences among the amounts of the actinides that are sent to the repository in the scenarios can be explained through the following reasons:

- The classification of the spent fuel as a waste and its transfer to the repository without reprocessing (UOX in A1 and MOX in A2) results in a big amount of uranium (also plutonium in lesser quantity) sent to the repository. The Uranium recovered in the reprocessing of the spent fuel has not been fully included as waste in this study, due to its potential as an energetic resource. But, since no use of it has been foreseen on the setup of these equilibrium scenarios, all this Uranium would necessarily be disposed of in a repository, due to its contents in long-lived isotopes. This would increase the number of long lived ILW waste packages in Scenarios A3, B1 and B2. If this Uranium were used, a new cycle analysis should be carried out. The assumptions of advanced reprocessing of spent fuel and further burning of minor actinides in Scenarios B1 and B2 lead to a reduction of the quantities of Np, Am and Cm sent to the repository.
- It is important to recall that the advanced cycle scenarios described in this section are so called equilibrium scenarios, which represent an approximation to what could be a long term potential situation. In order to have Pu available for the MOX fuel of fast reactors, a previous park of LWR's is needed.

6.3.4 Radioactive Inventory

Figure 6.6 shows the time evolution of the activity of HLWs and long lived ILWs per unit of produced electrical energy (Bq/TWh(e)) for each scenario.

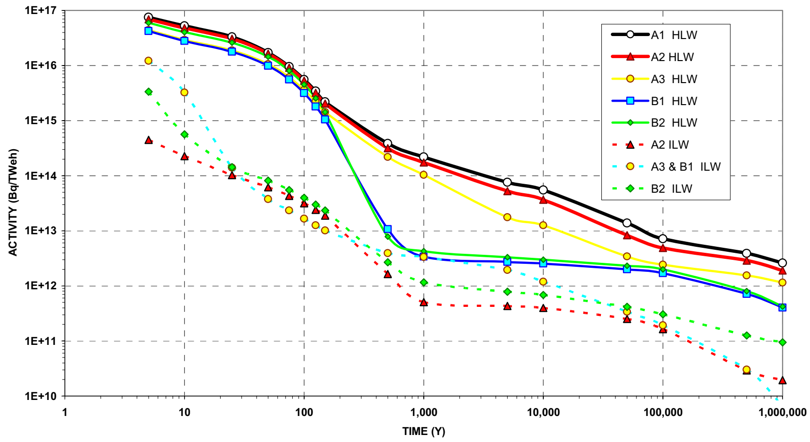

Figure 6.6: HLWs and long lived ILWs activity evolution with time

After about 150 years cooling time the total activity in Scenarios B1 and B2 decreases rapidly as the activity of the short-lived fission products Cs-137 and Sr-90, which are the most radioactive nuclides in the wastes, decreases. On the contrary, in Scenarios A1, A2 and A3, where the contribution of actinides to the total activity is more relevant, the decrease is more gradual. The highest differences in the total activity are found in the gap from 300 to 100 000 years cooling. In this period of time, the difference between conventional scenarios (A1, A2 and A3) and advanced scenarios (B1 and B2) is mainly due to the inventory of some nuclides as Pu-239, Pu-240, Am-241 and Am-243. After about 100 000 years the differences among the curves of all the scenarios are less significant.

In general, the activity of long lived ILWs is about two orders of magnitude below the activity of HLWs for each scenario. There is an exception in Scenarios B1 and B2, where the activity of long lived ILWs and HLWs during the period 300 yr to 10 000 yr are quite similar.

6.3.5 Radiotoxic Inventory

Figure 6.7 represents the time evolution of the radiotoxic inventory per TWh(e) due to HLWs and long lived ILWs. Also the radiotoxic inventory of the recovered irradiated uranium in the reprocessing is shown. The radiotoxic inventory of long lived ILWs is well below those of HLWs for the same scenario; only in Scenarios B1 and B2 the radiotoxic inventories of long lived ILWs and HLWs during the period from 300 to 1 million years are similar. In fact, they are approximately a factor five smaller in this period as they are dominated by the actinides and the hypothesis of reprocessing is that 0.1% of all actinides remain in the HLW and 0.02% is incorporated in the long lived ILW.

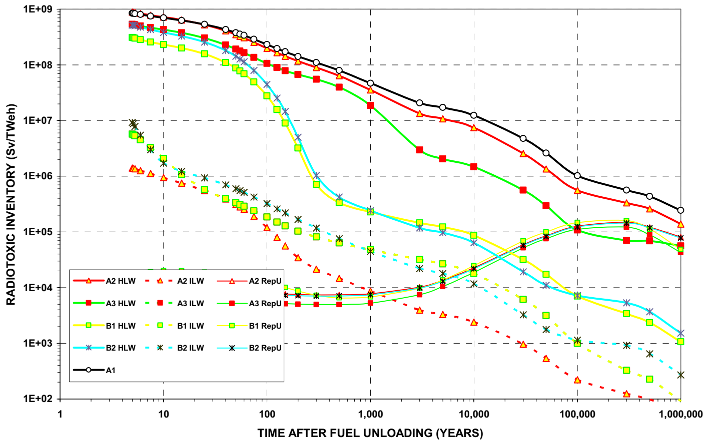

Figure 6.7: Radiotoxic inventory of wastes and reprocessed uranium

After about 100 years cooling time two different types of scenarios can be distinguished: transmutation scenarios (B1 and B2) and other scenarios (A1, A2 and A3). At the waste emplacement time (50 years) the radiotoxic inventory in Scenario A1 and A2 is dominated by the actinides (^{241}Am and ^{238}Pu), whereas in Scenario B1 and B2 the main contributors are the fission products (^{90}Sr and ^{137}Cs). In Scenario A3 both contributions are similar. For all the scenarios the medium and long-term radiotoxic inventory is sequentially dominated by ^{241}Am, ^{240}Pu and ^{239}Pu, although at a lower level in Scenarios B1 and B2.

With the exception of Scenario A2, where the total radiotoxic inventory of the whole long lived ILWs at the waste emplacement time is dominated by the fission products, in all the other scenarios the actinides (^{241}Am, ^{243}Am and ^{239}Pu) are the main contributors. The radiotoxic inventory of the long lived ILW generated in the operation of the ADS is quite different to other long lived ILWs. It is initially dominated by ^{210}Po which is generated in the eutectic Pb-Bi spallation target. Next, the decay chain ^{90}Sr/^{90}Y and ^{193}Pt are responsible for the radiotoxic inventory in the medium-term and ^{210m}Bi for the long-term (all of these radionuclides are formed by activation/spallation processes in the spallation target). ^{54}Mn and ^{55}Fe are the radionuclides responsible for the radiotoxic inventory of activated steel parts as the beam-window, beam-pipes and the target support structures.

It should be pointed out that the comparison in terms of radiotoxic inventory of wastes is biased against scenarios that include great amounts of uranium as waste. This is the case of Scenario A1 (and Scenario A2 although to a lesser extent) where the mass of the uranium is about 95% of the total mass of actinides plus fission products. The uranium isotope U-234 ($T½ = 2.5 \cdot 10^5$ years) and its daughter Th-230 ($T½ = 7.5 \cdot 10^4$ years) are the responsible nuclides for the medium to long term radiotoxicity. In order to properly evaluate final radiotoxic inventories, and the potential efficiency of these technologies on the reduction of the long term radiotoxic inventory, the recovered uranium in the reprocessing, should be accounted for and summed up.

6.3.6 Thermal Power

The HLW short-term thermal power is dominated mainly by the fission products ^{90}Sr and ^{137}Cs (and their respective daughters in secular equilibrium). The contribution of actinides (mainly ^{244}Cm and ^{241}Am) is only relevant in Scenarios A1 and A2.

The evolution of the HLW thermal power generated per TWeh in each scenario is shown in Figure 6.8. This figure shows a big difference between the Scenarios A1 and A2 and the rest of scenarios, in which the total thermal power of the HLW's is much lower. The reasons that explain the differences among scenarios in the total thermal power are:

- The higher electrical conversion efficiency of the fast reactors, with the corresponding smaller generation of fission products for a given electrical energy generation. LWR's have a 33% efficiency (Scenarios A1, A2 and B2) against the assumed 40% of FR's (Scenarios A3 and B1).
- The smaller production of Sr-90 in the fissions of plutonium (reference fuel for the FR of Scenarios A3 and B1) and heavy minor actinides as compared with the fission of U-235 (LWR of Scenarios A1, A2 and B2).
- The recovery of actinides (some of them are relevant heat producers). Scenarios where all the spent fuel is reprocessed by means of advanced reprocessing (Scenario B1 and B2) contain only small amounts of actinides in the HLW.

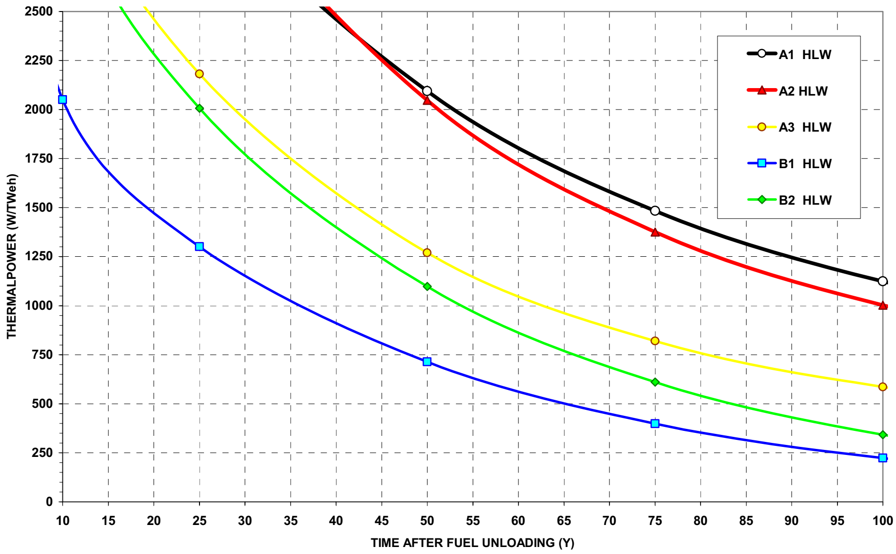

Figure 6.8: HLWs total thermal power evolution with time

Figure 6.8 can provide an idea about the relative dimensions of the HLWs repository emplacement area that could be achieved after optimization of the repository and waste package designs (including the waste mass per package), if only thermal criteria were considered. Another variable that could be optimized is the cooling time of the HLWs before disposal: the thermal power after 50 years in Scenarios A1 and A2 is similar to Scenarios A3 and B2 after 25 years, or even 10 years in the case of Scenario B1.

In general, at the HLW emplacement time (50 years of cooling), fission products ^{90}Sr and ^{137}Cs are the main contributors to the waste thermal power. Exceptions are the MOX SFA (Scenario A2) and the vitrified wastes (Scenario A3), where the thermal power of actinides (mainly ^{241}Am) is greater than the thermal power of fission products. Due to the small amount of actinides in the vitrified wastes resulting from advanced reprocessing (Scenarios B1 and B2); its contribution to the thermal power is low. Table 6.3 shows for each scenario the HLW packages thermal output at the emplacement time.

Table 6.3: HLW packages thermal power

Scenario	A1	A2		A3	B1	B2		
Fuel Type	LWR-UOX	LWR-MOX	LWR-UOX (reproc.)	FR-MOX (reproc.)	FR-MOX (reproc.)	LWR-UOX (reproc.)	LWR-MOX (reproc.)	ADS-TRU-Nitride (reproc.)
Waste Forms per HLW Package	4 SFA	1 SFA	1 UC-V	1 UC-V	1 UC-V	1 UC-V	1 UC-V	1 UC-V
Waste Packages per TWeh	1.34	0.54	2.48	2.40	2.27	2.07	0.26	0.33
Thermal Power (W) after 50 years (emplacement time)	1 564	1 549	486	529	314	420	331	438

The evolution of the thermal power of the different HLW packages is shown in Figure 6.9.

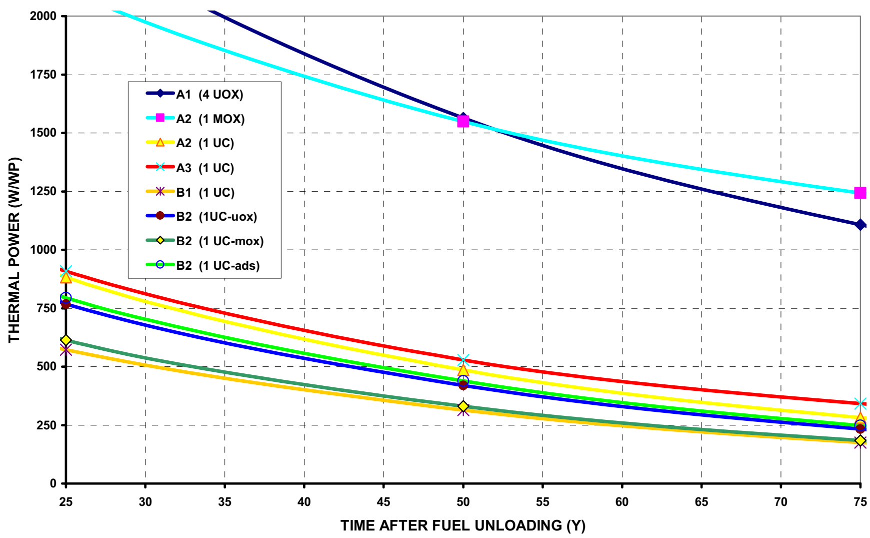

Figure 6.9: HLW packages thermal power evolution with time

6.3.7 Gamma and Neutron Emission

Table 6.4 provides the total gamma and neutron emissions per unit of produced electrical energy for the whole HLWs from each scenario. It shows that there are not important differences in the gamma emission, as it is mainly produced by fission products. On the contrary, the neutron emissions are very different: up two orders of magnitude lower can be found when comparing Scenario B1 to the rest of scenarios.

Table 6.4: HLWs Total Gamma and Neutron Emissions per scenario

		HLWs Radioactive Emissions per TWeh (1/s)				
Time	**Radiation**	**A1**	**A2**	**A3**	**B1**	**B2 (*)**
Initial cooling time (5 years)	Gamma	3.8E+16	3.8E+16	2.2E+16	2.2E+16	4.5E+16
	Neutrons	2.2E+09	6.1E+09	6.6E+09	7.0E+07	1.1E+09
Emplacement time (50 years)	Gamma	6.9E+15	6.7E+15	4.7E+15	4.5E+15	6.6E+15
	Neutrons	4.4E+08	1.3E+09	1.6E+09	1.7E+07	1.0E+08

(*) Initial cooling time is 2 y in the case of ADS spent fuels

Neutrons in the spent fuels from Scenarios A1 (UOX) and A2 (MOX) are produced by means of two major reactions: spontaneous fissions and (α, n) reactions on oxygen nuclei of the fuel matrix; in the vitrified waste they are due also to spontaneous fissions and (α, n) reactions on nuclei of the glass matrix light elements Li, B, O, Na, Al and Si. In general, in all the scenarios ^{244}Cm is the nuclide responsible for most of the neutron emissions. This is the reason why the HLW neutron emission has low significance in scenarios where the MAs are transmuted (Scenarios B1 and B2).

Long lived ILWs are much less radioactive than HLWs in the first years after discharge. Table 6.5 provides the total gamma and neutron emissions per unit of produced electrical energy from the whole long lived ILWs in each scenario.

Table 6.5: Long lived ILWs Total Gamma and Neutron Emissions per scenario

		longlived ILWs Radioactive Emissions per TWeh (1/s)				
Time	**Radiation**	**A1**	**A2**	**A3**	**B1**	**B2 (*)**
Initial cooling time (5 years)	Gamma	0	2.9E+14	9.8E+14	9.8E+14	1.2E+16
	Neutrons	0	4.0E+05	8.6E+05	6.3E+06	1.9E+07
Emplacement time (50 years)	Gamma	0	8.4E+12	1.3E+13	1.3E+13	1.5E+13
	Neutrons	0	7.5E+04	1.7E+05	1.4E+06	5.0E+06

(*) Initial cooling time is 2 y in the case of ADS spent fuels

Differences between the gamma emission in Scenario A2 and scenarios where FR fuel is reprocessed (Scenarios A3 and B1) can be explained because the much greater mass of non-fuel hardware in the FR fuel assemblies compared with LWR fuel assemblies. The pyro-reprocessing of ADS spent fuels with short cooling time (2y) is responsible for the high total emissions of gamma and neutrons in Scenario B2; the higher neutron emission at the waste emplacement time is due to heavy minor actinides, in particular Cm-244, Cm-246 and Cf-250.

6.3.8 Waste Package Radiation Levels

Dose rates of the HLW packages of the different scenarios are shown in Table 6.6. Radiation levels in contact with the packages are averaged over the cylindrical surface. Waste packages for SFA (Scenarios A1 and A2) are provided with a cast iron insert, which acts as an effective shielding.

Table 6.6: Radiation Levels of HLW packages per scenario after 50 y cooling time

Dose Rate (mSv/h)		Scenario and HLW package content							
		A1	A2		A3	B1	B2		
Distance	Radiation type	4 SFA LWR-UOX	1 SFA LWR-MOX	1 UC-V LWR-UOX (reproc.)	1 UC-V FR-MOX (reproc.)	1 UC-V FR-MOX (reproc.)	1 UC-V LWR-UOX (reproc.)	1 UC-V LWR-MOX (reproc.)	1 UC-V ADS-TRU-Nitride (reproc.)
Contact	Gamma	12	2.5	1 400	1 300	1 400	1 500	1 500	1 900
	Neutron	2.2	13	7.4	19	0.21	0.02	0.19	7.9
1 meter	Gamma	2.7	0.47	180	160	170	180	190	240
	Neutron	0.42	2.0	0.72	1.8	0.020	0.002	0.018	0.77

The results show the need for heavy shielding when handling the HLW packages loaded with vitrified HLWs. Scenario A1 and scenarios where no MAs are partitioned (A2 and A3) show high neutron dose rates. Scenarios where advanced reprocessing is applied show in general lower neutron dose rates.

Dose rates of the long lived ILW packages are shown in Table 6.7. Radiation levels in contact with the package are averaged over the surface. The table shows the need for additional gamma shielding when handling the long-lived ILW packages.

Table 6.7: Gamma Radiation Levels of long-lived ILW packages per scenario after 50 y cooling time

Dose Rate (mSv/h)		Scenario and Origin of long lived ILWs								
		A1	A2		A3		B1		B2	
Distance	Radiation type	LWR UOX	LWR MOX	4 UC-C (LWR-UOX reproc.)	4 UC-C (FR-SFA reproc.)	4 UC-C (FR-SFA reproc.)	4 UC-C (LWR-UOX reproc.)	4 UC-C (LWR-MOX reproc.)	4 UC-C (ADS-TRU-Nitride reproc.)	4 UC-C (ADS operation)
Contact	Gamma	-	-	59	26	25	59	63	23	13
1 meter	Gamma	-	-	13	5.5	5.6	13	12	4.8	1.8

6.4 Transition Scenarios

The Transition Scenarios consist of a step by step approach, evolving in the case of Transition Scenarios A3 and B1, from a pure UOX LWR nuclear park to a pure MOX FR, including in the meantime the burning of MOX fuel in LWR during some decades. In the case of Scenario B2 the initial park starts from an equilibrium situation of a mix of park of UOX LWR and MOX LWR evolving to a new equilibrium park that includes the operation of ADS. The power production in the nuclear park of the transition scenarios is constant and fixed to 800 TWeh per year. Year 2010 is the starting date for the study and year 2200 is the final date of these transition scenarios. The HLWs are vitrified and conditioned in Universal Canisters (UC) loading 40 kg of fission products plus actinides. Each waste package contains one UC.

6.4.1 Transition Scenario A3 (TSA3)

Figure 6.10 shows the process diagram for the Transition Scenario A3.

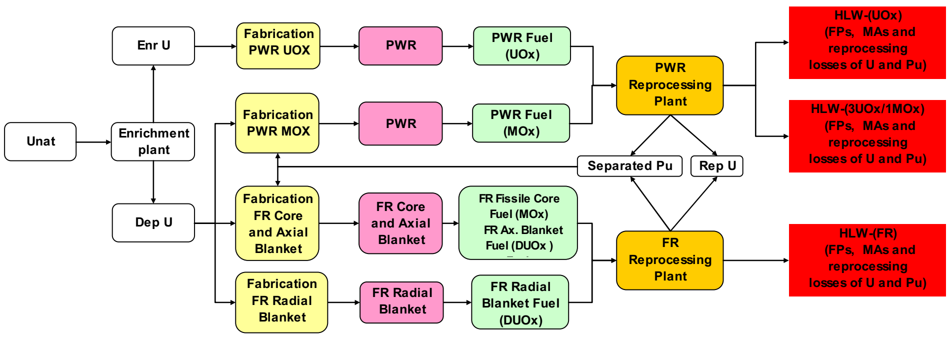

Figure 6.10: Process diagram for the Transition Scenario A3

In Transition Scenario A3 it is assumed that 3 types of HLW packages are produced:

1. Those generated in the reprocessing of LWR-UOX spent fuel.

2. Those generated in the combined reprocessing of LWR-UOX and LWR-MOX spent fuels, with a ratio of 3 UOX / 1 MOX assemblies.

3. Those generated in the reprocessing of spent fuel from the FR (fissile core + blankets jointly).

6.4.2 Transition Scenario B1 (TSB1)

Transition Scenario B1 is similar to TSA3. The differences are the recycling of minor actinides after year 2040 and the use of the Extended Purex reprocessing (Figure 6.111).

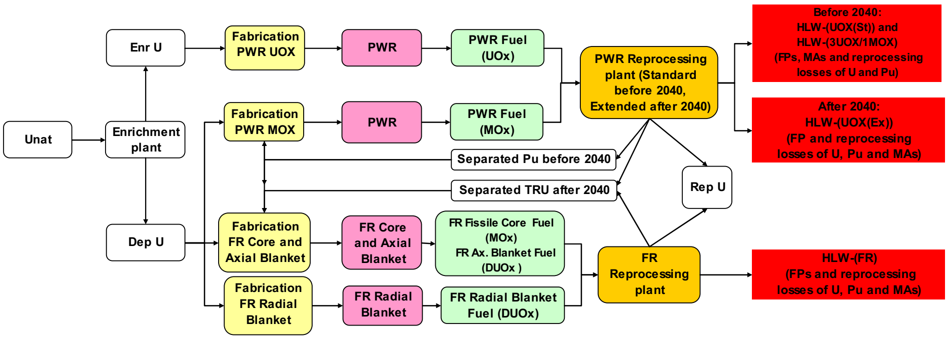

Figure 6.11: Process diagram for the Transition Scenario B1

In Transition Scenario B1 it is assumed that 4 types HLW packages are produced:

1. Those generated before 2040 in the reprocessing (Standard Purex) of LWR-UOX spent fuel.

2. Those generated before 2040 in the combined reprocessing (Standard Purex) of LWR-UOX and LWR-MOX spent fuels, with a ratio of 3 UOX / 1 MOX assemblies.

3. Those generated after 2040 in the reprocessing (Extended Purex) of LWR-UOX spent fuel. There will also be some few MOX-LWR SFAs that would be reprocessed jointly with UOX fuels.

4. Those generated after 2040 in the reprocessing (Extended Purex) of spent fuel from the FR (fissile core + blankets jointly).

6.4.3 Transition Scenario B2 (TSB2)

Figure 6.12 shows the process diagram for the Transition Scenario B2.

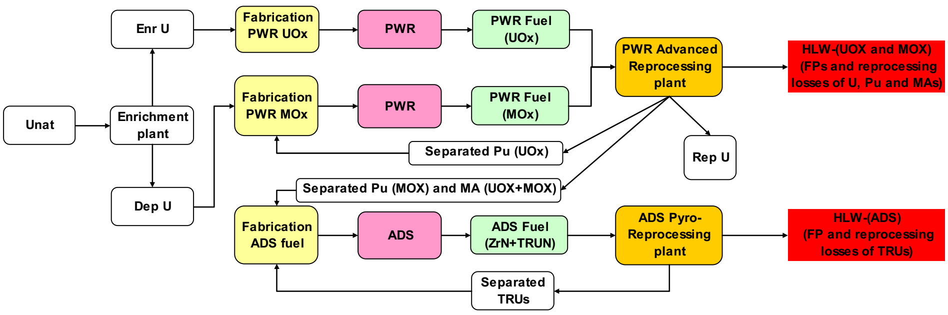

Figure 6.12: Process diagram for the Transition Scenario B2

In Transition Scenario B2 it is assumed that 2 types HLW packages are produced:

1. Those generated in the Extended Purex reprocessing of LWR-UOX and LWR-MOX spent fuels.

2. Those generated in the Pyro-reprocessing of spent fuel from the ADS. Note that the noble metals (Ru, Rh, Pd and Ag) and 99% of the Zr will be separately conditioned as long lived ILW.

6.4.4 HLW Packages Production

One type of HLW package has been assumed for final disposal of the waste forms (universal canisters) generated in all transition scenarios, each one containing vitrified HLW from the diverse reprocessing processes: Standard Purex, Extended Purex and Pyro-reprocessing. The annual production of waste packages (WPs) which are generated in the different transition scenarios is shown in Figure 6.13.

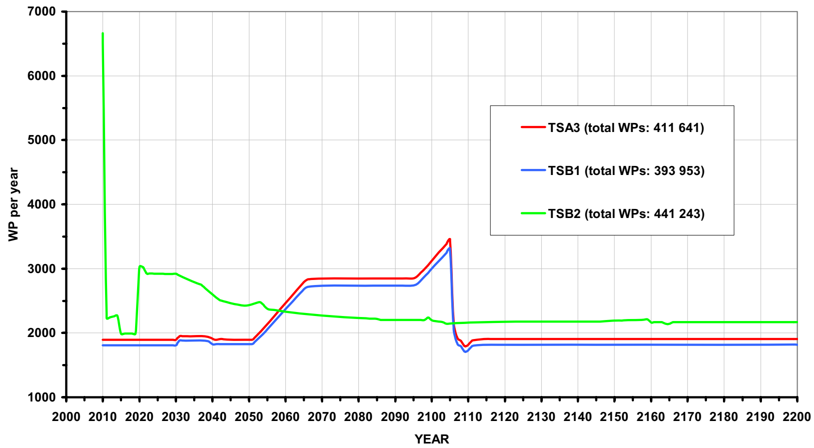

Figure 6.13: Annual production of HLW packages in Transition Scenarios

The initial peak in the production of waste packages in TSB2 is the consequence of the great mass of UOX LWR fuel that must be reprocessed in order to quickly feed the MOX LWR. The second peak is the consequence of the great mass of UOX and MOX LWR fuels that have to be reprocessed in order to fabricate fuel for the ADS operation from 2040. In TSA3 and TSB1 the reprocessing capacity for LWR fuel was fixed to 1 700 tHM/y, which is a little bit lower than the annual LWR spent fuel flux. It implies that LWR and FR spent fuel reprocessing plants will work simultaneously for some period, which explains the plateau and peak in the generation of HLW packages in these scenarios.

The content, production and volume of HLW packages from each transition scenario are summarized in Table 6.8. It should be noted that long lived ILWs have not been accounted for in the transition scenarios analysis.

Table 6.8: HLW package content and production in the transition scenarios

Transition Scenario		TSA3			TSB1				TSB2	
Reprocessed Fuel		UOX (LWR)	Mix of 3UOX MOX (LWR)	MOX (FR)	UOX (LWR)	Mix of 3UOX 1MOX (LWR)	UOX (LWR)	MOX (FR)	Mix of UOX MOX (LWR)	TRU+Zr Nitride (ADS)
Reprocessing Method		St. Purex	St. Purex	St. Purex	St. Purex	St. Purex	Ext. Purex	Ext. Purex	Ext. Purex	Pyro
Mass per Waste Package (kg)	FP	37.1	35.5	37.7	37.3	35.7	39.1	39.5	39.1	39.6
	U	0.84	0.81	0.37	0.89	0.84	0.87	0.38	0.84	0.006
	Pu	0.01	0.02	0.06	0.01	0.02	0.01	0.07	0.015	0.24
	MA	2.07	3.64	1.88	1.79	3.49	0.003	0.004	0.002	0.08
WP Production		164 910	18 540	228 190	38 306	16 584	121 503	217 560	397 290	43 953
WP Total Production		411 641			393 953				441 243	
Volume of WP (m^3)		~220 000			~210 000				~235 000	

As shown in the table, all the transition scenarios have similar total number of HLW packages. Small differences between TSA3 and TSB1 are due to the recycling of minor actinides after 2040 in TSB1. Differences with TSWB2 are due to the higher electrical conversion efficiency of the FR in TSA3 and TSB1 (40%) than LWR and ADS (33%) in TSB2. In TSB2 wastes with the partitioned noble metals in the pyroprocess are not included as they could be classified as long lived ILW.

Table 6.9 shows the total mass of fission products and actinides contained in the HLW generated in each transition scenario. The hypothesis of an advanced reprocessing technology for the separation and further recycling of minor actinides in FR after 2040 is the reason for the reduction of the mass of MA in TSB1 in comparison with TSA3. This is more noticeable in TSB2, in which the totality of MA being generated is assumed to be separated since year 2010 and recycled in the ADS after year 2040. In this last transition scenario, the assumption of the availability of advanced technologies of partitioning (Extended Purex) just at the very beginning of the transition scenario start-up, explains the lower amount of MA present in the HLWs, which represents only the losses of the corresponding partitioning process. On the other hand, the mass of FP and the losses of U in TSB2 are higher than in TSA3 and TSB1, while the losses of Pu are similar in the three scenarios.

Table 6.9: Mass of fission products and losses to HLWs in the transition scenarios

Transition Scenario		TSA3	TSB1	TSB2
Mass (t)	FP	15 370	15 375	17 292
	U	238	237	336
	Pu	16.9	17.7	16.7
	MA	839	128	4.4
Total Mass (t)		**16 465**	**15 758**	**17 649**

6.4.5 HLW Radiotoxic Inventory

Figure 6.14 shows the contribution of the different types of HLWs generated in the reprocessing of spent fuels in TSA3. Also the radiotoxic inventory of RepU is shown.

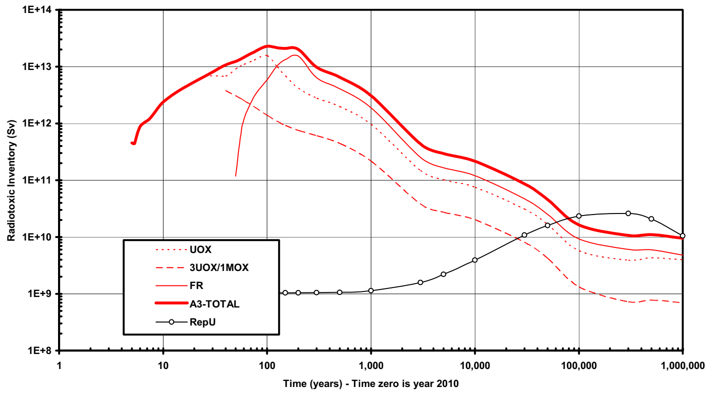

Figure 6.14: Radiotoxic Inventory of HLWs and RepU in TSA3

Figure 6.15 shows the contribution of the different types of HLWs generated in the reprocessing of spent fuels in TSB1. The radiotoxic inventory of RepU is also shown in the graphic.

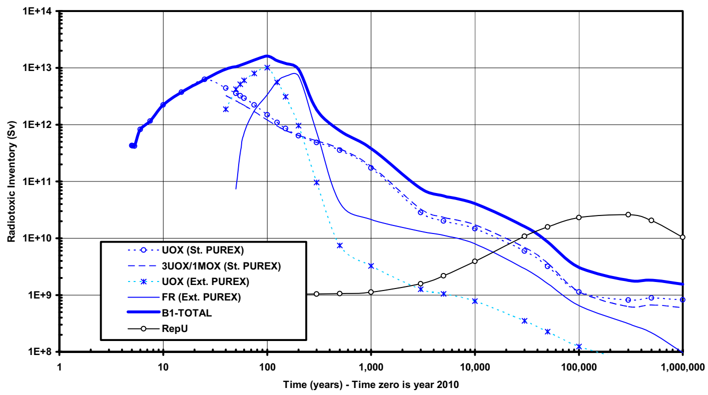

Figure 6.15: Radiotoxic Inventory of HLWs and RepU in TSB1

Figure 6.16 shows the contribution of the different types of HLWs generated in the reprocessing of spent fuels in TSB2. The radiotoxic inventory of RepU is shown too.

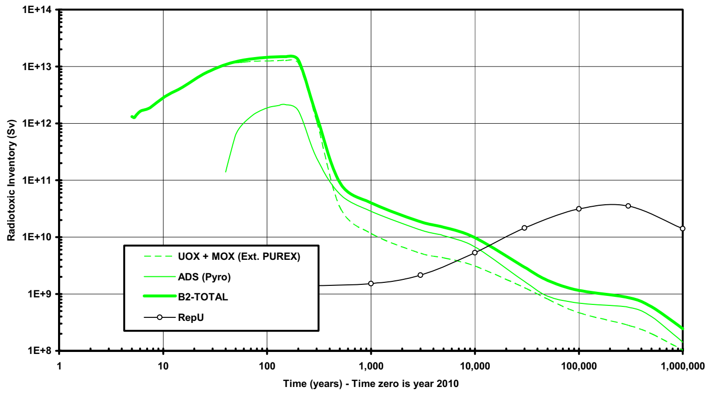

Figure 6.16: Radiotoxic Inventory of HLWs and RepU in TSB2

The hypothesis of the availability of advanced reprocessing technology (Extended Purex) at the beginning of TSB2 (by 2010) makes possible to reach a reasonably low HLW radiotoxic inventory in the short and medium term. If this technology were not industrially developed by that date, and its availability were delayed to 2040 as it has been assumed for TSB1, the radiotoxic inventory of the HLWs in scenario TSB2 would increase to levels higher than scenario TSB1 (see 6.17). The reason is that during the first 30 years (from 2010 to 2040) the vitrified HLWs would contain all the minor actinides of the reprocessed fuel, not only the reprocessing losses. The HLWs generated in this period from the reprocessing of LWR fuels (UOX and MOX) are responsible for most of the total radiotoxic inventory of the whole transition scenario (2010 to 2200). Then, the reduction in the radiotoxic inventory (compared with the Open Cycle) reached in the TSB2 would be strongly handicapped if the advanced partitioning technology were not available and could not be applied from the beginning of the scenario (2010).

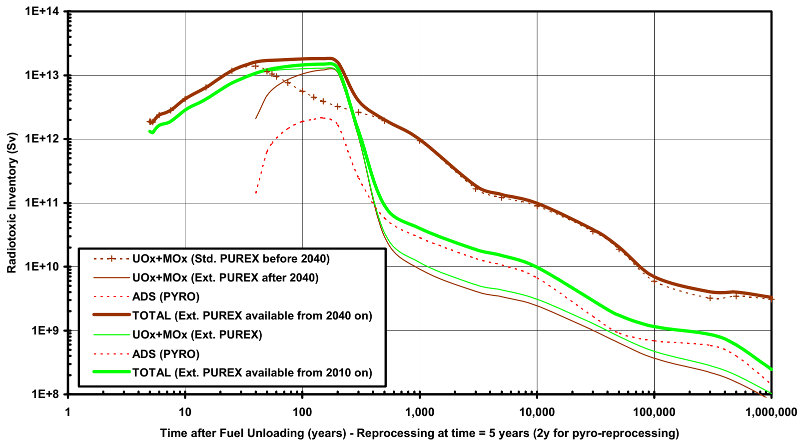

Figure 6.17: Effect of the date of Extended Purex availability on the radiotoxic inventory

Figure 6.18 represents the time evolution of the radiotoxic inventory present in the totality of HLWs generated in each transition scenario (year 5 in the figure corresponds to year 2010). For comparison purposes, the figure includes the curves of Scenario A1 and the corresponding Equilibrium Scenarios A3, B1 and B2 as if they had started in year 2010. The radiotoxic inventory of the reprocessed uranium is also shown in the figure.

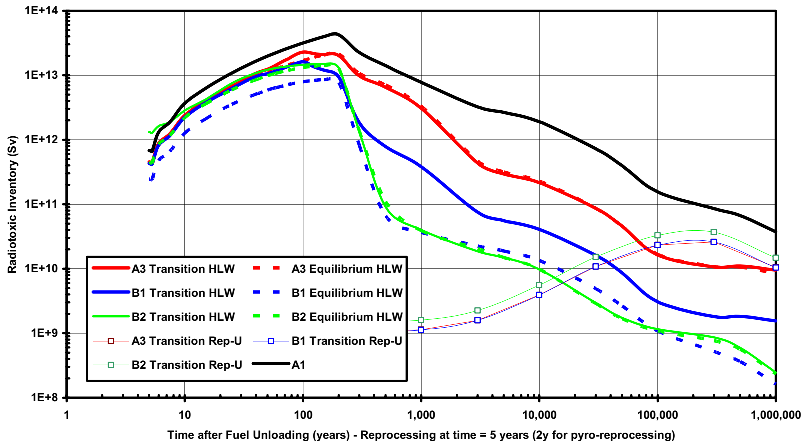

Figure 6.18: Radiotoxic Inventory in Transition and Equilibrium Scenarios

The HLW radiotoxic inventory in transition scenarios is very similar to that of equilibrium scenarios, except in scenario B1, in which the radiotoxic inventory in the transition scenario is higher than in equilibrium. The reason for this is the use of different partitioning technologies during some time intervals of the scenario. In fact, the equilibrium scenario B1 assumes the exclusive use of the Extended Purex, whereas the transition scenario, in a more realistic approach, considers that this technology will be available in 2040. It implies that during the first 30 years of the transition scenario (from 2010 to 2040) all the vitrified HLWs will contain the minor actinides of the reprocessed fuel, whereas in the equilibrium mode they will only contain small losses. The radiotoxic inventory of Equilibrium Scenarios B1 and B2, and Transition Scenario B2 are very similar, since they have assumed the use of advanced reprocessing for all spent fuel from the beginning (2010).

This reflects the importance that the deployment time for the advanced reprocessing technology has on the global reduction of the HLW radiotoxic inventory, if this was an objective. The reduction in the radiotoxic inventory reached in the innovative scenarios can be strongly diminished if the advanced technology of P&T is not available and can not be applied from the beginning of the transition scenarios. Currently, huge amounts of spent fuel have already been produced in many countries. If a significant reduction of the HLW radiotoxic inventory is desired, the transmutation of all the transuranics should be applied to all stored fuels, independently of their age or origin.

The radiotoxic inventory of the reprocessed uranium has to be accounted for, since it has not been used in the transition scenarios, constituting thus a long-lived radioactive waste to be disposed of. Figure 6.19 shows the radiotoxic inventory of HLWs plus RepU (long lived ILWs not included): not only the radiotoxicity reduction factors decrease, but also the period of time in which the reduction factors are significant. Moreover, all the curves approach each other, reaching the same level from year 100 000 on.

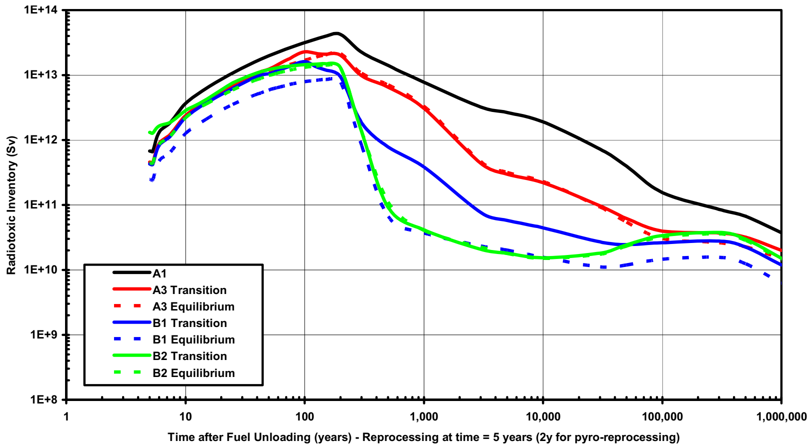

Figure 6.19: Total Radiotoxic Inventory (HLWs plus Reprocessed Uranium)

These data prove that if the transition period for all technologies is properly considered, and a realistic and feasible approach is made for the analyses, the potential benefit in terms of radiotoxicity reduction should be seriously reviewed.

6.4.5 HLW Thermal Power

The HLW packages must be placed in an interim storage in order to allow decaying the radioactivity and heating to an acceptable level before deep disposal. Table 6.10 shows the thermal power of the HLW packages after a 50 years interim storage period.

Table 6.10: HLW packages generation and thermal power

Transition Scenario	TSA3			TSB1				TSB2	
Reprocessed Fuel	UOX (LWR)	Mix of 3UOX 1MOX (LWR)	MOx (FR)	UOX (LWR)	Mix of 3UOX 1MOX (LWR)	UOX (LWR)	MOx (FR)	Mix of UOX MOX (LWR)	TRU+Zr Nitride (ADS)
Reprocessing Method	St. Purex	St. Purex	St. Purex	St. Purex	St. Purex	Ext. Purex	Ext. Purex	Ext. Purex	Pyro
WP Production	164 910	18 540	228 190	38 306	16 584	121 503	217 560	397 290	43 953
Thermal Power (W) 50 years after fuel unloading	533	661	489	517	645	419	315	410	445

The thermal power of packages with wastes resulting from the reprocessing of the mix of LWR UOX and MOX fuels with the Standard Purex is very high and perhaps more cooling time will be required before its final disposal, depending on the repository design and host rock thermal properties. Packages with wastes from advanced reprocessing are less heat producers, as it happens in transition scenarios B1 (after 2040) and B2.

Not only the thermal output of the individual packages, but also the total heat to be managed in the scenario is relevant. The curves in Figure 6.20 show for each year the total thermal power of the whole HLWs generated from year 2010 up to that year in each transition scenario. It could be concluded that the interim storage of wastes from TSB1 will need a lower heat removal system dimensioning than the other scenarios.

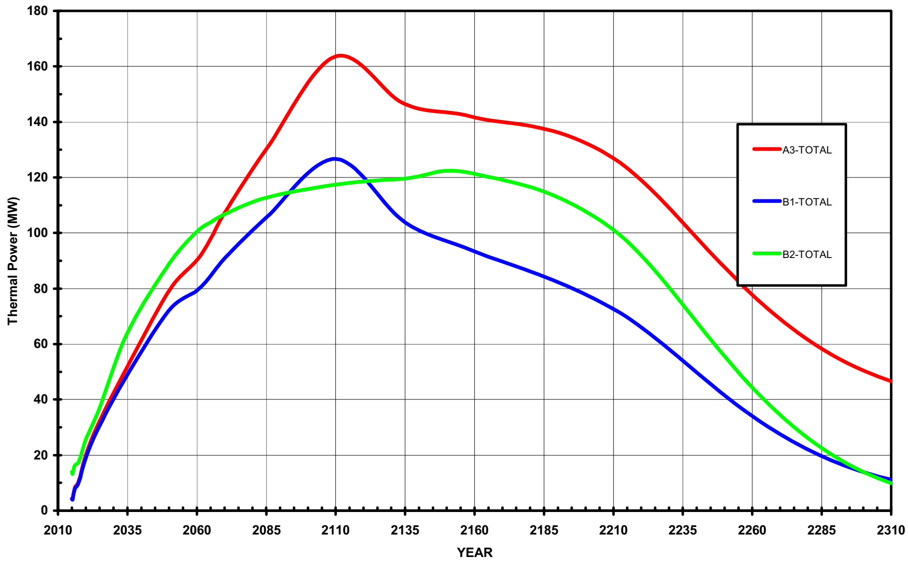

Figure 6.20: Thermal Power from HLWs in Transition Scenarios

6.5 Scenario Variants (Caesium and Strontium Partitioning)

Isotopes Cesium-137 and Strontium-90 are responsible of most of the short-term thermal power and radiation level of the HLWs, thus affecting the waste management operations and the repository design. Several variants of Scenario B1 have been analyzed regarding the possibility of partitioning and separate conditioning of Caesium and/or Strontium from the rest of the fission products. Separated Cs and Sr are assumed to be individually vitrified and conditioned in Universal Canisters (UC) loading 60 kg. The Scenario B1 variants that have been chosen are the following:

- Variant-B1.1 (40FP-60Cs-60Sr): Caesium and Strontium are removed from the fission products stream. It is assumed that only 1% of Cs and 1% Sr go to the glass. A mass of 40 kg of non-volatile fission products plus actinides are loaded per UC.
- Variant-B1.2 (40FP-60Sr): Only Strontium is removed from the fission products stream. It is assumed that only 1% Sr goes to the glass. A mass of 40 kg of non-volatile fission products plus actinides are loaded per UC.
- Variant-B1.3 (60FP): Scenario in which a mass of 60 kg of non-volatile fission products plus actinides are loaded per UC. Neither Cs nor Sr is partitioned from the fission products stream.
- Variant-B1.4 (60FP-60Cs-60Sr): Caesium and Strontium are removed from the fission products stream. It is assumed that only 1% of Cs and 1% Sr go to the glass. A mass of 60 kg of non-volatile fission products plus actinides are loaded per UC.
- Variant-B1.5 (60FP-60Sr): Only strontium is removed from the fission products stream. It is assumed that only 1% Sr goes to the glass. A mass of 60 kg of non-volatile fission products plus actinides are loaded per UC.

The differences with the HLW universal canisters produced in the Scenario B1 [6.1] are their content (fission products, Cs alone or Sr alone) and the amount of fission products per universal canister (40 or 60 kg, depending on the variant). The Table 6.11 summarizes the production of universal canisters in each variant, and the HLW volume in each variant. No waste packages are assumed for the conditioning of the Cs and Sr universal canisters when calculating volumes, as they have to be stored on surface for very long time (150-300 yr) prior to vitrification and final disposal in an eventual second deep geological repository.

Table 6.11: Production of universal canisters for HLW in the B1 scenario variants

SCENARIO B1-Variant	B1 Ref.	B1.1			B1.2		B1.3	B1.4			B1.5	
Waste	UC 40 kg FP	UC 40 kg FP	UC 60 kg Cs	UC 60 kg Sr	UC 40 kg FP	UC 60 kg Sr	UC 60 kg FP	UC 60 kg FP	UC 60 kg Cs	UC 60 kg Sr	UC 60 kg FP	UC 60 kg Sr
UC/TWeh	2.271	1.959	0.186	0.022	2.213	0.022	1.497	1.306	0.186	0.022	1.475	0.022
Package Volume (m^3)	0.53	0.53	0.18	0.18	0.53	0.18	0.53	0.53	0.18	0.18	0.53	0.18
HLW Volume (m^3/TWeh)	1.20	1.04	0.033	0.004	1.17	0.004	0.79	0.69	0.033	0.004	0.78	0.004

There is some reduction in the production and volume of HLWs when caesium is partitioned from the fission product stream, but the main reduction in the volume of wastes to be sent to the deep geological repository arises when the mass of fission products per universal canister is set to 60 kg instead of 40 kg. The partitioning of only Sr has a very small effect on the HLWs volume.

Table **6.20** shows the evolution with time of the thermal output of the wastes. After the removal of caesium and strontium, the thermal power of the waste packages is so low that they could be sent to the repository after a few years of cooling (Variants-B1.1 and B1.4). Although this very low thermal power of the wastes allows a reduction of the HLW emplacement area in the repository, it does not take advantage of the capacity of the repository to dispose of heat-producing wastes. In other words, a similar reduction could be obtained with a not so dramatic reduction in the thermal power of the wastes.

The removal of strontium alone would be a compromise solution, which results in the production of waste packages of medium thermal power (Variant-B1.2) in such a way that they could be sent to the repository after a shorter cooling time (less than 50 y). Another advantage of this medium thermal power would be, if technically possible, the optimization of the repository HLW emplacement area, either by designing new waste packages able to group several universal canisters, or by decreasing the distance between disposal galleries or boreholes. In any case these advantages depend on the repository host rock formation.

The Variant-B1.5 (removal of Sr and conditioning of 60 kg FP per canister) produces wastes with more o less the same power than the B1-Reference Scenario, but it produces one third less canisters per TWeh, due to the higher load of fission products per canister.

Table 6.20: Thermal output (W) of canisters for HLW in the B1 scenario variants

B1 Scenario Variant	B1 Ref.	B1.1	B1.2	B1.3	B1.4	B1.5	B1.1 and B1.4	B1.1, B1.2, B1.3 and B1.5
Waste content	40 kg (FP)	40 kg (FP-Cs-Sr)	40 kg (FP-Sr)	60 kg (FP)	60 kg (FP-Cs-Sr)	60 kg (FP-Sr)	60 kg (Cs)	60 kg (Sr) (*)
Time = 5 y	1 535	466	1 235	2 308	699	1 853	9 930	30 480
Time = 25 y	573	27	398	860	40	596	4 496	18 920
Time = 50 y	314	9	218	472	14	327	2 521	10 470
Time = 100 y	98	3	69	148	5	104	794	3 185
Time = 150 y	32	2	23	47	3	34	250	969
Time = 200 y	11	1	8	16	2	12	79	295
Time = 300 y	2	1	2	3	2	2	8	27

(*) The high values in this column just reflect the thermal power of the 60 Kg of Sr. Vitrification at these short times would be impossible

The separated Cs and Sr are strong heat-producers (HLWs), but after several centuries of cooling (about 300 y) their thermal power will have decreased enough to accept a different classification. The Sr-canisters could be sent to a shallow disposal facility after that period of time, but the presence of the long-lived Cs-135 in the Cs-canisters, which is a relevant nuclide for the long term safety, makes mandatory the deep geological disposal of these canisters. Due to the high thermal power of the Cs-canisters, they would require a long cooling time before its final geological disposal, which could delay significantly any repository operation schedule, or force the need of a second repository. All these canisters will need heavy shielding for the handling, in particular the Cs-canisters due to the strong gamma emission from the Cs-137 decay.

6.6 Conclusions

Five different “Equilibrium Scenarios” have been analyzed, showing the production and characteristics of Spent Fuel, HLWs and solid long lived ILWs arising in the reprocessing. Disposal canisters for these wastes have been selected and the resulting Waste Packages have been characterized. Several scenario variants considering the partitioning of Caesium and/or Strontium have been studied. Some “Transition Scenarios” have been analyzed. The main conclusions that can be drawn from the study performed for the RED-IMPACT Project are the following:

- The total mass of actinides per TWh(e) which is sent to the repository in the Open Cycle is much higher than the other scenarios. Most of this mass corresponds to uranium oxide. In the Closed Cycles studied this uranium is separated but not used, thus it must be indefinitely stored in solid form or definitely disposed of in a geological repository.

- Some radionuclides, especially I-129, Cl-36 and C-14, which have great significance for the long-term radiological impact of a deep geological repository, are almost completely released into the environment during the reprocessing. The fully recovering of these isotopes, their conditioning into stable matrices and proper disposing of would be necessary whatever reprocessing technique is implemented. This point is further developed in chapter 7.

- The recycling of plutonium alone decreases the mass of this element, but it increases the mass of minor actinides, mainly americium, that are sent to the deep disposal repository. The amount of minor actinides, which represents only a small fraction of the waste mass, can be reduced if they are burnt in fast reactors or accelerator driven systems, which requires the previous use of advanced partitioning technologies.

- The total volume of the solid waste packages (HLW and long lived ILW) to be disposed of in the deep geological repository in all the Advanced Cycle scenarios is much higher than in the Open Cycle (scenario A1) . Close Cycles result in a reduction of only the HLW packages volume compared to the waste package volume of the conditioned spent fuel in the Open Cycle. However, the high volume of long lived ILW generated in this process, reaches by itself values always higher than the spent fuel in the open cycle. This long lived ILW must be disposed of in a geological repository in addition to the HLW generated, producing total volumes considerably higher than in open cycle. The volume of the reprocessed Uranium, not included, must be recalled.

- The radiotoxicity is a magnitude whose relevance is limited to the case of accident during the handling of wastes (treatment, conditioning, transport, temporary storage) with release of radionuclides that could be ingested by humans. The present technologies of fuel reprocessing together with the burning of plutonium alone do not reach a significant reduction factor in the radiotoxic inventory of HLW wastes. A higher reduction in the radiotoxic inventory per TWh(e) can only be achieved by means of the use of advanced technologies for additional (not only plutonium) minor actinides partitioning and transmutation. This factor is drastically lowered if the radiotoxic inventory of separated reprocessed uranium is accounted for. In this case not only the factor decreases, but also the period of time in which the reduction factor is significant.

- The thermal power per TWh(e) of the total HLWs from each scenario is very sensitive to aspects as the electrical conversion efficiency of nuclear reactors, the reprocessing efficiency for the recovery of actinides and the fission yield for Sr-90 (relevant heat producer). Sr-90 fission yield decreases as the atomic weight of fissionable nuclides in the fuel increases (heavy transuranics). Scenario B1 (fast reactors with advanced reprocessing) takes advantage of all these aspects, although adequate optimizations of the repository size, waste packages design, etc. could be performed for all the scenarios.

- The analysis of the transition scenarios has shown that:

    - Any judgement or decision made on the basis of the advanced equilibrium scenarios analysis could be highly biased since they do not represent a realistic situation regarding waste production and characteristics.

    - If a substantial reduction of the short and medium term radiotoxicity compared to the open cycle is desired, the advanced reprocessing techniques (Extended Purex or Pyroreprocessing) should be applied to all spent fuels within a national or international program. As soon as the Standard Purex is applied over a small fraction of the spent fuel park, the potential benefits of the advanced P&T techniques on the reduction of part of the short term radiotoxicity is highly questionable.

- Isotopes Cesium-137 and Strontium-90 are responsible for most of the short-term thermal power and radiation of the HLWs. After the partitioning of caesium and strontium from the HLWs, the thermal power of the waste packages is so low that they could be sent to the geological repository after a few years of cooling. Nevertheless, a similar benefit could be obtained if only strontium were removed, which results in waste packages of medium thermal power, and avoids the shortcomings of the later management of Caesium and its interference with the operation schedule of the deep geological repository. In any case these advantages depend on the repository host rock formation.

7. Impact on Final Disposal

The main objective of this chapter is to assess the consequences on waste disposal of the application of partitioning and transmutation in advanced fuel cycles. To realise this objective, different aspects concerning the final disposal of the different high-level and intermediate level waste types, which were identified in Chapter 6, resulting from the application of advanced fuel cycles, are analysed by considering different disposal sites and host formations. For the high-level radioactive waste (HLW), i.e. vitrified high-level waste and spent fuels, first the potential impact on the repository concept was evaluated. Further, simplified assessments of the long-term radiological consequences in case of the expected evolution of the geological disposal system for disposal in granite and clay and for an altered evolution for disposal in salt, were carried out to illustrate how the doses are influenced by the fuel cycle options. Doses in case of a human intrusion into the sealed repository were also calculated, because it can be expected, on the basis of the radiotoxicity inventory, that human intrusion doses are much more sensitive to the fuel cycle options than are doses in the case of the expected evolution of the repository system. For the intermediate-level waste (long lived ILW), a limited number of impact assessments of its disposal in geological repositories were carried out.

7.1 Impact of advanced fuel cycle scenarios on repository designs for HLW disposal

In various countries repository concepts have been developed for the disposal of vitrified HLW and spent fuel. For the RED-IMPACT project the following repository designs have been used as starting point:

- disposal in granite:
  - Spanish repository concept used by Enresa (Spain);
  - Czech reference disposal concept used by NRI (Czech Republic);
- disposal in clay:
  - SAFIR 2 concept used by SCK•CEN (Belgium);
  - ANDRA concept used by CEA (France);
- disposal in salt:
  - German concept used by GRS (Germany).

Repository designs for HLW disposal have to take into account several aspects related to the characteristics of the disposed waste. Potentially important aspects are thermal output of the waste packages, radiation fields (γ, and neutrons), criticality and retrievability.

An analysis of <u>criticality</u> carried out by KTH has shown that the criticality problem in case of disposal of vitrified HLW is negligible, because of the very small amount of fissile materials present in that waste. However, in case of disposal of spent fuels (uranium oxide (UOX) and mixed oxide (MOX) fuels) and after ingression of groundwater into the disposal canister, the k_{eff} may rise to values of 0.63 and 0.68 for respectively UOX and MOX spent fuels. Consequently, as far as the fuels have been irradiated up to the burn-up for which they were designed, the k_{eff} will remain much lower than 0.95 in repository conditions and no criticality problems are expected. For the advanced fuel cycles, it is assumed that all spent fuels are reprocessed, and that only very small amounts of actinides are going into the repository; consequently, the possibility that criticality conditions will be reached in the repository is negligible.

NRG examined some aspects related to the retrievability of the disposed HLW types for the different considered fuel cycle scenarios. Various stages of the closure of a repository are considered. An important parameter for retrievability operations is the temperature in the repository. This might potentially conflict with an optimisation of the dimension of a repository from a definitive disposal point of view for which possibly less stringent temperature limitations in the near field have to be respected. The main conclusion of the study is that the HLW arising from advanced fuel cycles can be handled in the same way as the present spent fuel and vitrified waste with respect to retrievability. The most problematic waste type appears to be the MOX spent fuel arising from the industrial scenario A2, which assumes mono-recycling of Pu. However, even in this case, there would not be technical hindrance to retrieve such fuels from the repository.

An important aspect for the dimensioning of a geological repository for HLW is the thermal output of the waste. For all types of host formations, a number of temperature limitations have to be respected. Therefore, detailed analyses of the expected temperature evolution in the repositories have been made. It was assumed that the waste will be placed in the repository after a total cooling time of 50 years. The evolution of the thermal output of the HLW waste packages arising from the 5 base fuel cycle scenarios is given in Figure 6.9.

7.1.1 Impact of the thermal output on the repository design in case of HLW disposal in granite (Spain)

Enresa considered the repository concept shown in Figure 4.5. A longitudinal section of a disposal gallery is given in Figure 7.1. The disposal galleries have a diameter of 2.4 m and the distance between 2 disposal galleries is 35 m. The assumed depth of the repository is 500 m.

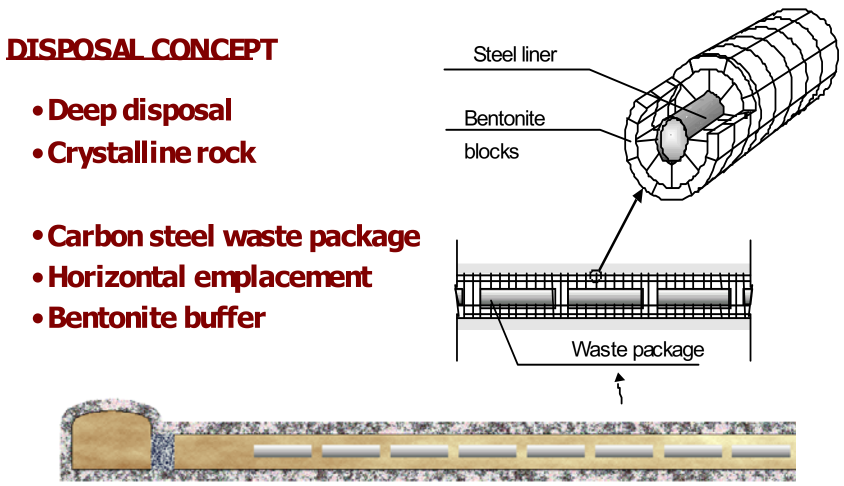

Figure 7.1: Longitudinal section of a disposal gallery (Spanish repository concept for disposal in granite)

Enresa used its numerical CODE_BRIGHT for heat transport calculations. CODE_BRIGHT stands for COupled DEformation, BRIne, Gas and Heat Transport. The code is a tool specifically designed for modelling thermo-hydro-mechanical processes in geological media in a coupled way. The main thermal constraint in the Enresa repository concept is that the temperature in the bentonite buffer has to remain lower than 100 °C. Heat transport calculations have been run several times for each scenario to determine the minimum acceptable distance between two waste packages. The calculated maximum temperatures are shown in Figure 7.2.

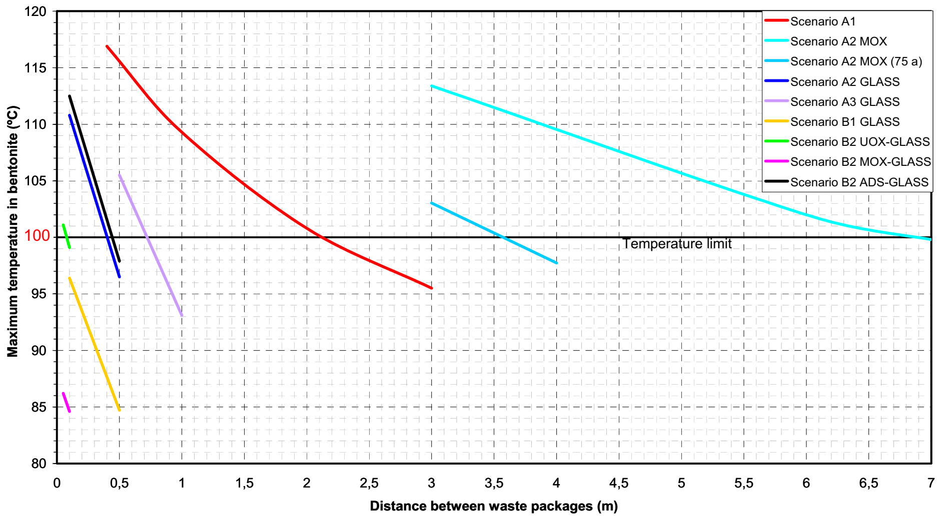

Figure 7.2: Maximum temperatures as a function of the distance between two waste packages (Spanish repository concept for disposal in granite)

The minimum distance between two waste packages is determined from Figure 7.2 by taking the shortest distance for which the maximum temperature remains below 100 °C. The resulting lengths of the disposal galleries for the 5 base fuel cycle scenarios are given in Table 7.1. Although the total thermal output (per TWh(e)) of the two high-level waste streams arising from scenario A2 is somewhat smaller than the one from scenario A1, the total needed gallery length is larger because of the high thermal output of the MOX fuel in case of scenario A2. Comparison of scenarios A1 and B1 shows that the introduction of an advanced fuel cycle can lead to a 60 % reduction in the needed disposal gallery length per unit of produced electricity.

7.1.2 Impact of the thermal output on the repository design in case of HLW disposal in granite (Czech Republic)

The considered repository concept is based on the Czech reference disposal project, relying on carbon steel canisters surrounded by bentonite in vertical arrangement in granite host rock (similar to the Swedish KBS 3 concept, with the exception of the considered canister material) for isolation of the disposed wastes (Figure 7.3).

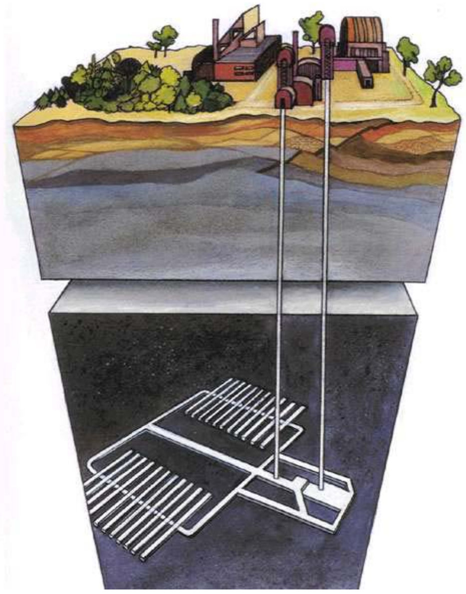

Figure 7.3: Czech reference concept of a repository in a granite formation

The repository layout consists of boreholes with spent fuel or high-level waste packages surrounded by bentonite bricks and located in tunnels 500 m under the surface. Since no candidate repository sites have been selected in the Czech Republic so far, the most pessimistic data from Swedish non-saline granite sites Ceberg and Beberg, possibly similar to future Czech sites, given in the SR 97 [7.1] performance assessment, have been used for modelling the radionuclide release from the repository. The total amount of spent fuel assemblies from scenario A1 and high-level wastes from scenario B1 disposed in the repository were assumed to correspond to the electricity production from the current Czech nuclear power plants considering a lifetime of 40 years (about 1190 $TW_{el}h$).

Thermal calculations aiming at determining the distance between 2 boreholes, at which the limiting temperature of 100°C in the bentonite buffer should not be exceeded, were performed for scenarios A1 and B1 by assuming an infinite host rock environment and boreholes containing one spent fuel canister (containing 4 uranium oxide assemblies) or two HLW canisters in a backfilled tunnel. The resulting maximum temperature on the outside surface of the canister with the highest thermal loading was determined by summation of the temperature increases calculated for individual boreholes. The ambient temperature at the depth of 500 m (23.5°C) was then added to the calculated temperature increase. The evolution of the maximum temperatures at the canister surface calculated for a constant distance (25 m) between 2 tunnels and different distances between the boreholes is given in Figure 7.4a for scenario A1 and in Figure 7.4b for scenario B1.

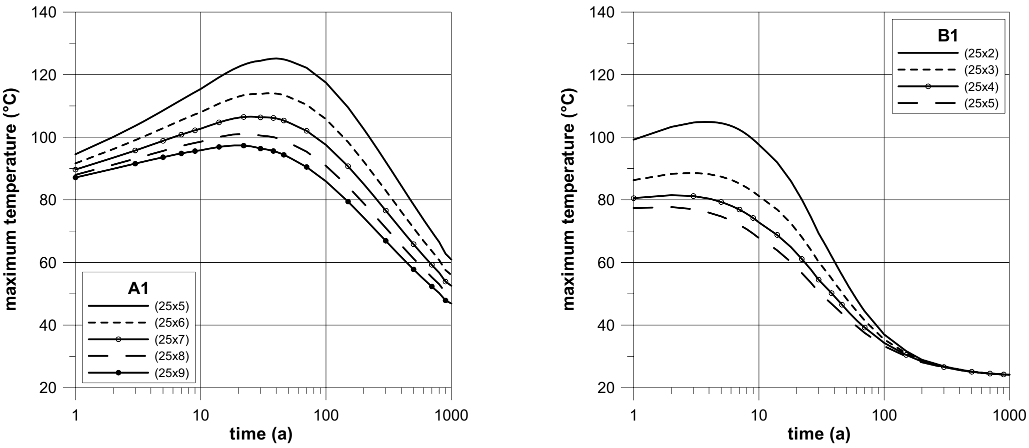

Figure 7.4: Evolution of the temperature at the surface of the canisters calculated for scenarios A1 and B1 (Czech repository concept for disposal in granite)

Figure 7.4a shows that the temperatures at the surface of the canisters will easily exceed the limiting temperature 100 ^{o}C at times before 100 years if the distance between 2 boreholes is shorter than 9 m. This means that not more than 20 canisters can be emplaced in tunnels of 200 m length. For the total number of 1478 canisters, 74 tunnels are needed. The size of such a repository will thus be larger than 370 000 m^2.

Due to the almost complete absence of actinides, the decrease of the thermal output of canisters containing high-level waste from scenario B1 is much faster than that of spent fuel (Figure 7.4b). The maximum temperature will be reached less than 10 years after repository closure. Figure 7.5b shows that the limiting temperature of 100 ^{o}C will not be exceeded even for a distance of 2.2 m between boreholes containing each 2 HLW canisters. This means that the area needed for the repository with HLW from scenario B1 would be about 6 times smaller (60 000 m^2) than the area needed for a spent fuel repository.

7.1.3 Impact of the thermal output on the repository design in case of HLW disposal in clay (Belgium)

The repository considered by SCK•CEN for the RED-IMPACT calculations is the reference repository that is described in detail in the Belgian SAFIR 2 report [7.2]. This reference repository is assumed to be excavated in the Boom Clay formation at the Mol site.

The considered repository will have a central access facility consisting of at least two vertical transport shafts and two transport galleries. The disposal galleries will be excavated perpendicular to the transport galleries. The HLW canisters will be placed one after the other or with a relatively small distance between two canisters to respect temperature limitations, in the centre of the disposal galleries. A clayey backfill, e.g. a mixture of swelling Ca-bentonite and sand, will be placed between the HLW canisters and the gallery walls. Because the Boom Clay is a plastic clay, a concrete lining is required to limit convergence of the gallery walls. A scheme of a HLW gallery configuration is given in Figure 7.5.

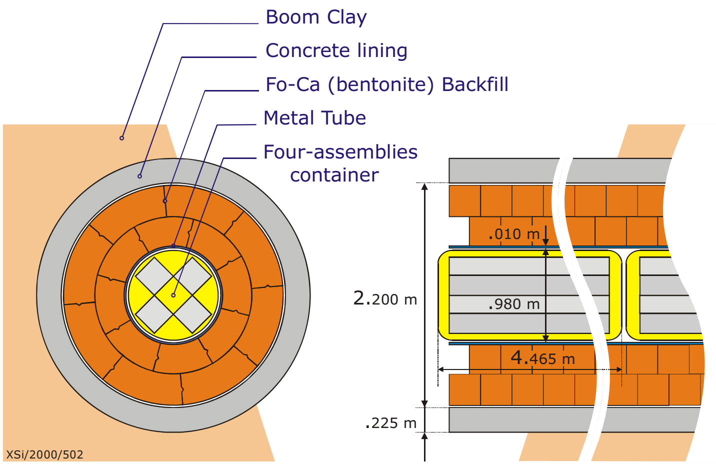

Figure 7.5: Gallery configuration with 4 assemblies per container for uranium oxide spent fuel disposal (Belgian repository concept for disposal in clay; from [7.2])

The maximum allowable disposal density for the SAFIR 2 repository configuration has been determined by making detailed thermal calculations with the PORFLOW code. In the case of disposal of vitrified HLW and spent fuel (comparable with the HLW from scenarios A1 and A2), the main thermal limitation for the repository concept is that the maximum temperature at the gallery lining / Boom Clay interface has to remain below 100 °C. In a first step, the evolution of the temperature at the lining / Boom Clay interface is calculated for a disposal configuration without spacing between waste packages (see Figure 7.6). When the calculated maximum temperature exceeds 100 °C, some spacing between waste packages is needed. When the maximum temperature remains below 100 °C, higher waste loadings of the canisters or denser disposal configurations can be considered. The estimated minimum needed gallery length per produced electricity is given in Table 7.1. However, in the dose calculations (cf. Section 7.2.5) waste loadings and dimensions of the waste canisters were used. This leads for most scenarios to longer gallery lengths.

The introduction of the advanced fuel cycle scenario B1 results in a reduction of the length of the needed HLW disposal galleries with a factor 3 in comparison with the reference "once through" fuel cycle scenario A1.

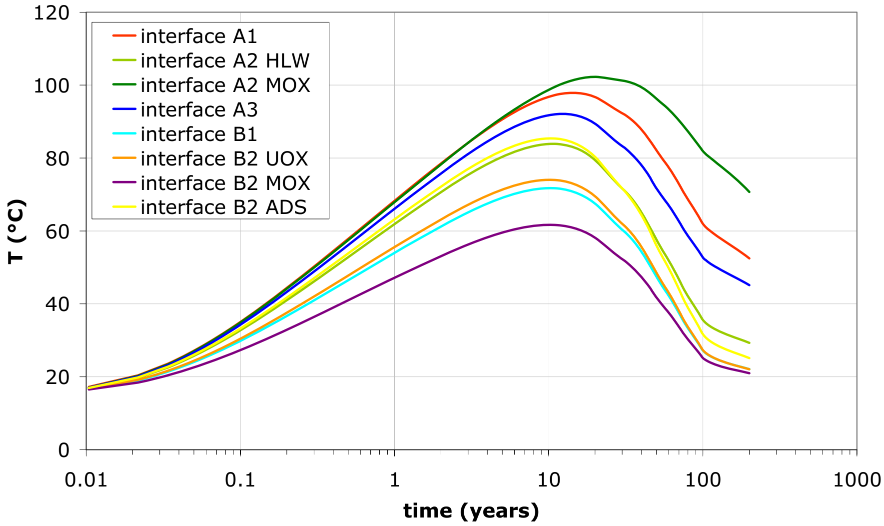

Figure 7.6: Calculated evolution of the temperature at the interface between the gallery lining and the Boom Clay for the various waste types (Belgian repository concept for disposal in clay)

SCK•CEN made also calculations for two variant scenarios of B1 in which it is assumed that only Sr or both Cs and Sr are separated from the HLW prior to vitrification. The removal of Cs and Sr has a considerable impact on the heat generation of the remaining HLW. Separating Sr only, results in a further reduction of the needed gallery length with a factor 1.5, or about a factor 5 in comparison with the "once through" reference fuel cycle A1. Separating Cs and Sr makes that the thermal output of the vitrified HLW is so low that cooling times are not longer needed, or that other repository concepts similar to those developed for the disposal of long lived ILW, in which several waste canisters are placed in a gallery section, can be taken into consideration. However, for the estimation of the needed gallery length one should not forget that the separated Cs has to be disposed of in a geological repository also, because it contains the long-lived ^{135}Cs isotope (half-life 2.3 million years). If it is assumed that the conditioned Cs-waste is disposed after a cooling time of 100 years (reminder: a 50 years cooling time was considered for all spent fuels and HLW types), the needed gallery length can be reduced with a factor 4 in comparison with scenario B1 and with a factor 13 in comparison with scenario A1.

Table 7.1: Overview of the estimated needed length of the HLW disposal galleries in case of disposal in granite and clay formations

Scenario		A1	A2-MOX	A2-HLW	A2-tot	A3	B1	B2-UOX	B2-MOX	B2-ads	B2-tot
General data											
number of canisters	(can/TWh)	1.3384	0.543	2.4825		2.399	2.2711	2.0708	0.2562	0.3253	
thermal output (50 a)	(W/can)	1564	1549	486		529	314	420	331	438	
canister length	(m/can)	4.54	4.54	1.335		1.335	1.335	1.335	1.335	1.335	
thermal output (50 a)	(W/m)	344.5	341.2	364.0		396.3	235.2	314.6	247.9	328.1	
Granite (Enresa)											
gallery length	(m/TWh)	8.89	6.16	4.96	11.12	5.52	3.63	3.52	0.41	0.56	4.49
relative gallery length	(-)	1.00			1.25	0.62	0.41				0.51
Clay (SCK•CEN)											
allowable thermal output (50 a)	(W/m)	353.5	332.3	375.6		364.8	378.8	379.3	379.1	378.1	
needed gallery length	(m/TWh)	5.92	2.53	3.21	5.74	3.48	1.88	2.29	0.22	0.38	2.89
relative gallery length	(-)	1.00			0.97	0.59	0.32				0.49
used gallery length	(m/TWh)	6.08	2.53	3.31	5.85	3.48	3.03	2.76	0.34	0.43	3.54

7.1.4 Impact of the thermal output on the repository design in case of HLW disposal in salt (Germany)

The disposal facility considered by GRS is assumed to be located in a salt dome at a depth of 870 m. The thickness of the overburden is about 300 m. The waste is disposed of in horizontal drifts in a single story (single layer) arrangement. The disposal facility consists of the central field, two flank drifts and five cross connections, with 100 disposal drifts (Figure 7.7). One cross connection and the adjacent 20 disposal drifts and the two segments of the flank drifts form one section. Access to the facility is provided by two shafts.

Disposal drift, connection drift and flank drift sealings and dams are constructed from pre-compacted salt. All void spaces are backfilled with crushed salt. It is assumed that the backfilling has an initial porosity of 35%, the sealings an initial porosity of 25 % and the dams of 5 %. Due to creep characteristic of salt fast encapsulation is ensured in the engineered barrier system.

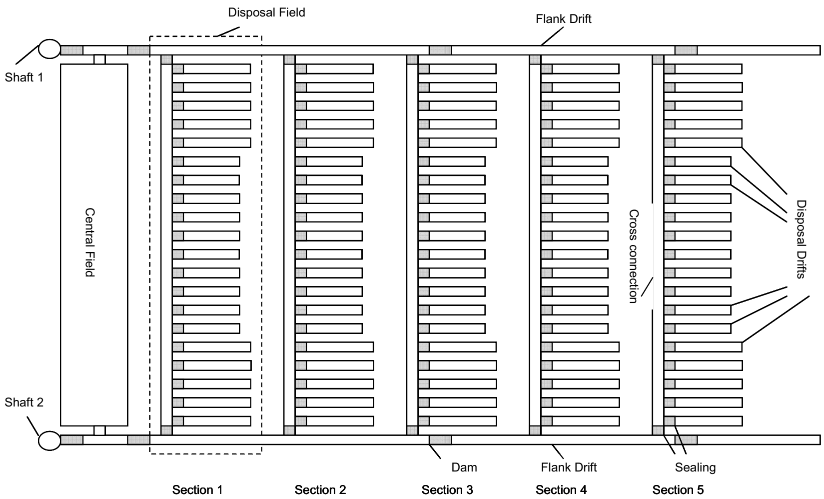

Figure 7.7: Sketch of the disposal facility (German repository concept for disposal in salt)

For the temperature field calculations the finite difference code HEATING is used. A three-dimensional model was generated consisting of the heat producing drifts and a surrounding of undisturbed homogeneous salt with a thickness of 500 m. The initial temperature is 30°C. The boundary temperature is kept constant at 30°C.

For the considered repository concept in salt the bulk temperature of the salt has to remain below 180°C. Calculations of the temperature field are needed to estimate the number of canisters that can be loaded in the disposal drifts. Furthermore, in case of disposal in salt the temperature evolution is needed for the calculation of the salt creep. Due to the very high temperature dependency of the salt convergence this calculation is an essential step in the long term performance assessment. The evolution of the calculated temperature is shown in Figure 7.8.

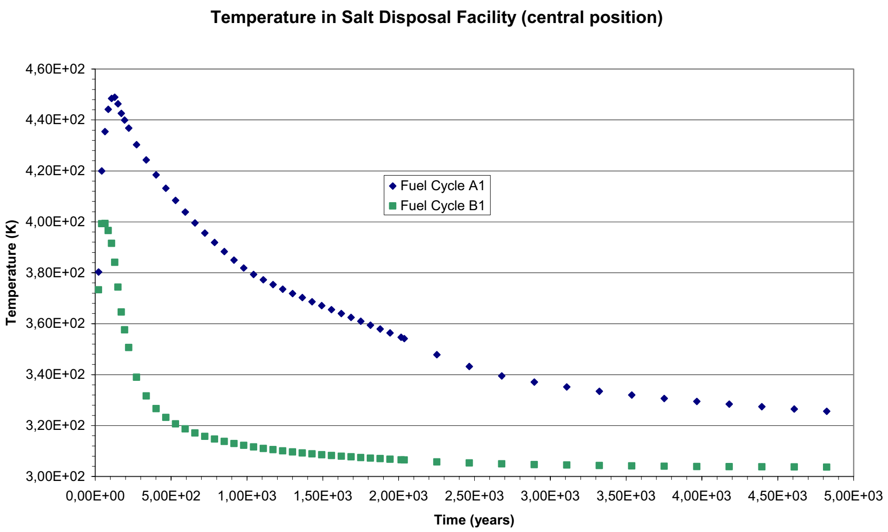

Figure 7.8: Evolution of the bulk temperature (relevant for salt creep calculation)

7.1.5 Conclusions concerning HLW repository designs

The HLW arising from advanced fuel cycles is not expected to be more difficult to dispose of in a geological repository than the HLW arising from existing industrial fuel cycles. On the contrary, the small amounts of actinides present in the resulting HLW means that this waste generates less heat than spent fuels or HLW arising from the currently applied fuel cycles, especially after one hundred years or more.

In case of repository concepts based on "in gallery disposal" (i.e. the waste packages are placed along the axis of a disposal gallery) in granite or clay formations, the needed length of the disposal galleries is about proportional to the thermal output of the waste at disposal time. A limited correction of about 15% can be made to take into account that the thermal output of HLW from advanced fuel cycles decreases much faster than the one of MOX spent fuel. The introduction of advanced fuel cycles can lead to a reduction of the needed length of disposal galleries with a factor 3, if the waste loading and or the dimensions of the waste canisters can be optimised.

Separation of Cs and Sr results in a drastic reduction of the thermal output of the remaining HLW. However, the Cs-waste has to be disposed of in a geological repository too, because it contains the long-lived ^{135}Cs isotope. The separation of Sr from the HLW would allow to further reducing the needed gallery length with a factor 1.5 or a factor 5 in comparison with the reference fuel cycle A1. The separation of both Cs and Sr results in a vitrified HLW with about negligible thermal output. This waste does not require a cooling time and it can be considered for disposal in repository concepts developed for the disposal of long lived ILW, in which several canisters are placed in a gallery section. By assuming a 100 years cooling time for the conditioned Cs-waste, the needed gallery length can be reduced with a factor 4 in comparison with scenario B1 and with a factor 13 in comparison with scenario A1.

The case of an "in borehole disposal" concept (i.e. the waste packages are placed in small boreholes that are drilled perpendicular upon the axis of a disposal gallery) is no longer a one-dimensional problem, but a two-dimensional problem. This explains the higher reduction (a factor 6) of the needed repository size calculated by NRI. One should notice that in case of "in gallery disposal" concepts it is possible, in principle, to optimise the distance between two disposal galleries. However, such an optimisation was considered to be out of the scope of the RED-IMPACT project, because it requires that various other aspects concerning repository design have to be taken into account.

The advanced fuel cycle scenarios A3, B1 and B2 can thus lead to a reduction in the total length of the excavated HLW disposal galleries or the size of the HLW disposal area for repositories in granite and clay in comparison with scenario A1. However, the other facilities of a repository (surface facilities, access ramps or shafts, elevators, ventilation shafts and galleries, etc.) remain mostly unchanged.

It is difficult to draw general conclusions concerning the impact of thermal output of the HLW on the dimensions of a repository located in a salt formation. On the one hand a maximum temperature limit has to be respected; on the other hand higher temperatures accelerate the convergence of the salt and thus contribute to limiting the possible consequences of a groundwater or brine intrusion scenario.

7.2 Impact of advanced fuel cycle scenarios on the long-term radiological consequences of HLW disposal in geological repositories

Post-closure safety assessment is the process of systematically analysing the performance of a repository and showing, with an appropriate degree of confidence, that it will remain safe over a prolonged period [7.4]. The arguments and evidence that describe, quantify and substantiate the safety and the level of confidence in the safety, are integrated in a safety case. Safety or performance assessments are the key component of a safety case. In general, a performance assessment will start with an analysis of the possible evolutions of the repository system; this step is called 'scenario development'. A second step consists in making analyses of the consequences of a relevant set of possible evolutions of the repository system.

It is evident that it is not possible to make a detailed performance assessment of the disposal of the main waste types arising from the considered fuel cycle scenarios within the RED-IMPACT project. In general, the considered repository systems will be strongly simplified and the consequence analysis can only be carried out for one evolution scenario and one pathway to the biosphere. The main objective of the calculations carried out within the RED-IMPACT project is to illustrate and evaluate the impact of the different radionuclide inventories associated with advanced fuel cycles on the long-term radiological consequences of the considered geological disposal systems. Doses due to the HLW arising in the 5 basic fuel cycle scenarios are calculated for different host formations and repository designs.

The calculations that are presented in the following sections are done for the expected evolution of the repository system, except in the case of the repository in salt. For a repository in salt, as long as the salt dome remains intact, it can be assumed that no groundwater will come in contact with the disposed waste and consequently no radionuclides would be released from the disposal system and doses would be always zero. Therefore, the consequences of an alternative scenario, i.e. an intrusion of groundwater into the disposal facility via an anhydrite vein, have been analysed. In other words, for a repository in salt an alternative, less likely evolution has been studied instead of the expected one.

The main safety indicator, which has been used for many years in safety assessments of radioactive waste disposal systems, is the annual effective dose (or dose rate) to an average member of the so called critical group[4] affected by the repository. To evaluate the safety of a repository the maximum dose rate can be compared with a dose constraint, the value of which has to be determined by the radiological protection authorities. ICRP recommends an upper value for the dose constraint of 0.3 mSv/year for application in normal exposure situations [7.9]. However, the evaluation of the safety of a repository cannot be reduced to a simple comparison of the calculated dose with the dose (or risk) constraint. Aspects relating to the time and the duration of exposure are to be taken into consideration as well.

For exposure situations with a probability of occurrence lower than one, annual risk (defined as the product of the dose rate and the probability of occurrence of that situation) can be used instead of dose rate.

During the time scales that have to be considered in the safety assessments of repositories for the disposal of HLW, various components of the repository system will undergo significant changes. Radiological exposure modes and surface environmental processes are expected to change considerably after a few decades or centuries, and the hydro geological system could change significantly after a few thousands of years (the extent of change varies dependent of e.g. on the geology and external factors that affect groundwater flow at a particular site). Such changes affect the confidence that can be placed in quantitative calculations at very long times (e.g. to one million years in the future). This has led to the concurrent use of complementary safety indicators, which are independent of the biosphere and the hydrogeology [7.3] in safety case studies. One of these indicators is the radiotoxicity flux released from the host formation or into the biosphere; another is the radiotoxicity flux integrated over time which takes into account the duration of the radionuclide releases.

In most recently published safety cases, e.g. [7.5], [7.6] and [7.8], results of dose calculations are presented for times up to 1 million years. However, in this time period no actinides are released from the host formation in case of disposal in granite or clay formations. Therefore, to show also contributions of actinides, results are presented in the following sections up to 10 million of years, although the significance of those results are strongly debatable as explained in the preceding paragraph.

The results of the assessments that are presented in the following sections are normalised per unit produced electricity ($TW_{el}h$) to allow for a comparison of the long-term consequences resulting from the different considered fuel cycles. In order to compare the calculated radiological consequences with a background dose or a dose constraint, it is necessary to multiply the normalised dose (in $Sv/a.TW_{el}h$) with the amount of electricity ($TW_{el}h$) that was generated from the HLW or spent fuel disposed of in the considered repository. For the repository in rock salt only the dose (not normalised to the produced energy) is presented.

[4] A critical group is a population living around a contaminated area in which each individual is supposed to be exposed to the maximum level of irradiation (via external irradiation, ingestion and inhalation) because of its way of living.

Results obtained for different repository concepts or host formations are deterministic, and cannot be compared with each other because the results of the assessments are subject to various types of uncertainty (were this work to form part of a safety case, a much more thorough analysis of uncertainty would be undertaken than is done herein). Some of the calculations presented are carried out for generic repository locations, others are site-specific and the level of characterisation of the repository systems can be very different. No attempt was made to harmonise the considered pathways to the biosphere, the treatment of uncertainty or the level of conservatism in the reported assessments.

Reprocessing of spent fuel also generates considerable amounts of intermediate-level waste (long lived ILW), which contains a number of mobile (in the geosphere) fission and activation products. Therefore, additional consequence analyses of disposal of long lived ILW have been made and the results are reported in Section 7.4.

7.2.1 Evaluations of the radiological impact for HLW disposal in granite (Spain)

The evaluations made by Enresa are based on the computer codes and parameter values that have been used for the assessment study of the Enresa reference disposal concept. The main code used is GoldSim (GoldSim Technology Group LLC). For the calculations it is assumed that the canisters will gradually fail between 1 300 and 10 000 years, that the HLW glass matrix corrodes at a constant rate during a 72 000 years lifetime (taken from the SAFIR 2 report [7.2]; for the spent fuels the corrosion rate is calculated by using an α-radiolysis dependent leaching model and this results in lifetimes of 10 million and 1 million years for respectively uranium oxide (UOX) and mixed oxide (MOX) spent fuel.

Figure 7.9 shows the doses per radionuclide for the reference fuel cycle A1 (UOX). ^{129}I controls the peak doses and the contribution of ^{135}Cs becomes significant after 1 million years. Doses due to actinides and daughters start only after several million years, because these radionuclides have a low solubility and are strongly sorbed on the bentonite buffer surrounding the waste package.

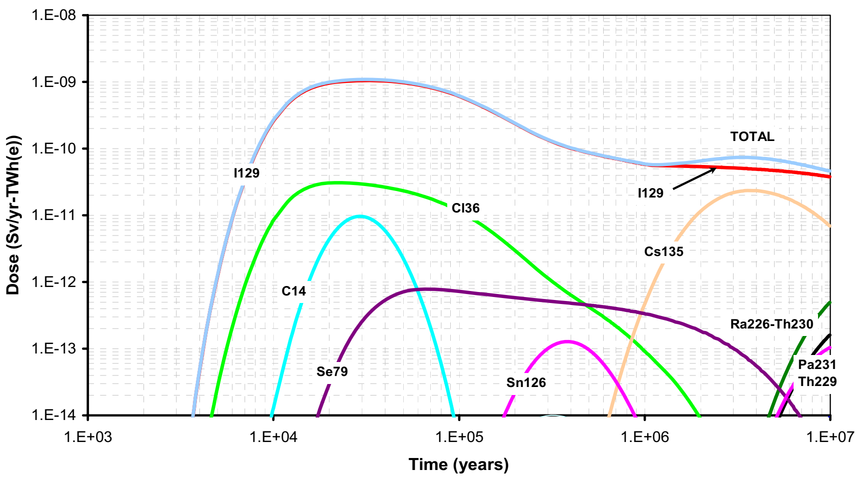

Figure 7.9: Total dose and main contributing radionuclides for fuel cycle scenario A1 (Spanish repository concept for disposal in granite)

The shape of the doses per radionuclide due to the disposal of MOX spent fuel from fuel cycle scenario A2 is similar to the one of scenario A1, but the values of the calculated doses are different. The difference in peak dose is a factor 3. Although the inventory of fission products (per TWhe) in the MOX fuel is roughly one order of magnitude smaller than in the UOX, the matrix alteration rate of MOX fuel calculated with the above mentioned model is 5 times greater than for UOX fuel, due to higher α activity.

Figure 7.10 shows the doses per radionuclide due the disposal of vitrified HLW from fuel cycle scenario A2. Since the inventory of ^{129}I is only 1% of the inventory in scenario A1, doses due to ^{129}I are much smaller than in the cases of spent fuel disposal. Doses due to the other fission products are larger than in scenario A1, because in the calculations the glass is assumed to dissolve completely in 72 000 years, while the spent fuel matrices last much longer. Peak doses due to ^{135}Cs and ^{129}I are about equal.

The doses per radionuclide for HLW from fuel cycles A3, B1 and B2 are similar to those shown in Figure 7.10, because the considered glass matrix degradation rate is the same for all scenarios, and there are only small differences in the inventory of fission products.

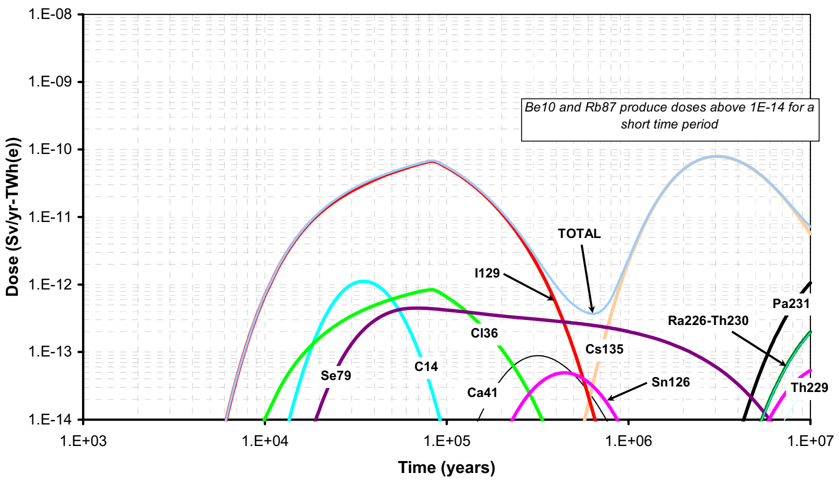

Figure 7.10: Dose per radionuclide due to vitrified wastes in scenario A2 (Spanish repository concept for disposal in granite)

Figure 7.11 shows the total doses due to HLW for the 5 analysed fuel cycle scenarios. The dose in case of fuel cycle A2 is the sum of doses due to the MOX spent fuel that controls the doses up to 800 000 years, and to the vitrified waste, which is dominant after 1 million years. The dose of fuel cycle B2 is the sum of the doses due to the 3 vitrified HLW types arising in this fuel cycle scenario.

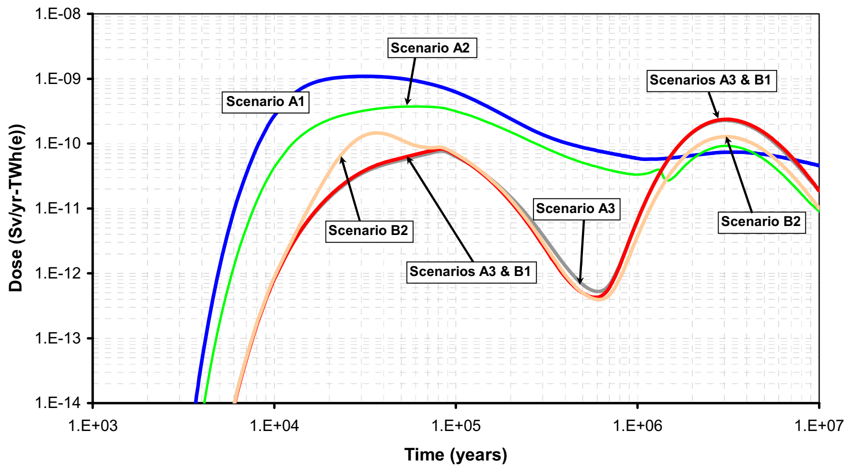

Figure 7.11: Normalised total doses due to HLW for the 5 considered fuel cycle scenarios (Spanish repository concept for disposal in granite)

The total dose curves for fuel cycle scenarios A3, B1 and B2 are very similar. This is a logical result because in the 3 scenarios the same glass matrix degradation rate is used and the inventories of fission products are quite similar. However, the maximum dose due to high level waste from fuel cycle scenarios A3 and B1 is about a factor 2 higher than the maximum dose from fuel cycles A2 or B2; this is due to the higher ^{135}Cs production in fast reactors than in light water reactors. The higher ^{135}Cs production can be explained by different fission yields of ^{235}U (in a light water reactor) and ^{239}Pu (in a fast reactor) and by spectrum effects (^{135}Xe decays to ^{135}Cs, but it can also capture a neutron; in a fast spectrum neutron capture by ^{135}Xe is lower and thus the ^{135}Cs production is higher).

In addition, the dose of fuel cycle B2 show a hump between 10 000 and 60 000 years due to ^{14}C, the inventory of which is much higher in fuel cycle B2 than in the others. The assumed use of nitride fuel (95% ^{15}N and 5% ^{14}N) in the ADS leads to a high ^{14}C inventory, generated by activation of ^{14}N; the generation of large amounts of ^{14}C can be avoided (if it is a problem for long-term safety) by selecting, e.g., oxide fuels.

Besides annual dose, the cumulative radiotoxicity fluxes to the biosphere have also been calculated in the project. Table 7.2 shows the cumulated radiotoxicity flux (in Sv/TWh(e)) released from the HLW repository into the biosphere during selected time periods for the 5 fuel cycle scenarios and Figure 7.12 shows the cumulative radiotoxicity fluxes as function of time.

In scenarios A3, B1 and B2 fluxes are controlled by ^{129}I up to 2 million years approximately, and ^{135}Cs in the long term. In the 2 scenarios with spent fuel (A1 and A2) cumulative radiotoxicity fluxes to the biosphere are greater than in the other 3 scenarios, and are due to the ^{129}I in the spent fuel (UOX or MOX).

Table 7.2: Cumulated radiotoxicity flux released into the biosphere [Sv/TW$_{el}$h].

Time period (years)	**A1**	**A2**	**Scenario** **A3**	**B1**	**B2**
10^3 to 10^4	0.2	0.0358	5.49 10^{-4}	5.34 10^{-4}	5.63 10^{-4}
10^4 to 10^5	43.6	16.0	2.4	2.6	2.8
10^5 to 10^6	62.2	36.0	2.4	2.2	2.3
10^6 to 10^7	234.8	18.2	26.6	26.5	14.4

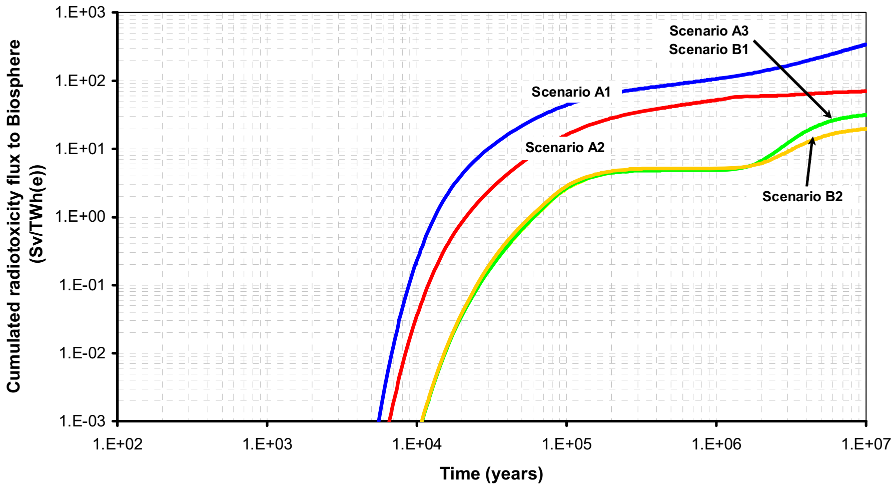

Figure 7.12: Cumulative radiotoxicity fluxes from the repository to the biosphere in the 5 scenarios. (Spanish repository concept for disposal in granite)

7.2.2 Evaluations of the radiological impact for HLW disposal in clay (Belgium)

The evaluations made by SCK•CEN are based on the computer codes and parameter values that have been used for the safety evaluations in the SAFIR 2 report [7.2].

Figure 7.14 shows the doses per radionuclide for the reference fuel cycle A1 (UOX). The total dose is dominated by ^{129}I. Smaller contributions are due to ^{126}Sn, ^{79}Se and ^{99}Tc. Doses due to actinides start only after several million years, because these radionuclides are strongly sorbed in the clay host formation (shorter lived actinides decay in the host clay formation).

The doses calculated for the disposal of the MOX spent fuel from fuel cycle A2 are comparable with those shown in Figure 7.13.

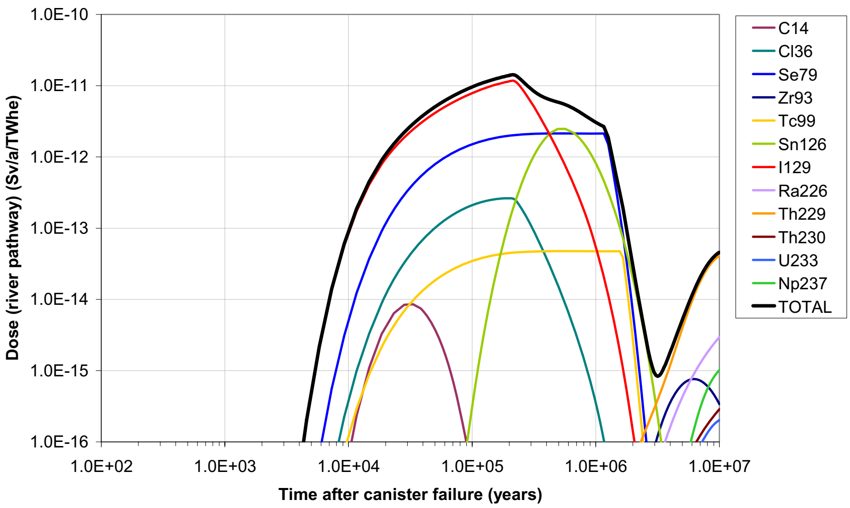

Figure 7.13: Total dose and its main contributors calculated for fuel cycle A1 (Belgian repository concept for disposal in clay)

When the spent fuels are reprocessed, which is the case for fuel cycles A2, A3, B1 and B2, a large fraction of the iodine escapes as gas during the fuel dissolution phase; in RED-IMPACT it is assumed that 1 % of the ^{129}I present in the spent fuel is transferred to the vitrified high level waste. For the vitrified high level wastes (see for example Figure 7.14, which shows the doses calculated for the HLW from fuel cycle A2) the calculated total doses are mainly due to ^{126}Sn and ^{79}Se; ^{129}I and ^{99}Tc give smaller contributions. Doses due to actinides start after a few million years and are always very small.

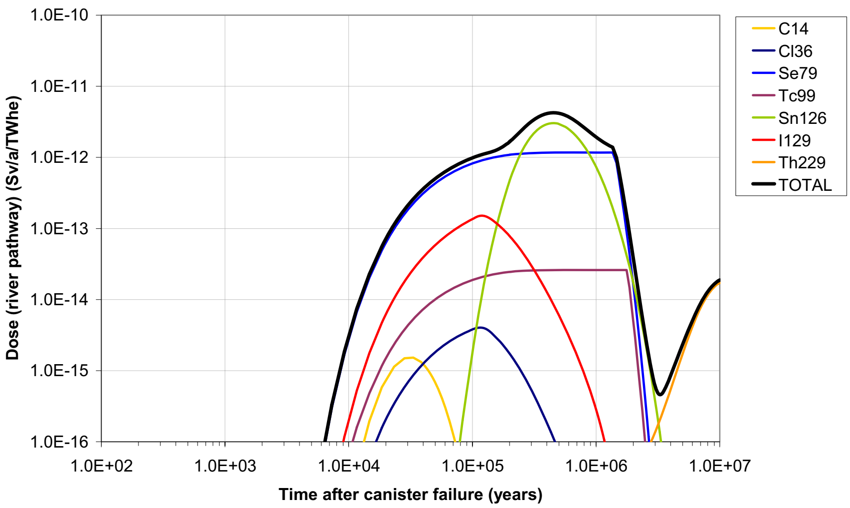

Figure 7.14: Total dose and its main contributors calculated for the HLW from fuel cycle A2 (Belgian repository concept for disposal in clay)

Figure 7.15 shows the total doses calculated for the 5 considered fuel cycle scenarios. The highest maximum dose is due to the reference fuel cycle A1 because in this case all the generated iodine is going into the geological repository. In the case of fuel cycle A2 a large fraction of the iodine was released from the high-level waste stream during reprocessing of the uranium oxide spent fuel, but the disposed MOX spent fuel still contains all the generated iodine. The maximum doses calculated for the three fuel cycles, for which only conditioned high level waste has to be disposed of (A3, B1 and B2) have a similar shape up to 3 million years. After 3 million years the doses due to the actinides depend somewhat on the amount of actinides present in the disposed waste. However, the actinide doses are for all fuel cycle scenarios more than 2 orders of magnitude lower than the maximum dose.

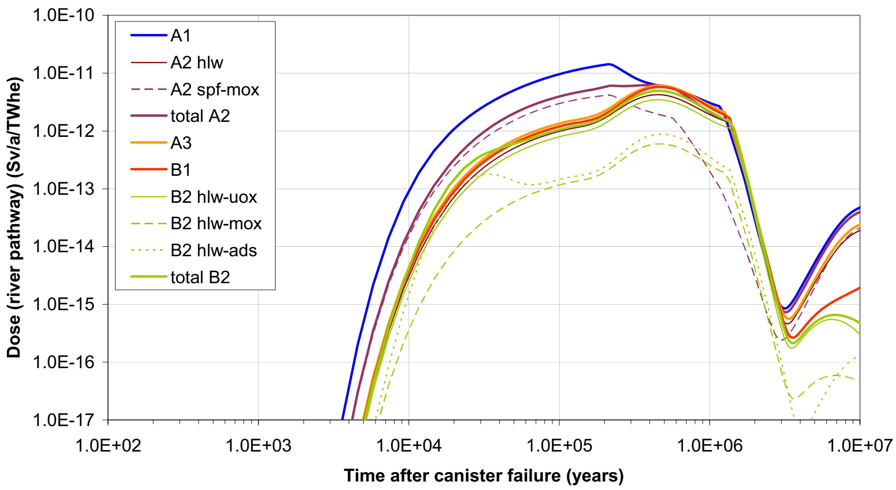

Figure 7.15: Total dose calculated for the 5 base fuel cycle scenarios (Belgian repository concept for disposal in clay)

7.2.3 Evaluations of the radiological impact for HLW disposal in clay (France)

The evaluations made by CEA are based on the computer codes and parameter values that have been used for the "Dossier 2005 Argile" report [7.5].

Figure 7.16 shows the doses per radionuclide for the reference fuel cycle A1 (UOX). The total dose is dominated by ^{129}I. Smaller contributions are due to ^{36}Cl, ^{79}Se and ^{41}Ca. The actinides are considered to be strongly sorbed (R > 10 000) by the clay minerals of the Callovo-Oxfordian formation and consequently the actinides remain confined in the host clay formation and are not released into the surrounding aquifers.

The doses calculated for the disposal of the MOX spent fuel from fuel cycle A2 are comparable with those shown in Figure 7.16.

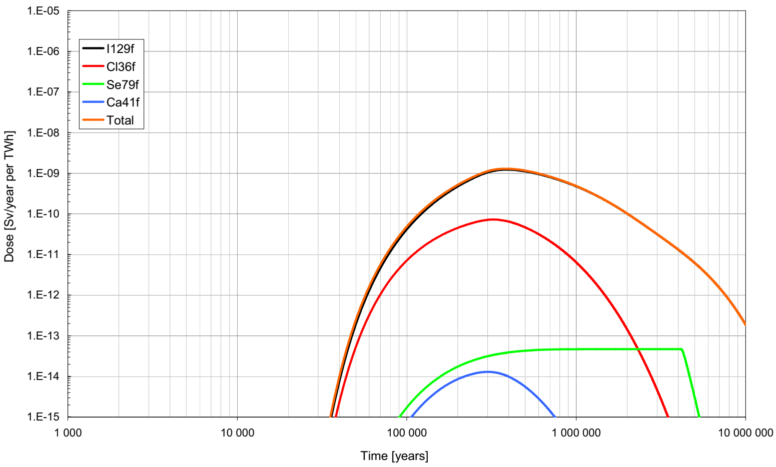

Figure 7.16: Total dose and its main contributors calculated for fuel cycle A1 (French repository concept for disposal in clay)

When the spent fuels are reprocessed, which is the case for the high level waste from fuel cycles A2, A3, B1 and B2, only a small fraction (1%) of the iodine is assumed to be transferred to the vitrified HLW. For the vitrified HLW (see for example Figure 7.17, which shows the doses calculated for the HLW from fuel cycle A2) the calculated total doses are still mainly due to ^{129}I and ^{36}Cl; ^{79}Se and ^{41}Ca give smaller contributions.

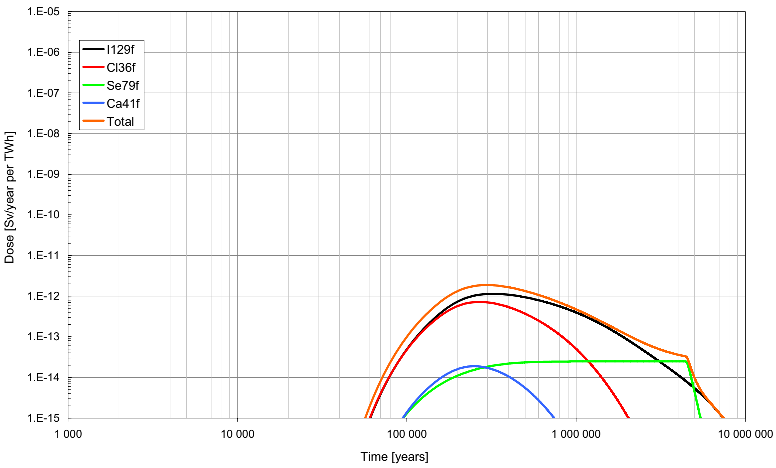

Figure 7.17: Total dose and its main contributors calculated for the disposal of HLW from fuel cycle A2 (French repository concept for disposal in clay)

Figure 7.18 shows the total doses calculated for the 5 considered fuel cycle scenarios. The highest maximum dose is due to the reference fuel cycle A1 because all the generated iodine is going into the geological repository. In the case of scenario A2, a large fraction of the iodine was released from the HLW during reprocessing of the uranium oxide spent fuel, but the disposed MOX spent fuel still contains all the generated iodine; the maximum dose is reduced with a factor 10. In the case of fuel cycle scenarios A3, B1 and B2, for which only conditioned HLW has to be disposed of, the maximum dose is reduced with a factor 100 in comparison with the dose from the reference fuel cycle A; this reduction is a consequence of the reduction of the ^{129}I inventory during reprocessing.

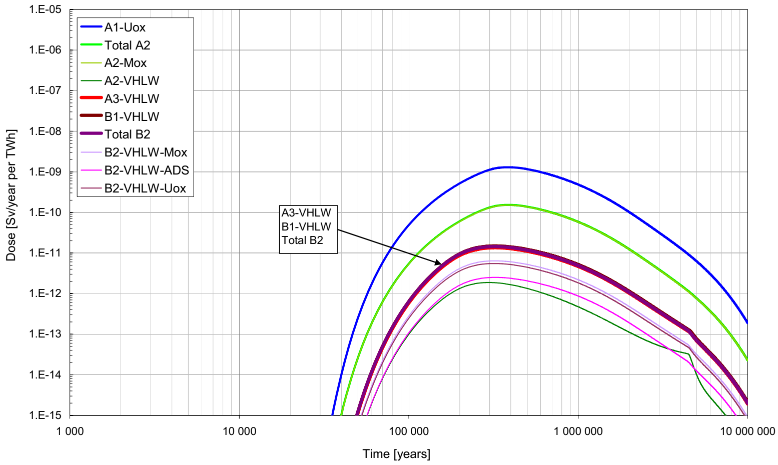

Figure 7.18: Total dose calculated for the 5 base fuel cycle scenarios (French repository concept for disposal in clay)

7.2.4 Evaluations of the radiological impact for HLW disposal in salt (Germany)

The evaluations made by GRS are made for an altered evolution scenario considering the intrusion of groundwater into the disposal facility via an anhydrite vein. The evaluations are based on the computer codes and parameter values that have been used by GRS for the assessment of spent fuel disposal in a salt formation within the framework of the SPA project of the European Commission [7.7].

The results of the performance assessment calculations for disposal facilities in salt for spent fuel from fuel cycles A1 and for vitrified HLW from fuel cycle B1 are shown in Figures 7.19 and 7.20. In case of disposal of spent fuel from fuel cycle A1, the main contributors to the total dose are ^{135}Cs, ^{129}I, ^{226}Ra (from the U decay chain 4N + 2) and ^{229}Th (from the Np decay chain 4N + 1). In case of disposal of HLW from fuel cycle B1, the main contributors to the total dose are ^{135}Cs, ^{79}Se and ^{226}Ra (from the U decay chain 4N + 2); the contribution of the Np decay chain (4N + 1) is here much smaller than in the case of scenario A1.

The total doses calculated for the 5 base fuel cycle scenarios are shown in Figure 7.21; the difference between the five curves is at any time smaller than two orders of magnitude (notice that the calculated doses are here not normalised to the produced energy).

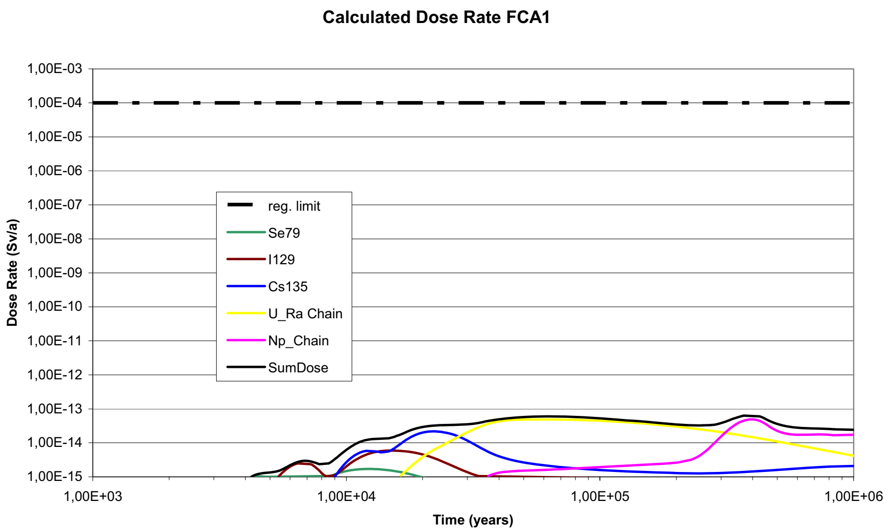

Figure 7.19: Calculated dose for spent fuel from fuel cycle A1 (German repository concept for disposal in salt)

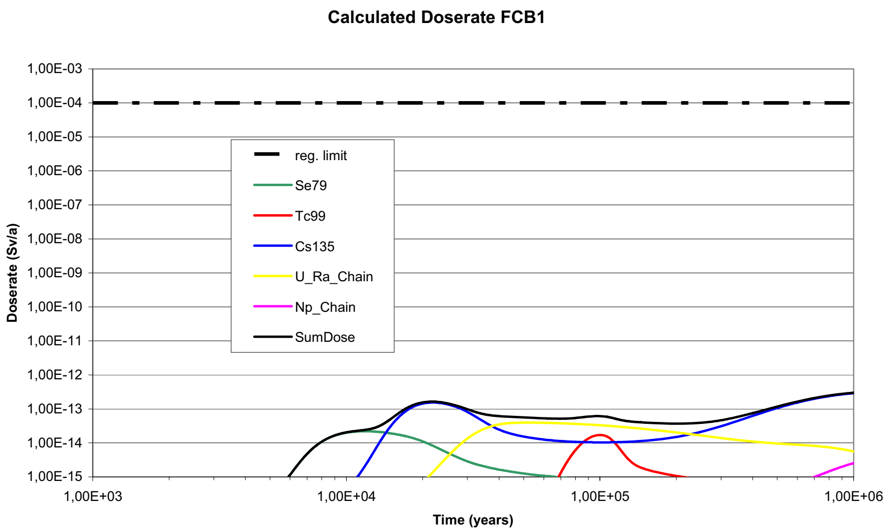

Figure 7.20: Calculated dose for HLW from fuel cycle B1 (German repository concept for disposal in salt)

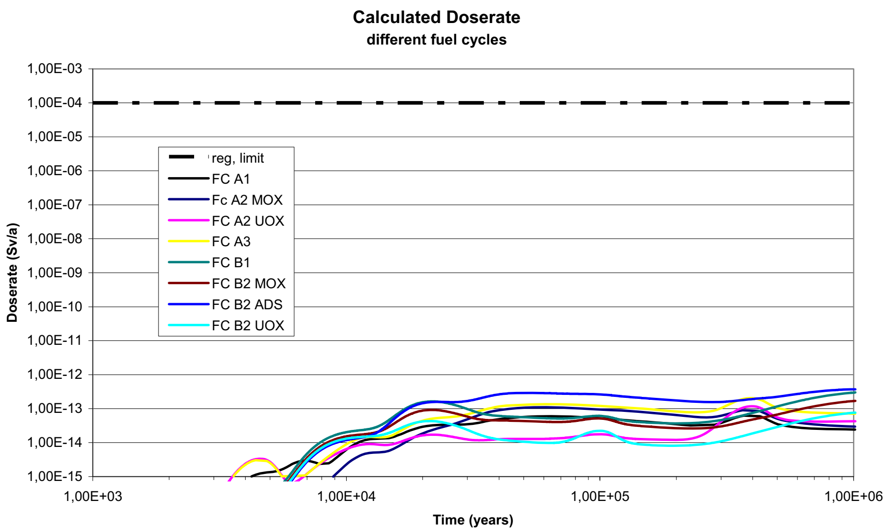

Figure 7.21: Total dose calculated for the 5 base fuel cycle scenarios (German repository concept for disposal in salt)

The calculated dose rates are extremely low. The maximum is about eight orders of magnitude below the regulatory limit of 0.3 mSv/a, which is applicable in Germany. It can be concluded that the differences in the calculated dose curves for the performance assessment in a rock salt formation are negligible. This is all the more the case because an altered evolution scenario was considered – the normal evolution would result in zero release for all fuel cycles anyway. Differences occur with respect to the number of disposal casks per drift and, thus, the capacity of the repository. For the evaluation of the calculation results, it should be borne in mind that the disposal facility is not optimised for waste with fast decreasing thermal load.

It must be remembered that the convergence rate of the salt is a strongly influential parameter in the assessment calculations of the brine intrusion scenario and that, as a consequence, a long-lasting thermal output has a positive effect on the convergence of the salt and, as a consequence, on calculated doses.

7.2.5 Calculations made for variant scenarios and sensitivity studies

The basic fuel cycle scenarios presented in the previous sections have been complemented with a number of variant scenarios and sensitivity studies. Whereas all the main participants in WP4 have carried out calculations for the 5 base fuel cycle scenarios, only two of them (Enresa for disposal in granite and SCK•CEN for disposal in clay) have made calculations for the variant scenarios and sensitivity studies. The reference fuel cycle scenario used for the sensitivity studies is scenario B1, because it is an advance fuel cycle scenario; the results obtained for the B1 variants are easily transferable to other fuel cycle scenarios with recycling of Pu and minor actinides.

a) Sensitivity of the calculated doses to a stable waste matrix

For the analyses of the base scenarios, a lifetime of the glass matrix of respectively 72 000 and 100 000 years has been assumed by respectively Enresa for disposal in granite and SCK•CEN for disposal in clay. For future fuel cycles it is possible that waste matrices will be used that are much more stable than the borosilicate glass, which is currently used by AREVA and BNFL for conditioning the HLW arising from reprocessing of spent fuel. An example of such a stable matrix is synrock, for which lifetimes of 1 million years or more seem to be realistic.

In order to evaluate if the use of waste matrices much more stable than borosilicate glass matrices, has an impact on the calculated dose rates, Enresa and SCK•CEN have made calculations for longer matrix lifetimes.

Enresa considered matrix lifetimes of 720 000 and 7.2 million years, whereas their reference values is 72 000 years. Figure 7.22 shows the doses calculated for the main radionuclides for the 3 considered matrix lifetimes. This figure clearly shows that the matrix lifetime has a strong effect on doses due to the mobile fission and activation products such as ^{129}I, ^{14}C and ^{36}Cl. The effect is rather limited on retarded radionuclides such as ^{135}Cs and about negligible for solubility limited radionuclides such as ^{79}Se, ^{126}Sn and the actinides.

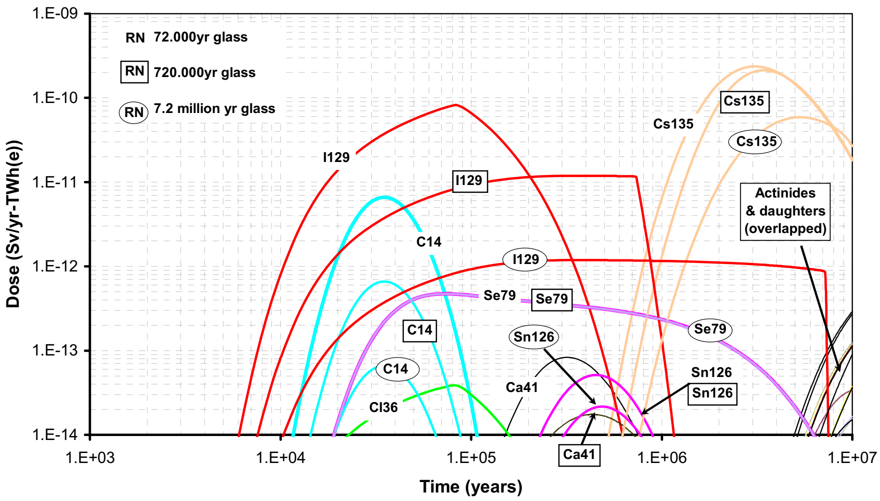

Figure 7.22: Effect of the lifetime of the glass matrix on the doses calculated for scenario B1 for a repository in granite.

SCK•CEN considered a matrix lifetime of 1 million years, whereas the reference values is 100 000 years. Figure 7.23 shows the doses calculated for the main radionuclides for the 2 considered matrix lifetimes.

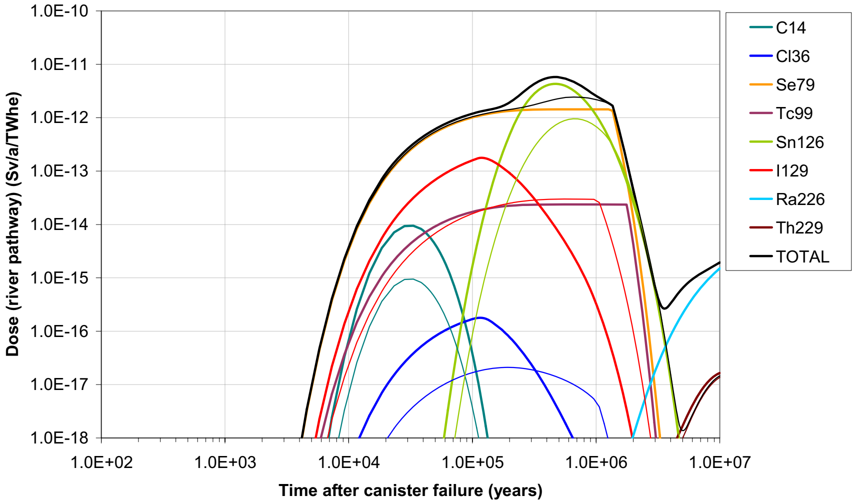

Figure 7.23: Comparison of the contributions to the total doses via the river pathway for scenario B1 (matrix lifetime 100 000 years; shown in thick lines) and its variant considering a more stable waste matrix (lifetime 1 million years; shown in thin lines) for a repository in clay.

Figure 7.23 shows that the effect of the longer matrix lifetime is very similar for disposal in granite and clay. The matrix lifetime has a strong effect on doses due to the mobile fission and activation products such as ^{129}I, ^{14}C and ^{36}Cl. The effect is rather limited on retarded radionuclides such as ^{126}Sn and about negligible for solubility limited radionuclides such as ^{79}Se, ^{99}Tc and the actinides.

b) Variant assuming the separation of Cs and Sr from the HLW

Two variants of scenario B1 are analysed:

- variant B1.4: Cs and Sr are separated during reprocessing and 3 different HLW waste packages (WP) are generated (60 kg/WP of Cs, 60 kg/WP of Sr and 60 kg/WP of the remaining fission products and actinides);
- variant B1.5: only Sr is separated during reprocessing and 2 different HLW waste packages are generated (60 kg/WP of Sr and 60 kg/WP of the remaining fission products and actinides).

The reduced thermal power (see 7.1.5) of the resulting HLW in these variants could allow a more compact repository. But when the doses are dominated by highly soluble radionuclides (^{129}I and ^{135}Cs), as it is the case in granite, any potential optimization of the repository layout does not necessarily lead to any decrease in these doses. This is illustrated in Figure 7.24. The fist hump of the doses is due to ^{129}I while the second one is caused by ^{135}Cs.

Total inventories of ^{129}I and ^{135}Cs to be disposed in the repository are the same in all the cases, although distribution between waste packages is different. Since iodine and caesium are highly soluble species their doses are independent of the particular distribution between different waste packages and doses do not change. Only in the valley between the two humps a difference can be observed, because in that period ^{79}Se contribution is relevant, and Se is a solubility controlled element.

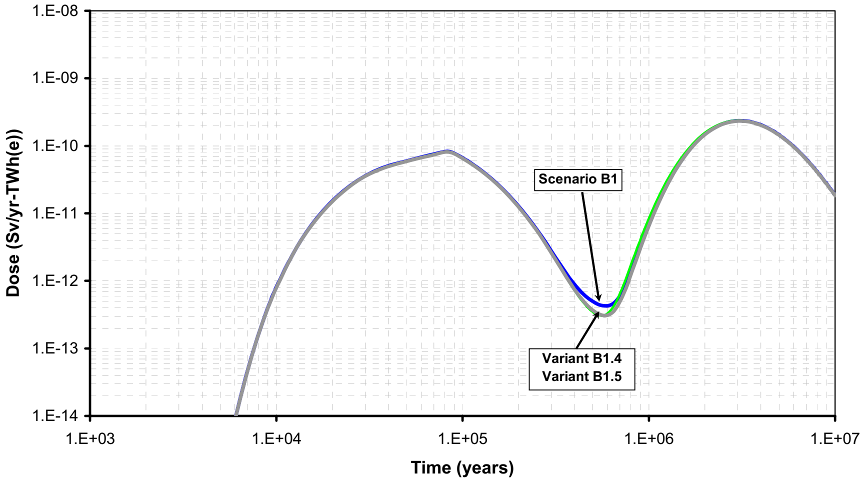

Figure 7.24: Total dose calculated for scenario B1 and its variants considering the separation of Sr (variant B1.5) and of Cs and Sr (variant B1.4) (Spanish repository concept for disposal in granite)

SCK•CEN has adapted the dimensions of the disposal galleries to the thermal output of the new waste types. Therefore, it made use of repository concepts developed within the Belgian radioactive waste management programme for the disposal of long lived ILW in clay formations. In this concept it is assumed that 19 canisters are placed in a gallery section. The use of such a dense repository configuration has a significant influence on the migration of solubility limited radionuclides, and, as a consequence, on the doses calculated for those radionuclides. Figure 7.25 shows the total doses calculated for scenario B1 and its variants B1.4 and B1.5. This figure shows that there is a small impact of the separation of Sr only on the total dose, but there is a significant impact, a factor 9, in case of separation of both Cs and Sr. Figure 7.26 shows the contributions of the main radionuclides to the total dose. This figure illustrates that in case of fuel cycle scenario B1 the main radionuclides are ^{79}Se and ^{126}Sn, which are both solubility limited; consequently, the total dose is significantly reduced when implementing separation of Cs and Sr. Figure 7.26 shows also that the dose due to non-solubility limited radionuclides ^{129}I, ^{14}C and ^{36}Cl is not influenced by the denser disposal configuration.

It can be concluded that for repositories for which the total doses are mainly due to solubility-controlled radionuclides (such as the Belgian repository in clay for HLW from the B1 fuel cycle) Sr and/or Cs separation can lead to a reduction in total doses. However, when the total doses are mainly due to highly soluble radionuclides (such as the Spanish repository in granite for HLW from the B1 fuel cycle) Sr and/or Cs separation has no significant effect on total doses.

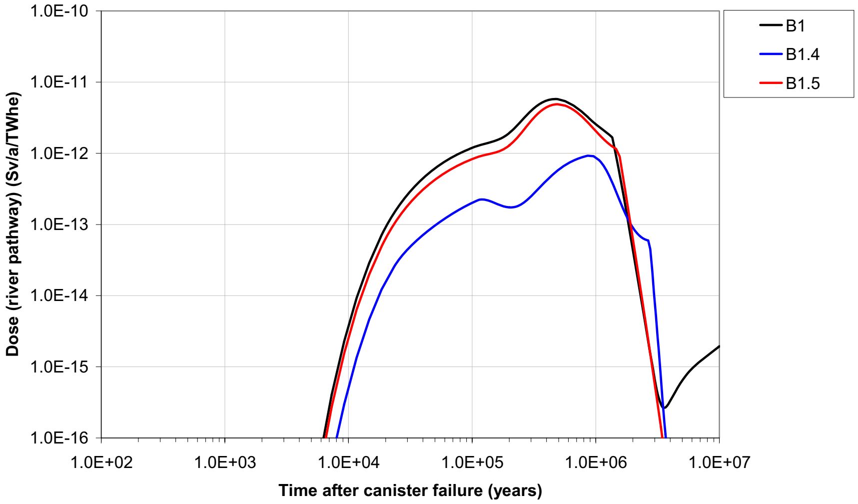

Figure 7.25: Total doses via the river pathway per TWh(e) calculated for the scenario B1 and its variants considering the separation of Sr (B1.5) and of Cs and Sr (B1.4) (Belgian repository concept for disposal in clay)

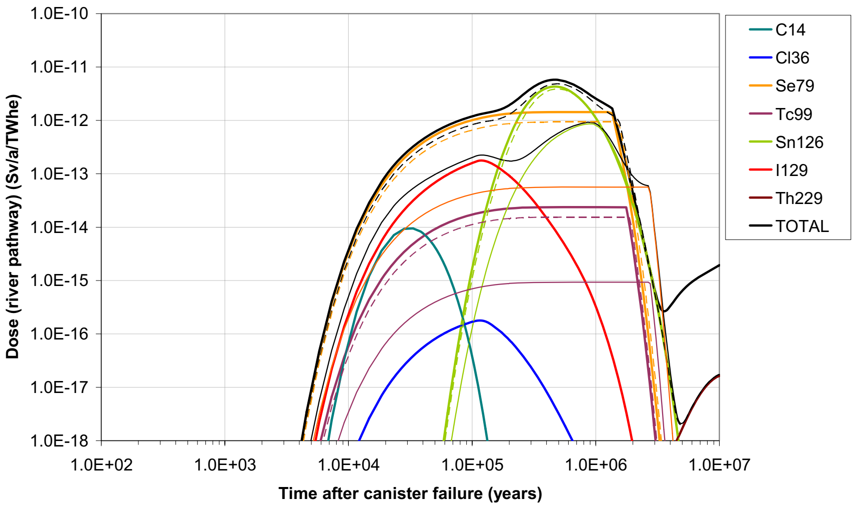

Figure 7.26: Contributions of the main radionuclides to the total doses via the river pathway per TWh(e) calculated for the scenario B1 (thick continuous line) and its variants B1.4 (thin continuous line) and B1.5 (thin dashed line) (Belgian repository concept for disposal in clay)

c) Variant assuming that iodine is captured at the reprocessing plant

During reprocessing of spent fuel, iodine escapes in the dissolver from the liquid HLW as a gas. At present, the reprocessing plants of Areva at La Hague (F) and BNFL at Sellafield (UK) discharge iodine into the sea. In future reprocessing plants iodine might be captured, conditioned in a stable matrix and also disposed of in a geological repository.

Calculations are performed for scenario B1 assuming that 98% of the ^{129}I present in the spent fuel is captured and conditioned in a stable matrix (1% is assumed to pass to HLW and 1% to long lived ILW). For the calculations, waste matrix lifetimes equal to 1/10, 1, 10 and 100 times the reference lifetime of a glass matrix are considered. It has to be noticed that at present no matrices for the conditioning of iodine are available that can guarantee such long lifetimes.

The ^{129}I doses calculated by Enresa for the 4 considered matrix lifetimes for a repository in granite are given in Figure 7.27. The ^{129}I doses calculated by SCK•CEN for the 4 considered matrix lifetimes for a repository in clay are given in Figure 7.28.

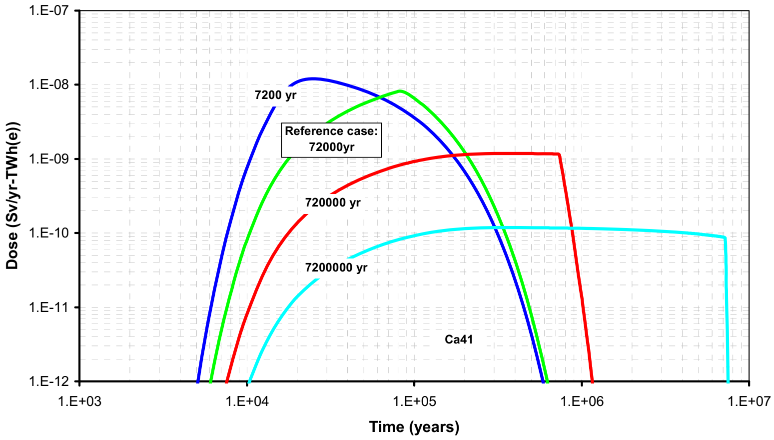

Figure 7.27: ^{129}I doses calculated for different waste matrix lifetimes for a repository in granite (scenario B1, assuming capture of iodine at reprocessing plant).

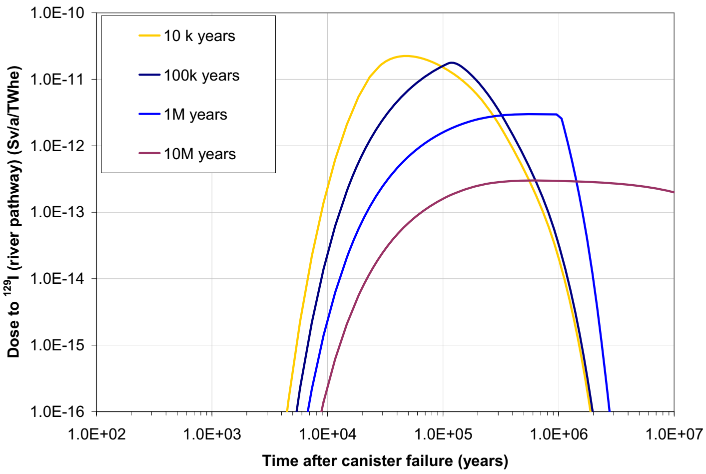

Figure 7.28: ^{129}I doses calculated for different waste matrix lifetimes for a repository in clay (scenario B1, assuming capture of iodine at reprocessing plant).

Figures 7.27 and 7.28 show that if the waste matrix lifetime is shorter than 72 000 or 100 000 years, e.g. 10 000 years, releases into the environment occur earlier, but the increase of the maximum dose is limited (just a factor of about 1.25). If extremely stable waste matrices can be developed, this can lead to a considerable reduction of the I-129 dose: a factor 6 for a matrix lifetime of 1 million years and a factor 60 for a matrix lifetime of 10 million years.

7.2.6 Conclusions concerning the radiological impact of disposal of HLW generated in advanced fuel cycles

For all considered host formations, the impact of advanced fuel cycles on the maximum dose resulting from the disposal of the high level radioactive waste generated in the considered fuel cycle scenarios is limited, because the maximum dose is essentially due to long-lived fission products. The amount of produced fission products is almost proportional to the thermal energy generated in the reactors; the higher thermodynamic efficiency of the heat to electricity conversion in the power plants and the different neutron spectrum in case of fast reactors can have a limited influence on the generated amount of fission products. One of the most important contributors to the total dose is ^{129}I. The amount of ^{129}I going into the repository as HLW very strongly depends on the fraction of spent fuel that is reprocessed.

For those long-lived fission products that pass to the HLW during reprocessing (^{79}Se, ^{126}Sn and ^{135}Cs) doses arising from the different scenarios are quite similar. For ^{129}I and ^{14}C and ^{36}Cl (activation products) the inventories in the HLW are much smaller than in the original fuel, because large fractions of those elements have been released as effluents. The decrease in doses due to ^{129}I, ^{14}C and ^{36}Cl in the scenarios with reprocessing is a consequence of the reduced inventory in the HLW; however, a significant amount of these radionuclides has been released into the environment during reprocessing.

The transmutation of most actinides in fast reactors or accelerator-driven systems in case of advanced fuel cycle scenarios has little impact on the resulting doses, because the low solubility of the actinides in reducing conditions and the strong sorption on minerals present in the buffer and host formation make that only extremely small amounts of actinides are released from the host formation into the surrounding aquifers after a few millions of years.

When the waste disposal configuration is adapted to the thermal output of the disposed waste, the higher disposal density in case of advanced fuel cycles can result in a decrease of the release rate of solubility limited radionuclides in case of disposal in argillaceous or crystalline formations, but there is no effect on highly soluble radionuclides.

7.3 Impact of advanced fuel cycle scenarios on the consequences of human intrusion into the geological HLW repositories

For the evaluations made in section 7.2 for disposal in granite and clay formations, it was assumed that the main engineered and natural barriers of the repository system function as expected. In the scenario considered for disposal in rock salt, failure of the main natural barrier is assumed. As the (remaining) other barriers of the disposal system are still assumed to perform as intended, this means that a large proportion of the most radiotoxic radionuclides present in the disposed waste will decay to negligible levels within the disposal system and only a limited number of mobile fission and activation products will reach the aquifers surrounding the host formation.

In the scenario analyses that are made for the safety assessments a number of disruptive scenarios (i.e. scenarios that can bring the disposed waste in contact with man) can be identified. It should be noticed that these scenarios have a very low probability of occurrence. An important group of disruptive scenarios are future human intrusions into the repository. The occurrence of human intrusion scenarios cannot be ruled out; the scenario assumes that information about the existence of the repository has been lost.

Three groups of human intrusion scenarios can be distinguished generally:

- doses to intruders: these are doses to geotechnical workers drilling a borehole or to geologists who are examining borehole cores containing fragments of the disposed waste;
- doses to residents: when a borehole is drilled above a sealed repository the cuttings are sometimes left at the surface of the drilling site; it is assumed that these cuttings contain fragments of the disposed waste;
- destruction of a number of barriers leading to an increased release of radionuclides via the groundwater pathway.

Recent performance assessments of HLW repositories essentially focus on the resilience of the repository system, i.e. the ability of the repository system to perform its main functions after having been affected by a borehole (the third group of human intrusion scenarios). In most performance assessments doses to the intruder are not calculated.

It is widely accepted that protection from exposures associated with human intrusion is best accomplished by efforts to reduce the possibility of such events, which may include sitting a disposal facility at greater depth and away from natural resources, incorporating robust design features to make intrusion more difficult or using active or passive institutional control.

The possibility of elevated exposures to intruders is an unavoidable consequence of the high concentration of radionuclides in the waste. Reduction of dose to an intruder by diluting the waste prior to disposal is not a viable option.

In this section intruder doses will be calculated because they represent an extreme case for which the effect of radiotoxicity reduction can be illustrated. It has to be remembered that the probability of occurrence of such scenario is considered to be very low, because repository sites are selected in areas that do not contain exploitable resources. The realism of the considered intrusion scenarios is also strongly debatable: on the one hand they assume the availability of advanced drilling techniques; on the other hand they assume that the presence of radioactive waste materials is not established.

7.3.1 Evaluations of doses to a geotechnical worker

In the geotechnical worker scenario it as assumed that a core from exploratory drilling is subjected to laboratory analysis by a geotechnical worker (although some difficulty might be expected in successfully coring the waste because of the presence of a thick metallic container). Such workers constitute the potentially exposed group. A number of activities may occur during laboratory analysis of core material that give rise to exposure, for example, examination procedures require the grinding or cutting of samples. These operations may raise substantial amounts of dust, leading to inhalation hazards. Specimen handling may also give rise to external irradiation and ingestion hazards.

The considered exposure pathways are:

- external exposure, from short-term working in close proximity to core samples, and from longer-term, more distant irradiation while materials are stored in the laboratory;
- ingestion, as a result of handling samples leading to activity being taken into the body orally from contamination of the hands;
- inhalation, of respirable dusts, generated as a result of laboratory analysis techniques, and of radon, associated with the decay of radium-226 that may be present in core samples stored in the laboratory.

Results are presented in Figure 7.29 for intrusion into waste derived from fuel cycle scenario A1 (one canister contains 4 UOX spent fuel assemblies). A total dose versus time is presented, as well as dose versus time for specific key radionuclides for this intrusion scenario. It is shown that the dose to geotechnical workers examining reduces if the human intrusion event occurs at later times. This is a typical result in human intrusion assessment calculations, and is due to the effect of radioactive decay reducing the radionuclide inventory. The maximum radiological doses are therefore associated with an intrusion event occurring immediately after institutional controls on the repository site have been lifted. This is conservatively assumed to occur 100 years after closure of the repository. The total peak individual dose is calculated to be 21 Sv. If an intrusion event does not occur until 10,000 years after repository closure, the total calculated dose falls to 1.6 Sv. For an intrusion occurring at 100,000 years, the total dose is calculated to be 0.9 Sv.

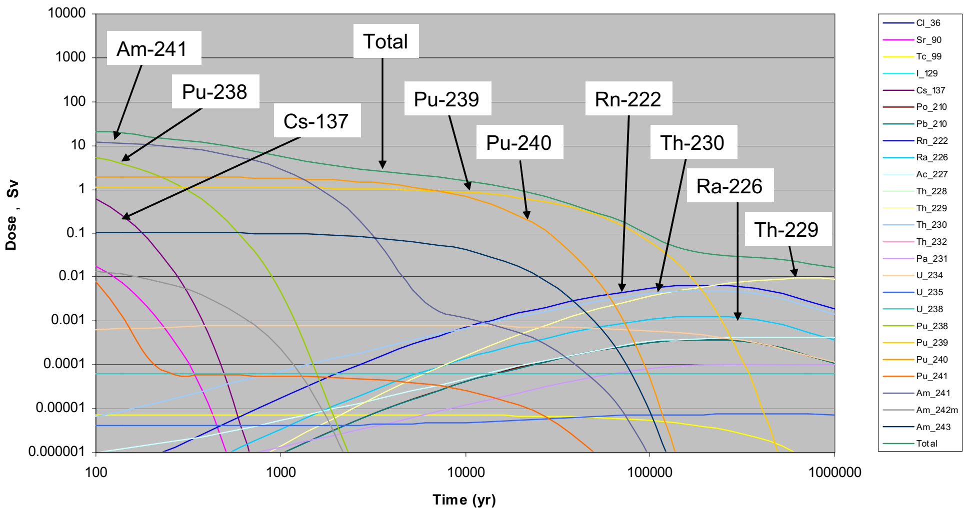

Figure 7.29: Calculated annual dose in the geotechnical worker scenario, for scenario A1 (1 canister contains 4 UOX spent fuel assemblies)

Figure 7.30 presents total dose versus time for all eight high-level waste and spent fuel types. Also shown is the calculated dose assuming that intrusion into a uranium rich ore body has occurred (the Cigar Lake uranium ore body is considered here; the rock contains 4.510^4 Bq/kg of ^{235}U and 10^6 Bq/kg of ^{238}U [7.8]). Additionally, ICRP intervention levels of 10 mSv and 100 mSv (from [7.9]) are shown.

As can be seen, calculated total dose varies considerably between HLW and spent fuel types, by up to three orders of magnitude at any one time. The MOX spent fuel from fuel cycle scenario A2 has the highest calculated total dose at all times, whereas the vitrified HLW from UOX fuel reprocessing in fuel cycle scenario B2 has the lowest calculated total dose at all times.

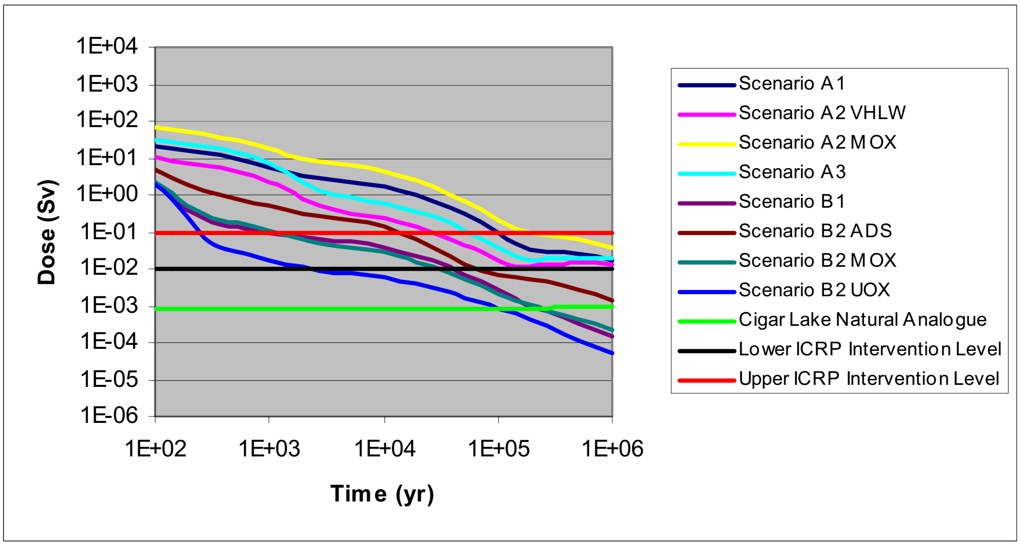

Figure 7.30: Comparison of calculated annual dose in the geotechnical worker scenario, for eight high-level waste and spent fuel types and for Cigar Lake Uranium ore body and ICRP intervention levels

Consideration of the calculated total dose associated with the Cigar Lake uranium ore body and with the ICRP intervention levels can be used to contextualise the dose values calculated for the eight high-level waste and spent fuel types. Applying a simple criterion of ranking when the calculated dose for each high-level waste and spent fuel type is bounded by that calculated from the Cigar Lake uranium ore body and ICRP intervention levels allows a 'ranking' of the high-level waste and spent fuel types to be undertaken. Table 7.3 gives the estimated 'required isolation times' for the most active HLW type arising from each of the 5 base fuel cycle scenarios. In order to verify whether the radiotoxicity present in the disposed waste can be used as an alternative indicator, Table 7.3 also gives the time after which the radiotoxicity in the disposed waste drops under the radiotoxicity in the fresh uranium required for the fuel production in fuel cycle scenario A1.

Comparison of the times obtained in Table 7.3 from the 100 mSv intervention levels with the times derived from the radiotoxicity shows that these times are often of the same order of magnitude. However, in case of scenarios generating different HLW types, waste types can occur in which a lot of radioactivity is concentrated, but of which only a very small number of waste packages are generated; this the case for the MOX spent fuel in scenario A2 and the vitrified HLW arising from the pyro-reprocessing of ADS spent fuel in scenario B2.

Table 7.3: Ranking of high-level waste and spent fuel types, on basis of calculated dose from geotechnical worker scenario in comparison with calculated dose from intrusion into Cigar Lake uranium ore body and ICRP intervention levels; for comparison the time after which the radiotoxicity in the disposed waste drops under the radiotoxicity present in the fresh uranium required for the fuel production of fuel cycle scenario A1 is also given.

Comparator / **HLW/SF Types**	**Cigar Lake natural analogue**	**ICRP 10 mSv intervention level**	**ICRP 100 mSv intervention level**	**Radiotoxicity**
Scenario B1: HLW	~200,000 a	~40,000 a	~1000 a	~300 a
Scenario B2: HLW from ADS fuel	> 1 Ma	~70,000 a	~13,000 a	~300 a
Scenario A3: HLW	> 1 Ma	> 1 Ma	~70,000 a	~24,000 a
Scenario A1: spent UOX fuel	> 1 Ma	> 1 Ma	~100,000 a	~200,000 a
Scenario A2: spent MOX fuel	> 1 Ma	> 1 Ma	~200,000 a	~90,000 a

7.3.2 Conclusions concerning the impact of human intrusion

Intrusion into a repository, in which spent fuels (UOX or MOX) have been disposed of, can result in a significant dose to a geotechnical worker during a few hundreds of thousands of years. Reprocessing of the spent fuel with recuperation of U and Pu only (scenarios A2 and A3) gives rise to vitrified HLW that is less radiotoxic and the period during which significant doses can occur is reduced to a few tens of thousands of years. When all the actinides are recycled in a fast reactor (scenario B1) or an ADS (scenario B2), the radiotoxicity in the waste is considerably reduced and the period during which significant doses can occur is limited to a few hundreds or thousands of years. However, concentrating the remaining radioactivity in a small number of waste packages sometimes masks the effect of the reduction of the total radiotoxicity in case of advanced fuel cycle scenarios.

7.4 Impact of advanced fuel cycle scenarios on the long-term radiological consequences of long lived ILW disposal in geological repositories

The preceding sections dealt only with the various HLW types arising from the different fuel cycle scenarios. However, the fuel cycles generate also different types of long lived ILW packages types. The estimated number and inventories of the long lived ILW packages have been documented in Chapter 6 of this synthesis report. Although the radiotoxicity of the long lived ILW is significantly lower than the one of the HLW, the long lived ILW packages contain non-negligible amounts of mobile fission and activation products; consequently they can give rise to doses comparable with those arising from the disposal of HLW. However, it should be noticed that the activation products, such as ^{14}C and ^{36}Cl, mainly result from the activation of impurities in the fuel or cladding. The considered concentrations of the impurities were conservative upper limits. It can also be expected that low activation materials will be developed for new reactor types such as fast neutron reactors. For the fission

product ^{129}I a "nominal" value of 1% was used, whereas, measurements indicate that the amount of ^{129}I present in the glass is about 0.2%. Consequently, the activities of a number of mobile fission and activation products in the considered long lived ILW types have to be considered as very conservative estimates.
It has also to be noticed that the repository concepts considered by NRI and Enresa were not specifically designed for the disposal of long lived ILW.

Three organisations have made evaluations of long lived ILW disposal:
- NRI and Enresa for disposal in granite;
- SCK•CEN for disposal in clay.

7.4.1 Evaluations of the radiological impact of long lived ILW disposal in case of disposal in granite (Czech Republic)

For the evaluations it was considered that all soluble fission products are immediately released from the long lived ILW after canister failure (2000 years) and contact of waste with water; radionuclides activated in structure materials (e.g. ^{14}C, ^{36}Cl, ^{94}Nb or ^{59}Ni) are released congruently with degradation of structure materials (lifetime of 10 000 years). It was assumed that a direct pathway exists from the repository to the biosphere with a travel time of 10 years only.

Figure 7.31 shows the doses (corresponding to the generation of 1 TWh(e)) arising from the disposal of 2.214 canisters with long lived ILW arising from fuel cycle scenario A2. It can be seen that the soluble radionuclides, such as ^{129}I, ^{14}C and ^{36}Cl play the most important role; follows ^{93}Mo, ^{79}Se and ^{126}Sn limited by solubility, but having very low distribution coefficients in granite host rock. The high doses from ^{93}Mo are presumably caused by a very low distribution coefficient for granite used for calculations due to the lack of more realistic data. In further period ^{59}Ni and ^{94}Nb from activated structure materials dominate biosphere doses. Only very small contribution is from actinides and their daughter products radionuclides (^{226}Ra, ^{237}Np and ^{229}Th) after about 300 000 years after disposal.

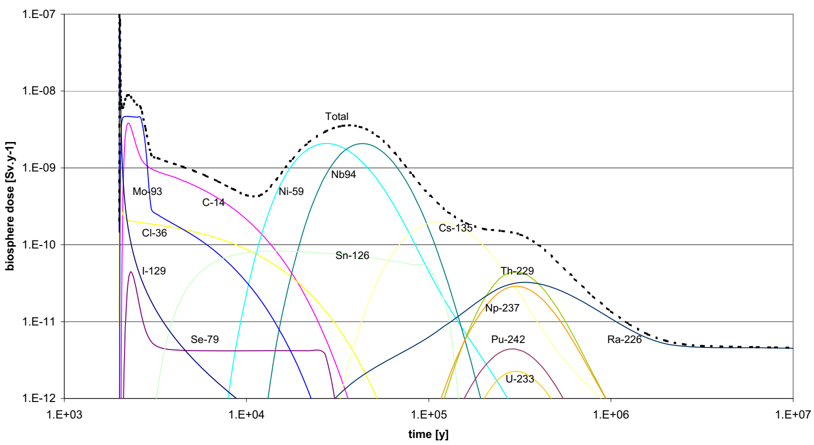

Figure 7.31: Doses due to disposal of long lived ILW from fuel cycle scenario A2 (Czech repository concept for disposal in granite)

In Figure 7.32, doses from all main waste streams coming from the A2 scenario (MOX spent fuel, HLW vitrified waste and long lived ILW) are compared. It can be seen that that the long lived ILW from A2 scenario has an important impact on the doses, particularly in time until 10 000 years after waste emplacement mainly due to contribution from ^{129}I and ^{93}Mo.

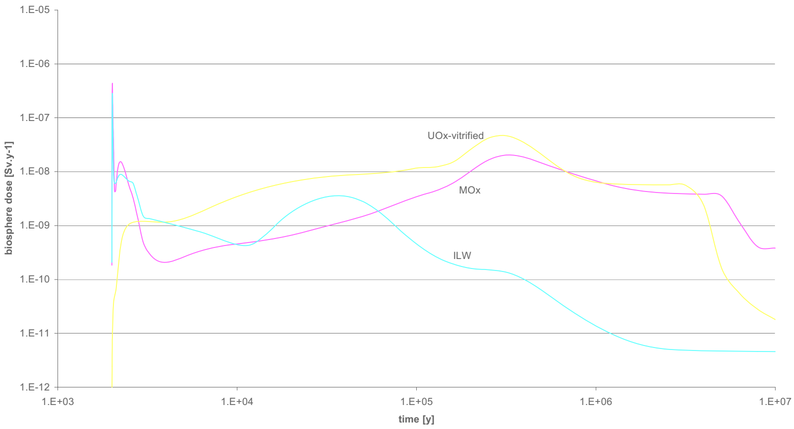

Figure 7.32: Comparison of doses arising from the main waste types generated in scenario A2 (Czech repository concept for disposal in granite)

The doses calculated for the long lived ILW from scenarios A3 and B1 are rather similar to those calculated for scenario A2 with the exception of ^{14}C, which occurs in much larger amounts in case of the use of fast reactors in the considered fuel cycle.

Four different long lived ILW types arise from scenario B2. The long lived ILW doses are compared with the HLW doses in Figure 7.33.

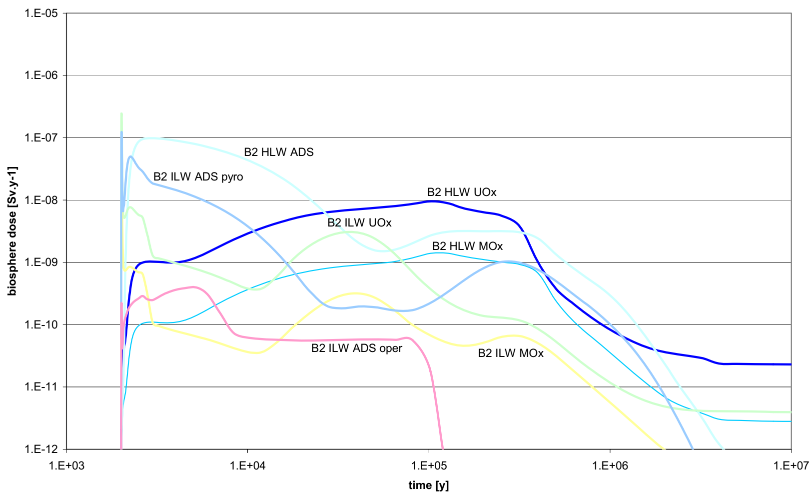

Figure 7.33: Comparison of the doses arising from long lived ILW and HLW from scenario B2 (Czech repository concept for disposal in granite)

The results obtained by NRI show that, because of the assumed presence of considerable amounts of mobile fission and activation products, the disposal of LONG LIVED ILW might give rise to doses that are higher or of the same order of magnitude as the doses due to HLW.

7.4.2 Evaluations of the radiological impact of long lived ILW disposal in case of disposal in granite (Spain)

For these evaluations it was assumed that the waste matrix does not provide any confinement, and that the radionuclides are released to the groundwater directly after sealing of the repository. A simplified near field model is used: 100% of the radionuclide inventory in a waste package is released to an (arbitrary) water volume of 0.1 m^3, where solubility limits are applied. Groundwater flows around the long lived ILW disposal are with an equivalent flow rate (Q_F) of 1 litre/WP per year, which is considered a typical value for a HLW repository in granite.

Figure 7.34 shows the doses due to long lived ILW in scenario A2. Only ^{79}Se doses show a "plateau", indicating that Se transport is solubility controlled.

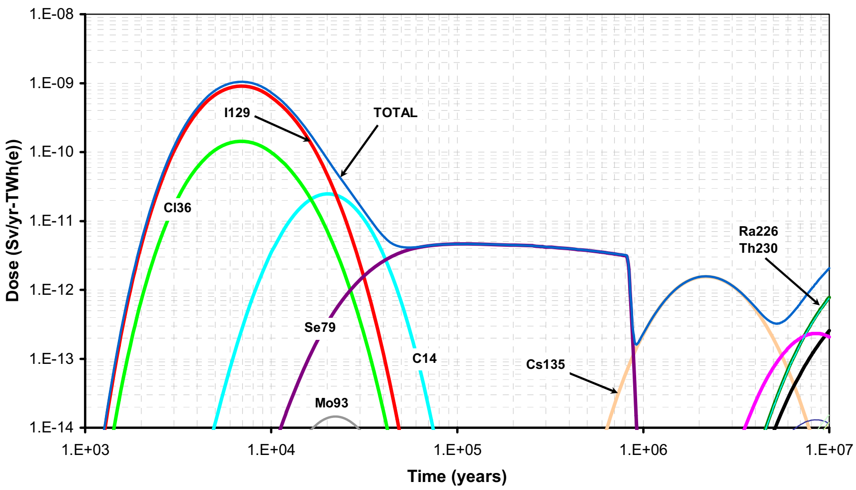

Figure 7.34: Total dose and main radionuclides for long lived ILW in scenario A2 (Spanish repository concept for disposal in granite)

Figure 7.35 presents the different contributions to dose in scenario A2. Peak dose due to long lived ILW is greater than the peak dose due to HLW and is controlled by ^{129}I in the hulls, but long term doses are controlled by HLW. Inclusion of long lived ILW in scenario A2 has a strong effect on doses:

- the peak dose increases a factor 3 compared to the "HLW only" case;
- the doses start much earlier;
- the doses are controlled by long lived ILW up to 20,000 years and by HLW thereafter.

Figure 7.36 presents the doses due to disposal of HLW and long lived ILW arising from scenario A3. Similar results would be obtained for scenario B1. The total dose due to long lived ILW is first controlled by ^{129}I (up to 7 000 years) and than by ^{14}C, which causes the peak dose. Inclusion of long lived ILW in scenario A3 has a dramatic effect on doses:

- the peak dose increases a factor 100 compared to the "HLW only" case;
- the doses start much earlier;
- the peak doses are reached much earlier;
- doses are controlled by long lived ILW up to 50,000 years and by HLW thereafter.

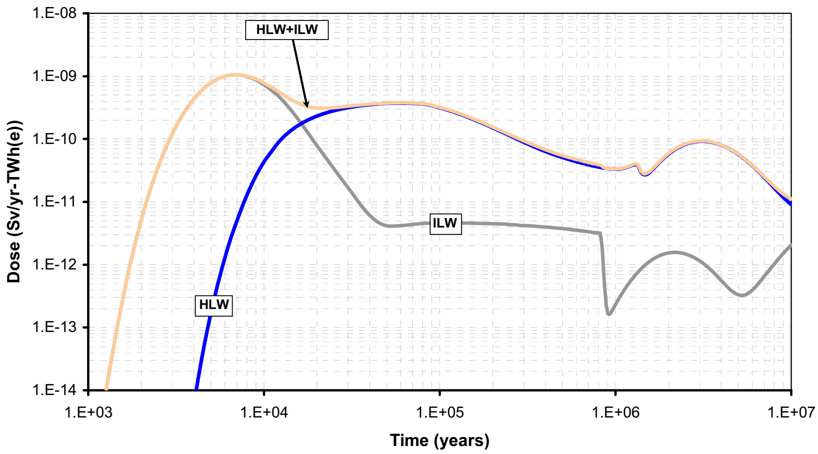

Figure 7.35: Doses due to disposal of long lived ILW and HLW from scenario A2 (Spanish repository concept for disposal in granite)

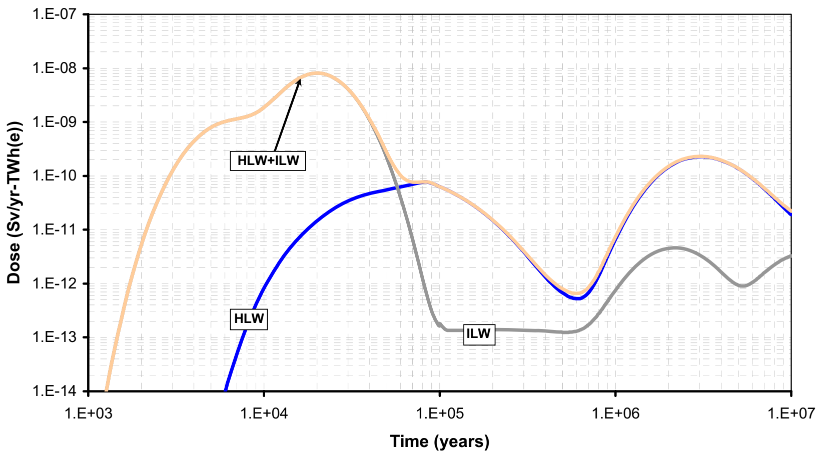

Figure 7.36: Doses due to disposal of long lived ILW and HLW from scenario A3 (Spanish repository concept for disposal in granite)

Figure 7.37 presents the different contributions to dose in scenario B2. Inclusion of long lived ILW in scenario B2 has a strong effect on doses due to the repository: peak dose increases a factor 6 (caused by ^{129}I) compared with the "HLW only" case, doses start much earlier, and peak doses are reached much earlier.

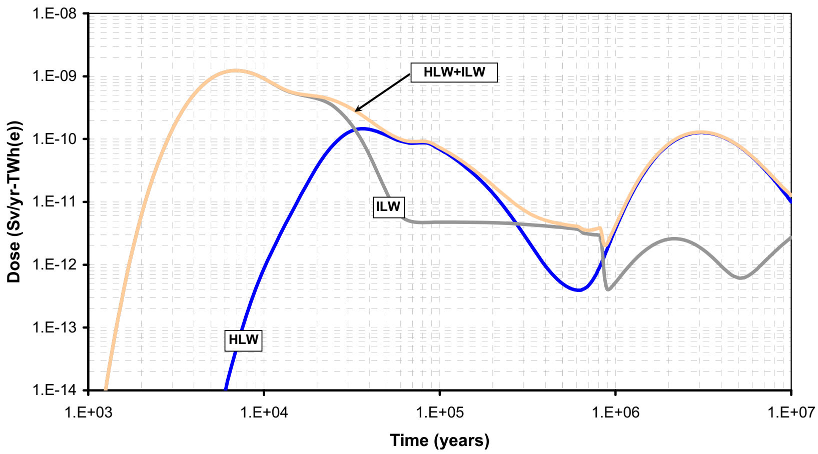

Figure 7.37: Doses due to disposal of long lived ILW and HLW from scenario B2 (Spanish repository concept for disposal in granite).

The total doses due to combined disposal of the HLW and long lived ILW types arising from the base fuel cycle scenarios are shown in Figure 7.38.
Figure 7.38 shows that when long lived ILW is taken into account peak doses of the scenarios A2 and B2 are nearly the same as the one of scenario A1, peak doses of scenarios A3 and B1 are 8 times greater than the one of scenario A1 (those high maximum dose are due to the high amount of ^{14}C generated in scenarios A3 and B1), and peak doses are reached much earlier than in scenario A1.

7.4.3 Evaluations of the radiological impact of long lived ILW disposal in case of disposal in clay (Belgium)

For the evaluations it was assumed that because the use of an overpack is not considered for the disposal of long lived ILW the groundwater comes in contact with the disposed waste directly after sealing of the repository. The matrix lifetime was taken equal to 1000 years.

The cumulative released radiotoxicity is dominated by ^{129}I; other important contributors are ^{36}Cl and ^{99}Tc. Figure 7.39 shows that the main contributors to the total dose are ^{129}I, ^{36}Cl and ^{79}Se. The ^{14}C, ^{126}Sn and ^{99}Tc contributions are one to two orders of magnitude lower.

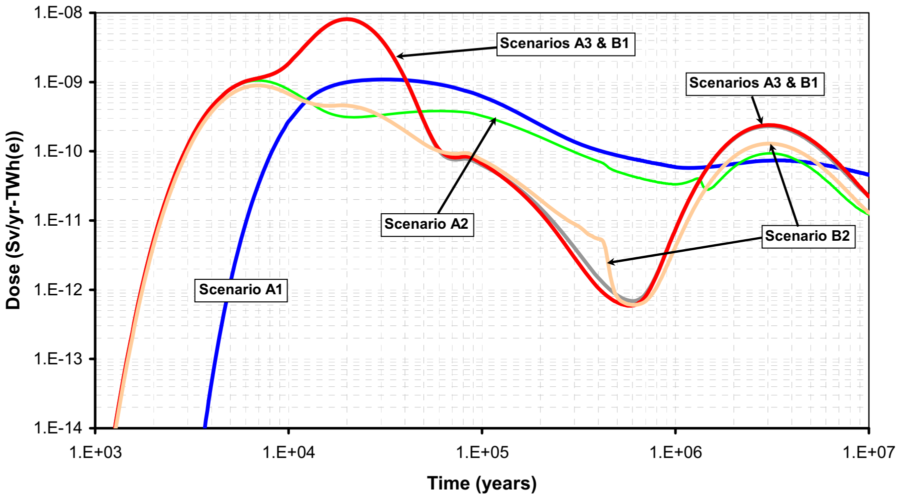

Figure 7.38: Doses due to HLW and long lived ILW for the 5 considered fuel cycle scenarios (Spanish repository concept for disposal in granite).

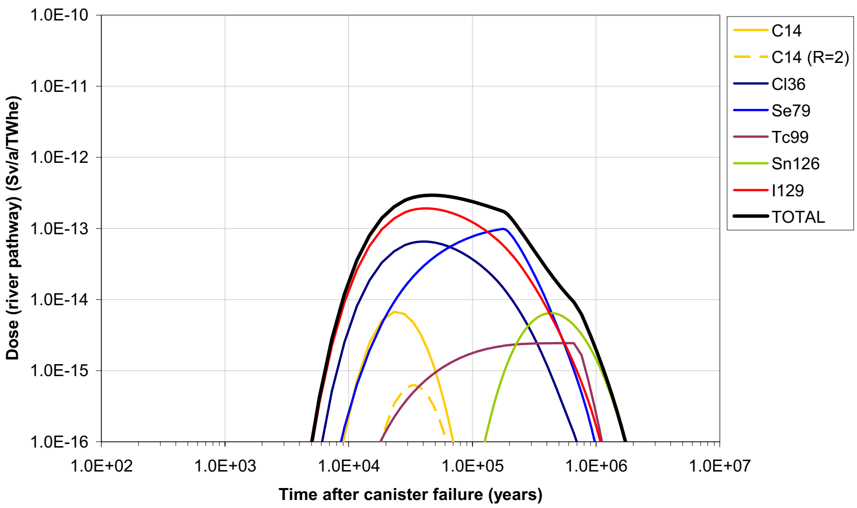

Figure 7.39: Dose via the river pathway per TWh(e) due to disposal long lived ILW of scenario A2 in a repository in clay

Figure 7.40 shows that in the case of a fuel cycle based on a fast reactor the main contributor to the total dose is ^{14}C. Other important contributors are ^{129}I and ^{79}Se. The ^{126}Sn and ^{99}Tc doses are two orders of magnitude lower than the ^{14}C dose.

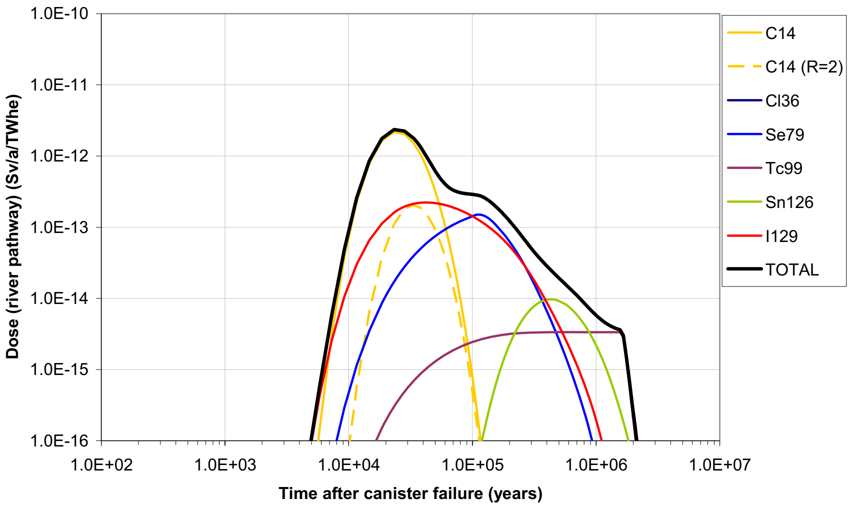

Figure 7.40: Dose via the river pathway per TWh(e) due to disposal long lived ILW of scenarios A3 and B1 in a repository in clay A3

Within the Belgian waste disposal programme the conservative value R = 1 is used for the retardation factor of C in clay. Hitherto, there was no need to consider a higher value, because of the low ^{14}C inventories of the considered waste types. However, there are several indications, such as results of migration experiments and literature data, that C is probably slightly retarded by the Boom Clay. As ^{14}C yields here the highest dose, the dose corresponding to a slight retardation of C in clay (R = 2) is also calculated. In this case, the ^{14}C dose is of the same order of magnitude as the ^{129}I and ^{79}Se doses.

In case of scenario B2, four different long lived ILW types are considered. The doses calculated for those long lived ILW types are shown in Figure 7.41.
The doses calculated for the different HLW and long lived ILW types arising from the base fuel cycle scenarios are shown in Figure 7.42. This figure shows that, in case of disposal in clay, the long lived ILW doses are for scenarios A2 and B2 one order of magnitude lower than the HLW doses. However, for scenarios A3 and B1 the maximum long lived ILW doses are only a factor 2.5 lower than the corresponding HLW doses.

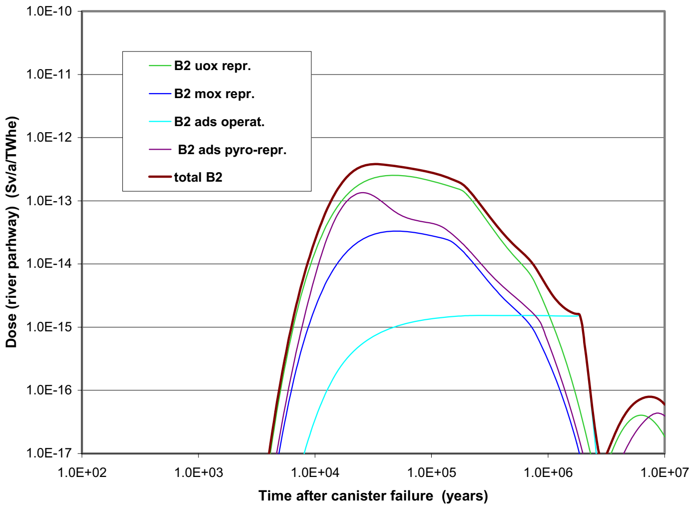

Figure 7.41: Dose via the river pathway per TWh(e) due to disposal of all long lived ILW types of scenario B2 in a repository in clay

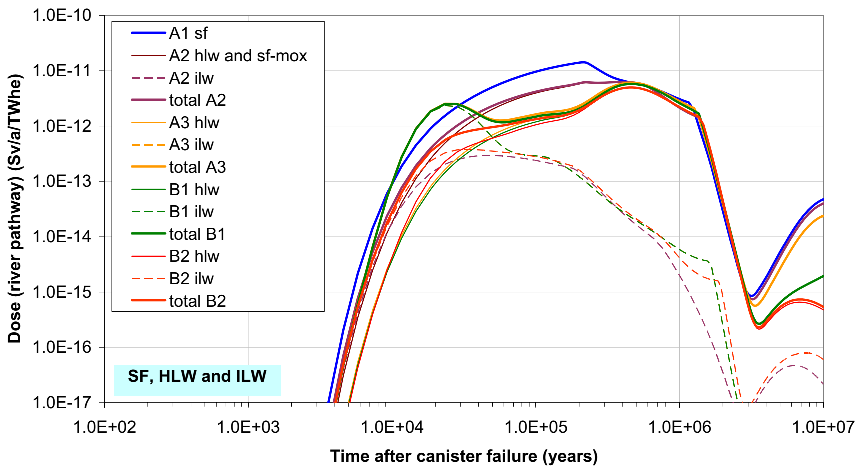

Figure 7.42: Comparison of the doses via the river pathway per TWh(e) due to disposal of HLW and long lived ILW of the 5 base scenarios in a repository in clay

7.4.4 Conclusions concerning the radiological impact of long lived ILW disposal

The high doses due to ^{14}C and ^{129}I calculated for long lived ILW disposal in granite make that the maximum doses due to the disposal of long lived ILW are higher than the doses due to

HLW. The increase of the maximum dose ranges from a factor 3 in case of scenario A2, over a factor 6 in case of scenario B2, to a factor 40 in case of scenarios A3 and B1.

In case of disposal in clay, the about 50 m thick clay barrier spreads the release of mobile fission and activation products from the host clay formation into the surrounding aquifer layers over several tens of thousands of years. The implication of this is that the calculated long lived ILW doses are a factor 2.5 for scenarios A3 and B1 to a factor 10 for scenarios A2 and B2 lower than the HLW doses for that fuel cycle scenario.

An important radionuclide for both disposal in granite and in clay is ^{14}C. This radionuclide is an activation product that is generated in considerable amounts in a fast reactor (scenarios A3 and B1). The use of a more realistic value of the retardation factor of C in the buffer and host formation materials can possibly avoid strongly conservative estimations of the radiological consequences of long lived ILW disposal. For advanced nuclear power systems it can be expected that the development of low activation materials will contribute to reducing the generated amount of ^{14}C.

It has also to be remembered that the amounts of ^{129}I present in the long lived ILW have been very conservatively estimated.

7.5 Impact of advanced fuel cycle scenarios on long-term radiological consequences of HLW, long lived ILW and iodine waste in geological repositories

In the preceding sections the doses arising from the disposal of HLW and long lived ILW have been presented. At present the reprocessing plants generally discharge the iodine that is released in the dissolver directly into the sea. It can be considered, however, that this iodine waste will, in future, be captured in reprocessing plants, conditioned in a waste matrix and disposed of in a geological repository. Therefore, in the following sections doses are presented that might result if all the waste containing significant amounts of long-lived radionuclides (HLW, long lived ILW and I-waste) were disposed of in the same geological repository.

7.5.1 Evaluations of the radiological impact in case of disposal in granite (Spain)

Figure 7.43 shows the doses due to HLW in the different scenarios, as well as doses due to long lived ILW and iodine-wastes. It is clearly seen that in all the scenarios with reprocessing peak doses due to long lived ILW and iodine-waste are much greater than doses due to HLW, and at least equal to peak dose in scenario A1.

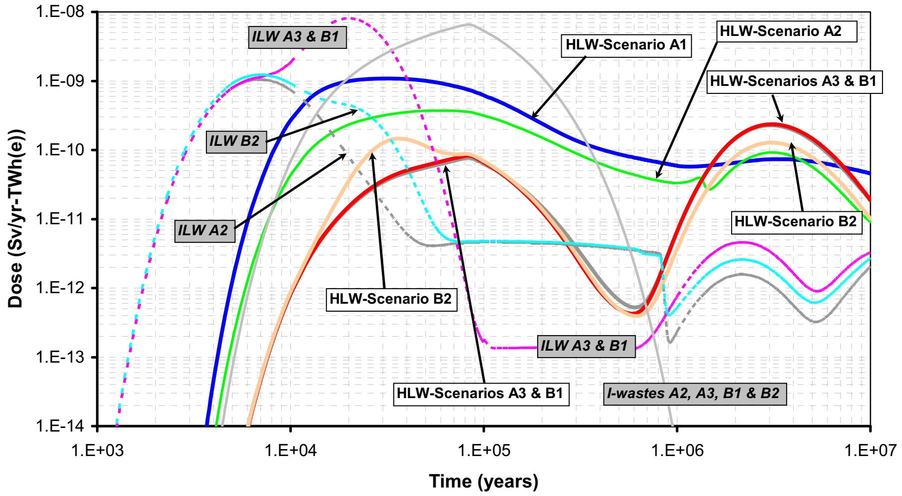

Figure 7.43: Doses due to HLW, long lived ILW and iodine-waste for the 5 considered fuel cycle scenarios (Spanish repository concept for disposal in granite).

Figure 7.43 shows total doses due to HLW, long lived ILW and iodine-waste for the 5 scenarios for the Spanish repository concept in granite. In these calculations it is assumed that 98% of the iodine in the reprocessed fuel is immobilized in a vitrified matrix identical to the one used for HLW (72 000 years of duration). When long lived ILW and iodine-waste are taken into account it is found that peak doses of all scenarios with reprocessing are quite similar, and at least 6 times higher than the peak dose of scenario A1. To lower the peak doses due to iodine-waste to a value similar to the peak dose of scenario A1 (about 10^{-9} Sv/y.TW$_{el}$h) a waste matrix with a duration of 1 million years would be necessary (see Figure 7.26).

The following conclusion can be drawn for the Spanish repository concept in granite: when all the wastes to be disposed of in the repository in the different scenarios are taken into account (not only the HLW), long term radiological consequences for scenarios with reprocessing are equal or greater than in scenario A1.

7.5.2 Evaluations of the radiological impact in case of disposal in clay (Belgium)

Figure 7.44 shows the doses due to the disposal of HLW, long lived ILW and iodine waste in the same geological repository located in a clay formation for scenario A2. A lifetime of 10000 years is assumed for the matrix of the iodine waste. Figure 7.44 clearly shows that the doses due to the iodine waste are responsible for the maximum dose rate and they are the main contributor to the total dose up to 250000 years; thereafter, the maximum dose is essentially due to radionuclides present in the HLW. A comparison of Figure 7.44 with Figure 7.27 shows that a matrix lifetime longer than 1 million years is needed for the iodine waste to bring the iodine waste dose under the HLW dose.

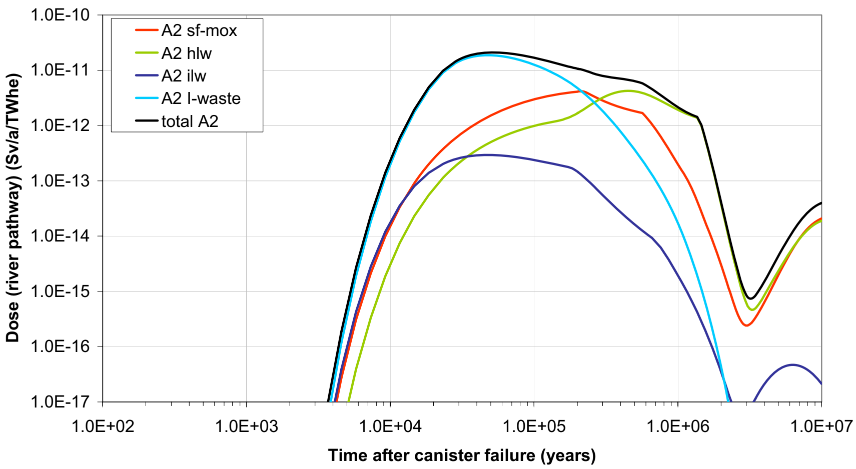

Figure 7.44: Comparison of the doses via the river pathway per TWh(e) due to disposal of HLW, long lived ILW and iodine-waste arising from scenario A2 in a repository in clay

Figure 7.45 gives the total dose, i.e. the dose that will occur if the main waste types (HLW, long lived ILW and iodine waste) arising from a fuel cycle are disposed of in the same geological repository, for the 5 base fuel cycle scenarios.

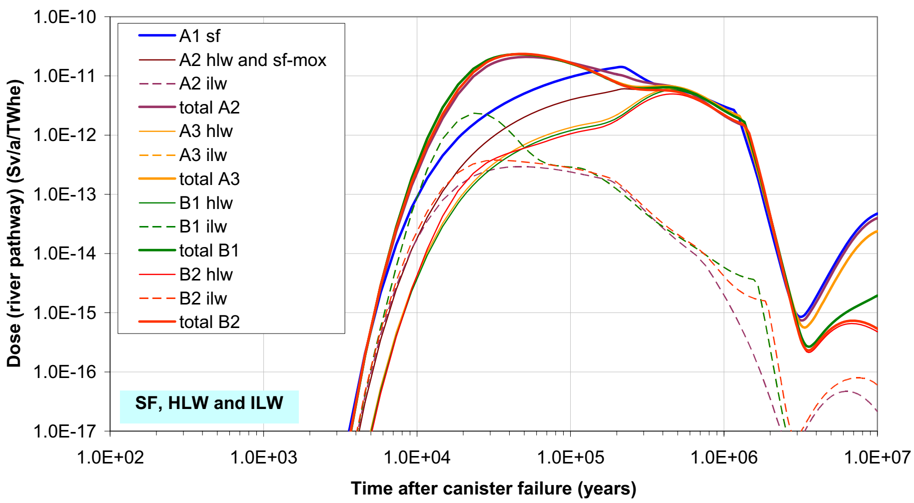

Figure 7.45: Comparison of the doses via the river pathway per TWh(e) due to disposal of HLW, long lived ILW and iodine-waste arising from scenario B2 in a repository in clay

Figure 7.45 shows that the "once through" fuel cycle A1 gives a lower maximum dose than the more advanced fuel cycles do. This somewhat surprising conclusion is explained by the fact that for the advanced fuel cycles about 98% of the generated ^{129}I is assumed to be conditioned in a waste matrix with a lifetime of 10000 years, whereas in the case of the "once through" fuel cycle the largest fraction of the ^{129}I is assumed to be retained in the uranium oxide matrix of the spent fuel, which is here assumed to have a lifetime of 200000 years. When the iodine waste matrix has the same lifetime as the spent fuel matrix, then all the considered fuel cycle scenarios will result in about the same maximum dose.

7.5.3 Comparison of the calculated doses with a dose constraint

To facilitate a comparison of doses from the considered fuel cycles, all doses have been presented as dose standardised per produced electricity with as unit Sv/year·TWh(e). However, those standardised doses cannot be compared with a dose constraint or with the dose from background radiation which are expressed in Sv/y. Therefore in the following section the standardised dose is multiplied by the electricity generated by nuclear power over a given time period within the national nuclear programme.

In Belgium the present nuclear power park consists of 7 reactors. The total installed capacity is 5.5 GW(e) and the reactors are expected to be operated during 40 years. By assuming an availability factor of 95% with estimate the amount of electricity produced by the existing nuclear reactor to be 1830 TWh(e). The resulting doses are given in Figure 7.46 together with the dose constraint of 0.3 mSv/y proposed by ICRP [7.9] and with the dose from natural background radiation, which is about 3 mSv/y in Belgium.

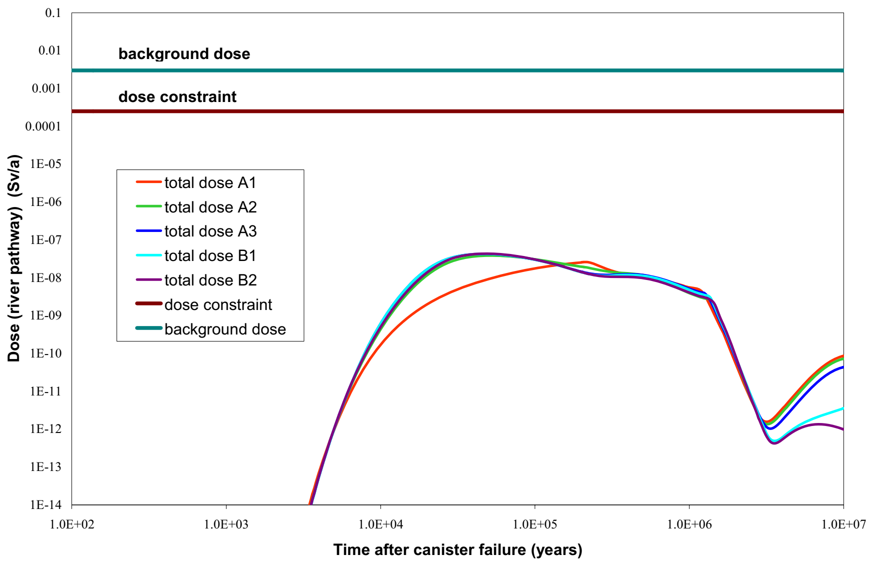

Figure 7.46: Doses due to the long-lived waste (HLW, long lived ILW and I-waste) arising from the generation of 1830 TWh(e) by the 5 considered fuel cycles; for comparison the dose constraint proposed by ICRP and the dose from natural background radiation are also given.

Figure 7.46 shows that the doses via a river pathway due to the long-lived waste arising from the present Belgian nuclear power programme are more than 3 orders of magnitude lower than the dose constraint proposed by the ICRP 81 publication [7.9].

7.6 Conclusions

Impact on repository size

High-level radioactive waste arising from advanced fuel cycles generates significantly less heat than does the equivalent amount of spent fuel arising from the "once through" fuel cycle, which is called scenario A1 in RED-IMPACT.

In case of disposal in crystalline or argillaceous formations the smaller thermal output of the waste allows a reduction in the size of the HLW repository needed per unit produced electricity. For instance, the greatest reduction in needed length of disposal galleries is observed for scenario B1 (fast reactor with recycling of the actinides), a reduction with a factor 2.5 (Enresa case for disposal in granite) to 3.2 (SCK•CEN case for disposal in clay) in comparison with scenario A1. The reduction of the needed surface of the repository can even be higher when the distance between two disposal galleries is also optimised, for so far as geomechanical limitations do not impose a minimum distance between two galleries.

The separation of Sr or of Cs and Sr allows a further reduction to be made in the gallery length or repository surface that is needed. E.g., in case of disposal in clay for the variant fuel cycle scenario B1 with separation of Cs and Sr, the gallery length for HLW disposal can be reduced with a factor 10 in comparison with the reference fuel cycle scenario A1. However, to take full benefit of the separation of Cs and Sr it will be necessary to increase the waste loading of the vitrified high-level waste and to develop repository designs adapted to dispose of waste packages with moderate thermal output.

In case of disposal in salt formations the impact of the reduced thermal output is less straightforward. On the one hand, a temperature limit in the host rock has to be respected; on the other hand, high temperatures accelerate convergence of the salt in the excavated galleries, what strongly contributes to limiting the consequences of a groundwater intrusion into the salt dome.

Impact on dose

The calculated doses are for all considered repository systems far below the dose constraint of 0.3 mSv/a proposed by ICRP. The impact of the application of P&T in the fuel cycle on the maximum dose resulting from the disposal of the generated HLW is rather limited, because the maximum dose is essentially due to mobile long-lived fission and activation products. The amount of produced fission products is about proportional to the heat produced by nuclear fission in the reactors; the higher thermodynamic efficiency of the heat to electricity conversion in advanced power plants and the different neutron spectrum in fast reactors or ADSs can have some influence on the generated amount of some fission products.

The amount of iodine that is going into the repository, which depends on the fraction of spent fuel that is reprocessed, strongly influences the maximum dose. In case of disposal in crystalline or argillaceous formations, the higher disposal density, which can be considered in case of disposal of HLW from advanced fuel cycles, can result in a small decrease of the dose from radionuclides of which the release is controlled by a solubility limit. Very long-term doses, i.e. after about three hundred thousand years for disposal in granite and after a few million years for disposal in clay, are lower in the case of advanced fuel cycle scenarios because smaller amounts of actinides are present in the waste arising from these scenarios.

Evaluations of the radiological impact of disposal of long lived ILW in the same geological repository reveal that in case of disposal in granite, the maximum dose due to the disposed long lived ILW can be an order of magnitude higher than the one due to the disposed HLW from the same fuel cycle scenario. The main reasons for the high long lived ILW doses are on the one hand the strongly conservative assumptions made for the estimation of the amounts of mobile fission and activation products, e.g. ^{14}C and ^{129}I, present in the long lived ILW, and on the other hand the absence of a stable waste matrix or of an appropriate engineered barrier limiting the release of mobile radionuclides.

In case of disposal in clay, the diffusive transport of radionuclides through the thick natural clay barrier results in continuous releases of mobile radionuclides from the host clay formation into the surrounding aquifers over several tens of thousands of years. As a consequence, the maximum long lived ILW dose is about one order of magnitude lower than the maximum HLW dose.

An important radionuclide occurring in long lived ILW is ^{14}C, of which considerable amounts were assumed to be generated in a fast reactor or an ADS. The development of high purity materials might reduce the generated amount of ^{14}C.

Maximum doses associated with future high-level waste repositories will strongly depend on the applied iodine management. If iodine is captured at future reprocessing plants and immobilized in matrices of which the stability is comparable to today's borosilicate glasses, doses due to these iodine-wastes would still control total doses resulting from a repository in which the main long-lived waste types, i.e. HLW, long lived ILW and iodine-waste, arising from a fuel cycle would be disposed of. The resulting peak doses for scenarios with reprocessing would even be higher than for scenario A1. As a consequence, if the iodine would be captured at the reprocessing plants, it would be necessary to develop extremely stable waste matrices, possibly in combination with long-lived containers, for conditioning and packing the iodine waste.

In a safety case of a geological repository for HLW or spent fuel disposal, the evaluations made for the expected evolution of the repository system are complemented with evaluations for a number of less likely evolution scenarios. A special group of less likely evolution scenarios are human intrusion scenarios, such as the geotechnical worker scenario. This scenario assumes that in the future, when the existence of the repository will be forgotten, a geological exploration borehole is drilled through the repository and that cores containing fragments of the disposed waste are taken and examined by a geotechnical worker, who is not aware of the radioactivity present in the cores. Intrusion into a repository, in which spent fuels (uranium oxide or MOX fuels) have been disposed of, might result in a significant dose to a geotechnical worker during a few hundreds of thousands of years after disposal. Reprocessing of the spent fuel, with recuperation of U and Pu, gives rise to vitrified HLW that is less radiotoxic and the period during which significant doses can occur in this scenario is reduced to a few tens of thousands of years. When all the actinides are recycled in a fast reactor or an ADS, the radiotoxicity in the waste is considerably reduced and the period during which significant doses can occur to a geotechnical worker is limited to a few hundreds or thousands of years.

In some of the considered fuel cycle scenarios, a large fraction of the remaining radioactivity is concentrated in a small number of waste packages. This concentration of radioactivity and, as a consequence, of radiotoxicity can mask the effect on intruder doses of the reduction of the total radiotoxicity in the disposed waste in case of advanced fuel cycle scenarios. Finally, it should be noticed that the realism of drastic human intrusion scenarios, such as the geotechnical worker scenario, is strongly debatable; those scenarios assume that, in spite of the fact that the drill-bit has to perforate a thick-wall metallic container, the geotechnical workers will not become aware of the presence of very exotic materials in the underground and will not take precautions when examining the extracted cores.

8. Assessment of the Results

8.1 Introduction

In order to make a choice between the scenarios developed in RED-IMPACT, it is necessary to parameterise the large amount of data resulting from the analysis of the scenarios. Cost benefit analysis is a well known tool in comparing the relative merits of different situations however it has some limitations. It is not always possible to ascribe a monetary value to certain properties of a scenario and these properties are therefore often left unaccounted for in the analysis. This can result in an intrinsically biased result.

From the beginning of the RED-IMPACT project, it was decided to use the established methods of Multi Criteria Analysis (MCA). This increasingly popular set of techniques typically combines a range of option impacts into a single framework for easier assimilation by decision makers. They are able to take account both of impacts measured in monetary units, such as construction costs, and of social and environmental impacts that may be quantified but not valued (such as proliferation risk) or assessed only in qualitative terms (such as visual impact). This provides an improved way of presenting monetised and non-monetised impacts of projects to decision makers. An important part of the process is the assessment of the sensitivity of the results to the input data. This will ensure the right data is being collected and address uncertainties.

Evaluating actinide minimization systems and industrialised P&T in general requires an assessment of relevant nuclear fuel cycles with particular regard to the economic, environmental and societal advantages/disadvantages (i.e. the sustainability of the fuel cycles). Thus, a set of indicators has been derived for each of these areas. However, in this study, there are several problems in applying MCA:

- Several indicators are missing (particularly representing societal considerations)
- Equilibrium not transition scenarios have been modelled to determine the indicators
- The uncertainties applied are very approximate
- Salt repositories have not been considered
- MCA weights used are from the RED-IMPACT participants only (other interested parties should also be involved in determining the weights)

Despite these problems, an example analysis has been performed in order to demonstrate how this technique can be used to assist the decision making process.

8.2 MCA Methodology

In the MCA technique any property can be used as an indicator. However, this means that many of the indicators will be in different units (€, KW, %population, etc.) depending on how each indicator is defined. It is therefore necessary to determine a way of combining the indicators to give a single value for each scenario. A scoring system is used to allow this. Each scenario is given a score for each indicator (between 0 and 100, for example). The indicators are weighted to represent the relative importance of each. The value of scenario j is then given by:

$$V_j = \sum_i w_i S_{ij}$$

where w_i is the weight given to indicator i and S_{ij} is the score given to scenario j for indicator i.

It is important to note that the value V_j determined here does not in any way represent a financial amount or an environmental value; it simply represents the preference for that scenario relative to the other scenarios. In this way, MCA allows comparisons to be made on the relative merits of each scenario, providing assistance in the decision making process. The weights used represent a qualitative assessment and certain groups or stake holders may have different opinions about this. Common practice is therefore to perform a MCA with different sets of relative weights (one representing one's own priorities, one representing perceived public priorities, one representing stake-holder priorities, etc.). In this way, any differences in the results will become clear and can be considered.

For the present analysis, the scores S_{ij} have been determined on the basis of the RED-IMPACT indicators. There are several methods that can be applied to determine the score. A simple and effective way is to determine the quantified indicator for each scenario giving the most favourable result a score of 100 and the least favourable a score of 0. The remaining scenarios are scored according to a linear relationship between the best and worst scores. If, in some instances, linear scoring does not describe the relationship adequately a non-linear scoring relationship can be defined to suit the circumstances.

By varying the weights w_i attributed to each indicator sensitivity analysis can be performed, by determining what effect these changes have on the relative value of each scenario. Uncertainty ranges associated with the evaluation of each indicator can also be used to track the effects of these uncertainties on the outcome of the analysis.

8.3 Computational Method

The computational method for the uncertainty analysis is performed in four steps:

- Grouping of the indicator values
- Determination of the scores S_{ij}
- Determination of the weights w_i
- Uncertainty analysis

8.3.1 Indicators

During the RED-IMPACT project, partners calculated the indicators which were determined for each of the five scenarios utilising each of the two disposal options; clay and granite. For some indicators, the partners also provided an estimate of the uncertainty related to the indicators. For other indicators however, uncertainties were not provided by the partners. Therefore an educated guess had to be made about uncertainties of these parameters.

8.3.2 Scores S_{ij}

The scores were determined on the basis of the RED-IMPACT indicators. For a given indicator the best and worst results over the five RED-IMPACT scenarios are given scores of 100 and 0 respectively. Indicator results that are between are given a score that is linearly proportional or logarithmically proportional (where the indicators span several orders of magnitude) to the minimum and maximum values.

This procedure applies to both the clay and granite disposal option.

8.3.3 Weights w_i

The weights w_i attributed to each indicator were obtained from a questionnaire that was sent around to the RED-IMPACT partners. For the determination of the scenario values V_j, average weights were used.

8.4 Environmental Indicators

8.4.1 Radiotoxicity of releases

Two indicators have been defined which account for the releases from all plants and processes in each scenario, during the operational period. These are:

- Env.1.1: Radiotoxicity of gaseous releases (gases and aerosols)
- Env.1.2: Radiotoxicity of non-gaseous releases (liquids)

These have been calculated by consideration of releases from operating nuclear facilities and from expert estimates of the expected releases of more advanced facilities. Facilities in all stages of the fuel cycles were considered in this analysis (including: mining and milling, conversion, enrichment, fuel fabrication, reactor irradiation, storage and reprocessing).

8.4.2 Fuel cycle related waste

Two indicators were defined to allow for the waste generated from the fuel cycle. These are:

- Env.2.1: Operational waste
- Env.2.2: Decommissioning waste

It was intended that a similar analysis would be performed as has been done for the radiotoxic releases indicators. However, it was not possible to provide these indicators for this analysis.

8.4.3 Pre-disposal waste management aspects

Five indicators have been defined to allow for the various implications related to pre-disposal waste management:

- Env.3.1a: Volume of high level waste (HLW)
- Env.3.1b: Thermal output of the HLW after 50 year's cooling
- Env.3.1c: Length of repository gallery required for waste
- Env.3.2: Volume of intermediate level waste (long lived ILW)
- Env.3.3: Volume of low level waste (LLW)

Env.3.1a alone would not be sufficient to describe the issues associated with HLW, as such, 3.1b and c were also defined. This allowed the full impact of the properties of the waste (specifically heat output) on the repository design and loading to be assessed. Unfortunately it was not possible to include Env.3.3 in this analysis, but the other indicators were considered sufficient to represent the pre-disposal waste considerations.

8.5 Long term consequences of radioactive waste disposal

A variety of different indicators have been defined to allow a comprehensive representation of the long-term impacts of disposal. These fall into several groups that are discussed below.

8.5.1 Radiotoxicity in the repository

The radiotoxicity of the waste in the repository is the first indicator. This is broken into three indicators, at different time periods:

- Env.4.1: Radiotoxicity at 500 years
- Env.4.2: Radiotoxicity at 10 000 years
- Env.4.3: Radiotoxicity at 1 000 000 years

This provides a reference point for the potential impact of the disposed material. However it is important to note that this figure does not represent a dose that will be received by the public; the material is trapped in a repository. However, it does have some bearing on the consideration of human intrusion into the repository at some point in the future.

8.5.2 Radiotoxicity flux release into the biosphere

Inevitably, some small quantity of waste will seep from the waste-form and eventually enter the biosphere. This indicator represents the impact of this waste release and was determined using various repository modelling tools.

Again, this indicator has been split into different time periods. However, the time-scales considered are different to those previously used. This is because very little (almost zero) would be released from repositories in the first 10 000 years (as a result of the waste-forms and barriers put in place within the repositories). The indicators are:
Env.5.1: Radiotoxicity flux released into the biosphere (0 – 10 000 years)
Env.5.2: Radiotoxicity flux released into the biosphere (10 000 – 100 000 years)
Env.5.3: Radiotoxicity flux released into the biosphere (100 000 – 1000 000 years)

Related to these indicators is Env.6.1: Maximum radiotoxicity flux released into the biosphere. This is recorded in order to ensure that the maximum impact of the scenario is captured. The maximum dose to an individual as a result of these releases is also determined in Env.7.1.

8.5.3 Human intrusion into the repository

Some indicators have been defined to allow for the impact of people intruding into the repository at some time in the future. The scenario envisaged here is that some geological workers drill a borehole through a waste canister and bring the sample to the surface for analysis.
The indicators are defined over various time-scales after the end of the scenario:
Env.9.1: Dose due to human intrusion at 300 years
Env.9.2: Dose due to human intrusion at 10 000 years
Env.9.3: Dose due to human intrusion at 1 000 000 years
Env.9.4: Required isolation time (to reach the commonly used 10mSv intervention level)

8.5.4 Doses received due to scenario operation

Some indicators were also defined to report the dose to the public and the nuclear industry workers due to scenario operation. These are:

Env.10.1: Maximum individual dose to public
Env.10.2: Collective dose to the public
Env.11.1: Maximum individual dose to workers
Env.11.2: Collective dose to workers

The collective dose to the public has been determined as part of the societal assessments. However the maximum individual dose to the public was unavailable. Env.11.1 and 11.2 were subsequently removed from the analysis as the dose to workers would always be kept below the regulatory limits by sufficient engineering measures at the various plants.

The final environmental indicator is Env.12.1: Potential impact of accidents. However, it has not been possible to determine this result due to the huge variety of contributing factors that would need to be considered (especially the safety features of the advanced processes and plants that have not yet been designed).

8.6 Economic Indicators

8.6.1 Research and development costs

The research and development cost, R&D, indicator includes costs for fuel development, research on reactors and costs associated with waste management and repository research. The fuel and reactor development costs are both assumed to be equal to certain fractions of the total fuel and construction costs respectively. The same assumption was made in previous studies both by the University of Chicago [8.1] and the Massachusetts Institute of Technology [8.2].

The cost of waste management research has mainly been estimated based on known costs for such research performed in Sweden. Data was provided by the Swedish waste management company SKB [8.3].

8.6.2 Facility construction costs

Facility construction costs are defined as the construction costs of the reactors in each scenario. For the first ten reactors of a kind, learning effects are considered. Studies performed on the nuclear expansion in France and in the US during the 1970s and 1980s suggest a four percent cost reduction relative to the last reactor to be assumed. The consequence of such an assumption is that the cost of the tenth reactor is reduced by one third as compared to the first.

8.6.3 Facility operational costs

In the facility operational cost indicator, costs for; mining and conversion, enrichment, fuel fabrication, fuel reprocessing and operation and maintenance, O&M, of reactors are included.

The influence of economies of scale has been disregarded in estimations of the indicator since the scale of most front end fuel cycle activities are determined by the scale of the global nuclear park. For these, economies of scale are already in play, and are not considered to be affected too much by variations in the mass flow in the scenarios studied. Whereas recycling is concerned, the limited experience of most of the advanced technologies suggested, provide very little basis for assumptions on economy of scale.

8.6.4 Waste operational costs

In the waste operational cost indicator, costs for waste handling, from arrival at the interim storage to emplacement in the geological repository, are included.

Construction costs of the geological repository and of its galleries have been excluded from this indicator. The data used to estimate this indicator have been provided by the Swedish waste management company SKB. Variations in the amount of waste arriving at interim storage account for most of the difference in this indicator in between scenarios.

8.6.5 Decontamination and decommissioning costs

The decontamination and decommissioning, D&D, costs include costs for shutdown and dismantling of the reactor. Experience from dismantling of light water reactors indicates that these costs may be estimated to have a linear dependence to the reactor construction cost. For light water reactors, the D&D cost are assumed to equal 15% of the original construction cost. For fast reactors and accelerator-driven systems, 25% is assumed. The LWR value is in agreement with the available literature [8.5, 8.6]. Regarding the FR-fraction (fast reactor fraction) only limited amounts of data have been published on D&D costs. Most of the data concerns FR decommissioning activities in the UK and France [8.5]. The estimated costs might become subject to reductions, as economies of scale and learning effects comes into play with large scale FR introduction. However, due to the complications associated with sodium recovery, they are not likely to decrease to the level seen for LWR decommissioning.

8.6.6 Total repository cost

The total repository costs include all costs for research on and construction of the repository, placing the waste in it and closing it. The differences in the amount of waste and heat generation in between the scenarios are accounted for in the cost model. The repository assumed is of the granite type and the cost estimates originate from the annual PLAN report from Swedish SKB where the costs associated with nuclear waste handling are estimated in Figure 8.1. The cost of a final repository in clay or salt has not been estimated. This is due to the small amount of detailed data published on the subject. Even though data exists, it is not possible to deduce the fixed and variable costs from it, and therefore not possible to estimate the impact of different scenarios on the repository costs.

First estimates have been based on the assumption that all waste goes to one repository only. However this assumption is rather hard to defend when considering the large amount of waste produced in scenario A1 and A2. Further it is impractical for a number of reasons. To investigate the influence of fixed costs on the overall cost for the repository, the one-repository-assumption compared two repositories limited in size. Limits of 10 000 and 100 000 tons respectively were investigated. Ten thousand tons correspond fairly well to the proposed Swedish and Finnish repositories, whereas one hundred thousand tons is closer to the suggested size of the Yucca Mountain repository.

Although not part of the MCA, it is interesting to estimate the cost of electricity (COE) for each of the scenarios.

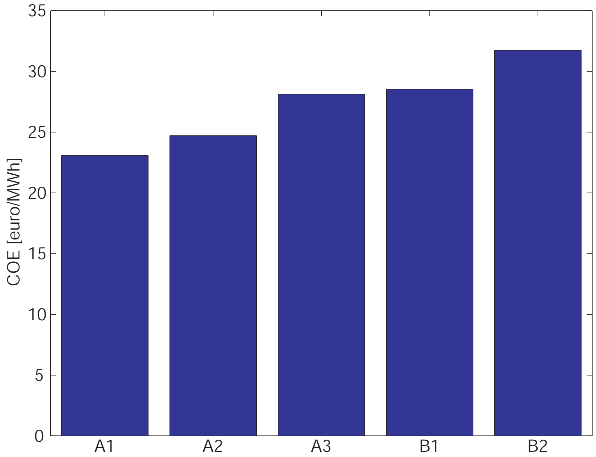

Figure 8.1: Total cost of electricity for the five scenarios

The COE is quite similar for all of the five scenarios studied, despite the fact that some indicators vary considerably. This may be understood based on the small fractions of advanced technologies even in the most innovative scenarios. Light water reactors still dominate the costs in most of the scenarios.

The contribution to these costs can be seen in Figure 8.2.

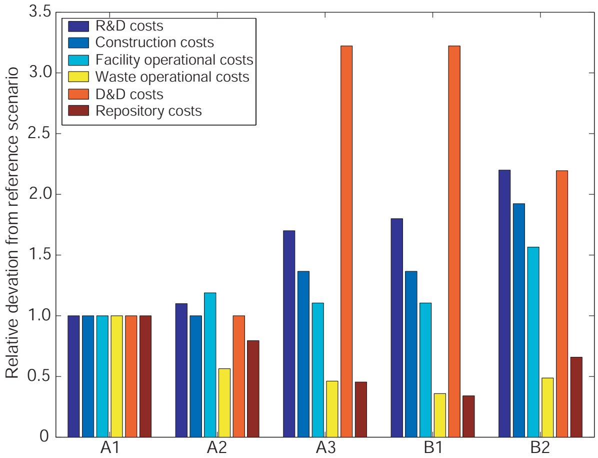

Figure 8.10: The relative deviation of the economic indicators normalized to the A1 scenario

The dismantling and decommissioning (D&D) costs are larger in the scenarios which have fast reactors since it is assumed that these costs will larger than the D&D costs for a LWR. In addition the R&D costs are larger for those scenarios which are more complex, although the actual costs of R&D are very much smaller than other costs considered.

8.7 Societal Indicators

8.7.1 Years of life lost due to power generation

The indicator Soc 1.1 exclusively considers radiological impacts of routine releases of effluents on the public. The generation of electricity by nuclear power involves the processing of materials that are naturally radioactive and creates larger amounts of artificial radionuclides. The release of radionuclides implies radiation exposures to the public. Years of life lost has been determined by evaluating the impact of the collective dose to the public.

8.7.2 Fatalities due to accidents in a nuclear fuel cycle

Severe accidents are one of the most contentious features of the social and general assessment of nuclear systems. Although sites at other stages of the nuclear fuel cycle handle more inventories of radioactive material than the generation plants, their activities are generally assumed to have lower risks. The transportation of the radioactive materials between all fuel cycle facilities involves relatively small risks.

8.7.3 Proliferation Resistance

Nuclear fuel cycles may contribute to the proliferation of nuclear weapons material. These potential contributions are complex to assess due to manifold options of diversion, transportation, transformation and weapon fabrication pathways. According to the set of established sustainability criteria of the economical and environmental issues, the proliferation of nuclear weapons is an essential social criterion.

Proliferation resistance is defined as the characteristics of nuclear energy systems that impedes the diversion or undeclared production of nuclear material, or misuse of technology, by States intent on acquiring nuclear weapons or other nuclear explosive devices.

8.8 Uncertainty and Sensitivity Analysis

Once the values of the indicators and the related uncertainties were estimated, the actual uncertainty analysis was performed on the impact of the uncertainties (i.e. the scores S_{ij} and the weights w_i) on the value V_j of each of the five RED-IMPACT scenarios. The analysis was done using the @RISK package.

@RISK is a “Risk Analysis and Simulation” add-in for Microsoft Excel [8.7]. As an add-in, @RISK is integrated with the spreadsheet, adding Risk Analysis to existing models. @RISK uses the Monte Carlo simulation technique, allowing all possible outcomes to be taken into account. For that matter, uncertainties in the spreadsheet model are replaced with @RISK functions to represent a range of possibilities. @RISK recalculates the values in the spreadsheet selecting random numbers from the @RISK functions that have been entered to represent the uncertainties. This results in distributions of possible outcomes and the probabilities of getting those results.

Sampling is used in an @RISK simulation to generate possible inputs from probability distribution functions. These sets of inputs are then used to evaluate the Excel worksheet. Because of this, sampling is the basis for the many "what-if" scenarios @RISK calculates for the worksheet. Each set of samples represents a possible combination of inputs which could occur. These are then determined by the input probability distributions allocated by the user. Probability distributions are represented in @RISK by probability distribution functions and sampling is performed by the @RISK program. Sampling in a simulation is done repetitively, with one sample drawn every iteration from each input probability distribution. With enough iterations, the sampled values for a probability distribution become distributed in a manner which approximates the known input probability distribution. The statistics of the sampled distribution (mean, standard deviation and higher moments) approximate the true statistics input for the distribution.

8.9 Results

8.9.1 MCA

The indicators have been calculated as discussed earlier and the results were then used to determine the score to be attributed to each scenario against each indicator. These are presented in Figure 8.3 and 8.4, for clay and granite repositories respectively.

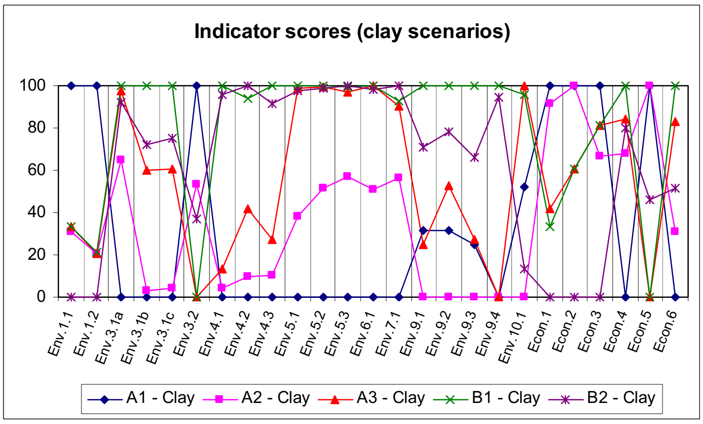

Figure 8.3: Graph of indicator scores for the scenarios with disposal in a clay repository

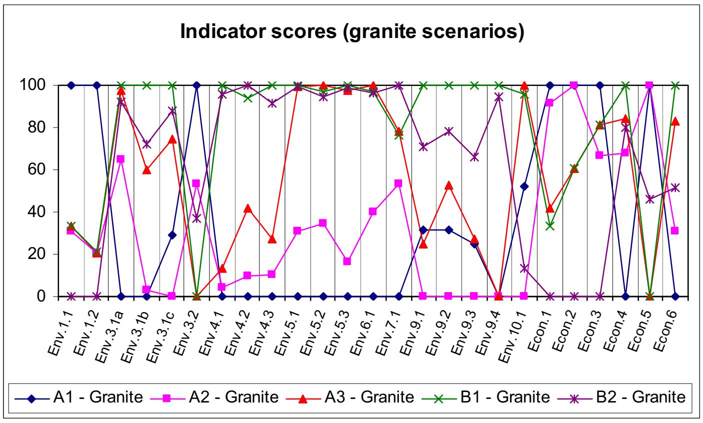

Figure 8.4: Graph of indicator scores for the scenarios with disposal in a granite repository

It is clear from these results that B1 performs the best for the majority of the environmental indicators but doesn't perform as well against several of the economic indicators. B2 performs almost as well for the environmental indicators but scores lowest for several economic indicators. Scenarios A1 and A2 (based on existing technologies) perform well against the economic indicators but do not do well in the environmental ones. These scores therefore can be used to give an indication of how the scenarios perform. However, account must be taken of the relative importance of the different indicators. To do this the weights were investigated and a full MCA applied. It should also be noted that the societal aspects of each scenario are not considered in this MCA system and will be discussed separately.

Twenty different weights sets were collected and these were grouped into six categories:

- average of all responses
- long-term environmental responses
- public opinion responses
- nuclear expert responses
- industrial responses
- economic responses

The average of all the weights sets in each category were then used in MCA assessments. The resulting values for each scenario are given in Figure 8.5 and 8.6, for clay and granite repositories respectively. From these it can be seen that scenario B1 is the preferred scenario from all of the categorised weights sets. It should be noted that some of the individual weights sets did not always show this. For example, if a very heavy economic bias is assumed (with the weights for all environmental indicators set to zero), then scenario A2 is the preferred option with A1 coming second and B1 third. However, this is not a very realistic case as it is unlikely that anyone would attach zero weight to environmental considerations. The economic bias presented below represents a less extreme bias that does include some weight for the environmental indicators. For the majority of weights sets, scenario B1 was the preferred option.

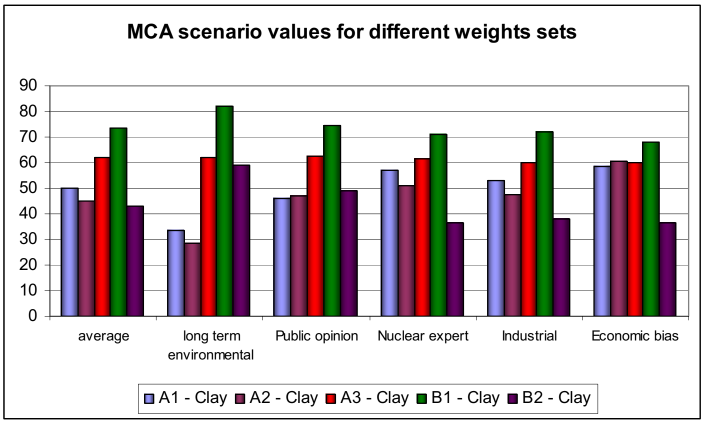

Figure 8.5: MCA values of each scenario (with disposal in clay) for a variety of weights sets

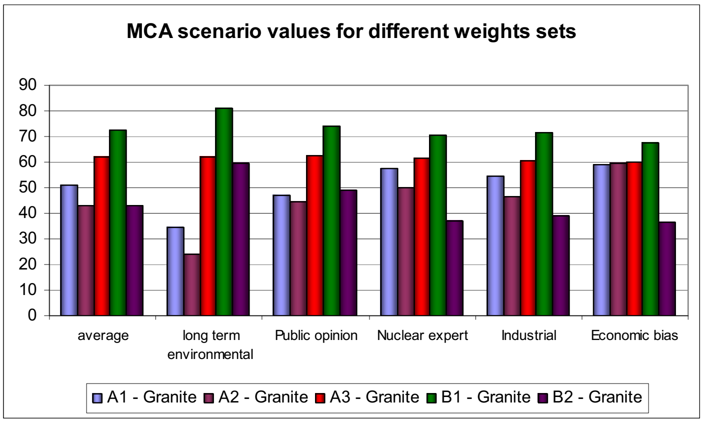

Figure 8.6: MCA values of each scenario (with disposal in granite) for a variety of weights sets

8.9.2 Uncertainties in MCA

A study has been made of how uncertainties affect the MCA technique. It should be noted that this study was done with a slightly less complete data set than was presented previously; as a result some of the values may differ slightly. However, this study was performed to provide an indication of the effects of the uncertainties on the reliability of the scenario ranking achieved using MCA (not to provide an absolute value of the uncertainty) and is suitable for this purpose, despite being based on slightly older data.

The @RISK add-in for Excel has been used to approximate the uncertainty distributions associated with each indicator and apply this to give an indication of the uncertainty associated with each MCA value as determined in the previous section.

Determining the uncertainty associated with each indicator is the main difficulty of this study; for many of the environmental indicators estimates of errors were applied. In this analysis only those contributions to the uncertainties that are different between each scenario are considered (those that are discriminatory). For example, in the dose due to human intrusion indicator, uncertainty in the repository depth is non discriminatory (as it is assumed to be the same for all scenarios), whereas uncertainty in the radiotoxic inventory extracted is discriminatory.

A set of reasonable uncertainties were determined and analysed. In order to demonstrate the effect if these are under- or over-estimations, analyses were also performed at half and double these uncertainties. The resulting distributions of the MCA values can be seen in Figures 8.7 to 8.9.

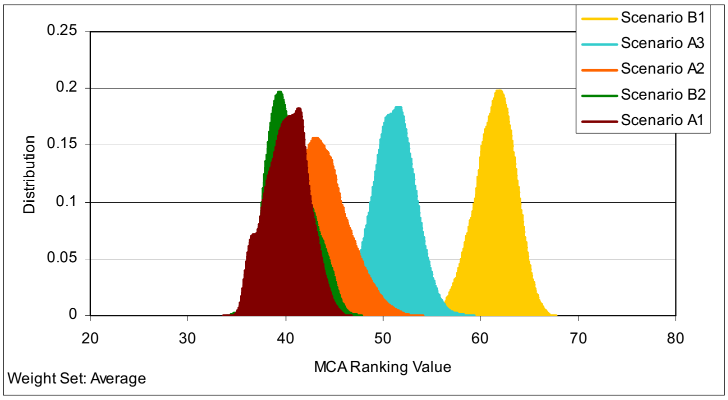

Figure 8.7: The uncertainties in the MCA values (using half the assumed uncertainties of each indicator)

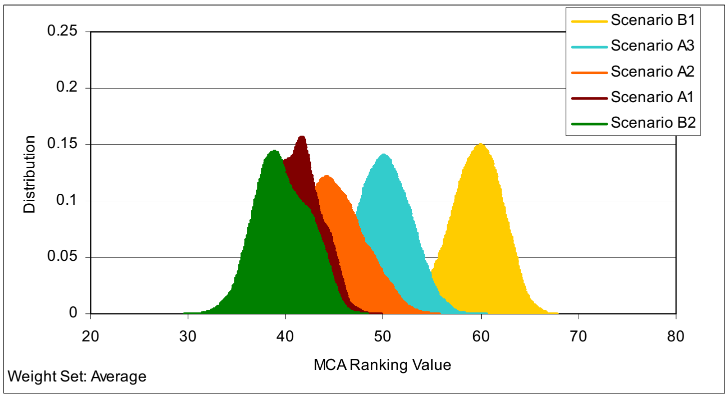

Figure 8.8: The uncertainties in the MCA values (using the assumed uncertainties of each indicator)

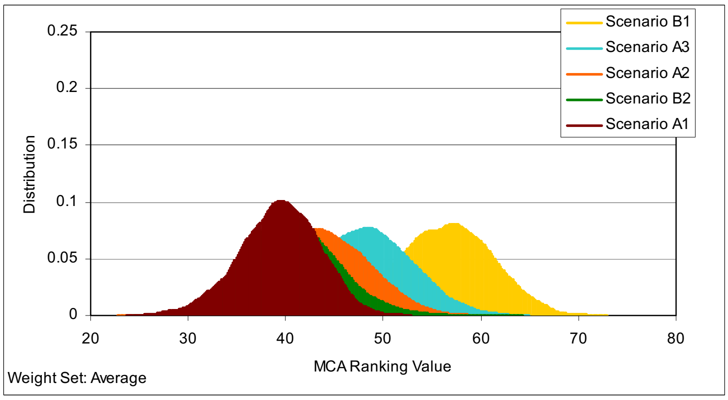

Figure 8.9: The uncertainties in the MCA values (using double the assumed uncertainties of each indicator)

From these graphs it can be concluded that at the uncertainty levels assumed, the MCA technique can be applied to demonstrate the preference for different scenarios, provided that their MCA values differ by 5-10% or more

8.10 Societal

It was recognised, from the beginning of the project, that the societal indicators would be difficult to incorporate into an MCA framework since most of these would be of a qualitative nature. Of these three indicators only Soc.1.1, Years of life lost due to power generation, can be included in the MCA. In this case, the results were produced in terms of collective dose to the public as shown below.

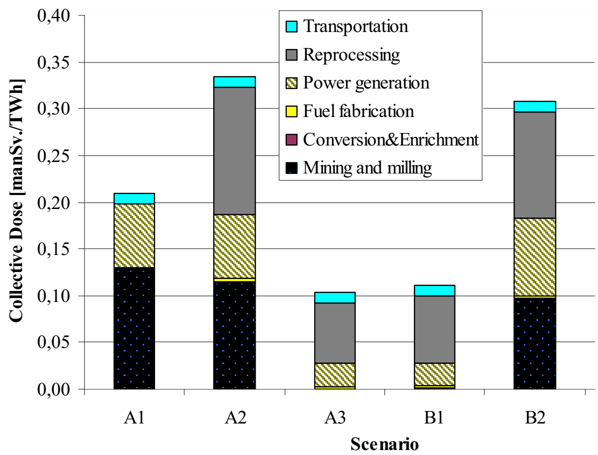

Figure 8.10: The collective dose to the public as a result of scenario operations The results from this were used in environmental indicator Env.10.1 and incorporated into the MCA results given above.

For indicator Soc.1.2, Fatalities due to accidents in a nuclear fuel cycle, relevant data was difficult to gather for some of the more ambitious scenarios and discussions within the project concluded that no great differences in these should be expected.

For indicator Soc.1.2, fatalities due to accidents in a nuclear fuel cycle, relevant data was difficult to gather. For A1 and A2 scenarios a value of 0,13 YOLL/TWh could be derived from [8.10].

After discussions within the project participants it was concluded that for more ambitious scenarios such as A3, B1 and B2 insignificant changes concerning this indicator should be expected. Hence, it can be concluded that no discrimination between the scenarios will be obtained using this indicator.

The final societal indicator, Soc.1.3, Proliferation Resistance, used an extension of the TOPS [8.8] methodology from France called SAPRA [8.9] to perform the analysis. The methodology produces a proliferation resistance index (PR-Index) by analysing the various barriers which would be in place in each of the scenarios and the effectiveness of the barriers. The scenario results are shown in Table 8.1 where the higher the index, the better the proliferation resistance. Because of the way the analysis is performed, each of the stages in the scenarios has its own PR-Index (this is what makes this indicator unsuitable for MCA). The PR-Index for each scenario marked minimum gives the value for the smallest PR-Index in the barrier analysis of each scenario.

Table 8.1: Proliferation resistance indices for each scenario

	A1	**A2**	**A3**	**B1**	**B2**
Minimum	0,25	0,32	0,32	0,32	0,25
Average PR-Index	0,40	0,45	0,47	0,47	0,44
Standard deviation (without reactor)	0,11	0,07	0,07	0,07	0,09

Given the inherent uncertainties in this type of analysis, it is difficult to provide a sharp distinction between the scenarios, but considering the average PR-Index and the minimum value, scenarios A3 and B1 would appear to be best. This is in agreement with the environmental and economic MCA results, that scenario B1 appears to be the best choice for waste minimisation.

8.11 Discussion

It is clear that for both clay and granite repositories and for the average weight sets considered here, scenario B1 (Gen IV fast reactor (FR) with infinite recycling of Pu and MA) has the highest value (i.e. the best result) – see Figures 5 and 6. Note that a salt repository was not considered explicitly in the MCA, however the long-term radioactive waste disposal indicators would score very closely to clay and granite and thus not influence the final rankings significantly.

B1 ranks highly because it recycles Pu and MA and therefore scores highly in the indicators referring to long term waste management and has a reasonable score in the economic area since the costs associated with FRs are relatively well known. The scores for individual indicators with clay and granite repositories are shown in Figure 8.3 and Figure 8.4 respectively.

In general scenario A3 (FR with infinite recycling of Pu alone) ranks second place in all the average weight sets. Although this has a FR as in B1, it does not recycle MA and therefore it's long term waste management scores are lower, however its economic scores are similar to scenario B1.

The rankings of the remaining three scenarios (A1, A2, B2) vary depending on the average weight sets applied. Scenario A1 (open cycle) scores very poorly in the indicators related to long term waste management since the irradiated nuclear fuel from the LWR is disposed of directly to a repository. However it scores highly in the economic area since the costs are well known and this is arguably the simplest scenario considered.

Scenario A2 (LWR with a single Pu recycle in a PWR) scores reasonably highly in some of the long term waste management indicators but very poorly in the human intrusion indicators because of the increased Pu in the repository waste from this scheme.

The final scenario, B2 (double strata with PWR and ADS) scores very poorly in the economic area. This is principally because this scenario is the most complex considered here and includes an ADS so that costs associated with R&D, construction and operation are the highest of all the scenarios. It has reasonable scores in long term waste management, comparable to scenario B1, but scores very badly in process dependant toxic emission releases because of its extremely complex nature.

As stated previously, information available for the societal indicators was quite sparse but the data that was available did not conflict with the MCA using economic and environmental indicators. However it is clear that the views of society in general are important for the choice of waste management techniques. Further work will certainly be required in this area.

The effect of uncertainties in the MCA parameters is important. The larger the uncertainties in these, the less distinct are the order of ranking of the scenarios. However the uncertainty level for most of the indicators is impossible to quantify currently. In order to illustrate the effect of indicator uncertainty on the scenario rankings, educated guesses were mainly used. Figure 8.7 to Figure 8.9 show the effect of half of the assumed errors, the assumed error and twice the assumed error respectively. From these it can be concluded that at the uncertainty levels assumed, the MCA technique can be applied to demonstrate the preference for different scenarios, provided that their MCA values differ by 5-10% or more. The other conclusion here is that more work is required in order to quantify these uncertainties.

Finally, it should be reiterated that there were several problems in applying MCA to the RED-IMPACT scenarios:

- Several indicators are missing (particularly representing societal considerations)
- Equilibrium not transition scenarios have been modelled to determine the indicators
- The uncertainties applied are very approximate
- Salt repositories have not been considered
- MCA weights used are from the RED-IMPACT participants only (other interested parties should also be involved in determining the weights)

As such, the work presented here is intended to be a demonstration of how the MCA technique could be applied to assist the decision making process. The results and conclusions of this MCA do not therefore represent an accurate or complete comparison of the different scenarios.

9. Summary & Conclusions:

The impact of partitioning, transmutation and waste reduction technologies on the final nuclear waste disposal was analysed within the EU-funded 'RED-IMPACT' project. The partnership of 23 organisations is drawn from European nuclear industry, waste agencies, research centres and universities. The system studies assume a potential evolution of P&T technologies and advanced fuel cycles, which can be deployed incrementally on an industrial scale as well as future developments such as thermal and fast reactors of the third and fourth generation (Gen III & Gen IV) and Accelerator Driven Systems (ADS). R&D needs for the development of processes and technologies have also been addressed.

A comprehensive inventory of existing and foreseen nuclear fuel cycle facilities in Europe has been made including a review of worldwide ongoing R&D programs on P&T. Thus, it was possible to select a set of three so-called "industrial scenarios", taking into account industrial feasibility of alternate strategies leading to increased actinide burning and reduced actinide generation. These are

- LWR with direct disposal (reference case A1),
- MOX fuel mono-recycling in LWR (A2),
- Plutonium recycling in Sodium Fast Reactors (A3).

In addition, three 'innovative scenarios' have been identified allowing multi-recycling of plutonium <u>and</u> minor actinides (MA) and making use of advanced reprocessing technologies like DIAMEX-SANEX or COEX^{TM} and PYRO processes:

- Fast neutron Gen IV system (B1),
- Simplified Double Strata with LWR and ADS (B2),
- Double Strata with Fast Reactors (FR), LWR and ADS (B3).

The selected fuel cycles had been identified to cover a wide range of representative waste streams, cycle facilities and performances. The assumed scenarios must be understood as reference possibilities and they do not exhaust further strategies including facilities combinations, designs, fuels and operations modes, as they are presently subject to R&D programmes.

With the exception of the B3 case, high and intermediate level waste streams have been calculated for all of these 'equilibrium' scenarios. In all of these scenarios, except for A1, intermediate level waste is assumed to be sent to a geological repository. The transition from the present situation develops towards new fuel cycle options. The scenarios are characterized as follows:

A1 – Once Through and direct disposal

The A1 scenario is the simplest one, serving as the reference case for our studies and used in some countries such as Sweden and Finland. Enriched uranium oxide fuel is loaded in nuclear reactors. Spent nuclear fuel is discharged and, after cooling and conditioning, disposed of in a deep geological repository. The scenario is characterised by its maturity and it is being industrialised on a fairly large scale. The calculations performed assume direct disposal of the present LWR spent fuel after an average burnup of 50 GWd/tHM. The initial ^{235}U enrichment is set at 4.2% and the reference cooling time is 50 years.

A2 – Mono-recycling of Pu in PWRs

The A2-scenario is based on single recycling of Pu in PWR. Pu is assumed to be separated from irradiated UOX-fuel through aqueous reprocessing. The recovered Pu is mixed with depleted uranium and manufactured into MOX fuel, which is burned in the same PWRs as the original UOX. Minor actinides (MA) and fission products (FP) are vitrified and sent to final (geological) disposal. The spent MOX, as well as the high level waste (HLW) from UOX reprocessing including MA, FP (except the volatiles) and reprocessing losses, are cooled before being sent to the geological repository. The losses in the reprocessing step are assumed to be 0.1% both for U and Pu. This scenario is characterised by its industrial maturity except for spent MOX disposal. UOX is reprocessed five years after discharge from the reactor. Two additional years of aging are supposed between Pu recovery (after reprocessing) and MOX-fuel loading in the reactor. The MOX consists of depleted uranium (0.25% ^{235}U) and Pu (8.5%). The burnup is assumed to be 50 GWd/tHM, for MOX and UOX in the same PWR used for the A1 scenario. Spent MOX, as well as HLW from UOX reprocessing, including MA, FP and the reprocessing losses, is cooled for 50 years before being sent to a repository. The liquid HLW from the industrial spent fuel reprocessing (as La Hague in France) contains all non-volatile fission products and minor actinides, plus small amounts of U and Pu. The liquid is calcinated, mixed with molten borosilicate glass, and poured into Universal Canisters. This scenario and its variants could be representative of a situation in which the FR introduction is postponed, for some time.

A3 – Multi-recycling of Pu in Fast Reactors

In scenario A3, energy is produced by fast reactors recycling Pu. The minor actinides are not separated, but remain in the final waste. Pu is recovered from the reprocessing of spent fast reactor fuel to prepare new MOX fuel. The reprocessing losses and all MA and most of the fission products are disposed of in a geological repository. Volatile fission products are released in a regulated way during reprocessing. The assumed loss rate during reprocessing is 0,1% for both U and Pu. The scenario simulation is based on the then available European Fast Reactor (EFR) CD9/91 configuration with MOX fuel in the core and with both axial and radial blankets. The MOX will include 23.2% Pu, reaching a burnup of 136 GWd/tHM in the core, 15 GWd/tHM in the axial blankets and 24 GWd/tHM in the radial blankets. The cooling time before reprocessing is five years, and the aging time before MOX manufacture is two years.

Advanced Scenarios

The second group of fuel cycle scenarios, referred to as the B-scenarios, involves more innovative technologies like the utilisation of fast spectrum Gen IV reactors and accelerator driven systems (ADS), as well as advanced reprocessing/partitioning techniques. Double strata approaches with combinations of LWR, FR and ADS are also included. It should be mentioned that the performance of the reprocessing/partitioning step will significantly govern the radiotoxic inventory discharged to the final repository. Its development would thus be as important as the different reactor and transmuter developments. This also applies to conditioning techniques, which will be decisive for the long-term retention and leaching of waste in a final repository.

B1 – Fast Neutron Gen IV Scenario

The scenario B1 is similar to A3 except that MA are partitioned and recycled, too. It represents a Gen IV solution for sustainable nuclear energy production with waste minimization based on fast reactors. The reactor has been chosen as a sodium-cooled European Fast Reactor, EFR, but loaded with an advanced fuel including MA (Np, Am and

Cm). Meanwhile, a new SFR system is under study. This new design was not available at the beginning of the RED-IMPACT investigations and was therefore not considered. The minor actinides are homogeneously dispersed in the fuel of the central fissile region. The core is equipped with depleted uranium blankets. The EFR configuration is identical to that of scenario A3, with initial addition of 2.7% MA to the MOX fuel. For this scenario, the feasibility of the (remote) fuel manufacture and of the partitioning processes has still to be demonstrated.

B2 – Simplified Double Strata Scenario (PWR and ADS)

The B2 scenario supposes the availability of ADS together with current or advanced LWRs. The scenario is similar to the scenario A2, but with the introduction of a "second stratum", based on a fast spectrum ADS, which transmute the remaining Pu and MA. All MA from the LWR fuel reprocessing plus the Pu and MA from the spent MOX fuel are recycled in the fast neutron spectrum ADS. After cooling, the ADS fuel is reprocessed using pyro-metallurgical processes.

Most of the power production in the B2 scenario occurs in an LWR-park with UOX fuel. Its characteristics are the same as for the A1 scenario, the burn-up is 50 GWd/tHM; the initial ^{235}U-enrichment is 4.2%. After 4 years of cooling, all the irradiated fuel is reprocessed by advanced PUREX where Pu is separated in one stream and MA in another. All the recovered Pu is recycled as MOX and used once in the LWR-park. The fuel is cooled during two years between reprocessing and fuel manufacturing. In the second stratum, all the MA from the LWR fuel reprocessing plus the Pu and MA from the spent MOX are recycled in a fast neutron spectrum ADS in an inert matrix as nitride fuel, reaching a burn-up of 150 GWd/tHM. This fuel is reprocessed by pyro-metallurgical methods two years after discharge from the core. The HLW from the UOX, MOX and ADS fuel partitioning, which include the FP and reprocessing losses is sent to the repository after 50 years of cooling time.

B3 – Double Strata (PWR, FR and ADS)

The B3 scenario is rather similar to B2 with an additional step of continuous Pu recycling in fast reactors within the first strata. It was decided not to perform detailed studies on this option because most of the results can be deduced from the other scenarios.

Transition Scenarios

Apart from studying the scenarios described above in a state of equilibrium, a set of transition scenarios were also investigated. The goal of doing so is first, to assess the feasibility of the selected scenarios and, second, to learn about the waste generation to provide basic information concerning strategic decisions with long-term effects. The three transition scenarios depart from a current nuclear power-park and end in equilibrium situations as close as possible to a corresponding equilibrium scenario:

- final situation corresponding to A3.
- final situation corresponding to B1.
- final situation corresponding to B2.

The power production in the 800 $TW_{el}h$ per year nuclear-park, roughly corresponding to the present (2007) annual European nuclear power production, is assumed to be constant. At year zero (assumed 2010), the entire nuclear-park consists of LWRs. It is supposed in this study that the end of the transition period will occur within 200 years, so results will be limited to this interval of time. Year 2040 is put as the year of industrial deployment of the new reactors, FR and ADS. The lifetime of all reactors is set to 60 years. The plant availabilities are postulated to 90% for all cores except the ADS, which is assumed to be of 70%.

Waste disposal

The data on the waste streams from all these scenarios provide the input to specific analyses on the impact on geological disposal in different host formations such as granite, clay and salt. The waste packages include the High Level Waste (HLW) as well as Intermediate Level Waste (long lived ILW) for the different options. The assessments do not only address the radiotoxicity of the waste but also the

- total radioactive inventory, discharges during reprocessing etc.
- thermal power of the waste packages
- gamma and neutron emissions of waste compositions,
- waste package radiation levels,
- corrosion of matrices,
- solubility of radioisotopes and
- transport of radioisotopes through engineered and geological barriers
- assessment of resulting doses from the repository

The investigations refer to the following geological formations and example sites:

Granite:	- Spanish concept development by Enresa - Czech reference disposal concept developed by NRI
Clay:	- SAFIR 2 concept, developed by ONDRAF/NIRAS, Belgium
	- Dossier, 2005 concept developed by ANDRA, France
Salt:	- German concept developed by GRS in Germany

Granite Repositories

For the Spanish granite repository concept, the leaching of radionuclides in case of the expected evolution of the disposal system was calculated with a view to comparing the total dose due to disposal of high-level waste and long lived ILW from fuel cycle scenarios A1 through B2. The annual releases to the biosphere are insignificant in all cases, as they are far below the natural background radiation level. The relative contribution of individual nuclides ^{129}I, ^{36}Cl and ^{14}C clearly dominate the dose to the biosphere for more than 100 000 years. After one million years, ^{135}Cs gives an important contribution to the total dose.

Clay Repositories

The repository considered for the RED-IMPACT calculations is the reference repository that is described in detail in the Belgian SAFIR 2 report and is assumed to be excavated in the Boom Clay Formation at the Mol site. Also for disposal in clay, doses are calculated for the expected evolution of the disposal system under the different scenarios. In the case of disposal in clay, the total dose is mainly due to contributions of ^{129}I, ^{79}Se and ^{126}Sn. Doses due to releases of actinides occur after several millions of years and are at least two orders of magnitude lower than the maximum dose, which is due to ^{129}I and ^{79}Se.

Salt Repositories

In the case of disposal in a salt dome, no releases occur in the case of the expected evolution of the disposal system; in other words the dose is equal to zero as long as the salt dome remains intact. Therefore, dose calculations are made for a disturbed evolution scenario, in which it is assumed that a layer of anhydrite is located in proximity to the final repository and that in response to mechanical stresses in the anhydrite, hydraulic connections between the aquifer and the repository are established and groundwater may flow into the repository. The resistance of the flow paths controls the amount of the water inflow. It is further assumed that the sealing of the shaft will fail. In addition, the existence of brine pockets in the vicinity of the disposal drifts cannot be excluded. The releases for all scenarios remain very low ($<10^{-12}$ Sv/yr).

Impact of P&T on Waste Disposal

In general, the results show that advanced fuel cycles reduce the thermal load and thus resulting in a smaller required size of the geological repository for disposal in clay, or hard rock formations. Recycling of all the actinides results in a reduction of the necessary gallery length for HLW (depending on geology and design) by a factor 3 to 6 after 50 years intermediate storage relative to the direct disposal of spent fuel.

If additionally caesium and strontium are extracted from the high-level waste for separate storage and decay, the reduction factor will be 10 or more. In the frame of the project, the feasibility and the impact of the Cs or Sr separation were not assessed.

For all repository concepts and for each analysed fuel cycle scenario, the impact evaluated as the post-closure maximum dose to an individual of the critical group, is a few orders of magnitude below the current dose constraints imposed by the regulatory authorities (0.1 – 0.25 mSv/yr) and far below the natural background radiation level of the order of 3 mSv/yr. The results presented show that this maximum dose is caused mainly by the mobile fission products ^{129}I, ^{79}Se, ^{135}Cs, ^{99}Tc and ^{126}Sn, and the mobile activation products ^{14}C and ^{36}Cl. In all repository cases, the possible dose from minor actinides (^{237}Np and ^{238}U chains) is always orders of magnitude lower than this maximum dose, occurring several millions of years after disposal; this is because of their very low solubility and the high sorption capabilities of the natural and engineered barriers for these elements. It should be indicated that neither the repository design nor the waste containers had been optimized to adapt to the P&T waste.

In the fuel cycles in which spent fuel assemblies are sent to the repository (UOX in scenario A1 or MOX in scenario A2) the main contributor to the dose is the fission product ^{129}I, followed by other fission or activation products depending on the specific behaviour and transport properties of the repository systems, under normal evolution.

When reprocessing steps are included in the fuel cycle (scenarios A3, B1, B2 and partially in A2), the main waste forms to be disposed off consists of vitrified HLW and compacted long lived ILW. Consequently, the maximum dose decreases in general by one order of magnitude. This lower dose is not a consequence of the application of any transmutation strategy, but is due to the removal of 98% of the iodine from the waste stream during reprocessing. In these scenarios, the main contributors are ^{135}Cs and ^{14}C (granite), ^{126}Sn and ^{79}Se (clay) or ^{135}Cs and the ^{238}U-^{226}Ra chain (salt). In the case of scenario B2, enrichment in ^{15}N or a different chemical form of the fuel is required to avoid larger contributions of ^{14}C, arising as an activation product in the nitride fuel assumed for the ADS. It should be noted that the iodine removed in present reprocessing facilities is released in a regulated way leaving the cumulative dose released to the biosphere in the different scenarios effectively unchanged.

The basic differences in the repository systems due to the different fuel cycle scenarios are the waste forms and their respective inventories. A simple sensitivity analysis of matrix stability was carried out for disposal in granite. This study showed the non-negligible effect of the dissolution rate (lifetime) on the radiological impact of key radio-nuclides in particular those which have no solubility limit.

A more complete comparison of the different fuel cycle scenarios can be made when strategies for management of intermediate level waste have been defined. Concerning the volatile isotopes, ^{129}I, ^{14}C and ^{36}Cl, which today are released to the environment, they could possibly be captured and immobilized in specific matrices, unless industrially applicable transmutation methods for these long-lived radioisotopes will be found, in the future.

An important argument in favour of geological disposal is that placing HLW waste deep underground is a robust method of reducing the potential for, and likelihood of, human intrusion, compared e.g. to the case of a surface storage. It cannot be guaranteed, however, that intrusion might not occur at some time in the future after administrative controls have been discontinued. Such events and their consequences have been assessed, even though their likelihood may be low.

Human Intrusion Scenarios

As a complement to the safety and performance analyses, investigations addressed the human intrusion e.g. by a geotechnical worker into a deep repository for high-level waste and spent fuel types arising from five advanced fuel cycle scenarios considered within the project. The Cigar Lake uranium ore is used here as a reference. This rock has a density of 2.8×10^3 kg/m^3 and contains 4.5×10^4 Bq/kg of ^{235}U and 1×10^6 Bq/kg ^{238}U. The comparison of the results indicates that the calculated dose for the high-level waste and spent fuel types, in the geotechnical worker human intrusion scenario, always exceeds the calculated dose from intrusion into the natural (Cigar Lake) uranium ore body for the assessment period of one million year duration. However, the radio-toxicity in the high level waste or spent fuel as well as human intrusion doses after 500 years are drastically reduced by the transmutation of the actinides. Although the calculated total dose varies considerably between high-level waste and spent fuel types (by up to two orders of magnitude at any one time), the variation is less pronounced in the comparison for the geotechnical worker scenario. Scenario A2 - 1 MOX spent fuel assembly - has the highest calculated total dose at all times, whereas Scenario B2 - 1 vitrified UOX-HLW universal canister – always has the lowest calculated total dose.

Economical, environmental and social aspects

Evaluating actinide minimization systems and industrialised P&T in general requires an assessment of relevant nuclear fuel cycles with particular regard to the economic, environmental and societal advantages/disadvantages (i.e. the sustainability of the fuel cycles). Thus, a set of indicators has been derived for each of these areas. Normally these indicators would be used in, for example, Multi Criteria Analysis (MCA) in order to rank the scenarios. However, in this study, there are several problems in applying MCA:

- Several indicators are missing (particularly representing societal considerations)
- Equilibrium not transition scenarios have been modelled to determine the indicators
- The uncertainties applied are very approximate
- Salt repositories have not been considered
- MCA weights used are from the RED-IMPACT participants only (other interested parties should also be involved in determining the weights)

Despite these problems, an example analysis has been performed in order to demonstrate how this technique can be used to assist the decision making process.

Variant scenarios:

Within the RED-IMPACT project, only a limited number of fuel cycle variants have been investigated, which are complementary or alternates to the selected scenarios. This applies to especially to Inert Matrix and Thorium-based fuels which show better Pu conversion ratios. The calculations were mainly based on PWR as a reference. Other reactor types like BWR, Heavy-Water Reactors, High-Temperature Reactors, Molten-Salt Reactors need to be included into further studies as they exhibit different neutron economies, neutron spectra and waste compositions. Reduced generation of actinides by alternate fuel cycles using e.g. thorium is another approach, which requires further attention.

Conclusions:

The impact of fuel cycle scenarios on the geological disposal is mainly governed by the heat generation of the waste, the geochemical stability of the waste packages and waste matrices under the specific geological conditions of a repository, the solubility of the individual radioisotopes and their mobility in the buffer and host formation etc. The performance of the reprocessing/partitioning processes also plays a major role with regards to the composition of the waste and the related releases to the environment (e.g. ^{129}I discharge) in the reprocessing step.

For the HLW, the normalized doses of all investigated types of repositories (granite, clay and salt) are strongly governed by the soluble and mobile long-lived fission and activation products (iodine, chlorine, radiocarbon, caesium, selenium etc.) and not by the contribution of the MA, which are considered to be strongly immobilized under the given geochemical (reducing) conditions.

However, in the case of human intrusion scenarios, there is a clear advantage for P&T due to the reduced radio-toxicity of the waste.

After a 50 years cooling time, high-level radioactive waste (HLW) arising from advanced fuel cycles generates less heat than does the equivalent amount of spent fuel arising from the "once through" fuel cycle; this smaller thermal output allows to reduce the length of the needed disposal galleries or the size of the repository.

When longer cooling times (e.g. 100 or 200 years) are considered, the thermal output of HLW from advanced fuel cycles decreases faster than the one of spent fuel from the once through fuel cycle, because of the smaller amounts of actinides that are present in the HLW.

Separation of Cs and Sr during partitioning in combination with an advanced fuel cycle results in HLW with a very limited thermal output and allows a strong reduction in the needed repository size. However, the separated Cs contains the long-lived (half-life 2.3 million years) ^{135}Cs isotope, which means that the Cs-waste has also to be disposed in a geological repository according to current European regulations. The feasibility of this option was not studied or assessed during the project.

Scenarios applying reprocessing show lower environmental impacts in the time periods below 1 Million years mainly due to the reduced iodine fraction in the vitrified waste. The resistance of the glass matrix in comparison to the leach resistance of UO_2 fuel matrix (direct disposal) is another parameter influencing the doses due to the radioactive waste disposal. This result also supports the development of more stable waste matrices for Conditioning (P&C) and more accurate models to better assess this parameter.

However, the different scenarios are also accompanied by different generation rates of long lived ILW. The contribution to the biosphere doses by long lived ILW is in some cases even higher than for HLW. The reason is that this type of waste is not embedded in a stable matrix but more or less directly exposed to leaching after closure of the repository. In the evaluations made for disposal in granite, peak doses may exceed those of the HLW much earlier and may even dominate in the first ten thousand years. Lower contribution of long lived ILW is expected for clay repositories. The high contribution of ^{14}C stemming from nitride fuels of fast systems and the activation of structural material as well as spallation products from ADS need to be included in an integral comparison. But, the contribution from the activation of structural materials can be minimised by suitable choices of structural materials in transmutation devices. Much more attention must be put on the generation of long lived ILW and the management of long-lived fission and activation products including their conditioning and the development of repository concepts including appropriate engineered barriers. For the transition scenarios it turned out that the early introduction of advanced reprocessing is of utmost importance.

In a full 'cradle-to-grave' evaluation, the radiological impacts should not only be focused on the final repository but should be extended to all stages of the fuel cycle starting from mining via conversion to fuel fabrication, reactor operation, transports, reprocessing and re-fabrication of transmutation fuel towards storage and disposal of the final waste. It has to be noted that the (remote) fabrication of transmutation fuel is a significant technical and economical challenge. The latter is an indispensable step to transmute long-lived actinides and possibly some long-lived fission products, too.

It can be stated that the radioactive waste from all investigated scenarios can be safely managed and that the radiological consequences all remain within the licensing limits of the geological repositories.

It is evident that for closed nuclear fuel cycles, reprocessing of the spent fuel is required to allow for a separation of different waste streams and potentially reusable material such as plutonium and uranium, from a fuel resource saving point of view. Disposal of long-lived waste components in geological repositories would be required no matter what scenario or national strategy is chosen for the spent fuel and HLW. However, the absence of a 'socially accepted decision path' to such a disposal is one of the main impediments to the continued and future use of nuclear energy. Concerns of the public related to the long life of the waste could possibly be overcome by P&T by reducing half-life to a couple of hundred years for most transuranic elements.

Efficient transmutation of MA can be carried out in an ADS or Fast reactors or a symbiotic combination of different systems with certain facilities being totally dedicated as waste burners and others as energy producers. Potential transmutation of long-lived fission products remains to be an open issue.

Reprocessing (or partitioning) of the spent fuel for more sustainable nuclear energy systems and fuel cycles would be required independent of the transmutation method being followed. Therefore, in the immediate future, efforts for the advanced partitioning processes should be reinforced towards pilot and test facilities for optimised separation processes in close cooperation with fuel fabrication teams and the geological disposal community. The latter should especially take account of the requirements to accommodate the waste streams emanating from the advanced (minor-actinide) reprocessing systems with a view to specific conditioning or transmutation whether in sub-critical or critical devices.

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ANNEX I

RED-IMPACT Partner Organisations

Partner No.	Participant Organisation	Acronym	Country
1	Kungliga Tekniska Högskolan	KTH	Sweden
2	Forschungszentrum Juelich GmbH	FZJ	Germany
3	Belgonucleaire	BN	Belgium
4	British Nuclear Fuels PLC	BNFL	United Kingdom
5	Commissariat à l'Energie Atomique	CEA	France
6	Centro de Investigaciones Energeticas Medioambientales y Tecnologicas	CIEMAT	Spain
7	Center Of Technology and Engineering for Nuclear Projects	CITON	Romania
8	Compagnie Générale de Matières Nucléaires	COGEMA	France
9	Empresarios Agrupados Internacional S.A.	EA	Spain
10	EnBw Kraftwerke AG, Kernkraftwerk Philippsburg	KKP	Germany
11	Empresa Nacional De Residuos Radioactivos S.A.	ENRESA	Spain
12	FRAMATOME ANP	FANP SAS	France
13	FRAMATOME ANP GmbH	FANP GmbH	Germany
14	Gesellschaft für Anlagen- und Reaktorsicherheit GmbH	GRS	Germany
15	Universitaet Stuttgart	USTUTT	Germany
16	European Commission, Joint Research Centre - Institute for Transuranium Elements	EC-JRC-ITU	Germany
17	United Kingdom Nirex Limited	Nirex	United Kingdom
18	Nuclear Research & consultancy Group	NRG	Netherlands
19	Sprava ulozist radioaktivnich odpadu	RAWRA	Czech Republic
20	Studiecentrum voor Kernenergie - Centre d'Etude de l'Energie Nucléaire	SCK-CEN	Belgium
21	Svensk Kaernbraenslehantering AB	SKB	Sweden
22	VUJE Trnava, a.s	VUJE	Slovakia
23	Nuclear Research Institute Rez	NRI	Czech Republic

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