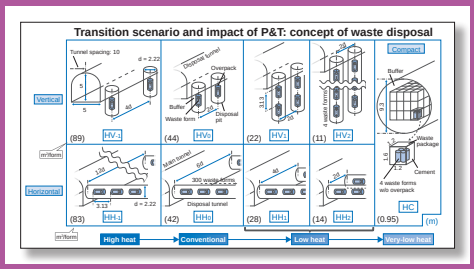
Nuclear Science
2011

OECD

Potential Benefits and Impacts of Advanced Nuclear Fuel Cycles with Actinide Partitioning and Transmutation



AEN
NEA

NUCLEAR ENERGY AGENCY

Nuclear Science ISBN 978-92-64-99165-1

Potential Benefits and Impacts of Advanced Nuclear Fuel Cycles with Actinide Partitioning and Transmutation

NEA No. 6894

NUCLEAR ENERGY AGENCY
Organisation for Economic Co-operation and Development

ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

The OECD is a unique forum where the governments of 34 democracies work together to address the economic, social and environmental challenges of globalisation. The OECD is also at the forefront of efforts to understand and to help governments respond to new developments and concerns, such as corporate governance, the information economy and the challenges of an ageing population. The Organisation provides a setting where governments can compare policy experiences, seek answers to common problems, identify good practice and work to co-ordinate domestic and international policies.

The OECD member countries are: Australia, Austria, Belgium, Canada, Chile, the Czech Republic, Denmark, Estonia, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Israel, Italy, Japan, Luxembourg, Mexico, the Netherlands, New Zealand, Norway, Poland, Portugal, the Republic of Korea, the Slovak Republic, Slovenia, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The European Commission takes part in the work of the OECD.

OECD Publishing disseminates widely the results of the Organisation's statistics gathering and research on economic, social and environmental issues, as well as the conventions, guidelines and standards agreed by its members.

This work is published on the responsibility of the OECD Secretary-General.
The opinions expressed and arguments employed herein do not necessarily reflect the official views of the Organisation or of the governments of its member countries.

NUCLEAR ENERGY AGENCY

The OECD Nuclear Energy Agency (NEA) was established on 1 February 1958. Current NEA membership consists of 30 OECD member countries: Australia, Austria, Belgium, Canada, the Czech Republic, Denmark, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Italy, Japan, Luxembourg, Mexico, the Netherlands, Norway, Poland, Portugal, the Republic of Korea, the Slovak Republic, Slovenia, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The European Commission also takes part in the work of the Agency.

The mission of the NEA is:

- to assist its member countries in maintaining and further developing, through international co-operation, the scientific, technological and legal bases required for a safe, environmentally friendly and economical use of nuclear energy for peaceful purposes, as well as
- to provide authoritative assessments and to forge common understandings on key issues, as input to government decisions on nuclear energy policy and to broader OECD policy analyses in areas such as energy and sustainable development.

Specific areas of competence of the NEA include the safety and regulation of nuclear activities, radioactive waste management, radiological protection, nuclear science, economic and technical analyses of the nuclear fuel cycle, nuclear law and liability, and public information.

The NEA Data Bank provides nuclear data and computer program services for participating countries. In these and related tasks, the NEA works in close collaboration with the International Atomic Energy Agency in Vienna, with which it has a Co-operation Agreement, as well as with other international organisations in the nuclear field.

Corrigenda to OECD publications may be found online at: *www.oecd.org/publishing/corrigenda*.

Cover illustration: Dr. Kenji Nishihara (Japan Atomic Energy Agency).

Foreword

Under the guidance of the OECD Nuclear Energy Agency (NEA) Nuclear Science Committee (NSC) and the mandate of the Working Party on Scientific Issues of the Fuel Cycle (WPFC), the Task Force on Potential Benefits and Impacts of Advanced Fuel Cycles with Partitioning and Transmutation (TFPT) carried out a comparative analysis of studies performed in several international laboratories on the impact of advanced nuclear fuel cycles, including partitioning and transmutation (P&T), on geological repository performance.

Several impact studies have been performed over the last few years that have underlined the role of radiotoxicity and heat load along with their potential reduction. The task force investigated the motivations, hypotheses and contexts of the different impact studies, and analysed the requirements, risks, objectives and criteria for P&T with respect to geological disposal and the relevant nuclear development scenarios.

This report examines the results obtained in previous studies in different international laboratories and provides further insight, for example, into the role of long-lived fission products, cooling times and the impact on americium build-up, strategies for caesium and strontium, and the extent of losses during fuel processing. In addition to assessing the impact on geological repository characteristics and performances, it indicates possible goals for future studies and provides recommendations. The outcomes of the analysis can be used to guide the development of appropriate P&T strategies in favourable combination with geological disposal. It should also help build bridges to help promote better collaboration across the fields of nuclear science and geological disposal.

Acknowledgements

The NEA Secretariat expresses its sincere gratitude to the members of NEA Nuclear Science Committee (NSC), the NEA Committee for Technical and Economic Studies on Nuclear Energy Development and the Fuel Cycle (NDC), and the Integration Group for the Safety Case (IGSC) of the NEA Radioactive Waste Management Committee (RWMC) for their valuable comments.

Table of contents

List of figures

List of tables

List of boxes

Executive summary

Partitioning and transmutation (P&T) studies have been performed since the early 1970s. The transmutation physics issues were the first to be successfully tackled. There was also an early recognition of the complexity of the chemistry associated with partitioning, and a significant number of experimental activities have been launched since the early 1980s in the few laboratories across the world that are equipped for that type of research. In parallel, major efforts have been devoted to the development of geological repository concepts, with some experimental realisations and the development of appropriate normative frameworks, mostly at national levels.

International organisations like the OECD Nuclear Energy Agency (NEA) and the International Atomic Energy Agency (IAEA) have promoted international collaboration and assessment studies in both the geological and P&T fields. The result has been a much better understanding of the issues at stake and a growing awareness that only a holistic approach, which should account for the performance of the reactor and the associated fuel cycle and for the specific geological repository characteristics and requirements, can help to shape a clear, optimised and plausible vision for nuclear waste management that can benefit from wide public acceptance.

In a scenario which projects a growing role for nuclear energy to meet energy demand worldwide, sustainability becomes a predominant concern, meaning that preservation of natural resources, providing for waste disposal and proliferation resistance are criteria as important as economy and safety; this was pointed out within the Generation IV International Forum and assessment of future fuel cycles.

Partitioning and transmutation of actinides has been considered as a means of reducing the burden on a geological repository. Since plutonium and the minor actinides are mainly responsible for the long-term radiotoxicity, when these nuclides are removed from the waste (partitioning) and fissioned (transmutation), the remaining waste loses most of its long-term radiotoxicity.

The P&T strategy of recycling actinides allows in principle a combined reduction of the amount of radioactive waste to be stored and the associated residual heat, although processing will increase the amount of intermediate and low-level waste. Despite a very large number of studies, both at national and international levels, there is not a general consensus on the impact of such P&T strategies on repository performance. This is due partly to the use of different repository environments and partly to the repository performance analysis approach and assumptions.

The major objective of the present study has been to gather and analyse the results of different studies performed to assess the potential impact of P&T on different types of repositories in various licensing and regulatory environments. This means that criteria, metrics and impact measures have been analysed and compared, in order to give as far as possible an objective comparison of the state of the art to help shape decisions on R&D needs for future advanced fuel cycles.

The consensual outcome of the present assessment reflects the varied expertise and backgrounds of the members of the Task Force, spanning from reactor safety and fuel cycle physics to radiochemistry and geological disposal concepts analysis. A major outcome is the recognition that one significant effect of P&T is that the inventory of the emplaced materials is

much lower on an energy-generated basis for the actinides that can have an impact on the repository, even if not overwhelmingly high, since by reducing waste heat production a more efficient utilisation of the repository space is expected. Moreover, the inventory reduction can have the effect of making the uncertainty about repository performance less important both during normal evolution and in particular in the case of hypothetical disruptive scenarios that can bring man in direct contact with the disposed waste; these scenarios seem to be affected by the hazard (radiotoxicity) and not so much by the geology, because the P&T of the actinides does reduce the hazard of the emplaced materials.

It is also strongly emphasised that, while P&T will never replace the need for a deep geological repository, it can be argued that P&T has the potential to significantly improve public perception of the ability to effectively manage radioactive wastes by largely reducing the transuranic (TRU) waste masses to be stored and, consequently, to improve public acceptance of the geological repositories. Both issues are important with regard to the future sustainability of nuclear power.

Regarding the organisation of the present report, Chapter 1 defines the objective of the study and its context and reviews radioactive waste management principles and basic concepts of geological disposal. Chapter 2 presents various issues related to waste management and disposal for which an impact from P&T can be expected. Chapter 3 gives an overview of the main results that are available from previous studies. Finally, conclusions that can be drawn from the results are presented in Chapter 4.

Chapter 1. Introduction

1.1 Objective of the study

By 2050, global electricity demand is expected to have increased by about a factor of 2.5 [1] over that of today. Simultaneously, the world faces environmental threats from climate change caused by anthropogenic CO_2 emissions. Nuclear energy offers the opportunity of meeting a significant part of the anticipated increase in electricity demand in combination with a reduction of the CO_2 emissions. Therefore, a strongly increasing demand for nuclear power can be expected:

- Nuclear installed capacity could be multiplied by a factor of 3 to 4 by 2050 (1 500-2 000 GWe). This will be made possible with light water reactors (LWR) using uranium fuel, which will dominate the world market for the first half of the 21st century (Gen-II, Gen-III).
- These reactors will have a minimum lifetime of 60 years. The countries which will build reactors in 2050, aimed at operating until 2110, might have to take into consideration uranium supply issues.
- There is a need for a clear and plausible vision for nuclear waste management.

In such a scenario, sustainability becomes a predominant concern; preservation of natural resources, providing for waste disposal and proliferation resistance have become criteria as important as economy and safety.

As for fuel cycles, different options have been promoted, mainly open or closed fuel cycles; the latter include a so-called partitioning and transmutation (P&T) strategy. With the open cycle, no sustainability is guaranteed, due to uranium availability and cost issues. Historically this option has been associated with LWR, which effectively use only ~1% of the mined uranium. The closed fuel cycle has historically been associated with enhanced resource utilisation, fuel reprocessing and recovery of plutonium for use in new fuel. As for P&T, it has historically been associated with the waste minimisation goal, and has mostly been discussed over the last two decades as a specific option.

Recently, within a wide consensus, the Generation IV International Forum (GIF) [2] has defined a set of more general goals for future systems in four broad areas:

- sustainability – more efficient use of the available uranium resources and waste minimisation;
- enhanced economics;
- safety and reliability;
- proliferation resistance and physical protection.

The objectives of the Generation IV nuclear systems include P&T (waste minimisation) and this option is no longer seen as independent, but as consistent with sustainability and non-proliferation objectives, i.e. as an integrated part of advanced fuel cycles including e.g. homogeneous and heterogeneous TRU recycling in fast reactors and accelerator-driven systems (ADS). Partitioning and transmutation is considered in principle a means of reducing the burden on a geological disposal. The purpose of this report is to review the extent to which P&T will impact geological disposal depending on the disposal environment and the details of the P&T approach.

Spent fuel from nuclear power plants has to be managed in a safe manner, respectful of the environment and acceptable for the public. At present two management options are considered worldwide: direct disposal of the spent fuel and reprocessing. This management of the spent fuel is a major challenge for all countries where nuclear energy has been developed, and whatever perspective is applied to its continued and future utilisation, from further development to progressive phase-out.

As will be discussed in detail in the following sections, there is a general consensus that the spent fuel (or, more precisely, some of its constituents) discharged from nuclear power plants comprises the main contribution to nuclear waste. Most of the hazard (see below) from the spent fuel stems from only a few chemical elements – plutonium, neptunium, americium, curium and some long-lived fission products such as iodine and technetium at concentration levels of kilograms per tonne.

These radioactive by-products, although present at relatively low concentrations in the spent fuel, are a hazard to life forms when released into the environment. As such, a worldwide consensus has been reached that their disposal requires isolation from the biosphere in stable deep geological formations for long periods of time [6].

A measure of the potential hazard of these elements is provided by the *toxicity* and in particular the *radiotoxicity* arising from their radioactive nature rather than their chemical form. Partitioning and transmutation (P&T) of the actinide elements has been considered as a way of reducing the burden on a geological disposal. Since plutonium and the minor actinides are mainly responsible for the long-term radiotoxicity, when these nuclides are removed from the waste (partitioning) and fissioned (transmutation), the remaining waste loses most of its long-term radiotoxicity. However, P&T of the actinide elements has no impact on the inventory of fission products in the radioactive waste. It has been shown by many studies that the radiotoxicity 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 (MA) are recycled. A prerequisite for these reduction figures is a nearly complete fissioning of the actinides, for which multi-recycling is a requirement. To maximise the benefit of P&T, actinide losses during reprocessing and re-fabrication must be well below 1% and probably in the range of 0.1%, mandatory for Pu.

Moreover, the P&T strategy of recycling actinides allows in principle a combined reduction of the amount of high-level radioactive waste to be stored and the associated residual heat, although processing will increase the amount of intermediate-level and low-level waste. That increase should be systematically evaluated since the amounts are determined by the technologies being used and the ability of the operators to minimise operational wastes. Despite a very large number of studies, both at national and international levels, there is not a general consensus on the impact of such P&T strategies on the repository performance, caused partly by the use of different repository environments and partly by the repository performance analysis approach and assumptions.

The major objective of the present study is to gather and analyse results of different studies performed to assess the potential impact of P&T on different types of repositories in different licensing and regulatory environments. This means that criteria, metrics and impact measures will also be analysed and compared, in order to give as far as possible an objective state of the art that can help to shape decisions on different options of future advanced fuel cycles.

The outcome of the present study should help to develop a common understanding about the P&T potential and indicate both to the P&T research community and to research policy makers the interest to pursue research on specific fuel cycles.

1.2 Radioactive waste management

All the operations associated with the production of nuclear energy generate radioactive waste. That waste has to be managed adequately according to internationally accepted principles [3]. The main objective of radioactive waste management is to deal with radioactive waste in a manner that protects human health and the environment now and in the future without imposing undue burdens on future generations.

Radioactive wastes arise in a wide range of concentrations of radioactive materials and in a variety of physical and chemical forms. Several methods can be used to categorise radioactive waste. Classification systems based on the IAEA waste classification are in use in many countries. In 2009, the IAEA updated its waste classification report of 1994 [4]. The new report [5] considers six classes of waste among which the following three can be considered relevant for the present study:

- *Low-level waste (LLW)*: Waste that is about clearance levels, but with limited amounts of long-lived radionuclides; such waste is suitable for disposal in engineered near-surface facilities.
- *Intermediate-level waste (ILW)*: Waste that, because of its content, particularly of long-lived radionuclides, requires a greater degree of containment and isolation than that provided by near-surface disposal. However, ILW needs no or limited provision for heat dissipation; it may contain long-lived radionuclides, in particular alpha-emitting radionuclides, that will not decay to a level of activity concentration acceptable for near-surface disposal during the time for which institutional controls can be relied upon. ILW requires disposal at greater depths, of the order of tens to a few hundred metres.
- *High-level waste (HLW)*: Waste with levels of activity concentration high enough to generate significant quantities of heat by radioactive decay or waste with large amounts of long-lived radionuclides that need to be considered in the design of a disposal facility for such waste. Disposal in deep, stable geological formations usually several hundreds of metres below the surface is the generally recognised option for disposal of HLW.

There are other waste classification systems, such as that for the United States as defined by the United States Nuclear Regulatory Commission and legislation. The main differences are in the definitions for low-level waste into Classes A, B and C, as well as a category of "greater than Class C" (GTCC) waste that is similar to the ILW category listed above. The functional definition of HLW is essentially the same in that it identifies highly hazardous long-lived materials, such as spent fuel, that require isolation as provided by deep geological disposal.

The present report will mainly focus on high-level wastes arising from the different fuel cycles; these can be, depending on the considered fuel cycle, conditioned high-level waste from reprocessing, e.g. vitrified waste, or spent nuclear fuel. Intermediate-level waste will be considered where relevant and insofar as sufficient information on these waste types is available.

1.3 Geological disposal

Whatever the future of nuclear power, it is universally recognised that a safe and acceptable final solution must be pursued for existing and projected inventories of high-activity, long-lived radioactive waste. In the past various options have been considered and discarded for political or safety reasons, such as disposal under the seabed or in geological subduction zones, or launching the waste into space. Transmutation of part of the waste, e.g. the higher actinides, through use of advanced fuel cycles, although perhaps feasible in the coming decades, would not eliminate the need to manage the currently existing waste (i.e. vitrified waste) and remaining high-activity, long-lived radioactive waste, e.g. fission products, and other activated materials from future fuel cycles [6].

Geological disposal has been investigated worldwide for several decades, and is now being further developed in several countries, as the ultimate waste management endpoint for high-activity, long-lived radioactive waste. The concept is that at depths of hundreds of metres, the host rock formation is intended to provide a stable geological and chemical environment that inhibits waste degradation and transport of hazardous radionuclides and sufficient isolation to protect the disposal facility from future human activities and from natural processes. Furthermore, careful selection of the host formation and the repository site aims to reduce as far as practicable the risks of perturbations from such processes.

A variety of geological settings have been investigated worldwide as potential host formations for geological disposal. The main types of geological settings considered are:

- *Hard rock formations*: The considered formations are often granite formations; this option is currently studied in e.g. Canada, Finland, Japan, Spain and Sweden.
- *Argillaceous formations*: The considered formations range from plastic clays, over indurated clays to mudstones; this option is currently studied in e.g. Belgium, France, Japan, the Netherlands, Spain, Switzerland and recently in Germany.
- *Salt formations*: Salt layers as well as salt domes are being considered. The only geological radioactive waste repository that is already in operation for the disposal of long-lived transuranium waste, the Waste Isolation Pilot Plant (WIPP) facility in the United States, is located in a salt layer. Disposal in salt domes is studied in Germany and the Netherlands.
- *Formations of volcanic origin*: Examples of considered formations of volcanic origin are welded tuff and basalt. The United States programme for high-level radioactive waste disposal has considered a welded tuff formation at Yucca Mountain.

The concept of geological disposal is based on the capabilities of both the local geology and the engineered barriers to fulfil specific safety functions in complementary fashion. Releases from the engineered barriers would only be expected to occur many thousands of years after disposal and to be very small due to the selection of the site and the design of the repository. Additionally, these releases may be retarded and diluted by the geological conditions in the formations surrounding the repository and are further reduced by radioactive decay.

A typical barrier system of a geological repository sited in a hard rock or argillaceous formation is shown in Figure 1.1. The main engineered barriers are the waste matrix, a metal container and a buffer. The geological barrier is provided by the host formation.

Figure 1.1. Safety barriers in a geological repository for the disposal of high-level waste

1: waste matrix, containing radioactive material; 2: metal container; 3: buffer; 4: host rock.

Source: Figure taken from the NAGRA website.

The functioning of a repository system can be explained by its safety functions. For example, the safety functions that were identified in the SPIN project [7] of the European Commission are:

- *Containment*: A watertight barrier isolates the radioactive waste from groundwater during the first phase after repository closure. As long as this safety function is effective, no release of radionuclides can occur from the waste form. Containment makes the disposal system more robust and easier to analyse by preventing dispersion of radionuclides during the strongly transient initial phase of the repository history (re-saturation processes, heat release, strong radiation, pressure rebuilding, etc.). In the case of disposal in a salt formation, the safety function containment is also provided by the host formation.
- *Slow release*: When the containment safety function is no longer available, i.e. after container failure, groundwater comes in contact with the conditioned waste, leaching of radionuclides from the waste matrix starts in combination with the degradation of the waste matrix. Various physico-chemical processes, such as corrosion of and lixiviation from the waste matrix, precipitation, sorption or co-precipitation control releases of radionuclides into the surrounding layers.
- *Retardation*: Radionuclides dissolved in the groundwater after contact with the waste will start to migrate through the buffer and the host formation. Because of the very low groundwater flow in potential host formations, this transport will be very slow; furthermore, many radionuclides will be sorbed onto minerals of the buffer and the host formation. Retardation delays the releases and limits the amounts of radionuclides that are released into the biosphere per unit of time.

If small amounts of long-lived radionuclides leave the repository's barrier system, they are released into the overlying or surrounding aquifers and eventually into the accessible environment. The dispersion and dilution processes in the aquifers and surface waters will further reduce the radionuclide concentrations in the waters that are directly accessible by man.

The concept of geological disposal, including its safety and ethical implications, have been debated and approved in many forums, including national legislatures, state and local discussions, by international organisations and by national scientific bodies. A broad consensus has been achieved through open and participative processes in many countries.

Some countries have reached important milestones. In the United States, a license application for a geological repository at Yucca Mountain has been developed and was submitted to regulatory authorities in 2008. However, the United States administration in 2009 proposed removing support for this repository and its future is uncertain. In Finland and Sweden a repository design has been developed and a site selected for the geological disposal of spent fuel in a granite formation. In Finland, the final disposal of spent nuclear fuel at Olkiluoto is planned to start in 2020. In March 2011, SKB (the Swedish nuclear fuel and waste management company) submitted applications to Swedish authorities to build a spent fuel repository at Forsmark [8]. It is expected that the Swedish geological repository will become operational in 2023. In France and Switzerland a safety case on the geological disposal of various high- and intermediate-level waste types in a clay formation has been submitted and reviewed by national and international committees. In Switzerland the first phase of the site selection process was initiated in 2008. Study and research are currently under way in France to select a site and to develop a design of a disposal facility. A safety case for a license application will be submitted to the authorities in 2015; depending on the decision of the authorities, it is expected that the facility will become operational in 2025.

1.3.1 Dimensions of a geological repository

Repository designs for high-level waste disposal have to take into account several aspects related to the characteristics of the waste and requirements from authorities and stakeholders.

Potentially important aspects are thermal output and volume of the waste packages, radiation fields (γ, neutrons), criticality and retrievability. In the case of disposal of high-level radioactive waste the dimensions of the geological repository will strongly depend on the thermal output of the disposed waste, because for all disposal sites one or more temperature limitations have to be respected. Those temperature limitations depend on the considered host formation and site.

For most repositories excavated in clay or granite, a first temperature limitation is that the temperature of the buffer has to remain below 100 °C. This limitation ensures that the favourable characteristics of the buffer will be maintained after the initial thermal transient of the repository due to the high initial thermal output of the disposed high-level waste. Depending upon the characteristics of the selected site, other temperature limitations can be imposed. For instance, in the case of disposal in the Boom Clay, which is the reference host formation study within the Belgian radioactive waste management programme, a maximum temperature increase of 10 °C at the top of the host formation is imposed to limit the impact of the repository on the overlying aquifer.

For repositories excavated in other host formations, e.g. tuff or salt, higher temperatures can be allowable. For Yucca Mountain two thermal design goals limit the repository's capacity: the host rock must remain below 200 °C everywhere in the repository to limit mineral alteration, and the temperature midway between storage drifts must remain below the boiling temperature of 96 °C at all times to permit drainage of water through the fracture network between drifts. In Germany, the maximum temperature of the salt must remain below 200 °C.

For several host formations, especially in the case of in-gallery disposal concepts, estimates of the approximate maximum values of the allowable thermal output per metre gallery at disposal time have been made; these limits can then be used for a preliminary estimation of the needed length of disposal galleries. Examples of such approximate limits of the thermal output are 200 W/m considered for a possible repository in Opalinus Clay in Switzerland, 300 W/m for a repository in Boom Clay in Belgium and about 1 kW/m for a repository in tuff at Yucca Mountain in the United States. Thereupon, detailed calculations of the temperature evolution in the repository and the host formation taking into account the evolution of the thermal output of the waste to be disposed are carried out to adjust the final allowable thermal loading of the disposal galleries and to determine the minimum distance between two disposal galleries.

1.3.2 Radiological protection in the post-closure period

Geological repositories are sited, designed, constructed, operated and closed so that the protection in the post-closure period is optimised, social and economical factors are taken into account, and a reasonable assurance is provided that doses or risks to members of the public in the long term will not exceed the dose or risk level that was used as a design constraint [9]. The dose limit for members of the public from all practices is an effective dose of 1 mSv in a year. However, in future situations an individual might be exposed to radioactivity from several sources. Therefore, the International Commission on Radiological Protection (ICRP) has introduced the concept of a source-related dose constraint [10]. A geological disposal facility is designed so that the estimated average dose to members of the public who may be exposed in a far future as a result of releases from the disposal facility does not exceed a dose constraint of more than 0.3 mSv in a year [11]. Various radiological protection authorities have set target limits of impact for most exposed individuals in their national regulations [12]. For instance, the Czech Republic and France apply a dose limit/constraint of 0.25 mSv for highly probable scenarios; in Finland, Hungary, the Republic of Korea, the Netherlands, the Slovak Republic, Spain and Switzerland the dose limit/constraint is 0.1 mSv/yr. The United States [13] applies a dose limit of 0.15 mSv/yr for the first 10 000 years after disposal and a limit of 1 mSv/yr after 10 000 years and up to 1 million years. Other countries apply a risk limit; in Sweden and the United Kingdom the risk limit is 10^{-6} per year. To estimate the risk from a calculated dose a risk conversion factor is used; the value of this risk conversion factor is 0.073 per Sv in Sweden and 0.06 per Sv in the United Kingdom.

1.3.3 Safety evaluations of geological disposal systems

Central to successfully implementing geological disposal is the ability to demonstrate and communicate the safety and security of the disposal system far into the future in a manner that is clear, scientifically sound and persuasive to decision makers and the public. There is now a wide consensus on the main elements of a safety case for a geological disposal system [14]. A safety case is an integration of arguments and evidence that describe, quantify and substantiate the safety and the level of confidence in the safety of the geological disposal facility. The safety assessment is an essential part of the safety case; it is the process of systematically analysing the hazards associated with the facility and the ability of the site and designs to provide the safety functions and meet technical requirements. Essential elements of the safety assessment are:

- presenting evidence that the geological disposal system, its possible evolutions and relevant events that might affect it are sufficiently well understood;
- providing convincing estimates of the performance of the geological disposal system and a reasonable level of assurance that all the relevant safety requirements will be complied with and that radiation protection has been optimised;
- identifying and presenting an analysis of the associated uncertainties.

Scenario development

The systematic analysis of the possible evolutions of the repository system is called scenario development. Its main objective is to identify a sufficiently representative set of possible evolution scenarios. Most safety evaluations focus on a reference or central scenario (also called normal or expected evolution scenario); this scenario corresponds to the expected evolution of the repository system and availability of its safety functions. However, a number of events and processes can affect the ability of the disposal system to fulfil its intended safety functions; thus, these events and processes can lead to significantly different evolutions of the disposal system, which can be analysed in the safety analyses as altered evolution scenarios.

In the case of disposal in a salt formation the expected evolution is that the barrier provided by the salt formation will remain intact during a very long time, and consequently no releases of radionuclides occur for the expected evolution of the repository system. Therefore, safety evaluations of disposal of radioactive waste in salt formations only present dose curves from altered evolution scenarios.

The derivation of the reference and altered evolution scenarios is based on scientific information that is collected and described in the so-called assessment basis of the safety case. However, future human actions cannot be predicted on a scientific basis. Therefore, human intrusion scenarios form a special category of scenarios that is treated separately from the other scenarios in most safety cases.

Treatment of human intrusion in the safety case

The publication of the report *Technical Bases for Yucca Mountain Standards* by the United States National Academy of Sciences [15] in 1995 was an important milestone for the treatment of human intrusion scenarios in safety cases of geological disposal systems. Between 1996 and 2008 a harmonisation occurred in the requirements imposed by national regulations and in the treatment of future human actions in the safety cases developed in various countries. Important issues related to human intrusion are [9, 16]:

- A safety case only has to address future human actions that can lead to an inadvertent intrusion into the sealed repository.
- Probabilities of human intrusion cannot be estimated; however the likelihood of intrusions should be reduced by an appropriate selection of the repository site, e.g. by not retaining sites at which there are known exploitable resources; and the depth of the repository.

- High doses to a small group of intruders are an inevitable consequence of the "contain and confine" waste management strategy, and this in contrast to a "dilute and disperse" strategy.
- As it is not possible to predict the evolution of technology in the future, the evaluations of the consequences of intrusion scenarios have to be based on existing technology and practices.

Future human actions that can lead to an intrusion into a geological repository are mainly limited to borehole drilling and mining activities, because of the depth at which those repositories will be excavated. Most human intrusion scenarios considered focus on the possible consequences of drilling a borehole through the repository.

A factor that strongly influences the consequences of an intrusion into a repository is the time of intrusion. Initially, this time is determined by the period of institutional control and by the knowledge of the existence of the sealed repository. Deep boreholes that can reach the depth of a geological repository generally are part of a large geological exploration programme. In this case, it is rather likely that the presence of a sealed geological repository will be detected before the drilling of boreholes within a preliminary site characterisation programme by investigation techniques such as 3-D seismic surveys. Nevertheless, if a borehole were drilled, the thick-walled metallic disposal canister is expected to deviate the drill-bit, thus preventing penetration into the disposed high-level waste as long as the canister is sufficiently intact.

Human intrusion can lead to exposure of different groups:

- A first group consists of members of the drilling team. In the case that cores are taken from the borehole, members of the drilling team can come in contact with cores containing fragments of the disposed waste. However, exposure times are considered to be limited. As natural γ is one of the standard geophysical borehole logging techniques, it can be expected that the drilling team will detect after relatively short times the presence of radioactive materials in the cores and will take measures to limit the exposure.
- In the case of a borehole drilled by a destructive technique, cuttings containing small fragments of waste might be left in the neighbourhood of the borehole. It can then be assumed that a few years after the drilling a small group of people settles in the drilling area and that they produce most of their food in that area and use water pumped from a shallow local aquifer as drinking water for men and cattle and as irrigation water.
- For the third group, it is assumed that a borehole is drilled through the repository and abandoned. Groundwater can flow through the borehole and come in contact with the disposed waste. Radionuclides transported by the groundwater flow can contaminate a neighbouring aquifer, the water of which is drained by a local river or pumped via a water well.

Recently, representatives of various radiological protection authorities discussed the treatment of human intrusion at an international workshop [17]. They concluded that:

> The results of human intrusion scenarios could also be used in a safety case to demonstrate the robustness of the repository system and the safety concept. The consequences to the intruder who comes into direct contact with the waste should not be required to meet regulatory protection goals. These consequences may need to be calculated but they should neither be evaluated against quantitative limit values nor be used as a crucial criterion for the repository optimisation process. [17]

Analyses of the radiological consequences

An important part of the safety evaluations in a safety case are the analyses of the radiological consequences. Because of the involved time scales (up to one million years), these analyses are

necessarily based upon model simulations. The used model of the repository system comprises, among other factors, releases of radionuclides from the waste matrices, dissolution/precipitation processes, migration of radionuclides through the buffer and host formation, transport of radionuclides in aquifer layers, transfers of radionuclides between the biosphere compartments and eventually an estimation of the dose to a member of the critical group. Consequence analyses do not make a prediction of the future doses, but they give an indication of the doses corresponding to the identified evolution scenarios and calculational cases.

Uncertainty management

A key output from the safety assessment is the identification of uncertainties that have the potential to undermine the safety of the repository [12]. Some uncertainties can be reduced by methods including site characterisation, design studies, demonstration tests and experiments both in the laboratory or underground research facilities. In other cases, it may be preferable to avoid the sources of uncertainty or to mitigate their effects by modifications to the location or design of the repository. Safety assessments must nevertheless capture, describe and analyse residual uncertainties that are relevant to safety, and investigate their effects. In most safety cases three types of uncertainty are considered:

- *Scenario uncertainty*: Generally treated within the scenario development as mentioned above; it includes uncertainty about whether all the relevant features, events and processes have been considered.
- *Model uncertainty*: Uncertainty in the relevant processes comes from insufficient knowledge or lack of understanding; it can be treated by considering alternative conceptual models.
- *Data uncertainty*: Uncertainties in model parameters can be due to insufficient knowledge or to time and spatial variability; uncertainties in model parameters are often described with probability density functions and their impact on the calculated doses can be evaluated with, e.g. stochastic calculations.

Some uncertainties that can have a significant effect on evaluated levels of safety are difficult to bound, and are less amenable to the above-mentioned methods. The evolution of the biosphere and the nature and timing of future human actions become highly speculative even over relatively short times into the future. A possible method to deal with uncertainties inherent in biosphere modelling is the use of stylised approaches (e.g. one or more reference biospheres) and complementary safety indicators such as radionuclide concentrations in groundwater or radionuclide fluxes released from the host formation [7, 18].

1.4 Potential role of P&T and organisation of the present report

It can be expected that the introduction of P&T techniques in future fuel cycles will have an impact on radioactive waste management and disposal. In 2003, the Nuclear Development Committee of the NEA set up an expert group consisting of experts in reactor physics of advanced fuel cycles and radioactive waste management to study the possible impact of P&T on radioactive waste management and disposal [19]. The expert group identified a number of representative fuel cycles based on current technology, partially closed fuel cycles and fully closed fuel cycles and it assessed the resulting waste streams, the impact on geological disposal and economic aspects.

After 2005 several more detailed studies on the impact of P&T were carried out in the European Union, France, Germany, Japan and the United States. Chapter 2 of the report presents various issues related to waste management and disposal for which an impact of P&T can be expected. Chapter 3 gives an overview of the main results that became available from the aforementioned studies. Finally, conclusions that can be drawn from the obtained results are presented in Chapter 4.

Chapter 2. Objectives and criteria for partitioning and transmutation

As discussed in Chapter 1, the use of nuclear power invariably creates long-lived, highly radioactive materials, many of which require disposal, such as fission products, activated elements and actinides. Depending on the nuclear fuel cycle, the materials to be disposed can include spent fuel, high-level waste (HLW) and other classes of radioactive waste. The most hazardous materials, spent fuel and HLW, are generally agreed to require isolation from the biosphere for periods upwards of hundreds of thousands of years or more, making deep geological disposal the preferred approach for spent fuel and HLW disposal, and the need for an effective solution to this issue has resulted in numerous repository concepts in various geological environments.

The disposal issue for spent fuel and HLW can be affected by the use of advanced nuclear fuel cycles, depending on the disposal needs. These advanced nuclear fuel cycles may reduce or eliminate the need for spent fuel disposal, with many approaches producing only HLW instead. Advanced fuel cycles can involve processing of spent fuel, fabrication of new nuclear fuels containing higher amounts of actinide elements, and recycling of the actinide elements in either conventional reactors, new types of advanced reactors, or other nuclear irradiation facilities such as accelerator-driven systems. They may also include separate treatment of certain fission products in the waste to better match HLW characteristics with the capabilities of the disposal environment.

In this study, the use of advanced fuel cycles that include partitioning and transmutation (P&T) of the actinide elements is being reviewed for the potential impact on the waste disposal issues as compared to the direct disposal of spent fuel. The radionuclide inventory, volume and decay heat generation from spent fuel and HLW may all have significant impacts on the performance of a geological repository and the mission of providing long-term isolation, with the specific impacts also depending on the characteristics of the geological disposal environment. It must be recognised that all activities related to P&T have the potential for creating additional radioactive wastes, although not necessarily of the same hazard level, and it is important to recognise both the benefits and detriments associated with advanced nuclear fuel cycles.

The different radionuclide inventory in the HLW resulting from P&T activities, especially a reduced actinide inventory, may have a favourable impact on the post-closure radiological consequences of a geological repository, depending on the importance of the actinide elements to the risk from the repository. In all evaluations, the considerations should include the effects of waste form characteristics, as this may have favourable or unfavourable implications. Similarly, the dimensions of a geological repository may be favourably impacted by a reduction in heat generation from the HLW. However, the specific impact of P&T of the actinides on the dimensions or capacity of a geological repository will depend on the importance of heat generation from the actinide elements in the specific repository environment, and other measures such as interim storage of HLW to allow for fission product decay may also be used effectively.

The function of geological disposal is to isolate spent fuel and high-level waste (HLW) from the biosphere for the time required for radioactive decay of the hazardous radionuclides to essentially eliminate the hazard. In reality, it is expected that the isolation may not be perfect and that releases may occur as a result of normal corrosion, degradation and transport processes as well as events that could disturb the repository.

The acceptability of the risk from these events is evaluated using metrics specified in regulations governing radioactive waste disposal. In many countries those regulations are based upon recommendations and guidelines from international organisations such the International Commission on Radiological Protection [11] and the International Atomic Energy Agency [3, 9], although the specific metrics can vary from one country to another. One typical metric is the estimated peak dose rate from such releases. The effect of P&T of the actinides on the overall risk from a geological repository will depend on the contribution of the actinides to the risk metrics. There may be other effects as well on any other metrics that are being used in the advanced fuel cycle that are affected by HLW content and characteristics. For these reasons, it is essential to review current performance of geological repositories for both spent fuel and HLW disposal, for a variety of geological environments, to determine the importance of the actinides and the potential impact of P&T, as well as the importance of fission products, especially shorter-lived fission products where decay storage can be effective.

2.1 Current repository performance

As part of a spent fuel management strategy, it is being proposed that spent fuel be processed to separate and recover actinide and fission product elements for further treatment, removing at least some of them from the waste stream so that the amounts placed in a geological repository are smaller than with direct disposal of spent fuel. In this chapter, some typical metrics are reviewed that describe the important spent fuel and HLW characteristics with regard to geological disposal in various disposal environments. Then the potential effects of actinide P&T on the metrics are discussed, leading to suggested objectives and criteria, recognising any synergistic effect that treatment of shorter-lived fission products may have.

As discussed above, the purpose of geological disposal is to isolate the radioactive spent fuel and HLW from the biosphere without imposing undue burdens or risks on future generations [3]. The risk from such disposal is determined by the dose rate for releases from the repository, which can be viewed as consisting of two parts: first, inherent to the disposed materials as expressed by radiotoxicity for the dose-response assessment; and second, the exposure pathways in the geological disposal environment that would transport any released radionuclides to the biosphere for the exposure pathway assessment [20]. Taken together, the overall risk characterisation of geological disposal in a repository may be represented by the estimated peak dose rate attributed to releases from the repository, although the two metrics can also be used individually or other variations of these metrics may be applied.

2.1.1 Radiotoxicity

Radiotoxicity of spent fuel and HLW is measured by the hazard to humans from ingestion, inhalation and exposure to the contained radionuclides, although only ingestion radiotoxicity is often used to assess the hazard since that may be viewed as the most probable pathway. The ICRP continues to publish and develop information that relates the radiation emanating from any given isotope to a health hazard, as measured by radiotoxicity, using designated “dose conversion factors” that are constantly being re-evaluated and updated [21]. The dose conversion factors are different depending on the postulated exposure pathway, whether by inhalation, ingestion or exposure. For some pathways, such as human intrusion, radiotoxicity can be a relevant indicator since the exposure may be intimate contact of humans with the contents of the spent fuel and HLW. For other repository evolution scenarios, such as groundwater contamination, it is important to account for solubility and transport effects as described in Section 2.1.2, since some radionuclides are not readily dissolved or transported even though they may be significant contributors to the radiotoxicity, and vice versa.

Measures for comparing the radiotoxicity of spent fuel and HLW to naturally-occurring conditions have been proposed as a means of evaluating the relative hazard. The most common measure is the radiotoxicity of the uranium and its decay products as occurs in natural uranium ore. Uranium ore exists in nature with a wide variation in the concentration of uranium and its decay products from one ore deposit to the next, resulting in a large variation in the hazard per tonne of uranium ore. To avoid this complication, it is typical to use the hazard from the uranium and its decay products instead, on a per tonne of uranium basis, eliminating ore quality from the comparison. However, it must be emphasised that in doing so, the natural uranium and its decay products only provide a comparative measure of hazard and should not be used to imply any measure of safety or acceptability of the hazard related to the natural uranium and its decay products. Natural uranium ore with very low concentrations of uranium may represent a small hazard to the public, while ores with much larger uranium and decay product concentrations may pose an increased hazard. Using only the natural uranium and its decay products for normalisation, without the diluting effect of the ore, maximises the hazard from the naturally occurring materials, and therefore has the effect of minimising the apparent hazard from spent fuel and HLW as a result of the normalisation.

Figure 2.1 provides an example of the ingestion radiotoxicity from LWR-UOX spent fuel with a discharge burn-up of 51 GWd/MTIHM. The fission products dominate the radiotoxicity up to about 30 years after discharge from the reactor. After that, the actinide elements dominate the radiotoxicity, and after about 100 years, the actinide elements are almost entirely responsible for the radiotoxicity, mainly due to the relatively short half-lives of many of the hazardous fission product elements. Radioactive decay gradually reduces the radiotoxicity with time, so that by about 250 000 years, the ingestion radiotoxicity of the spent LWR-UOX fuel is equivalent to the radiotoxicity of the uranium and its decay products as would be found in nature, on a per mass of uranium basis used in making the fuel.

In Figure 2.2, the effect of actinide P&T is shown using the example of processing the spent LWR-UOX fuel to recover the actinide elements and recycling the actinides continuously in fast reactors, with the effect of keeping all but process loss amounts of the actinides (assumed to be 0.1%) out of the resulting HLW. In this case, the very small actinide content of the wastes allows the ingestion radiotoxicity of the HLW to decrease much more rapidly after discharge,

Figure 2.1. Ingestion radiotoxicity for spent LWR-UOX fuel, 51 GWd/MTIHM, normalised to natural uranium and decay products as occurs in natural uranium ore

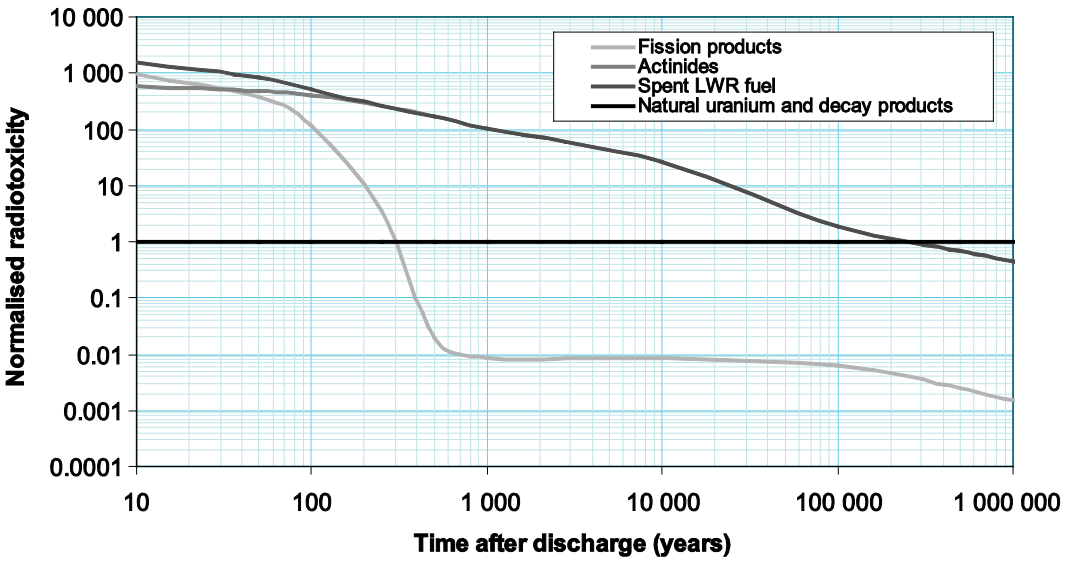
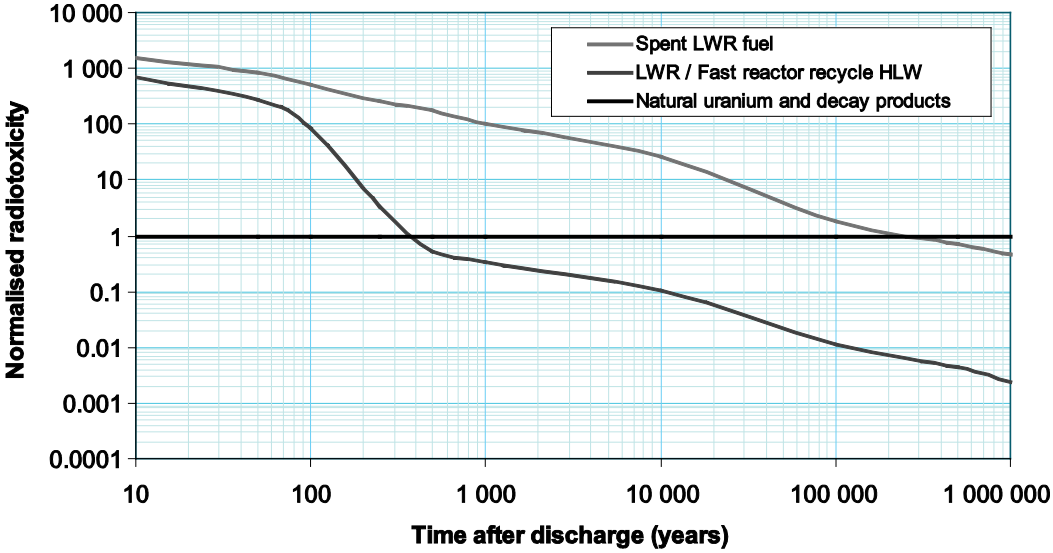


Figure 2.2. Ingestion radiotoxicity for spent LWR fuel, 51 GWd/MTIHM, and for the spent LWR fuel processing wastes where actinides are recovered for recycling, normalised to natural uranium and decay products as occurs in natural uranium ore



dropping below the radiotoxicity of natural uranium and its decay products by about 350-400 years after discharge from the reactor. However, as stated in the previous paragraph, it must be noted that the use of the radiotoxicity of natural uranium and its decay products for normalisation of radiotoxicity must be viewed with caution, and that the radiotoxicity of natural uranium and its decay products should not necessarily be viewed as benign.

Examination of the individual isotopes contributing to the radiotoxicity indicates that only a few actinide isotopes are responsible for the ingestion radiotoxicity to about 100 000 years, ^{241}Am (from the decay of ^{241}Pu), ^{238}Pu, ^{239}Pu-239 and ^{240}Pu. After that time, decay products of actinide isotopes such as ^{210}Pb and ^{226}Ra dominate. If one effectively removes the actinide isotopes, as can be done with many of the partitioning and transmutation approaches, one can greatly decrease the long-term radiotoxicity, as shown in Figure 2.2.

Another way to quantify the impact of P&T is to consider the reduction of the HLW radiotoxicity comparing the case of once-through LWR spent fuel to the case of fast reactors being used to multi-recycle the TRU from the same LWR spent fuel (recovery factor at reprocessing 99.9%). The reduction factor as a function of time is as follows:

Years after disposal	10	100	1 000	10 000	100 000	1 000 000
Reduction factor	2	6	400	250	200	180

2.1.2 Exposure pathway characteristics

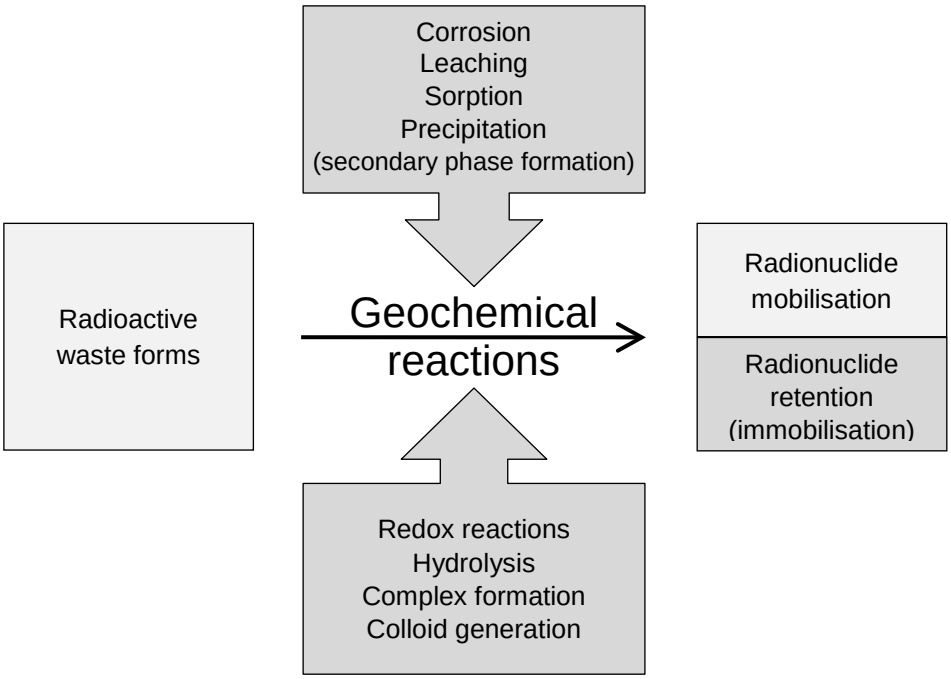
The exposure pathway affects risk from geological disposal since it reflects the potential of the geological environment to facilitate or hinder the movement of any released radionuclides from the repository to the biosphere. The specific characteristics vary considerably from one geological environment to another because the local geochemical conditions determine the mobility of each element, and have a dominant impact on the relative importance of the various radionuclides to the overall risk. After a short description of actinide chemistry for various geochemical conditions, the characteristics of some of the geological environments currently being studied are summarised in the following sections.

The quantification of radionuclide release to the biosphere from the repository containing the nuclear waste forms, such as high-level vitrified waste and spent fuel, requires the consideration of chemical processes:

- container corrosion under relevant groundwater conditions;
- waste form corrosion, alteration and degradation;
- radionuclide solubility;
- radionuclide complexation with groundwater ligands;
- radionuclide retention at waste form and container corrosion product surfaces;
- subsequent radionuclide transport from the repository through the geological environment, ultimately reaching the biosphere;

Geochemical processes have to be quantified taking those reactions into account in order to assess the mobility of individual radionuclides (see Figure 2.3).

Figure 2.3. Potential geochemical reactions in a repository system after groundwater access to the waste form [22]



The behaviour of the actinides in the nuclear waste is of primary interest in the context of discussions on P&T since transmutation of actinides can be performed in nuclear reactors, while transmutation by neutron irradiation is ineffective for most fission products. After breaching of the waste package, groundwater access to the waste form will initiate corrosion and partial dissolution of radionuclides. Under given geochemical conditions, solubilities of individual actinides are controlled by relevant solid phases. As an example, pH dependent solubilities for the actinides Am, Th, Pu, Np, U in equilibrium with their hydroxides in carbonate free aqueous solutions are shown in Figure 2.4 [23]. In general, tetravalent actinide ions exhibit low solubility for neutral to basic pH conditions. For trivalent actinides such as Am^{+3}, low solubility only occurs for very basic conditions. In comparison, hexavalent cations such as U^{+6} and pentavalent actinide ions such as Np^{+5} display relatively high solubilities. For the actinide ions Pu, Np, U, the oxidation state is controlled by the groundwater reducing/oxidising (redox) conditions. For

Figure 2.4. pH dependent solubilities for actinide ions in different oxidation states [23]

Ionic strength: 0.1 mol/L NaCl, T = 25 °C, absence of carbonate

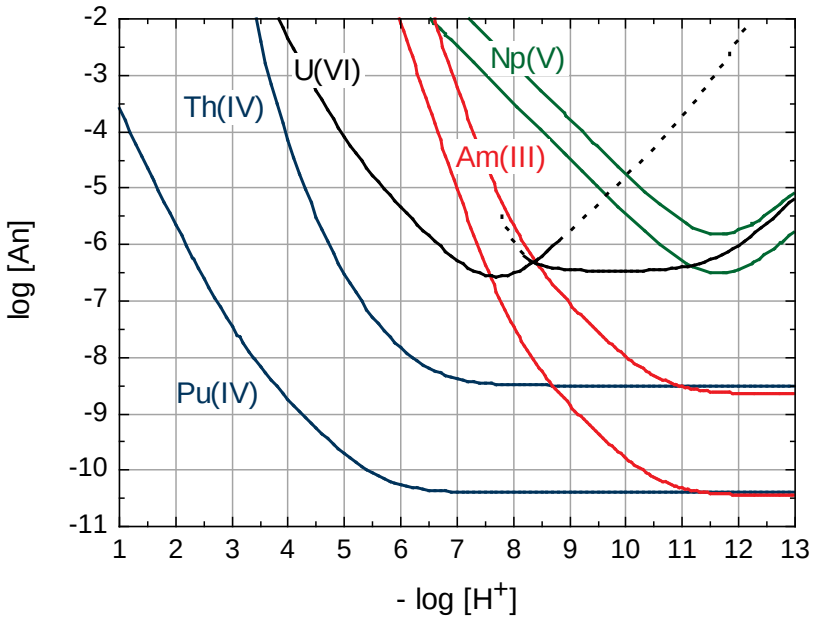
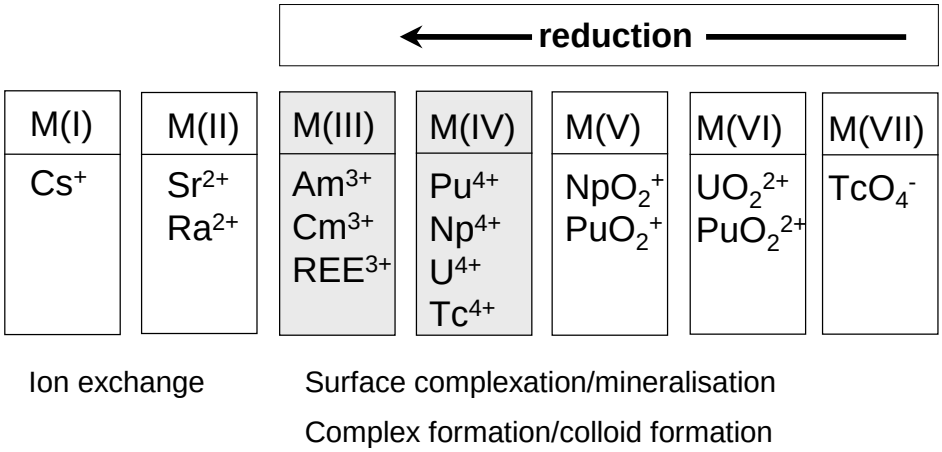


Figure 2.5. Chemical state of radionuclides controlling their mobility in natural aquatic systems [24]



example, Pu can exist as either Pu^{+3} or Pu^{+4}, Np as Np^{+4}, and U as U^{+4} or U^{+6}, depending on the redox conditions in the groundwater. For Am, the trivalent cation Am^{+3} predominates over a wide range of groundwater conditions. A very general scheme for geochemical reactions of waste derived radionuclides in their respective oxidation states is given in Figure 2.5 [24]. For reducing groundwater conditions, many actinide ions are generally found to be quite immobile. Their solubility is low and their affinity to sorption at mineral surfaces is high. This is the reason for the finding that in performance assessment studies on nuclear waste repositories with reducing groundwater conditions, it is mainly the long-lived fission and activation products that are mobile [25]. Actinides are predicted to remain immobilised in the vicinity of the repository. As a consequence, based on the current analyses by the developers, the impact of reprocessing and P&T of actinides on the outcome of long-term safety calculations for such repositories is believed to be negligible [26].

However, there are some geochemical processes that may significantly enhance actinide mobility. As conceivable from Figures 2.4 and 2.5, one can see that intrusion of oxidising groundwater will enhance U, Np and Pu solubility. Solubility enhancement may also occur under the action of a high radiation field at the surface of highly active waste (HAW) generating local oxidising conditions through radiolysis products such as H_2O_2 or ·OH radicals. At the same time, such radiolytic effects can be counteracted by the reducing conditions induced by the corroding canister. Even if such local radiolysis occurs, at a certain distance from the waste, the geochemical conditions would return to those of the groundwater. Access of oxidising groundwater to the depth of a repository depends on the groundwater flow, and in cases where relatively fast groundwater flow through water bearing fractures cannot be completely excluded, oxidising conditions may be expected in the vicinity of the waste packages. Such conditions have been investigated in detail for the Scandinavian repository concept in granite [27] and the Yucca Mountain site in the United States [28].

Actinide mobilisation is also known to be enhanced in presence of carbonate, which in principle can either be present in the groundwater or elevated carbonate concentrations may form in waste being rich in organics where CO_2 is one conceivable final product of microbial degradation. While low- and intermediate-level wastes can be rich in organic matter, such conditions are not expected for high-level nuclear waste consisting of glass product and spent fuel only.

It is important to note that radionuclide interaction with solid surfaces does not necessarily promote retardation and retention. Sorption to naturally abundant mobile inorganic or organic nanoparticles or colloids may on the contrary enhance migration of sorbed elements, notably the polyvalent actinides [29]. The mobility of colloidal radionuclides has been shown in many laboratory and field studies [30,31]. Prediction of colloid-assisted radionuclide transport in the geosphere is difficult in that the presence or absence of colloids is subject to large uncertainties; consequently, the relevance of colloids for nuclear waste disposal safety remains controversial. It is well known that the extent of colloid-borne actinide retention in an aquifer system depends on the solubility of the considered colloidal species, colloid agglomeration, filtration and colloid attachment to surfaces. Retention of colloids is favoured at the high ionic strength that is typical of deep groundwater, low pH and in impermeable rock such as clay rock. The mechanism of actinide-colloid interaction, in addition, has a decisive impact on actinide mobility.

The above-mentioned geochemical processes which may potentially enhance actinide mobility are usually not assumed to be relevant for the normal undisturbed evolution of a repository. However, the controversy surrounding the subject requires that the effects of such processes be quantified as they cannot be excluded *a priori*, especially for less probable repository evolution scenarios that may need to be considered. Therefore, reprocessing and P&T would certainly reduce the uncertainties related to such less probable actinide mobilisation scenarios.

2.1.2.1 Unsaturated volcanic tuff

The proposed Yucca Mountain repository in the United States is situated in unsaturated volcanic tuff, a rock formed by the consolidation of volcanic ash. The repository would be constructed using traditional mining techniques, with the emplacement locations being about 300 m below the surface and about 300 m above the water table for groundwater. The selection of volcanic tuff was based on the absorptive capabilities of the tuff, especially in the zeolitised regions, which would hinder movement of released radionuclides. It was also believed that volcanic tuff would provide a stable geology that would facilitate retrieval if that became necessary. The selection of the Yucca Mountain site was based on a number of factors, including the extremely dry climate and the ownership of the site by the United States DOE [20]. As the project progressed, the design of the repository evolved with lower temperature limits to allow greater predictability in the long-term performance of the repository. The transport characteristics in the volcanic tuff indicated that while soluble fission products such as ^{99}Tc and ^{129}I would be readily transported by

groundwater, the actinide elements would have very low solubility and are readily adsorbed on the volcanic tuff, limiting the transport of these isotopes. As a consequence, in the latest total system performance analyses, the estimated peak dose rate is dominated by both actinide and fission product isotopes, and actinide P&T may not have a significant effect on the estimated peak dose rate for this repository [32, 33].

However, the presence of small amounts of water, the corrosion characteristics of the waste packages, the groundwater chemistry and other technical issues related to the site contribute to the uncertainty about the ability to predict long-term performance. These uncertainties prompt questions about the ability of the selected site to provide the required isolation. As a result, even though unsaturated volcanic tuff may be a suitable environment for the disposal of radioactive waste, at the time this report is being written, political pressures in the United States have placed the future of the Yucca Mountain repository in doubt.

2.1.2.2 Argillaceous formations

For more than 25 years the possibility to dispose of high-level radioactive waste in argillaceous formations has been investigated in various European countries and Japan. Potential host formations cover a large spectrum of argillaceous media, i.e. from plastic, soft, poorly indurated clays to brittle, hard mudstones. Argillaceous media have a number of favourable properties relevant to long-term isolation of HLW and spent fuel, such as homogeneity, very low groundwater flow (i.e. sufficiently low that the flow can be assumed to be negligible), chemical buffering, a propensity for self-healing of fractures by swelling and plastic deformation and the capability to chemically and physically retard the migration of radionuclides. Furthermore, the sites being considered for geological disposal are in areas with low seismic and tectonic activity and where the potential host formation occurs with large thickness, extent and continuity [34]. Detailed site investigation programmes, including the construction of underground research laboratories on possible argillaceous host formations have been undertaken in various countries, e.g. Belgium, France and Switzerland, and iterative performance assessments and safety cases [35-37] have been conducted.

The disposal concept of geological repositories in clay formations relies heavily on the characteristics of the host clay formation, including negligible groundwater flow, reducing chemical conditions, the ability of the host clay to act as a colloid filter, and sorption potential of a large number of radionuclides on clay minerals to provide the desired isolation of the wastes. The metal container mainly has to ensure that groundwater does not come into contact with the disposed waste during the initial thermal transient phase of the repository. The waste matrix and the clay buffer both contribute to the isolation function.

In the case of disposal in clay formations, the impact of actinide P&T, i.e. a large reduction of the actinide inventory of the disposed waste, on the calculated estimated dose rates is currently negligible since in the reference scenario as well as in the assumed altered evolution scenarios [35-37], the total dose is mainly due to mobile fission and activation products, while the contribution of actinides is a few orders of magnitude lower. At this time, only in the case of a few unlikely human intrusion scenarios would actinides provide a more significant contribution.

2.1.2.3 Crystalline formations

The possibility to dispose of spent nuclear fuel in crystalline formations such as granite has been investigated in Canada, Finland, Sweden (since 1975) and Switzerland. Extensive research programmes, including the construction of underground research laboratories, on disposal in hard rock formations have been carried out. Geological settings selected as potential host formations for the deep geological disposal of radioactive waste are chosen for, among other assets, their long-term stability and buffering capacity against disruptive or destabilising events and processes [38]. Other favourable characteristics of stable hard rock formations are the ease

of constructing galleries, geo-mechanical strength enabling the use of strongly swelling buffer materials, relatively high thermal conductivity, modest groundwater flow, and reducing chemical conditions. At the time of this report, the spent fuel disposal programmes in Finland and Sweden have progressed to the stage of submitting the safety cases to the authorities [39, 40]. Assuming a favourable review from the safety authorities, and no other perturbations to the programmes, spent fuel disposal is expected to start in 2020 and 2023, respectively.

The safety concept of repositories in crystalline formations is mainly based on a number of engineered barriers complemented by the favourable characteristics of the selected host formation and site. The waste package is a copper canister that forms the primary barrier for the spent fuel and is expected to remain intact during several hundreds of thousands of years, such that no releases of radionuclides are expected during the first hundreds of thousands of years [39, 40]. Surrounding the copper canister is a bentonite clay buffer layer that is intended to protect the canisters from rock movements and potential detrimental substances and to limit or prevent groundwater flow around the canisters. In the event of canister failure, the clay buffer layer is also intended to limit and retard radionuclide releases and to act as a colloid filter to retard or prevent colloid transport. As a result, in the case of the reference scenario in which the failure of some canisters is assumed, the dose is mainly due to ^{129}I and to a lesser extent to ^{135}Cs and ^{14}C. However, for several variant scenarios, such as penetration of oxidising glacial melt water in the near field or a very high groundwater flow, ^{226}Ra, ^{231}Pa and ^{239}Pu are the dominant dose contributors, and in such cases, actinide P&T would have a favourable impact on the estimated dose rate.

2.1.2.4 Bedded salt and domed salt

The main advantage of rock salt as a repository host rock is its very high heat conductivity and the plasticity leading to the relatively rapid closure of fractures and veins, but also of artificial mine openings. Crushed salt used as backfill material has been demonstrated to be compacted under the convergence forces of the rock down to very low porosities. This property is the reason that no water access to the waste is considered in generic safety assessments for the expected and most likely evolution of the repository system. Less likely scenarios discuss:

- the impact of limited volumes of water inclusions in the genuine rock salt, which may move towards the waste in thermal and/or chemical gradients of the repository;
- limited water access through the excavated disturbed zone around seals and plugs in the access tunnel and the residual pore space of the potentially not entirely compacted backfill.

Brine fluids may then be transported by different processes: advective transport in a pressure gradient, diffusion in a concentration gradient and solubility-controlled brine migration in a temperature gradient. Additionally, in porous (crushed) rock salt, transport of water via the vapour phase may occur, if a chemical activity gradient prevails.

Advective transport processes

Advective transport in porous media depends on the permeability and on the pressure gradient of the fluid. Impacts on pressure gradients arise from the convergence of the rock salt and from the evolution of a hydrogen gas pressure due to the anaerobic corrosion of metallic canister materials. Undisturbed rock salt is very impermeable to advective fluid transport. Reported permeability of rock salt cores in laboratory are in the range of 10^{-20} m^2 [41, 42]. Under *in situ* conditions permeability is even lower [43]. In an excavated salt mine, stress fields develop which may cause deformation of the rock salt and increase of the permeability. However, due to the self-healing capacity of dry rock salt especially under high pressure load (or temperature) the increase of permeability decreases again with time.

Diffusion processes

Rock salt crystals show a very low diffusion coefficient for He/H_2 gases [44, 45]. Fluid diffusion in rock salt occurs on grain boundaries only. Detailed investigations of self-diffusion of the intergranular water in rock salt revealed values of $D = 10^{-7}$ cm^{-2} s^{-1} [46]. Investigations of the compositions of fluid inclusions in rock salt of the Gorleben salt dome showed that the fluids did not interact with solutions from outside of the salt dome due to uplift of the salt structure [47]. Less likely scenarios discuss the water access to the waste via residual pore space, e.g. in the excavation disturbed zones of closed access tunnels and not entirely compacted backfill material (crushed salt).

Brine migration

Small quantities of brine trapped in "negative crystals" (cavities) exist in most rock salt (halite) and have been found to migrate towards a heat source. The driving force for the migration is the difference in solubility between the warm and colder sides of the brine cavity. The theoretical model indicates that the migration rate is a function of temperature and is directly proportional to the temperature gradient. Brine inclusions having a size of < 20 μm move unhindered in NaCl crystals. Bigger droplets remain fixed, though show deformations forming tips in the direction of the cold side. The tips grow, resulting in small size droplets which can move through the crystal [48].

Calculated dose exposure to the population

Rock salt has very limited retention capacities for radionuclides. Therefore, the geochemistry of the very near-field controls the mobility of radionuclides in the host rock. Due to the redox barrier function of canister corrosion and the sorption and reduction capacity of the respective corrosion products, most of the actinides are retained. Sorel phases in the backfill/sealing systems can further contribute to radionuclide retention. Important contributions to the peak dose are mainly due to anionic species, such as ^{14}C, ^{79}Se or ^{129}I species. ^{135}Cs and some negatively charged actinide complexes are also assumed to undergo only negligible sorption and thus may also play a role.

2.1.2.5 Deep borehole disposal

The concept of using very deep drilled holes, "boreholes", for the disposal of radioactive waste has been suggested and studied periodically since the 1950s [49-51]. The boreholes are envisioned as being drilled in hard rock using standard well-drilling technology, although the extreme depth of 4-5 km has only recently been demonstrated sufficiently to be considered an existing technology. The main principle in using deep boreholes to provide permanent isolation from the inhabited environment is the behaviour of groundwater deep below the earth's surface. Near the surface, the groundwater has very little salt content and is used as a resource for fresh water, whether for consumption or irrigation. However, as depth increases, there is a threshold where salinity starts to increase, resulting in an increased water density. This increasing water density prevents deep subsurface water from interacting with the near surface groundwater, effectively isolating the lower region. Exploration of the lower regions has shown that the water has been in place for millions of years, and suggests that waste emplacement in this region would provide the required isolation during which the radioactive waste would decay.

There are a number of considerations regarding the ability to use deep boreholes for radioactive waste burial. Assuming that all dimensional issues can be resolved, it would be desirable for the site to have certain characteristics:

1. The groundwater should not be overpressurised with respect to hydrostatic pressure at that depth. If the groundwater is overpressurised, the action of drilling the boreholes will disturb the isolation environment, and continued stability of the isolation zone would be difficult to guarantee.

2. The salinity gradient should be sufficient to prevent any thermally-driven convection from the emplaced wastes. The balance is controlled by limiting decay heat generation and ensuring a sufficient salinity gradient, with any movement only occurring in the horizontal direction at depth.
3. The borehole must be capable of being sealed to prevent any vertical migration pathways. Of particular concern is the use of well casing materials that corrode with time, resulting not only in vertical migration pathways, but the corroded materials may have pathways with very small dimensions facilitating significant vertical movement by capillary action.

The advantage of sites with these characteristics is that all of the other local environment characteristics are not relevant to the isolation potential, which is provided by the extreme depth. As a result, issues such as rock permeability and fractures/fissures are not important, as they would only affect horizontal motion within the isolation zone and would not contribute to a potential release pathway. Similarly, the local effects of drilling the borehole, or even the waste forms, waste packages or waste containers themselves would also be irrelevant. This could greatly simplify the required site characterisation and performance assessment of such a repository. Operation of the repository may be complicated by deep well drilling technology limitations and the risk of difficulties with emplacing wastes down to such depths. However, the extreme depth may provide a convincing argument about the isolation potential meeting regulatory requirements.

The key issue appears to be demonstrating that the actions of drilling the boreholes and emplacing the wastes will not affect the underground environment, guaranteeing the existing hydrodynamic stability. The level of difficulty in proving this statement in practice does not appear to be known at this time. If stability can be shown, then actinide P&T, or even fission product treatment, would have no effect on the performance of the repository since there would be no transport of radionuclides out of the isolation zone. However, if decay heat becomes an issue in guaranteeing stability, then actinide P&T and interim decay storage for fission product decay could be considered for the reduction of decay heat, as discussed in Section 2.1.4.

2.1.3 Estimated peak dose rate

The function of the geological repository is to prevent the spent fuel and HLW from ever resulting in an exposure to the biosphere. Although the hazard of the spent fuel and HLW has been measured by radiotoxicity, as discussed in Section 2.1.1, which is relevant for direct contact with the materials (ingestion, inhalation or exposure), the effectiveness of the isolation provided by the repository can be evaluated by the estimated peak dose rate to exposed individuals caused by releases from the repository. Degradation and transport mechanisms in the repository environment can alter the relative importance of the radionuclides when considering the hazards of geological disposal, i.e. the radionuclides that are most radiotoxic may not necessarily dominate the expected dose rate, or even be relevant at all. As a result, the effectiveness of actinide P&T and fission product treatment can vary substantially from one country to another, depending on the preferred geological disposal environments, and may also be reflected in the regulations developed in each country depending on the relative importance given to normal evolution and disturbances in assessing the overall risk to the biosphere.

International organisations such as the ICRP [11] and the IAEA [9] have formulated recommendations on criteria for the protection of human health and the environment in the post-closure period of a geological repository. The IAEA safety requirements [11] state, “To comply with the dose limit, a geological disposal facility (considered as a single source) is designed so that the estimated average dose or average risk to members of the public who may be exposed in the future as a result of activities involving the disposal facility does not exceed a dose constraint of no more than 0.3 mSv in a year or a risk constraint of the order of 10^{-5} per year.” Many countries have followed these recommendations and have defined dose constraints ranging between 0.1 to 0.3 mSv per year or risk constraints of 10^{-6} per year [12].

In the United States, the licensing of the Yucca Mountain repository will be based on satisfying estimated peak dose rate limits, specified for the maximally reasonably exposed individual (MREI). Currently, the regulation is for a limit of 0.15 mSv/yr for the first 10 000 years and 1 mSv/yr out past the time of peak dose. To assess repository capability, an analysis process is used, referred to as a "total system performance assessment" (TSPA), where one result of the analyses is the estimate of peak dose rate as a function of time after the repository is closed, for times typically extending to one million years or more [32, 33]. These results are compared with the regulatory limits for releases in order to judge the adequacy of the protection provided by the repository.

The performance of a geological repository in such analyses is estimated using detailed computational models. The intent of the models is to simulate the repository environment, including the physical and chemical conditions of any groundwater in the repository, whether essentially immobile or mobile, as well as any natural phenomena that may occur such as seismic and igneous events. The groundwater presence and mobility can be especially important in determining the long-term integrity of any waste disposal packaging, barrier systems and the spent fuel and high-level waste forms, since corrosion and degradation of the waste packaging and its contents are key to estimating any releases from the repository. Few repository concepts utilise geological environments containing mobile groundwater, although many rely on the presence of essentially immobile groundwater, and some such as salt would have no groundwater at all. In addition, the chemical conditions of the groundwater are vitally important in determining the transportability of the radionuclides contained in the waste packages, since solubility is a function of the groundwater chemistry. Other factors include particulates in the groundwater, since even those radionuclides that may not dissolve into the groundwater can be adsorbed onto such particles. However, many repository environments are capable of retarding or preventing the movement of such particles from the repository.

Understanding the causes of the estimated peak dose rate is important if one is considering actinide P&T for treatment of spent fuel and high-level wastes prior to disposal with the expectation that the P&T will favourably affect estimated dose rates. As summarised above, previous studies have indicated that the radionuclides dominating the dose rate are a function of the repository geology. Depending on the repository project, there also appears to be a large uncertainty about the relative importance of actinide and fission product isotopes, since this has been shown to vary considerably during different analysis stages, and no licensing review has yet been completed. In general it appears that the fission products may be more important to the dose rate for releases from an undisturbed repository containing groundwater under reducing conditions, that actinides and fission products may be equally important for undisturbed repositories containing groundwater under oxidising conditions, and there do not appear to be releases from dry repositories under normal evolution. Actinides may be major contributors when the repository is disturbed, especially by human intrusion, but this can greatly depend on the assumptions made for the disturbed conditions [20, 32, 33, 52]. However, it appears that in all cases, uncertainty may be large if groundwater is present in the disposal environment.

2.1.3.1 Normal repository evolution

It is useful in the context of discussing repository performance to consider the normal, or undisturbed, evolution of the repository system. In this case, once the repository is closed, there is no disturbance to the repository and the emplaced waste packages containing spent fuel and HLW are only subjected to whatever corrosive mechanisms may occur given the chemical environment of groundwater that may be present. There are also repository concepts, such as those in salt, where the environment is essentially dry and such mechanisms may not occur.

If there is groundwater in the repository, the presence or absence of groundwater flow through the repository is very important in determining any potential for exposure. In some repository concepts, the groundwater has been stationary for many millions of years, providing a

level of assurance that if spent fuel and HLW were placed in such an environment, even if the waste packages and the contents were corroded and degraded to release the radionuclides into the groundwater, this would not necessarily result in exposure to the biosphere. Other proposed repositories may have sufficient groundwater flow through the repository such that the small releases of radionuclides into the groundwater are further diluted, and the resulting concentration in the water would be extremely low, sufficient to maintain dose rates well below any established limits even if the water were ingested.

This is especially true for repositories with significant water flow, so that even if the spent fuel and wastes were slowly degraded over time, the resulting releases over the decades relevant for human exposure could be extremely small. All of the details related to the timing of waste package corrosion, degradation of the contents and release of radionuclides can become important when there is a mechanism for transporting the radionuclides out of the repository. For repositories in relatively dry climates where water flow is low, the radionuclide concentrations in groundwater can be higher as a result, and the potential risk of exposure may be greater, although not necessarily high enough to be of concern. As a result, in all cases where there is water flowing through the repository, the estimation of peak dose rate far in the future would seem to be much more uncertain due to the existence of transport paths under nominal undisturbed conditions than for repositories without flowing water.

2.1.3.2 Disturbed repository conditions

The repository environment can be disturbed either by naturally-occurring phenomena such as seismic and igneous events, or by human actions. The naturally-occurring phenomena are highly stochastic and prediction of future events can only be approximated by examining past behaviour at the repository location. Analyses may include the effects of such events in a statistical manner, as in the TSPA for the proposed Yucca Mountain repository [32, 33]. Other approaches may also include detailed examination of the event to provide analysis results, but with no prediction on the likelihood other than to note that it is highly unlikely and should be considered as such when determining the acceptability of the disposal site. For such natural disturbances, in some cases it may be assumed that most or all of the repository is disturbed, while in others the assumption may be that only a small part of the repository is disturbed. Subsequent events would reflect the changes in repository conditions and exposure pathways in the geological environment.

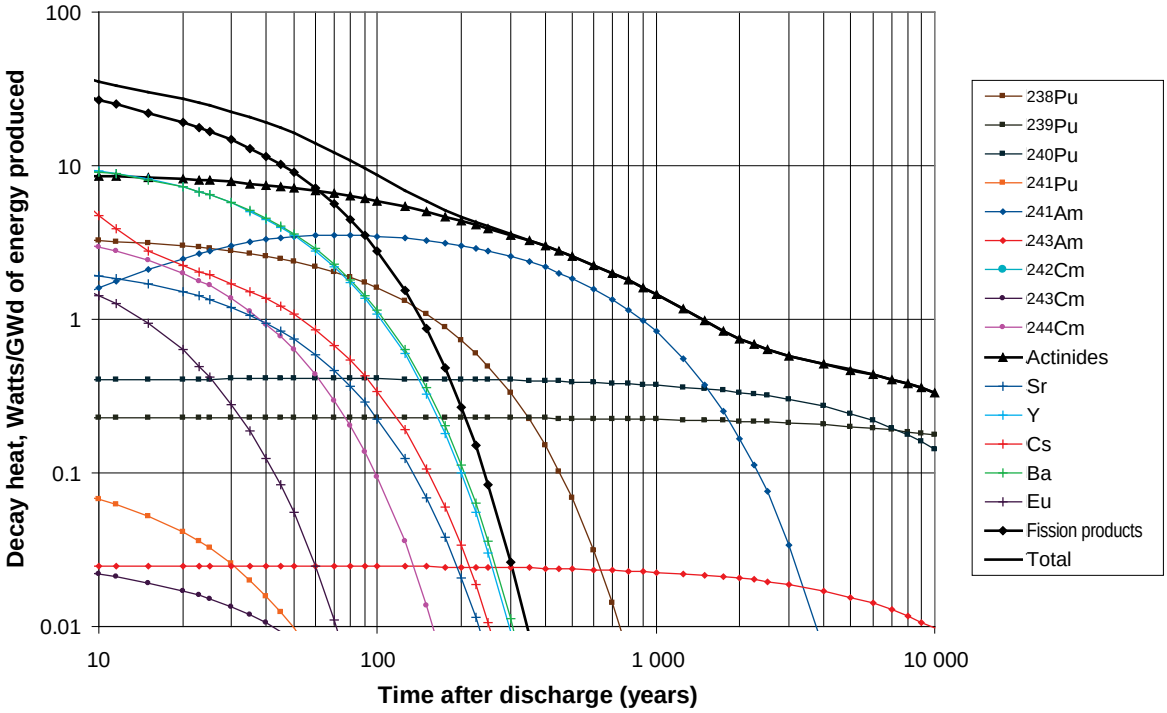
The potential human activities that can disturb the repository are even more difficult to predict. One scenario is intentional human intervention, either to retrieve the spent fuel for the content that could be usable as fuel, including uranium and plutonium, or if the materials are desired for potential weapons use. Mined geological repositories may be vulnerable to these actions, yet accounting for them in a detailed analysis in a scientific manner is considered to be impossible since they are completely determined by motive.

Another possibility is where the disturbance is unintentional and results from other activities. An example of this is drilling for natural resources such as oil, gas or water. It is conceivable that such drilling activities could penetrate the repository and make the contents available for exposure of individuals, either by direct contact or indirectly by subsequent transport mechanisms in the environment. For such scenarios, it is likely that only a single waste package would be penetrated, and the potential exposure would be determined by the package contents at that time. However, if the action of drilling through the repository had additional implications, those implications would also have to be estimated. An example could be for a salt repository where penetration of the overlying rock and the salt layer could change the isolation of the entire salt deposit with respect to groundwater, resulting in the potential for disturbing the entire repository and its geology.

2.1.4 Decay heat

Decay heat from spent fuel and HLW is essentially an engineering and operational issue for most, if not all, repository concepts as the decay heat is highest at the time the spent fuel is discharged from the reactor and decreases rapidly with storage time, as shown in Figure 2.6. It is seen that decay heat is dominated by fission products up to about 60 years after discharge and by the actinide elements afterwards. Higher decay heat makes storage of the spent fuel and HLW more difficult, since there are temperature limits for these materials to ensure their integrity.

Figure 2.6. Decay heat contributors in spent LWR fuel, 51 GWd/MTIHM discharge burn-up



Similarly, when placed in a geological disposal environment, there are additional temperature limits for the repository itself, designed to prevent changes in the repository environment and to increase the reliability of predictions about long-term performance. For example, in a repository situated in a saturated clay layer, it is essential for the clay layer to retain water to prevent drying and fracturing of the clay so that the clay provides a barrier of very low permeability. When spent fuel and HLW are placed in such a repository, the area loading of materials must account for the decay heat and any heat removal mechanisms to ensure that temperature limits are not exceeded. Similar constraints exist for repositories in granite, for which the clay-based buffer is an essential barrier, and volcanic tuff, among others. For a salt repository, decay heat may be beneficial in aiding long-term repository performance by using the high temperatures to cause the salt deposit to flow around the emplaced materials. In this manner, the waste packages would be completely surrounded by salt, essentially becoming part of the salt deposit. Such an approach would also distort any surrounding rock, and the implications of using decay heat in this manner would need to be assessed to guarantee that there are no negative performance impacts.

On the other hand, higher decay heat could cause migration of the included water, as discussed in Section 2.1.2.4, and there may be limits on decay heat to mitigate this effect. Interim storage to allow time for radioactive decay to lower the decay heat may be a useful approach, depending on the time span over which the decay heat affects the repository. If longer-term decay heat is a significant issue, past 75-100 years, either from an engineering and operations

viewpoint or from a more fundamental perspective in developing a repository concept, actinide P&T and interim storage prior to disposal can be effectively used to lower decay heat in the corresponding wastes, allowing for an increased utilisation of repository space [53, 54] or enabling some repository disposal concepts such as salt and deep boreholes.

2.1.5 Waste form, volume and mass

The form, volume and mass of the spent fuel and HLW may be important for disposal, depending on the repository environment. For most repositories analysed to date, the placement of the spent fuel and HLW is affected mainly by the decay heat and its effect on repository temperatures. In principle, the lower the decay heat, the smaller the amount of repository space required for disposal. Partitioning and transmutation can directly affect the decay heat of the waste materials, even if only actinide elements are considered, as discussed in the previous section.

However, it must also be emphasised that the purpose of a repository is to provide sufficient isolation of the emplaced materials from the biosphere. As discussed above, the hazard can be viewed as a function of the radiotoxicity of the emplaced materials, which is directly proportional to the mass of the hazardous isotopes in the disposed inventory. For normal evolution, there is likely to be sufficient temporal incoherence in waste package failures, waste form degradation, and radionuclide transport such that the inventory may only be a secondary effect, with releases from the repository spread over very long periods of time. Developing a convincing argument that this is the most probable behaviour may be a challenging prospect, and could be affected by more complex engineered systems and waste forms. For disturbed conditions caused by human intervention, many events do not involve more than a single waste package, so the number of waste packages may only be relevant in assessing the probability that one would be penetrated as a result of human actions. For other very low probability events, such as seismic or igneous events, the inventory assumed to be affected is arbitrary, and can range from a fraction of the repository inventory to the entire inventory.

With this background, actinide P&T can greatly lower the inventory of actinide elements, as well as separating out some of the shorter-lived fission products for separate disposal if desired. However, longer-lived fission products that are not amenable to transmutation would still be present at the same inventory per unit of energy generated. It must be recognised that if the partitioning and transmutation were used to increase the amount of wastes that could be disposed in a given repository as a result of the lower decay heat, then the inventory of the unseparated fission products would rise accordingly. The effect on the ability of the repository to provide the required isolation for the greater amount of fission products would need to be analysed to ensure that measures such as estimated peak dose rate have not been adversely affected.

2.1.6 Retrievability and reversibility

The issues of retrievability and reversibility from geological disposal are raised for several reasons. One reason is to allow for future developments in the science of geological disposal that may either negatively affect the predicted performance of the repository that was used to initially license the facility, or that may have resulted in a superior treatment or disposal option. In either case, there would be a need to access and remove the emplaced waste packages for further actions. Another reason is the desire to access the emplaced materials, especially the spent fuel, and remove them for processing to acquire uranium, plutonium and possibly other elements that would be useful for future energy production. This may happen as a consequence of changing from a “once-through” fuel cycle to a “closed” fuel cycle, with the spent fuel representing a valuable resource. There is also the possibility of retrieving the spent fuel and HLW for processing to obtain the weapons-usable materials, a prospect that becomes more feasible with time as the shorter-lived fission product and actinide elements decay, an approach described as the “future plutonium mine”.

In the first of these options, the actions of partitioning and transmutation would only affect the form and content of the materials to be retrieved for subsequent disposal. In the second option, the need for retrievability and reversibility would depend on the usable contents of the spent fuel or wastes in the repository, i.e. if there were sufficient usable fissile or fertile material, and if the waste form would need to be one that was amenable to further processing. In the third case, partitioning and transmutation of actinides would be able to eliminate this concern by not placing any significant amount of recoverable weapons-usable materials in a geological repository.

2.2 Summary of objectives and criteria for partitioning and transmutation

In general, partitioning and transmutation has referred to the separation and recovery of actinide elements for recycle, although there has been some examination of the potential for treating some of the long-lived fission products such as ^{99}Tc and ^{129}I. As indicated in the previous sections, the benefit of treating actinides may depend on the specific repository environment being considered, as for many of the options the actinides do not appear to be very important for the normal evolution of the repository since the exposure pathways may inhibit actinide transport from the repository to the biosphere.

When the disturbed events are considered, the situation may be different. For some altered evolution scenarios, actinides may dominate the risk since the radiotoxicity of the materials appears to be more of a determining factor for the risk, and the actinides dominate long-term radiotoxicity. In this case, partitioning and transmutation of the actinides could have a significant impact on the associated risk, as measured by changes in the radiotoxicity, and the candidate isotopes for partitioning and transmutation could be identified.

However, it should be recognised that radiotoxicity alone may not be the essential indicator of the effect of actinide partitioning and transmutation on the estimated dose rate for some repository concepts, since both radiotoxicity and exposure pathway characteristics are needed to evaluate repository performance. A detailed analysis must always be performed in order to judge the potential impacts on the estimated dose rate from partitioning and transmutation of actinides, since in some cases it is possible that the impacts may be small or even negligible, even if there is a substantial change in radiotoxicity, as has been shown in some of the previous sections.

As discussed above, decay heat presents an interesting engineering problem, and in many cases can be addressed with engineering solutions, such as providing ventilation, reduced loading of waste packages and proper placement of waste packages. Since all of these potential solutions directly increase the cost of spent fuel and HLW disposal, another criterion could be the impact on the decay heat from partitioning and transmutation. A subsidiary concern would be the volume of waste and the specific heat generation rate for wastes resulting from partitioning and transmutation, because inefficient loading capability for waste forms would result in less than optimal usage of space in the repository.

Another aspect of the waste disposal issue that can affect decisions on the usefulness of P&T is the inherent uncertainty in being able to convincingly predict the performance of geological disposal over very long time scales. Past geological conditions can help in making the case that the location will continue to behave in the same manner, and thus the predictions have a high probability of being sufficiently reliable. As described above, what matters for overall repository performance is the inventory of hazardous radioactive materials and the risk to the biosphere associated with these materials. One significant effect of partitioning and transmutation is that the inventory of the emplaced materials is much lower on an energy-generated basis for the actinide elements, which can have the effect of making the uncertainty about repository performance less important, especially with respect to the actinides. One can decide to use this benefit of P&T in making the case for siting and licensing a repository, although if one uses

partitioning and transmutation as an opportunity to increase the amount that can be placed in the repository, i.e. the wastes from more spent fuel is disposed in the repository, the beneficial effect of such an inventory reduction may be mitigated. Overall, the actinide inventory would be much lower, but the fission product inventory would be higher, and the consequences of this must be evaluated for each repository.

Chapter 3. Summary of past studies, analysis and comparison of results

3.1 Introduction

The impact of the P&T technology on the management of nuclear wastes has been discussed since the 1970s. Up until about the year 2000, the impact of P&T technology tended to be emphasised from a viewpoint of the "potential radiological toxicity" or "radiotoxic inventory" which is defined by the amount of the radioactive nuclides in the wastes normalised by their dose coefficients for ingestion [21] for individual nuclides. One of the important studies in this period was an OECD/NEA document issued in 1999, *Actinide and Fission Product Partitioning and Transmutation: Status and Assessment Report* [55].

For the decade following the year 2000, more emphasis was put on the realistic consideration of the fuel cycle analysis and the repository design. In this chapter, some past studies including the specific analysis of the possible benefit and impact of P&T are reviewed and compared in terms of several criteria mentioned in Chapter 2:

- peak dose rate;
- radiotoxicity;
- decay heat;
- waste form, volume and mass;
- uncertainty;
- miscellaneous aspects such as proliferation and cost.

The comparison among independent studies by different counties and regions in the world is expected to be beneficial in terms of extracting the essence of various results and building common understandings.

3.2 Reviews of past studies

3.2.1 First OECD/NEA P&T report in 1999 [55]

This report was edited to provide an authoritative analysis of the technical, radiological and, as far as possible, the economic consequences resulting from the proposed partitioning and transmutation operations at that time. The report was subdivided into a general part for non-specialist readers and a technical systems analysis discussing the issues in partitioning, transmutation and long-term waste management.

Part II of the report compiled the technical analysis and systems study including a comprehensive discussion of the impact of partitioning and transmutation on risk assessment and waste management, especially the long-term impact of proposed methods.

Chapter 4 of Part II of the report was devoted to "Impact of P&T on Risk Assessment and Waste Management". The executive summary of this chapter is extracted in Box 3.1. The report provided a comprehensive analysis of the impact of P&T on waste management, though it did not make any proposal concerning a means of waste management incorporating P&T.

Box 3.1. Executive summary of Chapter 4 of the OECD/NEA P&T report [55]

Impact of partitioning and transmutation on risk assessment and waste management activities

In order to set out very clearly the impact of partitioning and transmutation on long-term radiotoxicity, a special section in the report is devoted to the significance and definition of radiotoxic inventory.

The radiotoxic inventory depends on the physical inventory of radionuclides in the various fuel cycles and on the effective dose coefficients of individual radionuclides. Graphs (e.g. Figure 3.1) show the natural evolution of radiotoxic inventories corresponding to the once-through cycle, reprocessing fuel cycle and advanced fuel cycle.

The role of partitioning and transmutation in waste management depends primarily on the minor actinide inventories to be handled in the advanced fuel cycle facilities. The section discusses the reactor strategy to be implemented in order to achieve a significant reduction of the radiotoxic inventory. Three representative cases are considered with various combinations of light water reactors loaded with uranium dioxide and mixed-oxide fuel, and fast reactors loaded with mixed-oxide fuel. A substantial proportion of fast reactors is necessary. In each of the cases examined, the radiotoxic inventory in the waste streams is reduced by a factor of about 100. However, the decrease observed in the waste discharges is accompanied by a steady increase of the radiotoxic inventory of the reactor and fuel cycle facilities operated over a long period of time to accomplish this waste minimisation.

Where all minor actinides are recycled including the curium fraction, the radiotoxic inventory reduction factor, compared with once-through cycle, ranges from 77 to 100 after 10 000 years, and in the very long term (105 and 106 years) from 80 to 150.

If the curium fraction is not recycled, the radiotoxic inventory is reduced only by a factor of 7 to 14 at 10 000 years owing to the decay of ^{244}Cm into ^{240}Pu and ^{243}Cm into ^{239}Pu.

When making a global mass balance of the minor actinides involved in a long-term multiple recycling programme, the radiotoxic inventory of the nuclear materials in the reactors and fuel cycle facilities becomes overwhelmingly more important than the annual waste discharges. Various cases leave equilibrium inventories of several hundred tonnes of transuranic elements.

Based on data from different economic assessments, but principally relying on a comprehensive study carried out within the European Union's R&D programme on radioactive waste management and disposal, some very preliminary cost data for the advanced fuel cycle with partitioning and transmutation strategy are given in Annex F.

Waste management concepts

Estimates of the radiological benefit to be expected from partitioning and transmutation must take into account the mobilities of the elements in the geosphere, which may substantially modify the simple picture presented by the reduction in radiotoxic inventory alone. Other benefits, e.g. with respect to intrusion, may however be derived from the possibility of especially secure conditioning and emplacement of the long-lived radionuclides.

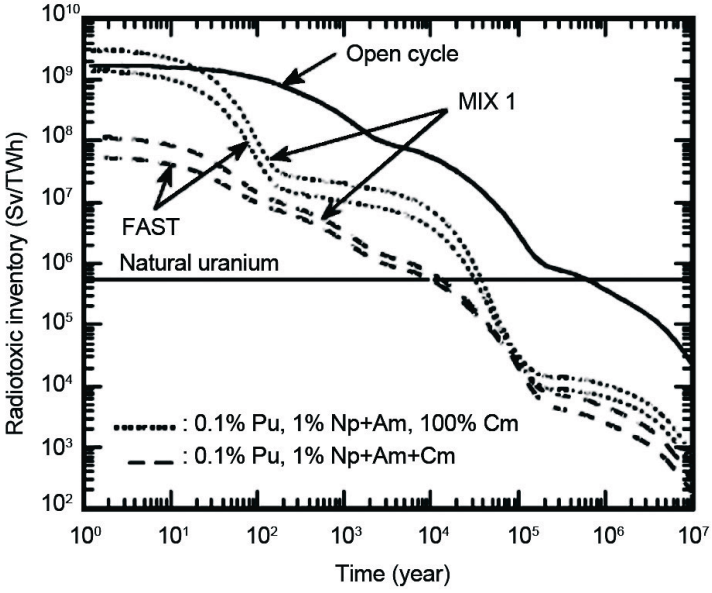
In order to fit the partitioning and transmutation option into the context of existing waste management strategies, the direct disposal concepts in granite formations of Spain and Sweden are briefly described as examples. A section discusses the issue of criticality safety in the disposal of spent fuel.

In the case of the reprocessing fuel cycle, the impact of the disposal of high- and medium-level waste as scheduled to be realised in Germany, Switzerland and Belgium, respectively in salt, granite and in clay, are given as illustrations. A number of options are being studied on disposal of waste forms resulting from the advanced fuel cycle incorporating partitioning and transmutation. Nevertheless, the report sheds some light on the technical and operational aspects of such a strategy; in particular, intermediate storage management, the fate of irradiated targets and the residual reactor cores from a prolonged nuclear energy production.

In addition to the highly radioactive materials resulting from an advanced fuel cycle, attention should be drawn on the large inventories of depleted uranium produced during the enrichment processes. The depleted uranium stocks can in a short time interval be considered as a strategic material to be reused in case the FRs would become an important fraction of the total electricity generating capacity.

If the FRs do not emerge as nuclear electricity producers, depleted uranium will be considered as a waste material whose radiotoxicity will eventually become equal to that of natural uranium ore.

Figure 3.1. Evolution of radiotoxicity in wastes



"MIX1" indicates reactor system consisting of PWR (70%) and CAPRA type FR (30%) for multi-recycling of actinides.

Source: OECD/NEA, 1999 [55], Figure II.25.

3.2.2 IAEA 2004 report Implications of Partitioning and Transmutation in Radioactive Waste Management *[56]*

The main purpose of this report was to provide useful technical information for decision makers on the expected long-term consequences of their decisions on waste management. Chapter 2 of the report described the potential impact of P&T on radioactive waste management. One of summary paragraphs in the concluding chapter stated:

> "The application of P&T would, if fully implemented, result in a significant decrease in the transuranic inventory to be disposed of in geological repositories. Currently, it is believed that the inventory and radiotoxicity can be reduced by a factor of 100 to 200 and that the time scale required for the radiotoxicity to reach reference levels (natural uranium) will be reduced from over 100 000 years to between 1 000 and 5 000 years. To achieve these results it is believed that it would be necessary for plutonium and neptunium to be multiple recycled and for americium (curium) to be incinerated in a single deep burn step." [56]

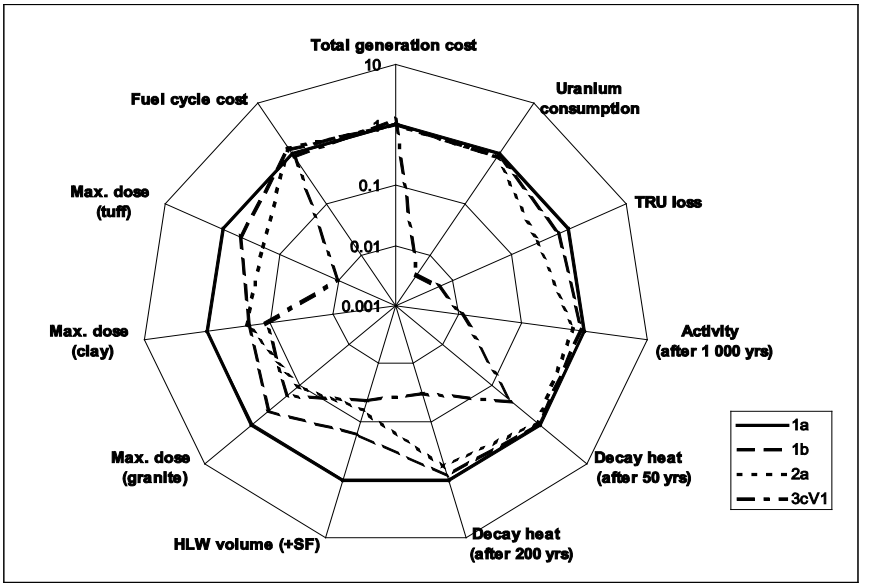
The report also discussed the non-proliferation aspects of P&T in Chapter 3. One of the summary paragraphs in the concluding chapter noted that "The proliferation risk potentially associated with P&C [1] and P&T development and application, and their impact on the non-proliferation regime and on IAEA safeguards, should be addressed at the early stages of P&C and/or P&T programmes." [56]

1. "P&C" means "partitioning followed by conditioning".

3.2.3 OECD/NEA 2006 report Advanced Nuclear Fuel Cycles and Radioactive Waste Management *[19]*

The OECD/NEA conducted a comprehensive assessment of various types of nuclear fuel cycle from the viewpoints of uranium consumption; TRU loss/transfer to waste; radioactivity, decay heat and volume of wastes; maximum dose for repositories in various geological formations; fuel cycle cost; and total electricity generation cost. The results of the assessment were summarised as a radar chart shown in Figure 3.2.

Figure 3.2. Comparison of 11 representative indicators for various fuel cycle schemes



1a: once-through PWR scheme (reference); 1b: 100% PWR, spent fuel reprocessed and Pu reused once; 2a: 100% PWR, spent fuel reprocessed and multiple reuse of Pu; 3cV1: 100% fast reactors and fully closed fuel cycle.

Source: OECD/NEA, 2006 [19], Figure 1.

The executive summary of the report indicated the impact of advances fuel cycles as follows:

- TRU loss is the indicator which is the most sensitive to the details of the schemes. Multi-recycling can divide the TRU flow to waste by six and deeper burning schemes allow for a further reduction of up to two orders of magnitude of this flow.
- The activity of HLW after 1 000 years describes the radioactive source term after decay of heat-generating isotopes. At this horizon, the short-lived fission products have decayed strongly and the removal of actinides from the waste flux is very efficient in decreasing the "cold" source term. Indeed, the fully-closed fuel cycle schemes reduce by almost two orders of magnitude the activity of HLW after 1 000 years.
- Decay heat is a major input for the design of underground repositories. For disposal in granite, clay and tuff formations the maximum allowable disposal density is determined by thermal limitations. HLW arising from advanced fuel cycle schemes generates considerably less heat than the spent fuel arising from the reference PWR once-through scheme. This lower thermal output of HLW allows a significant reduction in the total length of disposal galleries needed. Separation of caesium and strontium reduces even further the required repository size. For example, in the case of disposal in a clay formation the gallery length needed in the HLW disposal is reduced by a factor of 3.5 through a fully-closed cycle scheme as compared with the reference PWR once-through scheme and by a factor of 9 through a scheme including separation of caesium and strontium.

- After 50 years of cooling, variations in decay heat of HLW do not exceed a factor of 4 for any of the fuel cycle schemes considered in the study. After 200 years, the decay heat of HLW would be reduced by a factor of up to 30 in all MA-burning schemes as compared to the reference PWR once-through scheme. Extending the cooling time from 50 to 200 years will result in a drastic reduction of the thermal output of HLW from advanced fuel cycle schemes and, consequently, of the repository size needed.
- The volume of HLW to dispose of is a driving factor to determine the total capacity of a given repository site. This volume is reduced significantly by closed fuel cycle schemes as compared with the reference PWR once-through scheme. Further HLW volume reductions, by an order of magnitude, are achievable by progressing towards deeper burning of minor actinides and lesser use of uranium.
- Differences in heat load and waste volume may have a major impact on the detailed concept of the repositories. This in turn has technical and economic impacts. For example, a given repository could receive the waste issued by the production of 5 to 20 times more electricity if the electricity were produced by advanced reactors associated with advanced fuel cycle processes than if it were produced by light water reactors operated once through.
- The maximum dose released to the biosphere at any time in normal conditions remains well below authorised limits for all the schemes and all the repositories considered. For all the repositories examined in the study, the maximum dose resulting from the disposal of the high-level waste does not differ significantly for any of the various fuel cycle schemes envisaged.
- The doses resulting from the disposal of HLW from fuel cycle schemes with reprocessing are at most a factor of 8 lower than those from the reference PWR once-through scheme. However, the lower dose mainly results from the removal of ^{129}I from the liquid HLW during reprocessing; should it be captured and disposed of in the HLW repository, the doses resulting from all scenarios would be about equal. In the very long term, i.e. after a few million years, the total dose is lower in the case of the fully closed fuel cycle schemes, because much smaller amounts of actinides have to be disposed of in the repository.

3.2.4 European project "RED-IMPACT" [52]

The RED-IMPACT project was an EU-funded broad approach to analyse the impact of P&T and waste reduction technologies on the nuclear waste management and particularly on the final disposal. Twenty-three organisations including nuclear industry, waste agencies, research centres and universities from EU countries participated in the project.

Five representative scenarios, ranging from direct disposal of the spent fuel to fully closed cycles (including MA recycling) with fast reactors and accelerator-driven systems, were chosen to cover a wide range of representative waste streams, fuel cycle facilities and process performances. High- and intermediate-level waste streams were evaluated for all 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 the study can be summarised 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 significantly reduces 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 of 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 must be compared to potential savings on the repository, with 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.

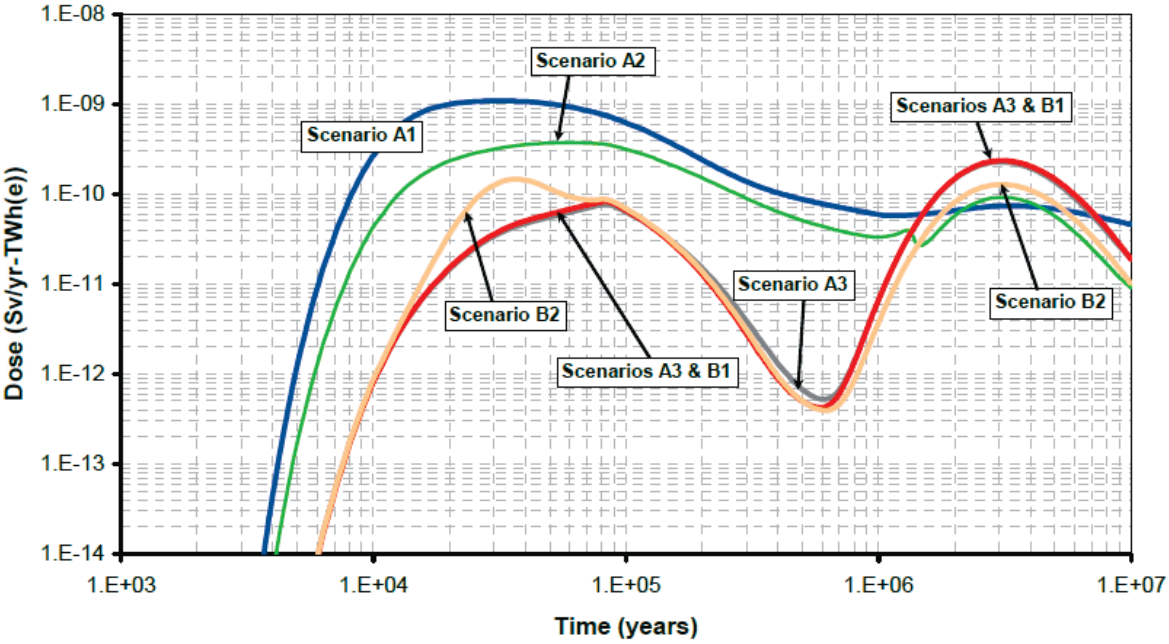
J. Marivoet *et al.* [57], as a part of RED-IMPACT, compared five fuel cycle scenarios and two environments (granite and clay). The results are summarised in Box 3.2.

In a paper based on results obtained in the OECD/NEA study [19] and the RED-IMPACT project [52], J. Marivoet and E. Weetjens compared six fuel cycle scenarios for a possible repository excavated in a clay formation at the Mol site [26]. A summary of the results can be found in Box 3.3.

Box 3.2. Summary of results extracted from a comparison of five fuel cycle scenarios and two environments, by Marivoet *et al*. [57]

The impact of advanced nuclear fuel cycles on radioactive waste management and geological disposal has been evaluated within the RED-IMPACT project. Five representative fuel cycles, which were considered in equilibrium, were identified and the resulting waste volumes and compositions were estimated. Repository designs for disposal in hard rock and clay formations, developed by national radioactive waste management agencies for today's waste types, were used as reference concepts. After a 50-year cooling time, the heat generated in the high-level radioactive waste arising from advanced fuel cycles is significantly lower than that in spent fuel from the present "once-through" fuel cycle. This would allow the dimensions of a geological repository to be comparatively reduced. The impact of advanced fuel cycles on the radiological consequences in the case of the expected evolution or reference scenario is rather limited (Figure 3.3). This is because the maximum dose in this scenario, which is calculated to occur a few tens of thousands of years after the disposal of the waste and is associated with radionuclide transport in groundwater, is essentially due to mobile fission and activation products; as geological disposal systems are very effective at retarding the migration of actinides, the contribution of the actinides to the effective dose is limited. The associated intermediate-level waste contains considerable amounts of mobile activation products; these species persist in giving relatively high post-closure doses. On the other hand, for the variant human intrusion scenario, calculated doses to a geotechnical worker resulting from inadvertent intrusion into a high-level waste repository are significantly reduced in the case of advanced fuel cycles, because of the much lower actinide content of the waste. However, it should be noted that the realism of human intrusion scenarios, and the weight that could be placed on the associated outcome, is strongly debatable.

Figure 3.3. Total doses due to disposal of SF and HLW for the five considered fuel cycle scenarios (disposal in granite)



Scenario A1: "once-through" (PWR) with uranium oxide fuel; Scenario A2: PWR and Pu is recycled once as mixed-oxide (MOX) fuel; Scenario A3: a sodium-cooled FR with MOX fuels, Pu is multi-recycled; Scenario B1: a sodium-cooled FR with MOX fuels, all actinides are recycled; Scenario B2: Scenario A2, and MA+Pu in the spent MOX fuel are recycled in ADS.

Source: Marivoet *et al.*, 2009 [57], Figure 3.

Box 3.3. Comparison of six fuel cycle scenarios for a possible repository excavated in a clay formation at the Mol site – summary of results [26]

This paper presents evaluations of the impact of six advanced fuel cycles, ranging from the present "once-through" fuel cycle in light water reactors to a gas-cooled fast reactor with fully recycling of all actinides, on geological disposal in a clay formation. Both the dimensions and the radiological consequences of a geological repository for the disposal of high-level radioactive waste (HLW) and spent fuel are estimated. After a 50-year cooling time, the thermal output of the HLW arising from advanced fuel cycles is significantly lower than that of spent fuel. This allows the dimensions of the geological repository to be reduced. The impact of advanced fuel cycles on the radiological consequences in the case of the expected evolution scenario is rather limited. The maximum dose, which is expected to occur a few tens of thousands of years after the disposal of the waste, is essentially due to fission products, and their amount is approximately proportional to the heat generated by nuclear fission. An important contributor to the total dose is ^{129}I; the amount of ^{129}I going into a repository strongly depends on the fraction of spent fuel that is reprocessed. By considering the evolution of the radiotoxicity of the waste, it can be expected that the radiological consequences of human intrusions into a repository will be significantly lower in the case of waste arising from advanced fuel cycles.

3.2.5 Study in Germany

H. Geckeis *et al.* [58] summarised their study entitled "Impact of Innovative Nuclear Fuel Cycles on Geological Disposal" as follows:

- The thermal output of high-level nuclear waste arising from present and future reactor concepts is determined for several hundreds of years by the decay of the fission products ^{137}Cs and ^{90}Sr. After solidification most likely by vitrification of waste containing those radionuclides, the highly active product has to be disposed in a geological repository. A considerable heat load reduction in a repository by P&T is thus only achievable by extended intermediate storage of spent fuel. P&T can, however, contribute to the optimisation of required repository space by a factor of up to 3 by separating minor actinides, notably ^{241}Am [19].
- The impact of P&T on performance assessment calculations made for a number of disposal concepts in most cases is limited. The reason lies in the low solubility and strong retention of actinides in a repository system in case of groundwater access. Dose-relevant radionuclides are thus in most cases the long-lived fission and activation products, which are difficult to be transmuted and which are considered as quite mobile under relevant geochemical conditions.
- Slow release of actinides can occur from the highly active wastes (spent fuel and glass) due to limited corrosion rates under relevant groundwater conditions. Actinides exhibit under reducing conditions relatively low solubility. They exist in tri- and tetravalent oxidation states and thus undergo either strong sorption to mineral surfaces or are efficiently scavenged in secondary solid phases, precipitating in the near or far field of a repository. There are however processes that may counteract actinide retention. Corrosion of spent fuel and release of actinides can be promoted by radiolysis which generates oxidising conditions. Waste glass converts to thermodynamic stable mineral phases in case of groundwater access and radionuclides potentially can be released. Formation of colloidal actinide species is in addition discussed as a potentially relevant process for actinide mobilisation.
- Research activities on partitioning and conditioning (P&C) strategies represent an alternative to P&T. Separation of actinides and other long-lived radionuclides and their subsequent incorporation into alternative tailored matrices may help to reduce the actinide source term from a repository in case of water access. Minerals and ceramics

investigated for P&C are characterised by high thermodynamic stability, and thus low solubility in groundwater. Among others monazite, pyrochlore, zircon and zirconolite are discussed as potentially appropriate matrices. Thermodynamic data on solubility of radionuclide-bearing ceramics under groundwater-relevant conditions needed to predict their long-term properties are, however, still rare.

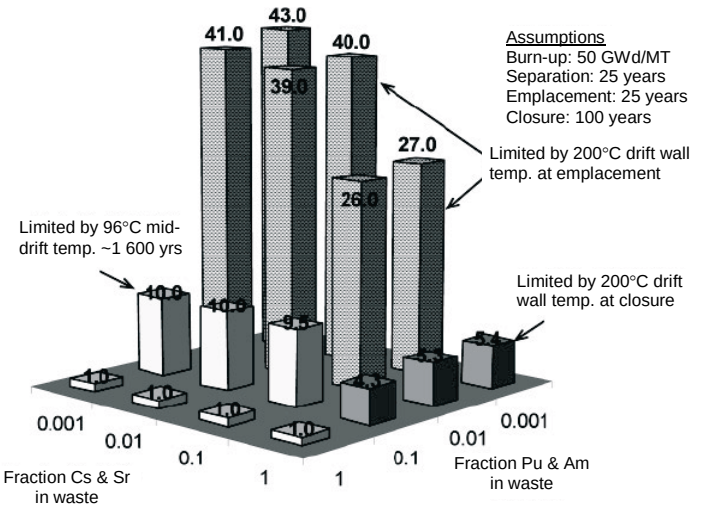
- Future P&T and P&C strategies will certainly influence nuclear waste disposal concepts to some extent. Notably for radionuclide release scenarios such as human intrusion or actinide mobilisation by colloids, P&T can minimise estimated resulting doses to the population. The expected waste from reprocessing plants and the vitrified high-level liquid waste already present today will however make geological disposal unavoidable.

3.2.6 Studies in the United States

Wigeland *et al.* [53] analysed the effect of P&T on the Yucca Mountain repository that was being considered in the United States. The summary of the paper said:

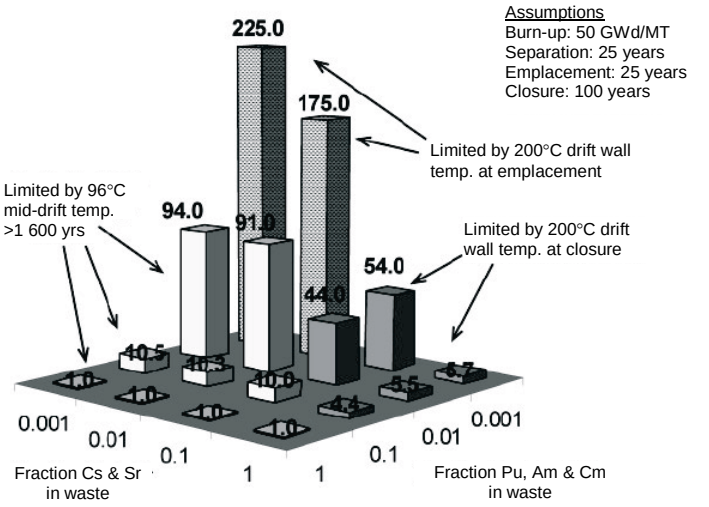
> "This paper describes the results of a study that uses the thermal performance of the repository to establish chemical separations and transmutation criteria for commercial spent nuclear fuel of benefit to a geological repository, as measured by the allowable increase in utilisation of repository space. The method for determining the chemical elements to be separated is based on the thermal performance of the repository. The important chemical elements are identified, the order of importance of the separated elements is established and the relationship between the efficiency of the chemical separation and the resulting increase in utilisation is determined. The proposed repository at Yucca Mountain is used as an example of a geological repository for the purposes of illustrating the magnitude of the benefits that are possible and the implications for repository size and operation. This work is being done in support of the US Department of Energy Advanced Fuel Cycle Initiative, where numerous reactor, processing and recycling strategies are being examined to determine the impact on issues important to the viability of nuclear electricity generation, including the disposal of spent nuclear fuel and nuclear waste."

Figure 3.4. Potential repository drift loading increase as a function of separation efficiency for plutonium, americium, caesium and strontium



Source: Wigeland *et al.*, 2006 [53], Figure 7.

Figure 3.5. Potential repository drift loading increase as a function of separation efficiency for plutonium, americium, curium, caesium and strontium



Source: IAEA, 2006 [9], Figure 8.

Figures 3.4 and 3.5 represent Wigeland's results on the potential repository drift loading increase as a function of the separation efficiency for actinides and fission products [53]. The difference between the two figures is the treatment of curium; Figure 3.5 shows that the high-efficiency recovery of actinides including curium combined with FP separation has a potential to increase the loading capacity by a factor of more than 100.

Wigeland *et al*. [59] also conducted a study on the impact of limited actinide recycle in PWR. The abstract of the paper said:

> "A project has been conducted as part of the US Department of Energy Advanced Fuel Cycle Initiative to evaluate the impact of limited actinide recycling in light water reactors on the utilisation of a geological repository where loading of the repository is constrained by the decay heat of the emplaced materials. In this study, it was assumed that spent PWR fuel was processed, removing the uranium, plutonium, americium and neptunium, along with the fission products caesium and strontium. Previous work had demonstrated that these elements were responsible for limiting loading in the repository based on thermal constraints. The plutonium, americium and neptunium were recycled in a PWR, with process waste and spent recycled fuel being sent to the repository. The caesium and strontium were placed in separate storage for 100-300 years to allow for decay prior to disposal. The study examined the effect of single and multiple recycles of the recovered plutonium, americium and neptunium, as well as different processing delay times. The potential benefit to the repository was measured by the increase in utilisation of repository space as indicated by the allowable linear loading in the repository drifts (tunnels). The results showed that limited recycling would provide only a small fraction of the benefit that could be achieved with repeated processing and recycling, as is possible in fast neutron reactors." [59]

An early 2000 study by C.W. Forsberg [60] indicated the impact of high-heat FP:

> "An alternative approach for disposal of high-level waste (HLW) is proposed. HLW would be separated into two fractions: a) the high-heat radionuclides (HHRs), e.g. ^{90}Sr and ^{137}Cs, and b) the low-heat radionuclides (LHRs), which are all the remaining radionuclides. These two categories of waste would be disposed of separately in different sections of the repository or different facilities.

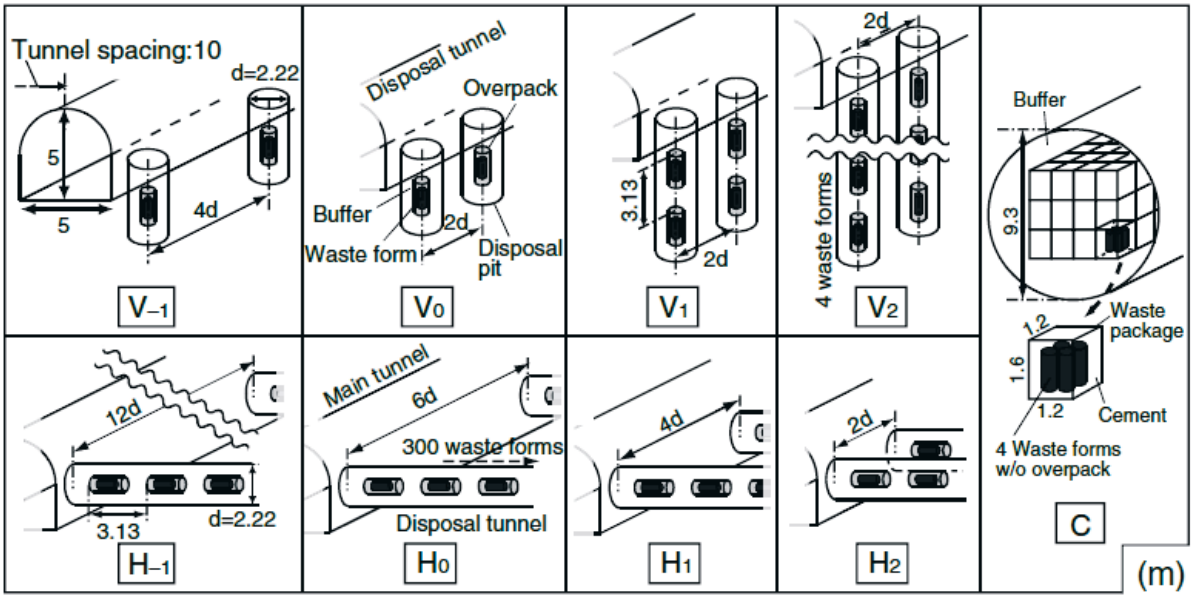
The LHRs in the HLW contain the long-lived radionuclides that control the repository performance requirements that in turn necessitate: a) expensive waste packages (WPs) and b) limiting the repository temperatures to avoid repository performance degradation. To limit repository temperature, the amount of HLW per WP is limited and the WPs are spread over a large area. If the decay-heat-generating HHRs are removed from HLW, the repository design is not controlled by decay heat. The resultant LHR repository size (area, number of WPs, total tunnel length) may be reduced to <20% of the size of a conventional repository. With a waste partitioning and transmutation process that includes removal of the minor actinides (americium and curium) from the LHR wastes, significant further reductions in repository size are possible. The minor actinides are the next largest heat generators in LHR wastes.

Separate management of HHRs does require: a) separation of the HHRs from the HLW and b) a separate HHR disposal facility. The HHRs are disposed of in a separate lower-cost facility made possible by the limited lifetimes ($T_{1/2}$ ~30 yr) of the HHRs. There are potentially significant gains in economics and repository performance for separate management of HHRs and LHRs in some types of fuel cycles." [60]

3.2.7 Studies in Japan

K. Nishihara *et al.* [54] made an analytical study on the impact of P&T for HLW from LWR based on the Japanese reference disposal configuration. They discussed some emplacement configurations shown in Figure 3.6 to explore the possibility of condensed disposal of wastes according to heat generation. The relationship between the emplacement area and the burden of the storage of the wastes before the disposal was also addressed (Figure 3.7). The abstract of the paper is presented in Box 3.4.

Figure 3.6. Emplacement configurations assumed for thermal analysis

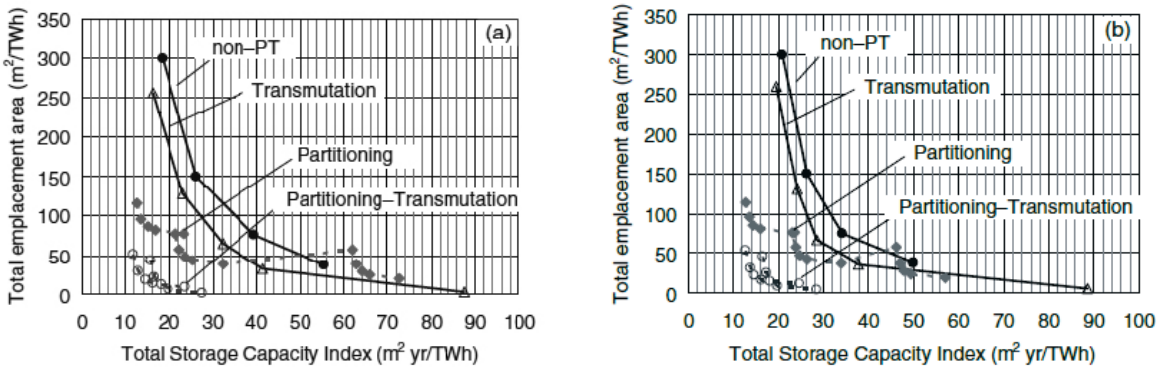


V_0 and H_0 are configurations regarded as reference cases of the study; d (= 2:22 m) is the diameter of the pit (vertical) or the disposal tunnel (horizontal).

Source: Nishihara *et al.*, 2008 [54], Figure 6.

Figure 3.7. Effects of P&T technology on predisposal storage and disposal of radioactive wastes in terms of the required storage capacity and the required emplacement area in a repository

(a) Vertical and compact emplacement and (b) horizontal and compact emplacement



The total storage capacity index is defined in this study as the summation of the area of storage pits multiplied by the storage period for each type of waste form.

Source: Nishihara *et al.*, 2008 [54], Figure 14.

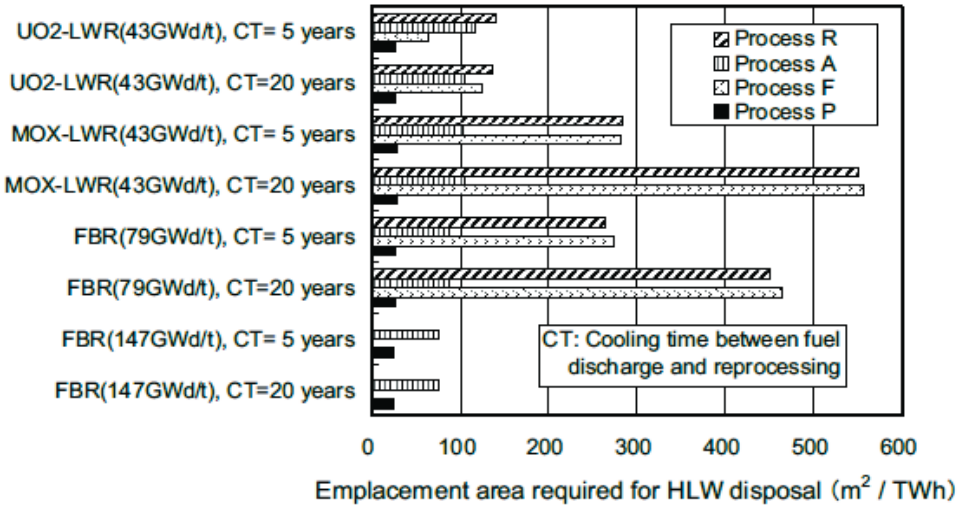
Box 3.4. Abstract of the Nishihara *et al*. analytical study [54]

Partitioning and/or transmutation (P&T) technology affects the disposal concept of high-level radioactive waste (HLW). We studied how cooling in the predisposal storage period may affect the design of the emplacement area in a repository for radioactive wastes produced by a light water reactor nuclear system that uses P&T technology. Three different fuel cycle scenarios involving P&T technology were analysed: *i)* partitioning process only (separation of some fission products); *ii)* transmutation process only (separation and transmutation of minor actinides); *iii)* both partitioning and transmutation. The necessary predisposal storage periods for some predefined emplacement configurations were determined through transient thermal analysis, and the relation between the storage period and the emplacement area was obtained. For each scenario, we also estimated the storage capacity required for the dry storage of the heat-generating waste forms. The contributions of P&T technology on the storage and disposal were discussed holistically, and we noted that the coupled introduction of partitioning and transmutation processes can bring an appreciable reduction in waste management size.

Oigawa *et al*. [61] made a parametric survey on the emplacement areas of HLW from various spent fuels. The results are summarised in Figure 3.8 and the abstract of the study noted:

> "In Japan, the partitioning and transmutation (P&T) technology is being studied and developed aiming at the reduction of the burden caused by the high-level radioactive waste (HLW) management. To assess the benefit of the P&T technology in the future nuclear fuel cycles, the repository area necessitated to dispose of the HLW was discussed quantitatively for the spent fuels from UO_2-LWR, MOX-LWR and MOX-FBR. Four options of the separation process were assumed in the analysis: *i*) conventional PUREX reprocessing; *ii*) minor actinide (MA) recycling without partitioning fission product (FP); *iii*) partitioning of FP without MA recovery; *iv*) full P&T for both MA and FP. The areas required to emplace waste forms per unit electricity production (m^2/TWh) were then compared. The results showed that MA recycling significantly reduced the emplacement area for MOX spent fuels from both LWR and FBR. The full P&T scheme can give further reduction of the emplacement area (i.e. the enhancement of the capacity of a repository site) independently on the fuel type, the reactor type and the cooling period." [61]

Figure 3.8. Estimated total emplacement area for each case (normalised by 1 TWh)



Process R: conventional PUREX reprocessing; Process A: MA recycling without partitioning FP; Process F: partitioning of FP without MA recovery; Process P: full P&T for both MA and FP.

Source: Clark *et al.,* 2009 [13], Figure 4.

Oigawa *et al.* [62] also made an analysis for the possibility of the very compact disposal by coupling P&T and long-term predisposal storage. The results are summarised in Table 3.1, and the abstract of the paper can be found in Box 3.5.

Table 3.1. Five typical concepts of waste management for HLW from UO_2-LWR (5-y cooling)

Normalised for 1 TWh

Case	Waste	Waste form	Volume of waste forms	Predisposal storage	Emplacement area
Process-R	HLW	Normal glass	479 L	50 y	140 m²
Process-A	HLW w/o MA	Normal glass	398 L	50 y	117 m²
Process-P	Ln	High-density glass	46 L	18 y	3.4 m²
	Precipitation	High-density glass	82 L	5 y	1.4 m²
	Sr, Ba	Calcined form	28 L	130 y	8.7 m²
	Cs, Rb	Calcined form	34 L	150 y	11 m²
	Tc-PGM	Alloy waste	44 L	7 y	3.0 m²
		Total	234 L	(av. 44 y)*	27 m²
Process-A + Long-term predisposal storage	HLW w/o MA	Normal glass	398 L	330 y	2.5 m²
Process-P + Long-term predisposal storage	Ln	High-density glass	46 L	60 y	0.3 m²
	Precipitation	High-density glass	82 L	9 y	0.5 m²
	Sr, Ba	Calcined form	28 L	320 y	0.2 m²
	Cs, Rb	Calcined form	34 L	330 y	0.2 m²
	Tc-PGM	Alloy waste	44 L	110 y	0.3 m²
		Total	234 L	(av. 122 y)*	1.5 m²

* Average period weighted by the volume of the wastes.

Source: Modified from Table 5 of Ref. [62].

Box 3.5. Abstract of the paper, by Oigawa *et al*. [62]

To illustrate the benefit of the partitioning and transmutation (P&T) technology, several types of the waste management and the geological disposal concepts incorporating P&T were discussed in terms of the repository size required to emplace the wastes. It was found that the transmutation of ^{241}Am, which is a long-term heat source, is effective to prevent the inflation of the repository size expected in the case of plutonium utilisation. If we intend to reduce the repository size or enhance the capacity to a larger extent, for example by a factor of 4-100, in comparison with the conventional Japanese repository design for glass waste forms, the partitioning of Sr and Cs followed by their long-term (100-300 years) storage should be adopted together with the transmutation of MA. In such cases, to reduce the burden of the long-term storage of Sr and Cs, they are recommended to be contained in heat-resistant waste forms such as calcined wastes. The long-term storage without MA transmutation cannot achieve significant reduction of the repository size such as factor of 100 because of long-term heat from actinides.

3.3 Analysis and comparison of the results

3.3.1 Peak dose rate

The peak dose rate is the most general indicator used to evaluate the performance of a radioactive waste disposal repository. As described in Chapter 2, the groundwater will play an important role to transport waste elements out of the disposal system in the case of the normal evolution of a repository excavated in granite or clay. Table 3.2(a) presents the descriptions in the past studies, and Table 3.2(b) summarises them.

It should be noted that in all the normal evolution scenarios the estimated peak doses can be significantly lower than the regulatory limits (e.g. 0.3 mSv/yr recommended by ICRP [11] as the upper value for dose constraint for application in normal exposure situation).

Table 3.2(a). Comparison of past studies on peak dose rate

Reference	Description
OECD/NEA report on advanced fuel cycles (2006) [19]	• ***Granite by ENRESA, Spain.*** ^{129}I dominates the peak dose 0.001 μSv/y/TWh at around 20 000 years for the direct disposal. The reprocessing of spent fuel lowers the contribution of ^{129}I because of its 99.9% release at the process, and the peak dose 10^{-4} μSv/y/TWhe at 300 000 years is dominated by ^{135}Cs. Introduction of fast reactors increases this peak dose by ^{135}Cs by about a factor of 2 because of a higher yield of ^{135}Cs by ^{239}Pu fission and lower neutron capture of precursor nuclide ^{135}Xe. • ***Granite by JNC, Japan***. ^{135}Cs dominates the peak dose of vitrified HLW which is well below 10^{-6} μSv/y/TWhe at around 500 000 years, where ^{129}I is assumed to be excluded from HLW. After about 5 million years, ^{229}Th dominates the dose. • ***Clay, SCK•CEN, Belgium***. ^{129}I dominates the peak dose of 5×10^{-5} μSv/y/TWhe at 200 000 years. The reprocessing of spent fuel lowers the contribution of ^{129}I assuming its 99.9% release at the process, and the peak dose 10^{-5} μSv/y/TWhe at 500 000 years is dominated by ^{126}Sn. Introduction of gas-cooled fast reactors lowers somewhat this peak dose (5×10^{-6} μSv/y/TWhe) because of its better efficiency. Separation of Cs and Sr results in the compact repository configuration, which strengthens the contribution of the solubility limit for Se, Tc, Sn and actinides, and lowers the dose. • ***Salt, GRS, Germany***. The normal evolution scenario in salt leads inevitably to zero release. • ***Tuff, ANL, United States***. By adopting TRU recycling by GFR, 2-order reduction of peak dose can be foreseen because it is dominated by actinides and their decay products. (Note that after the revision in 2008 both actinides and fission products are contributing to the dose [33].)

Table 3.2(a). Comparison of past studies on peak dose rate (*cont.*)

Reference	Description
European project RED-IMPACT (2007-8) [26, 52, 57]	• A deep geological repository to host the remaining HLW and possibly the long-lived ILW is unavoidable. • Even in the cases w/o P&T, dose levels at the surface are significantly lower than regulatory limits. • Comparisons were made for six scenarios and three environments (granite, clay and salt). • ***Granite, Spain***. ^{129}I dominates the peak dose of 10^{-3} μSv/y/TWhe at 30 k years after disposal for once-through, while ^{135}Cs dominates that of about 2×10^{-4} μSv/y/TWhe at 3 million years for fully-closed cycle. • ***Clay, France***. ^{129}I dominates the peak dose of 10^{-3} μSv/y/TWhe at 300 000 years after disposal for once-through, while the peak dose will be reduced by a factor of 100 for reprocessing because of 99% removal of ^{129}I. The actinides are considered to be strongly sorbed by the clay minerals. • ***Salt, Germany***. All the scenarios result in extremely low dose. • In general for granite and clay, total doses are dominated by mobile FP and activation products (AP), and removal or capture of ^{129}I could have stronger impact. MA transmutation reduces radiotoxicity drastically after a few centuries cooling time, which would lead to potential hazard in the case of inadvertent human intrusion. For example, the time necessitated to reduce the dose of geotechnical worker scenario below ICRP 100 mSv intervention level can be shortened from 100 000 years to 1 000 years by introducing TRU recycling.

Table 3.2(b). Comparison of evaluated impacts on peak dose rates

Host rock	Ref.	Direct disposal of UO_2 spent fuel	HLW (and MLW) of UO_2 spent fuel	HLW of FBR spent fuel
Granite (Spain)	[19]	10^{-3} μSv/y/TWh (^{129}I)	10^{-5} μSv/y/TWh (^{135}Cs)	2×10^{-4} μSv/y/TWh (^{135}Cs)
Granite (Japan)	[19]	–	$<10^{-6}$ μSv/y/TWh (^{135}Cs)	–
Clay (Belgium)	[19]	5×10^{-5} μSv/y/TWh (^{129}I)	10^{-5} μSv/y/TWh (^{126}Sn)	5×10^{-6} μSv/y/TWh (^{126}Sn)
Clay (France)	[19]	10^{-3} μSv/y/TWh (^{129}I)	10^{-4} μSv/y/TWh (^{129}I)	1.5×10^{-5} μSv/y/TWh (^{129}I)
Salt (Germany)	[19]	Extremely low dose	Extremely low dose	Extremely low dose
Tuff (United States)	[33]	~20 μSv/y due to ^{239}Pu, ^{242}Pu, ^{237}Np, and ^{226}Ra	~6 μSv/y from the ^{237}Np and ^{226}Ra	~4 μSv/y from ^{129}I

For clay and granite in the normal evolution scenarios, ^{129}I obviously dominates the peak dose if it is disposed of into the repository, either using direct disposal or the reprocessing case. When most of ^{129}I is removed from HLW in the reprocessing case, the peak dose is dominated by ^{79}Se, ^{36}Cl and ^{135}Cs. The effect of MA transmutation on the peak dose rate is therefore limited in the normal evolution scenarios for these types of host rock. In the case of the Yucca Mountain evaluation, changes made to input parameters for the performance assessment that substantially slowed waste package corrosion and delayed waste package failure, among other things, resulted in the current modest impact from actinide P&T as compared to the more significant impact reported in 2006. In the case of salt domes, no release of radionuclides is expected in normal evolution.

On the other hand, MA plays an important role in the human intrusion scenarios, especially in later years (after 1 000 years of disposal). MA transmutation will reduce the dose to an intruder by two orders of magnitude. It is therefore reasonable to regard the MA transmutation as a measure to reduce the future uncertainty owing to some unlikely disturbances of all the barriers of the repository system. From this point of view, the cost and the benefit of MA transmutation should be discussed further, though it will be difficult to estimate the future uncertainty quantitatively.

3.3.2 Radiotoxicity

Table 3.3 lists the descriptions in the past studies. The radiotoxicity has been used as the index to demonstrate the effect of MA transmutation. All the previous studies assumed 99-99.9% recovery of TRU, and eventually about two orders of reduction of radiotoxic inventory were foreseen after several hundred years and later. This index is also used to emphasise the effect of MA transmutation from a viewpoint of reduction of time period which is necessary to wait for the decrease of this index down below a reference level. Once this occurs, the radiotoxicity of the natural uranium is used. In general, MA transmutation shortens such a time period to several hundred years, though more than 10 000 years will be necessary in the case of direct disposal.

Table 3.3. Comparison of past studies on radiotoxicity

Reference	Description
OECD/NEA P&T report (1999) [55]	• Assuming the loss of TRU during reprocessing operations as 0.1% for Pu and 1% for MA, the reduction of radiotoxic inventory in the wastes can be a factor of 100 in general independent of the scenarios and time after the reprocessing. • There is no immediate advantage in recycling Np. • Cm must ultimately be considered if a maximum inventory reduction is intended.
IAEA P&T report (2004) [56]	• It takes 130 000 years before the radiotoxicity reaches the reference level. • It is reduced to 500 years by the recovery of 99.5% of Pu and 99% of Am+Cm. If Cm is left in the waste, this time is extended to 1 000 years. • Recovery of 99.5% of Pu and 95% of Am+Cm results in the crossover point of 1 000 years. • Recovery of 99.5% of Pu combined with one single recycling of Am+Cm with a 90% efficiency results in the crossover point of 1 500 years. If Cm is left in the waste, this time is extended to 3 000 years.
OECD/NEA report on advanced fuel cycles (2006) [19]	• Fully closed fuel cycle leads to the activity reduction by a factor of 3 500 from 50 to 1 000 years, while direct disposal and partially closed cycle leads to a factor of about 100 reduction.
European RED-IMPACT project (2007-8) [26, 52, 57]	• After several hundred years, about two orders of reduction can be foreseen with a fully closed fuel cycle in comparison to a once-through scenario. For example, at 1 000 years after the disposal, 5×10^7 Sv/TWh will be reduced to 3×10^5 Sv/TWh. • MA transmutation reduces radiotoxicity drastically after a few centuries cooling time. The time necessitated to reduce the radiotoxicity below the fresh uranium required for the production of fuel cycle can be shortened from 200 000 years to 300 years by introducing TRU recycling. • The radiotoxicity of HLW after 500 years is drastically reduced by MA transmutation, which indicates that the potential hazard in the case of inadvertent human intrusion is considerably reduced.

3.3.3 Decay heat

In comparison with the estimation of the peak dose rate, the decay heat of the waste can be calculated without major assumptions, and therefore realistic discussions on the repository design are very active. Table 3.4 lists the past studies on the decay heat and the repository design.

In general, reduction of gallery length by factor of 3-6 can be foreseen by the TRU transmutation in comparison with direct disposal. Additional gain can be also foreseen by separating Cs and Sr, and by storing them for 100-300 years.

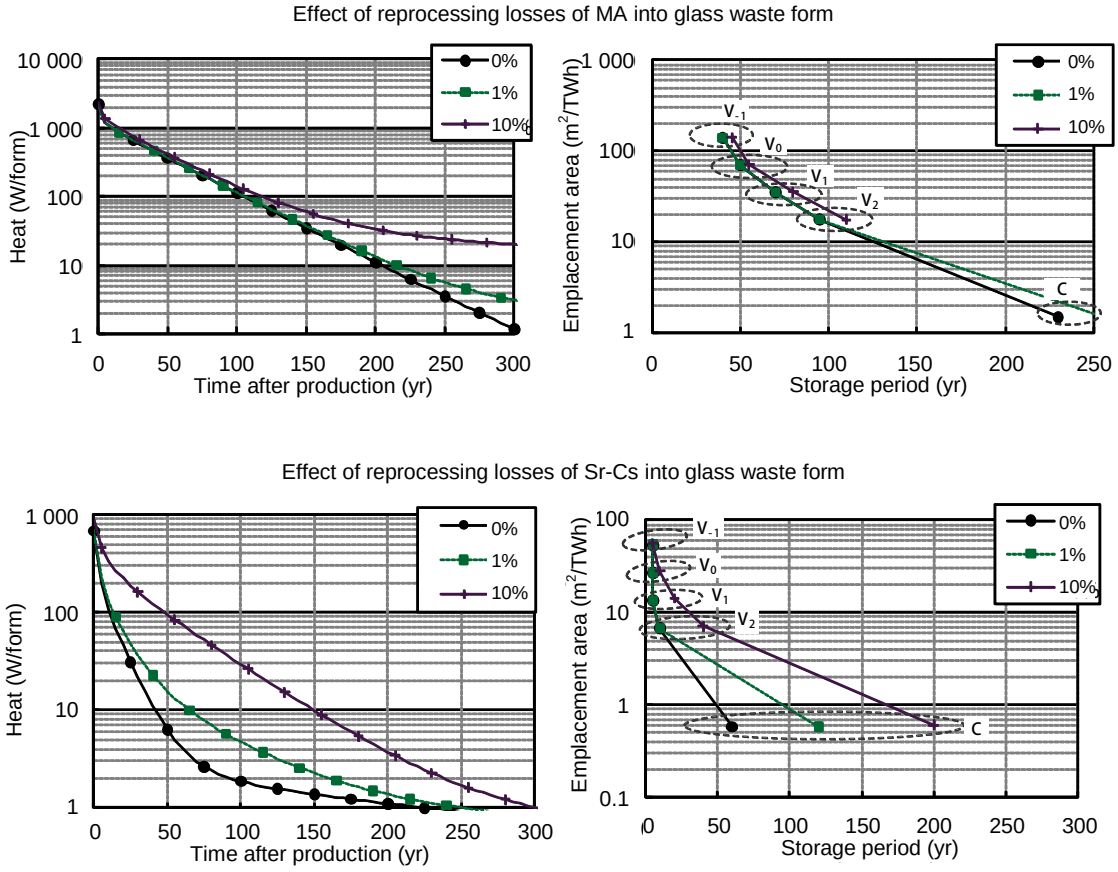
Table 3.4. Comparison of past studies on decay heat

Reference	Description
OECD/NEA report on advanced fuel cycles (2006) [19]	• Separation efficiency of 99.9% is assumed for all fuel types and reprocessing methods. • Length of disposal galleries can be reduced by: *i)* a factor of 4.2 by actinide recycle (granite); *ii)* a factor of 3.5 by actinide recycle and by a factor of 9.3 by additional separation of Cs and Sr (clay); *iii)* the footprint of the repository can be reduced by a factor of 15.9 by actinide recycle (tuff); *iv)* no description on gallery length (salt). • Extension of cooling time from 50 to 200 years provides the reduction of thermal output of waste by a factor of 3.6, while that of HLW with actinide recycle is reduced by a factor of 30. In the case of disposal in rock salt, the heat generation of the disposed waste contributes to a fast salt creep and void volume reduction. Therefore, a lowering of the thermal output of the high-level waste forms necessitates an optimisation of the waste packages and of the disposal configuration. • Separation and temporarily storage of Cs and Sr can reduce the number of vitrified HLW canisters by 25 to 40%. Additional cost of Cs/Sr separation and Cs/Sr interim storage results in a 5-10% increase in total cost. • Removing and sequestering Cs and Sr in a separate area of the repository or another facility would allow a further substantial increase in the drift loading of the repository, up to a factor of 43 in comparison with the direct disposal case for 99.9% removal of Pu, Am, Cs and Sr.
European RED-IMPACT project (2007-8) [26, 52, 57]	• Separation efficiency: 0.1% for U and Pu, 1% for MA by wet process, 0.1% for MA by dry process. • P&T of Pu and MA can reduce emplacement galleries length up to a factor of 3 in clay and hard rock formation, though there is no effect in salt formation. • Multi-recycling of actinides in FR reduces length of disposal galleries by a factor of 2.5 (granite) to 3.2 (clay) in comparison with once-through. • It can be significantly reduced by using longer cooling times or by separation of Cs and Sr. • By assuming a 100-year cooling time for conditioned Cs waste, the needed gallery length can be reduced by a factor of 4 in comparison with MA recycling and by a factor of 13 in comparison with once-through. Sr is assumed to be disposed of as short-lived waste.
Consideration on the impact of innovative fuel cycles in Germany (2008) [58]	• Repository capacities can be enhanced by a factor of ~3 by P&T with 50 years of cooling.
Investigation in the United States on the P&T impact on repository (2000, 2006-7) [53, 59, 60]	• Separation efficiency for Pu and Am: 99.9% – LF = 5.4, 99% – LF = 5.3, 90% – LF = 4.3 (LF = load factor relative to direct disposal; this factor may be influenced by an arbitrary condition of ventilation period). • Multi-recycle of Pu and Am with high recovery efficiency (more than 99%) is required for transmutation by FR. • Separation efficiency for Cs and Sr in addition to 99.9% removal of Pu and Am: 99.9% – LF = 42.7, 99% – LF = 40, 90% – LF = 26.8 (disposal of Sr and Cs is not discussed). • In addition to 99.9% removal of Pu, Am, Cs and Sr, Cm removal (99%) provides LF = 225. • The separations criteria proposed to benefit a repository at Yucca Mountain for the high-temperature operation of repository are also effective for low-temperature operation. • Cs and Sr are separated from process waste, and are stored separately for 200 to 300 years. • Transmutation in PWR is not very beneficial because the last assemblies, which will be directly disposed of, will occupy a large area due to the accumulated actinides. • The small LHR repository size allows more local flexibility in siting for disposal of the long-lived radionuclides. The best local rock can be used for these wastes. • If MA are additionally removed from LHR, very low heat radionuclide (VLHR) waste can be disposed of in silos. Using the example of YM, 10 silos would replace ~10 000 waste packages and ~100 km of tunnel.

Table 3.4. Comparison of past studies on decay heat (*cont.*)

Reference	Description
Investigation in Japan on the P&T impact on waste management (2007-8) [54, 61, 62]	• Storage (cooling) period before disposal of vitrified waste and Sr-Cs calcined waste are parametrically changed and the effect on the emplacement area for the wastes are discussed. The "storage capacity index" was defined to quantify the impact of long-term storage. • Full P&T combined with separated cooling (90-150 years) of Sr and Cs would reduce the emplacement area by a factor of 5. • In the case of MOX spent fuel for both LWR and FBR, transmutation of ^{241}Am is important especially for longer cooling times. • If we intend to reduce the repository size or enhance the capacity to a larger extent, for example by a factor of 4 to 100 in comparison with the conventional Japanese repository design for glass waste forms, the partitioning of Sr and Cs followed by their long-term (100-300 years) storage should be adopted together with the transmutation of MA. In such cases, to reduce the burden of the long-term storage of Sr and Cs, they are recommended to be contained in heat-resistant waste forms such as calcined wastes.

Figure 3.9. Decontamination factor and site area (vertical emplacement in crystalline rock)



V_{-1}, V_0, V_1, V_2 and C indicate the emplacement configurations in Figure 3.6.

Source: JAEA, 2009 [63], Figures 2-20, 2-22, 2-25 and 2-27.

Figure 3.9 shows the effect of the decontamination factor in terms of MA and Sr-Cs on the site area for vertical emplacement in crystalline rock in the case of TRU recycling in FBR. If very compact configuration is targeted, i.e. a reduction of the repository area by a factor of 100, more

than 99% of MA should be removed from the glass waste form. The cooling period necessary before the disposal in such a very compact configuration is dependent upon the decontamination factor of Sr and Cs.

These findings about the relation between decay heat and repository design are quite consistent among independent studies for different types of host rock. Therefore, P&T can be regarded as an effective measure to design compact repositories or to allow for larger capacity of one repository. It should be noted, however, that such condensed disposal may increase the peak dose rate because of larger loading of long-lived FP.

3.3.4 Waste form, volume and mass

By introducing fuel recycle, the volume and the mass of HLW can be significantly reduced mainly owing to the recovery of the uranium, while long-lived low- and intermediate-level waste would increase. A comprehensive and detailed comparison was presented in Ref. [19].

Conventional reprocessing adopts the vitrified waste form (or glass waste form) for the HLW. The introduction of P&T will increase the options of the waste forms. As for the vitrified waste, the density of the waste elements in the glass matrix might be increased because of the reduced heat source and removal of elements such as molybdenum and platinum group metals, which form separate phases in the glass [62]. Ref. [58] mentions the possibility of "partitioning and conditioning (P&C)", where MA is contained in thermodynamically stable host matrices such as monazite, pyrochlore, zircon and zircolite.

As for the Sr and Cs partitioned from HLW, there is no definite candidate for the waste form. Ref. [60] discusses the existing Hanford ^{137}Cs and ^{90}Sr capsules, where SrF_2 and CsCl were adopted as the chemical forms, respectively. In Ref. [54], Sr is assumed to be calcined into titanic acid and Cs into zeolite. These calcined forms are expected to show good performance in terms of high-temperature and long-term stability, and low leaching rate, which are all important characteristics for the waste forms with high heat generation.

Table 3.5. Comparison of past studies on waste form, volume and mass

Reference	Description
OECD/NEA report on advanced fuel cycles (2006) [19]	• The volumes of HLW for various schemes are compared in the unit of m^3/TWh. Direct disposal of the spent fuel leads to one-order-higher values than the cases of fuel cycle. On the other hand, those of long-lived low- and intermediate-level waste become larger in the cases of fuel cycle than the direct disposal.
European RED-IMPACT project (2007-8) [26, 52, 57]	• Total volume of HLW is 3.87 m^3/TWh for the once-through scenario, while it is 1.2-1.4 m^3/TWhe for the fully-closed fuel cycle scenario.
Consideration on the impact of innovative fuel cycles in Germany (2008) [58]	• Partitioning and conditioning (P&C) strategies were also discussed. Separation of actinides and other long-lived radionuclides and their subsequent incorporation into alternative tailored matrices may help to reduce the actinide source term from a repository in case of water access. • Minerals and ceramics investigated for P&C are characterised by high thermodynamic stability, and thus low solubility in groundwater. • Among others, monazite, pyrochlore, zircon and zirconolite are discussed as potentially appropriate matrices.

As for the low- and intermediate-level waste (LILW), Ref. [19] provided comprehensive comparisons among various fuel cycle schemes, where LILW was divided into two categories: short-lived (LILW-SL) and long-lived (LILW-LL). It was found that the volume of LILW-SL (7-20 m^3/TWh) is much larger than that of LILW-LL (0.3-3.3 m^3/TWh) and HLW (0.1-4.2 m^3/TWh). The volume of LILW-SL is dominated by the operation wastes from the power plants, while that of LILW-LL is increased by the operation of reprocessing plants, which on the contrary largely reduces the volume of HLW. The introduction of P&T of MA seems not to be very influential in

terms of the total volume of LILW in general, though it should be noted that uncertainties in these estimations are very large because no real and experimental data on secondary waste flows exist. That report also provided some information about LILW to be produced by the decommissioning of reactor plants and other fuel cycle plants. It is notable that the volumes of LILW-SL (25-45 m^3/TWh) and LILW-LL (06-1.7 m^3/TWh) from the decommissioning can be comparable to those from the normal operation [19].

3.3.5 Uncertainty

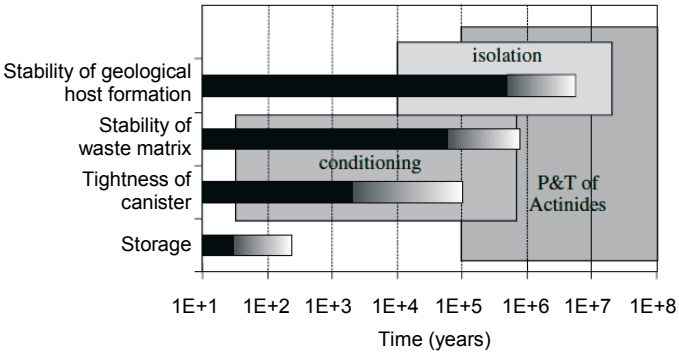
As listed in Table 3.6, some past studies mentioned the role of P&T as a possible measure to mitigate the uncertainty which is inherent to the long-term nature of the radioactivity.

Figure 3.10 illustrates the complementarity between the main components of a geological repository system and P&T. By strongly reducing the long-term radiotoxicity inventory of the disposed waste, P&T is expected to mitigate the potential radiological consequences of severe accidental situations leading to breaches into the repository system.

Table 3.6. Comparison of past studies on uncertainty

Reference	Description
OECD/NEA P&T report (1999) [55]	• The risk of contaminating the geosphere will be decreased if the conditioning of the toxic radionuclides is improved (e.g. by using ceramic matrices or improved glass compositions for the separated MA).
OECD/NEA report on advanced fuel cycles (2006) [19]	• Section 5.4.5 “Complementarities of conditioning, geological disposal and P&T in handling uncertainties”: The various courses of action on waste management are complementary means to achieve redundant confinement of waste, either during the natural loss of predictability of the retention barriers or during their accidental disturbance: partitioning followed by transmutation, storing in surface halls, embedding radionuclides in durable matrices, conditioning in canisters, emplacing canisters in underground galleries within selected host rocks, sealing galleries with engineered barriers, all these actions contribute to devise safe ways of dealing with radioactive waste. Many technical solutions achieve fulfilment of radiation protection criteria over the decay times of the very long-lived radioisotopes.
European RED-IMPACT project (2007-8) [26, 52, 57]	• Removal of MA can reduce the impact in the very unlikely scenario of accidental human intrusion. • Removal of MA has nearly no effect on long-term impact under normal evolution of the repository.
Consideration on the impact of innovative fuel cycles in Germany (2008) [58]	• P&T can minimise estimated resulting doses to population for less probable scenarios: human intrusion, colloid mediated actinide transport anionic actinide complexes increasing solubility and oxidising conditions in the repository environment.

Figure 3.10. Schematic illustration of the coverage of the expected reliability of the main components of a geological repository system by different elements of the waste management policies



Source: OECD/NEA, 2006 [19], Figure 5.21.

3.3.6 Miscellaneous aspects such as proliferation and cost

Table 3.7 lists some miscellaneous comments in the past studies.

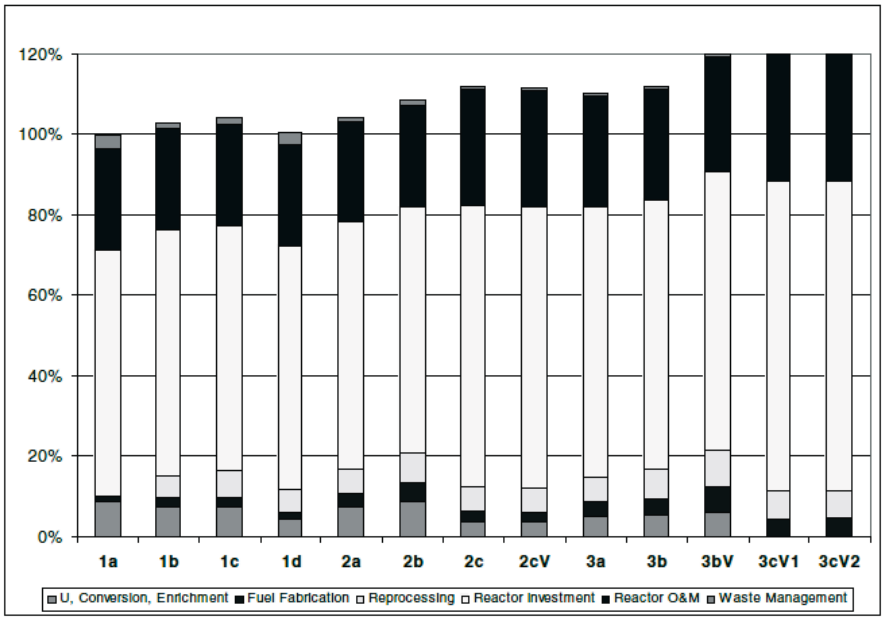
As for the proliferation problem, Ref. [52] said that the introduction of FBR would appear to be best based on an extension of the "TOPS" methodology: technological opportunities to increase the proliferation resistance of global civilian nuclear power systems. In the TOPS report [64], the transmutation technology is being picked up as one of the specific technical options for reactor and fuel cycle systems that have been proposed to improve proliferation resistance. It should be noted that "no single diplomatic, military, economic, institutional or technical initiative alone will be able to fully deal with this proliferation challenge. The best prospect for achieving non-proliferation goals while expanding nuclear power is to engage all appropriate means." [19] It is also notable that Ref. [56] gave warning about recovered Np and Am as the fissionable materials.

As for the cost, detailed discussion was made in Ref. [19]. It was concluded that the introduction of MA transmutation might result in a 10-20% increase of the electricity generation cost in comparison with direct disposal (Figure 3.11), though the uncertainty of the cost estimation is very large in comparison with this difference in the cost (Figure 3.12).

Table 3.7. Miscellaneous comments in past studies to be noted

Reference	Description
OECD/NEA P&T report (1999) [55]	• The term "radioactive inventory" was recommended instead of "potential radiotoxicity". • The impact of P&T is an improvement of the long-term hazard but it requires additional actinide handling facilities and does not eliminate the necessity of geological disposal. • Recycling Pu in LWR-MOX reduces the U needs by 20%, and if MA+Pu 25% is recycled, max. 25% benefit can be expected. • Compared with the reference fuel cycle, the advanced fuel cycle with P&T would moderately increase the collective dose to the workers in the fuel cycle, and particularly to those in fuel and target fabrication. However, appropriate measures must be taken to reinforce shielding, especially against neutrons, throughout all recycling facilities, and this will significantly increase the overall investment cost. • Not only natural uranium but also depleted and reprocessed uranium need to be considered when comparing the long-term radiotoxicity of the different man-made TRU.
IAEA P&T report (2004) [56]	• Non-proliferation aspects of recovered Np and Am should be carefully addressed at the early stage of the development.
OECD/NEA report on advanced fuel cycles (2006) [19]	• Chapter 6 is dedicated to the economic analysis. The conclusion says: – Total electricity generation costs are dominated by reactor investment costs and, therefore, do not vary widely between schemes. – Fuel cycle costs vary by a factor of 2 depending on the scheme and are significantly affected by uncertainties on unit costs for advanced technologies and processes. – The price of natural uranium, which is an important factor in fuel cycle cost for the reference once-through scheme, has a moderate impact on fuel cycle costs for schemes involving reprocessing and recycling. – The portion of waste management cost, including repository, in the total electricity generation cost is so low that uncertainties in unit costs for the waste management steps have no significant impact on those total costs. Monte Carlo simulations show that, taking into account the large uncertainties on unit costs for advanced technologies and processes, innovative schemes eventually may have lower costs than the reference once-through scheme.
European RED-IMPACT project (2007-8) [26, 52, 57]	• An extension of TOPS methodology was used as the indicator of proliferation resistance. The introduction of FBR would appear to be best, though there was large uncertainty in the analysis. • Total cost of electricity was quite similar for all the scenarios studied, despite the fact that some indicators vary considerably.

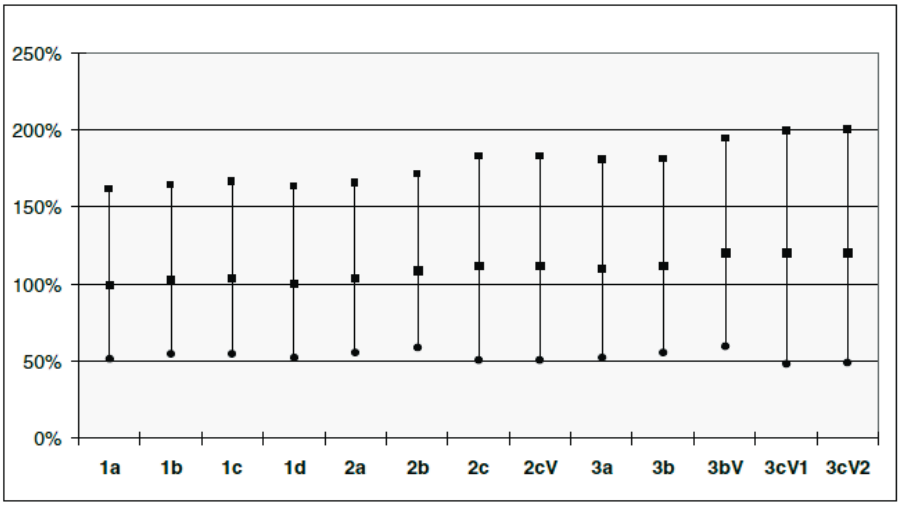
Figure 3.11. Relative total generation costs calculated with nominal values of the unit costs



1a: once-through (direct disposal of UOX-LWR spent fuel, reference scenario); 1b: conventional reprocessing (Pu is recycled once in MOX-LWR); 3a: burning of TRU from UOX-LWR in FR; 3b: double strata (Pu from UOX-LWR is recycled once in MOX-LWR, Pu from MOX-LWR is recycled in FR, and MA from them is recycled in ADS); 3c: all-FR (Generation IV gas-cooled FR recycles TRU).

Source: OECD/NEA, 2006 [19], Figure 6.2.

Figure 3.12. Ranges of relative total generation costs



1a: once-through (direct disposal of UOX-LWR spent fuel, reference scenario); 1b: conventional reprocessing (Pu is recycled once in MOX-LWR); 3a: burning of TRU from UOX-LWR in FR; 3b: double strata (Pu from UOX-LWR is recycled once in MOX-LWR, Pu from MOX-LWR is recycled in FR, and MA from them is recycled in ADS); 3c: all-FR (Generation IV gas-cooled FR recycles TRU).

Source: OECD/NEA, 2006 [19], Figure 3.13.

Chapter 4. Conclusions

4.1 P&T: a concept evolution with time

P&T has been historically associated with the waste minimisation goal, and has been mostly discussed in the last two decades as an option *per se*. The development of P&T can be summarised as follows (see also Figure 4.1).

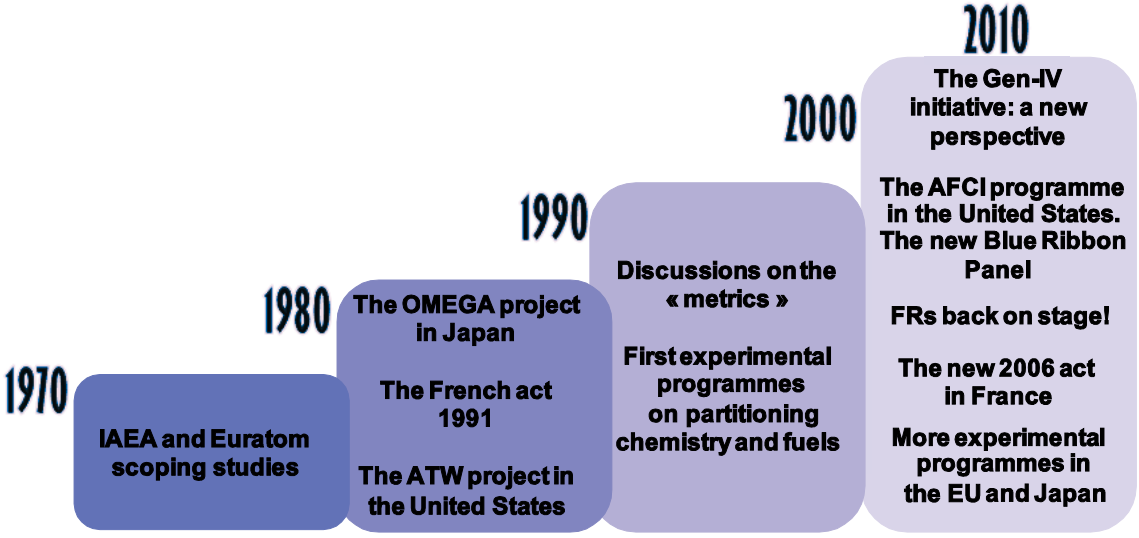
From the late 1970s to late 1980s:

- Early P&T studies are performed, mostly in Europe and in the United States. Early studies were conducted on the impact of P&T on the fuel cycle, P&T motivations and possible P&T "metrics" for cost/benefit evaluations.
- In the mid-1980s both the IAEA and EURATOM issued extensive reports with lukewarm conclusions: the challenge did seem to be formidable, without a clear strategy.

In the late 1980s-early 1990s:

- The "OMEGA" initiative is launched in Japan, motivated by a strong public opinion concern about waste management.
- At the same time in France, the waste management issue is discussed at the political level and a law is passed in 1991, in order to study possible strategies (including P&T) over a 15-year period (1991-2006).
- In the same period the ending of reprocessing of spent fuel in some countries (e.g. Belgium, Germany and Switzerland) and the stagnation in the development of fast reactors strongly hypothecated an industrial deployment of fuel cycles including P&T technologies for the foreseeable future.

Figure 4.1. Evolution with time of P&T focus and studies



Since the early 1990s:

- International involvement of the waste management and geological disposal communities led to discussions on "metrics" and motivations tend to focus on the waste doses and "radiotoxicity". However, the use of the radiotoxicity notion has to be considered in the light of geological disposal. Later studies indicated that the decay heat in the repository was a more significant parameter. The possibilities to include P&T technologies in future fuel cycles are pointed out.
- Very significant resources are deployed in particular in Japan, in France (in particular in the field of partitioning, in order to achieve scientific demonstrations of the feasibility of different separation processes) and in Europe.

From 2000, increasing oil prices and global warming concerns started a trend towards a nuclear renaissance:

- Within a wide consensus, the Generation IV International Forum (GIF) has defined a set of more general goals for future systems in four broad areas: sustainability (more efficient use of the available uranium resources and waste minimisation), enhanced economics, safety and reliability and proliferation resistance and physical protection. The objectives of the Generation IV nuclear systems do include P&T (waste minimisation) and P&T is no longer seen as an option *per se*, but as consistent with sustainability and non-proliferation objectives and mostly associated with future reactor deployment. [67]
- As for implementation, the P&T implementation roadmap in France is closely related to the FR deployment decision. Both homogeneous and heterogeneous TRU recycling in FR is considered as well as the use of other transmutation systems, e.g. ADS. The date of 2012 for decisions on the path forward is indicated by France, consistent with a parliamentary requirement, and this date is also considered at the European level. This makes the P&T implementation horizon to be in the range 2040-2050.

4.2 Objectives and criteria for P&T impact evaluation

The objectives and criteria for P&T were described in Chapter 2. As described in Chapter 3, the impact of the P&T technology on the management of nuclear wastes has been discussed since the outset of R&D activities performed on this technology. Up until about a decade ago (i.e. up to around the year 2000), the impact of P&T technology tended to be emphasised from a viewpoint of the "potential radiological toxicity" or "radiotoxic inventory" which is defined by the amount of the radioactive nuclides in the wastes normalised by the ingestion dose coefficients for individual nuclides.

The reduction of "potential radiological toxicity" was found by the researchers and engineers in the field of waste disposal to be important in the safety analysis for a number of repository disturbed evolution scenarios. Also, a remarkable effect due to the reduction of the radiological toxicity can be expected in the case of the human intrusion scenario.

For the decade following the year 2000, more emphasis was put on realistic considerations based on the fuel cycle analysis (including the impact of P&T operations on workers and routine releases of radionuclides into the environment) and the repository design. In Chapter 3 work was reviewed in the key areas defined in Chapter 2, such as peak dose rate, radiotoxicity, decay heat, waste form, volume and mass and uncertainty assessment.

4.2.1 Peak dose rate

Following international recommendations (ICRP, IAEA) and several national regulations the peak dose rate is the most general indicator used to evaluate the performance of a waste disposal repository.

It has been pointed out that in the normal evolution scenario and most disturbed scenarios the estimated peak doses from currently proposed repositories and with disposal of spent fuel and high-level waste can be significantly lower than the regulatory limits (e.g. 0.3 mSv/yr recommended by ICRP as the upper value for dose constraint for application in a normal exposure situation).

For clay and granite in the normal evolution scenarios, ^{129}I dominates the peak dose if it is disposed of into the repository in the cases of either direct disposal or reprocessing. When most of the ^{129}I is removed from HLW in the reprocessing case, the peak dose is dominated by ^{79}Se, ^{36}Cl and ^{135}Cs. The effect of MA transmutation on the peak dose rate is therefore limited in the normal evolution scenarios. In the case of Yucca Mountain, peak dose rate is dominated by ^{242}Pu, ^{226}Ra and ^{237}Np with important contributions from ^{129}I and ^{99}Tc. In the case of salt domes, no release of radionuclides is expected in normal evolution.

On the other hand, MA can play an important role in the unlikely case of human intrusion scenarios in which an intruder comes in contact with the disposed waste, especially after 1 000 years of disposal. MA transmutation will reduce the dose by two orders of magnitude. It is therefore reasonable to regard MA transmutation as a measure to reduce the future uncertainty related to some unlikely disturbances within the repository. From this point of view, the cost and the benefit of MA transmutation should be discussed further, though it will be difficult to estimate the future uncertainty quantitatively. In order to pursue the analysis, doses to workers and population occurring during the operation of the different facilities should be estimated in case of P&T deployment and compared to doses without P&T. Potential long-term doses to the public due to the repository with or without P&T should be compared.

4.2.2 Radiotoxicity

Radiotoxicity has been used as the index to demonstrate the effect of MA transmutation. All the previous studies assumed 99-99.9% recovery of TRU, and eventually about two orders of magnitude reduction of radiotoxic inventory were foreseen after several hundred years and later. This index is also used to emphasise the effect to shorten the time period necessitated to decay below a reference level. As a reference level, the radiotoxicity of the initial natural uranium in equilibrium with its decay products is used. In general, TRU transmutation shortens the time period needed to reduce the radiotoxicity to the reference level to several hundred years, though more than 100 000 years will be necessary in the case of direct disposal. However, one should not draw the conclusion that contact with the remaining waste would no longer be hazardous.

4.2.3 Decay heat

In general, reduction of gallery length by factor of 3-6 depending on the considered cooling time can be foreseen by the TRU transmutation in comparison with direct disposal. Additional gain can also be foreseen by separating Cs and Sr and storing them for 100-300 years or by lengthy intermediate storage of vitrified high-level waste without separation.

4.2.4 Waste form, volume and mass

By introducing P&T in the fuel cycle, the mass of actinides can be significantly reduced in the HLW, and it may be possible to reduce HLW volume and mass while long-lived low- and intermediate-waste must increase. Conventional reprocessing adopts a vitrified (or glass) waste form for the HLW. The introduction of P&T could increase the options for optimised waste forms. For example, as concerns the vitrified waste, the density of the waste elements in the glass matrix might be increased because of the reduced heat source and removal of elements that are forming particles within a glass matrix such as molybdenum and platinum group metals.

4.2.5 Uncertainty

The management of uncertainty is an essential feature of the safety case for a geological repository [66]. The role of P&T can be seen as a measure to mitigate the importance of the uncertainty which is inherent to the very long-term nature of the radioactivity. This is achieved essentially by the reduction of the source term.

4.3 Geological disposal and P&T

Remarkable progress in the last two decades has been made in the field of geological disposal. Some countries have reached important milestones and geological disposal (of spent fuel) is expected to start in 2020 in Finland and in 2022 in Sweden; licensing of the geological repositories in both countries is now entering the final phases. In France disposal of ILW and vitrified HLW is expected to start around 2025, according to the roadmap defined by a parliamentary act in 2006.

In this context, P&T hardly offers any significant extra advantage towards the solution of the management of existing wastes, e.g. vitrified wastes. Incentive to implement any significant P&T technology can only be seen in case of the existing spent fuel and in particular in the context of future innovative fuel cycles. However, it should be evaluated in view of the demonstrations still to be made and the potential additional cost and risk (separations, remote fuel fabrication, fuel handling and potential impact on reactor availability, etc.) and short-term doses to workers and releases of volatile radionuclides into the environment.

It has been recalled that in spite of the widespread agreement that such disposal would be acceptable, siting a repository for commercial spent fuel and obtaining a license to construct and operate the repository has proven to be much more difficult in practice. However, lessons learnt from delays or difficulties of repository siting in various countries have led to the introduction of a stepwise and transparent decision-making process involving a strong participation of all stakeholders [67, 68].

The concepts of reversibility of impact of advanced fuel cycles and retrievability of the spent fuel or waste from geological disposal are currently being discussed and defined within the national programmes of several countries and there are, as yet, varying views on the desirability and the methods and degree of their implementation [69]. In many countries that host nuclear power stations, the notions of reversibility and retrievability have come to occupy an increasingly central place in debates and decision making concerning the management of high- and medium-level radioactive waste. P&T apparently does not have a significant impact on the flexibility captured by the notions of reversibility and retrievability.

It is also worth pointing out that major nuclear countries still show a very cautious attitude about P&T and, to varying degrees, favour further study, technology and scientific demonstrations of the different aspects of P&T. This provides a much-needed flexibility as regards possible regulatory evolutions and the handling of uncertainties. Examples are given below.

4.3.1 France

The summary reports on the findings of the R&D performed in the period 1991-2006 as requested by the 1991 French Act were published (2005-2006) and peer reviewed in the framework of an OECD/NEA report [70]. The successive Act of 28 June 2006 foresees the implementation of a national plan for managing nuclear materials and radioactive waste and it defines a stepwise programme for long-lived waste (high- and intermediate-level activity) taking the complementarities of the various approaches into account.

In practice, the 2006 Planning Act requires that the research and studies on this waste shall be pursued according to the three following complementary objectives:

- *Partitioning and transmutation of long-lived radioactive elements*. The corresponding studies and research shall be conducted in relation with those performed on the new generations of nuclear reactors and those on accelerator-driven reactors devoted to waste transmutation, so that an assessment can be made in 2012 of the industrial prospects of these reactor types and a prototype installation set in operation before 31 December 2020.
- *Reversible disposal in deep geological formations*. The corresponding studies and research shall be performed in order to choose a site and design a disposal facility so that, on the basis of the results of the studies undertaken, an application for its authorisation can be filed in 2015 and, subject to said authorisation, the facility can be set in operation in 2025.
- *Storage*. The corresponding studies and research shall be performed in order to create new storage installations, at the latest by 2010, or modify existing ones to meet French requirements, in particular in terms of capacity and lifespan.

The act includes guarantees for long-term funding of radioactive waste management

In summary, in the framework of the French law for waste management, scenario studies are still carried out (very often with an active participation of industry) to compare different options of separation and transmutation of plutonium and minor actinides in the French fleet of reactors. The goal of these studies is to evaluate the consequences of these options, which involve the deployment of fast reactors, on indicators such as the plutonium and actinide inventories and waste production (HLW and ILW). Based on these results, the economic and technical impacts will be assessed, more specifically the impact on waste disposal.

The main scenarios that are being considered are:

- plutonium recycling in SFR in the fissile part of the core, minor actinides sent for disposal;
- plutonium recycling in SFR core, minor actinide recycling in SFR radial blankets (heterogeneous mode);
- plutonium recycling in SFR core, americium recycling in SFR radial blankets (heterogeneous mode), other minor actinides such as neptunium and curium sent for disposal;
- plutonium recycling in SFR core, minor actinide recycling in accelerator-driven system (ADS).

Technical and economic results are expected in 2012 in order to provide an industrial prospect assessment of minor actinide transmutation.

4.3.2 Japan

The Atomic Energy Commission (AEC) of Japan launched the Technical Subcommittee on P&T Technology under the Advisory Committee on R&D in August 2008, and conducted the discussion about the current state of the art concerning P&T technology in Japan and the future R&D programme. The subcommittee issued its final report on 28 April 2009, and the AEC endorsed it. The report covers the impact of P&T technology, the state of the art of the technology, evaluation on the progress of the R&D and recommendations for future R&D.

As for the impact of the P&T technology, the report mentioned addressed three main topics: *i)* reduction of the radioactive potential hazard; *ii)* reduction of the requirement for the geological repository; *iii)* enhancement of a degree of freedom for rational design of nuclear waste disposal systems. Conversely, it was emphasised that this technology is still far from commercial deployment and it is still desirable to accumulate basic data for the assessment of its feasibility. According to this report, more precise evaluation on the benefit and the cost of P&T will be studied together with basic data preparation for innovative waste management incorporating P&T technology.

4.3.3 United States

The United States government has been pursuing the development of a geological repository for spent fuel and high-level waste for almost 30 years, leading to the selection of the Yucca Mountain site in 2002 after years of characterisation and testing. In 2009, the United States administration proposed abandoning the Yucca Mountain project and revisiting the question of how to deal with spent fuel and high-level radioactive wastes. A "Blue Ribbon Commission" was appointed by the administration to revisit the question of geological disposal that would also "include an evaluation of advanced fuel cycle technologies that would optimise energy recovery, resource utilisation and the minimisation of materials", with the objective of recommending one or more strategies to resolve the waste management issue [71].

The United States Department of Energy is continuing with research on advanced fuel cycles, including partitioning and transmutation, in the Fuel Cycle Research and Development Programme and a programme on advanced reactor concepts. These programmes evolved from the previous Advanced Fuel Cycle Initiative and ongoing activities in the Generation IV International Forum. At the same time, the license application for a repository at Yucca Mountain is being reviewed and evaluated by the US Nuclear Regulatory Commission at the direction of the US Congress.

The future direction in the United States appears to be uncertain, but geological disposal is likely to remain the preferred solution for disposal of spent fuel and high-level radioactive wastes. P&T is being considered for its impact on geological disposal. A specific decision on the future direction for high-level waste management in the United States has not yet been made.

4.4 Concluding remarks

Many recent studies have demonstrated that the impact of P&T on geological disposal concepts could be significant even if not overwhelmingly high. In fact, by reducing waste heat production a more efficient utilisation of repository mines is likely. In practice, the reduction of the thermal output of the high-level wastes by a factor of ~3 can reduce the needed repository gallery length by a factor of ~3 and the repository footprint up to a factor of 9. Moreover, even if radionuclide release from the waste to the environment and related potential doses to the population are not significantly reduced by P&T since mostly dominated by some long-lived fission products for which no practical transmutation strategy is applicable, it is important to point out that a clear reduction of the actinide inventory in the highly-active waste (HAW) reduces risks arising from less probable evolutions of a repository, i.e. increase of actinide mobility in certain geochemical situations and radiological impact by human intrusion.

Since the inventory of hazardous radioactive materials present in long-lived radioactive waste necessitates its disposal in a geological repository, one significant effect of P&T is that the inventory of the emplaced materials is much lower on an energy-generated basis for the actinide elements, which can have the effect of making the uncertainty about repository performance less important. One could possibly use this in making the case for a repository, although if one uses P&T as an opportunity to increase the amount that can be placed in the repository, i.e. the wastes from more spent fuel is disposed in the repository, the potential beneficial effect is mitigated. In such a case, the actinide inventory would be much lower, but the fission product inventory would be higher.

As for uncertainty, P&T can reduce the importance of uncertainties both in normal evolution and in particular those related to hypothetical disruptive scenario that can bring man in direct contact with the disposed waste, since these scenarios seem to be affected by the hazard (radiotoxicity) and not so much by the geology. P&T of the actinides does reduce the hazard of the emplaced materials.

In summary, while P&T will never replace the need for a deep geological repository, it can be argued that it has the potential to significantly improve the public perception of the ability to effectively manage radioactive wastes by largely reducing the TRU waste masses to be stored (and, consequently, the hazard from the wastes per unit energy generation) and of the ability to effectively manage a geological repository. These are important issues to ensure the future sustainability of nuclear power.

References

[1] NEA (Nuclear Energy Agency), *Nuclear Energy Outlook 2008*, OECD/NEA, Paris (2008).

[2] US Department of Energy, *Technical Roadmap for Generation IV Nuclear Energy Systems*, Nuclear Energy Research Advisory Committee, Washington, Report GIF-002-00 (2003).

[3] IAEA (International Atomic Energy Agency), *The Principles of Radioactive Waste Management*, IAEA, Vienna, Safety Series No. 111-F (1995).

[4] IAEA, *Classification of Radioactive Waste: A Safety Guide*, IAEA, Vienna, Safety Series No. 111-G-1.1 (1994).

[5] IAEA, *Classification of Radioactive Waste*, IAEA, Vienna, General Safety Guide No. GSG-1 (2009).

[6] NEA, *Moving Forward with Geological Disposal: A Collective Statement by the NEA Radioactive Waste Management Committee*, NEA/RWM(2008)5, OECD/NEA, Paris (2008).

[7] Becker, D.A. *et al.*, *Testing of Safety and Performance Indicators (SPIN)*, EC, Luxembourg, Report EUR 19965 EN (2003).

[8] SKB, *Long-term Safety for the Final Repository for Spent Nuclear Fuel at Forsmark: Main Report of the SR-site Project*, SKB, Stockholm, Report TR-11-01 (2011).

[9] IAEA, *Geological Disposal of Radioactive Waste: Safety Requirements*, IAEA, Vienna, Safety Standards No. WS-R-4 (2006).

[10] ICRP (International Commission on Radiological Protection), *1990 Recommendations of the International Commission on Radiological Protection*, ICRP Publication 60, Annals of the ICRP, Vol. 21 (1-3) (1991).

[11] ICRP, *Radiation Protection Recommendations as Applied to the Disposal of Long-lived Solid Radioactive Waste*, ICRP Publication 81, Annals of the ICRP, Vol. 28 (4) (1998).

[12] NEA, *Regulating the Long-term Safety of Geological Disposal: Towards a Common Understanding of the Main Objectives and Bases of Safety Criteria*, OECD/NEA, Paris (2007).

[13] Clark, R.L. *et al.*, "Amendments to the US Environmental Protection Agency's Public Health and Environmental Radiation Protection Standards for Yucca Mountain, Nevada (40 CFR Part 197)", *Proceedings ICEM 2009*, Liverpool, 11-15 October 2009, paper 16156 (2009).

[14] NEA, *Post-closure Safety Case for Geological Repositories; Nature and Purpose,* NEA/RWM/IGSC(2004)7/REV1, OECD/NEA, Paris (2004).

[15] National Academy of Sciences, *Technical Bases for Yucca Mountain Standards*, National Academy Press, Washington (1995).

[16] Beuth, T., "Human Intrusion", Chapter 9 in *Handbook of the State-of-the-art of Safety Assessments Methods*, Deliverable D1.1 of the EC PAMINA project (2011).

[17] NEA, *Towards Transparent, Proportionate and Deliverable Regulation for Geological Disposal*, Tokyo, Japan, 20-22 January 2009, OECD/NEA, Paris (2010).

[18] IAEA, *Safety Indicators for the Safety Assessment of Radioactive Waste Disposal: Sixth Report of the Working Group on Principles and Criteria for Radioactive Waste Disposal*, IAEA, Vienna, Report IAEA-TECDOC-1372 (2003).

[19] NEA, *Advanced Nuclear Fuel Cycles and Radioactive Waste Management*, OECD/NEA, Paris (2006).

[20] Rechard, R.P., "Historical Relationship Between Performance Assessment for Radioactive Waste Disposal and Other Types of Risk Assessment", *Risk Analysis*, Vol. 19, No. 5, p. 763 (1999).

[21] ICRP, *Age-dependent Doses to Members of the Public from Intake of Radionuclides, Part 5, Compilation of Ingestion and Inhalation Coefficients*, Publication 72, Annals of the ICRP, Vol. 26/1 (1996).

[22] Kim, J.I., "Significance of Actinide Chemistry for the Long-term Safety of Waste Disposal", *Nucl. Eng. Technol.*, 38, pp. 459-482 (2006).

[23] Fanghänel, Th., V. Neck, "Aquatic Chemistry and Solubility Phenomena of Actinide Oxides/hydroxides", *Pure Appl. Chem.*, Vol. 74, No. 10, pp. 1895-1907 (2002).

[24] Kim, J.I., "Is the Thermodynamic Approach Appropriate to Describe Natural Dynamic Systems?", *Nucl. Eng. and Design*, 202, pp. 143-155 (2000).

[25] Grambow, B., "Mobile Fission and Activation Products in Nuclear Waste Disposal", *J. Cont. Hydrol.*, 102, pp. 180-186 (2008).

[26] Marivoet, J., E. Weetjens, "Impact of Advanced Fuel Cycles on Geological Disposal in a Clay Formation", *Nuclear Technology*, 163, pp. 74-84 (2008).

[27] Rasilainen, K. *et al.*, "Release of Uranium from Rock Matrix – a Record of Glacial Meltwater Intrusions", *J. Cont. Hydrol.*, 61, pp. 235-246 (2003).

[28] Ewing, R.C., "The Nuclear Fuel Cycle: A Role for Mineralogy and Geochemistry", *Elements*, 2, pp. 331-335, December (2006).

[29] Kim, J.I., "Actinide Colloid Generation in Groundwater", *Radiochimica Acta*, 52/53, pp. 71-81 (1991).

[30] Kersting, A.B. *et al.*, "Migration of Plutonium in Groundwater at the Nevada Test Site", *Nature*, 397, pp. 56-59 (1999).

[31] Geckeis, H. *et al.*, "Results of the Colloid and Radionuclide Retention Experiment (CRR) at the Grimsel Test Site (GTS), Switzerland – Impact of Reaction Kinetics and Speciation on Radionuclide Migration", *Radiochim. Acta*, 92, pp. 765-774 (2004).

[32] Sandia National Laboratory, *Total System Performance Assessment Model/analysis for the License Application*, MDL-WIS-PA-000005 Rev 00, AD 01, United States Department of Energy Office of Civilian Radioactive Waste Management, Las Vegas, NV (2008).

[33] Swift, P. *et al.*, "Broader Perspectives on the Yucca Mountain Performance Assessment", *Proc. of the International High Level Radioactive Waste Management Meeting*, Las Vegas, NV, 7-11 September (2008), p. 479.

[34] Marivoet, J. *et al.*, "Functions of Argillaceous Media in Deep Geological Disposal and Their Handling in a Safety Case", *Proc. Clay Club Workshop Stability and Buffering Capacity of the Geosphere for Long-term Isolation of Radioactive Waste*, Braunschweig, December 2003, OECD/NEA, Paris (2004).

[35] ONDRAF/NIRAS (Belgian Agency for Radioactive Waste and Enriched Fissile Materials), *SAFIR 2: Safety Assessment and Feasibility Interim Report 2*, ONDRAF/NIRAS Brussels, Report NIROND 2001-06 E (2001).

[36] ANDRA (French Agency for Radioactive Waste Management), *Dossier 2005 Argile, Synthesis Report*, ANDRA, Paris (2005).

[37] NAGRA (National Co-operative for the Disposal of Radioactive Waste), *Project Opalinius Clay: Safety Report: Demonstration of Disposal Feasibility for Spent Fuel, Vitrified High-level Waste and Long-lived Intermediate-level Waste*, NAGRA, Wettingen, Technical Report 02-05 (2002).

[38] NEA, *Stability and Buffering Capacity of the Geosphere for Long-term Isolation of Radioactive Waste: Application to Crystalline Rock,* Workshop Proceedings, Manchester, UK, November 2007, OECD/NEA, Paris (2009).

[39] Posiva Oy (Finnish Nuclear Waste Management Company), *Interim Summary Report of the Safety Case 2009*, Posiva Oy, Olkiluoto, Report 2010-02 (2001).

[40] SKB (Swedish Nuclear Fuel and Waste Management Company), *Long-term Safety for KBS-3 Repositories at Forsmark and Laxemar – A First Evaluation*, Main Report of the SR-Can Project, SKB, Stockholm, Report TR-06-09 (2006).

[41] Schulze, O., T. Popp, H. Kern, "Development of Damage and Permeability in Deforming Rock Salt", *Engineering Geology*, 61, pp. 163-180 (2001).

[42] Elliger, C., *Untersuchungen zum Permeationsverhalten von Salzlaugen in Steinsalz bei der Endlagerung wärmeentwickelnder nuklearer Abfälle*, Fachbereich Maschinenbau, Technischen Universität, Darmstadt, Germany (2005).

[43] Stormont, J.C., C.L. Howard, J.J.K. Daemen, "*In Situ* Measurements of Rock Salt Permeability Changes Due to Nearby Excavation", *Sandia Report*, Sandia National Laboratories, Albuquerque, NM, USA (1991).

[44] Alekseenko, N.N., P.V. Volobuev, P.E. Suetin, "Diffusion and Solubility of Hydrogen and Deuterium in NaCl Crystals", *Soviet Physics – Solid State*, 15 (5), pp. 589-590 (1973).

[45] Alekseenko, N.N., P.V. Volobuev, P.E. Suetin, "Diffusion and Solubility of Helium in NaCl Crystals", *Soviet Physics – Solid State*, 14 (8), pp. 2073-2975 (1973).

[46] Friedrich, A.J., *Selbstdiffusion von Wasser im intergranularen Raum von Steinsalz*, Institut für Umweltphysik, Ruprecht-Karls-Universität, Heidelberg, p. 79 (2000).

[47] Herrmann, A.G., L.E.V. Bostel, "The Composition and Origin of Fluid Inclusions in Zechstein Evaporites of Germany", *Neues Jahrbuch fuer Mineralogie*, Monatshefte (6), pp. 263-269 (1991).

[48] Müller, E., "The Migration of Gas-filled Brine Inclusions in Rock Salt Under a Temperature Gradient", *Cryst. Res. Technol.*, 20 (4), pp. 521-526 (1985).

[49] Juhlin, C., H. Sandstedt, *Storage of Nuclear Waste in Very Deep Boreholes: Feasibility Study and Assessment of Economic Potential,* SKB Technical Report 89-39, SKB, Stockholm, Sweden (1989).

[50] Chapman, N., F. Gibb,"A Truly Final Waste Management Solution: Is Very Deep Borehole Disposal a Realistic Option for High-level Waste or Fissile Materials?", *Radwaste Solutions*, pp. 26-37 (2003).

[51] IAEA, *Scientific and Technical Basis for the Geological Disposal of Radioactive Wastes,* Report TRS-413, IAEA, Vienna (2003).

[52] Forschungszentrum Julich GmbH, *RED-IMPACT – Impact of Partitioning, Transmutation and Waste Reduction Technologies on the Final Nuclear Waste Disposal*, European Commission Contract No. FI6W-CT-2004-002408, September (2007).

[53] Wigeland, R.A. *et al.*, "Separations and Transmutation Criteria to Improve Utilization of a Geological Repository", *Nuclear Technology*, Vol. 154, No. 1, April (2006).

[54] Nishihara, K. *et al.*, "Impact of Partitioning and Transmutation on LWR High-level Waste Disposal", *J. Nucl. Sci. Technol.*, 45 (1), 84 (2008).

[55] NEA, *Actinide and Fission Product Partitioning and Transmutation: Status and Assessment Report*, OECD/NEA, Paris (1999).

[56] IAEA, *Implications of Partitioning and Transmutation in Radioactive Waste Management*, Technical Report Series No. 435, IAEA, Vienna (2004).

[57] Marivoet, J. *et al.*, "Impact of Advanced Fuel Cycle Scenarios on Geological Disposal", *7th European Commission Conference on the Management and Disposal of Radioactive Waste (Euradwaste '08)*, Luxembourg, 20-22 October 2008, Report EUR 24040 (2009).

[58] Geckeis, H. *et al.*, "Impact of Innovative Nuclear Fuel Cycles on Geological Disposal", *Annual Meeting of Nuclear Technology* (2008).

[59] Wigeland, R.A. *et al.*, "Impact on Geological Repository Usage from Limited Actinide Recycle in Pressurized Light Water Reactors", *J. Nucl. Sci. Technol.*, 44 (3), 1 (2007).

[60] Forsberg, C.W., "Rethinking High-level Waste Disposal: Separate Disposal of High-heat Radionuclides (^{90}Sr and ^{137}Cs)", *Nucl. Technol.*, 131, 252 (2000).

[61] Oigawa, H. *et al.*, "Partitioning and Transmutation Technology in Japan and its Benefit on High-level Waste Management", *Int. Conf. on Advanced Nuclear Fuel Cycles and Systems (GLOBAL 2007)*, Boise, Idaho, 9-13 September (2007).

[62] Oigawa, H. *et al.*, "Concept of Waste Management and Geological Disposal Incorporating Partitioning and Transmutation Technology", *Actinide and Fission Product Partitioning and Transmutation, Proc. 10th Information Exchange Meeting*, Mito, Japan, 6-10 October 2008, OECD/NEA, Paris (2010).

[63] JAEA (Japan Atomic Energy Agency), *Present Status and Future Plan of Research and Development in JAEA on Partitioning and Transmutation Technology for Long-lived Nuclides*, JAEA-Review 2008-074 (2009) [in Japanese].

[64] TOPS Task Force, *Technological Opportunities to Increase the Proliferation Resistance of Global Civilian Nuclear Power Systems (TOPS)*, TOPS Task Force of the Nuclear Energy Research Advisory Committee, US Department of Energy; available on the Internet at: *www.nuclear.energy.gov/pdfFiles/TOPS-Final.pdf* (2000).

[65] Nuclear Energy Study Group, *Nuclear Power and Proliferation Resistance: Securing Benefits, Limiting Risk*, Nuclear Energy Study Group of the American Physical Society Panel on Public Affairs, available on the Internet at: *www.aps.org/policy/reports/popa-reports/proliferation-resistance/upload/proliferation.pdf* (2005).

[66] NEA, *Management of Uncertainty in Safety Cases and the Role of Risk*, Workshop Proceedings, Stockholm, Sweden, 2-4 February 2004, OECD/NEA, Paris (2005).

[67] NEA, *Stepwise Approach to Decision Making for Long-term Radioactive Waste Management: Experience, Issues and Guiding Principles*, OECD/NEA, Paris (2004).

[68] Dupuis, M.C., "Radioactive Waste Management: Where Do We Stand?", *7th European Commission Conference on the Management and Disposal of Radioactive Waste (Euradwaste '08)*, Luxembourg, 20-22 October 2008, Report EUR 24040 (2009), pp. 31-40.

[69] NEA, *Reversibility and Retrievability in Geological Disposal of Radioactive Waste: Reflections at the International Level*, OECD/NEA, Paris (2001).

[70] NEA, *French R&D on the Partitioning and Transmutation of Long-lived Radionuclides: An International Peer Review of the 2005 CEA Report*, OECD/NEA, Paris, NEA No. 6210 (2006), available on the Internet at: *http://home.nea.fr/ndd/reports/2006/nea6210-french-research.pdf*.

[71] Barack Obama, *Blue Ribbon Commission on America's Nuclear Future*, Memorandum for the Secretary of Energy, The White House, Office of the Press Secretary, 29 January 2010, available on the Internet at: *www.energy.gov/news/documents/2010nuclearfuture_memo.pdf*.

Members of the Task Force

Belgium

MARIVOET, Jan (SCK•CEN)

France

MILOT, Jean-François (CEA)

SALVATORES, Massimo (Chair) (CEA/INL)

SATURNIN, Anne (CEA)

ZAETTA, Alain (CEA)

Germany

GECKEIS, Horst (KIT)

GOMPPER, Klaus (KIT)

Japan

OIGAWA, Hiroyuki (JAEA)

United States

TAIWO, Temitope A. (ANL)

WIGELAND, Roald (INL)

International organisations

CHOI, Yong-Joon (Secretary) (NEA)

NEA PUBLICATIONS AND INFORMATION

Printed material

The NEA produces a large selection of printed material, part of which is on sale, and part of which is distributed free of charge. The full catalogue of publications is available online at www.oecd-nea.org/pub.

Internet and electronic products

In addition to basic information on the Agency and its work programme, the **NEA website** offers free downloads of hundreds of technical and policy-oriented reports.

An **NEA monthly electronic bulletin** is distributed free of charge to subscribers, providing updates of new results, events and publications. Sign up at www.oecd-nea.org/bulletin/.

You can also visit us on **Facebook** at www.facebook.com/OECDNuclearEnergyAgency.

OECD PUBLISHING, 2 rue André-Pascal, 75775 PARIS CEDEX 16
ISBN 978-92-64-99165-1

Potential Benefits and Impacts of Advanced Nuclear Fuel Cycles with Actinide Partitioning and Transmutation

This report provides a comparative analysis of different studies performed to assess the potential impact of partitioning and transmutation (P&T) on different types of geological repositories for radioactive waste in various licensing and regulatory environments. Criteria, metrics and impact measures have been analysed and compared with the goal of providing an objective comparison of the state of the art to help shape decisions on options for future advanced fuel cycles.

P&T allows a reduction of the inventory of the emplaced materials which can have a significant impact on the repository. Such a reduction can also make the uncertainty about repository performance less important both during normal evolution and in the case of disruptive scenarios. While P&T will never replace the need for waste repositories, it has the potential to significantly improve public perception regarding the ability to effectively manage radioactive waste by largely reducing the transuranic (TRU) waste masses to be stored and, consequently, to improve public acceptance of the geological repositories. Both issues are important for the future sustainability of nuclear power.

OECD Nuclear Energy Agency
12, boulevard des Îles
92130 Issy-les-Moulineaux, France
Tel.: +33 (0)1 45 24 10 15
nea@oecd-nea.org www.oecd-nea.org

ISBN 978-92-64-99165-1