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Nuclear spent fuel management scenarios

Status and assessment report

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Summary

The strategy for management of spent nuclear fuel from the Swedish nuclear power programme is interim storage for cooling and decay for about 30 years followed by direct disposal of the fuel in a geologic repository. In various contexts it is of interest to compare this strategy with other strategies that might be available in the future as a result of ongoing research and development. In particular partitioning and transmutation is one such strategy that is subject to considerable R and D-efforts within the European Union and in other countries with large nuclear programmes. To facilitate such comparisons for the Swedish situation, with a planned phase out of the nuclear power programme, SKB has asked the team at KTH to describe and explore some scenarios that might be applied to the Swedish programme. The results of this study are presented in this report.

The following scenarios were studied by the help of a specially developed computer programme:

- Phase out to 2025 with direct disposal.
- Burning plutonium and minor actinides as MOX in BWR.
- Burning plutonium and minor actinides as MOX in PWR.
- Burning plutonium and minor actinides in ADS[1].
- Combined LWR-MOX plus ADS.

For the different scenarios nuclide inventories, waste amounts, costs, additional electricity production etc have been assessed.

As a general conclusion it was found that BWR is more efficient for burning plutonium in MOX fuel than PWR. The difference is approximately 10%. Furthermore the BWR produces about 10% less americium inventory.

An ADS reactor park can theoretically in an ideal case burn (transmute) 99% of the transuranium isotopes. The duration of such a scenario heavily depends on the interim time needed for cooling the spent fuel before reprocessing. Assuming 10 years for cooling of nuclear fuel from ADS, the duration will be at least 200 years under optimistic technical assumptions. The development and use of advanced pyro-processing with an interim cooling time of only 2 years may decrease the duration for transmuting 99% of the transuranium to about 50 years. ADS reactors have turned out to be a necessary component to decrease the americium inventory because neither BWR nor PWR alone can provide prevalence of americium destruction over its production during the operation time. Nevertheless, the economic advisability of these scenarios calls for further investigation.

A scenario using in total six ADS reactors during a 100 year period from 2035 to 2135 is analysed in some detail. It would reduce the TRU-inventory[2] projected from the current LWRs from about 100 tonnes in 2025 to about 6 tonnes in 2135. The ADS reactors would produce on the average 840 MWe giving in total some 740 TWhe of electricity during the 100 year period. The costs for the system are assessed to about 95 GSEK for investments and about 62 GSEK for fuel cycle and waste costs. All these numbers depend on some optimistic assumptions concerning ongoing technical development. They are thus subject to large uncertainties.

In addition, a combination of LWR-MOX plus ADS has been found somewhat more efficient in reducing the transuranium inventory than ADS alone.

[1] ADS = accelerator driven system. [2] TRU = transuranium elements.

Sammanfattning

Strategin för hantering av använt kärnbränsle från de svenska kärnkraftverken är mellanlagring under ca. 30 år för avklingning och kylning följt av direkt slutförvaring av bränslet (utan upparbetning) i ett geologiskt slutförvar. I olika sammanhang är det av intresse att jämföra denna strategi med andra möjligheter som kan bli tillgängliga i framtiden som resultat av pågående forskning och utveckling. Separation och transmutation är en sådan strategi som tilldrar sig betydande FoU-insatser inom EU och i andra länder med omfattande kärnenergiprogram. För att underlätta sådana jämförelse för den speciella svenska situationen med en planerad kärnkraftsavveckling har SKB bett den grupp vid KTH som sysslar med transmutationsforskning att undersöka några scenarier som skulle kunna tillämpas i Sverige. Resultaten från denna studie presenteras i denna rapport.

Följande scenarier studerades med hjälp av ett speciellt för ändamålet utvecklat datorprogram:

- Avveckling till 2025 med direkt slutförvaring.
- Förbränning av plutonium och andra transuraner som MOX i BWR
- Förbränning av plutonium och andra transuraner som MOX i PWR
- Förbränning av plutonium och andra transuraner i ADS
- Kombination av LWR-MOX plus ADS

För de olika scenarierna beräknades nuklidinventarier som funktion av tiden, avfallsmängder, kostnader, tillkommande elproduktion m m.

Som en allmän slutsats observerades att BWR är mer effektiv att bränna plutonium än PWR. Skillnaden är ca. 10 %. Vidare byggs det upp ca. 10 % mindre mängd americium i BWR.

Ett system med ADS kan teoretiskt i ett idealt fall bränna upp (transmutera) 99 % av transuranisotoperna. Tidsåtgången för ett sådant scenario beror starkt på den erforderliga tiden för avsvalning av använt ADS-bränsle före upparbetning. Antar man 10 års avsvalningstid krävs det åtminstone 200 år med optimistiska antaganden om teknisk utveckling. Utveckling och användning av avancerad pyro-kemisk upparbetning med en kyltid på endast 2 år kan kanske minska tiden för transmutation av 99 % av transuranerna till ca. 50 år. ADS reaktorer är nödvändiga för att transmutera americium. Varken BWR eller PWR kan förbränna mer americium än vad som bildas under driften. I alla händelser kräver det ekonomiska incitamentet för dessa scenarier ytterligare undersökningar.

Ett scenario med totalt sex ADS reaktorer i drift under en 100 års period från 2035 till 2135 har studerats mer i detalj. Det skulle reducera det beräknade TRU -inventariet från nuvarande LWR från ca. 100 ton år 2025 till ca. 6 ton år 2135. ADS reaktorerna skulle producera i medeltal ca. 840 MWe och ge totalt ca. 740 TWh elektricitet under 100 års perioden. Kostnaden för detta system uppskattas till ca. 95 miljarder kronor i investeringar och ca. 62 miljarder kronor i bränslecykel- och avfallskostnader. Alla dessa värden beror på flera optimistiska antaganden om den pågående tekniska utvecklingen. De har därför stora osäkerheter.

Vidare har en kombination av LWR-MOX plus ADS visat sig något effektivare för att reducera TRU-inventariet än enbart ADS.

Contents

1 Introduction

Nowadays, the nuclear power provides about 17% of the world's electricity demand. The stockpiles of spent fuel and highly active waste materials steadily grow. The nuclear waste poses a potential high risk for the environment. There is a general consensus within OECD countries that geological disposal is an adequate solution to protect humans and their environment in the far future. However, difficulties encountered in siting, constructing and licensing of repositories as well as public opinion have caused delays in the development of these facilities. A promising alternative is partitioning and transmutation of radioactive and long-lived components from the highly radioactive waste.

Sweden's electricity consumption has been rising and it has one of the world's highest individual levels of consumption: about 18 MWh/head per year. About half of the domestic production is nuclear, and up to half hydro, depending on the weather. However, after the 1980 referendum the Swedish parliament decided to embargo further expansion of nuclear power and aim for decommissioning the 12 plants by 2010 if new energy sources were available realistically to replace them. In 1997 the target year 2010 was abolished by a new decision to start the phase out by closing the Barsebäck plant but recognizing the replacement production sources would not be available to 2010. No new time limit was established for the remaining units. The power companies are now planning for a scenario of at least 40 years of operation for these units and also for power increases in most of the units.

A **nuclear power phase-out** is the discontinuation of usage of nuclear power for energy production. It includes the closing down of nuclear power plants. It was introduced in Sweden (1980), in Italy (1987), in Belgium (1999), and in Germany (2000) and has been discussed in several other European countries. Austria, the Netherlands, and Spain have enacted laws not to build new nuclear power stations.

The current report presents possible phase-out scenarios in Sweden.

2 Assignment

The following conditions are assumed to describe a scenario with the transmutation of spent fuel:

- Barsebäck 1 and 2 are shut down.
- The remaining 10 units are in operation within 40 years from the respective start date.
- These two items give an amount of fuel that must be directly deposited. This amount is calculated according to the PLAN 2005 /1/ that is 9,350 tons of (initial) uranium weight[3]. This amount will thus be available year 2025.
- In a transmutation scenario, it is assumed that this amount (UOX-fuel) shall be transmuted instead of depositing.

The purpose of in such a transmutation scenario is to transmute at least 99% of trans-uranium elements that are contained in those 9,350 tons of spent fuel that otherwise must be deposited. The same demand applies also for the transuranium elements that are produced during the transmutation itself. The reprocessed spent fuel may thus contain at most 1% of the transuranium elements contained in those 9,350 tons of the unprocessed spent fuel.

The transmutation is assumed to proceed in the following way:

- LWR fuel (UOX and MOX) is reprocessed in a European country. Uranium-Plutonium as well as **M**inor **A**ctinides (MA = Neptunium, Americium, and Curium) are separated for the transmutation. High level waste are immobilized borosilicate glass matrix. Long-lived low- and middle-active waste including *I-129* should be addressed.
- Uranium and Plutonium is used in MOX-fuel to be burned up in one or several of the existing LWR reactors. *Following issues should be addressed: How many LWRs are needed? What is the difference between PWR and BWR nuclear power park? How long can one extend the lifetime of LWR (60 yr, 80 yr)?*

  Assumption: When the transmutation cannot be done in LWRs because of aging, the transmutation is performed in ADS.
- MOX-fuel for LWRs is produced in another European country.
- Plutonium recycling is done as many times as technically possible or suitable for transmuting at least 99% of all Trans-Uranium elements in the fuel.
- MOX-fuel from LWR is reprocessed in the same way as other fuel.
- MA as well as plutonium, that cannot be recycled in a reactor, is used in Uranium-free fuel that is transmuted in one or several specially designed reactors with fast neutron spectrum (ADS). These ADS-systems are built at the sites of the existing NPP (Forsmark or Oskarshamn).

  Assumption: the first such a reactor can be put into full operation year 2035 – (but probably later).
- Uranium-free MA-fuel (MA/Plutonium-fuel) is produced in another European country.
- Spent MA-fuel is reprocessed in a European country.

[3] 9,350 tons consists of 7,040 tons of BWR-fuel and 2,310 tons of PWR-fuel. Total power production is 2,820 TWh. This corresponds to a burn-up of 37 MWd/kgU assuming an average efficiency of 34%.

- Highly active waste from a reprocessing plant is assumed to be converted to borosilicate glass that is in principle disposed according to KBS-1.
- Long-lived low- and middle-active waste from reprocessing plants as well as production of MOX- and MA-fuel (MA/Pu-fuel) is conditioned and disposed of at a separate repository for long-lived LILW.
- Construction waste from new facilities is treated as a construction waste from the existing facilities.
- All the waste that arises from reprocessing and transmutation is brought to Sweden for disposal.

During the description of this scenario, one should establish and quantify the following:

- The number of LWRs for MOX recycling.
- The number of ADS for burning up MA and remaining plutonium.
- Required time to achieve 1% of Pu+MA.
- Additional electricity due to MOX and MA transmutation.
- Amount of waste as a result of reprocessing in the glass form to be disposed.
- Amount of TRU-LILW to be disposed of.
- Amount of left fuel to be disposed.
- Amount of construction waste from ADS facilities.
- Any volumes needed in Sweden for interim storage of vitrified waste before final disposal.
- Expenses for reprocessing and production of MOX- and MA-fuel.
- Cost of ADS facilities.

The above means that the aim is to transmute at least 99% of Pu and MA. This perhaps means that ADS must be in operation considerably longer than 50 years after 2035. In this case, one may assume that after 50 years, the first generation of ADS is substituted by a new one and so on.

It is conceivable that this aim, namely, to reduce the amount of Pu and MA to a level less than 1%, may not be achieved within 100 years for the amount of waste that is evaluated to exist by the year 2025 in those 9,350 tons of UOX. As an alternative, evaluate how much Pu and MA will still be present by the year 2060, when the fuel repository can be closed, as well as 50 and 100 year later

It should be noted here that Table 2-1 gives the current nominal values only. The thermal power of almost all the units has substantially been increased since their first start of operation. In addition, special measures have been taken to increase the thermal efficiency that resulted in higher efficiency and electrical power. This may explain some of the discrepancies between numerical results and data reported here and, for example, in /1/.

Table 2-1. Swedish reactor data.

Unit	Type	Power (MWt)	Power (MWe)	Eff (%)	Start (year)	End (year)	Burn-up (MWd/kg)	Fuel Mass (kg)	Enr (%)
Barsebäck-1	BWR	1,800	595	33.0	1975	1999	40	80,000	3.25
Barsebäck-2	BWR	1,800	595	33.1	1977	2005	40	80,000	2.97
Forsmark-1	BWR	2,929	970	33.1	1980	2020	40	121,000	3.25
Forsmark-2	BWR	2,929	970	33.1	1981	2021	41	121,000	3.20
Forsmark-3	BWR	3,300	1,150	34.9	1985	2025	43	126,000	3.30
Oskarshamn-1	BWR	1,375	445	32.4	1972	2012	40	82,000	2.80
Oskarshamn-2	BWR	1,800	602	33.4	1975	2015	40	82,000	3.00
Oskarshamn-3	BWR	3,300	1,160	35.2	1985	2025	40	127,000	3.10
Ringhals-1	BWR	2,500	830	33.2	1976	2016	43	125,000	3.21
Ringhals-2	PWR	2,652	870	32.8	1975	2015	44	80,500	3.92
Ringhals-3	PWR	2,775	920	33.2	1981	2021	43	82,100	3.94
Ringhals-4	PWR	2,775	915	33.0	1983	2023	44	82,100	3.92

3 Nuclear fuel cycle

3.1 General description of nuclear fuel cycle

The nuclear fuel cycle (NFC) refers to the overall scheme in which nuclear fuel is mined, enriched, fabricated into fuel assemblies, used in a reactor, and then reprocessed. Reprocessed fuel material may follow one of three routes: return of material to the reactor, return of the material to the reactor after it undergoes enrichment, or final storage as waste material.

It has turned out that low cost of fueling is the chief reason for the economic competitiveness of nuclear power. The principal steps of the fuel cycle are uranium mining and extraction from its ore (milling), uranium enrichment, fuel fabrication, loading and irradiation in the reactor (fuel management), unloading and cooling, reprocessing, waste packaging, and waste disposal.

Fuel is loaded into a reactor in a careful pattern so as to obtain the most energy production from it before it becomes no longer usable. Typically, a reactor is fueled in cycles, each cycle lasting one to two years, and a fuel batch is kept in the reactor for three to five cycles. At the end of each cycle, the oldest fuel is removed and fresh fuel loaded. The partially burned fuel that remains, however, is shuffled before the fresh fuel is installed. The objective of this procedure is to achieve a loading of maximum reactivity while keeping the power distribution among the different fuel assemblies within technical specifications.

A schematic picture of a nuclear fuel cycle is shown in Figure 3-1.

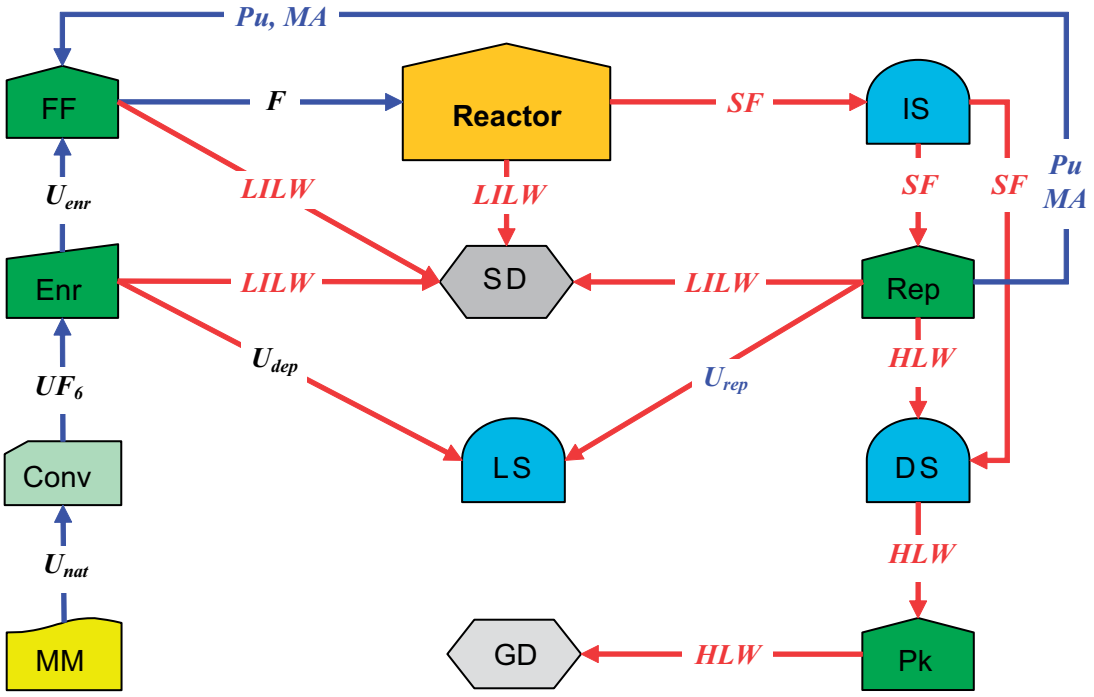

Figure 3-1. *Essential steps in nuclear fuel cycle.*

Here the following notation is used.

- **Conv**: Conversion.
- **DS**: Dry Storage.
- **Enr**: Enrichment.
- **FF**: Fuel Fabrication.
- **GD**: Geological Disposal.
- **HLW**: High Level Waste.
- **IS**: Interim Storage.
- **LILW**: Low and Intermediate Level Waste.
- **LS**: Long-term Disposal.
- **MA**: Minor Actinides.
- **MM**: Mining and Milling.
- **Pk:** Encapsulation/conditioning.
- **Rep**: Reprocessing facility.
- **SD**: Surface Disposal.
- **SF**: Spent Fuel.
- $\mathbf{U_{dep}}$: Depleted Uranium.
- $\mathbf{U_{enr}}$: Enriched Uranium.
- $\mathbf{U_{nat}}$: Natural Uranium.
- $\mathbf{U_{rep}}$: Reprocessed Uranium.

The complete set of technological processes to produce nuclear fuel from uranium ore is known as the *front end* of the nuclear fuel cycle. More specifically, the processes in the front end of the nuclear cycle are mining and milling, conversion, enrichment and fuel fabrication.

After producing energy in the reactor, nuclear fuel becomes spent fuel. The spent fuel has also to be processed in a storage facility or in a reprocessing facility if it is planned to be recycled. Temporary storage, reprocessing, long-term storage, or final disposal of spent fuel are together called the *back end* of the nuclear fuel cycle.

3.2 Essential steps in nuclear fuel cycle

Mining and milling: Uranium ore is mined by open-pit or underground mining methods and the uranium is extracted from the crushed ore in processing plants or mills using chemical methods. Sometimes it is possible to pass chemical solutions to the ore beds and dissolve the uranium from the ore directly. This process is known as in-situ leaching. This is the first step in a nuclear fuel cycle. The feed for mining and milling process is uranium ore and the product is U_3O_8 compound, which is mostly called yellowcake due to its color.

Conversion is the process of purifying the uranium concentrate and converting it to the chemical form required for the next stage of the nuclear fuel cycle. There are three such forms in common usage: metal, oxide (UO_2) and uranium hexafluoride (UF_6). UF_6 is the predominant product at this stage of the nuclear fuel cycle since it is converted to a gas by

heating for the enrichment stage, as employed in Light Water Reactors (LWRs), world's most common reactor types.

Enrichment: Uranium naturally consists of about 0.7% of ^{235}U isotope which is the main energy source in thermal reactors. A light water reactor cannot go critical with the natural occurrence of ^{235}U, so the ^{235}U content should be increased with a special process called enrichment. There are two commercially available technologies: gaseous diffusion and gas centrifuges. Both techniques are based on the slightly different masses of the uranium isotopes nuclei. So the enrichment is defined as the process of increasing the amount of ^{235}U contained in a unit quantity of uranium. The feed for this stage is natural UF_6 and the product is enriched UF_6. The other output of the process is the uranium which has lower fissile content than the natural uranium. It is known as enrichment tail or Depleted Uranium (DU).

Fuel fabrication: Enriched uranium in UF_6 form is converted to UO_2 powder to make fuel for LWR technology. This powder then is formed into pellets, sintered to achieve the desired density and ground to the required dimensions. Fuel pellets are loaded into tubes of zircaloy or stainless steel, which are sealed at both ends. These fuel rods are spaced in fixed parallel arrays to form the reactor fuel assemblies. The whole process is referred as Fuel Fabrication (FF). The similar procedure is adopted for natural uranium oxide fuel for some reactor types. The feed of this process is enriched or natural uranium oxide powder and the product is fuel assembly.

The reactor itself is an irradiator for nuclear fuel. It burns the fuel, produces energy and spent fuel. There are currently 7 types of thermal power reactors in the world: PWR, BWR, PHWR, RBMK, GCR, AGR, and VVER. The feed for reactor is fresh fuel containing uranium and sometimes plutonium, in case of Mixed Oxide (MOX) fuel, for existing nuclear fuel cycle options. The product is the spent fuel consisting of new nuclides such as fission products (Cs, I, etc), Minor Actinides (MA=Np, Am and Cm) and Pu as well as the remaining uranium. The biggest part of the spent fuel is still uranium.

Reprocessing: The spent nuclear fuel still consists of significant amount of fissile material that can be used to produce energy. The considerable amount of ^{235}U is still contained in the spent fuel and there are new fissile nuclides that were produced during normal operation of nuclear reactor such as ^{239}Pu. Some nuclear fuel cycle options consider taking out the fissile material from the spent fuel, refabricating it as fuel and burning in reactor. MOX fuel is the most common fuel that uses reprocessed material. Reprocessing process is based on chemical and physical processes to separate the required material from spent nuclear fuel. The feed of this process is spent fuel and the products are reusable material and High Level Wastes (HLW).

Spent fuel storage: Spent fuel can be temporarily stored for future use, or alternatively, it can be stored for final disposal. Spent fuel can be placed in water pools (wet type) or in air cooled containers or facilities (dry type).

HLW Storage: The waste from reprocessing facilities are classified as High Level Waste (HLW) and requires careful handling. HLW is stored in special storage facilities after proper treatment.

4 Methodology of the calculation

The calculation of the isotopic inventory in the whole system is a complex problem. The TRU isotopes may arise by neutron capture or by radioactive decay of another isotope. The isotopes may be changed into other isotopes by fission, neutron capture or radioactive decay. One can form the following Bateman equations for each isotopeas ${}^{Z}_{A}X$ as

$$\frac{dN({}^{Z}_{A}X)}{dt} = -\sigma_a({}^{Z}_{A}X)N({}^{Z}_{A}X)\Phi - \lambda({}^{Z}_{A}X)N({}^{Z}_{A}X) + \sigma_c({}^{Z-1}_{A}X)N({}^{Z-1}_{A}X)\Phi + \tag{1}$$
$$+\lambda_{\alpha}({}^{Z+4}_{A+2}X)N({}^{Z+4}_{A+2}X) + \lambda_{\beta-}({}^{Z}_{A-1}X)N({}^{Z}_{A-1}X) + \lambda_{\beta+}({}^{Z}_{A+1}X)N({}^{Z}_{A+1}X)$$

where

Φ	is one-group neutron flux.
$N({}^{Z}_{A}X)$	is a concentration of ${}^{Z}_{A}X$.
σ_a	is a one-group microscopic cross-section for neutron absorption (fission and capture).
σ_c	is a one-group microscopic cross-section for neutron capture.
λ	is a decay constant.

4.1 Fuel cycle analyzer (FCA)

In order to analyze the total actinide inventory over the whole fuel cycle of various reactor scenarios, we have built a MATLAB /2/ code FCA (Fuel Cycle Analyzer). The FCA code solves Bateman equation for the actinides in the whole inventory, which means the in-core inventory plus the waste cooling storage plus the waste repository storage. All isotopes are present in one vector during the calculation.

4.1.1 Methodology

Let us assume the neutron flux is proportional to the thermal power of a reactor ($\Phi = c \cdot P$), then one can write

$$\frac{dN({}^{Z}_{A}X)}{dt} = p({}^{Z}_{A}X)P + \sum_{A_2,Z_2} \lambda_{A_2Z_2 \to AZ} N({}^{Z_2}_{A_2}X) - \lambda({}^{Z}_{A}X)N({}^{Z}_{A}X) \tag{2}$$

where

$$p({}^{Z}_{A}X) = -\sigma_a({}^{Z}_{A}X)N({}^{Z}_{A}X)c + \sigma_c({}^{Z-1}_{A}X)N({}^{Z-1}_{A}X)c \tag{3}$$

represents a production/depletion rate of ${}^{Z}_{A}X$ caused by the neutron flux irradiation of the fuel.

In the vector form one can write the Bateman equation simply as

$$\frac{d\mathbf{N}(t)}{dt} = \mathbf{R}(t) + \mathbf{A}\mathbf{N}(t) \tag{4}$$

where $\mathbf{N}(t) = \left[N_1(t), N_2(t), \cdots, N_m(t)\right]^T$ is the actinide vector $N_i(t)$ represents a total number of atoms of the i-*th* isotope in the system); **R**(*t*) is a vector of total production/depletion rates (total change of the isotope number per time unit) for the actinide isotopes from the neutron absorption (by irradiation in the reactor); **A** is a matrix of decay constants (created from ENDF/B-VI.8).

Assuming time independent **R** and **A** within the time interval $[t_0, t_1]$, one can write the solution to equation (4) as

$$\mathbf{N}(t) = e^{\mathbf{A}t}\mathbf{N}_0 + \mathbf{A}^{-1}(e^{\mathbf{A}t} - \mathbf{I})\mathbf{R} \tag{5}$$

Here, $\mathbf{N}_0 \equiv \mathbf{N}(0)$ and the matrix exponential function is defined as

$$e^{\mathbf{A}t} = \mathbf{I} + \mathbf{A}\frac{t}{1!} + \mathbf{A}^2\frac{t^2}{2!} + \mathbf{A}^3\frac{t^3}{3!} + \ldots \tag{6}$$

In general, the production/depletion rate, **R**, varies in time. Our basic assumption is that this time dependency is piecewise constant, i.e. equation refers (5) only to a corresponding time period within which R may be considered time independent. On a large scale, the vector function **R** (*t*) represents the corresponding scenario. The next section deals with the way of computing of this function.

4.1.2 Production/depletion rates

The total production/depletion rate (**PD** rate) **R** (*t*) represents the sum of particular production/depletion rates of all reactors

$$\mathbf{R}(t) = \sum_j P_j(t)\mathbf{p}_j \tag{7}$$

where P_j (*t*) is the thermal power of the j-*th* reactor (if the reactor is shut down then it has no contribution to the PD rate); $\mathbf{p}_j$ is a vector of average production/depletion rate for the actinide isotopes from the neutron absorption in the j-*th* reactor per unit thermal power per unit time.

The vector $\mathbf{p}_j$ must be averaged over the whole fuel cycle. Vectors $\mathbf{p}_j$ can be evaluated from the ORIGEN2 /4/ calculation in the following way. First, one writes the equation for $\mathbf{p}_j$ as

$$\frac{d\mathbf{N}_j(t)}{dt} = P_j\mathbf{p}_j + \mathbf{A}\mathbf{N}_j(t) \tag{8}$$

with the initial and final conditions

$$\begin{aligned} \mathbf{N}_j(0) &= \mathbf{N}_{j0} \\ \mathbf{N}_j(T) &= \mathbf{N}_{jT} \end{aligned} \tag{9}$$

Second, the vector $\mathbf{N}_{jT}$ is computed directly by ORIGEN2. Finally, the solution is found by demanding that $\mathbf{p}_j$ in equation (8) provide the same final condition, $\mathbf{N}_{jT}$, as ORIGEN2 does. It has been found, by using equation (5), that the solution is given by

$$\mathbf{p}_j = \frac{1}{P_j}\mathbf{A}\left(\mathbf{I} - e^{-\mathbf{A}T}\right)^{-1}\left[e^{-\mathbf{A}T}\mathbf{N}_{jT} - \mathbf{N}_{j0}\right] \tag{10}$$

Numerical calculations have shown that the solution as presented by equation (10) is not sufficiently accurate probably due to large times, *T*–1 year or even greater. One can dramatically increase the numerical precision, first, by putting the calculated rate, $\mathbf{p}_j^{(0)}$, as given by (5), back into equation (10) to evaluate the expected final value

$$\mathbf{N}_{j,T}^{(0)} \equiv \mathbf{N}(T) = e^{\mathbf{A}T}\mathbf{N}_0 + \mathbf{A}^{-1}(e^{\mathbf{A}T} - \mathbf{I})P_j\mathbf{p}_j^{(0)} \tag{11}$$

Second, we calculate the discrepancy in the final value

$$\Delta\mathbf{N}_{j,T} = \mathbf{N}_{j,T} - \mathbf{N}_{j,T}^{(0)} \tag{12}$$

Third, we use this discrepancy to evaluate a small correction to the rate $\mathbf{p}_j$

$$\Delta\mathbf{p}_j = \frac{1}{P_j}\mathbf{A}\left(\mathbf{I} - e^{-\mathbf{A}T}\right)^{-1}\Delta\mathbf{N}_{j,T} \tag{13}$$

Finally, we calculate a numerical solution that is sufficiently accurate

$$\mathbf{p}_j = \mathbf{p}_j^{(0)} + \Delta\mathbf{p} \tag{14}$$

Vector $\mathbf{p}_j$ must be computed for each reactor type, and each reprocessing strategy.

4.1.3 Calculation data flow

A reactor can be loaded with various fuels (UOX, MOX), and each fuel can have generally a different PD rate. The PD rate of the reactor must therefore be weighted over the PD rates which correspond to the particular fuels. In order to be able to decide which fuel (MOX1, MOX2 ... ADS1, ADS2 ...) is to be loaded at a certain time into a certain core, the code creates several databases. Each reactor core has an in-core database of fuels and corresponding dates of loading. Each fuel type has also its own store database. If the fuel is unloaded from the core, the corresponding waste store notes which year the fuel was unloaded from the core and what the fuel energy equivalent was (released energy).

The input file assigns a preferable order in which a certain core should be loaded with different fuel, e.g. a core BWR-MOXa will have the following preferable order of fuel MOX1 MOX2 MOX3. That means the core will be loaded with MOX1 fuel preferably. The MOX1 fuel is a MOX fuel which is reprocessed from the UOX waste. If the UOX waste stores contain no sufficiently long cooled waste then the core can be loaded with MOX2 fuel which is reprocessed from the MOX1 waste, and so on. If the waste store contains sufficiently long cooled waste to reprocess it into a new fuel then a certain amount of fuel is removed and new fuel is loaded into a core. The energy equivalent of the waste removed from the store is not equal to the energy equivalent of the new fuel. The ratio of those equivalents must be stated in the input file for each fuel which is reprocessed from waste.

At each time step, the code has to find out which cores are to be loaded. When a core is to be loaded then a new PD rate is weighted from the PD rates of the fuels that form the in-core inventory database. After that the fuel is burned, and the time is increased by the time step. If a certain reactor can not be loaded because of lack of fuel, then the reactor is automatically shut down.

4.1.4 Code implementation

The code is named FCA that stands for Fuel Cycle Analyzer; it has been written in MATLAB programming environment that is suitable for vector and matrix operations. The input consists of a number of text files describing the nuclear reactors park as well as other technical parameters. MATLAB offers also a great variety of plotting options to display illustrative figures on both the screen and paper.

4.1.5 Considered isotopes

We consider the following 40 isotopes: ^{234}U, ^{235}U, ^{236}U, ^{237}U, ^{238}U, ^{239}U, ^{236}Np, ^{236m}Np, ^{237}Np, ^{238}Np, ^{239}Np, ^{240}Np, ^{240m}Np, ^{236}Pu, ^{237}Pu, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{243}Pu, ^{244}Pu, ^{245}Pu, ^{240}Am, ^{241}Am, ^{242}Am, ^{242m}Am, ^{243}Am, ^{244}Am, ^{244m}Am, ^{245}Am, ^{241}Cm, ^{242}Cm, ^{243}Cm, ^{244}Cm, ^{245}Cm, ^{246}Cm, ^{247}Cm, ^{248}Cm, ^{249}Cm.

We note here that the bulk of the results have been obtained by the FCA code that uses the nuclides listed in Table 4-1 only. However, some calculations have been performed by the ORIGEN2 code (see 4.2) when it was necessary, for example to trace individual isotopes such as ^{129}I or to calculate radiotoxicities. Because of this, some isotopes not listed in Table 4-1 may appear in plots.

The considered actinides and decay constants are stated in Table 4-1.

Table 4-1. Actinides and decay constants considered in the calculation.

Isotope	$T_{½}$	Fraction				
		α	β−	β+	SF	IS
^{234}U	2.46×10^{5} y	1.000				
^{235}U	7.04×10^{8} y	1.000				
^{236}U	2.34×10^{7} y	1.000				
^{237}U	6.75 d		1.000			
^{238}U	4.47×10^{9} y	1.000				
^{239}U	23.45 m		1,000			
^{236}Np	1.54×10^{5} y		0.089	0.910		
^{236m}Np	22.5 h		0.480	0.520		
^{237}Np	2.14×10^{6} y	1.000				
^{238}Np	2.11 d		1.000			
^{239}Np	2.36 d		1.000			
^{240}Np	61.9 m		1.000			
^{240m}Np	7.22 m		1.000			
^{236}Pu	2.86 y	1.000				
^{237}Pu	45.2 d			1.000		
^{238}Pu	87.7 y	1.000				
^{239}Pu	2.41×10^{4} y	1.000				
^{240}Pu	6.56×10^{1} y	1.000				
^{241}Pu	1.43×10^{1} y		1.000			
^{242}Pu	3.75×10^{5} y	1.000				
^{243}Pu	4.96 h		1.000			
^{244}Pu	8.00×10^{7} y	1.000				
^{245}Pu	10.5 h		1.000			
^{240}Am	50.8 h			1.000		
^{241}Am	4.32×10^{2} y	1.000				
^{242}Am	16.02 h		0.827	0.173		
^{242m}Am	1.41×10^{2} y					1.000
^{243}Am	7.37×10^{3} y	1.000				
^{244}Am	10.1 h		1.000			
^{244m}Am	26 m		1.000			
^{245}Am	2.05 h		1.000			
^{241}Cm	32.8 d	0.010		0.990		
^{242}Cm	162.8 d	1.000				
^{243}Cm	2.91×10^{1} y	0.998		0.002		
^{244}Cm	1.81×10^{1} y	1.000				
^{245}Cm	8.50×10^{3} y	1.000				
^{246}Cm	4.76×10^{3} y	1.000				
^{247}Cm	1.56×10^{7} y	1.000				
^{248}Cm	3.48×10^{5} y	0.917			0.083	
^{249}Cm	64.15 m		1.000			

4.1.6 Input files

The FCA code uses two main plain text input files. The "Reactor database" file contains all details to create the required reactor scenario, and the "PD_database" file contains all details to compute PD rates for each fuel. Apart from these input files, the FSA code uses a number of data files from ORIGEN2 calculations, which are also necessary to compute the PD rates.

4.2 ORIGEN2

The original version of the ORIGEN program was developed by the Chemical Technology Division of ORNL for use in computing the compositions and radioactivity of fission products, cladding materials, and fuel materials in LWRs, LMFBRs, MSBRs, and HTGRs. ORIGEN has four cross-section libraries designed for the analysis of fuel irradiated in each of the four reactor types mentioned above. Basic microscopic cross-section data are processed from the ENDF/B files. The primary application of ORIGEN is analysis of the fuel irradiation and decay history.

Mathematically, ORIGEN solves the following system of equations

$$\frac{dN_i(t)}{dt} = \left(\text{Production}\right)_i - \left(\text{Destruction}\right)_i - \left(\text{Decay}\right)_i \tag{15}$$

Specifically, these equations may be written as

$$\frac{dN_i(t)}{dt} = \sum_j \gamma_{j,i}\sigma_{f,j}N_j\phi + \sigma_{c,i-1}N_{i-1}\phi + \lambda_i' N_i' - \sigma_{c,i}N_i\phi - \lambda_i N_i \tag{16}$$

Here, $\sum_j \gamma_{j,i}\sigma_{f,j}N_j\phi$ is the yield rate of N_i due to the fission of all nuclides N_j;

$\sigma_{c,i-1}N_{i-1}\phi$ is the rate of transmutation of into N_i due to radiative neutron capture by nuclide N_{i-1};

$\lambda_i' N_i'$ is the rate of formation of N_i due to radioactive decay of nuclides N_i';

$\sigma_{f,i}N_i\phi$ is the destruction rate of N_i due to fission;

$\sigma_{c,i}N_i\phi$ is the destruction rate of N_i due to all forms of neutron capture (n,γ), (n,α), (n,p), $(n, 2n)$, $(n, 3n)$;

$\lambda_i N_i$ is the radioactive decay rate of N_i.

Equation (16) is written for a homogeneous medium containing a space-energy-averaged neutron flux, ϕ, with flux-weighted average cross sections, σ_f and σ_c, representing the reaction probabilities. In reality, the flux as a function of space, energy, and time is dependent upon the nuclide concentrations. ORIGEN assumes that the space-energy-averaged flux can be considered constant over time steps. Similarly, ORIGEN assumes that a single set of flux-weighted neutron cross sections is adequate for use over the entire fuel exposure time.

4.3 Nuclear fuel cycle simulation (NFCSim)

The NFCSim /5/ model simulates complex nuclear-fuel-cycle (NFC) scenarios characterized by a large array of interacting components of the NFC. A nuclear economy in NFCSim may consist of any number of reactors. The model's reactor and fuel cycle modeling capabilities include water-reactor oriented fuel cycles with the option of actinide recycling and a suite of fast reactors or accelerator driven systems for closure of the nuclear fuel cycle. NFCSim deploys new facilities as needed, subject to additional exogenously specified constraints such as limitations on the capacity of reprocessing facilities. Using a database of the current nuclear infrastructure (mines, conversion, and fabrication and enrichment plants) as the point of departure, NFCSim determines the time-dependent demand for these services. Unit costs for individual processes plus amortized capital costs for new facilities are assessed; a default cost database is provided. With the aforementioned information NFCSim calculates a system-wide time dependent annual cost of electricity as well as a discounted life-cycle cost.

4.4 Monteburns

Monteburns /6/ is an automated tool that links the Monte Carlo transport code MCNP with the radioactive decay and burnup code ORIGEN2. *Monteburns* produces a large number of criticality and burnup results based on various material feed/removal specifications, power(s), and time intervals. The program processes input from the user that specifies the system geometry, initial material compositions, feed/removal specifications, and other code-specific parameters. Various results from MCNP, ORIGEN2, and other calculations are then output successively as the code runs. The principle function of *Monteburns* is to transfer one-group cross-section and flux values from MCNP to ORIGEN2, and then transfer the resulting material compositions (after irradiation and/or decay) from ORIGEN2 back to MCNP in a repeated, cyclic fashion. Overall, *Monteburns* acts as a pre- and postprocessor for both MCNP4B and ORIGEN2.

5 Phase-out scenario

In this section, we analyze the time variation of the total inventory of actinides and trans-uranium nuclides (TRU) in the whole system of the Swedish nuclear power plants during the planned life time.

5.1 Basic initial data

Conditions:

- Average fuel burnup: 37 GWd/tHM.
- Energy availability factor (EAF): 0.836 (10 months in a year).
- ORIGEN2 libraries: BWRUE.LIB, PWRUS.LIB.

The average EAF of all Swedish NPPs is 0.792 according to IAEA (http://www.iaea.org/cgi-bin/db.page.pl/pris.charts.htm), but we assume the average factor will be slightly higher over the whole planned life time.

The calculation was performed by the FCA code. The input file was created according to Table 2-1 (except of the fuel burnup).

5.2 Total actinide inventory

The total mass of actinides produced at the Swedish Nuclear Power Plants was simulated by the FCA code. Figure 5-1 shows the total actinide inventory in time over a period beginning in 1972, when the first Swedish NPP in Oskarshamn was put into operation, until year 2050.

According to PLAN 2005, the actinide amount should be 9,350 tons in 2025. Our calculation estimates the total actinide inventory to be 9,826 tons in 2025. The difference is caused by various simplifying assumptions of the calculations. In particular, we assume a higher average EAF then the actual one to compensate for somewhat lower average burnup of 37 MWd/kg used in current model. In reality the burnup of the most Swedish reactors was gradually increasing sometimes reaching 45 MWd/kg for PWR nowadays.

Figure 5-2 shows that the total transuranics inventory will reach 100 tons in 2025. The TRU inventory mass is relatively independent on the fuel burnup.

5.3 Plutonium and americium inventory

Figure 5-3 depicts plutonium and americium inventory in the system. Americium is the second most frequent transuranic element in the waste, and must be taken into account if TRU is to be burned.

Detailed calculations give the following mass inventory of the most important uranium and plutonium isotopes.

Figure 5-4 is completely consistent with what we may expect from the half lives of the above radio isotopes. Indeed, these half lives are given in Table 5-1.

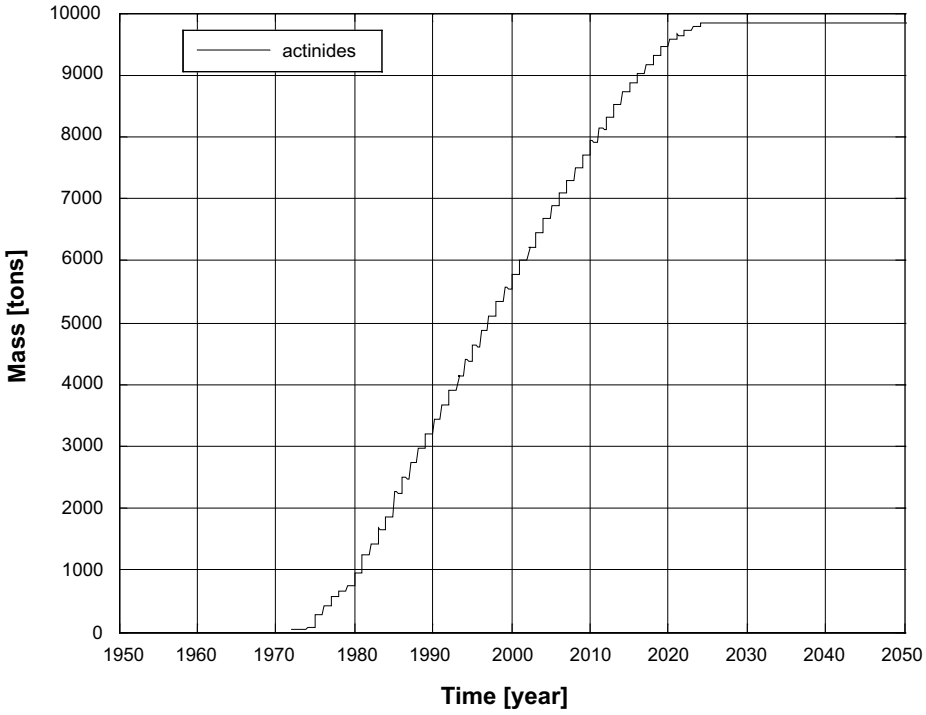

Figure 5-1. *Total inventory of actinides in the spent fuel from the Swedish reactors in the Phase-out scenario.*

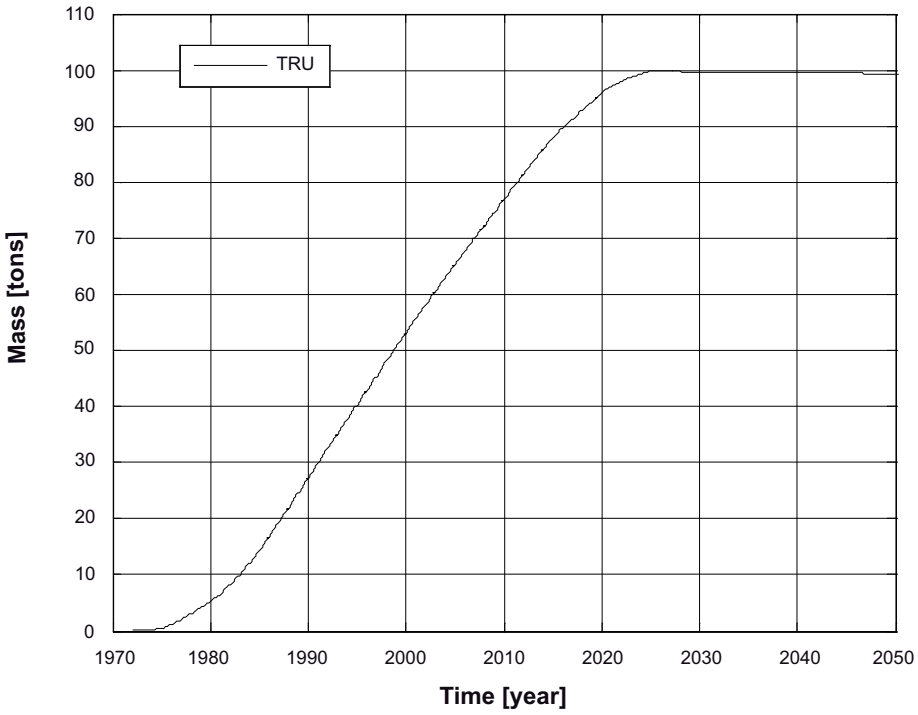

Figure 5-2. *Total inventory of transuranium elements in the spent fuel from the Swedish reactors in the Phase-out scenario.*

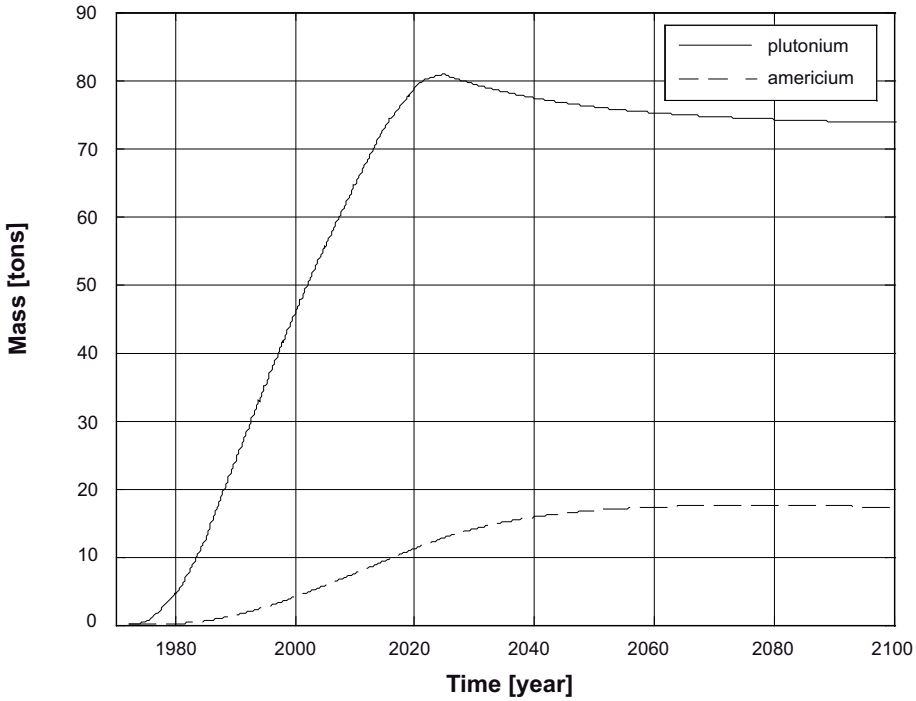

Figure 5-3. *Total inventory of plutonium and americium in the spent fuel from the Swedish reactors in the Phase-out scenario.*

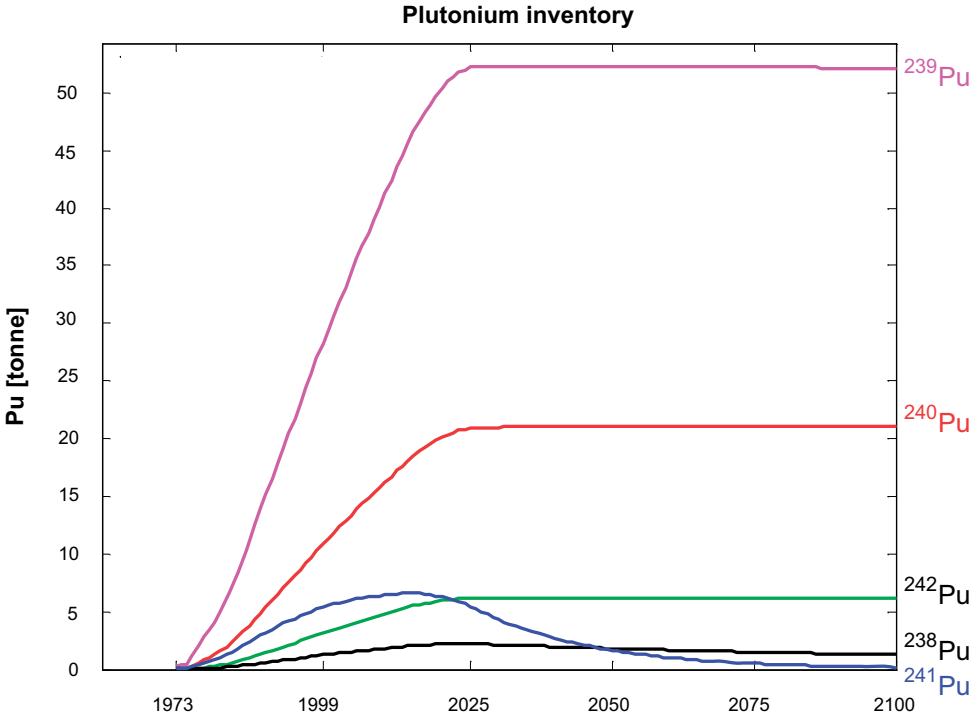

Figure 5-4. *Total inventory of plutonium isotopes in the spent fuel from the Swedish reactors in the Phase-out scenario.*

Table 5-1. Half lives of some plutonium isotopes.

Isotope	^{238}Pu	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu
Half life (yr)	87.7	24,110	6,563	14.35	373,300

^{239}Pu, ^{240}Pu, and ^{242}Pu do not change appreciably over a time interval of 100 years, whereas masses of ^{238}Pu and ^{241}Pu reflect the balance between their own decay and build-up from other isotopes.

5.4 Uranium inventory

Uranium inventory in the Phase-out scenario is presented in Figure 5-5.

We briefly note here that the depletion of both ^{235}U and ^{238}U is completely hidden by the income of these isotopes with fresh fuel. After year 2025, their amount does not change noticeably due to very long half-lives. As contrast, both ^{233}U and ^{234}U keep growing from the decay of other isotopes even after year 2025 when the last reactor is shut down. These rates are rather slow, about 2 g per year for ^{233}U and about 13 kg per year for ^{234}U.

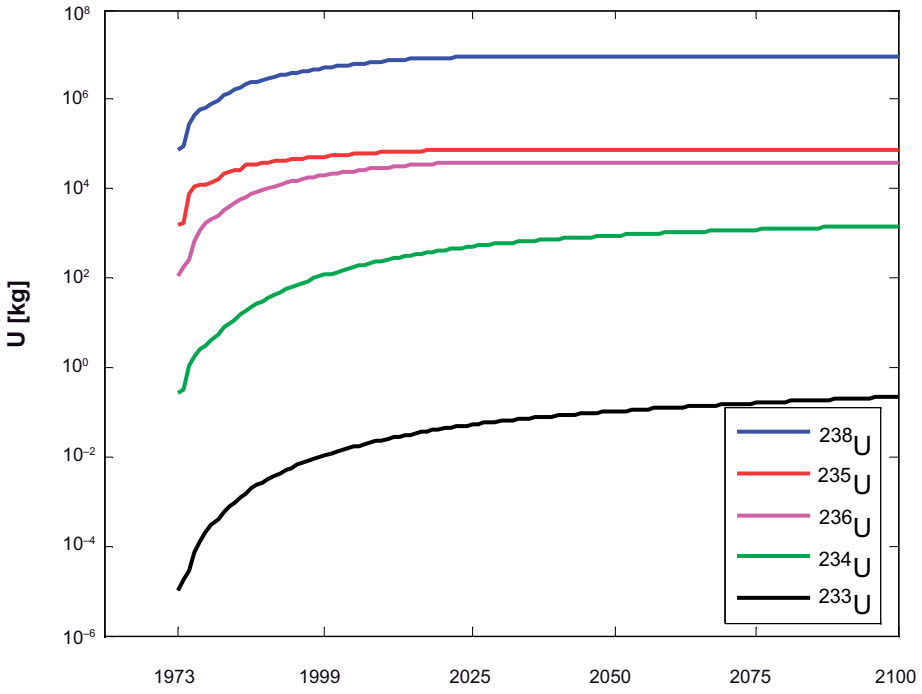

Figure 5-5. *Total uranium inventory in the spent fuel from the Swedish reactors in the Phase-out scenario.*

5.5 Minor actinides inventory

Masses of minor actinides accumulated in the Swedish repository in the Phase-out scenario are presented in Figure 5-6.

We note here that ^{243}Cm, ^{244}Cm, and ^{245}Cm are produced in relatively small quantities (as compared to ^{241}Am, for example), of order of several kg to hundreds kg. It is interesting to note that ^{241}Am exhibits, at first glance, a somewhat peculiar behavior growing for some time after 2025, but explanation is actually simple. Until year 2025, the mass of ^{241}Am develops due to the balance of production by decay from other isotopes and destruction by irradiation in the reactors. In 2025 the irradiation (or destruction) is abruptly stopped, since then the production – mainly from the ^{241}Pu β-decay – dominates until it is overwhelmed by natural decay of ^{241}Am.

Table 5-2. Half lives of minor actinides isotopes.

Isotope	^{237}Np	^{241}Am	^{243}Am	^{243}Cm	^{244}Cm	^{245}Cm
Half life (yr)	2,144,000	432.2	7,370	29.1	18.1	8,500

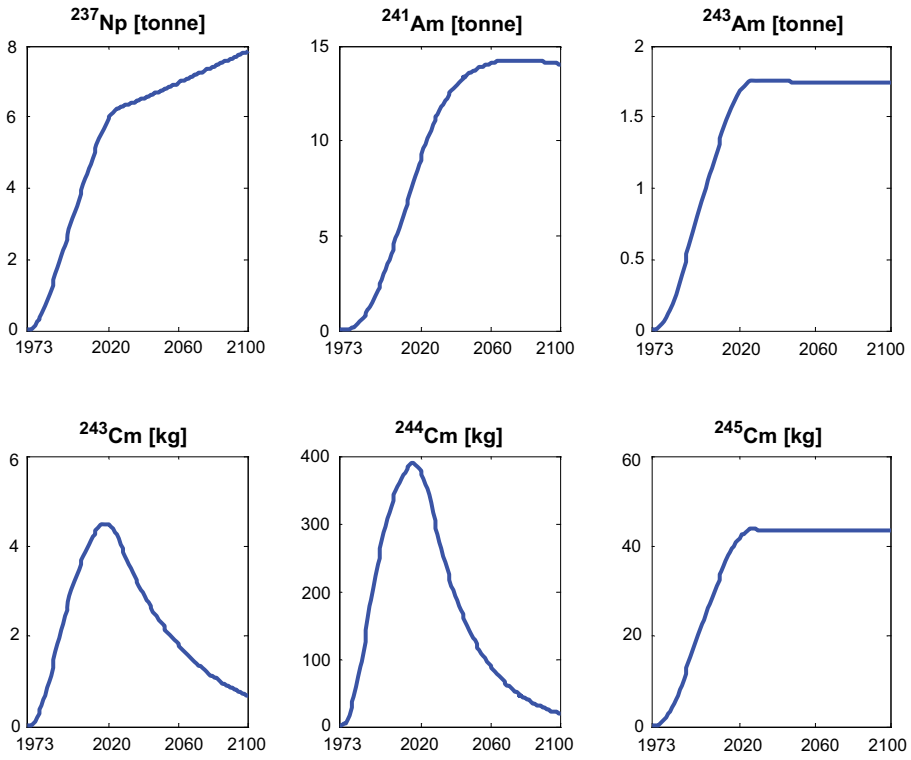

Figure 5-6. *Inventory of minor actinides in the spent fuel from the Swedish reactors in the Phase-out scenario.*

5.6 Long lived and short lived fission/activation products inventory

The most radiotoxic of long-lived fission/activation products are ^{99}Tc, ^{129}I, ^{135}Cs, ^{93}Zr. The mass inventory of these isotopes accumulated in the spent fuel from the Swedish Nuclear Power Plants in the Phase-out scenario is presented in Figure 5-7.

As a general observation, we note here that the above fission products are present in the spent fuel in considerable quantities of order of several thousand kilograms whereas their natural decay is relatively slow, cf Table 5-3.

Some short-lived fission products, namely ^{137}Cs, ^{90}Sr, ^{151}Sm, and ^{85}Kr are given in Figure 5-8.

We briefly note here that the peak of inventory of these short-lived isotopes happens somewhere about year 2015 when the bulk of the Swedish reactors are scheduled to shut down. The time behavior of these isotopes closely follow our expectation based on their respective half-lives, cf Table 5-4.

Table 5-3. Long lived fission/activation products.

Isotope	^{107}Pd	^{126}Sn	^{99}Tc	^{129}I	^{135}Cs	^{93}Zr
T½	6,500,000	100,000	211,100	15,700,000	2,300,000	1,530,000

Table 5-4. Short-lived fission/activation products.

Isotope	^{137}Cs	^{90}Sr	^{151}Sm	^{85}Kr
Half life (yr)	30.07	28.79	90	10.76

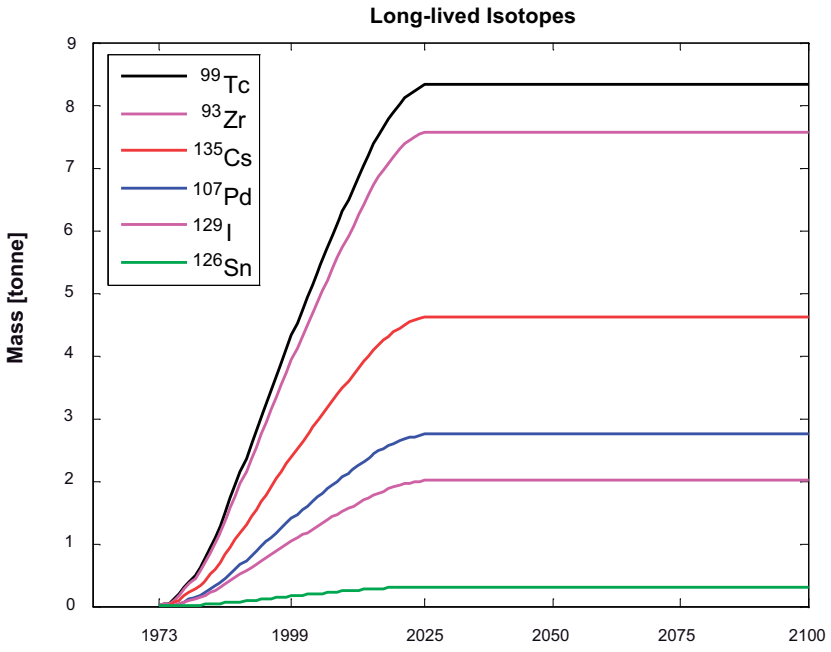

Figure 5-7. *Inventory of long lived isotopes in the spent fuel from the Swedish reactors in the Phase-out scenario.*

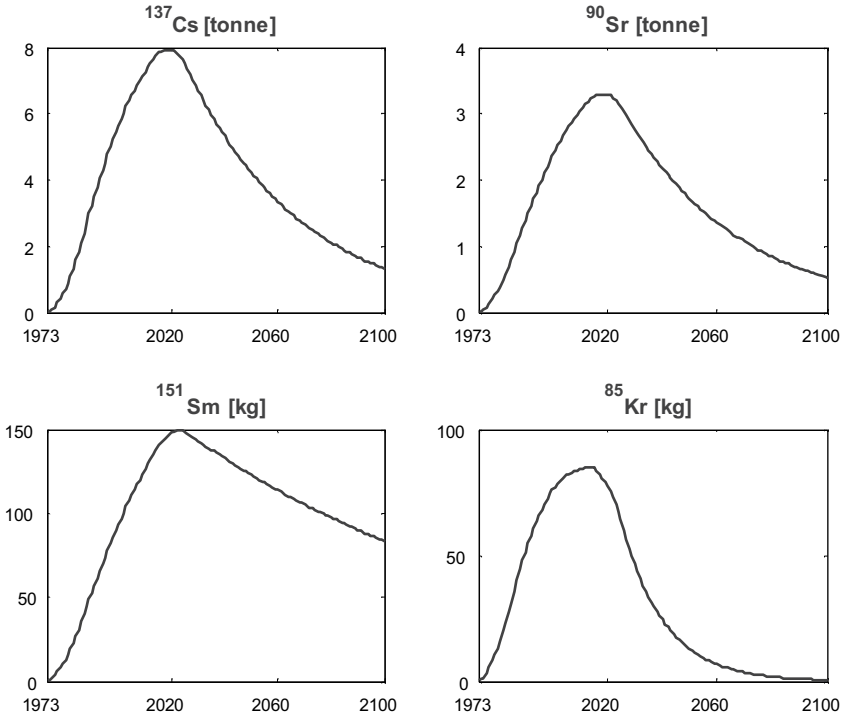

Figure 5-8. *Inventory of short-lived fission products in the spent fuel from the Swedish reactors in the Phase-out scenario.*

5.7 Radioactivity and radiotoxicity

The quantity which expresses the degree of radioactivity or radiation producing potential of a given amount of radioactive material is radioactivity denoted in the current report as *RA*. The radioactivity, *RA*, may be considered as the rate at which a number of atoms of a material disintegrate, or transform from one isotope to another which is accompanied by the emission of radiation. It is defined as the number of disintegrations per unit time:

$$RA(t) \equiv \frac{dN_{dis}(t)}{dt} \tag{17}$$

The radioactive decay law states that the number of isotopes of a certain kind, *N*, obeys the relationship

$$\frac{dN}{dt} = -\lambda N \tag{18}$$

Here, λ is the decay constant that is related to the half life, $T_{1/2}$, as

$$\lambda \cdot T_{1/2} = \ln 2 \tag{19}$$

Then the activity may simply be expressed in terms of *N*:

$$RA(t) = \frac{dN_{dis}(t)}{dt} = -\frac{dN(t)}{dt} = \lambda N \tag{20}$$

Each disintegration event is normally accompanied by emitting various kinds of radiation. Because of this, the radioactivity quantity, *RA*, does not correctly reflect the biological effect of radioactivity. For example, fission products decay mostly by β radiation, while transuranium elements decay essentially through α radiation. For the same disintegration rate, α emitters are much more radiotoxic than β emitters with the exception of ^{129}I which

has very peculiar biological properties, with a very high affinity for the thyroid gland. To account for different biological consequences of radiation, the radiotoxicity, *RT*, is defined as,

$$RT\left[\mathrm{Sv}\right] \equiv F_d \left[\frac{\mathrm{Sv}}{\mathrm{Bq}}\right] \cdot RA\left[\mathrm{Bq}\right] \tag{21}$$

Here, *RA* is the radioactivity, and F_d is a dose factor converting the radioactivity in Bq of a specific nuclide to its radiotoxicity in Sv. The dose factors are evaluated taking into account weighting factors, or quality factors, of different kinds of radiation, see Table 5-5.

Based on these values, the International Commission on Radiation Protection (ICRP) has evaluated /7/ the dose factors some of which are given in Table 5-6.

Table 5-5. Quality factors.

Radiation	Quality factor Q
Photons, all energies	Q = 1
Electrons and muons, all energies	Q = 1
Neutrons:	
energy < 10 keV	Q = 5
10 keV < energy < 100 keV	Q = 10
100 keV < energy < 2 MeV	Q = 20
2 MeV < energy < 20 MeV	Q = 10
energy > 20 MeV	Q = 5
Protons, energy > 2 MeV	Q = 5
α particles and other atomic nuclei	Q = 20

Table 5-6. Radiotoxicological data.

Isotope	Half-life (years)	Dose factor (Sv/Bq)	Radioactivity (Bq/kg)	Radiotoxicity (ingested) (Sv/kg)
^{99}Tc	2.11 10^{5}	7.8 10^{-10}	6.3 10^{11}	4.9 10^{2}
^{129}I	1.57 10^{7}	1.1 10^{-7}	6.5 10^{9}	7.0 10^{2}
^{135}Cs	2.30 10^{7}	2.0 10^{-9}	4.2 10^{10}	8.0 10^{1}
^{93}Zr	1.53 10^{6}	1.1 10^{-9}	9.3 10^{10}	1.0 10^{2}
^{233}U	1.59 10^{6}	2.5 10^{-7}	3.6 10^{11}	9.0 10^{4}
^{238}Pu	8.77 10^{2}	2.3 10^{-7}	6.3 10^{14}	1.4 10^{8}
^{239}Pu	2.41 10^{5}	2.5 10^{-7}	2.3 10^{12}	6.0 10^{5}
^{240}Pu	6.56 10^{3}	2.5 10^{-7}	8.3 10^{12}	2.1 10^{6}
^{241}Pu	1.43 10^{1}	4.7 10^{-9}	3.8 10^{15}	1.8 10^{7}
^{242}Pu	3.73 10^{5}	2.4 10^{-7}	1.5 10^{11}	4.0 10^{4}
^{237}Np	2.14 10^{6}	1.1 10^{-7}	2.6 10^{10}	3.0 10^{5}
^{241}Am	4.33 10^{2}	2.0 10^{-7}	1.3 10^{14}	3.0 10^{7}
^{243}Am	7.37 10^{3}	2.0 10^{-7}	7.4 10^{12}	1.5 10^{6}
^{243}Cm	2.91 10^{1}	2.0 10^{-7}	1.9 10^{15}	4.0 10^{8}
^{244}Cm	1.81 10^{1}	1.6 10^{-7}	3.0 10^{15}	0.5 10^{9}
^{245}Cm	8.50 10^{3}	3.0 10^{-7}	6.3 10^{12}	1.9 10^{6}

Specific radioactivity as well as specific radiotoxicity may easily be calculated using the relationship between the total number of nuclides and the total mass of nuclides

$$N = M\frac{N_A}{A\cdot 10^{-3}} \tag{22}$$

Here, N is the total number of atoms of an isotope with a mass number of A g/mole found in a piece of mass M kg, N_A stands for Avagadro's number of nuclear entities per mole. It follows then that the specific radioactivity may be expressed as

$$RA = \frac{\lambda N}{M} = \frac{\lambda N_A}{A\cdot 10^{-3}} = \frac{\ln 2\cdot N_A}{T_{1/2}\cdot A\cdot 10^{-3}} = \frac{1.32}{T_{1/2}\left[yr\right]}\frac{10^{19}}{A} \tag{23}$$

The total radioactivity as well as total radiotoxicity of the spent fuel accumulated by the Swedish reactor park is presented in Figure 5-9 where one can clearly see a sharp decrease of these quantities. This is due to the end of irradiation that resulted in the stop of production of highly radioactive nuclides. Another reason is a relatively large time step of one year within which a great fraction of short-lived isotopes has considerably decayed during a time interval 2025 through 2026.

Figure 5-10 presents the total radioactivity and radiotoxicity of the plutonium isotopes found in the spent fuel produced by the Swedish reactors in 1972 through 2025 years according the Phase-out scenario.

The radioactivity and radiotoxicity for individual plutonium isotopes accumulated in the Swedish Nuclear Power Plants under the same period of time are shown in Figure 5-11.

Figure 5-12 presents the total radioactivity and radiotoxicity of the uranium isotopes found in the spent fuel produced by the Swedish reactors in 1972 through 2025 years according the Phase-out scenario.

It is clearly seen a sharp decrease of the uranium radioactivity in Figure 5-12. This is due to the isotope ^{239}U. It is very short lived (about 23 min) and hence very radioactive which is additionally shown in Figure 5-12.

Radioactivity and radiotoxicity of individual uranium isotopes are presented in Figure 5-13.

Time behavior of uranium isotopes radioactivity and radiotoxicity closely follows our expectation as one may deduce from their halve-lives whose values are given in Table 5-7.

It should be noted here that ^{239}U actually exists only under the irradiation period due to its very short half-life that results in a very high radioactivity. This fact explains the sharp discontinuity in the total radioactivity/toxicity.

Figure 5-14 picture presents the radioactivity and radiotoxicity in the Phase-out scenario due to only minor actinides found in the spent fuel accumulated by the Swedish Nuclear Power Plants.

Individual radioactivities and radiotoxicities of minor actinides found in the spent fuel accumulated by the Swedish reactor park in the Phase-out scenario are presented in Figure 5-15.

Table 5-7. Halve-lives of uranium isotopes.

Isotope	^{233}U	^{235}U	^{236}U	^{238}U	^{239}U
Half life	1.59×10^{5} yr	7.04×10^{8} yr	2.34×10^{7} yr	4.47×10^{9} yr	23 min

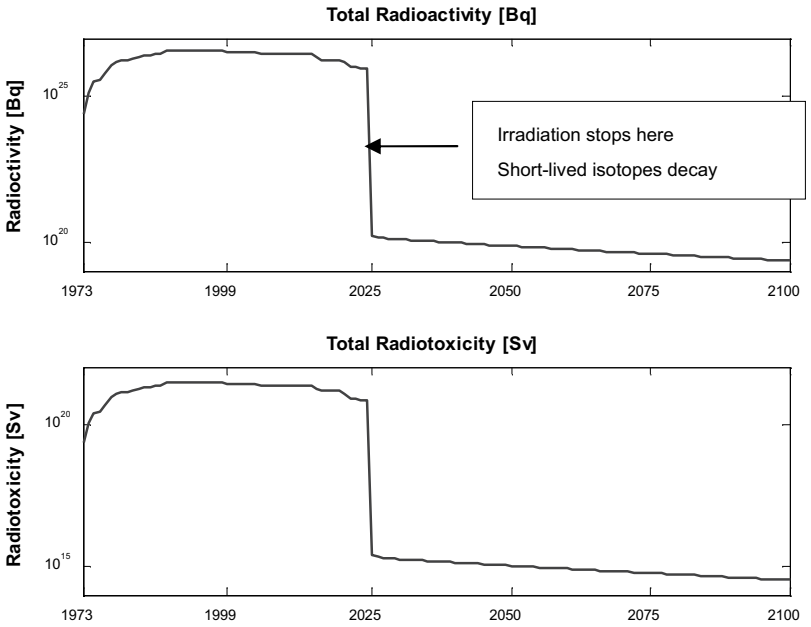

Figure 5-9. *Total radioactivity/radiotoxicity of the spent fuel accumulated by the Swedish Nuclear Power Plants in the Phase-out Scenario.*

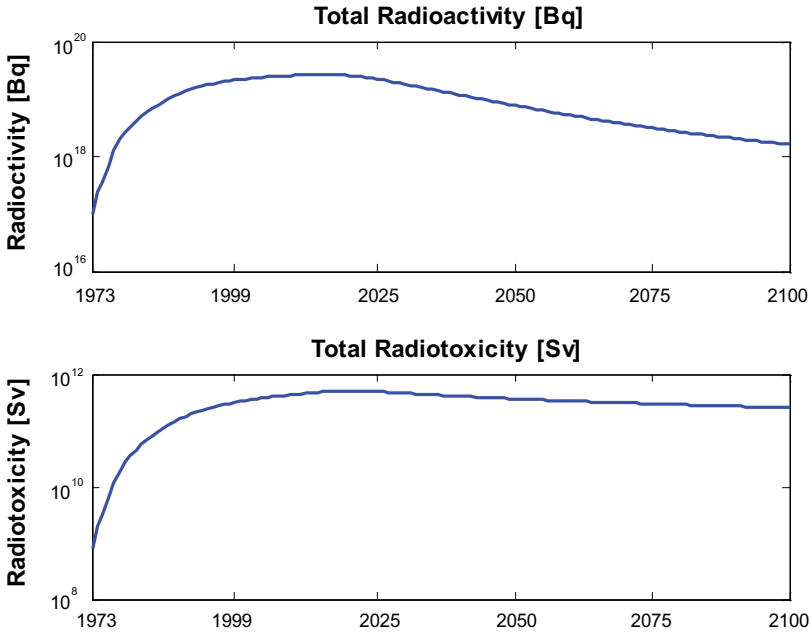

Figure 5-10. *Plutonium radioactivity/radiotoxicity constituent of the spent fuel accumulated by the Swedish reactors in the Phase-out Scenario.*

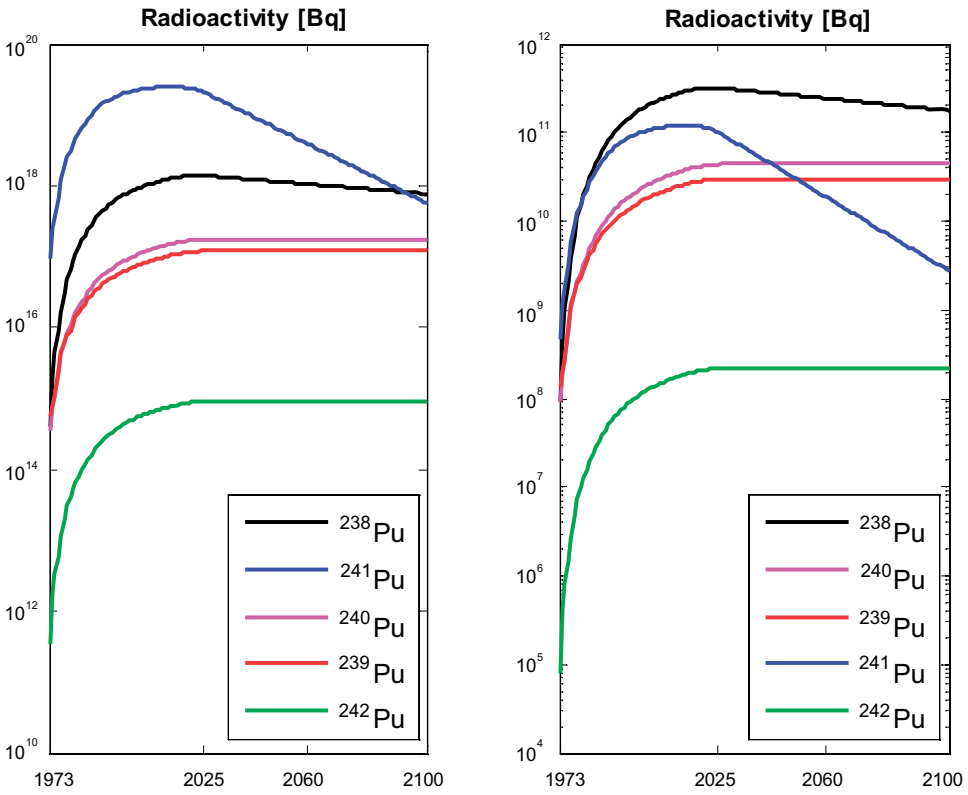

Figure 5-11. *Individual Plutonium Radioactivity/toxicity in the Phase-out Scenario.*

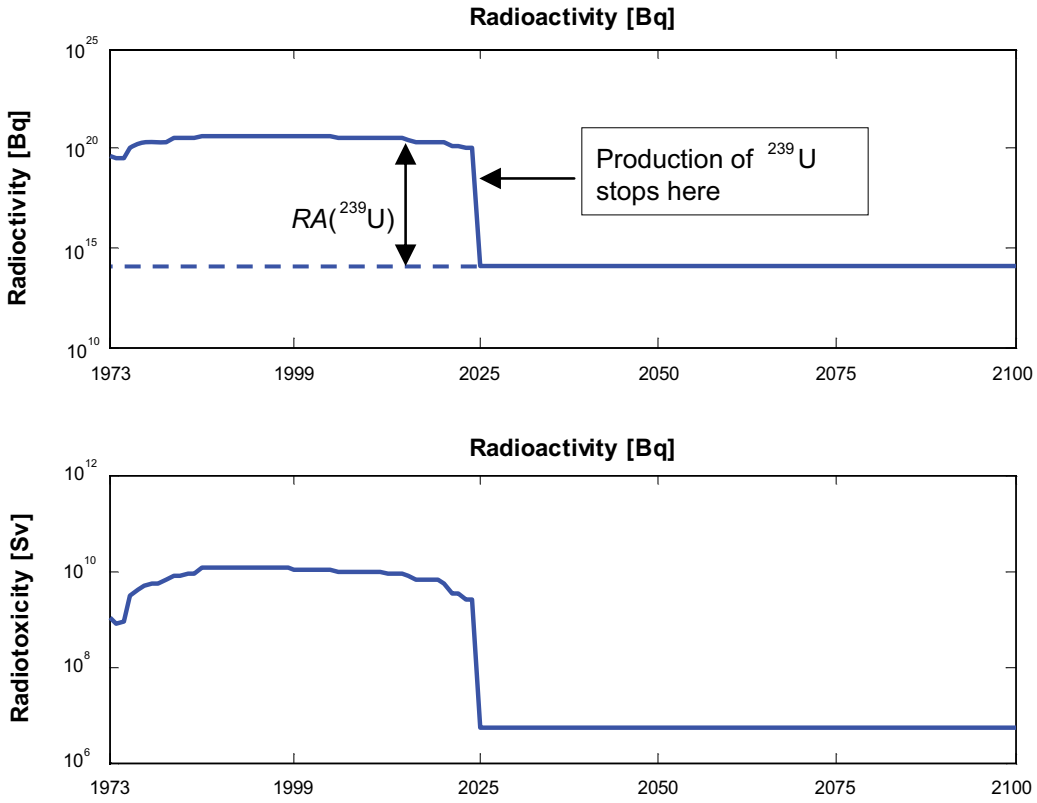

Figure 5-12. *Uranium total radioactivity/toxicity in Phase-out scenario.*

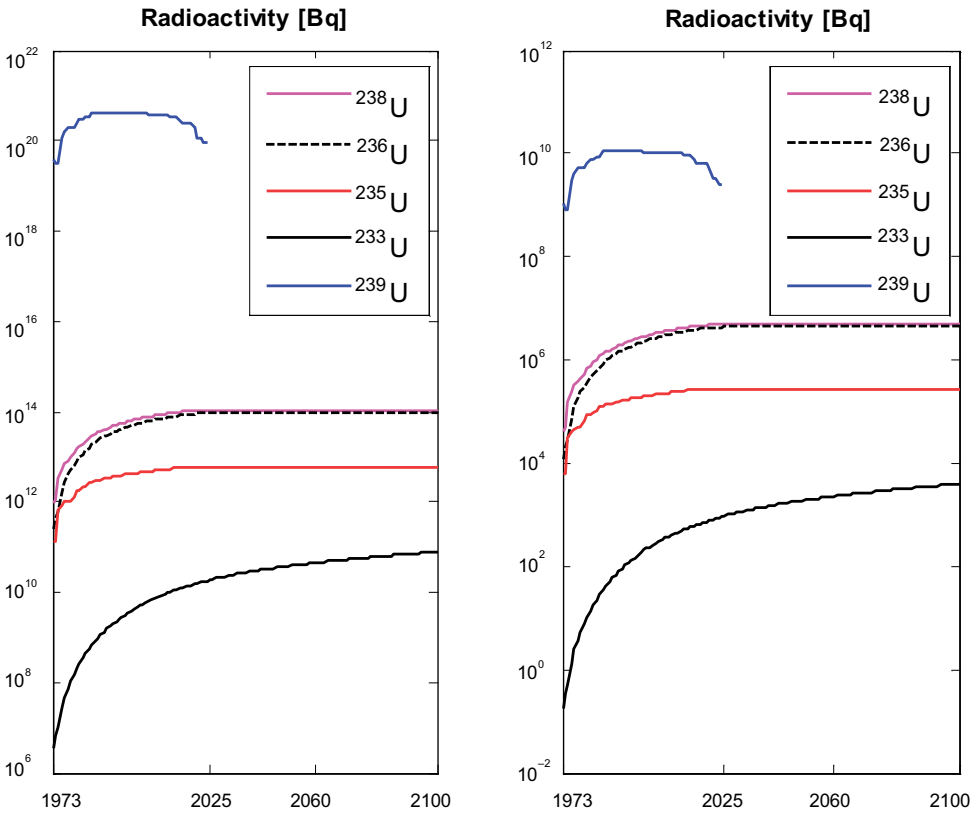

Figure 5-13. *Radioactivity/radiotoxicity of uranium isotopes of the spent fuel accumulated by the Swedish Reactor Park in the Phase-out scenario.*

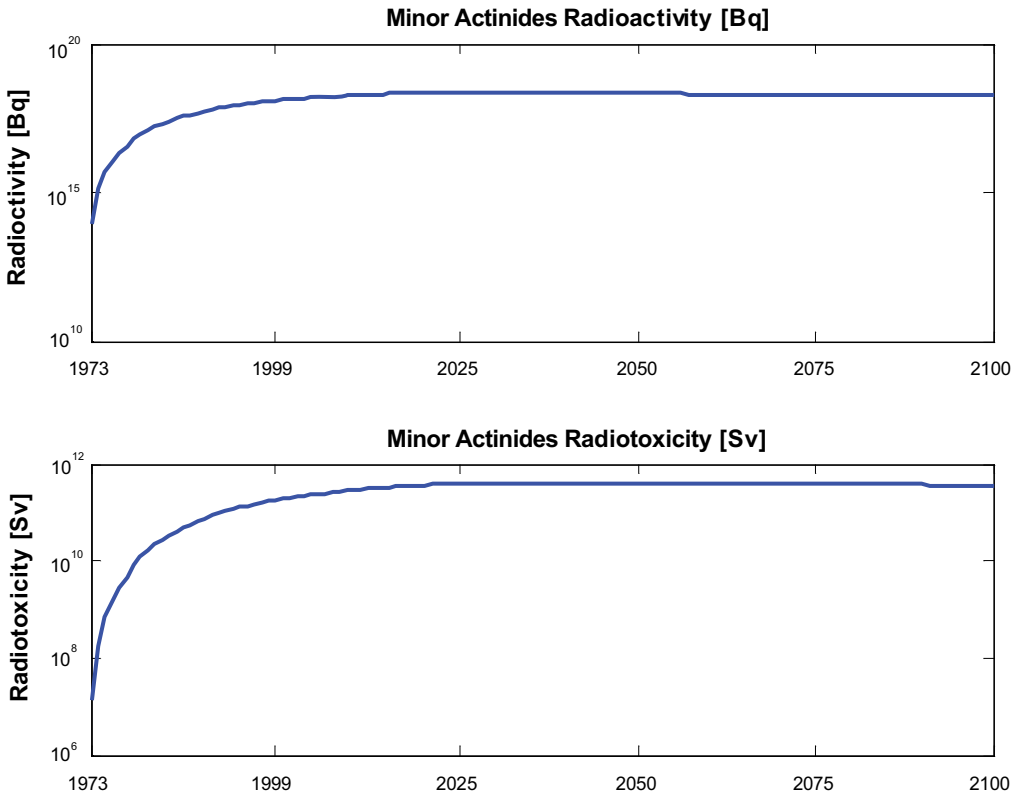

Figure 5-14. *Total radioactivity and radiotoxicity of Minor Actinides in the spent fuel accumulated by the Swedish NPPs in Phase-out Scenario.*

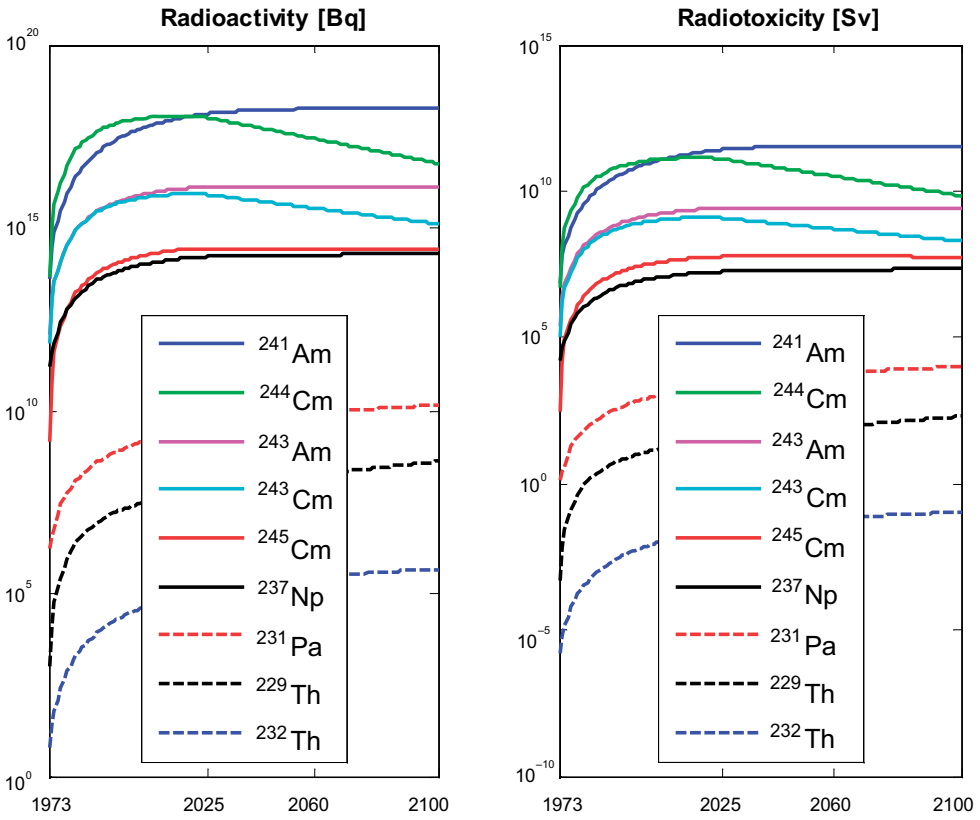

Figure 5-15. *Individual radioactivity and radiotoxicity of Minor Actinides isotopes found in the spent fuel from the Swedish reactors in the Phase-out Scenario.*

The radioactivity and radiotoxicity of some long lived fission/activation products present in the spent fuel accumulated by the Swedish Nuclear Power Plants since 1972 through 2025 in the Phase-out scenario is presented in Figure 5-16.

It may be noted that ^{129}I is the least radioactive out of these four isotopes are shown in Figure 5-16, but as one might envisage, it is much more radiotoxic when we account for the different dose factors.

The radioactivity and radiotoxicity of some short lived fission/activation products present in the spent fuel accumulated by the Swedish Nuclear Power Plants since 1972 through 2025 in the Phase-out scenario is presented in Figure 5-17.

The isotope of ^{85}Kr is absent from the radiotoxicity plot in Figure 5-17 due to missing data for the corresponding dose factor.

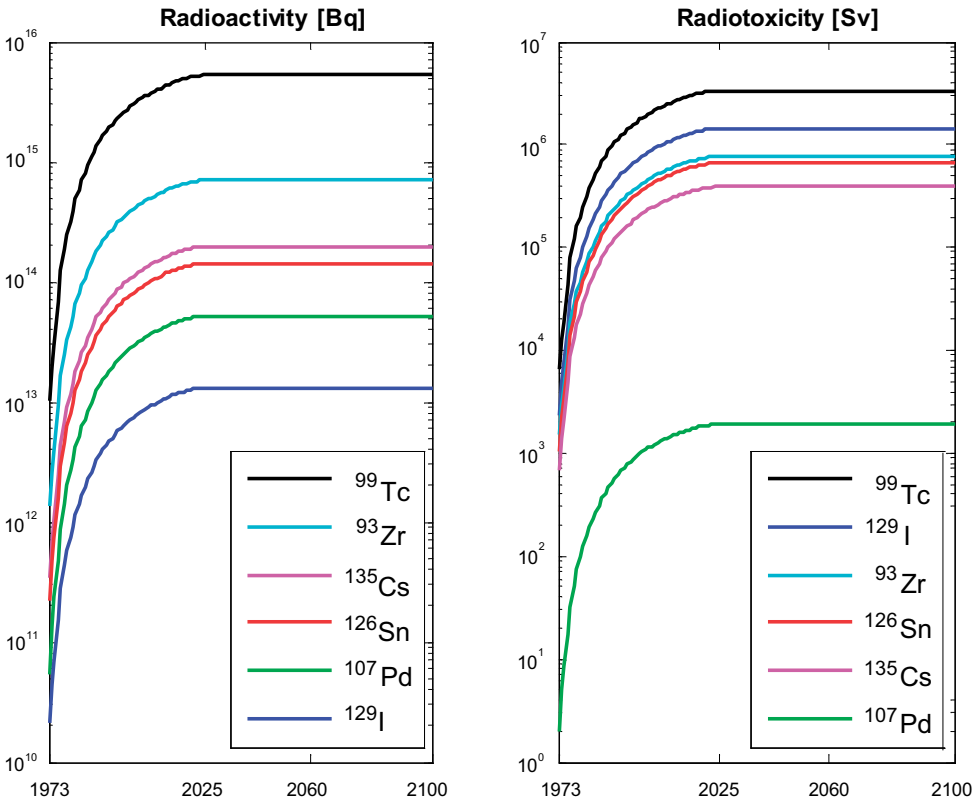

Figure 5-16. *Individual radioactivity/radiotoxicity of long lived fission/activation products found in the spent fuel from the Swedish reactors in Phase-out Scenario.*

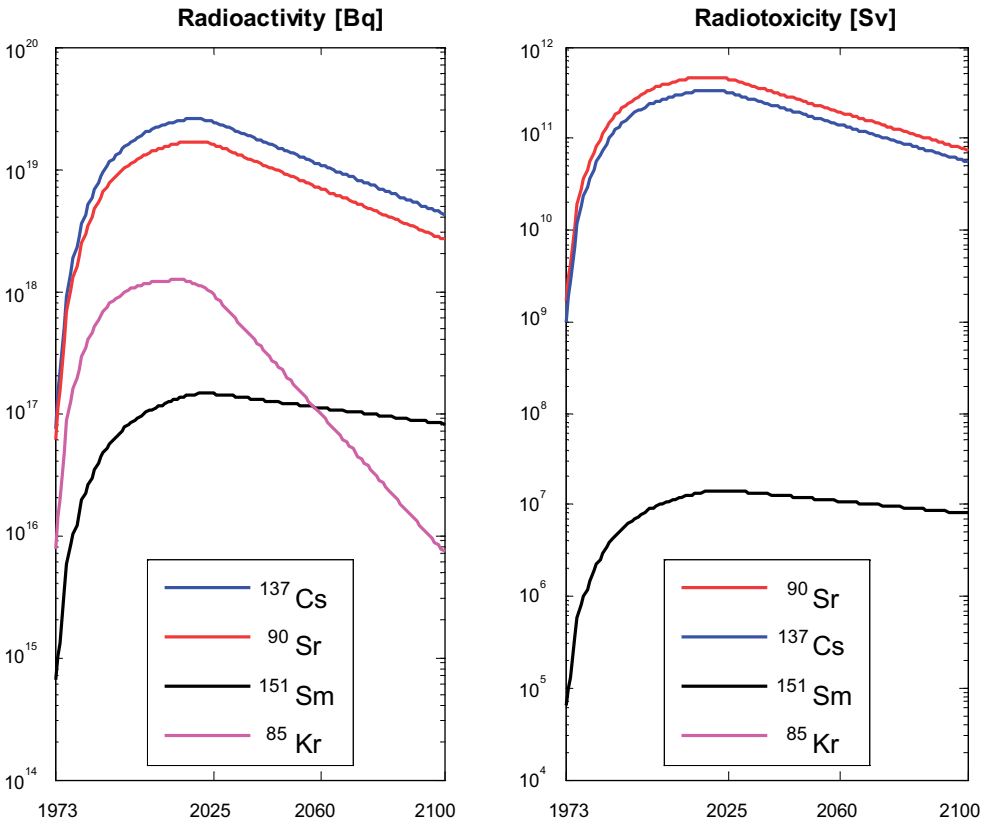

Figure 5-17. *Individual radioactivity/radiotoxicity of short lived fission/activation products found in the spent fuel from the Swedish reactors in Phase-out Scenario.*

6 Burning heavy metals in MOX fuel

In this section we analyze possible burning of heavy metals with emphasis on plutonium in MOX fuel. Let the MOX fuel reprocessed from the UOX fuel be denoted as MOX1; the MOX fuel reprocessed from MOX1 fuel as MOX2; and the MOX fuel reprocessed from MOX2 fuel as MOX3. In this calculation we assume that MOX3 is not reprocessed any more, and is disposed of in the repository. That corresponds to the nowadays technical abilities; however, we admit that multiple reprocessing of MOX fuel might be possible in the future. We reprocess the MOX1 and MOX2 fuel in such a way to keep reactivity of MOX2 and MOX3 respectively at the same level as MOX1 has. The plutonium enrichment of the MOX fuel is described in Table 6-1.

Conditions:

- Power history of Swedish reactor park is given by Table 2-1.
- UOX fuel burnup: 37 GWd/tHM.
- Energy availability factor until 2025: 0.836.
- Total thermal power of MOX reactors: 4,600MW.
- Number of MOX reactors: 2 (thermal power of one reactor is 2,300MW).
- Energy availability factor from 2,025 to 2,075 (MOX reactors): 0.918.
- MOX fuel burnup: 40 GWd/tHM.
- Lifetime of MOX reactors: 50 years (2025–2075).
- Core fraction of MOX fuel: 1.0.
- Recycling (cooling + fabrication) time of spent fuel to produce MOX fuel: 6 years.
- ORIGEN2 library for PWR core: PWRPUPU.LIB.
- ORIGEN2 library for BWR core: BWRPUPU.LIB.

The thermal power of the reactors was chosen in such a way that all the available MOX2 waste is reprocessed into MOX3 fuel and burned. That means the highest possible impact on Pu inventory is reached during the planned lifetime of 50 years. The only MOX2 waste which remains after 2075 is waste produced from 2069 to 2075 which could not be reprocessed because of its cooling. The calculation was performed by the FCA code. The following sections present results.

Table 6-1. Pu enrichment of MOX fuel.

Fuel	Pu enrichment of MOX fuel (%)	
	PWR	BWR
MOX1	7.65	7.65
MOX2	10.75	9.24
MOX3	12.23	11.97

6.1 Burning plutonium in PWR or BWR

In this section we analyze the impact of burning plutonium found in the spent fuel accumulated from the Swedish Nuclear Power Plants during 1972 to 2025 in PWR and BWR core respectively.

Figure 6-1 shows that BWRs are more efficient in burning the plutonium than PWRs. The plutonium inventory is decreased from 80.8 tons to 20.2 tons in the BWRs. The PWRs burned 3.4 tons of plutonium less than BWRs.

As seen in Figure 6-2, BWR is slightly more efficient in reducing the total radiotoxicity of plutonium present in the spent fuel from the Swedish Nuclear Power Plants. We stress here once more that ^{241}Pu is the most radioactive plutonium isotope as the left part of Figure 6-2 suggests, it is approximately by a factor of 20 greater than ^{238}Pu, but what regards the radiotoxicity, the situation is just the opposite, ^{238}Pu is by a factor of 2.5 more radiotoxic as compared with ^{241}Pu. This fact is due to the difference in the dose factors for these isotopes, as seen in Table 5-6, the dose factor for ^{238}Pu is two orders of magnitude greater than that of ^{241}Pu.

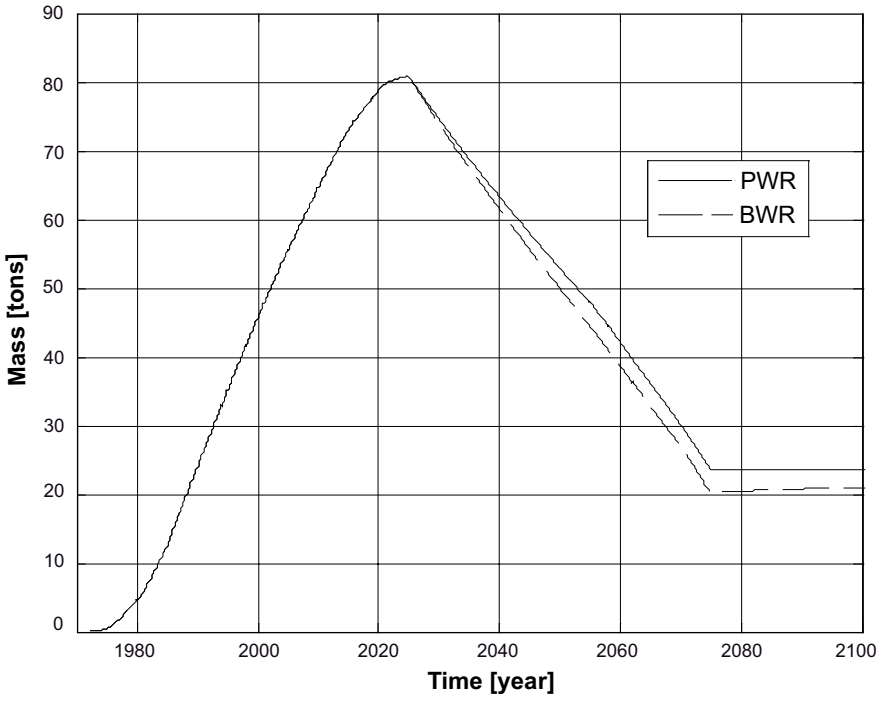

Figure 6-1. *BWR vs. PWR: Plutonium inventory.*

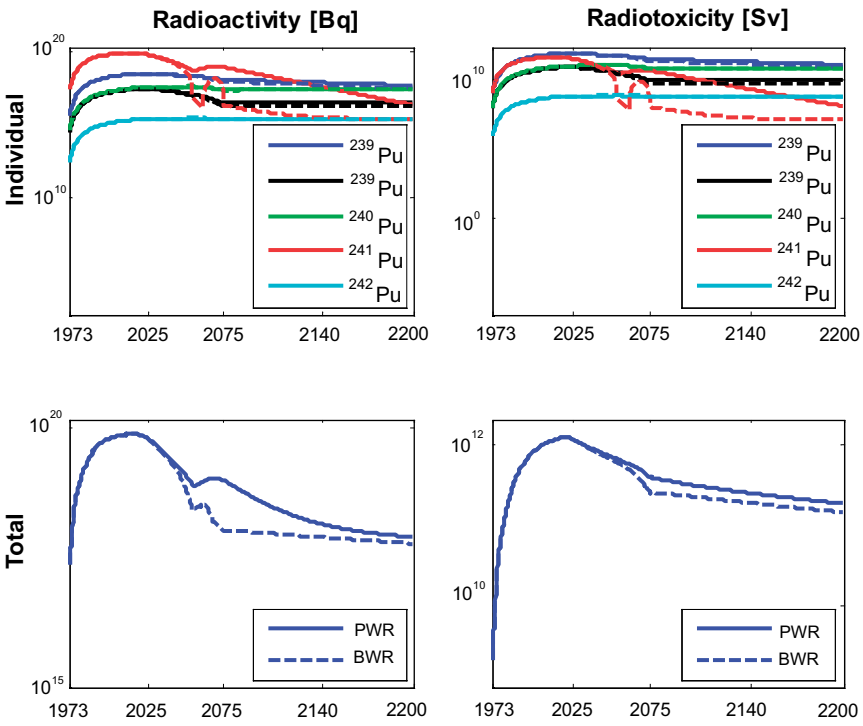

Figure 6-2. *PWR vs. BWR: Plutonium Radioactivity/toxicity. Solid line represents PWR, broken line stands for BWR.*

6.2 Burning americium in PWR or BWR

In this section we analyze the impact of burning americium found in the spent fuel accumulated from the Swedish Nuclear Power Plants during 1972 to 2025 in PWR and BWR core respectively.

Figure 6-3 shows that americium inventory is considerably increased from 12.8 tons in 2025 to 20.2 tons in 2075 (BWR case), which is a consequence of irradiating plutonium in the thermal cores. BWRs seems to be a better choice again taking into account the increase of americium inventory is not that big as in the PWR case. Around the year 2055 the MOX2 fuel was started to be loaded into the reactors, which explains the higher americium production compared to the previous years. The MOX2 fuel contains a higher amount of plutonium as showed in Table 6-1, which results in faster americium production. This process is even more accelerated around the year 2069 when the MOX3 fuel was started to be loaded. The difference in americium mass between BRW and PWR is 1.1 ton in 2075.

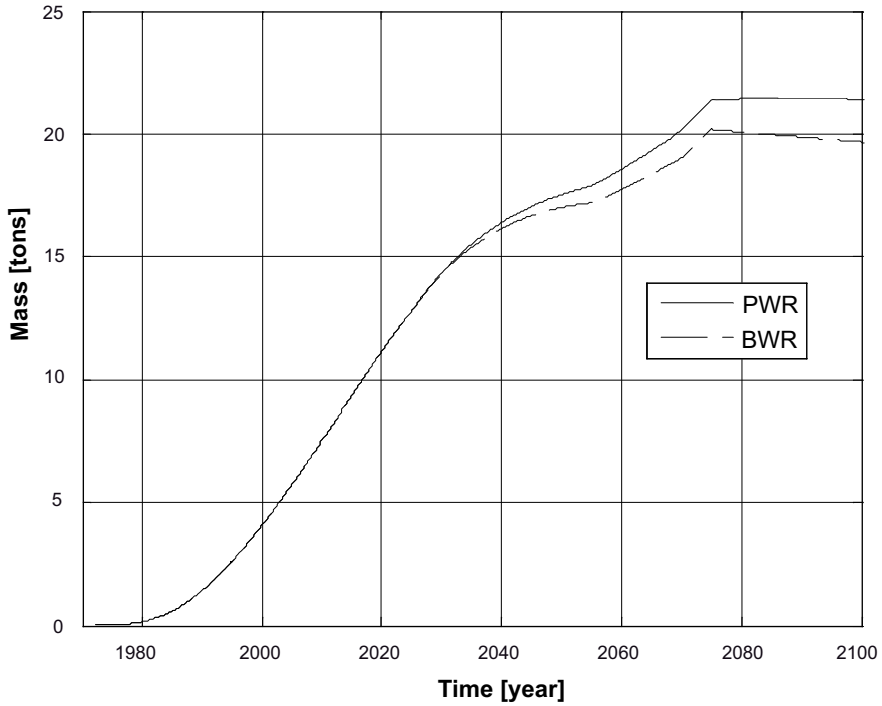

Figure 6-3. *BWR vs. PWR: Americium inventory.*

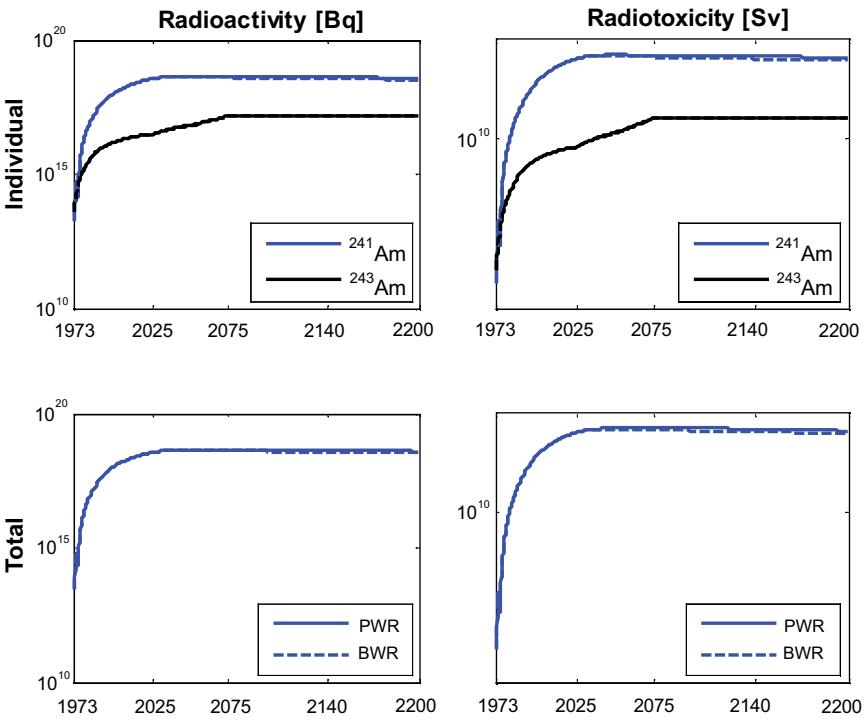

Figure 6-4. *PWR vs. BWR: Americium Radioactivity/toxicity. Solid line represents PWR, broken line stands for BWR.*

6.3 Burning transuranium elements in PWR or BWR

In this section we analyze the impact of burning transuranium elements found in the spent fuel accumulated from the Swedish Nuclear Power Plants during 1972 to 2025 in PWR and BWR core respectively.

Figure 6-5 shows that TRU inventory was decreased from 100 tons (year 2025) to 49 tons (year 2075) in the BWR case. PWRs burned 4.5 tons of transuranics less than BWRs. Thus, BWRs are approximately 10% more efficient in TRU burning than PWRs. After 2085 more TRU can be theoretically burned if MOX3 will be possible to reprocess into a new MOX fuel at that time, however one should be aware it has a negative impact on the americium inventory.

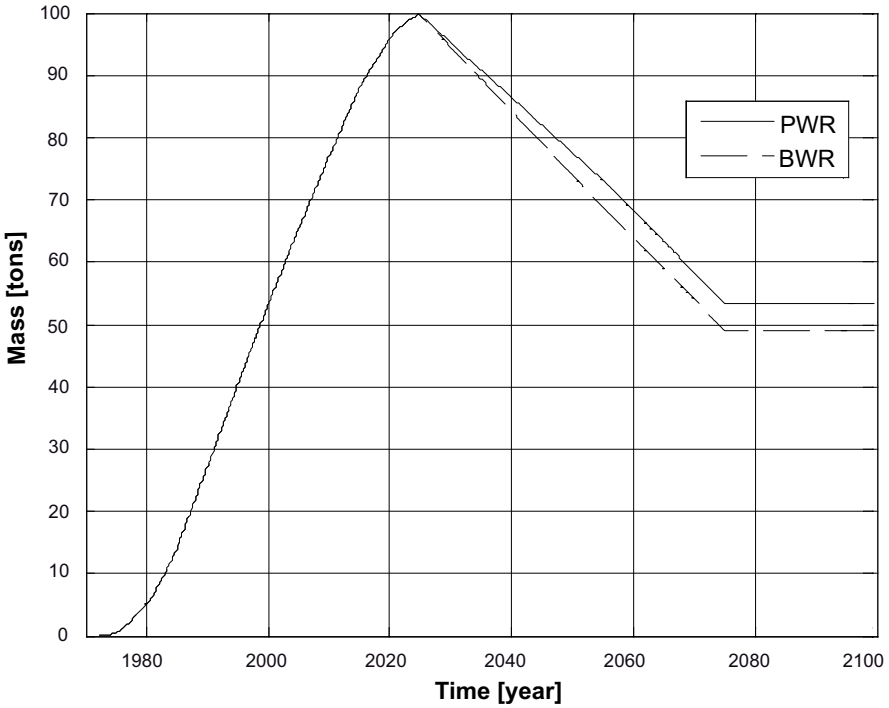

Figure 6-5. *BWR vs. PWR: TRU inventory.*

7 Burning TRU in ADS

In this section we analyze possible burning of TRU in ADS. Chapter 7 describes the ADS unit considered in the calculation.

The assignment raised the question of the possibility to burn 99% of TRU by ADS. We investigated three cases in the following sections. Section 7.2 deals with a limit case when TRU inventory is burned in ADS in the shortest possible time without an international cooperation, meaning an exchange of waste or plutonium. Section 7.3 describes what benefit the international cooperation might have on the TRU inventory. Finally Section 7.4 evaluates a realistic (not limit) case when ADS units operate over 50 years at a constant power.

7.1 Description of ADS model

The fuel for ADS should be fertile free, and be able to accommodate large amounts of minor actinides. An efficient transmutation can be facilitated only with a hard neutron spectrum. Several combinations of fuel matrices and coolant are possible. Nitride, metallic, and oxide fuels have been investigated in /18/. The requirement of fast neutron spectrum implies usage of coolants with low moderating power, such as Na, Pb, Pb/Bi, and He.

We have chosen a novel concept of using hafnium nitride, HfN, as a fuel matrix, and Pb/Bi as a coolant. Hafnium nitride was identified in /18/ as an attractive diluent option for highly reactive transuranium elements. (TRU,Hf)N fuels appeared to have a good combination of neutronic, burnup, and thermal characteristics: maintaining hard neutron spectra, yielding acceptable values of coolant void reactivity and source efficiency, and providing small burnup reactivity loss. An MCNP model of such a core /18/ was used to produce one-group neutron cross-sections for ORIGEN2 by MONTEBURNS 1.0 /6/.

7.2 Transmutation of 99% TRU without an international cooperation

We have evaluated the amount of TRU at 100 tons in 2025 (see Section 5.2). If 99% is to be transmuted then only 1 ton of TRU may stay in the waste. The time needed to transmute 99% of TRU can be easily estimated if one knows how much the TRU inventory is decreased by irradiating in the ADS core, and what time is necessary to cool and reprocess the waste. The ORIGEN2 calculation results in decreasing the TRU inventory by 21% if fuel is irradiated up to 200GWd/t. We assume the standard cooling time together with time needed for reprocessing and fuel fabrication can be approximately 10 years with conventional PUREX procedures, and totally 3 years of cooling with PYRO reprocessing. The whole TRU inventory can be once recycled during a time not shorter than 10 years, or 3 years respectively. In the ideal, limit case, the total TRU inventory can be decreased every 10, or 3 years, respectively, by 21%.

Figure 7-1 shows the TRU inventory over several decades when ADS is utilized inthe most efficient, ideal way. The fuel waste must be recycled 20 times in order to decrease the TRU content down to 1% of the original amount. Thus in an ideal case, 99% of TRU is theoretically possible to burn during time not shorter than 200 years, assuming that the ADS fuel can reach burnup of 200GWd/tHM, and that the waste from ADS must be cooled for 10 years. In case of PYRO processing, it is theoretically possible to burn 99% of TRU during 60 years.

We stress that ADS systems burn smaller amount of TRU at each following time period of 10 or 3 years. It is clear such a scenario would not be economically acceptable since the capacity of the ADS park would be utilized only for 10 or even 3 years and then decreased. In reality ADS systems should run at the full capacity for 50 years to optimize the economic issue, which means that transmuting 99% TRU would in reality take considerably longer time than shown in Figure 7-1. A realistic case without an international cooperation is presented in Section 7.4 and 7.5.

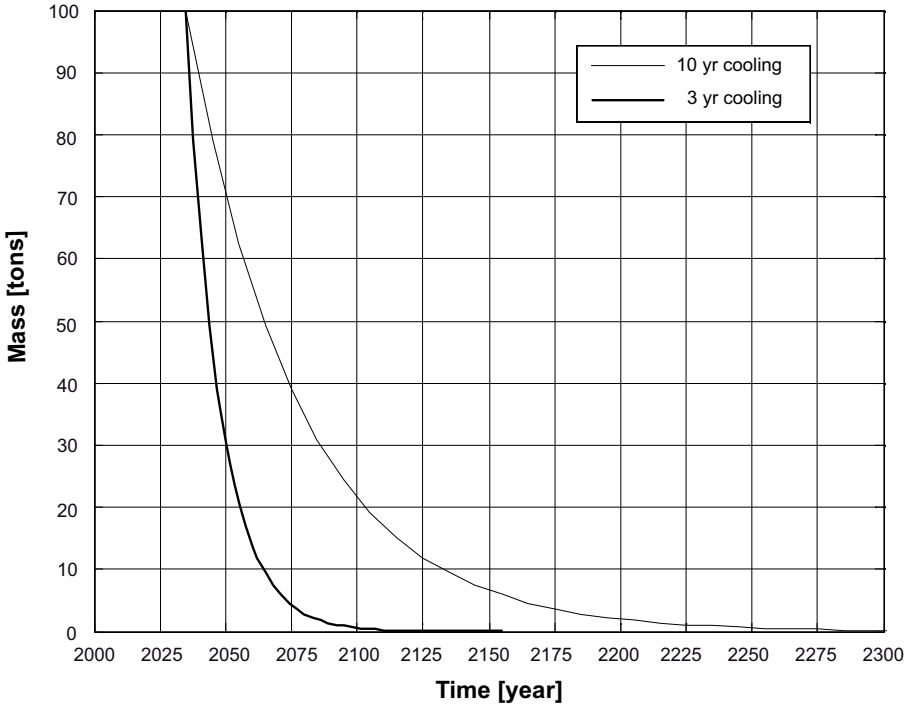

Figure 7-1. *Burning TRU in ADS with 3 and 10 years of fuel recycling.*

7.3 Transmutation of 99% TRU with an international cooperation

We refer to the international cooperation as a situation when we can always produce TRU fuel for ADS until ADS burns 99% of transuranium elements. In this scenario, the TRU fuel is preferably produced from the sufficiently cooled spent nuclear fuel available from the national programme otherwise we use TRU from other programmes, which later is exchanged by an equal amount of TRU from the Swedish programme.

The time needed to burn all 99% of TRU can be relatively short, depending only on the installed thermal power of ADS systems. Supposing that ADS fuel is burned up to 200 GWd/tHM, each installed GW of thermal power will decrease the TRU inventory by 347 kg every year (11 months working, 1 month reloading). If 99% (or even 100%) is to be burned over 50 years (planned lifetime of ADS) then Sweden needs 5.8 GWth of installed thermal power of all ADS systems.

7.4 ADS1 scenario: 2035–2085. Mass flows and cost estimate

An option when several ADS units give a constant power during a planned life time of 50 years is analyzed in this section. Similarly to the MOX case, we also want to set the ADS power in such a way that the TRU inventory in the spent nuclear fuel from the Swedish programme would be maximally eliminated after 50 years. In other words, the power of ADS is set at such a level that the ADS facilities can run for 50 years, and the waste repositories will contain only spent fuel under the cooling regime in the end of ADS operation. This condition gives the total thermal power of the ADS units.

Conditions:

- Power history of Swedish Reactor Park is given by Table 2-1.
- Energy availability factor until 2025: 0.836.
- ADS fuel burnup: 200 GWd/tHM.
- Lifetime of ADS units: 50 years (2035–2085).
- Energy availability factor from 2035 to 2085 (ADS units): 0.918.
- ORIGEN2 library for ADS core: prepared by MONTEBURNS 1.0.

Reprocessing (cooling + fabrication) time: 10 years.

- Total thermal power of ADS units: 3,150MWth.

Reprocessing (cooling + fabrication) time: 3 years.

- Total thermal power of ADS units: 4,450MWth.

Reprocessing (cooling + fabrication): 2 years.

- Total thermal power of ADS units: 4,700MWth.

First we compare the total power production offered by these three options of 2, 3 and 10 years of recycling period with the power production suggested by the scenario with MOX reactors with 3 recycling passes of reactor fuel. To this end, we define the thermal efficiency, ε, the thermal power, P_{th} and the total operational time, T. We label these quantities with a subscript i to distinguish between individual power units. The total electric energy can then be estimated as

$$W_{el} = EAF \cdot \sum_i \varepsilon_i P_{th,i} T_i \qquad (24)$$

We set a value ε = 33%, usual for LWRs, T = 50 years in the current scenario that gives results listed in Table 7-1.

We proceed further with cost estimates of fuel fabrication and reprocessing by defining first the unit costs for fuel fabrication and reprocessing /16, 17/.

Table 7-1. Unit costs for ADS and MOX fuel processing.

MOX_{fab} = 1,100 $/kgHM	ADS_{fab} = 7,000 $/kgHM
MOX_{rep} = 800 $/kgHM	ADS_{rep} = 11,000 $/kgHM

The total mass of the fabricated fuel, M_{fab}, may be estimated using the relationship between the thermal power, P_{th}, operational time, T_{op}, and burnup, B:

$$W_{th} = EAF \cdot P_{th} \cdot T_{op} = B \cdot M_{fab} \Rightarrow M_{fab} = \frac{EAF \cdot P_{th} \cdot T_{op}}{B} \qquad (25)$$

It is somewhat more involved to evaluate the mass of the reprocessed fuel. The situation is visually illustrated in Figure 7-2.

Then the fraction of the reprocessed fuel, η_{rep}, and correspondingly the fraction of the left fuel, η_{left}, may be evaluated as

$$\eta_{rep} = \frac{T_{rep}}{T_{op}}; \quad \eta_{left} = \frac{T_{cool}}{T_{op}} \qquad (26)$$

Thus the masses of the reprocessed and left (unprocessed) fuel are accordingly given by

$$M_{rep} = \eta_{rep} M_{fab} \qquad M_{left} = \eta_{left} M_{fab} \qquad (27)$$

This allows us to evaluated the cost of ADS fuel manufacturing and reprocessing as

$$Cost_{ADS} = ADS_{fab} \cdot M_{fab} + ADS_{rep} \cdot M_{rep} \qquad (28)$$

We estimate the cost of running MOX reactors in essentially the same way summarizing results in Table 7-2 where the abbreviation GSEK stands for 10^9 SEK.

Now we proceed to the evaluation of the transuranium LILW to be stored as a result of losses during reprocessing and fabrication. To this end, we first present a mass flow diagram corresponding to Partitioning and Transmutation (P and T) process. It is shown in Figure 7-3.

Here we follow the notation and values adopted in /9/:

- r = 0.002 is the loss factor for reprocessing.
- δ = 0.0001 is the loss factor for fuel fabrication.
- $\varepsilon = 10^{-8}$ is the transmutation efficiency.
- α = 0.001 is the loss factor for separation.

The transmutation efficiency, ε, may be estimated for relatively short times as

$$\varepsilon N = -\frac{dN}{dt} = \bar{\sigma}\bar{\phi}N \Rightarrow \varepsilon = \bar{\sigma}\bar{\phi} \tag{29}$$

One estimates the core mass using the relationship

$$W_{th} = P_{th} \cdot T_{FC} = B \cdot M_{core} \Rightarrow M_{core} = \frac{P_{th} \cdot T_{FC}}{B} \tag{30}$$

Here, T_{FC} is the Fuel Cycle (FC) duration, T_{FC} = 2 years for ADS and, T_{FC} = 4 years for LWR and MOX reactors; B is burnup; M_{core} is the mass of heavy metals in the core. As seen from Figure 7-3, the mass of the core is given by

$$M_{core} = (1-\delta)(1-r)A + (1-\alpha)(1-\varepsilon)(1-\delta)(1-r)A \tag{31}$$

The total mass loss per fuel cycle is evaluated by summing up partial losses:

$$Loss(T_{FC}) = rA + \delta(1-r)A + \alpha(1-\varepsilon)(1-r)(1-\delta)A \tag{32}$$

Finally, the total mass loss over time T is found as

$$Loss(T) = Loss(T_{FC})\frac{T}{T_{FC}} \tag{33}$$

Evaluation results as applied to the transuranium elements are summarized in Table 7-2.

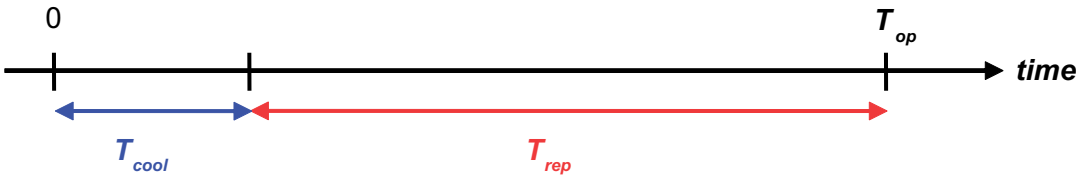

Figure 7-2. *Evaluation of reprocessed fuel mass.*

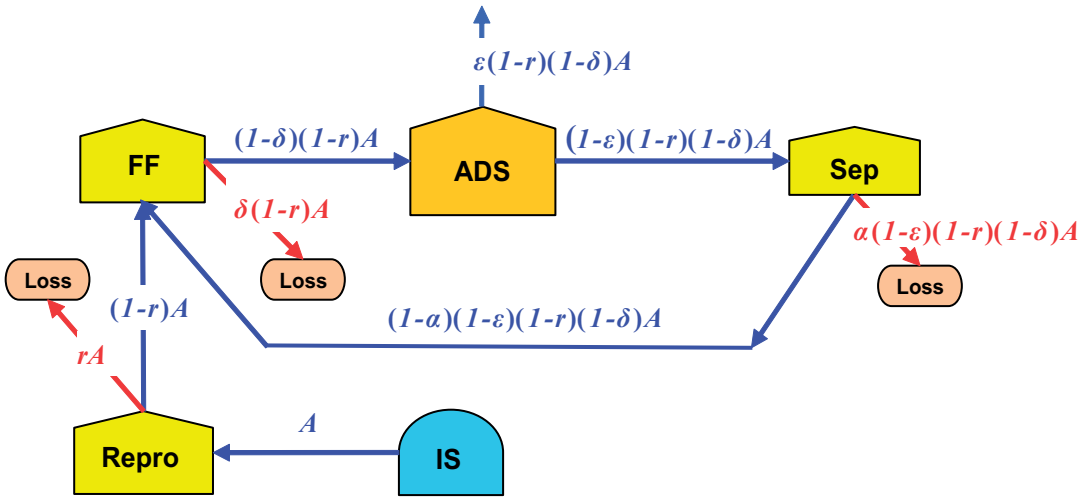

Figure 7-3. *Partitioning and Transmutation process.*

Table 7-2. Power production and fuel costs for three ADS options and 3 pass MOX reactor.

	ADS 2 years	ADS 3 years	ADS 10 years	MOX 3 passes
Total Pth (MW)	4,700	4,450	3,150	4,600
Total Wel (TWh)	642	607	430	556
Fuel Cost (GSEK)	49.8	46.5	30.0	25.0
Fuel Cost/year (GSEK)	1.00	0.93	0.60	0.50
Fuel left (tHM)	14	20	48	42
TRU-LILW (kg)	666	631	447	3,261

We conclude here briefly that the total electricity produced while incinerating Pu and minor actinides in this scenario may be estimated to 600 TWh for the advanced reprocessing technology of 2–3 years as well as for the MOX fuel with 3 recycling passes whereas the conventional cooling time of 10 years gives only 400 TWh.

Efficient burning of TRU in ADS requires the ADS waste to be reprocessed several times into a new ADS fuel. An ADS fuel cycle is therefore dependent on the cooling time of the spent fuel. If the cooling time is too long, e.g. 10 years, then the ADS will produce an amount of waste which can not be reprocessed during the long cooling period. Thus, long cooling periods decrease the efficiency of ADS.

The calculation was performed by the FCA code. The following figures depict the results, and compare them to the MOX case.

Figure 7-4 shows that if the ADS waste is cooled for at least 2 years then the plutonium inventory can be reduced from 78.3 tons (in 2035) to 16.0 tons, which gives a better performance than burning Pu in the MOX fuel.

As contrast to Figure 7-4 where the plutonium mass inventory is shown, Figure 7-5 gives the plutonium radiotoxicity for the same scenarios, i.e. 3 pass MOX fuel as well as ADS fuels with 2, 3 and 10 years of recycling period.

It should be noted here that both Figure 7-4 and Figure 7-5 shows a decrease of the plutonium inventory, correspondingly radiotoxicity, between 2025 through 2035 when there is no ADS facility running according to the scenarios in consideration. This is mostly due to the decay of ^{241}Pu to ^{241}Am with a half life of about 14 years. Much lesser contribution to this process comes from the decay of ^{238}Pu to ^{234}U with a half life of around 88 years.

At the first glance, Figure 7-5 is in a sharp disagreement with Figure 7-4. The explanation rests on the fact that ADS power plants generate in appreciable quantities various plutonium isotopes as by-products out of which ^{238}Pu is extremely radiotoxic, it has the dose factor of two orders of magnitude higher than that of ^{241}Pu, cf Figure 6-2.

Figure 7-6 shows the americium inventory which is always reduced in ADS whereas is increased by burning MOX fuel. The thermal reactors (BWR and PWR) change a considerable amount of plutonium into americium, which is partially the reason why BWRs can so efficiently decrease the plutonium inventory using the MOX fuel (see Figure 6-1 and Figure 6-3).

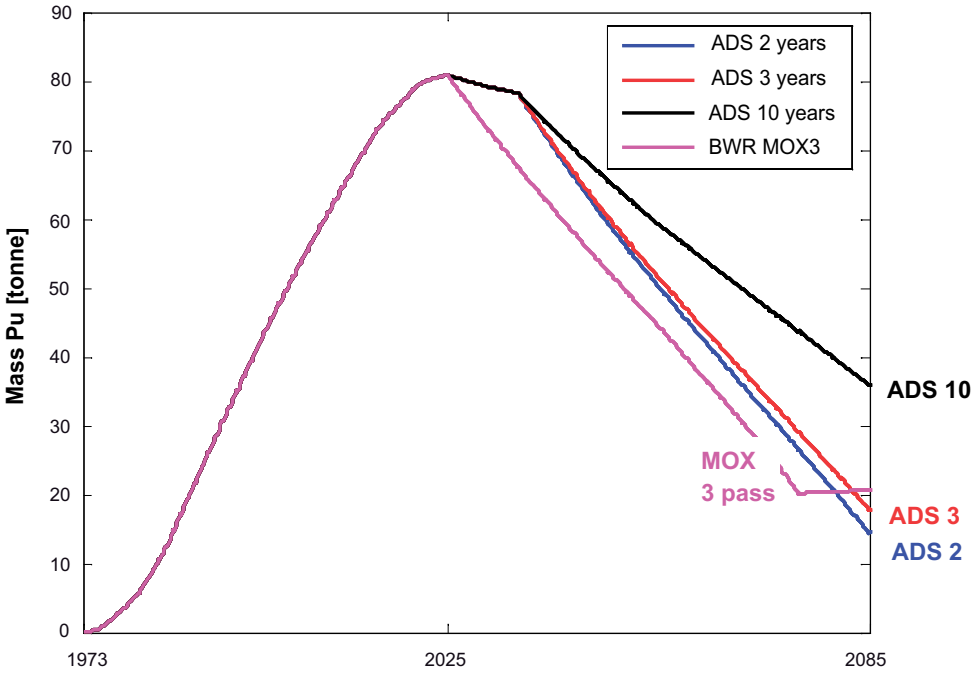

Figure 7-4. *Burning Pu: 3 pass MOX reactor vs. ADS units with 2, 3 and 10 years of fuel recycling.*

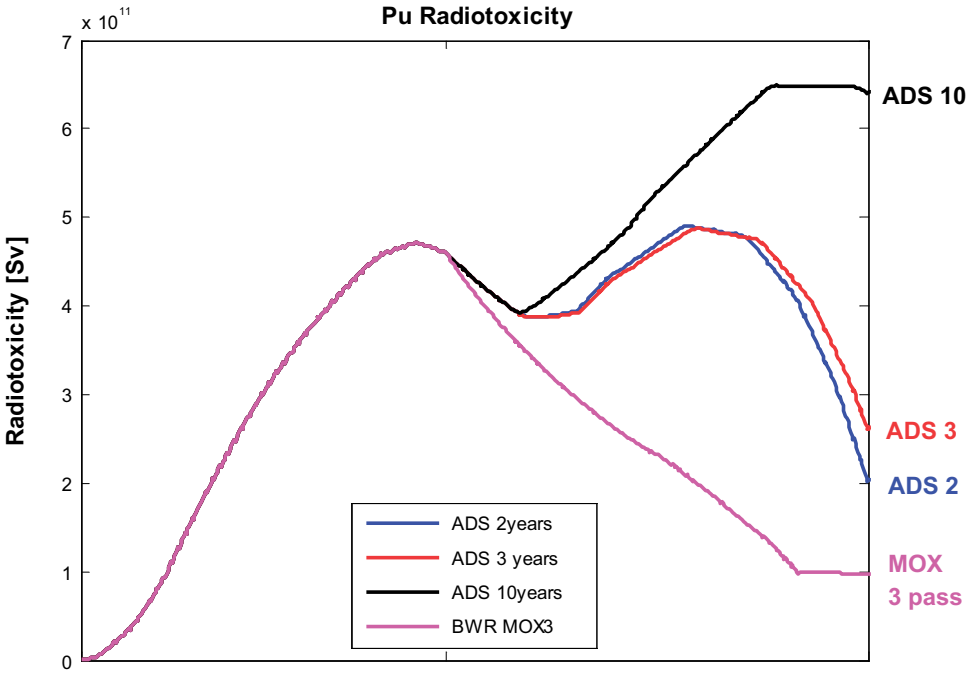

Figure 7-5. *Total Pu Radiotoxicity in case of 3 pass MOX reactor and 3 ADS units with 2, 3 and test on 10 years of fuel recycling.*

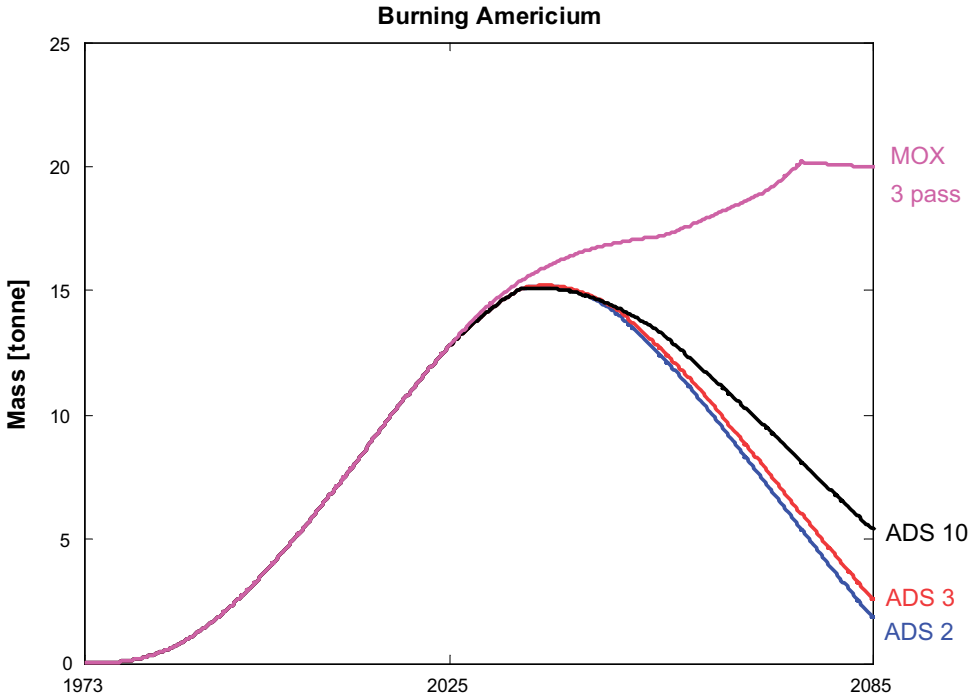

Figure 7-6. *Burning Am: 3 pass MOX reactor vs. ADS units with 2, 3 and 10 years of fuel recycling.*

One can conclude from Figure 7-7 that the ADS power plants are efficient regarding decrease of the Am radiotoxicity namely by a factor of 2 in case of ADS with 10 years of cooling time compared to the three pass MOX-cycle. This efficiency increases further when we come to shorter cooling times, for example an ADS burner with 2 years of cooling time is more than by order of magnitude more efficient than a three pass MOX reactor.

Figure 7-8 shows that if the ADS waste is cooled at most 2 years then the TRU inventory can be decreased from 100 tons to approximately 20 tons during 50 years in ADS units of constant thermal power of 4,700 MW. Moreover the remaining TRU inventory can still be reduced later with smaller size ADS afterwards (see Section 7.5).

Finally, we evaluate here the mount of waste as a result of reprocessing in the glass form to be disposed. In doing so, we rely on the concept of the universal canister loading criterion as given in /14/. It suggests slightly more than 400 kg of glass and waste mass per one Waste Package (WP) out of which at most 40 kg is allowed to be net waste (FP+Actinides). Let m_W denote the mass of waste in a waste package and m_{WG} stand for the mass of waste and glass in a waste package; in what follows, we will set m_{WG} = 400 kg/WP in compliance with /14/. Then the ratio r_{WG} obeys

$$r_{WG} \equiv \frac{m_{WG}}{m_W} = \frac{Waste + Glass}{Waste} \geq 10 \qquad (34)$$

The most important limitation proposed in /14/ is thermal power generation per one waste package hereafter denoted as H_{max}. Depending on the specific construction, it varies 2,000 to 2,300 watts per waste package (W/WP). We will be using a value of H_{max} = 2,000 (W/WP) in our study.

To evaluate the thermal power decay rate, we will need decay modes and Q values of the transuranium elements found in the waste in appreciable quantities. They are summarized in Table 7-3.

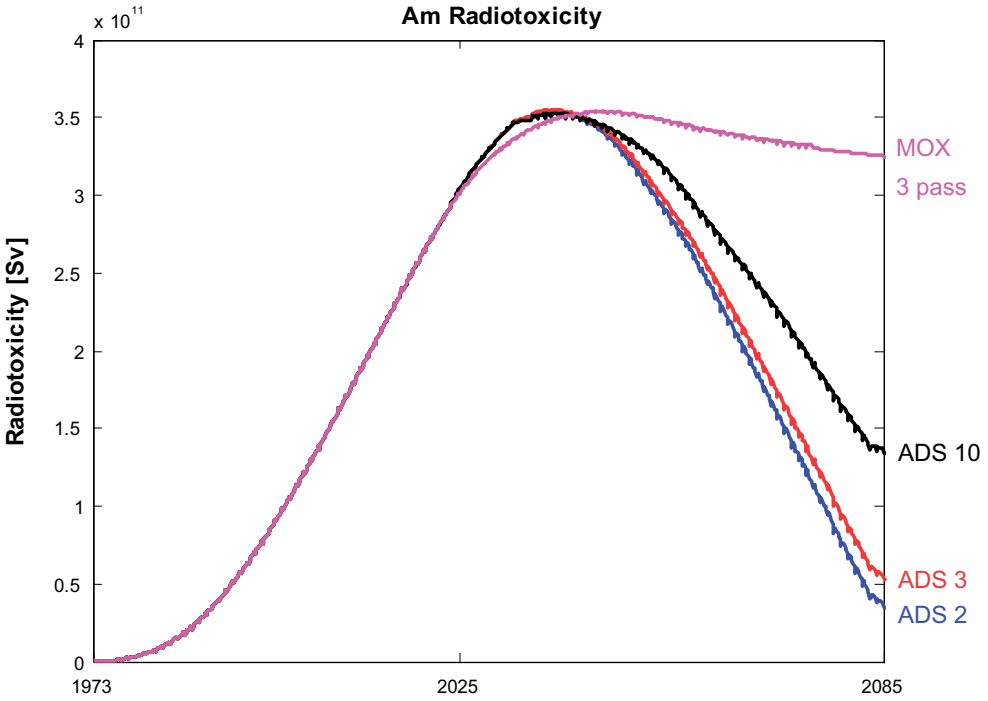

Figure 7-7. *Total Am radiotoxicity in case of 3 pass MOX reactor and 3 ADS units with 2, 3 and 10 years of fuel recycling.*

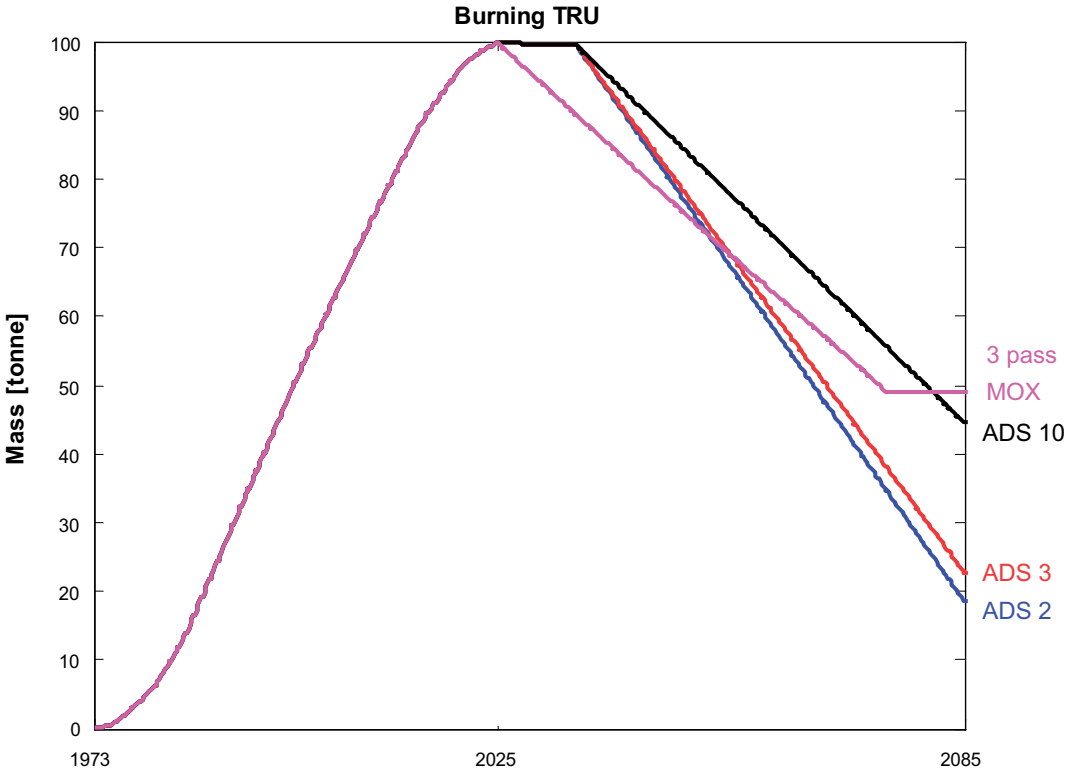

Figure 7-8. *Burning TRU: 3 pass MOX reactor vs. ADS units with 2, 3 and 10 years of fuel recycling.*

Table 7-3. Decay mode, branching and Q values for TRU and some FP.

Nuclide	Dec. Const. λ (s-1)	Decay Mode	Branching (%)	Q Values (KeV)
^{237}Np	1.0264×10^{-14}	α	100	4,958.30
^{238}Pu	2.5046×10^{-10}	α	100	5,593.03
^{239}Pu	9.1103×10^{-13}	α	100	5,244.43
^{240}Pu	3.3468×10^{-12}	α	100	5,255.82
^{242}Pu	5.8809×10^{-14}	α	100	4,984.40
^{241}Am	5.0763×10^{-11}	α	100	5,637.81
^{242m}Am	1.5578×10^{-10}	IT; α	99.55; 0.45	48.63; 5,635.3
^{243}Am	2.9803×10^{-12}	α	100	5,438.70
^{242}Cm	4.9236×10^{-8}	α	100	6,215.56
^{244}Cm	1.2135×10^{-9}	α	100	5,901.74
^{245}Cm	2.5841×10^{-12}	α	100	5,623.50
^{246}Cm	4.6438×10^{-12}	α	100	5,474.80
^{90}Sr	7.6003×10^{-10}	β–	100	546.00
^{137}Cs	7.3140×10^{-10}	β–	100	1,175.60

Radioactivity in Bq of a particular isotope *i* is deduced from (22) as

$$RA_i(t) = \lambda_i N_i = \lambda_i M_i(t)\frac{N_A}{A}10^3 \quad [\mathrm{Bq}] \tag{35}$$

Here, *t* in the parentheses denotes the time after discharge. To account for the cooling time, T_c, which is assumed to be 50 years, Assuming this isotope *i* to decay through several channels with branching ratios, b_i, and energy, Q_i, one evaluates the total energy deposited in the surroundings as

$$H = \sum_{i=1}^{Isotopes} RA_i \sum_{c=1}^{Channels} b_c Q_c \tag{36}$$

First we calculate the final, i.e. in 2085, transuranium inventory in these four scenarios. Figure 7-9 shows nuclides with considerable (more than 10 kg) mass.

Next, we calculate individual and total heat rates using . They are summarized in Table 7-4.

Figure 7-10 shows these rates visually.

Table 7-4. Mass inventory and individual heat generation in various scenarios.

	MOX 3 pass		ADS 10 years		ADS 3 years		ADS 2 years	
Nuclid	Mass (t)	Heat (kW)	Mass (t)	Heat (kW)	Mass (t)	Heat (kW)	Mass (t)	Heat (kW)
^{237}Np	7.183	0.149	2.180	0.045	1.132	0.023	0.917	0.019
^{238}Pu	0.333	18.934	2.781	1,579.100	1.088	617.730	0.834	473.440
^{239}Pu	3.195	6.162	11.773	22.706	2.693	5.194	1.269	2.447
^{240}Pu	10.535	74.522	14.802	104.700	10.941	77.393	9.730	68.826
^{242}Pu	6.340	0.738	5.522	0.642	4.250	0.495	3.816	0.444
^{241}Am	11.313	1,297.700	3.596	412.450	1.584	18.164	1.008	11.561
^{242m}Am	0.038	0.175	0.394	1.806	0.139	0.637	0.111	0.510
^{243}Am	7.629	49.096	1.010	6.499	0.726	4.672	0.644	4.141
^{243}Cm	0.002	0.000	0.004	0.000	0.001	0.000	0.001	0.000
^{244}Cm	0.145	411.820	0.095	268.410	0.110	312.800	0.118	333.030
^{245}Cm	0.146	0.837	0.168	0.963	0.278	1.592	0.282	1.611
^{246}Cm	0.014	0.140	0.022	0.214	0.058	0.573	0.007	0.664
Total	46.877	2,030.600	42.348	2,397.500	23.004	1,202.700	18.801	1,000.700

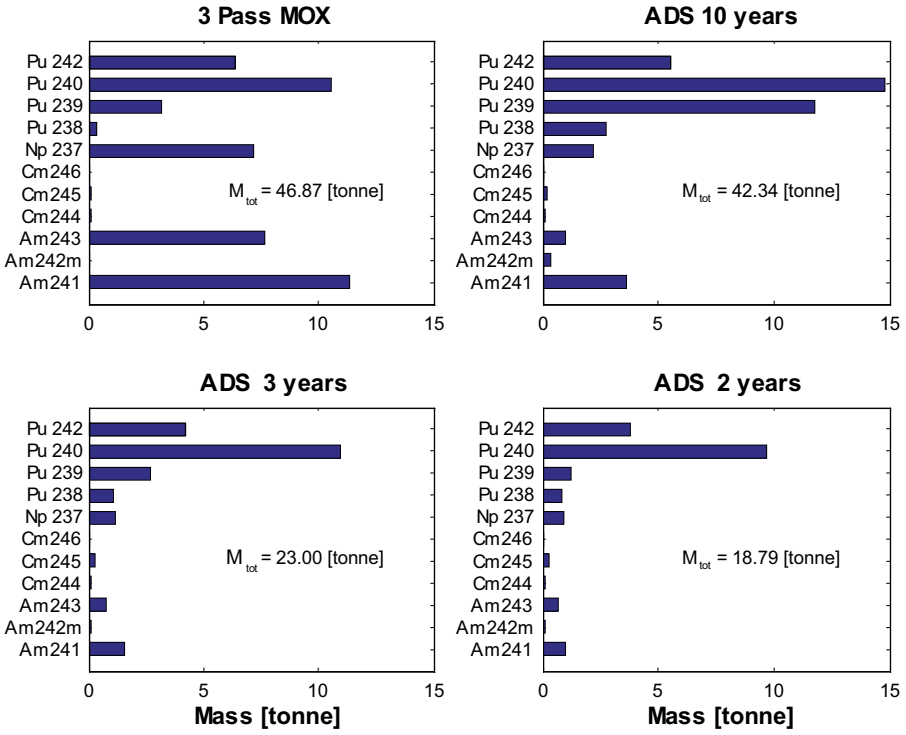

Figure 7-9. *TRU inventory accumulated in various scenario by 2085 and cooled 50 years.*

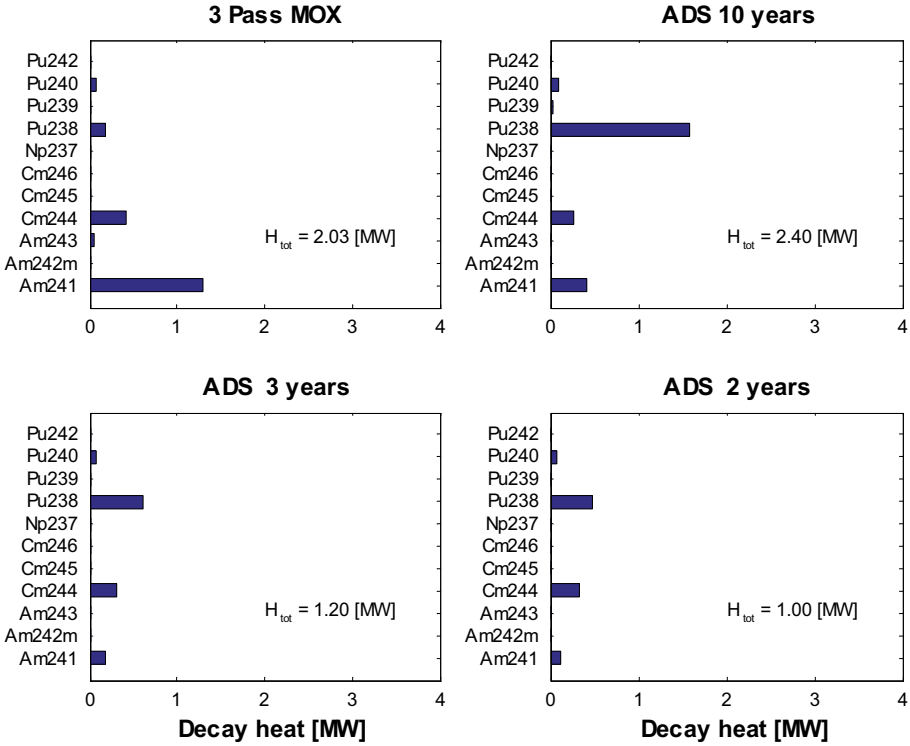

Figure 7-10. *Individual and total heat rates of TRU in various scenarios.*

One can estimate the allowable mass of TRU waste, m_W, per one waste package by first evaluating the specific heat generating rate in a scenario

$$h_{sp} \equiv H_{tot} / M_{\mathrm{TRU}} \tag{37}$$

and then equating

$$m_{WP} \cdot h_{sp} = H_{max} \Rightarrow m_W = H_{max} / h_{sp} = M_{\mathrm{TRU}} \cdot H_{max} / H_{tot} \tag{38}$$

Here, H_{max} is the maximal allowable decay heat generation per one waste package. The value for this quantity is set H_{max} = 2,000 W/WP /14/. Results of these calculations are summarized in Table 7-5.

As seen in Table 7-5, only MOX 3 scenario allows waste packages with more than 40 kg of TRU waste. In the three other scenarios we have to reduce the amount of TRU waste per one package down to 35–37 kg. However, the above calculations completely disregard the thermal energy deposited in the environment by fission/activation products.

Unfortunately, the FCA code does not calculate fission/activation products. To overcome this difficulty, we assume a linear in burnup dependence of the fission products. This assumption has reportedly turned out to be fairly accurate.

The most important heat generating fission products are ^{90}Sr and ^{137}Cs. Calculations give the amount of ^{90}Sr accumulated per 1 tU in a LWR reactor of 37 GWd/tU to be about 0.141 kg. This evaluates to m (^{90}Sr) = 0.0038 kg/GWd, accordingly, one estimates m (^{137}Cs) = 0.0090 kg/GWd. This allows us to evaluate the total masses of ^{90}Sr and ^{137}Cs in various scenarios using the total amount of energy produced

$$E_{th} = EAF \cdot P_{th} \cdot T_{op} \tag{39}$$

Here, EAF is Energy Availability Factor set to either 0.836 or 0.918 in the MOX or ADS scenarios respectively; T_{op} is time of operation assumed to be 50 years in each scenario.

These masses are corrected by decay to take into account the cooling phase, which is assumed to be 50 years. Next, one calculates the radioactivities and finally evaluates the decay energy using Q values from Table 7-3. The decay energy due to ^{90}Sr and ^{137}Sc should be added to that produced by TRU waste. Numerical results are given in Table 7-6.

Table 7-5. Allowable mass of TRU waste per 1 WP ignoring heat from FP.

	MOX 3 pass	ADS 10 years	ADS 3 years	ADS 2 years
M_{TRU} (t)	46.9	42.3	23.0	18.8
H_{tot} (MW)	2.03	2.40	1.20	1.00
h_{sp} (W/kg)	43.3	56.6	52.3	53.3
m_W (kg)	46.2	35.3	38.2	37.6

Table 7-6. Allowable mass of TRU waste per 1 WP with ^{90}Sr and ^{137}Cs accounted.

	MOX 3 pass	ADS 10 years	ADS 3 years	ADS 2 years
P_{th} (MW)	4,600	3,150	4,450	4,700
E_{tot} (GWd)	70,230	52,810	74,604	78,795
M(^{90}Sr) (kg)	80.7	60.7	85.7	90.5
M(^{137}Cs) (kg)	198.7	149.4	211.1	223.0
Q(^{90}Sr) (kW)	35.9	27.0	38.1	40.3
Q(^{137}Cs) (kW)	183.1	137.8	194.6	205.5
H_{tot} (kW)	2,249.7	2,562.2	1,435.4	1,246.5
m_W (kg/WP)	41.7	33.1	32.1	30.2

Once again, as seen in Table 7-6 only MOX 3 scenario allows waste packages with more than 40 kg of TRU waste. In the three other scenarios we have to reduce the amount of TRU waste per one package down to around $m_W = 33$ kg in ADS10, $m_W = 34$ kg in ADS3 and $m_W = 32$ kg in ADS2 scenarios. This means we may use the ratio, $r_{WG} \equiv$ (Waste+Glass)/ Waste, of 10:1 only in case of MOX3 scenario whereas for other scenarios we have to correct the ratio as $r_{WG} = 400/33.8 = 11.8$ for ADS10; $r_{WG} = 400/34.0 = 11.8$ for ADS3 and $r_{WG} = 400/32.5 = 12.3$ for ADS2 scenarios.

The number canister, N_{can}, is evaluated as

$$N_{can} = \frac{M_{TRU}}{m_W} \tag{40}$$

Finally, the mass of the vitrified waste, M_{WG}, is estimated by

$$M_{WG} = m_{WG} \cdot N_{can} = \frac{m_{WG}}{m_W} \cdot M_{TRU} = r_{WG} \cdot M_{TRU} \tag{41}$$

We conclude this section by calculating the area needed to store the TRU waste in universal canisters. The most common demand is a minimal distance between canisters, which varies 3 to 5 metres according to different technical specifications. For a rough estimate we assume a minimal distance of 4 metres; this results in an area of 16 square metres per a canister. The number of canisters is evaluated dividing the amount of TRU waste by the weight allowed per one canister, which was stated earlier to be of 400 kg/WP.

Table 7-7. Amount of TRU vitrified waste to be disposed in ADS1 scenario.

	MOX 3	ADS 10	ADS 3	ADS 2
M_{TRU} (t)	46.9	42.3	23.0	18.8
r_{WG}	10.0	12.1	12.5	13.3
M_{WG} (t)	469	512	287	249
N of canisters	1,170	1,280	720	625
Area (ha)	1.88	2.05	1.15	1.00

7.5 ADS-R scenario: 2035–2135. Mass flows and cost estimate

7.5.1 Technical description

A realistic scenario utilizing ADS over the period 2035–2135 is proposed in this section. The objective is to maximally decrease the TRU inventory under the condition that all ADS units work without interruption with operational times as close to 50 years as possible. We propose 6 ADS units totally out of which 5 units have a thermal power of 890 MW each and one unit has 800 MW. The first 5 ADS units will start in 2035 out of which one is shut down in 2077 (42 years of operation), another one is disconnected in 2083 (48 years of operation), 2 units are shut down in 2085 (50 years of operation), and finally the oldest one retires in 2107 (72 years of operation). The 6th ADS unit is put into operation in 2090 and is taken out of operation in 2135 (45 years). This scenario is not unique; it is possible to find different scenarios reaching approximately the same efficiency.

It should be noted that despite neptunium is a long-lived isotope ($T_{½} = 2.144\times10^{6}$ years) abundant in the spent fuel, it is not included in the current transmutation calculation model because of very low radiotoxicity of neptunium – see Figure 5-15. The current studies on ADS fuel do not suggest neptunium separation from the spent fuel as being economically not justified. However, should any further investigation reveal such a necessity for some reason, neptunium may easily be included into calculations. Moreover, it should be noted that neptunium transmutation would have a similar pace as destruction of americium in ADS-systems studied in this paper.

Conditions:

- Power history of Swedish Reactor Park is given by Table 2-1.
- Energy availability factor until 2025: 0.836.
- ADS fuel burnup: 200 GWd/tHM.
- Energy availability factor from 2035 to 2135 (ADS units): 0.918.
- ORIGEN2 library for ADS core: FFTFC.LIB.
- Recycling (cooling + fabrication) time for ADS waste: 3 years.
- Thermal power of ADS unit #1 (2035–2077): 890 MW.
- Thermal power of ADS unit #2 (2035–2083): 890 MW.
- Thermal power of ADS unit #3 (2035–2085): 890 MW.
- Thermal power of ADS unit #4 (2035–2085): 890 MW.
- Thermal power of ADS unit #5 (2035–2107): 890 MW.
- Thermal power of ADS unit #6 (2090–2135): 800 MW.

The proposed deployment of ADS units is visually shown in Figure 7-11.

7.5.2 Efficiency of converting thermal to electric power

We can assume thermal efficiency of ADS units, ε_{ADS}, to be approximately 40% due to higher coolant temperature. However, a certain fraction of electric power produced by ADS has to be returned back to feed ADS accelerator. The proton beam power is related to the core power as

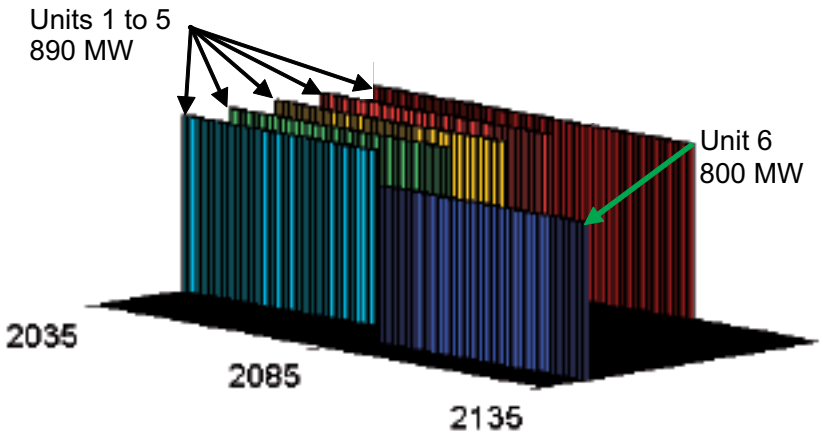

Figure 7-11. *Deployment of ADS units in ADS-R scenario, years 2035 through 2135.*

$$P_{Beam} = \frac{1}{G} P_{ADS,th} \tag{42}$$

Here, G is the gain factor, which in turn may be evaluated as /21/

$$G \equiv \frac{P_{ADS,th}}{P_{Beam}} = \frac{k_{eff}}{1-k_{eff}} \frac{\varepsilon_f}{\nu} \frac{\psi^*}{E_p} \tag{43}$$

We have made here use of the following notation as well as values recommended in /21/:

- $k_{eff} = 0.95$, effective multiplication factor.
- $\varepsilon_{f=}$ 187 MeV, average recoverable energy released per a fission.
- $\nu = 2.97$, average number of secondary neutrons per a fission.
- E_p, proton energy.
- ψ^*, proton source efficiency.

Typical values of the gain factors have been calculated in /21/:

Table 7-8. Gain factors for various proton energies.

Proton Energy, Ep	Proton Source Efficiency, ψ*	Gain Factor, G.
400	8.1	25
600	16.8	35
800	25.8	41
1,000	35.5	45
1,400	50.9	46
2,000	69.8	44

Taking the most anticipated proton energy of about 1 GeV, we may thus assume $G = 45$. Next, we calculate the electric power needed to feed the accelerator

$$P_{Acc,feed} = \frac{1}{\varepsilon_{Acc}} P_{Beam} = \frac{1}{\varepsilon_{Acc} G} P_{ADS,th} \quad (44)$$

Combining this with , we estimate the net electric power produced by ADS as

$$P_{ADS,net} = \varepsilon_{ADS} P_{ADS,th} - P_{Acc,feed} = \varepsilon_{ADS} P_{ADS,th} - \frac{1}{\varepsilon_{Acc} G} P_{ADS,th} \quad (45)$$

Finally, we estimate the net efficiency of converting thermal to electric power, ε_{net}, as

$$\varepsilon_{net} = \frac{P_{ADS,net}}{P_{ADS,th}} = \varepsilon_{ADS} - \frac{1}{\varepsilon_{Acc} G} \quad (46)$$

Assuming the accelerator thermal efficiency to be ε_{Acc} = 40%, we obtain finally $\varepsilon_{eff} \approx 34\%$.

7.5.3 Electricity production

We evaluate the electrical power produced in this scenario through multiplying the power rate of an ADS unit by the corresponding operational time followed by summation of individual energy contribution. This procedure yields

$$W_{el} = EAF \cdot \sum_i \varepsilon_i P_{th,i} T_{op,i} \approx 736 \text{ TWh} \quad (47)$$

The above technical data is summarized in Table 7-9.

In addition, we evaluate average power production over the period of 2035 through 2135

$$P_{el,ave} = \frac{W_{el}}{Time} = \frac{736TWh}{100years} \approx 840MW \quad (48)$$

The distribution of thermal power production in this scenario is shown in Figure 7-12. The bulk of the thermal power production of about 4.5 GW_{th} falls years 2035 through 2077, then it changes (mostly decreases) stepwise until 2135.

Table 7-9. Technical data for scenario ADS-R (3 years of cooling).

Unit	Pth (MW)	Net eff (%)	Start (year)	Stop (year)	Top (year)	Wel (TWh)
Unit-1	890	34%	2035	2077	42	102
Unit-2	890	34%	2035	2083	48	117
Unit-3	890	34%	2035	2085	50	122
Unit-4	890	34%	2035	2085	50	122
Unit-5	890	34%	2035	2107	72	175
Unit-6	800	34%	2090	2135	45	98
Ptot =	5,250		Pel,ave =	840 MW	Wtot =	736

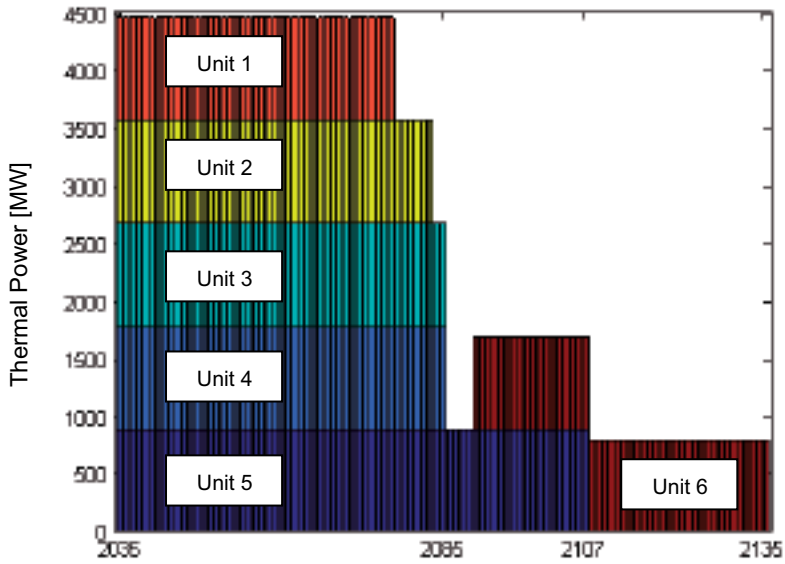

Figure 7-12. *Thermal power distribution in scenario ADS-R, years 2035 through 2135.*

7.5.4 Fuel fabrication cost

Finally, we apply the same cost analysis for the fuel manufacturing as was done for ADS1 scenario in Section 7.4 to yield 61.8 GSEK needed totally in this scenario to fabricate and reprocess nuclear fuel or equivalently 0.62 GSEK yearly. The mass of the fuel left in ADS-R scenario is estimated to be about 27 tonnesHM.

7.5.5 Burning plutonium, americium and curium

Detailed results about the isotopic content obtained with the help of ORIGEN code are presented in the rest of this section. In doing so, we assume the length of the fuel cycle for ADS facilities, T_{fc}, to be 2 years; furthermore, half of the core is replaced with a fresh fuel batch each year. We simulate the ADS fuel production, first, by separating plutonium, americium, and curium isotopes from the sufficiently cooled spent fuel, and second, by fabricating ADS fuel batches with a preferable isotopic composition of Pu:Am:Cm = 50:40:10. Whenever we run out of curium, the percentage of americium is increased up 50% in an extreme case. Moreover, if we suffer a shortage of americium we increase the concentration of plutonium up 100% in the most extreme case. As for the technical details presented in Table 7-9 such as the number ADS units, their thermal power, times of operation, they are the result of the manual optimization to incinerate as much of these heavy metals as possible under a constrain of an uninterruptible operation of each ADS facility. Details of this scenario are shown in the next two pictures. Figure 7-13 gives the distribution of fissile material (Pu, Am, Cm) between ADS cores (dashed purple line), cooling pool (red line), and fissile material stock (ready for fabrication – blue line). The optimal strategy pursued here is to obtain as low amount of these heavy metals as possible in the end of the scenario provided there is always enough sufficiently cooled spent fuel (blue line in Figure 7-13) to fabricate new fuel batches.

We evaluate first the efficiency of this strategy in burning plutonium, americium and curium. Figure 7-13 shows that the amount of plutonium, americium, and curium accumulated in the spent fuel from the Swedish reactor park may be considerably reduced by a factor of 16 from about 99 tonnes in 2035 down to around 6 tonnes in 2135. In addition,

Figure 7-13 demonstrates that americium and curium will be almost completely eliminated in this scenario. Another important observation is that we have to fabricate plutonium rich fuel raising the fraction of plutonium up to 100% towards the end of the scenario.

As a matter of fact, we note here that the amount of plutonium (blue line) decreases slightly between 2025 through 2035 when there is no ADS unit running in the current scenario. This decrease is accompanied by an almost equivalent increase in americium due to β^- decay of ^{241}Pu to ^{241}Am in such a way that the total amount of these heavy metals (black line) remains nearly constant.

Summarizing, the transuranium elements were evaluated to amount up to about 100 tonnes by year 2025 according to calculations by the FCA code. The simulation of ADS-R scenario has shown an appreciable decline of transuranium elements by a factor of around 16, namely down to approximately 6 tonnes.

With the same arguments as in 7.4, we evaluate the amount of the TRU waste to be disposed as the amount of the transuranium elements found in the spent fuel at the end of the current scenario. As seen in Figure 7-14, this amount may be estimated to about 6 tonnes.

We conclude this section with evaluation of the mount of waste as a result of reprocessing in the glass form to be disposed. The general procedure of doing so has already been discussed in detail in Section 7.4.

First we calculate the final, i.e. in year 2135, transuranium inventory in this scenario. It is shown on the left in Figure 7-15 whereas the right half of the figure gives individual heat rates.

Next, we perform essentially the same analysis as we did in 7.4; omitting details, we summarize our calculations in Table 7-10.

The following notation has been used here:

- M_{TRU} is the mass of the transuranium elements evaluated in this scenario.
- r_{WG} is the ration (Waste+Glass)/Waste.
- M_{WG} is the mass of vitrified waste to be disposed.

Table 7-10. Amount of TRU vitrified waste to be disposed in ADS-R scenario.

	M_{TRU} (t)	m_W (kg/WP)	r_{WG}	M_{WG} (t)	N of canisters	Area (ha)
ADS-R:	6.4	22.8	17.5	112	285	0.45

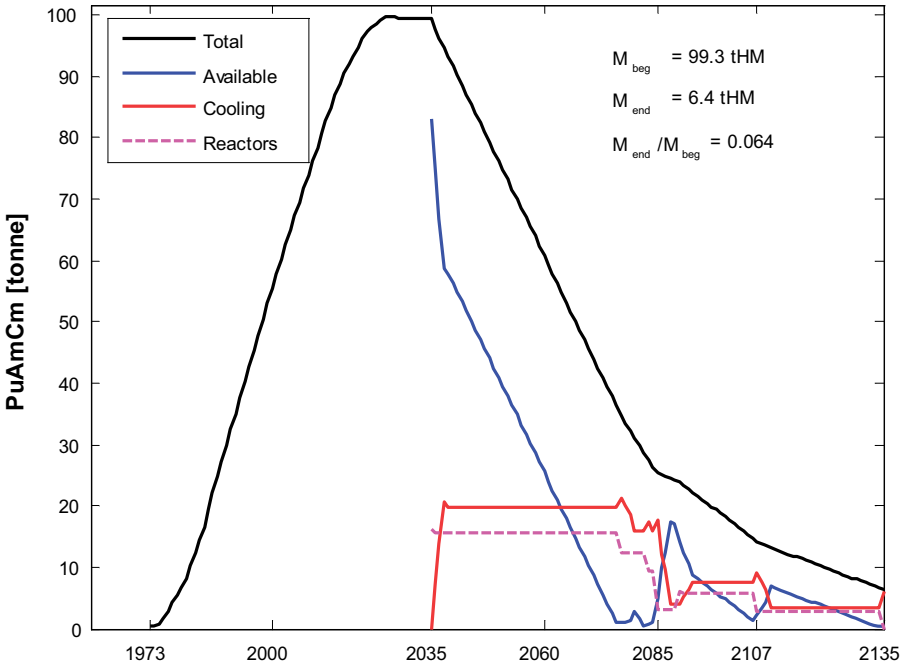

Figure 7-13. *Distribution of fissile material between reactors (broken purple), cooling pool (red), and fissile stock (available – blue) in ADS-R scenario, T_{cool}=3 years.*

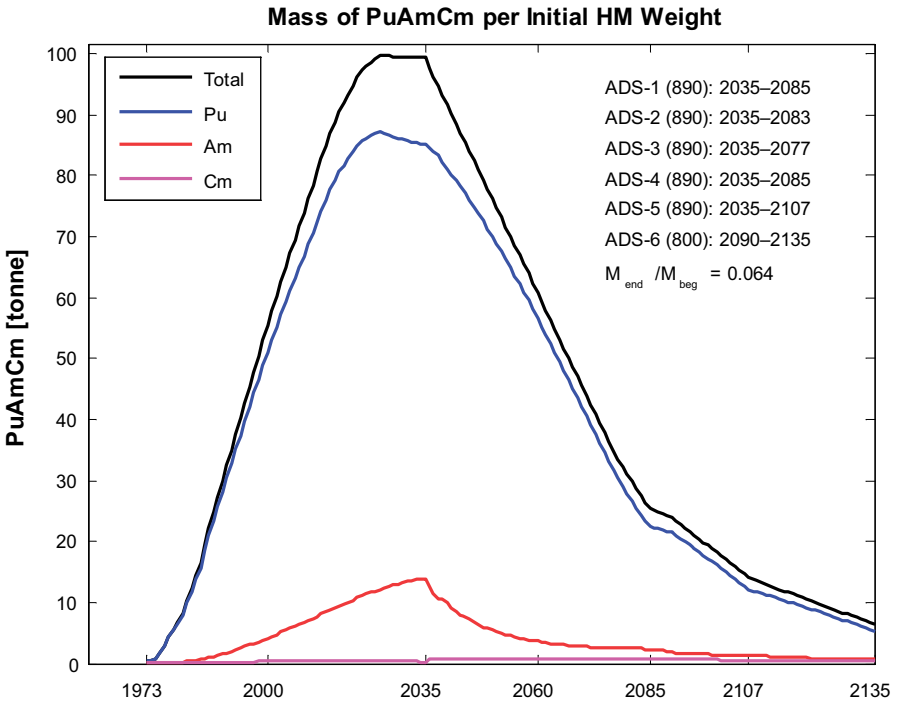

Figure 7-14. *Burning Pu, Am, and Cm in ADS-R scenario, T_{cool} = 3 years.*

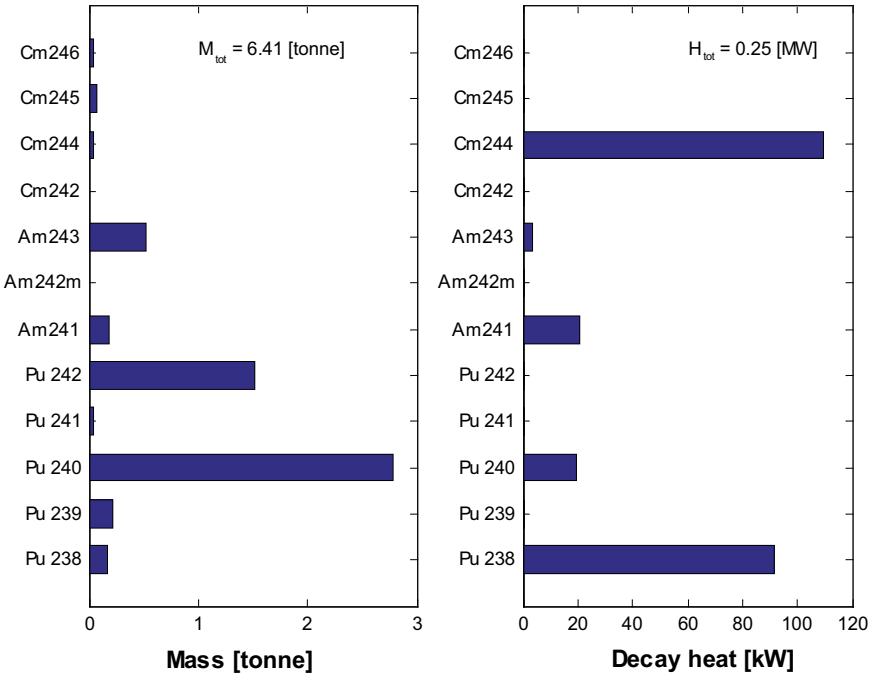

Figure 7-15. *Left: TRU mass inventory accumulated in ADS-R scenario by 2135 and cooled 50 years; Right: individual heat rates.*

7.6 Constructional waste from ADS facilities

Generally speaking, it is very difficult to evaluate the amount of constructional waste from dismantling an ADS facility mainly because the ADS facility concept is still under design. Conceptually, an ADS facility will be represented by two components: Fuel Irradiator and Accelerator. We assume the Fuel Irradiator to be similar to a Fast Reactor (FR). A major uncertainty here lies in the fact that no specific decision has been made regarding the type of fast reactor. Anyhow, we rely strongly on the experience obtained by Electricite de France (EDF) which probably possesses the greatest knowledge in this area.

EDF reports in /10/ the following Low or Intermediate Level Waste (LILW) from decommissioning the eight first FR units, see Table 7-11.

It gives totally 183,400 tonnes of constructional waste or approximately 23,000 tonnes per one unit.

It is much more difficult to evaluate the constructional waste from dismantling an accelerator due to very rare and scattered information. One such example is found in /11/ where the authors summarize the experience of decommissioning the 184-Inch Synchrocyclotron capable of generating 730 MeV protons, see Table 7-12.

All together, it gives slightly less than 9,000 tonnes of constructional waste. No further scaling for proton energy neither for beam power has been made because of large uncertainties underlying this estimate. As a very rough assumption, we conclude that one ADS facility leads to about 23,000+9,000 = 30,000 tonnes of constructional waste in the end of exploitation time. Similarly, realization of ADS-R scenario would result in

$$6\,\text{units} \times 30{,}000 \frac{\text{tonnes}}{\text{unit}} = 180{,}000\ \text{tonnes}$$

of constructional waste.

Table 7-11. Constructional waste from 8 FR units.

	Metallic waste	Graphite	Concrete	Miscellaneous
VLLW	20,000 t		70,000 t	1,800 t
SL-LILW			75,000 t	1,400 t
LL-LILW	200 t	15,000 t		

Table 7-12. Constructional waste from 184-inch synchrocyclotron.

Concrete	Steel and concrete	Cyclotron parts
5,626 t	2,455 t	900 t

Another literature source, /12/, describes the experience of dismantling the 6 GeV CEA-SATURNE synchrotron which resulted in slightly less than 1,200 tonnes of VLLW/ LLW. It leads to 23,000+1,200 = 24,200 tonnes of constructional waste hence giving 6×24,200 ≈ 145,000 tonnes in total. Based on these two examples, we conclude that an estimate of 150,000 tonnes of constructional waste would be reasonable one.

7.7 Impact of reprocessing time in ADS-R scenario

A realistic scenario with a reprocessing (sometimes called cooling) time of 3 years has been analyzed in Section 7.5. It feels that the overall performance of a specific scenario appreciably depends on this parameter. To confirm or reject this hypothesis, we repeat here the previous calculations with a new reprocessing time of 10 years. Once again the objective is to maximally decrease the TRU inventory under the condition that all ADS units work without interruption with operational times as close to 50 years as possible. It has been found than an optimal scenario is characterized by the following technical data, see Table 7-13.

Some of the results are briefly presented in the next two figures. Figure 7-16 gives the distribution of fissile material (Pu, Am, Cm) between ADS cores (dashed purple line), cooling pool (red line), and fissile material stock (ready for fabrication – blue line). The optimal strategy pursed here is to obtain as low amount of these heavy metals as possible in the end of the scenario provided there is always enough sufficiently cooled spent fuel (blue line in Figure 7-16) to fabricate new fuel batches.

Figure 7-16 shows that the amount of plutonium, americium, and curium accumulated in the spent fuel from the Swedish national programme may be reduced only by a factor of 4 as contrast to the factor of 16 in ADS-R scenario with 3 years of reprocessing time.

Table 7-13. Technical data for scenario ADS-R (10 years of cooling).

Unit	P_{th} (MW)	Net eff (%)	Start (year)	Stop (year)	T_{op} (year)	W_{el} (TWh)
Unit-1	650	34%	2035	2085	50	89
Unit-2	650	34%	2035	2085	50	89
Unit-3	700	34%	2035	2085	50	96
Unit-4	900	34%	2035	2103	68	167
Unit-5	900	34%	2055	2135	80	197
Unit-6	450	34%	2085	2135	50	62
Ptot =	4,250		Pel,ave =	798 MW	Wtot =	699

Specifically, the amount of the above mentioned heavy metals will be reduced from about 99 tonnes in 2035 down to around 25 tonnes in 2135. Moreover, we will find a considerable amount of americium (approximately 2 tonnes as seen in Figure 7-17) in the end of the scenario. It stands in a sharp contrast to the previous scenario, ADS-R, with 3 years of reprocessing time (Figure 7-13) when almost all americium and curium is eliminated in the end of the scenario.

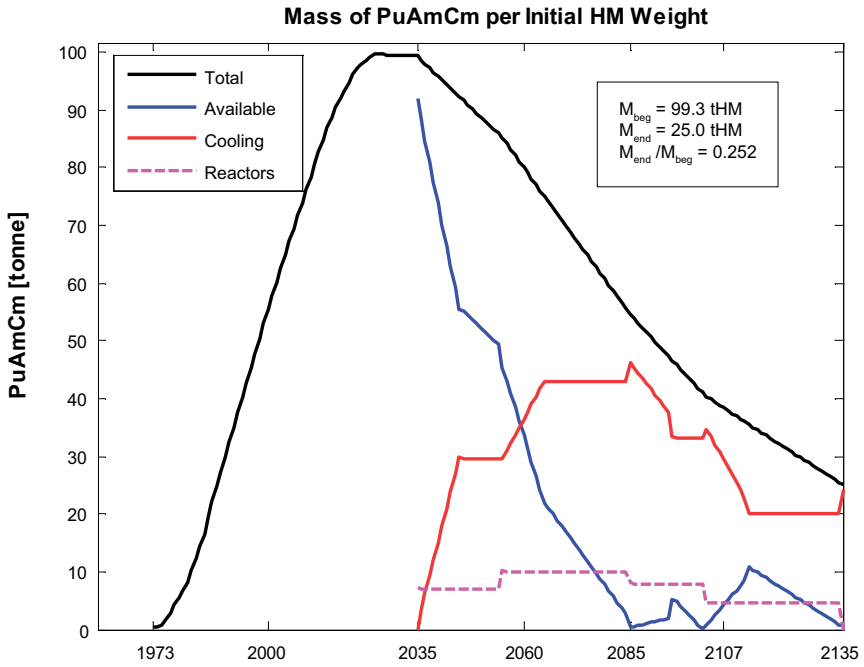

Figure 7-16. *Distribution of fissile material between reactors (broken purple), cooling pool (red), and fissile stock (available blue) in ADS-R scenario, T_{cool} = 10 years.*

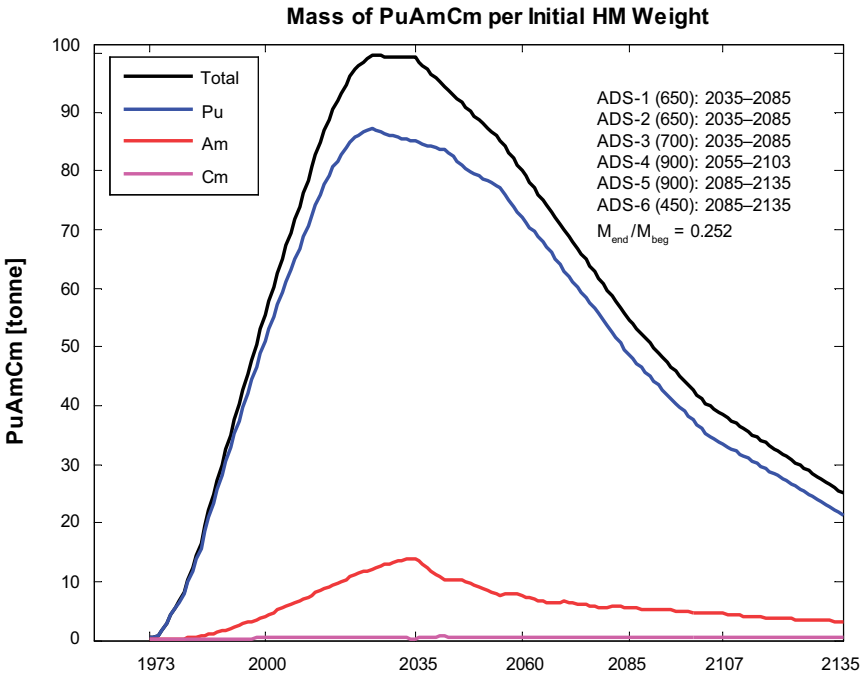

Figure 7-17. *Burning Pu, Am, and Cm in ADS-R scenario, T_{cool} =10 years.*

8 Combined MOX – ADS scenarios

In this section, we evaluate two scenarios combined from MOX and ADS units. The UOX waste is to be recycled into the MOX fuel, and the MOX waste is to be reprocessed into the ADS fuel. The first scenario assumes the MOX1 waste is reprocessed directly into the ADS fuel, whereas the second scenario requires the MOX1 waste to be recycled into the MOX2 fuel which waste is recycled into the ADS fuel.

There are more parameters to be found in these scenarios, like the total thermal power of MOX and ADS units, and the year when the ADS units should start. In order to find the unknown parameters, we pose the optimization requirements. First, the MOX and ADS units run for 50 years at a constant power. The power of MOX and ADS units is set at the highest reachable value to assure the highest possible impact on the TRU inventory. The start of the operation of the MOX reactors is in 2025.

The methodology of finding the unknown parameters is simple. First only MOX reactors are introduced into the scenario. The total power of the MOX reactors is adjusted to the highest possible value which can be assured over 50 years. Then the ADS is introduced into the scenario. The start of the ADS operation is temporally set at the shut down year of the MOX reactors. Then the power of the ADS is adjusted to the highest possible value, such, that the ADS can run for 50 years. Then the start of the operation of the ADS (of known power) is moved to the earliest date, in such a way, that the ADS can still work for 50 years.

8.1 UOX –> MOX1 –> ADS1 –> ADS2 –> … –>ADSx

In this section we analyze the scenario when TRU from UOX is recycled into MOX1 fuel after 2025, and TRU from spent MOX1 is recycled into ADS fuel. Recycling the ADS waste is not limited.

Conditions for UOX reactors:

- Power history of Swedish Reactor Park is given by Table 2-1.
- UOX fuel burnup: 37 GWd/tHM.
- Energy availability factor until 2025: 0.836.

Conditions for MOX reactor:

- Total thermal power of MOX reactors: 2,700 MW.
- Type of MOX reactor: BWR.
- Number of MOX reactors: 1.
- Energy availability factor from 2025 to 2075 (MOX reactors): 0.918.
- MOX fuel burnup: 40 GWd/tHM.
- Lifetime of MOX reactors: 50 years (2025–2075).
- Core fraction of MOX fuel: 1.0.
- Recycling (cooling + fabrication) time of spent fuel to produce MOX fuel: 6 years.
- ORIGEN2 library for BWR core: BWRPUPU.LIB.

Conditions for ADS units:

- Total thermal power of ADS units: 1,740 MW.
- Number of ADS units: 2 (power of one reactor is 870 MW).
- Energy availability factor from 2060 to 2110 (ADS units): 0.918.
- ADS fuel burnup: 200 GWd/tHM.
- Lifetime of ADS units: 50 years (2060–2110).
- Recycling (cooling + fabrication) of spent fuel to produce ADS fuel: 10 years.
- ORIGEN2 library for ADS core: prepared by MONTEBURNS 1.0.

The thermal power of MOX reactor is smaller than that in Section 7, because the fuel is recycled once only. The thermal power of the MOX reactor is set at such level that the whole UOX waste will be reprocessed until 2075. Start of ADS is moved form 2035 to 2060 to collect a sufficient amount of MOX1 waste for the ADS1 fuel production. We stress that during the first 10 years ADS has to be loaded with ADS1 fuel only (if ADS1 waste must be cooled for 10 years), which implies an increased consumption of ADS1 fuel (recycled form MOX1 waste) during the first 10 years. Our calculation shows that the total thermal power of the ADS units can be set at most at 1,740 MW during the assumed lifetime of 50 years, in which case all direct fuel waste will not be older than 10 years in 2110, and will be in the cooling storage.

8.1.1 Impact of one pass MOX reactor

As our calculation shows, only one MOX-BWR reactor with thermal power of 2,700MW is sufficient to burn all MOX1 fuel which was recycled form the whole UOX waste. First, we investigate the impact of this reactor on the TRU inventory. As Figure 8-1 shows the plutonium inventory was decreased from 80.8 tons to 44.9 tons. Section 7 showed the plutonium inventory is possible to decrease down to 20.2 tons if the MOX waste is twice recycled.

What may happen with americium found in the spent fuel from the Swedish reactors if we use only one pass MOX reactor is given in Figure 8-2.

From Figure 8-2 one can see that the americium inventory is increased less than if MOX2 and MOX3 fuel is loaded (Section 7).

Figure 8-3 presents simulation results regarding transuranium elements in the spent fuel from the Swedish Nuclear Power Plants if we run only one pass MOX reactors.

8.1.2 Impact of combination: one pass MOX + ADS

Our calculations show it is suitable to start to run ADS in 2060 at the thermal power of 1,740 MW. However, we stress it is possible to start sooner with ADS of lower total thermal power.

The following figures show the impact of ADS on the inventory of plutonium, americium, and TRU in total.

As seen from Figure 8-4, ADS significantly reduces the plutonium inventory, by more than a factor of 2 in this case, namely from approximately 45 tonne down to less than 22 tonnes. This is in accord with the previous conclusion what regards burning of plutonium.

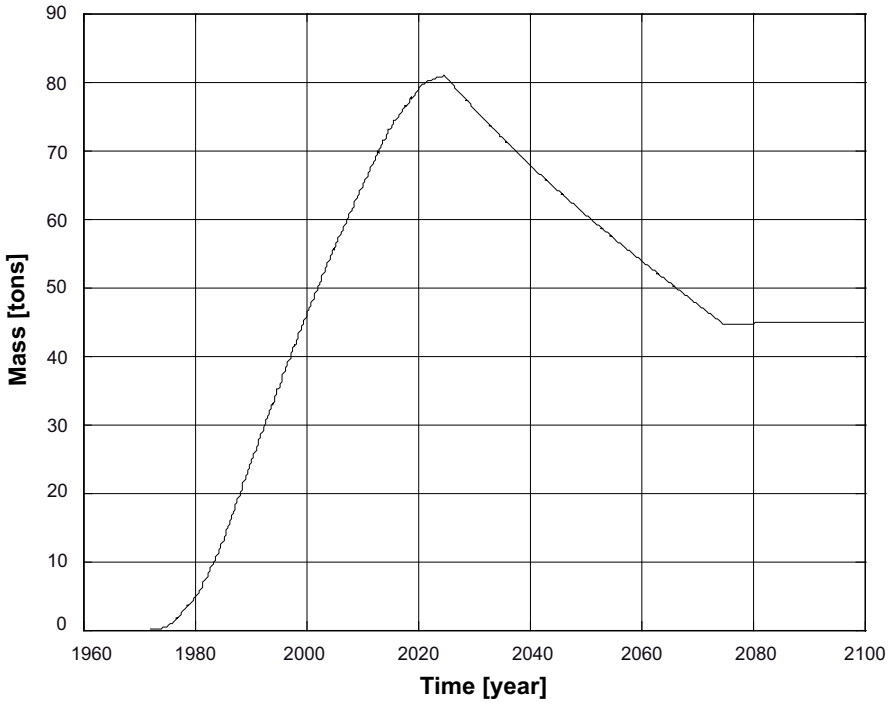

Figure 8-1. *Burning Pu with one pass MOX reactor of 2,700 MW.*

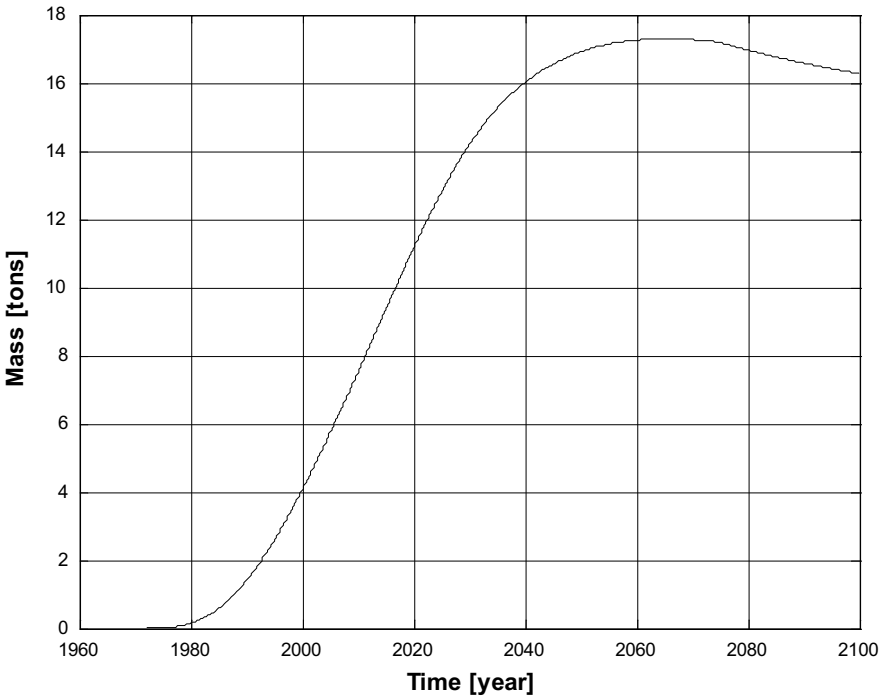

Figure 8-2. *Burning Am with one pass MOX reactor of 2,700 MW.*

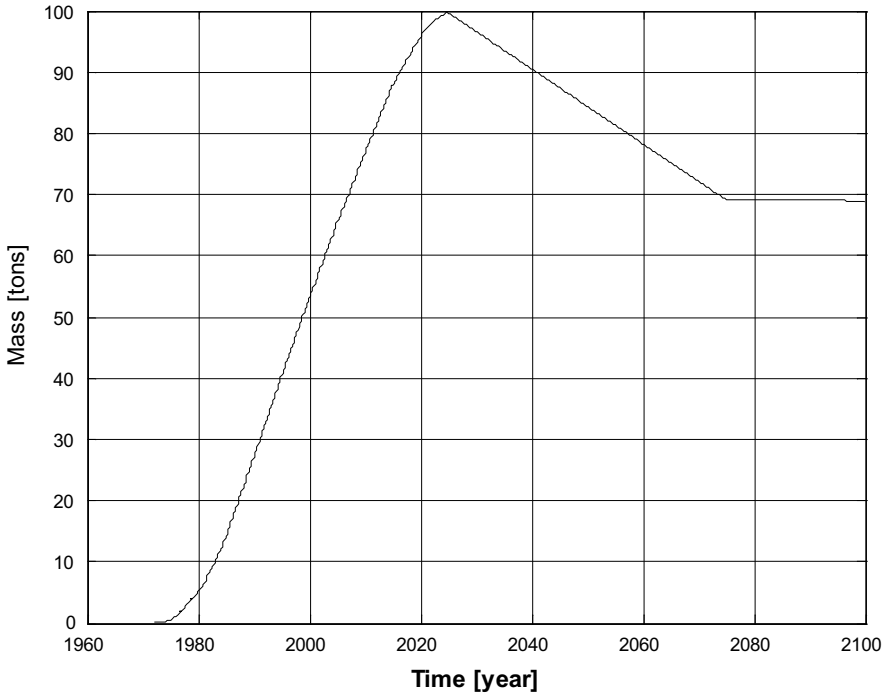

Figure 8-3. *Burning TRU with one pass MOX reactor of 2,700 MW.*

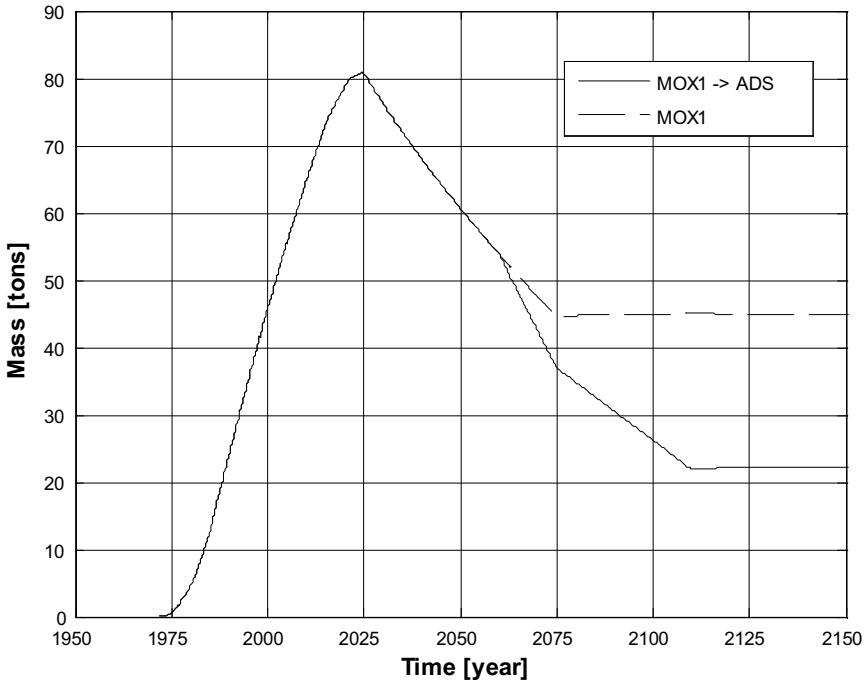

Figure 8-4. *Burning Pu with one pass MOX and ADS.*

Figure 8-5 proves once again that inclusion of an ADS incinerator significantly reduces the americium inventory, in our case about by a factor 2.

We briefly note here that an ADS burner appreciably reduces the total amount the transuranium elements as compared with only a MOX burner, once again approximately by a factor of 2.

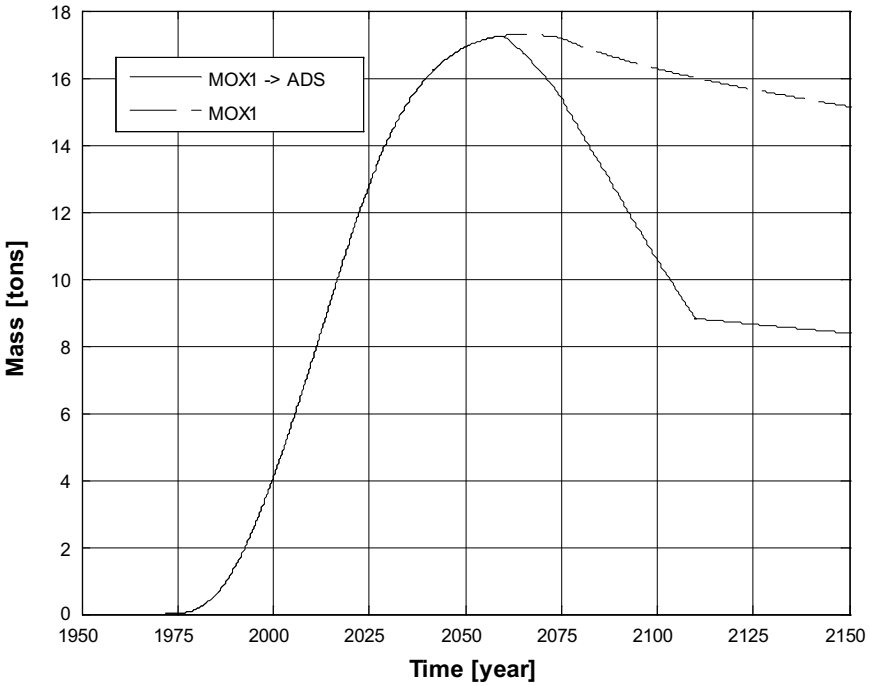

Figure 8-5. *Burning Am with one pass MOX and ADS.*

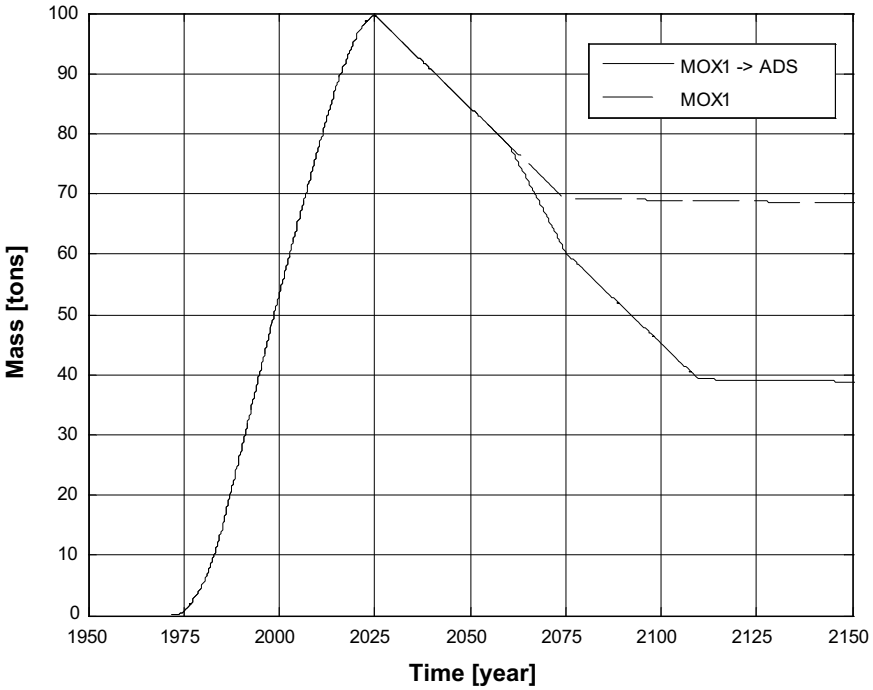

Figure 8-6. *Burning TRU with one pass MOX and ADS.*

8.2 UOX –> MOX1 –> MOX2 –> ADS1 –> ADS2 –> …–>ADSx

In this section we analyze the scenario TRU from UOX is recycled into MOX1 fuel after 2025. TRU from spent MOX1 is then recycled into MOX2 fuel. TRU from spent MOX2 is then recycled into ADS fuel. Recycling of TRU from spent ADS fuel is not limited.

Conditions for UOX reactors:

- Power history of Swedish Reactor Park is given by Table 2-1.
- UOX fuel burnup: 37 GWd/tHM.
- Energy availability factor until 2025: 0.836.

Conditions for MOX reactor:

- Total thermal power of MOX reactors: 4,100 MW.
- Type of MOX reactor: BWR.
- Number of MOX reactors: 2.
- Energy availability factor from 2025 to 2075 (MOX reactors): 0.918.
- MOX fuel burnup: 40 GWd/tHM.
- Lifetime of MOX reactors: 50 years (2025–2075).
- Core fraction of MOX fuel: 1.0.
- Recycling (cooling + fabrication) time of spent fuel to produce MOX fuel: 6 years.
- ORIGEN2 library for BWR core: BWRPUPU.LIB.

Conditions for ADS units:

- Total thermal power of ADS units: 1,140 MW.
- Number of ADS units: 2 (thermal power of one reactor is 570 MW).
- Energy availability factor from 2072 to 2122 (ADS units): 0.918.
- ADS fuel burnup: 200 GWd/tHM.
- Lifetime of ADS units: 50 years (2072–2122).
- Recycling (cooling+fabrication) time of spent fuel to produce ADS fuel: 10 years.
- ORIGEN2 library for ADS core: prepared by MONTEBURNS 1.0.

8.2.1 Impact of multi-pass MOX reactors

Again, we first evaluate the impact of the MOX reactors. The thermal power of the MOX reactors is higher than that in Section 8.1 because the MOX2 fuel can be loaded when no MOX1 fuel is available. Our calculations show the total thermal power of MOX reactors can be set at 4,100 MW, which corresponds to 2 reactors. At the end of the lifetime the system contains MOX2 waste and certain small amount of MOX1 waste in the cooling storage.

Figures in this section compare one MOX reactor of thermal power 2,700 MW burning only MOX1 fuel (form Section 8.1.1) to two MOX reactors of total thermal power 4,100 MW burning MOX1 and MOX2 fuels. Thus, the difference in the figures is caused mainly by the different power levels.

Figure 8-8 shows that americium is produced mainly in MOX2 fuel in the MOX reactors.

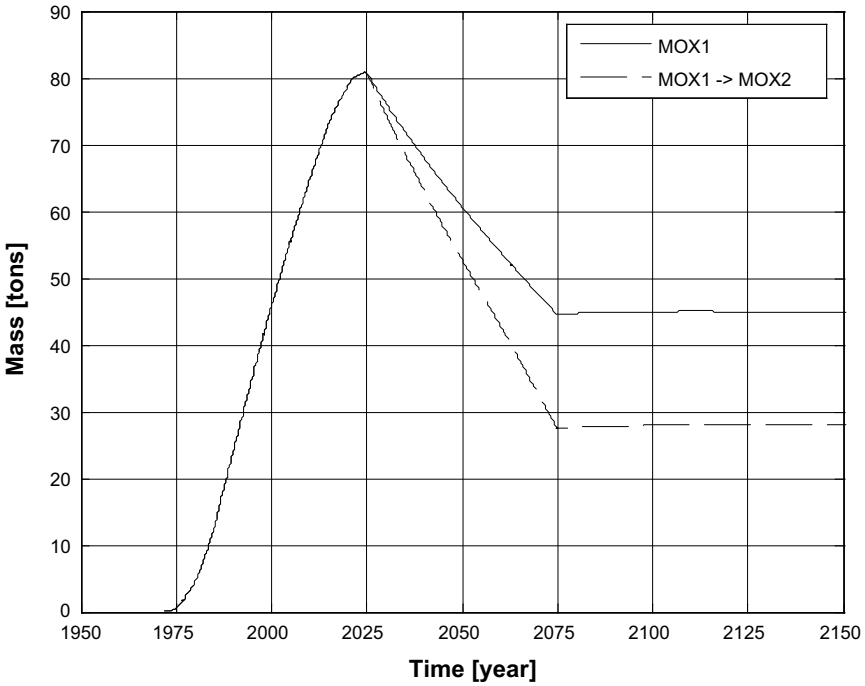

Figure 8-7. *Burning Pu with 2 two pass MOX reactors of totally 4,100 MW.*

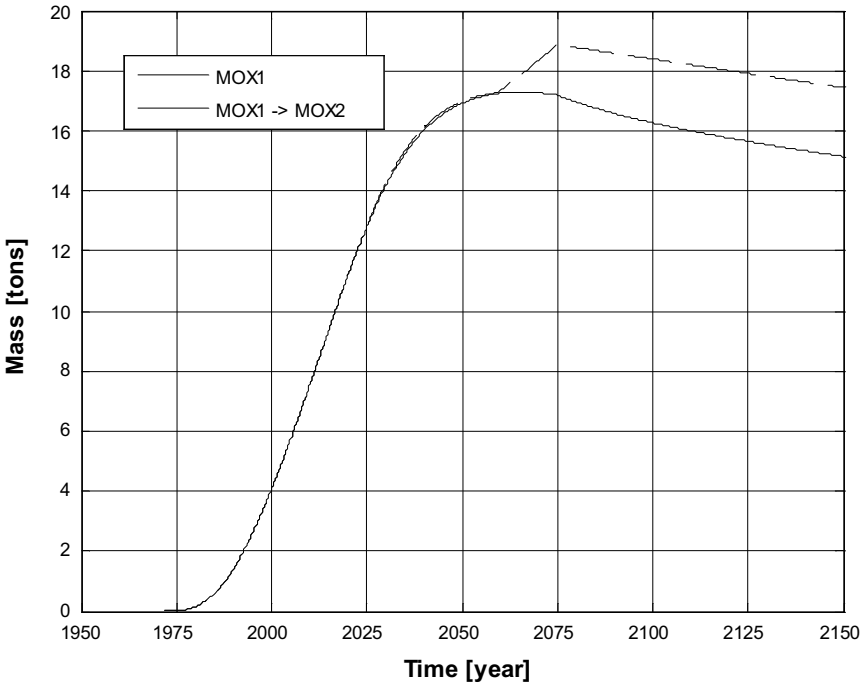

Figure 8-8. *Burning Am with 2 two pass MOX reactors of totally 4,700 MW.*

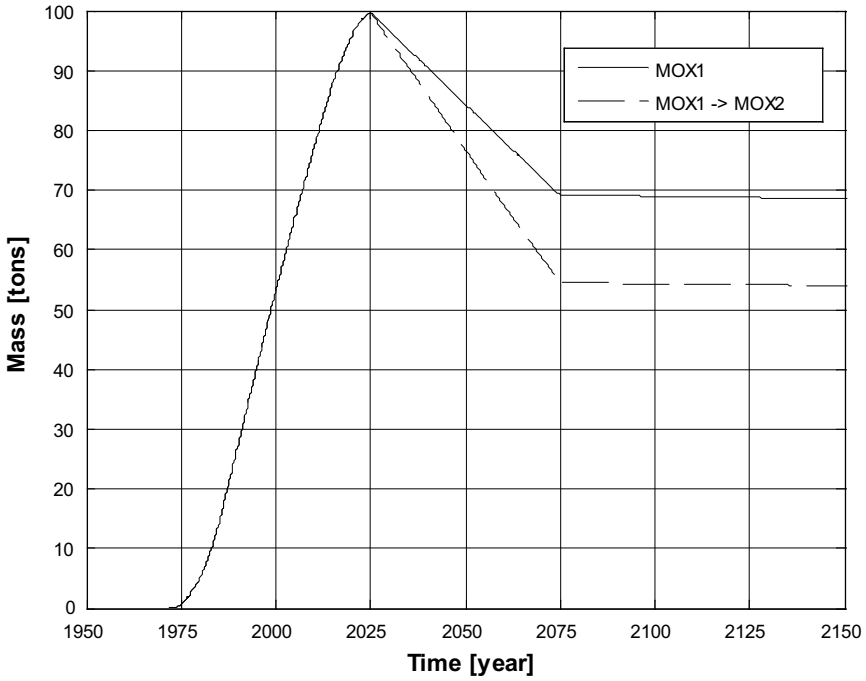

Figure 8-9. *Burning TRU with 2 two pass MOX reactors of totally 4,700 MW.*

8.2.2 Impact of combination multi-pass MOX + ADS

If the MOX2 fuel is to be loaded into the MOX reactors when all MOX1 fuel is spent, then this moment comes around 2060 (with assumed conditions) as computed in the previous section. The ADS technology has to wait until 2072 according to our calculations, when enough sufficiently cooled MOX2 waste is available. Comparing to MOX1 –> ADS scenario the total thermal power of ADS units has to be decreased because the MOX2 waste contains less amount of energy comparing to MOX1 waste. Our calculation shows it is necessary to decrease the ADS thermal power from 1,740 MW to 1,140 MW if the ADS lifetime of 50 years is preserved.

The figures below compare the MOX1 –> ADS scenario to MOX1 –> MOX2 –> ADS scenario. We stress that MOX1 –> MOX2 –> ADS scenario gives 18% more energy. This energy difference is caused by breeding of ^{239}Pu in MOX2 fuel (by neutron capture on ^{238}U) which is recycled in the ADS fuel.

Figure 8-10 shows that plutonium inventory is decreased more in the MOX1 → MOX2 → ADS scenario. Partially since more TRU was recycled in MOX reactors which are more efficient in Pu burning than ADS (if the waste is cooled for 10 years), and partially because of the difference in power levels. Naturally, MOX1 → MOX2 → ADS is worse in burning the americium, as shown in Figure 8-11. Finally Figure 8-12 shows that TRU inventory was more decreased in the MOX1→MOX2→ADS scenario, specifically by 12%, which was mainly caused by the different power levels.

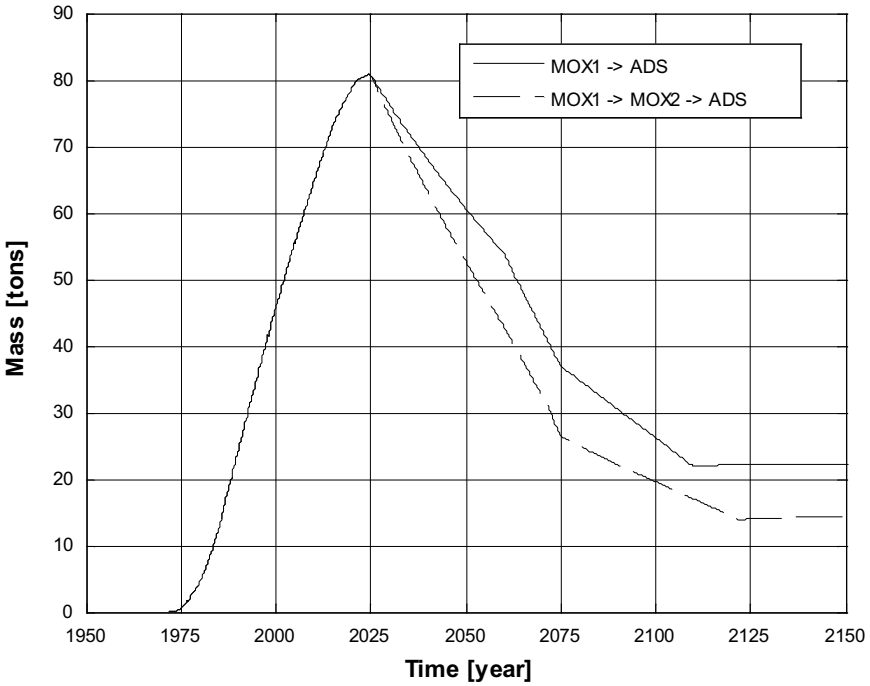

Figure 8-10. *Burning Pu with 2 MOX and 1 ADS reactors.*

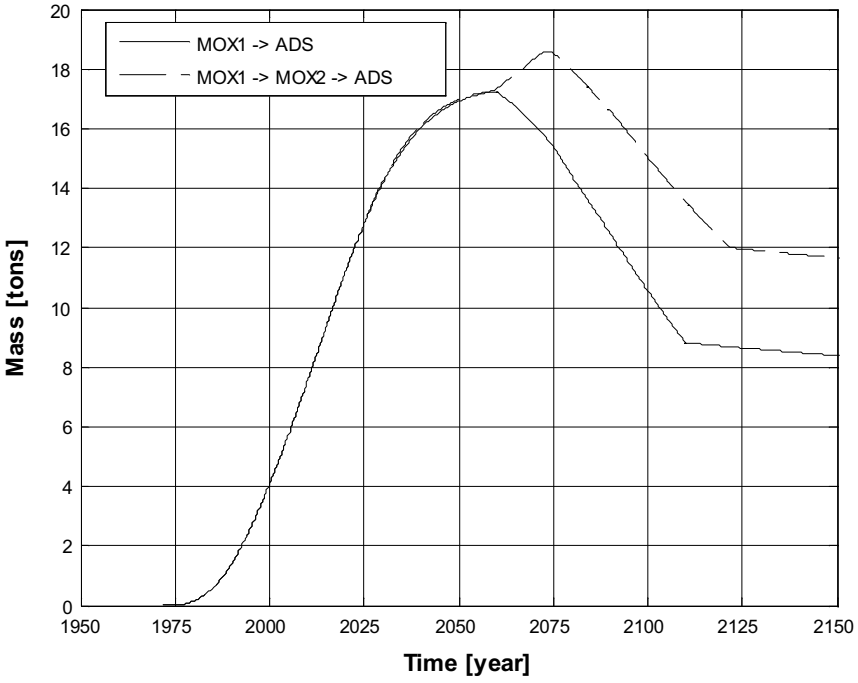

Figure 8-11. *Burning Am with 2 MOX and 1 ADS reactors.*

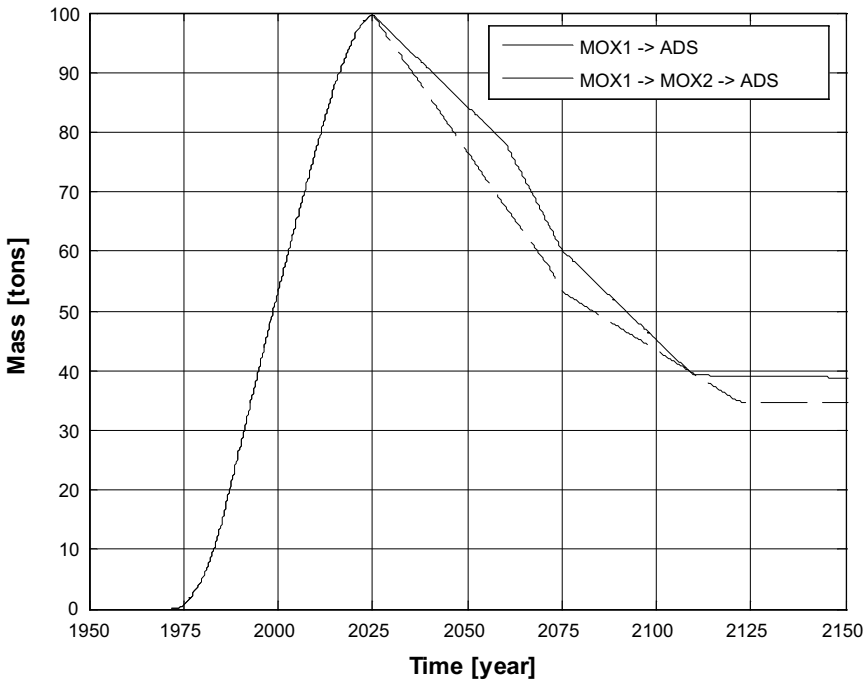

Figure 8-12. *Burning TRU with 2 MOX and 1 ADS reactors.*

9 Economic analysis

9.1 Introduction

The objective of this study is to analyze a range of possible fuel cycle options from the perspective of their effect on waste management policies. To do this we attempt to put the consequences of fuel cycles in terms of indicators, or metrics, which drive, or at least influence, waste management policy decision. A principal difficulty lies in the fact that there is no single, simple, and universally agreed upon indicator which can be used to measure waste management benefit. There are a wide variety of indicators, some of which are inter-related, and some of which are more independent. In addition, the relative importance of these indicators varies with the individual, the organization, nation or decision-making entity.

9.2 Cost as an integrating indicator

The use of cost as an indicator can be seen as both attractive and misleading. Estimates of total system cost can be used as a normalizing factor for widely different processes. It can also be used as a de-facto integrating factor to roll all the other indicators together into a single value. Another major benefit of the cost analysis is the ability to conduct parametric sensitivity studies of the various system components and features.

Along with the benefits of cost analysis come several complications. The process of assigning costs to fuel cycle components and processes may overlook individual indicators, or at least over or under value them. Also, assigning costs requires a number of assumptions, and these may unconsciously incorporate value judgments into the indicator that are more appropriately metrics for decision-makers.

All of this said, we will estimate costs for fuel cycle materials, processes and facilities – and roll these into total; fuel cycle costs normalized to a constant unit of energy produced. It must be kept in mind that while the relative costs of systems evaluated using consistent assumptions can provide useful guidance, the absolute value of the cost values, and any cost values taken out of context, may be misleading.

9.3 Unit costs

As explained above, unit costs have been allocated to each of the steps described in Figure 3-1. However, it is difficult or sometimes impossible to have the exact estimate of the unit cost since many uncertainties exist in terms of technological progress, financial arrangement, political and social environment etc. Moreover, sometimes a simplified scheme was used to evaluate the electricity cost in specific scenarios as will be indicated explicitly.

The methodology adopted in this report generally follows the cost study performed by NEA /15/. Another important source of information is a cost analysis conducted at the department of Nuclear and Reactor Physics, Kungliga Tekniska Högskolan, by Daniel Westlén /16/. In particular many unit cost estimates have been adopted with minor corrections from /16/.

Costs are given in USD 2001. The following unit costs have been used in this report.

- Cost of natural uranium, U_{nat}, = 30 $/kgHM.
- Cost of uranium conversion = 5 $/kgHM.
- Cost of uranium enrichment = 90 $/kgSWU.
- Cost of UOX fuel fabrication = 250 $/kgHM.
- Cost of MOX fuel fabrication = 1,100 $/kgHM
- Cost of ADS fuel fabrication = 7,000 $/kgHM.
- Cost of UOX fuel reprocessing = 800 $/kgHM.
- Cost of MOX fuel reprocessing = 800 $/kgHM.
- Cost of ADS fuel reprocessing = 11,000 $/kgHM
- Cost of Interim Storage = 120 $/kgHM.
- Cost of Geological Disposal = 300 $/kgHM.
- Cost of HLW Disposal = 200 $/kgHM.

It should noted here that the last two items, i.e. the cost of geological disposal and HLW disposal, refer to the weight of heavy metals placed in a repository for permanent storage.

In addition, the following economic data for investment and maintenance were found in /16/:

- Capital investment of LWR = 1,700 $/kWe.
- Capital investment of FR = 1,950 $/kWe.
- Fixed Charge Rate, FCR = 5.8%/year.
- Operation and Maintenance, LWR = 3%/year.
- Operation and Maintenance, ADS = 6%/year.

The basic costs of the U-Pu fuel cycle steps have been assessed in year 2001. However, since then the natural uranium prices have increased by a factor of 3 from about 30 $/kg up to nearly 90 $/kg. Because of this, we update several unit costs as follows /19/:

- Cost of natural uranium, U_{nat}, = 100 $/kgHM.
- Cost of uranium conversion = 11 $/kgHM.
- Cost of uranium enrichment = 120 $/kgSWU
- Cost of UOX fuel fabrication = 275 $/kgHM.

Our economic requires some technological parameters as well. They are

- Loss factor in Mines and Mills = 1.0%.
- Loss factor in Conversion = 0.1%.
- Loss factor in Enrichment = 0.1%
- Loss factor in UOX fuel fabrication = 0.1%.
- Loss factor in MOX fuel fabrication = 0.1%
- Loss factor in ADS fuel fabrication = 0.1%.
- Loss factor in UOX reprocessing = 0.1%.
- Loss factor in MOX reprocessing = 0.1%
- Loss factor in ADS reprocessing = 0.1%.

In what follows, the calculations have been done with the updated unit costs.

9.4 Capital investment for ADS facilities

It is not so easy to assess the capital investment for Accelerator Driven Systems, CAP_{ADS}. To do so we assume that

$$Cost_{ADS} = CAP_{FR} \cdot P_{ADS,el} + CAP_{Acc} \cdot P_{Beam} \tag{49}$$

Here, CAP_{Acc} is capital investment for Accelerator (Acc) and P_{Beam} is the beam power of the accelerator that is related to the ADS power through the gain factor, G, as, $P_{Beam} = P_{ADS}/G$, cf (42). Capital investment for ADS is assumed to be similar to that for Fast Reactor (FR), which is estimated in /16/ as CAP_{FR} = 1,950 \$/kW$_{el}$ (per installed electric power). According to /20/, it is reasonable to assume capital cost for accelerator to be CAP_{Acc} = 15 \$/W (per proton beam power). Finally, we deduce from

$$CAP_{ADS} = \frac{Cost_{ADS}}{P_{ADS}} = CAP_{FR} + \frac{1}{G} CAP_{Acc} \tag{50}$$

This gives us a reasonable estimate for the capital investment as

$$CAP_{ADS} = 1,950 + \frac{1}{45} 15,000 \approx 2,300 \frac{\$}{\text{kWe}} \tag{51}$$

This allows us to assess the total capital cost of ADS facilities in various scenarios. For example, assuming the total electric power of ADS reactors in ADS-R scenario to be 5,250×34% = 1,785 MWe (see Table 7-9), we evaluated the total cost for erecting new ADS facilities as

$$Cost_{CAP} = 18,200 \frac{\text{SEK}}{\text{kWe}} \times 1,785 \times 1,000\text{kW} \approx 32.5 \text{ GSEK} \tag{52}$$

We have assumed here the rate of conversion, US dollars to Swedish kronor, to be 7.9 SEK/ USD that resulted in 2,300 \$/kWe = 18,200 SEK/kWe. As result, we conclude that the total capital cost of ADS facilities may be evaluated to be of order 32.5 billions Swedish kronor for the realistic ADS-R scenario.

9.5 Phase-out scenario (once-through cycle)

This model follows the chart shown in Figure 9-1. The cost of electricity, *COE*, we define as

$$COE \equiv \frac{Cost_{tot}\left[\$/yr\right]}{E_{el}\left[kWh/yr\right]} \tag{53}$$

It is relatively straightforward to evaluate the total electrical energy produced by a reactor per one year:

$$E_{el} = \varepsilon \cdot P_{th} \cdot LF_{LWR} \cdot year \tag{54}$$

Here, ε is the thermal efficiency, which is 33–35% for Swedish light water reactors; LF_{LWR} is load factor for light water reactors. The total cost may be represented as a sum of 4 components.

$$Cost_{tot} = Cost_{FE} + Cost_{BE} + Cost_{OM} + Cost_{IntR} + Cost_{CAP} \tag{55}$$

Here *FE* and *BE* designates Front End and Back End respectively, *OM* stands for Operation and Maintenance and *IntR* denotes Interest Rate. The cost of operation and maintenance and the cost interest rates are evaluated first through the investment

$$Invest \equiv CAP_{LWR} \cdot P_{el} \quad \left(CAP_{LWR} = 1,700\$/kW\right) \tag{56}$$

Second, we estimate the cost of operation and maintenance as

$$Cost_{OM} = Invest \cdot OM_R \quad \left(OM_R = 3\%/yr\right) \tag{57}$$

Third, we calculate the cost of interest rate

$$Cost_{IntR} = Invest \cdot FCR \quad \left(FCR = 5.8\%/yr\right) \tag{58}$$

Finally, we evaluate the yearly cost due to the initial investment

$$Cost_{CAP} = Invest/T_{op} \tag{59}$$

Here T_{op} is assumed to be 50 years.

It is more involved to evaluate the cost of the Front and Back end. The evaluation begins with calculating the total mass of fuel, M_F, needed to provide a certain thermal power level during a year

$$M_F = \frac{LF_{LWR} \cdot P_{th} \cdot 1year}{B} \tag{60}$$

Here, P_{th} is the nominal thermal power of a reactor, LF_{LWR} is the Load Factor for a Light Water Reactor (LWR), and B is the fuel burnup.

Then we proceed evaluating the masses needed at each step indicated in Figure 9-1. The least straightforward is to estimate the mass of enriched uranium. The enrichment process splits the original amount of natural uranium into two parts as shown in Figure 9-2.

First, we write down two balance equations

$$\begin{aligned} &N_{nat}: \ N_{nat} = N_{enr} + N_{dep} \\ &^{235}N: \ \varepsilon_{nat} N_{nat} = \varepsilon_{enr} N_{enr} + \varepsilon_{dep} N_{dep} \end{aligned} \tag{61}$$

Then we deduce the ratio $r = M_{nat}/M_{enr}$ as

$$r \equiv \frac{M_{nat}}{M_{enr}} \approx \frac{N_{nat}}{N_{enr}} = \frac{\varepsilon_{enr} - \varepsilon_{dep}}{\varepsilon_{nat} - \varepsilon_{dep}} \tag{62}$$

The cost and mass requirement for uranium enrichment are expressed in terms of the number of Separative Work Units (SWU) for a specified enrichment and tails assay. The unit cost of enrichment, UC_{enr}, (per kg of U_{enr}) may be found by

$$UC_{enr} = UC_{SWU} \cdot \left[V\left(\varepsilon_{enr}\right) + \left(r-1\right) V\left(\varepsilon_{dep}\right) - rV\left(\varepsilon_{nat}\right) \right] \tag{63}$$

The value function, $V(\varepsilon)$, of each stream, i.e. natural, enriched and depleted uranium, is related to the fraction, ε, of ^{235}U in that stream by:

$$V(\varepsilon) = \left(1 - 2\varepsilon\right) \ln \frac{1-\varepsilon}{\varepsilon} \tag{64}$$

For example, assuming the current situation in the uranium market, we use UC_{SWU} = 120 $ per kgSWU that gives approximately 527 $/kg$U_{enr}$ for ε_{enr} = 3.30% and ε_{dep} = 0.25%.

The rest is trivial; we correct the masses for loss factors and then multiply the masses by the corresponding unit costs. The estimation of the Back End is quite straightforward if we know the corresponding unit costs and it is not shown here. The result is given in Figure 9-3 with an average cost of electricity of about 3.29 ¢/kWh. Presented in Figure 9-3 calculations are based only on the data shown in Table 2-1.

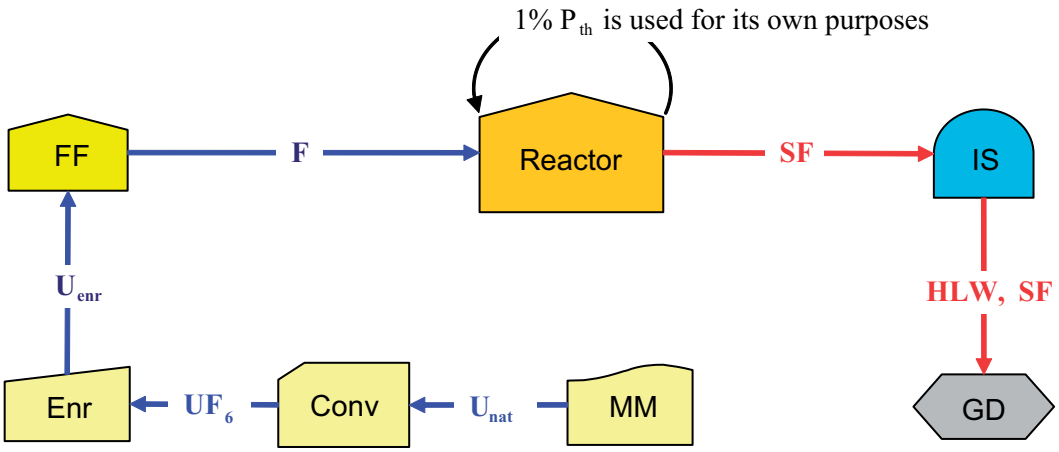

Figure 9-1. *Once-Through Fuel Cycle.*

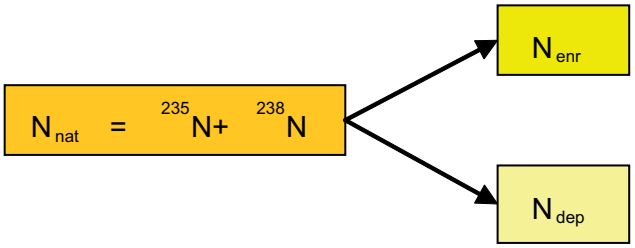

Figure 9-2. *Number balance of ^{235}U and ^{238}U atoms during enrichment process.*

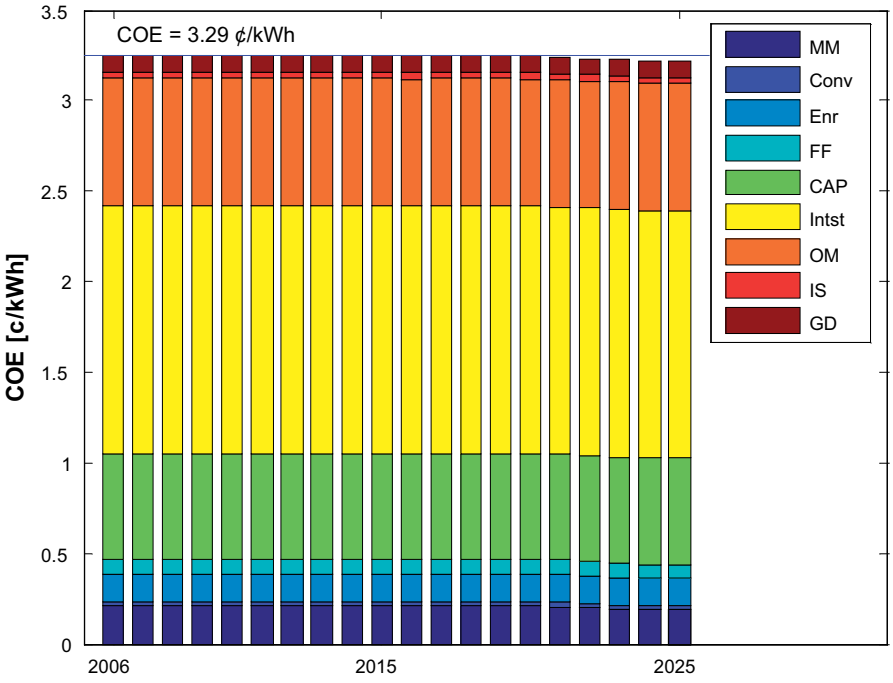

Figure 9-3. *Cost of "nuclear" electricity in Swedish Phase-out scenario. Existing reactors operate until the projected end of life.*

We note here as a matter of fact that a slight cost decrease by year 2025 reflects the situation when the most modern (and hence advanced) reactors are the last to be shut down. In other words, more efficient reactors (cf Table 2-1) are left into operation by year 2025. It is worth mentioning that the unit costs used here are not necessarily representative of the current costs for the Swedish system. In particular, the capital investments are much larger than the actual historical investments. This makes the costs presented here different and presumably higher that the actual coats for the current nuclear power production.

9.6 Steady state scenario with ADS

This fuel cycle model is visually given in Figure 9-4.

Conceptually, the current case is equivalent to the previous one with some evident modifications. Results are visually presented in Figure 9-5 where in addition we plot analogous calculation performed for the MOX fuel cycle with 2 passes of MOX fuel.

As seen in Figure 9-5, the most effective, from the economical point of view, way of producing electricity is the ordinary UOX Once-Through Cycle. Producing electricity with MOX40 reactors increases the cost of electricity by approximately 90% with the greatest share being associated with the reprocessing whereas the cost of ADS200 fuel cycle is about 60% higher than that of UOX. The real cost driver for the MOX scenario is the reprocessing of spent fuel and fabrication of new MOX fuel due to high unit costs and a big amount of fabricated and reprocessed fuel needed. In case of ADS, much less amount of fuel is needed because of higher burnup level; however the cost of fuel fabrication is 7 times that of MOX fuel, and what regards reprocessing, the cost is more than 13 times higher than that of MOX fuel reprocessing. In addition, the interest rate and operation and maintenance, in case of ADS, are appreciably greater due high capital investments.

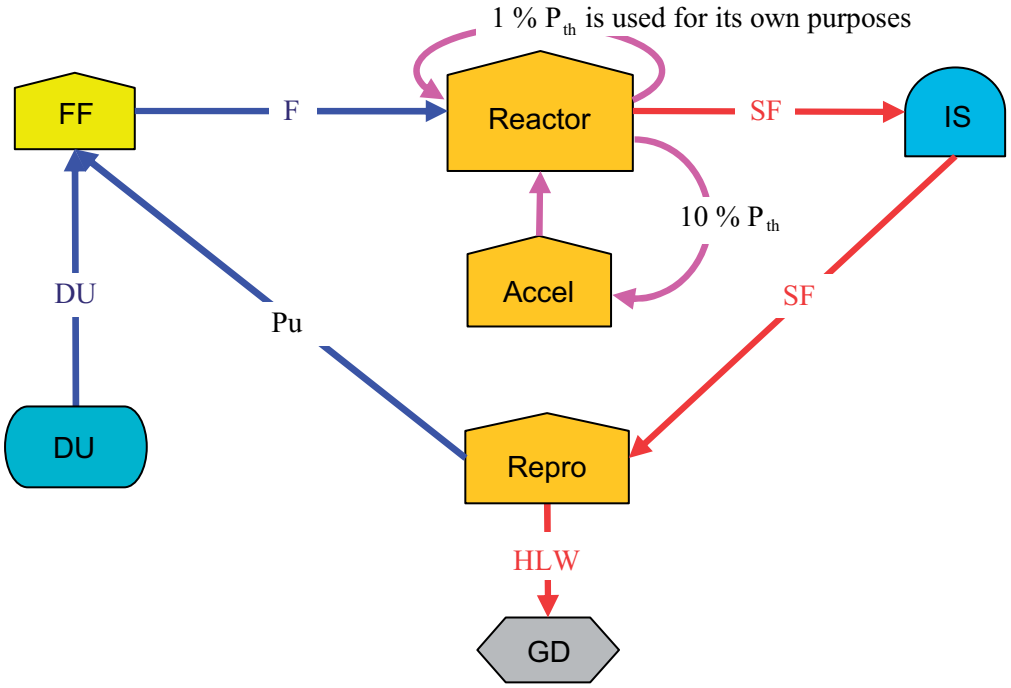

Figure 9-4. *ADS fuel cycle.*

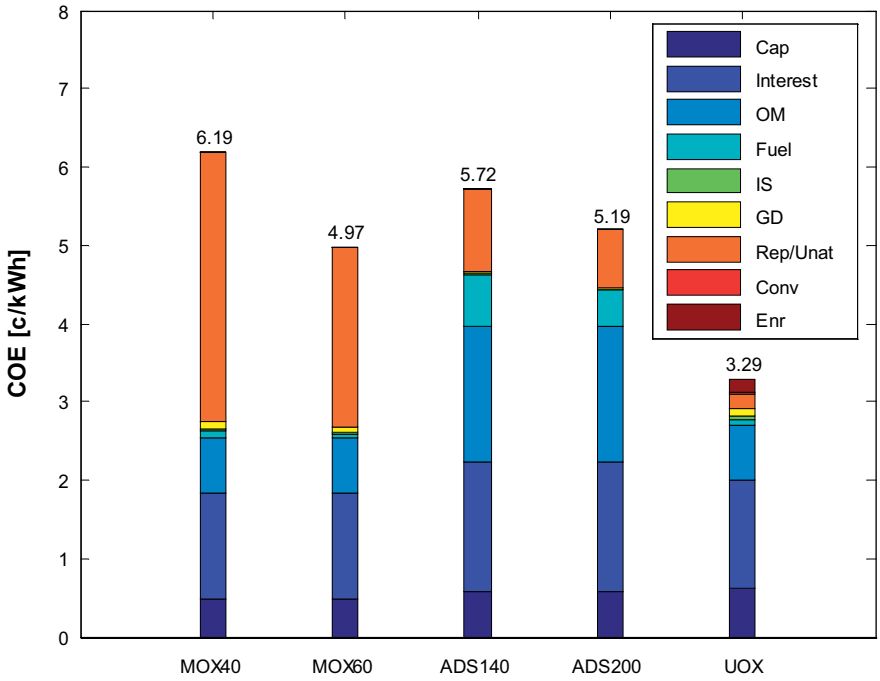

Figure 9-5. *Cost of electricity comparison for various scenarios with 40, 60, 140, and 200 GWd/tHM of burnup.*

On the other hand, raising the burnup of the MOX fuel up to 60 GWd/tHM, which seems plausible according to a recent NEA study, makes the situation more favorable for the MOX scenarios regarding the cost of electricity. The reduction of cost in this case amounts up to 20%. On the other hand, the impact of the burnup increase from 140 to 200 GWd/tHM in the ADS scenario amounts only up 10%.

It should be noted here that Figure 9-5 presents scenarios when nuclear facilities are build from the scratch thus giving a considerable share of the interest rate. However in the case of a MOX scenario, this constituent is perhaps lower because it is probable that existing BWR or PWR reactors will be converted to MOX burners thus making theses scenarios economically more attractive.

10 Conclusions

The purpose of this report was to analyze the potential to transmute at least 99% of transuranium isotopes found in the Swedish spent nuclear fuel by year 2025. To this end, a computer code, FCA has been developed at the department of Nuclear and Reactor Physics, KTH. The code implements a novel method to effectively solve point reactor equations of isotopic kinetics by using evaluated production-depletion rates.

Generally speaking, we are limited by only few options:

- To dispose the spent fuel as it is in a waste repository for permanent storage and monitoring (Phase-out Scenario).
- To restart several BWR or PWR units to burn/transmute existing plutonium and/or minor actinides in a form of Mixed Oxide or MOX fuel.
- To build one or several Accelerator Driven Systems (ADS) for use as stand alone reactors or more probably for use with a combination with BWR or PWR burners.
- Finally, to build one or several Fast Reactors (FR) which are reportedly very effective as transuranium burners.

The last option, Fast Reactors as transuranium burners, has been deliberately left unexplored in this report.

Several scenarios have been modeled with a help of the FCA code:

- Phase-out scenario.
- BWR MOX burners.
- PWR MOX burners.
- ADS MOX burners.
- BWR + ADS MOX burners.

As a very general conclusion, we may state BWR MOX reactors are more efficient in burning plutonium in the form of MOX fuel. In this respect BWR reactors supersede PWR ones by approximately 10%. In addition, BWR reactors produce about 10% less americium inventory. However, neither BWRs alone nor PWRs alone are capable of incinerating 99% of the TRU inventory.

It was found that an ADS reactor park can theoretically in the ideal case burn 99% of transuranium isotopes. The duration of this scenario heavily depends on the time needed for cooling the spent fuel. If we assume 10 years of cooling the nuclear waste from ADS, the scenario duration becomes at least *200* years under most optimistic technical assumptions. On the other hand, the use of advanced pyro-processing with a cooling time of only 2 years decreases the incineration time down to *50* years. Moreover, ADS reactors have turned out to be a necessary component to decrease the americium inventory because neither BWR nor PWR alone can provide prevalence of americium destruction over its production during the operation time. Nevertheless, the economic advisability of these scenarios calls for further investigation.

In addition, a combination of MOX1 –> MOX2 –> ADS has been found more efficient (approximately by 10%) in reducing the transuranium inventory.

Regarding the questions posted in the assignment, we summarize here briefly our simulation results.

1. The number of ADS for burning up MA and the left plutonium.
   Few options have been considered: the number of ADS varies 1 to 6 depending on the fuel burnup and, what is more important, on the thermal power. A realistic scenario based entirely on domestic resources suggests 5 ADS units of 890 MW of thermal power and 1 unit of 800 MW. The first 5 units are deployed year 2035 out of which two units are shut down in 2085; one unit is shut down in 2077 and another one in 2083. The smaller unit runs 2090 through 2135.
2. The time required to incinerate 99% of plutonium and minor actinides.
   This time depends heavily on the recycling (mostly cooling) time. A theoretical minimum time in an ideal case was evaluated to be 50 years provided an advanced (to be developed) PYRO technology offers the cooling time of about 2 years. The conventional PUREX technology with the cooling time of 10 years demands 200 years or more to decrease the amount of plutonium and minor actinides by 99%.
3. Additional electricity due to MOX and MA transmutation has been evaluated to be of order 600 to 950 TWh with an average power rate of 800–1,000 MWe.
4. What happens to ^{129}I?
   The total amount of the iodine isotope ^{129}I accumulated in the Swedish Nuclear Power Plants has been evaluated to be around 2 tonnes in the Phase-out scenario. Regarding ^{129}I in ADS1 and ADS2 scenarios, there are no technical solutions today for transmutation of this isotope, however assuming suitable form of the transmutation pellets/targets containing otherwise volatile iodine physically this isotope can be effectively transmuted through the neutron capture and a very fast beta-decay into a stable ^{130}Xe isotope.
5. Amount of waste as a result of reprocessing in the glass form to be disposed:
   a. Reprocessing gives 254 tonnes of vitrified waste for ADS with 2 years of recycle time.
   b. Reprocessing gives 291 tonnes of vitrified waste for ADS with 3 years of recycle time.
   c. Reprocessing gives 515 tonnes of vitrified waste for ADS with 10 years of recycle time.
   d. Reprocessing gives 469 tonnes of vitrified waste for 3 pass MOX reactors.
   e. Reprocessing gives 112 tonnes of vitrified waste for the ADS-R scenario.
6. Total number of canisters and eventual area in Sweden necessary for final storage of the waste as a result of reprocessing in the glass form and 50 years of cooling:
   a. 625 canisters in 1.00 ha for ADS with 2 years of recycle time.
   b. 720 canisters in 1.15 ha for ADS with 3 years of recycle time.
   c. 1,280 canisters in 2.05 ha for ADS with 10 years of recycle time.
   d. 1,170 canisters in 1.88 ha for 3 pass MOX reactors.
   e. 285 canisters in 0.45 ha for the realistic ADS-R scenario.
7. Amount of TRU-LILW to be stored has been estimated:
   a. 666 kgHM for ADS with 2 years of recycle time.
   b. 631 kgHM for ADS with 3 years of recycle time.
   c. 447 kgHM for ADS with 10 years of recycle time.
   d. 3,261 kgHM for 3 pass MOX reactors.
   e. 821 kgHM for the realistic ADS-R scenario.
8. Amount of left fuel to be disposed has been estimated:
   a. 16 tHM for ADS with 2 years recycle.
   b. 23 tHM for ADS with 3 years recycle.
   c. 53 tHM for ADS with 10 years recycle.
   d. 211 tHM for MOX with 6 years recycle.
   e. 27 tHM for the realistic ADS-R scenario.

9. Amount of construction waste from ADS buildings:
   a. 25,000 tonnes per one ADS facility however this amount would be dependent on the choice of accelerator for ADS facilities. Therefore, this number should be treated as indicative with at least 25% margin. Moreover, accelerator technology has not yet reached industrial maturity and significant progress is to be expected, particularly in reduction of mass of materials used for accelerator construction;
   b. 150,000 tonnes for the realistic ADS-R scenario.
10. Expenses for reprocessing and production of MOX- and MA-fuel in ADS and MOX scenarios have been evaluated to amount up to:
   a. 49.8 GSEK under 50 years or 1.00 GSEK/year for ADS with 2 years of recycle time.
   b. 46.5 GSEK under 50 years or 0.93 GSEK/year for ADS with 3 years of recycle time.*
   c. 30.0 GSEK under 50 years or 0.60 GSEK/year for ADS with 10 years of recycle time.
   d. 25.0 GSEK under 50 years or 0.50 GSEK/year for MOX with 6 years of recycle time.
   e. 61.8 GSEK or 0.62 GSEK/year for ADS with 3 years of recycle time in the realistic ADS-R scenario.

   The total fuel expenses in the more realistic ADS-R scenario have been evaluated to be about 61.8 GSEK under 100 years or equivalently 0.62 GSEK yearly. Electricity from this fuel would correspond to 383 GSEK over 100 years or equivalently 3.83 GSEK yearly in both cases excluding taxation, i.e. assuming electricity price without taxation for consumers of 52 öre/kWhe i.e. being twice of the projected costs of "nuclear" electricity production as shown at Figure 9-3 – this assumption is in the spirit of today's electricity market, and by chance this is electricity price produced at ADS-R scenario with burnup of 200MWd/tHM as shown on Figure 9-5.
11. Cost of ADS facilities has been evaluated to be of order of 32.5 GSEK in the realistic ADS-R scenario – having 6 ADS units. This numbers can be compared to economical estimations done for the European Spallation Source to be presumably constructed in Sweden – see /13/. A unique multipurpose research facility based on an accelerator of ADS-class is estimated to cost ~11 GSEK. In this context, an estimation of 5.5 GSEK for a single ADS unit (subcritical reactor + an accelerator) built as an industrial, single purpose facility is very reasonable.

Economic analysis of various scenarios has been performed under a number of simplifying assumptions. It must be stressed that the costs evaluated for various scenarios should be regarded as relative numbers. In this respect it has been found that the MOX scenario is about 90% more expensive as ordinary UOX fuel cycle. On the other hand, ADS burners reduce this factor down to about 60%, in other words electricity produced after year 2025 at a hypothetical combination of BWR MOX reactors together with ADS MOX reactors appears to be approximately 70% more expensive in comparison with the today's situation. It should be noted here that this conclusion refers to the currently achievable burnup level of 40 GWd/tHM for MOX reactors. Raising this technical parameter up to 60 GWd/tHM changes the economical situation by reducing the cost of electricity by approximately 20% thus making MOX scenarios more attractive from the economical point of view.

No uncertainties or confidence margins have been assessed in details for most of the results presented in this report and as mentioned before the results are mainly of interest for comparisons. However it is worth to note that input data and results concerning nuclear reactor technology are well founded, in contrast significant progress may be expected in accelerator technology having an impact on amount of waste and economical indicators. No attempts have been done to predict electricity market in the future.

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