Analysis of European fuel cycle transition and equilibrium scenarios with LWR and ADS

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Abstract

The 6th Framework Programme project RED-IMPACT explored the role of partitioning and transmutation technologies in the radioactive waste production and management, as well as design requirements of the final disposal. A series of open and closed fuel cycle scenarios for a representative nuclear power park were chosen and analysed by the partners, firstly in a theoretical equilibrium situation and secondly in a transition towards equilibrium. Results corresponding to each scenario were compared through certain key parameters.

CIEMAT has contributed with the detailed study of a fuel cycle scenario composed by two types of nuclear reactors in a so-called double strata scenario. It consists in a fully closed fuel cycle including a first stage of conventional UO_2 burning in LWR, followed by a second stage of plutonium usage in LWR with MOX fuel, and thirdly a cyclic burning in ADS of Pu and MA resulting from previous LWR and ADS irradiations.

Concerning the equilibrium scenario, CIEMAT has obtained the isotopic composition of all the major waste streams generated in the fuel cycle. Waste streams can be produced either in the normal reactor operation (fuel impurities, cladding and structure activation, creation of spallation products in the ADS target and activation of its components) or in the different types of spent fuel reprocessing activities (reprocessing losses, low and intermediate level waste). These results allow the estimation of the parameters affecting the final disposal, as the thermal power of the waste, activity, radiotoxic inventory and the neutron emission.

In the transition scenario, the main objective was the appraisal of the mass and composition of the waste going to the repository. In order to estimate these values, the detailed isotopic evolution of every source of waste was provided. This included the evolution with time of every reactor, reprocessing and storage facility and waste legacy, highlighting the different key stages of a complex fuel cycle scenario.

The objective of this paper is to present results as well as the tools and methods used to obtain them. An interpretation of the obtained results is also provided by means of comparison with current open cycle results to state the impact that P&T technologies have in the waste management and final disposal.

Introduction

The RED-IMPACT project has been performed within the auspicious of the 6th Framework Programme of the European Union. Its main objective was to assess the "Impact of Partitioning, Transmutation (P&T) and Waste Reduction Technologies on the Final Nuclear Waste Disposal". Other objectives of this three-year project were to assess economic, environmental and societal costs/benefits of P&T; to disseminate results of the study to stakeholders (scientific, general public and decision makers); and to iterate and refine the work based on stakeholders' feedback to achieve full impact of this study on the implementation of the waste management policy of the European Community [1].

For the completion of these objectives, a set of representative fuel cycles scenarios has been selected for which the different parameters of interest have been assessed. The choice of these scenarios has been based on an overview of previous studies of partitioning and transmutation/ conditioning (P&T/C) strategies performed in different EU framework programmes, NEA/OECD and other international organisations, taking into account the current national waste management policies of the different EU countries. The choice of scenarios has been repository driven looking for a good representativity of the different (primary and secondary) waste streams appearing in scenarios discussed in different EU countries. They were not chosen to be necessarily the best choices for any particular country or situation and not a proposal of cycles to be implemented. They described, however, coherent scenarios and should provide detailed information of the consequences of the key choices for the advanced fuel cycles and their waste management. In addition, the scenarios have been chosen after the feedback of the whole collaboration, and have been consequently re-evaluated and redefined, to make sure that they involve present practices or immediately deployable technologies.

Double-strata scenario with LWR and ADS

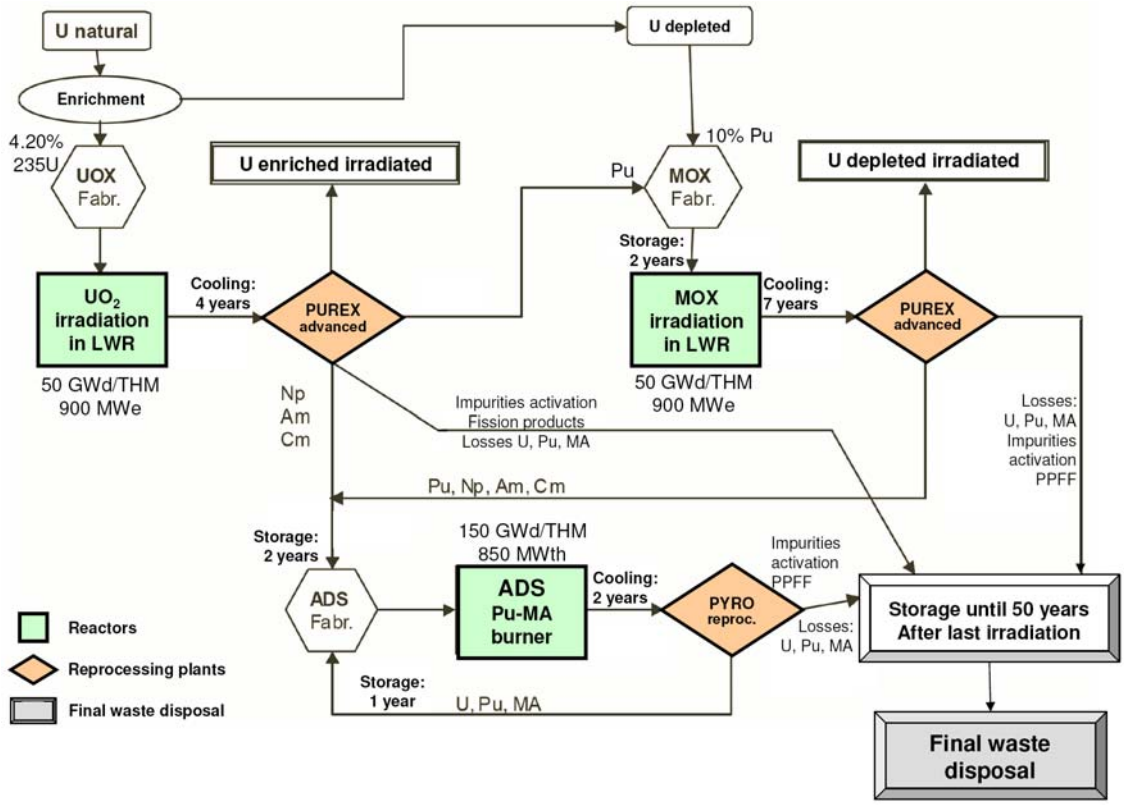
One of the fuel cycle scenarios studied by the collaboration is the double-strata scenario with light water reactors (LWR) and accelerator-driven subcritical systems (ADS), so-called Scenario B2. The scheme of this scenario can be found in Figure 1, where the main characteristics of each reactor also appear. The first stratum of this fuel cycle consists in the irradiation of UO_2 in LWR and the later advanced Purex reprocessing of the spent fuel for the Pu reutilisation, only once, as MOX, again in LWR. Recovered MA coming from the first partitioning process, and Pu and minor actinides (MA) coming from the reprocessing of the MOX spent fuel are reutilised as fuel for the ADS in the second stratum. This second stratum is based on a fast spectrum ADS, which operates with continuous recycling of the main actinides (U, Pu, Np, Am and Cm). This recycling is supposed to be a pyro-metallurgical process with assumed losses of 0.1% for all these actinides (same hypothesis as for Purex reprocessing). The case of Scenario B2 is of special interest because it introduces an additional element to the waste: a proton accelerator coupled to the subcritical core, producing spallation fragments.

This fuel cycle has been studied in detail, estimating the different waste streams coming from the different reactors. For the equilibrium (constant mass fluxes) situation, all the waste streams have been assessed, including both formed by high-level waste (HLW) and by intermediate-level waste (ILW). For the transition studies (from an actual nuclear park towards the equilibrium), the mass and waste flows have been estimated by means of two different calculation models, as will be explained in the following section.

Equilibrium calculations

The complete isotopic composition of the different waste streams can be obtained after the performance of a detailed simulation of every nuclear system. In the case of the LWR, this problem has been studied in detail many times (for instance [2]). Different standard simulation codes have been used to simulate this reactor so a precise set of cross-section libraries, checked with experimental data, has already been provided to the scientific community. Hence, there is no need of a simulation with an advanced code. For this task, the calculation has been made using the depletion code ORIGEN2 using the proper libraries for a LWR with high burn-up, for UO_2 fuel in the first reactor and for MOX fuel in the second one of the first cycle stratum. The case of the ADS is very different as

Figure 1: Scheme of RED-IMPACT Scenario B2 with LWR and ADS



it is a new concept of nuclear transmutation system. In order to obtain precise results of the isotopic composition of the different waste streams appearing after the irradiation, it is necessary to make a simulation of the system fulfilling several requirements:

- The computational code must be capable of simulating very complex designs of nuclear systems, including the coupling of a proton accelerator, a spallation system, etc.
- High fuel burn-ups are needed so corresponding large isotopic evolutions are expected and must be properly simulated.
- As simulations of this type of reactors have not been checked with experimental data, there are no standard cross-section libraries. The code must create them from the nuclear pointwise databases.
- The complete isotopic composition must be obtained for the different waste streams.

EVOLCODE2 [3] fulfils all these requirements so it has been used for the detailed simulation of the ADS. In particular, it has been used to obtain the isotopic composition of the nuclear fuel (actinides, fuel matrix, fission products and impurities activation) and the core structural elements during and after the irradiation, the evolution with the burn-up of the k_{eff} of the system, and the isotopic composition of the spallation system (target, window and other structural elements) after the irradiation [4]. In this document, only the results of those magnitudes having the strongest impact in the repository are included.

Transition calculations

The realisation of this study has been based in a detailed simulation procedure involving EVOLCODE2. Like in the equilibrium calculations, the isotopic composition of waste streams coming from the LWR has been obtained using ORIGEN2 with the default libraries. EVOLCODE2 has been in charge of the

simulation of all the waste streams coming from the ADS. With this methodology, the flows of mass and the amount of waste produced each year were calculated for the following streams:

- annual production of spent fuel, generated in LWR-UO_2, LWR-MOX and ADS;
- total amount of Pu and MA to be transmuted
- annual production of HLW (U, Pu, MA and fission products) going to the deep repository;
- total amount of Pu and MA in the fuel cycle.

A second calculation method consisted in a faster computational code by means of a simpler parametrical method. This code, called ECC, solve mass balances for fuel cycle scenarios and it is able to use the same main hypotheses that a fully detailed calculation. Nuclear fuel is supposed to be, in ECC, composed by four types of materials: U, Pu, MA as a whole (Np, Am and Cm) and fission products, also as a whole. Their burn-up in a reactor is modelled by the product of certain coefficients times the masses at the beginning of the irradiation. These coefficients depend on the fuel cycle and the reactor characteristics. Material pools are homogenised in this method for simplification reasons. The user proposes possible nuclear park compositions as a function of time and optimises the reprocessing capacity and the fabricated fuel. Parametrical analysis can be done by changing any of these factors to optimise the consumption of the waste.

For the realisation of the transition calculations, a series of hypotheses has been made for both tools:

- The total net energy production is 800 TWhe for the whole nuclear park.
- At year zero, the whole nuclear park is formed by LWR. The 89% of the energy is generated by reactors using UO_2 as nuclear fuel. The other 11% comes from LWR using MOX fuel. These ratios correspond to the equilibrium situation between these two reactors/fuels.
- Year 2010 is taken as starting point of the study. A certain legacy of nuclear irradiated fuel exists at this year, although there is an additional contribution at year 2022.
- Year 2040 is established as starting point for the industrial development of new reactors (ADS in the case of the double-strata scenario).
- The lifetime of every nuclear reactor is considered to be 60 years.
- The load factor is supposed to be 90% for LWR and 70% for ADS.
- It is assumed that the advanced reprocessing is operative since the beginning of the study (optimistic hypothesis). Thus, reprocessing losses are, in all cases, 0.1% for U, Pu and MA.
- After the corresponding cooling time, it is supposed that the spent fuel is sent to a storage or *pool*. For each reprocessing technology, the required material is extracted from the corresponding pool, considering that the isotopic composition of the pool is homogeneous. Reprocessing is then planned with the objective of minimising the fresh fuel in storage, excepting for the ADS, for which a certain amount of fresh fuel must be prepared beforehand in order to limit the advanced reprocessing necessities while having a moderated rate of introduction of ADS power in the nuclear park.

Results

With the aim of estimating the impact of P&T in the deep repository, the identification and detailed calculation of all the main waste streams have been performed. Table 1 shows the different waste streams appearing in Scenario B2 and the reactor where they are generated.

The detailed isotopic composition (more than 3 300 isotopes in all cases) of all these waste streams has been calculated with ORIGEN2 for the LWR and with EVOLCODE2 for the ADS. With these isotopic compositions, it was possible to estimate the value of those magnitudes having the strongest impact in the deep repository. We will hence concentrate, in this document, in the description of results concerning those magnitudes.

Table 1: Waste stream generated in the different reactors in Scenario B2, calculated using ORIGEN2 (streams coming from LWR) and EVOLCODE2 (streams coming from the ADS)

Waste stream	LWR-UO2	LWR-MOX	ADS
Actinides	Yes	Yes	Yes
Fission fragments	Yes	Yes	Yes Fuel matrix
Impurities activation	Yes	Yes	Yes
Components activation	Structural components	Structural components	Structural components Target Window Beam tube
Spallation products	No	No	Target Window

The assessment of the consequences on waste disposal of the application of partitioning and transmutation (in this study, in Scenario B2) can be achieved once the different aspects concerning the final disposal of the different waste streams have been identified. For the case of the HLW, one of the most potentially critical aspects having an impact in the deep geological repository dimensioning is the thermal output of the waste packages because for all types of host formations, a number of temperature limitations have to be respected. In particular, the host formation temperature has to remain below 100 °C in the underground facility and this is determined by taking a certain minimum distance between two HLW packages which is strongly affected by their thermal power emission.

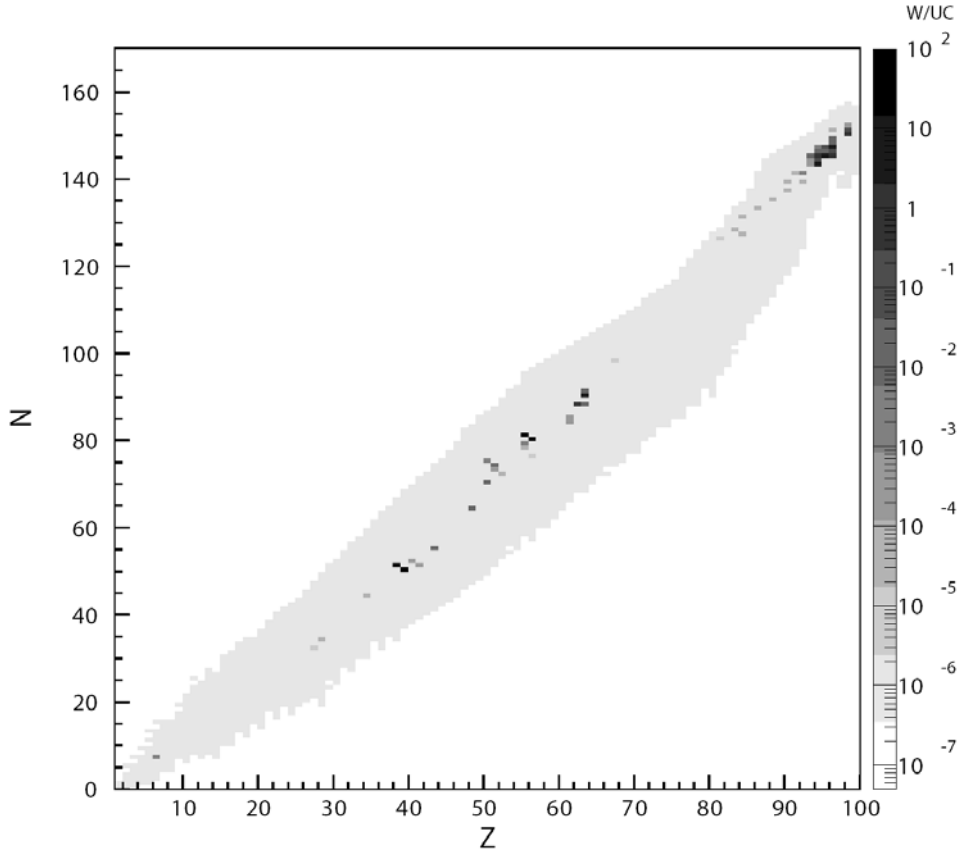
Besides, from the standpoint of safety assessment, significant changes in various components of the repository system are expected. Such changes affect the confidence that can be placed in quantitative calculations at very long times (*e.g.* to one million years in the future). This has led to the concurrent use of complementary safety indicators, such as the radiotoxic inventory flux released from the host formation or into the biosphere. Consequently, we have concentrated our efforts in obtaining detailed data concerning the isotopic composition of each waste stream (and the total amount) which lead to the related magnitudes having a potential impact in the repository: the thermal power and the radiotoxic inventory of the wastes.

The details of the HLW streams, in terms of the isotopic composition, result from the exact operating conditions along the fuel cycle mainly including the original (equilibrium) fuel composition and the burn-up level in the ADS. EVOLCODE2 has provided the list of isotopes (for all the considered waste streams) and the associated value of the thermal power and radiotoxic inventory. This isotope list was required to be as complete as possible due to the different reprocessing hypothesis applicable to each element. For the case of the ADS, different reprocessing factors were to be applied to actinides, noble metals, noble gases, some volatiles elements, Zr from the fuel matrix, etc. Details can be found in Ref. [5].

The contribution to the thermal power of the entire isotope list managed by EVOLCODE2 is shown in Figure 2 for a total cooling time of 50 years, which is when HLW packages are supposed to be sent to the final disposal (only the HLW coming from the ADS are considered here). The diagonal wide band of light grey markers represents all isotopes for which a certain value (even close or equal to zero) has been obtained (more than 3 300 isotopes). The thermal power of these isotopes has to be considered not to scale in the figure. It can be seen that indeed the fission products ^{90}Sr (half-life of 28.8 yr, Z=38 and N=52), ^{90}Y (daughter of ^{90}Sr, with very short half-life, Z=39, N=51), ^{137}Cs (half-life of 30.07 yr, Z=55 and N=82) and ^{137m}Ba (main daughter of ^{137}Cs, with very short half-life, Z=56, N=81) are the largest contributors to thermal power at a cooling time of 50 years. For a total cooling time of 500 years, the total thermal power is approximately two-three orders of magnitude inferior than the thermal power at 50 years of cooling decay time. At this time, unstable fission products with short and medium half-life have already decayed and the total contribution of the fission products becomes lower than the contribution of the actinides.

Figure 2: Thermal power of ADS HLW after 50 years of cooling time per isotope (in W/UC)

The diagonal wide band of light grey markers represents all isotopes for which a certain value (even close to zero) has been obtained (more than 3 300 isotopes). The thermal power of these isotopes has to be considered not to scale.



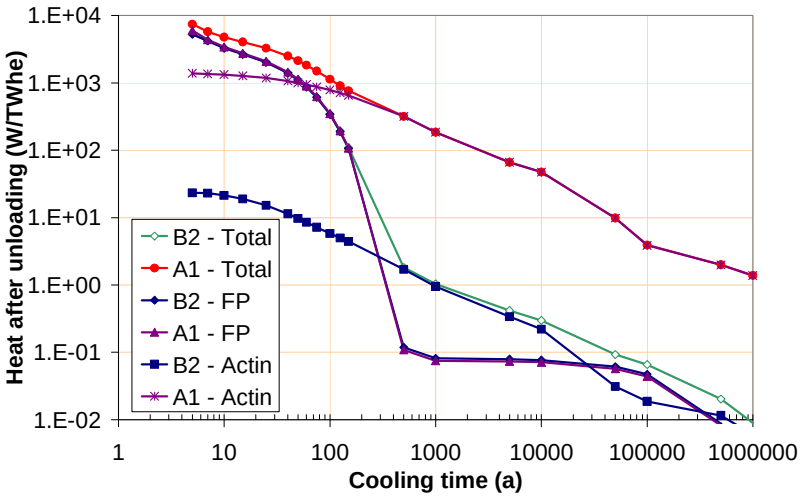
With this information, it has been possible to obtain very precisely the thermal power of the HLW generated per TWhe in Scenarios A1 (reference, the open cycle) and B2 (the double-strata described above) as a function of time (this is shown in Figure 3), identifying the contributions from fission fragments (FP in the figure) and actinides (the fuel impurities activation contributions are always negligible). The composed heat load contribution from the three different HLW packages existing in Scenario B2 have been taken into account, considering the number of waste packages per TWhe generated in each different reactor of the nuclear park.

This figure shows that the larger contributions at the first years of cooling time come, in both scenarios, from the fission products. For Scenario A1, in this period of time, the contribution of the actinides is also considerable, creating the difference between the total values for both scenarios (the same amount of fission products per TWhe are created in both scenarios because it is supposed that both LWR and ADS have the same efficiency of 33%). After several decades of cooling time, when the contribution of the fission products is decreasing due to their shorter half-life, actinides become the largest contribution to the total heat emitted in Scenario A1 (more than three orders of magnitude regarding fission products). As a consequence, the evolution of HLW thermal power is rather slow and takes about 500 years to reduce by a factor 10 and about 10 000 years for a reduction factor 100, with respect to the 50 years reference disposal time.

For Scenario B2, the contribution to the thermal power of the fission products also decreases with time, but it is not exceeded by the contribution of the actinides until a few hundreds of years, due to the reduced amount of actinides remaining in the waste packages. After this period of time, the contribution of the actinides becomes larger but only for several tens of thousands of years; since then both contributions become comparable. The total thermal power for this scenario is reduced by a factor 10 in only 100 years and by more than a factor 100 at year 300 after unloading.

Figure 3: Thermal power of the HLW streams produced in both Scenarios A1 and B2

Values in this graph, for Scenario B2, consider wastes after reprocessing only



As a conclusion, the total heat load per TWhe from scenarios with full Pu and MA recycling, such as B2, at 50 years after unload are approximately half of the open cycle. At 150 years, the heat from this advanced scenario is about 7 to 9 times smaller than in the reference cycle. After several hundreds of years (and later) Scenario B2 has about 100 times smaller heat emission than Scenario A1.

The resulting lengths of the disposal galleries for the fuel cycle Scenarios A1 and B2 depend on the type of host formation. For a deep repository in granite [4], at the reference disposal time of 50 years of cooling time, the required gallery length for the open cycle is 8.89 m/TWhe. The total required gallery length for Scenario B2 with fully recycling of Pu and MA is 4.49 m/TWhe, also considering the reference disposal time. The relative reduction factor is equal to 0.51. Considering the clay formation, the needed gallery length is 5.92 m/TWhe for the open cycle and 2.89 m/TWhe for Scenario B2. The relative reduction factor is 0.49.

The obtained results show the possibility, for scenarios with full Pu and MA recycling, of large gains in the reduction of the thermal load to the deep repository and on its associated capacity by delaying the disposal time 100 to 200 years more or by separating Cs and Sr from the main HLW stream.

EVOLCODE2 has also provided the radiotoxic inventory of the HLW coming from the ADS after a total cooling time of 50 years. Again, fission products ^{90}Sr and ^{137}Cs are the largest contributors at this time. For a total cooling time of 500 years, the contribution of the short- and medium-lived fission products has disappeared and the actinide radiotoxic inventory becomes the largest.

Taking into account the radiotoxic inventory of each single isotope, it has been possible to reconstruct the radiotoxic inventory flux released for Scenarios A1 and B2. It is displayed in Figure 4. The same analysis can be done for this magnitude that those for the thermal output of the wastes. The larger contribution for the radiotoxic inventory in the Scenario A1 comes from the actinides, excepting the very first years. For Scenario B2, the fission products play a more important role until the radioactive decay of the short- and medium-lived isotopes at a few hundreds of years. After several hundreds of years of cooling time, when the main fission products contributors have decayed, Scenario B2 has about 100 times smaller radiotoxic inventory than Scenario A1.

Concerning the transition studies, Figure 5 shows the evolution with time of the nuclear power for the different technologies, LWR-UO_2, LWR-MOX and ADS. It can be seen that, between years 2050 and 2160, the number of ADS must be larger than in the equilibrium, reached in 2160. In Figure 6, a similar result can be found for the ECC method, but showing small differences justifiable by its parametrical characteristics. The main difference in the working hypotheses is that, in ECC, ADS technology is introduced ten years later and with a lower power introduction ratio.

Figure 4: Radiotoxic inventory of the HLW streams produced in both Scenarios A1 and B2

Values in this graph, for Scenario B2, consider wastes after reprocessing only

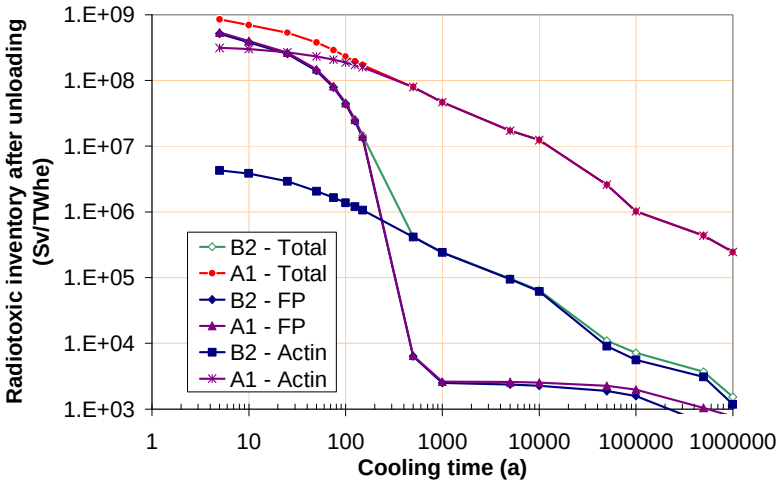


Figure 5: Time evolution of the nuclear power calculated by the EVOLCODE2 method

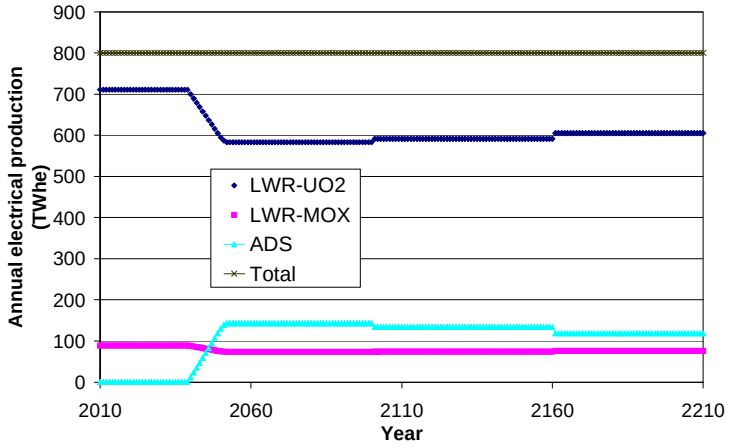


Figure 6: Time evolution of the nuclear power calculated by the ECC method

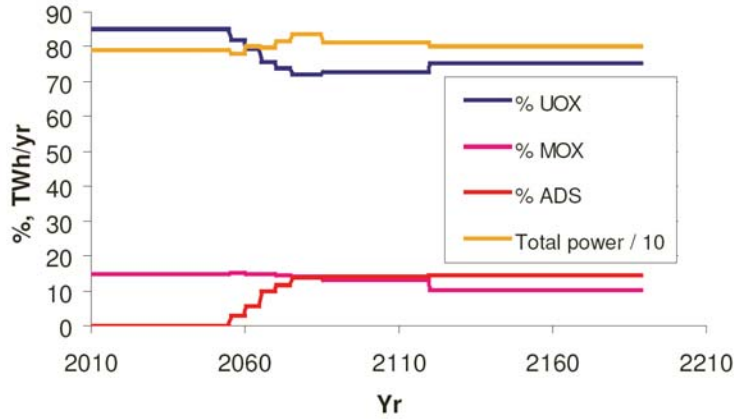
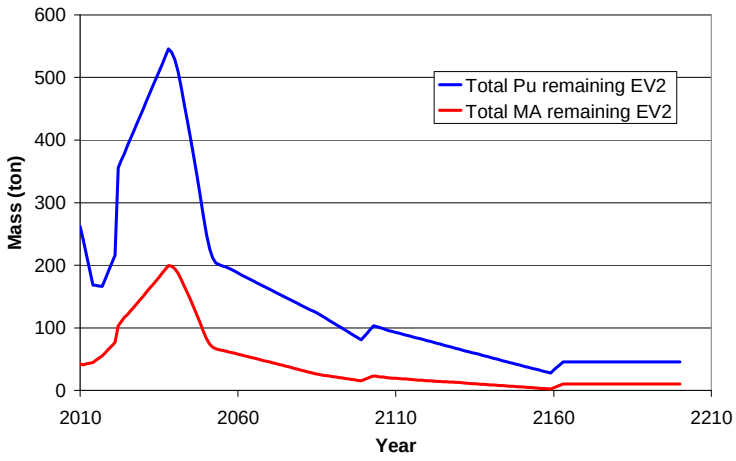


Figure 7 shows the evolution with time of the mass of Pu and MA, accumulated after the corresponding cooling time, that is, the remaining Pu and MA to be recycled or disposed as HLW. It mainly includes Pu and MA amounts waiting for reprocessing and, in smaller amounts, fresh fuel to be introduced in the reactors. In the Pu line, it can be seen that there is a reduction at the very first years. It represents the reutilisation of the Pu existing in the initial legacy pool for the MOX fabrication of these years. This reduction ends when the irradiated fuel coming from the LWRs of the cycle is available (it is supposed that there is no fresh MOX fuel before year zero). The following increasing in the amount of Pu comes from the appearance of the irradiated MOX fuel (after its cooling time), including Pu but also MA, so this increasing also can be seen in the MA curve. In year 2022, there is a strong increasing in both Pu and MA. It is due to the contribution of one of the partner countries to the legacy of LWR spent fuel, whose amount was given for this year instead of 2010. After this point, Pu and MA increase until the ADS technology is introduced in the park. A large amount of Pu and MA is introduced as ADS fresh fuel so their amount decreases. After this time Pu and MA are not considered waste but ADS fuel. The following smooth reductions in the amount of Pu and MA show the effective burning in the ADS. The small increases in Pu and MA in years 2100 and 2160 come from the closure of a certain number of ADS, needed to reach the equilibrium situation. With this closure, the remaining ADS cores are added to the pool of material to be transmuted. From year 2163, the equilibrium in the park is reached.

Figure 7: Evolution with time of the remaining mass of Pu and MA to be transmuted



Figures 8 and 9 show the amounts of reprocessing losses for both methods, with EVOLCODE2 and ECC. This material is supposed to be vitrified and sent to the repository. In addition to the TRUs and fission products, a certain amount of reprocessed Uranium (losses of 0.1%) from LWR-UO_2 and LWR-MOX is also sent to the repository. The rest reprocessed uranium does not have a clear destiny in this study as energetic resource or nuclear waste. In the latter case, less demanding and expensive solutions (than the deep repository) could be applied to the separated U. Since year 2060, curves of an element for both methods show the same slope, meaning that the same amount of material is reprocessed. The total amount (at a particular year) is different due to the different hypothesis of introduction of the ADS technology made by each method.

Conclusions

Activities developed by CIEMAT in the RED-IMPACT Project have allowed the preparation of two calculation models for the analysis of advanced fuel cycle scenarios. These models have been applied to a European scenario with LWR and ADS and provide useful information for the making of long-term strategic decisions. Results obtained in this study show that important reductions in the long-term radiotoxicity amount and heat load can be achieved, together with the generation of a considerable amount of energy. The comparison with other advanced scenarios appears in the final report of the project [4].

Figure 8: U and fission products losses as a function of time

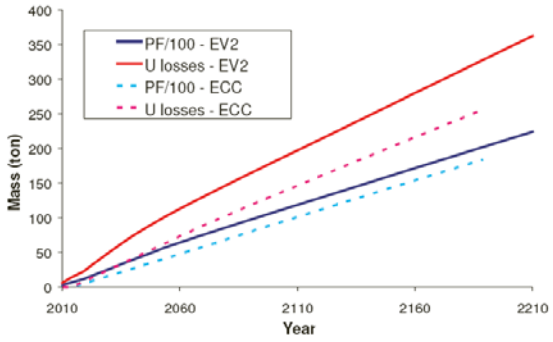
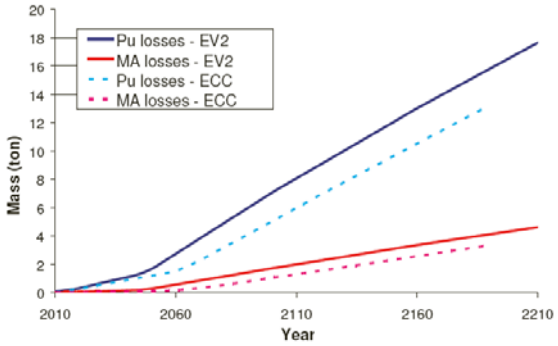


Figure 9: Pu and MA losses as a function of time



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On the consequences of a lifetime extension on the accumulated plutonium mass in Germany

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Abstract

The plutonium production is studied in the German reactor park under the actual political guidelines. The influence of different options (once-through scenario, single and double MOX recycling scenario) on the residual plutonium masses is analysed and compared to a close to reality scenario. All scenarios investigated are extrapolated for the case of a postulated lifetime extension of ten years. In contrast to former studies, a close to reality estimation with a time varying burn-up over the reactor operation period is invented. The consequences on the residual plutonium and minor actinide masses are given and as well the plutonium reduction as the change of the plutonium composition due to single and double recycling are demonstrated.

Introduction

In Germany the actual situation is characterised by a phase out scenario and the shutdown of all reactors after a theoretical life period of 32 years. This is regulated by the so-called "Atomkonsens" signed in 2000/2001 [1]. Due to this contract reprocessing is stopped by 2005 and the lifetime is fixed via the net amount of electricity (roughly 220 TWd) to be produced. The overall thermal energy produced can be calculated to 666 TWd gross in about 32 years [2-5]. These strictly defined boundary conditions offer the possibility for a credible estimation of the accumulated amounts of burnt fuel and the resulting amounts of plutonium and minor actinides. On this basis the influence of the variation of different parameters on the residual plutonium mass can be studied very well. The influence of different scenarios (once-through, idealised single and double plutonium recycling, close to reality plutonium recycling, and lifetime extension) on the residual plutonium and minor actinide mass has been studied on the basis of averaged lifetimes [2]. Owing to the actual discussions about lifetime extension for nuclear power plants in Germany [6,7] the study has been improved by accounting for the time varying burn-up over reactor lifetime [4,5]. Additionally the results have been recently validated with a licensing grade software module [8]. For a better direct analysis of the consequences of a postulated lifetime extension on the plutonium and minor actinide masses the calculation methodology is changed for the study. A time dependent evolution of the burn-up is used for the simulation.

Cycle study

Used calculation methods

The calculations are preformed with the standard software KAPROS (KArlsruher PROgramm System) [9]. The special procedure KARBUS (KARlsruher Burn-up System) [10], see Figure 1, is used. The flux and k_{∞} calculations are performed using 69 energy group cell calculations for best estimate weighting function determination. The calculation is based on the methods of the cell code WIMS [11] and uses collision probability method. The cross-sections are taken from a KAPROS master library based on ENDF/B-6.5. The calculated neutron fluxes are used for group collapsing to create the reaction rates, determined by the one group cross-sections and the one group neutron flux. The reaction rate is forwarded to the burn-up calculations in the module BURN-UP which is based on KORIGEN [12]. The calculation scheme shown in Figure 1 is interrupted when the criticality limit $k_{\infty} = 1.03$ is reached [13]. The end of life burn-up for the number of core burn-up stages (cycles) is then calculated by:

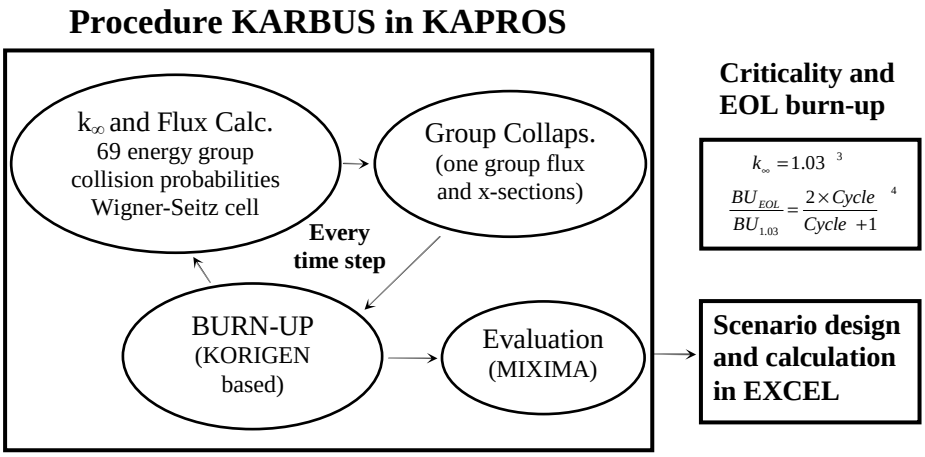
$$\frac{BU_{EOL}}{BU_{1.03}} = \frac{2 \times cycle}{cycle + 1}$$

to take into account the fuel reshuffling in a real reactor core [14]. A calculation up to the end of life (EOL) burn-up determines the final material composition at EOL. These material compositions are finally used for the scenario design performed in MS EXCEL. These burn-up calculations have to be performed for every used EOL burn-up.

The scenarios

The plutonium production is investigated for four different scenarios. The once-through scenario (direct disposal) is the upper limit for the plutonium production. Idealised single and double plutonium recycling scenarios provide the lower limits. Additionally a close to reality scenario which represents the situation in Germany is presented. This situation is characterised by an average MOX use of about 15% in the core and a limited reprocessing of roughly 40% of the used UOX fuel. This is due to the signed contracts for the reprocessing [16] which determines the stop of reprocessing in 2005.

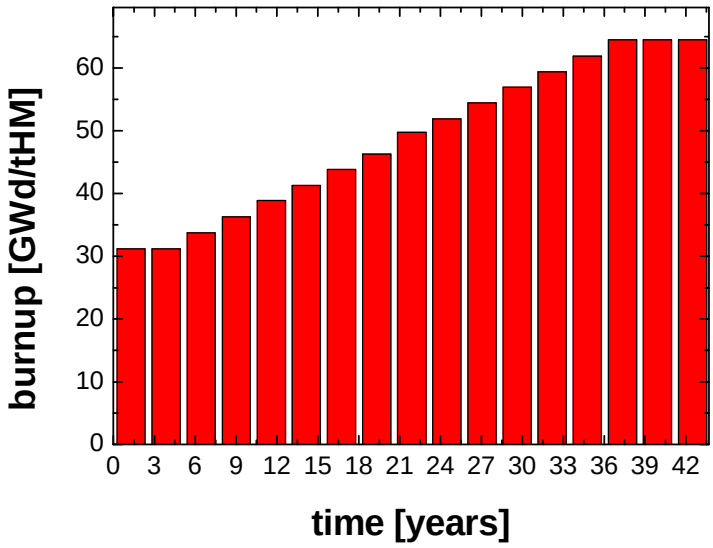
Figure 1: Schematic diagram of the calculation process



The scenarios are characterised by the following boundary conditions:

- *Once-through scenario (OT)*. The burn-up over the complete reactor operation period rises by roughly 1 GWd/tHM per year [15]. The starting value in the end of the seventies has been between 27 and 33 GWd/tHM [16]. An economic burn-up limit is expected for 65 GWd/tHM [17]. This leads to the estimated time dependent burn-up shown in Figure 2. The variable burn-up is achieved with an initial enrichment between 3.2 and 5.4% in the in core use in 716 to 1 480 days in three to five cycles (the averaged duration for fuel elements in the core is set to 940 days to produce a linear burn-up increase). The average linear pin power is 200 W/cm. This value is varied over the 940 days from 150% at beginning of life (BOL) to 50% at end of life (EOL).

Figure 2: Approximate time dependent burn-up in the German reactor park for the constituted lifetime and a postulated lifetime extension of ten years



- *Single plutonium recycling (MOX1)*. The burn-up structure, the cycle data, and the average linear pin power are identical to the once-through scenario. The following times are defined for the reuse of plutonium in MOX1 fuel: three years for storage in the nuclear power plant for cooling, the transport to the reprocessing plant, and the storage in the reprocessing plant and two years for reprocessing, MOX fuel production, and the storage in the nuclear power plant until the insertion into the reactor. This adds up to the overall time of five years until reuse.

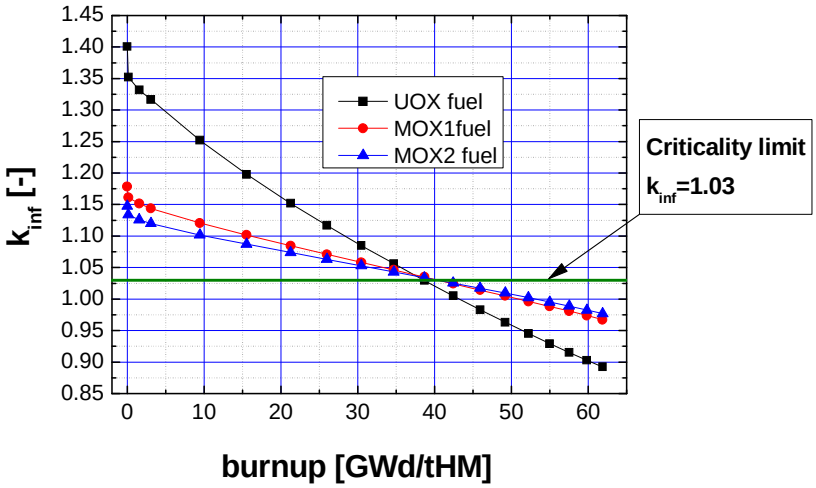
The MOX fuel is produced from roughly 4.4 Pu_{fiss} in depleted uranium to 7.1% Pu_{fiss} in natural uranium. This has to be done to avoid higher Pu_{fiss} contents, which could lead to positive void coefficients [14]. The average linear pin power is varied from 140% (BOL) to 60% (EOL). The averaged MOX content reduces for 18% in the beginning to 11% finally. This is due to the decreasing Pu_{fiss} fraction and the decreasing amount of unloaded fuel elements, both caused by the increasing burn-up.

- *Double plutonium recycling (MOX2)*.The burn-up structure, the cycle data and the average linear pin power are identical to the once-through scenario. The following times are defined for the reuse of plutonium in MOX2 fuel: 10 years for storage in the nuclear power plant for cooling, the transport to the reprocessing plant, and the storage in the reprocessing plant and two years for reprocessing, MOX fuel production, and the storage in the nuclear power plant until the insertion into the reactor. This adds up to the overall time of 12 years until reuse. The MOX fuel is produced from roughly 7.0% Pu_{fiss} in 1.1% enriched uranium to 7.0% Pu_{fiss} in 2.3% enriched uranium. The use of enriched uranium instead of depleted uranium is necessary to avoid an increase of the Pu_{fiss} content above 7% which could lead to problems in the safety parameters [14]. The average linear pin power is identical to MOX1.

- *Close to reality plutonium recycling (rMOX)*.The burn-up structure, the cycle data and the average linear pin power are identical to the once-through scenario. The following close to reality times are defined for the reuse of plutonium: eight years for storage for cooling, the transport to reprocessing, and storage in the reprocessing plant and five years for reprocessing, MOX fabrication, and the storage before insertion into the reactor. This adds up to the overall time of 13 years until reuse. The MOX fuel is produced from roughly 5.7 to 6.5% Pu_{fiss} in depleted uranium with a fraction of 0.3% ^{235}U. The average linear pin power is identical to MOX1. The starting time of the MOX use is calculated to roughly 15 years after the start of operation of the averaged reactor park and lasts for roughly 13.5 years. The average MOX content in the reactor park is 15% in every core. The available separated plutonium amount is 64.6 t from the reprocessing of 6 970 t burnt uranium oxide fuel. For this value implies the decay of ^{241}Pu during the long period of storage. This value is close to the value of roughly 70 t given in literature [16,18].

The time dependent behaviour of k_∞ versus burn-up for three representative different used fuel configurations is shown in Figure 3. To collocate the different fuels in one core all curves for k_∞ versus burn-up have to meet at k_∞ = 1.03 at the identical burn-up. Only under this conditions all fuels can be mixed in one hypothetical reactor core in the scenario design calculations in EXCEL. If this conditions would not be fulfilled a full core calculation would be needed to care for the burn-up behaviour. The difference between the critical value k_∞ = 1.0 and the used limit k_∞ = 1.03 accounts for the losses in a real core configuration.

Figure 3: Burn-up dependent k_∞ evolution for the different fuel materials

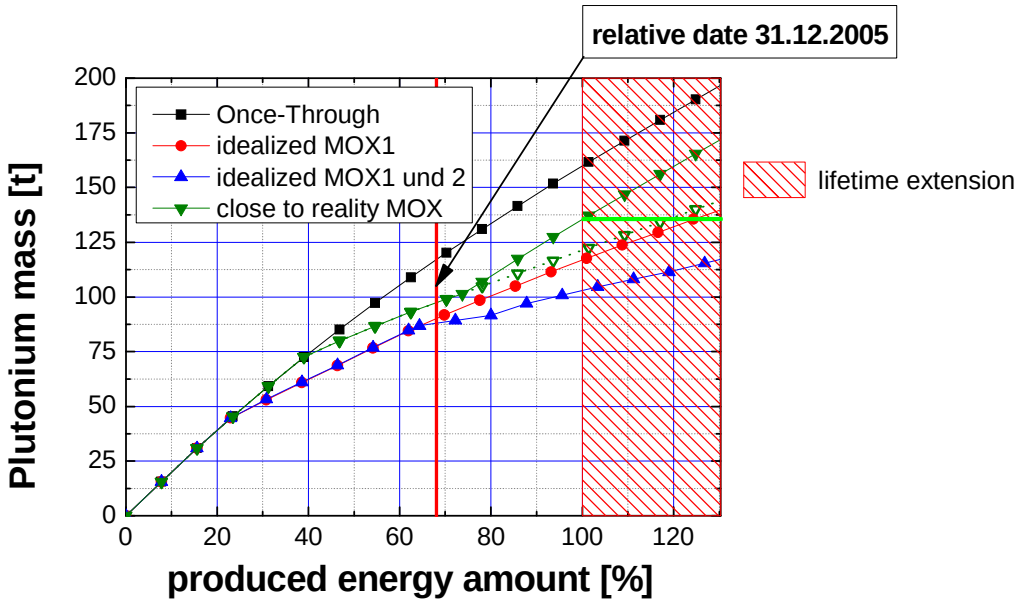


Regular lifetime

The following amounts of the residual plutonium and minor actinides are created during the regular operation of the German reactor park until phase out. In this time the net electric energy amount of roughly 220 TWd electric is produced, which is fixed in the "Atomkonsens". This value can be transformed following the VGB statistics on the operation of nuclear power plants in Germany monthly published in the ATW [19]. Additionally is a postulated thermal efficiency of 34.5% estimated to calculate a thermal energy amount of 666 TWd gross. Roughly 15 800 tonnes of uranium fuel are needed in the once through scenario to produce this amount of energy. Due to the use of MOX fuel this amount can be reduced for the MOX1 case by 10% and for the MOX2 case by 12.5%. The real use of MOX fuel in Germany will reduce the overall needed amount of UOX fuel by 5% for the complete defined electricity amount.

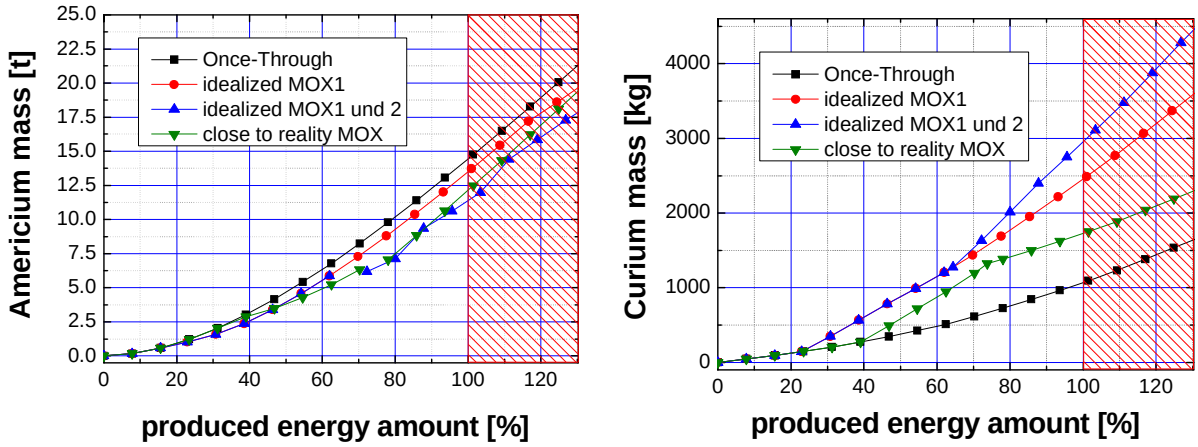
The time dependent accumulated plutonium mass is shown for the different scenarios in Figure 4. In the once-through (black squares) scenario the plutonium mass is rising with a decreasing gradient due to the time dependent increase of the burn-up. After the regular lifetime about 160 tonnes of plutonium are accumulated when 100% of the electric energy is produced. The plutonium mass rises similarly in the single MOX scenario (red circles) for the first roughly 20% of the energy production. In the following time the slope is decreased due to the reuse of plutonium. At the end of the reactor operation period the plutonium amount is reduced by 27% or more than 43 tonnes. In the case of the double plutonium (MOX2 scenario) recycling (blue triangles) the plutonium amount is reduced by roughly 35% or more than 57 tonnes. In the close to reality MOX scenario (green upside down triangles) the MOX use starts when already about 40% of the electricity is produced. At this point is a first break in the slope of the plutonium production due to the reuse in the MOX fuel. The second break is at about 75% of produced electricity. At this point the reuse of plutonium is stopped because no separated plutonium is available anymore due to the termination of reprocessing in the year 2005. The residual plutonium amount is reduced for the close to reality MOX scenario by about 15% or roughly 25 tonnes in the German reactor park. This result could be improved by the re-entry into the reprocessing of burnt uranium oxide fuel. In this case a result close to the result for the idealised single MOX scenario could be reached (see linear extrapolation, dotted green line). The plutonium content in the final storage will be reduced due to the decay of ^{241}Pu in the first 100 years after the shut down of the last reactor by roughly 8 to 12%. The asymptotic amount after the decay of ^{241}Pu is highest for the once-through scenario and lowest for the MOX2 scenario.

Figure 4: Residual plutonium mass for the different scenarios with identical energy production



The time dependent accumulation of minor actinides is shown in Figure 5. The americium amount (left figure) occurring over the regular lifetime is very close for all scenarios. A major reason for this behaviour is in the two fold nature of the americium production. On the one hand americium (^{243}Am) is produced during irradiation of the fuel in the reactor core, on the other hand americium (^{241}Am) is created due to the decay of ^{241}Pu (half-life 14.35 a) during the storage time of the fuel. For all scenarios the americium production is between 11 and 15 tonnes, highest for the once-through and lowest for the MOX2 scenario. The americium amounts in the final storage will be changed due to the different available ^{241}Pu amount at the end of operation for the different scenarios. The asymptotic values about 100 years after the shutdown of all plants are between 27 and 32 tonnes; highest for the close to reality scenario and lowest for the once through scenario. In contrast to americium, the curium amount is only dependent on the irradiation time of the fuel during reactor operation. The accumulation of curium over the produced energy amount is show in the right diagram of Figure 5. The residual curium amount is lowest for the once-through scenario, but even here rises the slope due to the successive prolongation of irradiation time caused by the increasing EOL burn-up. Every recirculation of plutonium into the reactor rises the slope for the curium mass. At end of regular lifetime is the residual curium mass for the rMOX scenario 45% higher, for the MOX1 scenario 118% higher, and for the MOX2 scenario 163% higher than for the once-through scenario. Over all, every scenario with recycling of plutonium causes a significant rise in the residual curium mass due to the longer irradiation time of the plutonium in the reactor core.

Figure 5: Residual minor actinide masses for the different scenarios with identical energy production



Lifetime extension

The postulated lifetime extension for 10 years for the German reactor park is defined in the following way. The electric energy produced in the five years between 2000 [19] and 2005 [20,21] in the 17 end of 2005 still operating nuclear power plants is doubled. This definition leads to an additional energy production of roughly 80 TWd net. The thermal energy production rises from 666 TWd in 32 years to 900 TWd in 42 years.

The energy production is increased by roughly 35%due to the lifetime extension. For this additional energy production is roughly 20% more fresh UOX fuel needed for all considered scenarios. Due to the significantly higher burn-up in the lifetime extension period is the energy production per used tonne of UOX fuel considerably higher than during the regular lifetime.

The accumulated plutonium masses after this additional time period are changed in the following way. In the once-through scenario additional 23% or roughly 37 tonnes of plutonium are created. In the single MOX scenario (MOX1) arise 20% or 23 tonnes more plutonium. In the double MOX scenario (MOX2) only 14% or 15 tonnes of additional plutonium are created. Finally 27% or 37 tonnes more plutonium arises for the close to reality scenario for the case of the termination of the reprocessing in Germany. Due to these numbers in the case of a lifetime extension a re-entry into reprocessing should be discussed. With this option and the continuation of the use of MOX fuel it

would be possible to reach after an additional operation period of 10 years nearly the same residual plutonium amount as for the regular lifetime (see the bright green mark). All scenarios show the identical tendency of under proportional plutonium production compared to the amount of additional energy production. This is a direct consequence of the higher averaged burn-up in the case of a lifetime extension.

Only the minor actinide production is over proportional in the case of a lifetime extension. There will be about 50% more americium for all scenarios and about 50% more curium too. Only in the rMOX case both values are a little bit lower, since it is nothing more than the continuation of a once-through scenario. For the suggestion of a lifetime extension with simultaneously re-entry into the reprocessing and continuation of MOX use the residual americium amount would raise by roughly 6 tonnes or 44%. The residual curium amount would rise by roughly 1.9 tonnes, this is about a doubling of the curium amount.

Conclusions

Basically it has to be mentioned that all presented calculations apply averaging over the German reactor park where every reactor core composition and every cycle is individual. The investigated scenarios are nevertheless reliable estimations of the plutonium production in the German reactor park under changing boundary conditions. The estimation of the accumulated plutonium mass for a scenario close to the reality in Germany shows the benefit of reprocessing and MOX fuel use. This strategy reduces the residual amount of plutonium that has to be stored. A significant reduction of the plutonium amount by roughly 20 tonnes is achieved during the limited time period of MOX use. A part of the possible achievement is lost due to the termination of reprocessing by law in 2005. Without this limiting condition a reduction of about 40 tonnes would be achievable in a single MOX strategy. With a double recycling strategy like in the MOX2 scenario, the plutonium amount can be reduced by roughly 40% or 57 tonnes compared to the one resulting from the once-through case.

This result demonstrates that for partitioning and transmutation (P&T) a first significant step in plutonium reduction can be done in today's standard light water reactors. It has to be mentioned that plutonium is the major contributor with about 90% of the produced actinide mass. New developments will be needed, but only for the further reduction steps for plutonium and minor actinides.

A postulated lifetime extension leads for all scenarios to an under proportional amount of plutonium produced, compared to the additionally produced amount of energy. While the energy production rises by about 35% the plutonium amount is only increased by 13 to 27% depending on the used scenario. However, all recycling scenarios lead to an over proportional increase in the amounts of curium and americium in the final storage. This is caused by both, the increasing discharge burn-up and the postulated lifetime extension.

An additional overview on the results for the different scenarios is given in Table 1. One fact has to be highlighted here. In the case of a lifetime extension with simultaneously a re-entry into reprocessing and continuation of single MOX recycling arises a nearly identical final plutonium amount as for the close to reality scenario defined by the actual situation without lifetime extension in Germany.

Table 1: Overview on the resulting actinide amounts for the different scenarios

	Once-through (OT)	Single MOX (MOX1)	Double MOX (MOX2)	Close to reality MOX (Creal)
Lifetime by law (LL)				
Plutonium tonnes	~160	~117	~103	~135
compared to OT	100%	-27%	-35%	-15%
Americium tonnes	~14.5	~12	~11.5	~13.5
compared to OT	100%	-17%	-21%	-7%
Curium tonnes	~1.1	~2.4	~2.9	~1.7
compared to OT	100%	+118%	+163%	+45%
Lifetime extension				
Plutonium tonnes	~37	~23	~15	~37
compared to LL	+23%	+20%	+14%	+27%
Americium tonnes	~7	~7.5	~6.5	~6.5
compared to LL	+48%	+63%	+57%	48%
Curium tonnes	~0.6	~1.2	~1.5	~0.6
compared to LL	+55%	+50%	+52%	+35%
Overall after 42 years				
Plutonium tonnes	~197	~140	~118	~172
compared to Creal LL	+45%	+3%	-13%	+27%
Americium tonnes	~21.5	~19.5	~18	~20
compared to Creal LL	+60%	+44%	+33%	+48%
Curium tonnes	~1.7	~3.6	~4.4	~2.3
compared to Creal LL	±0%	+112%	+159%	+35%

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Study on partitioning of high-level waste related to geologic disposal area

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Abstract

Partitioning is effective to reduce a burden of a geologic disposal of the high-level waste. Many studies have been performed for that. However, there is few quantitative study related to the reduction of a geologic disposal area. For examining the effect of partitioning on the reduction of the geologic disposal area, we have quantitatively studied about the partitioning method, the fuel-type and the temperature limit of a buffer material. In the study, we have analysed three partitioning scenarios: i) no partitioning scenario; ii) four-group partitioning scenario; iii) element partitioning scenario. The UO_2 and MOX fuels were considered with the fuel exposures of 45 and 70 GWd/ItHM. The temperature limits of a buffer material were investigated between 100 ℃ (373 K) and 200 ℃ (473 K).

The calculations of the fuel burn-up were done using the SWAT code, which is an integrated burn-up calculation code system developed by Tohoku University and Japan Atomic Energy Agency (JAEA). The following were set for calculating the geologic disposal area: i) the limit of oxide fraction; ii) the limit of heat generation; iii) the waiting period before disposing; iv) the heat generation at the site; v) the disposal area for each waste form. These five conditions were set based on Ref. [1]. The conditions of i) and ii) were set to maintain the capability of the glass waste form. The others were set to maintain the temperature of the buffer material less than 100 ℃ (373 K). To verify our calculation method, the disposal areas were compared with the areas for the cases mentioned in Ref. [1]. Both disposal areas were agreed within 5%.

The typical results are the following: i) The four-group partitioning scenario reduces the geologic disposal area to 37% of the area for the no partitioning scenario with UO_2 fuel. ii) For the MOX fuel, the reduction of the geologic disposal area is only 12% with the four-group partitioning scenario. Partitioning is not effective to reduce the disposal area. iii) The element partitioning scenario is not effective for both UO_2 and MOX fuel. This is because the heat generation of Am governs the disposal area for both fuels. iv) When the temperature limit of the buffer material is changed from 100 ℃ (373 K) to 200 ℃ (473 K), the disposal area was reduced to 21% and 32% of the area for both the UO_2 and the MOX fuel. To reduce the geologic disposal area, the reduction (the partitioning and transmutation) of Am and the setting of the temperature limit of buffer material should be studied.

Introduction

Nuclear power generation is expected for main electric sources in the world. Accompanied with the nuclear power generation, however, the problem of high-level radioactive waste (HLW) still remains. In Japan, the vitrification and the geologic disposal are considered for the HLW, however, it is difficult even to determine the site of geologic disposal.

Partitioning is studied for moderating the geologic disposal. Although there are many partitioning methods, the four-group partitioning is expected to be effective and realistic for the HLW. By the four-group partitioning, the elements in the HLW are partitioned into the following four groups: *i)* the minor actinide (MA) group, mainly Np, Am, Cm, etc.; *ii)* the Sr/Cs group; *iii)* the Tc/Pt group, mainly Tc, Pt, Pd, Rh, etc.; *iv)* the "others" group. The details are described later. The four-group partitioning has been studied for many aspects, though, the effect to the geologic disposal, such as the reduction of the area of the geologic disposal, is studied little. The study performed before is for the typical fuel such as the UO_2 fuel with the burn-up 45 GWd/ItHM. No study is made for the high burn-up fuel and the MOX fuel.

In this study, the effect of the partitioning to the area of the geologic disposal is studied quantitatively, not only for a typical fuel (UO_2 fuel, 45 GWd/ItHM) but also for high-burn-up fuel (70 GWd/ItHM) and MOX fuel. This paper first explains the details of the fuels, the burn-ups, the partitioning and the waste forms and the calculation method of the disposal area. Then this study examines the effect of the partitioning to the area of the geologic disposal. In addition, this paper also discusses the setting of the temperature limit for the waste disposal, which is very important issue related to the area of the geologic disposal.

Partitioning, fuel and burn-up

This study considers two fuel types of the UO_2 and MOX fuel and supposes two fuel burn-ups of the exposure of 45 and 70 GWd/ItHM. For these fuels and burn-ups, this study investigates the following three partitioning cases: *i)* the no partitioning case; *ii)* the four-group partitioning case; *iii)* the element partitioning case. In the no partitioning case, partitioning is not considered. In the four-group partitioning case, the HLW are partitioned into the four groups mentioned above. For the four-group partitioning case, the transfer rates of all the elements in the HLW are shown in Table 1 [1]. In the element partitioning case, which is imaginary, all the elements are individually partitioned without loss. The waiting period before reprocessing is set to five years.

Burn-up calculation

The calculation of fuel burn-up is performed by using the SWAT code [2]. The cross-section library is JENDL-3.3. By using the SWAT, the exact pin geometry (fuel, cladding, coolant), the fine treatment of neutron energy and the full inclusion of nuclides (all fission products and actinides) are considered since the SWAT code performs the neutron transport calculation (SRAC), and the burn-up equation calculation (ORIGEN2). The power density of each fuel is set to the 37.0 MW/ItHM (UO_2 fuel) and 36.6 MW/ItHM (MOX fuel).

Waste form

After the partitioning, three kinds of waste forms are supposed in this study: *i)* the original glass waste form; *ii)* the HL (high-waste loading) glass; *iii)* the calcined waste form. In the no partitioning case, all the elements of the HLW are put into the original glass waste form. In the four-group partitioning case, the MA group, the Sr/Cs group, the Tc/Pt group and the others group are separately transferred. Then the MA group and the others group are mixed and solidified to the HL glass. The Sr/Cs group is sintered and solidified to the calcined waste form. In the element partitioning case, except Sr and Cs, the elements whose radioactivity is higher than that of the natural uranium ore (1×10^{12} Bq/ItHM) are solidified individually to the HL glass. Sr and Cs are solidified to the calcined waste form. The others are disposed as the low level waste. The characteristics of each waste form are as listed below:

1) The original glass waste form has an oxide fraction limit of 15 wt.% and a maximum heat generation of 2 300 W per original glass waste form. These values are taken from Refs. [1,3-7].

2) Since the HL glass can contain more oxides than that of original glass, the limits are different. The limit of the oxide fraction for the HL glass is 35 wt.% [5,7]. The maximum heat generation is the same as that of the original glass (2 300 W).

3) The calcined waste form is sintered to the titanic acid or the zeolite since those are the good absorbers of Sr and Cs respectively. For this calcined waste form, the maximum heat generation is 10 000 W which is based on Refs. [1,4].

The specification of each waste form is summarised in Table 2 [1].

Table 1: Transfer rates of the four-group partitioning [1]

		Tc/Pt group	Sr group	Cs group	MA group others group	Secondary
Element	Cr	8.53E-01	1.45E-01	7.20E-06	1.15E-03	9.98E-04
	Fe	8.10E-03	9.89E-03	5.00E-07	1.09E-01	8.73E-01
	Ni	4.97E-03	9.89E-01	4.95E-05	1.11E-03	4.97E-03
	Se	4.45E-02	3.00E-07	3.00E-07	9.55E-01	2.50E-06
	Br	4.50E-05	5.00E-05	5.00E-05	1.00E+00	5.00E-05
	Rb	1.00E-06	5.00E-07	9.90E-01	1.00E-02	1.00E-04
	Sr	1.00E-02	9.90E-01	5.00E-05	3.00E-04	2.00E-11
	Y	5.00E-06	5.00E-05	2.00E-09	1.30E-02	9.80E-01
	Zr	1.00E-06	1.00E-08	5.00E-13	1.00E+00	2.00E-03
	Nb	9.89E-03	9.94E-05	0.00E+00	9.90E-01	5.00E-06
	Mo	5.00E-02	5.00E-07	2.00E-08	9.50E-01	2.00E-03
	Tc	9.80E-01	1.00E-06	1.00E-06	2.00E-02	4.00E-08
	Ru	5.30E-01	3.00E-02	3.00E-06	4.30E-01	4.00E-03
	Rh	8.95E-01	8.95E-02	5.00E-06	1.10E-02	4.97E-03
	Pd	8.90E-01	9.00E-02	5.00E-06	2.01E-02	4.00E-03
	Ag	9.88E-02	8.09E-01	4.50E-05	9.09E-02	9.98E-04
	Cd	9.88E-02	8.09E-01	4.50E-05	9.09E-02	9.98E-04
	In	8.89E-01	9.79E-02	5.40E-06	1.19E-02	9.98E-04
	Sn	6.56E-01	1.54E-02	7.70E-06	2.41E-01	8.73E-02
	Sb	6.56E-01	1.54E-02	7.70E-06	2.41E-01	8.73E-02
	Te	4.45E-02	3.00E-07	3.00E-07	9.55E-01	2.50E-06
	I	4.50E-05	5.00E-05	5.00E-05	1.00E+00	5.00E-05
	Cs	1.00E-06	5.00E-10	1.00E+00	1.00E-04	5.00E-12
	Ba	1.00E-02	9.90E-01	5.00E-05	3.00E-04	2.00E-11
	La	1.00E-05	9.00E-05	5.00E-09	1.00E+00	1.00E-04
	Ce	1.00E-05	9.00E-05	5.00E-09	1.00E+00	1.00E-03
	Pr	9.90E-06	9.00E-05	0.00E+00	1.00E+00	1.70E-04
	Nd	1.00E-05	9.00E-05	5.00E-09	1.00E+00	1.00E-04
	Pm	9.90E-06	9.00E-05	0.00E+00	9.99E-01	5.99E-04
	Sm	1.00E-05	9.00E-05	5.00E-09	1.00E+00	1.00E-03
	Eu	9.90E-06	9.00E-05	0.00E+00	9.98E-01	1.77E-03
	Gd	9.90E-06	9.00E-05	0.00E+00	9.98E-01	2.38E-03
	Tb	9.90E-06	9.00E-05	0.00E+00	9.94E-01	5.96E-03
	Dy	9.90E-06	9.00E-05	0.00E+00	9.94E-01	5.96E-03
	Ho	9.90E-06	9.00E-05	0.00E+00	9.94E-01	5.96E-03
	Er	9.90E-06	9.00E-05	0.00E+00	9.94E-01	5.96E-03
	Tm	9.90E-06	9.00E-05	0.00E+00	9.94E-01	5.96E-03
	Yb	9.90E-06	9.00E-05	0.00E+00	9.94E-01	5.96E-03
	U	4.00E-06	5.00E-05	2.00E-09	1.00E+00	1.00E-03
	Np	4.00E-05	5.00E-04	2.00E-08	1.00E+00	5.00E-05
	Pu	4.00E-05	1.00E-05	5.00E-10	1.00E+00	5.00E-05
	Am	5.00E-06	5.00E-05	2.00E-09	1.00E+00	3.00E-08
	Cm	5.00E-06	5.00E-05	2.00E-09	1.00E+00	3.00E-08
	Others	0.00E+00	0.00E+00	0.00E+00	1.00E+00	0.00E+00

Table 2: Specification of the original glass, the HL glass and the calcined waste form

Waste form	Original glass, HL glass	Calcined waste form	
Material/absorber	Borosilicate glass	Titanic acid (Sr)	Zeolite (Cs)
Density [g/cm^3]	2.67	4.20	2.47
Volume [m^3/waste form]	0.15	0.15	

Disposal area

We adopt one of the most standard disposal methods in Japan [1]. The method considers the vertical and horizontal emplacement of the waste form. In this study, we only suppose the vertical emplacement which the waste form are placed vertically inside the disposal pit in the crystalline rocks as shown in Figure 1. The parameters of the disposal are shown in Table 3 and the disposal area is 44.4 m^2 per waste form [1].

Figure 1: Schematic view of the vertical emplacement

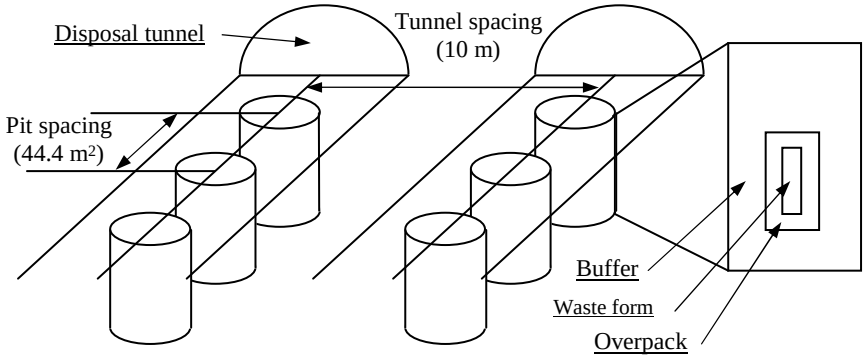


Table 3: Repository specifications for geologic disposal

Waste form	Original glass, HL glass, calcined waste form 0.43 m diameter, 1.34 m height
Overpack	Carbon steel, 0.19 m thick
Buffer material	Bentonite-silica sand mixture, 0.7 m thick
Emplacement configuration	Vertical
Rock	Crystalline
Repository	1 000 m
Initial temperature of repository	318 K
Temperature limit of buffer material	373 K

To maintain the integrity of the geologic disposal, the temperature limit is required to each waste form. The limit of the temperature is defined at the buffer material and is set to 373 K. To fulfil this condition, the heat generation for each waste form is restricted. The limits of the heat generation for the three waste forms are already determined: 350 W for the original glass, 161 W for the HL glass, 10 000 W for the calcined waste form [1].

Effect of partitioning

In this section, the effect of the partitioning to the disposal area is examined for the UO_2 fuel with the burn-up of 45 GWd/ItHM by comparing with the disposal areas for the three partitioning cases. The details of the disposal group, the waste form and the specification of the waste forms are shown in Table 4 [1].

Table 4: Conditions of waste forms

Case		No partitioning	Four-group partitioning			Element partitioning	
Waste form		Original glass	HL glass	Calcined waste form		HL glass	Calcined waste form
Disposal target		HLW	MA group others group	Sr group	Cs group	Elements other than Sr and Cs	Sr Cs
Waste form specifications	**Waste loading limit [wt.%]**	15.0	30.0	9.9	14.3	35.0	9.9 14.3
	Heat generation limit [W/waste form]	2 300	2 300	–		2 300	10 000
	Storage period [year]	85	85	130		85	130
	Heat generation limit at disposal [W/waste form]	161	161	–		161	–
	Disposal area per waste form [m^2/waste form]	44.4	44.4	44.4		44.4	44.4

The calculated results of the disposal area and the heat generation in each partitioning case are shown in Figures 2 and 3. The element partitioning case reduces the disposal area to 39% of that for the no partitioning case. The four-group partitioning case also reduces the disposal area to 37% of that for the no partitioning case. The effect of the partitioning is large for reducing the disposal area.

The reduction effect between the four-group and element partitioning is almost the same. This is reasonable considering the influence of the temperature limit as follows: *i)* The temperature limit restricts the heat generation. *ii)* The heat generation of Am (^{241}Am) is very large and Am dominates the total heat generation. *iii)* The concentration of Am per waste form is limited to be very small. *iv)* The concentration limit becomes constant independent to the partitioning method. *v)* Finally, in order to satisfy the temperature limit, the number of the waste form does not depend on the method of partitioning. This situation is shown in Figure 3.

Figure 2: Disposal area in each partitioning case

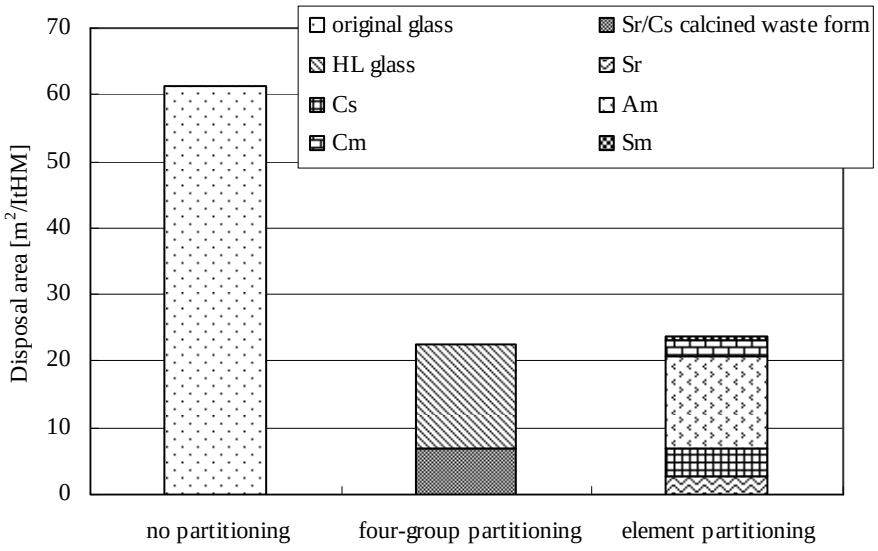
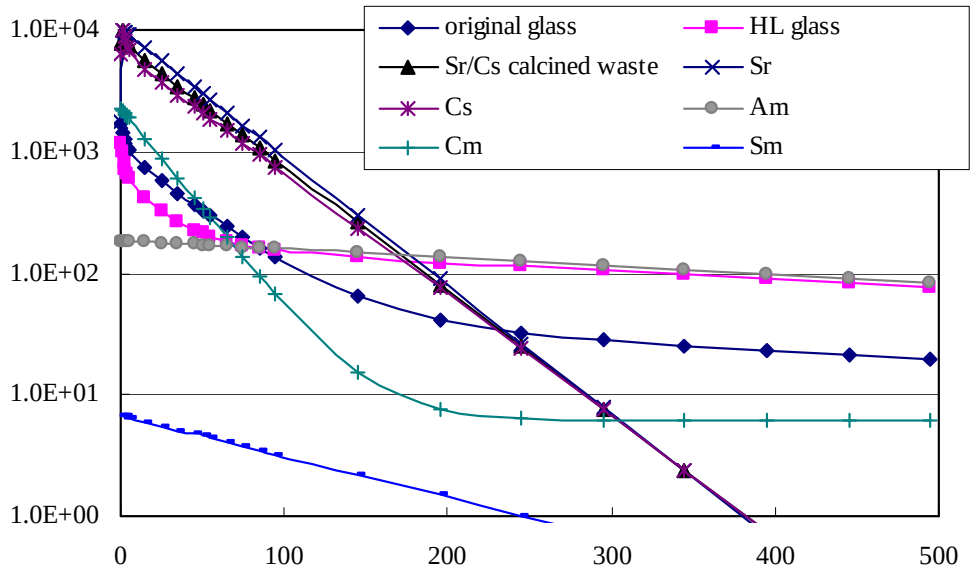


Figure 3: Heat generation of waste form in each partitioning case



Based on these results, it is found that the effect of the partitioning is large for reducing the disposal area. It is also noted that the reduction effect to the disposal area is not improved by changing the partitioning method because of high heat generation of Am. The transmutation of Am will be effective for the reduction of the disposal area.

Fuel burn-up

This section examines the effect of the fuel burn-up to the disposal area. We calculate the disposal area for different fuel burn-ups of 45 and 70 GWd/ItHM. The three partitioning cases mentioned above are considered.

Figure 4 shows the result of the disposal area for the burn-up 45 and 70 GWd/ItHM for the UO_2 fuel. Figure 5 is the same for the MOX fuel. The disposal area of 70 GWd/ItHM is 1.3~1.6 times larger than that of 45 GWd/ItHM. The ratios almost equal to the ratio of the fuel exposures (*i.e.* 70/45 = 1.5). It is concluded that the effect of the fuel burn-up is liner to burn-up exposure for both the UO_2 fuel and the MOX fuel and is not large.

MOX fuel

Figure 6 shows the result of the MOX fuel comparing the UO_2 fuel for the fuel exposure of 45 GWd/ItHM. Figure 7 is the same for the fuel exposure of 70 GWd/ItHM.

The disposal area of the MOX fuels largely increases compared with that of the UO_2 fuel. The disposal areas of the MOX fuels are 3.5~4.0 times larger than that those of the UO_2 fuel for the no partitioning case, and 8.1~9.6 times larger than those of the UO_2 fuel for the four-group and the element partitioning cases. The increase of the disposal areas are due to the increase of the amount of Am which has high heat generation. Owing to the effect of Am, the disposal area for the two partitioning of element and four-group becomes the same. It is concluded that the disposal area for the MOX fuels largely increase compared with the UO_2 fuel.

Figure 4: Disposal area for different burn-ups 45 and 70 GWd/ltHM (UO_2 fuel)

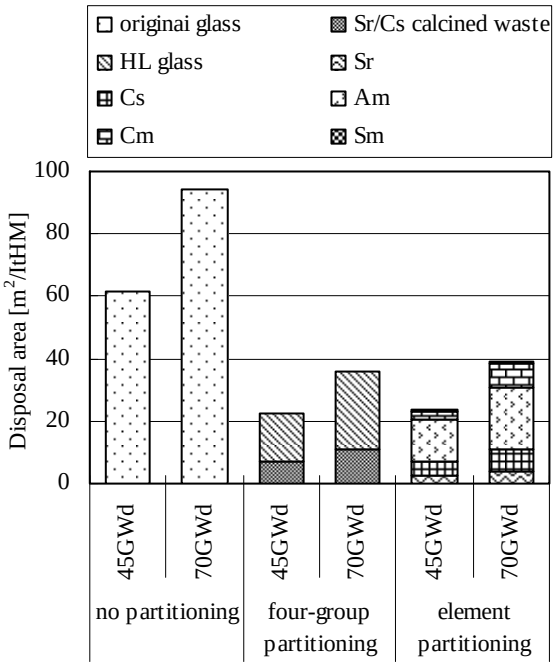


Figure 5: Disposal area for different burn-ups 45 and 70 GWd/ltHM (MOX fuel)

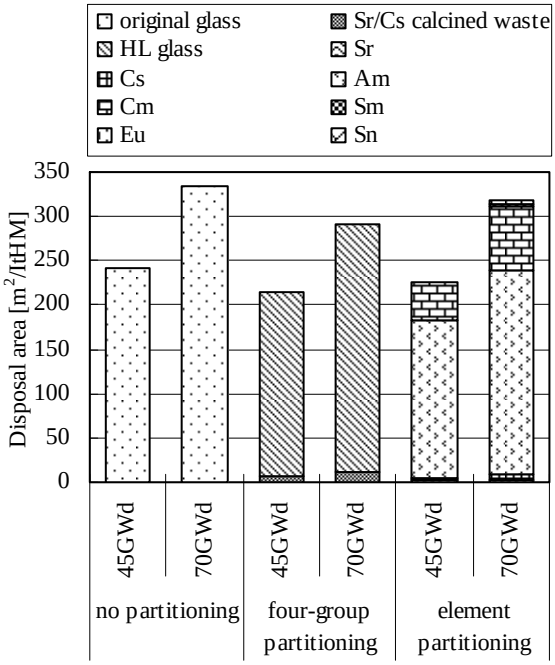


Figure 6: Disposal area for different the UO_2 and MOX fuel (45 GWd/ltHM)

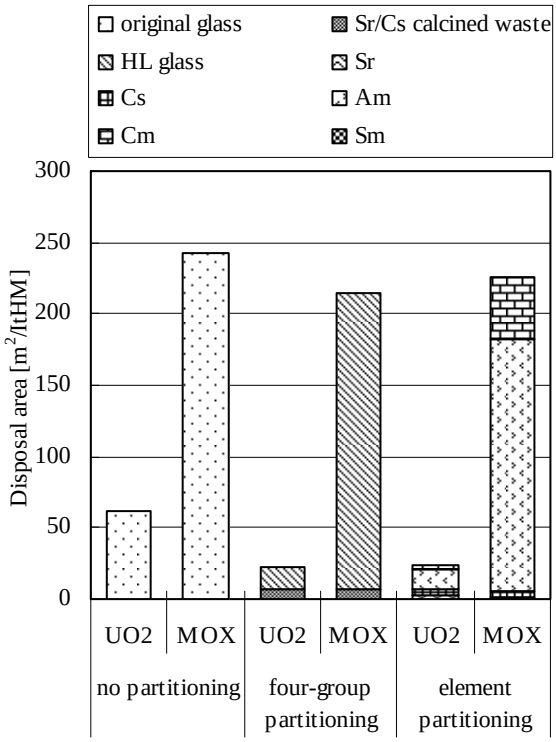
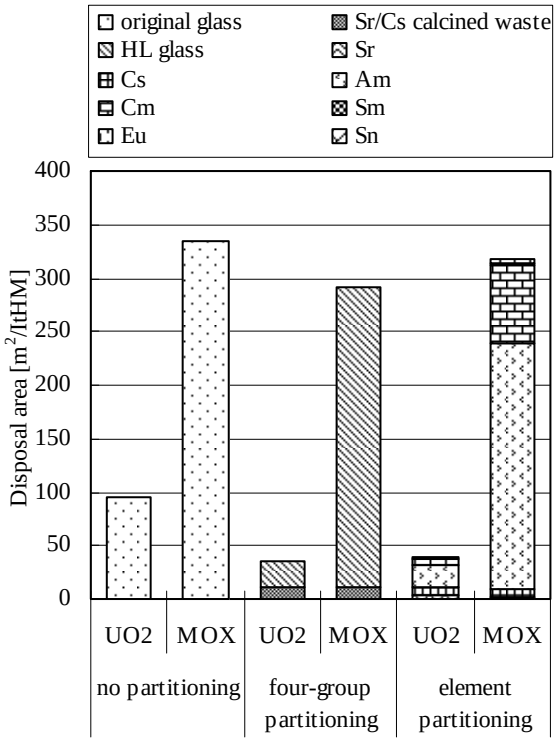


Figure 7: Disposal area for different the UO_2 and MOX fuel (45 GWd/ltHM)

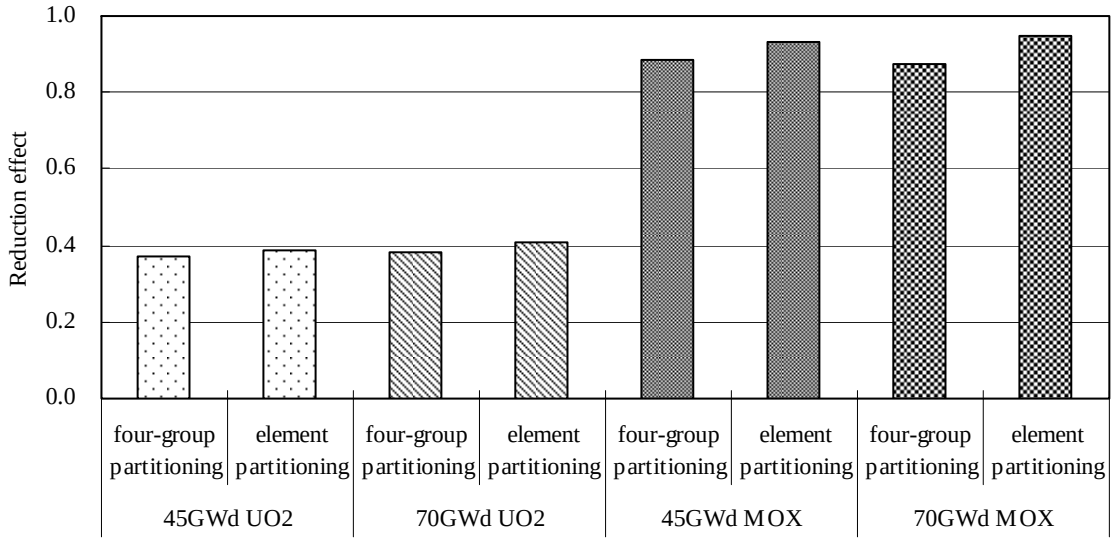


Reduction effect

Figure 8 shows the reduction effect of the partitioning to the disposal area compared the four-group partitioning case or the element partitioning case with the no partitioning case for the two fuels and the two burn-ups mentioned above.

For the UO_2 fuel, the reduction effect at the burn-up of 45 and 70 GWd/ItHM is about 40% for the four-group partitioning and the element partitioning case. The effect of the partitioning is not large since the disposal area mainly determined by Sr and Cs. However, the reduction effects for the MOX fuels remain to 87% for the four-group partitioning case and 93% for the element partitioning case at the burn-up of 45 and 70 GWd/ItHM. For the MOX fuel, the effect of Sr and Cs is relatively small and the Am contribution becomes very large. Therefore, the effect of the partitioning becomes little in the case of the MOX fuel. To reduce the area of the geologic disposal for the MOX fuel, the transmutation after partitioning will be important for MA, especially for Am.

Figure 8: Reduction effect of disposal area compared the four-group partitioning case or the element partitioning case to the no partitioning case



Temperature limit of buffer material at disposal area

As mentioned above, the disposal area is dominated by the temperature limit (373 K). A higher temperature limit will result in smaller disposal area. In this study, the effect of the temperature limit on the disposal area is also investigated quantitatively, considering the same partitioning case and the fuels.

In this study, the disposal areas for different temperature limits are determined based on a database which incorporates the analysis results of heat conduction equation for heat sources with various concentrations of waste element (mainly Am) [1]. The database is prepared for the temperature limit from 373 to 473 K with an interval of 25 K. By using the database, the concentration of waste elements can be determined for each temperature limit and the disposal area is obtained corresponded to the concentration of waste elements.

The results for the UO_2 and MOX fuel are shown in Figures 9 and 10 respectively. The disposal area largely varies by changing the temperature limit. For the temperature limit of 473 K, the reduction ratio of disposal area is about 20% and 33% for the UO_2 and MOX fuel, respectively. It is concluded that the influence of the temperature limit on the geologic disposal is very large. The effect is higher for the MOX fuel than that for the UO_2 fuel.

Figure 9: Disposal area changing temperature limit (45 GWd/ltHM, UO_2)

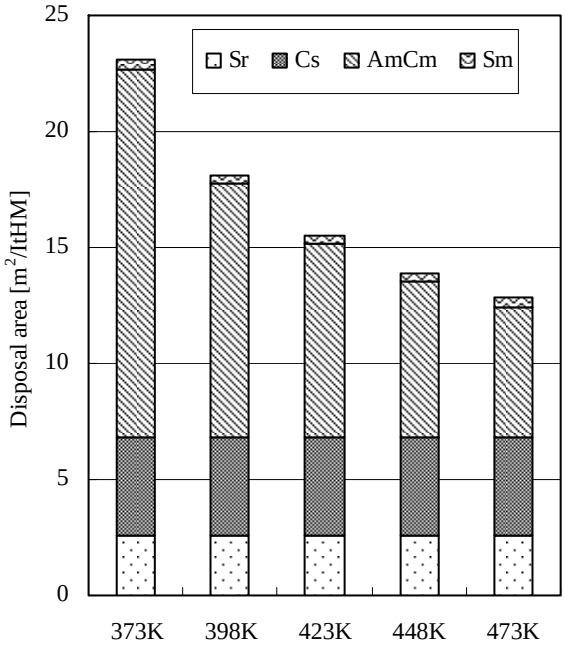
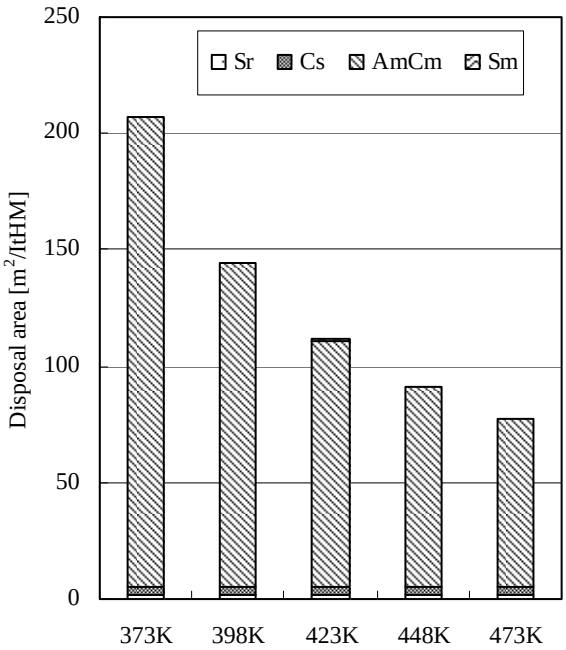


Figure 10: Disposal area changing temperature limit (45 GWd/ltHM, MOX)



Conclusion

In this study, the effect of the partitioning on the reduction of the area of the geologic disposal is studied quantitatively. We considered three partitioning methods and two fuel types and two burn-up exposures. The three partitioning methods are: *i)* the no partitioning case; *ii)* the four-group partitioning case; *iii)* the element partitioning case. The two fuel types are the UO_2 and MOX fuel and the two fuel burn-ups are the exposure of 45 and 70 GWd/ItHM. For the waste after the partitioning, three kinds of waste forms are supposed in this study: *i)* the original glass waste form; *ii)* the HL (high-waste loading) glass; *iii)* the calcined waste form. The amount of waste elements per waste form is determined considering the temperature limit at the buffer in the geologic storage. To estimate the disposal area, the most standard disposal method in Japan is adopted, and the disposal area is 44.4 m^2 per waste form.

In this study, we conclude the following:

- The effect of the partitioning is large for reducing the disposal area.
- The reduction effect to the disposal area is not improved by changing the partitioning method because of high heat generation of Am.
- The effect of the fuel burn-up is linear to burn-up exposure for both the UO_2 fuel and the MOX fuel and is not large.
- The disposal area for the MOX fuels largely increase compared with the UO_2 fuel.
- The effect of the partitioning to the disposal area becomes little for the MOX fuel.
- The influence of the temperature limit in the geologic disposal is very large. The effect is higher for the MOX fuel than that for the UO_2 fuel.

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