ANALYSIS OF THE TRANSMUTATIONAL CHARACTERISTICS OF A NOVEL MOLTEN SALT REACTOR CONCEPT

Gy. Csom, S. Fehér, M. Szieberth
Budapest University of Technology and Economics

ABSTRACT

One of the arguments most frequently brought up by the opponents of the utilization of nuclear energy is the requirement that the radioactive waste and the long-lived radioisotopes accumulated in the spent fuel should be isolated for a very long time from the biosphere. The solution is the elimination of long-lived actinides (plutonium isotopes and minor actinides) and long-lived fission products by transforming (transmuting) them into short-lived or stable nuclei. The high neutron flux required for transmutation can be realized in nuclear installations. These may be conventional thermal and fast reactors, furthermore dedicated devices, namely thermal and fast reactors and accelerator driven subcritical systems (ADSs), which are specifically designed for this purpose. Some of the most promising systems are the molten salt reactors and subcritical systems, in which the fuel and material to be transmuted circulate dissolved in some molten salt. In the present paper this transmutational device, as well as recommendations for the improvement are discussed in detail.

The goal is the development of a transmutational technique along with a device implementing it, which yield higher transmutational efficiencies than that of the known procedures and thus result in radioactive waste whose load on the environment is reduced both in magnitude and time length. The procedure is the multistep transmutation, in which transformation is done in several consecutive steps of different neutron flux and spectrum. In order to implement this, a multiregion transmutational device, i.e. nuclear reactor or subcritical system is proposed, in which several separate flow-through irradiation rooms are formed with various neutron spectra and fluxes.

In order to compare a "conventional" MSR and a proposed MRMSR in terms of efficiency, preliminary calcualtional results are shown. We carry on further calculations in order to find the optimal implementation of this new concept and to emphasize its other advantageous features.

INTRODUCTION

One of the arguments most frequently brought up by the opponents of the utilization of nuclear energy is the requirement that the radioactive waste and the long-lived radioisotopes accumulated in the spent fuel should be isolated for a very long time from the biosphere. In the past 40 years, deep geological repositories have been considered as the primary method of nuclear waste final disposal. However, concerns about this method delayed the application of geological repositories. There is still no consensus whether the nuclear waste processing with conventional methods and final geological disposal are acceptable practice for humanity. A solution, which is fundamentally different from the one mentioned and new in principle, is the elimination of long-lived actinides (plutonium isotopes and minor actinides) and long-lived fission products by transforming (transmuting) them into short-lived or stable nuclei. In this manner the amount of radioactivity present in the waste can be decreased and the time required for the radioactivity to decay to an acceptable level can be shortened radically. Although this would not make final storage (typically geological) unnecessary, the conditions of storage can be dramatically improved as well as the required isolation period can be reduced by several orders of magnitude. The storage facilities would thus be easier to design and operate and transmutation may help to form a more convincing image for the inhabitants. These issues are dealt with in our earlier work [1], while certain aspects are analyzed in our papers [2,3] being parallely presented at another international conferences.

The high neutron flux required for transmutation can be realized in nuclear installations. These may be conventional thermal and fast reactors, furthermore dedicated devices, namely thermal and fast reactors and accelerator driven subcritical systems (ADSs), which are specifically designed for this purpose. Some of the most promising systems are the molten salt reactors and subcritical systems, in which the fuel and material to be transmuted circulate dissolved in some molten salt. In the present paper this transmutational device, as well as recommendations for the improvement are discussed in detail.

1 - ANALYSIS OF THE KNOWN MOLTEN SALT REACTOR CONCEPTS IN RESPECT TO TRANSMUTATION

The Molten Salt Reactor (MSR) Program started in the US in 1956. At the end of the 50s and beginning of 60s the interest towards breeder reactors rose and support to the MSR program could be obtained by verifying the breeding capability. The design of the facility called MSR Experiment started in 1960 [5]. The reactor with 8 MW maximum thermal power went critical in 1965 and was operating almost continuously until December 1969, when it was shut down due to the lack of financial support. Later the plans of an installation with 1000 MW_e called MS Breeder Reactor Experiment (MSBR) were completed, but finally, in 1976 the entire molten salt reactor program was stopped due to financial reasons.

The basic idea of transmutation, which can be traced back to the 40s, was brought up again in the 70s. After a temporary slow down the research obtained new impulses at the end of 80s / beginning of 90s. To a large extent, this was backed by the R&D project initiated by the Japanese government in 1988 and the regular information exchange meetings held afterwards by OECD/NEA. The idea of molten salt reactors and, according to Rubbia's thoughts, the concept of accelerator driven molten salt subcritical systems were again proposed. Although the conception of the latter system was worked out at the Los Alamos National Laboratory (LANL), different concepts have become known [8-19; 26] since then. At the AER symposia of the VVER reactor operating countries several papers related to this topic have been presented in the past few years [20-25].

In the LANL concept, a 11 mA / 800 MeV particle beam generated by a linear proton accelerator is directed into a cylindrical lead target 22.5 cm in radius and 100 cm in height. This is surrounded by a vessel filled either with heavy water or graphite, with the aid of which a thermal neutron flux of 2.45×10^{14} n/cm^{2}s can be achieved. The molten salt containing the isotopes to be transmuted circulates in the subcritical system with k_{eff}=0.96. The warmed up molten salt transfers its heat to the secondary fluid in a heat exchanger. The following compositions were considered as candidates for molten salt: fluoride or chloride basis minor actinide and fission products salts.

According to the concept of the Japanese Atomic Energy Research Institute (JAERI), a target located in a subcritical environment would be bombarded using a similarly high energy (1.5 GeV) and high intensity (25 mA) proton beam. Therefore, differently from the LANL concept, in this case the target, fuel and coolant are the molten salt itself, in which spallation neutrons are produced. The goal to be achieved with this system was the transmutation of minor actinides, for which chloride basis salt compounds and NaCl and $PbCl_2$ salts have been found suitable. The highest neutron flux is approximately 10^{14} n/cm^{2}s, while the thermal power is about 800 MW, which makes the device capable of burning out about 250 kg of actinides per year. The outlet temperature of the primary molten salt is approx. 650 °C. Part of the device is a separation unit, which is connected to the primary circuit and operated continuously.

Considering the above written and other publications, it can be stated that the major advantage of the molten salt transmutational device over the conventional solid fuel heterogeneous systems is that it is far easier to connect a separation device to the transmutational one. On the other hand, the known molten salt systems have their disadvantages too. Since the molten salt containing the isotopes to be transmuted constitutes one single space in the reactor or subcritical system, it is impossible to utilize the advantages due to the spatial distribution of neutron spectrum and neutron flux in order to improve the transmutational efficiency. There is a possibility to utilize the above advantages in heterogeneous systems by appropriately arranging the fuel elements. A common feature of the known transmutational processes is a time scheduling which only modifies the duration that elapses between the removal of the spent fuel from the nuclear reactor to the start of the transmutation. A multistep time-scheduled transmutational strategy was not taken into consideration during which the transmutation of certain isotopes is suspended and only continued after some "cooling" time in the same or in a different transmutational device in order to enhance the transmutational efficiency and decrease the long-term radiotoxicity. Scheduling

and finding the proper number and order of steps make it possible to develop the optimal transmutational strategy. A common shortcoming of all of the known transmutational concepts is that they do not or only partly make use of this opportunity.

As a consequence of the above written facts, neither the molten salt reactors nor the accelerator driven molten salt subcritical systems are capable of achieving as high transmutational efficiency and as effective decrease in radiotoxicity as they would be capable of by the application of an optimal strategy. However, the optimal strategies require modifications to the known solutions of the molten salt systems.

2 - THE MULTIREGION MOLTEN SALT REACTOR AND SUBCRITICAL SYSTEM

The goal is the development of a transmutational technique along with a device implementing it, which yield higher transmutational efficiencies than that of the known procedures and thus result in radioactive waste whose load on the environment is reduced both in magnitude and time length.
The procedure is the multistep transmutation, in which transformation is done in several consecutive steps of different neutron flux and spectrum. In order to implement this, a multiregion transmutational device, i.e. nuclear reactor or subcritical system is proposed, in which several separate flow-through irradiation rooms are formed with various neutron spectra and fluxes. The characteristics of these rooms may differ significantly in the case of ADSs, particularly if certain regions are filled with some moderating material.

It is known that the long-lived radioisotopes found in spent fuel are categorized as actinides and fission products. The first group is made up of the so-called dominant actinides (plutonium isotopes) and minor actinides (mainly Np-237, Am-241, Am-242, Am-242m, Am-243, Cm-243, Cm-244, Cm-245 and Cm-246 isotopes), while the second group is the group of long-lived fission products (mainly Tc-99 and I-129 isotopes).

The fission products can typically be transformed by capturing thermal neutrons at adequate efficiency. For this purpose such transmutational devices are necessary, in which (in the total or part of the volume) the thermal neutron flux is high. This can be achieved in an ADS or in a part of it, practically an external region, which is surrounded by a reflector of suitable thickness from the outer surface.

As a result of neutron capture, the actinides in most case transform into isotopes with even longer half-life and larger mass number, which is an undesired change in the present case. Here the neutron induced fission reactions lead to beneficial changes. Certain actinides (Pu-238, Pu-239, Pu-241, Am-242m, Cm-243 and Cm-245) are fissile upon interaction with thermal neutrons (the cross sections are large), but, unfortunately, their neutron capture cross section is also large in this energy region, which results in the production of a significant amount of heavier actinides. Upon the effect of higher energy neutrons all the actinides are fissile, while the ratio of fission to

capture cross section is far greater than in the case of thermal neutron induced nuclear reactions. One can conclude from Figs. 1 through 9 that this ratio increases as the neutron energy increases. Table 1 also shows this tendency with figures. In view of these facts, those devices are advantageous for actinides, in or in part of which the fast neutron flux is high. Such candidate might be e.g. the innermost region (the region closest to the proton beam or target if any) of an ADS. In the case of some of the actinides specific solutions may be desirable. The Np-237, Am-241 and Am-243 go through the following nuclear reactions and decay processes:

$$^{237}Np(n,\gamma)^{238}Np \xrightarrow[2,1d]{\beta^-} {}^{238}Pu(n,\gamma)^{239}Pu(n,f)$$

$$^{241}Am(n,\gamma)^{242}Am \xrightarrow[16h]{\beta^-} {}^{242}Cm \xrightarrow[163d]{\alpha} {}^{238}Pu(n,\gamma)^{239}Pu(n,f)$$

$$^{243}Am(n,\gamma)^{244}Am \xrightarrow[10h]{\beta^-} {}^{244}Cm \xrightarrow[18,1d]{\alpha} {}^{240}Pu(n,\gamma)^{241}Pu(n,f)$$

The first (n,γ) reactions occur at high probability in devices with thermal spectrum. In the case of the reaction and decay product nuclei the fast neutrons are favourable. Accordingly, for these three isotopes the so called double step transmutation may yield the best results. The irradiation should be performed in thermal and fast fields in the first and second steps, respectively.

According to the above considerations, in an optimal transmutational strategy it may be useful to apply irradiations of different neutron spectra and combine these steps corresponding to an elaborate schedule, especially in the case of certain isotopes. For this purpose a multiregion transmutational device, the multiregion molten salt reactor or the multiregion accelerator driven molten salt subcritical system can be advantageous candidates.

Fig. 10 shows the scheme of a **multiregion molten salt reactor (MRMSR)**, the example being a three-region one. In this example the regions are concentric cylindrical rings signed 1, 2 and 3 from outside to inside. The warmed up molten salt containing the already transmuted and to be transmuted isotopes goes to the upper expansion tank (5) via the riser outlet (4) and then via the downcomer (8) to the heat exchanger (9), where it transfers its heat to the secondary fluid (which may also be some molten salt or eg. He gas) through the heating surface (13). The resulting cooled down molten salt is pushed back to the 1st region by the circulating pump (10) via the inlet pipe (14). The dimensions of the upper expansion tank are such that the level of the molten salt should remain above the prescribed minimal value in the case of the highest potentially occurring density and should not exceed the maximum value when the density is the lowest. The volatile and gaseous fission products getting out of the molten salt are carried off via the gas off-take pipe (7) along with the He gas present in the expansion tank. The secondary fluid enters the heat exchanger via pipe (11) and exits via pipe (12) when heated up. Circulation loops identical to the one described are connected to regions 2 and 3, as shown in Fig. 10. The three reactor regions are separated by partition walls (37 and 38). In the example shown, the upper expansion tanks are also separated by partition walls (40 and 41). The three-region molten salt reactor is surrounded

by a wall (39). The entire device outlined is located in a shaft (43) surrounded by a shell (44). The dimensions of the shaft must be chosen so that, in case the total amount of the molten salt present in the system got out, the molten salt accumulated at the bottom of the shaft be subcritical to a prescribed value under any conceivable circumstance.

In the example shown in Fig. 10 the implementation does not contain moderating material in any of the regions and therefore the neutron spectrum is fast in all of them. If a region with thermal spectrum is also necessary, the moderator, which is normally graphite, is usually contained in the outer region (1). In this case it is practical to surround the reactor with a graphite reflector.
Fig. 11 shows the side view of an **accelerator driven three-region molten salt subcritical system**. The device is driven by a beam of particles from an accelerator, either with the application of a target core or the target being the molten salt material of the innermost region. The effective multiplication factor of the system is between 0.95 and 0.98, consequently self-sustaining chain reaction cannot occur. The construction is similar to that shown in Fig. 10 with the difference that the innermost region (3) is not cylindrical shaped but rather a cylindrical ring inside which, i.e. around the axis of the system, the beam (45) is supplied to the target (46). If the target is the molten salt itself, the target denoted 46 is not necessary. The advantage of this method is that significantly harder neutron spectra can be achieved (especially in the innermost region) than in the same region of the reactor construction shown in Fig. 10. On the other hand, due to the fact that the effective multiplication factor is less than 1 and the accelerator can be switched off at any time, this system is safer than a critical reactor. The beam might be a proton or electron beam. In the case of electrons the beam passing through the vacuum (46) enters the target (47) where bremsstrahlung is produced. The gamma photons produce high energy photoneutrons in the (γ,n) reactions [27]. The created neutrons enter the subcritical system, where they induce fission reactions. The ratio of the number of fission neutrons to that of the spallation or photoneutrons is the neutron amplification, which depends basically on the k_{eff} of the system: the higher the k_{eff}, the larger the amplification and hence the neutron flux at given beam parameters. However, an increase in the amplification will result in a softer neutron spectrum. The numbers corresponding to the subcritical system and connecting equipment depicted by Fig 11 are the same as for Fig. 10.

3 - CALCULATIONS

In order to compare a "conventional" MSR and a proposed MRMSR in terms of efficiency, preliminary calcualtional results are shown in this section. The main characteristics of the devices are as follows.

Geometry, material composition and fuel cycle parameters

In the case of the MSR the molten salt is contained in a cylinder, whose diameter and height are both 4 m, surrounded by a steel reflector in order to achieve a harder spectrum. The average

temperature of the salt is 940 K, while that of the reflector is 800 K. The salt composition is 30.04 mol% BeF_2 + 69.62 mol% LiF + 0.34 mol% $(Pu+MA)F_3$. This is far enough from the solubility limit, which is approximately 1% for Pu + MA)F_3. The starting composition of the actinide isotopes corresponds to that found in the fuel of a VVER-440 at 42 MWday/kgU burnup (without U). In order to constantly maintain this salt composition, the burnt out actinides are continuously supplied in the same composition. In this manner the total mass of the actinide isotopes remains constant, only their ratio to each other changes. The evolving fission products are continuously removed using a chemical process. After a given period the whole actinide content of the core is discharged and sent to final disposal. In this preliminary study we supposed 5, 10 and 20 years as irradiation time.

In the case of the MRMSR the 4 m by 4 m cylindrical core is divided into 5 concentric cylindrical rings of equal volume. The reflector is 45 cm thick graphite in order to achieve criticality, and it is also favourable due to the more thermal spectrum of the external region. The initial salt composition is 32.17 mol% BeF_2 + 67.13 mol% LiF + 0.7 mol% $(Pu+MA)F_3$, which is still represents a safe distance from the solubility limit. The composition of actinides at loading is the same as with MSR. This goes into the outer region. The 1 year burnt out goes to the 2^{nd}, the 2 year to the 3^{rd} etc. The spent fuel removed from the innermost region is treated as waste and sent to final disposal. There is no continuos feed, but the fission products are removed continuously in this case too.

The calculational methods and codes applied

The calculations were performed with the aid of the codes of the SCALE system. The codes BONAMI and NITAWL produce the AMPX working library, which contains resonance self-shielded problem dependent cross sections. This is used by the Monte Carlo code KENO-VI for criticality calculations. The XSDRNPM 1-D discrete ordinates code produces a weighted AMPX library and calculates the spectrum and k_{eff}. The COUPLE code generates ORIGEN working libraries from the data present in the weighted library. The ORIGEN-S point-depletion code is suitable for calculating the burnup of liquid fuel reactors since, besides the nuclear reactions, it is also capable of modelling the continuous feed and blending of different partial flows. The burnup was calculated in one-year intervals. Based on the composition data obtained from ORIGEN, the spectrum and cross sections were calculated again at the end of each interval and the burnup of the next interval was calculated using the new data.

Results

According to the calculations, the five regions of the MRMSR will have the neutron spectra shown in Fig. 12. One can see from the figure that in the case of a critical reactor the spatial dependence of the neutron spectrum is not very strong. It can be concluded that the division into regions is expected to introduce some improvement, however its magnitude is moderate. The calculations aimed at the efficiency partly verified these assumptions, although the improvement is not negligible (especially regarding radiotoxicity).

Table 2 shows the number of VVER-440 reactors which can be "served" by an MSR and an MRMSR of 1000 MW_e (with respect to the transmutation of actinides).

Table 2 Number of VVER-440 reactors, which can be served by a 1000 MW_e molten salt reactor

Type of transmutational reactor	Initial load	Yearly load
MSR	15.5	5.8
MRMSR	18.6	6.2

Figs. 13 and 14 show the behaviour of radiotoxicity for different cases. OTC means the case without transmutation. Fig. 13 shows the radiotoxicity taking into account the plutonium and uranium isotopes, while in Fig. 14 the radiotoxicity calculated without these isotopes can be seen. From Fig. 13 one can draw the conclusion that in terms of the total radiotoxicity of all the actinides the largest and smallest radiotoxicity values are obtained without transmutation and with the application of the MRMSR, respectively. This is even better depicted by Fig. 15, which shows the so called relative residual hazard [1,2] as a function of storage time. Concerning the radiotoxicity and residual hazard index calculated without accounting for the uranium and plutonium isotopes (see Figs 14 and 16) one may come to different conclusions. This is comprehensible since the transmutational devices transform part of the plutonium isotopes into higher actinides, thus raising the amount of actinides. On the other hand, the results reflect the limited transmutational ability of the critical MSRs and MRMSRs. This is partly due to the comparatively soft neutron spectrum (see Fig. 12), where the fission cross section of some of the actinides is low and the σ_f/σ_c ratio is very low for all them. Nevertheless, examining Figs 14 and 16 it can be stated that the MRMSR is a better transmutational device than the MSR. This is backed up by the numerical figures listed in Table 3, in which the initial (corresponding to t=0) relative residual hazard values are summarized for different options.

Table 3 Initial relative residual hazard for different cases, %

Transmutational device or method	With plutonium isotopes	Without plutonium isotopes
OTC	100	2.89
MSR (5 years)	47.16	14.71
MSR (10 years)	28.38	10.35
MSR (20 years)	15.19	5.46
MRMSR	10.35	7.47

It is emphasized again that in the case of the multistep transmutation, which is realized in a multiregion reactor it is very important to optimize the number and order of the steps. The above shown results obtained using five regions are *not* the results obtained after optimization. Therefore, the results obtained for the MRMSR can not be taken as optimal. It is also seen from the results that it may be useful to connect the transmutation going on in the MSR and in the

MRMSR as a series. This fact raises the necessity of full-system performance analyses, about which we report in more detail in a paper being presented at another international conference [3].

4 - CONCLUSIONS

The paper analyses possibilities to improve the transmutation aimed molten salt reactor and accelerator driven molten salt subcritical system. It is concluded that the efficiency of transmutation can be increased and waste with lower radiotoxicity can be produced with the application of a multistep transmutational method, which takes into account the spatial changes in the neutron flux and spectrum of the transmutational device. This method can be implemented in the proposed multiregion molten salt reactor and particularly in the multiregion molten salt subcritical system. Preliminary calculational results for a system which have not been optimized are presented. This and the transmutation in the multiregion molten salt subcritical system are considered as the next targets of our investigation. The possibility to connect the various transmutational devices into a tandem should also be examined, which requires full-system performance assessment.

REFERENCES

1. Gy. Csom, S. Fehér, M. Szieberth: Perspectives Of Waste Recycling Symbiotic Nuclear Energy Systems ("A kétszeresen zárt tiszta atomenergia-rendszerek megvalósításának lehetõségei"), 5th International Symposium on Nuclear Technology, 2000. október 4-6. (in Hungarian)
2. Gy. Csom, S. Fehér, M. Szieberth: Improved Method For P&T Strategy Selection By Long-Term Risk Evaluation Of Nuclear Energy Systems, in: Proceedings of GLOBAL 2001, PARIS – FRANCE, September 9-13, 2001,
3. Gy. Csom, S. Fehér, M. Szieberth: Perspectives Of Waste Recycling Symbiotic Nuclear Energy Systems, in Proceedings of ICEM '01, Bruges (Brugge) – BELGIUM, September 30 – October 4, 2001 (to be published)
4. Gy. Csom, A. Aszódi, S. Fehér, M. Szieberth: Time scheduled multi step method for transmutation of radioactive wastes and a multiregional molten salt transmutational device for its realisation ("Radioaktív hulladékok idõprogramozott többlépcsõs transzmutációs eljárása és az azt megvalósító többrégiós sóolvadékos transzmutációs eszköz"), announced to Hungarian Patent Office under number P0103762, September 20, 2001 (in Hungarian)
5. MSRE (1962): Directory of Nuclear Reactors, Vol. V. Research, Test and Experimental Reactors. IAEA, Vienna, 1964, STI/PUB/73
6. Actinide and Fission Product Partitioning and Transmutation, Status and Assessment Report. OECD Nuclear Energy Agency (1999)
7. C. Rubbia: Energy amplifier for nuclear energy production driven by a particle beam accelerator. United States Patent 5,774,514 (issued June 30, 1998)

8. J. R. Ireland: Overview of Los Alamos concepts for accelerator transmutation of waste (ATW), Proceedings of the Specialists' meeting on Accelerator-Based Transmutation PSI Villigen, SW. March 24-26, 1992, PSI-Proceedings 92-02, ISSN 1019-6447
9. L. V. Tocheniy et al: Design Schemes of Blanket for Actinoids Transmutation, Proceedings of the Specialists' meeting on Accelerator-Based Transmutation PSI Villigen, SW. March 24-26, 1992, PSI-Proceedings 92-02, ISSN 1019-6447
10. H. Takahashi, H. Rief: Concepts of Accelerator Base Transmutation Systems Proceedings of the Specialists' meeting on Accelerator-Based Transmutation PSI Villigen, SW. March 24-26, 1992, PSI-Proceedings 92-02, ISSN 1019-6447
11. Y. Kato et al: Accelerator Molten Salt Target System for Transmutation of Long Lived Nuclides, Proceedings of the Specialists' meeting on Accelerator-Based Transmutation PSI Villigen, SW. March 24-26, 1992, PSI-Proceedings 92-02, ISSN 1019-6447
12. C.D. Bowman et al.: Nuclear energy generation and waste transmutation using an accelerator-driven intense thermal neutron source, Nuclear Instruments & Methods in Physics Research, Aug. 15, 1992, Vol. A 320 Nos. 1,2 ISSN 0168-9002, Section A: Accelerators, spectrometers, detectors and associated equipment
13. H. Katsuta et al: A continuous transmutations system for long lived nuclides with accelerator-driven fluid targets, Proceedings of the Second Internationa Exchange Meeting on Actinide and Fission Product Separation and Transmutation, Argonne, USA, 1992
14. W.C. Sailor, C.A. Beard: Neutronics Analysis for an Accelerator-Based Nuclear Waste Transmuter, Proceedings of GLOBAL'93 International Conference and Technology Exhibition, Sept. 12-17, 1993, Seattle, Washington
15. J.W.Davidson, M.E. Baltat: Neutronics-Processing Interface Analyse for the Accelerator Transmutation of Waste (ATW) Aqueous-Based Blanket System, Proceedings of GLOBAL'93 International Conference and Technology Exhibition, Sept. 12-17, 1993, Seattle, Washington
16. M.Cappiello et al: Los Alamos Aqueous Target/Blanket System Design for the Accelerator Transmutation of Waste Concept, Proceedings of GLOBAL'93 International Conference and Technology Exhibition, Sept. 12-17, 1993, Seattle, Washington
17. N.Watanabe et al: Highlight of OECD/NEA Workshop on "Utilisation and Reliability of High Power Proton Accelerator, Proceedings of the 5th International Information Exchange Meeting on Actinide and Fission Product Partitioning and Transmutation Mol, Belgium, November 25-27, 1998
18. M. Hron: The Czech National R&D Program of Nuclear Incineration of PWR Spent Fuel in a Transmuter with Liquid Fuel, Proceedings of the 5th International Information Exchange Meeting on Actinide and Fission Product Partitioning and Transmutation Mol, Belgium, November 25-27, 1998
19. C.D. Bowman: Once-through thermal-spectrum accelerator-driven light water reactor waste destruction without reprocessing, Nuclear Technology, Vol. 32 (Oct. 2000)
20. P.N. Alekseev et al: Concept of the Demonstration Molten Salt Unit for the Transuranium elements Transmutations, Proceedings of the ninht Symposium of AER, Demänovska Dolina, Slovakia, 4-8 October 1999
21. J. Svarny, P. Mikolás: Design Calculations of the Thermal-Spectrum Accelerator-Driven System for LWR Waste Destruction, Proceedings of the ninht Symposium of AER, Demänovska Dolina, Slovakia, 4-8 October 1999

22. V. Lelek: AER Working group F "Transmutations" activities in 2000, Proceedings of the tenth Symposium of AER, Moscow, Russia, 18-22 October 2000
23. V. Lelek: Establishing the Design Basis for a Molten Salt Demonstration Transmuter, Proceedings of the tenth Symposium of AER, Moscow, Russia, 18-22 October 2000
24. V. Lelek, R. Vocka: Open problems in reprocessing of a molten salt reactor fuel, Proceedings of the tenth Symposium of AER, Moscow, Russia, 18-22 October 2000
25. P.N. Alekseev et al: Nuclear Power Technology System with Molten Salt Reactor for Transuranium Nuclides Burning in Closed Fuel Cycle, Proceedings of the tenth Symposium of AER, Moscow, Russia, 18-22 October 2000
26. C. Bowman. Apparatus for transmutation of nuclear waste, United States Patent 6,233,298 (issued May 15, 2001)
27. P. Vértes, A. Brolly: Evaluation of Neutron Sources for ADTW Systems, Proceedings of ICENES 2000, Petten, The Netherlands, September 24-28, 2000, p. 118

Table 1 Fission and capture cross-section and its ratio for actinides isotopes at different energy

Parameter	σ_f, barn						σ_c, barn						σ_f/σ_c					
Energy, MeV	0,1	0,5	1	2	5	10	0,1	0,5	1	2	5	10	0,1	0,5	1	2	5	10
Np-237	0,017	0,543	1,144	1,66	1,43	2,19	1,17	0,379	0,156	0,0555	0,0182	0,0385	0,0145	1,43	9,22	29,9	78,8	56,8
Pu-238	0,62	1,52	2,07	2,26	2,03	2,73	0,384	0,154	0,0571	0,0351	0,0162	0,00213	1,61	9,87	36,3	64,4	126	1280
Pu-239	1,53	1,59	1,76	1,98	1,67	2,27	0,278	0,115	0,0676	0,0326	0,000968	$1,5\cdot10^{-8}$	5,48	13,9	26,0	60,9	1720	$1,51\cdot10^{8}$
Pu-240	0,0539	0,526	1,55	1,73	1,49	2,15	0,365	0,170	0,095	0,0676	0,00229	0,000696	0,148	3,09	16,3	25,6	654	3080
Pu-241	2,15	1,52	1,59	1,67	1,38	2,03	0,450	0,347	0,301	0,211	0,00525	$1,12\cdot10^{-8}$	4,78	4,37	5,27	7,91	263	$1,81\cdot10^{8}$
Pu-242	0,0182	0,287	1,50	1,43	1,25	1,95	0,267	0,140	0,0909	0,0762	0,00195	0,000697	0,068	2,05	16,5	18,7	642	2800
Am-243	0,00609	0,0565	1,26	1,50	1,50	2,05	1,40	0,504	0,302	0,0473	0,010	0,010	0,00434	0,112	4,18	31,7	150	205
Cm-244	0,0495	0,575	2,08	1,92	1,99	2,98	0,393	0,227	0,101	0,0399	0,00647	0,000858	0,126	2,53	20,6	48,1	308	3480
Cm-245	2,08	1,63	1,97	2,11	1,79	2,48	0,230	0,0862	0,0513	0,0236	0,00321	0,000346	9,06	19,0	38,3	89,4	559	7180

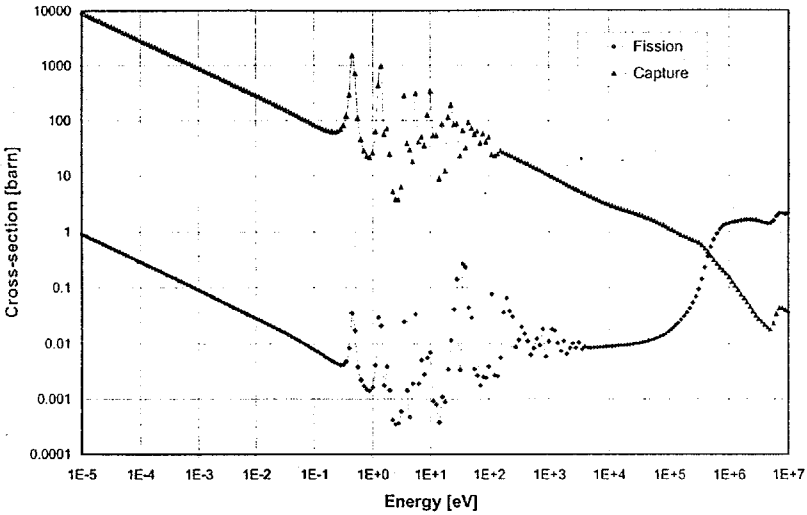

Figure 1 Fission and capture cross-section of Np-237

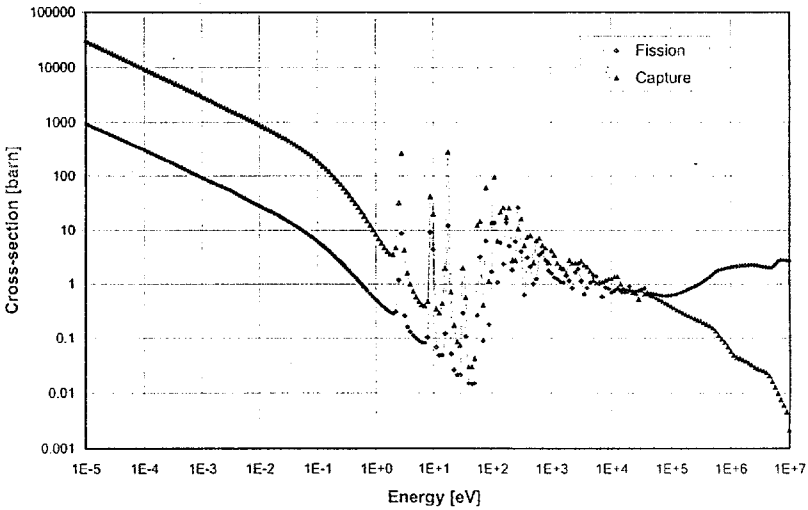

Figure 2 Fission and capture cross-section of Pu-238

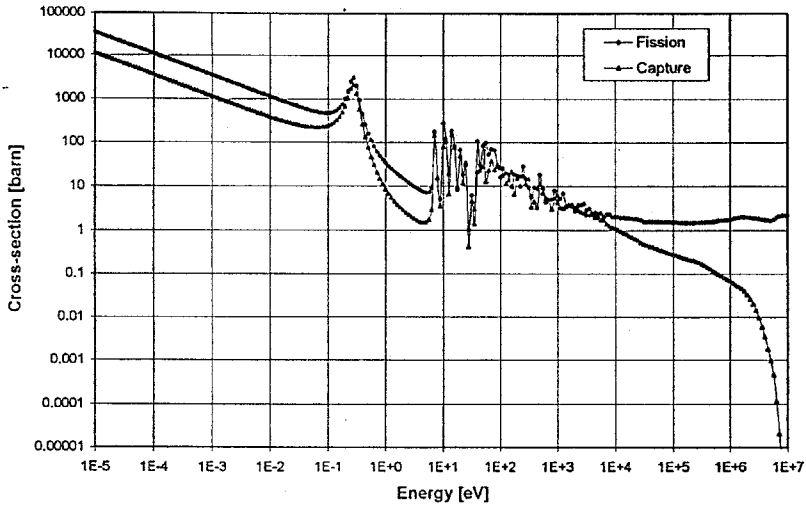

Figure 3 Fission and capture cross-section of Pu-239

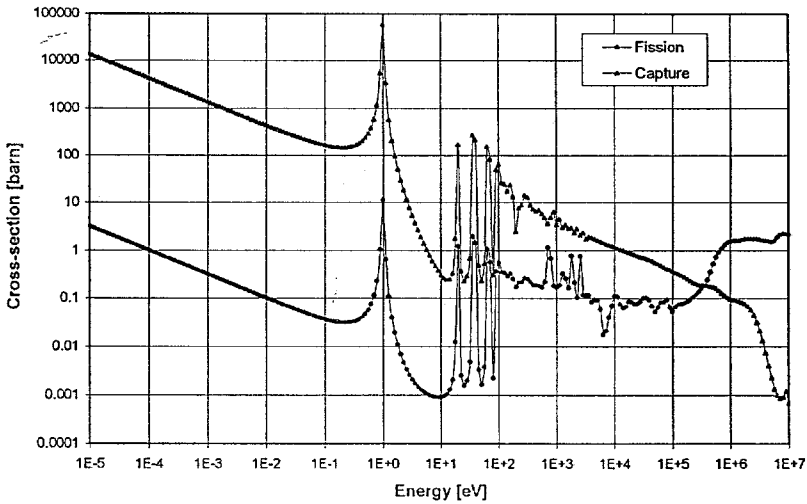

Figure 4 Fission and capture cross-section of Pu-240

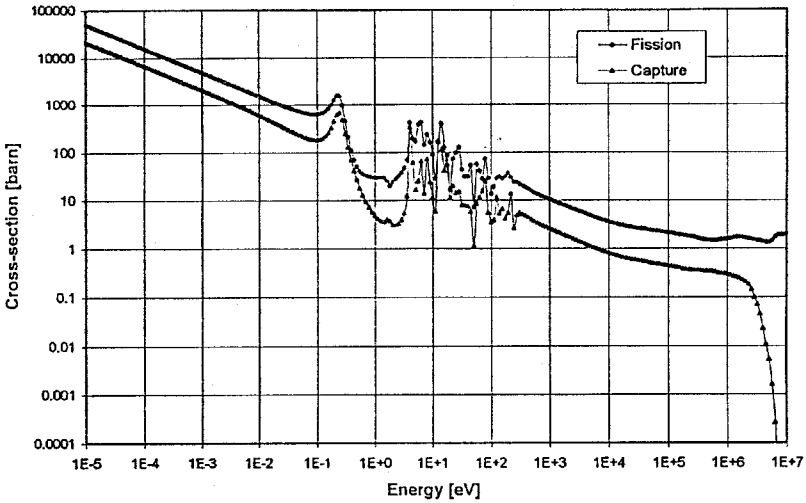

Figure 5 Fission and capture cross-section of Pu-241

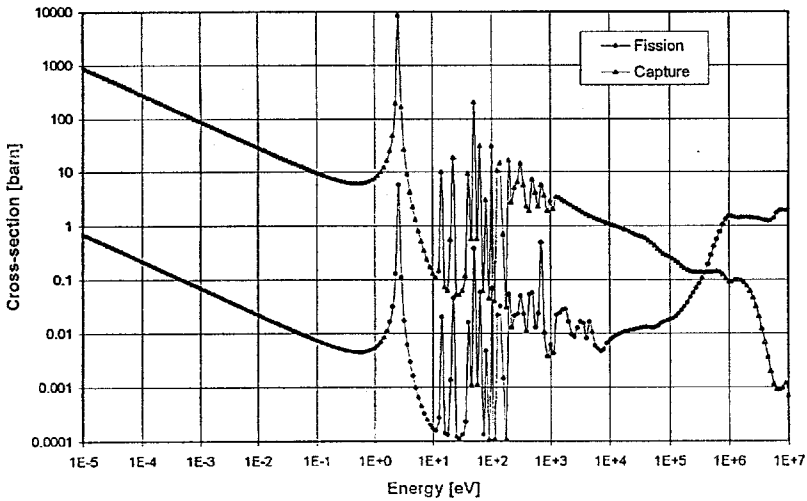

Figure 6 Fission and capture cross-section of Pu-242

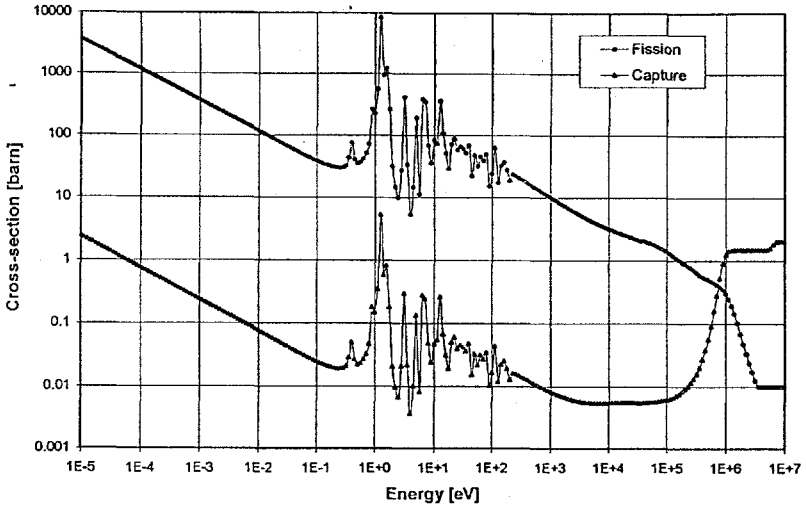

Figure 7 Fission and capture cross-section of Am-243

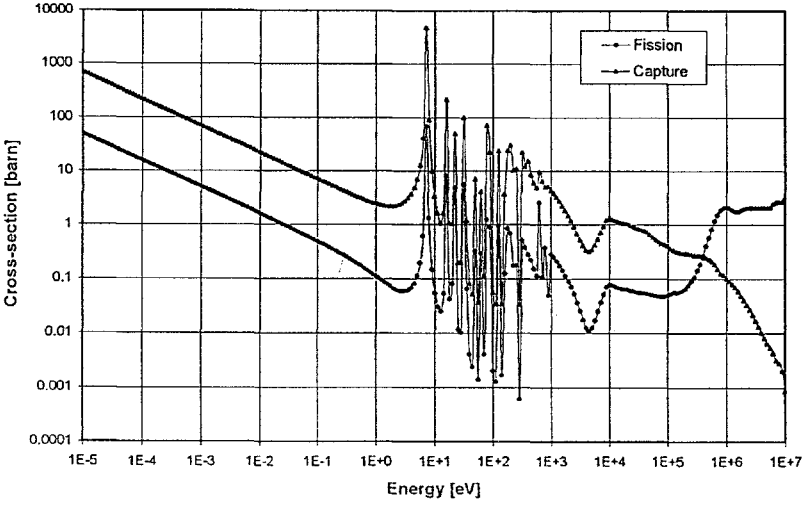

Figure 8 Fission and capture cross-section of Cm-244

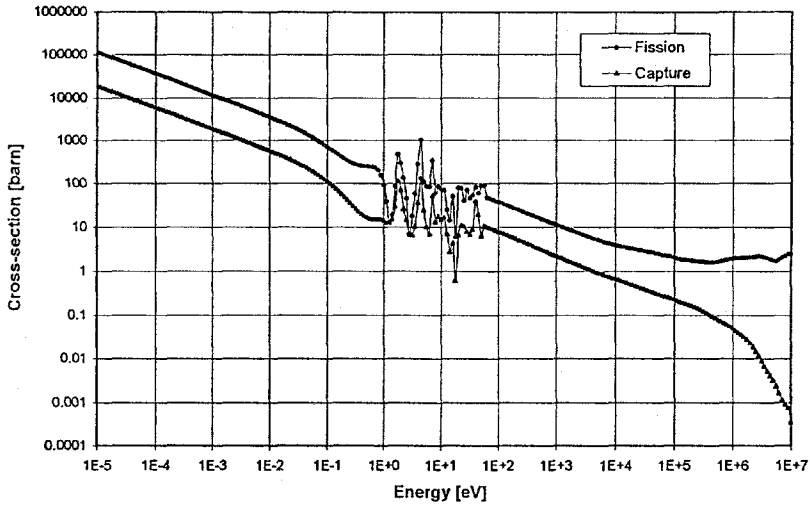

Figure 9 Fission and capture cross-section of Cm-245

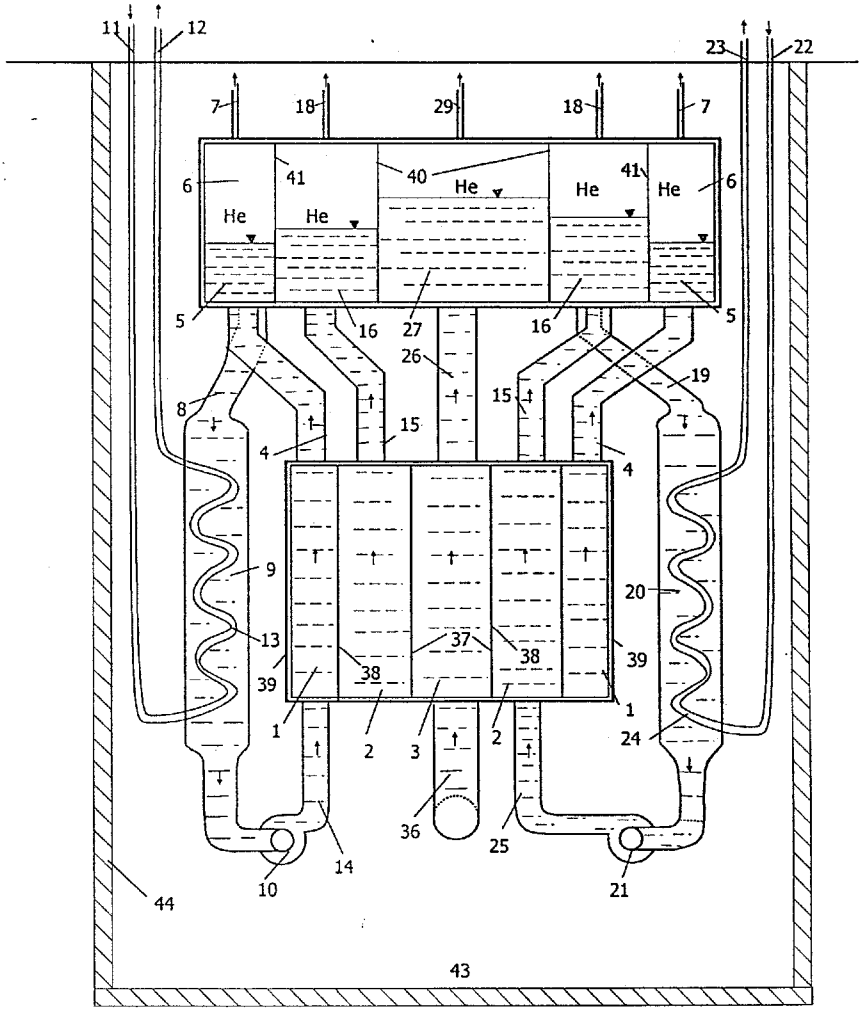

Figure 10 Scheme of a three-region molten salt reactor

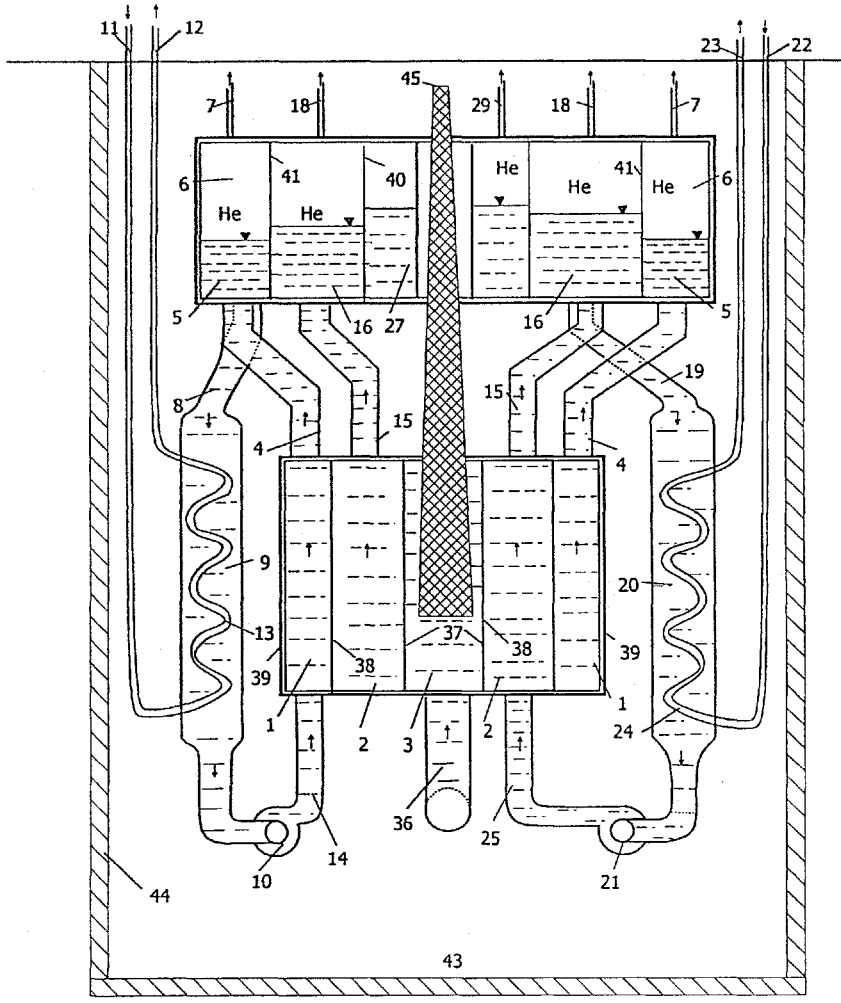

Figure 11 Scheme of a three-region accelerator driven molten salt subcritical system

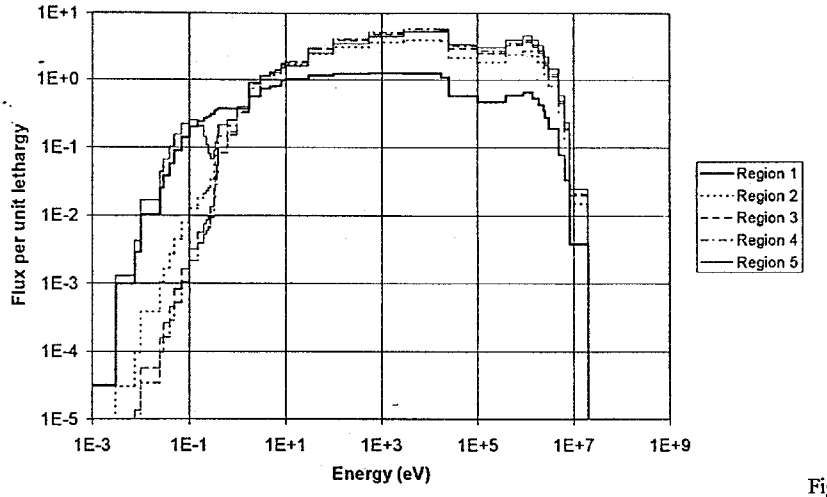

Figure 12 Spectra in the different regions of a MRMSR (Region 1 is the innermost)

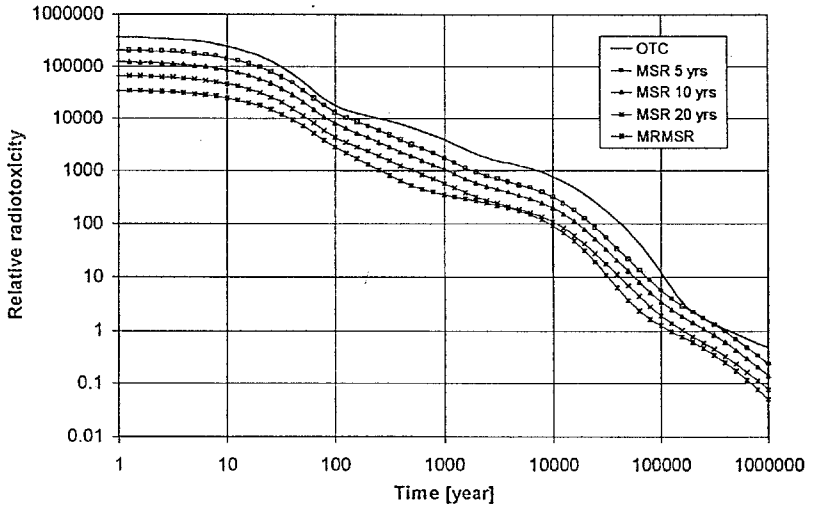

Figure 13 Relative radiotoxicity in the case of different scenarios

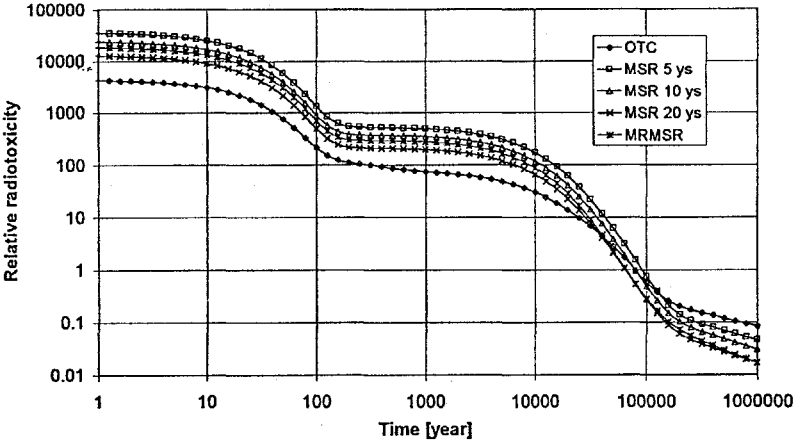

Figure 14 Relative radiotoxicity in the case of different scenarios without the contributions of U and Pu isotopes

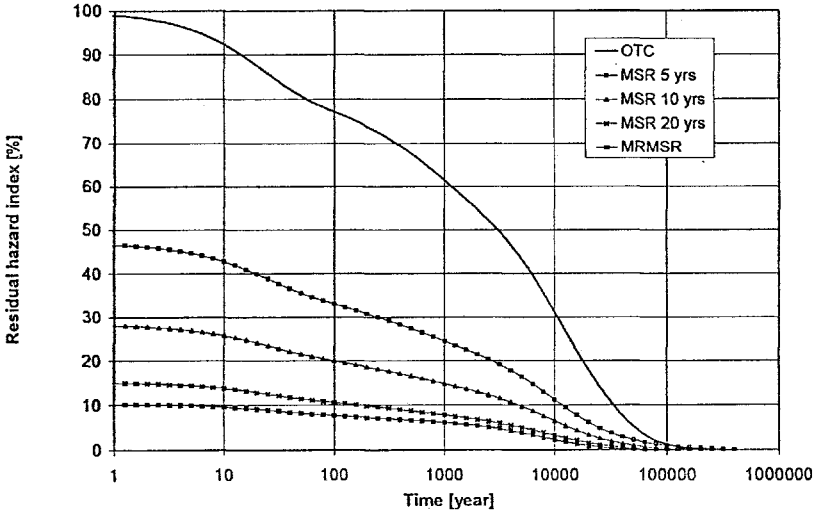

Figure 15 Residual hazard index in the case of different scenarios

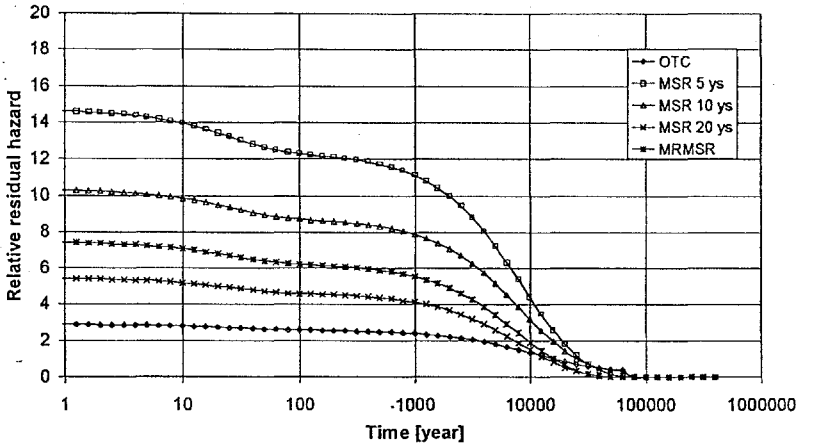

Figure 16 Residual hazard index in the case of different scenarios without the contributions of U and Pu isotopes