SESSION V

ACCELERATOR DRIVEN TRANSMUTATION

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The highlights of the OECD-NEA Workshop on "Utilisation and Reliability of High Power Proton Accelerator" at Mito, Japan on October 13 - 15, 1998 were reported.

- The objective of the workshop was to explore the efficient utilisation of high power proton accelerator (HPPA) in various fields and the future possibility of accelerator-driven subcritical systems (ADS).

- One serious problem is frequent beam trips of existing HPPAs. It is indispensable to understand the effects of beam trips on sub-systems, especially on ADS.

- For ADS application, beam trips should be reduced by two order of magnitudes.

- These severe requirements from ADS to HPPA are completely new for the accelerator community.

- Development of a reference subcritical system was recommended for benchmark analysis of possible problems.

- The comparison between Linear and Circular accelerators was discussed, and the conclusion was the following: (1) a linac was for tens mA and GeV, (2) a linac or a cyclotron was for 10 mA, and (3) a cyclotron was for less than 10 mA.

Effect of beam trips on sub-systems was investigated for the fuel pin and beam window in JAERI. It was concluded that fatigue damages caused by beam trips were negligible.

The CEA/EDF paper and the KAERI paper discussed the comparison between thermal neutron spectrum and fast neutron spectrum. They concluded that thermal neutron spectrum is not adequate for ADS because its short mean free path causes localisation of nuclear reactions which is not acceptable for safe operation of ADS. The former paper also discussed the comparison of coolant materials for a demonstration device of ADS and concluded that a gas cooled system is the first priority, lead-bismuth is the second, and the sodium.

SCK/CEN of Belgium proposes a medium-size ADS prototype, MYRRHA. The spallation target is a liquid lead-bismuth with a unique windowless design. The subcritical core is made of two consecutive zones of fast and thermal zone.

Conceptual safety analysis was carried out for the Fast Energy Amplifier design (ANP, Italy). For the analysis, three-dimensional simulation was carried out using MCNP since the point kinetic analysis cannot be applicable due to the existence of higher mode for ADS.

HIGHLIGHT OF OECD/NEA WORKSHOP ON "UTILISATION AND RELIABILITY OF HIGH POWER PROTON ACCELERATOR"

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Abstract

The first OECD/NEA Workshop on "Utilisation and Reliability of High Power Proton Accelerator" was held at Mito, Ibaraki, in Japan on October 13 -15, 1998. The participants were 92 persons, including 32 persons from the outside of Japan. There were two keynote talks and eleven invited talks from the high power proton accelerator (HPPA) projects. Twenty three technical papers were presented about the areas related to reliability of HPPA, new accelerators, effects of beam trips, and interface technologies. Technical discussion sessions were also arranged for accelerator and accelerator driven system (ADS), in parallel. In the last panel session, the commentators from both fields commented on technical problems.

Scope of the workshop

R&D activities and construction plans related to high power proton accelerators (HPPAs) are being considered in various countries to promote basic and applied sciences, including accelerator-driven nuclear energy system (ADS), using neutrons, protons and other secondary particles. Taking into account the fact that proton beams from existing HPPA trip (suddenly stop) very frequently, it is indispensable to understand the effects (e.g. thermal shocks) of such beam trips on different sub-systems, especially fission sub-systems. Additional R&D will be needed to accomplish a highly reliable HPPA and sub-systems resistant to thermal shocks.

The scope of the workshop comprises:

- The experiences and prospects of HPPA utilisation.
- Reliability of existing HPPAs, especially focused on beam trips and power fluctuations.
- Effects of resulting thermal shocks in fission sub-systems.
- Required accelerator reliability in various applications.
- R&D of sub-systems resistant to such shocks.
- Accelerator types suitable for ADS, interface technology between proton beam and sub-systems.
- Control system and safety concept for ADS and problems relevant to utilisation (multi-purpose vs. dedicated systems, etc.).

The purpose of the workshop is to exploit more efficient utilisation of HPPAs in various fields and to ensure the future possibility of ADS.

Major Presentations

Applications of High Power Proton Accelerators (HPPA)

Five presentations were given to the projects developing spallation neutron sources with HPPA, and six presentations to the accelerator driven system (ADS) with HPPA.

The SNS project in USA was approved, which aims at 1 MW pulsed spallation neutron source for neutron scattering and will be upgraded to 4 MW. The ESS project for 5 MW spallation neutron source in Europe has started an optional study of a superconducting proton linac, in addition to the reference normal conducting one. There are two projects in Japan and it was reported that both projects will join**:** the JAERI Neutron Science Project for neutron science and for transmutation of long-lived nuclear wastes with 1.5 GeV-5.3 mA superconducting linac, and the JHF project promoted by High Energy Accelerator Research Organisation (KEK) which includes two ring synchrotrons of 3 GeV-200 mA and 50 GeV and four research facilities, i.e., for high energy nuclear physics, neutron scattering, muon science and RI beam nuclear physics. The KOMAC project in Korea is a multipurpose accelerator complex aims at constructing a 1 GeV-20 mA HPPA in conjunction with ADS project of HYPER.

Many countries have ADS oriented projects. The Chinese project of a 150 MeV-3 mA HPPA is injecting a beam to 3.5 MWt LWR with criticality of 0.94-0.98. Russian activities of ADS development includes critical experiment with photo neutrons from a Pb or Pb-Bi target. The Czech program is for LLFP and TRU transmutation with mixture of target/MA and Flibe cooled graphite/LLFP blankets driven by a 35 MeV deuteron external neutron source. In France, there are GEDEON activities of development for nuclear waste transmutation. Those include spallation target experiments at SATURN and reactor experiments at MASURCA with a 14 MeV source, material research of structure and Pb , Pb-Bi target and 10 MeV-100 mA accelerator developments. Italy has TRASCO program by ENEA and INFN, and the industrial program by Ansaldo Corp. The TRASCO is developing a 1 GeV-30 mA accelerator, subcritical system like Energy Amplifier with nine sub-programs. The industrial program is focusing on a demonstration proto-type of an 80 MWt, a Pb-Bi target and subcriticality of 0.95. In USA, the ATW is conducting three blocks of developments: accelerator in APT (1 GeV-140 mA), pyrochemical processes and a subcritical burner (2000 MWt/Pb-Bi).

Table 1 **HPPA projects presented at the workshop**

Project	Country	Specification	Utilisation
SNS	USA	1 GeV, 1 mA , 1 MW(4 MW)	Neutron Scattering
ESS	EU	1.333 GeV, 3.75 mA, 5 MW	Neutron Scattering
JAERI-NSP	Japan	1.5 GeV, 5.33 mA 8 MW	Neutron Scattering, ADS
KEK-JHF	Japan	3 GeV-200 mA, 50 GeV-10 mA	Neutron Scattering, Muon, Kaon
KOMAC	Korea	1 GeV, 20 mA	RI production, Therapy, ADS
RCNPS	China	150 MeV, 3 mA	RI production, Therapy, ADS
TRASCO	Italy	1 GeV, 30 mA	ADS
GEDEON	France	not yet specified	ADS
ATW	USA	1 GeV, 140 mA (APT)	ADS
?	Russia	target & subsystem only	ADS
?	Czech	35 MeV deuteron	ADS

Reliability of HPPAs

The reliability of the accelerator was investigated by using a statistical code, based on the data from LANSCE, ISIS, PSI and TJNAF facilities. The accelerator down time predicted from MTBF (mean time between failure) and MTTR (mean time to repair) taken from those were agreed well with observations. The results of analysis of beam trip and down time of LANSCE during 1996 to 1997 showed that the most frequent trip is at H^+ injector with 77% but its down time is shorter than one minute. The next component with frequent trips is at RF systems. In Moscow Meson (500 MeV, 20 mA peak current, 50 Hz and 90 ms duration), it was shown that the average beam current loss is less than 0.2 % but localised along the linac. It was pointed out from the PSI cyclotron experience that the major cause of beam trips are due to micro-sparks of the RF systems. The shutdown time by the most frequent sparks is less than 200 ms and the beam is automatically restarted.

The concept adopted in the SNQ project in 1980s showed that the accelerator consists of independent single cavities so as to compensate one cavity by the others if a fault in a cavity occurs. The multiplexing of accelerators and ADS systems largely increase the reliability. A method to improve the reliability of an accelerator based on experiences of IPNS and APS at ANL is to operate the accelerator adequately below the upper limit of its ratings, i.e., with an enough margin. From experiences of LEP at CERN for cavities, couplers and RF systems, it was pointed out that the life of 34 crystrons are expected to be more than 10,000 hours and the reliability of superconducting electron linac is well established.

New Accelerators

A H_2^+ cyclotron concept was proposed to reduce the space charge effect and to eliminate deflection by a stripping foil. A separated orbit cyclotron with superconducting magnets and cavities was also proposed with the results of the test device TRITRON, in which three cascade rings can accelerate the beam up to 1 GeV. A FFAG (fixed field alternating gradient) accelerator was proposed to improve power efficiency of accelerator for ADS, because of the context of progress in cavity and magnet technologies.

Beam Trips/Fluctuations: Effect on ADS and ADS Resistance

From a preliminary analysis on a modest design concept of ADS, it was shown that thermo-mechanical effect on ADS components is the most important problem of the trip, but the effects on fuel pellets, fuel pins and beam window would be negligible. It was also shown from the results of temperature transient test for making structural design guide in FBR, that in repetition of temperature variation between 250°C and 600°C, the test piece was damaged by thermal fatigue for short period of cycles and by creep fatigue for long period of cycles.

From the analysis of the components with temperature variation during the trip based on the EFR concept, it was pointed out that above core structure (ACS) and intermediate heat exchanger (IHX) were important components for such analysis.

Interface Technology

In the IFMIF analysis, a decay constant of Li temperature is a few minutes for the two kinds of trips of two deuteron beam injections with 40 MeV-250 mA: two beams of 10 MW and one beam of 5 MW.

In transient thermal stress analysis in the window of mercury target made of SS316 steel, bombarded by pulsed-protons at a beam power of 5 MW with 50 Hz, it was shown that an asymptotic temperature at the beam window was quickly achieved within a couple of seconds, although the temperature fluctuates at 50 Hz.

A temperature decay constant in the lead incore and the cladding for a lead rod target is estimated to be 5-10 s for unscheduled beam trip or loss of coolant. Maximum stress was 90 MPa in the cladding through normal operation and beam trip transient.

In the lead-bismuth spallation target at a proton beams of 600 MeV-6 mA, the thermal stresses decoupled by the fluid dynamic transient showed the Mises stress (conserved quantity related to yield function) of 175 MPa and a fatigue damage induced by cyclic beam trip longer than 4-5 s leads to predict the allowable number of interruptions to failure.

Discussions

Parallel Discussion/Accelerator

The items discussed here were:

1. Origin of beam trips and fluctuations
2. Possible improvements for HPPA
3. Achievable reliability in future and necessary R&D
4. Linear vs. Ring accelerators as HPPA systems.

And the major conclusions are the followings:

1. For Beam trips, three types were categorised as: 1) short (< 1 min.), 2) medium (1 min. - 1 h), 3) long (>1 h). The most frequent trip is type (1). Most of (1) and (2) are caused by sparks in HV/RF systems. The down time of micro-spark is typically 100 ms, according to PSI data, and its recovery time is 600 ms. The frequency of trips depends on machine.
2. For possible improvements, to reduce sparks (then to reduce damage), it is considered to design carefully the devices and controls, to build and maintain as clean room, to keep good conditioning. The micro-sparks are not avoidable. It is important to note that accelerators are normally tuned to get maximum and "possibly more" performances, and this is the main reason of trips and faults.
3. To get reliability, overdesign is necessary, i.e., the same concept applied to the HV/RF systems should also be applied the other components besides those. The reliable accelerator is possible, but the meaning of "reliability" should be agreed. The participants agreed with: 1) MTBF (mean time between failures) can be reduced to about 100 hrs, and 2) MTTR (mean time to repair) depends on spares and redundancy, i.e., cost.
4. It is important to rely as much as possible on proven accelerators and to select considering specific application. For CW machines, "dream" cyclotron (1 GeV, 10 mA) was discussed and general opinions were collected. "Halo" aspect, as most important technical issue, was stressed. The compactness, modulability, flexibility and so on were also discussed as well as achievable power.

Parallel Discussion/ ADS and Sub-Systems

The discussion were summarised in the following categories:

(A) Definition and separation of problems

- System Test Facility (STF) vs. Transmutation Plant (TP).
- Driven Facility (DF) vs. Spallation Target (ST).
- Nature of trips.

(B) Issues not definitely resolved

(C) Issues not discussed but that need to be considered.

And the major conclusions are the followings.

(A) STF is defined as a demonstrator of combination of a subcritical facility with an accelerator. Only one reliable accelerator will be utilised in this case. The facility should have the maximised flexibility. TP is defined as the facility routinely carrying out actinide transmutation. The multiplexing of ADSs and HPPAs is conceivable, it is optimised for cost and reliability. In discussion, separated coolant loops are assumed for DF and TP. DF consists of above core structure (ACS), intermediate heat exchanger (IHX), primary coolant piping(PCP), fuel core (FC), etc. TP has a beam window and a liquid or solid spallation target. Trips are separated into two time ranges: 1) shorter than a system characteristic time t (40-60 sec or less) and 2) much longer than t. In the case of 1), self-recovering (automated) is considered and effect mainly takes place in ACS or PCP. In the case of 2), special restart protocol is necessary.

(B) Effect of pulse operation of accelerator and possible restriction to pulse structure were not clearly discussed. These questions are related to economy, possibility of multiplexing and beam control.

(C) The necessity of control rod in ADS needs to be discussed, relating to compensation of burn up and a safety device. Controllability of beam power and power surge are also to be discussed. Specific questions, partial Loss of Beam (LOB) for multiplexing, direct/indirect feedback of electricity to the system, and construction philosophy, were also left without any discussion.

Summary

The conclusions and recommendations from the workshop are summarised as follows:

[1] Required accelerator reliability

[2] Siginificant reduction of beam trips is inevitable for ADS application. (two order of magnitude) Regulatory requirement is to be considered. It may be comparable to "Critical Reactor". If once unscheduled stops happen, a long period will be needed to restart. Specially at the beginning of commissioning it should be very stable to get reliance on the technology. Power control must be studied.

[3] Achievable accelerator reliability

[4] MTBF (mean time between failures) can be reduced to about 100 hrs (85 trips/y) using recent technology.

[5] R&D needs in HPPA

[6] MTBF has not been important issue for current use of existing HPPAs. Severe requirement from ADS is completely new for the accelerator community. Beam trip that will cause the ADS restart would be less than once a year.

[7] R&D needs in ADS

[8] A reference subcritical system should be developed for benchmark analysis to get common understanding of possible problems. A way how to control ADS power level must be studied. Whether k_{eff} has to be stable or changeable should be discussed. R&D on structural materials for frequent thermal shock is required.

[9] Linear vs. Circular Accelerators

[10] It depends on current, power, energy. A linac is for tens mA and GeV, a linac or a cyclotron is for 10 mA and a cyclotron is for less than 10 mA. Beam shape and “Halo” are also to be considered. At the same time, a good core with stable k_{eff} is necessary.

[11] Multiplexing vs. Single Accelerator

[12] It is to be considered from the cost, the reliability and the number of components. The STF may be one accelerator and TP has options. The key question is economy and reliability in the commercial operation, including repairing and maintenance.

[13] Dedicated or Multi-purpose Facility

[14] A multi-purpose facility, such as JAERI-NSP and JHF, aims at multi-disciplinary or cross-disciplinary. Secondary particles such as neutrons and also muons are the purpose of HPPA utilisation. Many dedicated facilities are not allowed especially for small countries. But ADS is not in this domain. It is important to make a dedicated facility to demonstrate ADS as a promising system to a society.

[15] Future Collaboration

[16] International collaboration is fruitful to contest the ideas and to reduce the R&D cost. Large number of R&D could be shared. The frame work of OECD/NEA is also useful.

[17] Next OECD/NEA workshop will be held on November 1999 at Cadarache in France.

STUDIES ON ACCELERATOR-DRIVEN TRANSMUTATION SYSTEMS

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Abstract

Research and development on transmutation of long-lived radioactive nuclides are being carried out with an emphasis placed on the dedicated accelerator-driven systems at the Japan Atomic Energy Research Institute (JAERI) under the Japanese long-term program for research and development on partitioning and transmutation technology (OMEGA Programme). The preliminary design of the sodium-cooled solid system has been developed as a reference system based on the current LMFBR technology. Concurrently with the study on the sodium-cooled system, we made a preliminary design study of lead-bismuth cooled system as an option. Transient responses of temperatures and stresses due to accelerator beam trips were evaluated for the fuel pin and the beam window of the proposed experimental facilities

Introduction

Research and development on transmutation of long-lived radioactive nuclides are being carried out with an emphasis placed on dedicated accelerator-driven systems at the Japan Atomic Energy Research Institute (JAERI) under the Japanese long-term program for research and development on partitioning and transmutation technology (OMEGA Program). Design studies are being made for accelerator-driven transmutation systems (ADTSs). Design of the current reference ADTS follows that of contemporary sodium cooled fast breeder reactors (FBRs). The major reasons to chose sodium as coolant are its excellent thermal performance and technology maturity. However, a preliminary design study was recently started for a heavy liquid-metal cooling option of ADTS at JAERI. Monju (714-MWt sodium cooled prototype fast reactor) suffered a leakage and fire of secondary sodium in December 1995. This incident has caused the people to feel concern about safety of sodium coolant technology and stimulated much interest in alternative coolant options, such as lead, lead-bismuth, and He, for fast reactors. The main purposes of the design study of heavy liquid-metal cooling option are to determine the performance, to assess the feasibility and to identify the technical issues of a heavy liquid-metal cooled design in comparison with the sodium cooled reference design.

The ADTS was specially designed to transmute minor actinides (MAs) from about 10 units of 3000-MWt light water reactor (LWR) in a hard neutron energy spectrum and a high neutron flux. Such dedicated transmuter can be very efficient and effective for MA transmutation. In this context, JAERI has been pursuing the strategy of transmutation with dedicated transmuters, rather than recycling to commercial power reactors and proposing a concept of a double-strata fuel cycle consisting of a power reactor fuel cycle (the first stratum) and a P-T cycle (the second stratum) [1,2]. In this scenario, a reprocessing plant, a partitioning plant and a dedicated transmuter will be co-located in one site.

The experimental program for development and demonstration of ADTS technology is being planned under the JAERI Neutron Science Project. Pre-conceptual design study is being made for the experimental facilities (30 – 60 MWt experimental system and high-power target experimental facility). Large fluctuations and frequent trips of the incident proton beam may be inevitable in these facilities. These changes in beam intensity will cause changes in temperatures and stresses in reactor components. This will introduce thermal stress problems related specifically to accelerator-driven system. To assess the impact on the structural integrity and lifetime of the components, transient responses of temperatures and stresses were evaluated for the fuel pin and the beam window of the experimental system.

Conceptual design study of transmutation systems

The concepts of ADTS have been developed at JAERI. The ADTS is specially designed for transmutation purpose as a dedicated transmuter to be deployed in the P&T cycle of the double-strata fuel cycle. In the proposed ADTS, an actinide loaded subcritical core is driven by a high-intensity proton accelerator with several tens of MW beam and uses fast neutrons for efficient and effective transmutation of actinides. The current design aims at supporting about 10 units of large-scale LWRs with 1000-MWe capacity.

The preliminary design of the sodium-cooled solid system has been developed as a reference system based on the current sodium cooled FBR technology. The reference system employs solid tungsten for the spallation target and MA mononitride for fuel of subcritical core. Nitride is adopted as the fuel material because of its excellent thermal property. The other advantage of nitride fuel is that it can be processed with the pyrochemical reprocessing, [3] and, hence, the fuel cycle facilities can be very compact and cost effective.

Recently, JAERI started a design study of a heavy liquid-metal cooled ADTS as an alternative possible option to re-evaluate its technical viability. Although the heat transfer capability of lead or lead-bismuth is inferior to that of sodium, it has several advantages as coolant for accelerator-driven transmutation systems. Liquid lead or lead-bismuth is particularly suited to the target material, this eliminating the need for a distinctive solid target. It also offers the possibilities to achieve a harder neutron energy spectrum and to avoid a positive void reactivity coefficient. Intermediate heat exchangers and secondary heat transport loops will possibly be eliminated with lead or lead-bismuth coolant. Lead-bismuth eutectic coolant offers much lower system operating temperatures than lead coolant. The lower operating temperature will alleviate the severe problems of material corrosion/erosion in heavy metal coolants.

A computer code system ATRAS has been developed for the design of accelerator-driven transmutation system. [4] The cascade code NMTC/JAERI [5] simulates the proton-induced nuclear spallation, subsequent internuclear transport process for energies above 20 MeV. It also calculates high-energy fission reaction as a competing process with evaporation. Neutronic calculation below 20 MeV is carried out using transport codes, TWODANT and MCNP4A. The time evolution process of transmutation products is calculated by SPCHAIN code and by BURNER code for energies above and below 20 MeV, respectively.

Sodium cooled accelerator-driven transmutation system

The design of the sodium cooled solid system is based on a sodium cooled FBR. Proton beam is injected through a beam window into the tungsten target at the center of the target/core. The subcritical core loaded with actinide nitride fuel surrounds the target. The target consists of multiple layers of tungsten disk with through holes for coolant passage. The target is designed to maximise the neutron yield and to flatten the axial power distribution. The target and fuel subassemblies are cooled by forced upward flow of primary Sodium. Impinging flow of coolant from the target exit cools the beam window. The concept of the sodium cooled solid system is shown in Figure 1.

Figure 1 **Sodium cooled ADTS concept**

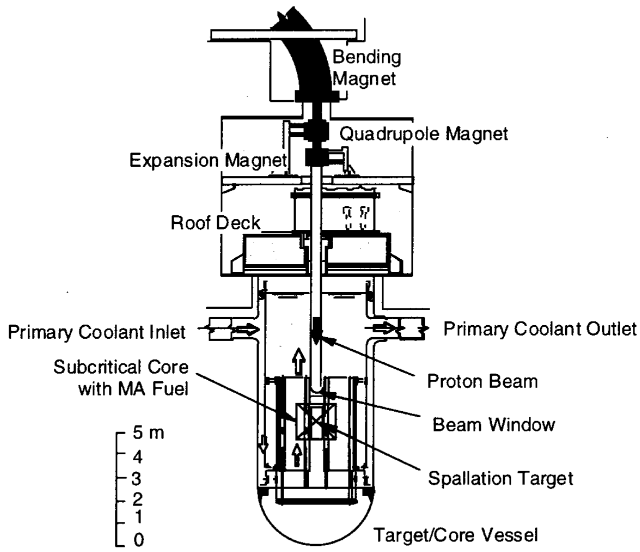
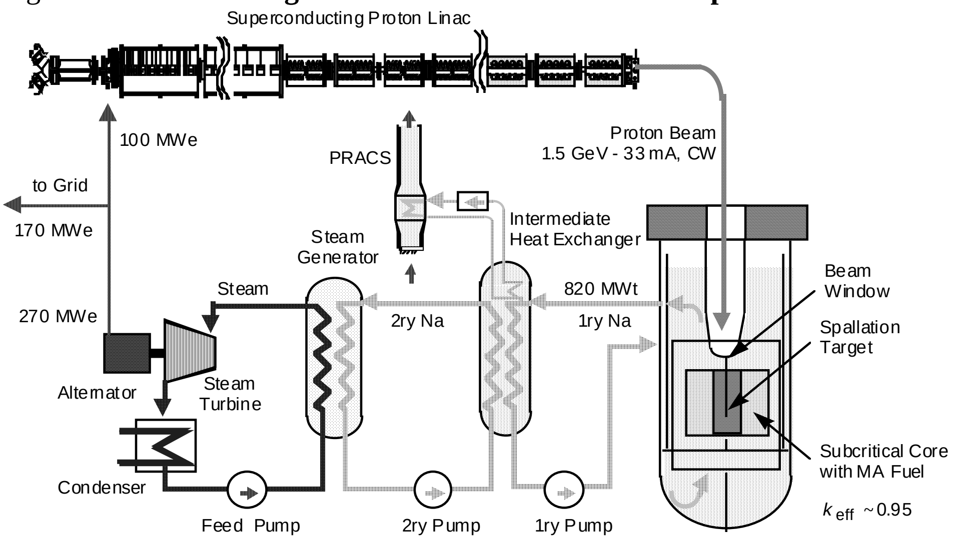


Table 1 **Major parameters of the sodium cooled ADTS**

MA/Pu Inventory	1950/1300 kg
k_{eff} (Initial/Max./Min.)	0.93/0.94/0.92
Sodium Void Coefficient	+4.5% dk/k
Doppler Coefficient	-2.2×10^{-4} T dk/dT
Thermal Power	820 MW
Transmutation Rate	250 kg/y
Power Density (Max./Ave.)	550/380 MW/m^3
Coolant Temperature (In/Out)	330/430 °C
Coolant Velocity (Max.)	8 m/s

With a 1.5 GeV - 33 mA incident proton beam, the target/core having an effective neutron multiplication factor of around 0.95 produces 820-MW thermal power. The net MA transmutation rate is approximately 10%/y, at a plant load factor of 80%. Heat transport and power conversion systems in the plant design are similar to those for a sodium cooled FBR plant. Electricity of 270 MW is generated through conventional steam turbine. One third of electric power is supplied to its own accelerator operation. The major parameters and the schematic diagram of the ADTS plant are shown in Table 1 and Figure 2, respectively.

Figure 2 **Schematic diagram of the sodium cooled ADTS plant**



Lead-Bismuth cooled accelerator-driven transmutation system

There are several advantages of heavy liquid metal coolants (lead or lead-bismuth) for fast neutron systems in comparison with sodium coolant, though their thermal property is inferior to that of sodium. In accelerator transmutation systems, lead or lead-bismuth can play roles of both coolant and spallation target material. Their neutron slowing down power is smaller than that of sodium, and hence neutron spectrum becomes harder in the lead or lead-bismuth cooled core. The hard spectrum is preferable for MA transmutation. Their chemical inertness is particularly attractive for safety. This also offers the possibility to eliminate secondary heat transport loops.

Corrosion which is one of the most important problems in lead or lead-bismuth cooled systems is significant at high temperatures. The melting point of lead-bismuth is almost the same as that of sodium while that of lead is considerably higher. So, we selected the lead-bismuth as the coolant material for the dedicated ADTS.

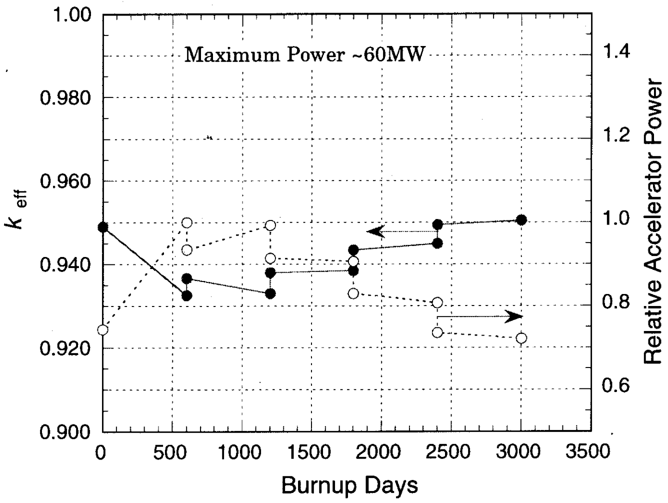
In this study, we investigated the basic characteristics of lead-bismuth cooled ADTS, such as the change of the multiplication factor, the fuel composition change and the transmutation rate. The MA inventory and the transmutation rate are especially important factors in the transmutation system. It is also significant to suppress the excess reactivity change during burnup for minimising the change of the proton beam current.

Calculation of neutronic characteristics

The survey calculation was performed to investigate the neutronic characteristics of lead-bismuth cooled ADTS. First, the fuel compositions were determined by adjusting the Pu content and the fraction of inert matrix to set the initial subcritical level at 0.95. Second, the burnup calculations were done for these fuel compositions. The MA transmutation rate and the burnup swing are especially important to estimate the ADTS performance. The purpose of the survey was to optimise the core parameters for maximising the MA transmutation rate and minimizing the burnup swing. The quantitative goals were the MA inventory below 2500 kg, the MA transmutation rate above 10%/y, and the burnup swing below 2% during 10 years full power operation.

The core sizes were determined by heat-balance calculation using the fuel pin pitch to diameter ratio (P/D) and coolant velocity as the parameters. The P/D value was varied in the range from 1.5 to 1.9. The coolant velocity was varied from 1.2 to 2.0 m/s. In this calculation, the core thermal power was assumed to be 800 MW. The coolant temperature rise through the core was set to 100 ºC, and the average linear power, 300 W/cm. The core height was fixed at 100 cm and the diameter of the spallation target region was set at 50 cm, in all cases.

Figure 3 **Changes of the multiplication factor and proton beam power with burnup**



Fuel is mixture of mono-nitride of Pu and MA and inert matrix (ZrN). The initial Pu content was varied from 0 to 60%. ^{15}N enriched nitrogen are used for both (Pu,MA)N and ZrN. The compositions of Pu and MA used in this study are those in the spent fuel of the 33 GWd/t burnup in PWR after 5 years cooling.

The multiplication factor was calculated by the diffusion calculation. The group constants of fuel region were obtained by the collision probability method considering heterogeneous effect by the SLAROM code. [6] The cross section library used in SLAROM was JFS-3 type [7] 70 group constants set based on the JENDL-3.2 library. [8] The diffusion calculation was done by the modified CITATION code [8].

The burnup calculation was done by the ABC-SC code system. The ABC-SC code system consists of SLAROM, CITATION and ORIGEN2. [10] The changes of effective cross section with burnup for about 40 actinide isotopes including the JENDL-3.2.2 library were considered in each burnup step in ABC-SC. For other isotopes, the values in ORIGEN's fast reactor library were used. The burnup calculations were done for 5 burnup cycles. We assumed one burnup cycle consists of 2 years burnup and 3 years cooling. After the cooling, FPs were removed and MA of equal mass to the FPs was added to the burnup fuel to recycle.

Figure 3 shows the changes of the multiplication factor and the proton beam power for the ADTS with optimum set of core parameters. In this case, the initial Pu fraction is 40% and P/D is 1.5. The coolant velocity is 2.0 m/s, corresponding to the power density of about 180 W/cm^3. The figure indicates that the change of proton beam power is about 30%, though the burnup swing in 10 years operation is only 1.8%. The maximum beam power is needed at the maximum system subcritical level for constant power operation, corresponding to about 60 MWb.

The major core parameters of the lead-bismuth cooled ADTS are shown in Table 2.

Table 2 **Major parameters of the lead-bismuth cooled ADTS**

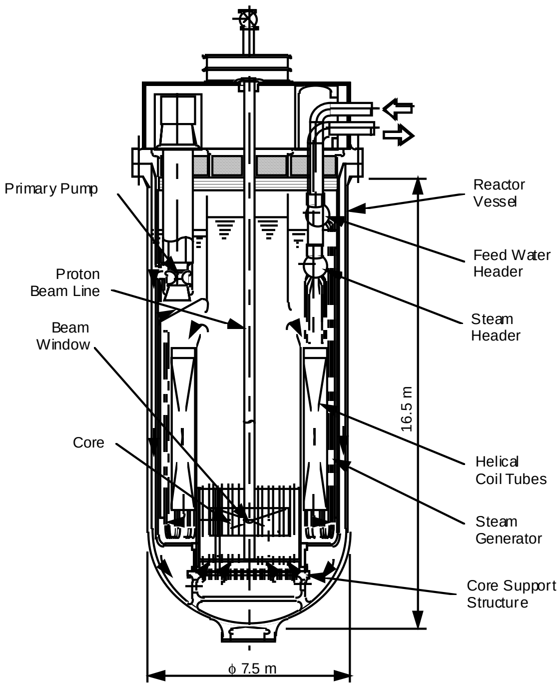
Core Thermal Power	800 MW
Core Height	1000 mm
Core Radius	1200 mm
k_{eff} (Initial/Max./Min.)	0.95/0.95/0.94
Linear Power Rating (Max./Ave.)	520/300 W/cm
Power Density (Max./Ave.)	310/180 MW/m^3
MA/Pu Inventory	2500/1660 kg
Fuel	(MA, Pu)N, inert matrix ZrN initial Pu fraction 40% N-15 enriched
Pin Pitch-to-Diameter ratio	1.5
Coolant Temperature (In/Out)	330/430 °C
Coolant Velocity (Max.)	2 m/s
Sodium Void Coefficient	-4.8% dk/k
Doppler Coefficient	-3.7×10^{-4} T dk/dT
Burnup Swing	1.8 %
Variation in Beam Power	30 %
MA Transmutation Rate	500 kg/cycle (20%/cycle)

Conceptual design of lead-Bismuth cooled ADTS plant

The plant design was principally based on the early design study of a 1500-MWt lead cooled fast reactor at JAERI. [11] For designing lead-bismuth cooled system, it is necessary to overcome the subject of aseismatic design and thermal stress resistance because the reactor will be excessively massive. The loop type configuration, which was employed in the sodium cooled ADTS design, is not suitable for lead-bismuth cooled system, because it will be difficult to realise the heavy piping system. So, the pool type reactor is selected for the lead-bismuth cooled ADTS. Intermediate heat transport system is possible to be eliminated with lead-bismuth coolant. Cr-Mo steel is used as the structural material considering the corrosion resistance in lead-bismuth, mechanical strength and ductility. The plant heat balance was calculated and the sizes of the primary pumps, the steam generators and the core vessel were determined.

The concept of the lead-bismuth cooled ADTS is shown in Fig. 4. The core, the core support structure and the primary heat transport system components are contained in the reactor vessel. The primary heat exchangers are the integral type steam generators consisting of helical coil tubes. This type of steam generator allows to reduce the reactor vessel size. The space between the outer shell of steam generator and the reactor vessel is the cold-leg flow path of primary coolant. The primary pumps are located at the cold region of the primary coolant flow path. The pumps are located above the helical coil steam generator, and the steam generators are suspended from the reactor upper flange from the viewpoint of maintenance. The plant has the water-steam cooled decay heat removal system, that is the primary reactor auxiliary cooling system (PRACS). The PRACS heat exchanger coils are located above the steam generator coils. Electrical heaters are wrapped around the PRACS piping to prevent lead-bismuth from freezing.

Figure 4. **Concept of the lead-Bismuth cooled solid ADTS**



The present study is preliminary one, and there remain many technical issues to be solved before finalising the design. One of the major technical issues is corrosion/erosion of material in lead-bismuth coolant. Candidate materials are 2.25Cr-1Mo steel, 9Cr-1Mo steel, etc. The dedicated ADTS does not have the primary purpose of power generation. It can be operated at about 100 °C lower temperatures than commercial sodium cooled fast reactors, sacrificing the efficiency of thermal to electricity power conversion. The low operating temperature is expected to mitigate the corrosion/erosion problem.

The beam window suffers from the stress due to the differential pressure between the accelerator vacuum and system operating pressure. In the lead-bismuth cooled system, the differential pressure is much larger due to high density of lead-bismuth than in the sodium cooled system. The pressure loads together with thermal stress cause a very difficult problem of beam window mechanical strength. This problem could further worsen in a high flux environment of protons and neutrons.

In lead-bismuth coolant, fuel subassemblies are subjected to upward buoyancy force that is larger than downward gravitational force. Design measure for mechanical constraint should be needed to keep the subassemblies fixed.

The problem of ^{210}Po generation by capture reaction of ^{209}Bi arises in lead-bismuth cooled system. The amount of radioactivity in the primary coolant was estimated to be the same level as conventional fast reactors. The estimation of radioactivity and toxicity of the spallation products is now underway. Some purification system to remove ^{210}Po and spallation products from primary coolant will be necessary.

Effects of accelerator beam trips on fuel pin and beam window

One of the most important effects of beam trips is thermal fatigue of ADS component materials. A beam trip changes temperatures in ADS components. Changes in temperatures causes changes in thermal stresses and thermal strains in the components. Repeated temperature changes (thermal cycling) due to frequent beam trips can cause thermal fatigue of the component materials. Damages caused by thermal fatigue can lead to degradation of their structural integrity and reduction of their lifetime. Rather frequent replacement of the components of concern and/or special design measures to mitigate the problem might be required, if the accelerator reliability could not be sufficiently improved.

Preliminary analyses were made on thermal and structural responses of a fuel pin and a beam window for the experimental facilities [12] planned under the Neutron Science Project.

Analyses of fuel pin

Two-dimensional thermal-hydraulic and structural analyses were made on the single fuel pin with the average power of the planned 60 MWt experimental system. Conditions of analyses are listed in Table 3.

The maximum fuel temperature calculated is 1 520°C, sufficiently lower than the melting point around 2 800°C. Tensile strength of the oxide fuel pellet is around 10 kg/mm^2 at 1 500°C. Calculated maximum tensile stress of the fuel pellet is around 37 kg/mm^2. It is expected that one or two radial cracks will be formed. The cracks will relieve the stress and preventing from further crack formation and propagation.

It is predicted there are no possibilities of tensile/creep rapture of the 316 SS cladding, since the internal pressure of fuel pin due to gaseous FP release is low. With an appropriate gap, fuel clad mechanical interaction can not occur. The major factor that can affect the integrity of cladding is fatigue caused by thermal cycling.

The maximum thermal strain of the cladding is 2.3×10^{-4}. Since the design fatigue strain range at 10^7 cycles is 10^{-3} at 430°C, it is evaluated that fatigue damage to cladding is negligible.

Table 3 **Conditions of analyses of fuel pin**

Fuel	UO_2
Bond	Helium
Cladding	316 SS
Pin Diameter	5.4 mm
Pin Pitch	7 mm
Active Core Height	850 mm
Coolant	Sodium
Coolant Temperature (In/Out)	330/430°C (normal operation, average)
Linear Power Rating	120 W/cm (normal operation, average)

Analyses of beam window

Two-dimensional thermal-hydraulic and structural analyses were made on the beam window of the planned high-power target experimental facility. The beam window is a hemispherical cap made of 316 SS, having a radius of 120 mm and a thickness of 1 mm. The incident proton beam with a power of 7 MW has a diameter of 200 mm and a uniform current density distribution. The beam window is cooled by upward flow of sodium coolant. The operating pressure of the sodium coolant at the level of the beam window is 3 Mpa.

The maximum operating temperature calculated is 525°C, and the temperatures at evaluated cross sections are in the range from 340 to 515°C. Results were evaluated for a high-temperature operating time of 8 760 h according to “High-temperature Design Guideline for FBR Prototype Class 1 Components” (STA) and “Technical Standard for Structures of Nuclear Facilities for Power Generation” (MITI).

All stresses and strains are well within their allowable limits, and the allowable number of operating cycles is greater than 10^6. Fatigue damage to the beam window was evaluated to be negligible.

Concluding remarks

A preliminary design study was made for a lead-bismuth cooled accelerator-driven system as an possible alternative option to the sodium cooled solid system concept based on the current LMFBR technology. The core parameters were determined to achieve minimum variation in the effective multiplication factor during burnup cycles. Pool-type configuration was selected for the lead-bismuth cooled system from the point of aseismatic design view rather than loop-type used for the sodium cooled reference system. Major technical problems to be resolved are material compatibility, beam window design, etc.

Preliminary analyses were performed on thermal and structural responses of a fuel pin and a beam window during a normal operation and a beam trip transient. It was estimated that fatigue damages to the fuel cladding and the beam window caused by beam trips are negligible. More detail design and further analyses are needed to formulate design considerations, R&D needs, and the requirements of accelerator reliability.

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REACTOR PHYSICS ANALYSIS OF HYBRID SYSTEMS

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Introduction

This paper presents two different approaches to analyse ADS concepts in order to optimise them. One is based on the analysis of external proposals with CEA codes (SPARTE, ERANOS) and using the same data (JEF 2). The second one is a broad study aiming at analysing the physics of different technologies (gas cooled, metal cooled, molten salt,...).

The analysis of hybrid reactors is based on the SPARTE system i.e. the High Energy code modules developed at CEA/DAM coupled to the ERANOS deterministic set of tools and adjusted data and the TRIPOLI Monte-Carlo transport code. The data essentially rely on JEF2.2 in the adjusted library ERALIB. These methods are qualified upon the MUSE experimental program in MASURCA, the mock-up facility at CEA - Cadarache.

External proposals analysis

The aim of the analysis of various concept is to compare the transmutation potential based on same data and method.

The analysis focused on 4 main projects. A solid fuel system like the Energy Amplifier and 3 proposals of fluid fuel systems. Each study presents the CEA vision of the proposal and should not substitute to the actual project. In all configurations the spallation target is liquid lead.

Energy amplifier

The aim of that study was to take inspiration from a project proposed by CERN (The Energy Amplifier of C. Rubbia et al.) in order to test the neutronics computation scheme we use for Accelerator Driven Systems. This computation scheme was tailored to the Energy Amplifier without any ambition to debate on its performances. We described a few distinctive features of ADS's among which φ^* : external neutron source importance in a subcritical core. The neutronics show a behaviour that is almost common for fast reactors as far as damages, flux and power rating are concerned. However, it seems that the uncertainties coming from DATA bases can affect the reactivity swing due to irradiation and the nominal values of most reactivity coefficients. This first phase should lead to a collaboration with CERN in the field of R&D related to ADS's.

Main system feature

Goal=Energy production in the $^{232}Th/^{233}U$ cycle.
Thermal Power: 1.5 $GW_{Th.}$.
Proton beam : E_p = 1 GeV, I_p = 12.5 mA.
K_{eff} = 0.98
Cycle length : 5 years without refuelling. (Burn-up: 115 GWJ/t).
Target, Coolant (natural convection) and Reflector : Lead (Total Mass : 10^4 T).
Annular Core with 2 fuel zones + 1 Breeding blanket.
Fuel : $^{233}UO_2$ + $^{232}ThO_2$.

Specifics

In the $^{232}Th/^{233}U$ cycle, the production of minor actinides is ~ 100 times less than in the conventional $^{238}U/^{239}Pu$ cycle. This, obviously, needs to be moderated by the long-term fuel radiotoxicity. In that type of nuclide fuel cycle, 2 elements drive the fuel behaviour as far as it contributes to the neutron balance over the operation cycle. There is, first, a fast effect due to the build-up in ^{233}Pa and, second, the accumulation of fission products versus breeding of ^{232}Th. The appearance of capturing ^{233}Pa is flux dependant, it is worth ~ 2000 pcm and appears ~ 1 month after Beginning Of Life.

Then, the reactivity loss is essentially due to the build-up in Fission Products. The uncertainties affecting the data concerning these nuclides should be deeper investigated, considering the effect a reactivity swing can have on the beam current at that level of subcriticality.

Operational parameters

	Energy Amplifier		*Superphénix*
	t=0	t = 1800 FPD = 5yrs = EOC	t=0
Mean flux (Internal region)	3.8×10^{15} n/s.cm²	3.9×10^{15} n/s.cm²	4×10^{15} n/s.cm²
Radial Peaking factor	1.8	2.2	1.2
Max Linear Heat rating	570 W/cm	670 W/cm	480 W/cm
Speed dpa_{max} Inner core	0.10 NRT/day	0.14 NRT/day	0.15
dpa_{NRT} , inner core	0	216	80
Keff	0.97978	0.94390	1
Source intensity	2.02×10^{18} n/s	6.75×10^{18} n/s	
φ *	1.17	1.03	
Beam current	13.2 mA	43.2 mA	

Accelerator Driven Transmutation technique (C. Bowman)

The molten salt accelerator-driven waste burner proposed by C. Bowman is dedicated to the destruction of transuranic elements. This system is a thermal sub-critical reactor (k_{eff}=0.96). It's composed of fluoride salt flowing through a lattice of graphite channels. It's a big reactor with 5% salt in volume in the graphite. The scenario in which the system is implied (closing of the nuclear cycle) leads to equilibrium notion. After explaining the principal characteristics of the molten salt reactors, we present a description of the Tier 1 system and its chronological evolutions. A neutronic study is made on the cell and on the core to show the principal characteristics of the system (spectra, reactivity, cross sections...). The cell calculations allow to show three important parameters. The graphite temperature has a big impact on the hardness of the spectrum. The proportion of FP's in the feed is also very important because their capture cross section plays a role in the system spectrum and reactivity. A set of parametric studies was conducted to find the best proportion of fission products to reach a target objective in reactivity. The cell diameter also has an effect because of spatial self shielding which impacts over the cell reactivity. The impact of that study over the strategy foreseen for these applications is not negligible. One finds an optimisation at the cell and core level can yield a new vision on the system.

Main system features

Accelerator	
Beam Energy	1 GeV
Beam Current	49.3 mA
Target	Liquid Pb (Cold window upstream)
Core	
Volume	35.8 cm^3
Flux	1.84×10^{15} n/cm^2s
$V_{salt}/ V_{graphite}$	5%
Channel diameter	7 cm
V_{salt} In & Outside	1.925 m^3
Power density (Salt)	390 W/cm^3
Power density (Channel)	19.5 W/cm^3
Salt speed	2 m/s
Salt Temperature	700°C-600°C
Number of salt channels	122
Carrier salt	$NaF-ZrF_4$
Salt density	3.5 g/cm^3
Salt feed flow	2.2 l/day
Graphite density	2.25 g/cm^3
Construction Materials	Hastelloy N modified
Performances	
k_{eff}	0.96
P_{th}	750 MWth
Carnot Thermal efficiency	42 %
Yearly Actinide burn-up	300 kg/year
Irradiation time	5 years

Accelerator Transmutation of Waste (ATW)

General description of ATW

Objective: Burn TRansUranics (TRU's) corresponding to a PWR spent fuel after 10 years of cooling time. This aim gives the actinides composition at equilibrium. The spectrum is superthermal. The reactor has a liquid fuel in which the carrier salt is a molten fluoride flowing through a lattice of graphite (80 m^3). The salt fraction is 13% amounting to ~ 100 kg of TRU. The spectrum is therefore Superthermal.

Characteristics of ATW

Target: Molten lead.
P = 500 MWth.
I = 11 mA; E_p = 800 MeV; K_{eff} = 0.96; Average.
Flux: 2.45×10^{14} n/s.cm² (Max/Mean flux: 7)

Strategy to equilibrium and performances

After one year of operation, the fraction of heavy nuclei in the refueling stream has to be superior to the fission rate in order to keep K_{eff} at an acceptable level. Otherwise, K_{eff} would rapidly (6 months) fall down to ~ 0.65. The feed rate will have to remain superior to the fission rate for 5.5 years and then decrease down to the level of equilibrium for the 8 following years. The equilibrium is reached after ~ 20 years for all nuclei except ^{246}Cm.

Toxicity: One 500 MWth ATW module can transmute the TRU production of 2/3 PWR (i.e ~ 180 kg).

Sensitivity of reactivity to the feed rate at equilibrium.

– 0.5 × assigned figure for 24 hrs : Δk = -32 pcm.

– 1.5 × assigned figure for 24 hrs : Δk = +46 pcm.

We can see that an accident on the feed stream will not have a very severe effect over the core. It also indicates both a necessity for the control of reactivity in that type of system on several time scales as well as a good knowledge of the composition.

Accidental situations (Salt Volume Fraction Constant). In that study, 4 configurations are characterised. The size of the graphite channel is modified, graphite and salt collapse one on the other and mix together & the spallation region is voided.

– Graphite lattice channels +30% : Δk = +1400 pcm.
– Graphite lattice channels -30% : Δk = -1700 pcm.
– Mix graphite and salt : Δk = -7500 pcm.
– Molten lead of the spallation target voided : Δk = -410 pcm.

The design of that type of system is obviously sensitive to the elementary cell dimensions. It should determine what is the most reactive situation in order to avoid positive reactivity transients.

JAERI

JMS : Burn* Minor Actinides. Fast *Spectrum

Objective: Burn Minor Actinides corresponding to a PWR spent fuel after 3 years of cooling time. The core is made of molten chloride. The Salt volume is 2.5 m^3 amounting to ~ 5 Tons of MA's. The spectrum is Fast as no significant moderation occurs onto the carrier salt.

Characteristics of JMS

Target : Molten Salt.
P = 800 Mwth.
I = 24 mA; Ep = 1500 MeV; K_{eff} = 0.95;
Average Flux : 2.06×10^{15} n/s.cm² Max/Mean flux : 15)

Sensitivity of reactivity to the feed rate at equilibrium.

- $0.5 \times$ assigned figure for 24 hrs : Δk = -1.5 pcm.
- $1.5 \times$ assigned figure for 24 hrs : Δk = +1.6 pcm.

Study on the effect of a shift in the spectrum of the spallation source. (φ^* is an indicator showing the relative efficiency -importance- of external neutrons as compared to fission neutrons)

- Softer spectrum : φ^* - 9 %.
- Harder spectrum : φ^* + 8 %.

The difference is not negligible. Again, the source configuration should correspond to the maximum efficiency so that any type of transient would not lead to an power excursion.

Strategy to Equilibrium and Performances.

The equilibrium is reached after ~ 15 years of operation.

The initial composition shows an important fraction of MA's. This will lead to a build up in fissile Pu, after ~ 3 years of operation. Then 2 strategies can lead to equilibrium if we intend to keep reactivity constant:

1. The feed rate remains constant : An absorber (^{10}B) has to be dissolved into the salt in order to anchor k_{eff} at 0.95 (Equilibrium).
2. The feed rate is changed : The feed rate is set to 0 for 6 years so that the inventory decays to keep k_{eff} = 0.95

Toxicity: One 800 MWth JMS module can transmute the MA's production of 10 PWR (i.e ~ 290 kg).

Conceptual designs of demonstration reactors.

Type of studies conducted

Based on the previous analyses, different concepts were proposed and analysed for a hard spectrum demonstration core.

The motivations for a demonstration hybrid reactor stand fairly simply as the following objectives :

1. Deployment of a facility using non traditional fuels, e.g. Inert matrix, High actinide content and enrichment.
2. Technological coupling of the main components i.e. Accelerator, Target and Sub-critical blanket, including continuous integrated control of the system.
3. Optimisation of an eventual burner in terms of neutron balance excess.

4. Materials behaviour under severe conditions (High and Hard Flux, strong gradients, High temperature and corrosion exposure, maximum damage production, innovative coolants).

A set of complementary applications, with parallel objectives, can further justify the project, i.e. neutron beams for biology or material physics applications and irradiation facility for space applications and future nuclear prospects.

In that perspective, we considered a set of ″Fuel/Coolant/Elementary design″ systems that could match the target objectives of a *Demonstration* device. The concept was applied to MOX, Nitride and Carbide type of fuels, combined with Sodium, Lead (or Lead Bismuth Eutectic), or Gas coolant in a Hexagonal base, a rectangular lattice or a pebble bed elementary cell. The power in each configuration was set to 100 MWth and the reactivity k_{eff} = 0.92 ÷ 0.98 with a computed overall efficiency of external source neutrons φ^* = 1.2 ÷ 1.4. No full thermal optimisation was conducted yet each individual system can still yield some indications as to how appropriate it would be to meet the objectives mentioned above. They all have very similar features i.e. high reactivity loss over the cycle, because of a high initial enrichment as well as of a small size, itself because of a high (necessary) neutron leakage rate. Peaking factors are connected to the reactivity and a specific study is addressing the question of what level of subcriticality is sound to an Accelerator Driven System for Transmutation (cf. same conference, paper on *Comparison study of critical vs ADS reactors from the point-kinetics standpoint*).

Apart from those common features, a deeper investigation will be necessary to optimise each type of system.

Basic conclusions.

A generic system with 34% Fuel, 49% Coolant, 17% Structural material, a fissile enrichment ~ 29% in an annular core of less than 400 l will give k_{eff} ~ 0.96. Provided the external source neutron efficiency is more than 1.2, the beam power required to produce 100 Mwth will be ~ 1.3 mA @1GeV. The flux level will be 2×10^{15} n/s.cm² (Peaking factor ~ 2) with more than 65% of the neutrons above 0.1 MeV. The core maximum power density will be ~ 440 W/cc and structural materials will have a damage rate ~ 0.1 dpa_{NRT_Steel}/day. At that level of enrichment, the reactivity swing will follow an average loss of the order of - 20 pcm/day. Of course, most of these features will degrade over the core cycle, just as the reactivity loss. For instance, with a 3 months long cycle, the peaking factor will deteriorate by about 10%.

There is a positive interest to limit the core thickness if the peaking factor has to be low. It would also be sound to put demonstration/irradiation sub-assemblies close to the target in order to maximise the flux efficiency.

This generic image has been pushed ahead and a gas-cooled demonstration reactor was optimised on a first order basis. The hypothesis were taken from existing projects like GBR-4 and a hole was inserted in the centre in order to put the spallation module.

A set of reflexions concerning these cores is still going on and includes the overall core safety as well as generic aspects of control and surveillance issues and long term behaviour.

Nuclear data and Computation tools

The data for actinides, fission products and structural materials are evaluated in the JEF2-2 library. This library is in turn adjusted, within the error bars, groupwise, to match the results of integral experiments. The corresponding adjusted data set is ERALIB. In the Intermediate Energy range the data are computed from double differential cross sections during the Monte-Carlo transport.

SPARTE stands for SPAllation Ralentissement (slowing-down) Transport & Évolution (decay). It is made of three blocks. *T*he spallation source is computed with a CEA **-** Bruyères le Châtel version of HETC. *C*ore calculations propagate the source in either ERANOS (European Reactor ANalysis Optimised System, validated for fast reactor calculations) or TRIPOLI (CEA – Monte Carlo). *D*epletion and fuel management are integrated either in ERANOS or connected to DARWIN (CEA Depletion stand alone tool).

In HETC, the Intra-Nuclear Cascade module can be tuned either on the standard Bertini or on a Cugnon type of model. Then an evaporation module, including the Fermi break-up takes charge of the excited nucleus. The transport is conducted from the energy of the incident particle down to the upper level of reactor physics evaluated libraries (i.e. 20 MeV). This threshold is called the cut-off energy. Whenever a particle is emitted lower than this limit, its main features (type, position, momentum) are "frozen" and stored in a dedicated file. For neutrons, this file is processed and then used into the lower energy transport code as a spallation source. It covers exactly the same geometry and compositions. It is obvious that the hypothesis of direct Nucleon-Nucleon interaction on which Intra-Nuclear Cascade models are based is crude in the range $20 < E < 200$ MeV. That is why a broad international effort has been initiated in order to raise the cut-off energy up to 150÷200 MeV. This ultimate set of evaluations, concerning a high priority list of key isotopes (potential spallation targets & Construction materials), will allow to fulfil reactor physics calculations up to 150÷200 MeV either with a deterministic or with a Monte-Carlo code.

In the Lower energy range, a comprehensive computation scheme has been developed in ERANOS to fully characterise systems in a semi-automatic mode. It includes cell, core (with spallation source), and depletion calculations. Further investigations will focus on the space kinetics features of ADS'. The MUSE experimental programme at MASURCA will build the necessary basis to qualify this tool.

Conclusion

We have focused our attention towards a 2 phase approach to the physics of hybrid reactors. The study and optimisation of external proposals in a first time and the pioneer characterisation of innovative multi domain concepts. The analysis of external systems, from a reactor physics standpoint, is helpful to focus more efficiently our effort towards an application that meets our needs. For that, we use a specific integrated tool, SPARTE, based on the ultimate ERALIB nuclear data library.

A FIRST EVALUATION OF THE FEA

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Abstract

An EA core model suitable for MCNP code has been built on the available reference data and engineering judgment, in case of lack of information. The EA was modelled in cylindrical geometry by subdividing the actual EA structure into five regions: target, buffer, inner core, outer core, breeder.

ORIGEN calculations were carried out for the core region at a specific power of 52.8 MW/t; for the breeder region a total flux value of 5.49×10^{14} $n.cm^{-2}.s^{-1}$, obtained from MCNP calculations, was used. The calculations were carried out using the LMFBR libraries available in order to select the best performing one comparing the results to the data available.

Calculations were performed to verify the nominal effective multiplication factor as well as the thermal power and neutron flux at BOL conditions for each region and whole systems.

General description

The first version of Rubbia's Energy Amplifier (EA) [1] was an Accelerator Driven System (ADS) in which the subcritical reactor was a Thorium fuelled PWR-like core. In the reactor side of the EA concept, that is the one considered in the present work, the evolution of the design moved from a thermal to a lead cooled fast concept maintaining the Th mixed oxides fuel. The physical basis and the conceptual design of the Fast Energy Amplifier (FEA) are illustrated in Reference 2, that was the main source of information used in the present work.

The general layout [2,3] can be grouped into three main components: the accelerator, the heat generating unit, the heat dissipation or utilisation devices. Their description can be found in the quoted references.

Model development

The main goals of the calculations were to verify the completeness and consistency of the data available and to set up base decks and models suitable to be used as reference for further calculations and verifications.

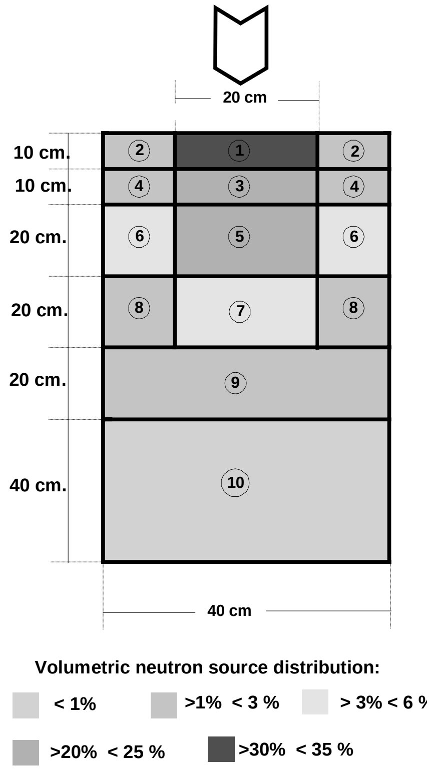
A preliminary EA three dimensional simulation was carried out using MCNP [4] version 4.2 Monte Carlo computer code. The EA was modelled in cylindrical geometry by subdividing the actual EA structure into three main regions: the target/source region, the core, the vessel and internals; the model was built on the available reference data [2,5,6] and engineering judgment, in case of lack of information.

Target/source region

The target and buffer have an external radius [2] of 40 cm. The target/source region, of 20 cm in diameter and 1.2 m in high, was modelled (Figure 1) with cylindrical meshes according to a previous calculation [7] performed with LAHET [8] code.

The MCNP neutron source distribution is derived from LAHET calculation performed with proton energy of 1 GeV and neutron energy cut-off of 20 MeV. These calculations showed an almost isotropic angular distribution and substantial spatial independence of the energy spectra. The neutron source was described in terms of volumetric distributions in the target meshes and energy spectrum for whole target.

Figure 1. **Target/source zone meshes**



Core

Core is made of hexagonal fuel assemblies similar to fast reactor ones. The EA core is divided into three regions:

1) An innermost ring with a larger pitch between pins to flat the power distribution, using pellets of ThO_2 enriched with 10% by weight of $^{233}UO_2$.

2) A middle annulus filled with fuel assemblies of ThO_2 enriched with 10% by weight of $^{233}UO_2$.

3) An outermost ring of breeder assemblies with pellets of ThO_2.

The main parameters of the EA core are summarised in Table 1. As the number of fuel assemblies in the inner and outer core zone are not available, they have been estimated from the pin volume, the total fuel mass and the power density. The solution leads to 14 and 106 fuel assemblies for the inner and outer core zone respectively.

Table 1. **Main design parameter of FEA core**

Fuel pellet inner diameter	1.1 mm
Fuel pellet outer diameter	7.3 mm
Gap thickness	0.1 mm
Fuel clad thickness	0.35 mm
Fuel active length	150 cm
Fuel bundle inner apothem	113 mm
Fuel bundle thickness	3 mm
Flat to flat of fuel assembly module	234 mm
Clad length above/below active zone	90 cm (each side)
Clad outer diameter above/below active zone	5.0 cm
Fuel bundle total length	5.3 m
Number of fuel bundles in the inner and outer core zone	120
Number of fuel bundles in the breeder zone	42
Number of fuel pins of inner core fuel assemblies	331
Number of fuel pins of outer core and breeder fuel assemblies	397
Total thermal power	1 500 MW
Specific power (power density by oxide mass)	52.8 W/g
Power density	523 W/cm^3
Initial fuel mass (ThO_2 + 0.1 $^{233}U_{O2}$)	28.41 ton
Initial breeder mass (ThO_2)	5.6 ton

The fuel elements were arranged in hexagonal geometry. While breeder zone has exactly 42 elements, the inner one has only 12 out of 14 fuel elements needed to fill the inner core zone. This geometrical inconsistency, due to the lack of detailed core description, is overcome, by the assumption of cylindrical geometry for the MCNP model (Figure 2).

The flat to flat length of fuel assembly module was assumed as thickness of the inner core and breeder regions. The core region was divided into five axial zones with constant cross sections:

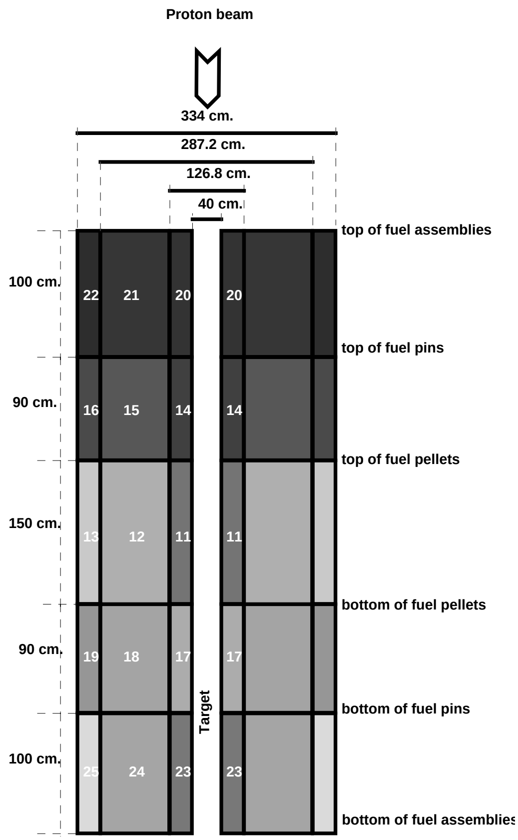
1) Active length [2] (1.5 m).
2) Fuel pins zone above and below the active length (0.9 m each): pins have an upper and lower plenum to ensure the required burn-up, but with a smaller diameter to reduce the pressure drop through the core.

Fuel bundle zone above and below fuel pins (1 m each).

The main geometrical parameters of MCNP core model are summarised in Figure 2; minimum required areas are calculated from actual dimension of the EA fuel assemblies to verify the spatial consistency of the assumptions. For each core zone, the cross sections filled with fuel, structural material and coolant are calculated from the actual geometrical data of the EA fuel assembly module.

The fuel mass was divided into the different core regions by weighting with the inner and outer core volume filled with fuel while the structural material (low absorption steel: HT-9) and coolant (lead) mass was calculated using the material density at their operative temperature.

Figure 2. **MCNP model of EA core**



Vessel and internals

The main design parameters of EA vessel and internals are listed on Table 2. An overall view of the EA MCNP model is provided by Figure 3.

Figure 3. **MCNP model of EA vessel and internals**

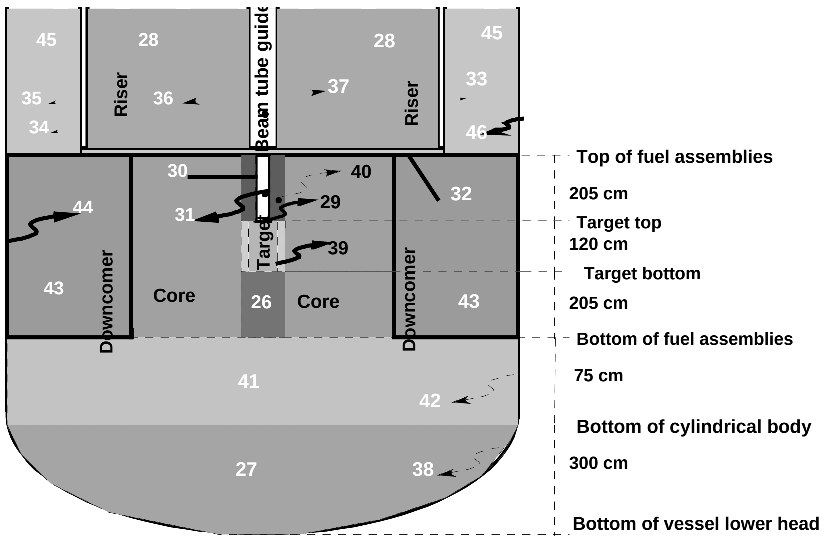


Table 2. **Main design parameter of EA vessel and internals**

Vessel outer diameter = 6 m. Vessel thickness = 7 cm. Vessel with an hemispherical lower head. Beam pipe external diameter = 30 cm. Beam pipe external diameter in the core zone = 20 cm. Beam pipe thickness = 3 mm. Beam window thickness = 3 mm at edges and 1.5 mm at centre (hemispherical shape). Riser external diameter = 3.65 m. Riser thickness = 3 cm (double wall with vacuum between, each region are 1 cm thick). Core support disk thickness = 15 cm.

The model extends to 25 m in height and 6 m in diameter and includes:

- The vessel wall.
- The downcomer near the riser and the core.
- The vessel lower head.
- The core.
- The target and the central buffer region.
- The beam tube guide walls and internal voided region in the riser and core zone.
- The beam window, modelled as a disk with the average thickness.
- The core support structure has been modelled as a disk 15 cm thick, made of 60% of steel, the remaining 40% being holes for various purposes.
- The cylindrical double wall barrel with vacuum between, which separates the riser from downcomer.

Though the model could seem excessively detailed from the point of view of criticality and flux calculation, it will save a lot of work for the future simulations of the system.

Irradiation model

The main data requested for ORIGEN [9] calculations are power history and material compositions. Concerning the power history, a 5 year (1 825 days) full power operation is used [2]. For the core region, a power irradiation was used at a specific power of 52.8 MW/t; for the breeder region, a flux irradiation was used with a total flux value of 5.49×10^{14} n/(cm² s) obtained from the MCNP calculations. For the material composition, the same core description of the Monte Carlo model is used to quantify the number of fuel assemblies in the inner and outer core.

EA steady state results at BOL

As a first step, some calculations were performed to verify the nominal effective multiplication factor as well as the thermal power and neutron flux at Begin Of Life (BOL) conditions.

In order to achieve the steady state value of the effective multiplication factor (k_{eff} = 0.98) [2] several calculations were performed by varying the enrichment in $^{233}U_{92}$ of the fuel loaded in the core at BOL. The results lead to a fuel enrichment of 10.33% which corresponds to the nominal EA multiplication factor. The fuel composition was selected for the subsequent power and flux calculations.

Table 3 lists the results for the flux and power calculations for each region and whole systems.

The energy released per fission in ^{233}U calculated by MCNP code is 180.84 MeV while, considering all contributions to the recoverable energy (kinetic energy of fission fragments and fission neutrons, β and γ rays from fission product decay, prompt γ rays and γ rays from capture of structural material) we have an energy ranging from 194 to 203 MeV/fission (depending upon the structural material in the reactor) [13].

Taking into account the above considerations and the beam heat depositions in lead and window (7.07 MW) [2], the total FEA power is estimated to range from 1 347 to 1 409 MWth which is in reasonable agreement with the nominal power of 1 500 MW_{th}. The standard deviation of the Monte Carlo calculations is 6.18% (from 83 to 87 MW) and many uncertainties already exists for the spallation neutron yield.

Table 3. **Fast energy amplifier steady-state calculation at BOL**

Region	**Neutron flux [n/(cm^2 s)]**	**Flux σ (%)**	**Fission prompt power (MW)**	**Fission power σ (%)**
Inner Core	3.42049×10^{15}	2.44	201.08	3.51
Outer Core	2.03134×10^{15}	2.87	1,047.10	4.08
Inner and outer core average	2.20805×10^{15}	3.77		
Breeder	5.63569×10^{14}	2.98	0.60	3.03
Core and breeder average	1.75340×10^{15}	2.80	1,248.78	6.18

System effective multiplication factor (K_{eff}) = 0.980022 (σ = 0.060 %).
Spallation neutron yield = 26.5.
Proton beam nominal current = 12.5 mA.

As regards the neutron flux, a direct comparison is not possible because its value is not explicitly indicated in the FEA nominal parameters of chapter 4 of Reference 2. On the other hand, there is a reasonable confidence that the calculated value is in good agreement with the FEA ones; in fact, Table 2.3 of chapter 2 in Reference 2 reports a neutron flux of 2.33 x 10^{15} n/(cm² s) for a power density of 60 W/g_{Th} at breeding equilibrium while the calculated (averaged inner and outer core) MCNP value is of 2.21 x 10^{15} for a power density of 52.8 W/g_{MOX} at BOL.

Irradiation calculations

For the core region (inner and outer), the calculations were carried out considering a core power of 1 500 MW with a specific power of 52.8 W/g of oxide for a 5 year cycle (1 825 Effective Full Power Days, EFPD). These data lead to a total mass of 28.409 t of oxide and to a burn-up of 96 GWd/t compared to 28.41 t and 100 GWd/t of Reference 2. Using the cross section values for ^{232}Th and ^{233}U, a flux value of about 2.6×10^{15} n/(cm^2 s) was estimated compared to an average neutron flux value from ORIGEN ranging from about 2.43×10^{15} n/(cm^2 s) to 2.49×10^{15} n/(cm^2 s) according to the used library (from 2.52×10^{15} n/(cm^2 s) to 2.56×10^{15} n/(cm^2 s) in the first time step of 3 days) and of about 2.21×10^{15} n/(cm^2 s) from the MCNP at BOL.

Figure 4 shows the obtained k_∞ values during the cycle. The previous values are normalised to 0.98 at BOL. It is very difficult to compare these results because we cannot find a figure with data such as to be qualitatively compared. The only comment we can make at the moment is that our values seem to be higher. As concerns the ORIGEN libraries used, the qualitative responses are similar, except for the N15 advanced oxide recycle one. That gives a weight of the neutron absorption in the core region higher by a factor of greater than two, compared to the results of the other two libraries.

In the following Figures 5 and 6 the mass ratio between ^{233}U and ^{232}Th and between ^{233}Pa and ^{233}U are represented respectively. As concerns the mass ratio, all the libraries give values very close to each other.

In order to compare correctly the numerical values in Figure 5 with those in the corresponding Figure 5.5 of Reference 2, we have to multiply the values by 0.9957, that is the ratio of the atomic masses of Th and ^{233}U. We first notice that the initial value at 0 burn-up is different; that means that we are starting from initial conditions that are different and that there is no full consistency among the data of Reference 2. Anyway some qualitative comparison can be done. We have about the same decrease in weight ratio of 0.001 at about the same time 5 GWd/t, but we have a relative higher ^{233}U build-up leading to a value of about 0.13 compared to about 0.116 at the same burn-up.

Figure 4. **ORIGEN II k_∞ values normalised to 0.98 at BOL**

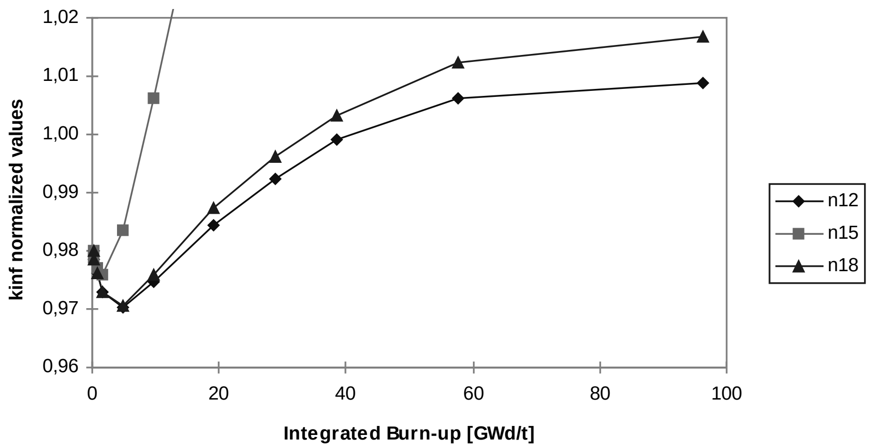


Figure 5. **Mass ratio between ^{233}U and ^{232}Th along irradiation in the inner and outer core**

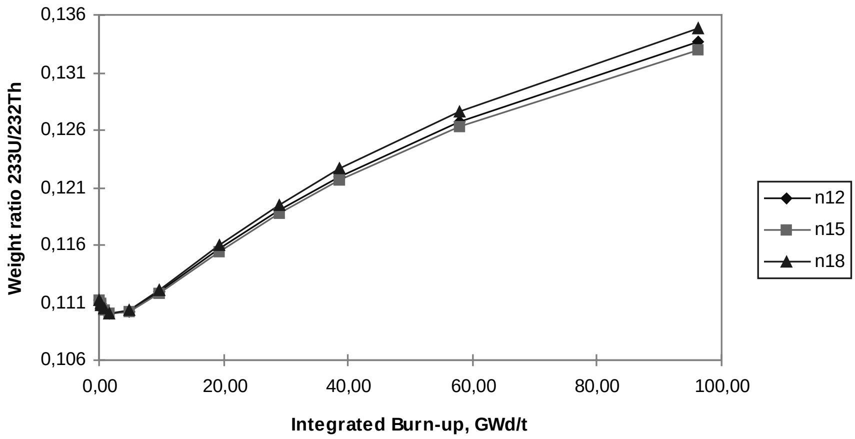
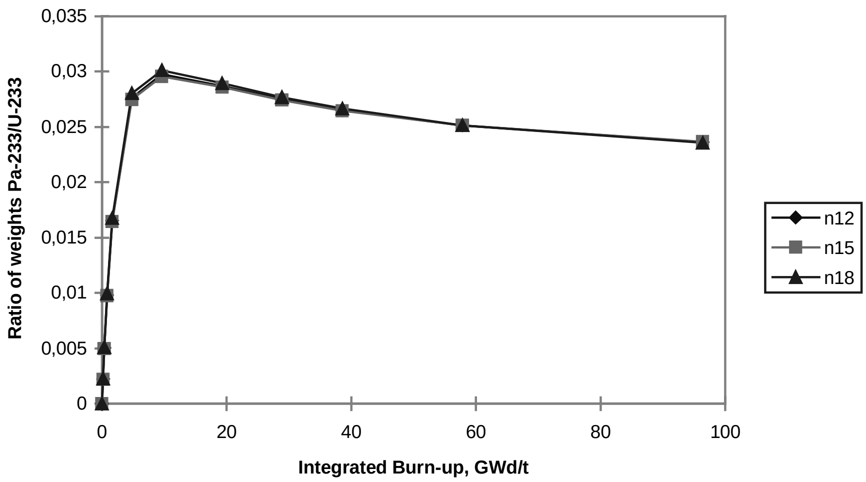


Figure 6. **Mass ratio between ^{233}Pa and ^{233}U along irradiation in the inner and outer core**



As concerns the ^{233}Pa and ^{233}U ratio we obtain a higher value of about 0.03 at 10 GWd/t and a value of 0.024 at 96 GWd/t compared to the maximum value of 0.026 at about 12 GWd/t and 0.024 at 96 GWd/t. The results show a higher maximum value at the same burn-up and in the flat region the results of Reference [2] seem linear.

The results obtained, for the breeder region, from the two libraries used are the same for all the parameters analyzed.

Concerning the specific power, in Figure 7, we get at end of cycle a value of 29.6 MW that corresponds to about 5.3 W/g to be compared to 3.0 W/g of Reference 2. For the ^{233}U amount, in Figure 8, we produced 226.5 kg compared to 242.7 kg of Table 4.1 in Reference 2, but there is no evidence that the two values refer to the same period.

Figure 7. **Breeder region: specific power behaviour**

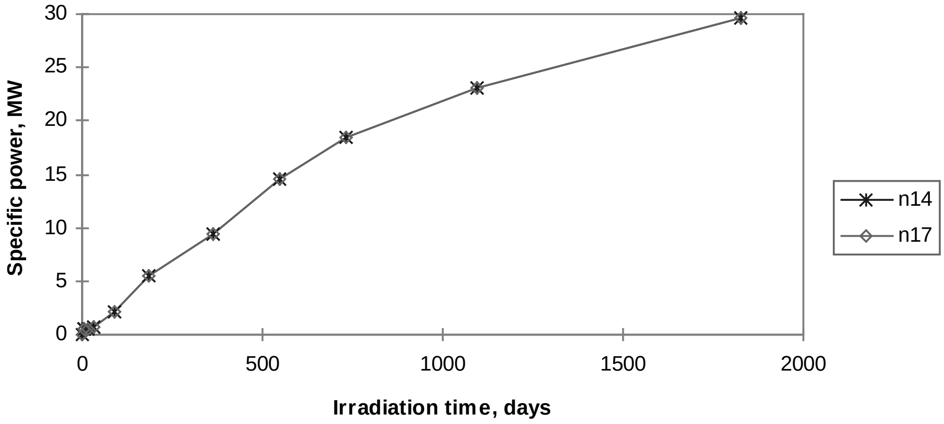
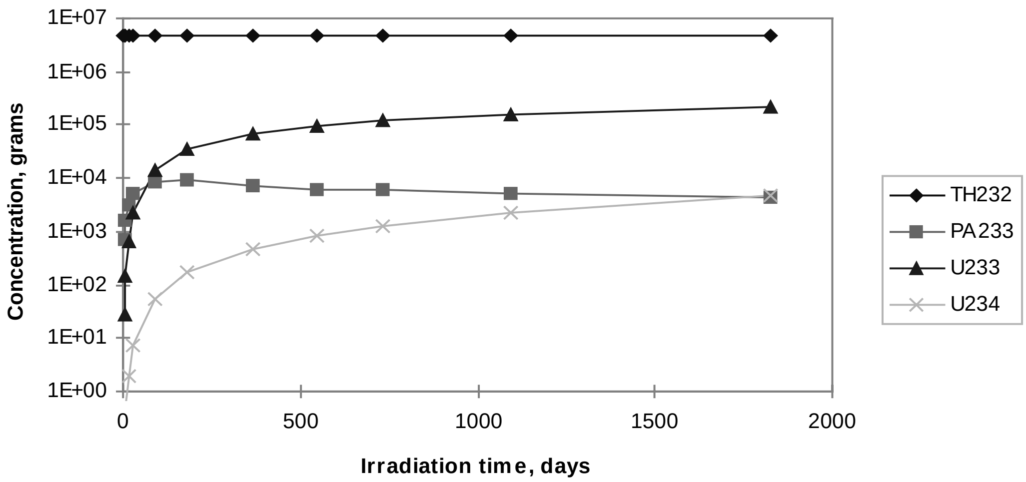


Figure 8. **Breeder region: concentration evolution of the actinides with the larger mass inventory at end of cycle**



Preliminary conclusions

This analysis represents only the first step of a conceptual safety analysis of Rubbia's FEA which should follow the steps listed below:

1) Evaluate, at BOL, the steady state multiplication factor, thermal power and neutron flux.
2) Perform several burn-up calculations in order to estimate the k_{eff} evolution and the system effectiveness in reducing the final waste inventory (burning performances).
3) To evaluate the plant behaviour under transient and accident conditions.

The ORIGEN or CINDER code coupled with LAHET and MCNP can, in principle, perform the calculations of point 2 but it will require a lot of computer time as well as calculation and typing work to interface the codes. The optimal solution is represented by the integrated code MCNPX (in development at LANL) which should be able to simulate the spallation process, the neutron transport and interaction (with Monte Carlo method) and the burn-up evolution.

For the simulation of the system (point 3), some work need to be done in order to develop a set of equations able to describe the power evolution during the transients/accidents and the simple point kinetic model based on reactivity coefficient seems to be no longer applicable: a system fed by a consistent source will not behave by following essentially the main harmonic because the source also excites the superior flux modes.

In general it can be said that the system concept does not have the chance to be less complex than the fast reactor, in terms of protection and control systems; in fact, though the system is subcritical, the following aspect plays a negative role in terms of safety and system simplification:

- Several critical masses are still present in the core: some core geometrical deformations could lead to a critical configuration.
- In case of fuel cladding failure the pellets will float on the liquid lead.
- The local power evolution during accidents and transients need to be deeply investigated: though in case of complete loss of coolant (lead) our previous calculation [11] indicates a decrease of the multiplication factor mainly due to the major influence of the (n, 2n) lead reactions in respect to the parasitic absorption and some local lead vaporisation (local void coefficient) could lead to a local power increase.

Finally, it should be noted that the choice of an operative multiplication factor so close to the unit (0.98) permits to achieve an acceptable thermal power with a relatively low proton beam current but it raises some safety concerns. The above considerations suggest that the k_{eff} evolution during the burn-up over the whole reactor life should be carefully evaluated, being a possible critical (both in sense of dangerous and neutronic) aspect from the safety margin point of view.

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PERFORMANCE ON ACTINIDE TRANSMUTATION OF LEAD-THORIUM BASED ADS

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Abstract

The FACET group at CIEMAT is studying the properties and potentialities of several lead-cooled ADS designs for actinide and fission product transmutation. The main characteristics of these systems are the use of lead as primary coolant and moderator and fuels made by transuranics inside a thorium oxide matrix.

The aim of the study is to analyse the effect of some operation parameters (fuel transuranics composition, thermal power and transuranics recycling scheme) in the ADS performance, mainly in the achieved transmutation rates and in the accelerator requirements.

The model selected enhances the energy production by the ^{233}U breeding from thorium seed. This breeding can maintain the neutron multiplication during long burnups, improving the transmutation capacity. The fuel inventory isotopic evolution during burnup will be presented illustrating the general capabilities of this strategic option for transuranics transmutation.

Introduction

One of the major problems of nuclear power production is the undesired long-lived radioactive waste that comes from the spent fuel used for electricity production. Currently many countries plan to deal with these products storing them into geological repositories. However, in the recent years an increasing attention is paid to the transmutation option as a complementary activity to geological disposal in a near future [1]. Among the foreseen advantages of transmutation are a reduction in volume of the high level waste and a reduction in the long-term radiotoxicity inventory, with an impact in reducing the final costs and potential risks of the geological repository.

The transmutation of radioactive waste could be applied to two main groups of nuclides, each group with its own elimination methods and different impact in the waste management strategy:

- The transuranics (TRUs) mainly neptunium, plutonium, americium and curium isotopes coming from the LWR's discharge. They are responsible of the long-term radiotoxicity and their elimination should be done by fission since every neutron capture just increases the mass number of the considered actinide. Any optimised TRUs transmutation strategy has to reach the highest fission to capture ratios, something not easy to achieve in the case of the fissile under threshold TRUs, and take into account the energy production during transmutation process.
- The fission products (FP), some of them with a very large radiotoxical potential in the short or even in the long term. Their elimination means to transform them by neutron capture and subsequent radioactive decay into stable isotopes, something that has some extra energy cost.

The impact of the transmutation in the waste management policy of each country will depend on the size of the local nuclear power industry and the characteristics of the current and foreseen nuclear fuel cycle, among others. An optimal use of nuclear waste transmutation is unavoidable connected to the development of partitioning methods due to the necessity to separate waste streams: TRUs and FP from the spent fuel uranium matrix. Currently, nuclear fuel reprocessing is available only in a few countries in the world. In addition, some countries assume that plutonium is another nuclear fuel instead of as waste, and burn it with the MOX fuel technologies. To them, the TRUs elimination means neptunium, americium and curium elimination in specially design systems, with different properties compared to the TRU-with plutonium charged ADS.

Due to the diverse points of view about transmutation, there is a wide scope of R&D strategies that has produced a large amount of proposed policies and systems for nuclear waste transmutation that could be available in the near future. However, in the last decade the Accelerator Driven Systems (ADS) appear among the most promising transmutation systems. Basically the common components of every ADS are:

- A subcritical core ($k<1$), loaded with the unwanted TRUs and (maybe) FP to eliminate.
- A proton accelerator, producing a beam of some mA of intensity and an energy of a few hundreds of MeV. This beam produces the spallation neutron source that the subcritical core need in order to maintain its operation regime.

The subcritical core configuration of the ADS would allow burning up some atypical actinide mixtures for long irradiation periods; with this kind of systems it is not necessary to reach some reactivity excess in order to start the operation, because the external source maintain the steady state.

The Energy Amplifier Model of Lead-thorium Based ADS

The ADS model considered in the present study is the so-called Energy Amplifier (EA), a concept developed at CERN by Prof. C. Rubbia's research group [2].

Basically an EA is an ADS cooled by molten lead, material that also works as spallation target and neutron diffusing medium. Because of the special neutronics properties of lead, the neutron regime is fast (an energy spectrum centred on a few hundreds of keV). The fast EA design is, in principle, very adequate for elimination of those TRUs non-fissile under thermal spectra (see discussion in refs. [2]). The EA concept is very flexible permitting:

- A neutron multiplication constant (k_{eff}) that could vary between 0.93 and 0.98. The chose of k_{eff} value has an important impact on accelerator requirements.
- Both metal and oxide fuel has been proposed, depending the final selection on the power density limits and the working temperatures. Typically in the EA core, the TRUs charge to transmute is mixed into a thorium matrix. The presence of thorium allows the system to maintain the neutron multiplication: the fissile TRUs burning can be compensated by the ^{232}Th to ^{233}U breeding, followed by ^{233}U fission.
- A wide range of operating conditions, with different thermal power (a few MW up to 1 500 MW) or (equivalently) proton beam conditions (300 MeV to a few GeV of energy), depending on the model considered. The operating conditions will play an important role in the transmutation efficiency.
- Options under study have been presented by Rubbia's team [2] for FP incineration, distributing them in the lead diffusive media outside the EA core.

The simulations performed in this study have been focused primarily in the effects of the initial TRUs inventory composition, the thermal power and the proton beam energy on the performance of the system. The table 1 summarised the main parameters of the models under study. Two thermal energy outputs have been considered: 200 MW, which corresponds to a large-size demo EA facility, and a medium-size energy production unit of 800 MW. For simulation purposes, it has been supposed that the increase in thermal power is gained not directly by beam intensity increase but by proton kinetic energy growth, therefore going from 200 MW_{th} to 800 MW_{th} would imply a proton energy rise from 380 MeV to 1 GeV.

The study on the nuclear fuel composition effect on the EA performance has been done for three selected TRU mixtures. These mixtures have been calculated with ORIGEN2.1 [3] (table 2) modelling a typical PWR discharge. The differences are due to the cooling down time, considered as the delay time between the PWR discharge and the EA load (10, 25 and 40 years). The hypothesis is that the PWR discharge reprocessing produces a transuranics stream without any specific element separation inside it. This means that neptunium, plutonium, americium and curium are extracted in almost the same relative percentage they have in the spend fuel.

Basically, increasing the cooling down time produces an increase in the ^{241}Am concentration by ^{241}Pu decay. This conversion will have an important effect in neutronic properties evolution of the EA because while the ^{241}Pu is a fissile actinide, the ^{241}Am is fertile (by neutron capture it produces some amount of high fissile ^{242*}Am). There is also a slight decrease of the ^{238}Pu concentration while other plutonium isotopes (^{239}Pu, ^{240}Pu and ^{242}Pu) remain near to constant. Among the curium isotopes another rapid decayed isotope is the ^{244}Cm.

Table 1 **EA Models simulated**

Number	1	2	3	4
Thermal output (MW)	200	800	800	800
Proton energy (MeV)	380	1000	1000	1000
Fuel composition		$(TRUs+Th)O_2$		
TRU mixture cooling down time (years)	40	40	25	10
Fuel oxide mass (kg)		~10 ton		
TRUs/Th	~0.32	~0.32	~0.35	~0.38
(Th+TRUs)/Pb mass fraction (core)	~0.20	~0.20	~0.20	~0.20
Cladding material		HT9 steel		

All the simulated EA models share the geometry characteristics (table 3). The geometry models used during simulation are:

- A complete detailed description of the EA core, where each single fuel pin of every fuel bundles, including its own cladding and the bundle wall, can be distinguish.
- A homogenous model, where the internal structure of a fuel bundle is homogenised to a mixture of materials, preserving the total mass and every isotope mass fraction during the conversion (figure 1).

The differences in the Monte-Carlo estimators when using heterogeneous or homogeneous descriptions of the EA has been discussed elsewhere [4].

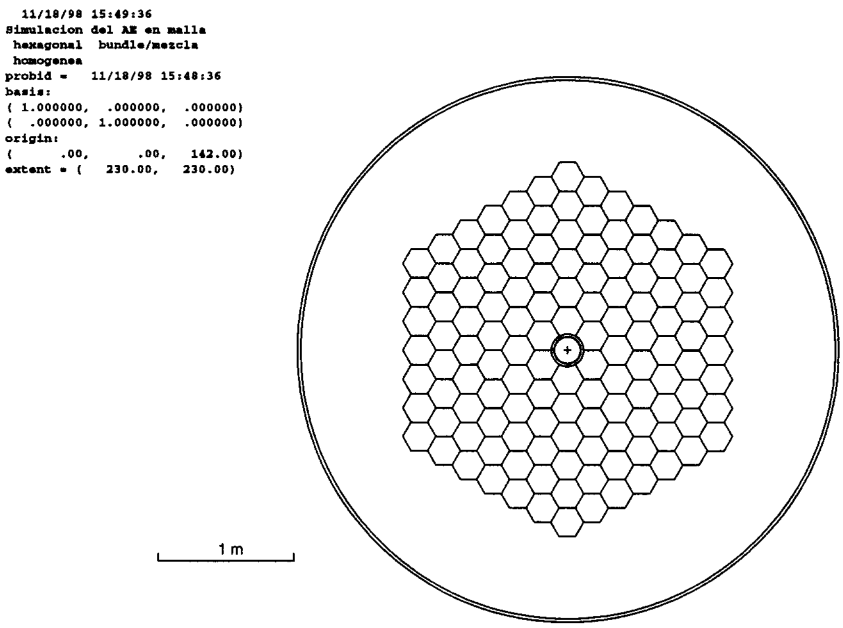
Table 2 **TRU mixtures mass fractions considered in the simulation**

Cooling down time	**10 years**	**25 years**	**40 years**
^{237}Np	5.31E-02	5.51E-02	5.78E-02
^{239}Np	0.00E+00	0.00E+00	0.00E+00
^{236}Pu	1.35E-08	3.57E-10	0.00E+00
^{238}Pu	1.71E-02	1.52E-02	1.35E-02
^{239}Pu	5.15E-01	5.16E-01	5.17E-01
^{240}Pu	2.13E-01	2.14E-01	2.15E-01
^{241}Pu	8.39E-02	4.08E-02	1.99E-02
^{242}Pu	4.96E-02	4.98E-02	4.99E-02
^{241}Am	5.65E-02	9.80E-02	1.17E-01
^{242*}Am	9.77E-05	9.10E-05	8.56E-05
^{243}Am	9.82E-03	9.83E-03	9.84E-03
^{242}Cm	2.55E-07	2.37E-07	2.07E-07
^{243}Cm	3.83E-05	2.68E-05	1.84E-05
^{244}Cm	2.04E-03	1.15E-03	6.52E-04
^{245}Cm	1.22E-04	1.22E-04	1.22E-04
^{246}Cm	1.38E-05	1.38E-05	1.38E-05
^{247}Cm	1.37E-07	1.37E-07	1.38E-07

Table 3 **Common geometry EA parameters**

EA core	
Configuration	Hexagonal
Number of fuel bundles	120
Fuel bundles	
Flat to flat	210.96 mm
Number of pin per bundle	169
Pitch between pins	15.8 mm
Total length	150 cm
Fuel pins external diameter	8.2 mm
Proton beam, spallation target	
Beam pipe material	HT9
Beam window material	W
Thickness	3 mm
External diameter	20 cm
Vessel	
Thickness	2.5 cm
Material	HT9
Lead column height (with regard to the core center)	6 m

Figure 1 **Homogenous model of the EA core fuel bundles array**



Simulation Procedure

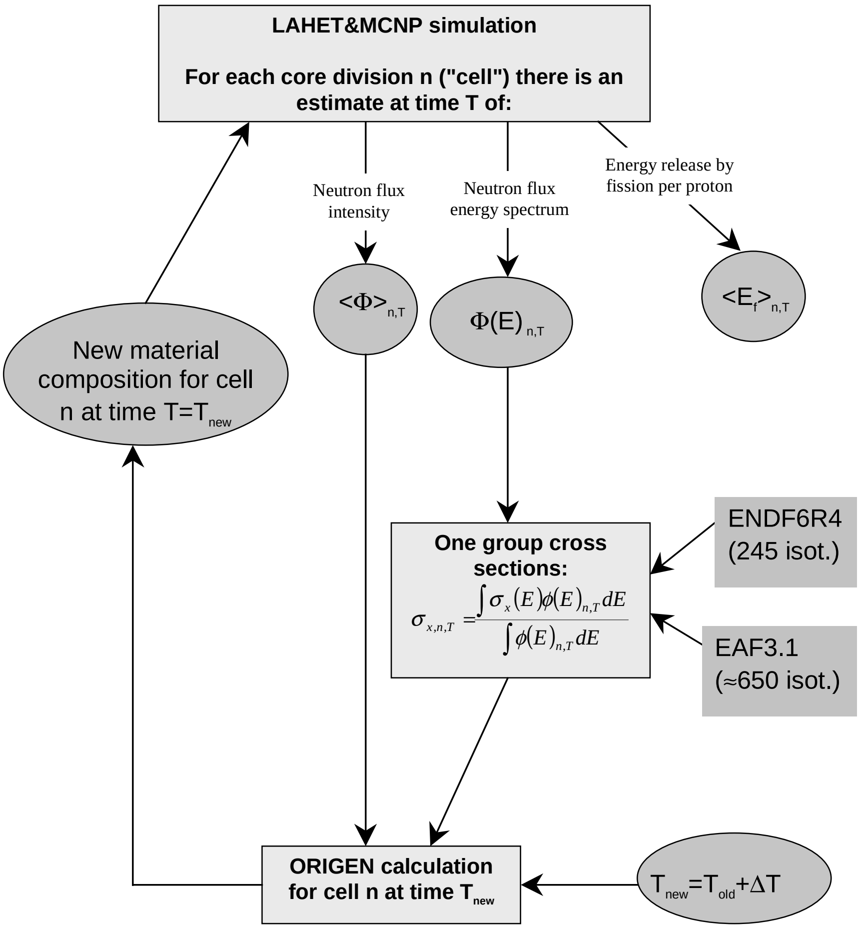
The simulations have been done using a combination of the following codes:

- NJOY94.61 for nuclear data processing [5]. The database used is the ENDF6R4 for neutron transport calculations and reaction rates calculation. During the simulations data of 245 isotopes have been used, being 196 fission fragments, and representing not less than 99% of the total mass inventory. The EAF3.1 database (about 650 isotopes) has been used for several reaction rates calculation whenever the isotopes considered are not available in the ENDF6R4 library. To handle with it, the EAF3.1 library has been converted to ENDF format.
- LAHET [6] for the simulation by Monte-Carlo of the proton beam interaction with lead. This code calculates the external neutron source.
- MCNP4B [7] for the complete neutron simulation of the spallation source produced by LAHET, including all the neutron progeny via any multiplication reaction below 20 MeV. It calculates the neutron multiplication, the energy release by fission that permits to know the beam intensity needed to work under a nominal power output, the neutron flux and specific power core distributions and the neutron flux energy spectra at every core positions.
- ORIGEN2.1 [3] for any burnup calculation.

The aim of the code combination is to perform a coupled neutronic and isotopic time evolution calculation, where the neutronics and fuel depletion simulation tools shared all the necessary data. The simulation procedure is as follows (figure 2):

- A mesh division of the EA core is performed, where each fuel element is divided into 10 axial zones. Each core division will have its own neutron flux estimate, energy release by fission and neutron flux energy spectrum estimators.
- Using LAHET and MCNP4B a complete simulation of the neutronics in steady state at considered conditions is done, obtaining the desired estimates.
- The following step is a set of burnup calculations, performing one for each core division with the previous resulted neutron fluxes. Special ad-hoc one-group integrated cross sections libraries are written in ORIGEN2.1 format and used for every burnup calculation. The one-group cross sections are obtained integrating the cross sections weighted by the spectrum obtained by MCNP4B for each core zone. The burn up time step considered is of the order of a few days.
- After all ORIGEN2.1 calculations have finished, an automatic procedure developed by FACET group translated the ORIGEN2.1 material descriptions formats to MCNP4B formats, creating a new MCNP input data file with an update of material composition for each core division.
- A new time step is performed beginning a new neutron transport simulation.

Figure 2 **Combined neutronic and isotopic combined time evolution calculation scheme**



During all the steps ORIGEN2.1 and MCNP4B shared the neutron flux data and material composition for all the core regions. Figure 2 shows a scheme of the procedure.

Definition of ADS parameters of interest for a performance evaluation

The main interest of the ADS in general and the EA in particular is their utilisation for TRUs transmutation. Therefore elimination efficiency parameters should be defined for system comparison purposes. In addition, from the neutronics point of view the most important aspects of the ADS performance are related with the accelerator requirements, once a fixed accelerator operation range is considered the maximum achievable burnup is fixed. Another EA relevant performance parameter is the energy gained by fission per unit of energy consumed in the accelerated beam.

The simplest way of defining the elimination efficiency is:

$$\eta = \frac{N_0 - N_f}{N_0} \frac{1}{Time} \; ; \; \tau = \frac{1}{\eta}$$

Where N_0 is the initial load; N_f is the remaining amount after irradiation and Time is the burnup period; τ could be defined as the characteristic elimination time. The Time units can be also burnup units as GWdt. This formula is valid whenever the isotope disappearing could be adjusted to a linear fit. This is not the usual case, normally the isotopic transmutations follows at least an exponential fit, if not some other more complex laws. Therefore, another calculation procedure for the elimination efficiency considers that the isotope time evolution is an exponential function, giving:

$$N_f = N_0 \exp(-\eta \times Time) \; ; \; \eta = -\ln\left(\frac{N_f}{N_0}\right)\frac{1}{Time} ; \; \tau = \frac{1}{\eta}$$

Again τ is the characteristic elimination time (burnup) as the time needed to reduce the isotope inventory in a factor e. The lower τ values mean the higher elimination capacity. Expressing τ in GWdt units different thermal power systems can be compared.

For isotopes that follows non-simple exponential function laws the parameters η and τ are not constant during the entire burnup considered. But in any case, for comparison purposes at equal burnup intervals, these parameters could be considered as logarithmic effective elimination time constants.

The transmutation is not possible under non-appropriate neutronic conditions. Considering that the external source is responsible of maintaining the operating regime, assuming a desired burnup period with the EA working at constant power output means that the ADS accelerator should provide the current needed at any time. The accelerator performance would be defined as the maximum and minimum beam intensity, binding its operational range. The lower ratio implies the less demanded accelerator requirements, with its foreseen impact on cost and maintenance.

Another important parameter in the ADS neutronics study is the evolution of its neutron net multiplication (M) and its multiplication constant with external source (k_s), defined both as:

$$M = \frac{n_{n,f} + n_{n,xn} + n_s}{n_s} \; ; \; k_s = 1 - \frac{1}{M}$$

Where $n_{n,f}$ is neutron fission production per proton at steady state, $n_{n,xn}$ is the neutron production by reaction such as (n,2n) or (n,3n) per proton and n_s is the spallation neutron source per proton. During all the burnup the EA should be in subcritical state with a reactivity margin that is fixed by security limitations. In Montecarlo simulations of ADS, the numbers $n_{n,f}$ and $n_{n,xn}$ can be estimated either directly (by counting the number of particles transported) or indirectly using track-length estimators.

The concept of energy gain is related with the neutron net multiplication per proton. The energy gain is the energy produced per unit of energy transported by the proton beam E_p:

$$G = \frac{\sum_{i=1}^{N}\left(n_{n,fi} / \nu_i \right) \times \left(E_{fi}[MeV] \right)}{\left(E_p [MeV] \right)} \quad ;$$

$$\sum_{i=1}^{N}\left(n_{n,fi} / \nu_i \right) = n_{n,f} / \nu \; ; \; \langle E_f \rangle = \frac{\sum_{i=1}^{N}\left(n_{n,fi} / \nu_i \right) \times \left(E_{fi}[MeV] \right)}{n_{n,f} / \nu} ;$$

$$n_{n,f} + n_{n,xn} + n_s \approx n_{n,f} + n_s \Rightarrow G \approx \frac{\langle E_f \rangle}{E_p \nu} \times (M-1) \times n_s$$

$$G = \frac{\langle E_f \rangle \times n_s}{E_p \nu} \frac{k_s}{1-k_s} = G_o \frac{k_s}{1-k_s}$$

Where G_0 is constant that depends on the nature of the fissile material loaded at EA core and the characteristics of the proton beam used; $n_{n,fi}$ are the neutrons per source proton that produce fission of isotope i; E_{fi} is the energy release by fission of isotope i and $\langle E_f \rangle$ is the energy release by fission averaged over all fissile isotopes weighted by their fission probability in the considered system. With these performance parameter definitions, both transmutation efficiency and operational valid range comparisons could be performed between different EA configurations.

Effect of the thermal power output on the neutronics and transmutation parameters

The first presented comparison is the thermal power output effect on the EA performance. The 200 MW EA (case 1 of table 1) and 800 MW EA (case 2 of table 1) have been simulated and analysed for this purpose. The differences in their initial properties are the thermal power considered (200 MW and 800 MW respectively) and the proton beam energy (380 MeV and 1 GeV). The beam current is supposed to vary adequately for constant power output working regime. The initial fuel composition is the same in both cases (see tables 1 and 2) and also they shared the same geometry characteristics.

The time evolution of the main neutronics parameters (k_s, beam current intensity, energy gain and G_0) are shown in figures 3 and 4. The required accelerator operational limits are compared in table 4, where the G_0 has been assumed as constant with time for each model. It is clear than the lowest k_s evolution point has the lowest energy gain and the highest beam intensity demand, and reciprocally the highest k_s point has the lowest beam intensity and the highest energy gain.

As can be seen in figure 3, in the 200 MW EA, after 2000 days of burnup, the maximum beam intensity required was near the starting point (150 days). This means than the burnup could be still extended far from 2000 days from the point of view of accelerator performance. This is not the case of the results presented in figure 4 for the 800 MW EA, where after 1000 days of burnup the beam intensity begins to grow almost linearly. After 1400 days of burnup, time that is assumed as the reference time for transmutation performance analysis, the necessary current reaches a maximum of 18.30 mA. On the other hand, considering the different thermal power outputs, it should be noted that 2000 days of burnup in the 200 MW EA is equivalent to 500 days in 200 MW EA. The maximums over minimum ratios for the required accelerator beam intensity are 1.92 in the 200 MW EA model and 1.99 in the case of 800 MW EA.

Table 4. **EA neutronics parameters at accelerator intensity highest demand, lowest demand and initial configurations for the two models (200 MW and 800 MW)**

	Max. Intensity	Min. Intensity	Initial
		200 MWth	
K_s	0.9669 ± 0.0026	0.9827 ± 0.0017	0.9714 ± 0.0024
I(mA)	15.7 ± 1.2	8.2 ± 0.7	13.6 ± 1.0
Energy gain	33.5 ± 2.4	64.3 ± 5.9	38.7 ± 3.0
G_0		1.12	
Time (days)	150	1000	
		800 MWth	
K_s	0.9505 ± 0.0021	0.9753 ± 0.0018	0.9706 ± 0.0014
I(mA)	18.3 ± 0.5	9.2 ± 0.4	11.1 ± 0.4
Energy gain	43.7 ± 1.2	87.8 ± 3.2	71.9 ± 2.4
G_0		2.21	
Time (days)	1400	600	

Figure 3. **Time evolution of the k_s, beam intensity, energy gain and G_0 parameter for the case 1 of table 1**

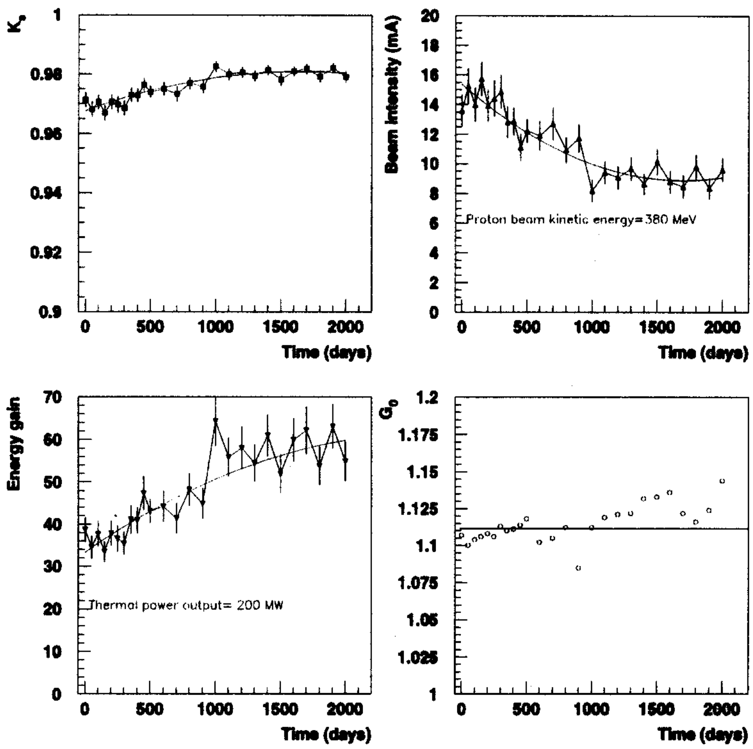


Figure 4 **Time evolution of the k_s, beam intensity, energy gain and G_0 parameter for the case 2 of table 1**

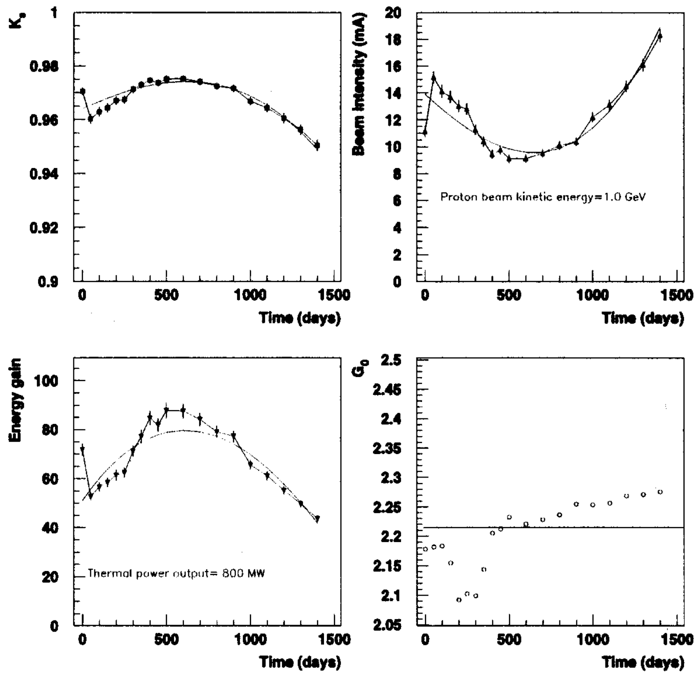


Figure 5. **Main plutonium isotopes time evolution for the EA 200 MW of case 1**

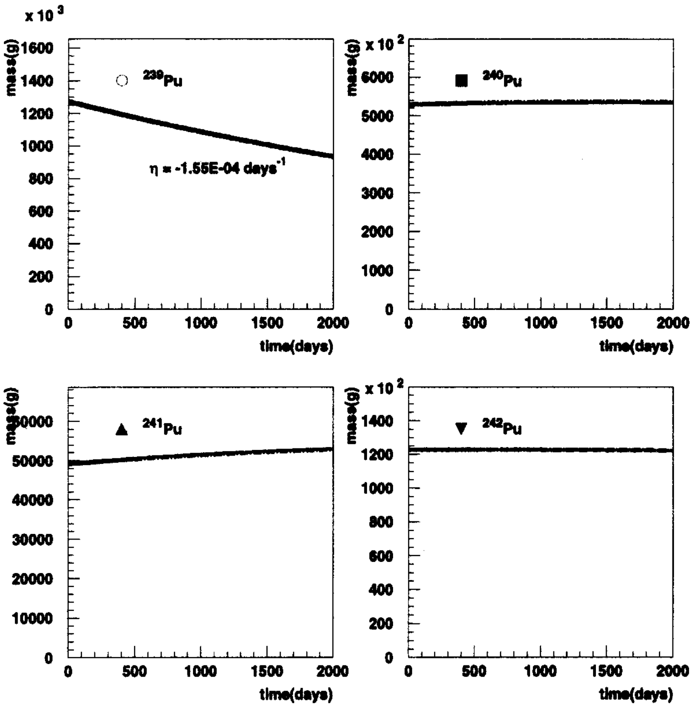
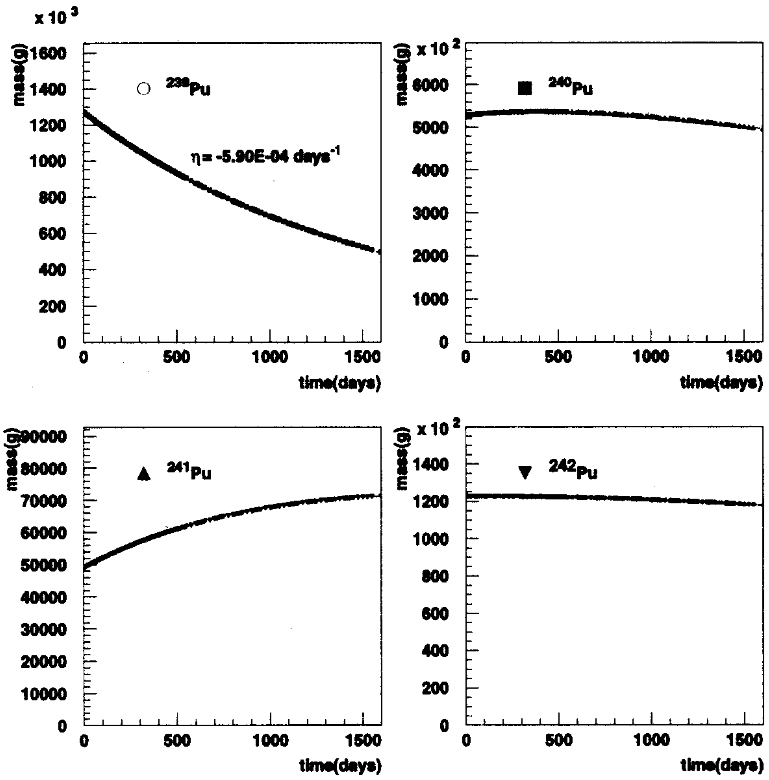


Figure 6. **Main plutonium isotopes time evolution for the EA 800 MW of model 2**



Considering the transmutation performance, the main plutonium isotopes evolutions with time appear in figure 5 for the 200 MW case and figure 6 for the 800 MW model. In both models the ^{239}Pu is eliminated at a rate that depends on achieved average fuel burnup. An exponential function adjustment can be done in both ^{239}Pu-evolution curves, giving characteristic transmutation time constants τ of 147.18 GWdt and 153.1 GWdt for 200 MW and 800 MW models respectively. These values are the average fuel burnup necessary for a ^{239}Pu depletion factor of e. In the case of the 800 MW EA, the average burnup after 1400 days has been of 127.4 GWdt, and the remaining amount of ^{239}Pu is near 0.43 times the initial load. Considering the 200 MW model, the average burnup after 1400 days is 32 GWdt, and the remaining ^{239}Pu is about 0.78 times the initial load.

In the 200 MW case, the ^{240}Pu, ^{241}Pu and ^{242}Pu isotopic concentrations grow slightly or are near by constant. On the other hand, in the 800 MW the ^{241}Pu mass grows up to 1.44 times the initial value, while both ^{240}Pu and ^{242}Pu isotopes slightly decrease.

In addition, both in the 200 MW_{th} and 800 MW_{th} EA cores, the ^{237}Np and ^{241}Am isotopes are eliminated, as figures 7 and 8 show. There is a rapid increase in the ^{242}Cm concentration, due to the ^{242}Am decay. The ^{242}Am are produced by ^{241}Am (n,γ) reactions that with some branching ratio produces ground state ^{242}Am. Another curium isotope that grows in some substantial way is the ^{244}Cm. In the model 2, after 1400 days of operation, the ^{244}Cm concentration grows more than twice and in the model 1 the increase is of 15%. In any case the total curium inventory increase is quite modest (in the 800 MW EA the final curium inventory is 1% of the remaining TRUs).

As has been said, the EA is a breeder ADS. Basically, the main breeding reaction is the ^{232}Th conversion to ^{233}U by neutron capture followed by two β decays (^{233}U and ^{233}Pa). Figures 9 and 10 summarised the ^{232}Th to ^{233}U breeding process rates for cases 1 and 2 respectively. In the 200 MW$_{th}$ EA the ^{232}Th characteristic disappearing constant is 693 GWdt, being the ^{233}U conversion rate of 1.13×10^{-3} kg per kg of ^{232}Th and GWdt. The ^{233}Pa to ^{232}Th ratio is almost constant after near 200 days, being this equilibrium ratio of about 9×10^{-4}. The results for the 800 MW$_{th}$ EA are shown in figure 10, where the ^{232}Th characteristic disappearing constant is 903.6 GWdt, and the ^{233}U conversion rate is 6.26×10^{-4} kg per kg of ^{232}Th and GWdt. The differences are because the ^{233}U production is less efficient with increasing irradiation time. The ^{233}U inventory growth is limited to a maximum given by the ^{233}U/^{232}Th asymptotic equilibrium. The ^{233}Pa to ^{232}Th ratio is almost constant after about 200 days, being this equilibrium ratio of near 3.9×10^{-3}. As expected this ^{233}Pa over ^{232}Th ratio is neutron flux intensity dependent. In addition, the ^{242}Cm over total curium inventory ratio are shown in figures 9 and 10. For case 1 this level reaches a maximum of near 0.68 after about 600 days, and in the model 2 it grows up to 0.85 after 250 days, time when the ^{242}Cm mass begins to decrease.

Table 5 summarised the transuranics evolution for the two considered cases. The elimination constant has been calculated using the equation of 3 instead of exponential adjustment to the evolution data.

Figure 7. **Time evolution of ^{237}Np, ^{241}Am, ^{242}Cm and ^{244}Cm for the EA 200 MW of case 1**

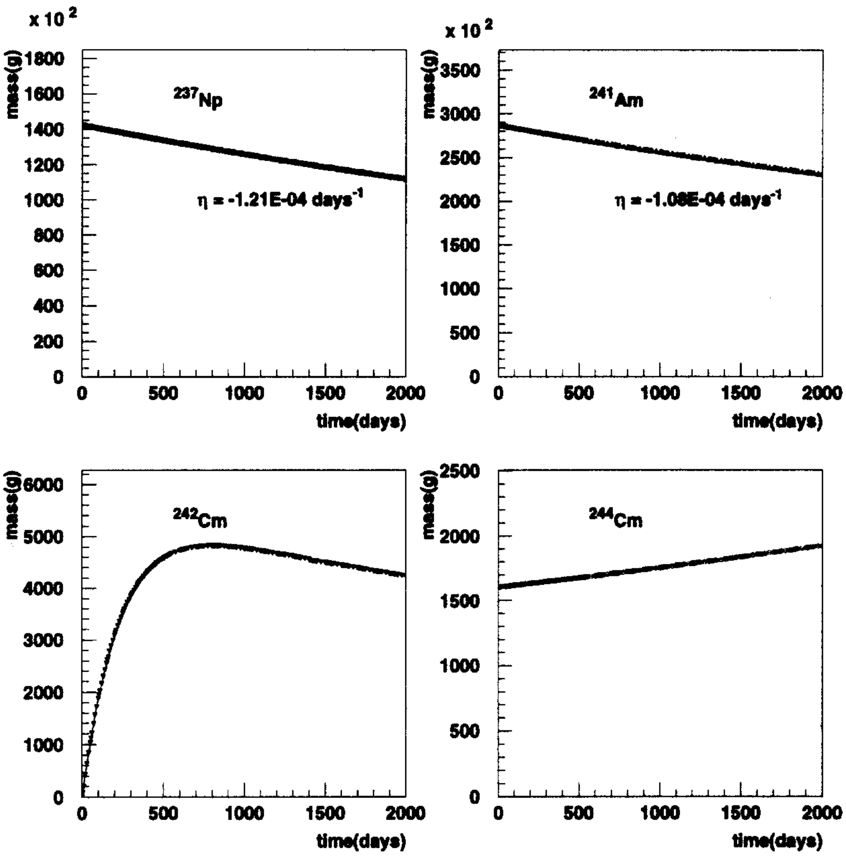


Figure 8 **Time evolution of ^{237}Np, ^{241}Am, ^{242}Cm and ^{244}Cm for the EA 800 MW of case 2**

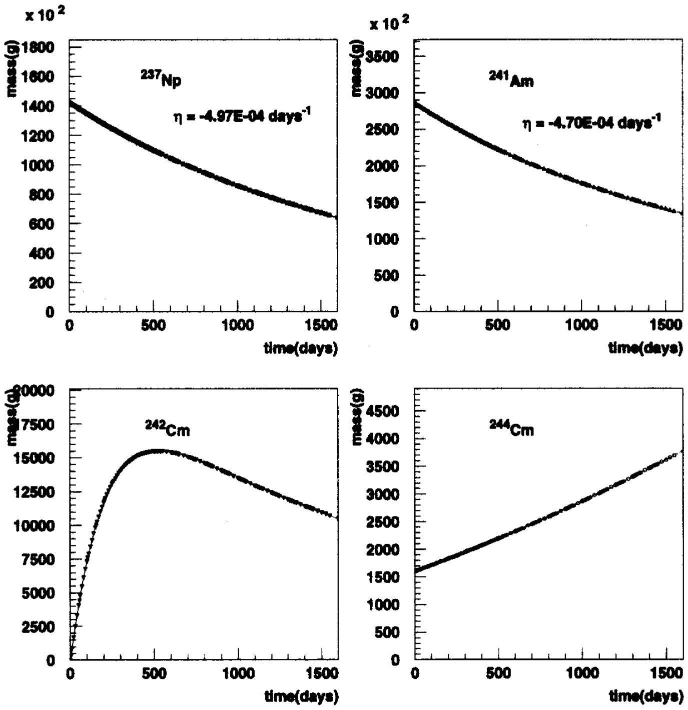


Figure 9 **Time evolution of ^{233}U, ^{233}Pa to ^{232}Th ratio, ^{233}U to ^{232}Th ratio and ^{242}Cm to total curium inventory ratio for the EA 200 MW of case 1**

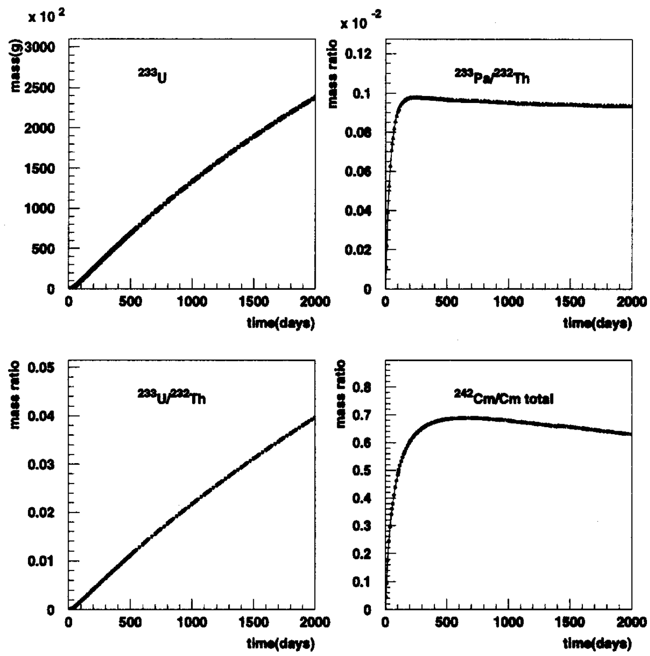


Figure 10 **Time evolution of ^{233}U, ^{233}Pa to ^{232}Th ratio, ^{233}U to ^{232}Th ratio and ^{242}Cm to total curium inventory ratio for the EA 800 MW of case 2**

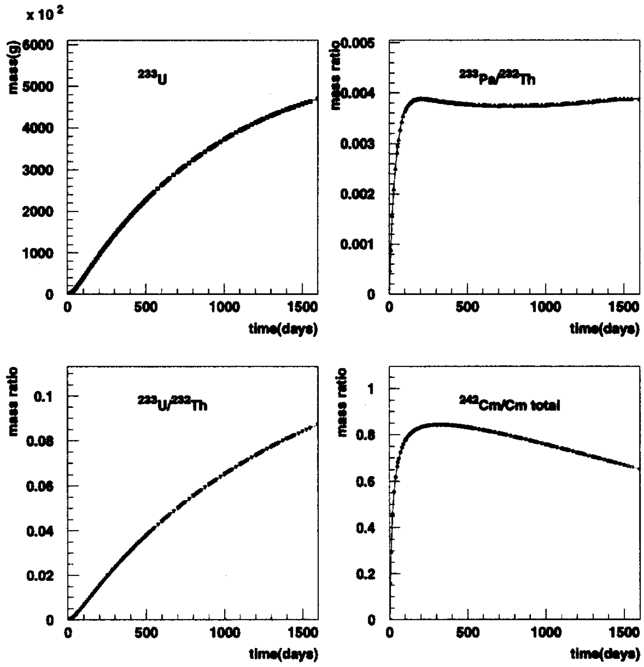


Table 5 **Time evolution parameters of the main transuranics in the 200 MWth EA model**

Isotope	Initial (g)	Final (g)	kg /kg initial/ GWdt	Elimination constant (y)	Elimination constant (GWdt)
^{237}Np	1.42E+05	1.20E+05	-4.95E-03	2.24E+01	1.86E+02
^{238}Pu	3.33E+04	6.70E+04	3.18E-02	—	—
^{239}Pu	1.27E+06	1.02E+06	-6.20E-03	1.74E+01	1.45E+02
^{240}Pu	5.29E+05	5.37E+05	4.50E-04	—	—
^{241}Pu	4.91E+04	5.20E+04	1.86E-03	—	—
^{242}Pu	1.23E+05	1.23E+05	-6.54E-05	1.84E+03	1.53E+04
^{241}Am	2.87E+05	2.46E+05	-4.52E-03	2.47E+01	2.05E+02
^{242}Am	0.00E+00	2.20E+01	—	—	—
$^{242*}Am$	2.11E+02	5.65E+03	8.10E-01	—	—
^{243}Am	2.42E+04	2.88E+04	5.98E-03	—	—
^{242}Cm	5.10E-01	4.54E+03	2.79E+02	—	—
^{243}Cm	4.52E+01	1.47E+02	7.09E-02	—	—
^{244}Cm	1.61E+03	1.82E+03	4.28E-03	—	—
^{245}Cm	3.02E+02	3.59E+02	6.03E-03	—	—
^{246}Cm	3.41E+01	4.03E-02	-3.14E-02	5.69E-01	4.73E+00
^{247}Cm	3.39E-01	1.77E-06	-3.14E-02	3.15E-01	2.62E+00

Table 5 (cont.) **Time evolution parameters of the main transuranics in the 800 MW_{th} EA model**

Isotope	Initial (g)	Final (g)	kg /kg initial/ GWdt	Elimination constant (y)	Elimination constant (GWdt)
^{237}Np	1.42E+05	7.06E+04	-1.80E-02	5.48E+00	1.82E+02
^{238}Pu	3.33E+04	1.09E+05	8.15E-02	—	—
^{239}Pu	1.27E+06	5.53E+05	-2.02E-02	4.61E+00	1.53E+02
^{240}Pu	5.29E+05	5.06E+05	-1.53E-03	8.78E+01	2.92E+03
^{241}Pu	4.91E+04	7.06E+04	1.56E-02	—	—
^{242}Pu	1.23E+05	1.19E+05	-1.13E-03	1.20E+02	3.95E+03
^{241}Am	2.87E+05	1.48E+05	-1.73E-02	5.77E+00	1.92E+02
^{242}Am	0.00E+00	5.26E+01	—	—	—
$^{242*}Am$	2.11E+02	9.02E+03	1.49E+00	—	—
^{243}Am	2.42E+04	4.17E+04	2.57E-02	—	—
^{242}Cm	5.10E-01	1.14E+04	7.99E+02	—	—
^{243}Cm	4.52E+01	9.58E+02	7.20E-01	—	—
^{244}Cm	1.61E+03	3.46E+03	4.13E-02	—	—
^{245}Cm	3.02E+02	6.65E+02	4.31E-02	—	—
^{246}Cm	3.41E+01	8.35E+01	5.18E-02	—	—
^{247}Cm	3.39E-01	3.79E+00	3.63E-01	—	—

Effect of the Delay Time between the PWR Spent Fuel Discharge and the EA Fuel Load on the Neutronics and Transmutation Parameters

As has been explained in section 2, another parameter of interest is the influence of the transuranics composition in the EA performance. For this purpose three different EA models have been considered (cases 2, 3 and 4 of table 1). The differences between the three are the TRUs load compositions that appear in table 2. All the three cases have a thermal power output of 800 MW.

From the point of view of time evolution of some neutronics parameters, figure 11 and 12 show the simulation results for cases 3 and 4 respectively. Case 2 results are presented in figure 4. As can be seen in figure 12, for the case 4 there is a progressive fall of the k_s and an increase of beam intensity demand. This means that the new fissile material breeding is not able to compensate reactivity fall because of fissile TRU burning. The beam intensity maximum to minimum ratio is as high as 3.86, considering that the irradiation period is of 1400 days.

Figure 11 indicates that for case 3 there is a reactivity recovery similar to that of case 2 (figure 4). The beam intensity maximum to minimum ratio is 1.78, less than in case 2 (1.99), with a maximum k_s of 0.9643 taking place at 600 days of burnup (also 600 days in the model 2) and a minimum at the end of the cycle (1400 days) of 0.9384.

Table 6 summarised the transmutation performance parameters for the main TRUs loaded in the EA model 3 and 4. A comparison of the characteristic elimination constants indicates that for ^{237}Np (value around 178 GWdt), ^{239}Pu (150 GWdt) and ^{240}Pu (2950 GWdt) are near the same for the three models (with differences smaller than 5%). For ^{241}Am this elimination is clearly a function of its initial concentration: while for the cases 2 and 3 the value is almost constant of 195 GWdt, for case 4 it rises to a value of 255 GWdt.

Evolution of TRU Transmutation Efficiency with the EA Burnup Cycle

Among the transmutation strategies there are two options for closing the fuel cycle:

- The one-through irradiation step: In this option the TRUs loaded fuel is burned up only once. The discharge will be stored almost directly in the secular repository.
- The closed cycle: Every spent fuel discharge is reprocessed and the remaining TRUs are reloaded in the transmutation system. This strategy allows increasing the final transmutation efficiency.

Basically the EA is designed to work in a TRUs transmutation closed cycle [2]. This strategy implies that every EA discharge will be reprocessed. In the case of the EA this reprocessing would produce four waste streams: fission and activation products, remaining ^{232}Th, produced ^{233}U and remaining TRUs. The ^{233}U is separated for other purposes; the remaining TRU are recovered altogether and mixed with the adequate amount of ^{232}Th and fresh TRUs coming from LWR spent fuel. The manufactured fuel should be able to maintain the nominal initial subcritical level once loaded in the new EA core. The fission and activation products stream will be processed for proper storage or eventually some LLFF can be also eliminated in appropriated devices (e.g. the EA core periphery [2]).

Figure 11 **Time evolution of the k_s, beam intensity, energy gain and G_0 parameter for the case 3 of table 1**

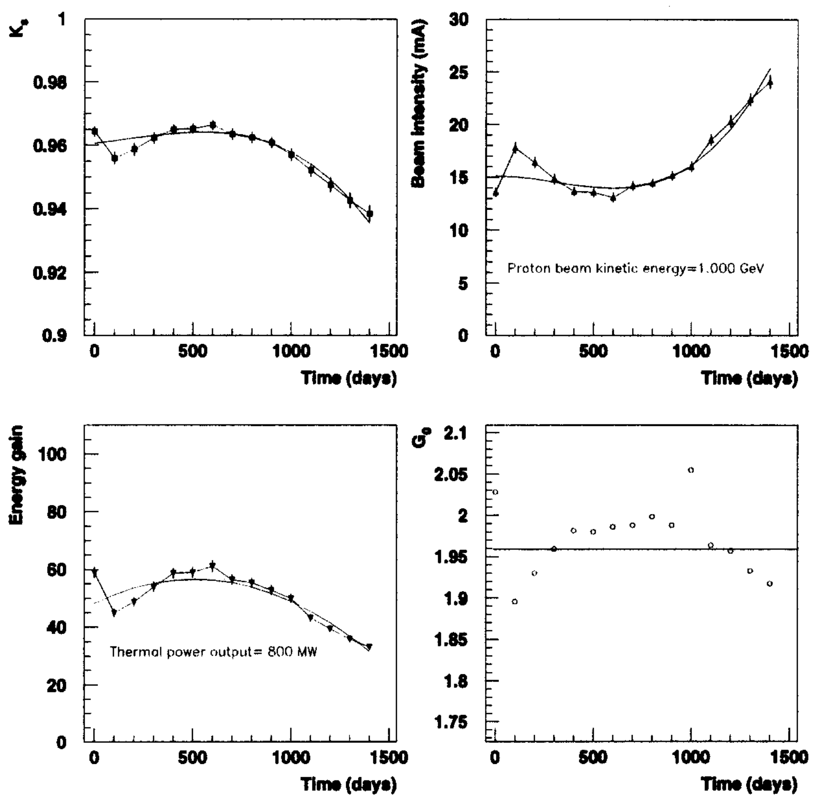


Figure 12 **Time evolution of the k_s, beam intensity, energy gain and G_0 parameter for the case 4 of table 1**

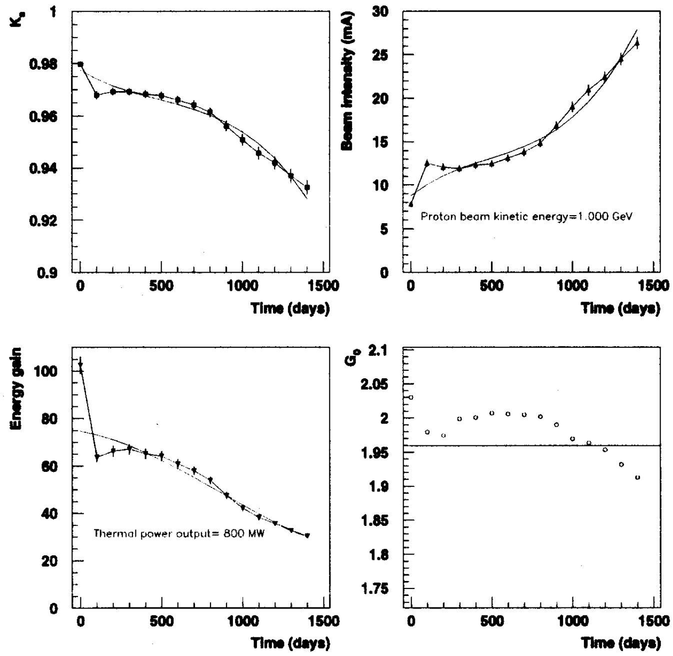


Table 6 **Time evolution parameters of the main transuranics in the 800 MW_{th}-25 years decay time EA model**

Model 3	**800 MW**		**25 years decay time**		
Isotope	**Initial (g)**	**Final (g)**	**kg /kg initial/ GWdt**	**Elimination constant (y)**	**Elimination constant (GWdt)**
^{237}Np	1.27E+05	6.21E+04	-4.01E-03	5.36E+00	1.78E+02
^{238}Pu	3.52E+04	9.62E+04	1.36E-02	—	—
^{239}Pu	1.19E+06	5.10E+05	-4.49E-03	4.53E+00	1.50E+02
^{240}Pu	4.95E+05	4.74E+05	-3.37E-04	8.75E+01	2.91E+03
^{241}Pu	9.43E+04	7.97E+04	-1.22E-03	2.28E+01	7.58E+02
^{242}Pu	1.15E+05	1.13E+05	-1.27E-04	2.35E+02	7.82E+03
^{241}Am	2.27E+05	1.19E+05	-3.73E-03	5.96E+00	1.98E+02
^{242}Am	0.00E+00	4.34E+01	—	—	—
^{242*}Am	2.10E+02	7.26E+03	2.63E-01	—	—
^{243}Am	2.27E+04	3.97E+04	5.89E-03	—	—
^{242}Cm	5.48E-01	9.38E+03	1.34E+02	—	—
^{243}Cm	6.20E+01	8.12E+02	9.49E-02	—	—
^{244}Cm	2.67E+03	3.94E+03	3.72E-03	—	—
^{245}Cm	2.83E+02	8.09E+02	1.46E-02	—	—
^{246}Cm	3.20E+01	9.52E+01	1.55E-02	—	—
^{247}Cm	3.18E-01	4.19E+00	9.55E-02	—	—

Table 6 (cont.) **Time evolution parameters of the main transuranics in the 800 MW_{th}-10 years decay time EA model**

Model 4	800 MW		10 years decay time		
Isotope	**Initial (g)**	**Final (g)**	**kg /kg initial/ GWdt**	**Elimination constant (y)**	**Elimination constant (GWdt)**
^{237}Np	1.14E+05	5.46E+04	-4.07E-03	5.22E+00	1.74E+02
^{238}Pu	3.66E+04	7.52E+04	8.23E-03	—	—
^{239}Pu	1.10E+06	4.64E+05	-4.53E-03	4.43E+00	1.48E+02
^{240}Pu	4.57E+05	4.38E+05	-3.25E-04	9.05E+01	3.02E+03
^{241}Pu	1.80E+05	9.86E+04	-3.54E-03	6.37E+00	2.12E+02
^{242}Pu	1.06E+05	1.08E+05	7.93E-05	—	—
^{241}Am	1.21E+05	7.33E+04	-3.09E-03	7.64E+00	2.55E+02
^{242}Am	0.00E+00	2.75E+01	—	—	—
^{242*}Am	2.10E+02	4.29E+03	1.52E-01	—	—
^{243}Am	2.11E+04	3.73E+04	6.01E-03	—	—
^{242}Cm	5.48E-01	5.81E+03	8.29E+01	—	—
^{243}Cm	8.21E+01	5.00E+02	3.98E-02	—	—
^{244}Cm	4.39E+03	4.68E+03	5.19E-04	—	—
^{245}Cm	2.62E+02	1.03E+03	2.30E-02	—	—
^{246}Cm	2.97E+01	1.14E+02	2.21E-02	—	—
^{247}Cm	2.94E-01	4.77E+00	1.19E-01	—	—

A simulation of this strategy applied to the EA model 2 (800 MW_{th} EA) has been performed under the considered hypothesis for six more burnup cycles. The first cycle was of 1400 days while the following six cycles was simulated for a burnup period of 1500 days. The resulting cumulative TRUs removals as a function of the number of cycle are presented in figure 13. As can be seen, there is substantial cumulative elimination of ^{239}Pu, ^{241}Am, ^{240}Pu, ^{237}Np and ^{242}Pu which grow almost linearly with the number of cycles. The ^{243}Am inventory increases from a production of 17.4 kg at the end of the first cycle up to 128.7 kg of cumulative production and the end of cycle seven.

Special behaviours come into view for the ^{238}Pu and the ^{241}Pu evolutions: both at the end of cycle number three starts to disappear, beginning with two production cycles. The ^{238}Pu is the main product of ^{237}Np conversion by neutron irradiation and the ^{241}Pu is produced by neutron capture in ^{240}Pu. These results indicate that both actinides have reached equilibrium ratios with their parents after the end of cycle three. In the case of ^{242}Cm and ^{244}Cm there is a linear increase of their cumulative production. Nevertheless, the produced masses are not comparable in scale with the transmutation rates of ^{239}Pu, ^{241}Am, ^{240}Pu and ^{237}Np.

The characteristic elimination constants for several TRUs of interest appear in figure 14. As can be seen, for ^{237}Np, ^{239}Pu and ^{241}Am these constants are nearly cycle independent, with average values of 0.19 year^{-1} for ^{237}Np, 0.21 year^{-1} for ^{239}Pu and 0.16 year^{-1} for ^{241}Am. The differences with average values are in the three cases less than 5%. Therefore, if the loaded masses at the beginning of every cycle are near the same, as it is the case, the elimination achieved for the same burnup is the same, and the cumulative elimination is a linear function of the number of cycles, as figure 13 has revealed.

Conclusions

The aim of this study is to analyse the effects of the thermal power output level and the TRUs loaded in the fuel on the EA performance. Also a first approach to define the transmutation efficiency of an EA operated in closed cycle is presented.

When comparing 200 MW and 800 MW EA systems its neutron multiplication constant can be maintained in an acceptable operation range for long burnup periods (up to 110 GWdt in average) in both cases. This is due to the ^{233}U breeding from the ^{232}Th loaded within the EA fuel. Therefore, in both cases accelerator maximum to minimum beam intensity ratios are limited to less than 2. From the point of view of TRU transmutation, the characteristic elimination constants for ^{237}Np, ^{239}Pu and ^{241}Am are near to constant in units of GWdt (~180 GWdt, ~150 GWdt and ~195 GWdt respectively). This result implies that higher thermal power generation needs less burnup periods (in days) to achieve near to similar ^{237}Np, ^{239}Pu and ^{241}Am elimination.

The use of the different proposed TRUs mixtures for this study in the EA fuel load has slight effect on the performance from the neutronics point of view (e.g. the accelerator maximum to minimum beam intensity demand). The key is once again the ^{233}U-breeding time evolution that should compensate reactivity decrease due to fissile TRUs burn up. The transmutation performance could depend on the initial mass loaded of the actinide under study, as the ^{241}Am case reveal in this study.

On the other hand, a closed fuel cycle strategy can be designed for the EA with substantial cumulative elimination of ^{239}Pu, ^{241}Am, ^{240}Pu, ^{237}Np and ^{242}Pu which grow almost linearly with the number of cycles .

Figure 13a **Cumulative elimination for several TRUs of interest as a function of the number of cycle**

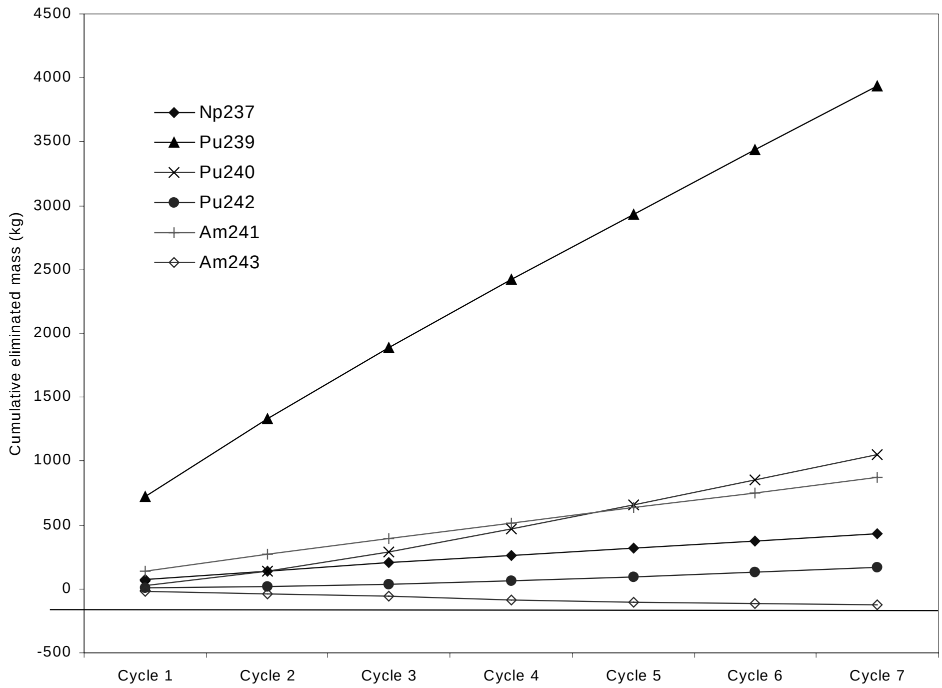


Figure 13b **Cumulative elimination for several TRUs of interest as a function of the number of cycle. (Note that there is a factor 40 reduction is the Y axis scale comparing with figure 13a)**

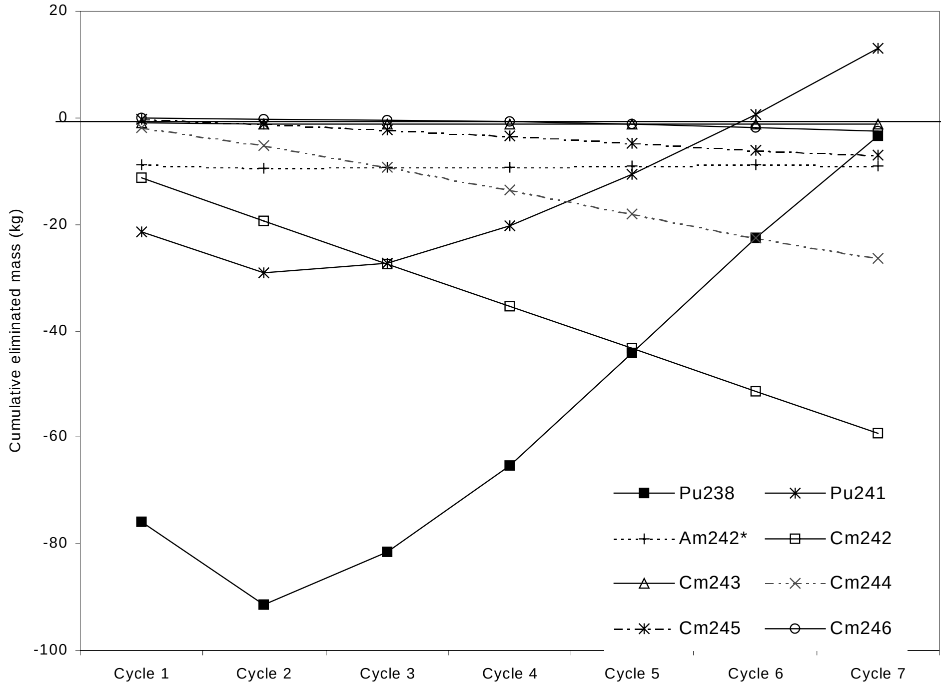
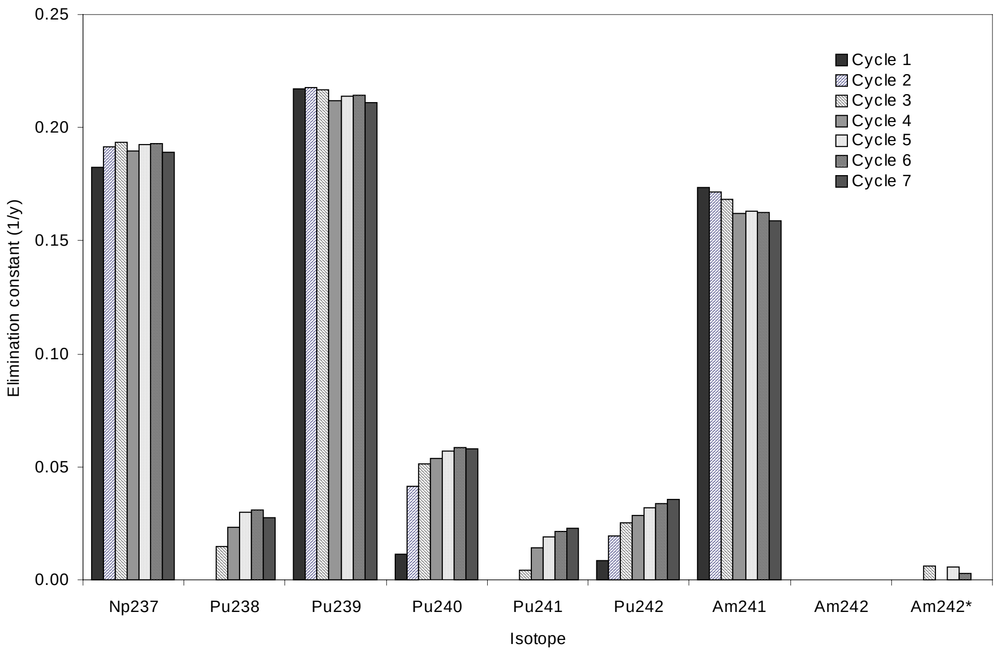


Figure 14. **Transmutation constants for the main TRUs of interest as a function of cycle number**



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ACHIEVABLE TRANSMUTATION RATES FOR TRUS AND LLFPS IN MYRRHA

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Abstract

SCK•CEN wants to fulfil a prominent role in the ADS field and is currently establishing a R&D programme to finalise the design of an ADS prototype. Therefore, the ADONIS system, initially intended for dedicated radioisotope production, was revisited to give birth to the Myrrha system. The need for ADS related R&D as well as the extension of current competencies within SCK•CEN and related institutes led to the study of a prototype ADS which would focus primarily on ADS related research, i.e. on materials and fuel research, on the utilisation of liquid metals and associated aspects, on reactor physics, and subsequently on applications such as transmutation and safety research on sub-critical systems. In this respect, the Myrrha system should become a new major research infrastructure for SCK•CEN supporting and enabling the international R&D programmes.

Currently, the study and preliminary conceptual design of the Myrrha system is being finalised and the basic engineering phase has started. This study will also define the final choice of the characteristics of the facility depending on the selected fields of application to be achieved in this machine. The applications which are considered can be grouped in three blocks; i) continuation, and later on extension towards ADS, of the ongoing R&D programmes in the field of reactor materials, fuel and reactor physics research; ii) enhancement and triggering of new R&D activities such as waste transmutation, ADS technology, liquid metal embrittlement; iii) initiation of new competencies such as medical application (proton therapy, PET production,..), neutron beam applications.

Myrrha will use an accelerator delivering a proton beam of (2 mA, 250 or 350 MeV). The spallation target is a liquid Pb-Bi volume circulating in a single closed loop. In its present design the spallation source is a windowless design. The sub-critical assembly is made of two consecutive zones of fuel rod lattices. Due to the main interest in research on the transmutation of TRUs, the first zone, consisting of MOX-type fuel rods with high Pu contents, forms a fast neutron spectrum region. In the present design, the fuel pins are inserted in a lead environment. This lead environment is either a solid block (cooled with a gas-circuit) or liquid lead, depending on the total power to be removed. The fast neutron zone is surrounded by a thermal neutron zone made of LWR UO_2 fuel rods inserted in a water moderator. This thermal zone will allow to increase the multiplication factor to perform experiments related to LWR fuel research, radioisotope production and transmutation of LLFPs.

In this paper we will report on the performances one can achieve in the present design of Myrrha concerning transmutation of TRUs such as Np, Pu, Am, and Cm in the fast neutron zone as well as of LLFPs such as ^{129}I, ^{99}Tc, ^{135}Cs, and ^{93}Zr in the thermal neutron zone.

Introduction

Nuclear energy has to cope with some topics to resolve the economical question of increasing energy demand and more specially the public acceptability requirements:

– Increasing the absolute safety of the installations.
– Managing more efficiently the nuclear waste.

In that respect, the development of a new type of nuclear installation coping with the above constraints as well as with those of technological, social and economical nature is most important for the future of sustainable energy provision. Accelerator driven systems are coping with the above constraints and can pave the way to a more environmentally safe and acceptable nuclear energy production. Fundamental and applied R&D are crucial in the development of these technologies and demand the availability of appropriate prototype installations. These prototype installations have to enable and have to deal with these R&D-issues related to accelerator driven system development.

The Myrrha project aims at investigating the design, development and realisation of a versatile neutron source based on an accelerator driven system. It focuses on the realisation of a radiation source, well-matched to both regional R&D needs and international fundamental research programmes in the field of accelerator driven systems.

The initial project leading to the current Myrrha project focused on the dedicated application of radioisotope production relying on an accelerator driven system. This project, called ADONIS, has been shifted to the Myrrha project to extend the scope of applications towards :

– material irradiation studies;
– fuel research (transient and high burn-up accumulation);
– radioisotope production;
– waste transmutation studies; and,
– ADS system prototyping from the technological point of view.

These studies lead to refocusing the scale of the system for both the sub-critical assembly and the accelerator performances to be considered.

Present Design of Myrrha

Accelerator

The accelerator considered up to now is a six sector cyclotron generating a 2 mA current at 250 or 350 MeV considered in two stages, the first stage being an injector ranging between 40 and 70 MeV. The positive ion (H^+) acceleration technology will be used for this machine. IBA is considered as the most potential partner for the design of the accelerator. The two stage accelerator option is not yet a frozen option: it is kept for potential applications with low energy protons such as radioisotope production using protons or, to a limited extent, proton therapy.

Spallation target

The spallation target is made of liquid Pb-Bi circulating in a double concentric cylindrical circuit with a dump tank at the lower end of the circuit. At the upper part of the target system a free surface is in contact with the incoming proton beam. No conventional window is foreseen between the Pb-Bi free surface and the beam in order to keep the energy losses at their minimum. The Pb-Bi is circulating bottom-top in the outer tube and going down in the central tube leading to the creation of a jet pump effect which will help in trapping the Pb and Po vapours. However, to reduce potential transport of volatile substances from the spallation source into the accelerator and related damage, a very thin He-cooled double walled barrier is placed in the beam trajectory at a certain distance from the spallation source.

The choice of a windowless design has been influenced by the following considerations :

- With 250 MeV, an incident proton delivers 7 MeV kinetic energy per spallation neutron. Almost 85% of the incident energy is lost as « evaporation » energy of the nuclei in the target. The addition of a window would only diminish the fraction of the incident energy delivered to the spallation neutrons.
- A windowless design avoids vulnerable parts in the concept, increasing its reliability.
- The low working temperature in the Pb-Bi eutectic. Indeed, if we consider an inlet temperature of 150°C, we end up with working temperatures of 250°C average temperature in the central part and 170°C in the outer part, with a circulating speed of 1.8 m/s in both sections. These low temperatures will result in very low Pb-Po evaporation rates (within the range of $2.\ 10^{-8} \sim 10^{-7}$ kg/m².s).

The thermo-hydraulics simulation of the windowless design is foreseen in a first stage to be carried out with hot water which is a good equivalent fluid for the Pb-Bi from the comparison of their dimensionless numbers (Reynolds, Prandtl, Weber)

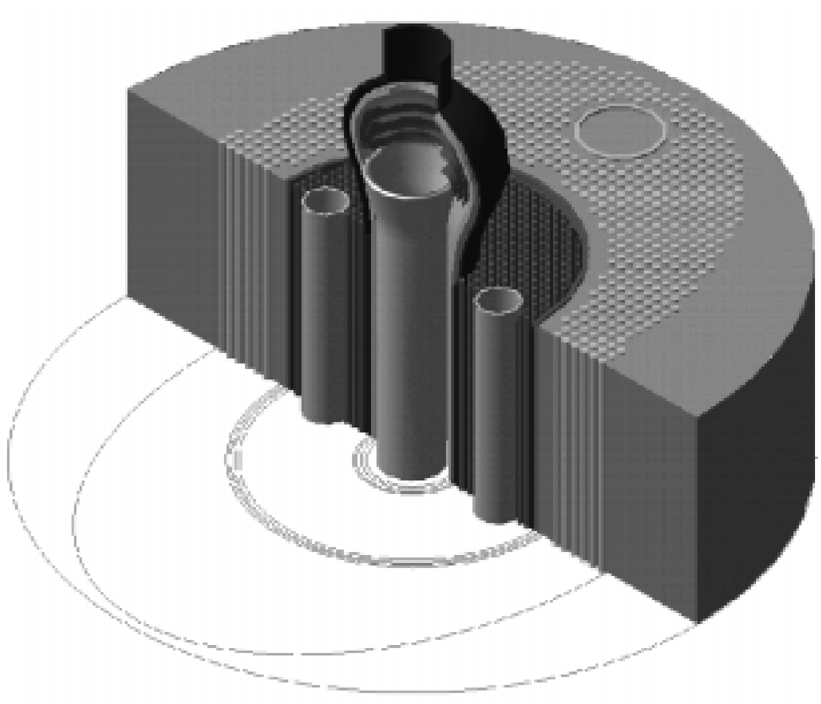
Sub-critical system

The design of the sub-critical assembly to be set up around the spallation source is application-dependent. Indeed, one should meet the neutronic performances as well as the volumes needed by the considered applications. The present design of the Myrrha sub-critical system is shown in Figure 1 below.

In that respect, to meet our goals of material studies, fuel behaviour studies, radioisotopes production, transmutation of MAs and LLFPs studies, we came to the conclusion that the sub-critical system of Myrrha should have two spectral zones ; a fast neutron spectrum zone and a thermal spectrum one. We should have irradiation channels having sufficiently large volumes for housing irradiation rigs such as the ones used in MTRs. Therefore, we should have a spallation source with axial distribution leading to a reasonable axial active length. This resulted in choosing a higher incident proton energy: 350 MeV instead of 250 MeV leading to a spallation-source axial distribution of 15 cm instead of 10 cm.

Besides these considerations, our objectives during this pre-design were to avoid the use of revolutionary technologies from the point of view of the components to be used. The targeted K_{source} value is 0.9: this value could be increased after having performed the necessary safety studies. The resulting thermal power of the sub-critical system is within 25 to 30 MW_{th}. The maximum power density estimated in the fast zone fuel pins is around 180 W/cm, whereas in the thermal zone the value does not exceed 150 W/cm.

Figure 1. **3-D view of the spallation source and the sub-critical system of Myrrha**



Fast zone description

The fast zone is made of MOX FBR-type fuel pins with a Pu content ranging between 20 and 30%, arranged in a square lattice with a 1 cm pitch. The fuel cladding is in stainless steel. The active fuel length is 50 cm. The coolant presently considered is liquid Pb or Pb-Bi or a solid Pb matrix cooled by a circulating gas. This will depend on the total amount of power to be extracted from this zone. In the present configuration, we have foreseen two irradiation channels with an inner diameter of 48 mm, where the radial form factor is nearly 1.0 and the axial form factor is 1.2 for 50 cm length.

Thermal zone description

The thermal zone is made of UO_2 PWR type fuel pins with 4% ^{235}U enrichment cladded with stainless steel. They are arranged in a typical 1.26 cm PWR square lattice. The active fuel length is 50 cm. The fuel pins are arranged presently in a light water pool. Two larger irradiation channels, with 84 mm inner diameter, are considered. Other irradiation channels can be easily accommodated as this zone is large enough.

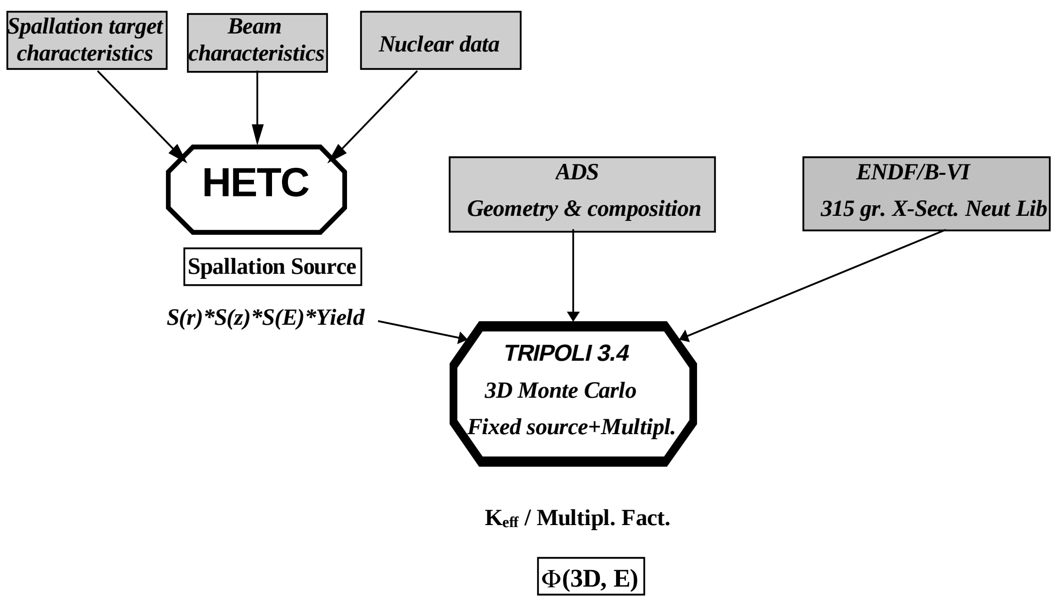
Neutronic performances of Myrrha

Based on our objectives in the project (fast and thermal flux optimisation in the sub-critical system) and our experience in the field of reactor physics, we decided at SCK•CEN to couple the HETC part of the HERMES Program System [1] high energy particle code to the S_N DORT/TORT [2], and to the Monte Carlo TRIPOLI [3] neutron transport codes. The generic calculations are performed using the DORT/TORT code whereas the TRIPOLI Monte Carlo code is used for more precise results as nearly no geometry simplifications are made thanks to the precise geometry description allowed by Monte Carlo codes. The latest resulting calculational scheme as applied in the Myrrha project is summarised in Figure 2.

The HETC code is used to compute the space and energy distribution of the primary spallation neutron source, also including all other particles involved. The high energy cascade is calculated down to 20 MeV neutrons, whereas the neutrons below this energy limit are stored as primary particles (without any interaction in the spallation medium) in the 315 multigroup energy structure of the low energy cross-section neutron library to be used with the TRIPOLI code and will be treated as a fixed neutron source in the Monte Carlo transport code.

A full detailed modelling of the sub-critical system with a heterogeneous representation of the fuel pins (fuel, cladding and coolant around) as well as of the spallation target was introduced into the TRIPOLI code. The code was run in a fixed source (in the spallation target volume) mode with possibility of having multiplication in the sub-critical system. Tallies were defined at various locations in the Myrrha system to record the fast (> 1 MeV or > 0.1 MeV) and thermal (< 0.5 eV) fluxes.

Figure 2 **Computational scheme used for the modelling of Myrrha**



The calculated values are summarised in Table 1 where we are giving also the very well known values of the BR2 materials testing reactor in different irradiation channels.

Table 1 **Comparison of the thermal and fast neutron fluxes in Myrrha and in the BR2 MTR**

Irradiation position	Myrrha Irradiation Positions: 1st Fuel FZ	Irr. Chan. FZ	last Fuel FZ	1st Fuel ThZ	Irr. Chan. ThZ	BR2 Irradiation Channels: Core Centre	Core Periphery	Reflector
Active Length (cm)	50	50	50	50	50	76.2	76.2	76.2
Diameter (cm)	1	4.8	1	1	8.4	8.4	8.4	8.4
Axial Form Factor	1.2	1.2	1.2	1.2	1.2	1.4	1.4	1.4
$\phi_{> 1\ MeV}$	9.5	2.3	5.4	7.2	1.2	2~3	0.2~0.4	0.03~0.2
$\phi_{> 0.1\ MeV}$	25.1	6.8	13.3	16.1	2.2	4~6	0.5~1	0.1~0.5
$\phi_{> 0.1} / \phi_{> 1}$	2.6	2.9	2.5	2.2	1.8	~2	~2.5	~3
$\phi_{Thermal}$	0.003	0.003	0.13	3.8	34.8	2~5	1~3	1~2

The Flux values given in Table 1 correspond to a K_{source} of 0.85, a proton current of 2 mA and a spallation yield of 2 neutrons/proton (corresponding to the calculated value for protons of 250 MeV). The flux values should be read $\times 10^{14}$ n/cm².s.The locations assessed in the Myrrha system correspond to:

- The first fuel pin in the fast zone, the closest to the spallation source.
- The irradiation channel in the fast zone.
- The last fuel pin in the fast zone.
- The first fuel pin in the thermal zone.
- The irradiation channel in the thermal zone.

Achievable transmutation rates of TRUs and LLFPs in Myrrha

The fast neutron flux positions in Myrrha are characterised by fast flux values (ϕ>1.0 MeV) estimated to be of the order of 1.0×10^{15} n/cm^2.s. The burn-out of the MAs ^{237}Np, ^{241}Am, ^{243}Am and ^{244}Cm in targets submitted to a fast neutron flux ($\phi_{>1.0\ MeV}$) of 1.0×10^{15} n/cm^2.s is plotted in Figure 3, as a function of irradiation time (the evolution curves only indicate the consumption of the nuclides concerned and do not indicate the formation of daughter products).

The thermal neutron flux positions in Myrrha, on the other hand, are characterised by thermal flux values ($v_0 \int_0^{0.5} n(E)\, dE$) estimated to range from 3.0×10^{14} to 3.0×10^{15} n/cm^2.s. The burn-out, as a function of irradiation time, of selected LLFPs (^{93}Zr, ^{99}Tc, ^{129}I and ^{135}Cs) in targets submitted to thermal flux values ($v_0 \int_0^{0.5} n(E)\, dE$) of 3.0×10^{14} and 3.0×10^{15} n/cm^2.s is plotted in Figure 4 and Figure 5, respectively (the evolution curves only indicate the consumption of the nuclides concerned and do not indicate the formation of daughter products).

Figure 3 **Evolution of selected MAs in the fast flux positions in Myrrha, with ϕ>1.0 MeV = 1.0×10^{15} n/cm^2.s**

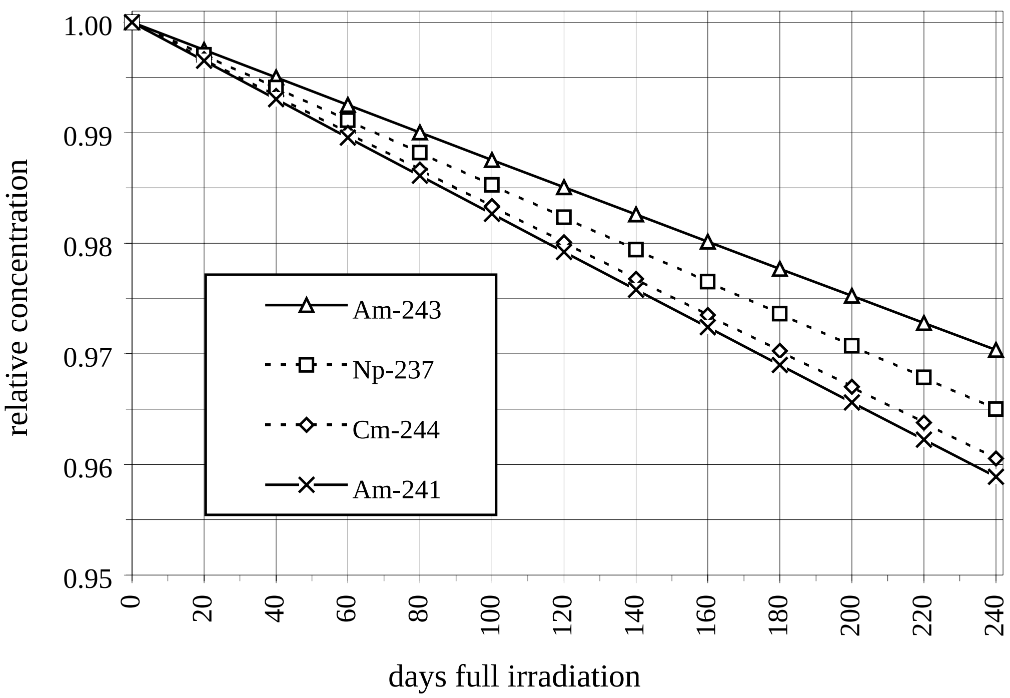


Figure 4 **Evolution of selected LLFPs in low thermal flux positions in Myrrha, with $v_0 \int_0^{0.5} n(E)\, dE = 3.0\times10^{14}$ n/cm^2.s**

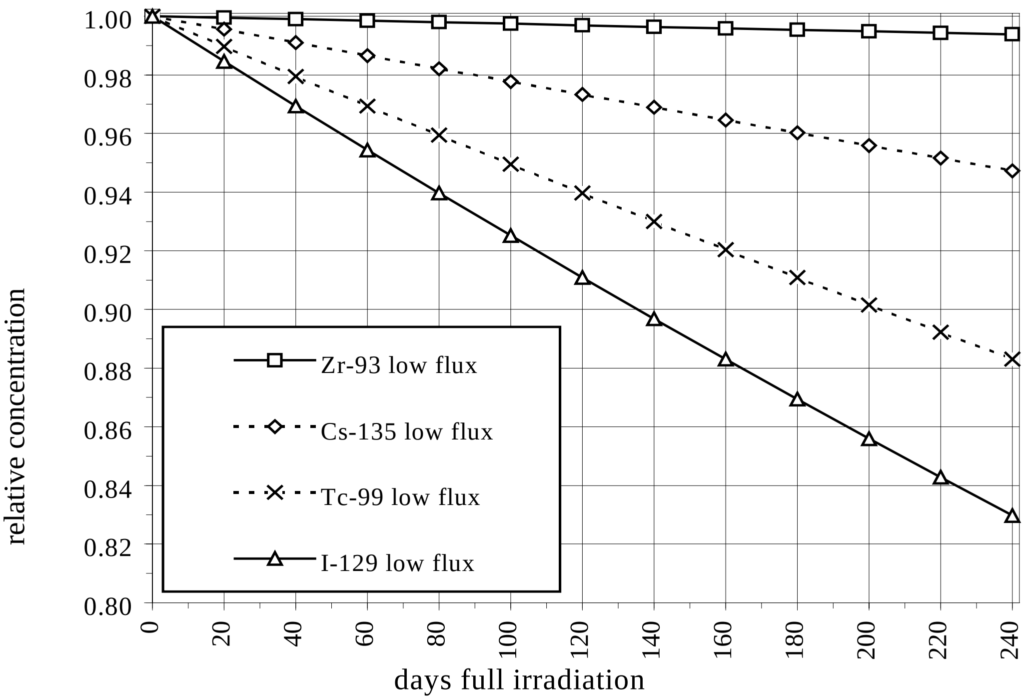
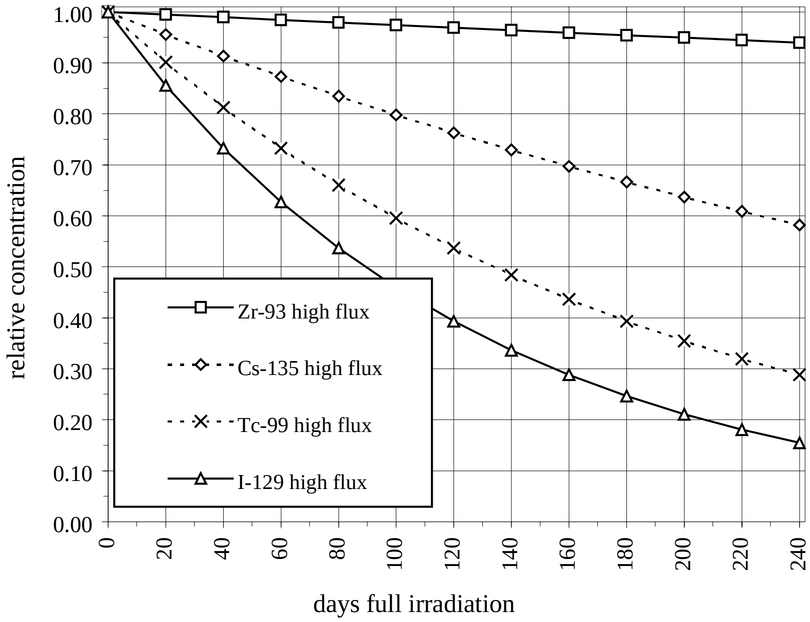


Figure 5 **Evolution of selected LLFPs in high thermal flux positions in Myrrha, with $v_0 \int_0^{0.5} n(E)\, dE = 3.0\times10^{15}$ n/cm^2.s**



Conclusions

As our purpose with Myrrha concerning the transmutation research is to provide the possibilities for performing relevant integral demonstration experiments in reasonable irradiation times, one has to look at the radiochemical detection limits and the related depletion rates needed for the different MAs and LLFPs to allow the delivery of relevant data for the validation of the evolution codes. The minimum depletion rates for the ^{99}Tc and the MAs needed by radiochemical labs for delivering 2σ precision data are : 3~5 % whereas for the remaining LLFPs considered here the depletion rate reduces to 1 %.

Given the above limits, we can conclude that transmutation experiments of MAs would be feasible in Myrrha within irradiation times ranging between 100 and 250 days. Whereas for the LLFPs the irradiation time reduces to very short periods of few days (10 to 40 days) if we consider the highest thermal flux position.

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THE CZECH NATIONAL R&D PROGRAM OF NUCLEAR INCINERATION OF PWR SPENT FUEL IN A TRANSMUTER WITH LIQUID FUEL

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Abstract

The principle drawbacks of any kind of solid nuclear fuel are listed and briefly analysed in the first part of the paper. On the basis of this analysis, the liquid fuel concept and its benefits are introduced and briefly described in the following parts of the paper allowing to develop new reactor systems for nuclear incineration of spent fuel from conventional reactors and a new clean source of energy. As one of the first realistic attempts to utilise the advantages of liquid fuel, the reactor/blanket system with molten fluoride salts in the role of fuel and coolant simultaneously, as incorporated in the accelerator-driven transmutation technology (ADTT) being proposed in [1], has been proposed for a deeper, both theoretical and experimental studies in [2]. There will be a preliminary design concept of an experimental assembly LA-0 briefly introduced in the paper which is under preparation in the Czech Republic for such a project [3].

Introduction

There are principle drawbacks of any kind of solid nuclear fuel listed and analysed in the first part of this paper. One of the primary results of the analyses performed shows that the solid fuel concept, which was to certain degree advantageous in the first periods of a nuclear reactor development and operation, has guided this branch of a utilisation of atomic nucleus energy to a death end (not having been able to solve principle problems of the corresponding fuel cycle in an acceptable way). On the basis of this, the liquid fuel concept and its benefits are introduced and briefly described in the following part of the paper.

As one of the first realistic attempts to utilise the advantages of liquid fuel, the reactor/blanket system with molten fluoride salts in the role of fuel and coolant simultaneously, as incorporated in the accelerator-driven transmutation technology (ADTT) being proposed in [1], has been studied both theoretically and experimentally. There is a preliminary design concept of an experimental assembly LA-0 briefly introduced in the following para which is under preparation in the Czech Republic for such a project.

Finally, there will be another very promising concept [4,5] of a small low power ADTT system introduced which is characterised by a high level of safety and economical efficiency. This subcritical system with liquid fuel driven by a linear electron accelerator represents an additional element - nuclear incinerator - to the nuclear power complex (based upon thermal and partly even fast critical power reactors) making the whole complex acceptable and simultaneously giving an alternative also very highly acceptable nuclear source of energy and even other products (e.g. radionuclides, etc.). In the conclusion, the overall survey of principal benefits which may be expected by introducing liquid nuclear fuel in nuclear power and research reactor systems is given and critically analysed. The other comparably important principles (e.g. the general subcriticality of reactor systems principle) are mentioned which being applied in the nearest future may form a basis for an absolutely new nuclear reactor concept and a new nuclear power era at all.

Solid nuclear fuel concept drawbacks

In spite of the fact that all what is following is well known it seems to be worth to remind it in the new circumstances of nuclear power at the end of the 20th century while starting to search new nuclear energy systems and fuel cycle options for the 21st century. Since the discovery of the reaction of atomic nucleus fission, the main goal of all efforts was to utilise it for an energy generation. As one of the most important conditions for an efficient achievement of this goal self-sustaining of fission chain reaction was demanded in an assembly containing fissionable nuclei of nuclear fuel without an external source of neutrons. If this was reached, the assembly was defined as being critical. Let us note that it was by definition (theoretically) critical on prompt neutrons released, immediately, from fission reactions only. Very early, it was observed experimentally that the assembly reaching criticality is in fact very slightly subcritical on prompt neutrons and that there is a not very strong natural source of delayed neutrons originated from radioactive decay of some of the fission products always added (which, fortunately, allowed easier control of the system).

At the early stages, the reaching of criticality was one of the most difficult tasks and all the effort and ideas had been devoted to this aim. The reason was that there were only small amounts of fissionable materials available in those times in the form of the low (0.7%) content of U235 in natural uranium. Therefore, solid phase metallic uranium with highest as possible density was used and in the form of blocks with a specifically defined size arranged in a heterogeneous lattice filled in by a solid (graphite) or liquid (heavy water) moderator with a certain pitch determined by optimal neutronic conditions. This arrangement has remained nearly exclusive one being used even in latter systems with fuels enriched by U235 content up to much higher levels than the content of natural composition of uranium. The reasons were of different nature, however, the designs have mostly started from what became already an approved conventional principle - solid fuel blocks in a heterogeneous lattice - which has been kept even in the case of pure or high enriched fuel in a fast neutron system without moderator.

One of the next consequences of the adoption of the solid fuel concept has been a type of control system which has been mostly applied for a short term control of nuclear reactors - the concept of solid absorbers - and what is more, the concept of a negative neutron source (neutron poison) at all. This, and a number of other consequences, can be traced to start all from the initial tension in neutron economy when the principle of a self-sustaining fission chain reaction and consequently the concept of a critical reactor have been adopted in nuclear reactor technology . They all begin to form a magic circle of convention in which the short term and finally even long term operational behaviour of nuclear, namely power, reactors is being imprisoned and limited in its ability to give a positive and broadly acceptable development. Let us explain this thesis in some following more see-through examples.

The adoption of the solid fuel concept leads to the principal necessity to keep the fuel blocks at a certain position in the reactor core for a shorter or longer period of time. This in-core residential time is especially long in power systems where at least a quasi-continuous exchange of fuel would be very complicated and expensive. Therefore, the following very inconvenient consequence arises: the whole time, the block of solid fuel remains at a certain position in the reactor core, there are fission fragments and by neutron capture induced radionuclides (let us call them altogether products) being accumulated in the volume of the fuel block. There are several secondary consequences caused by this fact which contribute to the above mentioned magic circle forming:

1. Reactivity margin for a short term as well as long term negative influence of the increasingly accumulated products has to be applied which has to be compensated by another artificial negative source of neutrons. It has, in principle, another consequence in the greater amount of fuel being present in the core than really necessary for the demanded power and then the more products including actinides is generated in the system.
2. The original fuel is finally so heavily poisoned by the products that it cannot keep the self-sustained fission chain reaction any more and a further operation of the reactor under original conditions is impossible. A principle change in the operation and structure of the reactor becomes to be unavoidable what means an outage and exchange of at least a part of the original fuel charge.
3. The most controversial problem what to do with spent solid fuel arises and a vicious circle has been closed or a solid fuel concept “trap” snapped.

The above briefly described solid fuel concept shows its most important and sensitive drawbacks: 1) continuous accumulation of products during the whole residential time of fuel blocks in the core, 2) following necessity to stop the operation, discharge spent fuel and store it for a necessary period of time (in order of magnitude of years until it reaches a desirably low level of radioactivity) in a specific storage, 3) the last and the most difficult drawback is the need of an optimal decision of the following destiny of spent fuel.

Up to now, the only two possible solutions were developed either to reprocess (chemically) it and to prepare next generation of solid fuel (it means with basically the same class of drawbacks) or to dispose it in a depository of a corresponding quality (which sometimes is called repository because a possible reuse of the disposed product is supposed). In the former case, mostly chemical methods and processes are applied. In the latter, a lot of branches is involved, however, nearly all of them are of a classical (non-nuclear) nature. The only nuclear process which is employed is the natural radioactive decay.

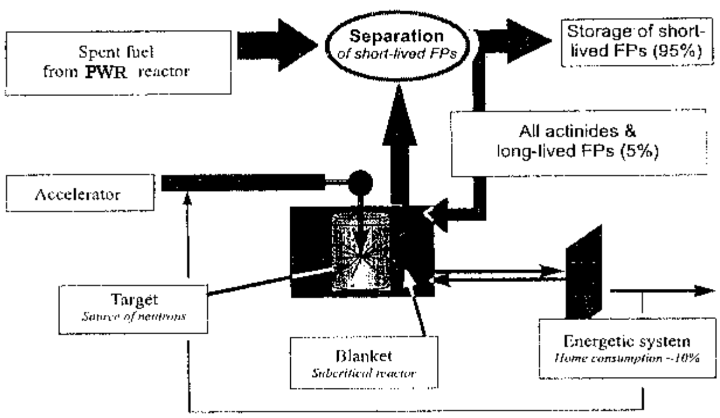
This fact contains one very controversial principle or better say a violence of a basic principle which can be described as follows: The energy generation in nuclear reactors utilises enforced nuclear process which are simultaneously producing products or nuclear waste (including secondary raw materials e.g. actinides). The treatment of the products needs to apply an adequate technology in an adequate scale. This principle has not been applied and fulfilled in those so far developed and designed systems for spent solid fuel management. There is an adequate technology which only one can utilise nuclear processes and which can transfer the high level and long-lived radionuclides towards short-lived or even stable nuclides-- the transmutation technology performed in a suitable nuclear reactor device and combined with an at least quasi-continuous separation of certain components of its core or reprocessing of the reactor fuel as to avoid the consequent induction of radioactivity by neutron irradiation of stable and short-lived nuclides. One of the principle concepts allowing to reach such a technology in an industrial scale is the concept of liquid nuclear fuel.

Liquid fuel concept for neutron source-driven transmutation technology

Molten fluoride salt fuel for neutron source-driven transmutation technology

The concept of a neutron source - driven subcritical blanket for a nuclear incineration of nuclear waste is well known for a several recent years [1] (see Figure 1). Let us recall at least very briefly the main features of the last developed version of this concept and let us show a part of a proposed research program to approve its ability for an efficient realisation in the industrial scale.

Figure 1. **Principle scheme of a transmuter system**



The fuel material is in the form of the fluoride salt AcF_4 dissolved in a molten salt carrier whose composition is a mixture of ^7LiF and $^9BeF_2$. The carrier's melting point and operating temperature are about 500°C and 650°C, respectively. The molten salt flows over either the outside of a close-packed set of cylindrical high-purity graphite blocks or inside cylindrical channels coaxially situated in e.g. hexagonal graphite blocks - Figures. 2 and 3.

Figure 2. **Single module of transmuter blanket**

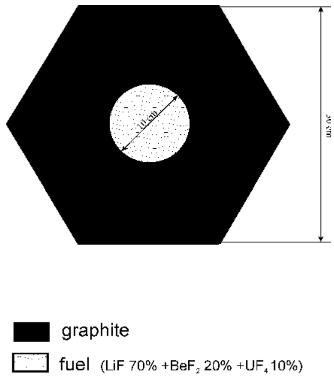
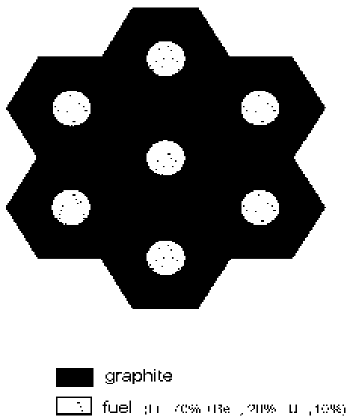


Figure 3. **The 7-module core of transmuter blanket**



There has been an experimental research system designed by the author preconceptually in [3] which should be developed and realised in the Nuclear Research Institute Rez plc in the Czech Republic. The final purpose of the system would be an experimental testing of a given type of transmuter reactor/blanket core neutronics and possibly also other physical and technological characteristics and properties including time behaviour. For the very first stage, the following scheme can be applied which will allow to reach the first results very cheaply and relatively soon. There can be an elementary, however, a sufficiently representative sample of the investigated reactor blanket lattice inserted into an existing experimental reactor core serving like a driver and the basic set of its characteristics can be experimentally measured and verified. The suitable experimental reactor can be e.g. the LR-0 experimental reactor (a full-scale VVER type core modelling zero power reactor operated in Nuclear Research Institute Rez) or VR-1 (a training reactor operated at the Faculty of Nuclear Science and Physical Engineering of the Czech Technical University in Prague) which have been successfully operated for core analyses of thermal reactors since 1982 and 1990, respectively.

Low power ADTT system

The molten salt reactors (MSRs) with the continuous control of nuclide composition almost do not require an initial reactivity margin. In such reactors, subcriticality may be reduced up to the minimum value ß where ß is the effective delayed neutron fraction. However, with such a small subcriticality and in view of available uncertainties in nuclear data and nuclide concentrations, the difference between subcritical and critical MSR in a great extent disappears: in both cases the nuclear safety is ensured by the large negative temperature reactivity effect. The deeper subcriticality is of course substantiated by the fact that under such conditions we exclude the necessity to control a reactor - burner in a dynamic mode, that is a bit difficult and poorly known.

In this case, the e.g. accelerator-driven positive source performs only one of the usual functions - the function of a reactor control system without inertia, an alternative to, up to now usually used as reactor control organs, negative sources like e.g. absorbers or decreasing of the dimensions of the system, etc. The high level parameter proton accelerator with its all disadvantages (like e.g. the length ~ 1 km, the investments ~ U$ 1 billion, etc.) having been applied e.g. in the Los Alamos concept is not necessary more in the system and a low level parameter accelerator can be employed.

Neutron sources for transmuters based on low parameter accelerators

In various concepts of the accelerator driven transmutation technologies (ADTT) the distinct effort is devoted to an employment of external neutron source other than spallation reactions initiated on (mostly future proposed) high energy proton linacs (>1 GeV, 100 mA, 10^{19} n/s). The obvious reason is that the lengths (1 km) and expected cost (above 1 B U$) of these facilities, which are an inherent and insuperable weak point of high energy proton linacs [4], could make unreliable the wide application of the ADTT. In the "subcritical enhanced safety molten salt reactor concept" the effective way of reducing the external neutron source power (below 3.5×10^{16} n/s) is accomplished by the cascade neutron multiplication in the system of coupled reactors with suppressed feedback between them. For such a "burner" reactor scheme the possibility of replacing proton linac with 100 MeV electron linac (of substantially lower length and cost than proton linac) has been argued [4]. In the similar scheme an employment of the isochronous AVF cyclotron-based neutron source was also considered [5].

Although external beams in the mA range (10^{14} n/s) have been already demonstrated for conventional AVF cyclotrons, the cryogenic technology for compact cyclotrons and also a wide class of commercially available, low-cost (below 3 M U$) cyclotrons of a type CYCLONE seem to be well methodical and technical basis for further beam-intensity improvements. Nevertheless, for a significant increase of beam intensity and neutron source strength the drift tube of Alvarez linear accelerators with the RFQ (radio-frequency quadruple) injector are more promising. The most developed proposal for 3×10^{16} n/s source strength projected originally for fusion material irradiation tests (FMIT) is based on 35 MeV,100 mA deuteron linac [6]. A fast flowing lithium jet is considered to be the best target material [7] for megawatt powers of proton and deuteron beams with medium energy. The medium energy (E < 100 MeV) proton and deuteron induced reactions on thick Li target produce a forward-directed fast-neutron fields (the fluency averaged energy E_n of about 15 MeV is to be compared with E_n ~ 3 MeV which corresponds to the spectral yield from a spallation reactions). Therefore, it seems appropriate to perform any computational and experimental study of target - blanket systems employing primary neutron sources, which in general have energy spectra with suppressed contribution of low energy neutrons.

In the NPI Rez, the d(18 MeV)+Be neutron generator, originally developed for the military directed research is now being upgraded with the main task to take advantage of H^- and D^- negative ions recently implemented on the NPI isochronous cyclotron U-120M. Conversion of the cyclotron into H^-, D^- machine enabled to utilise a high efficient extraction by means of the stripping, which resulted in substantially increased extracted beam currents of positive p^+ and d^+ ions. Nowadays, up to 20 μA currents of the 15 - 30 MeV protons are routinely extracted to various types of targets for radionuclides production. The purpose of employing this beams for fast neutron production is to perform a broad range of experiments closely related to the activation analysis and ADTT program as well.

For ADTT empirical research in NPI, an experimental study of the spectral and yield characteristics of various neutron produced reactions between light nuclei is now under way, the main tasks of which are as follows: a) to verify yield calculations from cross-section data, b) to determine an empirical shape of spectral yield of neutrons from deuteron break-up processes so as it can not be predicted reliably from simple phase-space calculations, and c) to determine the contribution of the target- v station arrangement to the background part of produced neutron fields. Knowledge of these characteristics is needed to evaluate the target and beam options for the best simulation of ADTT external neutron source mentioned above. The results of first experiments were found to be in a good agreement with calculations based on updated cross-section database (EXFOR). The preliminary calculations show that for the present neutron facility the neutron source strength up to 6×10^{12} n/s.sr and fluency averaged neutron energy E_n =15 MeV could be achieved from thick deuterium target irradiated by 30 MeV protons.

Blanket concept of at Transmuter for PWR spent fuel incineration

There has been a convenient blanket concept for an efficient nuclear incineration of PWR spent fuel developed as a combination of those two ideas described in two paragraphs above. The concept is illustrated by the Figure 4, where two zones are indicated, one under-moderated and thus better equipped for actinides burning and the second well-moderated (fuel channel in a graphite block) and thus more convenient for fission products incineration.

Experience in fluorine technology and application of fluoride liquid fuel in ADTT

There has been a technological process worked-out for the separation of uranium and plutonium from the spent fuel by a dry (fluorine) method in the NRI, Rez, in a close collaboration with the Kurchatov Institute, Moscow, during the 80s. The whole process was upgraded to a pilot plant scale with a capacity of 1-3kg of processed fuel/hour. There was a part of the technological equipment built and verified at an inactive scale at the NRI Rez. The whole technological line called Fregat (Figure 5) was then realised in the Institute of Atomic Reactors at Dimitrovgrad, Russia, and all processes proved by reprocessing of spent fuel from the fast reactor BOR 60. All equipment including fittings, measuring instruments and accessories have been built in the former Czechoslovakia, the plutonium part of the pilot plant has been built in the former USSR. A certain experience has also been obtained on the uranium isotopes separation by ultra-centrifugation and electrochemical processes.

Figure 4. **The blanket concept of a transmuter**

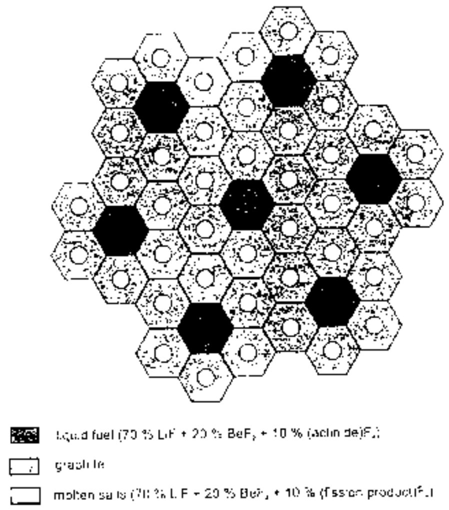
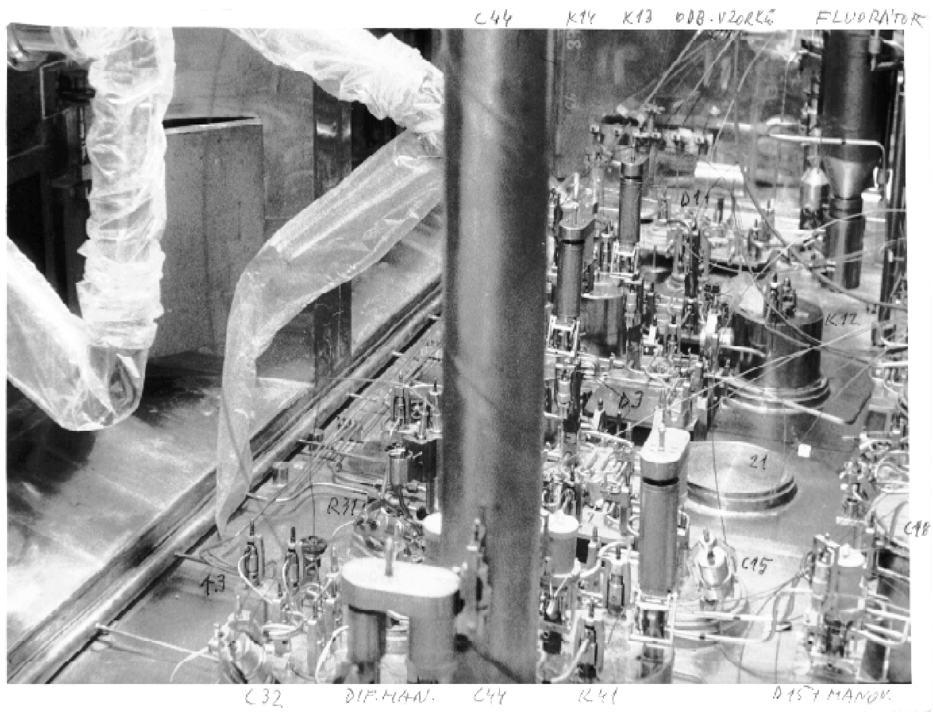


Figure 5. **The fluorine technology reprocessing line Fregat**



Technological development and testing

The experience gained in the course of the research is going to be applied in developing fluoride based liquid fuel as well as fluoride chemistry separation processes for the use in the Accelerator Driven Transmutation Technology (ADTT)

Beside this, the project “Experimental Molten Salt Loop for ADTT program” is carried out in the Škoda Works, Nuclear Machinery Ltd. in Pilsen, Czech Republic. In the frame of this project technological loop for studying of molten fluoride salts characteristics was designed and fabricated and then installed in the NRI Rez which will start up an experimental program till the end of 1998.

Conclusions

The analyses of spent fuel management from PWRs as well as all other nuclear reactors employing solid fuel concept have showed the principal drawbacks of that concept causing a series of consequences leading to a crucial issue of a nowadays nuclear power – spent solid fuel with accumulated actinides and long-lived fission products (without regard whether the open nuclear fuel cycle or a multiple reprocessing is applied). The necessity of an employment of nuclear processes and an adequate nuclear technology (nuclear incineration) in an efficient solution of that problem is definitely evident. The national R&D programme which was very briefly introduced in this paper is closely connected with the world - wide effort of research teams in leading countries of nuclear power like e.g. France, USA, Russia and Japan and forms a contribution to the common effort in the solution of that global issue. The overall co-operation in this field gives a real chance to make nuclear power acceptable as a clean source of energy for the 21st century.

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THE EFFECTIVENESS OF FAST AND THERMAL SPECTRUM FOR TRU INCINERATION IN SUBCRITICAL SYSTEM

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Abstract

An investigation to compare the relative effectiveness of fast neutron with thermal neutron in a subcritical system has been performed. Accelerator driven thermal and fast neutron systems are modelled on a CANDU reactor and a typical LMR with Pb-Bi coolant, respectively. TRU mixed with Thorium was selected as a fuel in both systems. The ratio of TRU to Thorium is adjusted to make the system subcriticality about 0.97 and the system output power is set to be 1000 MW_{th}. The thermal system was found to have unacceptable beam fluctuation and power peaking variations. The characteristics of thermal neutron are believed not to allow the employment of solid fuel concept in a thermal neutron subcritical system. In addition, the sensitiveness of thermal neutron to the concentration of TRU and fission products is believed to inevitably require on-line refuelling for reducing the beam power fluctuation. From the overall comparison, a fast neutron is concluded to be much better for the operation of the subcritical system.

Introduction

Korea Atomic Energy Research Institute (KAERI) is performing the project to develop an accelerator driven transmutation system, «HYPER (Hybrid Power Extraction Reactor)». As the first step to decide the neutronic characteristics of the HYPER system, a neutron energy spectrum study was performed. Many studies already have been done to decide which neutron spectrum is better for the transmutation. [1,2] However, as most of them were conducted on a theoretical basis, a more realistic investigation has been performed in this study. Two different types of an accelerator driven systems were developed. The thermal neutron system was developed using CANDU design values and the fast neutron system was constructed using typical LMR design data. Pb-Bi was adopted as a coolant for the fast system instead of sodium.[3] TRU mixed with thorium was used as fuel for both systems.

An optimum neutron energy spectrum would be something that minimises or maximises the following objective function,

$$F(\chi) = f(w_a a(\chi), w_b b(\chi), w_c c(\chi), \ldots\ldots\ldots) \qquad (1)$$

where,

- χ : neutron energy spectrum
- w : weighting factor for the parameter
- a, b, c, .. : system parameters.

In general, the system parameters to be considered for the determination of the neutron energy spectrum are; 1) transmutation capability, 2) system safety (reactivity coefficient, power shape control), 3) neutron economy, 4) TRU inventory, 5) total heavy metal inventory to be processed, 6) a required accelerator beam power, 7) toxicity variation, etc. The neutron energy spectrum effects on the TRU incineration were analysed based on individual parameters rather than the system performance index expressed in Eq. (1).

System model description

The basic core geometrical specifications for the thermal and fast systems were derived from a CANDU reactor [4] and a typical LMR [5], respectively. The proton energy was assumed to be 1.0 GeV and the beam powers were adjusted to produce 1 000 MWth system power. The composition of TRU was that of spent fuel being depleted up to 33 000 MWD/MTU and having 10 years cooling time. Table 1 shows the weight fraction of each nuclide in TRU.

Table 1.**Nuclide fraction in TRU**

Nuclide	W. Fraction
^{237}Np	0.046
^{238}Pu	0.014
^{239}Pu	0.521
^{240}Pu	0.237
^{241}Pu	0.077
^{242}Pu	0.045
^{241}Am	0.050
^{243}Am	0.008
^{244}Cm	0.002

Thermal Neutron Subcritical System

As it is in the CANDU system, the oxide form was employed as a fuel type for the subcritical thermal neutron system. Thorium was mixed with TRU for the fabrication of the fuel rod (mechanical strength) and for the minimisation of the reactivity swing as the TRU burns up. The geometrical specifications of a fuel rod and unit assembly in radial direction are described in Figure 1, 2, respectively. Total 37 Fuel rods are positioned on concentric circles of which the diameters are 2.9769 cm, 5.7506 cm, and 8.6614 cm.

Figure 1. **Fuel rod for thermal system**

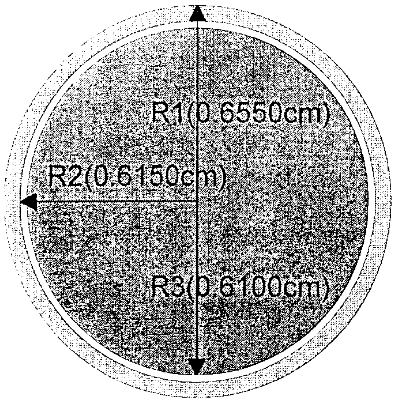
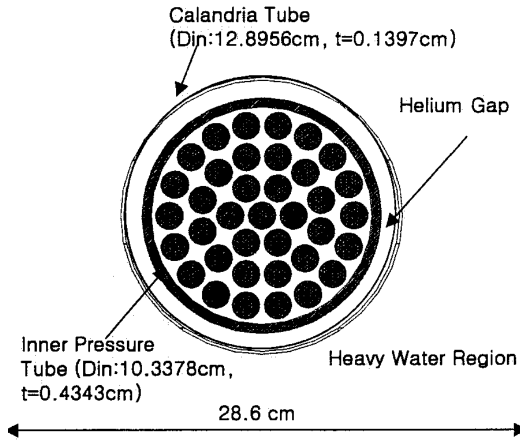


Figure 2. **Unit assembly for thermal system**



The core size was determined from the following conditions and assumptions;

1. Average linear power density of the fuel rod is 24kW/m.
2. Total core shape should be a type of square cylinder.
3. Spallation target region should be placed in the centre with about 1 m diameter.

The number of assemblies required was found to be 244 and the active core height was assumed to be 5 m. The composition of nuclides in the fuel meat was adjusted to make system subcriticality ~0.97 in eigenmode calculation at BOL condition. The fuel composition was determined to be $(Th(98.4\%)\text{-}TRU(1.16\%))O_2$. The major design parameters are described in Table 2.

Fast neutron subcritical system

The basic design parameters for a fuel rod were obtained from the design values of a typical liquid metal reactor. Fuel rods are arrayed in a triangular shape. Figure 3 shows the array of fuel rods and their geometrical specifications. HT-9 was selected for the cladding and the fuel form was determined to be a solid metal. The assembly consists of 331 fuel rods and its specifications are in Figure 4.

Lead bismuth (44.5Pb-55.5Bi) was employed to remove heat from the system. The reason for the selection of Pb-Bi as a coolant is that Pb-Bi can be used also for the spallation target. The spallation region was placed in the core centre. The reflector assemblies with the size of the fuel assembly and the Pb-Bi filled were loaded around the periphery of the active core region. The shield assemblies were placed in the outer most region of the core.

Figure 3. **Rod array for fast system**

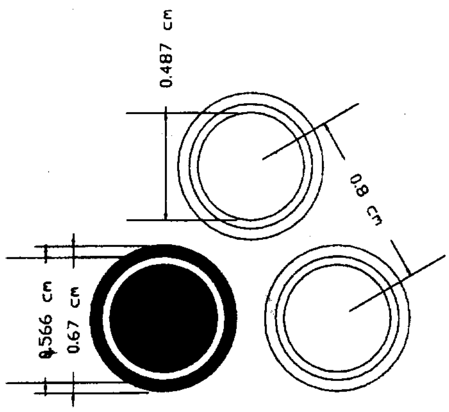
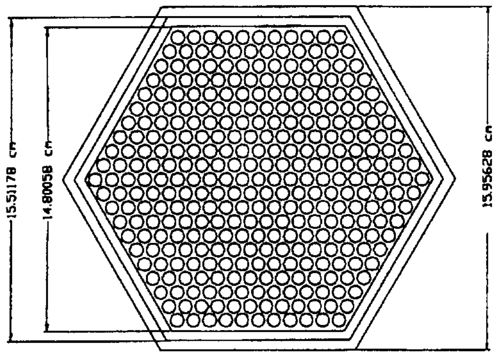


Figure 4. **Assembly for fast system**



For improvements in core safety, most of the fast neutron systems have a core height of 1~1.5 m. In this study, core height was assumed to be 1.0m. Core average linear power density was 16 kW/m which is a very common value in metal fuel.[4] Based on these limitations and using the condition of total core output, the core size was determined. The system was found to have 180 fuel assemblies. Fuel chemical composition was xTh-yTRU-0.1Zr. The sum of x and y was set to be 0.9 and they were adjusted to make the system subcriticality 0.97 in eigenmode calculations at BOL condition. From the calculations, x and y were determined to be 0.72 and 0.18, respectively. Table 2 shows the design parameters of the fast neutron system.

Table 2. **Design parameters of fast and thermal systems**

System Parameter	Fast System	Thermal System
Fuel Rod Design Parameter		
– Fuel Type	Th(0.72-TRU(0.18)-Zr(0.1)	$(Th(0.984)-TRU(0.016))O_2$
– Cladding Material	HT-9	Zr
– Fuel Meat Diameter (cm)	.4887	1.22
– Clad Outer Dimater (cm)	0.67	1.31
– Cladding Thickness	0.052	0.045
Fuel Assembly		
– Array Type	Triangular	Concentric
– Pitch-to-Diameter Ration	1.194	–
– Lattice Pitch (cm)	15.9563	28.6
– No. of Fuel Rods	331	37
Core		
– Power (MWth)	1 000	1 000
– Subcriticality (Eigenmode)	0.97	0.97
– Coolant	Pb-Bi	Heavy Water
– No. of Assembly	180	244
– Active Height (cm)	1.0	5
– Effective Radius (cm)	151.58	252

Target system

The targets for both of the thermal and fast systems were assumed to be Pb-Bi and have cylindrical shapes with the height of 50 cm and the radius of 15 cm. A proton with the energy of 1 GeV was found to produce 26.1 neutrons when it has spallation reactions with Pb-Bi.

Calculational results and discussion

The MONO (Monte-carlo Origen coupling) system was developed for HYPER system analysis at KAERI. The basic logic flow of MONO is very similar to that of other Monte Carlo depletion codes. The thermal and fast systems were loaded uniformly with the fuel assemblies described in section 2.

Figure 5 shows the variation of system multiplication factors as TRU burns up. Both spectrum systems were adjusted to have eigenvalue ~0.97 at zero burnup. However, the multiplication factor of the fast neutron system is larger from the beginning. The build-up of fission products explains the sharp drop of multiplication at the first burnup step. Because of a large absorption cross-section of fission products in thermal energy, the fluctuation of multiplication factors in the thermal system is much more severe than in the fast system. Figure 6 shows the variation of beam power to keep the system power 1 000 MWth. The fast system is believed to have higher multiplication than the thermal system for the same K_{eff} (eigenvalue) condition. In addition, the fluctuation of beam current required to keep the system power constant is supposed to be unacceptable for the thermal system.

Figure 5. **Multiplication factor vs burnup**

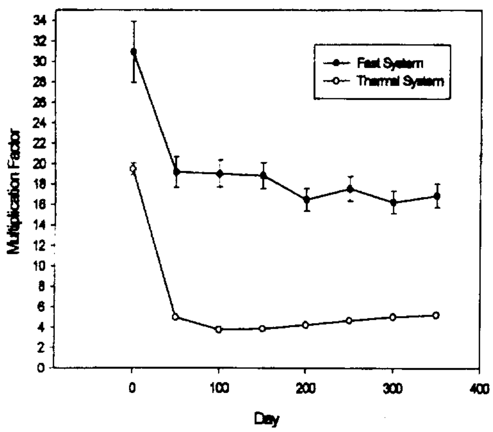


Figure 6. **Beam current fluctuation vs burnup**

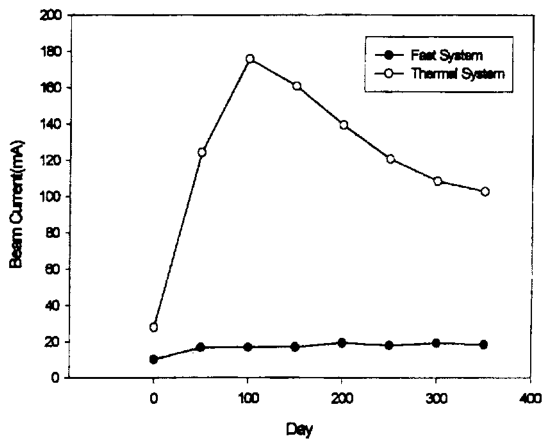


Figure 7 shows the variation of TRU inventory in the system. As expected, the inventory of the thermal system is about 1/3 of the fast system. From the figure, the thermal and fast systems incinerate about 207.13 and 318.01 kg of TRU a year, respectively. U-233 makes a contribution to that difference. U-233 build-up rates are shown in Figure 8. Both systems have almost the same amount of U-233 after one year operation. However, the total amount of U-233 produced in the thermal system is considerably different from that of the fast system. In order to produce 1 000 MW for a year, approximately 360 kg of fissile material has to be consumed. Thus, about 160 kg and 40 kg of U-233 are supposed to be depleted in the thermal and fast systems, respectively. Figure 9 shows the relative toxic variations in the system as a function of TRU burnup. As expected, the thermal system has a higher capture-to-fission ratio and makes TRU more toxic.(higher actinide)

Figure 7. **TRU inventory variation vs burnup**

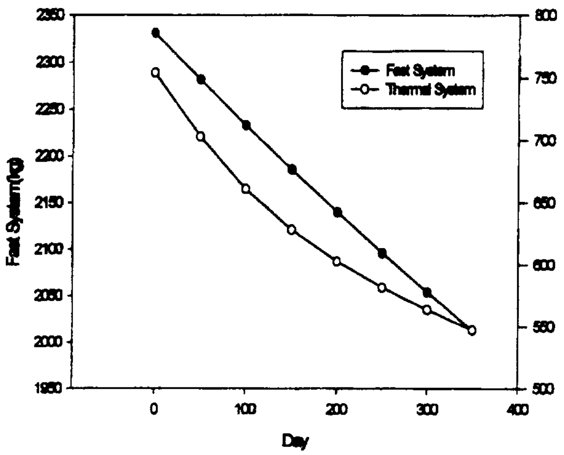


Figure 8. **The amount of ^{233}U produced**

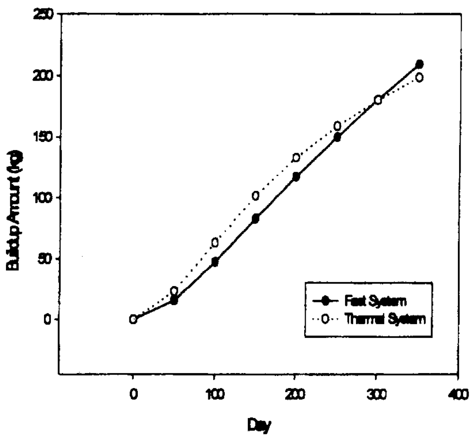


Figure 9. **Relative radioactive ingestion hazard variation**

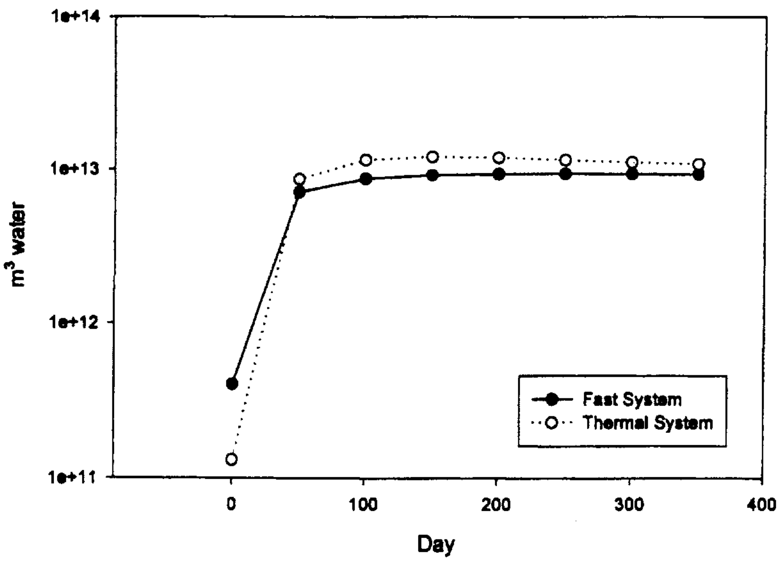


Figure 10 and 11 show the relative assembly power variations versus TRU burnup for fast and thermal systems, respectively. The radial power shape of the fast system is not perturbed considerably as TRU burns up. On the other hand, the burnup of TRU shows a totally different trend in the thermal system. The build-up of fission product prevents the spallation neutrons from being propagated to outer regions of the system. Thus, the power peaking of the thermal system becomes something unacceptably high.

Figure 10. **Relative assembly power variation in fast system**

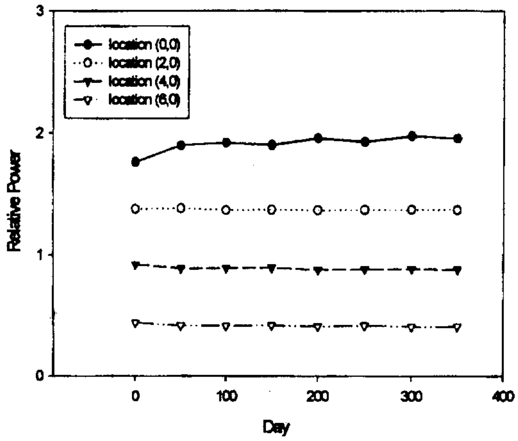
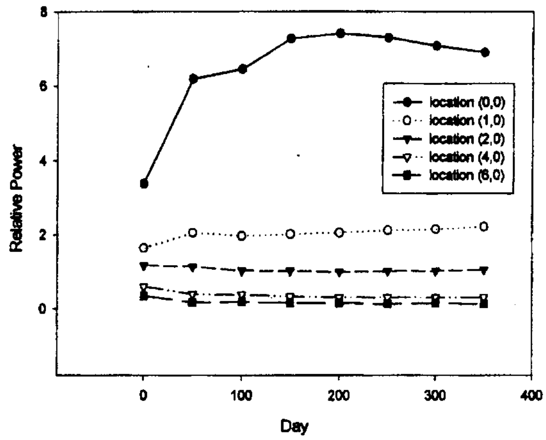


Figure 11. **Relative assembly power variation in thermal system**



Summary and conclusion

A type of system comparison study was performed to investigate which neutron system is more effective for the incineration of TRU. Table 3 shows the results of the comparison.

Table 3. **System performance parameters for TRU incineration**

Parameters	Thermal System		Fast System	
Multiplication Factor	19.469 5.216	(BOC) (EOC)	30.905 16.880	(BOC) (EOC)
Beam Fluctuation (Max/Min)	6.28		1.89	
Transmutation Capability – TRU(BOC) – TRU(EOC)	207.13 kg 754.3 kg 547.2 kg	 (BOC) (EOC)	318.01 kg 2 331 kg 2 012.99 kg	 (BOC) (EOC)
Power Peaking	3.39 6.91	(BOC) (EOC)	1.76 1.96	(BOC) (EOC)
Toxic Variation	1.3×10^{11} m^3 water 1.1×10^{13} m^3 water	(BOC) (EOC)	4.0×10^{11} m^3 water 9.3×10^{12} m^3 water	(BOC) (EOC)

Unacceptable beam fluctuation and power peaking variations of the thermal system come from the characteristics of the thermal neutron. In general, TRU and fission products have larger neutron fission/absorption cross-section in thermal energy than high energy. Therefore, a small change of TRU or fission product concentration disturbs the system multiplication or power shape considerably. Especially, this kind of phenomena becomes much more severe in a subcritical system because large absorption cross-section localises the influence of the external neutron. Figure 11 shows such localisation. As TRU burns up, the radial power shape is skewed to the central region of the system. Zoning of TRU fuel could lessen these kinds of power peaking problems. However, the characteristics of the thermal neutron would not allow the employment of a solid fuel concept in a thermal neutron subcritical system. In addition, the sensitiveness of thermal neutron subcritical system to the concentration of TRU and fission products is believed to inevitably require on-line refuelling for reducing the beam power fluctuation.

From the overall comparison, a fast neutron is concluded to be much better for the operation of the subcritical system.

Acknowledgements

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