NUKLEONIKA 2010;55(3):317–322

ORIGINAL PAPER

On the neutronics of European lead-cooled fast reactor

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Abstract. The perspective of nuclear energy development in the near future imposes a new challenge on a number of sciences over the world. For years, the European Commission (EC) has sponsored scientific activities through the framework programmes (FP). The lead-cooled fast reactor (LFR) development in the European Union (EU) has been carried out within European lead-cooled system (ELSY) project of the 6th FP of EURATOM. This paper concerns the reactor core neutronic and burn-up design studies. We discuss two different core configurations of ELSY reactor; one loaded with the reference – mixed oxide fuel (MOX), whereas the second one with an advanced fuel – uranium-plutonium nitride. Both fuels consist of reactor grade plutonium, depleted uranium and additionally, a fraction of minor actinides (MA). The fuel burn-up and the time evolution of the reactor characteristics has been assessed using a Monte Carlo burn-up code (MCB). One of the important findings concerns the importance of power profile evolution with burn-up as a limiting factor of the refuelling interval.

Key words: fuel burn-up • plutonium • minor actinides • oxide fuel • nitride fuel • LFR • ELSY • MCB

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Received: 15 December 2009
Accepted: 19 April 2010

Introduction

In search for an optimal configuration of a nuclear reactor with high burn-up, many tentative parameters should be followed. The physics of the reactor core is tightly connected with the geometry and material composition, so it is important to investigate all crucial parameters collectively as they evolve with time and fuel burn-up. Table 1 presents the main neutronic, thermohydraulic, thermomechanic and material requirements to be met in the pre-designing process.

The power profile depends also on the mentioned requirements, so its estimation could give many benefits in terms of reactor core optimization. The safety of nuclear system depends on many processes which take place inside the reactor [10, 11]. The reaction cross sections influence many physical processes as the nuclear reactions generate heat, which must be removed by liquid coolant which, in turn, affects corrosion of the structural and fuel assembly materials. These processes are coupled, therefore the core performance analysis needs to be carried out using coupled calculation models which are interconnected through power density distribution. In the paper we compare the power distributions in two ELSY core concepts depending on the fuel – oxide fuel vs. nitride fuel. The described cases are representative for each concept at the current level of optimization that has been achieved within the ELSY project.

Table 1. Reference parameters of ELSY lead-cooled fast reactor [4]

Characteristic	Value
Thermal power	1500 MW
Efficiency	~ 40%
Maximum clad temperature	550°C
Coolant inlet temperature	400°C
Coolant outlet temperature	480°C
Maximum radial form factor	1.3
Maximum Pb velocity	2.0 m/s
Expected fuel residence time	5
Cladding material	FMS T91
Fuel material	$(U,Pu,MA)O_2$, (U,Pu,MA)ZrN

ELSY

The ELSY specific targeted research and training projects (STREP) being realized since September 2006 and sponsored by the EC within the 6th EUROATOM FP aims at designing an innovative lead-cooled reactor. In particular, the project focuses on the European harmonization of actions essential to the development of new-generation nuclear reactors [4].

The specific features of pool-type LFR fully comply with four main goals of Generation IV nuclear energy systems, that are: sustainability, economics, safety and reliability, proliferation resistance and physical protection [5]. ELSY is characterized by the unique engineering features, which provide simplicity and compactness of the plant and reduce its capital cost.

The fast neutron flux ensures the capability of burning self-generated MA, which is one of the leading criteria of waste minimization and management. Additionally, the close fuel cycle criterion imposes synergy with the fleet of light-water reactors (LWR) and contributes to the reduction of spent fuel inventory.

The project consortium is constituted by many partners from the industry, universities, national research organizations and the EC's Joint Research Centre. The coordinator of the project is Ansaldo Nuclear from Italy.

The lead reactor technology was developed in the former EC project preliminary design study of experimental accelerator driven system (PDS-XADS). The development of accelerator driven system (ADS) in the 5th FP has given a lot of experience and has shown a new direction in the research. Lead-cooled European advanced demonstration reactor (LEADER) – the ELSY follow-up project – will be carried out in the 7th FP, where the main objective is to decide on the final configuration of a lead-cooled industrial reactor prototype and to propose a conceptual design of a scaled down demonstrator, the construction of which is foreseen for about 2020. AGH University of Science and Technology (AGH), Kraków will take part in this ambitious project.

AGH contribution

The main task of AGH was to perform a neutronic core analysis considering the usage of innovative nitride fuel and to asses its influence on the core performance. The potential of MA burning in the fast neutron spectrum and its in-core presence has also been investigated. The optimization of the core structure, its division scheme and fuel composition using MCB has been a significant part of the defined activities.

The other calculations done by the AGH team are related to the production of C-14, which is a considerable issue in terms of the radiotoxicity of nuclear waste. The criticality safety analysis was performed in order to assess the appropriate working conditions of the proposed system, which included the analysis of reactivity coefficients. In this paper we demonstrate two chosen core configurations, oxide and nitride one, and power profiles in each of them.

Lead impact

Pure lead is considered as the best available coolant for ELSY, which has impacted the reactor safety, thus imposing new requirements on the design. Lead demonstrates a set of specific advantages among a few possible coolants for fast reactor systems. Its chemical inertness limits the undesirable effect of interaction with air and the secondary coolant (e.g. water, CO_2), in the case of loss-of-flow accidents during operation. Therefore, the design of the cooling system is facilitated and there is no necessity of an intermediate cooling loop. Operating at atmospheric pressure enables the decay heat to be removed by natural circulation. High lead density and low operating pressure are other important features in case of primary circuit rapture. The ejection of low pressure lead is less dangerous than that of high pressure. In addition, this kind of coolant is less expensive, presents less radiological concern than, for instance, lead-bismuth eutectic (LBE). Furthermore, from the neutronic point of view, lead has a low neutron absorption cross section and low moderating power, which are basic features needed to form fast neutron flux, required for MA burning.

On the other hand, some problems with lead are posed by its high density and lack of transparency. Repair works, reloading, reshuffling and internal inspections of the core elements are more difficult. Fuel pin cladding material must withstand high temperatures and ensure high corrosion resistance. It was shown that dilution of a minor amount of oxygen in molten lead starts the passivation process of the cladding. The other problem is concerned with high melting temperature of pure lead (601 K) that poses a risk of the coolant freezing in the system, therefore high temperature must always be kept.

Core geometry

In our model we assume that fuel rods are nested in square fuel subassemblies [1]. Huge experience with such a fuel lattice brings about its common usage in LWR. Figures 1 and 2 plot the cross cuts of an arranged core with the fuel zone placement for both cases. The crossed squares represent control rod (CR) subassemblies whose position is regulated during irradiation time in order to maintain the thermal power at a required

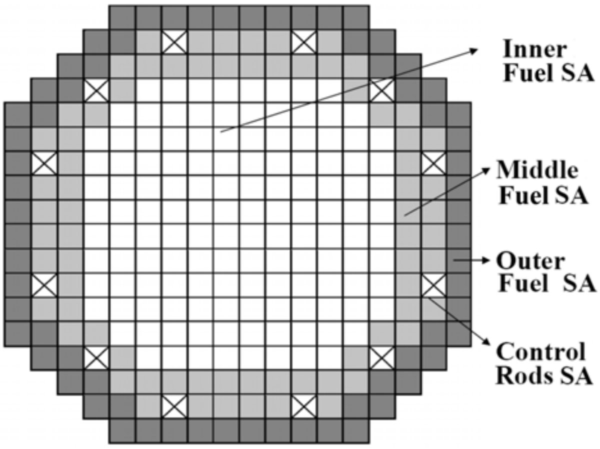

Fig. 1. The three-zone division of ELSY core loaded with MOX fuel.

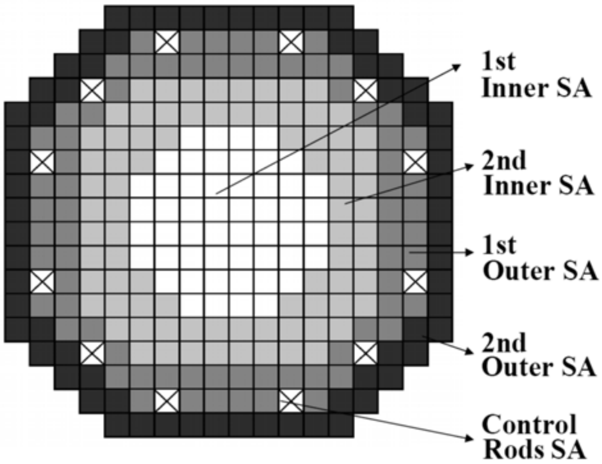

Fig. 2. The nitride core configuration with four zones of fuel SA.

level. The absorbing material consists of 70% B_4C volume fraction while the rest is lead and stainless steel. In the case of a serious accident all 12 CRs subassemblies (SA) are fully inserted, thus stopping the chain reaction. Above the active core another absorbing element, so-called control cloak, is placed. Its task is to provide an additional negative-reactivity margin. The control process is envisaged by changing its distance to the active core. The material of this system contains 15% B_4C. The B-10 enrichment in the CR and control cloak is 90%. Each fuel rod contains a plenum located under the fuel pellet stack for the accommodation of fission gases. This compact core together with the elements of the reactor cooling system pumps and heat exchangers are immersed in a pool of liquid lead. The radius of the active core is about 216 cm, while the total radius of the reactor footprint is 290 cm. The axial dimension of the active column depends on the fuel type, optimization of which was also a part of our work. The fuel SA contain 284 pins with fissionable material and five structural pins of stainless steel. The total radius of the pin is about 5.3 mm and SA side length is 24 cm. The T91 ferritic-martensitic steel (FMS) shows good corrosion and irradiation resistance during burn-up time [1]. Hence, we use it for pin cladding material.

Fuel characterization

In the preliminary ELSY analysis the MOX PuO_2+UO_2 with various enrichment of Pu atomic fraction and addition of MAs at a level of 2.5–5% at. has been considered [1, 4, 8].

In comparison to the oxide fuel, the nitride fuel shows a higher thermal conductivity, which enables lower operating temperature, thus mitigating the fuel swelling and the fission gas release [10]. Higher density improves the safety characteristics, particularly through the decrease of the fuel active column height, with the radial dimension unchanged. In addition, dense fuels provide better neutron economics and thus, it is possible to extend the duration of the fuel cycle with the same nominal power level without refuelling and reshuffling. The next benefit of nitride fuel usage in ELSY core is holding constrains of allowed reactivity swing and maximal power density during irradiation time. The production of radiotoxic C-14 from N-14 is a drawback of nitride fuels. In order to mitigate this feature the application of N-15 enriched nitride is foreseen. The N-15 enrichment of 99% has been found sufficient to bring the C-14 generation comparable to that in the oxide fuel.

Several types of inert fuel matrices (IFM) are envisioned for LFR. The fuel with IFM is a special kind of nuclear fuel, where fissionable material is dissolved in the matrix of non-fissionable one. However, the main reason for the usage of IFM is limiting the breeding of Pu-239 from U-238. For such a material, the assessment of its physical and chemical properties under irradiation is a necessary requirement. Especially the thermal conductivity at high temperatures and the reduction of the fuel reactivity play an important role. The fabrication and reprocessing technologies are also a significant criterion for its choice. As the IFM material we use ZrN, with high melting temperature and thermal conductivity. Simultaneously, the small slowing down power and absorption cross section improve its neutronic characteristics.

The isotopic composition of the fuel elements contains reactor grade plutonium and uranium unloaded at the burn-up of 45 $MW \cdot kg^{-1}$ from classical LWR and cooled down for 15 years. The enrichment of the initial UO_2 fuel in LWR is 4.5% of U-235. The amount of neptunium, americium and curium in heavy metal (HM) is a result of mixing the above spent fuel (90%) with the fuel whose cooling period reaches 30 years (10%) [1, 4]. The main fissionable isotope, which provides self-sustaining chain reaction for such a fuel is Pu-239, created from U-238 in LWR core, through successive β^- decays of U-239 and Np-239.

Calculation tools

For the core time dependent calculation, we use Monte Carlo continuous energy burn-up code (MCB) [2]. The MCB couples well-known MCNP code (a general Monte Carlo n-particle transport code) with novel TTA code (trajectory transmutation analysis). MCNP is used for static, time independent particle transport simulation and TTA calculates nuclide density evolution during irradiation and cooling time [3].

The code enables eigenvalue calculations of critical or sub-critical system and there are no limitations in the geometry and material content of the physical model. Fully three-dimensional (3D) heterogeneous geometry with high level of complexity may be simulated by MCB. Moreover, the code allows for adopting a bulk of cross section data for the reaction rates calculation and radiotoxicity assessment. In the fuel cycle of high burn-up it is important to evaluate the impact of the CRs operation on the power distribution. This impacts the burn-up distribution and power profile, thus implicating the reshuffling and reloading times. Neglecting this may lead to false conclusions. This imposes the requirement on the calculation system to simulate fuel cycle with CRs operation, that is CRs insertion level adjustment with time and burn-up. The MCB provides such a solution, which has been exploited in the reported studies.

The fuel cycle in MCB is analysed step-wise. At each step, MCNP modules are invoked first for transport calculation, where reaction rates, heating rates and neutron flux are evaluated, then MCB makes normalization and invokes TTA transmutation calculations, where material densities and composition are obtained using extended linear chain method based on the Bateman equation. Between time steps the system geometry rearrangement takes place according to the user specification, which can include changes due to CR operation, fuel reshuffling or reloading. This approach is repeated until all time steps are advanced and the final results are printed in output files.

Core optimization

For oxide fuel, we have proposed three fuel zones with different enrichment of plutonium, but without addition of minor actinides (Fig. 1). The density of applied fuel equals 95% of the theoretical one; 5% of porosity was added in order to accommodate the fuel swelling. The Pu content in HM increases outwards from 13.4 wt.% in the inner zone, 15.0 wt.% in the intermediate zone and 18.5 wt.% in the outer one, which is shown in Table 2. The number of SAs varies in each zone, and is, respectively: 132 in the inner, 72 in the intermediate and 68 in the outer, which gives the total of 272 fuel SAs in the core. This kind of division has been applied to receive a flat power profile, which is fully connected to the core thermohydraulics and thermomechanics.

Table 2. Enrichment in Pu in MOX fuel [4]

Zone	Weight fraction in HM (%)
	Pu
I	13.4
II	15.0
III	18.5

Table 3. Nitride fuel composition and MA and Pu enrichment in HM

Zone	Volume fraction (%)		Atomic fraction in HM (%)	
	ZrN	(U,Pu,MA)N	Pu	MA
I	30	70	15.0	2.5
II	20	80	15.0	2.5
III	10	90	17.5	5.0
IV	0	100	17.5	5.0

In the second option we have postulated the usage of a more advanced nitride fuel with the addition of MAs and 85% theoretical density [9]. The total number of SAs has not changed, but the enrichment in Pu and the innovative inert matrix material ZrN has been introduced. In this model there are four fuel zones, with different enrichment in Pu, and different amount of Zr and MAs. The first, innermost region consists of 52 SAs, the second – 80 SAs, the third – 72 SAs and the fourth, outermost – 68 SAs (Fig. 2). In the two inner fuel zones the plutonium atomic fraction is 15.00% and MAs – 2.5%. In the two outer ones this amount increases to 17.5% for Pu and to 5% for MAs. The zirconium volume fraction is the highest in the core centre and decreases outwards from zone to zone, being 30, 20, 10 and 0%, respectively.

The considerable result of nitride fuel usage, in comparison to oxides, is the reduction of the active column height from 90 to 77 cm. This means that with the higher actinide density the lower relative concentration of fissile nuclide occurs, and this property may be responsibly used in the design of the lead-cooled fast reactor.

Power profiles

The power profile is one of the most important parameters which should be properly calculated in the design of the nuclear reactor core and there are many ways for its estimation [6, 7]. The requirement of the flat power profile is necessary to avoid the creation of hot-spots. In such a place the released power is higher than the average and the probability of material damage increases. Thus, it is necessary from the safety perspective to find local power density (PD) in the hottest region. To avoid hot spots, the subdivision of the core into various enrichment regions, reshuffling, application of burnable poison and CR movement is applied. All these methods may be considered to make power profiles as flat as possible.

Figure 3 plots the radial PD distribution for the oxide core without MA fraction, in function of core radius. The PD has been calculated in each fuel pin, in a symmetric row of fuel subassemblies (9 SAs) and homogenized over the assembly volume. The calculations of the power profile have been performed by MCNP4 equipped with Jeff2.2 and Endf6.8 nuclear data tables, processed for temperatures from 300 to 1800 K, with a step of 300 K. We have used two independent nuclear data libraries in order to check how the results depend on various cross section tables. Some statistical fluctuation in the range of Monte Carlo (MC) relative error may be observed, but the difference is insignificant. The Jeff2.2 libraries lack the reaction cross section for lead and silicon, instead we have applied the Jendl2.2 tables.

The 1500 MW core model has been characterized only at the beginning of life (BOL) state. Neutronic analysis shows that power peaks do not occur in the core centre, but at the core boundaries. The maximum

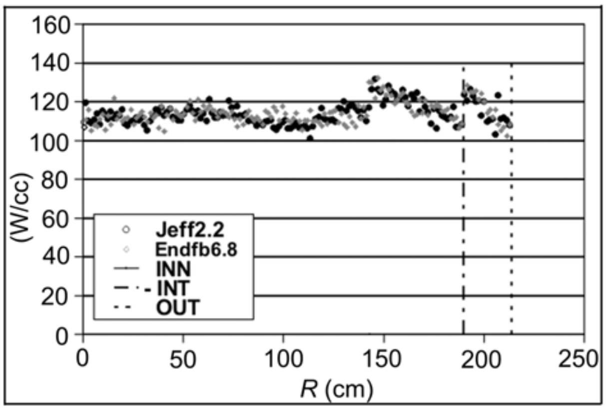

Fig. 3. Power density at BOL for oxide core specification (W/cc).

value of about 130 W/cc is obtained at the beginning of the second mid-enriched fuel zone. In the first fuel region the power shape is almost flat and in the second and third decreases outwards.

The dynamic, time-dependent simulation of nitride core configuration has been performed by MCB. In this case we have used Jeff3.1 cross section files. The set of material temperatures is the same as in the oxide core. The MCB automatically estimates axially integrated power factor, which is the ratio of average PD in selected radial region to average PD in the whole reactor core, in a specified time step. In order to provide more accurate results in the reactor centre, we have introduced six radial burn-up regions and seven axial ones. Each of the two most inner enrichment zones, was splitted into two burn-up zones, whereas the 3rd and 4th enrichment zones make subsequent 5th and 6th burnable regions, as shown in Fig. 2.

We have investigated the power profile evolution during 5 y of irradiation. Figure 4 presents the dependence of axially integrated power factor (AIPF) on radial regions from BOL to the end of life (EOL) for seven time points, with a step of 300 d. It is noticeable that the power profile does not vary significantly over the reactor operation cycle. Two mild peaks can be observed in the third and fifth radial regions. Let us note that AIFP increases in the first three burn-up regions, is almost levelling in the 4th one and decreases in the 5th and 6th, during the life time of the core. Its maximum value lies in the 5th region, at BOL, and is approximately 1.2.

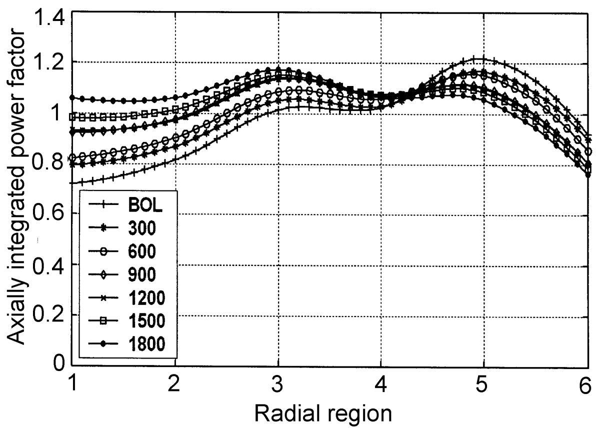

Fig. 4. Axially integrated power factor for nitride core configuration in time function.

Conclusions

Both core specifications, the applied fuel composition and division schemes fulfil major assumption, provide the appropriate power profile. The power distribution is flat and only mild peaks at the fuel zone boundaries are observed. All this justifies the usage of various zones with different fuel enrichment in Pu and MAs.

The results obtained by MCNP code are insensitive to nuclear data libraries. Two independent simulations, using Jeff2.2 and Endf6.8, present almost the same values of PD. The maximum value of PD in the oxide core appears in a few first pins in the second fuel zone.

The ZrN nitrate fuel matrix shows a new trend in fuel technology for Generation IV fast reactors. Our studies confirm that nitrides present a number of advantages under irradiation and considerably stabilize the reactor behaviour in terms of power profile distribution and the reduction of axial fuel dimension. The creation of hot-spots is avoided, which constitutes a great potential of the nitride core, and eliminates the need for fuel reshuffling during operation time.

Nevertheless, many technical issues are open for discussion and future international activities should be prepared for further development. In long terms, the construction of reactor prototype should be foreseen, and the confrontation of numerical results with experimental values should be curried out.

Acronyms

3D	– three-dimentional
ADS	– accelerator driven system
AGH	– AGH University of Science and Technology, Kraków
AIPF	– axially integrated power factor
BOL	– begin of life
CR	– control rod
EC	– European Commission
ELSY	– European lead-cooled system
EOL	– end of life
EU	– European Union
FMS	– ferritic-martensitic steel
FP	– framework programme
HM	– heavy metal
IFM	– inert fuel matrix
LBE	– lead-bismuth eutectic
LEADER	– lead-cooled European advanced demonstration reactor
LFR	– lead-cooled fast reactor
LWR	– light-water reactor
MA	– minor actinides
MC	– Monte Carlo
MCB	– Monte Carlo burn-up code
MCNP	– a general Monte Carlo n-particle transport code
MOX	– mixed oxide fuel
PD	– power density
PDS-XADS	– preliminary design study of experimental accelerator driven system
SA	– subassembly
STREP	– specific targeted research and training projects
TTA	– trajectory transmutation analysis

Acknowledgment. This work has been financially supported by the following grants: 1) FP6 ELSY, contract no. FI6W-2006-036439; 2) SPUB 348/EURATOM/2007/7; 3) ACK CYFRONET AGH, ELSY FP6 "Lead cooled system", MNiSW/IBM_BC_HS21/AGH/052/2008.

References

1. Artiori C, Sarotto M, Massara S (2007) Preeliminary ELSY core design (Open square fuel assembly option), Derivelable 6, FP6 ELSY, contract no. FI6W-2006--036439
2. Cetnar J (2006) Development and applications of MCB – Monte Carlo continuous energy burn-up code. AGH University of Science of Technology, Kraków
3. Cetnar J (2006) General solution of Bateman equations for nuclear transmutations. Ann Nucl Energy 33:640–645
4. ELSY Work Program, European lead cooled system (ELSY) (2006) Project, FP-036439, 6th Framework Program, EURATOM, Management of radioactive waste
5. GEN-IV (2002) A technology roadmap for generation IV nuclear energy systems. US DOE Nuclear Energy Research Advisory Committee and the Generation IV International Forum, GIF-002-00
6. In Ho Bae, Man Gyun Na, Yoon Joon Lee YJ, Goon Cherl Park (2008) Calculation of the power peaking factor in a nuclear reactor using support vector regression models. Ann Nucl Energy 35:2200–2205
7. Montes JL, François JL, Ortiz JJ, Martín-del-Campo C, Perusquía R (2009) Local power peaking factor estimation in nuclear fuel by artificial neural networks. Ann Nucl Energy 35:121–130
8. Sobolev V, Malambu E, Aïd Abderrahim H (2007) Preeliminary fuel pin, hexagonal assembly and core design for ELSY-600, Derivelable 5, FP6 ELSY, contract no. FI6W-2006-036439
9. Thetford R, Mignanelli M (2003) The chemistry and physic of modeling nitride fuels for transmutation. J Nucl Mater 320:44–53
10. Tuček K (2004) Neutronic and burnup studies of accelerator-driven system dedicated to nuclear waste transmutation. Doctoral thesis, Royal Institute of Technology, Department of Physics, Stockholm
11. Tuček K, Carlsson J, Wider H (2005) Comparison of sodium and lead-cooled fast reactors regarding severe safety and economical issues. In: Proc of the 13th Int Conf on Nuclear Engineering, 16–20 May 2005, Beijing, China, ICONE13-50397