BURNUP IN A SUB-CRITICAL SYSTEM WITH FLAT POWER DENSITY

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Abstract

Burnup characteristics of a sub-critical system devoted to the incineration of LWR spent fuel have been studied. Depleted uranium and burnable absorbers were introduced into the core in order to minimize power peaking and reactivity loss during a burnup period of 300 days. In addition, the burnable absorber served the purpose of blocking unwanted neutron captures in even neutron number actinides. As a result, reactivity loss for a fuel having a ^{239}Pu/TRU ratio of 50% was limited to about 6.5% for a 8% TRU burnup. This is to be compared with 12% for a uranium-free system. Due to rather low capture-to-absorption rate the effective TRU elimination rate was about 0.65 times that of a pure TRU-fuel, in spite of the U-concentration being $\sim$50%. Further, Am and Cm destruction rates became similar to Pu elimination rates. The main drawback of the introduction of uranium and burnable absorbers is a lowering of the neutron source importance which, however, can be counteracted by use of a higher pitch-to-diameter ratio in the innermost core region.

Introduction

Investigation of the transmutation potential of accelerator-driven systems (ADS) is an essential part in the assessments of their capabilities to reduce the environmental and safety threats arising from long-term radiotoxicity of spent nuclear fuel as well as possible proliferation of nuclear materials.

In recent years, a number of two-component and multicomponent transmutation strategies involving accelerator-driven systems have been proposed [1, 2]. While the two component approach suggests destruction of the entire TRU inventory in dedicated reactors, the multi-component strategy relies on the reprocessing of uranium and plutonium which are then burned in a form of MOX fuel in ordinary critical light-water and burner reactors. Residual minor actinides are destined for incineration in a sub-critical transmuter.

The objective of the present study was to investigate neutronic and transmutation characteristics of a lead-bismuth cooled accelerator-driven system as a part of two-component scenario. The aim was to achieve flat power density in the reactor core over the whole cycle length. In standard ADS without power flattening arrangement the radial power peaking factor would be about 1.8 at BOC. A lowering of the power peaking factor would reduce radiation damage imposed on structural materials, fuel swelling rates as well as balance burnup in individual sub-assemblies.

System concept

A preliminary survey of desirable system parameters has been carried out, defining conditions and limitations for reactor design:

- average burnup about 10% h.a.
- TRU transmutation rate higher than 200 kg/y
- maximum temperature of the fuel less than half of the melting temperature
- an average uranium (^{238}U) content around 50%

The model of ADS investigated in this study is a sub-critical liquid lead-bismuth accelerator-driven transmuter [3], baptized as the Sing Sing Core (SSC). The proton beam (E_p=1 GeV) enters the reactor at top and strikes the lead-bismuth target at the center of the core. The target region is surrounded by four core regions with active fuel length of 100 cm. Transmutation of long-lived fission products takes palace in the radial reflector and 5 cm wide upper and lower plenum regions.

Fuel

Mononitride fuel enriched by ^{15}N was chosen due to the excellent thermophysical properties, especially thermal conductivity, high disintegration (2000 K for UN) and melting temperatures (3035 K for UN, 2843 K for PuN) allowing to achieve linear power ratings by factor of two higher than for oxide fuels [4]. A very important factor, limiting the maximum power ratings in transuranic nitride fuel is rapid decrease of thermal conductivity of minor actinides in comparison with UN, see Figure 1. A transition from purely uranium fuel to plutonium nitride fuel halves the thermal conductivity and, hence, maximum linear power ratings.

Implementation of transuranic fuel is inevitably associated with reduction of fuel fraction in the core volume. This is done by addition of diluting material in a form of inert materials (like spinel) or moderators ($^{11}B_4C$). The choice of dilutent effects neutron multiplication in the core and influences neutron spectrum with consequences for the neutronic and transmutation performance of the system.

Coolant

Lead-bismuth eutectic has been investigated since 50's as a promising option for liquid-metal reactors intended to use on submarines. Though thermophysical properties of Pb-Bi are somewhat inferior to those of sodium, it offers, on other hand, low chemical activity with outer environments (water,air) excluding the possibility for fire or explosions. The development of oxygen content control of structural material corrosion has been a main breakthrough which enabled successful construction and operation of eight nuclear submarines and two full-scale on ground lead-bismuth reactors [5] with total operating experience of about 80 reactor-years.

Effect of absorbers on neutron spectra

The typical averaged neutron energy in liquid metal cooled breeder reactor (70–80% of uranium or thorium) is about 150 keV. Due to the adverse fission-to-absorption ratio, the lower energetic part of the spectra (under 10 keV) enhances unfavorable build-up of higher actinides. If we

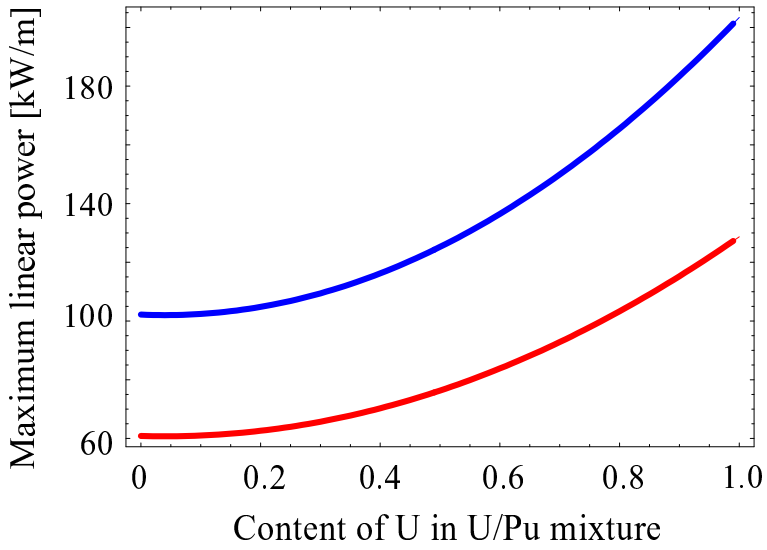

Figure 1: Maximum linear power of (U/Pu) nitride fuel with respect to the amount of uranium in the fuel. Blue bar denotes the linear power corresponding to decomposition temperature of the fuel (2000 K) which is the operational limit for fuel lattice design before the gap is closed. The gap closure occurs within 5–10 days after the start of the reactor. Then, in order to reduce fuel swelling rates it is favorable to maintain linear power such that fuel temperatures do not exceed a half of the melting temperature (red bar)

could suppress this part of the spectrum by usage of appropriate absorber the effective fission-to-absorption ratio would increase, resulting in reduction of unwanted captures. An additional demand on a chosen absorber is its partial capacity to burn out and, hence, contribute to the power flattening in the core. An extensive survey of available absorber materials was carried out with respect their neutronic properties and material compatibility. Two promising candidates have been found: $^{10}B_4C$ and ^{177}Hf.

$^{10}B_4C$ is a standard absorber material used in fast reactors. It features high absorption cross-section over the complete thermal to fast region and low induced radioactivity after irradiation. A major drawback is tritium production in the reaction $^{10}B(n,t)2\alpha$.

Natural hafnium consists of 18.6% ^{177}Hf, with the remainder being ^{174}Hf, ^{176}Hf, ^{178}Hf, ^{179}Hf, and ^{180}Hf. ^{177}Hf has high absorption cross-section in the thermal and resonance regions where it competes with the absorption reaction in actinides. Further ^{178}Hf, which is built upon the neutron capture reaction on ^{177}Hf, has an effective capture cross section lower by factor of about 15. Hafnium is a widely used absorption material, primarily in elemental form. Although fast reactor experience seems quite limited, the good temperature performance which hafnium shows in LWR is promising. Nevertheless, the cladding of hafnium pellets seems to be inevitable.

Power flattening

The effect of the fuel composition on the power flattening and burnup was investigated assuming two TRU mixtures. First of them was simulating typical LWR discharge at burn-up level of 41.6 GWd/tU after 40 years of cooling-down. This option is relevant for the countries as e.g. Sweden without MOX reprocessing technologies. The second fuel TRU mixture was assumed to have

Isotope	UOX, 40 years	MOX, 4 years
^{237}Np	0.051	0.059
^{238}Pu	0.017	0.025
^{239}Pu	0.490	0.330
^{240}Pu	0.218	0.247
^{241}Pu	0.020	0.120
^{242}Pu	0.062	0.085
^{241}Am	0.117	0.072
^{243}Am	0.016	0.043
^{244}Cm	0.001	0.015
^{245}Cm	0	0.001

Table 1: Composition of TRU mixtures assumed in simulations; UOX is a typical LWR discharge after a burnup of 41.6 GWd/tU allowing for 40 years of cooling-down; MOX denotes one-through Pu recycled spent fuel burned up to 33 GWd/tU

a composition of once-recycled MOX fuel after 4 years of discharge [6], see Table 1. MOX fuel is less reactive mainly due to changed isotopic composition of plutonium, especially ^{239}Pu whose content in the TRU mixture is lowered to 67% of the original amount.

The calculation scheme implemented in our investigations consists of nuclear data processing done by the NJOY code [7], high energetic particles and neutron transport calculations accomplished by MCNPX [8] and burnup simulations performed by an extended version of MCNP – MCB (Monte Carlo continuous energy Burnup) [9].

Power flattening arrangement has essential influence on system performance. It increases maximum fuel residence time in the core, balances linear power and, hence, the temperatures of the fuel. Following power-flattening techniques have been considered in our study:

- multiple enrichment zoning
- spatial variation of burnable poison
- variation of fissile atom density by lattice spacing

The multiple enrichment zoning is accomplished by variation of ^{238}U content in the U/TRU mixture and changing thus the proportion of fissile material in fuel matrix. ^{238}U has a dual role, acting as fuel dilutent (consequently increasing maximum linear power of the fuel pellet) and as a source of a new fissile material. The concentration of uranium in the fuel was varied between 30% and 70%. Simultaneously, burnable absorbers ($^{10}B_4C$, ^{177}Hf) were placed into core zones aiming to obtain harder neutron spectra favorable for efficient MA transmutation, see Table 2.

Burnup

Burnup characteristics of two typical model configurations which have been investigated are illustrated in Table 3. Total transuranic consumption in the MOX–$^{10}B_4C$ design is about 240 kg/year and, 20% higher than for UOX system. However, because of better neutron economy of UOX fuel, burnup rates in UOX-fueled system are about 17% higher than for MOX design. At the same time, the reactivity loss is in both cases about 6500 pcm.

Design	UOX with ^{177}Hf				UOX with $^{10}B_4C$			
	zone 1	zone 2	zone 3	zone 4	zone 1	zone 2	zone 3	zone 4
Pitch-to-diameter ratio	1.75	1.45	1.45	1.45	1.75	1.56	1.56	1.56
Number of fissile pins	271	397	264	264	271	331	276	248
Number of absorption pins	0	0	133	133	0	0	55	83
TRU/(TRU+U) concentration	32	25	45	58	27	30	43	67

Table 2: SSC fuel sub-assembly design with $^{10}B_4C$ and ^{177}Hf absorbers. The amount of fissile material was varied in individual core zones in order to minimize power peaking, averaged TRU concentration is 42% and 47% in ^{177}Hf and $^{10}B_4C$ UOX designs, respectively. The concentration of ^{238}U in the "source booster" (zone 1) was reduced in order to lower breeding in the innermost core region and hence balance power peaking during burnup

	UOX	**MOX**
Fuel inventory	5800 kg	5800 kg
TRU inventory	2440 kg	3370 kg
Absorber	^{177}Hf	$^{10}B_4C$
Thermal power	1200 MW	1200 MW
Cycle length	300 efpd	300 efpd
k_{eff} at BOC/EOC	0.965/0.900	0.963/0.897
Source intensity at BOC/EOC	$4.47\cdot10^{18}/1.26\cdot10^{19}$ n/s	$3.91\cdot10^{18}/1.26\cdot10^{19}$ n/s
Burnup rate	8.3% h.a./y	7.1% h.a./y
Power peaking at BOC/EOC	1.32/1.53	1.24/1.46
Coolant temperature	$330/550^{o}$C	$330/550^{o}$C

Table 3: Burnup characteristics of SSC cores fueled by UOX discharge (^{177}Hf absorber) and MOX discharge ($^{10}B_4C$ absorber)

The introduction of ^{238}U and burnable absorbers to the core design unfavorably influences the multiplication of spallation neutrons in the core. The source neutron importance [10] describes the external spallation neutron source multiplication with respect to the reference, concomitant multiplication in a source problem with eigenvalue source distribution. Neutron source importance is proportional to the probability of spallation neutron to induce the first fission in the core, i.e. the probability to avoid non-fission absorption in the fuel, coolant, and structural materials.

This drawback can be addressed by modified design of the innermost core zone (the *source booster*). Higher pitch-to-diameter ratio of the lattice allows to raise content of TRU in U/TRU mixture while achieving same power peaking. Relative probability of inducing fission in the *booster* zone can be thus enhanced by 10%, while reflection of neutrons back to the target remains about the same.

Conclusions

Spatial variation of ^{238}U and burnable absorbers $^{10}B_4C$ in the core allows to reach radial power peaking factor lower than 1.5 over the burnup period of 300 days. Furthermore, the reactivity loss is as low as 6.5% which is about factor of two smaller than for uranium/burnable-absorber

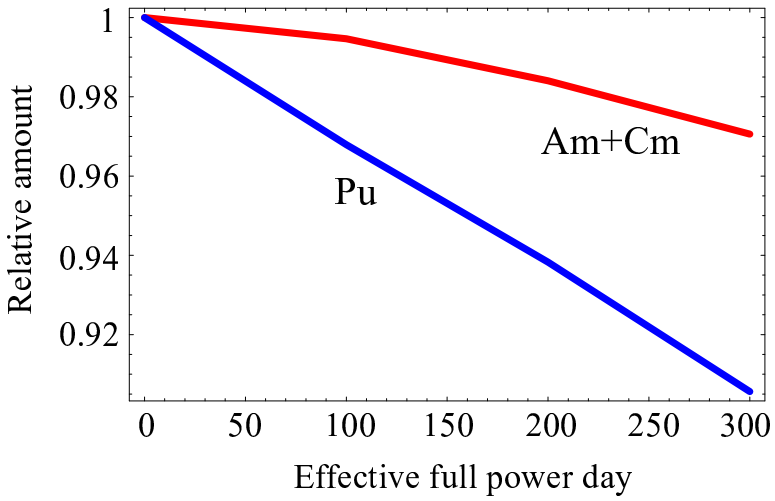

Figure 2: Relative amount of actinide (Pu and Am&Cm) vectors in SSC

free cores. At the same time, accelerator current increases 3 times.

The introduction of $^{10}B_4C$ into the third and fourth core zones significantly hardened neutron spectra in the system achieving average neutron energies as high as 350 keV. Minor actinides fission probabilities rise and their content decrease, see Figure 2. After 300 days, the Am+Cm inventory has decreased by 3%, while Pu inventory is down by 10%.

Further optimization of uranium and burnable absorber content is needed in order to accommodate reactivity loss during burnup. Additional studies aiming to minimize the "flux peaking" and increase the source importance will be carried out in the near future.

Acknowledgements

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