

MULTIRECYCLING MINOR ACTINIDES IN A GAS-COOLED FAST REACTOR

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The Gas-Cooled Fast Reactor (GFR) is a concept proposed within the Generation-IV initiative. The default core design of the GFR within the Sixth Framework Programme of the European Union is a so-called 'self-breeder' core that converts enough U-238 to plutonium to compensate the losses by fission and reprocessing. However, due to the hard neutron spectrum, the GFR has a potential for transmuting Minor Actinides (MA) as well. In this study, the CEA fuel design of the 'efficient' GFR-600 reactor core, is adapted to contain 5% to 10% of MA. The standard fuel contains 84% uranium and 16% plutonium with prescribed composition. Adding MA reduces the uranium contents correspondingly. The main results are that addition of MA reduces the reactivity swing, allowing a longer irradiation interval reaching 10% FIMA. Safety parameters remain acceptable. If a proper mix of MA material and depleted uranium is added to the reprocessed material, a fuel can be made with a pre-defined average value of k_{eff} and virtually no reactivity swing. It is shown that addition of a small amount of MA to the fuel increases the Breeding Gain, offsetting reprocessing losses. This opens the possibility of obtaining a closed fuel cycle. Decay heat is about 6.5% for steady state operation; long after shutdown the decay heat is higher for the MA fuel than for the reference U/Pu-fuel.

I. INTRODUCTION

The Gas-Cooled Fast Reactor is a concept proposed within the Generation-IV initiative¹. Within the European Union Sixth Framework Programme GCFR-STREP^a the reference design is GFR600, a 600 MWth reactor with self-breeding, i.e. breeding just enough new fissile material during operation to offset the losses of fission and reprocessing. The reactor is envisaged to run in a closed fuel cycle, requiring only natural or depleted uranium to be added to the reprocessed material to make new fuel. The fuel elements are ceramic plates arranged in hexagonal wrapper tubes. The fuel is a UPuC mixture,

^a Specific Targeted Research Program

clad with SiC. All structural materials are SiC, with Zr_3Si_2 as the reference material for the reflector. The coolant is helium and the reactor operates at high temperature, i.e. 850 °C outlet temperature. An illustration of the GFR600 fuel elements is given in figure 1.

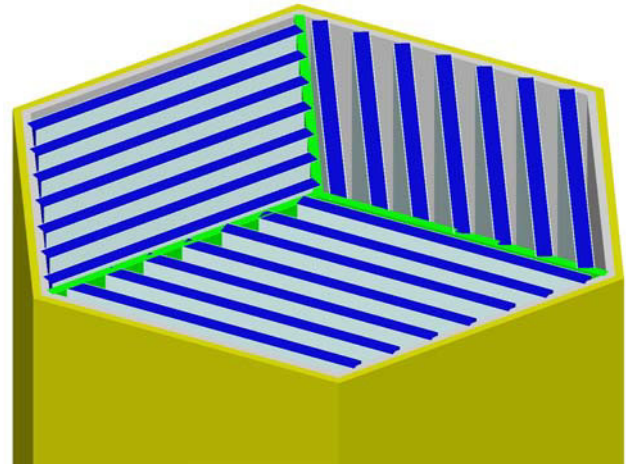


Fig. 1. GFR600 fuel assembly cross sectional view. Fuel plates (UPuC/SiC) are blue. All other materials (clad, wrapper, central mechanical restraint) are SiC.

Running the reactor in a closed fuel cycle will lead to increasing MA content in the fuel. Studies have shown² that the Minor Actinide concentration in a fast reactor running with a closed fuel cycle remains limited to about 1 or 2%. Therefore, this paper investigates higher MA loading in GFR600 to transmute existing stockpiles.

Section II gives the results for MA addition on a single irradiation cycle, and Section III reports on the effect on repeated recycling. Section IV details the results on constraint maximization of the MA loading. Finally, Section V discusses the effects on Decay Heat Power. All depletion calculations were performed with the TU-Delft in-house code BURN1D, coupling various modules of the SCALE 4.4a code system³, using a custom-made AMPX library in a 175-group fast reactor structure (VITAMIN-J) based on JEFF-3 and JENDL-3.2 data.

II. SINGLE CYCLE IRRADIATIONS WITH ELEVATED MA CONTENT

The reference fuel of GFR600 is a UPuC fuel with a small amount of Am-241. A porosity of 15% is assumed for the fuel material, leading to a density of 85% of Theoretical Density (TD). The isotopic composition of uranium, plutonium and MA material is given in Table 1. The MA vector is based on twice recycled MOx-fuel, as expected to be available from 2016. In all calculations, the uranium content is lowered corresponding to the amount of MA added. Adding extra MA to the fuel reduces the initial (fresh fuel) reactivity of the fuel, leading to the requirement of using a somewhat denser fuel at 88% TD to reach criticality. In figure 2, the result for depletion calculations with various percentages of MA is given.

TABLE 1. Fuel compositions and isotopic vectors used in the GFR600 MA depletion calculations.

Fuel Compositions and isotopic vectors			
84% U, 16% Pu, 0% MA, 85% TD			
79% U, 16% Pu, 5% MA, 88% TD			
74% U, 16% Pu, 10% MA, 88% TD			
Uranium		Minor Actinides	
U-235	0.7%	Np-237	16.86%
U-238	99.3%	Am-241	60.64%
Plutonium		Am-242m	0.23%
Pu-238	2.70%	Am-243	15.69%
Pu-239	56.00%	Cm-242	0.02%
Pu-240	25.90%	Cm-243	0.07%
Pu-241	7.40%	Cm-244	5.14%
Pu-242	7.30%	Cm-245	1.25%
Am-241	0.70%	Cm-246	0.10%

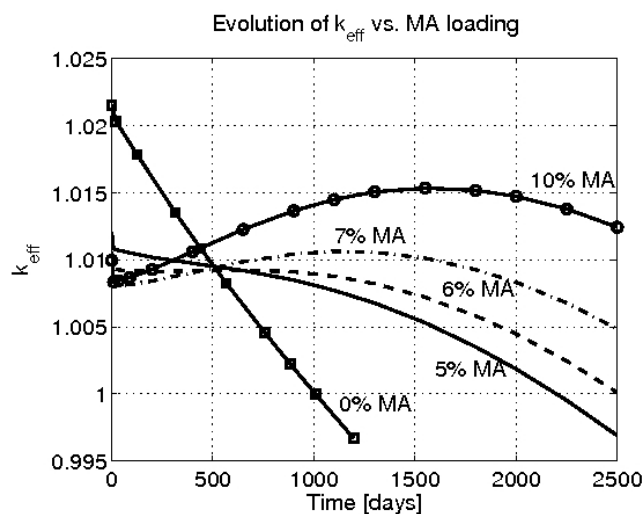


Fig. 2. Evolution of k_{eff} for various MA loadings

As seen in figure 2, the standard GFR600 fuel (0% MA, 85% TD) results in a monotonously decreasing reactivity during irradiation. The standard irradiation cycle is 1300 days, resulting in 4.5% FIMA (calculations show subcriticality because leakage is somewhat overestimated in the 1-D spectrum calculations). Addition of MA reduces the reactivity swing considerably, leading to the possibility of extending the irradiation interval to 2500 days for a 5% MA loading (corresponding to 9% FIMA), and even longer for higher MA contents, if the fuel integrity can be maintained.

The MA added to the fuel consists mainly of Np-237 and Am-241, which both transmute to fissile isotopes (Pu-238 and Am-242 respectively). As shown in section III, Np-237 and Am-241 improve the Breeding Gain of the fuel, by lowering the initial reactivity (at the expense of a higher initial fuel density), and increasing the discharge reactivity of the fuel. This effect explains the behaviour of the curves in figure 2.

The effect of adding extra MA is to increase the cycle length, yielding an increased discharge burnup, which is attractive. However, it is vital that the reactor safety parameters do not deteriorate unacceptably. The MA actinides which are added to the fuel generally produce a lower number of delayed neutrons than the uranium they replace⁴, and especially the addition of americium has the effect of reducing the effective fraction of delayed neutrons. In figure 3, the value of β_{eff} is given for 0%, 5%, and 10% MA in the fuel, calculated with the VAREX code⁵ for various levels of burnup. The reference β_{eff} of about 400 pcm is reduced by 15% to 340 pcm for the 10% MA fuel.

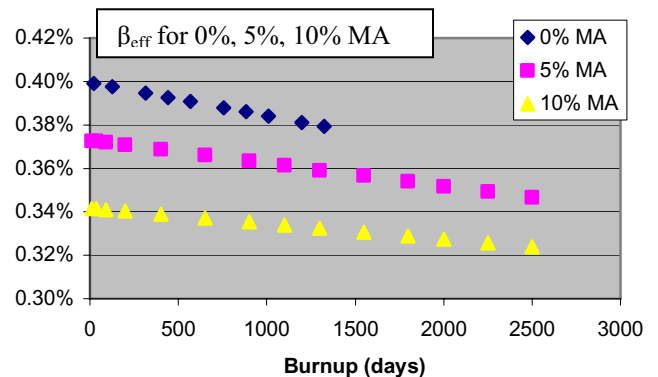


Fig. 3. Evolution of β_{eff} for GFR600 with 0%, 5% and 10% MA.

VAREX determines the contribution of individual nuclides to β_{eff} . As illustrated in table 2, U-238 gives by far the biggest contribution to β_{eff} , because although it is a threshold fissioner, it has a very high delayed neutron

production of about 1.69% in the GFR600 spectrum. Using the values of β_{eff} , the reactivity of GFR600 during irradiation can be plotted in dollars, as in figure 4. From this figure, it is seen that the reactivity of all GFR600 fuel mixtures is in a small bandwidth. Note that the reactivity swing for the standard fuel seems large, but is in fact very small compared to typical values for a PWR, for example: -24\$ (Ref. 2).

TABLE 2. Isotopic contributions to β_{eff} for a fuel with 5% MA loading.

Isotope	Contribution to β_{eff}
U-235	3.9%
U-238	51.1%
Np	2%
Pu-238	0.8%
Pu-239	23.4%
Pu-240	3.7%
Pu-241	11.5%
Pu-242	1.6%
Am	2%

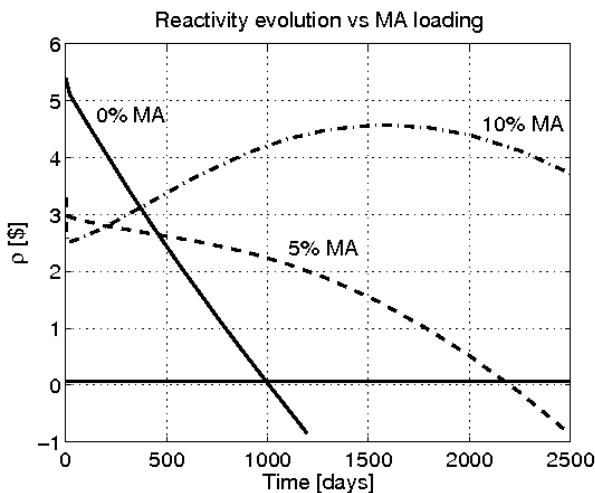


Fig. 4. Reactivity evolution for various MA loadings

Other important safety parameters are the coolant void effect and fuel temperature coefficient (Doppler). Fast reactors (especially liquid-metal cooled fast reactors) generally have positive void coefficients, which are counteracted by choosing a ‘leaky’ core configuration. GFR600 on the other hand has the optimal height over diameter ratio H/D of 1, to reduce neutron leakage, in order to reach criticality with a fuel composition adequate for self-breeding. Void-coefficient calculations were performed at each time step in the irradiation history by setting the coolant pressure from the nominal value (7.0 MPa) to atmospheric pressure. The Fuel Temperature Coefficient (FTC) was calculated by setting the fuel temperature 100 K above the reference temperature. The results are given in table 3. As can be seen, introduction

of MA increases the (positive) void coefficient from about +4.2pcm/bar to +6.4pcm/bar for the 10% MA fuel. The FTC decreases: absorption by Np-237 and Am-241 reduces the flux in the resonance energies, leading to spectrum hardening and a reduction of the Doppler effect.

TABLE 3. Values of the void and Fuel Temperature coefficients for GFR600.

Void coefficient			
	Average [pcm / bar]	Range	
0% MA	4.23	4.1 – 4.4	
5% MA	5.29	5.1 – 5.5	
10% MA	6.41	6.3 – 6.5	
Fuel Temperature Coefficient			
	pcm / K	Beta [pcm]	dR / 100 K
0% MA	-0.91 / -0.80	390	-0.22\$
5% MA	-0.56 / -0.50	370	-0.14\$
10% MA	-0.37 / -0.34	330	-0.10\$

The increase of the void coefficient is due to the somewhat harder neutron spectrum in the MA cores. One study⁶ found that the magnitude of the void effect can be tuned by carefully adjusting the spectrum in a Gas-Cooled Fast Reactor. Thus, the positive void coefficient of the MA cores could potentially be reduced with the addition of some moderating material (BeO, CaH₂)

In figure 5 the amount of helium in the fuel is illustrated for 0%, 5% and 10% MA. It is clear that increasing the initial MA loading significantly increases the Helium gas production. The consequences thereof will be investigated in further research.

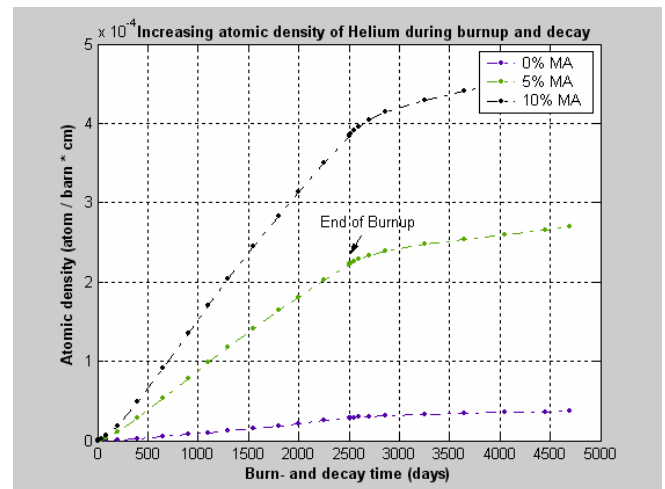


Fig 5. Gas production for the GFR600 MA-holding fuel mixtures.

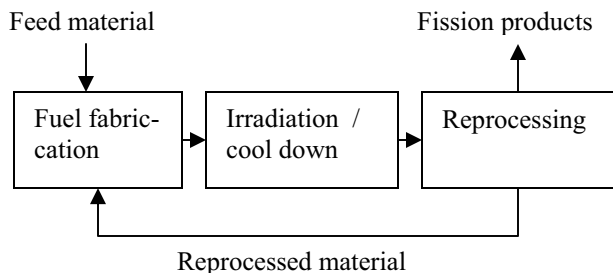


Fig 6: Material flow in the closed fuel cycle.

III. REPEATED RECYCLING WITH MA ADDITION, BREEDING GAIN

In a closed fuel cycle, a new fuel loading is made using reprocessed material, to which material is added to make up for the fission and reprocessing losses. In this process, reactivity of the fuel may be gained or lost by:

- reprocessing losses (+/- ρ)
- extraction of absorbing fission products (+ ρ)
- makeup material (+/- ρ)

An illustration of the material flow in a closed fuel cycle is shown in figure 6. As shown in Ref. 7, the reactivity of the reprocessed material generally decreases during cool down, due to decay of Pu-241 to Am-241 and due to reprocessing losses. Thus, a somewhat positive Breeding Gain (BG) during irradiation is required for a closed fuel cycle. To calculate the BG, the nuclide inventory at Begin of Cycle, End of Cycle and Begin of Cycle for the new irradiation can be calculated, based on a certain reprocessing strategy and makeup material. To accurately calculate BG, a reactivity weight w_i is necessary to quantify how each nuclide contributes to the overall reactivity of the fuel. Reactivity weights can be defined using a first-order perturbation theory⁴ approach, resulting in the following definition for w_i (in operator notation):

$$w_i = \frac{\Delta\rho}{\Delta N_i} = \frac{\left\langle \phi_0^+, \left[\lambda_0 \frac{\partial P}{\partial N_i} - \frac{\partial L}{\partial N_i} \right] \phi_0 \right\rangle}{\left\langle \phi_0^+, P \phi_0 \right\rangle} \quad (1)$$

The brackets denote scalar products, P is the neutron production operator, L is the neutron transport and loss operator, λ_0 is the reactor eigenvalue, ϕ and ϕ^+ are the forward and adjoint fluxes, respectively, and N_i is the number density of nuclide i . Equation (1) expresses the change of the reactivity of the reactor upon a small introduction of nuclide i . A rigorous derivation of this result can be found in Ref. 7. Using an infinite

homogeneous, 1-group approach for the operators appearing in eq. (1), w_i reduces to:

$$w_i = \frac{1}{\nu \Sigma_f} (\lambda_0 \nu_i \sigma_{f,i} - \sigma_{a,i}) \quad (2)$$

Which is similar to existing definition for reactivity weights (see for instance Ref. 8). In table 4 some values of w_i are given for GFR600, calculated based on eq. (1) with the TSUNAMI sensitivity module in SCALE 5⁹. Notice that w_i for Pu-238 is almost as high as for U-235 in this spectrum. Spectral hardening upon introduction of MA is evidenced by the w_i for threshold fissioners like U-238, which have a more positive (or less negative) w_i for a fuel with 5% MA.

TABLE 4: w_i for various isotopes, for GFR600 with 0% and 5% MA in the fuel.

	w_i , 0% MA	w_i , 5% MA
U-235	0.794	0.785
U-238	-0.079	-0.067
Pu-238	0.639	0.659
Pu-239	1.000	1.000
Pu-240	0.077	0.120
Pu-241	1.560	1.486
Pu-242	0.026	0.063

With the reactivity weights w_i , a ‘fissionability’ f can be defined by:

$$f = \sum_{i=1}^N N_i w_i \quad (3)$$

Where N_i is the number density of nuclide i .

In the closed fuel cycle the composition of the new fuel is dictated by the reprocessed material from the previous cycle, as indicated in figure 6. For this system, the fissionability of the new fuel is the sum of the f of the reprocessed and feed materials. It is possible to calculate the effect of changing the initial fuel composition on the value of f of the new fuel, using a first order perturbation theory approach. This approach requires solving the adjoint transmutation equations, and is detailed in Ref. 7. Calculations have been done for GFR600 in an integral fuel cycle (all actinides recycled indiscriminately), assuming reprocessing losses of 1% - 5%, and adding depleted uranium to make the new fuel. These calculations have shown that some of the absorbing MA nuclides, like Np-237 and Am-241 have positive adjoint values at BOC, which means that their presence at BOC results in a higher reactivity of the reprocessed material. This is due to transmutation of these nuclides to fissile materials (Pu-238, Pu-239 and Am-242) during

irradiation. In figure 7 an example is given of adjoint depletion calculations for a representative GFR600 mixture.

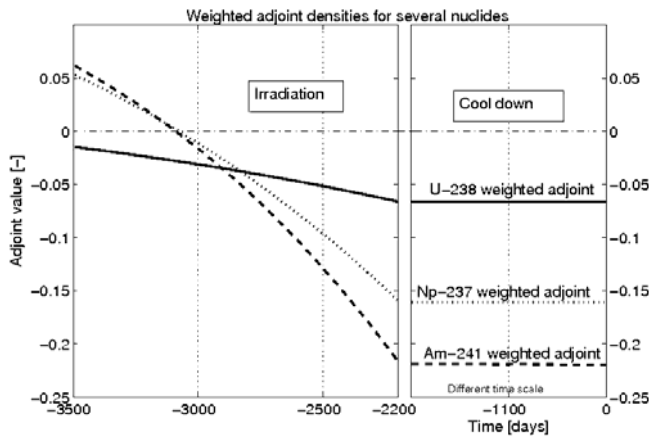


Fig. 7. Adjoint reactivity weights for several absorbing nuclides. A positive value means that the nuclide has a positive contribution to the ‘fissionability’ f of the new fuel. Np-237 and Am-241 present at BOC contribute positively through transmutation.

In figure 8 a multiple irradiation – reprocessing cycle is shown, keeping the fuel at a constant 5% MA at the beginning of an irradiation. As can be seen, the fissionability f is a useful estimator for the actual reactivity of the reactor, justifying the definition of the reactivity weights of eqs. (1) and (2). In table 5, an illustration is given of how the change of reactor composition influences f . From the table, it is seen that during irradiation, an increase of fissionability due to depletion of absorbing HM is offset by a decrease of fissionability due to fission product buildup. During the cool down period (EOD), more fissile material is lost by decay of Pu-241. After reprocessing (Repro), the fissionability is increased due to removal of fission products.

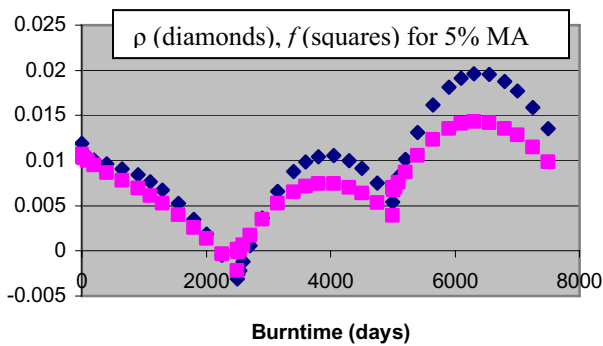


Fig 8. Reactivity and fissionability compared for consecutive irradiations in a closed fuel cycle.

TABLE 5. Contributions to f during irradiation. – HM: absorbing Heavy Metal, +HM: neutron producing HM, F.P. Fission Products

	- HM	+ HM	F.P.	f
BOC	0.0%	0.0%	0.0%	0.0%
EOC	18.4%	-4.0%	-22.3%	-7.9%
EOD	17.2%	-9.6%	-22.3%	-13.3%
Repro.	17.2%	-9.6%	0.0%	9.0%
Manif.	9.7%	-9.4%	0.0%	0.3%

In figures 9 and 10 are illustrated the reactivity and fissionability f for multiple irradiations, for different refueling options: keeping the BOC MA content constant at 5%, starting with 5% MA then adding DU, or by keeping the fissionability f of the new fuel constant. In all cases the reactivity of the reactor increases slightly from one cycle to the next. The constant MA scenario shows an increasing reactivity from cycle to cycle, which is contributed to transmutation of absorbing nuclides. As seen in figure 10, the fissionability of the fuel is more or less constant during irradiation, and increases slightly from one cycle to the next, showing that a zero Breeding Gain is almost reached.

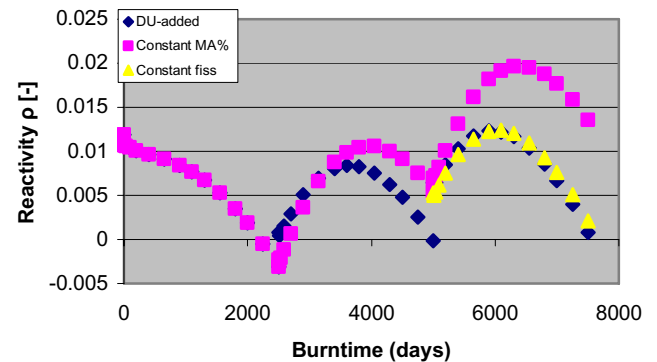


Fig. 9. Reactivity, three consecutive irradiations in a closed fuel cycle, three different feed materials.

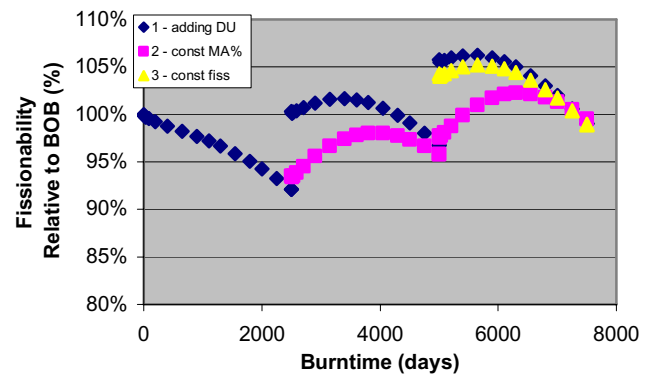


Fig. 10. Evolutions of f for consecutive irradiations in a closed fuel cycle.

In figures 11 and 12 the values of k_{eff} and f for 15 cycles of the integral fuel cycle are illustrated. The reprocessing strategy either assumes 1% loss and addition of depleted uranium, or 5% loss and addition of 90%DU+10%MA. As can be seen, both cycles result in a constant k_{eff} from cycle to cycle, showing that a closed fuel cycle is reached (strictly speaking, for 95% efficient reprocessing scheme, the cycle is nearly closed, because some MA from existing stockpiles is required as feed material).

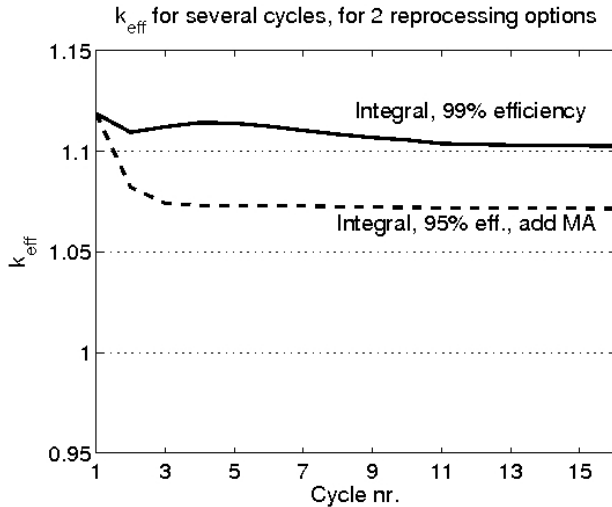


Fig. 11. Beginning of Cycle k_{eff} for two closed fuel cycle scenarios: reprocessing losses are either 1%, after which depleted uranium is added, or 5%, after which a mix of 10% MA and 90% DU is added.

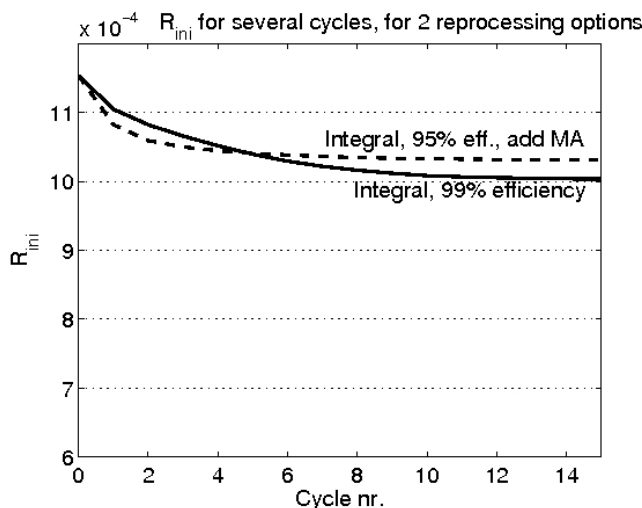


Fig. 12. The fissionability f at Beginning of Cycle (here denoted as R_{ini}) for the same fuel cycle schemes as figure 11.

In table 6 the BOC fuel composition of the tenth cycle is given corresponding to the two fuel cycle options illustrated in figures 11 and 12. As can be seen from this

table, in the 99% efficient + DU scheme, the MA loading of the fuel is very low. In the 95% efficient + (MA + DU) scheme, the MA loading is of course larger, but is still not higher than 3.9 %, confirming that in a fast reactor in a closed fuel the MA loading remains limited.

TABLE 6. Isotopic abundances at BOC of the 10th cycle. Strategy 1: 99% efficient reprocessing, DU added; Strategy 2: 95% efficient reprocessing, 90%DU+10%MA added.

	Strategy 1	Strategy 2
U	82.512%	80.78%
U-235	0.072%	0.091%
U-238	81.933%	80.272%
Np	0.119%	0.588%
Np-237	0.119%	0.588%
Pu	16.513%	15.355%
Pu-238	0.457%	1.173%
Pu-239	9.258%	8.410%
Pu-240	5.607%	4.648%
Pu-241	0.489%	0.384%
Pu-242	0.702%	0.740%
Am	0.838%	2.709%
Am-241	0.577%	1.952%
Am242(m)	0.038%	0.106%
Am-243	0.223%	0.651%
Cm	0.219%	0.568%
Cm-243	0.003%	0.007%
Cm-244	0.153%	0.419%
Cm-245	0.033%	0.095%
Cm-246	0.023%	0.038%
Cm-247	0.005%	0.007%
Cm-248	0.002%	0.002%

IV. CONSTRAINT MAXIMALIZATION OF MA LOADING

A study was performed to analyze how much MA material needs to be added to the reprocessed fuel to obtain a cycle-averaged value of k_{eff} of 1.05. For this study, an integral recycling of all actinides is assumed, reprocessing losses are assumed to be 1% or 2%, and a fuel TD of either 85% or 90% is used. The initial fuel is in all cases the reference UPuC fuel without extra MA (see table 1), and the reference irradiation interval of 1300 days is used, followed by a cool-down period of 5 years. In figure 13 the result for k_{eff} is given for 85% TD and 1% reprocessing loss. The first cycle does not meet the k_{eff} target, and 100% MA is used as a feed material. In subsequent cycles, the target k_{eff} value is reached, and a mix with 10% to 50% MA is used as the feed material in subsequent cycles. In figure 14 an overview is given of the material added to the reprocessed material for each cycle. For the other scenarios (with 90% fuel TD), a similar behaviour is observed.

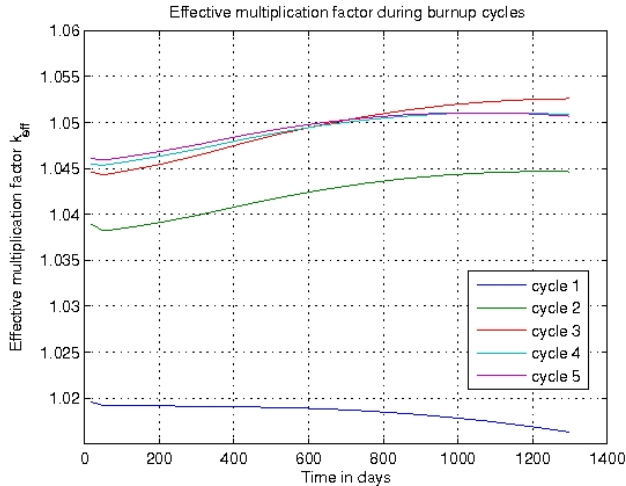


Fig. 13. Values of k_{eff} for a fuel cycle to reach a target average value of k_{eff} of 1.05. The reactivity of the fuel is adjusted by varying the amount of MA in the feed material to obtain the target k_{eff} .

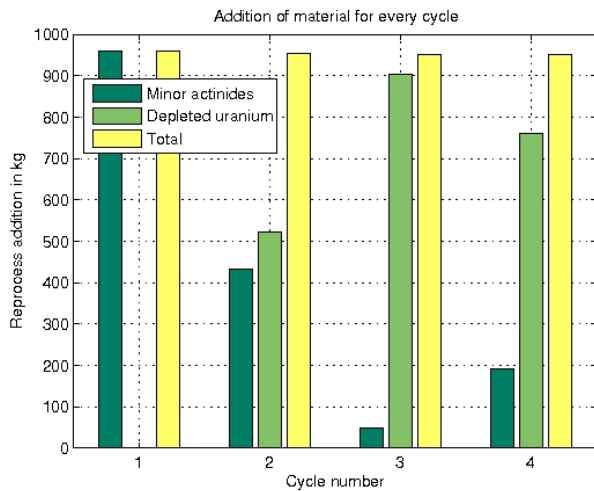


Fig. 14. Feed material added to the reprocessed material to meet the pre-defined average k_{eff} value.

In table 7 an overview is given of the total amount of materials added to the fuel cycle over 5 consecutive irradiation cycles. The fuel cycles with 85% TD have the largest MA consumption, which is expected (higher leakage, requiring larger fissile mass to obtain target k_{eff} , and larger specific power, resulting in higher burnup). The net MA consumption is about 1400 kg, for 5 irradiation cycles of 1300 days each. A modern UO_2 fuelled PWR (1000 MWe) produces some 27 kg of MA material per year; hence the support ratio for GFR600 is between 3 and 6 PWRs if the PWR plutonium is recycled as MOX or in other fast reactors.

TABLE 7. Overview of DU and MA consumption if MA is added to reach a time averaged k_{eff} of 1.05. Totals are given for 5 irradiation cycles, of 1300 days each.

	85%TD 1% loss	85%TD 2% loss	90%TD 1% loss	90%TD 2% loss
Total mass added	3816.7	4415.1	3846.5	4481.5
DU added	2184.9	2316.4	2924.7	3126.0
MA added	1631.6	2053.7	921.8	1355.5
MA consumed	1435	1541	951	1373

V. DECAY HEAT CALCULATIONS

The radioactive decay of unstable nuclides (fission products and actinides) is the source of so-called Decay Heat, which amounts to $\approx 7\%$ of the total power in an LWR during steady state operation. Decay heat production is usually expressed as the total power released by all fission products due to 1 fission of isotope j :

$$f(t) = \sum_{k=1}^K \alpha_{jk} \exp(-\lambda_{jk} t), \quad (4)$$

which is expressed in MeV/fission/s, and where it is assumed that all fission products can be lumped into K groups, with a respective decay constant λ_{jk} and contribution α_{jk} to the total decay heat. Equation (4) represents an impulse response, and should be convolved with the fission rate $\psi(t)$ to find the decay heat at any moment during or after reactor operation:

$$P_d(t) = \int_0^{t_0} \psi(t_0 - \tau) f(\tau) d\tau \quad (5)$$

Example decay heat standards based on eqs. (4) and (5) are the ANS¹⁰, and DIN¹¹ standards. To find the decay heat at time t after operation with a unit fission rate over a time interval I , integration of eq. (5) gives:

$$P_d(t) = \int_t^{t+I} \sum_{k=1}^K \alpha_{jk} \exp(-\lambda_{jk} \tau) d\tau = \sum_{k=1}^K \frac{\alpha_{jk}}{\lambda_{jk}} \exp(-\lambda_{jk} t) [1 - \exp(-\lambda_{jk} I)] \quad (6)$$

For an infinitely long interval I the last term on the RHS equals zero, and the decay heat curve is given by:

$$P_d(t) = \sum_{k=1}^K \frac{\alpha_{jk}}{\lambda_{jk}} \exp(-\lambda_{jk} t) \quad (7)$$

For lack of specific data for fast reactors in the standards, a synthetic decay heat curve based on the ANS standard was used in previous work¹². However, it is possible to construct a specific set of coefficients α_{jk} and λ_{jk} for each burnup state of a nuclear fuel, by fitting equation (7) to a calculated decay heat curve, and correcting for the total power release per fission. The data sets found in this way can be used for instance in thermal-hydraulic codes like CATHARE2¹³ in which a decay heat model based on eq. (4) is available. For 3 fuel compositions (0%, 5%, 10% MA) data sets were calculated for GFR600 with 10% burnup. In figure 15, the resulting curves are plotted together with the ANS standard for thermal fission of Pu-239 and Pu-241. Decay heat in the GFR600 fuel falls between Pu-239 and Pu-241 for short time scales, and is higher for longer time scales, which is attributed to the presence of MA with alpha-decay in the fuel (Am and Cm). In figure 16 a plot is shown of $t^*f(t)$ for the GFR600 fuel mixtures and for the ANS Pu isotopes. As can be seen from this figure, the fitting parameters for GFR600 leave room for improvement.

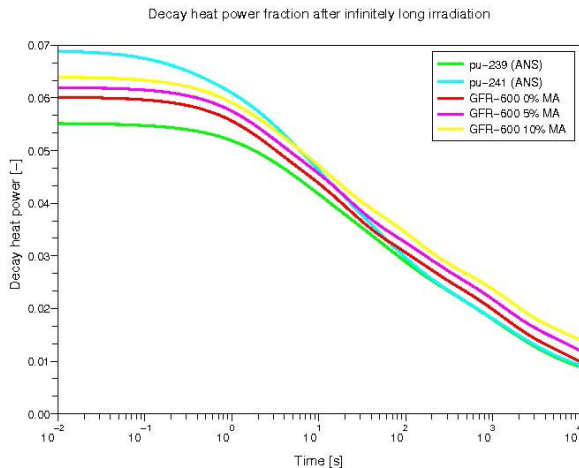


Fig. 15. Decay heat as a fraction of total power after an infinitely long irradiation interval for three fuel mixtures in GFR600, and for the 2 Pu isotopes of the ANS standard.

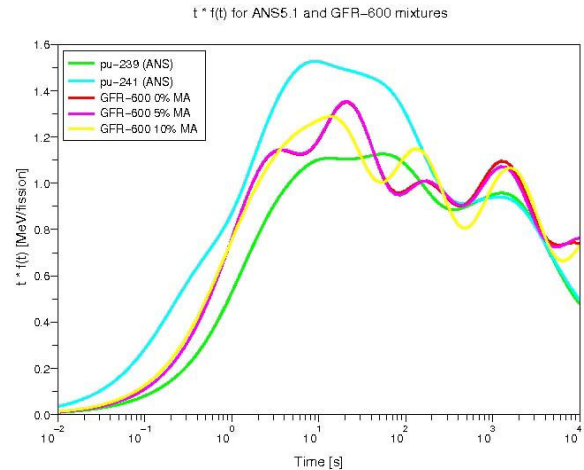


Fig. 16. A plot of $t^*f(t)$ for three GFR600 fuel mixtures and the two Pu-isotopes in the ANS standard.

The decay heat immediately after shutdown depends on the irradiation interval (eq. (6)). For GFR600, the decay heat immediately after shutdown was calculated at various burnup levels. The result is shown in figure 17. As can be seen, the decay heat peaks after about 500 days of irradiation. This effect is thought to arise from the increase of MA nuclides from begin of cycle to 500 days followed by a net destruction of MA nuclides from between 500 and 2500 days. A more detailed investigation is planned for the future.

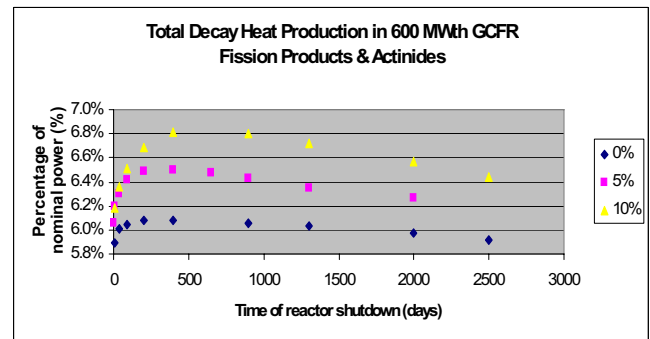


Fig. 17. Decay heat immediately after shutdown, at various burnup levels for fuels with 0%, 5% or 10% MA.

VI. CONCLUSIONS

The possibilities of obtaining a closed fuel cycle in a Generation IV Gas-Cooled Fast Reactor are reported. The reactor under investigation is GFR600, a 600 MWth GCFR using a ceramic plate fuel and helium cooling. Extra MA (up to 10% max) is added to the fuel by replacing a corresponding amount of uranium. Adding MA decreases the reactivity swing considerably, yielding longer irradiation intervals and higher discharge burnup.

The values of β_{eff} , void coefficient and Fuel Temperature Coefficient deteriorate, but not to an unacceptable level.

Multiple irradiations were performed for closed fuel cycle scenarios. Addition of MA increases the Breeding Gain of the reactor. If the reactor is started with 5% MA, addition of depleted uranium allows a closed fuel cycle. The fissionability parameter f is a good estimator for the reactivity of the reactor in the closed fuel cycle. Using a proper mix of Depleted Uranium and MA as feed material allows obtaining a pre-defined average value of k_{eff} , with almost no reactivity swing. In such a fuel cycle scenario, one GFR600 can transmute the MA from 3 to 6 PWRs.

Decay heat of the MA fuels is around 6.5% during steady state operation, with a time evolution comparable to existing standards. For long time scales ($>10^4$ s), the decay heat is larger, due the presence of alpha-emitting MA in the fuel.

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