Annals of Nuclear Energy 40 (2012) 106–115

Contents lists available at SciVerse ScienceDirect

Annals of Nuclear Energy

journal homepage: www.elsevier.com/locate/anucene

Near-complete transuranic waste incineration in a thorium fuelled pressurised water reactor

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ARTICLE INFO

Article history:
Received 22 August 2011
Received in revised form 28 October 2011
Accepted 1 November 2011
Available online 1 December 2011

Keywords:
Thorium
PWR
Transuranic
Waste
Plutonium
Incineration

ABSTRACT

The production of long-lived transuranic (TRU) waste is a major disadvantage of fission-based nuclear power. Incineration, and virtual elimination, of waste stockpiles is possible in a thorium (Th) fuelled critical or subcritical fast reactor. Fuel cycles producing a net decrease in TRUs are possible in conventional pressurised water reactors (PWRs). However, minor actinides (MAs) have a detrimental effect on reactivity and stability, ultimately limiting the quality and quantity of waste that can be incinerated. In this paper, we propose using a thorium-retained-actinides fuel cycle in PWRs, where the reactor is fuelled with a mixture of thorium and TRU waste, and after discharge all actinides are reprocessed and returned to the reactor. To investigate the feasibility and performance of this fuel cycle an assembly-level analysis for a one-batch reloading strategy was completed over 125 years of operation using WIMS 9. This one-batch analysis was performed for simplicity, but allowed an indicative assessment of the performance of a four-batch fuel management strategy. The build-up of ^{233}U in the reactor allowed continued reactive and stable operation, until all significant actinide populations had reached pseudo-equilibrium in the reactor. It was therefore possible to achieve near-complete transuranic waste incineration, even for fuels with significant MA content. The average incineration rate was initially around 330 kg per GW_{th} year and tended towards 250 kg per GW_{th} year over several decades: a performance comparable to that achieved in a fast reactor. Using multiple batch fuel management, competitive or improved end-of-cycle burn-up appears achievable. The void coefficient (VC), moderator temperature coefficient (MTC) and Doppler coefficient remained negative. The quantity of soluble boron required for a fixed fuel cycle length was comparable to that for enriched uranium fuel, and acceptable amounts can be added without causing a positive VC or MTC. This analysis is limited by the consideration of a single fuel assembly, and it will be necessary to perform a full core coupled neutronic–thermal-hydraulic analysis to determine if the design in its current form is feasible. In particular, the potential for positive VCs if the core is highly or locally voided is a cause for concern. However, these results provide a compelling case for further work on concept feasibility and fuel management, which is in progress.

1. Introduction

The estimated world stockpile of TRU waste is over 3 kt (Rubbia et al., 1995). This represents a major economic and environmental liability. Large-scale incineration of waste has been proposed in a critical or subcritical fast reactor. This approach allows virtual elimination of TRU waste (Rubbia et al., 1995) but the need for a low moderation coolant presents challenges and is not commercially proven. A highly corrosive environment in a liquid metal reactor may necessitate the use of new materials and limit product life (Furukawa et al., 2009; Zhang, 2009). Fast gas-cooled reactors may have problems safely enduring loss of coolant accidents (LOCAs), and high temperatures may cause material life problems (Pope et al., 2009). Subcritical reactors require reliable and economic accelerator technology. Given the large number of reactors required to incinerate waste, this represents a substantial barrier to effective implementation. In contrast, light water reactor (LWR) technology is well-proven, with the materials technology available to design for 60 years of operational life.

The thorium fuel cycle is of considerable interest for reducing long-term nuclear waste liability, as only a small quantity of TRUs is bred from ^{232}Th; thorium is relatively abundant and Th-based fuel does not require enrichment (International Atomic Energy Agency, 2005). Incineration of waste in an LWR partially or completely fuelled with uranium (U) reduces TRU destruction as additional TRUs are bred from ^{238}U (Aquien et al., 2006).

An extensive study of TRU waste incineration in PWRs was conducted by (Shwageraus et al., 2004). This concluded that increasing moderation in a conventional LWR was advisable to utilise Th effectively. Unaltered or reduced moderation was concluded to result in short cycle lengths and therefore uneconomic amounts

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doi:10.1016/j.anucene.2011.11.002

of reprocessing as the plutonium (Pu) burned out quickly. Using increased moderation, a limited proportion of waste could be incinerated at a substantial rate.

In this paper, we extend the work of Shwageraus et al. (2004) to consider a thorium-retained-actinides fuel cycle with an unaltered moderation ratio. All actinides are returned to the reactor after reprocessing to provide theoretically unlimited recycling of transuranic waste. The capability of a PWR, such as an EPR or AP1000, to achieve rapid and near-complete destruction of TRU waste when loaded with a mixed oxide (MOX) fuel of Th and 'dirty' Pu is demonstrated.

The study is performed for a single fuel assembly operating on a one-batch fuel management strategy for computational simplicity and concept demonstration. From these results, the performance of a four-batch strategy is inferred.

2. Method

An assembly-level calculation was carried out on a standard 17×17 fuel assembly with 12.6 mm square pitch, 0.418 mm inner fuel pin radius and 0.475 mm outer fuel pin radius. There were 265 fuel rods per assembly and control rod guide tubes were assumed to be empty. The core equivalent diameter was 3767 mm and the active fuel height was 4200 mm (Pairot et al., 2011a). The core was assumed to be bare. The fuel temperature was 900 K and the coolant temperature 586 K. The clad was zircaloy. The reactor power was 30.4 MW t^{-1}.

Two TRU compositions were considered. The first case (Table 1) was 'dirty' Pu as considered by Rubbia et al. (1995) in their Energy Amplifier concept.

The second case (Table 2) contained more MAs and was taken from (Shwageraus et al., 2004).

The fuel was Th/TRU MOX assumed to have a uniform density of 9.8 g cm^{-3}, slightly less than the theoretical density, as is common practice. The fuel was assumed to be $MO_{1.98}$ to limit oxygen release on fission, again common practice.

At the end of each fuel cycle (defined as the point when the effective multiplication factor k_{eff} dropped to 1, to the nearest 60 days, rounded down), the fission products were removed and replaced with Th/TRU MOX. The reloading enrichment was typically 60–70 at%. This is the enrichment of dirty Pu in the dirty Pu/Th MOX added to the retained thorium and actinides in the reactor to replace burned isotopes during subsequent cycles. The fuel was assumed to be homogenous. In reality, in-core fuel management would be used to ensure adequate form-factors and reactivity. In practice, the time taken to reprocess the fuel may mean that the reloaded fuel is the output of a different reactor operating the same fuel cycle. The start-of-cycle (SOC) Th content of the fuel never dropped below 70 wt%. The starting enrichment was 13 at% for dirty Pu and 22 at% for dirty Pu/MA. The reload enrichments were selected on a cycle-by-cycle basis and are plotted in Fig. 1. The resulting SOC proportions of Th in the core each cycle are plotted in Fig. 2.

Table 1
Assumed dirty Pu composition (at%) (Rubbia et al., 1995).

Nuclide	Proportion (at%)	Nuclide	Proportion (at%)
^{237}Np	7.507	^{241}Am	0.486
^{239}Np	7.54×10^{-7}	^{242}Am(M)	4.38×10^{-3}
^{236}Pu	9.24×10^{-5}	^{243}Am	0.913
^{238}Pu	2.204	^{242}Cm	0.0489
^{239}Pu	52.988	^{243}Cm	7.21×10^{-4}
^{240}Pu	21.747	^{244}Cm	0.335
^{241}Pu	10.193	^{245}Cm	0.0204
^{242}Pu	3.55	^{246}Cm (not modelled)	2.29×10^{-3}

Table 2
Composition (wt%) of dirty Pu/MA fuel (Shwageraus et al., 2004).

Nuclide	Proportion (wt%)	Nuclide	Proportion (wt%)
^{234}U	0.0001	^{242}Pu	5.0330
^{235}U	0.0023	^{241}Am	4.6540
^{236}U	0.0019	^{242}Am(M)	0.0190
^{238}U	0.3247	^{243}Am	1.4720
^{237}Np	6.641	^{242}Cm	0.0000
^{238}Pu	2.7490	^{243}Cm	0.0050
^{239}Pu	48.6520	^{244}Cm	0.4960
^{240}Pu	22.9800	^{245}Cm	0.0380
^{241}Pu	6.9260	^{246}Cm (not modelled)	0.0060

The refuelling was assumed to be instantaneous. In reality, decay processes will occur, notably beta decay of ^{241}Pu and ^{233}Pa. This is not thought to greatly affect reactor performance, especially given the ability of the reactor to handle dirty Pu/MA fuel with high ^{241}Am content.

The void coefficient (VC), i.e. the increase in k_{eff} with void fraction (VF) at constant coolant temperature, was evaluated as:

$$VC = \frac{k_{eff,5\%VF} - k_{eff,0\%VF}}{5 k_{eff,5\%VF} k_{eff,0\%VF}} \quad (1)$$

The moderator temperature coefficient (MTC), i.e. the increase in k_{eff} with moderator temperature and corresponding density decrease, at constant VF, was evaluated as:

$$MTC = \frac{k_{eff,606\ K} - k_{eff,586\ K}}{20 k_{eff,606\ K} k_{eff,586\ K}} \quad (2)$$

The Doppler coefficient, i.e. the increase in k_{eff} with fuel temperature, was evaluated as:

$$DC = \frac{k_{eff,920\,K} - k_{eff,900\,K}}{20 k_{eff,920\,K} k_{eff,900\,K}} \quad (3)$$

The build-up of MAs has a tendency to make the core more reactive with 100% VF than at 0% VF. The constraint that the core is less reactive at 100% VF than at its operating condition was applied to the Japanese Reduced Moderation Water Reactor, which is a BWR (Fukaya et al., 2009). This constraint was not applied to the PWR considered here, as the condition of a PWR core entirely filled with steam without having been shut down was thought to be implausible. It was, however, illustrative to evaluate it.

The 100% VC was evaluated as:

$$VC_{100\%} = \frac{k_{eff,100\%VF} - k_{eff,0\%VF}}{100 k_{eff,100\%VF} k_{eff,0\%VF}} \quad (4)$$

This is an illustrative calculation to demonstrate the issues associated with high transuranic enrichment in PWRs. It is relevant for a very serious LOCA without scram, which is extremely unlikely but may be important from a regulatory standpoint.

In addition, in future work it would be worth considering the VC of a fully voided assembly surrounded by unvoided assemblies. This results in a more thermal neutron spectrum than a fully voided core (OECD Nuclear Energy Agency, 1995) but removes the increased radial neutron leakage from the voided assembly. Therefore this condition may be more or less limiting than the 100% VC case. While this scenario is not considered possible in PWRs (Takagi et al., 1997), it is desirable to avoid any possibility of local positive reactivity feedback that could lead to accidents. Ultimately, the stability of the reactor must be verified using a full core coupled neutronic–thermal-hydraulic model.

The calculation was carried out using the commercial reactor physics code WIMS 9 (Newton and Hutton, 2002) which uses data from JEF-2.2 (OECD Nuclear Energy Agency, 2000). The nuclear

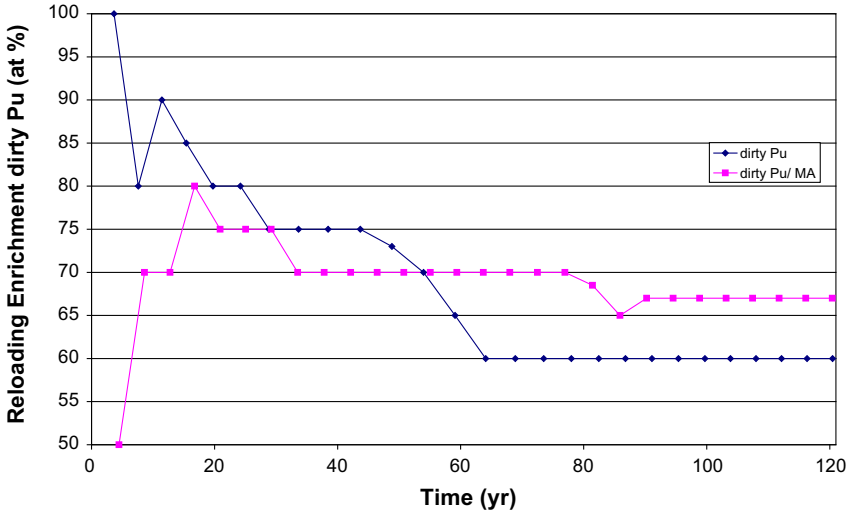

Fig. 1. Refuelling strategies.

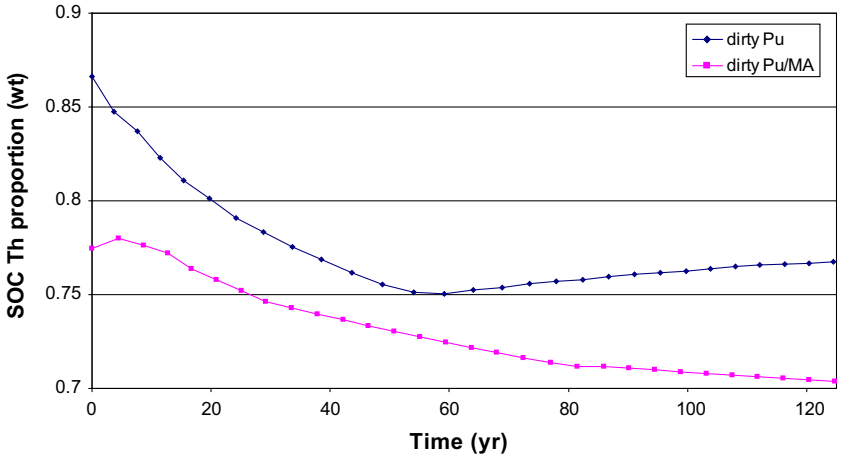

Fig. 2. SOC Th proportions.

data was benchmarked against single cell and 17 × 17 lattice calculations found in (International Atomic Energy Agency, 2003), giving results consistent with other calculations for Th–Pu fuels.

An approximate flux solution was determined using 172 neutron energy groups, and then the final solution was evaluated using l1 groups. A burn-up calculation was performed. The effect of buckling was evaluated using the module CRITIC. One limitation of this analysis is the accuracy of the data available for some MAs, leading to some variation among international benchmark calculations.

Results were benchmarked against a 3D MCNPX (Los Alamos National Laboratory, 2011) model with infinite radius and finite height to give equivalent geometric buckling, using data at 900 K from JEF-2.2 (OECD Nuclear Energy Agency, 2000). The standard deviation of the MCNPX calculations was ~0.0004. The fuel compositions were taken from SOC conditions after the specified number of years of operation. The isotope populations are given in Section 3.

Table 3 compares the WIMS and MCNPX benchmark calculations. The difference in k_{inf} and k_{eff} values between the two codes was initially negligible, but rose as the MA population rose in the

Table 3
WIMS and MCNPX benchmark calculations.

Case	$k_{inf,WIMS}$	$k_{inf,MCNPX}$	Relative error	$k_{eff,WIMS}$	$k_{eff,MCNPX}$	Relative error
12% dirty Pu, 0% VF	1.12688	1.12718	−0.00027	1.11308	1.11354	−0.00041
12% dirty Pu, 5% VF	1.11910	1.11886	0.00021	1.10468	1.10583	−0.00104
12% dirty Pu, 100% VF	0.95399	0.95391	0.00009	0.90183	0.90199	−0.00017
49 year dirty Pu, 0% VF	1.14440	1.14846	−0.00355	1.13112	1.13714	−0.00532
49 year dirty Pu, 5% VF	1.14a176	1.14654	−0.00419	1.12784	1.13283	−0.00442
49 year dirty Pu, 100% VF	1.18784	1.18507	0.00233	1.12367	1.12757	−0.00347
86 year dirty Pu/MA, 0% VF	1.10444	1.11207	−0.00691	1.09182	1.10033	−0.00779
86 year dirty Pu/MA, 5% VF	1.10294	1.11084	−0.00716	1.08970	1.09794	−0.00756
86 year dirty Pu/MA, 100% VF	1.20265	1.20198	0.00056	1.13982	1.14593	−0.00536

reactor. For high MA loadings, WIMS predicted the multiplication factor to be lower by about 0.7%. The VC did not appear to be significantly different. Therefore the WIMS calculation was deemed satisfactory for the purposes of this analysis.

Data for ^{246}Cm was available from a beta version of WIMS 10, so the effect of ^{246}Cm could be estimated. From the capture cross-section of ^{245}Cm, the ^{246}Cm population was expected to rise to around 0.5% after 60 years of operation and 0.8% after 120 years of operation. 0.5% ^{246}Cm reduced the reactivity in the 49 years dirty Pu case by ~0.2%, and 1% ^{246}Cm reduced the reactivity in the 86 years dirty Pu/MA case by ~0.3%. The VC became relatively more negative by ~2% per wt% ^{246}Cm. It therefore appears unlikely that ^{246}Cm or heavier isotopes will affect the feasibility of any of the models presented here, but it would be advisable to consider such isotopes in further work.

The soluble boron worth (SBW) was evaluated as:

$$\mathrm{SBW} = \frac{k_{\mathrm{inf},X} - k_{\mathrm{inf},0}}{X k_{\mathrm{inf},X} k_{\mathrm{inf},0}} \tag{5}$$

where X is the concentration of soluble boron in ppm, which was taken as 1000 ppm at SOC and 500 ppm at middle-of-cycle (MOC).

From Eq. (5), the concentration of soluble boron required to control the reaction at SOC could be calculated. In practice, burnable poisons would be used for partial reactivity control. This was then normalised against the concentration of soluble boron required to control a one-batch ^{238}U/4.5 wt% ^{235}U fuel.

For each fuel cycle of Th/Pu fuel, the cycle length was normalised against the cycle length of the same U reference case. The relative soluble boron required was calculated as the ratio of the normalised boron concentration to the normalised fuel cycle length.

So for example, if 1500 ppm of boron was required to set k_{eff} to 1 for a Th/Pu cycle and 1000 ppm was required for the U reference case, then the normalised soluble boron requirements would be 1500/1000 = 1.5. However, this does not take into account the relative fuel cycle length. So if the cycle length was 3.2 years for Th/Pu and 2 years for U (at the same power), then the Th/Pu cycle would be 1.6 times longer and the relative soluble boron required would be 1.5/1.6 = 0.9375. This methodology was used to indicate the relative controllability of the Th/Pu fuel. The SBW is instructive but does not account for the much lower starting k_{eff} of Th/Pu fuel. As the fuel cycle length varies throughout the analysis and is often much greater than that for the U reference calculation, a simple ratio of boron concentrations is misleading. Similarly the absolute amount of soluble boron required is misleading because in practice the core is not controlled using just boron and the fuel cycle length presented may differ from that of a final design. In the final design, the fuel cycle length would be set by specifying an enrichment to give a desired burn-up (which may be influenced by a need to limit Pu enrichment to improve the reactivity coefficients).

3. Incineration performance

The cumulative average incineration in kg (metal) per GW_{th} year is plotted in Fig. 3. The slight difference for the different fuels is probably a reflection of the higher TRU loading and hence higher fissile enrichment of the dirty Pu/MA case. The decrease over time is due to the increase in ^{233}U fissions as it accumulates in the reactor.

The initial incineration rate for the dirty Pu/MA case is 337 kg per GW_{th} year falling to 259 kg per GW_{th} year after 120 years; for the dirty Pu case the corresponding values are 328 and 249 kg per GW_{th} year. For comparison, in a 1500 MW_{th} Energy Amplifier, 420 kg of dirty Pu are incinerated per year (280 kg per GW_{th} year) (Rubbia et al., 1995). A 2-year fuel cycle has been proposed (Rubbia et al., 1997). The high moderation PWR approach proposed by Shwageraus et al. (2004) gives theoretical destruction rates of up to ~1000 kg per GW_e year (i.e. about 330 kg per GW_{th} year) not including the effect of TRUs bred from Th. Fuel cycles of up to 36 months were considered. However, only 75% Pu destruction was demonstrated in Pu–Th fuel, and this figure drops to 50% for Th–Pu–MA fuel. Further reprocessing was not considered.

Eliminating 3000 tonnes of TRU waste requires ~10,000 GW_{th} year of reactor operation (equivalent to 40 EPRs operating for 60 years each). A large number of reactors are therefore needed, so power generation should be economically competitive. In this context, the approach proposed here appears very competitive relative to using Energy Amplifiers, as the waste incineration rate is only slightly lower and the concept does not require new technology, beyond the ability to manufacture Th/dirty Pu MOX. It also appears to be beneficial to accept a ~25% decrease in the ultimate waste incineration rate in order to achieve complete destruction, instead of 50–75% destruction in a PWR with increased moderation. This choice is ultimately a policy decision.

The evolution of isotopes within the reactor at SOC is shown for both cases in Figs. 4–7. For the dirty Pu case, the TRU population

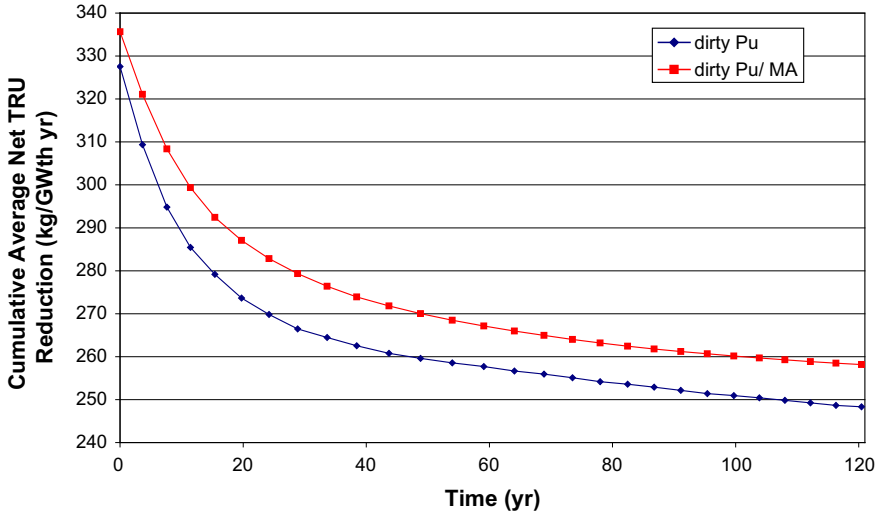

Fig. 3. Cumulative average TRU incineration.

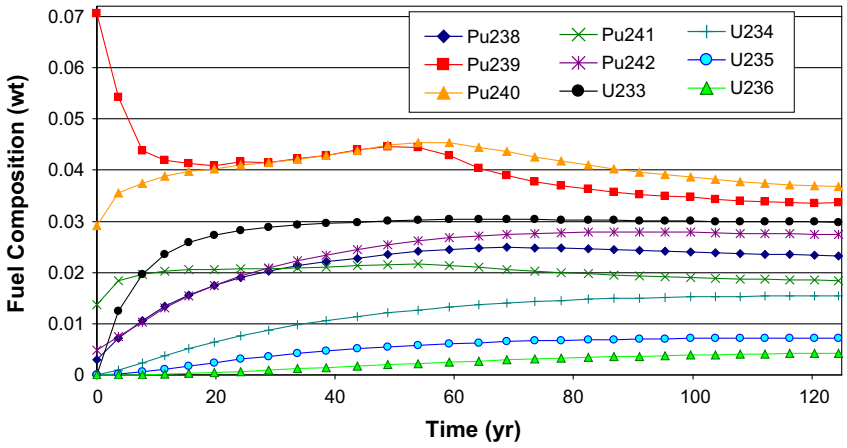

Fig. 4. Evolution of Pu and U isotope populations for the dirty Pu case.

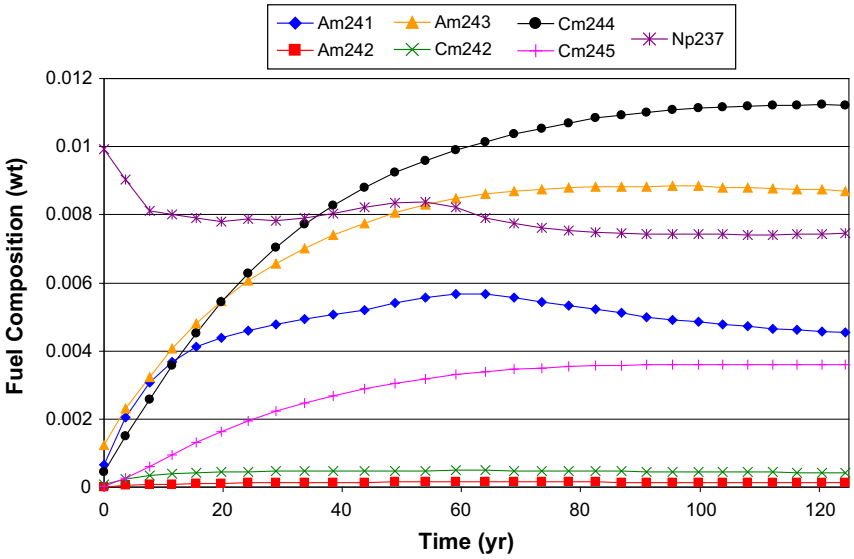

Fig. 5. Evolution of significant MA populations for the dirty Pu case.

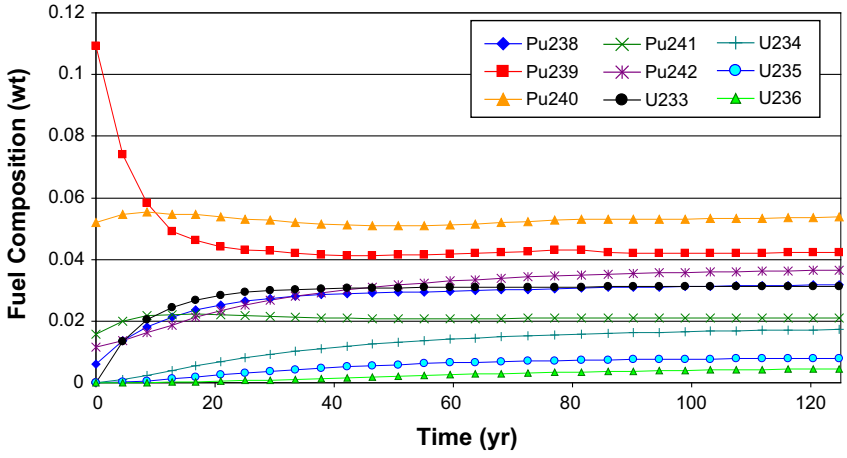

Fig. 6. Evolution of Pu and U isotope populations for the dirty Pu/MA case.

was 'ramped down' after 60 years of operation to demonstrate the feasibility of a lower stable operating point. For the dirty Pu/MA case, a higher equilibrium concentration was selected by increasing TRU loading. These two cases therefore indicate that an even-

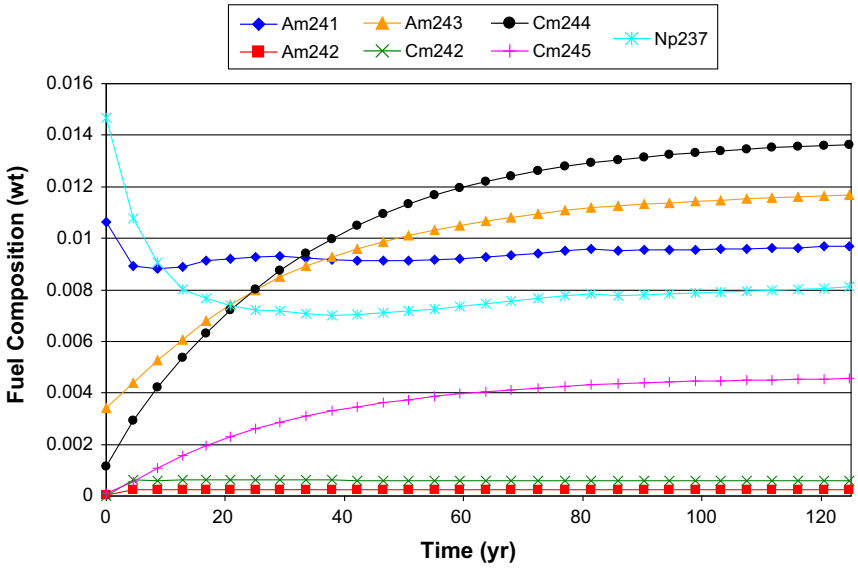

Fig. 7. Evolution of significant MA populations for the dirty Pu/MA case.

tual pseudo-equilibrium point can effectively be chosen by the operator without the required fissile Pu loading becoming unmanageably high.

One hundred and twenty five years of operation were needed to observe the MA populations tending to equilibrium. It takes a long time for these populations to accumulate through breeding of lighter isotopes. Notably, ^{244}Cm is bred via ^{243}Am and ^{242}Pu and takes a long time to accumulate. This then transmutes into ^{245}Cm, which is fissile and therefore the equilibrium population of ^{245}Cm is lower. Higher isotopes of Cm will be produced in the reactor. The effect of ^{246}Cm has been considered. Even in the thermal spectrum, higher isotopes ultimately decay or fission, and hence will eventually be expected to reach their own equilibrium populations. One possible source of concern is spontaneous fission of some heavier isotopes such as ^{248}Cm during reprocessing and fuel fabrication.

The equilibrium behaviour suggests that the long-term evolution of reactivity and isotope population is stable relative to earlier fuel management decisions. This is important because dirty Pu becomes less reactive over time in the reactor, so it appears plausible that overloading with dirty Pu at the start of operation could require higher fissile loading later on to compensate, and subsequent escalation until the Pu enrichment becomes unmanageably high or criticality is unobtainable. The equilibrium position appears to be obtainable because of a combination of factors: the build-up of ^{233}U; a harder neutron spectrum due to high Pu loading; and increased resonance self-shielding of MAs as they accumulate, reducing their detrimental effect on reactivity.

From the equilibrium position, continued loading of TRUs is possible indefinitely: in the reactor and in identical successors. TRUs would only need to be stored in the event that fission-based nuclear power is phased out (hence *near*-complete incineration is claimed). The TRU proportion in the reactor can be ramped-down further in this eventuality, although this decreases burn-up. While the final TRU load of an Energy Amplifier would be lower, it would not be zero (Coates and Parks, 2010). There will also inevitably be some TRU losses during reprocessing.

4. Reactivity and control

The one-batch end-of-cycle (EOC) burn-up was calculated for each fuel cycle (Fig. 8). This was dependent on the TRU enrichment of the fresh fuel. In practice, this may be constrained by reactivity control considerations (see below). However, neglecting control requirements, the one-batch burn-ups were generally between 40 and 55 GWd t^{-1}. Given that modern reactors achieve 55–65 GWd t^{-1} and often operate a four-batch strategy (Ardron, 2010), these figures are encouraging. If reactivity varies linearly with burn-up (without neutron poisons), operating a four-batch strategy improves burn-up by 60% (Driscoll et al., 1990). Therefore, from a reactivity point of view, burn-up can be competitive with the burn-up achieved in existing reactors. For a given TRU enrichment, the burn-up is generally lower for the dirty Pu/MA loading. The variation shown here is necessarily a reflection on the different reloading strategies used in each case. Notably, the dirty Pu case was driven to a different equilibrium position after 60 years of operation.

The SBW is reduced relative to a U-fuelled reactor (Table 4). The value is generally similar at SOC and MOC. However, the reactivity decreases much less rapidly over the fuel cycle. This is consistent with the findings of Shwageraus et al. (2004) and due to a greater proportion of power being produced from bred isotopes (most significantly ^{233}U). In addition, when pseudo-equilibrium is established, the population of isotopes such as ^{240}Pu, which has a high capture cross-section, decreases over the fuel cycle. These effects result in the relative soluble boron requirement to control the reactor being initially lower, and then similar to a U-fuelled reactor after 10–20 years (Fig. 9). Fig. 9 was produced using the methodology described at the end of Section 2 and shows in effect the relative proportion of SOC excess reactivity that is suppressed by 1000 ppm of soluble boron in the Th/Pu cycle relative to the reference U cycle. The soluble boron requirement is lower with MA loading. This is thought to be in part due to the increased loading of ^{240}Pu.

In Table 4, calculated values of the soluble boron required to entirely control the reactor for a one-batch cycle are displayed. These values are far higher than those implemented in practice and would be impractical. In addition, at higher soluble boron concentrations the SBW is reduced and therefore the calculated values would be underestimated. In practice, a four-batch strategy reduces the SOC reactivity and therefore the soluble boron requirements, and burnable-poisons (and possibly control rods) are used for partial reactivity control. In particular, the value for dirty Pu after 54 years is very high due to the very long cycle length (and

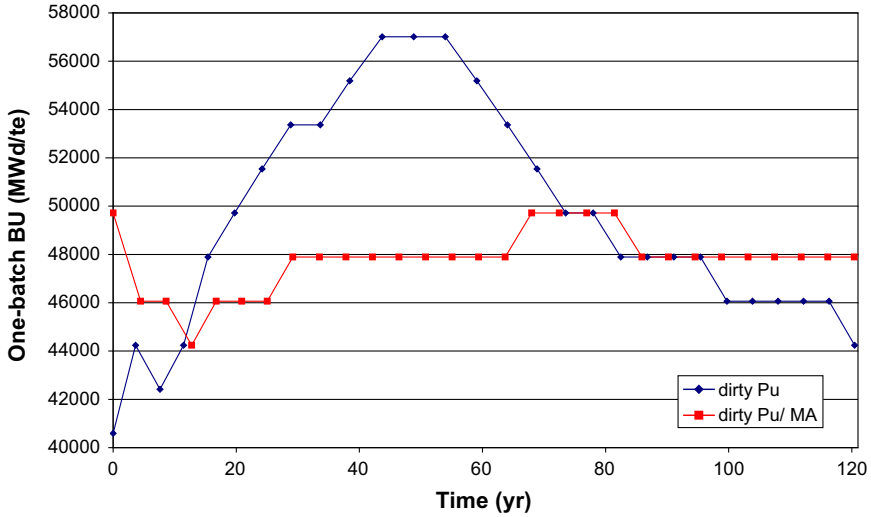

Fig. 8. EOC burn-up variation for the two cases.

Table 4
Solution boron worth (note that SOC soluble boron will be high due to one-batch operation).

Case	SBW at SOC (pcm ppm^{-1})	SBW at MOC (pcm ppm^{-1})	Cycle length (year)	SOC soluble boron (ppm)
^{238}U/4.5 at% ^{235}U	−5.53		3.2	6523
Dirty Pu year 0	−1.79	−1.90	3.7	6731
Dirty Pu year 54	−1.10	−1.22	5.1	11,773
Dirty Pu year 120	−1.25	−1.36	4.0	8409
Dirty Pu/MA year 0	−1.28	−1.29	4.5	5587
Dirty Pu/MA year 51	−1.08	−1.16	4.3	8722
Dirty Pu/MA year 120	−1.02	−1.09	4.3	8882

hence high burn-up) of this fuel cycle. These soluble boron concentrations would also result in a positive MTC for U (Pairot et al., 2011b) and Th/Pu fuels. It can, however, be seen that for many of the cases in Table 4 the soluble boron requirements are roughly comparable to the U reference case. The SBW alone is not particularly illustrative due to the much larger starting k_{eff} for low enriched U fuel compared to Th/Pu fuel.

Enriched boron is typically used in an EPR to limit the critical boron concentration (with burnable poisons but without control rods) to ~1400 ppm. For U fuel, the critical natural boron concentration is 2195 ppm and is slightly larger for MOX (Pairot et al., 2011b).

5. Reactivity coefficients

The VC, 100% VC and MTC for both fuel loads at EOC are plotted in Figs. 10 and 11. The VC and MTC remain significantly negative throughout, so stable operation can be maintained. The SOC VC for the reference case was −98 pcm per% void and the SOC MTC was −50 pcm K^{-1}. Adding dirty Pu resulted in a less negative VC and MTC.

Adding MAs was found to make these coefficients worse, which is consistent with the findings of Fukaya et al. (2009). In particular, the 100% VC becomes positive for the dirty Pu/MA loading case after ~20 years and is nearly zero at one point for dirty Pu. This may be the result of the high ^{241}Am content of the dirty Pu/MA

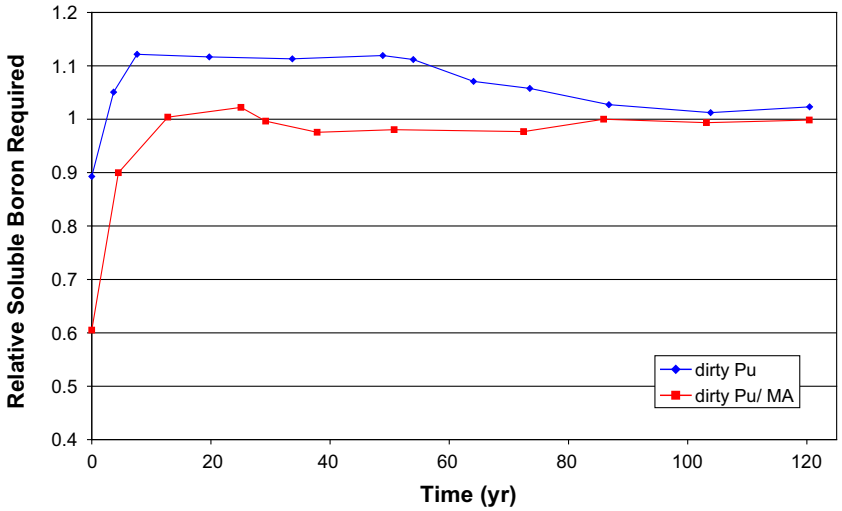

Fig. 9. Relative soluble boron required to control reactivity.

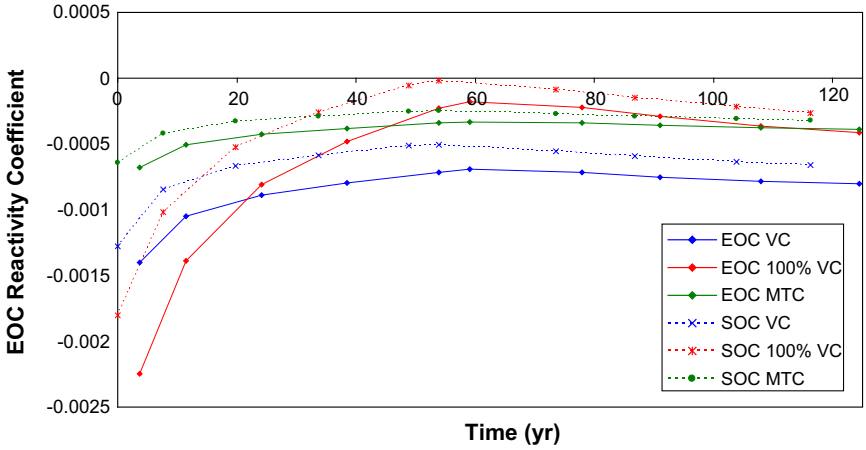

Fig. 10. Moderator reactivity coefficients in the dirty Pu case.

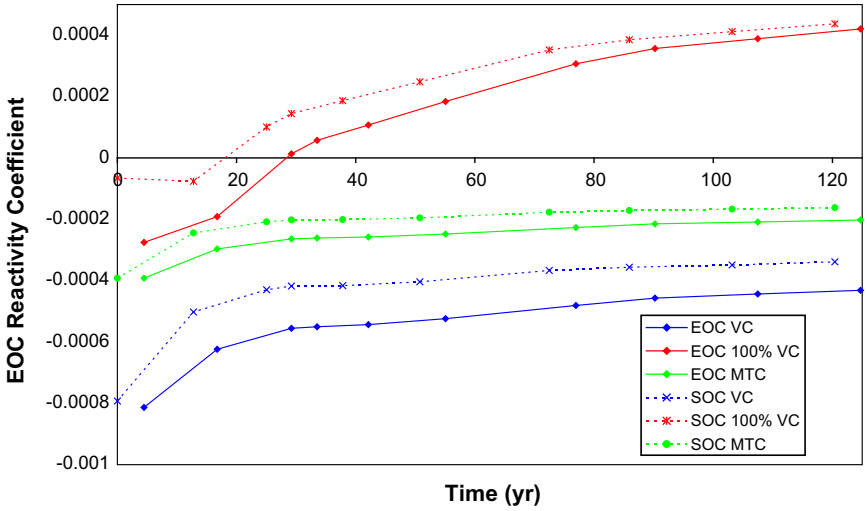

Fig. 11. Moderator reactivity coefficients in the dirty Pu/MA case.

fuel. This is not thought to represent a safety risk, but may require attention from a regulatory perspective. The reactivity coefficients can be improved by reducing the Pu enrichment in the fuel. As the burn-up of the fuel cycles presented here is generally high, an acceptable burn-up should be achievable with lower Pu enrichment. In addition, it may be advisable to limit the average MA content of the loaded dirty Pu across the core in order to keep the 100% VC negative.

Th fuel is known to improve stability relative to U fuel (Downar and Xu, 2001). ^{233}U is fissile in the thermal spectrum and hence improves stability. The VC, 100% VC and MTC were all generally more positive (less negative) at SOC than at EOC. The leakage effect (Fukaya et al., 2009) was necessary for the 100% VC to be negative in the dirty Pu case, but VC and MTC were always negative in both cases for zero buckling. The MTC was similar to that of a U-fuelled reactor (Shwageraus et al., 2004). Without the radial leakage effect, the 100% VC is more positive by ~29 pcm, which would result in a positive 100% VC for the dirty Pu case after about 30 years. This is indicative of a positive 100% VC for a fully voided assembly surrounded by unvoided assemblies. However, in this case the neutron spectrum in the voided assembly would be more thermal which would improve the 100% VC. A full consideration of this scenario is beyond the scope of this assembly level analysis.

If a core was loaded with a mix of dirty Pu and dirty Pu/MA fuel it would be very sensible to place assemblies with MAs closer to the edge of the core to improve the reactivity coefficients. A similar argument applies to fresh fuel and fuel which has been in the reactor for multiple cycles (and/or recycles). Work on the in-core fuel management of these concepts will be performed to determine acceptable loading patterns. A coupled neutronic–thermal-hydraulic analysis of the full core will be performed to further examine issues related to high VFs both locally and during a severe LOCA.

These coefficients were calculated assuming zero soluble boron. A reduction of coolant density also reduces the quantity of dissolved boron and the SBW, and hence makes the VC and MTC worse. This makes the 100% VC coefficient positive at some points for the dirty Pu case. The maximum SOC soluble boron was calculated as

$$SB_{SOCmax} = \frac{5\% \times VC \times k_{eff,5\%VF} \times k_{eff,0\%VF}}{\Delta SBW} \quad (6)$$

where ΔSBW denotes the change in SBW. An equivalent calculation was performed for the MTC.

Fig. 12 shows that the amount of boron that can be added without the VC or MTC going positive (solid lines) is at some points less than the amount required to completely control the reactivity (broken lines) for both fuels. This assumes that a four-batch strategy would reduce soluble boron requirements by 60% relative to a one-batch strategy. For the dirty Pu case, this only occurs over a

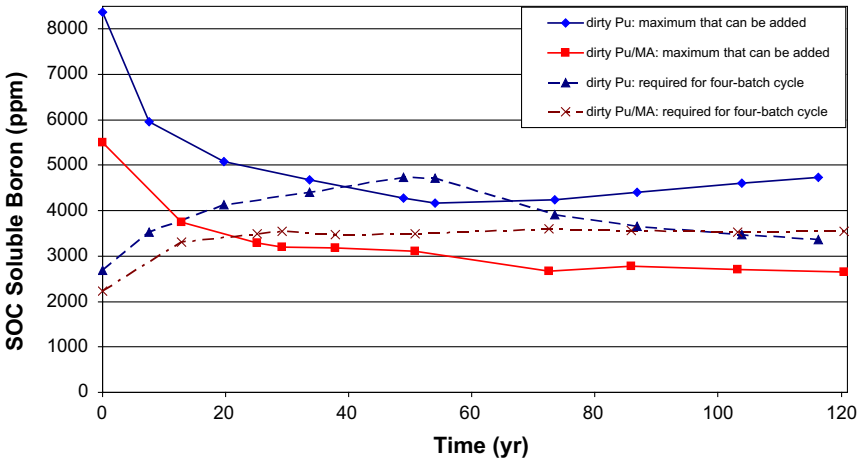

Fig. 12. Maximum soluble boron for negative VC and MTC and soluble boron requirements for the two cases.

few cycles with high four-batch burn-up (e.g. 91 GWd t^{-1} for the cycle at 54 years).

The amount of soluble boron which can be added is more limiting in the dirty Pu/MA case due to the more positive reactivity coefficients: the required amount slightly exceeds the maximum from 20 years of operation onwards. For the 120 year cycle in the dirty Pu/MA case, the required soluble boron would be 3553 ppm, more than the maximum permissible 2693 ppm. A four-batch cycle at this operating point would result in an EOC burn-up of ~76.6 GWd t^{-1}.

These boron concentrations are higher than those typically used in PWRs so the reactivity coefficient limit on soluble boron concentration is unlikely to cause problems with reactivity control. In practice, there will therefore be a substantial margin between the soluble boron concentration in the core and the limiting value.

In an EPR, use of burnable poisons is necessary to limit boron concentration to prevent a positive void coefficient (Pairot et al., 2011b), whereas this is not the case for some of the fuel cycles presented here. The limiting boron concentration is generally higher than with natural U fuel, which follows from the reduced SBW. Control rods can also be used to control reactivity, if necessary.

The use of control rod materials with high resonance integrals, such as silver [see, for example, JENDL-4.0 (Shibata et al., 2011)], may be advisable.

The Doppler coefficient was between −2.2 pcm K^{-1} and −2.9 pcm K^{-1} for both fuel loads.

6. Conclusion

This study shows that rapid and near-complete TRU incineration appears possible in a PWR such as an EPR or AP1000. Incinerating low quality waste appears feasible. Thorium can be introduced into the nuclear fuel cycle to improve fuel sustainability. Incineration rates are comparable to fast or subcritical reactors, and new coolant and materials technology is not required, significantly reducing the commercial risk and the timeframe for implementation. In addition to potentially ending the need for storing TRU waste, this fuel cycle may potentially allow burn-up to be increased. This analysis is limited to the consideration of a single fuel assembly, and it is necessary to perform a full core coupled neutronic–thermal-hydraulic analysis to determine if the design in its current form is feasible. In particular, the potential for positive void coefficients if the core is highly or locally voided is a cause for concern and further work is needed. The in-core and out-of-core fuel management of the concept will also be examined in detail to determine whether feasible, practical core designs can be achieved. However, the results presented here provide an extremely compelling case for further work, to investigate whether the concept is safe, feasible and economically viable. This work is ongoing at Cambridge.

Acknowledgement

We would like to thank Dr Matthew Eaton at Imperial College London for providing access to WIMS and for his guidance and support.

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