Th/^{233}U multi-recycle in pressurised water reactors: Feasibility study of multiple homogeneous/heterogeneous assembly designs

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Abstract

The objective of this study is to evaluate the potential of the Th/^{233}U fuel multi-recycle in current LWR, focusing on pressurised water reactors. In this work, approaches for ensuring a sustainable multi-recycle without the need for external source of make-up fissile material have been investigated. The intent is to achieve a design that allows existing pressurised water reactors (PWR) to be used with minimal modifications. Because this is considered a difficult proposition, the potential modifications that would be needed for PWR to ensure a sustainable multi-recycle system have been investigated and characterised. Additionally, the implications of the use of thorium on the LWR fuel cycle have been discussed.

Preliminary studies have been conducted. The investigation on homogeneous assembly designs led to the conclusion that a sustainable multi-recycling of Th/^{233}U oxide fuel cannot be achieved with a homogeneous PWR assembly within the parameter space of ^{233}U content in the fuel (2-4%) and the moderation ratio (2.0-0.7) obtained by varying the fuel pin size. Various heterogeneous designs have also been investigated with the seed unit (SU) to blanket unit (BU) ratio varied. It was found that designs with lower specific power densities tend to better meet the requirements of the Th/^{233}U multi-recycle. A 17 × 17 heterogeneous design with de-rated specific power density was selected as an optimum point design.

The modelling results for multiple homogeneous and heterogeneous assembly designs are presented. Discussions on the fuel cycle design characteristics, including assembly design details, fuel cycle lengths, material flow, decay heat, radiotoxicity, etc., are provided.

Introduction

The use of thorium in current or advanced light water reactors (LWR) has been of interest in recent years. These interests have been associated with the need to increase nuclear fuel resources and the perceived non-proliferation advantages of the utilisation of thorium in the fuel cycle [1]. Various options have been considered for the use of thorium in the LWR fuel cycle [2-5]. The possibility for thorium utilisation in multi-recycle system has also been considered [6], primarily because of the potential for near breeders with Th/^{233}U in the thermal energy range.

The objective of this study is to evaluate the potential of the Th/^{233}U fuel multi-recycle in current LWR, focusing on pressurised water reactors. The design requirements imposed on this study are:

- Fuel cycle length of 1.5 years ideally, but 1 year is acceptable.
- A sustainable fuel cycle (breeding must be achieved so that no external fissile material is needed within the fuel cycle). Clearly, for the transition or start-up cycle using thorium fuels, an external source of fissile material would be required.

The focus in the current paper is to explore a number of assembly designs, both homogeneous and heterogeneous, on their feasibilities towards Th/^{233}U multi-recycle in PWR and to derive a point design that can achieve the above stated goals. This paper also aims to provide fuel cycle impact data associated with the derived point design.

Calculation methods

In this study, unit assembly calculations with the WIMS-9 code [7] were used with 172 energy groups based on the JEF-2.2 library to estimate the performance of the PWR core. The core leakage was assumed to be ~3.0%.

Fuel cycle impact data for the 17×17 heterogeneous assembly design are also compiled. These include masses, radiotoxicity and decay heat. ORIGEN-2 [8] calculations were used to generate the pertinent mass flow data. A two-step process was utilised: the assembly lattice calculation was done using the WIMS-9 code to generate one-group cross-sections and fluxes (along with the other core parameters), and the ORIGEN-2 depletion calculations were done using the one-group fluxes and the effective cross-sections (which override base one-group cross-sections on ORIGEN-2 libraries). It needs to be pointed out that separate ORIGEN-2 calculations were run with the seed unit mass isotopic data and the blanket unit mass isotopic data. The results for the radiotoxicity, decay heat are then obtained by averaging the results from both units according to their heavy metal weight ratio in the core.

Investigations of homogeneous assembly designs

Fuel assembly level data for various homogeneous assemblies have been calculated for the infinite core multiplication factor (k-inf) and the fissile inventory ratio (FIR). The FIR is defined as the ratio of fissile isotopic mass at discharge over fissile isotopic mass at charge state. The FIR is used in this study as an indicator of conversion ratio. Nuclei considered as fissile in the FIR calculations are ^{233}U, ^{235}U, ^{239}Pu, ^{241}Pu and ^{233}Pa (which is counted as potential ^{233}U due to its short half-life: ~27 days).

The Th/^{233}U fuel cycle performance parameters (such as k-inf, cycle length and FIR) in a homogeneous PWR assembly were found to be dependent on the ^{233}U content and MR (moderator- to-fuel volume ratio). Increasing the initial ^{233}U content helps to improve k-inf but decreases the FIR. Decreasing the MR helps to improve the FIR but decreases k-inf. This study led to the conclusion that a sustainable multi-recycle with Th/^{233}U oxide fuel cannot be achieved with a homogeneous PWR assembly design within the parameter space of ^{233}U content (2-4%) and MR (2.0-0.7) by increasing the fuel pin size. The estimated discharge burn-up of a self-sustainable

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assembly design from the homogeneous scenario is less than 5 GWd/t, which is too small. It is necessary, therefore, that a heterogeneous assembly model be considered to obtain a feasible point design to realise a self-sustainable Th/^{233}U fuel multi-recycle scheme.

Investigations of heterogeneous assembly designs and the 17 × 17 heterogeneous assembly design

In order to derive an optimum assembly design to best achieve the goal of self-sustainable Th/^{233}U fuel cycle, various heterogeneous assembly designs were studied. A systematic investigation of the parameter space was carried out by varying each of the parameters in order to observe the trend leading to the final selection of the optimum heterogeneous assembly design. The parameters considered are the followings: seed unit and blanket unit fuel pin sizes, initial ^{233}U content in the seed (blanket fuel contains ThO_2 only) and seed unit over blanket unit (SU/BU) ratio. Assembly level performance parameters, k-inf and FIR, were evaluated for each heterogeneous design investigated. It was found that designs with lower SU/BU ratio and larger seed unit and blanket unit pin size tend to better meet the requirements of the Th/^{233}U multi-recycle. A 17 × 17 heterogeneous assembly design was selected as the optimum point design for this study as the result of this systematic parameter search.

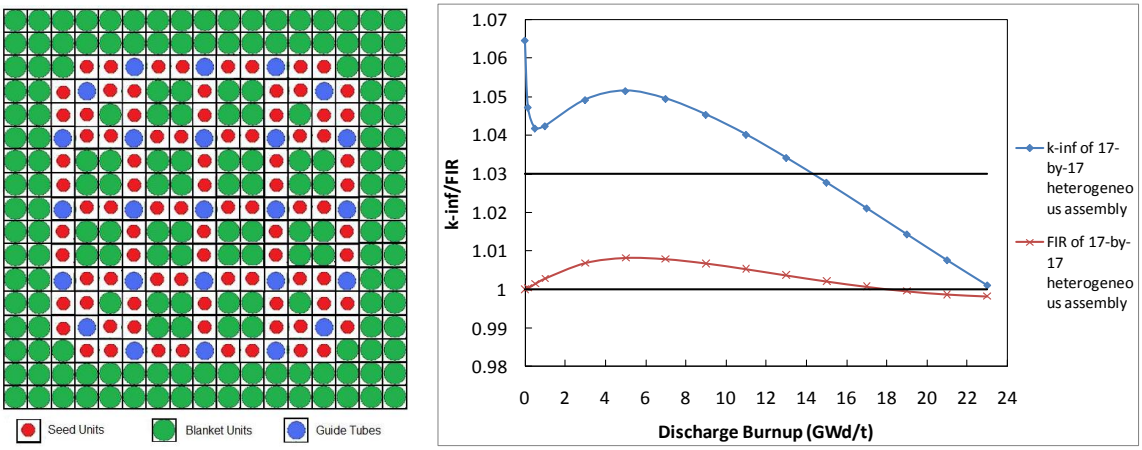
Design parameters for the 17 × 17 heterogeneous design are provided in Table 1. Figures 1(a) and 1(b) show the schematic drawing of the 17 × 17 heterogeneous assembly design and the kinf and FIR versus burn-up, respectively. The solid lines in Figure 1(b) represent the 1.0 cut-off for the FIR (when FIR falls below the cut-off, 1.0, the reactor is no longer breeding) and the leakage cut-off, which is 1.03 (assuming 3% neutron leakage/losses for the PWR).

Table 1: Design parameters for 17 × 17 heterogeneous Th/^{233}U fuel cycle analysis

Parameter	Value	
Assembly geometry	17 × 17	
Core power, MWt	1 000	
Number of assemblies	193	
Capacity factor, %	90	
Number of batches	3	
Active core height, cm	366	
Number of seed fuel rods per assembly	88	
Number of blanket fuel rods per assembly	176	
Assembly gap, cm	0.04	
Fuel pitch, cm	1.26	
Assembly pitch, cm (including water gap)	21.5	
Seed fuel composition	(Th/^{233}U)O_2 (7.8 wt.% ^{233}U)	
Blanket fuel composition	ThO_2	
Seed fuel pellet density, g/cm^3	9.6	
Blanket fuel pellet density, g/cm^3	9.5	
Cladding material	Zr-4	
Cladding density, g/cm^3	6.5	
Moderator to fuel volume ratio (MR)	0.7 (seed region)	0.3 (blanket region)
Fuel pellet radius, cm	0.51 (seed region)	0.57 (blanket region)
Heavy metal per assembly, kg	770.33 (221.46 kg in seed and 548.86 kg in blanket)	
Specific power density, W/g	6.73	

Figure 1(a): Schematic drawing of 17 × 17 heterogeneous assembly design and (b) k-inf and FIR versus burn-up

(a) (b)



Both the assembly pitch and the fuel pitch are similar to those of current PWRs (21.5 cm and 1.26 cm respectively) in order to reduce the modifications of the assembly to a minimum level. The discharge burn-up, estimated by a simplified point reactivity model assuming a three-batch fuel management scheme, is about 18 GWd/t. This corresponds to a fuel cycle length of about 990 days (assuming a 0.9 capacity factor).

There are three attributes of this design that deserve some discussions. Firstly, the core contains higher amount of fuel material than the current PWR. The core power has to be de-rated in order to match the linear power density of the current PWRs at BOL (due to the initial distribution of fissile ^{233}U in seed units only). As a result, the specific power density of this design is significantly lower than that of the PWR. Secondly, the MR values are low in both the seed and the blanket regions. This raises concerns for thermal-hydraulic properties of this specific assembly design. The low MR values are results of large seed and blanket fuel pins and short distances between fuel pins. Finally, the normalised neutron spectra of both the seed region and the blanket region were examined. The spectra are harder than the spectra of standard thermal systems both at the BOL and at the EOL (for the Th/^{233}U system). This indicates that in order to achieve high conversion of ^{232}Th, a hard neutron spectrum is necessary.

Table 2 presents the WIMS-9 assembly level fuel cycle mass flow data for the Th/^{233}U system. It is noted that the production of plutonium (Pu) and other minor actinides (MA) is extremely low, indicating a probable benefit of implementing the Th fuel cycle.

Th/^{233}U multi-recycle fuel cycle impact: Radiotoxicity

A variety of measures are available for quantifying the radiotoxicity of used nuclear fuel. In this work, the radiotoxicity data has been calculated using the ingestion dose coefficients obtained from the ICRP 72 database [9]. The used nuclear fuel (UNF) radiotoxicity values for the standard UOX system (the reference PWR system) and the Th/^{233}U system were estimated up to 10 million years after discharge and compared. For the purpose of comparison, the UNF radiotoxicity is normalised to unit electricity generated in one year (Sv/GWe-yr) and then normalised again to the radiotoxicity of the natural uranium ore that is needed to produce the low-enriched uranium fuel for the reference standard PWR UOX fuel cycle.

Figure 2(a) shows the normalised radiotoxicity for the standard UOX system (50 GWd/t discharge burn-up) broken down by the leading contributors to the radiotoxicity. It can be seen that from 10 years after discharge to about 100 years after discharge, fission products (FP) and TRU are the main contributors to the total radiotoxicity. The contribution from the fission

Table 2: Mass flow data for 17 × 17 heterogeneous Th/^{233}U Fuel (kg/assembly)

Nuclide	Charge	Discharge (18 GWd/t discharge burn-up)
^{232}Th	753.16	737.02
^{233}Pa	0	0.22
^{233}U	17.17	16.70
^{234}U	0	1.64
^{235}U	0	0.25
^{236}U	0	1.53E-02
^{237}U	0	1.30E-05
^{238}U	0	1.83E-06
^{237}Np	0	8.67E-04
^{239}Np	0	4.68E-10
^{238}Pu	0	1.17E-04
^{239}Pu	0	5.77E-06
^{240}Pu	0	5.16E-07
^{241}Pu	0	2.60E-07
^{242}Pu	0	1.65E-08
^{241}Am	0	9.60E-09
^{242m}Am	0	1.07E-10
^{243}Am	0	1.34E-09
^{242}Cm	0	7.00E-10
^{243}Cm	0	7.41E-12
^{244}Cm	0	1.27E-10
^{245}Cm	0	5.02E-12
Total	770.33	755.86

* Fission product mass at discharge is 14.4 kg/assembly.

products to the radiotoxicity decreases sharply between 100 and 1 000 years due to their natural decay. The TRU remains the main contributor to the radiotoxicity up to about a few hundred thousand years. It can also be observed that contribution from uranium isotopes is consistently low throughout the time range investigated.

Figure 2(b) shows the UNF normalised radiotoxicity for the Th/^{233}U system, again broken down to indicate the leading contributors. This figure contains data for all the nuclear elements in the UNF (i.e. no assumption on which elements are removed during fuel separations) and are provided in this form for illustration. (For comparison to the UOX system in Figure 3, the actual elements and separations losses assumed to be going to the repository are included.) It can be seen that from 10 years to about 100 years after discharge, FP and uranium isotopes are the main contributors to the total radiotoxicity. The nuclide ^{232}U is the main contributor within the uranium isotopes. The FP contribution to the radiotoxicity decreases sharply between 100 and 1 000 years. Contribution from the uranium isotopes then decreases and ^{233}U decays gradually into ^{229}Th, which becomes the main contributor from 1 000 years to about a million years. The TRU, different from the earlier scenario of standard UOX fuel cycle, contributes very small amount to the total radiotoxicity. This is because in the Th/^{233}U cycle, very small amount of TRU are produced. It should be pointed out that although ^{210}Po, decay daughter of ^{234}U, is present in the system only in very small amount, its contribution cannot be ignored due to its very high ingestion coefficient.

Figure 3 compares the total high-level waste (HLW) radiotoxicity of the standard UOX fuel system and the Th/^{233}U fuel system (18 GWd/t discharge burn-up). For the UOX system, the UNF is assumed sent to the repository as HLW. In the Th/^{233}U system, only the nuclear elements not recycled are assumed to be sent as HLW to the repository. These include the fission products, all non-uranium and non-thorium heavy elements, and 0.1% of the discharged uranium and thorium mass, which is assumed as system-loss during fuel separations. It is assumed that separation is done five years after discharge. The results indicate that the normalised radiotoxicity of the Th/^{233}U system is slightly lower than that of the UOX system from after separations.

Figure 2: Normalised radiotoxicity for (a) standard UOX fuel cycle (50 GWd/t discharge burn-up) and (b) 17 × 17 heterogeneous assembly design (18 GWd/t discharge burn-up)

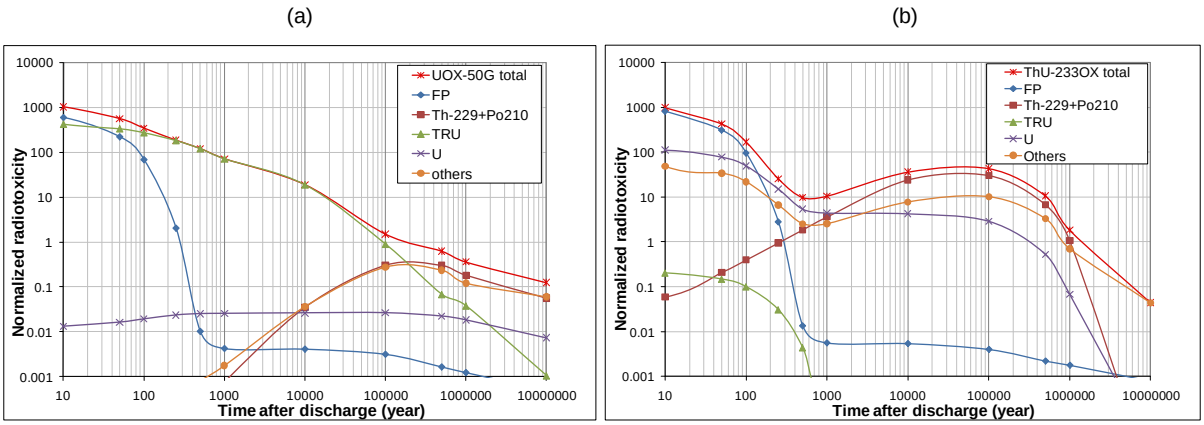
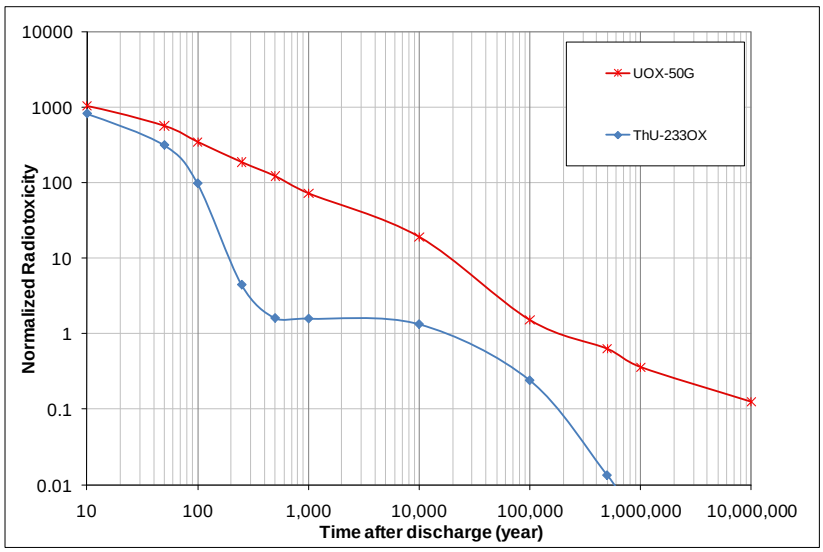


Figure 3: Comparison of normalised radiotoxicity values



After approximately 50 years, due to the natural decay of the fission products and the removal of 99.9% of Th and U isotopes in separation, the normalised radiotoxicity of the Th/^{233}U system decreases sharply. The radiotoxicity of the standard UOX system drops below natural uranium level in about 200 000 years, while the radiotoxicity of the Th/^{233}U system drops below natural uranium level in about 20 000 years.

Th/^{233}U multi-recycle fuel cycle impact: Decay heat

The decay heat of both the standard UOX system and the Th/^{233}U system have been assessed in this study as well. Figure 4(a) shows the decay heat of the standard UOX system (50 GWd/t discharge burn-up) broken down by the leading contributors to the decay heat. The decay heat is normalised to unit electricity generated in one year (W/GWe-yr). As for the UNF decay heat, it can be observed that from 10 years after discharge to about 100 years after discharge the FP and TRU are the main contributors to the total decay heat. The FP contribution to the decay heat decreases sharply between 100 and 500 years due to their natural decay. Contribution from TRU remains the primary contributor to the decay heat up to about 200 000 years. It can also be observed that contribution from uranium isotopes is consistently low throughout the time range.

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Figure 4: Decay heat curves for (a) standard UOX fuel cycle (50 GWd/t discharge burn-up) and (b) 17 × 17 heterogeneous assembly design (18 GWd/t discharge burn-up)

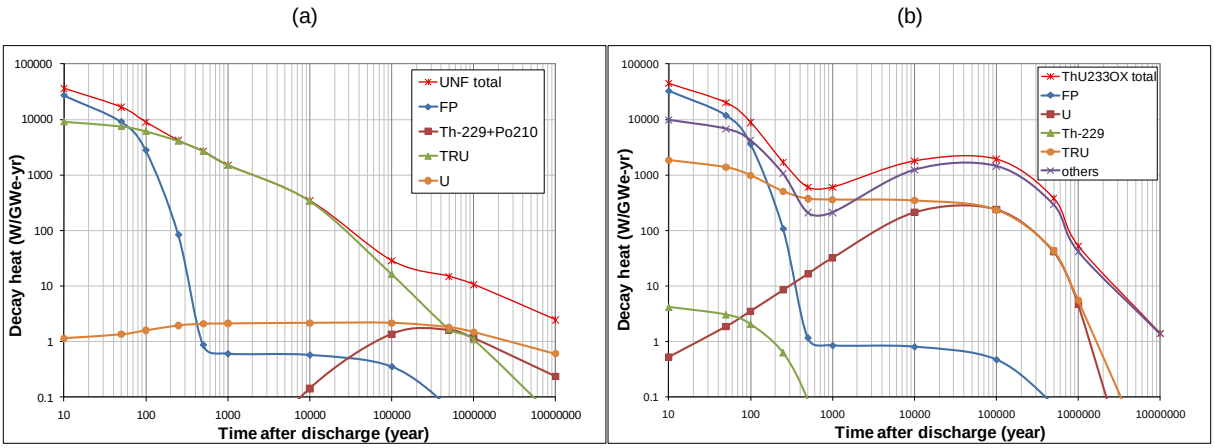

Figure 4(b) shows the normalised decay heat for the Th/^{233}U system, again broken down to indicate the leading contributors. This figure includes all the nuclear elements in the UNF, without consideration for the elements removed during fuel separations (provided in this form for illustrative purposes). From 10 years after discharge to about 100 years after discharge, FP is the main contributor to the total decay heat. The contribution of decay heat from FP decreases sharply between 100 and 300 years due to their natural decay. Contribution from the uranium isotopes decreases and ^{233}U decays gradually into ^{229}Th, which becomes the main contributor from 1 000 years to about a million years.

The TRU, different from the earlier scenario of standard UOX fuel cycle, contributes very small amount to the total decay heat. This is because in the Th/^{233}U cycle, very small amount of TRU are produced. It is noted that other isotopes are the main contributors of the decay heat as well for the whole time range covered. The contribution is mainly from decay daughters of Th and U isotopes, for instance, ^{225}Ac, which is a beta decay daughter of ^{225}Ra (beta decay half-life: 14.9 days), the alpha decay daughter of ^{229}Th (alpha decay half-life: 7 340 years).

Figure 5: Comparison of decay heat values

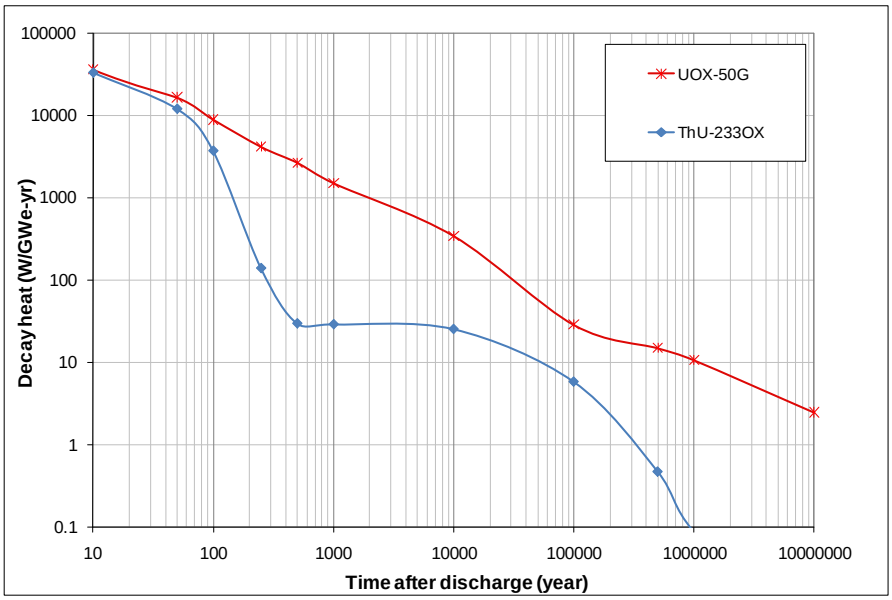

Figure 5 presents the calculation result for normalised HLW decay heat of the Th/^{233}U system and compares it to the standard UOX fuel cycle data as well. The time scale is in the unit of years after discharge and it is assumed that the separation of the Th/^{233}U system UNF is done five years after discharge. For the UOX system, the UNF is assumed sent to the repository as HLW. In the Th/^{233}U system, only the nuclear elements not recycled are assumed to be sent as HLW to the repository.

Similar to the trend of radiotoxicity, it is observed that compared to the standard UOX system the HLW decay heat of the Th/^{233}U system is slightly lower following UNF separations. After approximately 50 years, the decay heat of the Th/^{233}U system decreases sharply due to natural decay of the fission products, which are the main decay heat contributors in the short term. The trend after 50 years is due to the recycle of 99.9% of the Th and U isotopes and the low production of transuranic elements in the Th/^{233}U system.

Conclusions

Investigations have been conducted to assess the feasibility of using both homogeneous and heterogeneous PWR assemblies to achieve Th/^{233}U fuel multi-recycle. The conclusions are summarised as follows:

- A practical sustainable fuel cycle cannot be achieved with a homogeneous PWR assembly within the parameter space of initial ^{233}U content and reasonable moderator to fuel volume ratio obtained by varying the fuel pin size.
- A 17 × 17 heterogeneous assembly design achieved the following attributes and met the requirements defined in this study for sustainable Th/^{233}U multi-recycle:
  - A de-rated core power of 1 000 MWt.
  - A 2.7-year cycle length with sustainable fissile inventory.
  - Discharge burn-up of 18 GWd/t, which is significantly lower than the burn-up of current PWR.
  - The Th/^{233}U system requires hard neutron spectra to support high conversion of ^{232}Th. The neutron spectra of both the seed and the blanket region are harder compared to the standard thermal systems.
  - Compared to the UNF of the standard UOX system, the HLW of the Th/^{233}U multi-recycle system has lower normalised radiotoxicity and decay heat at discharge, but these become significantly lower post-separations, due to the recycle of 99.9% of Th and U isotopes in the nuclear system and the low production of transuranic elements.

Acknowledgements

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