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ABSTRACT

In this study, we have sought to determine the advantages, disadvantages, and viability of open cycle thorium–uranium-fuelled (Th–U-fuelled) nuclear energy systems. This has been done by assessing three such systems, each of which requires uranium enriched to ~20% 235U, in comparison to a reference uranium-fuelled (U-fuelled) system over various performance indicators, spanning material flows, waste composition, economics, and proliferation resistance. The values of these indicators were determined using the UK National Nuclear Laboratory’s fuel cycle modelling code ORION. This code required the results of lattice-physics calculations to model the neutronics of each nuclear energy system, and these were obtained using various nuclear reactor physics codes and burn-up routines. In summary, all three Th–U-fuelled nuclear energy systems required more separative work capacity than the equivalent benchmark U-fuelled system, with larger levelised fuel cycle costs and larger levelised cost of electricity. Although a reduction of ~5% in the required uranium ore per kWh was seen for one of the Th–U-fuelled systems compared to the reference U-fuelled system, the other two Th–U-fuelled systems required more uranium ore per kWh than the reference. Negligible advantages and disadvantages were observed for the amount and the properties of the spent nuclear fuel (SNF) generated by the systems considered. Two of the Th–U-fuelled systems showed some benefit in terms of proliferation resistance of the SNF generated. Overall, it appears that there is little merit in incorporating thorium into nuclear energy systems operating with open nuclear fuel cycles.

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1. Introduction

At the time of writing, the majority of the world’s electricity generating nuclear energy systems are fuelled with low-enriched uranium (LEU) and operate on open nuclear fuel cycles, where the U-based fuel is used only once with a view to being directly disposed of after a cooling period, i.e. the spent nuclear fuel (SNF) is neither reprocessed nor reused. Uranium reserves are typically graded in terms of their economic viability (United Nations, 2007). According to the most recent OECD report (OECD-NEA Report, 2012) there are 3.08 × 1012 tonnes of uranium recoverable for less than US$80/kgU, 5.33 × 1010 tonnes of uranium recoverable for less than US$130/kgU and 7.10 × 109 tonnes of uranium recoverable for less than US$260/kgU. Present estimates for global uranium reserves, including unconventional resources such as coal ash and phosphates (but excluding seawater), range from 1.92 × 107 tonnes (Tulsidas, 2011) to 3.93 × 107 tonnes (Romanello et al., 2012). At 2012 consumption rates of 67,990 tonnes of uranium per year (World Nuclear Association, 2013), this global supply would last 78.3 years for ore recoverable for under US$130/kgU and 578 years for assumed global uranium reserves of 3.93 × 107 tonnes. The work of Romanello et al. (2012) suggests that a burgeoning demand for electricity across the world will yield a significant increase in nuclear energy capacity, and correspondingly a significant increase in uranium consumption. Their predic-
tions (assuming only open nuclear fuel cycles) suggest that the reserve of ore recoverable for under US$130/kgU would be exhausted by ~2060 and a total reserve of 3.93 × 10^7 tonnes would be exhausted by ~2160.

One way in which this finite resource could be extended is by incorporating thorium as a nuclear fuel. Present estimates suggest that global thorium reserves total 6.4–7.5 million tonnes; however, due to its long half-life, it is expected that thorium is 3–4 times more abundant in the Earth’s crust than uranium. Thorium is traditionally associated with closed nuclear fuel cycles, where fissile isotopes (such as 233U formed from 232Th, and 239Pu and potentially other minor actinides primarily formed from 238U) are recovered from the SNF and reused. Advocates claim that Th-based fuels offer advantages over U-based fuels. Foremost, since 232Th has a larger thermal neutron capture cross-section than 238U, 232U can be bred more efficiently from 232Th within thermal spectra than 239Pu can be bred from 238U. Given that 233U is formed, it is often mentioned that less plutonium and fewer minor actinides are formed and these elements form the bulk of long-lived radiotoxicity, spontaneous neutron emission and decay heat (over 1000–100,000 years) of SNF, see e.g. Kamei and Hakami (2011). In terms of economics, thorium is currently characterised as a waste by-product, typically from rare-earth element processing, and it is suggested that the introduction of thorium has the potential to suppress the volatility of uranium prices. Thorium has been commonly ascribed as having enhanced proliferation resistance due to the facts that: (1) less LEU fuel is contained within the reactor, yielding smaller amounts of plutonium; (2) the 231U that is bred within the fuel is denatured with unreacted 235U; and (3) the short-lived isotope 232U is also formed, the daughter products of which (particularly 208Tl) add an additional radiological barrier.

For countries that want to adopt new nuclear energy systems, open nuclear fuel cycles are typically considered due to their lower infrastructure requirements and the significantly greater costs of reprocessing and refabrication than those of direct disposal. Therefore, the question arises as to whether thorium can be utilised in an open nuclear fuel cycle and incorporated in existing or novel nuclear energy technologies. It should be noted that this work treats the open nuclear fuel cycle in its strongest sense, i.e. plutonium disposition fuel cycles are considered to be out of scope, due to the need for prior SNF reprocessing. From a recent review paper (Ashley et al., 2013), a number of technology families have been highlighted that could prospectively use open Th-U-based fuel cycles. These include: (1) existing light water reactors (LWRs), (2) novel heavy-water-moderated, light-water-cooled reactors, and (3) novel high-temperature gas-cooled reactors.

This paper seeks to compare three candidate technologies operating with open Th-U-based nuclear fuel cycles to a ‘reference’ U-fuelled nuclear energy system over various performance indicators. The Th-U-fuelled technologies include AREVA’s European Pressurised Reactor (EPR), the Indian Advanced Heavy Water Reactor (AHWR), and General Atomics’ Gas-Turbine Modular Helium Reactor (GT-MHR). The reference U-fuelled system chosen was an EPR. Section 2 provides an overview of the fuel cycle modelling software, ORION, that was used in this work to derive mass flows and separative work units, isotopes associated with the SNF, and uranium and plutonium vectors for assessing the proliferation resistance of the SNF. Section 3 outlines the reactor systems further, the simulation techniques used and the parameters adopted in the neutronic analyses. Section 4 covers the mass flows of uranium, thorium, and separative work units for each of the four nuclear fuel cycles that are reported. Section 5 covers the isotopes of the SNF and the corresponding volumes, radiotoxocities, spontaneous neutron emission rates, and decay heats, and the potential issues surrounding deep geological disposal. Section 6 outlines an economic analysis of the fuel cycles and the corresponding nuclear energy systems to yield levelised nuclear fuel cycle costs and levelised costs of electricity. Section 7 covers the proliferation resistance methodology that was developed at the UK National Nuclear Laboratory (NNL) and is applied to each of the four nuclear fuel cycles. Finally, Section 8 provides a discussion of the results of Sections 4–7 and indicates the areas where open Th-U-based nuclear fuel cycles will need to be more competitive if they are to compete with open U-based cycles.

2. Fuel cycle simulation with ORION

ORION is a fuel cycle modelling code developed at NNL. The code performs inventory analysis to determine the throughput of material throughout a number of facilities in the nuclear fuel cycle, including storage buffers (that can represent the mine, mill and deep geological repositories), fuel fabrication facilities, reactors, and reprocessing facilities. For modelling the isotopic inventories within a reactor, ORION requires burn-up-dependent, shielded cross-sections produced by post-processing the results from deterministic or Monte-Carlo-based neutronic analyses of the reactor core. ORION has the capability to calculate the radiotoxicity, toxic potential, activity, spontaneous neutron emission rate and decay heat throughout the fuel cycle, as ~2500 isotopes including fission products and actinides are tracked. For radiotoxicity calculations, doses are evaluated using ingestion conversion coefficients provided in International Commission on Radiological Protection (1996). For decay heat and neutron emission rates, data from the JEFF-2.2 Nuclear Data Library (DOE-NEA, 2000) are used.

A major strength of ORION is its ability to model complicated multi-reactor, multi-recycle options that can be used in energy pathway analyses, as outlined in Gregg and Grove (2012). This paper will outline its use in novel, open Th-U-based nuclear fuel cycles.

3. Selection of nuclear energy systems fuelled with thorium and LEU

Nuclear energy technologies that can potentially utilise open Th-U-based nuclear fuel cycles are described in Ashley et al. (2013) and references therein. Three different reactor technologies that broadly cover LWRs, heavy-water-moderated, light-water-cooled reactors, and high-temperature gas-cooled reactors have been selected for this study, these are respectively: AREVA’s EPR, the Indian AHWR, and General Atomics’ GT-MHR. In each case, the maximal enrichment of the uranium component of the fuel is ~20%. As a reference for comparison, these three cases will be compared with a U-fuelled (with 235U = 5%) EPR. In general, detailed simulations for these reactor configurations have previously been performed by a number of groups and these are highlighted in the following sub-sections. The neutronic simulations outlined in this work were performed to generate burn-up-dependent cross-sections required to determine the composition of the SNF and mass flows of feed materials.

3.1. Reference LEU-fuelled nuclear reactor: The EPR

The reference system considered in this study is AREVA’s EPR, which is prospectively going to be constructed in the UK at Hinkley Point in Somerset and Sizewell in Suffolk. The parameters used to model this reactor are detailed in Table 1, the majority of these being taken from the submission to the UK Generic Design Assess-
ment (AREVA, 2007) with gaps being filled from the earlier design in Sengler et al. (1999). Neutronic simulations of this system were performed using WIMS 9 (Newton and Hutton, 2002), with nuclear data from the JEF-2.2 library (OECD-NEA, 2000). Results from these simulations were cross-checked with a separate set of calculations using CASMO-4 (Rhodes and Edenius, 2003) with nuclear data also from the JEF-2.2 library (OECD-NEA, 2000). Results from these simulations were cross-checked with a separate set of calculations using CASMO-4 (Rhodes and Edenius, 2003) with nuclear data also from the JEF-2.2 library (OECD-NEA, 2000).

During equilibrium operation, fresh nuclear fuel containing $^{235}\text{U}$ = 5.0% is loaded every 18 months on a three-batch scheme. For the reactor start-up, the associated uranium enrichments of the initial core are $^{235}\text{U}$ = 2.1%, 3.2%, 4.2%, and 5.0% (AREVA, 2007). The initial enrichments used in the reactor start-up phase are accounted for when determining the mass flows of the feed material. However, only the isotopic composition of the SNF from the equilibrium phase of the reactor has been analysed. This means that the absolute radioactivity, neutron emission rates and decay heats would be very slightly overestimated (when averaged over the whole life of the reactor).

### 3.2. The EPR fuelled with LEU and thorium

The starting point for this comparison was a Th–U core design based on an optimised Westinghouse four-loop PWR designed to minimise waste and enhance proliferation resistance (Todosow and Kazimi, 2004). As the AREVA EPR was used as the reference technology in this study, the Westinghouse core design was upscaled and mapped to match that of the EPR core. The main parameters used in these simulations are listed in Table 2.

In the original 193-assembly Westinghouse core design, there were 84 seed assemblies (containing annular UO$_2$ fuel pellets enriched to 20% $^{235}\text{U}$ with axial reflector blankets of natural U) and 109 blanket assemblies (containing blended ThO$_2$/UO$_2$ fuel, in the ratio 87:13, with the uranium enrichment set to 10% $^{235}\text{U}$). For the 241 assemblies within the EPR, this corresponds to 105 seed assemblies and 136 blanket assemblies. In both the Westinghouse and EPR design, 17 × 17 assemblies with identical cross-section of the Westinghouse assembly described in Todosow and Kazimi (2004) is 3.81 m compared to the active length of 4.2 m in the EPR assembly.

In a similar manner to the up-scaling of the number of assemblies, the fractional lengths of the poisoned, unpoisoned, and reflective regions (shown in Fig. 1) remain the same. The burnable poison used in these assemblies is Er$_2$O$_3$. Similarly, the same refuelling scheme is assumed for the EPR. Due to the high discharge burn-ups of the seed and blanket assemblies, silicon carbide has been assumed as the cladding material. Although silicon-carbide-based cladding requires significant research and development, such materials may become viable in the next 10–20 years. Further details on the development status of silicon-carbide-based cladding materials can be found in Hallstadius et al. (2012).

For the neutronic simulation of this core, the deterministic code WIMS 9 (Newton and Hutton, 2002) was used with cross-sections from the JEF-2.2 library (OECD-NEA, 2000). Cross-sections were generated using a supercell corresponding to an infinite ‘chequer-board’ of seed and blanket assemblies (as shown in Fig. 2), such that interface effects between assemblies were explicitly modelled in WIMS. The subgroup method was used for 172-group cross-section preparation for treatment of heterogeneous geometry resonance interaction effects between $^{232}\text{Th}$, $^{233}\text{U}$, $^{235}\text{U}$ and $^{238}\text{U}$ (Powney and Newton, 2004). As the assemblies were treated only in two dimensions, the axial blanket region and poisoned region of

<table>
<thead>
<tr>
<th>Table 1</th>
<th>Parameters used in modelling the AREVA EPR.</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reactor parameters</td>
<td></td>
</tr>
<tr>
<td>Thermal power</td>
<td>4500 MW(th)</td>
</tr>
<tr>
<td>Electrical power</td>
<td>1630 MW(e)</td>
</tr>
<tr>
<td>Load factor</td>
<td>90%</td>
</tr>
<tr>
<td>Number of assemblies</td>
<td>241</td>
</tr>
<tr>
<td>$^{235}\text{U}$ content (equilibrium)</td>
<td>5.0%</td>
</tr>
<tr>
<td>$^{235}\text{U}$ content (start-up)</td>
<td>2.1%, 3.2%, 4.2%, 5.0%</td>
</tr>
<tr>
<td>Core mass</td>
<td>127.15 tHM</td>
</tr>
<tr>
<td>Refuelling scheme</td>
<td>3-batch over 4.5 years</td>
</tr>
<tr>
<td>Number of assemblies with 8% Gd$_2$O$_3$</td>
<td>144</td>
</tr>
<tr>
<td>UO$_2$ density</td>
<td>10.2 g/cm$^3$</td>
</tr>
<tr>
<td>Power density</td>
<td>35.4 W/tHM</td>
</tr>
<tr>
<td>Active core length</td>
<td>4.2 m</td>
</tr>
<tr>
<td>Coolant and moderator</td>
<td>Light water</td>
</tr>
<tr>
<td>Average discharge burn-up</td>
<td>51.2 GWd/tHM</td>
</tr>
<tr>
<td>Operating temperature for fuel</td>
<td>626.9°C</td>
</tr>
<tr>
<td>Operating temperature for coolant</td>
<td>313.7°C</td>
</tr>
<tr>
<td>Assembly parameters (all dimensions reported are at 20°C)</td>
<td></td>
</tr>
<tr>
<td>Number of pins</td>
<td>265</td>
</tr>
<tr>
<td>Assembly layout</td>
<td>17 × 17</td>
</tr>
<tr>
<td>Fuel rod diameter</td>
<td>9.50 mm</td>
</tr>
<tr>
<td>Fuel pellet diameter</td>
<td>8.19 mm</td>
</tr>
<tr>
<td>Diametral gap</td>
<td>0.17 mm</td>
</tr>
<tr>
<td>Cladding thickness</td>
<td>0.57 mm</td>
</tr>
<tr>
<td>Pin pitch</td>
<td>12.6 mm</td>
</tr>
<tr>
<td>Assembly pitch</td>
<td>215 mm</td>
</tr>
<tr>
<td>Cladding material</td>
<td>Zircaloy M5</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Table 2</th>
<th>Parameters used in modelling the AREVA EPR whole-assembly seed-blanket design, fuelled with LEU and thorium.</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reactor parameters</td>
<td></td>
</tr>
<tr>
<td>Thermal power</td>
<td>4500 MW(th)</td>
</tr>
<tr>
<td>Electrical power</td>
<td>1630 MW(e)</td>
</tr>
<tr>
<td>Load factor</td>
<td>90%</td>
</tr>
<tr>
<td>Number of seed assemblies</td>
<td>105</td>
</tr>
<tr>
<td>Number of blanket assemblies</td>
<td>136</td>
</tr>
<tr>
<td>Seed $^{235}\text{U}$ enrichment</td>
<td>20%</td>
</tr>
<tr>
<td>Blanket $^{235}\text{U}$ enrichment</td>
<td>7.65%</td>
</tr>
<tr>
<td>Blanket UO$_2$/ThO$_2$ ratio</td>
<td>83:17</td>
</tr>
<tr>
<td>Seed mass</td>
<td>33.27 tHM</td>
</tr>
<tr>
<td>Blanket mass</td>
<td>84.85 tHM</td>
</tr>
<tr>
<td>Seed refuelling scheme</td>
<td>3-batch over 4.5 years</td>
</tr>
<tr>
<td>Blanket refuelling scheme</td>
<td>1-batch over 15 years</td>
</tr>
<tr>
<td>Seed fuel density</td>
<td>10.3 g/cm$^3$</td>
</tr>
<tr>
<td>Blanket fuel density</td>
<td>9.4 g/cm$^3$</td>
</tr>
<tr>
<td>Seed power density</td>
<td>87.91 W/tHM</td>
</tr>
<tr>
<td>Blanket power density</td>
<td>18.56 W/tHM</td>
</tr>
<tr>
<td>Active core length</td>
<td>4.2 m</td>
</tr>
<tr>
<td>Coolant and moderator</td>
<td>Light water</td>
</tr>
<tr>
<td>Seed discharge burn-up</td>
<td>140 GWd/tHM</td>
</tr>
<tr>
<td>Blanket discharge burn-up</td>
<td>88 GWd/tHM</td>
</tr>
<tr>
<td>Operating temperature for fuel</td>
<td>626.9°C</td>
</tr>
<tr>
<td>Operating temperature for coolant</td>
<td>313.7°C</td>
</tr>
<tr>
<td>Assembly parameters (all dimensions reported are at 20°C)</td>
<td></td>
</tr>
<tr>
<td>Number of pins</td>
<td>265</td>
</tr>
<tr>
<td>Assembly layout</td>
<td>17 × 17</td>
</tr>
<tr>
<td>Seed fuel rod diameter</td>
<td>9.00 mm</td>
</tr>
<tr>
<td>Seed pellet radius (outer annulus)</td>
<td>3.85 mm</td>
</tr>
<tr>
<td>Seed pellet radius (inner annulus)</td>
<td>2.20 mm</td>
</tr>
<tr>
<td>Blanket fuel rod diameter</td>
<td>10.60 mm</td>
</tr>
<tr>
<td>Blanket pellet radius</td>
<td>4.65 mm</td>
</tr>
<tr>
<td>Interstitial gap</td>
<td>0.08 mm</td>
</tr>
<tr>
<td>Cladding thickness</td>
<td>0.57 mm</td>
</tr>
<tr>
<td>Pin pitch</td>
<td>12.6 mm</td>
</tr>
<tr>
<td>Assembly pitch</td>
<td>215 mm</td>
</tr>
<tr>
<td>Cladding material</td>
<td>Silicon carbide</td>
</tr>
</tbody>
</table>

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1 In Ref. (Sengler et al., 1999) a four-batch fuelling scheme is posited. We believe the discharge burn-ups for this scheme would be too large (~70 GWd/tHM) compared to the present maximal discharge burn-up of 65 GWd/tHM.

4 In Table B2.1.4 of (Todosow and Kazimi, 2004) the active length of the fuel pin is quoted as 385.76 cm, which differs from the lengths provided in Fig. B2.1.9 of the same work which total 381 cm.
the seed assembly were “smeared” into a homogeneous region, which is appropriate for fuel cycle calculations.

On initial simulations of the blanket assembly, it was observed that the $^{233}$U content exceeded the 12% limit for LEU containing $^{233}$U, as described further in Section 7 and Ashley et al. (2012). Therefore, the ratio of thorium to uranium was changed from 87:13 to 83:17, with the $^{235}$U enrichment decreased from 10% to 7.65% to match the fissile fraction and to denature the uranium component. We note that the discharge burn-up of the blanket may be slightly affected by this change.

For standard commercial reactors, especially typical PWRs, codes exist for performing full-core analyses to determine the optimal loading patterns and exact discharge burn-ups. However, the nested Th–U fuel assembly design is novel and the highly heterogeneous nature of the core, with large differences in occupancy times of the seed and blanket assemblies, make full-core calculations non-trivial. Determining an optimal in-core fuel management scheme and corresponding discharge burn-ups was beyond the scope of this work. We therefore assume the same discharge burn-ups as that of the Westinghouse design, i.e. a discharge burn-up of 140 GWd/tHM for the seed assemblies and a discharge burn-up of 88 GWd/tHM for the blanket assemblies. We note that this particular design is not optimised and that further full-core calculations are required to determine accurately the exact discharge burn-ups of seed and blanket assemblies. For completeness, the thermal–hydraulic feasibility of the up-scaled design should also be evaluated.

3.3. India’s AHWR fuelled with LEU and thorium

The parameters used to model the AHWR are listed in Table 3. These parameters are taken from Bhabha Atomic Research Centre (2012), International Atomic Energy Agency (2012b), Thakur et al. (2011). Due to the relatively large central displacer rod, as shown in Fig. 3, the lattice code available was not able to model this particular geometry. Therefore, the Monte Carlo computer code MCNPX version 2.7.0 (Pelowitz, 2007; Pelowitz et al., 2011) was used instead. Nuclear data for these calculations were taken from the JEF-2.2 library (OECD-NEA, 2000). Further details on this model can be found in Cowper (2012).

Table 3

<table>
<thead>
<tr>
<th>Reactor parameters</th>
<th>Parameters used in modelling the AHWR fuelled with LEU and thorium.</th>
</tr>
</thead>
<tbody>
<tr>
<td>Thermal power</td>
<td>920 MW(th)</td>
</tr>
<tr>
<td>Electrical power</td>
<td>300 MW(e)</td>
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<tr>
<td>Load factor</td>
<td>90%</td>
</tr>
<tr>
<td>Number of fuel clusters</td>
<td>444</td>
</tr>
<tr>
<td>$^{233}$U enrichment</td>
<td>19.75%</td>
</tr>
<tr>
<td>Start-up $^{233}$U/$^{233}$ThO$_2$ ratio</td>
<td>13:87</td>
</tr>
<tr>
<td>Equilibrium $^{233}$U/$^{233}$ThO$_2$ ratio</td>
<td>21.8:78.2</td>
</tr>
<tr>
<td>Core mass</td>
<td>51.77 t$_{\text{HM}}$</td>
</tr>
<tr>
<td>Refuelling scheme</td>
<td>10-batch over 10 years</td>
</tr>
<tr>
<td>Fuel density</td>
<td>9.4 g/cm$^3$</td>
</tr>
<tr>
<td>Power density</td>
<td>17.77 W/g$_{\text{HM}}$</td>
</tr>
<tr>
<td>Active core length</td>
<td>3.5 m</td>
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<td>Moderator</td>
<td>Heavy water</td>
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<td>Coolant</td>
<td>Light water</td>
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<tr>
<td>Discharge burn-up</td>
<td>64 GWd/t$_{\text{HM}}$</td>
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<td>Operating temperature for fuel</td>
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<td>Operating temperature for coolant</td>
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<tr>
<td>Moderator temperature</td>
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<td>Moderator density</td>
<td>1.089 g/cm$^3$</td>
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Assembly parameters (all dimensions reported are at 20 °C)

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<thead>
<tr>
<th>Cluster pin arrangement</th>
<th>Number of pins with 5% Gd content</th>
<th>Fuel rod diameter</th>
<th>Pellet radius</th>
<th>Interstitial gap</th>
<th>Cladding thickness</th>
<th>Cluster pin-pitch circle diameter</th>
<th>Zircaloy centre (outer radius)</th>
<th>Zircaloy tube (outer radius)</th>
<th>Cladding material</th>
<th>$^{233}$U/$^{233}$ThO$_2$ ratio</th>
</tr>
</thead>
<tbody>
<tr>
<td>12 (inner)</td>
<td>2 (inner)</td>
<td>11.20 mm</td>
<td>4.90 mm</td>
<td>0.10 mm</td>
<td>0.60 mm</td>
<td>51.4 mm (inner)</td>
<td>9.00 mm</td>
<td>18.00 mm</td>
<td>Zircaloy-2</td>
<td>16.84 (outer)</td>
</tr>
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<td>18 (middle)</td>
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<td></td>
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<td>103.7 mm (outer)</td>
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<td>24 (outer)</td>
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</table>

Fig. 1. Schematic diagram of the seed fuel pin axial zoning for the AREVA EPR whole-assembly seed-blanket design, fuelled with LEU and thorium.

Fig. 2. Schematic diagram of the infinite chequerboard whole-assembly seed-blanket design (with translational boundary conditions) used in the neutronic calculations of the EPR with UO$_2$ and ThO$_2$. The dashed line shows the smallest unit cell (with reflective boundary conditions).
and an estimated 46 of the 444 bundles discharged VF, and VT refer to the masses and
/\0\0\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00\00
respectively. The value function for each of these components, \( V_X \), is:

\[
V_X = \left( \frac{1}{C_0^2 w_X} \right) \ln \left( \frac{1}{C_0 w_X} \right)
\]

where \( w_X \) is the relative weight of each component.

The EPR and GT-MHR are listed as having lifetimes of 60 years (AREVA, 2012; General Atomics, 2002) and the AHWR 100 years (International Atomic Energy Agency, 2012b). Therefore, to allow for a fair comparison as possible, and due to the blanket component of the EPR fuelled with UO\(_2\)/ThO\(_2\) lasting for 13.5 years, for all fuel cycles in this study, the reactor lifetimes were centred as close to 67.5 years as possible (i.e. 67.5 years for the EPR fuel cycles, 67 years for the AHWR, and 66.7 years for the GT-MHR). This should yield a fair comparison as each of the fuel cycles are complete and in equilibrium, and the results are normalised to each kWh generated (noting that the economics analysis in Section 6 accounts for discounting). For all technologies considered in this work, the lifetimes are significant increases on those of current technologies; therefore it remains to be seen whether such lifetimes are achievable. The amounts of uranium, thorium, and separative work required for each nuclear fuel cycle in this study are presented in Fig. 5, with the corresponding amounts of uranium, thorium, and separative work capacity to generate 1 kWh(e) being shown in Fig. 6.

From Fig. 6 it is evident that the Th–U-fuelled AHWR requires the least amount of uranium ore per kWh(e) generated. To highlight the sensitivity of this metric, if the core occupancy of the U-fuelled EPR were to be extended from 4.5 to 5 years (with the average discharge burn-up increasing from 51.5 GWd/tHM to 58.7 GWd/tHM, and assuming the same initial enrichments and load factor) then the U-fuelled EPR would require less uranium ore per kWh(e). Due to the requirement of uranium enriched to \( 20\% ^{235}\text{U} \) for the Th–U-fuelled systems, the U-fuelled reference EPR requires the smallest amount of separative work units per kWh(e) generated.

In the case of the AHWR, the ratio of the average discharge burn-up (64 GWd/tHM) to the fissile fraction of the fuel initially loaded (4.3%) is greater than that seen for the reference EPR in this study (51.5 GWd/tHM for 5\% \( ^{235}\text{U} \)). The requirement of \( 235\text{U} \) enriched to 19.75\% yields a greater amount of separative work per kWh than for the reference EPR. An open question is how a
It is particularly striking that the Th–U-fuelled GT-MHR requires significantly more uranium per kWh. This is due to the very low average discharge burn-up (~45 GWd/tHM) compared to the initial fissile fraction of the loaded fuel (12%) for this configuration. Further work in determining the optimal balance between uranium, thorium and the discharge burn-up of the fuel in such systems is required.

Finally, the EPR design with thorium and uranium was optimised such that the discharge burn-up was maximised with a view to minimising the amount of SNF. Thus, a Th–U-based core design where equivalent discharge burn-ups to that of the reference EPR are obtained could provide a fairer comparison. This is especially true if the required enrichment is well below 20%.

It should be stressed that this study is not exhaustive and there may be nuclear energy systems with different fueling arrangements that could utilise thorium more successfully. One possible emerging technology where thorium could be considered in an open nuclear fuel cycle is the reduced-moderation boiling water reactor, where initial fuel cycle studies have shown promise (Lindley et al., 2013). However, a significant amount of research and development is needed, with a robust appraisal of the technical barriers that need to be overcome, before such technologies could be considered licensable. It should also be stressed that there is a significant gulf between that which is licensable and that which is viable.

Fig. 6. Mass flows and separative work to generate 1 kWh(e) for the four reactor systems. Normal text denotes either the mass of uranium (in kgU) or the mass of thorium (in kgTh). Italicised text denotes the separative work units required in kgSWU.

Fig. 7. Decay heats per assembly discharged for the four nuclear energy systems studied in this work. (S) denotes the seed assemblies and (B) denotes the blanket assemblies of the Th–U-fuelled EPR. (DF) denotes the once-irradiated “driver” fuel and (PDF) denotes the “post-driver fuel” in the Th–U-fuelled GT-MHR. (For interpretation to colours in this figure, the reader is referred to the web version of this paper.)

Table 6
Parameters used in the analyses of mass flow of uranium and separative work required. Losses are taken from OECD-NEA (1994).

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Loss in conversion</td>
<td>0.5%</td>
</tr>
<tr>
<td>Loss in enrichment</td>
<td>0.0%</td>
</tr>
<tr>
<td>Loss in fuel fabrication</td>
<td>1.0%</td>
</tr>
<tr>
<td>235U content in tailings</td>
<td>0.25%</td>
</tr>
<tr>
<td>Lifetime of the EPR with UO2</td>
<td>67.5 years</td>
</tr>
<tr>
<td>Lifetime of the EPR with UO2/ThO2</td>
<td>67.5 years</td>
</tr>
<tr>
<td>Lifetime of the AHWR with UO2/ThO2</td>
<td>67.0 years</td>
</tr>
<tr>
<td>Lifetime of the GT-MHR with UO2/ThO2</td>
<td>66.7 years</td>
</tr>
<tr>
<td>Load factor for all nuclear energy systems</td>
<td>90%</td>
</tr>
<tr>
<td>Construction time for all reactors</td>
<td>5 years</td>
</tr>
<tr>
<td>Lifetime electricity generated by EPR with UO2</td>
<td>868.0 TWh</td>
</tr>
<tr>
<td>Lifetime electricity generated by EPR with UO2/ThO2</td>
<td>868.0 TWh</td>
</tr>
<tr>
<td>Lifetime electricity generated by AHWR with UO2/ThO2</td>
<td>158.6 TWh</td>
</tr>
<tr>
<td>Lifetime electricity generated by GT-MHR with UO2/ThO2</td>
<td>150.5 TWh</td>
</tr>
</tbody>
</table>

Th–U-fuelled AHWR would compare to an equivalent U-fuelled AHWR, noting that fuel-cycle schemes for a U-fuelled AHWR are yet to be published. It is particularly striking that the Th–U-fuelled GT-MHR requires significantly more uranium per kWh. This is due to the very low average discharge burn-up (~45 GWd/tHM) compared to the initial fissile fraction of the loaded fuel (12%) for this configuration. Further work in determining the optimal balance between uranium, thorium and the discharge burn-up of the fuel in such systems is required.
5. Waste forms

The waste-form analysis covers four separate aspects of the SNF: decay heats, neutron emission rates, radiotoxicities, and volume. Sections 5.1–5.3 report the properties of the decay heats, spontaneous neutron emission rates and radiotoxicities for an individually discharged fuel assembly. Section 5.4 discusses, in absolute terms, the total volume of all assemblies discharged over the life of the reactor. Section 5.5 reports these four aspects in terms of normalised electrical output. It should be stressed that in the absence of a complete repository design and licensed package for each system being studied the following assessment should be seen as indicative rather than normative.

5.1. Decay heats per discharged assembly

The decay heat, as a function of time, of a single assembly discharged from each of the nuclear energy systems is shown in Fig. 7. It is evident that the decay heat of a single discharged assembly from the U-fuelled EPR is lower than that of a single discharged seed assembly and single discharged blanket assembly from the Th–U-fuelled EPR. For the Th–U-based blanket of the Th–U-fuelled EPR [\(\text{UO}_2/\text{ThO}_2\) (B)], it is evident that for \(-120\) years the decay heats are higher than those from the seed [\(\text{UO}_2/\text{ThO}_2\) (S)]. In this time period, although less SNF is discharged from the Th–U-fuelled EPR, the decay heats may either affect the amount of SNF which can be housed within a waste-form package (see Section 5.4) or would require the fuel to be kept in wet- or dry-storage for a longer period of time. This is explained by comparing the contribution of each individual isotope to the total decay heat for a seed assembly and a blanket assembly (as in Fig. 8), which shows that daughter products from \(235\text{U}\), e.g. \(239\text{Po}\), contribute \(-10\)% of the decay heat in the blanket assembly at year 50. For all Th–U-based fuels, an increase in decay heat after 2000 years due to the \(233\text{U}\) decay chain could potentially have a detrimental effect on the performance of the backfill. For the AHWR and GT-MHR, the lower decay heats may allow for greater packing fractions in the waste form, as discussed further in Section 5.4.

5.2. Spontaneous neutron emission rates per discharged assembly

The spontaneous neutron emission rates, as a function of time, of a single assembly discharged from each of the nuclear energy systems are shown in Fig. 9. Neutron sources consist mainly of actinide nuclei which can undergo spontaneous fission, as seen in comparing the seed and blanket assemblies of the Th–U-fuelled EPR (shown in Fig. 10). Due to its longer irradiation time, the blanket contains greater amounts of minor actinides; \(244\text{Cm}\) is the dominant contributor for the first 150 years and thereafter \(246\text{Cm}\) up to \(-40,000\) years. In line with the decay heat observations in Section 5.1, the AHWR and GT-MHR have smaller spontaneous neutron emission rates than those seen in the EPR assemblies.

5.3. Radiotoxicities per discharged assembly

The radiotoxicity, as a function of time, of a single assembly discharged from each of the nuclear energy systems is shown in Fig. 11. Due predominantly to its lower discharge burn-up, the radiotoxicity of a single discharged assembly from the U-fuelled EPR is lower than that of a single discharged seed assembly and single discharged blanket assembly from the Th–U-fuelled EPR. After \(-1000\) years, contributions from the \(233\text{U}\) decay chain begin to dominate, as indicated by the \(233\text{U}\) contribution in Fig. 12. Similarly, the radiotoxicities of the AHWR and GT-MHR SNF are lower, mainly due to the smaller amounts of material discharged.
5.4. Volume of spent nuclear fuel

The parameters used in ascertaining the volumes of spent fuel for each nuclear system are presented in Table 7. In Table 8, data for spent fuel analysis. AHWR cluster dimensions are taken from Sinha and Kakodkar (2006).

In accounting for the total numbers of SNF packages, various assumptions were needed to account for the dimensions of the waste-form package.

Table 7
Dimensions of the nuclear fuel assemblies for the EPR, AHWR, and GT-MHR required for spent fuel analysis. AHWR cluster dimensions are taken from Sinha and Kakodkar (2006).

<table>
<thead>
<tr>
<th>Reactor</th>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>EPR</td>
<td>Fuel assembly cross-sectional area</td>
<td>0.0458 m²</td>
</tr>
<tr>
<td></td>
<td>Fuel assembly length</td>
<td>4.80 m</td>
</tr>
<tr>
<td>AHWR</td>
<td>Cluster diameter</td>
<td>0.118 m</td>
</tr>
<tr>
<td></td>
<td>Cluster cross-sectional area</td>
<td>0.0109 m²</td>
</tr>
<tr>
<td></td>
<td>Cluster length</td>
<td>4.30 m</td>
</tr>
<tr>
<td>GT-MHR</td>
<td>Fuel block diagonal length</td>
<td>0.416 m</td>
</tr>
<tr>
<td></td>
<td>Fuel block cross-sectional area</td>
<td>0.112 m²</td>
</tr>
<tr>
<td></td>
<td>Fuel block length</td>
<td>7.93 m</td>
</tr>
</tbody>
</table>

Table 8
Expected numbers and volumes of nuclear fuel assemblies discharged over the lifetime of the four nuclear energy systems studied in this work. The italicised text shows the individual contributions of the seed assemblies and blanket assemblies for the Th–U-fuelled EPR.

<table>
<thead>
<tr>
<th>Nuclear energy system</th>
<th>Volume of single assembly (m³)</th>
<th>Expected number of assemblies discharged</th>
<th>Total volume of assemblies discharged (m³)</th>
</tr>
</thead>
<tbody>
<tr>
<td>EPR with UO²</td>
<td>0.2200</td>
<td>3761</td>
<td>8.27 × 10²</td>
</tr>
<tr>
<td>EPR with UO₂/ThO₂ (total)</td>
<td>0.2200</td>
<td>2325</td>
<td>5.12 × 10²</td>
</tr>
<tr>
<td>EPR with UO₂/ThO₂ (seed component only)</td>
<td>0.2200</td>
<td>1645</td>
<td>3.62 × 10²</td>
</tr>
<tr>
<td>EPR with UO₂/ThO₂ (blanket component only)</td>
<td>0.2200</td>
<td>680</td>
<td>1.50 × 10²</td>
</tr>
<tr>
<td>AHWR with UO₂/ThO₂</td>
<td>0.0470</td>
<td>3375</td>
<td>1.59 × 10²</td>
</tr>
<tr>
<td>GT-MHR with UO₂/ThO₂</td>
<td>0.8900</td>
<td>2592</td>
<td>2.31 × 10³</td>
</tr>
</tbody>
</table>

Two separate canisters for two different deep geological repositories were selected: the Svensk Kärnbränslehantering (SKB) KBS-3 (SKB, 2011) and Yucca Mountain (U.S. DOE, 2002). Details of the dimensions of the various waste-form packages are presented in Table 9. It is also assumed that the canisters can be filled completely, i.e. the fuel can be sufficiently cooled for a long enough period before encapsulation. If the fuel cannot be cooled for long enough, there will be limits on the allowable decay heat such that the inner bentonite surface temperature in the deep geological repository does not exceed 100 °C (temperature limit taken from Nuclear Decommissioning Authority (2009)).

For the EPR assemblies within the SKB design, slight modifications were needed to account for the comparatively longer fuel assembly than the generic PWR assembly length listed. This was done by adding the difference in canister length and assembly length of the generic PWR to the EPR assembly length. For the Yucca Mountain container, it is assumed that the ‘long’ container which houses 12 assemblies would suffice as is (given that it is designed to hold volumetrically larger Combustion Engineering and South Texas Project assemblies).

Table 9
Dimensions of the spent fuel canisters for both the SKB KBS-3 and Yucca Mountain repositories.

<table>
<thead>
<tr>
<th>Waste-form package</th>
<th>Parameter</th>
<th>Value (m)</th>
</tr>
</thead>
<tbody>
<tr>
<td>SKB PWR</td>
<td>Assembly length</td>
<td>4.443</td>
</tr>
<tr>
<td>SKB (2010)</td>
<td>Canister length</td>
<td>4.835</td>
</tr>
<tr>
<td></td>
<td>Canister diameter</td>
<td>1.050</td>
</tr>
<tr>
<td>SKB EPR (assumed)</td>
<td>Assembly length</td>
<td>4.800</td>
</tr>
<tr>
<td></td>
<td>Canister length</td>
<td>5.227</td>
</tr>
<tr>
<td></td>
<td>Canister diameter</td>
<td>1.050</td>
</tr>
<tr>
<td>Yucca Mountain (long)</td>
<td>Assembly length</td>
<td>4.491–5.111</td>
</tr>
<tr>
<td>U.S. DOE (2002)</td>
<td>Canister length</td>
<td>5.651</td>
</tr>
<tr>
<td></td>
<td>Canister diameter</td>
<td>1.330</td>
</tr>
<tr>
<td>Yucca Mountain EPR (assumed)</td>
<td>Assembly length</td>
<td>4.800</td>
</tr>
<tr>
<td></td>
<td>Canister length</td>
<td>5.651</td>
</tr>
<tr>
<td></td>
<td>Canister diameter</td>
<td>1.644</td>
</tr>
<tr>
<td>SKB AHWR (assumed)</td>
<td>Cluster length</td>
<td>4.300</td>
</tr>
<tr>
<td></td>
<td>Canister length</td>
<td>4.692</td>
</tr>
<tr>
<td></td>
<td>Canister diameter</td>
<td>1.050</td>
</tr>
<tr>
<td>Yucca Mountain AHWR (assumed)</td>
<td>Cluster length</td>
<td>4.300</td>
</tr>
<tr>
<td></td>
<td>Canister length</td>
<td>4.840</td>
</tr>
<tr>
<td></td>
<td>Canister diameter</td>
<td>1.318</td>
</tr>
<tr>
<td>Yucca Mountain GT-MHR</td>
<td>Fuel block total length</td>
<td>7.930</td>
</tr>
<tr>
<td>General Atomics (2002)</td>
<td>Fuel block element length</td>
<td>0.793</td>
</tr>
<tr>
<td></td>
<td>Canister length</td>
<td>5.144</td>
</tr>
<tr>
<td></td>
<td>Canister diameter</td>
<td>1.397</td>
</tr>
</tbody>
</table>
As the AHWR is traditionally associated with closed nuclear fuel cycles, no spent fuel packaging form has been suggested. The following is therefore an ansatz as to how such fuel may be arranged within a canister. As the AHWR has cylindrical assemblies, the optimal periodic packing fraction could be obtained by hexagonal packing. With this geometry, 7, 19, or 37 assemblies could be housed within one SKB canister. From Sections 5.1 and 5.2, the decay heat release rate and spontaneous neutron emission rate (both at 50 years) of an AHWR cluster are, respectively, roughly four and six times lower than those of an EPR assembly. Therefore, it is conceivable (as an upper estimate) that 19 assemblies could satisfactorily be housed in one canister. The only other adjustment required for the canister is a reduction in its height, as detailed above.

For the Yucca Mountain design, it has been assumed that 44 clusters could be housed within a single canister, i.e. the same canister geometry as for discharged assemblies from boiling water reactors.

For the GT-MHR, only the Yucca Mountain repository design is considered. It should be noted that, although the GT-MHR was treated as a continuous fuel block in Section 3.4, the fuel block is constructed of ten separate elements, held together by a support infrastructure. In General Atomics, (2002), it is suggested that 42 elements (4.2 fuel blocks) would be housed within a single canister design. The basis for choosing only this design is that even with an optimal packing fraction, the volume of waste is considerably greater than for EPR assemblies within the heavily self-shielded SKB canister.

The total volumes of packaged SNF from each nuclear energy system within this study (over their respective lifetimes) are presented in Table 10.

5.5. Comparison to normalised electrical output

The decay heats, neutron emission rates, radiotoxicities, and volume of SNF (presented in Sections 5.1–5.4 respectively) per kWh are presented in Fig. 13 and Table 10. It is observed that there is a ~40% reduction in the volume of SNF for the Th–U-fuelled EPR compared to the reference U-fuelled EPR. As seen in Figs. 7 and 9, this is counter-balanced by the blanket assemblies having larger decay heat and spontaneous neutron emission rates, due to the significantly larger burn-up than in the reference EPR. However, the higher spontaneous neutron emission rate is counter-intuitive, given that Th-based fuels are normally expected to generate fewer minor actinides than equivalent U-based systems. The increase in the minor actinide contribution is attributed to the facts that the fuel is irradiated for significantly longer than conventional nuclear fuels and that the inner portion of the blanket assemblies appears more thermalised than the equivalent seed assemblies. In summary, although volumetrically less waste is generated, this advantage may be negated, as it may not be possible to use the same waste-form packaging as in the EPR reference scenario.

For the AHWR, studies involving the waste-form packaging have yet to be undertaken and so the analysis here is presented as a guide. A waste-form package containing 19 AHWR clusters would correspond to ~7% less SNF generated per kWh than for the reference U-fuelled EPR. A waste-form package containing 16 AHWR clusters (i.e. matching the total decay heat of the AHWR

<table>
<thead>
<tr>
<th>Nuclear energy system</th>
<th>Volume of package (m³)</th>
<th>Assemblies per package</th>
<th>Total volume of packaged SNF (m³)</th>
<th>Normalised volume of packaged SNF (m³/kWh)</th>
</tr>
</thead>
<tbody>
<tr>
<td>EPR with UO₂</td>
<td>4.53</td>
<td>4ª</td>
<td>4.26 × 10³</td>
<td>4.91 × 10⁻⁹</td>
</tr>
<tr>
<td></td>
<td>12ª</td>
<td>12b</td>
<td>3.76 × 10³</td>
<td>4.33 × 10⁻⁹</td>
</tr>
<tr>
<td>EPR with UO₂/ThO₂</td>
<td>4.53</td>
<td>4ª</td>
<td>2.63 × 10³</td>
<td>3.03 × 10⁻⁹</td>
</tr>
<tr>
<td></td>
<td>12ª</td>
<td>12b</td>
<td>2.33 × 10³</td>
<td>2.68 × 10⁻⁹</td>
</tr>
<tr>
<td>AHWR with UO₂/ThO₂</td>
<td>4.06</td>
<td>7ª</td>
<td>1.96 × 10³</td>
<td>1.23 × 10⁻⁸</td>
</tr>
<tr>
<td></td>
<td>4.06</td>
<td>19ª</td>
<td>7.21 × 10²</td>
<td>4.55 × 10⁻⁹</td>
</tr>
<tr>
<td></td>
<td>4.06</td>
<td>37ª</td>
<td>3.70 × 10²</td>
<td>2.34 × 10⁻⁹</td>
</tr>
<tr>
<td></td>
<td>6.60</td>
<td>44ª</td>
<td>5.06 × 10²</td>
<td>3.19 × 10⁻⁹</td>
</tr>
<tr>
<td>GT-MHR with UO₂/ThO₂</td>
<td>7.88</td>
<td>4.2b</td>
<td>4.86 × 10³</td>
<td>3.23 × 10⁻⁸</td>
</tr>
</tbody>
</table>

ª Waste-form package derived from the SKB canister design in Table 9.

b Waste-form package derived from the Yucca Mountain design in Table 9.

Fig. 13. Comparison of the decay heats, spontaneous neutron emission rates, and radiotoxicities (normalised to each kWh generated) for the four reactor systems considered in this work. (For interpretation to colours in this figure, the reader is referred to the web version of this paper.)
waste-form package to the reference U-fuelled EPR waste-form package) would correspond to \(\sim10\%\) more SNF generated per kWh than the reference U-fuelled EPR. Therefore, it is fair to conclude that the volume of SNF per kWh discharged by the AHWR would be similar to that of the reference U-fuelled EPR.

Due to the considerably lower burn-up for the Th–U-fuelled GT-MHR, the amount of waste generated from this reactor is significantly greater than for the EPR and AHWR.

### 6. Economics

#### 6.1. Estimation of the levelised fuel cycle costs

The methodology adopted for determining the levelised fuel cycle cost (LFCC) is based on that presented in OECD-NEA (1994). From this, the LFCC, \(C\), can be determined using:

\[
C = \sum \frac{F_i(t)}{E(t)} \frac{C(t)}{[1 + \epsilon_i]^t_0} (3)
\]

where \(i\) is the stage of the fuel cycle, \(t\) denotes the year, \(t_0\) denotes the commissioning year, \(F_i\) is the cost incurred in the \(i\)th stage of the fuel cycle, \(E\) is the electricity generated, and \(r\) is the discount rate (in this work, \(r\) is taken to be the weight-adjusted cost of capital). The cost incurred at each stage of the fuel cycle is given by:

\[
F_i = x_0C_i l_i (1 + s_i)^{t_0-1} (4)
\]

where \(x\) denotes either the mass of fuel, enriched product, or separative work units; \(c\) is the cost for each section of the fuel cycle; \(l\) denotes the loss of material at that stage; \(s\) is the rate of increase in uranium price that is exogenous to discounting; \(t\) denotes the date and \(t_0\) is the baseline date. It is assumed that the baseline date is 01/01/2012, with the reactor starting operation on 01/01/2018. For front-end processes, \(t\) is defined as the date when fuel is loaded less the lead time; for back-end processes, \(t\) is defined as the date when the fuel is loaded, plus the residency time and lag time.

Each of the costs assumed here is listed in Table 11, where the currency is taken to be 2012 US$. As suggested in OECD-NEA (1994), only uranium has an escalation rate attributed, with a rate of 1.2% adopted. Note that it is assumed that the cost of enrichment per separative work unit for uranium enriched to \(\sim20\%\), 235U is the same as that for uranium enriched to 5% 235U. For blended ThO2/UO2 fuels, it is assumed that the cost of fabrication is 1.5 times that for normal UO2 fuels, which is the upper limit provided in Shropshire et al. (2009), Lahoda (2004). For the annular UO2 in the seed assemblies of the EPR, the fuel fabrication cost is assumed to be the same as for UO2 fuel in the reference EPR.

Fig. 14 shows the comparison of the LFCC for each of the nuclear energy systems over the whole life of the reactor and normalised to each kWh generated. The normalised fuel cycle costs (assuming just the “middle” values quoted in Table 11) are 0.77c/kWh for the U-fuelled EPR, 0.90c/kWh for the Th–U-fuelled EPR, 0.99c/kWh for the Th–U-fuelled AHWR, and 2.37c/kWh for the Th–U-fuelled GT–MHR. From a Monte Carlo analysis using Simlab (EC JRC IPSC, 2011), the mean levelised costs (and uncertainties to one standard deviation) are: 0.97 ± 0.21 c/kWh for the U-fuelled EPR, 1.12 ± 0.25 c/kWh for the Th–U-fuelled EPR, 1.20 ± 0.22 c/kWh for the AHWR, and 2.61 ± 0.48 c/kWh for the GT-MHR.

For all Th–U-fuelled systems in this study, increases in LFCC are seen due to increased enrichment and greater fuel fabrication costs. The levelised cost of uranium ore is also slightly higher for the AHWR, due purely to discounting the longer fuel cycle length (i.e. a greater quantity of uranium is needed upfront than for the reference U-fuelled EPR). Due to increased uranium requirements for the Th–U-fuelled EPR and Th–U-fuelled GT-MHR in this study, any increase in the uranium price will cause further divergence in the LFCC for these systems.

Although the LFCC for Th–U-based fuels are higher, these values could be significantly underestimated. The assumed fuel fabrication cost for Th–U-based fuels is 1.5 times that of U-based fuels. This could also be significantly underestimated given: (1) the fact that fabrication of silicon-carbide cladding for the Th–U-fuelled EPR fuelled is currently not commercially viable (Hallstadius et al., 2012) and (2) the requirements of different U/Th ratios in the individual rings of the AHWR cluster (see Fig. 3).
The levelised cost of electricity (LCOE) can be calculated using the formalism derived by De Roo and Parsons (2011a)\textsuperscript{5}. In particular, the LCOE is split into four levelised components: the reactor cost, fixed operation and maintenance (O&M) costs, variable O&M costs, and fuel cycle costs (as calculated in Section 6.1). The levelised cost components for the nuclear fuel cycles under consideration were calculated using the corresponding Excel spreadsheet (De Roo and Parsons, 2011b) with the parameters listed in Tables 6, 11, and 12. It should be noted that the decommissioning costs were included in the levelised net present value associated with reactor construction. Using a similar procedure to the Monte Carlo analysis performed in Section 6.1, the mean LCOE (uncertainty to one standard deviation) was found to be 121 ± 16 US$/MWh for the reference, U-fuelled EPR, 122 ± 17 US$/MWh for the Th–U-fuelled EPR, 137 ± 18 US$/MWh for the Th–U-fuelled AHWR, and 157 ± 14 US$/MWh for the Th–U-fuelled GT-MHR. The breakdown of the LCOE is presented in Table 13.

Estimating the LCOE is notoriously difficult given the limited information associated with the capital and operation costs for all of these reactor systems. Construction costs for nuclear energy systems in the West have significantly increased since the early 2000s, as detailed in Massachusetts Institute of Technology (2011), and there is a large geographical variation in construction costs (International Energy Agency, 2010) which could yield further inconsistencies. Due to the variation in the sources of information, as described in Table 12, the analysis presented should be considered as indicative (rather than definitive); with different assumptions the differences between the relative costs of the systems considered narrow or broaden.

7. Proliferation resistance assessment

Proliferation resistance assessments of nuclear energy systems usually consist of qualitative assessments of quantitative data, and are by no means an exact science. In this work, the NNL proliferation resistance assessment methodology (Hesketh and Worrall, 2010; UK National Nuclear Laboratory, 2009) was used to assess the desirability of the plutonium and uranium components within the spent fuel for potential state proliferators. The methodology does not factor in the separative work capacity required for each system nor does it account for the amounts of dual-use materials as described in International Atomic Energy Agency (2012a).

The salient points of the NNL methodology are as follows. The NNL proliferation resistance score, $U(x)$, is defined as:

$$U(x) = -\log(V(x)A(x))$$

where $V(x)$ is defined as the value function and $A(x)$ is defined as the access function. It should be noted that a higher value of $U(x)$ corresponds to a more proliferation resistant system.

The value function consists of the significant quantities (SQs) of specific nuclear materials discharged per GWy. From International Atomic Energy Agency (2002), one SQ is defined as: 8 kg for plutonium (which contains ≤80% 238Pu) or 233U; 25 kg for highly-enriched uranium (HEU) (235U content >20%); 75 kg for LEU (235U content ≤20%); 10,000 kg for natural U; and 20,000 kg for natural Th. Although other minor actinides (such as neptunium, Americium, and protactinium) can be considered as alternative nuclear materials, of which SQs can be attributed, such nuclei are not considered in this analysis.

From the fuel cycle modelling analysis using ORION, the quantities of uranium and plutonium per discharged amount of SNF for each nuclear fuel cycle, and their respective isotopic vectors, are presented in Tables 14 and 15. The corresponding SQs per GWy for each nuclear fuel cycle are presented in the first row of Table 16 for plutonium and Table 17 for uranium. It should be noted that the 232U content is 884 ppm for the EPR blanket assemblies containing UO$_2$/ThO$_2$, 556 ppm for the AHWR with UO$_2$/ThO$_2$, 2 ppm for the GT-MHR UO$_2$/ThO$_2$ DF assemblies and 4 ppm for the PDF assemblies. The very low values for the GT-MHR can be explained by: (1) the increased amount of 235U yielding a lower neutron fluence (a factor of ~3 less than for the AHWR, corresponding to ~1/9 the amount of 232U), (2) a smaller fast neutron component of the spectrum (yielding an equilibrium 232U cross-section 1/3 that of the AHWR), and (3) the low discharge burn-up for the driver fuel.

The access function is defined as follows:

$$A(x) = \frac{1}{\sqrt{10}}\exp(|MT| + |TD| + |PC| + |PT| + |DP|)$$

where each of the terms listed in square brackets corresponds to accessibility barriers defined by the Proliferation Resistance and Physical Protection Evaluation Methodology Working Group of the Generation IV International Forum (Generation IV International Forum, 2006). An abridged overview of these indicators, based on the aforementioned reference, with justification of the values used in this study, is presented below.

\textsuperscript{5} LCOE is commonly represented in units of US$/MWh, whereas LFCC is commonly represented in units of US$/kWh. The conversion between these units is: 1US$/MWh = 0.1US$/kWh.
MT describes the material type on a five-point Likert scale. In decreasing desirability, the material types are: HEU \( [MT = 1] \), weapons-grade plutonium (WG-Pu) \( [MT = 2] \), reactor-grade plutonium (RG-Pu) \( [MT = 3] \), deep-burn plutonium (DB-Pu) \( [MT = 4] \), and LEU \( [MT = 5] \).

For plutonium: in Generation IV International Forum (2006), WG-Pu is ascribed as typically containing \( \sim 94\% \) fissile plutonium content (i.e. \( ^{239}\text{Pu} \) and \( ^{241}\text{Pu} \)), RG-Pu as typically containing \( \sim 70\% \) fissile plutonium content, and DB-Pu as typically containing \( \sim 43\% \) fissile plutonium content. Other works also look at the isoto-

### Table 12
Parameters used in calculating the levelised cost of electricity.

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Lower estimate</th>
<th>Middle estimate</th>
<th>Upper estimate</th>
<th>Distribution</th>
</tr>
</thead>
<tbody>
<tr>
<td>Overnight capital construction cost ($/kWe) for EPR</td>
<td>3860¹</td>
<td>6442²</td>
<td>9861³</td>
<td>Triangular</td>
</tr>
<tr>
<td>Fixed O&amp;M costs ($/kWe.yr) for EPR</td>
<td>45.00</td>
<td>56.25</td>
<td>67.50</td>
<td>Uniform</td>
</tr>
<tr>
<td>Variable O&amp;M costs ($/kWh) for EPR</td>
<td>0.23</td>
<td>0.27</td>
<td>0.30</td>
<td>Uniform</td>
</tr>
<tr>
<td>Decommissioning cost ($/kWe) for EPR</td>
<td>644</td>
<td>805</td>
<td>966</td>
<td>Uniform</td>
</tr>
<tr>
<td>Overnight capital construction cost ($/kWe) for GT-MHR</td>
<td>3818</td>
<td>5455</td>
<td>8180</td>
<td>Triangular</td>
</tr>
<tr>
<td>Fixed O&amp;M costs ($/kWe.yr) for GT-MHR</td>
<td>105</td>
<td>131</td>
<td>157</td>
<td>Uniform</td>
</tr>
<tr>
<td>Variable O&amp;M costs ($/kWh) for GT-MHR</td>
<td>0.23</td>
<td>0.27</td>
<td>0.30</td>
<td>Uniform</td>
</tr>
<tr>
<td>Decommissioning cost ($/kWe) for GT-MHR</td>
<td>344</td>
<td>430</td>
<td>516</td>
<td>Uniform</td>
</tr>
<tr>
<td>Overnight capital construction cost ($/kWe) for AHWR</td>
<td>4136</td>
<td>5910</td>
<td>8862</td>
<td>Triangular</td>
</tr>
<tr>
<td>Fixed O&amp;M costs ($/kWe.yr) for AHWR</td>
<td>45</td>
<td>101</td>
<td>157</td>
<td>Uniform</td>
</tr>
<tr>
<td>Variable O&amp;M costs ($/kWh) for AHWR</td>
<td>0.23</td>
<td>0.27</td>
<td>0.30</td>
<td>Uniform</td>
</tr>
<tr>
<td>Decommissioning cost ($/kWe) for AHWR</td>
<td>591</td>
<td>739</td>
<td>887</td>
<td>Uniform</td>
</tr>
</tbody>
</table>

¹ Lower estimate for the EPR CAPEX cost assumed for Flamanville 3 in International Energy Agency (2010).
² Middle estimate for the EPR CAPEX cost taken from the one of the Hinkley Point C reactors (British Broadcasting Corporation, 2013).
³ Upper estimate for the EPR CAPEX cost taken from “Option D” in Table 3 of Harris et al. (2012) for one of the Hinkley Point C reactors.

### Table 13
Breakdown of the LCOE for the four nuclear fuel cycles under consideration, giving mean values (and uncertainties to one standard deviation). Reactor cost also accounts for decommissioning costs. Fuel cycle costs accounts for both front- and back-end costs and are values taken from Section 6.1.

<table>
<thead>
<tr>
<th>Option</th>
<th>Levelised reactor cost ($/MWh)</th>
<th>Levelised O&amp;M cost ($/MWh)</th>
<th>Levelised fuel cycle cost ($/MWh)</th>
<th>Levelised cost ($/MWh)</th>
</tr>
</thead>
<tbody>
<tr>
<td>EPR with UO₂</td>
<td>88 ± 16</td>
<td>23 ± 3</td>
<td>10 ± 2</td>
<td>121 ± 16</td>
</tr>
<tr>
<td>EPR with UO₂/ThO₂</td>
<td>88 ± 16</td>
<td>23 ± 3</td>
<td>11 ± 3</td>
<td>122 ± 17</td>
</tr>
<tr>
<td>AHWR with UO₂/ThO₂</td>
<td>84 ± 13</td>
<td>41 ± 13</td>
<td>12 ± 2</td>
<td>137 ± 18</td>
</tr>
<tr>
<td>GT-MHR with UO₂/ThO₂</td>
<td>78 ± 12</td>
<td>53 ± 6</td>
<td>26 ± 5</td>
<td>157 ± 14</td>
</tr>
</tbody>
</table>

### Table 14
Quantities of uranium heavy metal and its isotopic composition, per discharge, for the four nuclear energy systems. For the Th–U-fuelled EPR, the compositions of the seed [S] and blanket [B] assemblies are treated separately. For the Th–U-fuelled GT-MHR, the compositions of the once-irradiated driver fuel [DF] and twice-irradiated post-driver fuel [PDF] are treated separately.

| Reactor | Quantities Vector (%)
<table>
<thead>
<tr>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>( U_{\text{TOT}} ) (t)</td>
</tr>
<tr>
<td>EPR with UO₂</td>
<td>39.6</td>
</tr>
<tr>
<td>EPR [S] with UO₂</td>
<td>9.28</td>
</tr>
<tr>
<td>EPR [B] with UO₂/ThO₂</td>
<td>12.9</td>
</tr>
<tr>
<td>AHWR with UO₂/ThO₂</td>
<td>0.97</td>
</tr>
<tr>
<td>GT-MHR [DF] with UO₂/ThO₂</td>
<td>1.16</td>
</tr>
<tr>
<td>GT-MHR [PDF] with UO₂/ThO₂</td>
<td>1.23</td>
</tr>
</tbody>
</table>

### Table 15
As Table 14, but with quantities of plutonium heavy metal and its isotopic composition, per discharge, for the four nuclear energy systems.

| Reactor | Quantities Vector (%)
<table>
<thead>
<tr>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>( \text{Pu}_{\text{TOT}} ) (kg)</td>
</tr>
<tr>
<td>EPR with UO₂</td>
<td>516</td>
</tr>
<tr>
<td>EPR [S] with UO₂</td>
<td>217</td>
</tr>
<tr>
<td>EPR [B] with UO₂/ThO₂</td>
<td>462</td>
</tr>
<tr>
<td>AHWR with UO₂/ThO₂</td>
<td>14.2</td>
</tr>
<tr>
<td>GT-MHR [DF] with UO₂/ThO₂</td>
<td>23.6</td>
</tr>
<tr>
<td>GT-MHR [PDF] with UO₂/ThO₂</td>
<td>37.9</td>
</tr>
</tbody>
</table>
pic properties of plutonium grades: (1) Mark et al. (2009) gives typical values of 94.2% fissile plutonium content (inclusive of \(^{241}\text{Am}\)) for WG-Pu, 69.4% RG-Pu (from U-based fuel, enriched to 3.0\% \(^{235}\text{U}\) with a discharge burn-up of 33 GWd/t), and spent \(^{233}\text{U}\) limit, \(^{233}\text{Pa}\) uranium, as described in Eq.(7). N.B. A higher NNL mass of the given isotope(s), shows the redefined approach. A significant question remains as to the variation of desirability of the plutonium in the SNF. In this analysis, it is as-
sumed that the plutonium isotopic vectors of each nuclear fuel cycle (including seed/blanket assemblies and driver/post-driver blocks) correspond to RG-Pu \([\text{MT} = 3]\). For the discharged uranium, the question remains as to the potential for re-enrichment with and without separation by laser enrichment techniques. This significantly affects whether such material can be considered as LEU \([\text{MT} = 5]; 1 \text{ SQ} = 75 \text{ kg}\) or equivalent to natural uranium \((^{238}\text{U (eq)})\) \([\text{MT} = 5]; 1 \text{ SQ} = 10,000 \text{ kg}\).

The remaining indicators \((\text{TD, PC, PT, DP})\) are described by qualitative descriptors in the form of five-point Likert scales with the scores: “very low” [1], “low” [2], “medium” [3], “high” [4], and “very high” [5]. There is the potential for more quantitative indicators (requiring analyses containing confidential/classified information) to be used. Given that this a comparative study that is limited to four open nuclear fuel cycles, and that analyses for some of the technologies using this methodology exist in UK National Nuclear Laboratory (2009), differences in the scoring of these components will be based on this previously published work. \(TD\) corresponds to “technical difficulty” which, from a state perspective, could be considered as the probability of the proliferation pathway failing. From a nuclear fuel cycle perspective this can include material properties such as the spontaneous neutron emission rate, decay heat, radiotoxicity, and the material form. From UK National Nuclear Laboratory (2009) for a reference U-fuelled PWR (with discharge burn-up of 45 GWd/t), \(TD\) was described as “medium” \([\text{TD} = 3]\).

For the four nuclear fuel cycles considered here, each of the systems provide more than one SQ of plutonium. It should be noted that the fuel cycle modelling approach for the AHWR assumes that there is an annual discharge of fuel per year (totalling one tenth of the core). In practice, it is proposed that the refuelling scheme would be based around an “on-line” approach, and so each discharge could potentially be performed on a cluster-by-cluster approach. A significant question remains as to the variation of desirability of the plutonium in the SNF. In this analysis, it is assumed that the plutonium isotopic vectors of each nuclear fuel cycle (including seed/blanket assemblies and driver/post-driver blocks) correspond to RG-Pu \([\text{MT} = 3]\). For the discharged uranium, the question remains as to the potential for re-enrichment with and without separation by laser enrichment techniques. This significantly affects whether such material can be considered as LEU \([\text{MT} = 5]; 1 \text{ SQ} = 75 \text{ kg}\) or equivalent to natural uranium \((^{238}\text{U (eq)})\) \([\text{MT} = 5]; 1 \text{ SQ} = 10,000 \text{ kg}\).

The remaining indicators \((\text{TD, PC, PT, DP})\) are described by qualitative descriptors in the form of five-point Likert scales with the scores: “very low” [1], “low” [2], “medium” [3], “high” [4], and “very high” [5]. There is the potential for more quantitative indicators (requiring analyses containing confidential/classified information) to be used. Given that this a comparative study that is limited to four open nuclear fuel cycles, and that analyses for some of the technologies using this methodology exist in UK National Nuclear Laboratory (2009), differences in the scoring of these components will be based on this previously published work. \(TD\) corresponds to “technical difficulty” which, from a state perspective, could be considered as the probability of the proliferation pathway failing. From a nuclear fuel cycle perspective this can include material properties such as the spontaneous neutron emission rate, decay heat, radiotoxicity, and the material form. From UK National Nuclear Laboratory (2009) for a reference U-fuelled PWR (with discharge burn-up of 45 GWd/t), \(TD\) was described as “medium” \([\text{TD} = 3]\). For the plutonium component, \(TD\) is also described as medium for the reference U-fuelled EPR and the seed

\[
\frac{M^{233}\text{U}}{M^{238}\text{U}} + 0.6M^{235}\text{U} \leq 0.12
\] (7)
assemblies of the Th–U-fuelled EPR. For the blanket assemblies of the Th–U-fuelled EPR and for the Th–U-fuelled AHWR, TD is classed as "high" [TD = 4] due to the enhanced content of $^{238}$Pu, which provides additional complication to conventional aqueous-based reprocessing techniques, and the enhanced radiotoxicity associated with the decay of $^{232}$U contained within the nuclear fuel. For the Th–U-fuelled GT-MHR, TD is ascribed as "high" [TD = 4] due to the additional process steps required to access the nuclear fuel kernels. In UK National Nuclear Laboratory (2009) TD for the HTGR is attributed as "very high" [TD = 5], this stronger classification being mainly due to the discharged material being equivalent to DB-Pu. For the uranium component, each of the values is increased by a single point on the Likert scale due to the additional re-enrichment step required, which, due to the content of $^{232}$U and $^{235}$U in some of the nuclear energy systems, may only be possible by using laser enrichment techniques.

$PC$ corresponds to "proliferation cost" which could be described in terms of a country's military expenditure per year. As this study is not state-specific using a more pertinent metric would involve the differences within the infrastructure required. As only open nuclear fuel cycles are considered here, it is assumed that the reprocessing infrastructure required to generate a SQ of either plutonium or uranium, even including the additional process steps associated with the GT-MHR-based fuel, for each nuclear fuel cycle would be near identical. Therefore, for all fuel cycles $PC$ is scored identically as "medium" [$PC = 3$].

$PT$ corresponds to "proliferation time" which is described as the amount of time required to generate a SQ of material. Although reprocessing stages can potentially be performed coverts within a 3–12 month time frame (Generation IV International Forum, 2006), the time required to build up the infrastructure and expertise can take many years. In this analysis, the differences in reprocessing infrastructure required for the different fuel cycles under consideration are assumed to be negligible and therefore each cycle is scored identically as "high" [$PT = 4$].

$DP$ corresponds to "detection probability" which is described as the probability that illicit movement of material is detected (in concordance with IAEA safeguards). From a fuel cycle perspective, this boils down to a series of factors including: the refuelling scheme, the size and composition of nuclear fuel assembly, the ability for non-safeguarded material to be irradiated, and the external radiation field. For the reference EPR this is considered as "high" [$DP = 4$] due to: (1) the large volume of the fuel assemblies; (2) the impossibility of on-line refuelling, and the length of time (over a year) between refuelling outages; (3) thermal-hydraulic instabilities from irradiating driver assemblies; and (4) the sizeable external radiation field due to the volume of material irradiated. These same factors are associated with the seed assemblies of the Th–U-fuelled EPR, while, for the blanket assemblies, $DP$ is assessed as being "very high" [$DP = 5$] due to the presence of a sizeable $^{232}$U component whose daughter product $^{208}$TI emits a 2.6 MeV $\gamma$ ray that is difficult to shield. For the Th–U-fuelled AHWR, although the refuelling scheme is typically described as being "on-line", given that fuel can be continuously shuffled around the reactor, which lessens $DP$, this is offset by: (1) the low number of nuclear fuel clusters discharged per year; (2) the sizeable volume of each nuclear fuel cluster; and (3) the high $^{232}$U content. Therefore $DP$ is also attributed as being "very high". For the GT-MHR, due to the lower burn-up of the driver fuel, the smaller amounts of $^{232}$U generated, and, most importantly, the fact that the 7.93 m nuclear fuel hexagonal block has to be separated into ten smaller blocks before being disposed of, $DP$ for both driver and post-driver fuel is listed as "high" [$DP = 4$].

The results from this discussion are summarised in Table 16 for plutonium and Table 17 for uranium. In summary, although open-cycle U-fuelled PWRs can be considered highly resistant against proliferation, it is evident that there are small proliferation resistance advantages for the SNF discharged from the open-cycle Th–U-fuelled EPR and open-cycle Th–U-fuelled AHWR, due primarily to the lower amount of plutonium discharged per GWy. For the open-cycle Th–U-fuelled GT-MHR, the low discharge burn-up contributes towards a greater amount of plutonium being generated per GWy. For the Th–U-fuelled EPR, the limiting factor in terms of its proliferation resistance score comes from the discharged seed assemblies, although this score is higher than that of the U-fuelled EPR. For the AHWR, due to all the assemblies being comprised of blended $\text{UO}_2/\text{ThO}_2$, the proliferation resistance is comparably greater. It should be stressed that the overall proliferation score is on a qualitative non-linear scale. The benefits arising due to the nature of the SNF have to be set against the enhanced separative work capacity per kWh for these fuel cycles, the need for $^{232}$U enriched to 20%, and the requirement of heavy water for the AHWR.

8. Conclusions

This paper has set out to compare a number of Th–U-fuelled nuclear energy systems operating with an open nuclear fuel cycle to a reference system. Reactor physics simulations were performed for one U-fuelled system (an AREVA EPR, which was treated as the reference) and three Th–U-fuelled systems (an AREVA EPR fuelled with a seed-blanket configuration, an Indian AHWR, and General Atomics' GT-MHR). Shielded reaction cross-sections from these simulations were then used by the NNL fuel cycle modelling code ORION to determine the front-end fuel cycle requirements (in terms of the uranium, thorium, and separative work capacity required) and characteristics of the back-end (in terms of the amount of spent fuel discharged, its isotopic composition and corresponding characteristics). These values have subsequently been used to assess the properties of the spent fuel, fuel cycle economics and its proliferation resistance.

In terms of the material flow, it is evident that although there is a smaller amount of uranium contained within the nuclear fuel, all of the Th–U-fuelled systems within this study, require more separative work units per kWh than the U-fuelled benchmark. This is predominantly due to the requirement of uranium enriched to ~20%. For the AHWR, ~6% less uranium ore (per kWh) is required than for the reference U-fuelled EPR. For the Th–U-fuelled EPR and Th–U-fuelled GT-MHR, more uranium ore is required than for the reference U-fuelled EPR. Although this is not an exhaustive study of all possible open cycle Th–U-fuelled nuclear energy systems, it is evident that significantly lower uranium enrichments are required for a notable advantage in resource utilisation to be realised in open cycles. Even if such designs can be realised, any advantage may not be significant enough to warrant a switch in fuelling options, particularly if the ratio of burn-up to uranium enrichment required for U-fuelled systems can be further enhanced. It should be stressed that the Th–U-fuelled GT-MHR needs further development and optimisation to ensure that there is greater utilisation of the nuclear fuel.

In terms of the spent fuel generated, the Th–U-fuelled EPR yields the smallest volume of SNF per kWh generated. This is principally due to the very long dwell times of blanket assemblies (~13.5 years). Correspondingly, the quantities of minor actinides within the discharged blanket assemblies are greater than for the U-fuelled seed assemblies. This could potentially complicate the encapsulation and long-term disposal. Overall, per kWh generated there are minimal differences in the decay heat and radiotoxicity of the fuel for each system considered.

In terms of the economics, the U-fuelled EPR has the lowest LFCC. This is predominantly due to the greater fuel fabrication
costs and additional separative work required by the Th–U-fuelled systems. In the analysis it is noted that the fuel fabrication costs for the Th–U-fuelled EPR seed and blanket fuel pins and the AHWR could be underestimated due to the need for novel fabrication techniques and variations in the ratio of UO₂ and ThO₂ required. Such variations are exemplified with the estimates for the GT-MHR fuel fabrication costs. In terms of the LCOE, although accurate estimates for capital and O&M are difficult to come by, we suggest that the U-fuelled EPR yields the lowest LCOE, on the assumption that the reactor is constructed in the West. We stress that this analysis should be treated only as indicative.

In terms of the proliferation resistance of the SNF, small advantages are observed for the Th–U-fuelled EPR and greater advantages are noted for the Th–U-fuelled AHWR. The low discharge burn-up of the GT-MHR design compromises its proliferation resistance score.

Overall, it appears that there is little merit in incorporating thorium into nuclear energy systems operating with open nuclear fuel cycles. In this study, three Th–U-fuelled reactor systems were considered. Two of these offer benefits in comparison to the reference U-fuelled system in terms of proliferation resistance, but this must be set against their need for uranium enriched to ~20% ²³⁵U and thus more separative work capacity, with limited savings in uranium ore and waste generated, than for the U-fuelled reference. The economics also appear to favour the reference case. These downsides are in addition to various technical and licensing barriers for such reactors and fuel cycle infrastructures to be commissioned, as outlined in detail in Nelson (2012).

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