The Use of Zirconium Hydride Blankets in a Minor Actinide/Thorium Burner Sodium-Cooled Reactor for Void Coefficient Control with Particular Reference to UK's Plutonium Disposition Problem

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The use of zirconium hydride (Th-$ZrH_{1.6}$) blankets in a thorium-fuelled sodium-cooled reactor for void reactivity control with particular reference to UK's plutonium disposition problem is proposed and considered. It is shown that, with the use of such blankets, a mild moderation effect is produced during voiding which compensates for the general hardening of the spectrum, enabling a net negative void coefficient at pin level to be attained without the need to rely on traditional neutron leakage enhancement techniques or neutron poisons, and with negligible impact on transmutation capabilities. One important difference in comparison with the traditional methods is that the void coefficient is obtained at the pin level, eliminating or mitigating substantially the spatial dependencies on the location of the void. Combining the use of such blankets with a suitable n-batch fuelling scheme yields a negative void reactivity coefficient throughout the life of fuel. Additional research and development are required to explore further this concept's potential.

Keywords. *fast reactors, zirconium hydride blanket, safety analysis, sodium void reactivity, plutonium disposition*

I. INTRODUCTION

The UK has the largest civil stockpile of separated plutonium in the world, totalling some 112 tonnes, most of it from reprocessing spent fuel over the years. This represents a major storage liability and a proliferation risk, costing the government ∼£80M/yr [1]. The question of what to do with this plutonium has vexed successive governments for decades. Plutonium management options include continued long-term storage, disposal in a long-term repository, and reuse as nuclear fuel [2]. A number of different plutonium storage options have been proposed [3].

Following a public consultation, in December 2011 the UK Department of Energy & Climate Change announced that the favored management strategy is to use the plutonium in mixed-oxide (MOX) fuel at an unspecified future date [4]. This fuel could be used to fuel conventional reactors, such as those planned for the UK's new nuclear build. This strategy would convert the plutonium into spent fuel that would still need storage, but it would no longer pose a proliferation risk.

However, manufacturing MOX fuel is not straightforward and the UK's recent record is not good. A MOX plant at Sellafield, part of the Thermal Oxide Reprocessing Plant (THORP) facility, cost £1.2G but only produced five tonnes of fuel per year for its first five years, despite a 120-tonne-per-year design capacity, and closed down due to lack of business.

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In order to minimize fuel fabrication costs associated with manufacturing MOX, the use of metallic fuel has been recently proposed using a fast burner reactor, taking advantage of the wide range of conversion characteristics achievable in a fast spectrum system [5–7], and several reactor designs have been developed for weapons-grade plutonium disposition [8]. However, the void coefficient in sodium-cooled fast reactors is positive, which means that reactivity increases if loss of the sodium coolant occurs due to boiling or gas intrusion. Safety concerns are exacerbated further by the fact that the delayed neutron fraction in fast reactors is smaller than for thermal reactors [9].

Traditional measures to maintain a negative void reactivity coefficient are mostly based on the use of neutron leakage enhancement techniques, such as the use of neutron streaming channels [10], the installation of fixed absorbers, increasing the operating reactivity margin, or increasing the fuel enrichment [11]. However, these techniques, and especially those relying on leakage enhancement, incur an important economic penalty, because of the reduction of available fissile material in the core.

Moreover, leakage enhancement techniques have a strong spatial dependence on the location of the void: a stream channel yielding a negative void coefficient if a void occurs in the center of the core could be useless if a void occurs in the core periphery. Weaver et al. [12, 13] have shown that the local void reactivity worth is ∼$3–5 worth of positive reactivity due to the very small delayed neutron fraction (∼2.5 times smaller than in a light water reactor). Thus, the nuclear designer is compelled to apply streaming fuel assemblies (or other neutron leakage enhancement techniques) extensively throughout the core, with the inevitable consequent load-factor penal-

ties.

An alternative approach to mitigate the positive void coefficient has been proposed with the use of thorium-based fuel. Thorium-based fuel is an especially attractive option not only because of its beneficial impact on the void coefficient but also because its use results in the production of fewer transuranics (TRU) than for uranium-based fuels [14, 15].

In the specific context of the UK's plutonium disposition problem, a thorium-based fuel in a PRISM sodium-cooled fast reactor has been suggested, because of the potential performance benefits arising from increasing the TRU enrichment in the fuel (replacing the uranium with ^{232}Th) and with a reduced void coefficient in comparison to uranium-fuelled cores [16, 17]. However, despite an improvement in the void coefficient, it was not possible to maintain a negative void reactivity coefficient throughout the life of fuel, thus limiting the maximum attainable burn-up. Moreover, the calculations in this study [16] were based on the assumption of homogeneous voiding.

In this paper the use of zirconium hydride (Th-$ZrH_{1.6}$) blankets to improve the void reactivity coefficient is proposed. The simple idea that lies behind the use of hydride blankets is to take advantage of the shape of the curve of the number of fission neutrons produced per absorption in the fuel, a parameter normally known as η, as a function of neutron energy, and especially for ^{239}Pu, which is the main contributor to the positive void coefficient. Briefly, when a void occurs, the neutron spectrum is hardened and η increases (see Fig. 1). However, with the presence of hydride blankets at top and bottom (in a so-called 'parfait' configuration), some mild moderation may be produced when a void occurs, and depending on the size of the blanket, this mild moderation could compensate the general hardening of the spectrum, thus resulting in a net negative void coefficient.

The use of zirconium hydride in uranium-fuelled fast reactors has been proposed in the past with the aim of reducing the positive void coefficient [18, 19]; however, a poor breeding ratio was reported because of the decrease in the value of η due to spectrum softening. In contrast, in the context of the use of thorium-based fuel and plutonium disposition, where the destruction of ^{239}Pu is sought, this is a point in favor of the use of such blankets. Moreover, the poor breeding of ^{239}Pu is compensated by the breeding of the isotope ^{233}U, which is able to breed effectively in hard and epithermal spectrums [20].

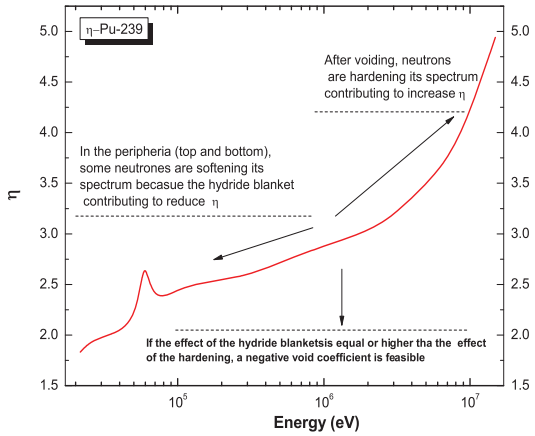

FIG. 1: The parameter η for ^{239}Pu.

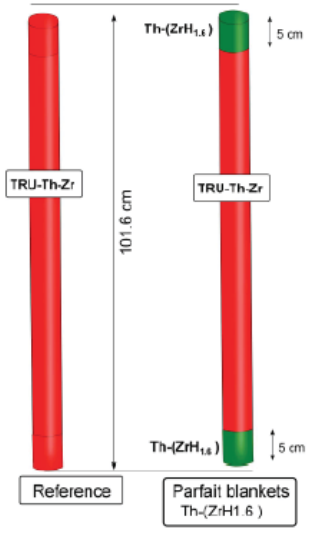

FIG. 2: Reference thorium-fuelled unit cell and 'parfait' core using Th-($ZrH_{1.6}$) blankets.

II. REFERENCE CORE

The core considered in this paper features the same tight lattice radial dimensions as the PRISM sodium-cooled fast reactor designed by General Electric Hitachi Nuclear Energy (GEH) proposed for construction at Sellafield in the UK as a means of dealing with a portion of the UK's civil plutonium stockpile [14].

The reference core unit cell consists of a 101.6 cm long active fuel height which is composed entirely of TRU(30%)-Th-Zr. For the 'parfait' version, the core unit cell is divided into an axially-zoned configuration composed of 3 axial sections: a central seed of length 91.6 cm composed by TRU(30%)-Th-Zr, a 5 cm long bottom Th-$ZrH_{1.6}$ blanket and a 5 cm top Th-$ZrH_{1.6}$ blanket. The comparison between the reference and the proposed parfait design is shown in Fig. 2.

The fuel pellet diameter is 0.547 cm and the distance between the centers of the rods (or pitch), arranged in a hexagonal lattice, is 1.191 cm. The 0.06 cm thick cladding is made of HT9 ferritic-martensitic steel. The inlet and exit sodium density is assumed to be 0.8 g/cm^3. The fuel and blanket compositions at beginning of life (BOL) are given in Table 1.

TABLE I: Fuel and blanket compositions at beginning of life.

TRU(30%)-Th-Zr fuel	
Component	Proportion
^{238}Pu	0.00741
^{239}Pu	0.13737
^{240}Pu	0.06783
^{241}Pu	0.03954
^{242}Pu	0.02115
^{237}Np	0.01638
^{241}Am	0.00150
^{242m}Am	0.00003
^{243}Am	0.00576
^{242}Cm	0.00054
^{243}Cm	0.00003
^{244}Cm	0.00234
^{245}Cm	0.00012
^{232}Th	0.6
Zr	0.1
Total Pu	**0.2733**
Total TRU	**0.3000**
Th-$ZrH_{1.6}$ blanket	
Component	Proportion
^{232}Th	0.400
Zr	0.589
H	0.011
Total Pu (parfait core)	**0.2460**
Total TRU (parfait core)	**0.2700**

A. Computer codes

The preliminary feasibility study presented in this work is based on a 3D unit-cell analysis. The computations are performed with the SCALE® code system using TRITON for the driver code [21], the KENO-V Monte Carlo code for neutron transport calculations [22] using ENDF/B-V derived 238 energy group cross-sections, and the ORIGEN-S code for burn-up calculations [23]. The active fuel portion of the fuel pin is divided into sixty axial depletion zones: 15 in the lower blanket, 30 in the fissile area and 15 in the upper blanket. The core performance is estimated from the single unit-cell burn-up analysis.

III. RESULTS AND DISCUSSION

Figure 3 shows a comparison between the void coefficient for the reference case and the parfait fuel design using Th-$ZrH_{1.6}$ blankets. It can be seen that a substantial reduction in the void coefficient of up to \$0.6 is attainable. Moreover, this reduction in the void coefficient was obtained without significantly affecting the plutonium consumption rates, as shown in Fig. 4.

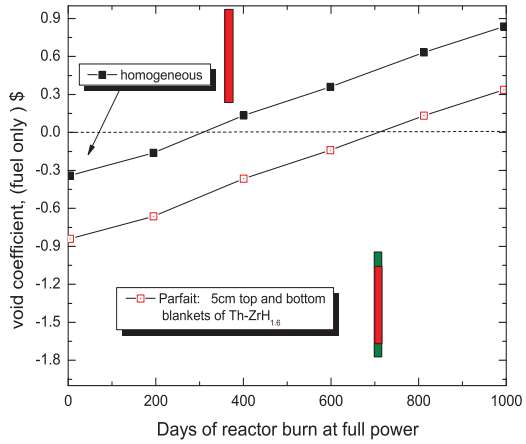

FIG. 3: The effect on the void coefficient as function of burn-up time of using Th-($ZrH_{1.6}$) blankets.

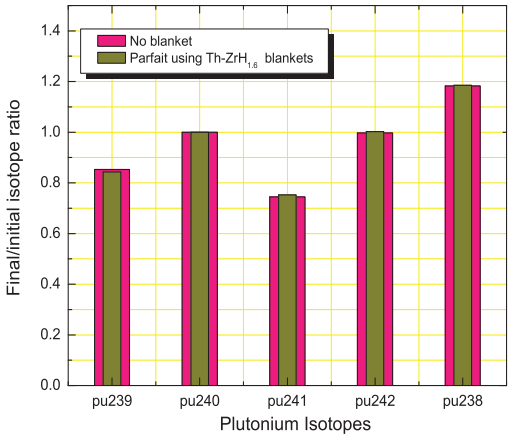

FIG. 4: The ratio of final (after 1000 full power days) to initial plutonium isotope quantities with and without Th-$ZrH_{1.6}$ blankets.

A. *n*-batch fuelling

Although a substantial improvement in the void coefficient resulted from the use of Th-$ZrH_{1.6}$ blankets (see Fig. 3), nevertheless the void coefficient becomes positive after 650 full power days or thereabouts. However, an additional improvement can be achieved if the use of

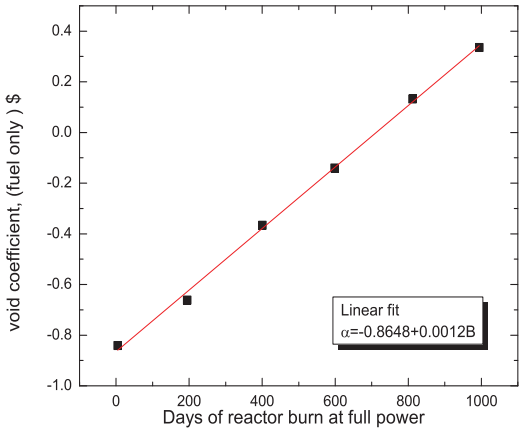

FIG. 5: Linear best fit of the void coefficient as a function of burn-up.

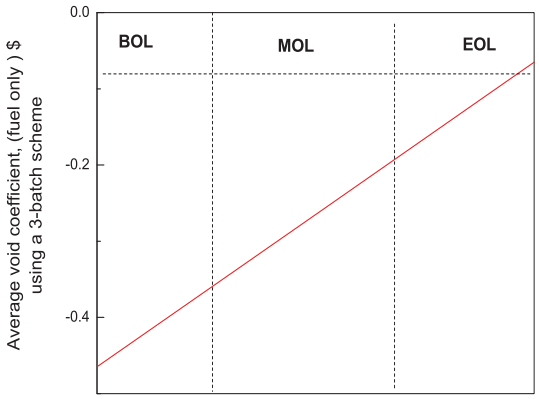

FIG. 6: Variation over a cycle of the average core void coefficient using a 3-batch fuelling scheme.

Th-$ZrH_{1.6}$ blankets is applied in combination with a n-batch fuel management scheme.

First, it is apparent from Fig. 3 that the relationship between void coefficient (α) and burn-up (B) is approximately linear. Therefore a linear best fit can be found as (see Fig. 5):

$$\alpha = \alpha_0 + AB \tag{1}$$

yielding $A = 0.0012$ and $\alpha_0 = -0.8648$.

For the general case, and for preliminary calculations, we can assume that all fuel operates at the same core-average power density. Then, for an n-batch core at steady state, at the end of a burn-up cycle the freshest batch will have burn-up B_c, the next oldest batch $2B_c$, etc., and the oldest batch, which is ready for discharge, has an accumulated burn-up $nB_c = B_d$, where B_d is the burn-up of the batch at discharge. The mean void coefficient of the mixture (α_s) can be computed merely by averaging the batch void reactivities:

$$\alpha_s = \frac{1}{n}\sum_{i=1}^{n}\alpha_i \tag{2}$$

The end-of-cycle (EOC) void reactivity α_i of batch i given by Eq. (1) is then:

$$\alpha_i = \alpha_0 + iAB_c \tag{3}$$

and the core EOC void reactivity α_n for an n-batch core given by Eqs. (2) and (3) is:

$$\alpha_n = \frac{1}{n}\left[n\alpha_0 + \frac{n(n+1)}{2}AB_c\right] = \alpha_0 + \frac{(n+1)}{2}AB_c \tag{4}$$

where the standard result $\sum_{i=1}^{n} i = \frac{n(n+1)}{2}$ has been used.

Figure 6 shows the variation of the core void reactivity over a cycle for a 3-batch fuelling scheme for the values of A and α_0 in the case considered. It is clear that α_n is negative not only at beginning of cycle (BOC) but also at middle of cycle (MOC) and EOC.

IV. CONCLUSIONS

The use of zirconium hydride (Th-$ZrH_{1.6}$) blankets in a thorium-fuelled sodium-cooled reactor for void reactivity control with particular reference to UK's plutonium disposition problem has been proposed and subjected to an initial assessment based on 3D unit-cell analysis. The following conclusions can be drawn from this work:

- With the use of zirconium hydride (Th-$ZrH_{1.6}$) blankets in a parfait configuration, it is possible obtain a core design with a negative void coefficient at beginning and middle of life.
- Because of the small amount of hydride used, the plutonium transmutation capabilities are not significantly affected by the use of these blankets.
- In combination with a 3-batch fuelling scheme, a core design with a negative void coefficient throughout each cycle is attainable.
- Because the negative void coefficient is obtained at the pin level, spatial dependence on the location of any void is mitigated, in contrast to traditional neutron leakage enhancement techniques.
- Preliminary results are encouraging and motivate further research and development to explore the potential of this concept.

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