Thorium-Plutonium LWR Fuel – Activity Summary

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Thor Energy generates specialized information that is needed for the future licensing of thorium fuels for light water reactors (LWRs). The company assesses that the introduction of thorium as a fertile component for LWR MOX fuels is by far the shortest route toward deriving appreciable energy share from thorium.

The company has two broad lines of work: (i) the design and planning of an irradiation experiment for a candidate thorium-plutonium oxide fuel (thorium-MOX), (ii) the design and modelling of thorium-MOX and thorium-^{233}U fuel assemblies for BWRs.

Irradiation Testing of Prototypical Thorium-MOX Fuel

Thor Energy has designed a sophisticated data collection program in which a number of thorium-plutonium oxide fuel pins will be irradiated in simulated LWR conditions in the fuel-testing reactor in Halden, Norway. The fuel will be typical of that which can be fabricated commercially as a variant of today's uranium-plutonium (MOX) fuels. The irradiation will be performed by the *Institutt for Energiteknikk* (IFE), operators of the Halden reactor.

Thor Energy is undertaking this thorium fuel test activity in the knowledge that:

- There is a paucity of data on the irradiation behaviour of thorium-plutonium oxide fuels – especially those with characteristics akin to current MOX fuels. Licensing such a fuel will call for high quality data to support the verification & modelling of its behaviour under operating conditions.
- Due to the length of the fuel licensing process, testing activities should be commenced as soon as possible, with a view to lead-test rod/assembly (LTA) testing in a commercial power reactor.

The data collection plan is summarised in the following tables according to the fuel behaviours that need to be characterized [1]. The ultimate goal for the activity is to create a set of quality data that demonstrates that the fuel ceramic operates safely in normal operating conditions. This data will support follow-on testing of thorium-MOX fuel segments in a commercial power reactor – leading to the testing of the fuel in transient conditions. Collectively, this data is vital for the safety licensing of such a new fuel.

Thermal Behaviour	Data Collection to Characterize Behaviour	
	On-line Experimental Measurables	*Data from Pre/PIE Measurements*
Thermal Conductivity Decay	Centerline temperature, including from a pin operating at higher temperature	Thermal conductivity of fresh pellet, Thermal conductivity of spent fuel ceramic
Thermal Pathway Changes	Centerline temperature, Cladding elongation (gap closure)	Neutron radiography, Microscopy

Fission Gas Release (FGR) Behaviour	Data Collection to Characterize Behaviour	
	On-line Experimental Measurables	*Data from Pre/PIE Measurements*
FGR Onset	Rod pressure, Centerline temperature, Heat generation rate (pin power)	Ceramic microstructure examination
FGR Amount	Rod pressure	Rod-puncture assessment
FGR Composition		Rod-puncture gas analysis

Mechanical Behaviour	Data Collection to Characterize Behaviour	
	On-line Experimental Measurables	*Data from Pre/PIE Measurements*
Cracking		Neutron radiography, Gamma scanning
Densification	Fuel-column elongation, Temperature	Resintering
Swelling & Pellet-Cladding Mechanical Interaction (PCMI)	Fuel-column elongation, Cladding elongation, Centerline temperature	Microscopy, Fuel rod profilometry
High Burn-up Re-structuring	Rod pressure, Fuel-column elongation, Cladding elongation, Temperature	Thermal conductivity, Microscopy

Chemical Behaviours	Data Collection to Characterize Behaviour	
	On-line Experimental Measurables	*Data from Pre/PIE Measurements*
Stress Corrosion Cracking		Microscopy & EPMA, Rod puncture gas analysis, Neutron radiography
Oxygen Potential in the Fuel Ceramic		Microscopy & EPMA on phases in the fuel ceramic, XPS if available
Pellet-Cladding Chemical Interaction		Microscopy & EPMA on pellet-cladding interface

The thorium-MOX fuel to be tested in this program is a prototype of commercial MOX fuel in its microstructure and its composition. Fuel behaviours are sensitive to the physical make-up of the fuel pellet, so it is important that a commercially oriented fuel program tests fuel material that is closely matched to that which can be produced industrially. If not, the irradiation data may be inadmissable for demonstrating the safe performance of the fuel. This is even more important for a two-phase plutonium-bearing fuel because there are more variability options in terms of: (i) plutonium isotope vector, (ii) plutonium homogeneity, (iii) americium content, (iv) non-metal impurities (C, N, H).

These 'MOX parameters' exist in addition to the normal pellet properties of density, grain-size and oxygen stoichiometry which also need careful specification and for which manipulation and control is different in a thoria-based ceramic. *Thor Energy* has defined a thorium-plutonium MOX test fuel specification taking careful account of all desired properties, control parameters & logistical restrictions relating to handling plutonium fuels. The thorium-MOX fuel for this experiment will be made using a co-milling process that gives a micro-heterogeneity of plutonium distribution that is similar to the MOX fuel produced in commercial plants in France or the UK.

Thor Energy is building an international Consortium whose members will collectively steer the irradiation experiment, co-fund the undertaking and share all resulting data. The Consortium is open to all interested parties. The irradiation experiment will commence in late 2011 and run for ~5 years.

Thorium Fuel Design for BWRs

Thor Energy has undertaken and sponsored a number of studies into optimal thorium fuel designs for the BWR platform. One key design direction has been to maximize the utilization of plutonium in a BWR fuel assembly. The other direction has been to maximize the conversion of thorium into ^{233}U in a reduced-moderated BWR (RBWR). These optimization modes are quite distinct and invoke very different design strategies.

Thorium-Plutonium BWR Fuel Design

The operation of thorium-MOX BWR fuel has been modelled using the *Studsvik-Scandpower* CASMO-5 – SIMULATE-3 simulation codes[1]. Thorium-MOX in a standard 10-by-10 'GE14-N' assembly was compared with other related oxide fuels in the same geometry [2]: LEU, uranium-MOX, thorium-^{233}U, thorium-LEU and thorium-plutonium + recycled-U from thoria fuel[2]. Burnup profiles for these are shown in Figure One. The flatter depletion

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[1] a multigroup two-dimensional transport theory code for performing depletion calculations & collapsing cross-sections for subsequent input to less rigorous models. The unadjusted JEFF 2.2 cross section library was used in this study.
[2] ~89% ^{233}U and 11% ^{234}U

profile for thorium-MOX fuel is evident – this being a favorable feature. Operating safety indicators were computed, including: power peaking, reactivity coefficients and control rod worths. Findings include:

- Thorium-MOX fuel is neutronically similar to uranium-MOX: power peaking factors are similar (higher than UOX fuel at end-of-life) and control rod worths are similar (lower than for UOX fuel).
- If efficient destruction of stockpile plutonium is sought then thorium-MOX is superior to uranium-MOX, but more plutonium is needed.
- It is not possible to achieve uranium savings using thorium-LEU fuels.
- Moderator temperature coefficients (MTC) and coolant void reactivity coefficients (CVC) for thorium-^{233}U fuels are significantly more positive over the life of the fuel, as compared to other fuel types. Special design measures would be needed to ensure a negative MTC and CVC for such fuels.
- Useful power share of 25 – 30% of energy output is derived from thorium in BWR fuels containing this fertile element. This seems indicative of what can be attained in optimized designs.

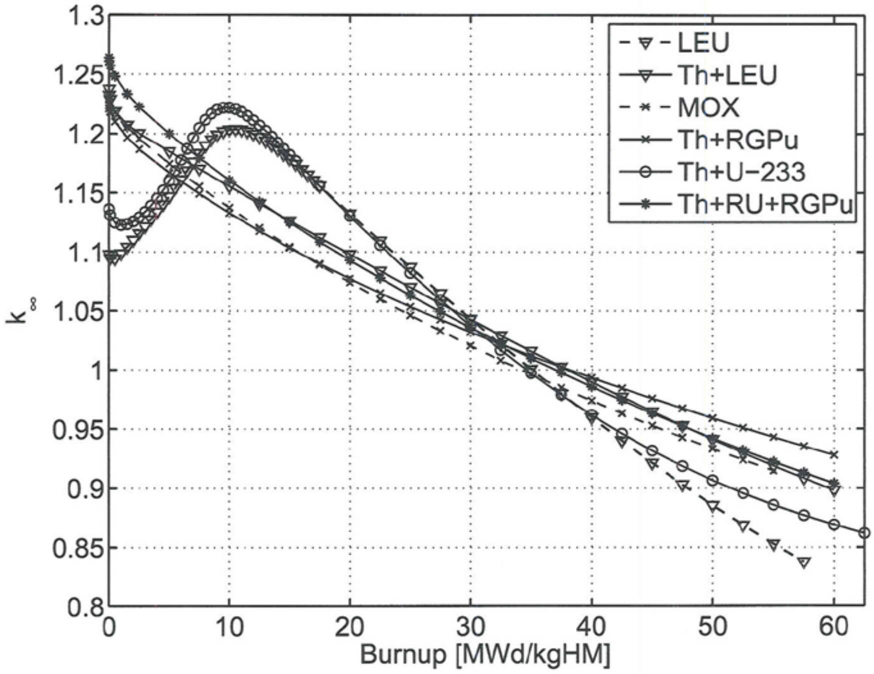

Figure One: Infinite multiplication factor (k_{eff}) dependence on burn-up for different fuels in the 10-by-10 GE14 BWR assembly arrangement.

These results illustrate the broad feasibility of designing viable thorium fuels for BWRs. The similarity of thorium-MOX fuels to uranium-MOX fuels means that there is an excellent basis for the design of thorium-MOX BWR assemblies (*eg*, measures used for today's MOX fuels to address control rod worths and power peaking factors will be applicable for thorium-MOX fuels).

Thor Energy has extended its BWR work to assess the effect of fuel composition, fuel geometry and moderation ratio on plutonium utilization and operating safety margins. Main findings included:

- The highest reactivity (and thus efficiency of plutonium use) for a thorium-MOX BWR fuel is reached with a *moderation ratio (H/HM) that is significantly higher* than those for UOX fuel.
- Higher plutonium fractions tend to give higher efficiency of use of that plutonium – within practical limits.
- Highly moderated fuel designs, however, have a reduced total fuel mass and thus a concommitant higher relative fuel cost.
- The different design options for varying the H/HM ratio – modifying the fraction of coolant or moderator in the fuel assembly – have significantly different effects on reactivity coefficients.
- Generally, when the H/HM ratio exceeds the value giving the highest reactivity for the thorium-MOX fuel, reactivity coefficients become positive.

A firm basis now exists for continuing efforts to design thorium-MOX fuels that can demonstrate high plutonium utilization, assembly fabricability and good cycle length. This requires finding an appropriate balance between maximum fuel content and having a large amount of moderator (water) in the assembly.

Thorium-^{233}U fuel for a Reduced-Moderation BWR

There is continuing motivation to explore the limits of fissile breeding from thorium fuels in thermal spectrum systems. A self-sustaining ^{233}U – thorium fuel cycle is a highly desirable trans-actinide free goal. Work has been done to this end, most notably in the Indian program [*eg*, 3], for the 'CANDU' heavy water reactor platform [*eg*, 4], and in the Shippingport light water breeder reactor program [5].

Thor Energy undertook to explore the operation of thorium-^{233}U oxide fuel in a Gen III+ reduced-moderation BWR (RBWR)[3]. This reactor platform should be well suited for achieving high ^{233}U conversion factors due to it's hard / epithermal neutron spectrum. It was designed as a platform for flexible uranium-plutonium fuels in which high conversion or actinide destruction can be achieved. Physically it is based on the ABWR architecture but has a shorter, flatter 'pancake' shaped core and a tight lattice to ensure sufficient fast neutron leakage and thus a negative void reactivity coefficient in a loss-of-coolant scenario.

The study aimed to shed light on: (i) the characteristics of uranium that is multi-recycled in a RBWR – and specifically which uranium isotopes attain equilibrium, (ii) the extent to which ^{233}U conversion can be maximized without trading-off too much achievable burn-up. Uranium from the fifth recycle was taken as the fissile feed for studies on RBWR fuel configurations.

Neutronic analyses were performed using the MCNP-4C transport code [7] and a 3D model of the RBWR core developed from the open literature. A coupled code gives burn-up calculations and uses the JEFF-3.1 evaluated cross-section library. A core configuration designed for a uranium-plutonium fuel was initially used, however this is adequate for demonstrating the evolution of the uranium vector.

Main results include:

- The multi-recycled uranium vector was found to attain approximate equilibrium after 6 – 7 cycles, though not with the content of ^{236}U and ^{238}U which are seen to rise with each recycle stage.
- The equilibrium content of ^{232}U is 1700ppm, which is high from the point of view of radiation protection requirements, but it is reassuring that it saturates at some level.
- Equilibrium for ^{234}U is still not attained after 5 cycles but is probably reached after 7 cycles at ~27at%.
- The uranium vector reactivity worth deteriorates with each successive re-cycle and top-up loading is needed at each stage such that the uranium content in the fuel rises from 13% to 18% after five cycles. The reactivity penalty from ^{234}U and ^{236}U is somewhat compensated by ^{235}U production.
- The extent of ^{233}U conversion is very sensitive to core configuration. Modifying the size and number of blanket zones can improve breeding, but trades-off with an inherent negative impact on k_{eff} and achievable burn-up.
- Maintaining a negative void coefficient in this system seems achievable but power peaking needs careful attention, especially when high enrichment is used in the seed fuel.
- The largest single positive impact on conversion ratio comes from increasing the heterogeneity of the core and matching the size of seed-blanket zones with void-dependent mean neutron path-lengths.
- The thorium absorption rate depends on neutron energy and is higher in blanket material in lower void regions, and in all cases it peaks several centimeters from a seed region.

Conversion (expressed as fissile inventory ratio) and k_{eff} are shown in Figure Two as a function of burnup for a number of different core configurations. These results show that it is possible to find a configuration that is an acceptable comprimise between conversion and cycle length. This study was undertaken in the infinite radial

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[3] The "Resource-renewable BWR" concept has been developed in Japan by Hitachi Ltd and JAEA [6] and is also referred to as the Reduced Moderation Water Reactor (RMWR), or the Flexible Fuel Cycle LWR (FLWR).

approximation and more detailed studies are warranted into, *eg*; thermal hydraulics, power density variations, seed/blanket compositions, core simulations and uranium vector refinements.

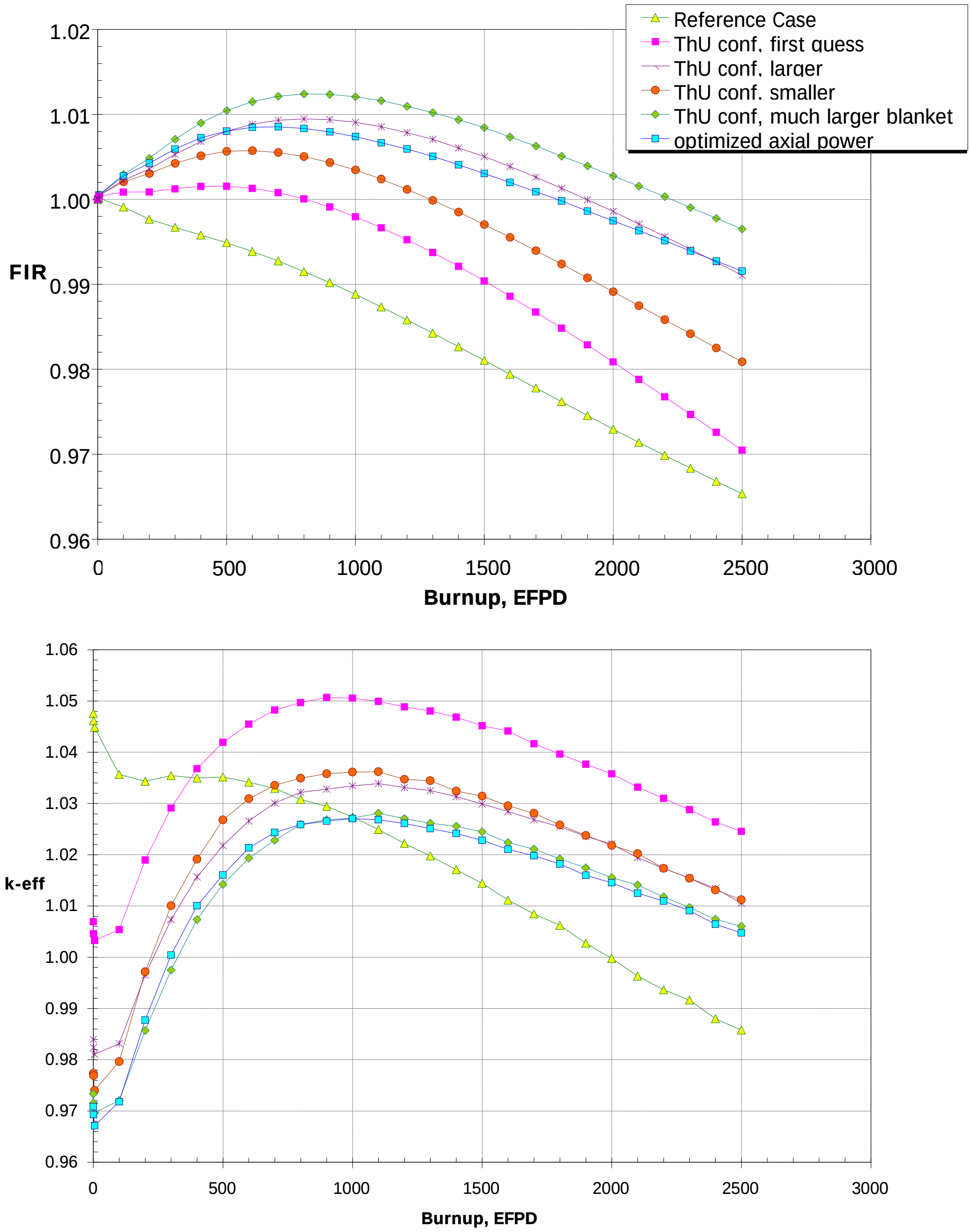

Figure Two: Fissile Inventory Ratio (FIR) and Reactivity as a function of burnup (effective full power days).

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