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# A BWR fuel assembly design for efficient use of plutonium in thorium–plutonium fuel

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ABSTRACT

The objective of this study is to develop an optimized BWR fuel assembly design for thorium–plutonium fuel. In this work, the optimization goal is to maximize the amount of energy that can be extracted from a certain amount of plutonium, while maintaining acceptable values of the neutronic safety parameters such as reactivity coefficients, shutdown margins and power distribution. The factors having the most significant influence on the neutronic properties are the hydrogen-to-heavy-metal ratio, the distribution of the moderator within the fuel assembly, the initial plutonium fraction in the fuel and the radial distribution of the plutonium in the fuel assembly. The study begins with an investigation of how these factors affect the plutonium requirements and the safety parameters. The gathered knowledge is then used to develop and evaluate a fuel assembly design. The main characteristics of this fuel design are improved Pu efficiency, very high fractional Pu burning and neutronic safety parameters compliant with current demands on UOX fuel.



## 1. Introduction

The objective of the work described herein is to develop a Boiling Water Reactor (BWR) fuel assembly design adapted to thorium–plutonium (Th/Pu) fuel, as a part of the fuel development program at the Norwegian company Thor Energy. The broad goal of our company's research is to develop a safe, reliable and cost effective Light Water Reactor (LWR) fuel which effectively consumes the world's growing stockpiles of plutonium, either it is present in nuclear waste or as surplus weapons material. The fuel is intended for use in currently operating and future LWRs and will be similar to currently used uranium oxide (UOX) and mixed oxide (MOX) fuel, except that the fuel pellets will consist of a homogenous mixture of thorium and plutonium oxides. The specific goal of this work is to develop a fuel assembly design which maximizes the amount of energy that can be extracted from a certain amount of plutonium, a property hereafter referred to as Pu efficiency. This goal is based on the perception that an entity in possession of a certain amount of plutonium would like to maximize its value, which effectively means extracting as much energy from it as possible.

The general viability of Th/Pu fuel in LWRs (mainly PWRs) has been confirmed by several recent studies (Trellue et al., 2011; Fridman and Kliem, 2011; Tsige-Tamirat, 2011; Insulander Björk et al., 2011; Todosow and Raitses, 2010). In these studies, conventional fuel assembly designs are used, since the main goal is to investigate the inherent differences between Th/Pu fuel and other fuel types. However, it is known that when Pu is the main fissile element, the optimal (for reactivity) hydrogen-to-heavy-metal ratio (H/HM ratio) is much larger than it is for UOX fuel (Puill, 2002; Kloosterman and Bende, 2000). This fact applies to Th/Pu fuel as well as to MOX fuel, and has been used to some extent for creating more optimized MOX fuel assembly designs (Ramirez-Sanchez et al., 2008; Bairiot et al., 2003; Hamamoto et al., 2001). New fuel designs for Th/Pu fuel have also been presented, aiming to improve conversion of $^{232}Th$ to $^{233}U$ (Galperin et al., 2000). There are however, to our knowledge, no previous studies of fuel assembly designs for Th/Pu fuel aiming to improve Pu efficiency.

The current study concerns BWR fuel assembly design. This choice was made since the location of the cruciform control rods outside the BWR fuel assembly and the channel allows for comparatively large freedom in designing the layout of the fuel within the channel. It is our intention to also create an optimized PWR fuel assembly design later.

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Varying the H/HM ratio affects the reactivity of the fuel assembly, but also several important safety parameters such as the control rod worth, shutdown margins and reactivity coefficients. These effects are investigated and, where necessary and possible, mitigated. An effort is made to keep the power distribution in the assembly even, in order to avoid excessive individual rod Linear Heat Generation Rate (LHGR). Ultimately, the LHGR which will be achievable with Th/Pu fuel will depend on the thermal, mechanical and chemical properties of the fuel material. Investigations of these properties are underway, but are not part of this study.

The implications of the changed design for the thermal hydraulic parameters, such as pressure drop and margin to dryout, will need to be evaluated too. This is also left for later study.

The constraints for the fuel design and the reactor system that is simulated are described in Section 2. The calculation tools and methods used in this study are described in Section 3. Within this framework, a number of different studies were carried out, investigating how different parameters affect the properties of the fuel assembly. Using the results of these, a fuel assembly was designed. This process is described in Section 4. The properties of the new fuel assembly design were calculated and compared to those of reference fuel assemblies. This phase of the work is reported in Section 5. In Section 6, finally, the conclusions are presented along with an outlook where future investigations and directions for the assembly design work are suggested.

## 2. Premises

### 2.1. Design constraints

In order to design a fuel assembly that would be commercially feasible in the sense that it is fabricable with current manufacturing facilities and licensable in current BWRs, certain constraints were put on the fuel assembly design. These constraints are summarized in the list below.

- The neutronic safety parameters should be within currently practiced limits.
- The size and shape of the fuel assembly should allow it to be loaded into a normal BWR.
- Normal cylindrical oxide fuel pellets should be used.
- Well-known materials should be used for the cladding, external channel and internal water channel(s).
- Geometrical parameters such as the fuel rod thickness, rod pitch, rod-channel distance etc. should not be outside current experience.

### 2.2. Reactor operation parameters

As a framework for the calculations, the Swedish BWR Forsmark 3[1] was considered. The standard operation parameters of this reactor are listed in Table 1. Fuel assembly mass and burnup depend on the fuel design and are listed separately in Table 3 in the Results section.

The modern and commonly used fuel assembly design GE-14N was used as a reference. For comparing the properties of the improved design with those of normally used fuel, reference designs were created with both UOX and MOX fuel, as well as with Th/Pu fuel. The fissile content of the reference fuel designs, as well as the new fuel assembly design, was adapted to release the same amount of energy during the lifetime of the fuel assemblies. This gives a fair comparison between the fuel types since the released energy is directly related to the revenue of the reactor operation. Due to different fuel masses in the different designs, equal energy release implies different burnup, which will be disussed in Section 3.4. The higher discharge burnup implied by the lower fuel mass in the Th/Pu cases is discussed and justified in Section 5.4.

**Table 1**
Reactor parameters. The numerical value of the neutron leakage is based on experience of normal reload design for Forsmark 3.

| Parameter | Notation | Value |
|---|---|---|
| Number of fuel assemblies | $N$ | 700 |
| Number of fuel batches | $n$ | 6 |
| Cycle length [days] | $t_C$ | 350 |
| Reactor thermal power [MW] | $P_{th}$ | 3300 |
| Neutron leakage [pcm] | $\Delta\rho$ | 2000 |
| At hot full power (HFP) | | |
| Power density [kW/dm$^3$] | | 53.5 |
| Coolant temperature [K] | | 559 |
| System pressure [MPa] | | 7.02 |
| Average void fraction [%] | $V$ | 40 |
| At cold zero power (CZP) | | |
| Power density [kW/dm$^3$] | | 0 |
| Coolant temperature [K] | | 293 |
| System pressure [MPa] | | 0.1 |
| Void fraction [%] | $V$ | 0 |

The UOX design had an average enrichment of 4.1 wt% $^{235}$U, yielding the required amount of energy as described in Section 3.4. The Pu content of the reference MOX and Th/Pu reference designs was adjusted to reach the same energy release, demanding 7.0% Pu in the MOX case and 9.2% Pu in the Th/Pu case. The Pu isotope vector used both in the reference fuel and in the created new designs was: 2% $^{238}$Pu, 53% $^{239}$Pu, 25% $^{240}$Pu, 15% $^{241}$Pu and 5% $^{242}$Pu. This corresponds to the Pu vector in spent LWR fuel burnt to approximately 42 MWd/kgHM, if reprocessed and used immediately after discharge (World Nuclear Association, 2009). In all three reference cases, a number of different enrichment levels (8–9) was used in order to create an even power distribution. Gadolinium was used as a Burnable Absorber (BA) in all cases, in order to suppress the reactivity of the fuel at the Beginning Of Life (BOL). Due to the small slope of the reactivity curve in the Th/Pu and MOX reference cases, only very small amounts of BA were required.

## 3. Calculation tools and methods

### 3.1. Neutronic simulation software

All calculations were carried out on a 2D infinite lattice level. For these calculations, the fuel assembly burnup simulation program CASMO-5 (Rhodes et al., 2007) was used, together with the cross section library ENDF/B-VII.0 (Chadwick et al., 2006). It should be noted that some changes were made to the $^{232}$Th and $^{233}$U cross sections in ENDF/B-VII short before its release (Mosteller, 2004 ; Mosteller, 2008). These data sets have not been subject to the same level of testing as the other data sets in the library. Benchmarks for $^{233}$U cross sections in the thermal spectrum show some improvement over the last ENDF/B-VI version. For $^{232}$Th cross sections, there are not many well-vetted benchmarks, so we must settle with regarding the results with some scepticism and await further testing and possibly improvements of the data sets.

These lattice calculations do not take leakage or spectral interactions between assemblies of differing type or age into account, except for that the neutron energy spectrum is corrected for the energy dependence of the leakage probability. The numerical values of the calculated parameters are thus only correct for the

[1] The ongoing power uprate at Forsmark 3 is not accounted for.