CTH-NT-285

Thesis for the degree of Doctor of Philosophy

Improved PWR Core Characteristics with Thorium-containing Fuel

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ISBN 978-91-7385-990-5

Doktorsavhandling vid Chalmers tekniska högskola
Ny serie nr 3671
ISSN 0346-718X

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Chalmers Reproservice
Göteborg, Sweden 2014

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ABSTRACT

Thorium is an element for possible use as nuclear fuel, because of the fertile property of Th-232, which could convert into U-233 after neutron capture and beta decays. However, thorium based fuels have not been successfully implemented, because of the high cost and the higher complexity compared with once-through uranium fuel. Therefore, the focus of this thesis is on thorium for use in traditional light water reactors for improving core characteristics by: achieving a more homogenous power distribution to provide larger thermal margins, high plutonium incineration efficiency to reduce the plutonium stockpile, and improved load-following capacity.

The chosen methodology is similar to the one commonly used in the industry. The model is based on Ringhals-3 PWR, and Studsvik Scandpower Ltd. codes are used to calculate the neutronics of the nuclear fuel and the reactor cores.

The thesis is divided into three parts. The first part addresses the fertile absorber concept, which is achieved by adding small amounts of thorium in traditional uranium fuel pellets to improve the thermal margins. The benefits are a reduced need of traditional burnable absorbers, and a slower decrease of reactivity as function of burnup.

The second part describes the use of thorium-plutonium mixed oxide fuel to reduce the plutonium stockpile, since thorium barely generates any new plutonium compared with uranium-plutonium mixed oxide fuel. However, the thorium-plutonium mixed oxide cores alter the core behavior compared with the traditional uranium core such as stronger Doppler feedback and lower fraction of delayed neutrons, which require boron (B-10) enrichment, and increased number of control rods with improved efficiency.

The third part of the thesis is focusing on improved load-following capacity by thorium-containing fuel. Th-232 has about three times higher thermal neutron absorption cross section compared with U-238, which hardens the neutron spectrum. This leads to less sensitivity to neutron spectrum changes, lower xenon poisoning, and lower axial power fluctuations upon load changes. Therefore, thorium-containing fuel improve the load-following capacity, and could be a positive complement to intermittent power sources, such as solar or wind power.

Keywords: Thorium, core calculations, PWR, load-following, xenon poisoning.

Appended papers

This thesis consists of an introduction to and a summary of the work published in the following papers:

Paper I

C.W. Lau, C. Demazierè, H. Nylén and U. Sandberg, "Improvement of LWR thermal margins by introducing thorium"
Progress in Nuclear Energy, **61**, 48–56 (2012).

Paper II

C.W. Lau, H. Nylén, C. Demazierè and U. Sandberg, "Investigation of the Equilibrium Core Characteristics for the Ringhals-3 PWR with Improved Thermal Margins Using Uranium-thorium Fuel"
2013 International Congress on Advances in Nuclear Power Plants (ICAPP2013), Jeju Island, South Korea, April 14-18 2013.

Paper III

K. Insulander Björk, C.W. Lau, H. Nylén and U. Sandberg, "Study of Thorium-Plutonium Fuel for Possible Operating Cycle Extension in PWRs"
Science and Technology of Nuclear Installations, Article ID 867561 (2013).

Paper IV

C.W. Lau, H. Nylén, K. Insulander Björk and U. Sandberg,"Feasibility Study of 1/3 Thorium-Plutonium Mixed Oxide Core"
Science and Technology of Nuclear Installations, Article ID 709415 (2014).

Paper V

C.W. Lau, H. Nylén, C. Demazierè and U. Sandberg, "Reducing Axial Offset and Improving Stability in PWRs by Using Uranium-Thorium Fuel"
Submitted to Progress in Nuclear Energy, (2013).

Paper VI

C.W. Lau, V. Dykin, H. Nylén, K. Insulander Björk and U. Sandberg, "Conceptual Study of Axial Offset Fluctuations upon Stepwise Power Changes in a Thorium-Plutonium Core to Improve Load-Following Conditions"
Submitted to Annals of Nuclear Energy, (2014).

Author's contribution

The present author calculated and analyzed all results reported in the above papers. In addition, the author also wrote the manuscripts of all papers, except paper III.

Related works not included in this thesis

C.W. Lau, C. Demazierè, H. Nylén and U. Sandberg, "Innovative use of Thorium in LWR fuel assemblies"
2011 International Congress on Advances in Nuclear Power Plants (ICAPP2011), , Nice, France, May 2-5 2011.

C.W. Lau, C. Demazierè, H. Nylén and U. Sandberg, "Improvement of the Thermal Margins in the Swedish Ringhals-3 PWR by Introducing New Fuel Assemblies with Thorium"
PHYSOR 2012, Knoxville, United States of America, April 15-20 2012.

CONTENTS

CHAPTER

1

INTRODUCTION

The introduction explains the motivation of the work and the thesis. Thereafter, the contents of the following chapters are outlined.

1.1 Objective

The main driving force of the thorium research started as the uranium price increased rapidly in the middle of 2000s. The uranium spot price increased more than 10 times in just four years (2003-2007). During this period, finding substitutes to uranium fuel were of high priority, and thorium based fuel was pointed out as one of the solutions, because thorium generates the fissile isotope U-233 from neutron capture [1]. Moreover, thorium resources are about four times more abundant and more easily accessible compared with uranium. However, the knowledge about thorium were far lower compared with uranium, including cross section data for Th-232, U-233, as well as experimental data from mechanical, thermal or fission gas release in irradiated samples. Therefore, a strong wave of research started in the middle of 2000s. However, as the uranium price decreased to more acceptable levels in 2008, new discoveries of uranium deposits and the limitations of thorium fuel started to surface from the research projects. The interest of thorium started to decrease, and fewer research projects were continued. The only exception is the thorium research in India. This is discussed in more detail in the next chapter.

One of limitations of thorium fuel is the expensive reprocessing process of spent nuclear fuel. There are limited places that have reprocessing facilities, because the low interest of reprocessing spent nuclear fuel. Also, the thorium based fuel reprocessing process is estimated to be more expensive compared with traditional uranium based fuels, because thorium dioxide is more chemically stable compared

with uranium dioxide [2]. Moreover, thorium has about three times higher thermal absorption cross section compared with uranium, and requires more fissile isotopes to remain critical, but the advantage of thorium is the slower decrease of fissile materials in thermal reactors using thorium based fuel. Therefore, it is possible to achieve the same discharge burnup for uranium and thorium fuel with the same amount of fissile materials, but it requires considerable higher discharge burnup and exceeds the limitations on current cladding materials. Implementing thorium is difficult compared with the existing uranium based fuel, because most of the nuclear reactors are based on uranium fuel. Therefore, the main focus of this research is to find thorium applications in nuclear reactors without considering thorium as fuel to generate power, but instead as additive or carrier to improve specific tasks in LWRs, such as improved thermal margins, plutonium incineration and load-following purposes.

This research project was funded by Swedish Centre for Nuclear Technology, Oskarshamnsverkets Kraftgrupp AB, Vattenfall AB, the Swedish Radiation Safety Authority, Ringhals AB, Forsmarks Kraftgrupp AB and Westinghouse Electric Sweden AB. The funders main interests are in current LWRs. Therefore, this project has been focused on LWRs. Larger part of this project has been in collaboration with Ringhals to get a deeper understanding on LWRs potential improvements by a realistic methodology used in the industry.

1.2 Outline of the thesis

This research project focused on three different thorium fuel applications: the concept of fertile absorber, plutonium incineration and load-following cores. The fertile absorber and load-following core concepts originated from this research project, while the plutonium incineration is a continuation from other research projects.

This thesis is divided into seven chapters including this one. The second chapter explains some of the background on thorium, such as the history, different types of thorium based nuclear reactors, thorium dioxide properties, difficulties in reprocessing and fuel fabrication, and the thorium strategy in India. The third chapter is about the methodology used in this research project. The computer codes and the Ringhals-3 PWR model are briefly described. Chapter four is about the fertile absorber based on PAPERS I-II, and describes the basic ideas, fuel designs, applications, fuel and core characteristics, and other interesting remarks. Chapter five (PAPERS III-IV) describes the core characteristics of thorium-plutonium mixed oxide (Th-MOX) fuel, and its good plutonium incineration capability. Two proposed cores, a full Th-MOX core and a 1/3 Th-MOX core are presented for comparison with a traditional UOX core. The differences between the cores are described to give a good understanding of the Th-MOX cores benefits and drawbacks. Moreover, Th-MOX cores require different reactor conditions to work properly. Chapter

six (PAPERS V-VI) gives the idea of using thorium-containing fuel to improve the load-following core capacity by achieving a more balanced axial offset (AO) from flatter power distributions and smaller impact from xenon oscillations. The chapter mostly focuses on comparing the full Th-MOX core and the UOX core, followed by explanations of the core behaviors. The seventh chapter briefly summarizes the most important information in chapter 4-6. It should be emphasized that all the results, statements, figure, tables without reference to the literature in chapter 4-6 are a result of the present research.

The aim of this thesis is to show additional thorium-containing nuclear fuel applications beside using it as fuel to generate power, and explain its potentials and limitations. These applications could possible contribute to solutions to a more demanding power supply for sustainable global welfare.

CHAPTER

2

BACKGROUND - THORIUM

Thorium is an element discovered by the Swedish chemist Jöns Jakob Berzelius in 1828, who named it after the god Thor in the Norse mythology. Natural thorium has mainly one isotope, Th-232, and a half-life of 14 billion years. Thorium applications have been very few, but nuclear fuel is one.

2.1 History

The interest of using thorium as nuclear fuel began in the 1950s. United States, USSR and Europe started investigating the possibility to harness power from thorium. In the early 1960s, Elk River (BWR) and Indian Point (PWR) [3, 4] were the first two reactors that were loaded with thorium-uranium fuel. The reactors were using high enriched uranium in combination with thorium dioxide. However, it was not until Shippingport in the late 1970s that breeding could be demonstrated, i.e. more fissile materials were produced compared with the consumed fissile materials by using thorium-uranium oxide pellets and thorium blankets. The reactor was the first light water breeding reactor [3, 5], but was only achieved at a high cost, with sophisticated core designs, and reduced reactor performance [2]. Thorium has also been tested in high temperature gas-cooled reactors with prismatic and pebble bed fuels such as Peach Bottom 1, Fort St Vrain, AVR and THTR, and have mostly been built and tested in USA, Germany and United Kingdom. Those reactors were using thorium in an open cycle, which means the reactors never used U-233 from reprocessed fuel [1]. Therefore, a complete closed fuel cycle with thorium was never demonstrated.

2.2 Thorium in nuclear reactors

The majority of the thorium research has been focused on thermal reactors, because it is the most common reactor type, and thorium fuel could achieve breeding, which is not possible with uranium fuel in thermal reactors. The thorium based fuels could be in homogenous and heterogeneous structures. Heterogeneous fuel has areas with only thorium, while homogenous fuel has always fissile materials with thorium. Heterogeneous fuel offers better neutronic characteristics compared with homogenous fuel [6], where fissile seeds are surrounded by thorium blankets. The seeds could be enriched uranium, plutonium or recycled U-233. Even though heterogeneous fuel often gives higher discharge burnup because of the better neutronics, the more heterogeneous power distribution and uneven burnup in the fuel assemblies lead to drawbacks on the thermal-hydraulic side. On the contrary, homogenous fuel offers better thermal-hydraulic properties, but need to reach discharge burnup of 120 GWd/tHM to be comparable to homogenous uranium fuel [7], which is about twice the presently allowed maximum discharge burnup. A well known heterogeneous fuel design is the Radkowsky thorium fuel concept. The seed (fissile material) is loaded in the center, and is surrounded by thorium in a fuel rod developed by Alvin Radkowsky [8].

The majority of the thorium research focus on achieving the highest breeding in LWRs. Unfortunately, this will nevertheless lead to a lower power because the thorium blankets generate considerably lower power or no power at the beginning. Therefore, light water reactors that reach breeding and have comparable burnups require larger cores because of the lower average power density compared with traditional UOX cores. Consequently, this kind of concepts are economically uncompetitive [9]. However, it has to be emphasized that thermal reactors do not have the possibility to reach breeding in the uranium-plutonium fuel cycle.

Heavy water reactors have better neutron economy and more efficient thermal neutron spectra compared with LWRs, and represent a great benefit for the thorium fuel cycle. There are two types of heavy water reactors, Advanced Heavy Water Reactor (AHWR) and Canada Deuterium Uranium (CANDU). The difference between those two reactors types is that AHWR has vertical pressure tubes instead of horizontal as in CANDU. Studies have shown that thorium-based fuel in CANDU reactors could increase the discharge burnup to 30 - 40 GWd/tHM from current 20 GWd/tHM [1], because of the higher conversion ratio. Moreover, thorium could reduce the positive void coefficient in CANDUs and improve safety performance [1].

The fissile isotope U-233 produced in a thermal reactor has the best thermal fission factor η compared with other fissile isotopes, because U-233 has the highest ratio between the number of absorbed neutrons that will undergo fission and the total number of absorbed neutrons in the thermal region. In fast reactor systems, the U-233 in the thorium cycle has a lower η compared with Pu-239 in the

uranium-plutonium cycle. Therefore, thorium fuel in fast breeder reactors are not as competitive as uranium-plutonium fuel. However, thorium could be utilized to reduce the void coefficient, which could be positive in fast reactors such as large sodium fast reactors, and by adding thorium the void coefficient could be improved.

2.3 Thorium dioxide properties

Thorium oxide has better properties compared with uranium oxide, such as better thermal conductivity, higher melting point and is chemically a more stable material. The melting points in unirradiated thorium oxide, uranium oxide and plutonium oxide are about 3651 K [10], 3120 K [11] and 2701 K [11], respectively. However, small amounts of thorium have shown to inherit slightly lower melting points from two different measurements of Christensen [12] and Latta et al. [13] at 2 and 5 mol% ThO_2, respectively. The higher melting point in thorium oxide allows thorium fuel to operate at higher fuel temperatures and to higher discharge burnup.

The density of the fuel pellets is an important property. The theoretical densities of thorium oxide and uranium oxide are 10.00 g/cm^3 and 10.96 g/cm^3 at 298 K, respectively. However, at fuel operational temperature of about 873 K, the theoretical density will decrease both in thorium oxide and uranium oxide to about 9.83 g/cm^3 and 10.76 g/cm^3 [3], respectively. The theoretical density shows a linear molar fraction behavior in a mixture between uranium oxide and thorium oxide, and could be calculated by the molar percentage of the uranium oxide and thorium oxide at the desired temperature [14].

The thermal expansion is similar in thorium oxide and uranium oxide at lower temperatures (below 1000 K), but at higher temperatures (above 1000 K), the thermal expansion in thorium oxide is lower compared with uranium oxide [3]. In the mixtures between uranium oxide and thorium oxide, a higher concentration of uranium oxide has shown to increase the thermal expansion [15]. Plutonium oxide has higher thermal expansion compared with thorium oxide. Therefore, by having larger share of thorium oxide in Th-MOX fuel, the thermal expansion will also become lower [16].

Thermal conductivity is a very important parameter for determining the fuel temperature profile, since higher thermal conductivity results in a lower average fuel temperature. Among three fuel materials: thorium oxide, plutonium oxide and uranium oxide, the thorium and plutonium oxides have the highest and lowest thermal conductivity, respectively. Even though thorium oxide has a better thermal conductivity compared with uranium oxide, a mixture of uranium oxide and thorium oxide could result in a lower thermal conductivity compared with uranium oxide [17]. Plutonium oxide has low thermal conductivity, and therefore, Th-MOX fuel could has higher thermal conductivity compared with uranium-plutonium mixed oxide (U-

MOX), and it is possible to have a higher share of plutonium oxide in the Th-MOX fuel compared with U-MOX.

For irradiated fuel, fission products change the fuel properties, such as lowering the melting point and thermal conductivity. The higher thermal conductivity in most of the thorium based fuel causes a lower fission gas release, since it mostly depends on fuel temperature and burnup [16, 18]. Current MOX fuel discharge burnup is often limited by the fission gas release from the pellet matrix, and thorium-plutonium has a low fission product migration, which is favorable for higher burnup [19]. Additionally, fission gas release in homogenous uranium-thorium fuel is lower compared with the uranium fuel [20].

2.4 Difficulties in reprocessing and fuel fabrication

The closed thorium fuel cycle requires THOREX reprocessing of the thorium-based spent nuclear fuel to separate the fission products, fissile and fertile isotopes [2]. The fissile and fertile isotopes are later reused in new fuel. However, the chemically more stable thorium oxide is more difficult to dissolve during the reprocessing process compared with UREX and PUREX reprocessing processes for uranium and uranium-plutonium fuel. Therefore, THOREX requires additional catalysts to improve the dissolution process and increases the corrosion in reprocessing plants. Consequently, more waste is needed to be stored. In addition, the chemically more stable thorium oxide is a great advantage for core and waste storage since it is more difficult to dissolve and oxidize [20].

The essence of the thorium fuel cycle is the fissile isotope U-233. However, fast neutrons can have a (n,2n) reaction with Th-232, Pa-233 and U-233, and accumulate U-232. U-232 has the daughter nuclide Thallium-208 that emits a 2.6 MeV gamma photon [2]. The 2.6 MeV gamma source requires extra shielding both in the reprocessing plants and fuel fabrication factories [21, 22]. Fabricating U-233 based fuels require more gamma shielding compared with MOX fuel, and will complicate the fuel fabrication process. However, the high gamma source is an advantage from a proliferation resistance viewpoint since it will be more difficult to divert as nuclear weapon material.

2.5 Strategy in India

India is the only country in the world that has established a fully committed thorium program. The main reasons are the small limited amount of domestic uranium resources and large easily accessible thorium resources. Before 2006, India was not allowed to acquire uranium on the international market, because India has not signed

the nuclear non-proliferation treaty. Consequently, India was forced to develop a thorium strategy in order to have a nuclear power program. However, in the last few years, some of India´s reactors have been subjected to IAEAs safeguards in order to be able to supply the India reactors with civil nuclear technologies and materials from other countries. India is mostly having pressurized heavy water reactors (PHWR) in its nuclear reactor fleet, but the current PHWR are mostly low power and small size reactors. The power for each reactor is around 220 MWe. However, larger PHWRs (about 700 MWe) are under construction.

Dr Homi Bhabha was the founding father of the Indian nuclear programme for developing the thorium cycle for future independence [23]. The first stage of a thorium cycle is to extract plutonium from natural uranium in existing PHWRs or enriched uranium in LWRs. The second stage is to use plutonium as seed and thorium/uranium as blankets in fast reactors. The blankets produce new plutonium and U-233, and the plutonium is reused in fast reactors. In the last stage, PHWRs use the U233-thorium fuel to obtain a closed thorium fuel cycle by generating enough U-233 from PHWRs. India just started the second stage, and needs at least another 10 - 20 years before thorium has an important role in the Indian thorium nuclear programme.

India is using thorium bundles for power flattening in PHWRs during start up of new reactors [24, 25], because online reloading without thorium or depleted uranium induces a power peak at the center of the core. Depleted uranium has been used to decrease the power at the center of the core, but is replaced by thorium in order to start up the thorium cycle studies. The first critical PHWR core with thorium bundles was initiated in 1992, and it is now a common procedure in the PHWR for power flattening.

CHAPTER

3

METHODOLOGY

The methodology used for the calculations in this thesis is presented in this chapter. The model is based on Ringhals-3 PWR and the main tools used are CASMO-4E and SIMULATE-3, widely used in the nuclear industry.

3.1 Models

The calculations in this thesis is based on Ringhals-3 PWR. Ringhals is located in the west cost of Sweden and is the biggest nuclear power plant in Scandinavia. Ringhals produces 28 TWh per year and has four nuclear reactors, three Westinghouse PWRs and one ASEA Atom BWR. Ringhals-3 generates about 1070 MWe in a three loop system and has a thermal output of 3135 MWth after the modernization and power uprate projects [26]. The core has 157 fuel assemblies and the height of the active core is 365.76 cm. The core uses silver-indium-cadmium control rods and natural boron in the moderator to control the reactivity. Ringhals-3 is typically using a 3-batch reload scheme with 12 months cycle. The fuel assemblies are 17x17 with 364 fuel rods and 25 guide tubes for control rods and instrument thimbles. The fuel assemblies are about 22 cm wide and each fuel rod is about 9 mm in diameter. The fuel assemblies have octagonal symmetry, and the cladding of the fuel rods is made of Zircaloy. The fuel assemblies are typically using UOX with a weight of 461 kg in heavy metal.

3.2 Tools

The tools for the calculations are from Studsvik Scandpower such as CASMO-4E, SIMULATE-3, CMS-Link, INTERPIN-3 and XIMAGE. All of the codes are necessary for calculating accurate neutronic properties of the core.

3.2.1 CASMO-4E

CASMO-4E is a multigroup two-dimensional neutron transport (infinite) lattice code used to determine the neutron transport solution in a two dimensional plane of a fuel assembly [27]. The fuel assemblies have an octagonal symmetry, and CASMO-4E uses reflective boundary conditions to reduce the calculational time. The calculations use the JEF 2.2 library, because it is normally used for Ringhals-3 PWR. The main purpose of CASMO-4E is to generate macroscopic cross section data with a variety of parameters or conditions such as: boron concentration, fuel temperature, moderator temperature, with or without control rods, and burnup. The macroscopic cross section data is then fed into XIMAGE and SIMULATE-3. Moreover, the infinite multiplication factor (k_∞) and other results from the fuel assembly level are calculated by CASMO-4E with the base operational conditions: fuel temperature is set to 928 K, moderator temperature to 575 K, boron concentration to 600 ppm and power density to 105.50 kW/liter.

3.2.2 SIMULATE-3

SIMULATE-3 is a three dimensional nodal two group diffusion code for calculating all the necessary data to demonstrate the safety margin during core operation [28]. Calculations determine the boron concentration, power distribution, xenon concentration, feedback coefficients, fraction of delayed neutrons, cycle length etc. A transient code SIMULATE-3K is used to calculate the fast transient events such as rod ejection.

3.2.3 Other codes

CMS-Link is a code used to extract data from CASMO-4E to generate the necessary data in different operational conditions and insert these into SIMULATE-3 and XIMAGE in the correct format.

XIMAGE is a graphical fuel management code for finding optimal loading patterns for predetermined requirements such as cycle length, homogeneous power distribution and feedback coefficients. It will quickly evaluate the corresponding patterns, but the results are not accurate enough to be used for safety evaluations.

Therefore, it requires full SIMULATE-3 calculations for better accuracy.

INTERPIN-3 is a code used to calculate the fuel temperature based on geometry of the fuel rods, size of the fuel pellets, fuel pellets density, burnup, etc.

3.3 Reload safety evaluation

In paper II - IV, the safety assessments are based on the reload safety evaluation (RSE) and the limits contained therein determining core safety during normal operation and transients. The Swedish Radiation Safety Authority requires that every core reload is demonstrated to be within the limits of RSE. The RSE used in Ringhals-3 is based on the Westinghouse reload methodology and is licensed by the US NRC in the topical report WCAP-92724 [29].

Some of the results presented in this thesis are the reactivity coefficients, α, such as: the moderator temperature coefficient of reactivity (MTC), Doppler temperature coefficient of reactivity (DTC), isothermal temperature coefficient of reactivity (ITC) and boron worth. In Eq. (3.1), P is the perturbed core parameter such as core average moderator temperature, core average fuel temperature and boron concentration. ρ and k_{eff} are the reactivity and effective multiplication factor, respectively.

$$\alpha = \frac{\Delta\rho}{\Delta P} \tag{3.1}$$

$$\rho = \frac{k_{eff} - 1}{k_{eff}} \tag{3.2}$$

Eq. (3.3) is obtained by combining Eq. (3.1) and (3.2). The calculation of α requires the information about k_{eff}, which is estimated in SIMULATE-3 for different P.

$$\alpha = \frac{\frac{k_{eff}(P_1) - k_{eff}(P_2)}{k_{eff}(P_1) \times k_{eff}(P_2)}}{P_1 - P_2} \tag{3.3}$$

CHAPTER

4

FERTILE ABSORBER

The new concept about fertile absorber could offer a more homogenous power distribution, safer reactor fuel, and is relatively easy to implement. An overview about the concept, designs, usage areas, applications and benefits is given.

4.1 The concept of fertile absorber

The purpose of burnable absorber materials is to suppress local reactivity and improve core power distribution. Traditional burnable absorbers such as gadolinium and boron, have high neutron absorption cross section and are depleted before end of cycle (EOC) to maximize the cycle length and fuel assembly discharge burnup.

Fertile materials cannot easily undergo fission, but have the possibility to become fissile by neutron capture or neutron capture followed by radioactive decays. The most abundant natural occurring fertile isotopes are U-238 and Th-232. Both of the fertile isotopes U-238 and Th-232 require neutron capture and two beta decays to become fissile isotopes Pu-239 and U-233, respectively.

The idea behind fertile absorber is to select a fertile material to be used as burnable absorber. Among the two natural occurring fertile isotopes, Th-232 has about three times higher neutron capture cross section in the thermal region compared with U-238. Moreover, U-233 has a higher thermal fission factor compared with Pu-239 and is a better fissile isotope in terms of higher reactivity. The reason is the higher ratio between the thermal fission cross section and the thermal capture cross section in U-233.

The majority of nuclear fuel fabrication factories are licensed to handle low enriched uranium, with a maximum 5 % enrichment of U-235. Therefore, the enrich-

ment of the U-235 should not exceed 5 %. Moreover, a higher amount of thorium in the fuel will require a higher quantity of U-235 to sustain the same discharge burnup about 55 GWd/tHM, as shown in Fig. 4.1. Consequently, the amount of thorium should be kept to a minimum to achieve the highest discharge burnup. The rule "Less is More" should apply.

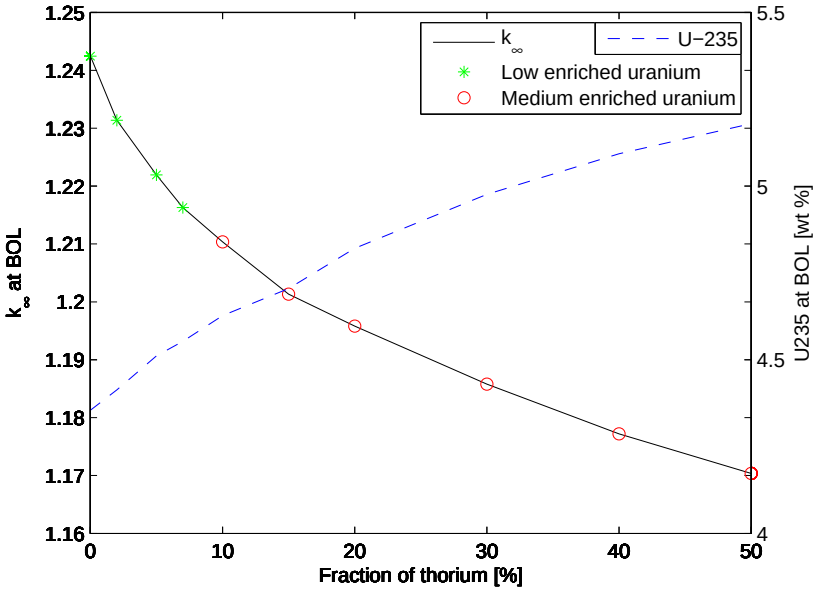

Figure 4.1: The infinite multiplication factor and U-235 weight percentage (wt%) at beginning of life (BOL) as function of the fraction of thorium.

4.2 Fuel assembly designs

Two different types of fuel assemblies are presented in order to compare fertile absorber with traditional burnable absorber in a simulated core. The first is the uranium-gadolinium (U-G) fuel assembly used as reference. The second is the uranium-thorium (U-Th) fuel assembly using fertile absorber.

The U-G fuel assembly has two different kinds of fuel pellets, 4.4 weight percentage (wt%) of U-235, referred to as FUE 1 and 2.6 wt% (2.8 % of enrichment) of U-235 with 6 wt% of gadolinium, referred to as FUE 2. The U-Th fuel assembly has two kinds of fuel pellets, FUE 2 and 4.6 wt% (4.95 % of enrichment) of U-235 with 7 wt% of thorium, referred to as FUE 3, as shown in Tab. 4.1. No fuel pellet contains more than two elements, in order to avoid complicated estimations of the thermal properties.

The densities of FUE 1, FUE 2, FUE 3 at operational conditions are 10.165 g/cm^3, 9.940 g/cm^3 and 10.126 g/cm^3, respectively. The estimated density in the

mixture FUE 3 is based on assuming that the thorium density is 9.79 g/cm^3 [17,30]. This is discussed in more detail in paper I.

	FUE 1	FUE 2	FUE 3
Nuclide	Weight percentage		
Th-232	0	0	7.00
U-234	0.04	0.02	0.04
U-235	4.40	2.63	4.60
U-238	95.56	91.43	88.36
Gd	0	5.91	0
Density [g/cm^3]	10.165	9.940	10.126

Table 4.1: Isotopic compositions and density in FUE 1, FUE 2 and FUE 3.

There are four different kinds of fuel rods in the U-G and U-Th fuels. The active length of the fuel rods is 365.76 cm. The U-G fuel assembly has uranium (U) and U-G fuel rods. The U fuel rod has only FUE 1 fuel pellets. The U-G fuel rods have FUE 2 fuel pellets, except in the 30 cm of the top and bottom of the fuel rod that have FUE 1 fuel pellets, to increase the power at the top and bottom of the fuel rod. The U-Th fuel assembly has U-Th and uranium-thorium-gadolinium (U-Th-Gd) fuel rods. The U-Th fuel rod has only FUE 3 fuel pellets. The U-Th-Gd fuel rod has FUE 2 fuel pellets, except in the top and bottom 30 cm of the fuel rod that are FUE 3 fuel pellets, as shown in Fig. 4.2. Both of the U-G and U-Th-G fuel rods have the traditional burnable absorber gadolinium.

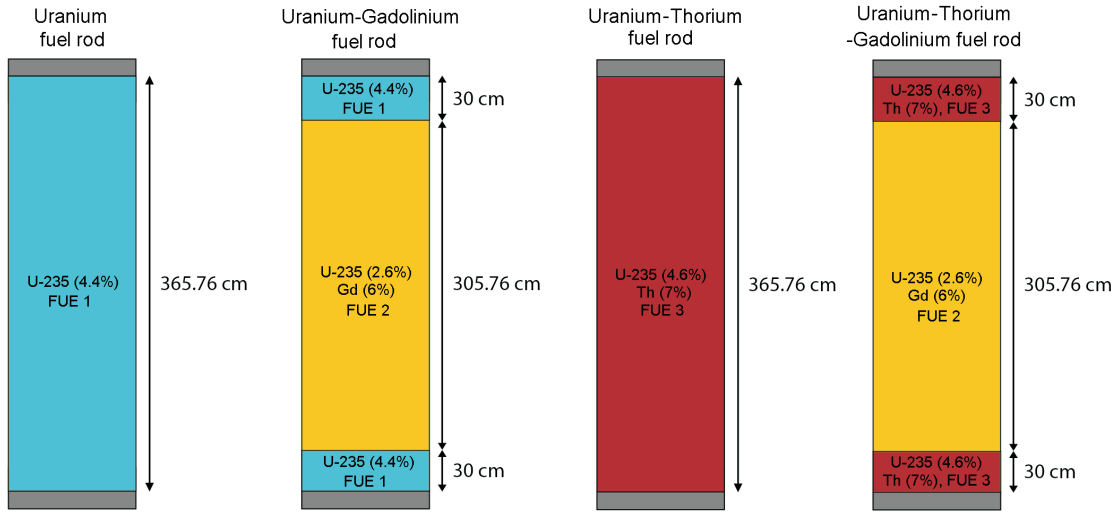

Figure 4.2: Fuel rod designs in the axial direction.

The fuel assemblies are based on a 17x17 lattice, and have octagonal symmetry. Each fuel assembly has 264 fuel rods and 25 guide tubes. The U-G fuel assembly has 12 U-G fuel rods, which contain gadolinium, and the rest are U fuel rods. The U-Th fuel assembly has 8 U-Th-G fuel rods, thus a lower amount of gadolinium, and the rest are U-Th fuel rods, as shown in Fig. 4.3

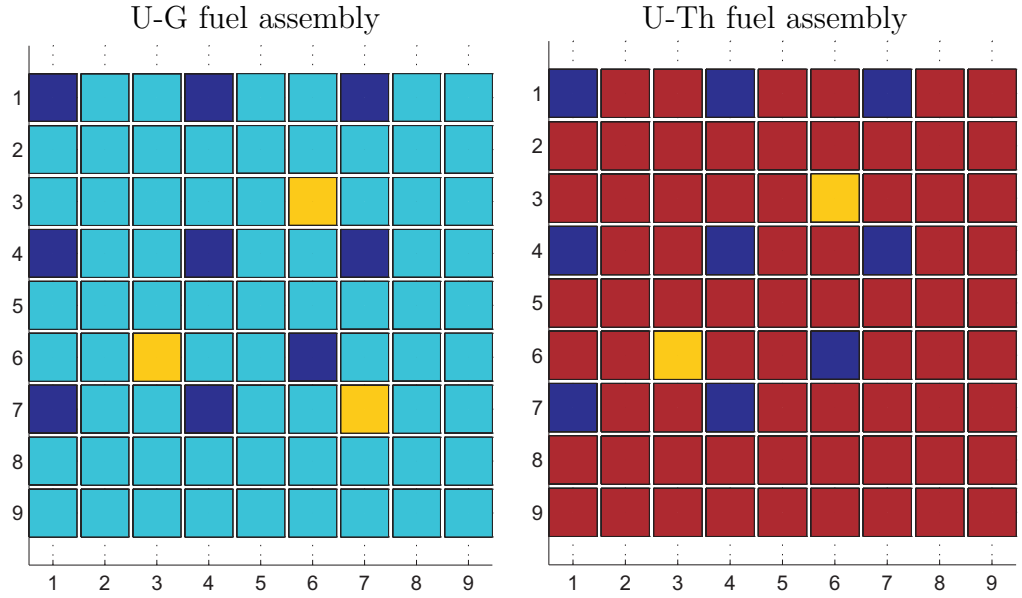

Figure 4.3: Lower right quarter of the 17x17 U-G and U-Th fuel assemblies. The dark blue color represents the guide tubes. The light blue color represents the U fuel rods and the red color represents the U-Th fuel rods. In the U-G fuel assembly, the orange color represents the U-Th fuel rods, while in the U-Th fuel assembly, the orange color represents the U-Th-G fuel rods.

4.3 2D fuel lattice characteristics

The 2D infinite lattice calculations of the fuel assemblies is based on a two dimensional horizontal plane in the middle of the fuel assemblies, which is the most significant part. k_∞ and other important parameters are presented in this section.

4.3.1 k_∞

The fertile absorber in the U-Th fuel assembly displays a different behavior in k_∞ compared with traditional burnable absorbers in the U-G fuel assembly, and thorium reduces the decreasing slope of k_∞. The lower and higher k_∞ at the first and second halves of the burnup, respectively, result in lower power at BOL and higher relative power at the end of life (EOL). This property is favorable for core operation since

one of the most restrictive parameters is the high power in the fresh fuel assemblies. Therefore, reducing the power in the fresh fuel assemblies, and increasing the power in the old fuel assemblies, enable larger thermal margins. However, because of the relative low absorption cross section in thorium compared to gadolinium, fuel assemblies with fertile absorber still require gadolinium to achieve the optimal k_∞ during the cycle. Fig. 4.4 shows the k_∞ in the U-Th fuel assembly with fertile absorber and the traditional U-G fuel assembly.

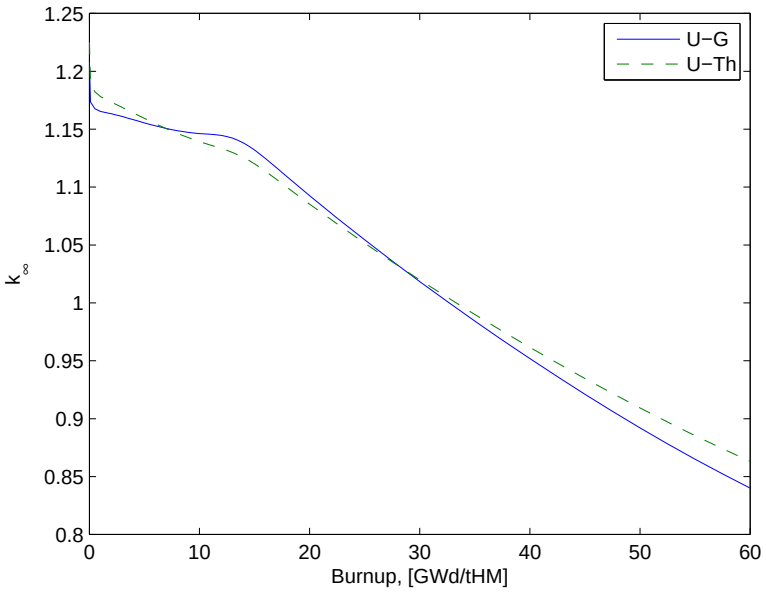

Figure 4.4: The k_∞ as function of burnup in the U-G and U-Th fuel assemblies.

4.3.2 Other remarks

The combination of uranium and a small amount of thorium in the U-Th fuel enables optimization of LWR operation by means of improved efficiency in neutron absorption. Thorium has higher absorption cross section compared with U-238 in the thermal region, which hardens the neutron spectrum and contributes to lower control rod worth. Additionally, the neutron absorption during slowing down of fast neutrons is more efficient, because of the additional resonance peaks brought by thorium, although U-238 has more efficient resonance peaks. However, fuel containing only uranium have U-238 resonance peaks fully saturated, meaning a reduced amount of U-238 will barely affect the resonance neutron absorption caused by the U-238 shielding effect. Meanwhile, adding small amounts of thorium, the number of total resonance peaks increase, which causes a more negative Doppler temperature coefficient of reactivity and could decrease the shutdown margin (SDM). This is discussed in more detail in paper I.

4.4 Core designs

The U-G and U-Th cores contain U-G and U-Th fuel assemblies, respectively. In Fig. 4.5, both of the cores have almost identical designs such as:

- Octagonal symmetry.
- Minimized neutron leakage by placing the highest burnup fuel assemblies at the periphery.
- Cycle length.
- Even radial power distribution obtained by placing fresh fuel assemblies without burnable absorber at semi-peripheral positions.
- Shielding fuel assemblies (consist of steel rods, low enriched and depleted uranium fuel rods) at the position closest to the reactor vessel to protect the belt-line-weld from fast neutron flux [31].
- Fresh fuel assemblies with burnable absorber (U-G and U-Th fuel assemblies) closer to the center of the core, but symmetrically distributed to suppress power peaks.

The differences between the core loading patterns arrive from achieving the lowest power peaks for each individual core with different burnable absorber designs.

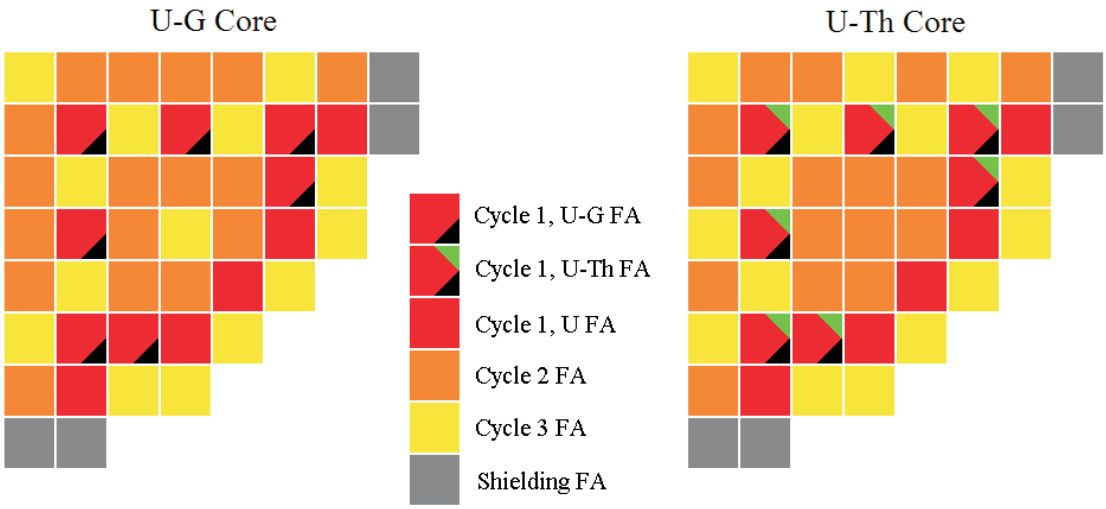

Figure 4.5: Lower right corner of the U-G and U-Th core designs.

4.5 3D core characteristics

The 3D nodal core calculations supply necessary data to demonstrate safe core operation. In this section, one of the most important and significant parameters for the U-Th fuel is presented. The results are from the U-G and U-Th cores reaching equilibrium conditions, after seven consecutive cycles with the same loading patterns.

4.5.1 Power distributions

The importance of an even power distribution is related to the maximum cooling capability during normal operation and transients, because insufficient cooling in combination with high local power could lead to fuel damages. Therefore, the importance to maintain sufficient cooling capability at all time is highly prioritized. To guarantee sufficient cooling in the whole core, the core total power is restricted by the highest local power, and therefore, an even power distribution is highly desired.

The U-Th core could achieve a more even power distribution by having a lower pin peak power and axial offset (AO). The pin peak power is the highest fuel pin power divided by the average fuel pin power. The AO is defined as:

$$AO = \frac{P_{top} - P_{bottom}}{P_{top} + P_{bottom}} \tag{4.1}$$

where P_{top} and P_{bottom} are the average powers at the top and bottom halves of the core, respectively [32].

The U-G and U-Th cores have about 3.9 % and 6.8 % margin to the constrained pin peak power limit, respectively, as shown in Fig. 4.6. In Fig. 4.7, the maximum AO in the U-Th core is about 33 % lower compared with the U-G core for Xe-equilibrium conditions. Therefore, the power in the U-Th core (fertile absorber) is more evenly distributed compared with the U-G core (traditional burnable absorber). The more even power distribution in the U-Th core allows for less core power restrictions and more flexible core designs.

4.5.2 General remarks

The U-Th core showed improved power distribution, but requires about 3.2 % more natural uranium compared with the U-G core, which increases the total fuel cost. In Tab. 5.3, the U-Th core requires a few remarks, but no essential differences:

- The MTC is more negative in the U-Th core, because the increase of moderator

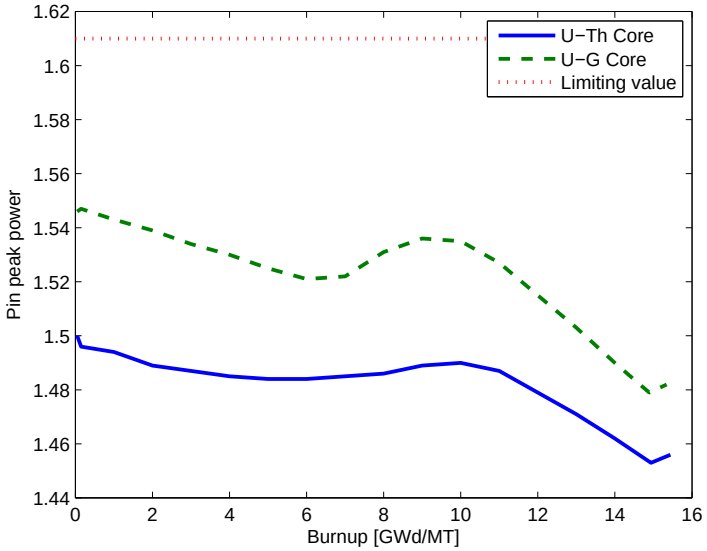

Figure 4.6: Pin peak power as function of burnup for the U-G and U-Th cores.

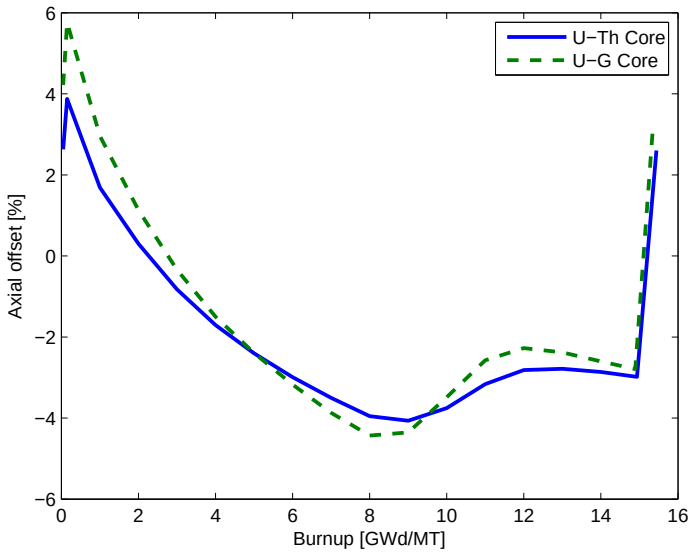

Figure 4.7: Axial offset as function of burnup for the U-G and U-Th cores.

temperature results in more neutrons being absorbed in the Th-232 resonance areas.

- The more negative DTC could be explained by the additional resonance peaks brought by the Th-232, which broadens the total resonance area.
- The ITC stronger negative feedback is contributed by the more negative DTC

and MTC.

- The boron worth is lower in the U-Th core, because of Th-232 slightly hardens the neutron spectrum.
- The SDM shows no significant differences between the cores.
- The fraction of delayed neutrons will not change, because of Pu-239 and U-233 have similar fractions of delayed neutrons [33].

	Limit	U-G Core	U-Th Core
Max MTC (BOC) [pcm/K]	-	-1.37	-2.02
Max ITC (BOC) [pcm/K]	< 0	-4.60	-5.52
Min boron worth [pcm/ppm]	≥ -15	-7.96	-7.71
Max boron worth [pcm/ppm]	≤ -5	-5.99	-5.88
Min DTC [pcm/K]	≥ -4.00	-3.73	-4.04
Max DTC [pcm/K]	≤ -1.70	-2.11	-2.24
Min SDM [pcm]	≥ 2000	2453	2443
Max $\beta_{eff}(BOC)$ [pcm]	≤ 720	645	644
Min $\beta_{eff}(EOC)$ [pcm]	≥ 430	502	502

Table 4.2: Significant key parameters for the U-G and U-Th cores. Their limits is based on the reload safety evaluation specific for Ringhals-3.

CHAPTER

5

THORIUM-PLUTONIUM CORE CHARACTERISTICS FOR PLUTONIUM INCINERATION

The last few decades use of uranium based fuels in thermal reactors have accumulated a large stockpile of reactor graded plutonium. This chapter presents two different types of core configurations to reduce the plutonium, a full Th-MOX core and a 1/3 Th-MOX core. The full Th-MOX core has high efficiency in plutonium incineration, but the 1/3 Th-MOX core has a more traditional core characteristics and is easier to implement.

5.1 Fuel designs

The two types of core configurations utilize different kinds of fuel assembly designs, Th-MOX and Th-MOX uranium compatible (Th-MOX-UC) fuels. The full Th-MOX core uses only Th-MOX fuel assemblies, while the 1/3 Th-MOX core uses Th-MOX-UC, U-G and U fuel assemblies. The U-G and U fuel assemblies have been presented in the previous chapter. The Th-MOX and Th-MOX-UC fuel assemblies use thorium-plutonium fuel pellets, but different concentrations of plutonium and thorium. The fuel assemblies have a 17x17 lattice with 264 fuel rods and 25 guide tubes. The active height of each fuel rod is 365.76 cm. The Th-MOX fuel assemblies has three fuel pellet zones in the horizontal plane, which varies from 9.0 to 13.7 wt% plutonium of the heavy metal and the remaining part is thorium, as shown in Fig. 5.1. The Th-MOX-UC fuel assembly has different concentrations of plutonium and more fuel pellet zones in order to be compatible to uranium fuel assemblies.

Pu-239 has higher fission cross section and requires a lower neutron flux for the

same power level compared with U-235. The 1/3 Th-MOX core has both uranium and plutonium in different fuel assemblies. Consequently, the Th-MOX-UC fuel assembly needs to be compatible with neighboring uranium based fuel assemblies. The Th-MOX-UC fuel assembly has four fuel pellet zones with bigger differences in plutonium concentration, from 2.3 to 10.2 wt% of heavy metal. The highest concentration of plutonium is at the center of the fuel assembly and the lowest plutonium concentration is at the corner of the fuel assembly, as shown in Fig. 5.2.

The plutonium vector in both of the fuel assemblies is: 2 % Pu-238, 53 % Pu-239, 25 % Pu-240, 15 % Pu-241 and 5 % Pu-242. This plutonium isotope vector corresponds to light water fuel with a discharge burnup around 42 MWd/kgHM if reprocessed immediately [34].

The choice of burnable absorber for the fuel assemblies is integral fuel burnable absorber (IFBA). The reason not to use other burnable absorber, such as gadolinium, is to avoid more than two elements mix into the same fuel pellet, because this complicates the estimation of thermal and mechanical properties. It is necessary to use a burnable absorber to reduce the need of boron concentration in the moderator and to obtain a negative isothermal temperature coefficient of reactivity.

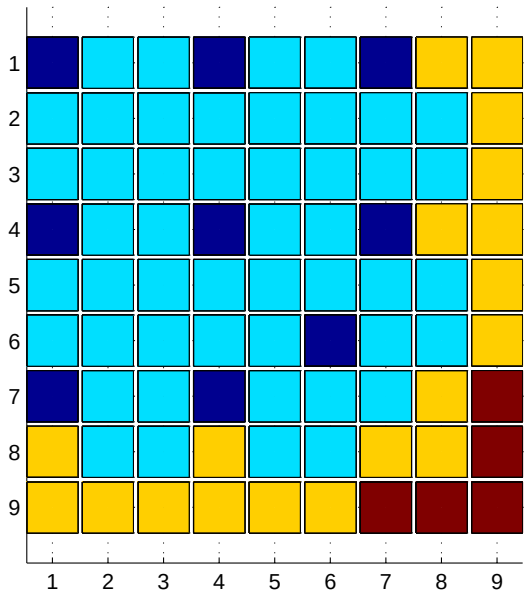

Figure 5.1: Lower right quarter of the Th-MOX fuel assembly. The light blue color, yellow color, dark red color represent the fuel rods that have 13.7 wt%, 11.7 wt%, and 9.0 wt% plutonium of the heavy metal, respectively. The dark blue color represents the guide tubes.

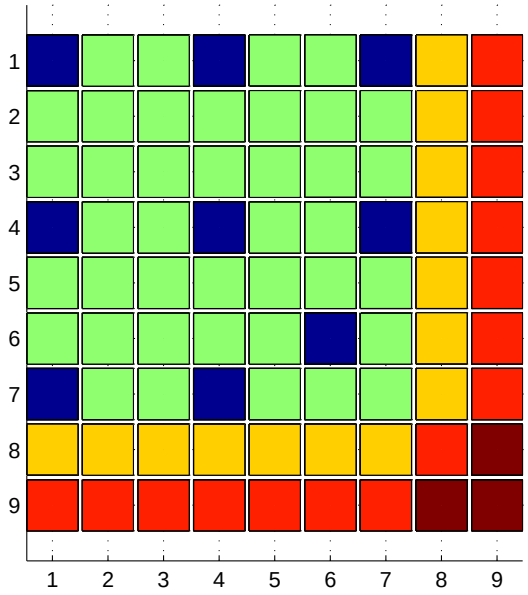

Figure 5.2: Lower right quarter of the Th-MOX-UC fuel assembly. The dark red color, light red color, yellow, green color represent the fuel rods that have 2.3 wt%, 4.6 wt%, 6.8 wt%, 10.2 wt% plutonium of the heavy metal, respectively. The dark blue color represents the guide tubes.

5.2 2D fuel lattice characteristics

The 2D infinite lattice calculations of the fuel assemblies are based on a horizontal plane of the fuel assemblies, and simulates k_∞ and other important parameters.

5.2.1 k_∞

One of the biggest differences between uranium and thorium based fuel in light water reactors is the reduction of k_∞ as function of burnup. Thorium based fuel has a lower reduction of k_∞ compared with uranium based fuel, because of the higher thermal absorption cross section and the presence of the fissile isotope U-233. Thorium and uranium based fuels are mostly influenced by the fertile isotopes Th-232 and U-238, respectively. Th-232 has about three times higher thermal neutron absorption cross section compared with U-238, and contributes to a lower k_∞. Th-232 and U-238 create U-233 and Pu-239 from neutron capture, respectively, and the higher neutron capture cross section in Th-232 results in a more effective breeding of U-233.

In Fig. 5.3, it is shown that k_∞ in Th-MOX and Th-MOX-UC fuel assemblies decreases more slowly compared with an U-G fuel assembly.

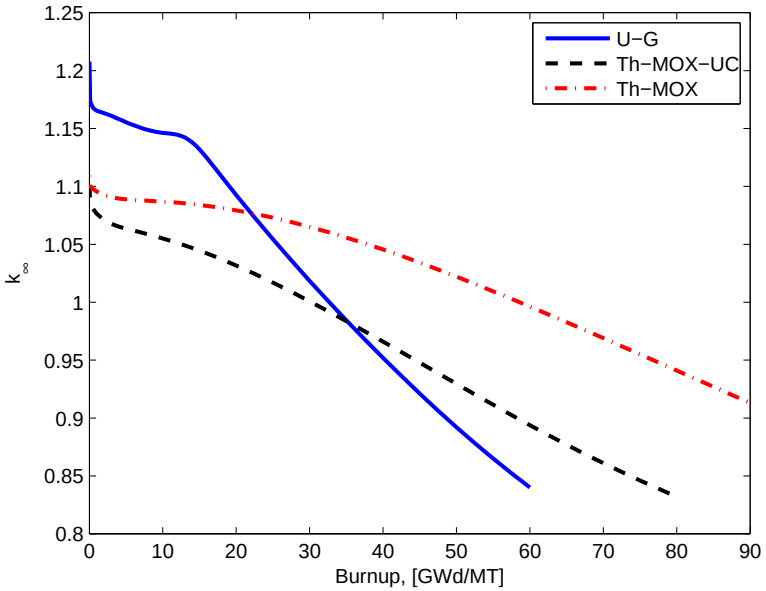

Figure 5.3: k_∞ in the U-G, Th-MOX and Th-MOX-UC fuel assemblies.

5.2.2 Other remarks

The higher thermal absorption cross section in Th-232 demands higher concentration of fissile isotopes to maintain core criticality, but Th-232 has higher conversion ratio (the average number of fissile atoms created per consumed fissile atom) compared with U-238. Therefore, thorium based fuels require higher initial costs, because of the necessary higher initial fissile concentration. The higher thermal absorption cross section in thorium and plutonium hardens the neutron spectrum, and therefore cores with the Th-MOX and Th-MOX-UC fuels have different core characteristics compared with cores using only U-235 fuel.

5.3 Core designs

The full Th-MOX core uses Th-MOX fuel assemblies and has a three batch loading scheme with octagonal symmetry, as shown in Fig. 5.4. The high concentration of fissile isotopes allows for an 18 months core cycle. The 1/3 Th-MOX core has a more complicated core loading scheme, since it uses three types of fuel assemblies and alternates between 80 and 76 new fuel assemblies in each batch, because of the 2 batch loading scheme in a 15 months core cycle. The three types of fuel assemblies are Th-MOX-UC, U-G and U fuel assembly with octagonal symmetry. Both of the cores use the low neutron leakage loading pattern to achieve high discharge burnup by placing the oldest fuel assemblies at the periphery. However, to achieve a radially

homogenous power distribution, the majority of the fresh fuel assemblies are located at semi peripheral positions.

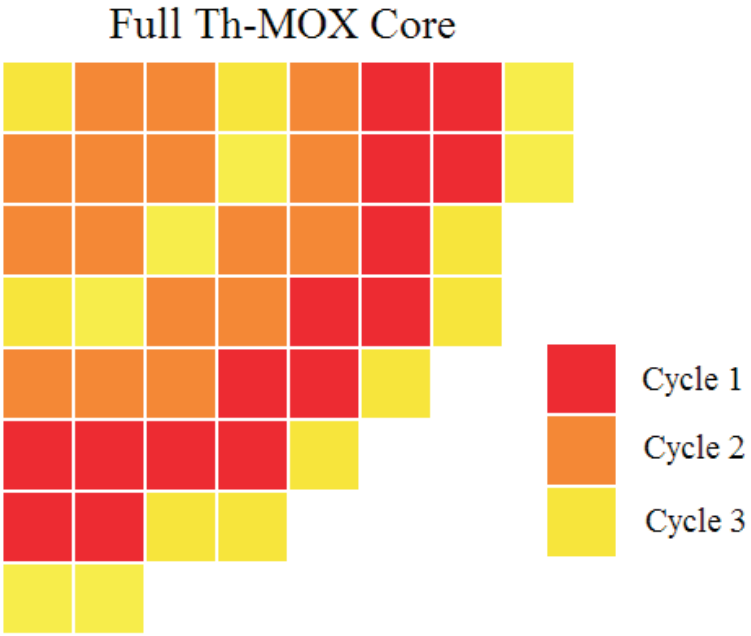

Figure 5.4: The figure shows the lower right corner of the full Th-MOX core loading pattern.

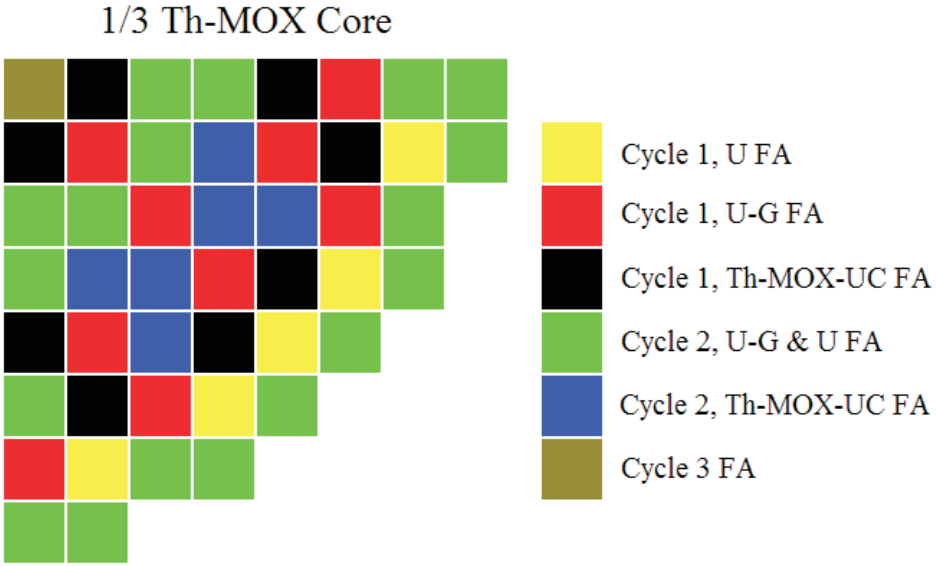

Figure 5.5: The figure shows the lower right corner of the 1/3 Th-MOX core loading pattern.

5.4 Core characteristics

Th-MOX fuel contributes to different core characteristics compared with UOX fuel. In this section, the most important differences are presented to understand Th-MOX cores characteristics such as: plutonium incineration rate, boron worth, control rods worth, feedback coefficients and fraction of delayed neutrons.

5.4.1 Plutonium incineration rate

Uranium-plutonium mixed oxide (U-MOX) fuel has been used to minimize the plutonium stockpile, but the uranium in U-MOX will continue to produce new plutonium and makes the plutonium incineration less efficient. In recent studies, Th-MOX fuel has shown to be a viable choice to reduce the plutonium stockpile. The thorium in the Th-MOX fuel acts as a plutonium carrier and produce barely any new plutonium. Therefore, Th-MOX fuel is very efficient on plutonium incineration [35–40].

UOX cores generate about 30 kg of plutonium per TWhe, while full or partial Th-MOX or U-MOX cores incinerate plutonium. However, the full Th-MOX core is about two times more efficient on plutonium incineration compared with a full U-MOX core, as shown in Tab. 5.1. Moreover, the 1/3 Th-MOX core and the 30 % U-MOX core incinerate about 16 and 1 kg of plutonium per TWhe, respectively, concluding that Th-MOX fuel is more efficient on plutonium incineration.

	Plutonium consumption rate [kg/TWhe]
UOX core	-30
1/3 Th-MOX core	16
Full Th-MOX core	116
30 % U-MOX core [41]	1
Full U-MOX core [41]	62

Table 5.1: The plutonium consumption rate.

5.4.2 Boron concentration and control rods

Thorium and plutonium have higher thermal absorption cross section compared with uranium, which hardens the neutron spectrum. Therefore, Th-MOX fuel decreases the thermal neutron absorption efficiency in other materials, and require: enriched boron, more effective and higher number of control rods. The full Th-MOX and 1/3 Th-MOX cores require enriched boron not to exceed the desired initial boron concentration, and the maximum boron worth, as shown in Tab. 5.2 and Fig. 5.6. The 1/3 Th-MOX core requires 30 % enrichment of B-10 compared with natural

boron (natural boron has 19.8 % of B-10), while a full Th-MOX core requires 60 % enrichment. The UOX core uses 48 silver-indium-cadmium control rods, but the 1/3 Th-MOX requires four additional control rods to guarantee sufficient SDM. The full Th-MOX core requires besides four additional control rods, also boron carbide control rods to stay within the SDM limit.

	Limit	U-G	1/3 Th-MOX	Full Th-MOX
Min boron worth [pcm/ppm]	≥ -15	-7.96	-9.94	-8.69
Max boron worth [pcm/ppm]	≤ -5	-5.99	-6.65	-5.29
Min SDM [pcm]	≥ 2000	2453	3243	3895

Table 5.2: Boron worth and SDM for the U-G, 1/3 Th-MOX and full Th-MOX cores. Their limits is based on the reload safety evaluation specific for the Ringhals-3.

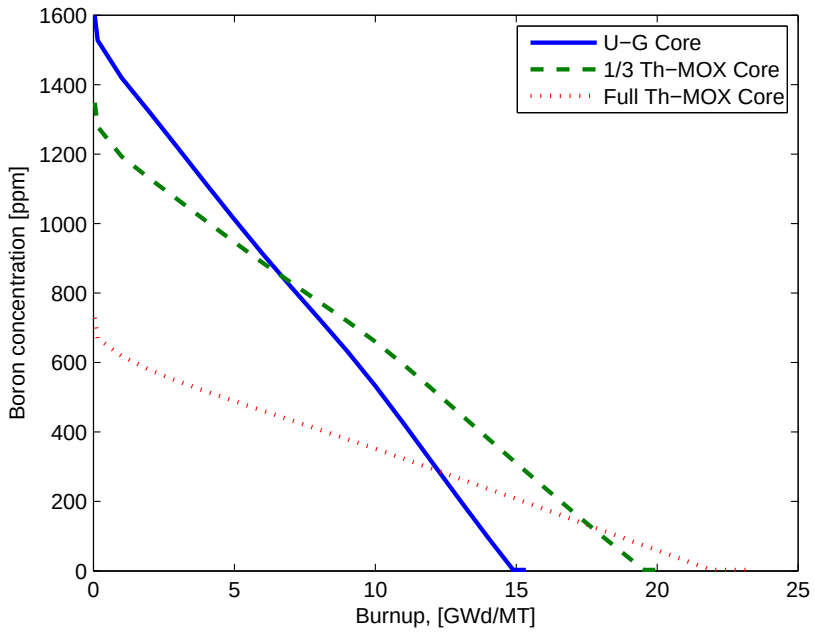

Figure 5.6: The boron concentration as function of burnup for the U-G, 1/3 Th-MOX and full Th-MOX cores.

5.4.3 Feedback coefficients

Feedback coefficients are essential for demonstrating reactor safety. Th-MOX fuel affects two important feedback coefficients, MTC and DTC. In Tab. 5.3, the 1/3 Th-MOX and full Th-MOX cores show weaker MTC feedback at end of cycle

(EOC) compared with the UOX core, because of less sensitivity to neutron spectrum changes and less dependence on thermal fission. Therefore, Th-MOX based cores are less sensitive to coolant-temperature-induced transients compared with UOX based cores. The Doppler coefficients of reactivity feedback is stronger throughout the cycle in the Th-MOX based cores compared with UOX based cores. The stronger Doppler coefficient of reactivity is explained by higher number of resonance peaks, from more isotopes in the Th-MOX fuel.

	Limit	U-G	1/3 Th-MOX	Full Th-MOX
Max MTC (BOC) [pcm/K]	-	-1.37	1.80	-4.02
Min MTC (EOC) [pcm/K]	-	-92.11	-89.27	-76.70
Max ITC (BOC) [pcm/K]	< 0	-4.60	-1.68	-7.92
Min DTC [pcm/K]	≥ -4.00	-3.73	-4.12	-4.53
Max DTC [pcm/K]	≤ -1.70	-2.11	-2.25	-2.36

Table 5.3: Feedback coefficients for the U-G, 1/3 Th-MOX and full Th-MOX cores. Their limits is based on the reload safety evaluation specific for the Ringhals-3.

5.4.4 Fraction of delayed neutrons

Pu-239 and U-233 have lower fraction of delayed neutrons compared with U-235, which makes the reactor more difficult to control upon reactivity perturbations, but could be avoided by having a lower amount Th-MOX fuel as in 1/3 Th-MOX core, as shown in Tab. 5.4. This type of strategy is commonly used in U-MOX cores to avoid low fraction of delayed neutrons. However, it is possible to use a full Th-MOX core, but this would require different reactor conditions and additional transient analysis to assure that all safety criteria are fulfilled.

	Limit	U-G	1/3 Th-MOX	Full Th-MOX
Max $\beta_{eff}(BOC)$ [pcm]	≤ 720	645	560	370
Min $\beta_{eff}(EOC)$ [pcm]	≥ 430	502	452	350

Table 5.4: The fraction of delayed neutrons for the U-G, 1/3 Th-MOX and full Th-MOX cores. Their limits is based on the reload safety evaluation specific for the Ringhals-3.

CHAPTER

6

LOAD-FOLLOWING WITH THORIUM-CONTAINING FUEL

This chapter describes how thorium-containing fuel could be utilized to improve the load-following capacity, by means of reducing the sensitivity to changes in neutron spectrum and xenon poisoning.

6.1 Load-following

Renewable energy is necessary for reducing the emission of green house gases, and achieve sustainable environmental goals. In 2012, the world wide investment on renewable energy was $244 billions and the majority of the investment was in wind and solar power [42]. However, the intermittent power sources such as solar and wind create imbalance between the production and consumption of electrical energies, and demand for more load-following power to compensate the need [43]. Therefore, the need and interest of load-following reactors might increase. The majority of the French nuclear reactor fleet utilizes load-following to stabilize the power grid, but has its limitation such as maximum allowed amplitude of induce fluctuation in the axial power upon rapid changes in total core power [44] or maximum allowed xenon poisoning. However, thorium mixed fuel indicates improved load-following capacity.

Improve load-following capacity in thorium mixed fuel is demonstrated by simulating step wise power changes with the full Th-MOX core and the U-G core at end of full power (EOFP), which are the most sensitive condition for AO instability. The simulation has two stepwise power changes from 100 % to 50 % and from 50 % to 100 % that occur at 0 and 200 hours, respectively. The instantaneous steady state 3D-calculations are performed for each time step of 6 minutes throughout the transient

within the adiabatic approximation, and updated the xenon and iodine concentrations according to built-in capabilities in SIMULATE-3. The full Th-MOX core shows considerably less fluctuations in AO compared with the traditional U-G core, as shown in Fig. 6.1. The full Th-MOX core shows no axial power oscillations and a more even axial power distribution at equilibrium state, while the U-G core shows large axial power oscillations and a more uneven axial power distribution at equilibrium state. It should be noted that no control rods are used to compensate for an uneven AO, because the objective is to demonstrate the core instability characteristics. Analysis on active methods to stabilize xenon oscillations for PWRs have previously been performed in great detail, based on continuous monitoring of AO and automatically adjusting the control rod positions to minimize the axial power swing [45–47]. However, PWRs have no control rods inserted from the bottom of the core and usually no axially graded control rods, thus a highly negative AO in the U-G core is hard to adjust into a positive regime.

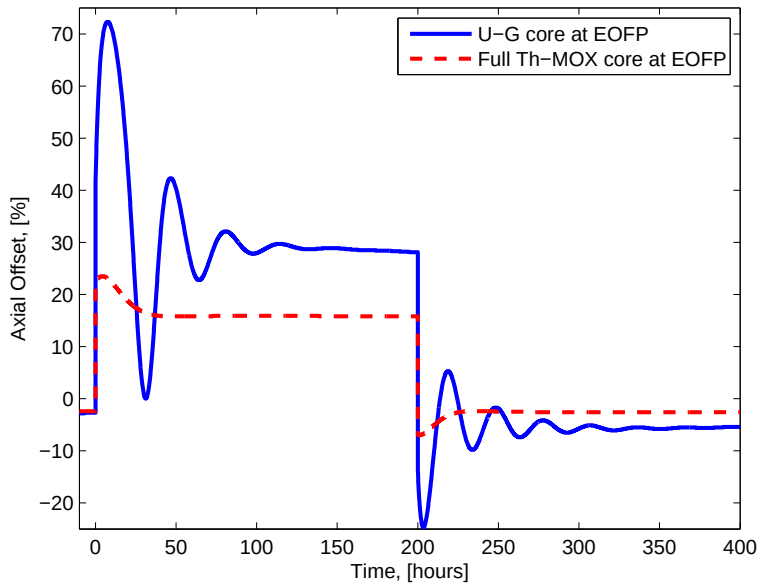

Figure 6.1: The AO as function of time in the U-G and Th-MOX cores.

6.2 Sensitivity to changes in the neutron spectrum

Stepwise power changes alter the moderator temperature gradient and profile, as shown in Fig. 6.2. The reduced core power mostly decreases the moderator temperature and soften the neutron spectrum at the top half of the core, independent of fuel types or cores, as shown in Fig. 6.3. However, different fuel types could respond differently to changes in the neutron spectrum. In Tab. 6.1, from comparing rele-

vant cross sections, it is found that the U-G core is about 3 times more dependent on thermal fission compared with fast fission, while the full Th-MOX core is about 2 times more dependent on thermal fission compared with fast fission. Therefore, the U-G core is more sensitive to neutron spectrum changes compared with the full Th-MOX core.

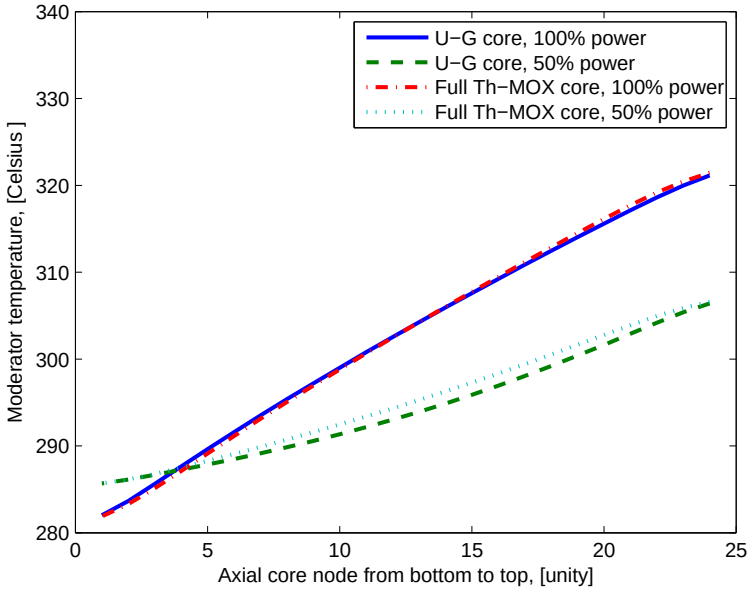

Figure 6.2: Moderator temperature as function of the axial core node from bottom to top in the U-G and full Th-MOX cores at 50 % and 100 % power.

	UOX core	Full Th-MOX core
$\frac{\nu\Sigma_{f,1}}{\Sigma_{a,1}}$	0.47	0.71
$\frac{\nu\Sigma_{f,2}}{\Sigma_{a,2}}$	1.45	1.44

Table 6.1: The $\frac{\nu\Sigma_{f,1}}{\Sigma_{a,1}}$ and $\frac{\nu\Sigma_{f,2}}{\Sigma_{a,2}}$ in the U-G and Th-MOX cores.

6.3 Xenon poisoning and oscillations

Xe-135 is the most important reactor poison, because the high neutron capture cross section, σ_{aX}, has a big effect to the reactivity. Therefore, the Xe-135 concentration, X, is important to track. The xenon poisoning, π_X, is given by:

$$\pi_X = \frac{\sigma_{aX} X}{\Sigma_{aF}} \tag{6.1}$$

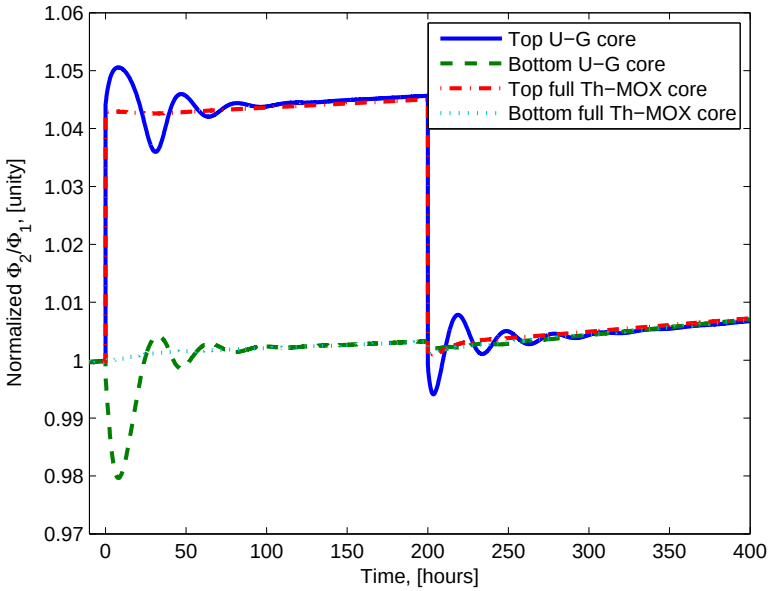

Figure 6.3: Normalized thermal flux, Φ_2, divided by the fast flux, Φ_1, as function of time at the top and bottom in the U-G and full Th-MOX cores. The thermal flux divided by the fast flux is normalized to the value before the onset of the transient.

where the Σ_{aF} is the macroscopic absorption cross section in the fuel without the poison. Therefore, the xenon poisoning is dependent on the xenon concentration and the macroscopic absorption cross section.

Xenon oscillations occur as xenon-, iodine- concentration and neutron flux exhibit out of phase behavior in two or more regions in a core. The period of xenon oscillations varies from 15 to 30 hours [33] and could lead to regional power oscillations [48]. There are three types of regional power oscillations: azimuthal, radial, and axial. The most dominant oscillation in PWRs is the axial oscillation. Higher power levels are more unstable with respect to axial power oscillation, because of the higher xenon production.

In the study of simulated stepwise power changes, the fluctuation of thermal absorption cross section, $\Sigma_{a,2}$, induced by the xenon in the U-G core is significantly larger compared with the full Th-MOX core, since the latter shows no signs of $\Sigma_{a,2}$ fluctuation contributed by the xenon, as shown in Fig. 6.4. However, the total xenon concentration is about 2.5 times higher in the full Th-MOX core compared with the U-G core, because of the three times lower thermal neutron flux, as shown in Fig. 6.5 and Tab. 6.2. However, the Th-MOX core has about 2.5 times higher $\Sigma_{a,2}$ and the 3 times lower thermal neutron flux compared with U-G core. Therefore, the $\Sigma_{a,2}$ fluctuation effect induced by xenon is almost insignificant. Additionally, the full Th-MOX core has less then half of the xenon poisoning compared with the U-G core.

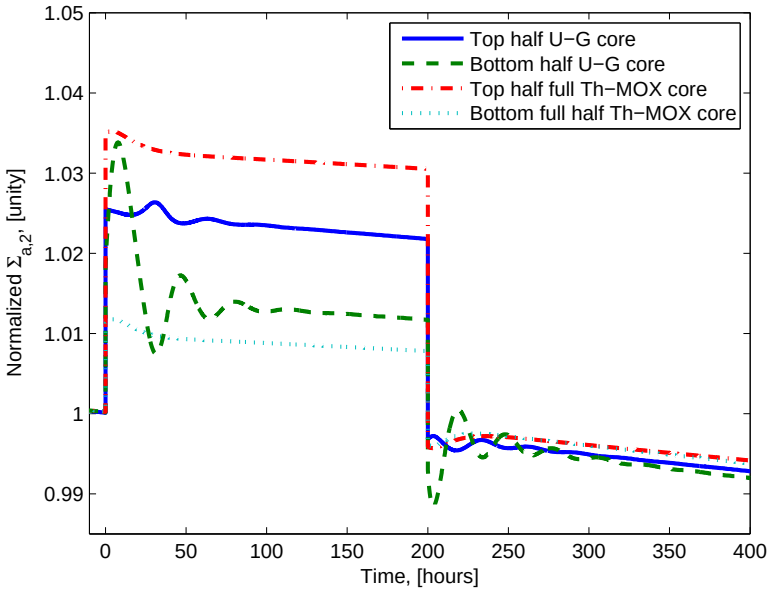

Figure 6.4: The normalized $\Sigma_{a,2}$ as function of time in the U-G and Th-MOX cores. The $\Sigma_{a,2}$ is normalized to the value before the onset of the transient.

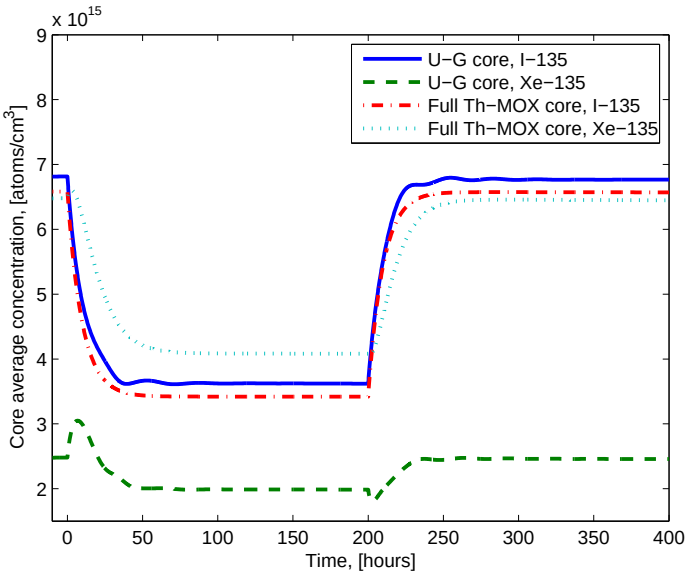

Figure 6.5: The average core I-135- and Xe-135 concentration as function of time in the U-G and Th-MOX cores.

6.4 Reactor deadtime

Load-following cores with flexible operation capacity should be able to regain full power without being restricted by recent significant reduction of power or scram.

	UOX core	Full Th-MOX core
Fast neutron flux	$3.6*10^{14}$ neutrons/(cm^2s)	$3.7*10^{14}$ neutrons/(cm^2s)
Thermal neutron flux	$4.7*10^{13}$ neutrons/(cm^2s)	$1.5*10^{13}$ neutrons/(cm^2s)
$\Sigma_{a,1}$	0.011502 cm^{-1}	0.015790 cm^{-1}
$\Sigma_{a,2}$	0.10986 cm^{-1}	0.26314 cm^{-1}
Xenon poisoning	2950 pcm	1339 pcm

Table 6.2: The $\Sigma_{a,1}$, $\Sigma_{a,2}$, fast and thermal neutron fluxes in the U-G and Th-MOX cores.

At the later half of the cycle, UOX cores are often restricted to regain full power from a scram within 24 hours, because of the build up of xenon poisoning. One of the interesting features in Th-MOX cores is the insensitivity to Xe-135 as described above. In a demonstration on the reduction of excess reactivity after reducing the power from 100 to 0 %, a large difference between a traditional U-G core and a full Th-MOX core is seen. The loss of excess reactivity is about 1884 pcm in the U-G core compared with 61 pcm for the Th-MOX core. Additionally, the full Th-MOX core regains the lost reactivity after 7 hours, while it requires 24 hours to restore the U-G core, as shown in Fig. 6.6. The lower sensitivity to xenon and the lower increase of xenon concentration after the decrease of power, contribute to the smaller loss of excess reactivity. After the decrease of power, the maximum xenon concentration compared with the initial xenon concentration at 100 % power in the full Th-MOX core is about 4 %, while the maximum xenon concentration increases by 67 % in the U-G core, as shown in Fig. 6.8. The reason is the three times lower thermal neutron flux, which gives about 2.5 times higher initial xenon concentration in the full Th-MOX core compared with the U-G core. Moreover, after the decay of Xe-135, the excess reactivity in the U-G core is about 1200 pcm higher compared with the full Th-MOX core, and thus the U-G core requires more SDM to guarantee the subcriticality compared with the full Th-MOX core.

6.5 U-Th and 1/3 Th-MOX cores

The U-Th and 1/3 Th-MOX cores also indicate improved load-following capacity compared with the U-G core, because of the presence of thorium and plutonium. The small amount of thorium in the U-Th core shows about 30 % improved damping in the axial stability compared with U-G core, and is explained in more detail in paper V. Meanwhile, in paper IV, the 1/3 Th-MOX core shows almost five times more efficient damping of the axial power oscillation induced by the xenon. Therefore, it could be concluded that fuel with thorium will have improved load-following capacities.

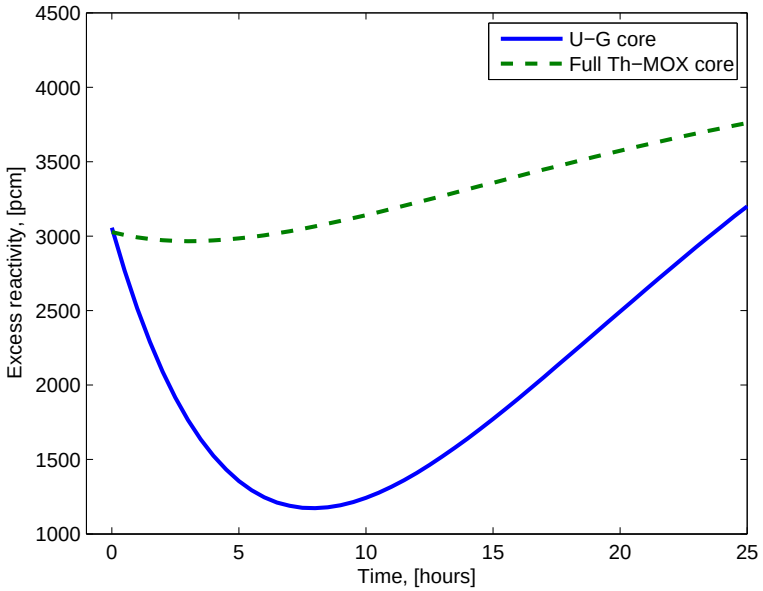

Figure 6.6: Excess reactivity as function of time in the U-G and Th-MOX cores.

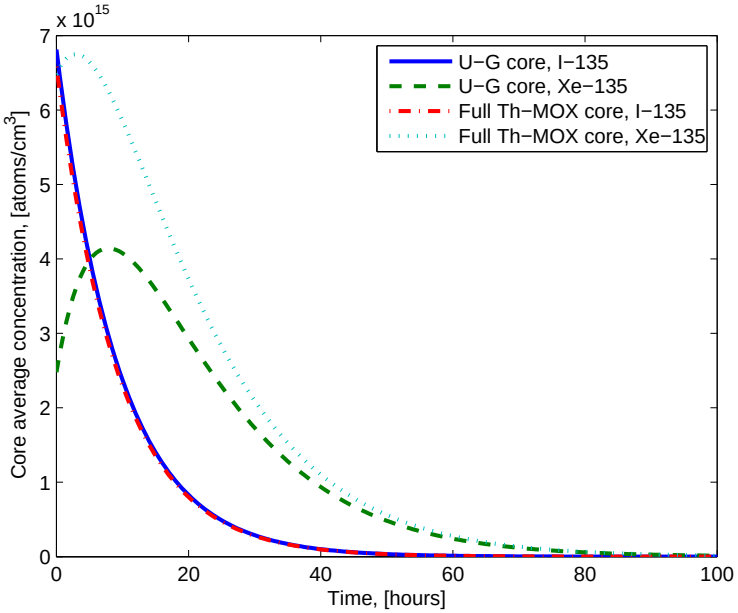

Figure 6.7: I-135 and Xe-135 concentration as function of time in the U-G and Th-MOX cores.

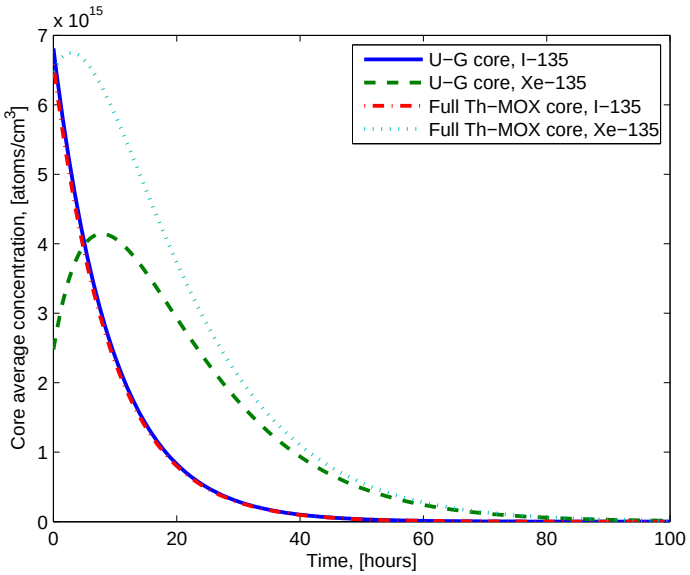

Figure 6.8: I-135 and Xe-135 concentrations as function of time in the U-G and Th-MOX cores.

CHAPTER

7

SUMMARY AND OUTLOOK

This last chapter summarizes the fertile absorber, thorium-plutonium core characteristics for plutonium incineration and load-following with thorium-containing fuel presented in the thesis. Thereafter, a short outlook is presented.

7.1 Summary

The fertile absorber concept uses a small amount of thorium mixed in with enriched uranium (below 5 % enrichment of U-235) fuel pellets to suppress local reactivity. The objective for this is to achieve a more homogenous power distribution, and to reduce the needs for traditional burnable absorbers such as gadolinium or boron. The concept still requires traditional burnable absorbers to reach the best performance, because the absorption cross section in thorium is considerably lower compared with traditional burnable absorbers. However, the fuel pellets have maximum two elements, because it is complicated to estimate the thermal properties without any experimental data. There are a few important remarks about thorium as fertile absorber, such as slower decrease of reactivity as function of burnup, higher thermal absorption cross section, more efficient neutron absorption during the slowing-down process, more homogenous power distribution and higher amount of natural uranium required for the same discharge burnup. The slower decrease of reactivity results in a more homogenous core power distribution since fresh fuel assemblies have lower power and older fuel assemblies have higher power compared with traditional fuel. Moreover, the required amount of gadolinium in the fertile absorber is lower and improves the intra-assembly power distribution. The inclusion of thorium contributes to a higher thermal absorption cross section and a correspondingly lower thermal neutron flux, control rod worth, and boron worth. The more efficient neutron ab-

sorption during the slowing-down process causes a stronger Doppler temperature feedback and a larger reactivity swing from hot full power to cold zero power. The more even power distributions provide for larger thermal margins, better conditions for power-uprate programs or flexible core designs. However, the extra fuel cost for introducing thorium and the increasing U-235 enrichment should be carefully considered.

Th-MOX fuel is efficient to reduce the plutonium stockpile accumulated in the last few decades. A full Th-MOX core consumes about 116 kg plutonium per TWhe, but requires 60 % enrichment of B-10 in the reactor coolant, and increased number of control rods with improved efficiency, utilizing e.g. boron carbide. A 1/3 Th-MOX core consumes about 16 kg plutonium per TWhe, and requires 30 % enrichment of B-10 in the reactor coolant, and more traditional control rods. However, Th-MOX fuel is about twice more efficient on plutonium incineration compared with U-MOX fuel. The higher absorption cross section in Th-232 demands higher fissile concentration to maintain criticality. Thorium and plutonium harden the neutron spectrum, and in the full Th-MOX core, the thermal neutron flux is three times lower compared with the UOX core. There are two additional features about the Th-MOX fuel: stronger feedback in the Doppler coefficients of reactivity and lower fraction of delayed neutrons. However, a lower fraction of delayed neutrons could be avoided by substituting parts of Th-MOX with ordinary UOX fuel, which will cause a higher fraction of delayed neutrons and a more similar UOX core characteristics. Therefore, partial Th-MOX cores are easier to implement compared with full Th-MOX cores, but have less efficient plutonium incineration compared with full U-MOX cores.

Load-following nuclear reactors are used in France, but this concept has its limitations as e.g. allowed amplitudes of induced axial power fluctuations from rapid changes in total core power. Regulating the core power, changes the moderation and the neutron spectrum, which shifts the axial power distribution. However, the Th-MOX fuel is less sensitive to changes in the neutron spectrum since they are less dependent to thermal fission compared with UOX fuel. Moreover, the higher absorption cross sections and lower thermal neutron flux cause a lower xenon poisoning, and reduce any xenon induced oscillations. Also, the xenon poisoning effect is especially reduced during rapid power reduction or scram. Therefore, Th-MOX fuel could improve the load-following capacity in nuclear reactors even further. Fuel containing smaller amounts of thorium has also shown similar tendency of improved stability, but is dependent on the amount of thorium for notably improving load-following capacity.

7.2 Outlook

The concept of fertile absorber and load-following with thorium-containing fuel could be further investigated with simulations and experiments. Here are some suggestions

for possible future research:

- The concept of fertile absorber could be utilized in BWRs or other types of reactors.
- For commercializing the concept of fertile absorber, experiments on U-Th fuel pellets are needed to obtain fission gas release, thermal and mechanical properties.
- Transient analysis with U-Th and Th-MOX fuels is needed in order to provide a deeper understanding on the core performance.
- Current PWRs use automatic adjustment of control rods positions to minimize the axial power swing. Therefore, it is interesting to investigate a thorium-containing core in load-following scenarios using control rods, because it is possible that a thorium-containing core responds differently to control rod adjustments compared with a traditional U core.

ACKNOWLEDGEMENTS

I want to first thank my main supervisor Henrik Nylén for all the helps, supports and encouragements. He has always wise answers and interesting perspectives to problems. He is much more down to earth than me, and when I rush into new ideas, he always helps me to see the reality.

Christophe Demazierè is a very kind person. He is very supportive and I can always count on him.

Imre Pàzsit really know how to motivate and encourage his colleagues. I really appreciate that he entrusted me to work on others tasks beside scientific projects.

Urban Sandberg is always helpful with the technical matters. I really appreciate the discussions in Ringhals. I have learned a lot. I want to thank Rune Axelsson to allow me to visit RTHH. The personal in RTHH made me feel very welcome during my time in Ringhals. I hope I will have chance to work in Ringhals in the future.

Victor is a perfect roommate. It has been a lot of fun times and interesting discussions. I really enjoy our chess games, even though that he beat me most of the time.

Viktor and Eva helped me a lot and made me feel very welcome during my first period in Chalmers.

Anders N is a nice boss. He taught me a lot about the administrations issues.

Joska, I just to want to say. Thank you Sir :D

Paolo, thanks for all the interesting talks.

Lennart and Lasses are fun to have around and thanks for making servers work so perfectly.

Albert and Irina, it was really fun to organize the lunch seminar together. It would not been so successful without them.

Thanks to all my former and current colleagues in Chalmers for making my days

much more fun and enjoyable. I will miss the coffee breaks, floor ball and football.

Vattenfall AB, OKG AB (Oskarshamns Kraftgrupp AB) & SKC (Swedish Centre for Nuclear Technology) are acknowledged for the financial support, as well Ringhals AB for the scientific and technical support.

I want to thank my girlfriend, brother, father, grand mother and mother for all the support. You have all contributed so much to my life. I will always cherish you.

ABBREVIATIONS

AO	Axial offset
BOL	Beginning of life
CANDU	Canada deuterium uranium
DTC	Doppler temperature coefficient of reactivity
EOC	End of cycle
EOFP	End of full power
EOL	End of life
IFBA	Integral fuel burnable absorber
ITC	Isothermal temperature coefficient of reactivity
k_∞	Infinite multiplication factor
MTC	Moderator temperature coefficient of reactivity
SDM	Shutdown margin
Th-MOX	Thorium-plutonium mixed oxide
Th-MOX-UC	Thorium-plutonium mixed oxide uranium compatible
U	Uranium
U-G	Uranium-gadolinium
U-MOX	Uranium-plutonium mixed oxide
U-Th	Uranium-thorium
Wt%	Weight percentage

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