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Material requirements for a thorium based nuclear fuel

Final report being equivalent to a Master´s thesis required in partial fulfillment for the degree of Master of Science in Technology in the Degree Programme in Engineering Physics and Mathematics.

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Abstract:

The increase in the energy consumption and the expected growth in the nuclear capacity make it necessary to look for alternative fuels to replace uranium. The fuel chosen, which was also considered in the early stages of nuclear energy, is thorium. Thorium has some characteristics that make it valuable as a fuel, like its abundance, the low radiotoxicity of the waste generated, the higher economy regarding its larger absorption cross-section and higher burnups and the proliferation resistance as compared to uranium. Despite these benefits it also raises some questions relating its safe operation in the reactor.

The aim of this work is to offer an overview about the use of thorium as a fuel element in a power reactor and the critical issues that the cladding faces. The programs run in different countries to use thorium, the benefits and challenges that presents and the physical configurations inside the reactor are explained. This work focuses in the configuration proposed by A.Radkowsky which is to have thorium (blanket) and enriched uranium (seed) in different assemblies. The physical schemes in the reactor core are the seed-blanket unit and the whole-assembly seed and blanket core.

The increased power density, higher burnup and longer residence time in the reactor of thorium fuel enhance some potential failure mechanisms which are presented in this work. This thesis also seeks to give a general idea about the materials used in the reactor, focusing on the cladding that is the first barrier and the element subjected to toughest operating conditions.

A modeling program called FEMAXI is used to simulate the interaction between the fuel element and the cladding in the high burnup region. Two physical phenomena are modeled, inner pressure and cladding corrosion, showing that the limiting factor would be corrosion due to the long residence time in the reactor.

In order to understand the difficulties to reach the operating conditions of thorium fuel, an overlook at the licensing process is done. It shows the strict safety conditions which have to be accomplished, especially with postulated accidents.

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Preface

This Final Report was carried out at the Department of Applied Physics of Aalto University School of Science.

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Finally, I would like to express my appreciation and gratitude to my family, especially my parents, for their support, encouragement and advice during my studies and during the realization of this project.

Contents

List of Abbreviations

BOL	Beginning Of Life
BWR	Boiling Water Reactor
CANDU	CANada Deuterium Uranium
CCS	Fossil Carbon Capture and Storage
CFCC	Continuous Fiber Ceramic Composite
DNB	Departure from Nucleate Boiling
FGP	Fission Gas Products
FR	Fast Reactor
GHG	Green House Gas
HBS	High Burnup Structure
HEU	Highly Enriched Uranium
HRP	Halden Reactor Project
HTR	High Temperature gas-cooled Reactor
HWR	Heavy Water Reactor
IAEA	International Atomic Energy Agency
IPCC	Intergovernmental Panel on Climate Change
LEU	Low Enriched Uranium
LOCA	Loss of Coolant Accident
LMFR	Liquid Metal Fast breeder Reactor
LWR	Light Water Reactor
PCMI	Pellet to Cladding Mechanical Interaction
PHWR	Pressurized Heavy Water Reactor
PWR	Pressurized Water Reactor
PWRT	Pressurized Water Reactor Thorium
RBMK	Reaktor Bolshoy Mashchnosti Kanalniy
RIA	Reactivity Initiated Accident
RTR	Radkowsky Thorium Reactor
SBU	Seed-Blanket Unit
SCC	Stress Corrosion Cracking
SPP	Second Phase Particles
SWU	Separative Work Unit
UN	United Nations

VVER (WWER)	Vodo-Vodyanoi Energetichesky Reactor
WASB	Whole-Assembly Seed and Blanket core
WPu	Weapons grade Plutonium

List of Symbols

Ψ	Apparent average number of gas atoms per unit intergranular volume
A	Area of the pellet
q^i	Average heat generation density of axial segment j (W/cm^3)
N_A	Avogadro's number
Bu	Burnup
dS/dt	Corrosion rate
ρ	Density
D'	Diffusion coefficient of gas atoms (cm^2/s)
E_f	Energy generated per one fission
φ	Fast neutron flux
P^{ij}	Fission gas rate per unit length in element ij (mol/cm^3s)
Y	Fission yield of fission gas (Xe+Kr)
L	Fuel column length
F	Fuel rods per assembly
$\phi(r)$	Heat generation profile function in the radial direction
$r_{i-\frac{1}{2},j}$	Inner radius of region i,j (cm)
h	Linear heat rate
M_{Th}	Mass of thorium
A_{Th}	Mass number of Thorium-232
M_{TOTAL}	Total mass of fuel
A_{ThO2}	Mass number for thorium dioxide
N	Number of thorium fuel assemblies
$r_{i+\frac{1}{2},j}$	Outer radius of pellet region i,j (cm)
S	Oxide layer thickness
d	Pellet diameter
P	Power of the reactor
t	Refueling time
T	Temperature
V	Volume fuel for a single rod

1. Introduction

One of the main issues that concerns the world is the increasing need of energy. Some studies predict that by 2030 the energy consumption worldwide will be 35% to 49% higher than in 2010. This increase is due to the growing population and mainly to the energy consumption growth of developing countries, especially in Asia. Such an increase shows the necessity of an energy source that can reduce the dependence on fossil fuels, providing at the same time a viable economics and mitigating the global warming. The UN panel, composed of the world's leading Earth scientists (IPCC), warns that global greenhouse gas (GHG) emissions must, by 2050, be cut by 70% to avert the risk of catastrophic change in our planet's climate system [1].

The challenges of energy consumption growth and GHG make nuclear power a prominent major energy source for the next several decades according to the projections made by the International Atomic Energy Agency (IAEA). There are currently 438 nuclear power plants in operation around the world, producing 16% of the world's electricity which is the largest share provided by any non-greenhouse gas emitting source [2]. Figure 1 shows the necessity of clean energies. The term Fossil CCS means Fossil Carbon Capture and Storage.

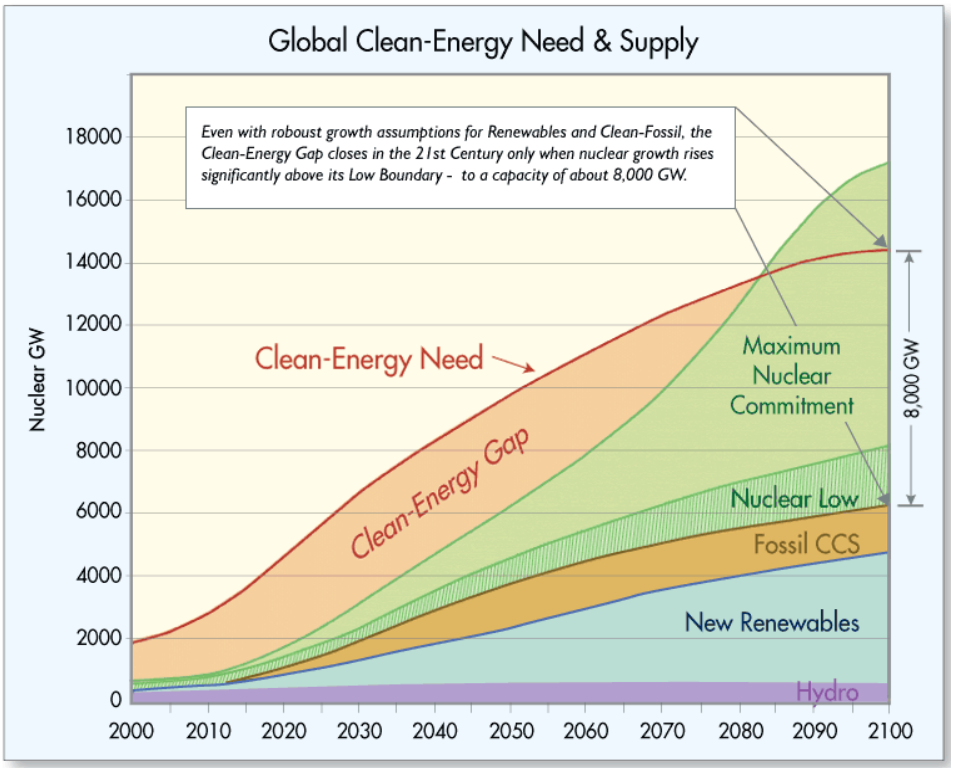


Fig. 1: Global Clean Energy Need and Supply [1]

The IAEA predicts two scenarios concerning the growth of nuclear power, a low and a high. The low projection assumes that all nuclear capacity which is currently under construction or firmly in the development pipeline, gets completed and attached to the grid, but no other capacity is added. It states that, there would be growth in capacity from 370 GWe at the end of 2006 to 447 GWe by 2030. This means a growth in the nuclear capacity of 20.8% [3].

In the IAEA's high projection, which adds additional reasonable and promising projects and plans, the global nuclear capacity is estimated to rise to 679 GWe by 2030. This translates to a growth in the nuclear capacity by 83.5%.

On the other hand the growth in the nuclear consumption also arises some questions about the viability to accomplish the expectations with the current fuel reserves, to respect the non-proliferation treaty, to reduce the radiotoxicity of the waste and to maintain and enhance the safe and reliable operation.

One of the solutions to face these issues is the use of thorium as a fuel. The first part of this Master’s Thesis offers the reader an outlook about thorium application in different countries, it shows the benefits and challenges for its use as a fuel and it describes the physical schemes available. The second part focuses on the critical problems that face the cladding as a result of the longer periods of residence in the reactor and the higher power of some parts of the fuel elements, it also discusses the properties of structural materials in the core and the cladding options that have been used or that will be used in the future. The third part shows the results obtained from the rod behavior with a simulating program using thorium dioxide as fuel. Finally the difficulties of the licensing process, regarding its technical limits and correct behavior against postulated accidents, are presented.

2. Background

The aim of this section is to present a general overview about thorium as a fuel regarding its use in the past and in particular to describe the conditions and drawbacks of the fuel element. It also states the possible core configurations and physical schemes.

2.1 Thorium. Historical outlook

The use of thorium fuel cycle has been studied more than 40 years. Many incentives have been identified for its use, including the fact that public concerns have increasingly focused on the high radiotoxicity of the long-lived waste of spent nuclear fuel. Also the large stockpiles of plutonium produced in civil and military reactors raised questions about weapons proliferation.

Even though thorium was considered, since the beginning of the nuclear power development, to be the nuclear fuel to follow uranium, the use of thorium-based fuel cycles has been studied on a much smaller scale as compared to uranium or uranium/plutonium cycles. The technology to use thorium in nuclear reactors was thought to be similar to that of uranium.

In the 1960s and 1970s, the development of thorium fuel for nuclear energy was of great interest worldwide. A large amount of research was carried out with the result of many interesting developments, including prototype High Temperature Reactors, Light Water Reactors and Molten Salt Reactors.

Basic research and development on thorium fuel cycles have been conducted in Germany, India, Japan, Russia, United Kingdom and the USA. A basic review of the most important programs held in these countries is done. Those studies include the determination of materials data, fabrication tests on laboratory scale and irradiation of thorium-based fuels in material test reactors with post-irradiaton examinations. Investigations on the use of thorium-based fuel for LWRs, LMFBRs (Liquid Metal Fast Breeder Reactor) and HTRs (High Temperature gas-cooled Reactor) are also included.

In the last few years peculiarities, problems and aspects of thorium fuel cycle have been discussed in the international nuclear community much more actively than it was 15-20 years ago. Experts have realized that application of thorium-based fuel cycle at least in the nearest future will be most probably evolutional, which does not demand any radical change in the existing fuel cycle. That means that it will be possible to implement the thorium fuel in the actual PWR, BWR and CANDU reactors with a minimum modification of the components, facilitating the process of licensing and testing.

Several reactors types other than LWRs have been tested with the use of thorium, like the gas cooled graphite moderated reactor or the pebble bed reactor, but this outlook focuses on the first type. Some of the countries which have been playing a more active role in the development of thorium fuel cycles are described below:

USA

Since its discovery, thorium was considered as a fuel for nuclear power reactors in the United States, but the initial interest decreased when it was found out that it was difficult to separate the fissile ^{233}U component compared with ^{235}U or ^{239}Pu. During the initial research of thorium as a nuclear fuel, the USA made the decision to merge their military and civil programs with the following implications [4]:

- The fuel infrastructure would be the same as their weapons material infrastructure which used $^{238}U/^{235}U$ fuel cycle instead of $^{232}Th/^{233}U$.
- Other countries followed the example of the United States and developed similar systems for their nuclear programs.
- Uranium resources were thought to be abundant enough to support the infrastructure, thorium was dismissed and uranium fuel cycle was adopted worldwide.

Despite this fact in the 1960s and 1970s whole core demonstrations of thorium-uranium oxide fuels in LWRs were explored in two types of arrangements [5]:

- Mixture of thorium oxide with highly enriched uranium oxide in a uniform lattice in the Borax-IV, Indian Point I PWR and Elk River BWR.
- Heterogeneous arrangement of seed and blanket regions tested in the Shippingport reactor.

In the mid 1970s EPRI (Electrical Power Research Institute) commissioned a study of the improvements in the nuclear fuel cycle where thorium was included with minimum modifications in modern LWRs [6,7]. Table 1 shows the characteristics of the main LWR that experimented with thorium. Some of the conclusions of the study are presented:

- Thorium with recycle can increase energy obtained per ton of uranium by about 85% beyond the once-through uranium cycle, and by 22% beyond plutonium recycle.
- Comparison of the characteristics of uranium and thorium based cores indicates that thorium fueling is feasible and major modifications to a PWR do not appear to be required.
- Even with the above, the thorium fuel cycle may not be economically attractive.
- The introduction of a totally new system of advanced converters into the USA would require more effort and funding than can be justified.

	Elk River	Indian Point I	Shippingport (LWBR)
Reactor Type	BWR	PWR	PWR
Reactor Power	28 MWe	270 MWe	70 MWe
Operation Dates	1962-1968	1962-1965	1977-1982
Fuel Assembly	Square 5×5	Square 14×14	Hexagonal
Assembly Radial Configuration	Homogeneous	Homogeneous	Seed and Blanket regions
Fuel Composition	Mixed ThO_2-UO_2	Mixed ThO_2-UO_2 and ThO_2	Mixed ThO_2-UO_2 and ThO_2
U Enrichment	92% ^{235}U	93% ^{235}U	98% ^{233}U
Cladding Material Thickness (mm) Max/Ave Temp (ºC)	Stainless 304 0.51 318/299	Stainless 304 0.51 293/-	Zircaloy-4 0.56 Seed 0.71 Blanket 1.06 Reflector -
Fuel Length (m) Pellet Diameter (mm)	1.52 10.35	1.9 6.6	2.66 7.8 Seed 14.5 Blanket 21.1 Reflector
Burnup (MWd/kg) Max Ave.	 8.5 -	 32 14.8	 60 Seed 30 Blanket For pin average divided by 2
Peak Linear Power (kW/ft) Heat Flux Peak (kW/m^2)	- 987	- 1700	22 Seed 29 Blanket -

Table 1: Main LWR experiments in the USA [7]

From all these reactors, the most well-know is Shippingport Atomic Power Station in Pennsylvania. It was the first commercial-scale nuclear power station in the world to operate and the fuel achieved a maximum burnup of 60 MWd/kgTh without fuel failure.

Russia (Soviet Union)

Russia has been one of the countries where the development and research of thorium fuel cycles has been most intense. Despite this, fuel cycle with a full scale use of thorium is still only a far perspective. Uranium resources are sufficient for decades in Russia and involvement of fast neutron reactors into power systems makes the problem of raw resources less actual. In contrast to other countries which abandon its thorium programs, Russia research work on uranium-plutonium cycle have closely been

followed by those on thorium-based one, although both research scales are of course not comparable. The work on thorium cycle was conducted for both studying aspects of development of nuclear power and the ways of involving thorium into it as an additional resource, and studying the beneficial qualities which can be introduced by the use of thorium in operating reactors [8].

Since the early 1990s Russia has had a program carried on at Kurchatov Institute to develop a thorium-uranium fuel. The Russian program involves the U.S. Company Thorium Power, Inc. (founded by Radkowsky) which is working in the fuel design for the conventional Russian VVER-1000 reactors (Vodo-Vodyanoi Energetichesky Reactor, which in english would be translated as Water-Water Energetic Reactor). Unlike the usual nuclear fuel, which uses enriched uranium oxide, the new fuel assembly design has two parts, the *seed* which provides the fissile material for the reaction and the *blanket* which provides the fertile material [9]. This application of thorium fuel is undergoing the most investigation nowadays and will be discussed further in the next sections of this document.

The VVER-1000 design, cf. Fig. 2, was developed between 1975 and 1985 based on the requirements of a new Soviet nuclear standard that incorporated some international practices, particularly in the area of plant safety.

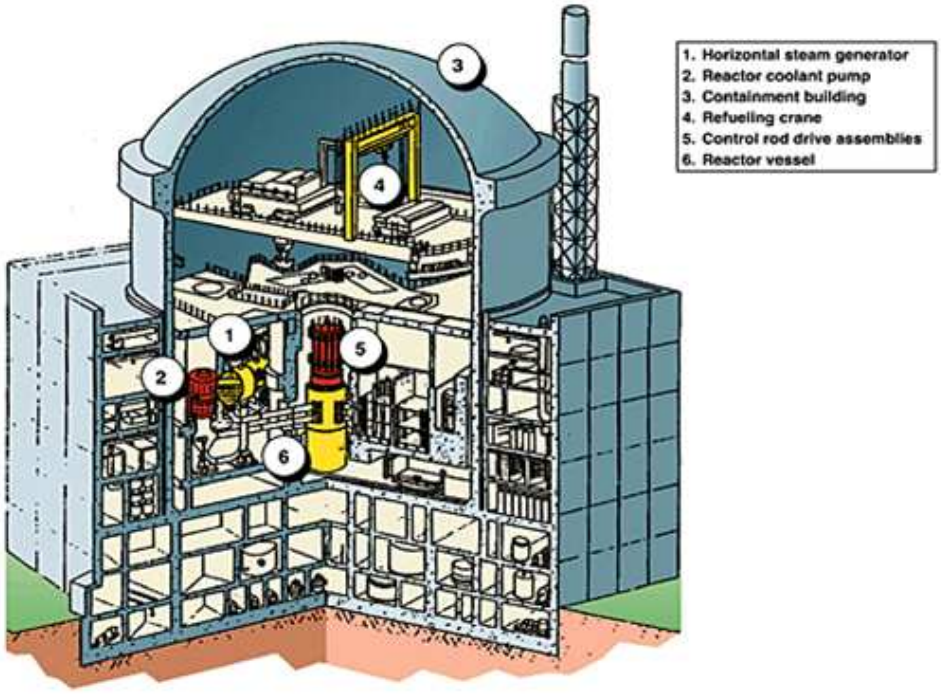


Fig. 2: VVER-1000 reactor [9]

India

India is the most committed country to the study and use of thorium fuel; by far most of the work in neutron physics on thorium has been done by Indian nuclear scientists [9]. This interest is not only motivated for the characteristics of thorium as a fuel element but also for the fact that India is outside the Nuclear Non-Proliferation Treaty due to its weapons program, which resulted in his exclusion from the trade in nuclear materials for more than 34 years. Due to these trade bans and lack of indigenous uranium, India has uniquely been developing a nuclear cycle to exploit its reserves of thorium [10].
The use of thorium as a nuclear fuel is also a result of the objective of India to be independent and self-sufficient. India has one of the largest resources of thorium in the beach sands of Southern India (its reserves of monazite are about 24% of worldwide reserves) and consequently any long term planning of the growth of nuclear power programme has to be based in thorium as a fuel [11].

During the 1950s the Indian physicist Homi Jehangir Bhabha set out a three stage development program for Indian nuclear technology with the generation of nuclear power through utilization of thorium as its ultimate goal. The first phase of the programme consists of Pressurized Heavy Water Reactors (PHWR) using natural uranium as a fuel. The second phase is based on the utilization of plutonium generated as a product in the first phase, in Fast Breeder Reactors (FBR) for fissile generation and to increase the fissile inventory of ^{239}Pu and ^{233}U. Finally the third phase is based on thorium fuelled thermal reactors. Several studies have been carried out at this stage on thorium fuel cycles in Heavy Water Reactors (HWR) [12].

Plans for building the first PHWR were finalized in 1964 and this prototype, Rajasthan 1, was built with Canada's Douglas Point reactor as a reference unit. It started up in 1972 and was duplicated. Subsequent PHWR development has been based on these units [10]. Table 2 lists the operating reactors in India.

Reactor	Type	MWe net each	Commercial operation
Tarapur 1 & 2	BWR	150	1969
Kaiga 1 & 2	PHWR	202	1999-2000
Kaiga 3	PHWR	202	2007
Kakrapar 1 & 2	PHWR	202	1993-1995
Kalpakkam 1 & 2	PHWR	202	1984-1986
Narora 1 & 2	PHWR	202	1991-1992
Rajasthan 1	PHWR	90	1973
Rajasthan 2	PHWR	187	1981
Rajasthan 3 & 4	PHWR	202	1999-2000
Rajasthan 5 & 6	PHWR	202	2010
Tarapur 1 & 2	PHWR	490	2005-2006
Total (19)		4183	

Table 2: India's operating nuclear power reactors [10]

2.2 Benefits and challenges of Thorium

Thorium fuels and fuel cycles have the following benefits and challenges:

2.2.1 Benefits

- Thorium is 3 to 4 times more abundant than uranium, widely distributed in nature as an easily exploitable resource in many countries and has not been exploited commercially so far. Thorium fuels, therefore, complement uranium fuels and ensure long term sustainability of nuclear power.

- Thorium fuel cycle is an attractive way to produce long term nuclear energy with low radiotoxicity waste. In addition, the transition to thorium cycle could be done through the incineration of weapons grade plutonium (WPu) or civilian plutonium.

- The absorption cross-section for thermal neutrons of ^{232}Th (7.4 barns) is nearly three times higher than that of ^{238}U (2.7 barns). Hence, a higher conversion (to ^{233}U) is possible with ^{232}Th than with ^{238}U (to ^{239}Pu), cf. Fig. 3. Thus, thorium is a better 'fertile' material than ^{238}U in thermal reactors but it is inferior to depleted uranium as a 'fertile' material in fast reactor. ^{233}U has a higher tendency to generate neutrons by fission in thermal and epithermal neutron fluxes.

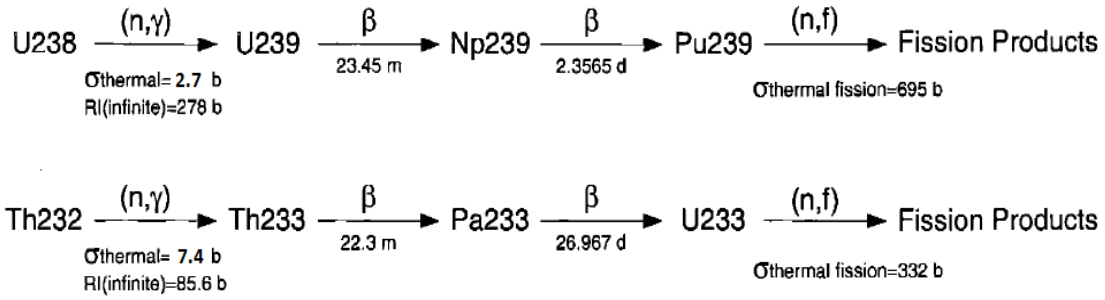


Fig. 3: Transmutation of ^{238}U into ^{239}Pu and ^{232}Th into ^{233}U [5]

- During long term irradiation the in-core fissile generation of ^{232}Th can be higher than that of ^{238}U. This will reduce the need for fuel ore and/or fuel enrichment per unit energy generation. Thus the fuel cost and the amount of spent fuel per unit energy generation can be reduced.

  Also higher burnups can be achieved with Th resulting in an improvement of the fuel cycle management, reducing the amount of waste produced and reducing the fuel cost. Although there are benefits for extended burnup, there are some limitations which become a major constraint to continuing the fuel cycle cost improvement. Figure 4 shows this fact and the relation between high burnups and the fuel cost reduction.

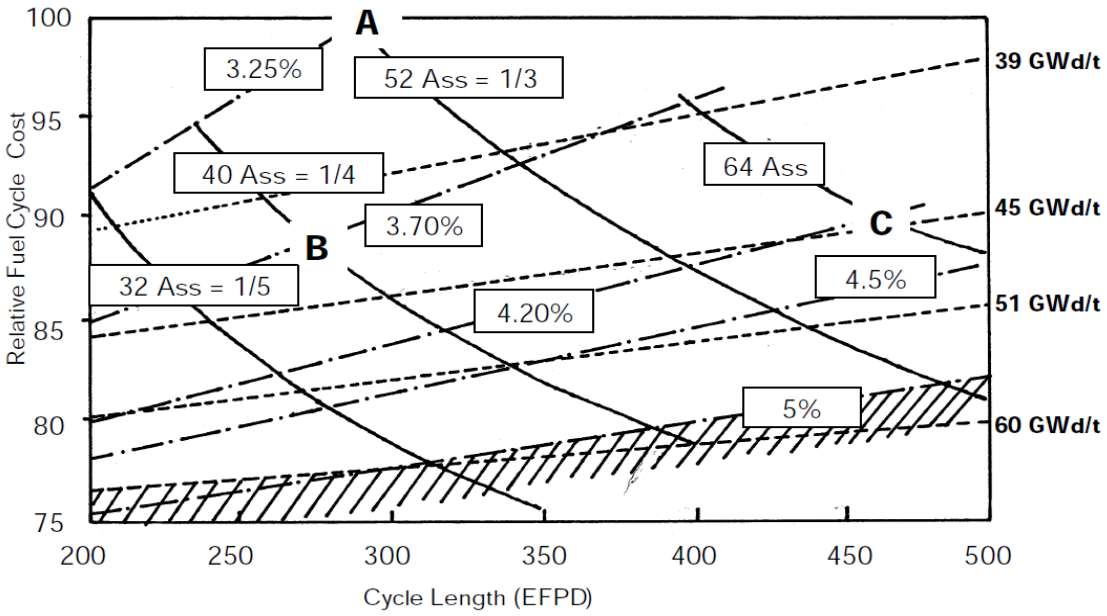


Fig. 4: Relative fuel cycle cost [1] [13]

Although in the results plotted in Fig. 4 uranium was used, they are valid also for the thorium fuel cycle. As can be seen, the tests were conducted for a different number of assemblies (Ass) and for different enrichments. Fig. 4 shows that the existence of enrichment limits is a major constraint to continuing the fuel cycle cost improvement. The letters A, B, C illustrates the entire range which has been covered for the utilities. The letter A refers to an average case of 52 assemblies enriched at 3% ^{235}U for 12 months cycle while the letters B and C show the fuel management used in 2000 in most of the European PWRs. At that time the enrichment was situated between 3.70% and 4.2%, for 40 and 64 assemblies respectively, with cycle lengths between 12 and 18 months [13]. It can be concluded that a higher enrichment in an increasing number of fuel rods can increase the burnup and consequently the cycle length. Despite this fact there are some limitations in the assemblies that can be enriched as a result of the limitations in the rods residence time in the reactor.

- For the fissile ^{233}U nuclei, the number of neutrons liberated per neutron absorbed (represented as η) is greater than 2.0 over a wide range of thermal neutron spectrum, unlike ^{235}U and ^{239}Pu, cf. Fig. 5. Thus, contrary to the ^{238}U–^{239}Pu cycle in which breeding can be obtained only with fast neutron spectra, the ^{232}Th–^{233}U fuel cycle can operate with fast, epithermal or thermal spectra. Another advantage of ^{233}U is the low value of its epithermal resonance capture fission ratio, cf. Table 3.

[1] The term EFPD refers to equivalent full power days.

Parameter		U-233	U-235	Pu-239	Pu-241
Thermal (barns)	σa	364	405	1045	1121
	σf	332	346	695	842
α=σc/σf		0.096	0.171	0.504	0.331
ηth		2.26	2.08	1.91	2.23
Epithermal Resonance Integral (barns)	RIa	882	405	474	740
	RIf	746	272	293	571
α=RIc/RIf		0.182	0.489	0.618	0.296
ηepi		2.10	1.63	1.77	2.29
Neutron Yield υ		2.48	2.43	2.87	2.97
Delayed Neutron Yield β		0.0031	0.0069	0.0026	0.0050

Table 3: Fissile neutronic properties [5]

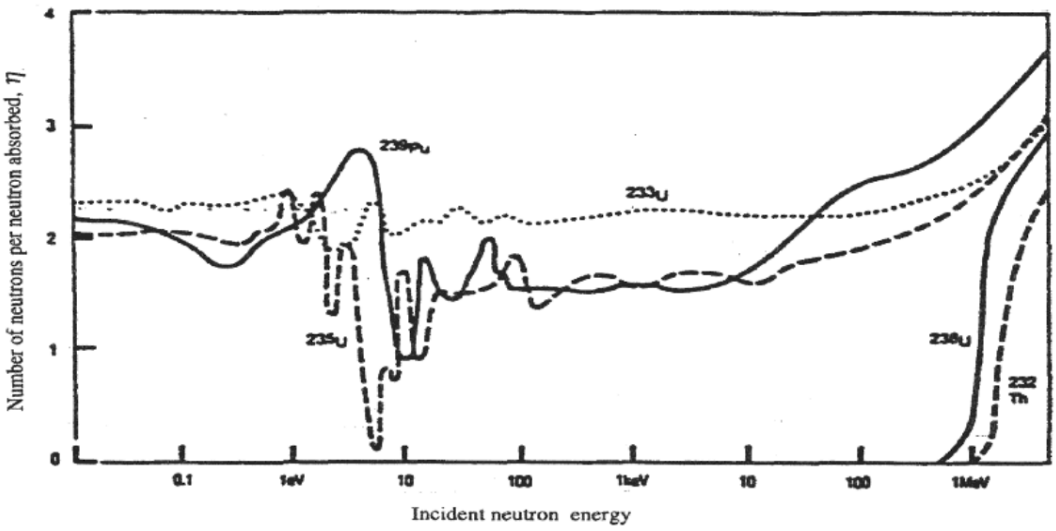


Fig. 5: Fission yield of various fuel isotopes [14]

- Thorium dioxide is chemically more stable and has higher radiation resistance than uranium dioxide. The production of fission products for ThO_2–based fuels is one order of magnitude lower than that of UO_2. ThO_2 has favorable thermophysical properties because of its higher thermal conductivity and lower coefficient of thermal expansion compared to UO_2. Thus, ThO_2–based fuels are expected to have better performance than that of UO_2 and UO_2–based mixed oxide.

- ThO_2 is relatively inert and does not oxidize unlike UO_2, which oxidizes easily to U_3O_8 and UO_3. Hence, long term interim storage and permanent disposal in repository of spent ThO_2–based fuel are simpler without the problem of oxidation.

- Th–based fuels and fuel cycles have intrinsic proliferation-resistance due to the formation of ^{232}U via (n,2n) reactions with ^{232}Th, ^{233}Pa and ^{233}U. The half-life of ^{232}U is only 73.6 years and the daughter products have very short half-life and some like ^{212}Bi and ^{208}Tl emit strong gamma radiations. The hard gamma rays from ^{208}Tl

cause ionization of materials destroying the explosives and electronics of a nuclear weapon, and heavy lead shielding is required to protect personnel assembling the warhead. From the same consideration, ^{232}U could be utilized as an attractive carrier of highly enriched uranium (HEU) and WPu to avoid their proliferation for non-peaceful purpose.

The Th-U fuel cycle has also the advantage to produce less plutonium than the conventional fuel cycles. Table 4 presents the production of plutonium isotopes for a conventional cycle and for a Th-U based cycle for a different burnups.

Plutonium Production in U and Mixed Th-U Cycles				
235-U 8% enrichment				
	U 4.5 yr	U 6 yr	Th-U 6 yr	Th-U 10 yr
Burnup (MWd/kg)	45	72	72	100
Production (gram/kg ihm[2])				
Pu-238	0.276	0.712	0.461	0.871
Pu-239	6.632	8.798	1.657	2.274
Pu-240	2.520	3.162	0.842	1.214
Pu-241	1.770	2.485	0.633	0.872
Pu-242	0.692	0.943	0.662	1.016
Total Pu	11.890	16.101	4.255	6.247
Production (MWd)				
grams Pu/MWd	0.264	0.224	0.059	0.062
relative	4.47	3.78	1.00	1.06
grams Pu-239/MWd	0.147	0.122	0.023	0.023
relative	6.48	5.37	1.01	1.00

Table 4: Plutonium production for conventional and Th-based cycles [15]

The plutonium produced in Th-U fuel is high in ^{238}Pu (about 14%). It is a strong source of neutrons and of decay heat, with a decay heat 5 times greater than that of plutonium derived from conventional fuel and 40 times greater than weapon grade plutonium. The decay heat would melt the internal components of a crude nuclear weapon, while the spontaneous neutrons would cause the artifact to predetonate. These two characteristics make the plutonium produced by the thorium-uranium fuel very undesirable for its use in nuclear weapons [15].

- For incineration of WPu or civilian Pu in 'once-through' cycle, (Th, Pu)O_2 fuel is more attractive, as compared to (U, Pu)O_2, since plutonium is not bred and the ^{232}U formed after the 'once-through' cycle in the spent fuel ensures proliferation resistance.

[2] The term ihm refers to "initial heavy metal".

- The fission product absorption, which is the main contributor to reactor poisoning, is about 25% less for ^{233}U than that for ^{235}U or ^{239}Pu [5].

- The thermal conductivity of ThO_2 is about 10% higher than that of UO_2 over a large temperature range, and its melting point is about 500 ºC higher than that of UO_2. As a consequence, fuel operating temperatures will be lower than those of UO_2 and all thermally activated processes such as diffusion of fission gas from the fuel will be decreased.

- In ^{232}Th–^{233}U fuel cycle, much less amount of plutonium and long-lived minor actinides (Np, Am, Cm) are formed as compared to the ^{238}U–^{239}Pu fuel cycle, as can be seen from Table 5, thereby minimizing the radiotoxicity associated in spent fuel. It is claimed that the radiotoxicity of the transuranics elements in the spent fuel of conventional LWR's is about 100 times higher than that in the spent fuel of the thorium fuel cycle [16]. However, in the back end of ^{232}Th–^{233}U fuel cycle, there are other radionuclides such as ^{231}Pa, ^{229}Th and ^{230}U, which may have long term radiological impact.

Different Fuel Composition				
Minor Actinides	^{235}U + ^{238}U	^{235}U + ^{232}Th	^{233}U + ^{238}U	^{233}U + ^{232}Th
^{237}Np	9.0E+02	9.6E+02	1.2E+02	2.7E+01
Am	4.7E+02	1.3E+00	5.5E+02	8.5E-03
Cm	2.2E+02	3.0E-01	2.9E+02	1.4E-03

T*able 5: Production of Minor Actinides in Uranium and Thorium cycles in g/t of heavy metal at 60 GWd/t [17]*

2.2.2 Challenges

- The melting point of ThO_2 (3350 ºC) is much higher compared to that of UO_2 (2800 ºC). Hence, a much higher sintering temperature (>2000 ºC) is required to produce high density ThO_2 and ThO_2–based mixed oxide fuels. The mixing of 'sintering aid' (CaO, MgO, Nb_2O_5, etc) is required for achieving the desired pellet density at a lower temperature.

- ThO_2 and ThO_2–based mixed oxide fuels are very inert and, unlike UO_2 and (U, Pu)O_2 fuels, do not dissolve easily in concentrated nitric acid. Addition of small quantities of HF in concentrated HNO_3 is essential but causes corrosion of stainless steel equipment and pipings in reprocessing plants. The corrosion problem is mitigated with addition of aluminum nitrate.

- The irradiated Th or Th–based fuels contain significant amount of ^{232}U, which has a half-life of 73.6 years and is associated with strong gamma emitting daughter products, ^{212}Bi and ^{208}Tl with very short half-life. As a result, there is significant buildup of radiation dose with storage of spent Th–based fuel or separated ^{233}U,

needing remote and automated reprocessing and refabrication in heavily shielded hot cells and increase in the cost of fuel cycle activities. The dose rate on recovered ^{233}U increases rapidly with time, as shown in Table 6, so the fuel operation should be taken up as soon as possible after the recovery of ^{233}U [16].

Aging Time (day)	^{232}U content in ^{233}U	
	100 ppm	1000 ppm
10	0.8	8
100	14.0	140
1000	100.0	900

Table 6: Dose rate (mGy/h) from 1 kg of ^{233}U at a distance of 30 cm [17]

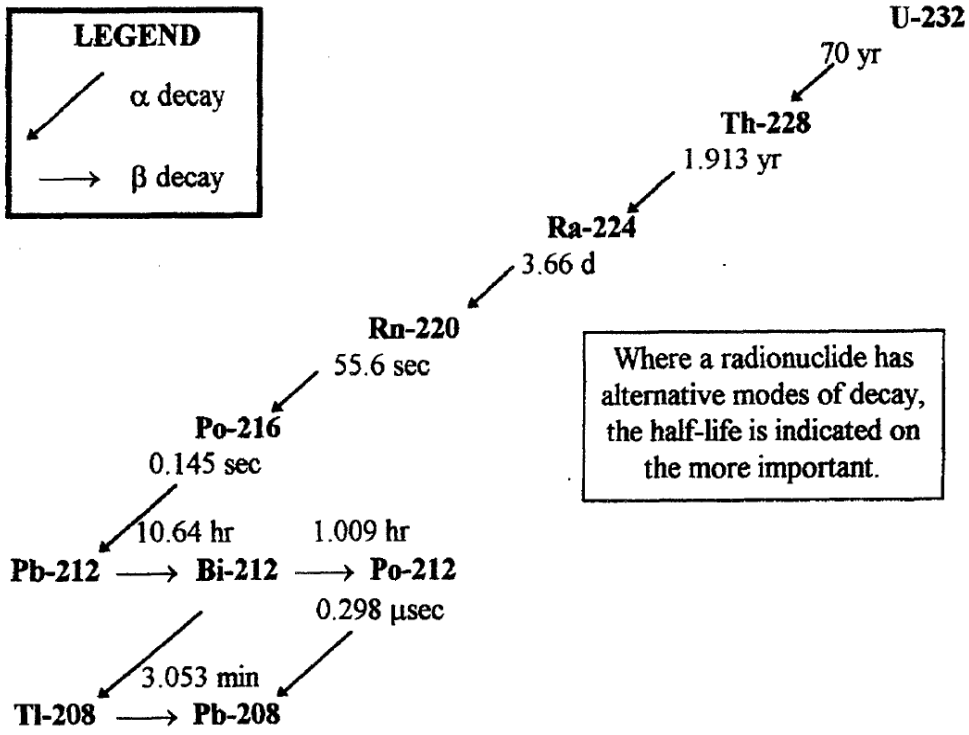


Fig. 6: Uranium-232 decay series [18]

- In the conversion chain of ^{232}Th to ^{233}U, ^{233}Pa is formed as an intermediate product. It has a relatively long half-life (~27 days) compared to ^{239}Np (2.35 days) in the uranium fuel cycle thereby requiring longer cooling time, at least one year for completing the decay of ^{233}Pa to ^{233}U. In the same neutron flux, the concentration of ^{233}Pa would be about 10 times that of ^{239}Np, resulting in greater losses, but this effect is limited. However the larger quantity of ^{233}Pa supposes a problem after the reactor shutdown, due to the increase of the reactivity for the accumulation of ^{233}U from the decay of ^{233}Pa [5].

- From a neutronic point of view, the epithermal resonance absorption in ^{232}Th is lower than that in ^{238}U, cf. Table 7. This may reduce the negative Doppler reactivity feedback in overpower transients. Furthermore, ^{233}U has a smaller delayed neutron fraction, β, than that of ^{235}U but comparable to that of ^{239}Pu, thus creating a need for faster response of control systems to transients.

Parameter	^{232}Th	^{238}U	^{234}U	^{240}Pu
Thermal σ_a	4.62	1.73	63	203
Epithermal	85.6	278	660	8500
Shielded	17	24		

Table 7: Fertile Neutronic Properties [5]

- The energy yield per ^{233}U fission is somewhat less than that of ^{235}U and ^{239}Pu. Hence, more fissions are needed per unit energy production. Also, more fission gases are produced per ^{233}U fission.

- There is a lack of ^{233}U in nature. It is necessary to initiate the cycle by using fissile material such as ^{235}U or ^{239}Pu, which can be mixed with the thorium fuel or spatially separated as a driver.

- The process of separation of uranium, plutonium and thorium from spent (Th, Pu)O_2 fuel, though viable, is yet to be developed.

- The database and experience of thorium fuels and thorium fuel cycles are very limited, compared to UO_2 and (U, Pu)O_2 fuels, and need to be augmented before large investments are made for commercial utilization of thorium fuels and fuel cycles.

2.3 Physical schemes of thorium in the reactor

The option of introducing thorium as a fuel in the actual configuration of reactors has been studied for two physical schemes; assembly of the active zone using homogeneous fuel rods and using heterogeneous fuel rods. The scope of this section is to focus in the heterogeneous assembly which, as some studies suggest, offers better properties regarding the use of the fuel and the non-proliferation resistance.

2.3.1 Homogeneous fuel rod

The homogeneous version consists of a mixture of thorium and enriched-uranium in the same bundle. Studies have shown that the amount of plutonium produced in this case, compared with a conventional reactor using uranium, is roughly halved [19]. The thermal performance is likely to be similar to present day reactors. The main difference would be the high burnup and high number of fissions for the same operation power.

This configuration was tested with a conventional geometry of fuel rods and fuel elements by the Kurchatov Institute but it failed in providing an economic advantage over VVER-1000 reactors [20]. For this reason some studies focused on the performance and economics of using micro-heterogeneous fuel rods, where some small distance physically separates the uranium and thorium. When this approach is compared to the equivalent homogeneous case, an increase in burnup is observed, which improves the economics of using thorium-based fuels. However these economic benefits are not enough to compensate for the costs of the increased Separative Work Units (SWUs) required for thorium oxide fuels. These micro-heterogeneous fuel rods have been used to burn weapons grade plutonium [21].

The mixed-fuel bundles have been the approach mainly taken in the CANDU reactors. The reason for this approach is the sophisticated fuel-management schemes required to shape the channels and bundle power distribution in mixed core. This is due to the difficulty to handle the disparity in reactivity and power output between driver channels and thorium channels [22].

2.3.2 Heterogeneous fuel rod

In the early 1990's A.Radkowsky proposed the concept of a light-water uranium-thorium reactor, taking account of present day requirements in nuclear power production. Radkowsky worked for a long time in the USA where he took an active part on military programs (he was chief scientist of the Bureau of Ships nuclear propulsion division) and in a program of thorium experiments on the Shippingport reactor in the USA [20].

Using as prototypes the well-known concept of a reactor with a heterogeneous active zone arrangement of the German-Brazilian thorium reactor project he proposed the idea of a reactor based on the following principles [20]:

- Neither the fuel loaded nor unloaded from the reactor can be used for the production of nuclear weapons.
- Thorium can be used economically as a fuel.
- The spent fuel contains considerably less high active toxic waste, including plutonium, compared to operating reactors.
- The reactor can be loaded with weapons uranium and plutonium.

In 1994 a program to study the seed-blanket concept was revived with the introduction of the Radkowsky Thorium Reactor (RTR) concept with a once-trough fuel cycle using thorium. This was a result of the collaboration between the Radkowsky Thorium Power Corporation and the Kurchatov Institute in Moscow with technical support from Brookhaven National Laboratory. The collaboration was initially focused on the suitability of the RTR concept for application as a whole core in a Russian VVER light water reactor.

The main idea of this concept is the utilization of a seed-blanket unit (SBU) fuel assembly geometry having separated uranium and thorium fuel zones. The central part of the assembly (seed) with the fuel using a uranium-zirconium alloy contains about 20% enriched U (a percentage generally accepted as being non-proliferative) while

peripheral region (blanket) with fuel based on uranium and thorium dioxides contains a small amount of 10 to 20% enriched uranium. Having ^{238}U in the blanket prevents anyone from withdrawing these rods using only chemical means to separate out the fissionable ^{233}U that is created over time. The high enrichment is necessary to compensate for the smaller volume of uranium present in the core and to compensate for the high thorium capture rate [5]. Calculations indicate that there are additional advantages to the use of highly enriched uranium [20]:

- There is a reduction in plutonium production by about a factor of 20 relative to conventional reactors and by a factor of 5 relative to a thorium reactor operating with uranium of 20% ^{235}U enrichment.
- The increase of enrichment implies a reduction in the toxicity and radioactivity of transuranic elements in the spent fuel. The production of transplutonic elements is reduced by a factor of 20 compared with a thorium reactor operating with uranium of 20% ^{235}U enrichment.

This arrangement provides the necessary flexibility for designing the seed as an efficient supplier of well thermalized neutrons to a subcritical blanket which, in turn, is designed for efficient generation and in-situ burning of ^{233}U. The spatial separation of the seed and blanket sub-assemblies permits local optimization of the lattice: an over-moderated seed region and an under-moderated blanket region.

The design calls for the use of the seed part in the reactor for three years but leaving the blanket fuel rods in the core for about ten years. The seed-blanket assembly is shown in Fig. 7.

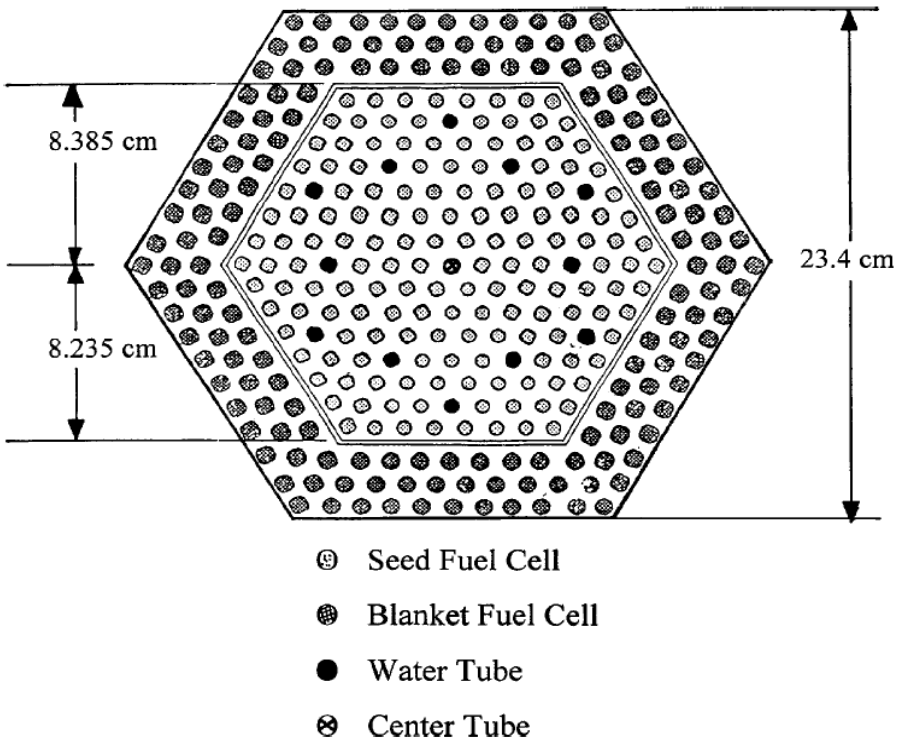


Fig. 7: Radkowsky Thorium Reactor SBU Fuel Assembly Geometry [23]

This concept has some advantages but also presents some problems or difficulties that will need further demonstration and research [7]:

- The radicalism of exposing the blanket fuel and cladding to 10 years of operation, more than twice that of the fuel elements of the conventional water-cooled reactors, and to a high burnup.
- Requirements to develop a seed-zone fuel element possessing a high power density and permitting deep depletion of the fuel. With the use of a metallic seed, it could have an average power of about 140% of the PWR and reach a high burnup of about 150 MWd/kg.
- The segmentation of the system of movable seed parts of the fuel rods over the height of the active zone would introduce considerably perturbations in the spatial distribution energy release, and this would lower the power density of the reactor. Furthermore, the need to displace them as means of compensation should be object of a deep study from the point of view of reliability and safety.

Despite these difficulties both parts of the fuel rod jointly satisfy the reactor criticality conditions during the operating life with a relatively weak change in reactivity.

The concept of a heterogeneous fuel assembly was applied to the design of a PWRT compatible with a current Western technology and to a Russian PWR design, using hexagonal assemblies, designated as VVER. The possible schemes in the core, the seed-blanket unit and the whole assembly seed and blanket configuration, are shown in Fig. 8.

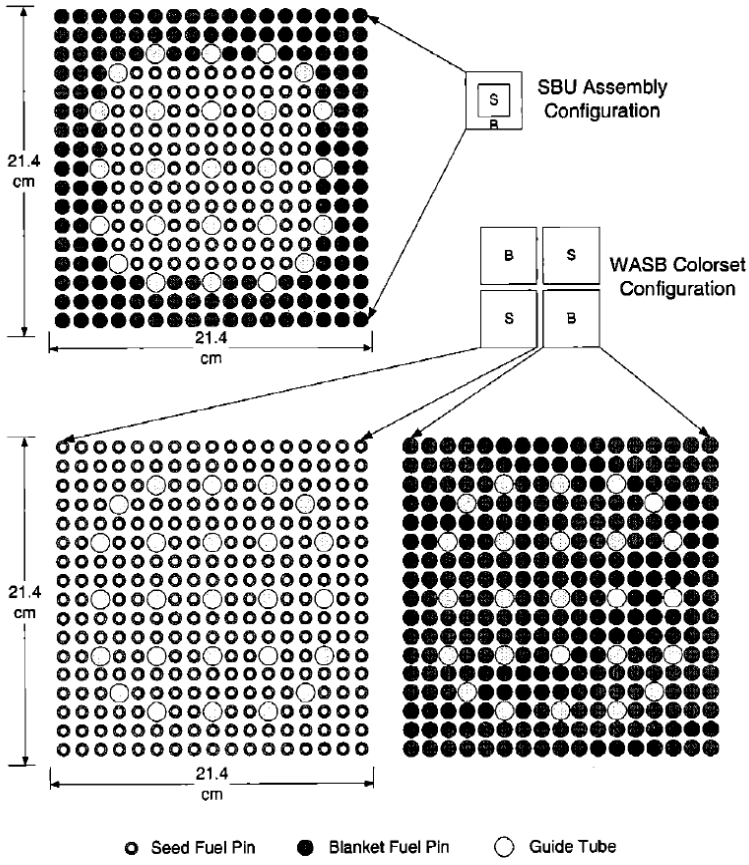


Fig. 8: SBU and WASB Assembly Configurations [5]

Thorium-based nuclear fuels can be designed in different ways. The first configuration proposed by Radkowsky was to have each nuclear assembly (squares) composed of a seed and blanket rods (SBU). This mixing of fuel types within an assembly complicates the refueling of a nuclear reactor, because the seed rods need to be replaced much more frequently than the blanket rods. The second configuration called the whole-assembly seed and blanket core (WASB), utilizes fuel assemblies that each contain only uranium-rich seed rods or thorium-rich rods, making easy the shuffled and replacement at prescribed intervals. Figure 9 shows the two types of configurations and the different burnups of the fuel elements.

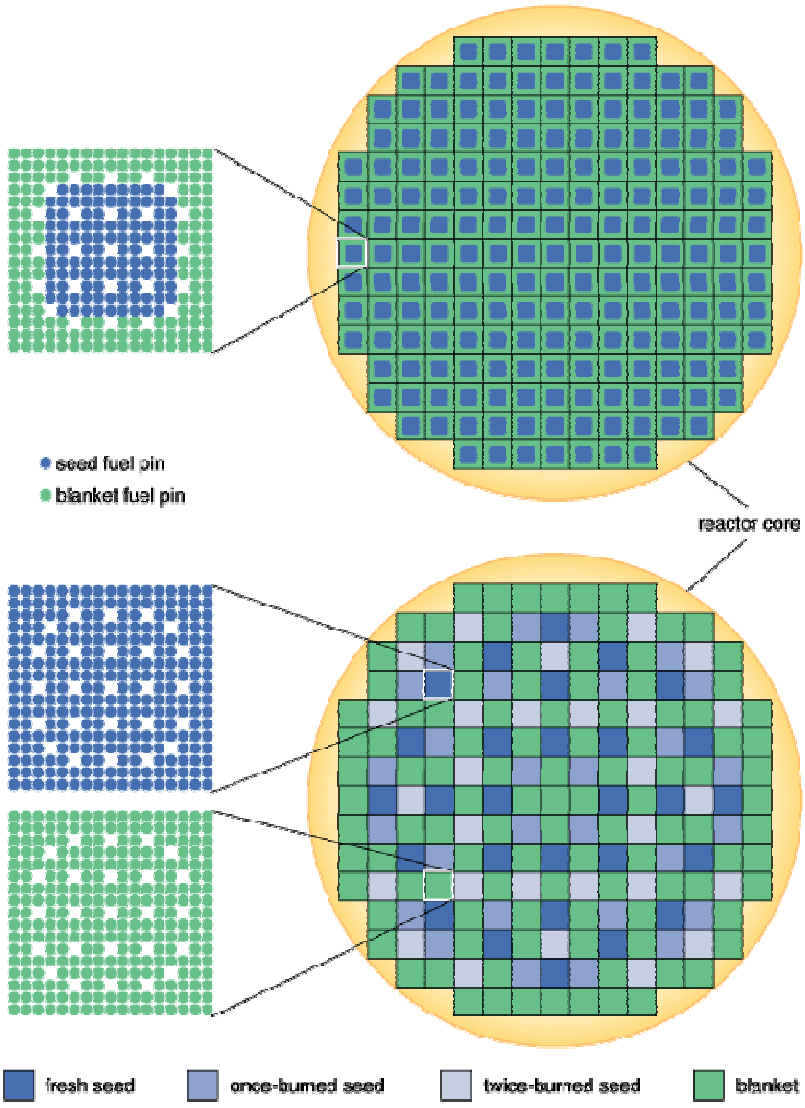


Fig. 9: Designs of the Thorium-based nuclear fuels [16]

The WASB configuration offers better characteristics relating the spent fuel discharged. One third of seed assemblies are refueled every cycle, and all the blanket assemblies are refueled every nine cycles. The discharged waste per unit energy is summarized in Table 8. The discharged assemblies per GWe-yr of WASB are nearly 40% less than those of a typical 18 month cycle PWR, and the discharged waste mass is about half of that of a typical PWR.

	WASB			Typical PWR
	Seed	Blanket	S+B	
Assemblies/Gwe-yr	16.2	7.0	23.2	37.3
MTHM/Gwe-yr	4.5	3.8	8.3	17.4

Table 8: Spent fuel discharged rate [5]

Regarding the neutronic performance, in principle, the seed-blanket arrangement does provide the designer with two added degrees of freedom: the ability to enhance the desired neutronic performance and to change the burnup limits of the seed and the blanket independently.

The SBU initially was designed to use U/Zr as the seed fuel, which could also be used for the WASB concept. However the thermal performance of UO_2 annular seed fuel pins was found acceptable for the WASB design to accommodate a higher linear power. The use of UO_2 should simplify the licensing process. In addition the neutronic and economic performance is slightly better than metallic uranium because of parasitic neutron absorption [5].

Despite the benefits of a uranium oxide seed, the use of a mixture of uranium and zirconium also offers good characteristics. The use U/Zr provides the seed with better thermal characteristics (high thermal conductivity) which makes the fuel-operating temperatures lower than that of UO_2 and all thermal activated process, such as diffusion of fission gas from the fuel, is decreased. The use of metallic fuel in the high energy density region also reduces the thermal energy stored in the fuel and enhances its safety response.

The heterogeneous configuration also offers a good proliferation resistance. Radkowsky calculated that his scheme would reduce the amount of plutonium by 80% compared with a conventional fuel reactor, cf. Table 9.

	PWR	Th-Homogeneous	Th-Heterogeneous
Total Pu Discharged (kg/Gwe-year)	250	150	70-90
Spontaneous Fission Source $(\text{cri.mass-sec})^{-1}$	$1.6 \cdot 10^6$	$3.0 \cdot 10^6$	$4.0 \cdot 10^6$
Decay Heat Emission (watts/crit.mass)	90	200	350

Table 9: Parameters related to the proliferation resistance for the two main implementation scenarios [5]

3. Fuel rod analysis

In this section the effects which have a major impact in the cladding are presented along with the options available to influence these effects. These considerations constitute the body of this document. It shows the conditions that the cladding has to face due to the use of thorium as a fuel element and also the properties that the materials have to possess in order to be used in the reactor. It also presents the cladding options available and the studies which are being taken in order to improve the operating condition.

3.1 Critical issues of the cladding

One of the major issues that face the implementation of thorium fuel cycles in current reactors is the structural integrity and safety of the cladding. To stay competitive the industry needs to reduce maintenance and fuel costs, while enhancing safety features. This document aims to focus in a heterogeneous fuel rod configuration, which offers more competitive properties in power distribution, depletion of the uranium-thorium fuel and in non-proliferation than a homogeneous fuel rod. For such an assembly there are several thermal design issues that need attention like the high burnup conditions and the high number of fissions required to obtain the same operational power.

Before presenting the potential failure mechanisms it is necessary to discuss briefly the thermal-hydraulic and safety considerations of the Radkowsky Thorium Fuel. The seed and blanket configuration provides the designer with more freedom to optimize the use of the fuel and to obtain an improvement in the neutron balance and a reduction in the production of plutonium and actinides. Related with the fuel performance there are two aspects of the fuel behavior that need to be examined: the fission gas release from the fuel grains into the fuel pin plenum and the corrosion of the cladding. The high seed power density will also be difficult to combine with the expected safety margins in advanced reactor designs. These characteristics are different from the critical issues that face conventional reactor fuel and also arise some critical questions that need to be studied.

The seed part of the assembly contains the fissile material to produce the nuclear reaction. This element operates at much higher local power density, about 1700 W/cm^3, than conventional PWR fuel which translates to a maximum linear heat rate of about 55 kW/m. This value is 1.3 times higher than typical LWR fuel thus challenging the critical heat flux margins [7]. As a result of these conditions the cladding has to be designed to prevent excessive release of fission gas which would over-pressurize the fuel rods internally.

The blanket part of the assembly contains fertile material which provides fissile nuclei for the nuclear reaction. This element operates at lower linear power, about 70%, than typical PWR fuel and is expected to remain in core for up to 10 or 13 years depending on the burnup degree. To face the operating conditions the cladding has to be designed to prevent excessive corrosion on the outside surface during its long residence time. Fission gas release may also be a problem because of the high burnup of the blanket. However, corrosion is expected to be the primary problem.

The large difference between seed and blanket power densities (the seed power density is 30% higher and the blanket power density is only 70% of a standard PWR) also makes the appropriate cooling of both elements a critical issue to avoid large coolant temperature asymmetries between the seed and the blanket subchannels [7].

Despite the critical issues that can occur, in the early days of the nuclear industry there were some incentives for operating fuel to high burnups. Most of the incentives are still valid; however the value of each one is slowly changing with time. The incentives for high burnups are [26]:

- Economic benefits. Lower fuel cycles costs.
- Capability for longer cycles.
- Improved resource utilization.
- Increased margin to storage capacity limits.
- Eventual decreased offsite shipping and storage costs. However, the significantly increased time required for high burnup fuel to decrease its decay heat in a spent fuel pool, before it can be loaded into an intermediate dry storage cask, and the unknown schedule for shipping the fuel from the dry cask to a permanent storage site prevents a reliable estimate for the capacity and cost required for the intermediate wet and dry storage facilities.

One of the largest challenges in this new approach is to assure the fuel and the cladding integrity due to the highest burnups. The discharged seed burnup is about 145 MWd/kgHM, and the peak assembly average burnup is about 160 MWd/kgHM. The discharge burnup of the average blanket assemblies is about 88 MWd/kgHM, and the peak assembly burnup is about 100 MWd/kgHM. These values are significantly higher than current fuel burnups (40-60 MWd/kgHM). Factors limiting fuel burnup include: cladding corrosion, internal gas pressure and dimensional changes of both fuel assembly and fuel rods [5].

	Discharged burnup (MWd/kgHM)	Peak assembly average burnup (MWd/kgHM)
Seed	145	160
Blanket	88	100

Table 10: Burnup of seed and blanket [3]

The main potential failure mechanisms that the fuel rod presents at high burnup are discussed in the following subsections.

[3] It must be made clear whether the burnup is expressed in kilogram or ton of oxide or heavy metal tons. In France, the usual units are MWd/t metal for PWR and MWd/t oxide for FR, according to the ratios between oxide and metal [25]:

1MWd/t metal = 0,8815 MWd/t oxide

3.1.1 Fission gas release

High burnup leads to an increase of the fission products produced and more fission gas is released. The presence of a large amount of fission products in the oxide plays an important role in reactor behavior. The thermal, mechanical and physicochemical properties of the uranium oxide fuel vary in a continuous manner owing to the presence of these fission products and gas release which thus directly influence fuel temperatures, cladding corrosion, oxide-cladding interaction as well as reactions with the coolant in case of cladding failure [24]. These gaseous fission products introduce some problems in the fuel safety, being a significant challenge for the cladding. The most important effects of the fission gas behavior on the fuel are:

- Increase of rod pressure.
- Decrease of gap thermal conductivity.
- Changes in the fuel dimension via swelling.

The high burnup is the main reason for the internal pressure. Fission gas production as Xe and Kr which represents about 30% of the total gaseous products [25], increases as a function of burnup and temperature, but the released fraction depends on other factors. These gas elements tend to accumulate, being almost completely insoluble in the oxide lattice and forming bubbles, either intra or intergranular. This problem along with the irradiation growth and irradiation creep contributes to dimensional changes of fuel components and may facilitate excessive clad embrittlement during the LOCA oxidation phase.

The gas production and related release phenomena have a significant importance during postulated accidents. As an example, during the LOCA oxidation phase the increased internal pressure will accelerate the bursting of the cladding, which results into two-sided oxidation (and thereby accelerated embrittlement) of the rod. Moreover, the increased transient fission gas release with higher burnup increases the clad strain rate and consequently embrittle the material during a class II transient and a RIA event PCMI loading [31].

Fission gases as Xe and Kr also have a deleterious effect on the fuel-clad gap conductance because their thermal conductivity is much lower than that of the filler gas helium. Deterioration of fuel-clad gap conductance leads to an increase in the fuel center temperature causing more gas release and compromising the integrity of the fuel rod during postulated accidents.

3.1.2 Corrosion

High burnup will increase the corrosion of zirconium alloy materials since it also means longer residence time in the reactor. As mentioned before, the blanket part of the assembly stays in the reactor much longer than a conventional PWR fuel rod which makes corrosion a critical issue. There are several factors that enhance corrosion such as:

- Increased LiOH coolant content.
- Boost in the hydrogen absorbed by the cladding.
- Intense flux of high energy electrons which occur mainly in the seed part of the assembly.

The reactor operation relies on a balance between the reactivity in the core and the absorber in the water. The high burnup which can be achieved with thorium fuel cycle implies more excess reactivity than in a conventional PWR. Consequently, the chemical and radiological conditions of the fuel are abnormal and must be studied. During initial operation, the higher core reactivity can be adjusted by addition of thermal neutron absorbers (like gadolinia, erbia or ZrB_2) to the fuel rod. When all these poisons have been consumed, the higher reactivity must be controlled by addition of more boron (^{10}B) which leads to an increase of the lithium concentration in order to maintain the levels of pH. Lithium is a very effective pH controller but, at the same time, is chemically aggressive against steel walls and claddings. It boosts the chemical activity of the walls and the claddings and can make that some radioactive particles get stuck to the walls, which makes its capture by the treatment system impossible. The concentrations of lithium-boron for a conventional PWR are shown in Fig. 10 and Fig.11.

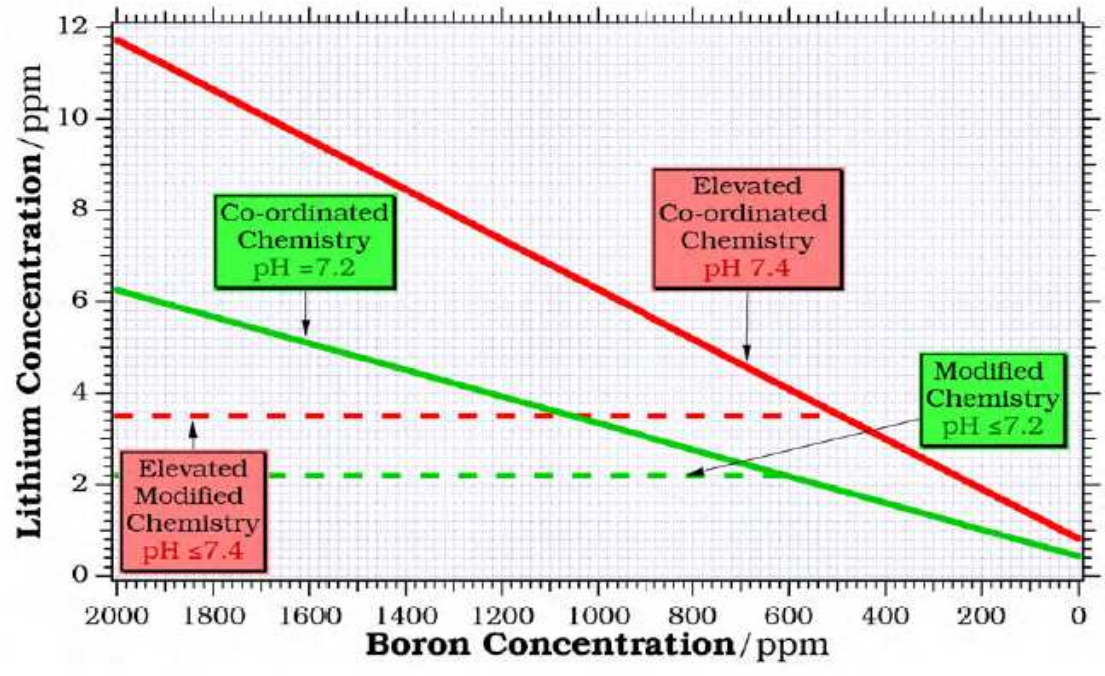


Fig. 10: Various lithium-boron models of operation for conventional PWR [26]

It is important to keep in mind that the maximum boron content is usually limited by the core design since higher boron content would result into a positive temperature reactivity coefficient.

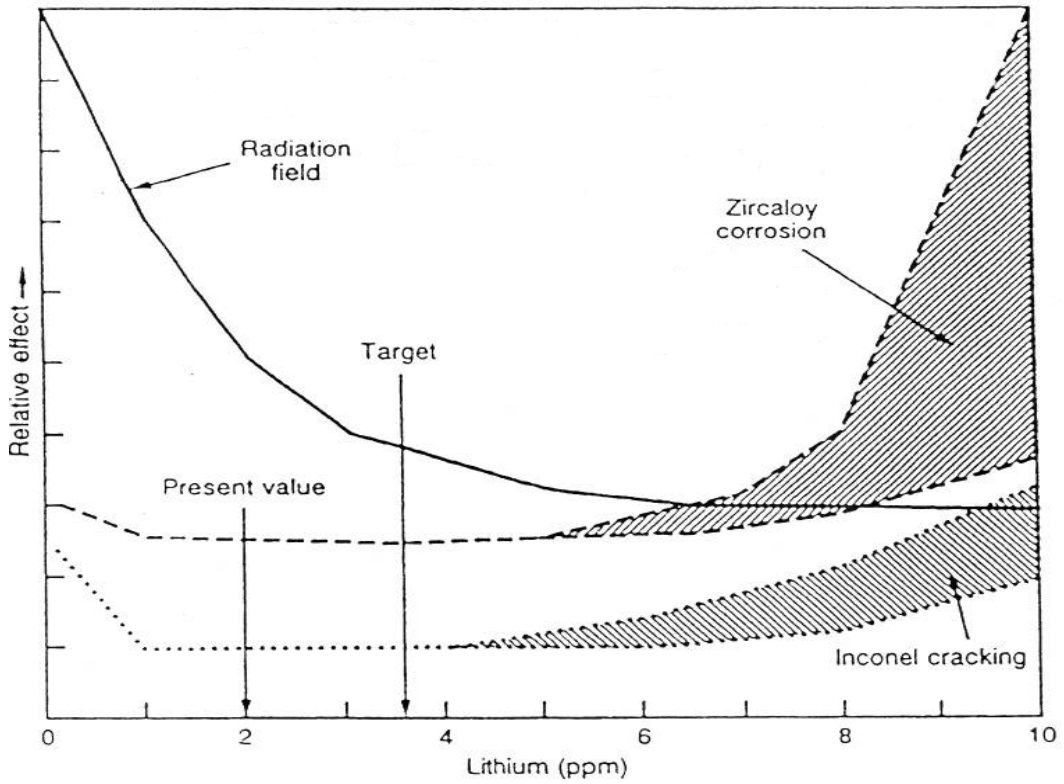


Fig. 11: Schematic diagram of PWR primary coolant chemistry; Effect of lithium concentration [27]

Zirconium reacts actively with oxygen and reduces water to form an oxide, zirconia (ZrO_2). At ambient temperature, the zirconia layer formed is dense and adhesive, making this metal practically stainless. Nevertheless, at operating temperatures and conditions the corrosion behavior is more complex. The initial corrosion phase corresponds to the formation of a protective zirconia layer. After this first stage the corrosion rate becomes constant and the oxygen begins to diffuse in the zirconia. Indeed the oxidation reaction occurs at the zirconia-zirconium interface and the progression of the oxidation front is ensured by the vacancy diffusion of oxygen through the zirconia layer [24].

The high enrichment of the seed fuel increases the radiolysis mechanisms which affects the cladding. The source of this radiation are electrons as a result of β active decay and those obtained by the creation of β^+, β^- pairs created by the materialization of high energy photons released during neutron capture. Due to the effect of this electron flux, the coolant is subjected to a radiolysis phenomenon: the H_2O molecule is dissociated into radical species which recombine with each other. In the case of Zircaloy undergoing oxidation, the existence of the porous zirconia requests to consider the free surface of the pores near the protective layer as capture sites of oxidizing species. An increase of the oxygen potential at the bottom of the zirconia pores follows and thus an acceleration of the corrosion rate [24].

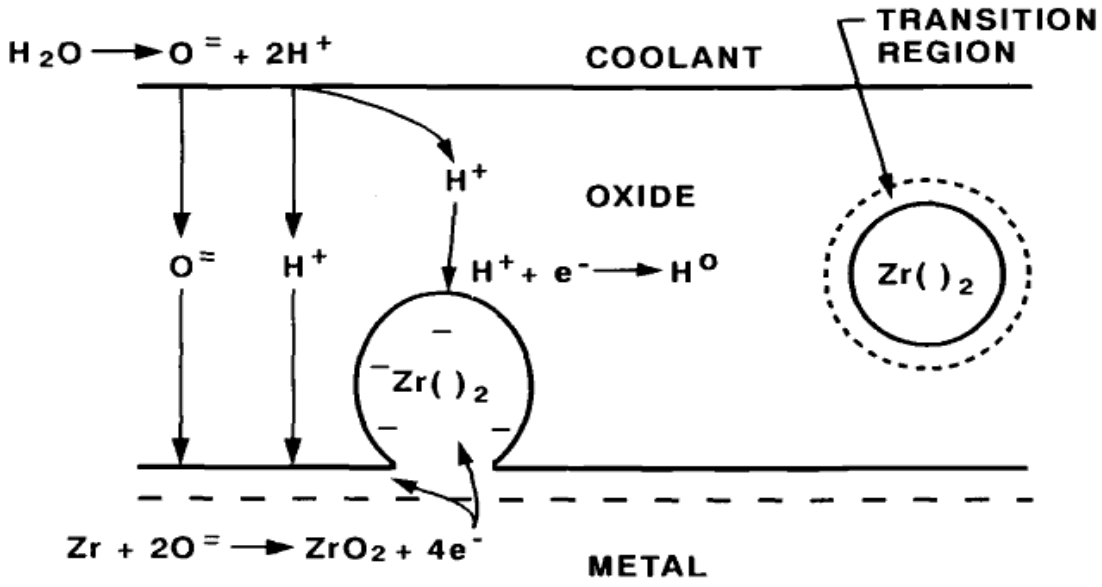


Fig. 12: Schematic of oxidation process in zirconium alloys [28]

Despite the good corrosion resistance of the zirconium alloys, Zircaloy-2 and Zircaloy-4 that are commonly used in present LWR, an effect of local corrosion is observed on them. This local corrosion, known as nodular corrosion cf. Fig. 13, is due to the high degree of burnup of the nuclear fuel. The growth of the local nodular corrosion makes that at one point these areas connect to each other and finally exfoliate from the material. The prevention of the nodular corrosion becomes essential to the operation of nuclear reactor with high degree of burnup of the nuclear fuel [29]. The ductility of the cladding is also reduced as the thickness of the ZrO_2 layer increases [35].

The nodule corrosion usually occurs in oxides with such poor electron conductivity that many H^+ ions are not reduced until they have diffused to the oxide-metal interface or close to it. The location of H^0 in the oxide determines the susceptibility to nodule formation in most proposed mechanisms. If the positive hydrogen ion which is reduced, by an electron, to H^0 is produced close to the oxide-coolant surface, the hydrogen is liberated to the coolant and the oxide grows uniformly but if H^0 is produced far into the oxide, the hydrogen is left in the oxide lattice, damaging it and eventually causing a nodule to form, cf. Fig. 12 [28].

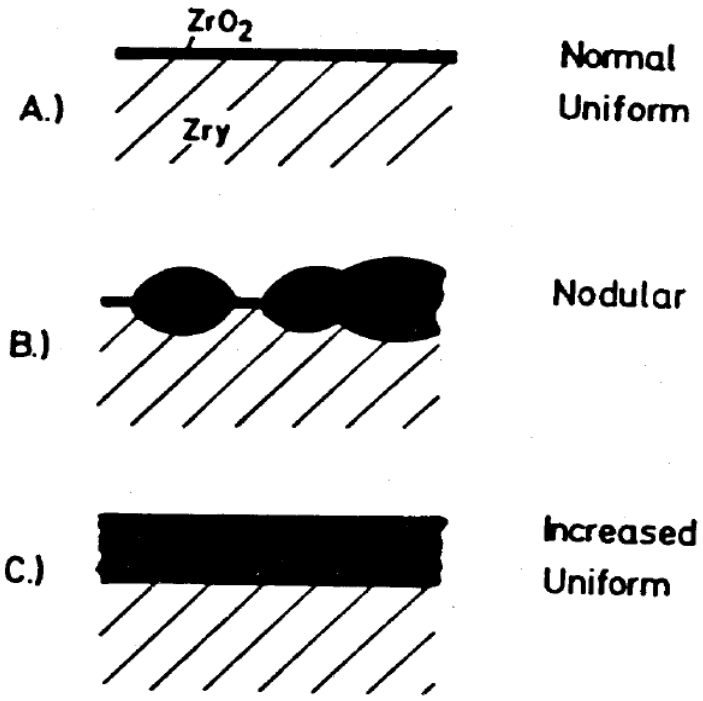


Fig. 13: Type of oxides layer formed on Zircaloy in BWR [28]

The increase in the fuel enrichment due to the high burnup leads to an increase of the fuel rod power over its life time. This situation tends to increase the fuel clad temperature which at the same time increases the corrosion rate.

In PWR and BWR it appears that hydrides at metal/oxide interface may accelerate corrosion rate, thus with increased burnup, and corrosion produced hydrogen absorbed in the zirconium alloy material will increase. This hydrogen may eventually precipitate out as hydrides and accelerate the corrosion rate.

The modifications in the operating conditions of the reactor and in the chemistry make necessary a further study to determine the concentration and margins for a safe operation without compromising the integrity of the cladding and the materials of the reactor.

3.1.3 Hydrides

Increased burnup results in more corrosion produced hydrogen that will be picked up by all zirconium alloy materials. Most of the hydrogen is released in the water but a small part (10 to 20% for Zircaloy-4) is incorporated in the metal after diffusing through the residual zirconia layer [24].

Zirconium-based alloys, in general, have a strong affinity for oxygen, nitrogen, and hydrogen. As seen in Fig. 14 zirconium alloys have a tendency to pick up corrosion hydrogen and that released from the water due to the radiological effect. The thickness of the external oxide scale and the amount of hydrogen ingress into the alloy are strongly dependent on exposure time and temperature. Fig. 14 shows the Zr-H phase diagram, indicating the low solubility value for hydrogen in the alloy (α phase).

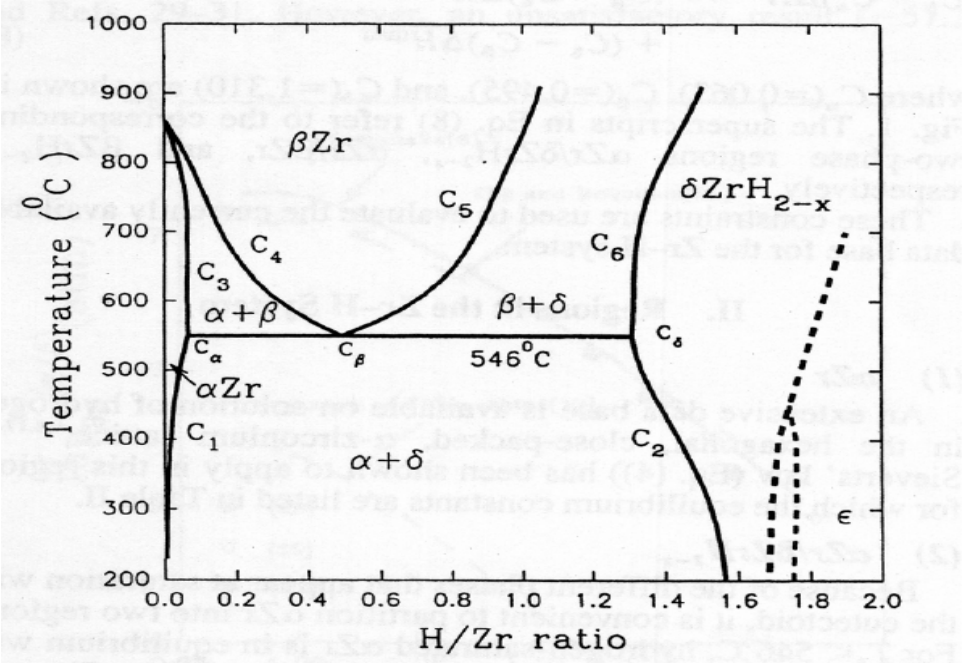


Fig. 14: H-Zr phase diagram indicating the low solubility point [24]

Hydrogen in excess will precipitate out as zirconium hydride that may embrittle the material in different degrees, depending not only in the concentration but also on how the hydrides are distributed and orientated in the material. At irradiation temperatures, hydrides are ductile and do not contribute to the embrittlement of the cladding, but at room temperature, when the solubility is lower than 1 ppm and the hydrides are brittle, an excessive concentration of hydrogen can lead to a dramatic mechanical embrittlement of the cladding [24]. In general the following considerations are applicable for this issue [30]:

- Increased fraction of hydrides reduces ductility and fracture toughness. The embrittlement effect of the hydrides is very temperature dependent.
- Nonuniform distribution of hydrides reduces ductility and fracture toughness more than uniformly distributed hydrides. Nonuniform hydride distribution is only found in components subjected to a heat flux. This effect is driven by a thermal gradient and consequently it is only seen in the fuel rods where hydrogen in soluble form tends to locate at areas with lower temperature.
- The increased surface heat flux, which may be a consequence of the higher reactivity due to the high burnup, increases the tendency for hydride rim formation.
- The embrittlement of the cladding is increased by the formation of hydrides which are orientated perpendicular to the major tensile stress directions.

During irradiation, the thermal gradient in the cladding is responsible for the heterogeneous redistribution of hydrogen. It tends to migrate towards the cold areas, which means to the external surface of the cladding.

3.1.4 PCI/PCMI

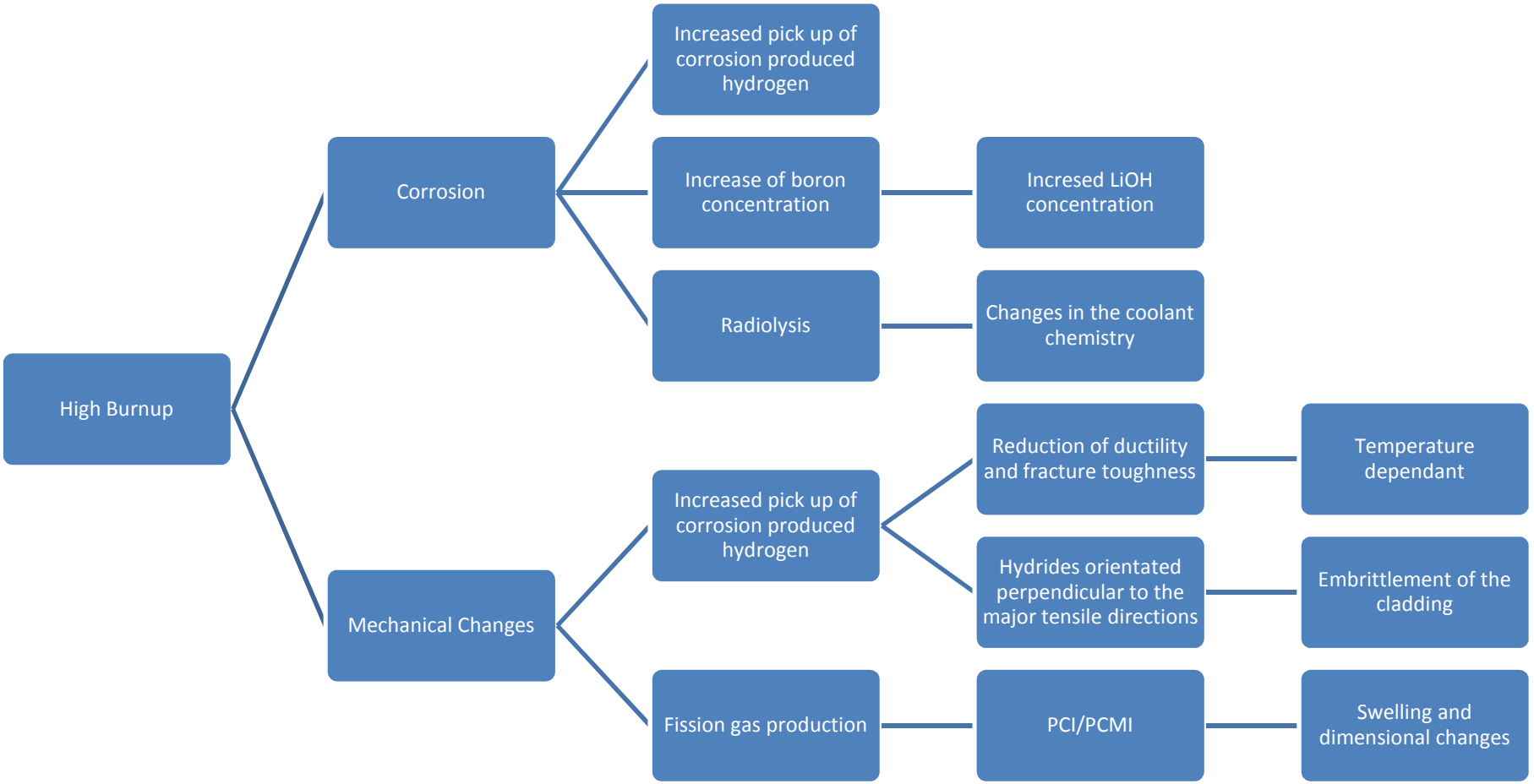
In case of a power increase PCI/PCMI (pellet-cladding mechanical interaction) constitutes a challenge for the cladding due to the induced stress which might lead to fuel rod failure. The pellet-cladding interaction takes place when the pressure of the water creeps down the cladding making contact with the fuel pellet [30]. This phenomenon occurs after a few operating cycles [34]. Although, as mentioned in the benefits and challenges of thorium, the fuel produces more fission products due to the high burnup and it may seem to decrease the tendency to PCI, there are several effect that enhance this effect.

On one hand, despite the increase in fission gas release (FGR) might mitigate the pressure of the water in the cladding, the fission products also create a more aggressive environment for the cladding (Cs, Cd, I) and there is an increased transient FGR during ramp with burnup which results in an increase in the tensile stress.

On the other hand with the burnup achieved in the thorium fuel cycles there is an increase tendency to rim formation and hydride concentration, which will embrittle the clad. Also the dissolution of SPPs may have this effect in the cladding [32].

The PCI/PCMI effect constitutes a major issue in the seed part of the Radkowsky fuel assembly. In this element the high burnup of the fuel and the high local power density results in an increase of the fission gas release which affect the cladding and can lead to pellet-cladding interaction. The elevated power density, which directly relates to the linear heat rate, also could be a problem in case of pellet-cladding contact during postulated accidents like LOCA.

3.1.5 Graphic of the critical issues of the cladding



3.2 Properties of structural materials

The selection of the materials used as cladding, or as structural material in general, must have specific properties relating their mechanical, thermal, chemical and neutronic behavior. The desired properties and interaction with other elements of the reactor are listed below:

- Mechanical properties have to be adapted to the imposed operation conditions (eg. resistance to the internal pressure of fission gases). Some of the properties that should present are ductility, high resistance to break and high resistance to fatigue.
- Thermal properties: high conductivity, low dilatation coefficient and high melting temperature.
- Satisfactory chemical compatibility with the coolant.
- Compatibility with the fuel.
- Resistance to corrosion and to the effects of the interaction with photons and neutrons of high energy.
- Low absorption cross section to avoid the neutron activation of the materials and to take into account the economic level.

The choice of fuel and structural materials is closely linked to the type of reactor in which they would be used. The kind of reactor will impose the level of stress and the conditions to which the materials will be subjected: temperature, coolant pressure, neutron flux, etc. The points below show the range of possibilities and the different characteristics available for structural materials [27].

- It is essential that the absorption cross section of neutrons is low and the melting temperature high. Only Aluminum (Al), Beryllium (Be), Magnesium (Mg) and Zirconium (Zr) satisfy these two requirements for thermal neutrons of 0.025 eV Al (0.23 barn), Be (0.0092 barn), Mg (0.063 barn), Zr (0.185 barn) respectively.
- Iron, chromium and nickel have absorption cross sections of 2.6 barn, 3.1 barn and 4.4 barn respectively, but stainless steel have better resistance to corrosion of water and good mechanical properties.
- In normal conditions the materials must not interact chemically either with the fuel or the coolant. As an example at 300 ºC the zirconium reacts with water creating a thin oxide layer. At 1000 ºC the oxidation is faster and can produce damage in the fuel.
- For reactors in which the temperature reaches or exceeds 700 ºC the use of stainless steel or a ferronickel alloy is necessary in order to obtain adequate mechanical characteristics. For high-temperature reactors in which the temperature exceeds 1000 ºC, metallic materials are replaced by refractory materials.

3.3 Cladding materials

Fuel cladding is the first physical barrier of a nuclear power plant to ensure its safe operation. Of all the physical barriers, the cladding is the one subject to the most intense and challenging environment. This element not only has to be able to work in extreme conditions with corrosion problems, fission gas release which increase the clad strain and dimensional changes and micro-structural changes, but also must keep all its thermal properties and mechanical integrity during its operation life time and in front of a postulated accident like RIA (reactivity initiated accident) or LOCA (loss-of-coolant accident).

The cladding has been one of the elements of the reactor on which most studies have been made as a result of the harsh operation environment and the specific characteristics that should feature. During the nuclear history different kinds of materials have been used. Some of them are still in use but others were substituted by more advanced materials which offer better properties and characteristics. This section provides an overview over cladding materials, the new challenges they face and their future development.

3.3.1 Materials and alloying elements

Zirconium

Zirconium is a commercially available refractory metal with excellent corrosion resistance, good mechanical properties, very low thermal neutron cross section and which can be manufactured using standard fabrication techniques [36]. The metal is present in two allotropic forms: the low temperature stable structure α is hexagonal close-packed, whereas the high temperature β phase is body-centered cubic. For pure zirconium the alpha-beta transition takes place at 863 ºC and the melting point is at 1855 ºC. This characteristic places zirconium at the limit of refractory metals [24].
The main physical properties are described in Table 11.

Property at room temperature	Unit	Average (or for one tube)	*a* direction	*c* direction
Density	$kg\ dm^{-3}$	6.5		
Young's modulus	GPa	axial= 102 radial= 92	99	125
Coefficient of expansion	K^{-1}	axial= $5.6 \cdot 10^{-6}$ radial= $6.8 \cdot 10^{-6}$	$5.2 \cdot 10^{-6}$	$7.8 \cdot 10^{-6}$
Lattice parameter	nm		0.3233	0.5147
Specific heat capacity	$J\ kg^{-1}\ K^{-1}$	276		
Thermal conductivity	$W\ m^{-1}\ K^{-1}$	22		
Thermal neutron absorption cross section	barn	0.185		

Table 11: Main physical properties of zirconium [24]

The early reactors used stainless steel as a cladding but due to its superior neutron economy, corrosion resistance and good mechanical strength at high temperatures, zirconium alloy became the principal cladding material by the 1960s. In fact the mechanical strength and corrosion are the determining factors. Other metals also have a low thermal neutron absorption cross section, but their resistance to corrosion is reduced as soon as the temperature rises.

Despite its good properties, zirconium has some drawbacks relating to its fabrication and hydrides absorption. Reactor-grade zirconium alloys must be made of purified zirconium free of hafnium contamination. Hafnium resembles zirconium and is found in zirconium minerals. Their chemistry is so similar that they are two of the most difficult elements to separate. The problem is that hafnium has a very high neutron absorption cross section, about 600 times higher than that of zirconium, which makes its removal a necessity.

Regarding its hydride absorption, zirconium has a tendency to pick up corrosion produced hydrogen.

$$Zr + 2\mathrm{H}_2O \rightarrow ZrO_2 + 2\mathrm{H}_2$$

This behavior reduces the ductility and fracture toughness and increases also the embrittlement of the rod.

There have been various zirconium alloy grades used in water-cooled nuclear reactors. The main idea of these alloys is to combine the good thermal and neutronic properties of zirconium with other elements, like chromium, niobium, iron, nickel, etc. They offer high corrosion resistance, hardness and less interaction with hydrides. The improvement of the zirconium metal has lead to select several main alloying elements, some of them also used in other cladding materials like stainless steel:

- **Oxygen**: The addition of some oxygen into the zirconia structure results in a solid solution which increase the hardness and thermal shock resistivity of this metal. The oxygen atom is in an interstitial solid solution on the octahedral sites and stabilizes the alpha phase.

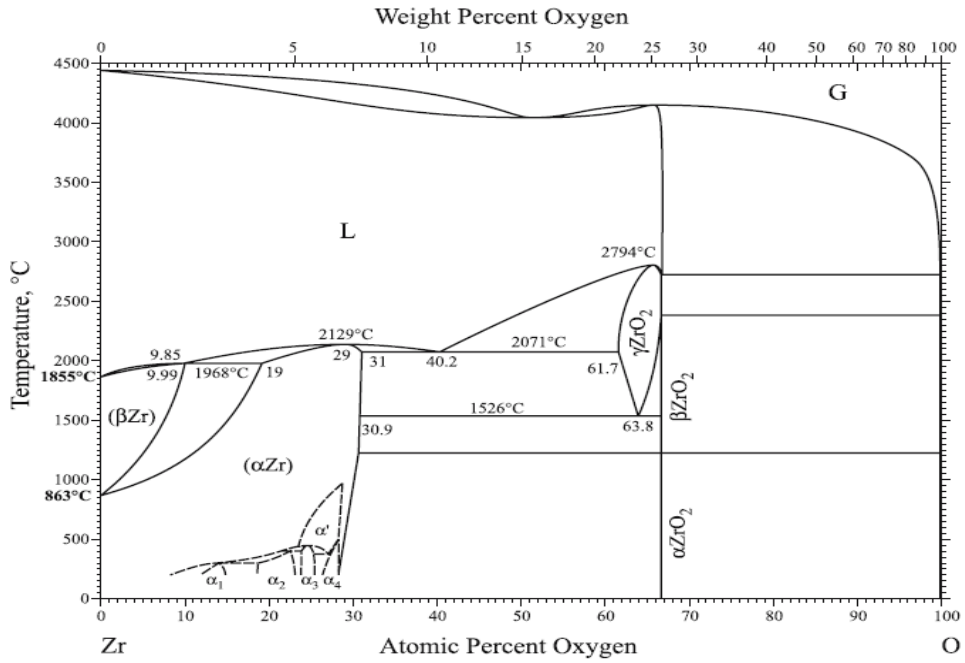


Fig. 15: O-Zr phase diagram [37]

- **Tin**: Is the basis of a large class of alloys, the Zircaloys. Its α phase in the solid solution is substituted for zirconium and also leads to a reduction of the β region. Tin is used to coat the zirconium to improve its corrosion resistance.

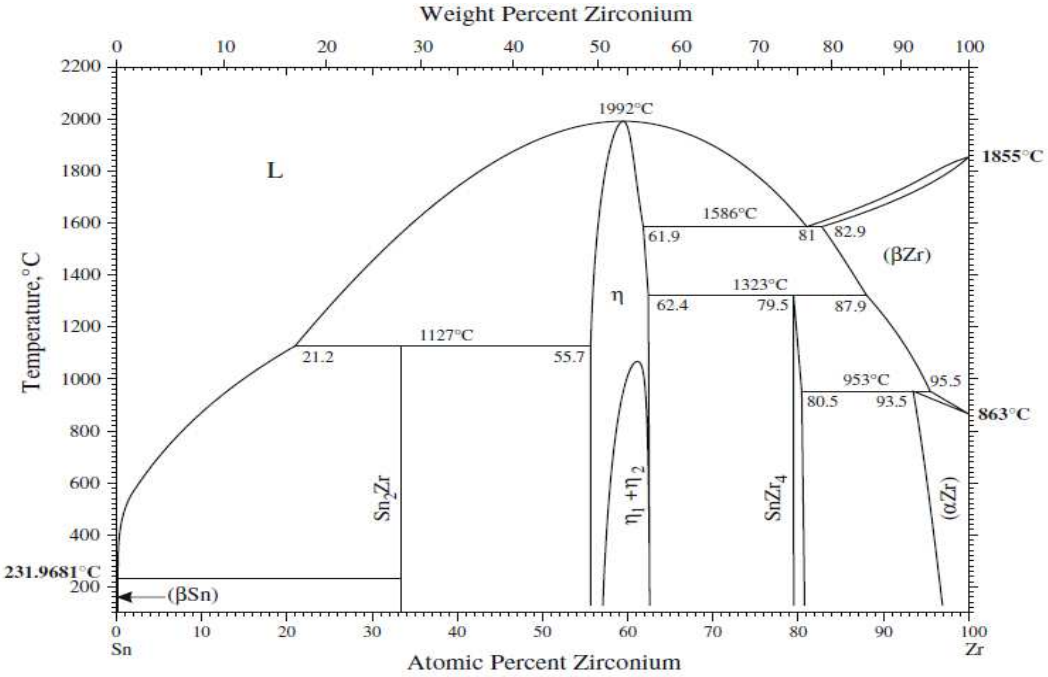


Fig. 16: Sn-Zr phase diagram [38]

Tin contributes to compensate the deleterious effects of nitrogen on corrosion. As a result of the improvement in the control of the nitrogen content, the general tendency for new alloys is to reduce the tin content, even though it also contributes to the improvement of the mechanical properties.

- **Niobium**: This element started to be used in Russian VVER and RBMK reactors with the aim of improving the efficiency of the fuel cycles and raising the operating characteristics. Niobium is also soluble at any concentration in the beta-phase.

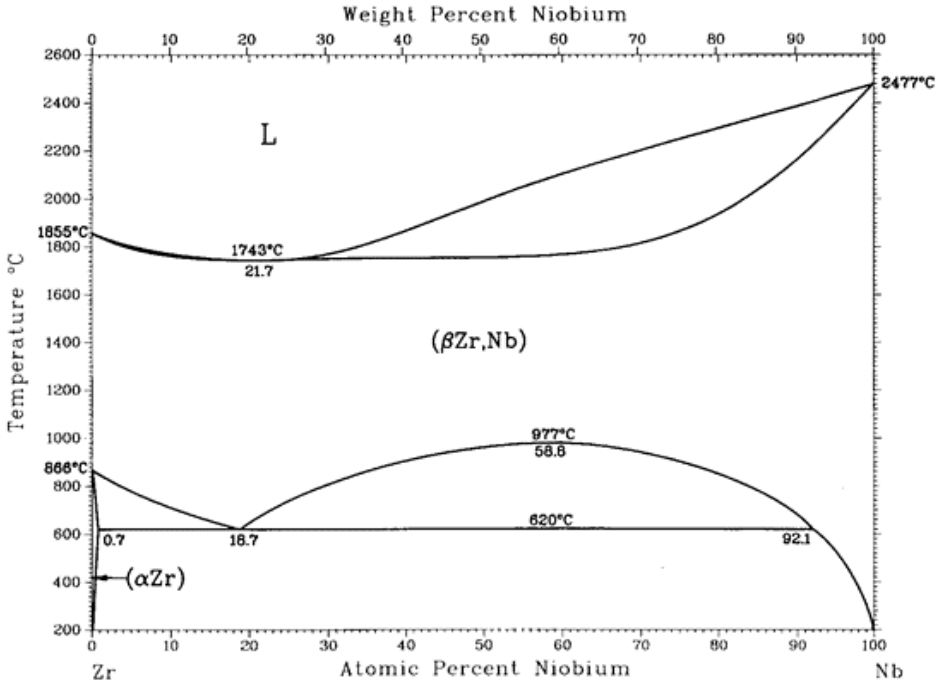


Fig. 17: Zr-Nb phase diagram [39]

Niobium provides the zirconium alloy with high corrosion resistance and deformation stability. The superior resistance of the Zr-Nb alloys is due to their post-transition behaviour. The oxide does not reach a thickness at which it breaks down, instead the oxide forms porosity at the interface between the pre-transition and post-transition oxides. This provides the cladding with a protective layer between the coolant environment and the metal substrate. The Zr-Nb alloys are presently considered potential substitutes for the Zircaloys in PWR [40].

- **Nickel**: This element has similar properties than iron. It has slightly lower strength and hardness and is magnetic, but in contrast to iron nickel presents a high resistance to corrosion and that is the reason for its use in the fuel cladding. The zirconium-nickel alloys are mainly used in the BWR. The reason is that nickel has a great tendency to pick up hydrogen which makes it worthless in PWR, where hydride corrosion is a main problem.

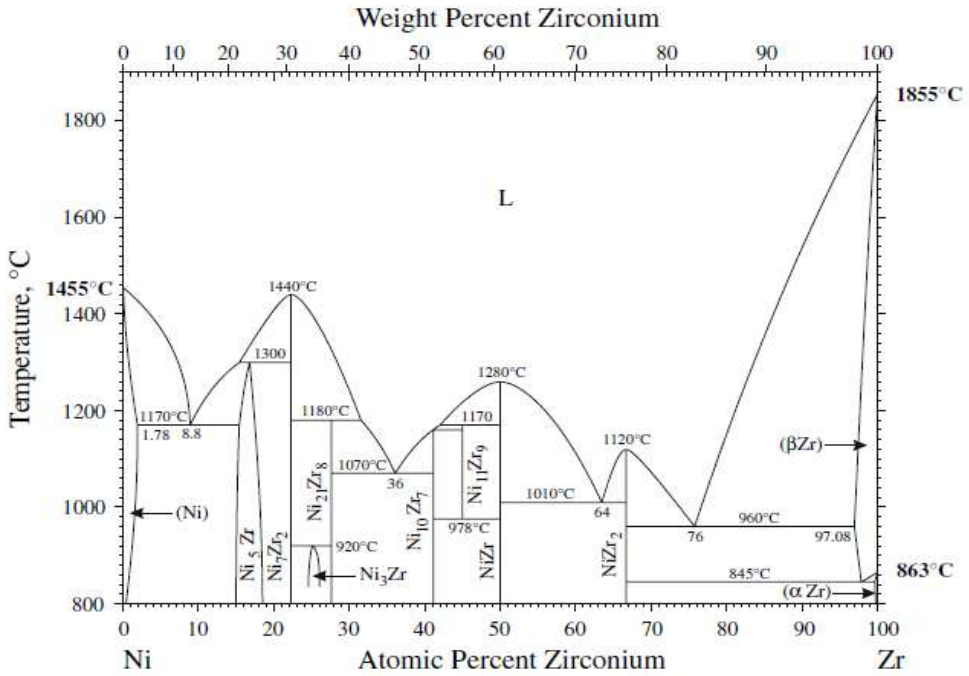


Fig. 18: Zr-Ni phase diagram [41]

- **Chromium**: The introduction of small amounts of chromium increase the sensitivity of the alloy to oxidation. This is because the diffusion rate of oxygen is increased. The oxidation resistance can be attributed to the formation of a highly adherent protective layer.

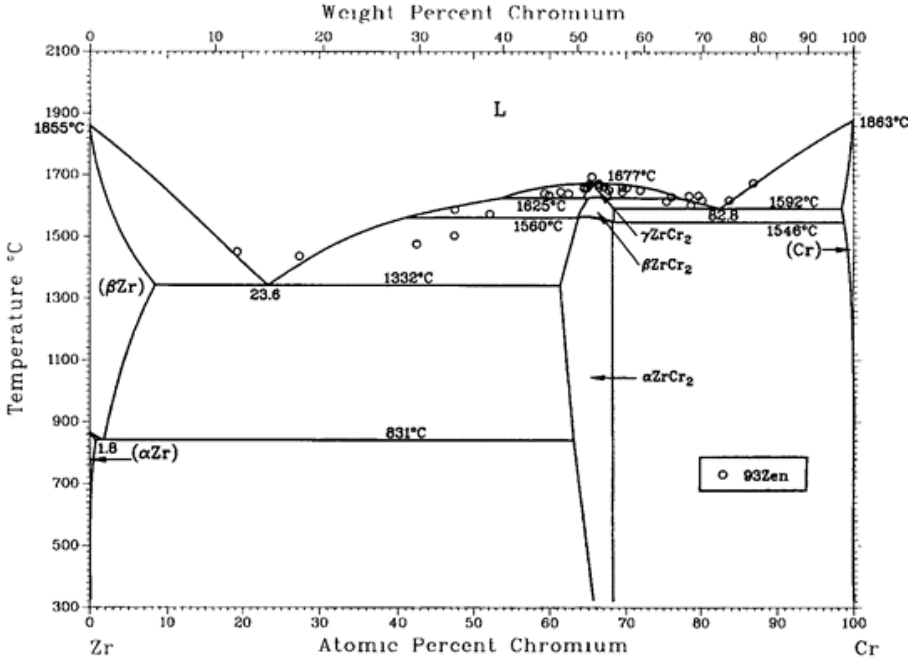


Fig. 19: Zr-Cr phase diagram [42]

The addition of chromium to the alloy also increases its hardness and in case of steel makes it highly resistant to corrosion and dislocations.

- **Iron**: This material is added to the alloy to increase the corrosion resistance and to improve the mechanical properties. It has a lower capture cross section than chromium and nickel which makes iron a good option to enhance the corrosion resistance of the zirconium alloys.

Table 12 summarizes all the main alloying elements and the benefits that offer to the alloy.

	Enhance Corrosion Resistance	Enhance Mechanical Properties
Tin	X	X
Iron	X	X
Chromium	X	X
Nickel	X	
Oxygen	X	X
Niobium	X	X

Table 12: Benefits of the main alloy materials

3.3.2 Cladding options

Zircaloy-2

The classical zirconium alloy for nuclear applications in western boiling water reactors (BWR) is Zircaloy-2. The difference to Zircaloy-4 is that it contains nickel, as shown in Table 13, to increase the resistance to water corrosion. Nickel has a tendency to absorb hydrogen, which is more likely to occur in PWR than in BWR [24].

Alloys	Content (%)					
	Tin	Iron	Chromium	Nickel	Oxygen	Nitrogen
Zircaloy-2	1.20/1.70	0.07/0.20	0.05/0.15	0.03/0.08	0.08/0.15	<0.008
Zircaloy-4	1.20/1.70	0.18/0.24	0.07/0.13	<0.007	0.08/0.15	<0.008

Table 13: Composition of Zircaloy-2 and Zircaloy-4 [24]

Zircaloy-4

The classical zirconium alloy for nuclear applications in western pressurized water reactors (PWR) is Zircaloy-4 with approximately 1.5 wt% tin as the major alloying element. It is used for fuel cladding, control rod guide tubes, and grid spacers. Currently Zircaloy-4 is being substituted by advanced cladding materials which are optimized for high burnup and long operation times in nuclear power plants [43].

Zirlo

Zirlo is a special developed alloy of zirconium with niobium, tin and iron. The use of this alloy intends to face the trends towards high fuel burnups, extended cycles and higher primary coolant temperatures with higher lithium demands. The corrosion resistance and enhanced structural stability of Zirlo cladding enable longer cycle length at higher temperatures without reducing operation margins in a standard PWR.

As mentioned before, the use of Zircaloy-4 presents some drawbacks relating its tendency to pick up corrosion hydrogen and its lower corrosion resistance due to the substitution of nickel for iron [33]. These reasons motivated an extensive search for a successor to Zircaloy-4 for cladding in PWR. In fact Zirlo has replaced Zircaloy-4 as a cladding in almost all of Westinghouse fuel deliveries since the introduction in the 1990's.

The main differences between Zircaloy-4 and Zirlo, which makes this alloy more resistant to corrosion, retaining at the same time its physical good properties of zircaloy, is the reduction in the tin content and the addition of niobium as an alloying element. The alloying elements of Zirlo are shown in Table 14.

Element	Sn	Fe	Cr	O	Nb
Composition	1	0.01	0.08	0.01	1

Table 14: Chemical composition of Zirlo (wt%) [44]

The tin content has an impact in the corrosion behavior and the decrease of this element in Zirlo compared to Zircaloy-4 which has about 1.3% more tin is one of the reasons for the improved corrosion properties. Optimized Zirlos have been tested, lowering the content of tin. With this tendency the oxide thickness is significantly lowered even compared to the conventional Zirlo. This is especially significant at high burnup, above approximately 50 MWd/kgU [45]. The most promising aspects of this alloy are the low corrosion rate, best creep properties under irradiation and increase of mechanical resistance [46].

Ferritic alloys

Ferritic based alloys like stainless 304 were one of the first cladding options for nuclear reactors. With the discovery of the good properties of zirconium alloys, ferritic alloys became rarer as a cladding material but are still used in some applications in which the temperature reaches or exceeds 700 ºC. In this kind of reactors the corrosion resistance and neutron economy are not so important as to keep the adequate mechanical characteristics [24].

There are many types of ferritic alloys depending on the alloying elements used. For example some of the stainless steel used for cladding are austenitic steel, ferritic steel, the types 302, 304, 304L, 305, 308, etc. It is not the goal of this report to describe in any detail the components and characteristics of these alloys but to offer a general overview of their main characteristics and problems. Table 15 presents some of the alloying elements and some mechanical properties of stainless steel.

C	Mn	Fe	S	Si	P	Cu	Ni	Cr	Co
0.08	2.00	---	0.03	1.00	0.045	---		19-21	---
Yield Strength				Tensile Strength				Elongation	
30.0 ksi (205 MPa)				75.0 ksi (515 MPa)				30%	

Table 15: Type 308 Stainless Steel at room temperature. Chemical composition and mechanical properties

Stainless steel presents correct mechanical properties over a wide range of temperatures and conditions but does not present such a good neutron and corrosion properties as zirconium alloys. The stress corrosion cracking (SCC) of stainless steels is an important degradation phenomenon not only in boiling water reactors (BWRs) but also in pressurized water reactors (PWRs) [47].

In order to improve the corrosion resistance of stainless steel chromium was added, which makes the alloy more resistant to corrosion and dislocations. Table 16 shows steel alloys with different amounts of chromium, and in Fig. 20 are shown the oxide film in each case.

	C	Si	Mn	P	S	Ni	Cr	Mo	Fe
5% Cr	0.050	0.54	1.54	0.023	0.001	12.09	5.12	2.34	Balance
10% Cr	0.049	0.50	1.50	0.022	0.001	12.10	9.45	2.31	Balance
12% Cr	0.059	0.49	1.54	0.005	0.006	14.06	11.91	2.27	Balance
15% Cr	0.059	0.47	1.51	0.006	0.005	13.88	14.99	2.23	Balance
SUS 316	0.051	0.51	1.48	0.028	0.001	13.20	16.47	2.34	Balance
20% Cr	0.044	0.5	1.48	0.008	0.005	13.70	19.81	2.24	Balance
Carbon steel	0.19	0.19	0.62	0.011	0.005	0.02	0.04	0.01	Balance

Table 16: Chemical compositions of stainless steel alloys (wt%) [48]

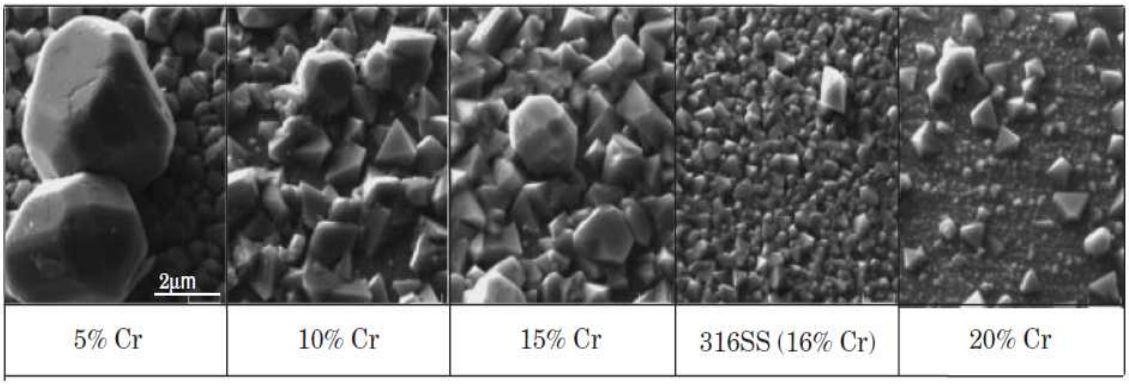


Fig. 20: Oxide film after immersion in simulated PWR primary water at 320 ºC for 380 h [48]

E110

In Russia the niobium-bearing alloy E110 (Zr1%Nb) has been used for cladding and structural materials in RBMK and VVER nuclear reactors for many years. It also contains a small amount of oxygen, about 0.05 wt%O, to increase the hardness and thermal shock resistivity of this metal. This material has been extensively investigated, mainly in Russian and Eastern European laboratories [43]. Niobium improves the mechanical strength and contributes to the resistance against water corrosion.

Some studies have demonstrated that E110 shows good mechanical properties and a better corrosion resistance than Zircaloy alloys. The improvement is mainly due to the lowering of the tin content.

Many alloys and material options have been tried as nuclear cladding. Some of them are still in use and constitute an acceptable option for standard reactors. Alloys like Zirlo or E110 could be used for higher burnups and longer resident times in the reactor like those related with thorium fuel cycles but their use under these new conditions and configuration (seed-blanket or whole core assembly) have to be tested. While these studies are being carried on, new types of cladding materials are considered. The following cladding materials are some of the options that are being considered in order to face high burnups and enhance corrosion resistance. They could be a valid option for its use in thorium reactors.

M5

M5 is an advanced fuel cladding and fuel assembly structural material for high burnup fuel applications. This alloy is composed of zirconium (99%) and niobium (1%) with 0.15 wt% O. The inclusion of niobium and the reduction of tin make this alloy less affected to hydrides than Zircaloy-4 and Zirlo, while still maintaining the same good mechanical properties and low thermal neutron cross section.

This alloy has been tested by the company Framatome showing a good behavior against corrosion at high burnups. In Fig. 21 some results of these experiences, comparing M5 with Zircaloy-4, can be seen.

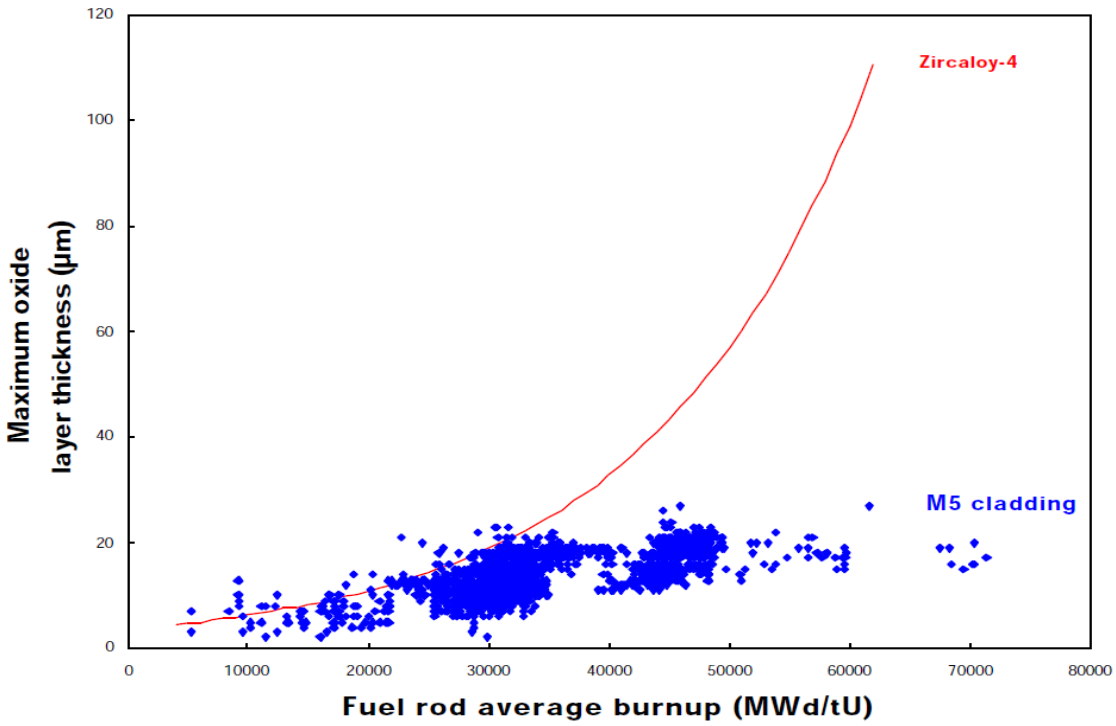


Fig. 21: Corrosion behavior of M5 compared to Zircaloy-4 [14]

The excellent corrosion resistance of M5 cladding makes this alloy a promising option for its use as a blanket in a thorium fuel cycle. This alloy could resist the long resident time of the fuel element in a harsh environment.

Duplex alloys

With the achievement of high burnup more challenging conditions will be faced in the reactor including higher pH and higher temperatures in the primary coolant. Under these operating conditions the corrosion resistance of Zircaloy-4 could be a performance-limiting factor especially for high temperature reactors. In order to face this problem without changing the basic structure and cladding material, a new alloy was developed, the duplex alloys. The concept is to introduce a thin external layer of a high corrosion resistant new zirconium-based alloy bonded to an inner Zircaloy-4 substrate layer. This concept was modified to the use of duplex cladding, which is composed of two layers of distinct composition. The inner 90% of the cladding wall is Zircaloy-4, but the outer 10% of thickness consists of an alloy of entirely different composition. With this concept, the design keeps the strength and neutron economy of Zircaloy-4 while enhancing corrosion resistance, offering a cladding design alternative for PWR applications [52].

Ceramic clad

One of the options which has been considered in order to face the new operating challenges is the replacement of zirconium alloys by ceramic materials. The use of such claddings would enable power levels higher than those achieved today while reducing or eliminating the consequence of design basis accidents such as loss-of-coolant-accident [53]. The ceramic composite silicon carbide is also more durable and less reactive than zircaloy cladding.
The use of zirconium-based cladding is widely used and has shown good operating behavior with no fuel failures through end-of-life, even as burnups approach currently licensed limits. Although future improvements in the cladding can extend the reliable use of zirconium alloys to burnups of 75 GWd/MtU and beyond, the initiative of replacing it by ceramic alloys has been taken in the United States.

In early investigations the study was carried out with ceramic composites made from alumina fibers and an alumina matrix, known as continuous fiber ceramic composite (CFCC). However, these investigations concluded that alumina composites were not acceptable for two reasons: the composite was permeable to fission gases and the alumina lost much of its strength during irradiation. Regarding these results a new approach was taken. The attention was turned to multilayered composites that could embody the hermeticity to retain fission gas, and the ductile behavior for robust in-pile service. The inner layer would be a high-density material to hold fission gases, and the outer layer would be a composite with the required strength. The material used was also switched from alumina to silicon carbide, cf. Fig. 22. This material would retain its strength under irradiation and has better mechanical properties at high temperatures, thus assuring survival with minimum damage during LOCA events and failure during DNB [53].

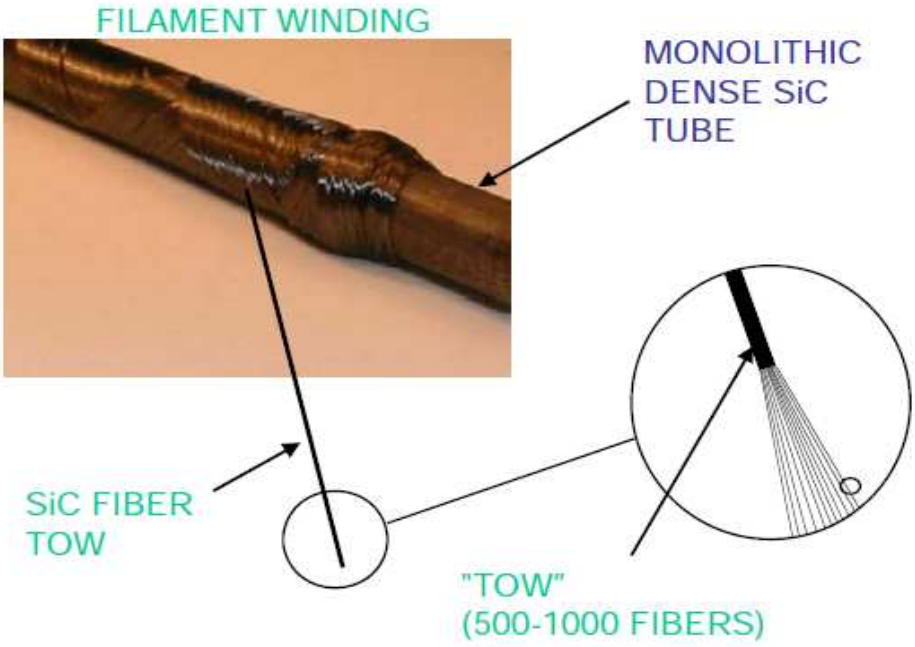


Fig. 22: Multilayered SiC clad fuel [49]

The SiC cladding has also the advantage that unlike zircaloy cladding the gap between the fuel pellet and the cladding would not close during the first few operating cycles. Therefore, the SiC clad is expected to operate at higher temperatures and to release less fission gas than zirconium fuel clad. Some of the advantages and technical characteristics of this cladding are [49].

- Strength retention to at least 1500 ºC appears to be DNB proof, and therefore can facilitate power levels of 30% or more.
- Minimal exothermic water reaction or H_2 release during LOCA's.
- Fully retain fission gases, no creep and fission gas retention to at least 5000 psi (34.5 MPa).
- Composite layer solves ceramic brittleness problem.
- Can operate in LWR coolant for over 10 years with no appreciable corrosion.
- Very hard, resists fretting and debris failure, further reduction in operation failures.

SiC has also higher yield and ultimate strength than Zircaloy-4 under the same primary stresses, which give higher primary safety margin. Despite the good mechanical properties, the brittle nature of SiC can be an important issue to be investigated. In addition, the degradation of SiC thermal conductivity after irradiation can also hurt its performance. A comparison of the properties of Zircaloy-4 and SiC is shown in Table 17.

	Zircaloy-4	SiC
Coefficient of Thermal Expansion	6 μm/m·K	3 μm/m·K
Modulus of Elasticity	99.3 GPa	410 GPa
Poisson's Ratio	0.37	0.21
Yield Strength	170 MPa	450 MPa
Ultimate Tensile Strength	241 MPa	450 MPa

Table 17: Properties of Zircaloy-4 and SiC [50]

Although all the tests have been conducted for UO_2 fuel pellets, all the properties and characteristics listed above are applicable to thorium fuel cycles and make this cladding an interesting option for thorium based fuel reactors. Nevertheless the main interests seem to be its use in the seed assembly part of the thorium fuel cycles. According to the studies the good thermal and mechanical behavior under irradiation and at high temperatures would be of great interest in the seed part where the maximum local power density is higher than in a usual PWR and the pellet cladding interaction and the fission gas release are more critical.

In Table 18 there is a comparison of some of the alloys mentioned in a LOCA scenario. The tendency to high burnups can be seen and the effect that it has in the corrosion layer and the hydrogen content. The improved alloys and materials can operate at high burnups with less corrosion and embrittlement due to hydrogen pick up.

Cladding	Reactor (Discharge)	Burnup (GWd/MTU)	Corrosion Layer (microns)	Hydrogen Content (wppm)
15×15 Zircaloy-4	H.B. Robinson (Apr. 1995)	64	71-75 95	550±100 740±110
17×17 ZIRLO	North Anna (Mar. 2001)	70	43±2 (43±2)	620±140 670±40
17×17 M5	Ringhals (Jul. 2003)	63	12±1	110±5
	North Anna (May. 2004)	68, 72	10-20	100±30

Table 18: High-burnup cladding for LOCA embrittlement tests [54]

The alloys and cladding materials showed in this part of this document are only the main options used or considered. It is beyond the scope of this document to try to comment all the alloys and options used as cladding materials. Table 19 shows some of the options considered in the past and the variability in the alloying elements.

Alloys	Content of elements (%)				
	Nb	Sn	Fe	Cr	O
E110 (Russia, 1985)	0.95-1.05	---	0.006-0.0112	---	≤ 0.10
E110K (Russia, 1972)	0.95-1.05	---	0.006-0.0112	---	0.12-0.16
E635 (Russia, 1971)	0.95-1.05	1.10-1.30	0.30-0.40	---	0.05-0.12
M5 (France, 1996)	0.8-1.2	---	0.015-0.06	---	0.09-0.13
Zircaloy-4 (USA, 1952)	---	1.2-1.7	0.18-0.24	0.07-0.13	0.09-0.13
Zirlo (USA, 1990)	0.9-1.13	0.9-1.2	0.1	---	0.09-0.15
MDA (Japan, 1990)	0.5	0.8	0.2	0.1	---

Table 19: Chemical composition for zirconium alloys for PWR and VVER reactors [51]

According to the information found and the properties of the different materials considered, the behavior of the cladding options against different situations and properties are summarized in Table 20. The good behavior of the material in these situations is marked with an X while the unsatisfactory behavior is left blank. For some cladding options the behavior of the material is not clear and has to be tested further, in those cases a question mark is used. Despite in some cases the cladding might present a satisfactory behavior, the degree is not the same for all the materials (eg. M5 and Zircaloy-4 present good corrosion resistance compared with ferritic alloys but M5 has more enhanced corrosion resistance than zircaloy).

	Properties				
Cladding Materials	Resistance to hydride absorption	Good neutron economy	Resistance to corrosion	Longer residence time in the reactor	Good mechanical properties at high temperatures (over 700ºC)
Zircaloy-2	X	X	X		
Zircaloy-4		X	X		
Zirlo	X	X	X	?	
Ferritic Alloys	X				X
E110/M5	X	X	X	X	?
Duplex Alloys	X	X	X	X	?
Ceramic	X	?	X	X	X

Table 20: Comparison of the cladding alternatives

4. Fuel modeling

Nuclear fuel elements are subject to a large number of physical and chemical processes that take place during reactor operation. Reliable predictions of fuel behavior are important in order to improve design and economics of the nuclear fuel cycle and to ensure safe operation margins.

In this section a program called FEMAXI-6 is used to model the fuel rod behavior, using thorium as a fuel in a light water reactor. The reactor fuel analysis code FEMAXI-6 has been provided by the Nuclear Energy Agency of the OECD [55]. The conditions and characteristics are those found in a conventional PWR using the whole-assembly seed and blanket core (WASB) configuration. The objective of the thorium fuel cycle is to introduce thorium in a conventional light water reactor with minimum modifications, for that reason all the operative conditions are the same as if the reactor would use uranium as fuel. In particular, the modeled part of the assembly is the blanket, which contains the thorium fuel. The main difference with a conventional PWR operation is a higher burnup and a longer residence time. The purpose is to see the effect that longer time and burnup have on the physical conditions of the cladding.

The research carried out in the thorium fuel cycle is not as extensive as for the uranium. For this reason the data and the models available are in some cases difficult to access or they are not fitted for the physical and chemical characteristics of thorium.

Regarding the huge number of interrelated processes, modeling programs are subject to uncertainty and are continuously being developed and improved. The thermal and mechanical behavior of a fuel rod depends strongly on complex phenomena that vary with burnup.

The results obtained in this section are rough approximations due to the difficulties in modeling: the lack of data to contrast the values obtained and, as mentioned, to the impossibility to find particular models validated for thorium.

4.1 Programs used for modeling

FEMAXI-6 is a light water reactor code which predicts the thermal and mechanical behaviour of a fuel rod during normal and transient (not accident) conditions using finite element method. It can analyze the integral behavior of a whole fuel rod through its life as well as the localized behavior of a small part of the fuel rod. Temperature distribution, radial and axial deformations, fission gas release, and inner gas pressure are calculated as a function of irradiation time and axial position. Stresses and strains in the pellet and cladding are calculated and PCMI analysis is performed. Also, thermal conductivity degradation of pellet and cladding waterside oxidation are modeled. Figure 23 describes the FEMAXI mechanical analysis system in a cylindrical geometry.

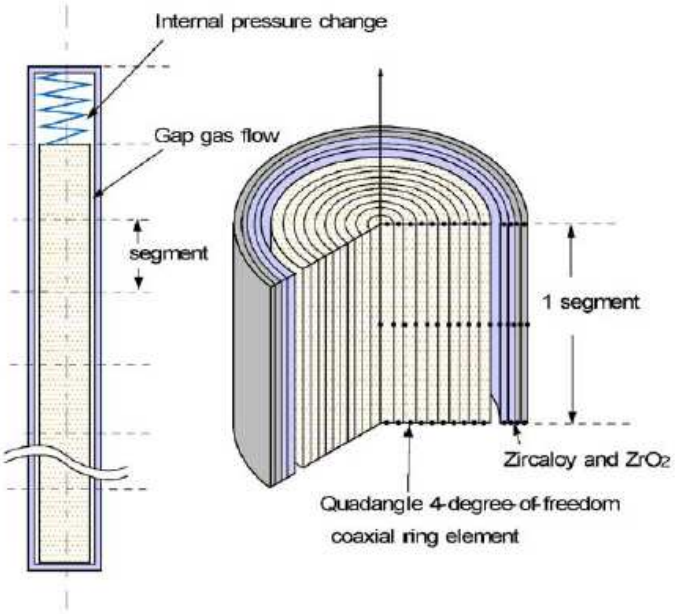


Fig. 23: Mechanical analysis using cylindrical elements in FEMAXI-6 [56]

The code is validated up to high burnup region (around 80 MWd/kgU). However, it does not cover such accident conditions as RIA and LOCA. Figure 24 shows schematics of thermal analysis model, where one single rod is divided into several axial segments. At each segment linear heat rate is provided and temperature profile is calculated using burnup-dependent material properties.

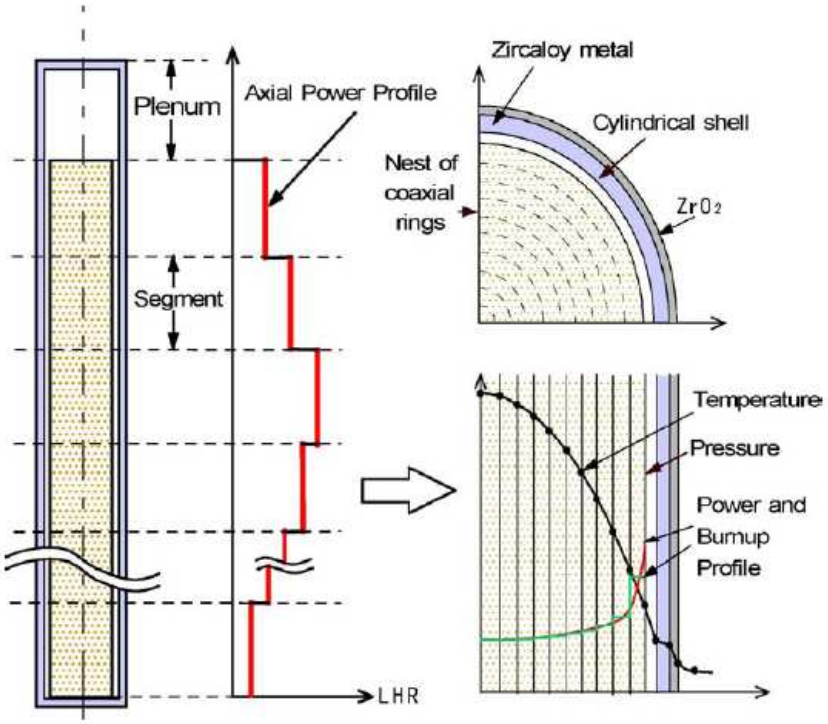


Fig. 24: Geometrical model for thermal analysis in FEMAXI-6 [56]

Power profiles in the radial and axial directions of a pellet and the burnup profile both change with burnup from Beginning of Life (BOL). These profiles affect the temperature distribution of pellets and also affect the material properties, the performance of which depends on the burnup. FEMAXI-6 includes a function that reads the output files of the burning analysis codes, RODBURN and PLUTON. In the simulations performed the code used was PLUTON. The relation between the different programs is shown in Fig. 25.

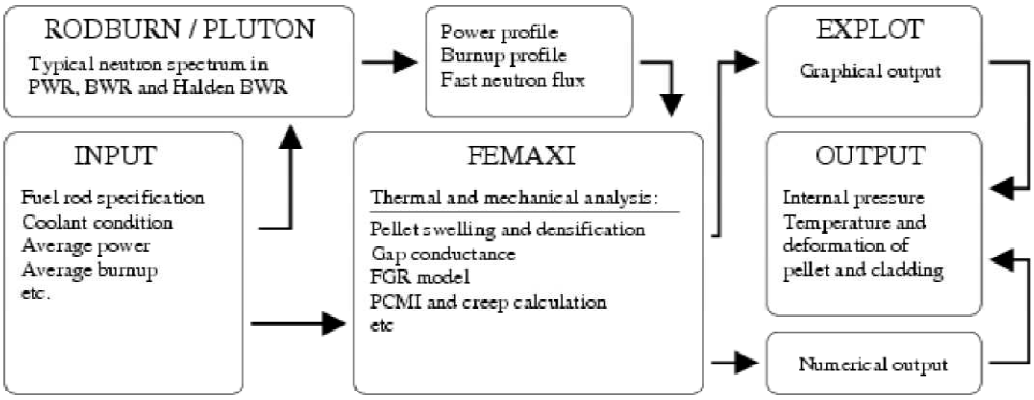


Fig. 25: FEMAXI-6 analysis system [57]

PLUTON is a three-group neutronic code which analyzes, as functions of time and burnup, the change of radial profiles, together with average values of power density, burnup, concentration of trans-uranium elements, plutonium buildup and fission product generation. The code has been verified up to 83 MWd/kg with satisfactory agreement and can handle $ThO_2 – UO_2$ fuels [57].

4.2 Input and output parameters

In this section the parameters needed to model the fuel rod are described. The approximations made are explained regarding the operational parameters and conditions and also the models used to simulate the physical behavior of thorium.

Input

FEMAXI-6 allows to choose between two cladding options: either Zircaloy-4 (IRM=0) or stainless steel SUS304 (IRM=1). During the simulation, the Zircaloy-4 cladding was chosen due to the impossibility to find reliable data about other cladding materials considered like M5.

As mentioned before, the operating conditions are those of a conventional PWR. FEMAXI-6 input file asks for the main parameters of the reactor. In this case time, linear heat rate, coolant temperature, coolant pressure and coolant velocity. The majority of those values were kept constant during the rod history, near the typical average values of a PWR.

- Time

The objective was to model the effect of higher burnup and longer residence time in the reactor so the blanket part was chosen. The time of residence considered is 10 years divided in 41 time steps of 2191.5 hours.

- Linear heat rate

For this parameter some assumptions were made, due to the lack of data, in order to approximate this value to the one that might be obtained in a conventional PWR using the WASB configuration. In a typical PWR using this configuration, the number of assemblies is 193 from which 111 are filled with thorium fuel, being the blanket part of the assembly, cf. Fig. 26. The discharged burnup of the blanket part is about 88 MWd/kg.

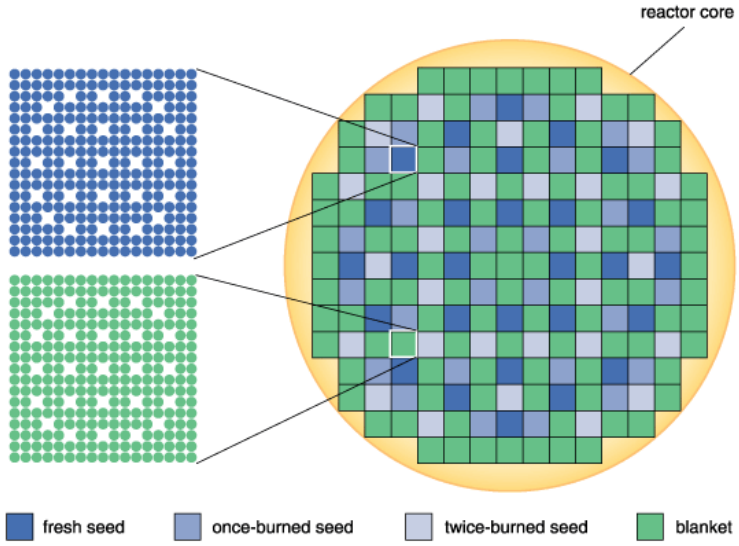


Fig. 26: WASB configuration [16]

Knowing the number of rods per assembly, the geometrical characteristics of the fuel rod, the time of residence and the density of thorium it is possible to find the average linear heat rate. The expressions and values used are listed in Table 21.

Symbol	Description	Value
N	Number of thorium fuel assemblies	111
F	Fuel rods per assembly	264
L	Fuel column length	4267 mm
d	Pellet diameter	8.2 mm
ρ	Density	10 g/cm^3
t'	Refueling time[4]	90 days
Bu	Burnup	88 MWd/kg
t	Time of operation of the reactor	
h	Heat rate	
P	Power	
M	Mass of fuel	

Table 21: List of symbols

The expressions and the calculations made to determine the average linear heat rate are presented. The time of operation of the reactor regarding the blanket part is considered to be of 10 years minus the refueling time.

$$t = 10 \text{ years} \cdot \frac{365.25 \text{ days}}{1 \text{ year}} - t' = 3562.5 \text{ days}$$

To determine the power of the reactor, the following expression is used:

$$Bu = \frac{P \cdot t}{M}$$

With the geometrical parameters and the density it is possible to determine the total mass of thorium in the reactor:

$$A = \pi r^2 = 0.5281 \text{ cm}^2$$

$$V = A \cdot L = 225.34 \text{ cm}^3$$

$$M_{\text{fuel rod}} = V \cdot \rho = 2.25 \text{ kg}$$

$$M_{\text{total}} = M_{\text{fuel rod}} \cdot N \cdot F = 2.25 \cdot 111 \cdot 264 = 66033.93 \text{ kg}$$

$$M_{Th} = M_{\text{total}} \cdot \frac{A_{Th}}{A_{ThO_2}} = 66033.93 \cdot \frac{232}{264} = 58.03 \text{ t}$$

[4] The refueling time considered was 30 days each 3 years, due to the replacement of the seed rods. However, it is an optimistic estimation

Using the operating time of the reactor, the mass of thorium and the burnup, the power of the reactor is found:

$$88\ \frac{\mathrm{MWd}}{\mathrm{kg}} = \frac{\mathrm{P} \cdot 3562.5\ \mathrm{d}}{58029.82\ \mathrm{kg}} \quad \rightarrow \quad \mathrm{P} = 1433.44\ \mathrm{MW}$$

Dividing this value with the rod length of all the assemblies the average linear heat rate is obtained:

$$\mathrm{h} = \frac{1433.44 \cdot 10^6}{264 \cdot 111 \cdot 426.7} = 114.64\ \frac{\mathrm{W}}{\mathrm{cm}}$$

With this value of reference and using the linear heat rate data for a PWR, it is possible to approximate the operation characteristics of a real light water reactor using thorium as a fuel. It is important to consider those variations in the linear heat rate as a result of the movement of the fuel rod to a different location at each reload. Figure 27 shows the profile used to adapt the variation of the heat rate.

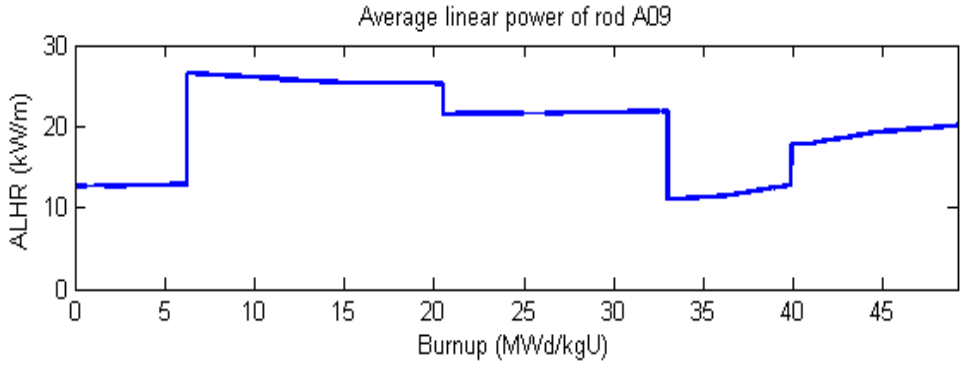


Fig. 27: Linear heat rate for PWR [58]

The axial heat rate profile of the fuel rod is considered to follow a cosinus, although it would be more correct to consider the effect of the reflectors. 16 axial segments were used to ensure enough accuracy and the axial profile was kept constant for all the history time points.

- Coolant temperature, pressure and velocity

Three parameters are considered constant in all the historical time points: the average coolant temperature is 579 K, the pressure 15.5 MPa and the velocity 3.8 m/s. All those values are representative to common PWRs [24]. The operating conditions considered during the simulation are shown in Table 22.

Operating Conditions	
Time of residence	10 years
Temperature	579 K
Pressure	15.5 MPa
Coolant velocity	3.8 m/s
Burnup	88 MWd/kg
Linear heat rate	115 W/cm

Table 22: Operating conditions considered for a PWR

FEMAXI-6 is not adapted to the use of $ThO_2 – UO_2$ as fuel and its adaptation needs gathering and implementing necessary material properties. This work was previously done by Vanhanen [57], although open literature cannot currently provide all properties needed in FEMAXI and thus it is not possible to completely adapt FEMAXI to $ThO_2 – UO_2$ fuels. The approximations and considerations made during the simulations and during the implementation of thoria properties are listed in the Appendix.

In the work of Vanhanen, two thermal conductivity models were implemented. Those are Belle and Berman and Bakker et al.[57], which differ in the temperature range of validity and on the value of the coefficients used to fit the lattice conduction. The value of the lattice conduction of the material is inversely proportional to the temperature. The selection of one model or another determines how other parameters behave due to the interdependence of many properties on temperature.

The validity of the two models was contrasted against two thorium rods that were irradiated in the Halden reactor in Norway. From the results obtained in “Implementing thorium fuel option to fuel performance code FEMAXI-6” it can be concluded that the model that fits better is the Bakker et al.. It is important to point out that the accuracy of the model depends on the burnup degree of the fuel and that it has only been compared to the data obtained from the Halden reactor.

Probably the parameter that influences most on all the other physical phenomena is the temperature. The results of Vanhanen of fuel temperature as a function of burnup seems to suggest that despite Bakker’s correlation overestimates temperature, from about 100 ºC to 150 ºC, until 23.5 MWd/kg the error is acceptable [57]. Once again these values of reference only can be applied to that particular model and conditions, which have not been validated. Using the program EXPLOT it is possible to plot the main parameters obtained from FEMAXI output. Regarding the great amount of parameters which affect the cladding and the impossibility to discuss them all, this work will only take into account two of them, the inner pressure of the cladding and the zircaloy outer corrosion.

4.3 Results and discussion

The results obtained are showed in this section. FEMAXI generates a great amount of data, from which only a small part is used in this work. Figure 28 shows the average burnup reached and the residence time of the rod in the reactor.

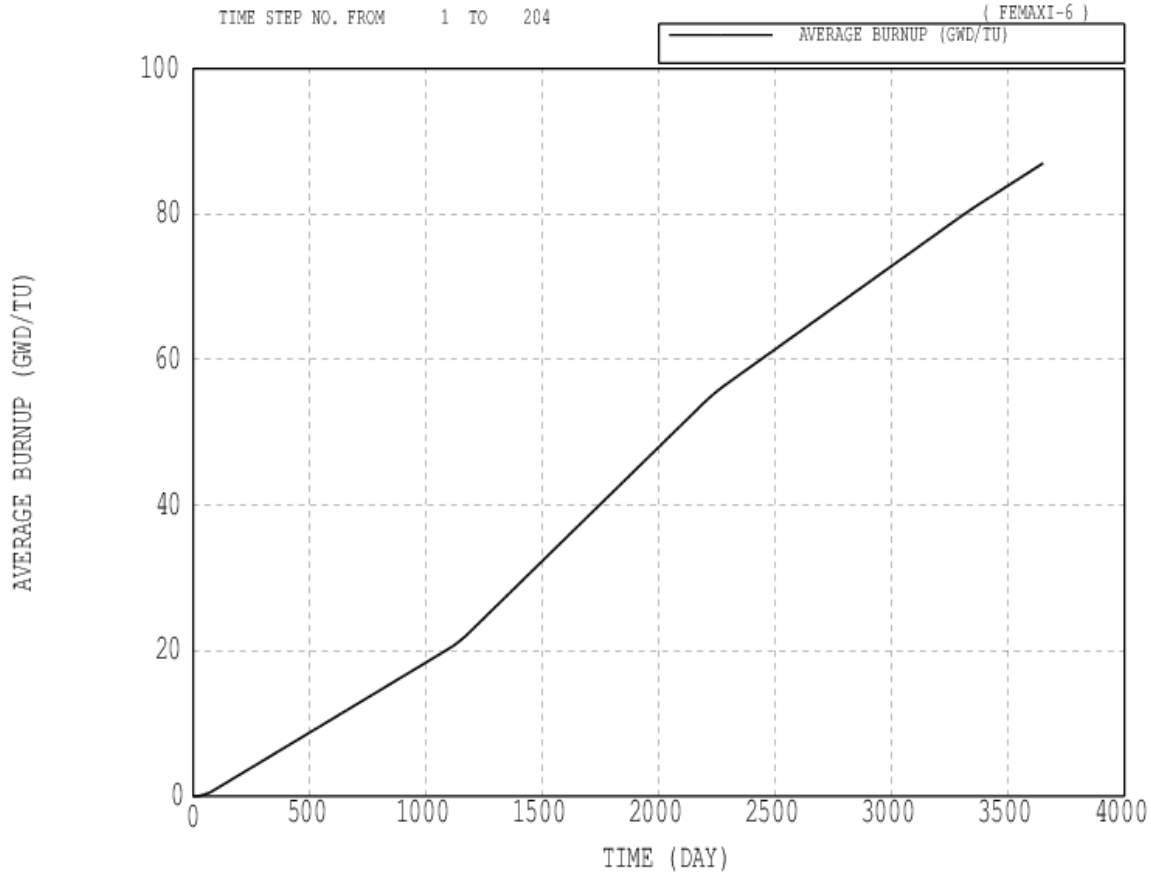


Fig. 28: Average burnup for thorium fuel as a function of residence time of the rod in the reactor calculated with FEMAXI

Inner pressure of the cladding

One of the problems which the cladding has to face at high burnups is the pressure, and the factors that affect this parameter are mainly the generation and release of fission gas. The equations that describe these phenomena are implemented in FEMAXI, but in the case of thorium dioxide, the problem to find open data about thorium led to the use of some properties of uranium. The models used in both cases can be found the FEMAXI manual [59].

- Generation rate

The expression used to calculate the generation rate is the following [59]:

$$P^{ij} = \frac{Y \cdot f^{ij} \cdot q^{i}}{E_f \cdot N_A} \quad (4.1)$$

$$f^{ij} = 2\pi \int_{-0,5}^{+0,5} \phi(r) r dr \quad (4.2)$$

Symbol	Description	Value
P^{ij}	Fission gas rate per unit length in element ij ($mol/cm^3 s$)	
$\phi(r)$	Heat generation profile function in the radial direction	
q^{j}	Average heat generation density of axial segment j (W/cm^3)	
E_f	Energy generated per one fission	$3.024 \cdot 10^{-11}$ J (200 MeV)
Y	Fission yield of fission gas (Xe+Kr)	0.3
N_A	Avogadro's number	$6.02 \cdot 10^{23}$

Table 23: List of symbols

As can be seen, the generation rate varies according to the heat density profile and with the average heat generation density. The average heat generation density depends on the linear heat rate of an axial segment, which is given in the input file of FEMAXI, while the heat generation density profile is proportional to the thermal neutron flux, which is given by PLUTON. PLUTON code used properties of thorium, making the generation rate of fission gas a good approximation for ThO_2 –UO_2 modeling.

- Fission gas release

The diffusion coefficient model, which determines the relocation and movement of gas bubbles in a pellet, uses UO_2 properties. The diffusion equation in the intra-granular spherical coordinate system including trapping is [59]:

$$\frac{\partial \Psi}{\partial t} = D' \cdot \left(\frac{\partial^2 \Psi}{\partial^2 r^2} + \frac{2}{r} \frac{\partial \Psi}{\partial r} \right) + \beta \quad (4.3)$$

$$\beta_{ij} = \frac{P^{ij} \cdot N_A}{\pi (r^2_{i+\frac{1}{2},j} - r^2_{i-\frac{1}{2},j})} \quad (4.4)$$

Where P_{ij} is the fission gas release per unit length (4.1). The values used in expressions 4.3 and 4.4 are listed in Table 24.

Symbol	Description
Ψ	Apparent average number of gas atoms per unit intergranular volume
D'	Diffusion coefficient of gas atoms (cm^2/s)
$r_{i+\frac{1}{2},j}$	Outer radius of pellet region i,j (cm)
$r_{i-\frac{1}{2},j}$	Inner radius of region i,j (cm)

Table 24: List of symbols

From the expressions used in both cases it can be seen that they are interrelated. This correlation shows the dependence of these models form thorium dioxide and uranium dioxide properties. Both materials have the same crystal structure, which is cubic fluorite. Due to this fact, they have similar properties and this is the reason that makes Vegard's law [57], explained in the Appendix, applicable to estimate ThO_2 properties from UO_2. Figure 29 and Fig. 30 show the average fission gas release and plenum pressure depending on the burnup degree obtained in the simulation.

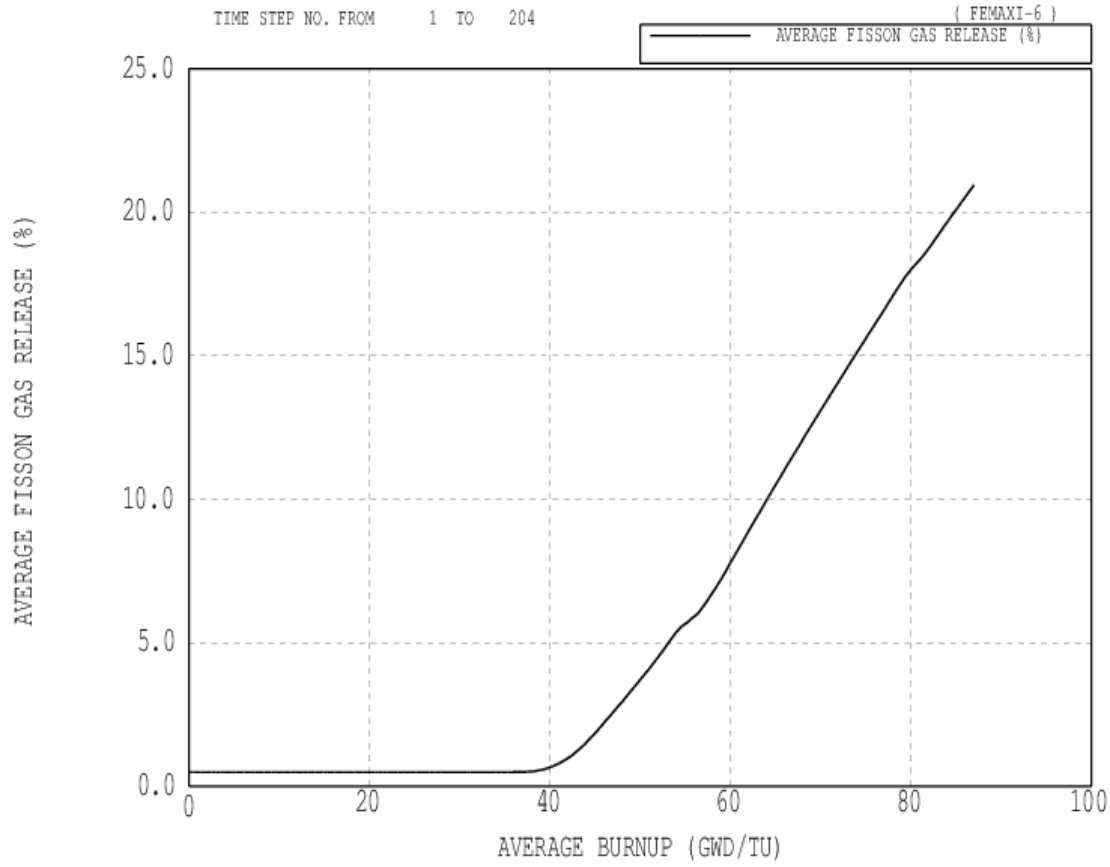


Fig. 29: Average fission gas release for thorium-uranium fuel as a function of burnup calculated with FEMAXI

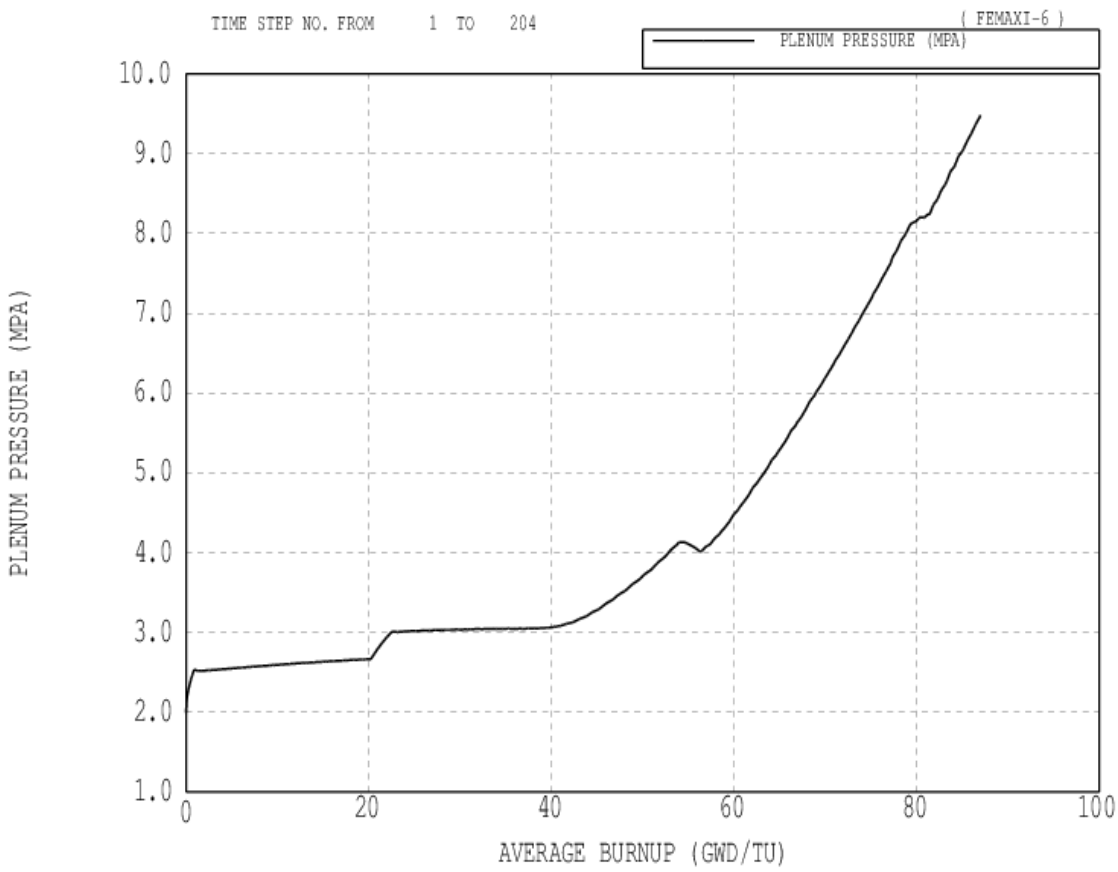


Fig. 30: Plenum pressure for thorium-uranium fuel as a function of burnup calculated with FEMAXI

The results obtained in the simulation of the blanket part of the assembly using ThO_2-UO_2 as fuel can be considered an acceptable approximation in the range of validity of FEMAXI. It can be seen that at a burnup of about 40 GWd/t there is an increase in the fission gas release and in the plenum pressure.

The internal pressure of the rod has effects on its resistance to creep, to buckling and to ballooning. Under normal reactor operation this limit must be lower than the limit leading to dimensional instability or heat transfer impairment. This criterion is such that the hot internal pressure, due to the accumulation of the pressure of the filling helium of the new rod and that of the fission gas released during irradiation, must be lower than the pressure which would cause, during normal operation, the reopening of the diametrical gap between the pellets and the cladding, by tensile creep of the cladding [24].

Despite this increase in the plenum pressure of the fuel rod, the value is still under the critical margins of operation and licensing. The internal pressure of the rod should not exceed the pressure of the coolant in order to avoid the swelling of the cladding. Some properties of un-irradiated Zircaloy-4 are listed in Table 25 to have a general reference of the values. The objective of reaching burnups of 60 MWd/kg seems possible and evens the expected burnups of 88 MWd/kg or higher would not compromise the integrity of the cladding with the pressure generated by fission gases.

Physical Properties	Value	Comments
Density	6.56 g/cm^3	
Mechanical Properties		
Tensile Strength, Ultimate	514 MPa	Room Temperature, Transverse
	541 MPa	Room Temperature, Longitudinal
	241 MPa	288ºC, Transverse
	271 MPa	288ºC, Longitudinal
Tensile Strength, Yield	381 MPa	Room Temperature, Longitudinal
	467 MPa	Room Temperature, Transverse
	152 MPa	288ºC, Longitudinal
	177 MPa	288ºC, Transverse
Modulus of Elasticity	99.3 GPa	
Poisson Ratio	0.37	
Shear modulus	36.19 GPa	
Thermal Properties		
Thermal conductivity	21.5 W/mºC	
Melting Point	1850 ºC	

Table 25: Zicraloy-4 properties [60]

Outer oxide thickness

The corrosion effect is one of the major issues which the blanket part of the assembly has to face, due to the long residence time in the reactor, and is one of the most studied processes in all the nuclear industry.

The equation that defines the cladding corrosion rate and that can be found in FEMAXI manual [59] is the following:

$$\text{Pre} - \text{transition corrosion rate: } \frac{dS}{dt} = \left(\frac{A}{S^2}\right) \cdot e^{\left(-\frac{Q_1}{RT}\right)} \quad (4.5)$$

$$\text{Post} - \text{transition corrosion rate: } \frac{dS}{dt} = C \cdot e^{\left(-\frac{Q_2}{RT}\right)} \quad (4.6)$$

$$C = C_0 + U \cdot (M \cdot \phi)^P \quad (4.7)$$

$$\text{Oxide layer thickness at transition: } S_t = D \cdot e^{\left(-\frac{Q_3}{RT-ET}\right)} \quad (4.8)$$

Symbol	Description	Units
dS/dt	Corrosion rate	μm/day
S	Oxide layer thickness	μm
T	Temperature at oxide - metal interface	K
φ	Fast neutron flux	n/cm^2s

Table 26: List of symbols

Symbol	Value	Units
R	1.987	cal/mol·K
A	$6.3 \cdot 10^9$	μm^3/day
Q_1	32289	cal/mol
Q_2	27354	cal/mol
Q_3	10763	cal/mol
C_0	$8.04 \cdot 10^7$	μm/day
U	$2.59 \cdot 10^8$	μm/day
M	$7.46 \cdot 10^{-15}$	cm^2s/day
P	0.24	
D	$2.14 \cdot 10^7$	μm
E	$1.17 \cdot 10^{-2}$	K^{-1}

Table 27: List of symbols

From equations 4.6, 4.7 and 4.8 it is possible to verify that the parameters which affect the formation of the oxide layer in the zircaloy cladding are the temperature at the oxide-metal interface and the fast neutron flux.

The average fast neutron flux is given by PLUTON, using thorium properties. Those conditions make the results obtained in the simulation an acceptable approximation to the expected corrosion behavior in the cladding. Figure 31 shows a plot of the results obtained during the simulation of this phenomenon.

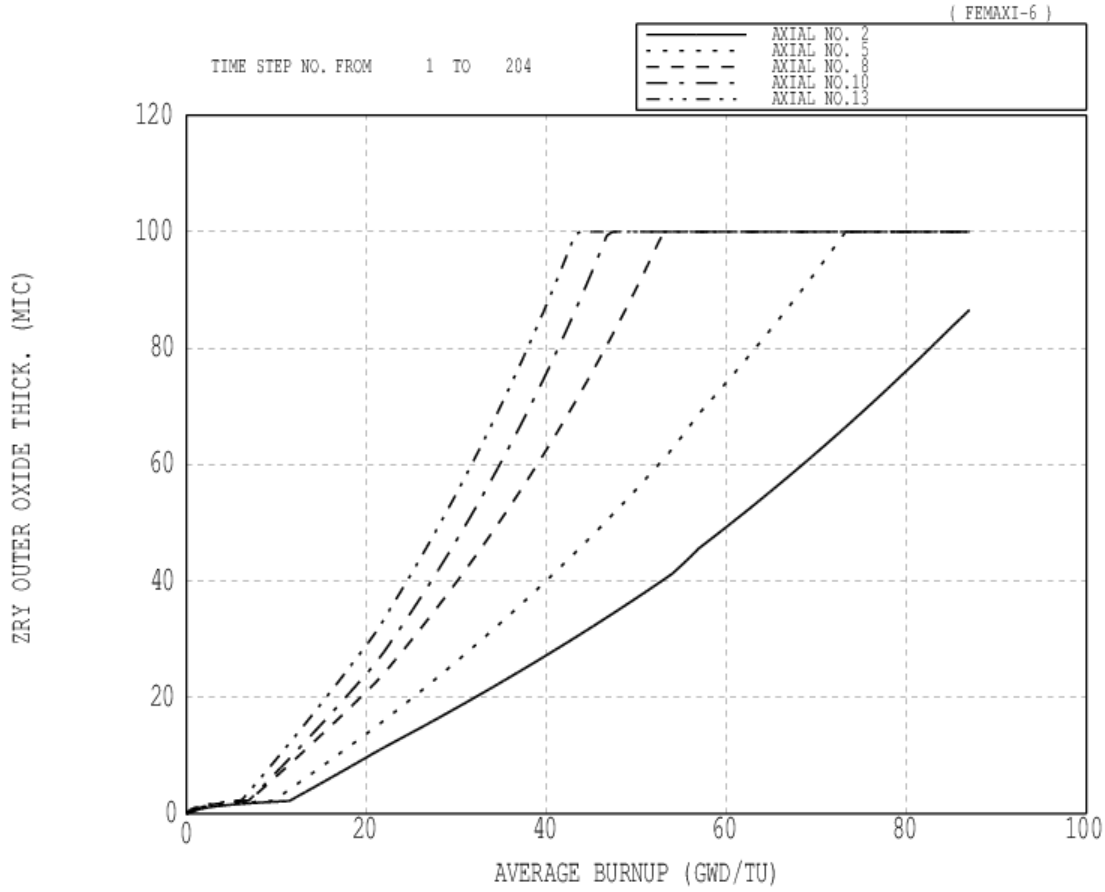


Fig. 31: Average burnup and Zircaloy-4 outer oxide thickness for thorium fuel calculated with FEMAXI

Regardless of the type of oxidant (oxygen, water, steam, CO, etc) to which zirconium or its alloys such as Zircaloy-2 and Zircaloy-4 are exposed, the general behavior of the process is more or less the same. The initial oxidation process is referred to as “pre-transition” and lasts until the oxide layer reaches a thickness of approximately 2 to 3 μm. After the oxide layer has grown to that point, the oxidation process enters the “post-transition” phase where the density of the layer decreases and becomes less protective [61].

From the results obtained, presented in Fig. 31, it seems that the pre-transition phase stands until about 10 MWd/kg. For high burnup the post-transition phase starts, compromising the protection of the cladding against the oxidants. Despite this process, a limit of 100 μm is generally applied, and if the oxide layer is allowed to grow above this, the protective oxide layer breaks down. Studies suggest that this limit is reached in standard Zircaloy-4 cladding at an average burnup of around 45 MWd/kg [62]. Figure 31 shows that the burnup in which the oxidation limit is reached is also about 45 MWd/kg. This agreement in the expected results and the values obtained give certain validity to the model used.

4.4 Summary

The possibility to model thorium fuel using FEMAXI-6 is briefly described in this section. The thorium properties needed to simulate the fuel rod behavior in the reactor were previously implemented in another work [57]. The interdependence of many properties makes simulating a challenge and causes uncertainty of the results.

The analysis of the models and the parameters used show that there is a lack of open literature about data of thorium properties and that more research should be done. Current correlations cannot satisfactorily reconstruct the temperature of an irradiated thorium fuel rod. The results obtained are only approximate as regards expected reactor operating conditions.

The two effects which, in the opinion of the author, have a major influence in the rod behavior and at the same time were the best estimation and had less uncertainty are the inner pressure of the cladding and the zircaloy outer corrosion. The equations for each effect are included in order to show the interdependence of the values and also to discuss the validity of the model, depending on the properties used.

The results obtained for both effects provide with two different conclusions. On one hand, the results from the inner pressure of the cladding show that this effect would not be a major issue to reach high burnups, about 88 MWd/kg. On the other hand, corrosion of the cladding seems to be a limiting factor for high burnups and longer residence time. The maximum burnup in order to avoid the formation of the post-transition phase is about 10 MWd/kg but the limiting burnup is about 45 MWd/kg. The limit in the oxidation process might be a limiting factor in order to reach the operating conditions expected for the blanket of the assembly.

More research must be done in order to validate these models, to gather more reliable properties and to ensure a better estimation of the physical and chemical processes in the reactor when using ThO_2–UO_2.

5. Fuel licensing

The increase of efficiency in the production of nuclear electricity has been one of the goals for many countries which have developed this energy source. Increase of the fuel burnup has been the approach taken to improve the efficiency. It leads to a reduction in the volume of the spent and discharged fuel due to the longer fuel cycles in the reactor. In order to be able to increase the burnup, the design and manufacture of the fuel, which is the source of the majority of radioactivity produced and the element that faces the most critical conditions during operation, has to be sufficiently robust and safe. The fuel rods must operate under normal conditions without incidents but also withstand any transient or accident which could occur in the plant. This is ensured through a licensing process that accounts not only the operation but also the design and manufacture of the nuclear fuel, which is carried out to extremely high standards and based on experimental validation studies.

Although this document focuses in the use of thorium as a fuel, the licensing process and the critical issues that the fuel rod faces are similar for both thorium and uranium fuel cycle. For that reason this section will offer a general view about the licensing process without focusing into a particular fuel cycle. The main difference between the two fuel cycles is the higher burnup of the thorium fuel rods which makes its licensing process complex.

Nowadays for both WWERs and PWRs, the strategy proposed in terms of core management and fuel design evolution is based to reach a burnup of about 60 MWd/kg [63]. Efforts are directed to improve the cladding and pellet materials, because of the more challenging conditions to be faced and the higher neutron flux prevailing.

Several technical committee meetings have been held, in which experts from different countries discuss and share their experience and studies relating higher burnups. These studies are taken in order to estimate the limitations from a physical and economical point of view. The recommendations drawn from these meetings include [63]:

- As a consequence of the variety of core designs (PWR, BWR, WWER, PHWR), the conditions of the fuel rods are different. This means that the qualifications for fuel management must be conducted for each of these separately.
- It is necessary to improve the chemistry modeling at high burnup. The fuel chemistry controls the local fuel properties evolution.
- The evaluation of the local hydrostatic pressure must be improved, as far as it appears to be a key parameter in the fission gas modeling.
- Efforts must continue towards and understanding of the rim formation mechanisms and on its fission gas retention capability during RIA experiments.
- Research is needed on alternative materials for the fuel and the cladding or on alternative assembly designs.
- More investigations are required in order to identify basic mechanisms, thereby enabling the extension of the validity domain of the existing models or the development of more mechanistic models.

The overall criterion for licensing is that under all conditions the fuel damage must remain below a small number of fuel rods. The need to demonstrate the operation under safety margins makes that any improvement to the fuel or the cladding design (fuel and cladding composition, enrichment, etc) and operation is carefully considered and implemented incrementally. All these changes are tested with extensive experimental demonstrations which take a great amount of time and resources. The incremental approach to burnup extension has been a feature of nuclear fuel development as limitations on burnup extension have been identified and overcome.

The increase in the fuel burnup is done by increasing the enrichment level due to the proportional relation between the burnup of the nuclear fuel and the level of enrichment. The limiting factor to increase the enrichment of the fuel has been the physical robustness of the fuel assemblies, and hence burnup levels of about 40 GWd/t have required only around 4% enrichment. But with better equipment and fuel assemblies, 55 GWd/t is possible (with 5% enrichment), and 70 GWd/t is in sight, though this would require 6% enrichment [64]. Figure 32 shows how the increase of burnup has been continuous as nuclear power developed.

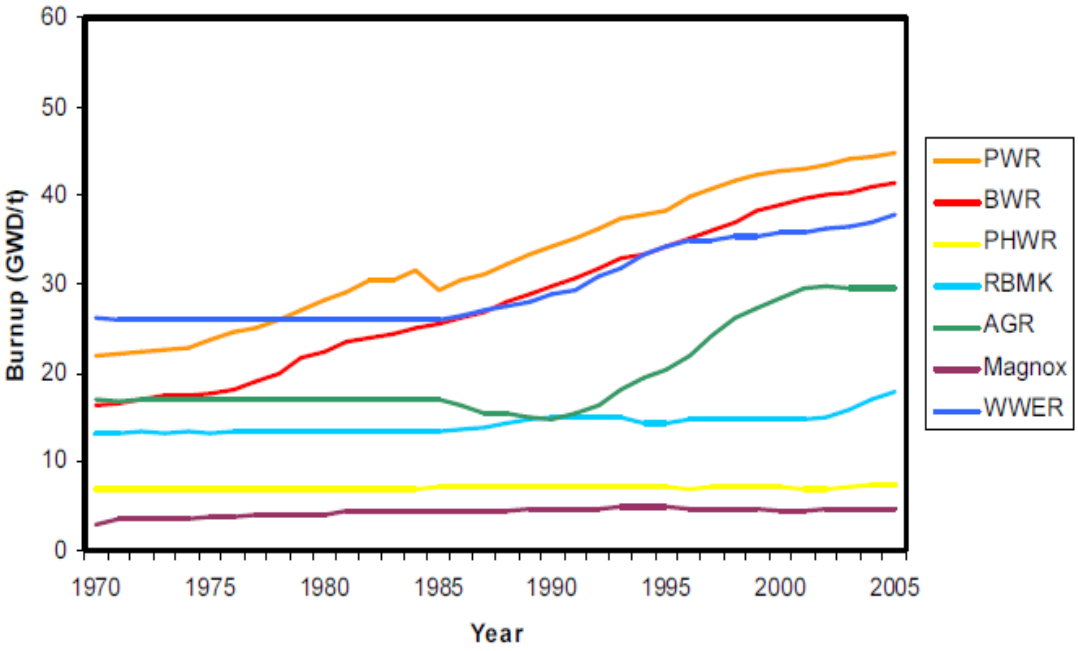


Fig. 32: Burnup over time for different types of reactors [64]

Some of the problems in the licensing process relating the technical limits of fuel and cladding, the enrichment limit or the economic limit are discussed below.

5.1 Technical limits

The technological challenges posed by increased burnup are mainly related to the corrosion and hydrogen pick up of the clad, the properties of the fuel and the dimensional changes of the fuel assembly structure due to fission gas production.

The high burnup behaviour of the fuel has been extensively investigated and the decrease of thermal conductivity with burnup, the rim effect of the pellet and the increase of fission gas release can be described, with acceptable accuracy, in fuel rod computer codes.

Most of the questions about the fuel operational behaviour and reliability in the high burnup range have been solved. Some of them are still in the process of verification or the solutions are visible. This fact is largely acknowledged by regulators too. The main licensing challenges for high burnup fuel are currently seen for accident condition analyses, especially for RIA and LOCA [65].

Studies and developments in the fuel rod and core materials have been carried on, which seem to show further technical potential to increase discharged burnup. Advanced fuel assemblies offer sufficient margins to be used with higher enrichments and with increasingly heterogeneous fuel rods. Cladding and structural materials have been developed and irradiated to burnups far above today's average values. The majority of these materials and tough conditions have been tested in research reactors, where values of 160 GWd/tU or above were achieved. In fact, it is believed that current technology could support LWR burnup up to 100 GWd/tU [62]. In the particular case of thorium fuel rods this value would allow to use current assemblies as the blanket part of a Radkowsky Thorium Reactor.

It is necessary to point out though, that the integrity of the fuel and cladding materials under high burnups has been tested in research reactors. The different conditions of operation in power reactors make it necessary to do further research and testing.

Despite these favorable technical conditions to increase it, the 5% enrichment still represents a worldwide established limit for fabrication, transport and storage for nuclear fuel for light water reactors. All the critical assessments have been made on the basis of a maximum enrichment level of 5% ^{235}U. For this reason any change in the fuel enrichment must be deeply studied before it can be applied, because it implies a great modification on the operating parameters and conditions of many reactors.

The experimental programme of the Halden Reactor Project (HRP) has focused on high burnup effects. The aim of this test programme is to identify the effects that high burnup has in the fuel rod, its behavior under these conditions and possible solutions. Some of the objectives of the test programme include [73]:

- Extend the data base of UO_2 fuel performance.
- Asses the influence of fuel microstructure on pellet-cladding mechanical interaction (PCMI) and fission gas release in medium and high burnup fuel.
- Investigate integral fuel rod behaviour at high burnup.
- Investigate rim effects.

Fuel fabrication

Higher enrichment means that fuel rods operate with a high power density with a very demanding environment of high radiation field, high temperature and high coolant flow. The first fuel and cladding designs were adequate for the initial low burnups, where the time that the fuel was in the reactor was limited, but as burnup increased it has been necessary to improve the fuel rods. The ways to do it listed below must also take into account the fact that material properties change with time under intense radiation fields in the reactor. All these improvements are necessary in order to enhance safety and to face the licensing tests [62].

The critical issues of the cladding have been previously discussed and they become a more critical element due to the higher burnup of thorium fuel compared with uranium fuel.

- Cladding oxidation

One of the limits on PWR operation is the cladding oxidation. A limit of 100 μm is generally acceptable and even protects the metal from further corrosion, but if the oxide layer grows thicker it breaks down. This limit is reached with standard Zircaloy-4 at an average burnup of around 45 GWd/tU. For this reason several studies have been made changing the composition of zirconium based alloys and testing other cladding options, in order to increase oxidation resistance. The introduction of new alloys containing 1%Nb and new cladding options as M5 or duplex cladding has led to a major reduction in oxidation. The 100 μm limit is not expected to be reached at the target burnup of 100 GWd/tU or even above [62].

The time needed to introduce these new alloys is slow and reflects the precaution taken in introducing new materials into nuclear fuel. The reasons are due to the necessary testing of the new materials in a reactor. Such tests are very expensive and take many years to carry out. It takes typically six years for a lead test assembly of a new fuel variant to reach the extended burnup necessary before experimental testing can even start.

- Fission gas release

Fission gases constitute a big problem for increased burnup. Fission gas is generated within the fuel during operation and the amount produced is roughly proportional to the burnup. The release of fission gas from the fuel pellet causes an increase in the pin pressure, changes in the fuel dimension, cladding expansion and a decrease in the gap thermal conductivity. For the thorium fuel this problem is even more critical due to the higher fission gas production compared with uranium fuel.

One of the options considered to face this issue is to add dopants to the fuel pellet to control the microstructure and to reduce the release rate during operation. Current WWER fuel pellets have annular geometry, with a central hole providing lower centre temperatures and a free volume to allow fission gas to expand and thereby reduce internal pressure. Relating the physical phenomena, a high burnup structure (HBS) with high porosity develops on the rim of the pellet, affecting fuel temperature distribution as the pellet burnup exceeds 45 GWd/tU [62].

- Water chemistry

Water chemistry has a great influence on the corrosion rate and on the mechanical behavior of the cladding. As a result of high burnup the residence time in the reactor is longer and the mechanisms that affect water, as radiolysis, are more intense. The recommendations for water chemistry have evolved over the years. In Fig. 33 it can be seen that the major events started with the introduction of lithium as pH controller in the 1980s, and more recently zinc addition for steam generator corrosion control followed by elevated pH.

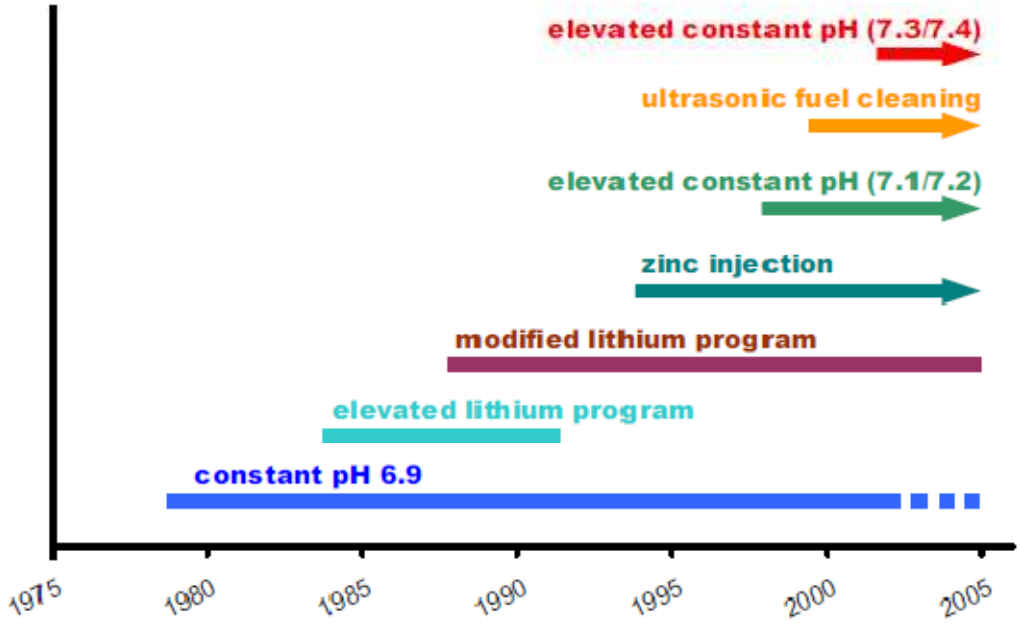


Fig. 33: Evolution of recommended PWR water chemistry [62]

- Rim structure

As the burnup increases at the periphery of a pellet, a microstructure begins to form with subgrains (reducing the size of the grains) and gas bubbles along grain boundaries. This effect results in larger releases of fission products and in a stronger pellet-cladding interaction (PCI) effect.

The best way to solve this problem is to use fuel with bigger grains and an optimized pore structure, capable of retaining FGP inside the pellet. Some analyses have shown that in order to reduce fission gas release as the burnup increases the mean grain size has to be not less than 20 μm and the volume fraction of open pores has to be not more than 0.5% (or not less than 10% of the total quantity) [66]. The problem to this approach is that standard fuel has grains of about 10-15 μm in size and no technological procedures are good enough to increase it without inflicting significant economic losses.

The approach taken to solve this inconvenience is the application of microadditives that accelerate the growth of grains, although it also raises some concerns relating its impact in the reactor physics and its effect in other properties of the fuel.

Finally, one of the options considered for innovative fuel design consists of uranium metal embedded in a zirconium-hydride matrix. The main advantages of this design relative to oxide fuel are the higher thermal conductivity, higher retention of fission products and the flat neutron flux shape inside the pellet due to the moderating effect of the hydrogen. The problem that it presents is the higher irradiation swelling at temperatures exceeding 700 ºC and the hydrogen-embrittlement of the zirconium cladding from the inside if pressed against the pellet. One possible solution to this problem is to limit the fuel pellet temperature below 650 ºC without limiting the linear heat rate, and to fill the cladding-to-pellet gap with a liquid metal (lead-tin-bismuth) to reduce the thermal resistance across the gap and to protect the cladding from hydrides [66].

Enrichment limit

Uranium processed for electricity generation is not applicable to nuclear weapons. The uranium used in power reactor fuel for electricity generation is typically enriched to about 3-4% of the isotope ^{235}U, compared with weapons-grade which is over 90% ^{235}U. For safeguards purposes uranium is deemed to be "highly enriched" when it reaches 20% ^{235}U [67]. Only a few countries possess the technological knowledge or the facilities to produce weapons-grade uranium.

The concept of low-enriched uranium (LEU) is used by the International Atomic Energy Agency (IAEA) to define “enriched uranium containing less than 20% of fissile material”. The IAEA classifies LEU as a so-called indirect use material, which in turn is defined as a nuclear material that cannot be used for the manufacture of nuclear explosive devices without transmutation or further enrichment [68].

Regarding the use of the fuel to produce nuclear weapons, there are two possible scenarios which have to be considered. The first one assumes that the fuel used to build a nuclear device is uranium recovered from the spent fuel, while the respective plutonium inventory is discarded from further use. The quantity of fissile material used as a reference in these assumptions is noted as M_B, which is about the amount of fissile material needed for a weapon. The reduction in the utility of uranium with reduced ^{235}U is taken into account applying the factor η_1 to the critical mass ratio m/ M_B. The total strategic value CM of the material extracted for its use as weapon is defined as follows [68].

$$CM = \eta_1(\epsilon_{FF}) \cdot \frac{m_{FF}}{M_B(\epsilon_{FF})} + \eta_1(\epsilon_{SF}) \cdot \frac{m_{SF}}{M_B(\epsilon_{SF})} \qquad (5.1)$$

The sub indexes FF and SF of mass (m) and enrichment level (ϵ) refer to the uranium contained in the fresh and the spent fuel, respectively. The strategic value for different enrichments is shown in Fig. 34. It can be seen that when the fuel is enriched beyond 20% its capacity as a weapons fuel increases.

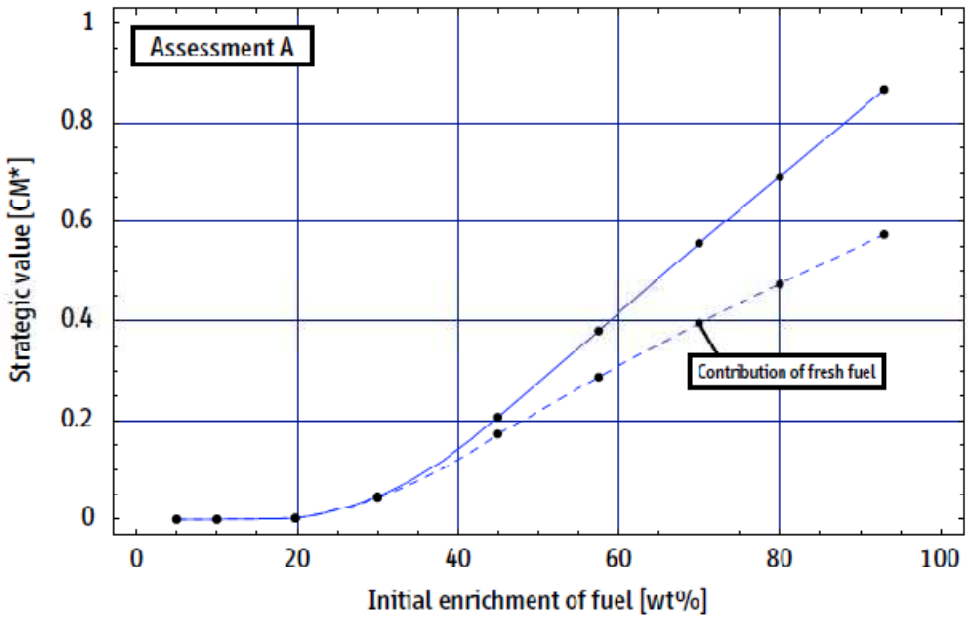


Fig. 34: Strategic value of fissile material assuming that only uranium is extracted from the fresh and irradiated fuel [68]

Declining the enrichment levels makes the strategic value to decrease for two reasons: the critical mass ratio m/ M_B and the usability factor η_1 drop simultaneously.

The second scenario considers the possibility that both uranium and plutonium are extracted and used for weapon purposes. Although the formula to estimate the strategic value varies from the first case in which only uranium would be used, the conclusions that can be extracted are the same. Figure 35 shows the relation between the strategic value and the enrichment, and it can be seen that 20% enrichment still constitutes the inflexion point.

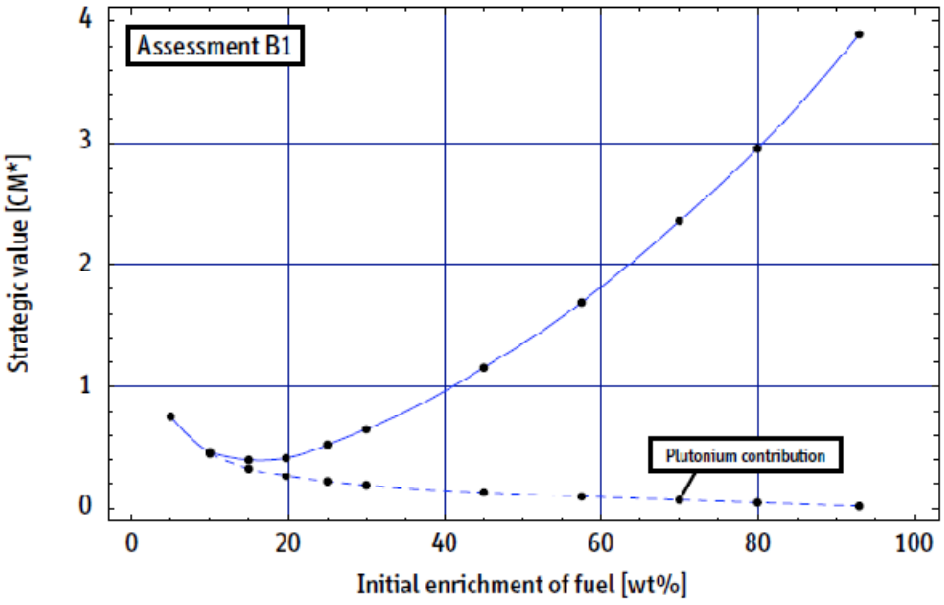


Fig. 35: Strategic value of fissile material assuming that uranium and plutonium are used for weapon production [68]

The capacity of enrichment plants is measured in terms of “separative work units” (SWU). This unit is a function of the amount of uranium processed and the degree to which it is enriched, which means the magnitude of effort needed to separate two isotopes of an element, so that the proportion of one isotope is increased in one of the resulting products. The unit is strictly: Kilogram Separative Work Unit and it measures the quantity of separative work performed to enrich a given amount of uranium. It is thus indicative of energy used in enrichment when feed and product quantities are expressed in kilograms. The unit “tons SWU” is also used [69].

Fig. 36: Cost of uranium enrichment [69]

The amount of uranium obtained from natural uranium depends on the enrichment. One ton of natural uranium might end up: as 120-130 kg of uranium for power reactor fuel, as 26 kg of a typical research reactor fuel, or as 5.6 kg of weapon-grade material. Figure 37 shows these facts and the relatively small increment of effort needed to achieve weapons-grade enrichment from normal levels. That is the reason to consider enrichment plants a sensitive technology in relation to prevent weapons proliferation and its tight supervision. The values presented are valid also for the thorium fuel cycle in order to have a general idea.

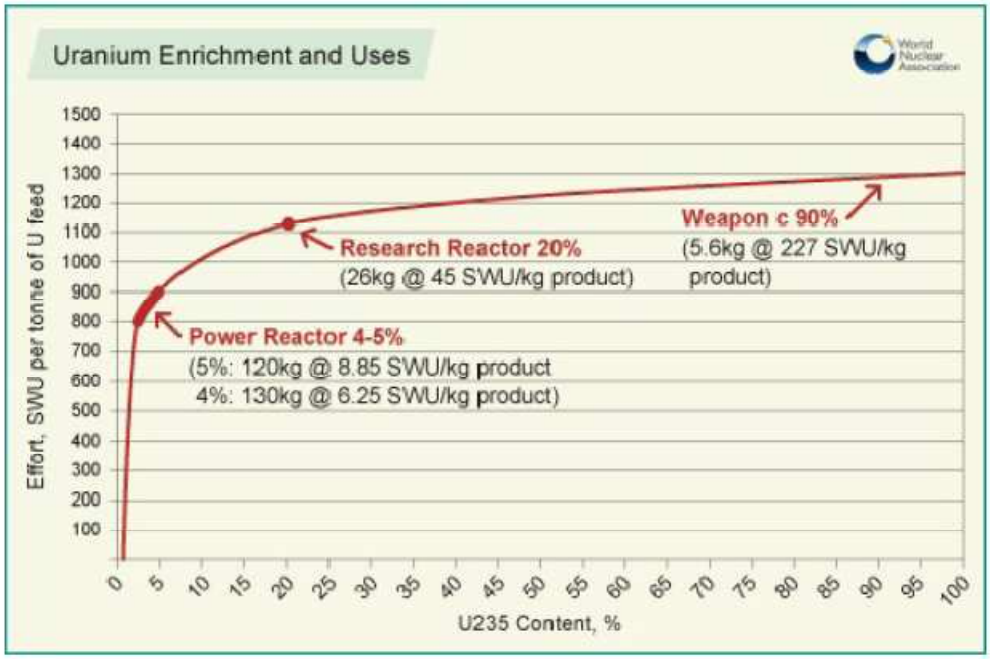


Fig. 37: Uranium obtained depending on the enrichment [69]

In order to have a general idea of how much work it takes to feed a power plant, studies have showed that about 140000 SWU is required to enrich the annual fuel loading for a typical 1000 MWe light water reactor at today high enrichment levels cf. Table 28. Enrichment costs are substantially related to the electrical energy used. The gaseous diffusion process consumes about 2500 kWh (9000 MJ) per SWU, while modern gas centrifuge plants require only about 50 kWh (180 MJ) per SWU [69].

Current Burnup Licensed Limits		
Belgium	UO_2	55 GWd/t ave. ass.
	MOX	50 GWd/t ave. ass.
Finland		45 GWd/t ave. ass.
France	UO_2	52 GWd/t ave. ass.
	MOX	42 GWd/t ave. ass.
Germany		55 GWd/t ave. ass.
Japan	UO_2	55 BWR/48 PWR GWd/t ave. ass.
	MOX	40 BWR/45 PWR GWd/t ave. ass.
Korea		60 GWd/t ave. rod
Netherland		55 GWd/t ave. ass.
Switzerland		50 BWR/60 PWR GWd/t ave. ass.
UK		55 GWd/t peak pellet
USA		62 GWd/t ave. rod

Table 28: Current burnup limits for different countries [73]

Enrichment accounts for almost one third of the cost of nuclear fuel and about 5% of the total cost of the electricity generated. Table 29 shows the cost of the different components and processes in the fuel cycle. The back-end of the fuel cycle includes the reprocessing and the management of the waste originating from reprocessing [24].

Natural Uranium	24%
Conversion	4%
Enrichment	31%
Fuel Fabrication	17%
Back-end of the Fuel Cycle	24%

Table 29: Breakdown of fuel cycle cost by step, in France [24]

5.2 Postulated accidents LOCA and RIA

All the critical issues that affect the fuel rod (pellet and cladding) are especially important when it comes to postulated accidents like LOCA or RIA. The purpose of this section is not to treat reactor accidents in detail but to give a general outlook about certain aspects of fuel and cladding evolution in an accident, and the reason that makes them a key factor in the fuel licensing process.

LOCA

The loss-of-coolant accident (LOCA) is a major postulated accident considered in licensing. During LOCA, the fuel cladding is subjected to high temperature oxidation and quenching. This last effect is due to the actuation of the emergency core cooling system. In the safety analysis for a postulated LOCA, it is generally estimated that the peak clad temperature would reach a value between 900 and 1400 K. The fuel cladding would be exposed to high temperature steam for several minutes until emergency core cooling system quenches the fuel bundle. The important aspects of a postulated LOCA are the time-temperature transient experienced by the fuel cladding before it is quenched and the differential pressure across the cladding during the transient.

The LOCA criteria is a key licensing issue for testing conventional and new materials (Zircaloy-4, Zircaloy-2, Zirlo or M5) at high burnup levels.

In this type of postulated accident there are three dominant phenomena which have to be taken into consideration:

- Fission product release.
- Cladding oxidation.
- Cladding deformation by internal pressure.

These challenges have to be taken into account, moreover when two of them, cladding oxidation and swelling, are main problems in thorium fuel rods. The oxidation produces hydrogen and, if very extensive, makes the cladding brittle during the quenching following reflooding of the core. In case of cladding deformation the depressurization of the primary circuit creates an internal pressure inside the rod due to the helium initially inside and the fission gases created. At the temperature to which the cladding is raised, localized ballooning of a significant number of rods can occur, thus reducing the coolant channels through the assembly. This can compromise cooling of the assembly by emergency systems [24].

As a result of the higher power density, fission gas release, temperature and residence time in the reactor, the cladding used in the thorium fuel cycles have to maintain safety operation characteristics not only during usual operation parameters but also while a design-basis accident takes place. The properties that the cladding has to present are more demanding than the usual reactors as a result of the tough conditions of postulated accidents. Furthermore the conditions to face postulated accidents are higher, making the licensing process of thorium fuel rods for high burnups a difficult and long process.

The pellet composition also has to show good behavior under accident conditions. In order to see the difficulties to prove correct characteristics while subjected to high burnup it is good to take a look at one of the effect of high burnup, which in fact is not the most challenging. Tests were done using the TESPA code which determines the temperature distribution in a cross-section of a fuel rod and deduces the strain from the resulting cladding temperature.

One of the problems mentioned before which affects the pellet as the burnup increases is the reduction of grain size and the formation of gas bubbles along grain boundaries at the periphery. The consequences of this effect during LOCA conditions were tested. Figure 38 shows the rapid temperature rise in a rim near to the pellet periphery within the first ten seconds of a LOCA transient.

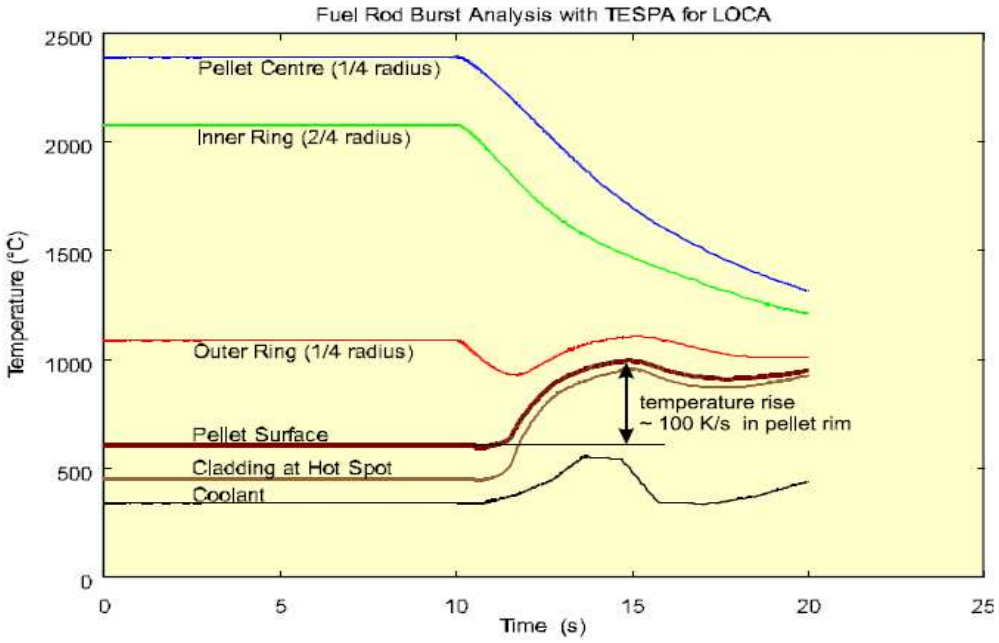


Fig. 38: Temperature development of the pellet during the LOCA transient [63]

In Fig. 38 can be seen that the temperature in the pellet surface increases rapidly which can enhance the release by the rim region of the stored fission gas and therefore support the fission gas pressure development during the LOCA transient. Despite this assumptions fission gas release by the rim region during accidents conditions is not well understood which makes that a certain margin should be taken into account until a better estimation of fission gas release is found.

RIA

The reactivity-initiated accident (RIA) involves an unwanted increase in fission rate and reactor power, capable of having a disruptive nature. A few such accidents occurred in the early days of research reactors which led to design improvements implemented in later generations of research reactors and power generating reactors. The most famous reactivity-initiated accident was the one in the Chernobyl power plant.

The most restrictive reactivity accident considered in the design of pressurized water reactors is the ejection of a control cluster which leads to a rapid increase in core reactivity and the deposit of a significant amount of energy into the fuel in a few tens of milliseconds. The energy deposited in the fuel causes swelling of the fuel due to the fission gases accumulated in it and fragmentation. This is a fast accident compared to the LOCA. Figure 39 shows the rapid increase in the rod power and how it produces the rod failure.

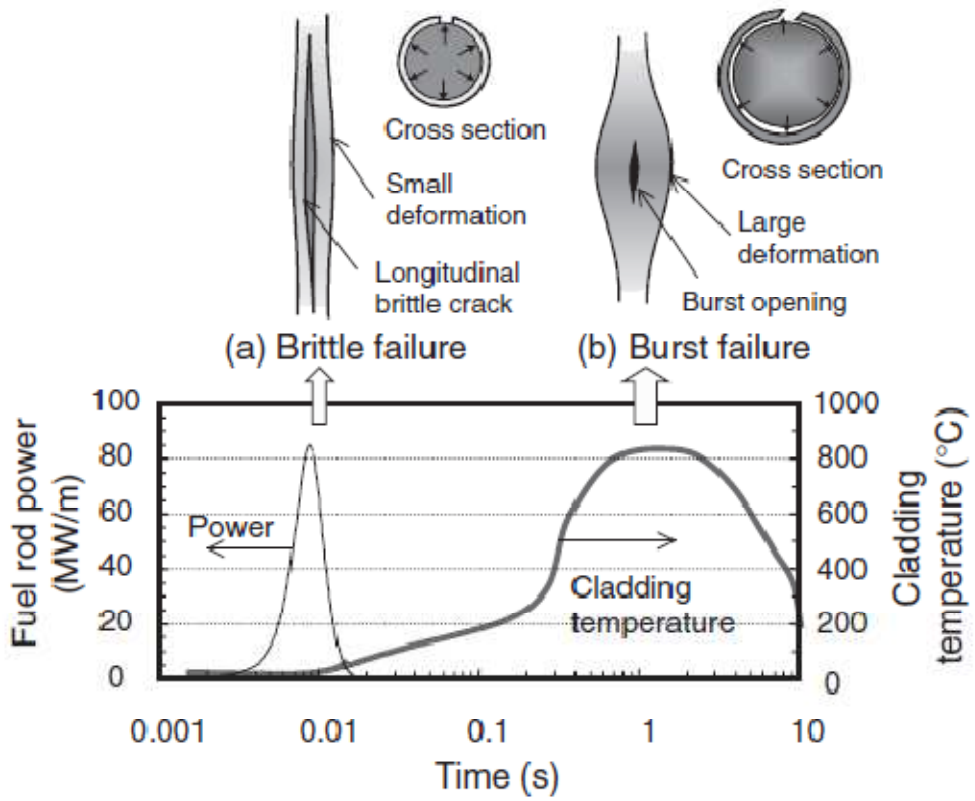


Fig. *39: Schematic illustration of two types of failure modes in early and late phases of RIA transients [70]*

In the licensing process of fuel or cladding materials and new operational conditions, the acceptance criteria for such an accident is defined by regulatory authorities. The limiting amount of damage is settled by the requirements to meet regulatory limits on radiation dose to the public, and to ensure integrity of the coolant pressure boundary and long-term coolability of the fuel [71].

The criteria are commonly defined in terms of limits on the radially averaged fuel pellet specific enthalpy, or the increment of this property during the reactivity-initiated accident. Regulatory authorities usually postulate two kinds of enthalpy limits:

1. Definite limit for core damage, which must not be transgressed at any axial position in any fuel rod in the core.
2. Fuel rod failure thresholds that define whether a fuel rod should be considered as failed or not in calculations of radioactive release.

Regarding the characteristics of this postulated accident, the increase in the enrichment to achieve high burnup would lead to an increase in the rod power. The fast power increase in RIA accidents combined with the higher rod power could challenge the integrity of the fuel assembly, especially the seed part of a Radkowsky Thorium Reactor which has a power of about 1.3 or 1.4 times higher than a conventional reactor. Further studies should be done in order to ensure safe behavior under these conditions for higher enrichments.

6. Summary and conclusions

A description of using thorium in nuclear fuel elements and the cladding requirements of such elements have been done in this work. An overview of the use of thorium in power reactors in different countries was given in order to have an outlook about the possibilities and the perspectives which this fuel element presents. The benefits and drawbacks of thorium fuel were also exposed. The objective was to show the differences between uranium and thorium fuels. The two fuel element configurations that thorium can present in the reactor, homogeneous and heterogeneous, were explained. This work focuses in the heterogeneous fuel bundles, proposed by Radkowsky, and their possible physical distributions in the reactor core. These two schemes are the seed and blanket unit (SBU) and the whole-assembly seed and blanket core (WASB). Both schemes were explained, showing the distribution inside the reactor and the benefits and challenges. Due to the easier shuffling and replacement of the fuel rods and to the easier geometry, the WASB characteristics were chosen in the modeling part of this work. The main idea of this scheme is to use it in a conventional PWR with minimum modifications.

The rod analysis focused on the effect that thorium operating conditions have on the cladding. The configuration proposed by Radkowsky is composed of two parts, one element called *seed* contains the fissile material, enriched uranium at 20% which is the percentage generally being accepted as non-proliferative, and operates at much higher local power density than a conventional PWR. The other part is the *blanket* and contains the fertile material, in this case thorium-232 with a small amount of uranium-238. The use of the Radkowsky concept arises some questions due to the large difference of power densities between the seed and the blanket and mainly as a result of the higher burnup and residence time of the rod in the reactor. From all the structural elements of the reactor, the cladding is the one which has to face the most challenging conditions regarding its physical integrity and corrosion resistance. The main problems related with the operating conditions, which are fission gas release, corrosion, hydride absorption and PCMI, were explained in this work.

The selection of the structural materials in the reactor is a critical decision. For that reason a brief presentation of the materials used and the mechanical, neutron, thermal and chemical characteristics that should present was given. The cladding options were explained, not only those used in past or in present reactors but also those considered as a cladding material for future reactors. Current alloys are mostly divided in ferritic alloys like stainless steel and zirconium based alloys like Zircaloy-2, Zircaloy-4 or Zirlo. Zirconium alloys are the main option as cladding material due to the superior neutron economy, corrosion resistance and good mechanical properties compared with stainless steel. The alloying elements and their effect were also presented, with the conclusion that the inclusion of niobium and the reduction of tin improve the behavior of the cladding. As a result of this combination new alloys are appearing, which present better properties than the usual cladding materials and which might resist the longer residence time and high burnup of the Radkowsky configuration. Some other kind of claddings are being tested like duplex alloys, which introduce an external layer of high corrosion resistance and an inner layer of zirconium alloy to keep the good neutron economy, and ceramic clad which has better mechanical properties under irradiation and at high temperatures.

In order to test the behavior of the fuel rod using thorium, a modeling program, FEMAXI, was used. FEMAXI has been provided by OECD/NEA and it allows to predict the thermal and mechanical behavior of a fuel rod during normal and transient conditions. It was not possible to gather information about the operating conditions of a PWR so during the simulation some approximations were made but always keeping the parameters around the typical for that kind of reactor. The values, the auxiliary programs used and the properties for thorium and uranium were presented in the summary of section four of this work. From the results obtained, the author focused on two effects which would give a better estimation and at the same were critical issues. The analysis of both, inner pressure and cladding corrosion, showed that despite the cladding seems to stand the pressure from the fission gases, the corrosion effect could be a limiting factor (pag. 52, 55). Although the results have not been validated the burnup limit that the fuel rod may reach in order to operate within safety margins is around 45 MWd/kg.

Finally in order to understand the process of licensing an overview was offered, presenting the technical limits of the fuel and the cladding, the enrichment limit of the fuel and also the challenges which has to stand during postulated accidents. This section aimed to explain the difficulties to reach higher burnups and longer residence time for thorium fuel rods.

After the realization of this work some conclusions can be extracted. Thorium seems to present good properties relating its physical and chemical stability, neutron economy, proliferation resistance, abundance and low radiotoxicity. Despite these promising characteristics thorium presents some inconvenience, the limited database and experience being the most important ones.

The main challenges for the use of thorium are the effect of the high burnup and long residence time in the cladding. Studies show that typical cladding materials such as Zircaloy-4 can stand only burnups of about 45 MWd/kg while new cladding materials such as M5 or duplex cladding could reach 100 MWd/kg. These values are lower than expected for a thorium fuel cycle, showing the impossibility nowadays to use thorium as a fuel in conventional reactors. More research must be done in order to overcome those limits, especially the corrosion effect and the hydride absorption from the cladding.

The utilization of thorium in a conventional LWR is a long process due to the licensing requirements. Many of the questions about the reliability at the high burnup range have been identified and overcome. Despite this fact, the need to implement gradually any change in the reactor, the great amount of time and resources for experimental tests and the need to demonstrate operation under safety margins in front of postulated accident conditions (LOCA and RIA) request that any change in the operation conditions of the reactor is carefully considered. Nowadays, the aim is to approach burnups of about 60 MWd/kg. This value is still far from that required for a thorium reactor.

The commercial use of the thorium fuel cycle is difficult to predict. The time needed depends on the research effort and the amount of resources used but is not likely to be in a near future. The high burnup reached and the challenging environmental conditions that the cladding has to face make the licensing a dilatory process.

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Appendix: Modeling parameters and considerations

The program used to model the behavior of a fuel rod was FEMAXI-6, provided by the OECD/NEA [55]. In this section, the estimations and considerations used during the simulation are explained with more detail.

FEMAXI uses auxiliary programs for burnup analysis. Values of power density profiles, burnup profiles and fast neutron flux are given by either RODBURN or PLUTON. Both of them need fuel rod specifications and operating characteristics, which are provided in the input file Throd4.inp. In this work PLUTON was used. The code PLUTON has been verified up to 83 MWd/kg with satisfactory agreement and can handle ThO_2-UO_2 fuel. The input parameters needed to run PLUTON are shown in Fig. 40. The data given by FEMAXI can be numerical Throd4.out and Throd4.plt output usable by plotter EXPLOT.

```
*******************************************************************************
  Input parameters for the PLUTON code
*******************************************************************************
```

TABLE 1. Physical Parameters:

List of variables	Units	Values
FRDENS, fractional fuel density	n/a	0.82
Bustep, burnup power-step increment	GWd/t	0.20
Burnup, upper burnup limit	GWd/t	88.0
EnriU5, initial enrichment with U-235	fraction	0.930
CDiam1, the outer clad diameter	cm	0.95
CDiam2, the inner clad diameter	cm	0.893
FDiam1, the outer fuel diameter	cm	0.819
FDiam2, the inner fuel diameter	cm	0.0
WEPITH, 0.304 - PWR/UO2; 0.366 - PWR/MOX		0.200
WFast, 2.010 - PWR/UO2; 2.390 - PWR/MOX		0.750
TWater, water temperature	K	578.
DWater, water density at lower end plug	g/cc	0.860
Vratio, fuel to water ratio by volume	N/A	0.628
VVoids, void fraction	N/A	0.350
TFuelC, fuel temperature in resonant region	C-degree	1000.
QLLBOL, BOL reference LHGR	w/cm	114.0
POWOFF, Reactor operation coefficient	n/a	0.700
DAYPIE, Time gap between EOL and PIE	days	0.0
URANIUM ISOTOPIC COMPOSITION (Initial wt% of uranium group)		
WtF_U_233	Wt.%	0.000
WtF_U_234	Wt.%	0.000
WtF_U_236	Wt.%	0.000
Wt_ThO2, weight content of thorium dioxide	fraction	0.883
Wt_NpO2, weight content of neptunium dioxide	fraction	0.000
Wt_PuO2, weight content of plutonium dioxide	fraction	0.000

Fig. 40: Main parameters needed in PLUTON input file

The ThO_2 properties needed to model this fuel were implemented by Risto Vanhanen in "Implementing thorium fuel option to fuel performance code FEMAXI-6" [57]. The difficulties to find open data and models for thorium properties led to some approximations. The thoria-urania properties were estimated using Vegard's law, which states that in cubic or amorphous solid solutions (like ThO_2 and UO_2) most properties vary linearly with atomic fractions. The use of this law is accepted because thorium dioxide and uranium dioxide have the same crystal structure. Similar crystal structure is important so thorium atoms are directly replaced by uranium atoms without disturbing the lattice. The differences in the properties of thoria and urania are due to their different electron configurations.

One of the sources of uncertainty in the fuel modeling is the different behavior in high and low burnup regions. Implemented models do no account for high burnup phenomena or effects which affect the long term behavior of the fuel. In some cases it was not possible to find specific models for thorium, in that case uranium models were used or a mix between uranium and thorium models.

Regarding the thermal conductivity, which is one of the values that has a major influence in the other properties of the material, two models were available, Belle and Berman and Bakker et al. [57]. The differences between both models are the value of the coefficients which define the lattice conduction and the range of validity of the correlations. During the simulation the correlation of Bakker et al. was used.

FEMAXI input

The lack of empirical data to contrast the values and to follow the variations in the parameters of the reactor leads to the use of some approximations. Certain values are assumed to be constant and the numbers of history points is reduced in order to make the simulation simpler and faster. The different possibilities and models available to model thorium are selected in the input file of FEMAXI.

During the simulation 41 history points were selected. These points are the time intervals in which the residence time of the rod in the reactor is divided. The linear heat rate points of the rod, which is the heat profile in the axial direction of the rod and is assumed to follow a cosinus, were set to 16. A general heat profile was used due to the lack of a rod history to model.

The physical characteristics of the fuel pellet and the level of enrichment are also included in the input file of FEMAXI. The fuel element was considered to have the characteristics presented in Table 30.

Cladding inner diameter (cm)	0.893
Cladding outer diameter (cm)	0.95
Nº of dishes	2
Chamfer	No chamfer
Pellet center hole diameter (cm)	0
Pellet diameter (cm)	0.819
Length of one pellet (cm)	1
U-235 enrichment	0.1
Pellet theoretical density ratio	0.82
Axial segment length of pellet stack part (cm)	26.7

Table 30: Physical characteristics of the pellet

The variation of the heat rate in the whole reactor was modeled using a graph which relates heat rate and burnup from a conventional PWR. The values were adapted to the higher heat rate of thorium fuel. Other parameters from the reactor as temperature, pressure or coolant velocity were assumed to be constant during operation but always in the range of a conventional PWR. Figure 41 shows some parameters needed in the input file.

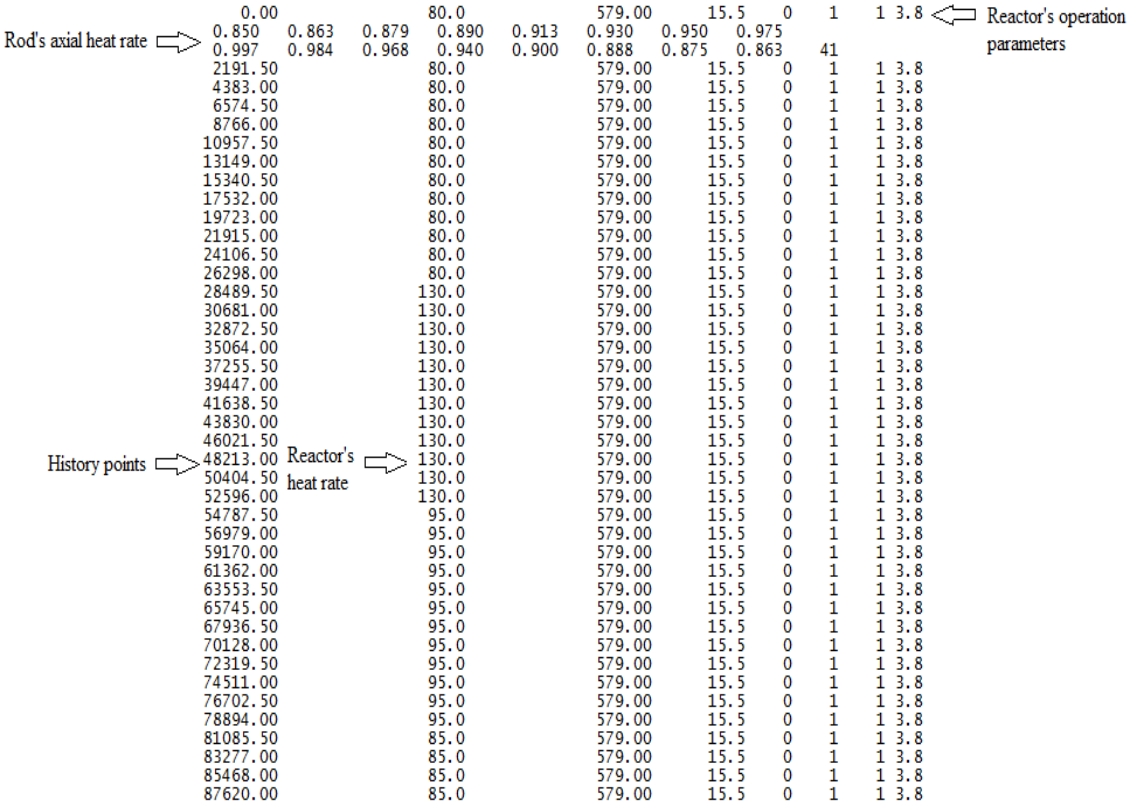

Fig. 41: FEMAXI input file