

Thorium-based fuel cycles: Saving uranium in a 200 MWth pebble bed high temperature reactor.

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- Summary -

The predominant nuclear fuel used globally at present is uranium which is a finite resource. Thorium has been identified as an alternative nuclear fuel source that can be utilized in almost all existing uranium-based reactors and can significantly help in conserving limited uranium reserves. Furthermore, the elimination of proliferation risks associated with thorium-based fuel cycles is a key reason for re-evaluating the possible utilization of thorium in high temperature reactors. In addition to the many advantages that thorium-based fuel has over uranium-based fuel, there are vast thorium resources in the earth's crust that up until the present have not been exploited optimally.

This study focuses on determining the amount of uranium ore that can be saved using thorium as a nuclear fuel in HTR's. Four identical 200 MW_{th} high temperature reactors are considered which make use of different fuel cycles. These fuel cycles range from the conventional uranium fuel cycle to a thorium-based fuel cycle in which no U-238 is present and have been simulated using the VSOP-A system of computer codes. This study also considers the effect that protactinium, an isotope that occurs in thorium-based fuel cycles, will have on the decay heat production in the case of a depressurized loss of coolant (DLOFC) accident.

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- Chapter 1 -

Introduction

The last ten years have seen a renewed interest in nuclear energy after the pessimistic years of the late 1980's and 1990's as a result of an accident in Chernobyl in 1986. This renewed interest saw many new reactors being developed and construction orders being placed. The Chinese currently have 26 reactors under construction and need 80 GW of nuclear energy by 2020. Therefore, an increased demand for uranium is expected which is a limited resource and is a major concern for the future of nuclear development. Apart from the concern about uncertain uranium supply in the future, there is anxiety about nuclear proliferation due to the plutonium isotopes that are produced in the uranium fuel cycle.

An alternative fuel cycle that addresses both the concerns of secure fuel supply and proliferation risks is the thorium fuel cycle because thorium is by far more abundant in the earth's crust than uranium and produces significantly less plutonium and radioactive transuranic isotopes than conventional uranium fuel cycles (Mulder, 2010). If the growing interest in nuclear energy continues and if nuclear energy is to remain a sustainable source for the future, thorium fuel cycles will have to be implemented unless vast uranium resources are to be discovered that may be extracted viably. If uranium is extracted from sea water or there is a change to fast breeder reactors there will be enough uranium for about 5000 years and the need for thorium will be alleviated. If, however, these options are not implemented there will be enough uranium for only 100 years at the current consumption and, therefore, thorium will be required as nuclear fuel.

Results from tests performed on the experimental AVR high temperature reactor (HTR) at Jülich in Germany have shown the degree of flexibility that HTR's have in utilizing different types of fuel with varying enrichment's and very superior levels of safety compared to existing nuclear reactors. Therefore, HTR's are ideal candidates for utilizing thorium-based fuel cycles in which large amounts of uranium ore can be conserved, thereby prolonging the sustainability of nuclear energy.

Kazimi et al., 1999 state that using thorium as a fertile material for nuclear fuel has been of interest since the beginning of nuclear development due to the associated neutronic advantages and the abundance of thorium ore. Due to finite uranium resources the thorium-based fuel cycle became an

attractive replacement option in the 1960's and 1970's (Lung et al., 1997). As a result of this interest a number of high temperature reactors (HTR) were developed which utilized thorium-based fuel cycles and it was found that most existing reactors could also utilize thorium as a nuclear fuel (Lung et al., 1997). The thorium fuel cycle has, however, not made a notable breakthrough in the nuclear industry, until recently, with India being the exception, in which extensive work in this field is being conducted.

According to Lung et al., 1997 thorium-based fuel cycles allow excess plutonium to be used as nuclear fuel. The radio toxicity of the spent fuel also lasts about a factor 10 shorter. They further found that thorium-based fuel cycles achieve higher burn-ups than conventional uranium-based fuel cycles. These findings along with the fact that uranium resources are limited should, according to Lung et al., 1997 promote long term interest in utilizing thorium as a nuclear fuel. The use of thorium-based fuel cycles do however require the use of a closed fuel cycle which poses certain reprocessing challenges as thorium (ThO_2) is chemically very stable and has radiological challenges attached to it as opposed to uranium-based fuel cycles that allow open fuel cycles without the need to reprocess spent fuel (Anantharaman et al., 2008).

Thorium-based fuel in the form of ThO_2 , $(\text{Th}+\text{U})\text{O}_2$, ThC_2 and $(\text{Th}+\text{U})\text{C}_2$ coated particles known as TRISO fuel has been used in HTR's in Germany, the USA, Japan and Russia in the past with success (Sokolov et al., 2005). Two pebble bed HTR's namely THTR (300 MW_e) and AVR (15 MW_e) using spherical shaped fuel elements in which the fuel particles are embedded in a graphite matrix were in operation in Germany until the late 1980's. In the USA two HTR's called Fort St. Vrain (330 MW_e) and Peach Bottom (40 MW_e) using TRISO coated fuel in the form of prismatic blocks were operational (Sokolov et al., 2005). Table 1.1 summarizes the experimental and power reactors in which thorium-based ceramic fuels have been used in the form of coated particles embedded in a graphite matrix or as Zircaloy clad fuel pin assemblies.

However, details of the performance of thorium-based fuel cycles in the HTR-MODUL 200 MW_{th} design for pebble bed HTR's still have to be established. Sokolov et al., 2005 state that the experience and data available concerning thorium-based fuel cycles are extremely limited when compared to uranium-based fuel cycles. This database needs to be improved before thorium-based fuel can be utilized commercially.

Table 1.1: Thorium-based experimental and power reactors (Sokolov et al., 2005).

Name and Country	Type	Power	Fuel	Operation period
AVR, Germany	HTGR Experimental (Pebble bed reactor)	15 MW(e)	Th+ ²³⁵ U Driver Fuel, Coated fuel particles Oxide & dicarbides	1967 – 1988
THTR, Germany	HTGR Power (Pebble Type)	300 MW(e)	Th+ ²³⁵ U, Driver Fuel, Coated fuel particles Oxide & dicarbides	1985 - 1989
Lingen, Germany	BWR Irradiation-testing	60 MW(e)	Test Fuel (Th,Pu)O ₂ pellets	Terminated in 1973
Dragon, UK OECD-Euratom also Sweden, Norway & Switzerland	HTGR Experimental (Pin-in-Block Design)	20 MWt	Th+ ²³⁵ U Driver Fuel, Coated fuel particles Dicarbides	1966 - 1973
Peach Bottom, USA	HTGR Experimental (Prismatic Block)	40 MW(e)	Th+ ²³⁵ U Driver Fuel, Coated fuel particles Oxide & Dicarbides	1966 – 1972
Fort St Vrain, USA	HTGR Power (Prismatic Block)	330 MW(e)	Th+ ²³⁵ U Driver Fuel, Coated fuel particles Dicarbide	1976 - 1989
MSRE ORNL, USA	MSBR	7.5 MWt	²³³ U Molten Fluorides	1964 - 1969
Borax IV & Elk River Reactors, USA	BWRs (Pin Assemblies)	2.4 MW(e) 24 MW(e)	Th+ ²³⁵ U Driver Fuel Oxide Pellets	1963 - 1968
Shippingport & Indian Point, USA	LWBR PWR (Pin Assemblies)	100 MW(e) 285 MW(e)	Th+ ²³³ U Driver Fuel, Oxide Pellets	1977 – 1982 1962 - 1980
SUSPOP/KSTR KEMA, Netherlands	Aqueous Homogenous Suspension (Pin Assemblies)	1 MWt	Th+ HEU Oxide Pellets	1974 - 1977
NRU & NRU, Canada	MTR (Pin Assemblies)		Th+ ²³⁵ U Test Fuel	Irradiation- testing of few fuel elements
KAMINI, CIRUS, & DHRUVA, India	MTR Thermal	30 kWt 40 MWt 100 MWt	Al- ²³³ U Driver Fuel 'J' rod of Th & ThO ₂ 'J' rod of ThO ₂	All three research reactors in operation
KAPS 1&2, KGS 1&2, RAPS 2,3&4, India	PHWR (Pin Assemblies)	220 MW(e)	ThO ₂ Pellets For neutron flux flattening of initial core after start-up	Continuing in all new PHWRs
FBTR, India	LMFBR (Pin Assemblies)	40 MWt	ThO ₂ blanket	In operation

Figure 1.1 outlines the four high temperature reactors and their fuel cycles considered in this study. The performance of these four fuel cycles are simulated using the multi-physics, multi-scale characteristics of the HTR-MODUL core with the aid of the VSOP-A system of codes.

The first objective is to determine the uranium ore requirement of each of the four HTR's used in this study (see Figure 1.1). The first of these HTR's shown in Figure 1.1 makes use of a conventional uranium fuel cycle and is the reference HTR to which all the other thorium-based HTR fuel cycles are compared. The second and third of these are so called pre-breeder (PB) HTR's and their function is to breed new nuclear fuel, namely U-233. This fuel is then recovered and inserted into the fourth reactor shown in Figure 1.1 which then additionally breeds U-233 and, therefore, is called a net-breeder (NB) HTR. The relationship of these two HTR's in which the pre-breeder supplies the net-breeder with fuel is called a reactor symbiosis.

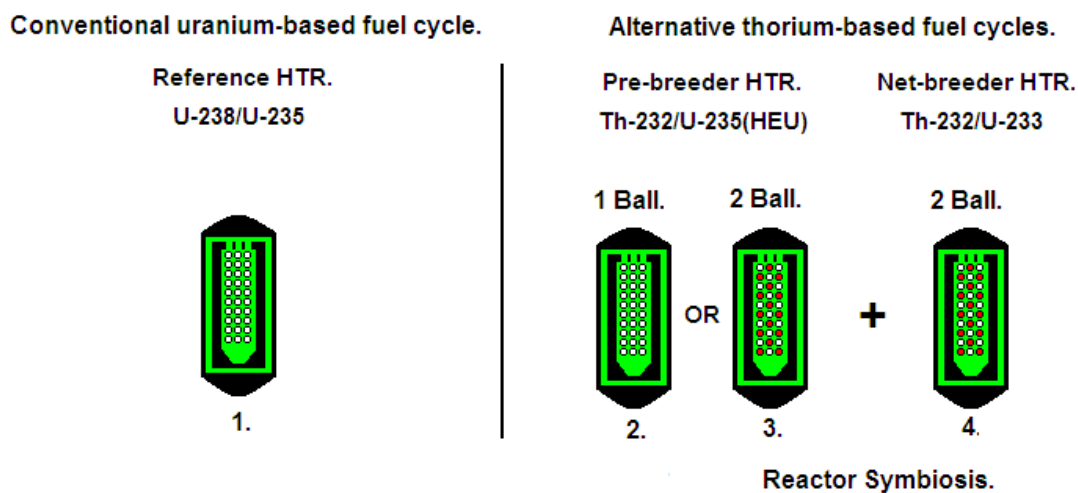


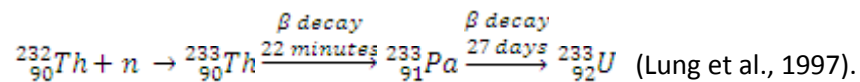
Figure 1.1: Outline of HTR's and their fuel cycles used for the simulations.

The second objective of this study is to make a comparison of the neutronic parameters between the two pre-breeder (PB) pebble bed HTR fuel cycles shown in Figure 1.1 namely a Th-232/U-235(HEU) fuel cycle in which the fuel elements contain a mixture of (Th+U)O₂ fuel particles, and a Th-232/U-235(HEU) fuel cycle in which two separate fuel elements are used where one contains only ThO₂ fuel particles and the other only UO₂ fuel particles, respectively. From this comparison it can be determined which one of these pre-breeder fuel cycles would be favourable to supply a separate net-breeder pebble bed HTR utilizing a Th-232/U-233 fuel cycle with the bred U-233 obtained from the spent fuel inventory of these respective PB fuel cycles. Of significant importance is the conversion ratio (CR) of the fissile U-233 produced in the PB fuel cycles by the fertile Th-232.

This symbiotic system in which a net-breeder pebble bed HTR is supplied by a Pre-Breeder pebble bed HTR is compared with the reference HTR in which a low enriched uranium (LEU) fuel cycle is used.

The third objective of this study is, therefore, to determine the amount of uranium ore that can be saved by breeding an alternative fissile fuel namely U-233 in a pre-breeder reactor that will then be used by a net-breeder reactor which additionally breeds U-233 (see Figure 1.1).

In the reactor design the effect of protactinium-233 (Pa-233) that appears in thorium-based fuel cycles is also taken into consideration as it could have a significant impact on the decay heat production during a DLOFC accident due to its long half-life of 27 days. Additionally the reactivity state of the reactor could be influenced by Pa-233 with this long half-life because it has a larger absorption cross-section than that of Np-239 present in the uranium-based fuel cycle. Pa-233 is formed when Th-232 absorbs a neutron to become Th-233 which then decays with the emission of gamma (γ) rays. This is depicted below:



The fourth objective of this study is to determine the influence of the protactinium-233 (Pa-233) on the decay heat production of the Th-232/U-235 and Th-232/U233 fuel cycles.

These objectives entail the use of the VSOP-A computer system of codes in which the different fuel cycles are simulated. The simulations are performed on the basis of the 200 MW_{th} HTR-MODUL calculational design model shown in Figure 1.2.

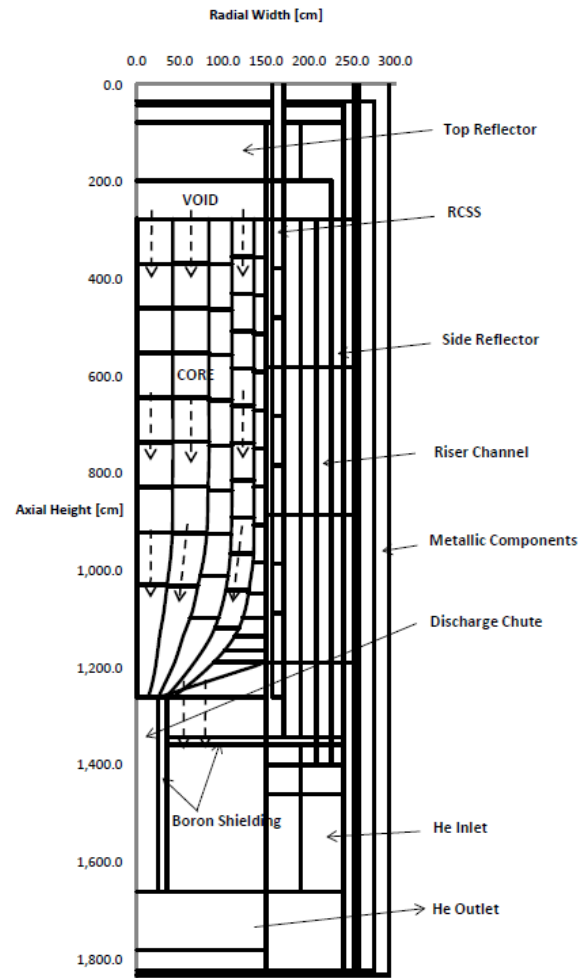


Figure 1.2: Calculational design model for VSOP-A (Mulder et al., 2010).

- Chapter 2 -

Literature study

2.1 Principal aspects of the 200 MW_{th} HTR-MODUL pebble bed High temperature reactor.

The 200 MW_{th} HTR-MODUL is the reactor selected for this study to model and simulate the various fuel cycles (Teuchert, et al., 1992). It is a pebble bed high temperature reactor and was designed and developed in Germany in 1979 (Siemens, 1987). This reactor makes use of spherical shaped fuel elements where the fuel is in the form of a TRISO-coated particle, i.e. a fuel kernel coated by a number of carbon and SiC layers. These coated particles are then imbedded in a graphite matrix which forms the spherical shape.

2.1.1 200 MW_{th} HTR-MODUL core design.

The 200 MW_{th} HTR-MODUL differs significantly from the more popular pressurized water reactors (PWR) currently used in the world today. One noteworthy difference is that HTR's use a gas as coolant, usually helium, as is the case with the 200 MW_{th} HTR-MODUL, whereas PWR's use water(H₂O). Furthermore, the materials used for the moderation of fission neutrons differ. HTR's make use of graphite for moderation whereas PWR's use either light water(H₂O) or heavy water(D₂O).

As a result of these two fundamental differences, significantly higher coolant temperatures are possible in HTR's resulting in an increased thermal efficiency. Further benefits of helium are that it is chemically inert and has a large specific heat capacity in comparison to water used in PWR's (Kugeler et al., 2003). One very significant advantage of the modular pebble bed HTR is that it is flexible at utilizing a wide range of different fuel cycles. These fuel cycles can be comprised of fissile fuel isotopes U-233, U-235, Pu-239 and Pu-241 and fertile isotopes Th-232, U-238 and Pu-240 which are used in either mixed or separate form (Kara et al., 2008).

The physical design of the 200 MWth HTR-MODUL is given in Figure 2.1 and shows the main structural components. The pebble bed core, which contains the spherical shaped fuel elements, is surrounded at the sides by side reflectors and at the top and bottom by a top and bottom reflector respectively. The reflectors are made from graphite and are enclosed in a core barrel that is formed from a number of steel rings that hold the reflectors in position. The side reflectors contain borings into which the control rods and shutdown elements are inserted. Surrounding the core barrel is a thermal shield and a reactor pressure vessel (RPV) (Kugeler et al., 2003).

Cold Helium enters the top of the core at 250°C, passes through the spherical shaped fuel elements where it is heated and then exits the core at the bottom at approximately 700°C before it is transported to the steam generator via a coaxial hot gas duct (Kugeler et al., 2010).

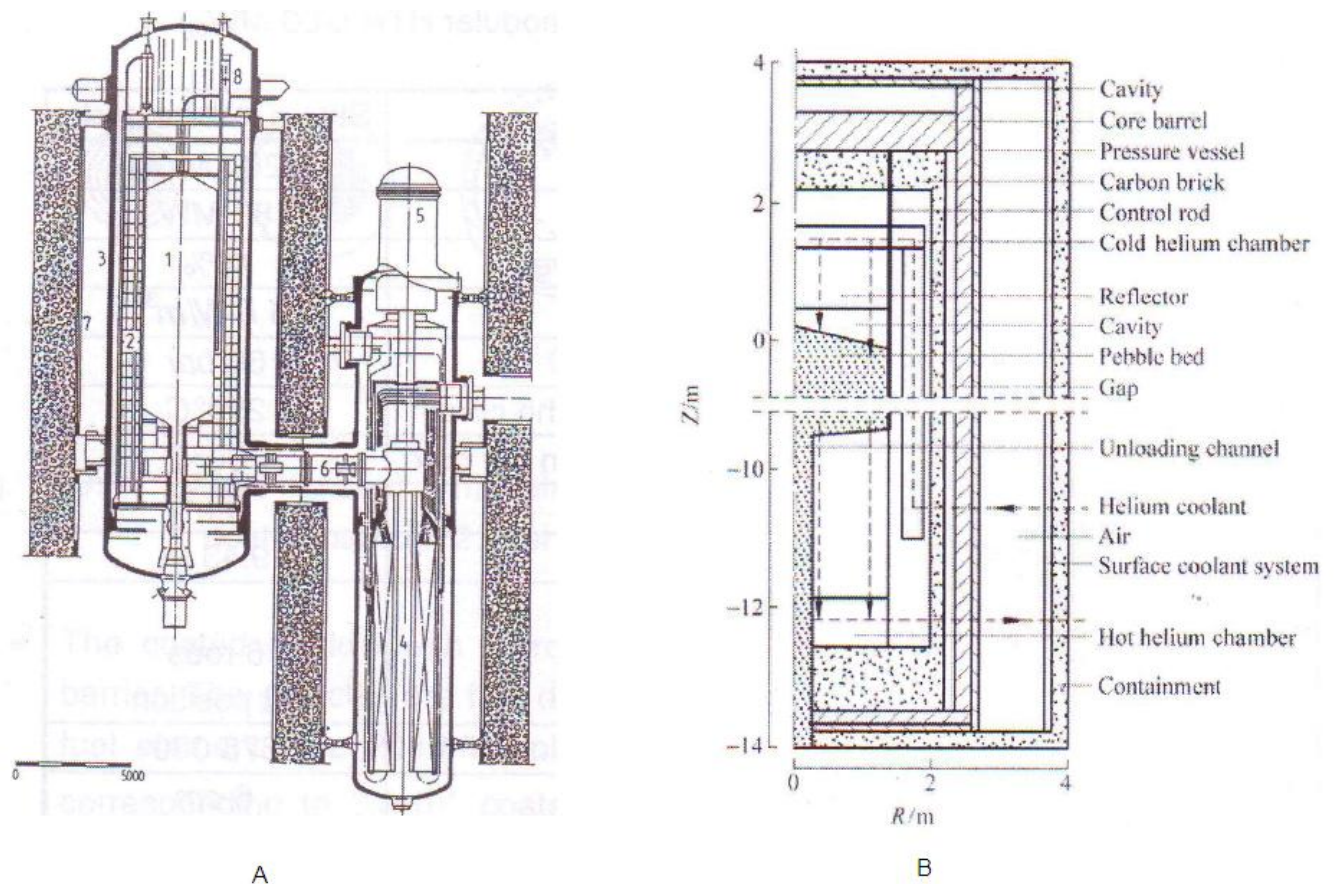


Figure 2.1: Physical design of the 200 MWth HTR-MODUL.

A: Reactor Design and Component Arrangement (1. Core, 2. Reflector, 3. Reactor Pressure Vessel, 4. Steam Generator, 5. Helium Circulator, 6. Coaxial Duct, 7. Surface Cooler) (Kugeler, et al., 2010).

B: Structural Configuration of HTR Core (Hong, et al., 2006).

A very important design feature of this pebble bed HTR is that the core cannot melt, even in the case of a severe accident in which there is a depressurized loss of forced cooling (DLOFC). The reason that this significant safety feature can be realized is that the pebble bed HTR is designed with the inherently safe characteristic of self reliant decay heat removal in which the decay heat is removed by natural means of conduction, radiation and free convection in the case of an accident. For this to be realized certain parameters are of utmost importance. Kugeler et al., 2010 state that the essential parameters that must be taken into consideration to achieve self reliant decay heat removal are the power density of the core, heat storage capacity of the core, the heat transfer coefficients and conductivities of the core structures and the surface to volume ratio of the core. Table 2.1 gives the main design parameters for the 200 MW_{th} HTR-MODUL core.

Table 2.1: Design Parameters of the 200 MW_{th} HTR-MODUL (Kugeler et al., 2010).

Thermal Power	[MW]	200
Electrical Power	[MW]	80
Power Density	[MW/m ³]	3
Fuel Element Power	[KW/FE]	0.56
Core Height	[m]	9.43
Core Diameter	[m]	3
Heating of Helium	[°C]	250-700
Helium Pressure	[MPa]	60
Mass Flow Rate	[kg/s]	85.4
Net Efficiency	[%]	40
Burn-up	[MWd/t _{HM}]	80000

2.1.2 HTR spherical fuel element design.

Pebble bed HTR's are distinguishable from other reactor types by their unique fuel design. The pebble bed HTR fuel sphere is 6 cm in diameter and consists of a graphite matrix containing thousands of TRISO coated particles that are about 1 mm in diameter. These coated particles consist of either a (Th+U) carbide or oxide or UO₂ kernel, normally between 240 µm and 500 µm in diameter coated by a low density C buffer layer, inner pyrolytic C layer, SiC layer and outer pyrolytic C layer respectively. Figure 2.2 shows the spherical shaped pebble bed HTR fuel elements and the TRISO coated fuel particles contained within them.

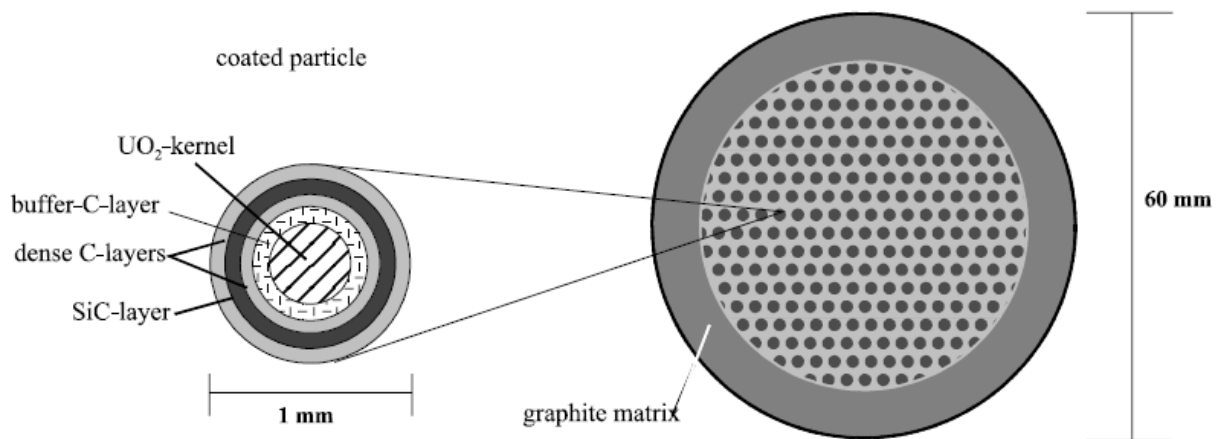


Figure 2.2: Pebble bed HTR fuel sphere and TRISO coated particle (Kugeler et al., 2003).

The inner low density carbon buffer layer acts as a storage containment for fission gases that build up due to the fission process within the fuel kernel, while the high density carbon and silicon carbide layers act as impenetrable containment layers that are very effective at preventing fission products diffusing out of the coated fuel particle at high temperatures (Kugeler et al., 2003). The TRISO coated particles retain all of the fission products almost completely as long as the fuel temperature is limited to 1600°C (Schenk, et al.). In the case of a severe accident in which there is a loss of active coolant the fission product release rate is less than 1×10^{-5} for several hundred hours at a temperature of 1600°C (Kugeler et al., 2003). Table 2.2 gives some important design parameters for the spherical fuel elements and TRISO coated particles used in the 200 MW_{th} pebble bed HTR-MODUL.

Furthermore, pebble bed HTR's can operate very flexible fuel cycles, as was shown with the AVR pebble bed HTR in which fuel spheres containing 22 different fuel types were tested with great success. These reactors also allow very high burn-up rates because the fuel elements are continuously recycled through the core during operation (Kugeler et al., 2003). This is of significant relevance to increasing the proliferation resistance since the plutonium inventory is extensively reduced and Pu-239 denatured with Pu-240 and Pu-238 with increased burn-up.

In the case of thorium-based fuel cycles the excellent neutronic characteristics of U-233 bred from fertile Th-232 at thermal energies combined with superior irradiation properties of graphite used for the

fuel spheres as moderator and the absence of neutron poisons from cladding materials result in even higher burn-ups (Greyvenstein, 2007).

Table 2.2: Design parameters for (Th+U)O₂ (HEU) and UO₂ (LEU) TRISO coated particle fuel elements (Kugeler et al., 2003).

Parameter	Dimension	(Th+U)O ₂ (HEU)	UO ₂ (LEU)
Coated Particles			
Kernel Composition		(Th+U)O ₂	UO ₂
Kernel Diameter	μm	500	500
Coating Layer Thickness	μm	95/40/35/35	95/40/35/35
Coating Layer Sequence		Buffer/PyC/SiC/PyC	Buffer/PyC/SiC/PyC
Fuel Element			
Heavy Metal Loading	g/FE	11	8-12
U-235 Enrichment	%	93	7-13
No. of Particles per Element		19000	13000-20000
Volume Loading of Particles	%	13	10-15
Operational Requirements			
Mean Operation Time	d	1100-1500	700
Maximum Burnup	MWd/t _{HM}	120000	90000
Maximum Fast Dose	1x10 ²⁶ m ⁻²	4.5	3.3
Maximum Fuel Temperature	°C	1020	1030
Maximum Power/FE	kW	2.7	4.1

2.2 Concept of reactor symbiosis.

The idea behind the symbiotic reactor system, in which thorium fuel cycles are utilized to conserve finite uranium resources, is to make use of two separate modular pebble bed HTR's namely a pre-breeder (PB) HTR and net-breeder (NB) HTR. The PB-HTR utilizes a thorium fuel cycle that is comprised of fuel spheres containing (Th+U)O₂ (HEU) fuel particles. An alternative PB-HTR fuel cycle can be considered in which the thorium and uranium are isolated from each other in which breed spheres containing only ThO₂ and drive spheres containing only UO₂ (HEU) are used (Mulder, 2009). The PB-HTR is used to convert the fertile Th-232 into fissile U-233 which will then be inserted into another thorium fuel cycle used by the NB-HTR.

Once the fuel spheres of the PB-HTR have reached their final burn-up they are removed from the fuel cycle and are reprocessed to recover the U-233 that was bred. This U-233 is then inserted into the NB-HTR driver spheres. In the case of the PB-HTR where the UO_2 and ThO_2 are inserted into separate fuel elements, it could be possible to insert the used breeder elements of the PB directly, which now contain U-233, into the NB-HTR as driver spheres without any reprocessing (Mulder, 2009). The NB-HTR fuel cycle also consists of driver spheres and breeder spheres containing fertile Th-232. In this reactor the fertile Th-232 is also converted into U-233 and can once again be recovered and used in further thorium fuel cycles in the driver elements. Figure 2.3 shows this symbiotic concept schematically.

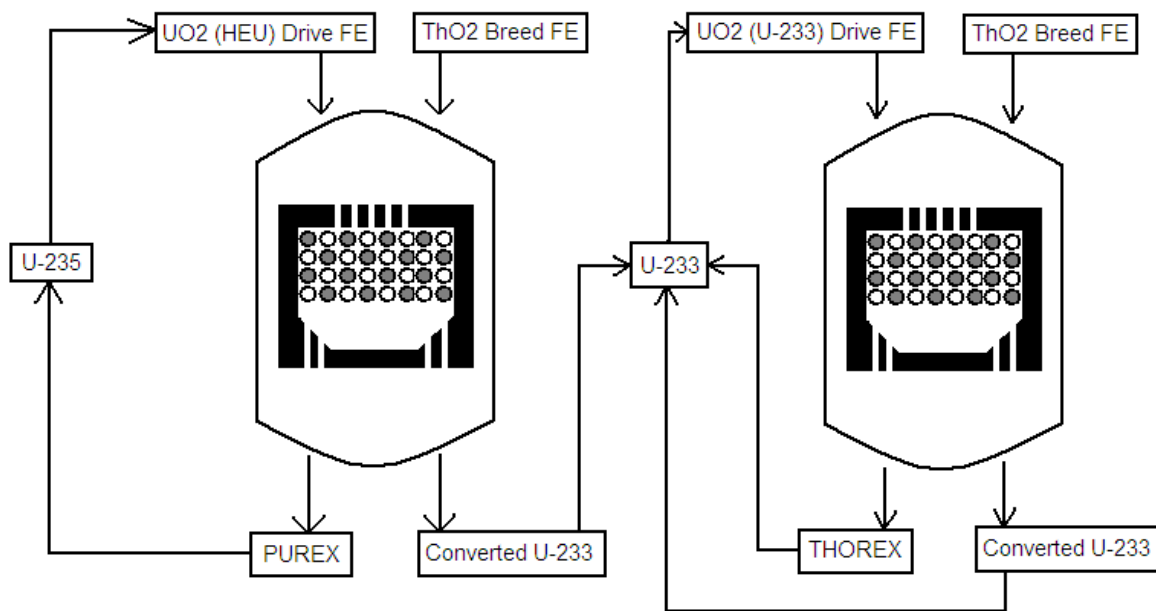


Figure 2.3: Schematic of the symbiotic concept of a pre-breeder HTR and a net-breeder HTR.

This symbiotic system of HTR's could utilize both uranium and thorium in a more efficient way, thereby conserving the earth's finite uranium resources. According to Hong et al., 2006, the U-233 recovered from three PB-HTR's, with a conservative conversion ratio of around 0.446, could supply one NB-HTR with fissile fuel. Teuchert et al., 1979 state that the U-233 produced in the Th-232/U-235 fuel cycle per uranium ore (U_3O_8) requirement is a factor 2.3 higher than the Pu produced in LWR for a U-238/U-235

fuel cycle. Furthermore, Teuchert et al., 1979 state the uranium ore requirement for the PB-HTR utilizing a thorium based fuel cycle is 31% of that needed for the conventional uranium fuel cycle. This concept could, therefore, contribute significantly to the sustainability of nuclear energy for the future.

2.3 Thorium resources.

It is estimated that 3 to 4 times more thorium than uranium is present in the Earth's crust (Anantharaman et al., 2008). According to Kara et al., 2008 the average concentration of thorium present in the Earth's crust is 7.2 [ppm]. Furthermore, the half-life of Th-232 is (1.4×10^{10}) years whereas U-238 has a half-life of (4.5×10^9) years.

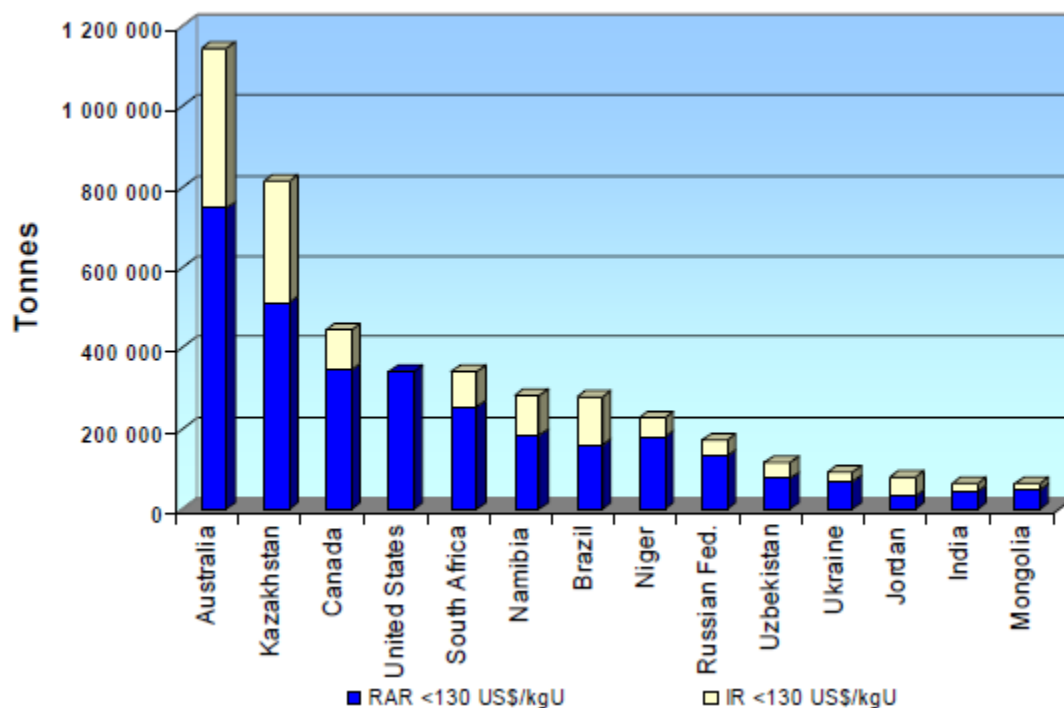


Figure 2.4: The world's Uranium resources (Kara et al., 2008).

Kara et al., 2008 state that 2.6 million tons of uranium resources with mining costs up to 80 [US\$/kg] have been globally identified (see Figure 2.4). This amount is very equally matched when compared to identified thorium resources. In Table 2.3 it can be seen that for the same mining cost of up to 80

[US\$/kg] the identified reasonably assured and inferred global resources recoverable for thorium are about 2.61 million tons (World Nuclear Association, 2009).

Table 2.3: Estimated global thorium resources – 80 [US\$/kg] (World Nuclear Association, 2009).

Country	Tons	% of Total
Australia	489 000	19
USA	400 000	15
Turkey	344 000	13
India	319 000	12
Venezuela	300 000	12
Brazil	302 000	12
Norway	132 000	5
Egypt	100 000	4
Russia	75 000	3
Greenland	54 000	2
Canada	44 000	2
South Africa	18 000	1
Other countries	33 000	1
World total	2 610 000	

In nature thorium is found in diverse rock types as thorite, thorianite and uranothorite. It also occurs as monazite in granite, syenites and pegmatites. Monazite happens to be the most popular source of thorium and it is estimated that resources amounting to 12 million tons are available (Unkown, 2009). The extraction of thorium from monazite is fairly unproblematic and differs significantly from the extraction of uranium. The radioactive waste produced from mining thorium is magnitudes less than in the case of uranium mining (Sokolov et al., 2005).

2.4 Thorium as a nuclear fuel.

Similar to uranium, thorium can also be used as nuclear fuel. Fissile U-233, which is bred from fertile Th-232 through radiative capture of epithermal neutrons, can be recovered from irradiated fuel elements. This U-233 can then be inserted into another reactor as fissile fuel in a closed fuel cycle (World Nuclear Association, 2009).

Lung et al., 1997 state that thorium is a superior fuel with regard to converting a greater fraction of loaded fuel into energy especially when utilized by a high temperature reactor because of the high burn-up achievable and the stability of thorium-based fuel.

According to Sokolov et al., 2005, high temperature reactors with harder neutron spectra and segregated fuel configurations permit neutrons to slow down through the resonance energy region and avoid being parasitically captured. For these reactors Th-232 is a better fertile material. Sokolov et al., 2005 state that the fertile isotopes Th-232 and U-238 have fission cross-sections equal to zero for neutron energies up to 0.1 MeV. The fission cross-section for Th-232, however, remains zero up to neutron energies of 1 MeV and only reaches a value of 0.01 barn above neutron energies of 1.4 MeV, whereas the fission cross-section for U-238 is significant at neutron energies below 1 MeV. This results in Th-232 isotopes being conserved for conversion purposes and in turn contributes to U-233 generation in thermal reactors which is of significant importance for closed fuel cycles.

According to Anantharaman et al., 2008, Thoria (ThO_2) shows superior qualities to Urania (UO_2). These qualities include higher thermal conductivity, lower fission gas release, superior dimensional stability, lower thermal expansion and stable stoichiometry. A few advantages and disadvantages of thorium as a nuclear fuel will now be discussed.

2.4.1 Advantages of thorium.

U-233 has the highest fission neutron yield per neutron absorbed (η) in the thermal and epithermal neutron energy range when compared to any of the other fissile fuel isotopes, namely U-235, Pu-239 and Pu-241 found in the uranium/plutonium fuel cycle (see Figure 2.5) and, therefore, U-233 will be a superior fuel in any thermal reactor type. Th-232 has a much lower fast fission cross section than both U-238 and Pu-240 and, therefore, the replacement of U-238 with Th-232 reduces the neutron economy and thus the conversion ratio in reactors with large fractions of fast neutrons, such as PWRs and fast breeder reactors. Since, however, fast fissions are almost absent in pebble bed reactors, the replacement of U-238 with Th-232 has a beneficial effect on their conversion ratios because more Th-232 is available to capture neutrons and there is no detrimental effect on the neutron economy.

According to Lung et al., 1997, the fission products produced from U-233 are less poisonous to neutrons than those created from U-235. This implies that less neutrons will be parasitically absorbed by the fission products produced from U-233, thereby benefiting the neutron economy.

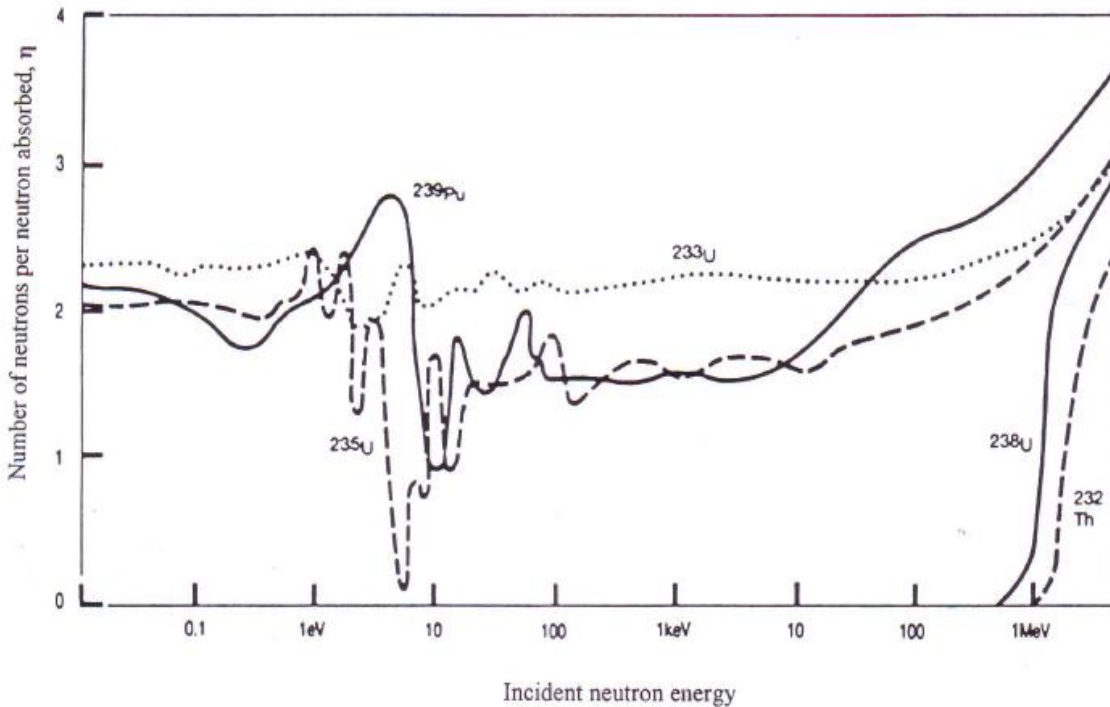


Figure 2.5: Fission neutron yield for fissile isotopes in the thermal and epithermal neutron energy range (Lung et al., 1997).

The conversion ratio, which is defined as the number of fissile isotopes produced per fissile isotope consumed, is higher in the thorium fuel cycle during long term irradiation due to a higher fission neutron yield and a larger absorption cross section of Th-232 than U-238 (see Table 2.4). As a result the fissile generation capability of Th-232 is higher over long periods. Therefore, Th-232 is a better fertile fuel than U-238 in terms of breeding fissile fuel. As a result the fuel ore requirement and the fuel enrichment per unit energy for the thorium-based fuel cycle is reduced (Kazimi et al., 1999).

Table 2.4: Fertile neutronic properties (Kazimi et al., 1999).

Parameter	Th-232	U-238	U-234	Pu-240
Thermal† Absorption Cross Section σ_a [barns]	4.62	1.73	63	203
Epithermal Absorption Resonance Integral RI_a [barns]	85.6	278	660	8500

†Average over Maxwellian spectrum at 300°C (0.05 eV).

Lung et al., 1997 state that at high temperatures U-233 retains its favourable neutronic properties over the other fissile isotopes U-235 and Pu-239 which is of specific relevance to HTR's as they operate at higher temperatures than LWR's (see Table 2.5).

The capture cross section in the thermal energy range of U-233 is the lowest with a value of 54 [barns] compared to the other fissile isotopes U-235 with a value of 100 [barns] and Pu-239 with a value of 267 [barns], which is a significant factor in making U-233 the superior fissile fuel (Sokolov et al., 2005).

Table 2.5: Values of η_{th} , the average number of fission neutrons emitted per neutron absorbed in a thermal flux at varying temperatures (Lamarsh et al., 2001).

T [°C]	U-233	U-235	Pu-239
20	2.284	2.065	2.035
100	2.288	2.063	1.998
200	2.291	2.06	1.947
400	2.292	2.05	1.86
600	2.292	2.042	1.811
800	2.292	2.037	1.785
1000	2.292	2.033	1.77

The potential neutronic advantages of using a thorium-based fuel cycle due to the higher fission neutron yield per neutron absorbed of U-233 at thermal energies and the lower epithermal resonance capture to fission ratio of U-233 can be seen in Table 2.6. Pu-241 is the only isotope that performs slightly better when the fission neutron yield per neutron absorbed at epithermal energies is considered (see Table 2.6).

Table 2.6: Fissile Neutronic Properties (Kazimi et al., 1999).

Parameter		U-233	U-235	Pu-239	Pu-241
Thermal† Absorption and Fission Cross Section [barn]	σ_a	364	405	1045	1121
	σ_f	332	346	695	842
	$\alpha = \sigma_c / \sigma_f$	0.096	0.171	0.504	0.331
Neutron Yield per Neutron Absorbed in Thermal Range	η_{th}	2.26	2.08	1.91	2.23
Epithermal Absorption and Fission Resonance Integral [barn]	RI_a	882	405	474	740
	RI_f	746	272	293	571
	$\alpha = RI_c / RI_f$	0.182	0.489	0.618	0.296
Neutron Yield per Neutron Absorbed in Epithermal Range	$\eta_{epithermal}$	2.1	1.63	1.77	2.29
Neutron Yield	ν	2.48	2.43	2.87	2.97
Delayed Neutron Yield	β	0.0031	0.0069	0.0026	0.005

†Average over Maxwellian spectrum at 300°C (0.05 eV).

ThO₂ has superior stability to radiation, a higher thermal conductivity and a lower coefficient of thermal expansion than UO₂ and, therefore, benefits the safety characteristics of thorium-based fuel cycles. It melts at a very high temperature of 3300°C giving it higher thermal stability than UO₂ which melts at 2850°C (Mulder, 2009). Higher burn-up and temperatures are, therefore, achievable in HTR's using thorium-based fuel cycles (Lung et al., 1997).

According to Kazimi et al., 1999, the void and temperature reactivity coefficients are lower in fuel-rod based reactors using thorium-based fuel cycles than in the uranium-based fuel cycles because Th-232/U-233 fuel benefits less from fast fissions in the case of a hardened neutron spectrum.

Sokolov et al., 2005 state that the intermediate and final storage of thorium-based spent fuel is significantly less problematic than that of uranium based fuel because ThO₂ is inert and does not oxidize whereas UO₂ is prone to easily oxidize to U₃O₈. Sustainable thorium-based fuel cycles will, however, almost always make use of reprocessing which greatly reduces the spent fuel inventory. This is an additional advantage as far as nuclear waste is concerned.

The spent fuel of thorium-based fuel cycles is less of a long-term radiologic threat as significantly less long-lived minor actinides are produced in this fuel cycle than in the uranium-based fuel cycle (Sokolov et al., 2005). If the spent fuel is reprocessed, however, these minor actinides are fissioned in future fuel cycles and only their fission products remain which have half-lives that are a factor 10 shorter. This additionally benefits the final storage of nuclear waste from thorium-based fuel cycles.

Hong et al., 2006 found that HTR's utilizing thorium-based fuel cycles can reduce initially loaded weapons-grade plutonium by around 95 % and civil-grade plutonium by around 88 %. Hong et al., 2006 further state that HTR's can incinerate up to 6 times more plutonium than what they produce over the fuel elements life-time. This is a significant advantage as proliferation risks are dramatically reduced.

2.4.2 Disadvantages of thorium.

Thorium however does have a few shortcomings when compared to uranium. The greatest disadvantage of the thorium-based fuel cycle is that natural thorium does not contain a fissile isotope to drive the nuclear chain reaction in thermal reactors. Uranium contains the only natural occurring fissile U-235 isotope. The thorium fuel cycle is, therefore, dependent on natural occurring fissile isotopes like U-235 or artificially produced Pu-239 from the uranium fuel cycle to initiate the nuclear chain reaction (Anantharaman et al., 2008).

Lung et al., 1997 state that in order for thorium-based fuel cycles to be sustainable the produced U-233 will have to be recovered and, therefore, reprocessing of spent fuel elements will become an essential component in thorium-based fuel cycles. Unfortunately chemical reprocessing is currently very expensive and, therefore, reprocessing is about an order of magnitude more expensive than similar fuel produced from fresh natural uranium. Therefore, reprocessing-based fuel cycles will only become economical if the reprocessing costs were to be reduced dramatically or the price of natural uranium were to increase dramatically.

A further disadvantage of specifically the NB-HTR's Th-232/U-233 fuel cycle, is the significantly smaller value of the delayed neutron fraction (β) for U-233, in comparison to U-235 used in the uranium fuel cycle (see Table 2.6). The delayed neutron fraction is a very important parameter associated with reactor control. The smaller the value, the more difficult it would be to control a reactor. The smaller value for U-233, therefore, necessitates that the control and shutdown systems respond more rapidly to nuclear transients (Kazimi et al., 1999). In neither the operation of the AVR or the THTR controllability was not experienced to be a problem at all.

In the decay chain in which Th-232 absorbs a neutron to become U-233 there are a number of possibilities that U-232 is formed which is associated with highly energetic gamma radiation. This radiation poses problems in handling the spent fuel. Figure 2.6 shows the nuclear reactions that form U-232. This characteristic is however seen as providing the intrinsic proliferation attributes to Th-based fuels.

The two nuclear reactions that are the major contributors to the formation of U-232 are the (n,2n) reaction with Th-232 and the (n,2n) reaction with Pa-233 and U-233. However, the predominant radiological impact is caused by the daughter isotopes of Th-228, which originate by the alpha decay of U-232, namely Pb-212, Bi-212 and Tl-208. These daughter isotopes are short-lived and decay by emitting highly energetic gamma radiation requiring remote and automated reprocessing of spent fuel resulting in elevated fuel cycle costs (Sokolov et al., 2005). Figure 2.7 shows how these daughter isotopes of Th-228 are formed from the alpha decay of U-232.

"Slow" neutron

Thorium-230
(Naturally present up to 100 ppm;
~3x Th-232's cross-section)

"Fast" neutron;
ie, >6.36 MeV

Thorium-232
(Naturally occurring isotope;
the fast neutron knocks out
another neutron; the "n, 2n"
reaction)

"Slow" neutron

Thorium-231
(Beta decay;
half-life 25.5 hrs)

Protactinium-231
(Half-life 32500 yrs;
~100x Th-232's cross-section)

Protactinium-232
(Beta decay; half-life
1.31 days)

Uranium-232
(Half-life 73.6 yrs;
parent of Tl-208 a 2.6
MeV gamma emitter.)

"Fast" neutron

Uranium-233
(The thorium based fuel;
fast neutron knocks out
another neutron; the "n, 2n"
reaction)

Uranium-232
(Half-life 73.6 yrs;
parent of Tl-208 a 2.6
MeV gamma emitter.)

"Fast" neutron

Neptunium-237
(Transuranic waste in uranium
fuels; fast neutron knocks out
another neutron; the "n, 2n"
reaction)

Neptunium-236
(50% beta decay;
half-life 22 hrs)

Plutonium-236
(Half-life 2.85 yrs;
alpha particle decay)

Uranium-232
(Half-life 73.6 yrs;
parent of Tl-208 a 2.6
MeV gamma emitter.)

Bruce Hoglund © 1996

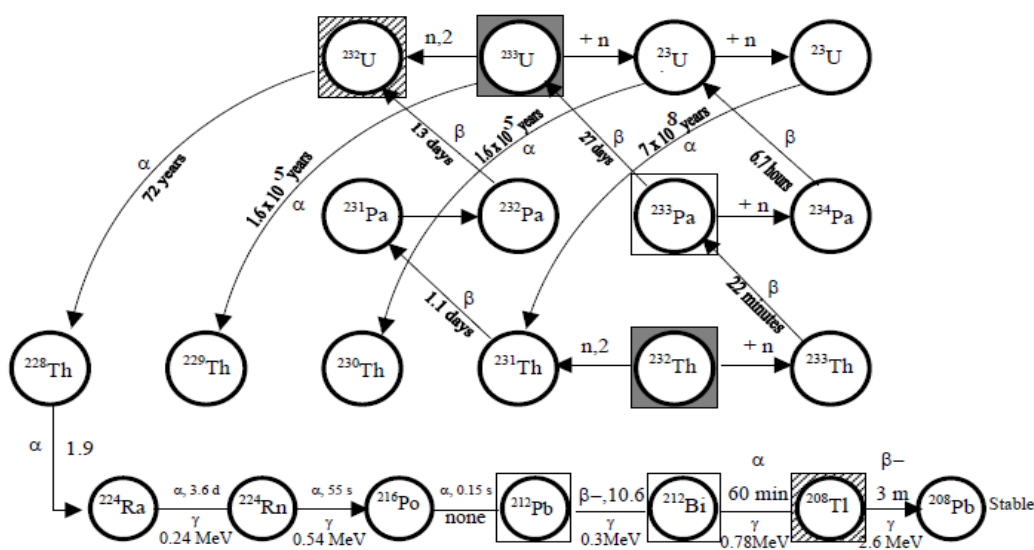


Figure 2.7: The decay chain of the daughter isotopes of Th-228 (Sokolov et al., 2005).

2.5 Protactinium in the thorium fuel cycle.

In both the uranium and thorium-based fuel cycles the fertile isotopes U-238 and Th-232 absorb neutrons during reactor operation and are transformed to isotopes with greater mass numbers (see Figure 2.8). These isotopes exhibit different radioactive properties and influence the reactivity state of the reactor in different ways which depend among other factors on their absorption cross section and half-life.

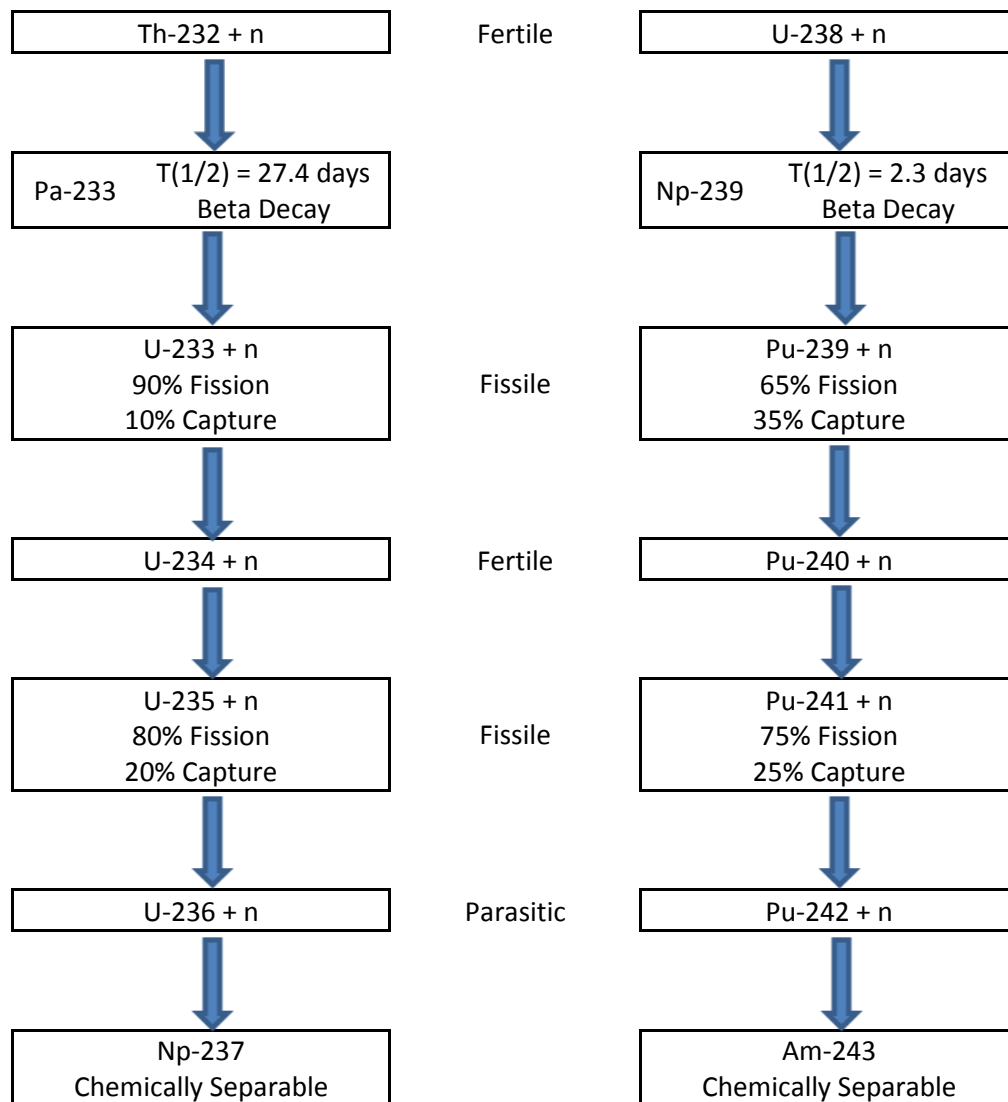


Figure 2.8: Isotope build-up in thorium and uranium fuel cycles (Lung et al., 1997).

A significant difference between these two fuel cycles that should be pointed out is the variation in the half-lives of the isotopes Pa-233 and Np-239. In the conversion of fertile Th-232 to fissile U-233 the Th-232 absorbs a neutron to transform to Th-233 which decays with a β -ray to become Pa-233 (see Figure 2.9). This isotope will in turn decay to U-233 emitting a β and γ -ray with a half-life of 27 days and is about an order of magnitude greater than its U-238/U-235 fuel cycle counterpart Np-239 which decays with the emission of a β -ray and a half-life of 2.3 days (Lung et al., 1997). As a result of this large half-life of Pa-233 an extended period of cooling is required before reprocessing can take place to allow for the completion of U-233 formation from the decay of Pa-233 (Sokolov et al., 2005). This longer half-life also results in the build-up of much higher concentrations of Pa-233, which possibly plays a significant role in the decay heat production in the case of a depressurized loss of forced coolant accident (DLOFC) due to among other factors its longer half-life and must be considered (Mulder, 2009).

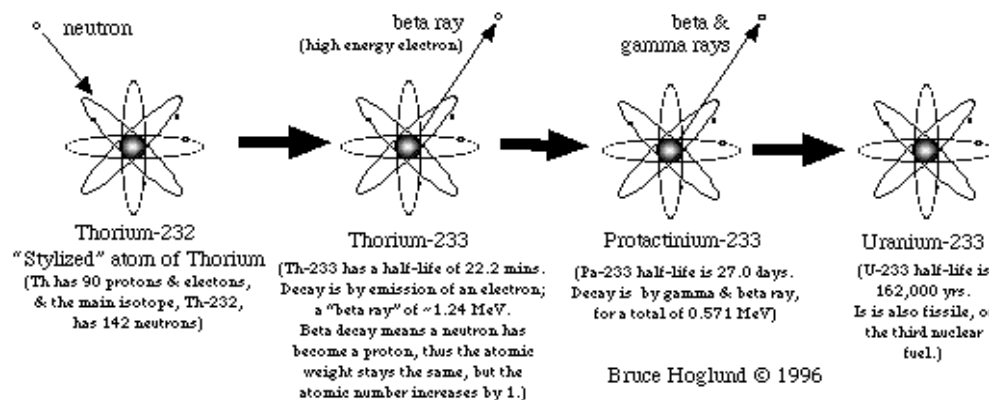


Figure 2.9: The decay chain of Th-232 to U-233 (Mulder, 2009).

An additional problem in this regard is that due to the long half-life of Pa-233 the U-233 concentration will increase for about two months after reactor shutdown and, therefore, the effective multiplication factor will also increase as the concentration of U-233 continues to rise. This, therefore, increases the possibility of the reactor spontaneously starting up and becoming critical after shutdown.

- Chapter 3 -

Method of investigation

3.1 The VSOP computer code system.

The VSOP system of codes has been developed continuously for more than three decades (Mulder et al., 2006). Although the VSOP computer code system can be applied to simulations involving light water reactors (LWR), heavy water reactors and other nuclear reactor systems, it was primarily designed for the research and development of high temperature reactors (HTR) that are cooled by a gas and moderated by graphite (Hong et al., 2006).

Hong et al., 2006 state that the VSOP-A computer code system is used to perform numerical simulations involving the physics of thermal reactors. The code entails a reactor and fuel element set-up, cross-section processing, neutron spectrum evaluation and 2D or 3D neutron diffusion calculations. In response to the input definition, fuel burn-up, fuel shuffling, reactor control, thermal hydraulics and fuel cost calculations are continuously performed (Mulder et al., 2006).

The VSOP-A system of codes allows the analysis and simulation of a reactor from the initial core loading to equilibrium core conditions. With quasi-static approximations transients can be simulated and decay power distributions can be calculated using the fuel element record data (Mulder et al., 2006). Figure 3.1 shows the VSOP code flow sheet for HTR core physic calculations.

Mulder et al., 2006 state that VSOP uses the present worth method for the evaluation of fuel cycle costs over the reactor life-time. By including the isotopic decay and consistent fuel inventory control VSOP can simulate fuel reprocessing and closure of fuel cycles. Furthermore, the results of each calculation can be stored and implemented to perform new investigations (Mulder et al., 2006).

The significant nuclides for albeit the Th/U and the U/Pu fuel cycles are incorporated into the fuel burn-up calculations in the VSOP system of computer codes. Validation studies for various HTR's have shown that the VSOP codes are very suitable for the analysis of both thorium-based and uranium-based fuel cycles (Mulder et al., 2006).

3.2 The 200 MW_{th} pebble bed HTR-MODUL VSOP model.

With the implementation of the VSOP-A computer system of codes the different fuel cycles are simulated to enable a neutronic comparison to be drawn between the different pre-breeder (PB) pebble bed HTR fuel cycles. A further comparison using the VSOP-A codes is made between the reference HTR utilizing a U-238/U-235 fuel cycle and the symbiotic reactor system in which two thorium-based fuel cycles are used. The HTR-MODUL models are setup up using the multi-physics, multi-scale properties of the core configurations (Mulder, 2010). The 200 MW_{th} pebble bed HTR-MODUL VSOP model could be visually imagined as shown in Figure 3.2. This HTR model is characterized by a cylindrical core in either a 2D or 3D coordinate system.

3.3 VSOP code functionality.

The relevant theory and implementation of the sub-programs comprising the VSOP system of codes is discussed in this section. This discussion will be based on the relevant calculational procedures involving the definition of the fuel constitution and homogenization process as well as the geometric setup and material composition for the calculational domain (Mulder et al., 2006).

Mulder et al., 2006 explain that a description of the spherical fuel element motion from the beginning to the end of the reactor core through the so-called flow lines is included in the geometry of the calculational domain. They further state that for a specific design problem these flow lines are experimentally determined and discretised in compliance with relevant numerical parameters. A calculational mesh relating to the fuel shuffling process is formed from the discretisation of the space left in between these so-called flow lines depending on among other factors the reactor to be modelled (Mulder et al., 2006).

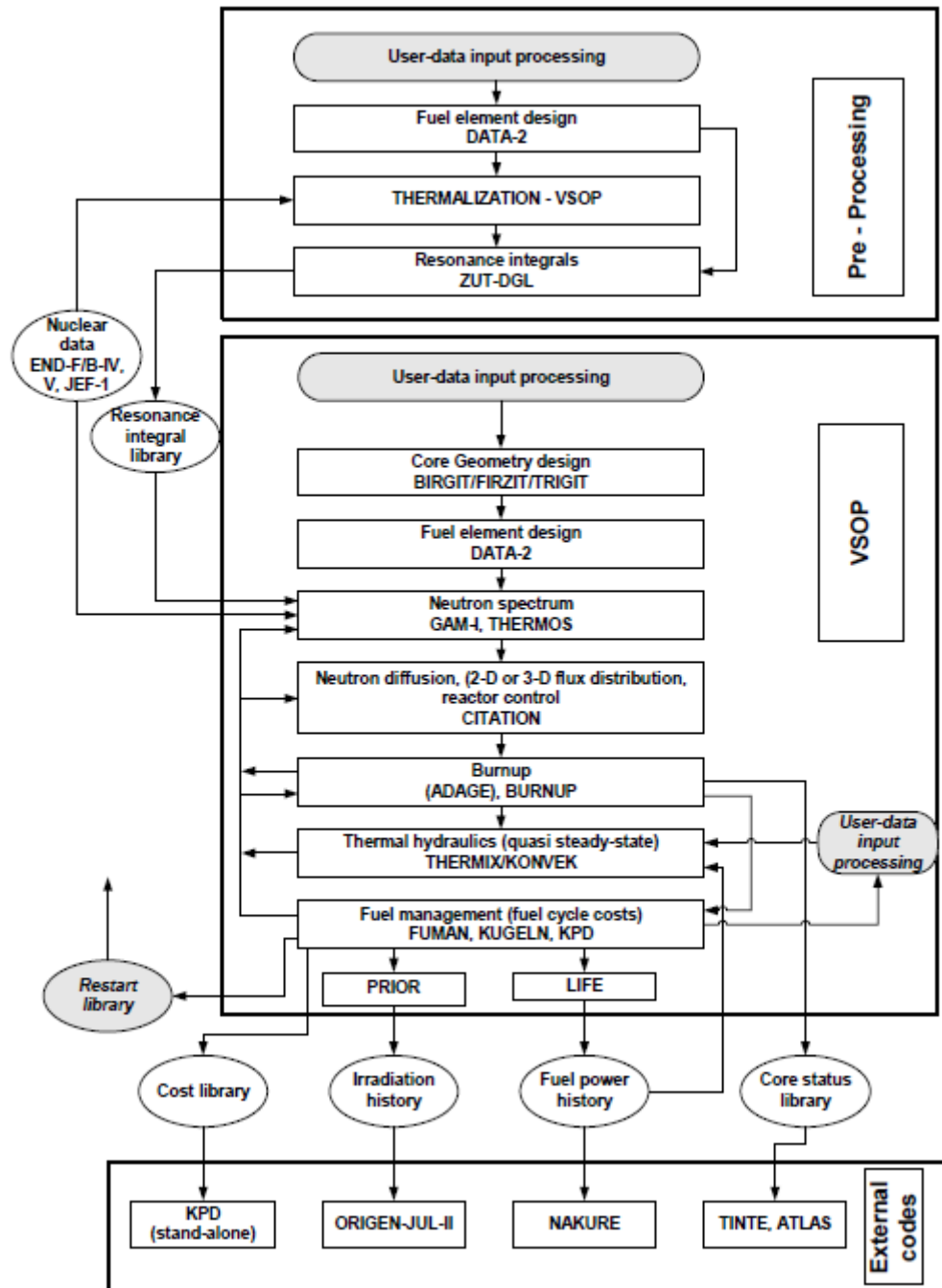


Figure 3.1: VSOP code flow sheet for HTR core physic calculations (Mulder et al., 2006).

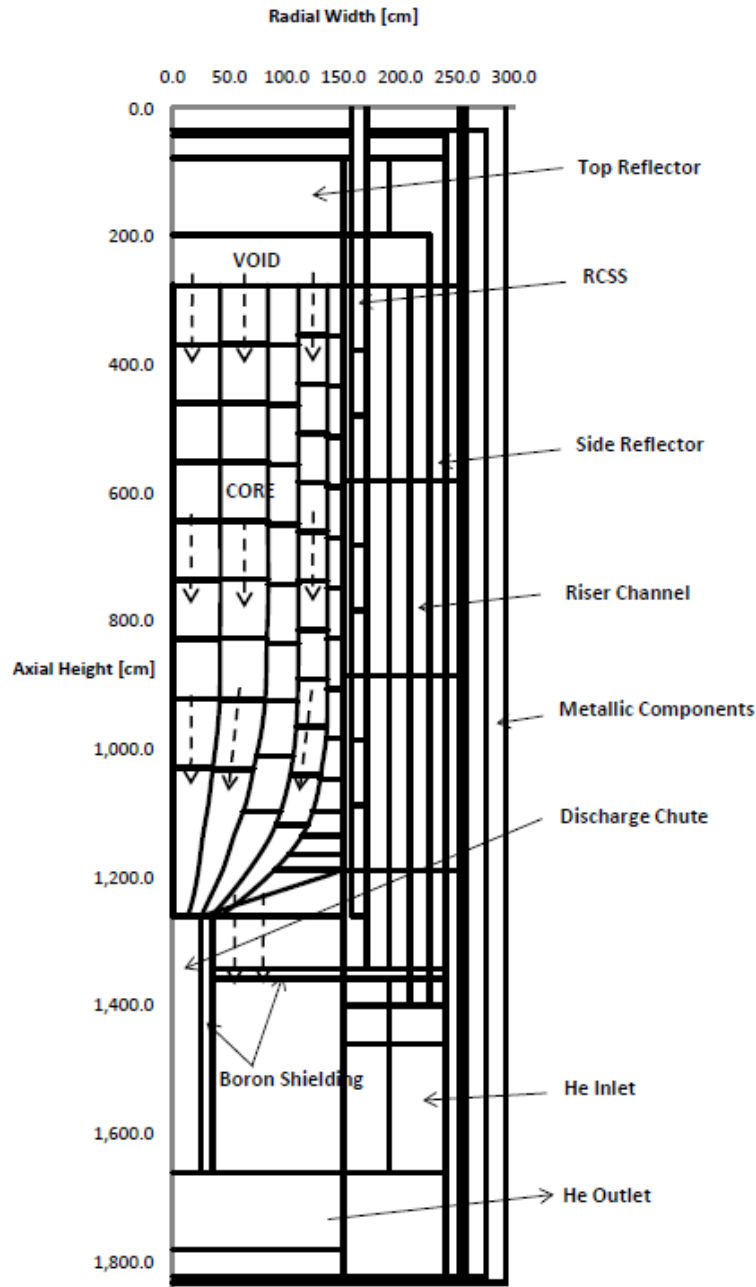


Figure 3.2: Schematic VSOP model used in the simulation (Mulder, et al., 2010).

According to Mulder et al., 2006, converged sets of data can be saved in binary form which then can be used to perform quasi-steady state analysis as part of a restart run, provided input data are arranged accordingly. The simulator has the ability to alter the geometric layout of the core during restart runs.

Mulder et al., 2006 further state that coupled neutronics and thermal-hydraulic calculations are performed in 2-D in VSOP.

3.3.1 Fuel constitution and definition.

Prior to running the actual VSOP simulation the relevant fuel element information must be generated. This information is obtained by means of a subroutine called DATA-2. This code uses the basic fuel element design data and prepares the fuel element information which is later used by the VSOP code (Mulder et al., 2006).

Table 3.1: Design parameters for TRISO coated particle fuel elements.

		Reference HTR	1-ball PB-HTR	2-ball PB and NB-HTR		
Parameter	Dimension	UO2 (LEU)	(Th+U)O2 (HEU)	UO2 (HEU)	ThO2	(U-233)O2
Coated Particles						
Kernel Composition		UO2 (LEU)	(Th+U)O2 (HEU)	UO2 (HEU)	ThO2	(U-233)O2
Kernel Diameter	μm	500	500	240	500	240
Coating Layer Thickness	μm	95/40/35/40	95/40/35/40	95/49/35/40	95/40/35/40	95/49/35/40
Coating Layer Sequence		Buffer/PyC/SiC/PyC	Buffer/PyC/SiC/PyC	Buffer/PyC/SiC/PyC	Buffer/PyC/SiC/PyC	Buffer/PyC/SiC/PyC
Fuel Element						
Heavy Metal Loading	g/FE	7	12	2	16	2
Enrichment of fissile isotopes	%	12	93	93	0.01	93

The basic fuel element design data consist of amongst other specifications the type of fuel element, the heavy metal content in the fuel element, the enrichment and the nuclear material contained in the fuel element as well as the physical dimensions of the fuel element (see Table 3.1). These data are contained in an input file which is read by DATA-2 which in turn generates an output file that is then used as input to the following subroutine of VSOP namely ZUT-DGL which calculates the relevant resonance integrals.

Mulder et al., 2006 state that at present DATA-2 is applicable to HTR's using both prism type and pebble type fuel elements as well as being able to handle 8 different fuel types. These 8 fuel types are UO₂, UC, UC₂, ThO₂-UO₂, ThC-UC, ThC₂-UC₂, PuO₂ and ThO₂-PuO₂.

3.3.2 Resonance treatment.

For a given fuel assembly the resonance absorption cross sections must first be prepared for different temperatures and absorber concentrations before the VSOP program can be executed (Mulder et al., 2006). The resonance integrals for the fertile isotopes Th-232 and U-238 are calculated by the ZUT-DGL code which is a supplementary program (Mulder et al., 2006). The above mentioned resonance absorption cross section data at different temperatures are contained within an input file which is read by ZUT-DGL. The results of this code are then permanently stored in a data set called “resona.d30” which is later used by the VSOP code to perform the different spectrum calculations required for the final results.

3.3.3 Reactivity.

The 2-D spatial flux distribution is calculated with CITATION which is a subroutine of the VSOP code that addresses neutron diffusion. CITATION is also used to calculate the multiplication factor K_{eff} (Mulder et al., 2006). The diffusion EQ employed for these calculations is used in the following form:

$$-\bar{\nabla} D(\bar{r}, E) \bar{\nabla} \Phi(\bar{r}, E) + (\Sigma_a(\bar{r}, E) + \Sigma_s(\bar{r}, E)) \Phi = \int_{E'} \left(\Sigma_s(\bar{r}, E' \rightarrow E) + \frac{\chi_E (\nu \Sigma_f(\bar{r}, E))}{k_{eff}} \right) \Phi(\bar{r}, E') dE'$$

Mulder et al., 2006 further explain that CITATION makes use of a group subdivision containing 33 separate energy groups. After these energy groups have been identified the solution domain is defined by a discrete fine mesh that enhances the numerical process.

Mulder et al., 2006 state that four neutron energy groups are sufficient for the simulation of high temperature reactors. These energy groups consist of one group that represents neutron energies in the thermal range 0 to 1.86 eV, two epithermal groups with neutron energies of 1.86 to 29 eV and 29 eV to 1.11 MeV respectively and one fast group with neutron energies between 1.11 to 10 MeV (Mulder et al., 2006).

3.3.4 Burn-up.

Mulder et al., 2006 state that the core power distribution, the neutron flux solution and the multiplication factor in a discrete domain consisting of mesh points for the energy groups are required

to perform the burn-up calculation. Mulder et al., 2006 explain that burn-up cycles are subdivided into course time intervals in which spectrum and diffusion calculations are done. In order for the normalization of neutron fluxes relative to core power distributions to be performed these course time intervals are divided into a number of finer time intervals. During each of these intervals the flux and power distribution, the weight of all heavy metals and the nuclide concentration of each region are calculated (Mulder et al., 2006). At present 91 individual isotopes can be simulated in these calculations per region. These isotopes include among others the chain from Th-232 to U-236 and then the chain from U-238 to Pu-242.

3.3.5 Decay heat.

According to Mulder et al., 2006, a power histogram of every fuel element as well as its irradiation time and out of pile period is required to calculate the decay heat power of the fuel elements. These fuel element power histograms that are needed for the decay power evaluation are calculated by the auxiliary code LIFE (Mulder et al., 2006). Furthermore a detailed knowledge of each fuel spheres history with regard to the contribution of the fissionable isotopes to the power production as well as the neutron capture rate of Th-232 and U-238 are required (Mulder et al., 2006).

The irradiation history of the fuel batches are contained within the code NAKURE which calculates the local and integral decay power. This code plays a significant role in following the reactor heat-up during a DLOFC accident (Mulder et al., 2006).

3.4 Simulation procedure.

The simulation procedure used to achieve the objects set out in the introduction of this report is divided into two parts. The first part of the procedure addresses the U_3O_8 ore requirement of each reactor and, therefore, the U_3O_8 saving of the symbiotic PB-NB system can be determined. The first part also provides the data required to draw up a comparison of each reactors neutronic performance.

The second part of the procedure deals with the decay heat generation of each of the four fuel cycles. This provides one with the data required to determine what effect protactinium will have on the decay heat produced in the case of a DLOFC accident.

3.4.1 U_3O_8 ore requirement and neutronic performance.

The simulation procedure used to determine the U_3O_8 ore requirement as well as the neutronic performance of each fuel cycle consists of three separate sequences that execute the relevant subroutines of the VSOP-A system of codes discussed above. These sequences will now be briefly discussed.

Sequence 1:

The first step in this sequence is to run the subroutine DATA-2 that prepares the fuel element information that is required by the ZUT-DGL code.

The second step in this sequence is to run the ZUT-DGL subroutine that uses an additional input file that contains information about the resonance absorption cross sections at different temperatures for the relevant fertile isotopes. This code then generates the “resona.d30” and “un.29” files which are needed by the VSOP code to perform the final calculations.

Sequence 2:

In this sequence the auxiliary code called BIRGIT is run which prepares the discrete mesh pattern for the geometry diffusion calculations (Mulder et al., 2006). This code uses an input file setup by the user which describes the reactor core geometry and dimensions. In this file the reflector material, vessel material and void regions are also defined.

The discrete mesh pattern generated by BIRGIT provides data for burn-up, cost evaluation and decay heat production parameters required by VSOP.

Sequence 3:

This is the final sequence of the procedure in which the VSOP code is run to produce the final results of the fuel cycle. These final results contain the relevant neutronic parameters, fuel cycle ore requirement as well as the fissile material required to achieve the necessary thermal power. This code uses all the prior data generated by the subroutines previously discussed as well as an additional input file describing among other specifications the number and thickness of the axial and radial meshes of the core as well as the layer and channel dimensions.

The simulation procedure used to determine the decay heat generation and associated temperatures in the case of a DLOFC accident will now be briefly discussed.

3.4.2 Decay heat generation in the case of a DLOFC accident.

The simulation procedure used to determine the decay heat production and associated temperatures during a DLOFC accident is divided into 3 sequences that execute the relevant sub-routines of the VSOP-A system of codes. These sequences will now be briefly discussed.

Sequence 1:

This sequence is run to determine and verify the temperatures of the equilibrium cycle prior to the DLOFC accident. This sequence is previously discussed under “sequence 2” of the previous simulation procedure.

Sequence 2:

In this sequence the VSOP-A code is run using an input file containing data describing the starter cards. The height and radius of the core as well as the thermal power are indicated on these cards. This input file also contains fuel management data that gives information about fuel reprocessing as well as the decay of the Pa-233 and Np-239 isotopes. Joint diffusion calculation data are also provided in this input file so that the CITATION code which is a sub-routine of the VSOP code can calculate the criticality and neutron flux distribution.

The next step in this sequence is to execute the auxiliary code called LIFE which uses the fuel life history for the decay power evaluation. This is done to save the power histograms of all the fuel spheres in the equilibrium cycle on the hard disk. These histograms will later be used to calculate the decay heat and associated temperatures during a DLOFC accident.

Sequence 3:

This is the final sequence in which a VSOP restart run along with the THERMIX and NAKURE codes are used to calculate the decay heat produced in the different fuel cycles during a DLOFC accident. The associated temperature values of the core regions and fuel are also calculated in this sequence (Mulder et al., 2006).

- Chapter 4 -

Results and discussion

4.1 HTR fuel cycle ore requirements.

The U_3O_8 ore requirement of each of the high temperature reactor fuel cycles are shown in Figure 4.1. These are the results that were obtained from the final VSOP-A simulation.

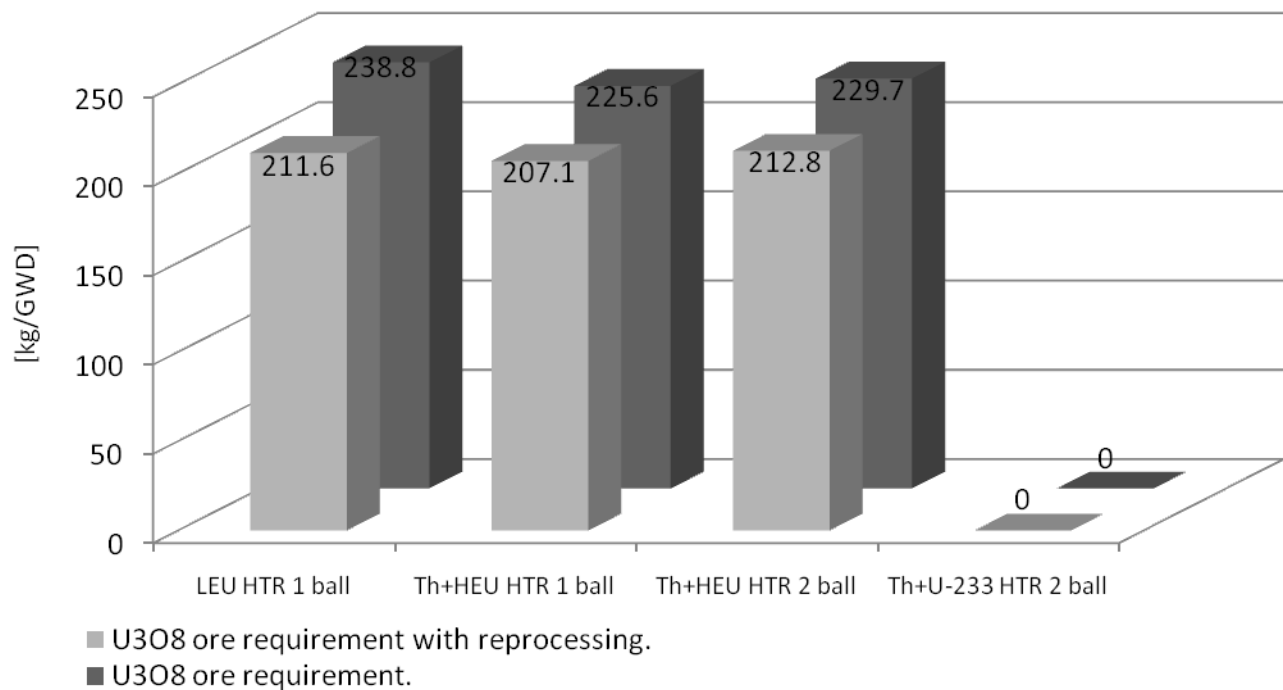


Figure 4.1: HTR fuel cycle ore requirements.

From Figure 4.1 it can be seen that although the pre-breeder (PB) reactors require less U_3O_8 than the reference reactor, this difference is not significant. The difference in U_3O_8 requirement between the net-breeder (NB) and the other reactors is, however, significant as the NB reactor requirement is essentially zero when its dependence on the PB reactor is neglected. Furthermore, it can be seen that there is no noteworthy difference in U_3O_8 ore requirement between the 1-ball and 2-ball PB reactor fuel cycles.

The fissile U-235 requirement per GWD is slightly less in the PB reactors than in the reference HTR (see Table 4.1). This is due to the higher conversion ratio attainable in thorium-based fuel cycles. The NB-HTR required almost no U-235. This is because the U-233, produced in the PB reactors, replaces the U-235 in the fresh fuel of the NB-HTR. This means that a significant fraction of the U_3O_8 requirement of the PB reactors is used to produce U-233 for the NB-HTR and that this extra U_3O_8 should be attributed to the NB reactor. It should further be noted from Table 4.1 that the plutonium discharge from the thorium-based PB-HTR's and NB-HTR fuel cycles is significantly less than that from the reference HTR fuel cycle and therefore limits the risks regarding proliferation associated with Pu-239.

Table 4.1: Fuel supply and discharge of relevant isotopes in HTR fuel cycles.

	Reference 1 ball HTR U-238+LEU-235		Pre-breeder 1 ball HTR Th-232+HEU-235		Pre-breeder 2 ball HTR Th-232+HEU-235		Net-breeder 2 ball HTR Th-232+U-233	
Isotope	<i>Fuel supply [kg/GWD]</i>	<i>Discharge [kg/GWD]</i>	<i>Fuel supply [kg/GWD]</i>	<i>Discharge [kg/GWD]</i>	<i>Fuel supply [kg/GWD]</i>	<i>Discharge [kg/GWD]</i>	<i>Fuel supply [kg/GWD]</i>	<i>Discharge [kg/GWD]</i>
U-233	0	0	0	0.2316	0	0.21	0.7016	0.2323
U-235	0.9456	0.1332	0.8652	0.0713	0.8809	0.0651	0.0011	0.0168
Pu-239	0	0.0644	0	0.0007	0	0.0012	0	0
Pu-241	0	0.0251	0	0.0002	0	0.0004	0	0
Th-232	0	0	11.6169	10.9634	11.3595	10.7595	11.3484	10.6746
U-238	11.5526	11.033	0.0562	0.0555	0.0572	0.0494	0.0008	0.0007
U-236	0	0.1261	0	0.1205	0	0.1229	0	0.0047
Pu-240	0	0.0462	0	0.0003	0	0.0007	0	0
Pu-242	0	0.0211	0	0.0001	0	0.0005	0	0

4.2 Pre-breeder HTR fuel cycle neutronic comparison.

The fractional fissions that each of the relevant isotopes in the pre-breeder HTR fuel cycles experience is listed in Table 4.2. From Table 4.2 it can be seen that more U-233 is fissioned in the 1-ball PB-HTR whereas the fractional fissions of U-235 are greater in the 2-ball PB-HTR. This gives the impression that the 2-ball PB-HTR would be better at breeding U-233 because it burns less than its 1-ball PB counterpart.

This is, however, not the case as the 1-ball PB-HTR proves to have a better conversion ratio (see Table 4.3) than the 2-ball PB and, therefore, discharges slightly more U-233 (see Table 4.1). This is to be expected since the 2-ball PB-HTR burned more U-235 and more Pu-239 and therefore worsens the neutron economy because the fission neutron yield is lower in the thermal and epithermal spectra for these isotopes.

Table 4.2: Fractional fissions of relevant isotopes in the pre-breeder HTR fuel cycles.

Neutron balance	Pre-breeder 1 ball HTR Th-232+HEU-235			Pre-breeder 2 ball HTR Th-232+HEU-235		
<i>Fractional fissions of:</i>	Th-232	0.1	%	Th-232	0.09	%
<i>Fractional fissions of:</i>	U -233	36.11	%	U -233	33.53	%
<i>Fractional fissions of:</i>	U -234	0.01	%	U -234	0.01	%
<i>Fractional fissions of:</i>	U -235	63.5	%	U -235	65.5	%
<i>Fractional fissions of:</i>	U -236	0.03	%	U -236	0.03	%
<i>Fractional fissions of:</i>	Pu-239	0.19	%	Pu-239	0.65	%
<i>Fractional fissions of:</i>	Pu-241	0.04	%	Pu-241	0.19	%

Table 4.3: Global data for PB-HTR fuel cycles.

Global Data.		Pre-Breeder 1 ball HTR Th-232 + HEU-235	Pre-Breeder 2 ball HTR Th-232 + HEU-235
<i>Effective multiplication factor:</i>	k-eff	1.0009	1.0001
<i>Average enrichment:</i>	%	3.83	3.74
<i>Average fuel residence time:</i>	Days	1719	1519.1
<i>Average burn-up:</i>	MWD/T	79990.2	81306.8
<i>Conversion ratio:</i>		0.544	0.508
<i>Fission neutron yield per fissile absorption:</i>	η	2.095	2.095
<i>Capture versus fission in fissile atoms:</i>	α	0.174	0.175

Although it can be seen in Table 4.1 that the 2-ball PB-HTR discharges about two times more Pu-239 than the 1-ball PB-HTR it must be pointed out from Table 4.2 that the 2-ball PB burns three times more Pu-239 and almost five times more Pu-241 than the 1-ball PB. Furthermore, the fact that one has the Pu isotopes now in a separate pebble simplifies proliferation management. This makes the 2-ball PB a

favourable reactor when proliferation is considered. Furthermore, as far as fractional fissions of isotopes are concerned, the two PB-HTR fuel cycles are relatively equally matched.

Table 4.4: Neutrons lost in heavy metals and through core leakage in the PB-HTR fuel cycles.

Neutron balance	Pre-breeder 1 ball HTR Th-232+HEU-235		Pre-breeder 2 ball HTR Th-232+HEU-235	
<i>Neutrons lost in heavy metals:</i>		75.56 %		73.79 %
<i>Neutrons lost in:</i>	Fissile isotopes	47.73 %	Fissile isotopes	47.73 %
<i>Neutrons lost in:</i>	Th-232	25.64 %	Th-232	23.6 %
<i>Neutrons lost in:</i>	Pa-233	0.4 %	Pa-233	0.4 %
<i>Neutrons lost in:</i>	U -233	16.12 %	U -233	14.93 %
<i>Neutrons lost in:</i>	U -234	0.69 %	U -234	0.66 %
<i>Neutrons lost in:</i>	U -235	31.49 %	U -235	32.35 %
<i>Neutrons lost in:</i>	U -236	0.74 %	U -236	0.71 %
<i>Neutrons lost in:</i>	U -238	0.03 %	U -238	0.3 %
<i>Neutrons lost in:</i>	Pu-239	0.11 %	Pu-239	0.36 %
<i>Neutrons lost in:</i>	Pu-240	0.03 %	Pu-240	0.11 %
<i>Neutrons lost in:</i>	Pu-241	0.02 %	Pu-241	0.09 %
<i>Neutrons lost in:</i>	Np-237	0.29 %	Np-237	0.28 %
<i>Neutrons lost in:</i>	Fission products	8.49 %	Fission products	8.7 %
<i>Neutrons lost in:</i>	Xe-135	2.05 %	Xe-135	2.14 %
<i>Neutrons lost in:</i>	Core-leakage	12.51 %	Core-leakage	13.4 %

Once again it can be seen in Table 4.4 that there are no major differences between the two PB-HTR fuel cycles in term of neutron balance. The 1-ball PB-HTR in which the fuel elements contain a mixed (Th+U)O₂ shows slightly better characteristics when the fraction of neutrons lost in Th-232 and the fraction of neutrons lost through core leakage are considered. The higher fraction of neutrons lost in Th-232 leads to greater U-233 yields which, therefore, is a significant advantage for the PB-HTR fuel cycles. Furthermore, less neutrons are lost in Xe-135, which is extremely poisonous to neutrons, and in fission products in the 1-ball PB-HTR which improves the neutron economy.

4.3 Uranium saved in the PB-NB reactor symbiosis.

4.3.1 Two-ball PB-NB HTR symbiosis without reprocessing.

Only the 2-ball PB-HTR fuel cycle will be considered for the scenario where no reprocessing takes place in which the 2 types of fuel elements contain ThO₂ and (HEU-235)O₂ respectively. The reason that only this fuel cycle is considered is because it is the only one in which it would be possible to directly load the spent breeder elements from the 2-ball PB-HTR into the NB-HTR as driver elements without the need for reprocessing whereas reprocessing will be an essential component in the 1-ball PB-HTR fuel cycle.

Table 4.5: Energy, U-233 and U₃O₈ balance for the 2-ball PB-NB reactor symbiosis without reprocessing.

Energy and U-233 balance	Energy [GWD]	U-233 consumption//production [kg/GWD]	U ₃ O ₈ requirement [kg/GWD]
2 ball net- breeder HTR:	1	-0.7016	0
2 ball pre-breeder HTR:	1	0.21	229.7
	3.340952381	0.7016	767.4167619
Balance:	4.340952381	0	767.4167619

Table 4.5 shows that the NB-HTR consumes 0.7016 kg of U-233 per GWD. The 2-ball PB-HTR, however, only produces 0.21 kg of U-233 and consumes 229.7 kg U₃O₈ per GWD. In order to reach the required production of 0.7016 kg of U-233 the PB-HTR will be required to be operated at 3.34 GWD. Another way of interpreting this is to say that 3,34 PB-HTR's operating at 1 GWD each are needed to produce the required amount of U-233 to fuel one NB-HTR to produce 1 GWD of thermal power. The U₃O₈ ore requirement then increases to 767.417 kg U₃O₈ (see Table 4.5).

One can therefore calculate the U₃O₈ ore requirement per GWD of the PB-NB reactor symbiosis as:

$$U_3O_8 \text{ required} = \frac{(767.416 + 0) \text{ kg}}{(3.34 + 1) \text{ GWD}}$$

$$U_3O_8 \text{ required} = 176.7853444 \text{ kg/GWD}$$

From Figure 4.1 it can be seen that the reference HTR consumes 238.8 kg/GWD U₃O₈. One can now determine the amount of U₃O₈ saved with this PB-NB reactor symbiosis as:

$$U_3O_8 \text{ saved} = 1 - \frac{176.7853444}{238.8}$$

$$U_3O_8 \text{ saved} = 1 - 0.7403071374$$

$$U_3O_8 \text{ saved} = 25.969 \%$$

From this calculation it is seen that 25.969 % U_3O_8 ore can be saved by the 2-ball PB-NB reactor symbiosis. Although we save U_3O_8 and avoid the expenses of reprocessing, this saving might not be substantial enough to make this reactor symbiosis an economically viable option. The PB-NB reactor symbiosis where reprocessing of fuel elements is taken into consideration will now be discussed.

4.3.2 Two-ball PB-NB HTR symbiosis with reprocessing.

As mentioned earlier this PB-HTR consists of two different fuel elements containing ThO_2 and (HEU-235) O_2 . In this case, however, we will consider what the U_3O_8 ore requirement will be if the spent fuel elements are reprocessed to recover the essential U-233 and U-235 isotopes. Table 4.6 lists the U_3O_8 supply and discharge quantities for both the 1-ball and 2-ball PB-HTR fuel cycles.

Table 4.6: Pre-breeder HTR U_3O_8 ore requirement.

Ore Requirement	Pre-breeder 1 ball HTR Th-232+HEU-235		Pre-breeder 2 ball HTR Th-232+HEU-235	
	<i>Fuel</i>		<i>Fuel</i>	
	<i>supply</i> [kg/GWD]	<i>Discharge</i> [kg/GWD]	<i>supply</i> [kg/GWD]	<i>Discharge</i> [kg/GWD]
<i>U₃O₈ requirement:</i>	225.6	18.5	229.7	16.9
<i>Seperative work units:</i>	197.6	15.1	201.2	13.7

Table 4.7: Energy, U-233 and U_3O_8 balance for the 2-ball PB-NB reactor symbiosis with reprocessing.

Energy and U-233 balance	Energy [GWD]	U-233 consumption//production [kg/GWD]	U3O8 requirement [kg/GWD]
<i>2 ball net- breeder HTR:</i>	1	-0.4693	0
<i>2 ball pre-breeder HTR:</i>	1	0.21	212.8
	2.234761905	0.4693	475.5573333
<i>Balance:</i>	3.234761905	0	475.5573333

From Table 4.1 it can be seen that the NB-HTR requires 0.7016 kg U-233 per GWD and discharges 0.2323 kg U-233 per GWD. This means that the NB in effect only consumes 0.4693 kg U-233 per GWD if the fuel elements are recycled (see Table 4.7) and the U-233 discharged recovered. The 2-ball PB-HTR produces 0.21 kg U-233 per GWD. As mentioned earlier it requires 229.7 kg U₃O₈. It does, however, discharge 16.9 kg U₃O₈ per GWD as is shown in Table 4.6. This means that if the fuel elements are recycled and the U₃O₈ recovered the PB-HTR only consumes 212.8 kg U₃O₈ as can be seen in Table 4.7. The 0.21 kg U-233 produced is, however, not sufficient. In order to reach the required production rate of 0.4693 kg U-233 per GWD the PB-HTR will in this case have to be operated at 2.23 GWD as Table 4.7 indicates. This increases the U₃O₈ consumption to 475.557 kg/GWD.

One can now therefore calculate the U₃O₈ ore requirement per GWD of the PB-NB reactor symbiosis with reprocessing as:

$$U_3O_8 \text{ required} = \frac{(475.557 + 0) \text{ kg}}{(2.23 + 1) \text{ GWD}}$$

$$U_3O_8 \text{ required} = 147.0146327 \text{ kg/GWD}$$

Once again from Figure 4.1 it can be seen that the reference HTR consumes 238.8 kg/GWD U₃O₈. One now can determine the amount of U₃O₈ saved with this PB-NB reactor symbiosis with reprocessing of fuel elements as:

$$U_3O_8 \text{ saved} = 1 - \frac{147.0146327}{238.8}$$

$$U_3O_8 \text{ saved} = 1 - 0.6156391654$$

$$U_3O_8 \text{ saved} = 38.436 \%$$

From this calculation it can be seen that with reprocessing of fuel elements 38.436 % U₃O₈ is additionally saved. However, when the costs of reprocessing are taken into consideration, which are currently not accurately known since the technology for reprocessing spent Pebble Bed reactor fuel has not yet been demonstrated on a commercial scale, but which can be expected to be high, this saving might not make financial sense and thus the previous scenario in which the fuel is inserted directly from the PB into the NB without reprocessing of fuel elements would probably be the better option. The 1-ball PB-NB reactor symbiosis will now be discussed.

4.3.3 One-ball PB-NB HTR symbiosis with reprocessing.

This PB-HTR differs from the previous one discussed as it uses only one kind of fuel element instead of two different kinds. The fuel element contains fuel particles in the form of a mixed (Th + HEU)O₂.

Table 4.8: Energy, U-233 and U₃O₈ balance for the 1-ball PB-NB reactor symbiosis with reprocessing.

Energy and U-233 balance	Energy [GWD]	U-233 consumption//production [kg/GWD]	U ₃ O ₈ requirement [kg/GWD]
2 ball net- breeder HTR:	1	-0.4693	0
1 ball pre-breeder HTR:	1	0.2316	207.1
	2.026338515	0.4693	419.6547064
Balance:	3.026338515	0	419.6547064

It was previously shown that the NB-HTR consumes 0.4693 kg U-233 per GWD when the fuel elements are reprocessed. The 1-ball PB-HTR, however, only produces 0.2316 kg U-233 per GWD. Furthermore, it can be seen in Table 4.6 that this HTR fuel cycle requires 225.6 kg U₃O₈ and discharges 18.5 kg U₃O₈ per GWD. The 1-ball PB therefore only consumes 207.1 kg U₃O₈ per GWD if the U₃O₈ discharged is recovered by reprocessing the fuel elements. In order to reach the required production rate of 0.4693 kg U-233 per GWD this PB-HTR will have to operate at 2.02 GWD (see Table 4.8). This results in the U₃O₈ consumption to increase to 419.654 kg/GWD.

One can now therefore calculate the U₃O₈ ore requirement per GWD of this PB-NB reactor symbiosis with reprocessing of fuel elements as:

$$U_3O_8 \text{ required} = \frac{(419.654 + 0) \text{ kg}}{(2.02 + 1) \text{ GWD}}$$

$$U_3O_8 \text{ required} = 138.6674704 \text{ kg/GWD}$$

Once again from Figure 4.1 it can be seen that the reference HTR consumes 238.8 kg/GWD U₃O₈. One now can determine the amount of U₃O₈ saved with this PB-NB reactor symbiosis with reprocessing of fuel elements as:

$$U_3O_8 \text{ saved} = 1 - \frac{138.6674704}{238.8}$$

$$U_3O_8 \text{ saved} = 1 - 0.5806845493$$

$$U_3O_8 \text{ saved} = 41.932 \%$$

From this calculation it can be seen that this case offers the greatest saving of U_3O_8 . Although this saving is significant it involves the expenses associated with reprocessing and could prove to be too expensive. The cost evaluation of these respective symbiotic reactor systems is outside the scope of this study but should be performed in the future to determine their financial viability. Figure 4.2 and Figure 4.3 summarize the U_3O_8 ore requirements and savings of the different PB-NB HTR symbiosis considered.

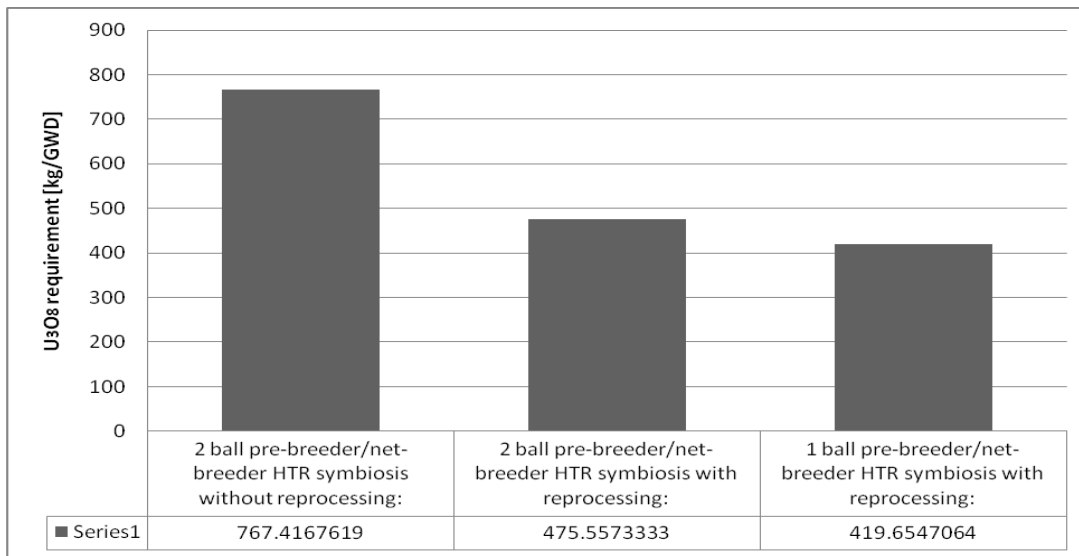


Figure 4.2: PB-NB HTR symbiotic U_3O_8 requirements.

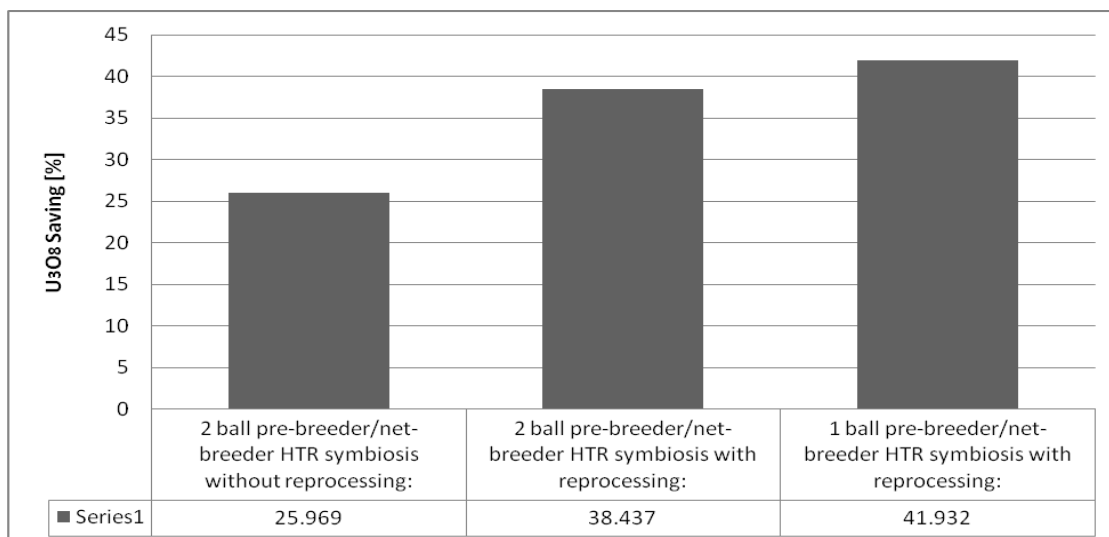


Figure 4.3: PB-NB HTR symbiotic U_3O_8 saving.

4.4 The effect of Protactinium on the decay heat production in the case of a DLOFC accident.

As mentioned earlier there is a significant difference in the isotope build-up between the thorium-based and uranium-based fuel cycles. The most noteworthy difference is the presence of Pa-233 in the thorium-based fuel cycles (see Figure 2.8). This isotope has a significantly longer half-life than its counterpart Np-239 in the uranium-based fuel cycle and, therefore, may contribute more significantly to the decay heat power during a DLOFC accident (Mulder, 2009).

Figures 4.4, 4.5, 4.6 and 4.7 show the decay heat powers and associated fuel temperatures over a 72 h period for each of the four fuel cycles. From these figures it can be seen that the thorium-based fuel cycles have higher fuel temperatures slightly exceeding 1600 °C. Although the allowable maximum fuel temperature in the case of an accident is limited to 1600 °C to prevent fission products escaping from the TRISO coated fuel particles, it can be seen in Figure 4.8 that the release rate from TRISO type spherical fuel elements for the important fission product Cs-137 is below 1×10^{-4} for up to 200 h at 1700 °C and the release rate of Kr-85 is below 1×10^{-5} at a temperature of 1700 °C. According to Kugeler et al., 2003, Ag-110 is the only fission product that diffuses through the TRISO coating layers at slightly higher temperatures. They further state that this isotope is however irrelevant for accident analysis.

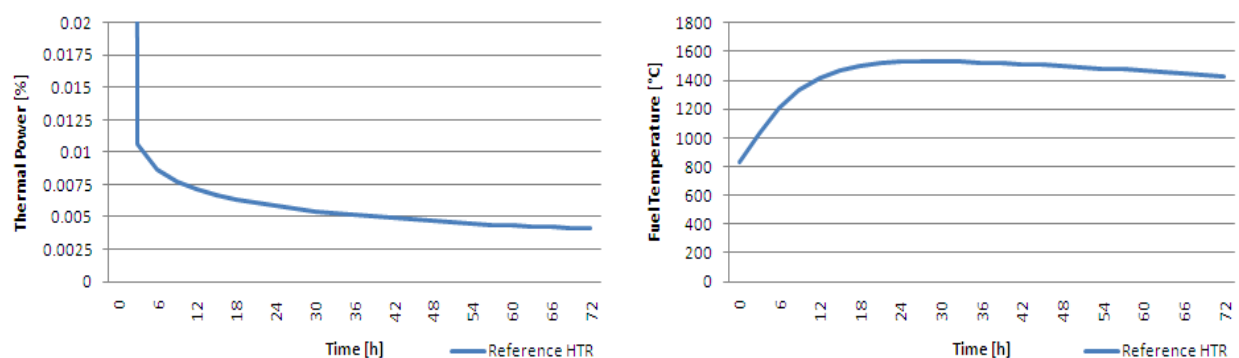


Figure 4.4: Decay heat power and associated fuel temperature of reference HTR fuel cycle.

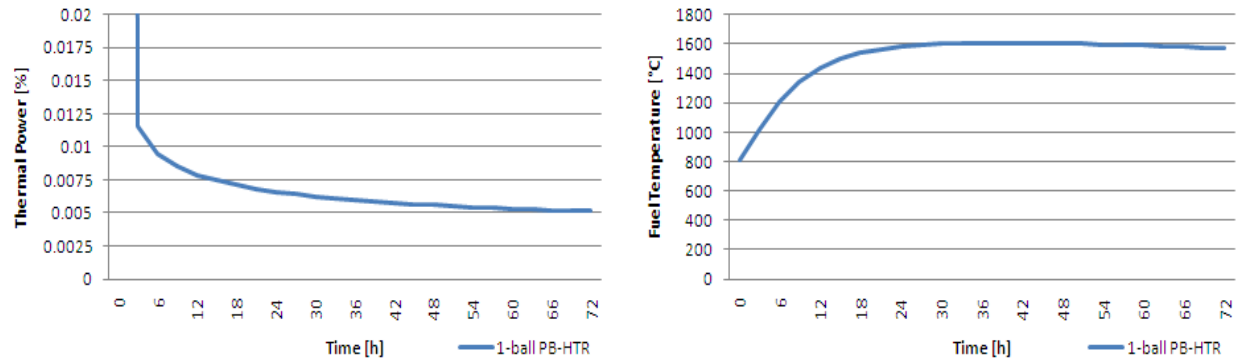


Figure 4.5: Decay heat power and associated fuel temperature of 1-ball PB-HTR fuel cycle.

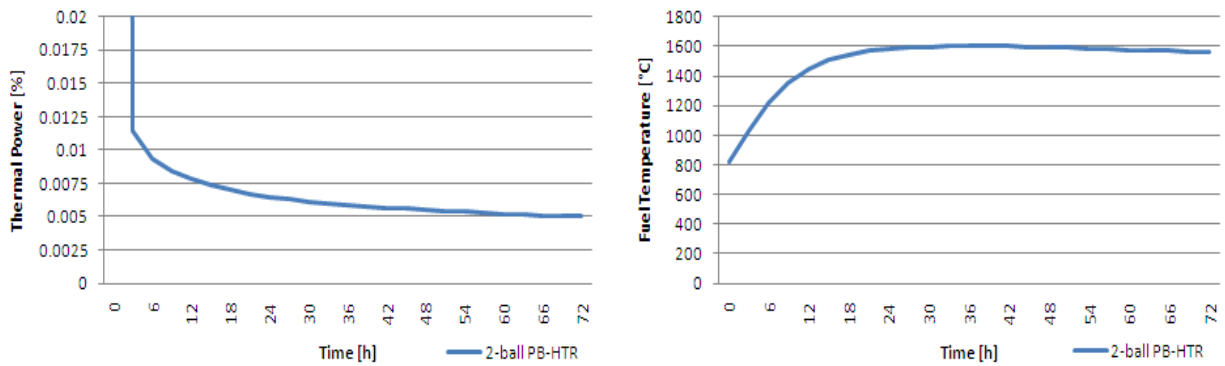


Figure 4.6: Decay heat power and associated fuel temperature of 2-ball PB-HTR fuel cycle.

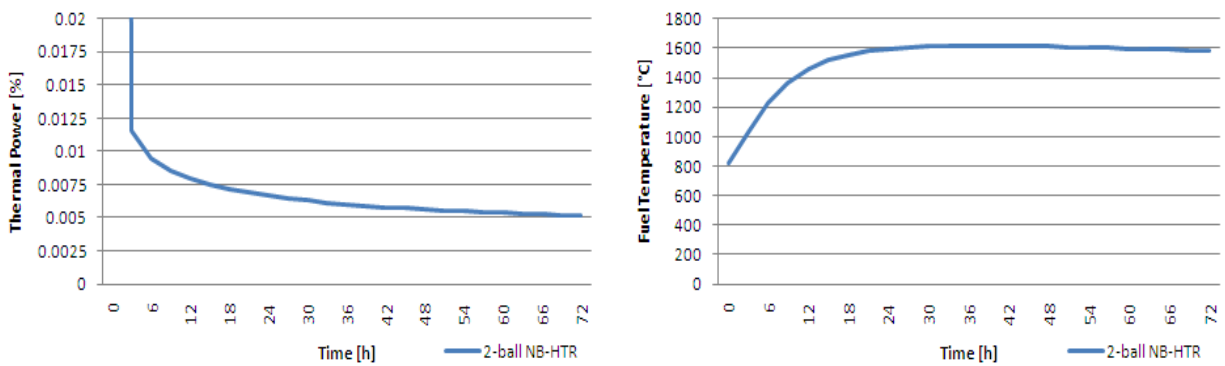


Figure 4.7: Decay heat power and associated fuel temperature of 2-ball NB-HTR fuel cycle.

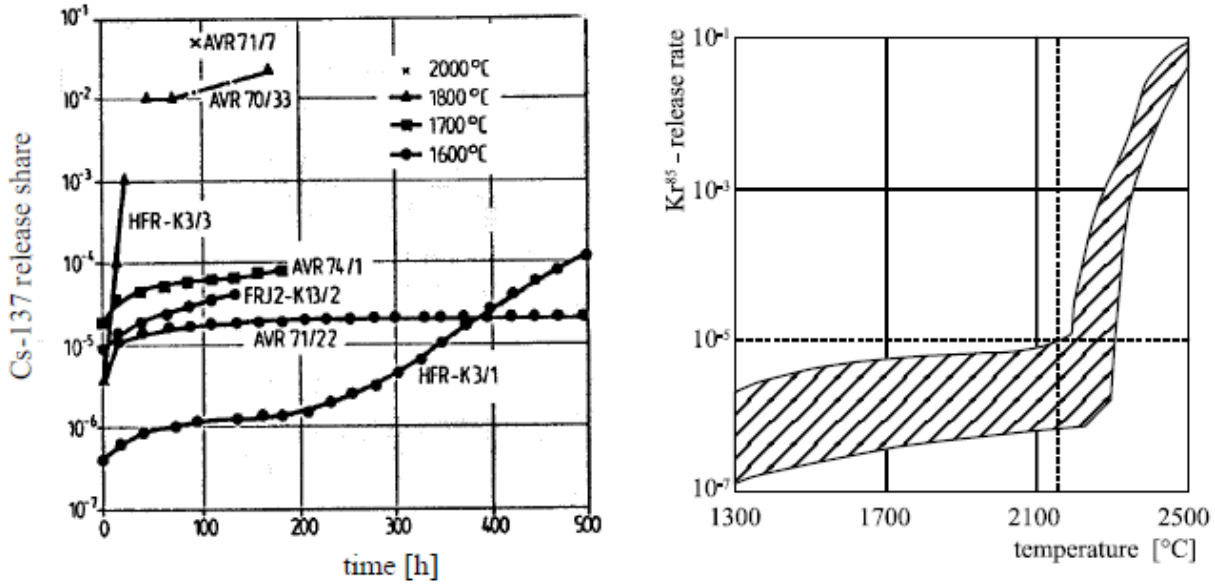


Figure 4.8: Release rates of Cs-137 and Kr-85 from TRISO coated fuel at temperatures above 1600 °C (Kugeler et al., 2003).

One can therefore conclude from Figure 4.8 that no major threat with regard to the release of Cs-137 and Kr-85 from the fuel within the thorium-based fuel cycles will occur at the temperatures shown in Figures 4.4, 4.5, 4.6 and 4.7. We will now discuss the major contributors to the decay heat power over a 72 h period and determine the effect of Pa-233 that occurs in the thorium-based fuel cycles. Table 4.9 lists the decay heat power at four specific time intervals after a DLOFC for each of the fuel cycles.

Table 4.9: Contributors to decay heat power after DLOFC over a 72 h period.

Time after DLOFC:	0.002 h	Dimension	Reference HTR	1-ball PB-HTR	2-ball PB-HTR	2-ball NB-HTR
Decay heat power of fission products:		W	9.78E+06	1.03E+07	1.03E+07	1.03E+07
Decay heat power of U-239,Np-239,Th-233 and Pa-233:		W	4.71E+05	5.79E+05	5.40E+05	5.95E+05
Decay heat power of rest-actinides:		W	5.08E+04	5.38E+04	5.37E+04	5.38E+04
Decay heat power of Cs-134:		W	2.85E+03	7.19E+03	8.60E+03	7.85E+03
Decay heat power of rest-neutron capture products:		W	2.44E+05	2.58E+05	2.57E+05	2.58E+05
Total decay heat power:		W	1.06E+07	1.12E+07	1.12E+07	1.13E+07

Time after DLOFC:	10.375 h	Dimension	Reference HTR	1-ball PB- HTR	2-ball PB- HTR	2-ball NB- HTR
<i>Decay heat power of fission products:</i>		W	1.18E+06	1.24E+06	1.23E+06	1.24E+06
<i>Decay heat power of U-239,Np-239,Th-233 and Pa-233:</i>		W	1.96E+05	2.73E+05	2.54E+05	2.80E+05
<i>Decay heat power of rest-actinides:</i>		W	4.19E+04	4.40E+04	4.39E+04	4.39E+04
<i>Decay heat power of Cs-134:</i>		W	2.85E+03	7.18E+03	8.60E+03	7.85E+03
<i>Decay heat power of rest-neutron capture products:</i>		W	6.37E+04	6.69E+04	6.67E+04	6.68E+04
<i>Total decay heat power:</i>		W	1.48E+06	1.63E+06	1.61E+06	1.63E+06
Time after DLOFC:	24.125 h	Dimension	Reference HTR	1-ball PB- HTR	2-ball PB- HTR	2-ball NB- HTR
<i>Decay heat power of fission products:</i>		W	9.01E+05	9.45E+05	9.42E+05	9.42E+05
<i>Decay heat power of U-239,Np-239,Th-233 and Pa-233:</i>		W	1.66E+05	2.69E+05	2.50E+05	2.76E+05
<i>Decay heat power of rest-actinides:</i>		W	3.81E+04	4.00E+04	3.98E+04	3.98E+04
<i>Decay heat power of Cs-134:</i>		W	2.85E+03	7.18E+03	8.60E+03	7.84E+03
<i>Decay heat power of rest-neutron capture products:</i>		W	5.49E+04	5.76E+04	5.73E+04	5.74E+04
<i>Total decay heat power:</i>		W	1.16E+06	1.32E+06	1.30E+06	1.32E+06
Time after DLOFC:	71.875 h	Dimension	Reference HTR	1-ball PB- HTR	2-ball PB- HTR	2-ball NB- HTR
<i>Decay heat power of fission products:</i>		W	6.50E+05	6.89E+05	6.86E+05	6.86E+05
<i>Decay heat power of U-239,Np-239,Th-233 and Pa-233:</i>		W	9.23E+04	2.56E+05	2.37E+05	2.63E+05
<i>Decay heat power of rest-actinides:</i>		W	3.24E+04	3.44E+04	3.42E+04	3.42E+04
<i>Decay heat power of Cs-134:</i>		W	2.84E+03	7.17E+03	8.58E+03	7.83E+03
<i>Decay heat power of rest-neutron capture products:</i>		W	3.63E+04	3.85E+04	3.83E+04	3.83E+04
<i>Total decay heat power:</i>		W	8.14E+05	1.03E+06	1.00E+06	1.03E+06

At this stage it is important to once again refer to Figure 2.8 and point out that the contribution to the decay heat power of U-239 and Np-239 is relevant to the reference fuel cycle utilizing U-238/U-235 fuel, whereas Th-233 and Pa-233 is relevant to the thorium-based fuel cycles. It is also important to note that the half-life of Pa-233 is 27.4 days and decays by emitting β radiation with an energy of 0.571 MeV, whereas Np-239 has a half-life of only 2.3 days and decays with the emission of β radiation with an energy of 0.722 MeV (Nuc10). Since these decay energies are similar and half-lives significantly different one would expect that as time goes on the effect of the Np-239 on the total decay heat in the uranium-

based fuel cycle will wear off faster in relation to the effect that the Pa-233 will have on the total decay heat in the case of the thorium-based fuel cycles.

From Table 4.9 it can be seen that throughout the 72 h period the contribution to decay heat power of rest-actinides and Cs-137 stays relatively constant for all four fuel cycles. It can also be seen that the effect of the rest-neutron capture products on the decay heat is very similar for all fuel cycles throughout the 72 h period. The decay heat power due to fission products, Np-239 and Pa-233 are by far the largest contributors to the total decay heat power of each fuel cycle throughout the 72 h period. It is important to note from Table 4.9 that the decay heat power contribution of Np-239 and Pa-233, relative to the total decay heat power becomes larger with time over the 72 h period. This can be seen in Table 4.10.

Table 4.10: Contribution to total decay heat power of U-239, Np-239, Th-233 and Pa-233 over 72 h.

<i>Decay heat power fraction of total decay heat power:</i>	<i>Time after DLOFC:</i>	<i>Dimension</i>	<i>Reference HTR</i>	<i>1-ball PB-HTR</i>	<i>2-ball PB-HTR</i>	<i>2-ball NB-HTR</i>
<i>U-239, Np-239, Th-233 and Pa-233:</i>	0.002 h	%	4.468	5.151	4.826	5.284
<i>U-239, Np-239, Th-233 and Pa-233:</i>	10.375 h	%	13.217	16.772	15.811	17.165
<i>U-239, Np-239, Th-233 and Pa-233:</i>	24.125 h	%	14.251	20.411	19.273	20.883
<i>U-239, Np-239, Th-233 and Pa-233:</i>	71.875 h	%	11.336	24.959	23.588	25.530

At this point in order to avoid confusion it must be pointed out that even though the isotopes U-239 and Th-233 are listed in Tables 4.9 and 4.10 along with Np-239 and Pa-233 their contributions are negligible as their half-life is extremely small relative to the half-life of the isotopes Np-239 and Pa-233. It is therefore reasonable to state that the decay heat power contributions given for these isotopes are dominated by Np-239 in the case of uranium-based fuel and by Pa-233 in the case of thorium-based fuel.

It can be seen in Table 4.10 that the fraction of decay heat power associated with Np-239 and Pa-233 to the total decay heat power is similar and relatively small at the start of a DLOFC. This is because the fission products with short half-lives at this stage play a significant role. As these short-lived fission products decay the effects of Np-239 and Pa-233 become more dominant. This can be seen from 10 h onwards in Table 4.10. It is clear in Table 4.10 that Pa-233 contributes significantly more to the decay heat power in thorium-based fuel cycles than Np-239 in uranium-based fuel cycles from 24 h onwards. This effect can be seen in Figure 4.9 that shows the associated fuel temperature over the 72 h period.

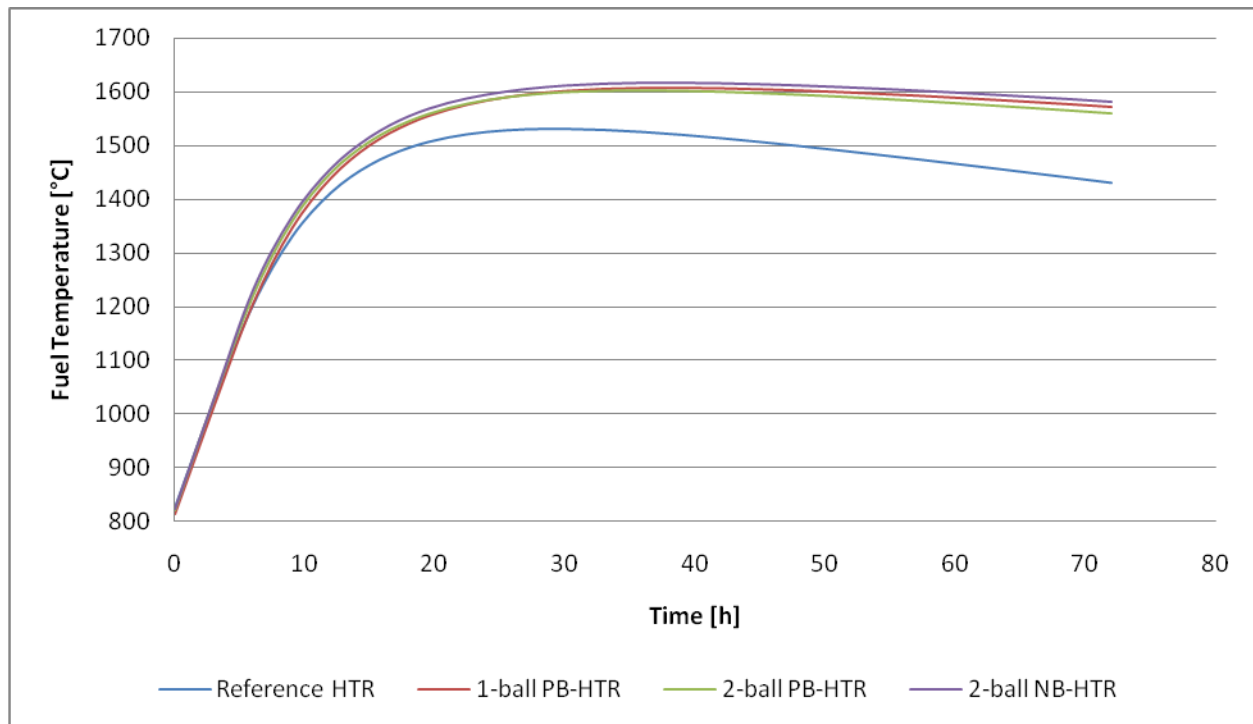


Figure 4.9: Associated decay heat fuel temperatures over a 72 h period.

It can be seen in Figure 4.9 that at around 24 h after DLOFC there is a distinct difference in fuel temperatures between the uranium-based reference HTR and the thorium-based PB and NB-HTR fuel cycles. This divergence occurs due to the significantly larger contribution of Pa-233 to the total decay heat power as time after DLOFC increases and can clearly be seen in Table 4.10. This increasing decay heat contribution can be related to the difference in half-lives between the Np-239 in the uranium-based fuel cycle and the Pa-233 in the thorium-based fuel cycle. At the 24 h mark after a DLOFC the decay heat power due to Pa-233 is around 5% higher than the decay heat produced by the Np-239. This margin increases to 13.5% within the next 48 h. One therefore can conclude that Pa-233 will indeed increase the decay heat power generation in thorium-based fuel cycles during a DLOFC accident and therefore additionally extend the required cooling period. One recommendation will be to reduce the continuous power rating of the reactor.

- Chapter 5 -

Conclusions and recommendations

5.1 HTR fuel cycle ore requirements.

From the results presented in Chapter 4 it was found that the 1-ball PB-HTR fuel cycle utilizing fuel in the form of $(\text{Th-232}+\text{HEU})\text{O}_2$ required the least U_3O_8 for the fuel cycles dependent on U-235 as fissile fuel. The NB-HTR using Th-232/U-233 fuel requires no U_3O_8 ore but is, however, indirectly dependent on U_3O_8 from the PB-HTR. The reference HTR utilizing the conventional uranium fuel cycle on the other hand requires the most U_3O_8 .

5.2 Pre-breeder HTR fuel cycle neutronic comparison.

It was found that the 1-ball PB-HTR has a better conversion ratio than the 2-ball PB-HTR and, therefore, discharged more U-233 making it the favourable PB to supply the NB-HTR with fissile fuel. It was, however, determined that the 2-ball PB-HTR could supply the NB-HTR with fissile fuel without the need of reprocessing the spent fuel elements as would be the case with the 1-ball PB-HTR. This is a very attractive option as the large expenses associated with reprocessing spent fuel elements would thus be avoided.

Although the 1-ball PB-HTR shows a better neutronic performance than the 2-ball PB-HTR in terms of conversion ratio the overall performance of the two PB-HTR's is very close. In addition to being able to supply the NB-HTR with U-233 without the need of reprocessing the 2-ball PB-HTR burns three times more Pu-239 and almost five times more Pu-241 than the 1-ball PB-HTR. Proliferation is directly impacted due to the fact that the amount of Pu is reduced drastically in the Th-based fuel cycle on the one hand but on the other the U-containing pebbles can be separately managed. It is therefore recommended that a further follow-up study be conducted in which the economical viability of a symbiosis involving the different PB-HTR's be determined.

5.3 Uranium saved in the PB-NB reactor symbiosis.

The results show that the greatest U_3O_8 saving is made with the 1-ball PB-NB reactor symbiosis in which 41.932 % U_3O_8 is saved. Although this is a significant saving the expenses associated with reprocessing are included. The 2-ball PB-NB reactor symbiosis saves the least U_3O_8 with a saving of 25.969%. The costs of reprocessing spent fuel are, however, avoided in the latter case which could possibly result in this symbiotic system being the only economically viable option.

5.4 The effect of Protactinium on the decay heat production in the case of a DLOFC accident.

It was found from the data shown in Table 4.10 that Pa-233 present in thorium-based fuel cycles contributes significantly more to the total decay heat production during a DLOFC accident than Np-239 present in the uranium-based fuel cycle. The significant effect that Pa-233 has on the decay heat power becomes increasingly evident as time passes. This is because Pa-233 has a significantly longer half-life than Np-239 and decays with similar energies.

It can further be concluded that the effect that Pa-233 has on the decay heat production results in the fuel temperature slightly exceeding the permitted limit of 1600 °C as can be seen from Figure 4.9. This slight increase in fuel temperature will, however, not have any detrimental effects on the release of radioactive fission products from the TRISO coated fuel particles as this fuel can be exposed to relatively higher temperatures of around 1700 °C and still exhibit satisfactory radio-active release rates as shown in Figure 4.8. It may be suggested if necessary that the continuous power rating of the reactor be reduced in order to maintain the maximum fuel temperature during a DLOFC event to remain below 1600 °C.

5.5 Additional conclusions and recommendations.

The implementation of thorium-based fuel cycles can help conserve finite uranium resources especially if used to breed U-233 in a symbiotic PB-NB reactor system. The thorium-based fuel cycles produce very little plutonium and other transuranic isotopes and, therefore, inherently reduce plutonium proliferation risks. In addition HTR's utilizing thorium-based fuel cycles are very effective at burning Pu isotopes and can therefore be used to incinerate Pu from conventional uranium-based fuel cycles.

Thorium-based fuel cycles will always be dependent on uranium ore to a certain extent as uranium contains the only natural occurring fissile isotope needed in nuclear chain reactions. This dependency should be alleviated to a certain extent by using fissile Pu isotopes instead of U-235 in thorium-based fuel cycles as large stockpiles of Pu from conventional fuel cycles already exist. This will serve more purposes than just reducing dependency – it will reduce Pu stockpiles thereby reducing proliferation risks as previously pointed out and additionally contribute in conserving uranium ore that would have otherwise been mined for fissile U-235 needed in thorium-based fuel cycles. It is also suggested that a Th-based fuel cycle be investigated with reactor grade plutonium as driver material.

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