

**ON THE USE OF THORIUM IN
LIGHT WATER REACTORS**

**M.S. Kazimi, K. R. Czerwinski, M.J. Driscoll,
P. Hejzlar and J.E. Meyer**

April 1999

MIT-NFC-TR-016

Department of Nuclear Engineering
Massachusetts Institute of Technology
Cambridge, Massachusetts 02139

On the Use of Thorium in Light Water Reactors

Abstract

The advantages and disadvantages of the use of thorium bearing fuel in light water reactors have been examined several times from the beginning of the nuclear energy era until the late seventies. The recent motivation for re-examining the use of thorium in light water reactors' once-through fuel cycle is enhancing the cycle proliferation-resistance due to reduced plutonium production. Additionally, economic benefits from reducing the initial enrichment needs of high burnup fuels may be obtained. Similarly, it may be possible to rely on the higher melting point and higher thermal conductivity of ThO₂ to enhance the safety margin of the core. Thorium dioxide is the highest stable oxide form of thorium, which may further improve the spent fuel repository performance. The information obtained in previous studies is reviewed to assess its suitability for application to the current fuel cycle conditions. It is concluded that the thorium fuel experience of the past is insufficient to make a judgement on the feasibility and performance of the thorium bearing fuels in the reactors operating under current conditions. The needs for new research and development efforts in the areas of neutronics, fuel behavior, safety and waste performance are outlined.

Acknowledgement

The authors are grateful for the comments they received on this report from Dr. Steve Shultz, until recently vice-chairman of Duke Energy Services Industries.

This review was partially funded by Idaho National Engineering and Environmental Laboratory (INEEL) and by Brookhaven National Laboratory.

Table of Contents

	Page
Abstract	2
Acknowledgements	3
Table of Contents	4
List of Figures	5
List of Tables	6
1. Introduction	1
2. An Historical Review of Thorium Use in Reactors	10
3. Recent Re-Examinations of the Thorium Cycle	13
4. Neutronic Performance R & D Considerations	16
5. Thorium Based Fuels R & D Considerations	17
5.1 Thorium Fuel Element Lessons	17
5.2 Fuel Performance Models	19
6. Thermal-hydraulics and Safety R & D Considerations	21
6.1 High seed power density	21
6.2 Significant differences between seed and blanket power densities and geometries	21
6.3 Tight lattice option for the blanket	23
6.4 Flooding during LOCA	23
6.5 More intensive decay heat generation	24
6.6 Fuel centerline temperature reduction	25
7. Thorium Ex-Core Considerations and Waste R & D Considerations	25
7.1 Waste Characteristics	25
7.2 Natural Thorium	26
7.3 Repository Behavior	27
References	28

List of Figures

	Page
Figure 1. The fission yield of various Fuel Isotopes [Ref. 7]	33
Figure 2. Average Annual ORE Demand vs. Equilibrium Cycle ORE Demand for 2 GWe Plant (75% Capacity Factor and 0.2% Tails) [Ref. 4].....	34
Figure 3. ORE Demand Components for 2 GWe Plant (75% Capacity Factor and 0.2% Tails) [Ref. 4].....	35
Figure 4. Effect of Leakage on Thorium Assessment	36
Figure 5. Neutronic Characteristics of U – Th Mixtures [Ref. 16]	37

List of Tables

	Page
Table 1. Fissile Neutronic Properties	38
Table 2. Fertile Neutronic Properties	38
Table 3. Characteristics of Thorium Experience in Large LWRs.....	39
Table 4. Plutonium Production Rates and Associated Plutonium Compositions for the RTR and a PWR. (Weapon grade composition given for comparison.).....	40
Table 5. Probability of a "Nominal" and a "Fizzle" Yield versus Plutonium Grade	40
Table 6. Fuel Compositions of RTR-Assembly and ABB/CE-Assembly [Ref. 7].....	41
Table 7. Plutonium Production in U and Mixed Th-U Cycles*	42
Table 8. Spontaneous Neutron and Decay Heat*.....	43
Table 9. Overview of the Ex-Core Thorium Fuel Cycle Relative to Its Uranium Counterpart	44

1. Introduction

The use of thorium as a fertile material in nuclear fuel has been of interest since the dawn of nuclear power technology due to the abundance of thorium ore and to the potential for some important neutronic advantages. The absorption of a neutron in thorium-232 produces uranium-233 which has a high tendency to generate neutrons by fission in thermal and epithermal neutron fluxes. Also, the thermal absorption cross section of thorium-232 is higher than that of uranium-238. Therefore, the in-core fissile generation capability of thorium-232 during long term irradiation can be higher than that of uranium-238. 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. The potential economic benefits as well as the enhanced proliferation resistant features of the spent fuel due to reduced plutonium content and to a less reactive mix of the plutonium isotopes have led to a recent resurgence of interest in this fuel.

Early studies have pointed out the potential neutronic advantages of using a thorium based fuel cycle because of the higher net number of fission generated neutrons per thermal neutron absorption in U-233 (η) and the lower value of its epithermal resonance capture to fission ratio (Table 1 and Figure 1). At epithermal energies, the value of η changes the least among all fissile isotopes with the increase in neutron energy. This reduces the reactivity effects of changes in the neutron spectrum due to coolant transients. In principle, the Th-232/U-233 fuel is less affected by spectrum hardening which reduces its void and temperature coefficients. There are other non-neutronic advantages to thorium dioxide. First, it is a highly stable oxide, not subject to oxidation beyond its stoichiometric constitution. Second, its thermal conductivity is about 10% higher than uranium dioxide. Finally, its melting point is about 500 °C higher than uranium dioxide (3300 °C instead of 2800 °C).

The main disadvantage of the thorium cycle is the lack of U-233 in nature, which necessitates the use of a fissile material such as U-235 or Pu-239 to prime the thorium

fuel. Given present day aversion to the use of highly enriched U-235 in commercial fuels, some U-238 would always be included in the fuel. Thus, thorium would only be used as part of a thorium-uranium fuel mixture. Naturally, a question arises about the wisdom of using two materials instead of one, if one alone can do the job. The main advantage of using a mixed Th-U oxide is the significant reduction in plutonium content of the fuel cycle, despite the presence of some U-238. One way to avoid the need for U-238 is to use only thorium-based fuels to make U-233 in accelerators. But that will be highly inefficient at the beginning of the cycle, and will introduce concern about U-233 proliferation. Therefore, an advantage of having U-238 in the fuel to begin with is to denature the U-233 and avoid the proliferation issues.

There are some disadvantages for the thorium fuel. From a neutronic point of view, the epithermal resonance absorption in Th-232 is lower than that in U-238 (Table 2). This may reduce the negative Doppler reactivity feedback in overpower transients. Furthermore, U-233 has a smaller delayed neutron fraction, β , than that of U-235 but comparable to that of Pu-239, thus creating a need for faster response of control systems to transients should uranium-235 not be sufficiently present. The energy yield per U-233 fission is somewhat less than that of U-235 and Pu-239. Hence, more fissions are needed per unit energy production. Also, more fission gases are produced per U-233 fission, although experience has shown that less of that gas is released outside the thorium fuel matrix. Yet another difference between the thorium fueled reactors and the present fuel cycle is the more pronounced positive reactivity introduction during long shutdowns due to the 27 day long half-life of Pa-233, an intermediate product of the absorption on a neutron in Th-232. By contrast, the Np-239 intermediate between the U-238 and Pu-239, has a half-life of 2.3 days. Finally, if reprocessing is involved, a higher radiation field than the uranium cycle fuel will be encountered. The higher field is due to the U-232 present which decays with 1.92 yr half life to Th-228 whose decay chain includes high energy gamma emitters.

With a mindset that focused on reprocessing of spent fuel to recover the fissile materials, early reviews of the comparative performance of fuel cycles in various reactors

(see for example Ref. 1 and 2) tended to conclude that the uranium cycle was more preferable than the thorium cycle in light water reactor systems. In recent years, the operating conditions of nuclear power have changed, bringing more focus on once-through fuel cycles and heightened concerns about proliferation issues of spent fuel storage and disposal. Hence, there is considerable merit for reexamining thorium-based fuels for light water reactors. In what follows, the lessons of past thorium experience will be summarized and the needs for research and development will be outlined.

2. An Historical Review of Thorium Use in Reactors

In the 1960s and 1970s whole core demonstrations of thorium-uranium oxide fuels in LWRs were explored in two types of arrangements:

- a) Mixing thorium oxide with highly enriched uranium oxide in a uniform lattice (the BORAX-IV, Indian Point I PWR, and Elk River BWR reactors), and
- b) Using a heterogeneous arrangement of seed and blanket regions, where the blanket has less uranium and is responsible for most of the in-core fissile generation (second generation core of Shippingport Reactor). In fact, the Shippingport experiment was aimed at confirming the feasibility of net breeding of fissile isotopes in the core and is often referred to as the Light Water Breeder Reactor (LWBR).

Table 3 summarizes some of the characteristics of the three large LWRs that experimented with a thorium based fuel cycle in the US.

The early work was done under a considerably different set of circumstances than at present. First, it preceded the current proscription of a 20% uranium enrichment limit for proliferation-resistance purposes, hence it relied on mixing highly enriched uranium with the thorium. Furthermore, the work was done when the expected average burnup of LWR fuel was about 25 MWD/kg, about half of the average burnup expected in today's reactors. Finally, the work was done while reprocessing of the spent fuel for recycling of its fissile content was the prevailing expectation. Given the high gamma energy associated with uranium-232, radiological protection would be more demanding for Th-232/U-233 recycling in comparison to plutonium recycling. That of course would now be viewed as an advantage from a spent fuel proliferation resistance point of view.

The Light Water Breeder Reactor (LWBR) program at the Shippingport station, the very station that first demonstrated the Pressurized Water Reactor concept for commercial power generation, operated the core from 1977 until 1982. The results confirmed that the ratio of the fissile content of the fuel at the end of operation to that at

the beginning of operation was about 1.0139. The effort also identified some shortcomings in the LWBR technology relative to LWR practices at that time, including lower power density of the core (30%), the need for high U-235 enrichments in the early stage of deployment, the more complicated design of a movable seed region, the more complicated recycling of U and Th compared to recycling of U and Pu and the extra shielding needed in the fabrication process [3].

In the mid 1970s EPRI commissioned a study of the prospects for improvements in the nuclear fuel cycle if thorium is included but with minimum modifications of modern LWRs [4]. The conclusions of the EPRI sponsored study on thorium cycle applications in Combustion Engineering System 80 PWRs can be summarized as follows:

- Use of thorium with recycle can in the long run increase energy output per mined ton of uranium by about 85% beyond the once-through uranium cycle, and by 22% beyond plutonium recycle.
- Even with the above, the thorium cycle may not be economically attractive relative to cycles with poor fuel conservation features because the early years' fuel demand is high, and savings occur in later years (Figs. 2 and 3).
- Comparison of the characteristics of uranium and thorium based cores indicates that thorium fueling is feasible, and modifications to a PWR designed to accommodate plutonium recycle do not appear to be required.
- The introduction of a totally new system of advanced converters into the US would probably require more effort and funding than can be justified.

In past attempts to improve LWR fuel cycle efficiency, it has been recognized [5 and 6] that the advanced converter concepts attain superior fuel utilization by elimination of major parasitic reactivity control through one of three methods: adoption of on-line refueling (CANDU, molten salt and pebble bed), special geometry to capture leakage neutrons (seed-blanket), or moderation control of neutron capture in fertile material (spectrum shift). At this stage only the CANDU reactor has been successfully deployed worldwide. All these cores can have improve fuel efficiencies when recycling the fissile material as well as with introduction of thorium in addition to uranium in the fuel.

Several reactor types other than the LWRs have experimented with the use of thorium. A brief review can be found in Ref. 7. The most notable among these is the first gas cooled, graphite moderated reactor in the US (Peach Bottom, 40MWe,1967-69) and the first pebble bed reactor in Germany (AVR,15MWe,1966-72). The AVR reactor demonstrated the ability of the thorium fuel encapsulated by pyrolytic graphite to operate up to burnups about 100 MWD/kg. The industrial follow-up reactors (the 300MWe Fort St.Vrain in the US and THTR in Germany) did not prove to be successful enough to produce further orders. Today gas cooled reactor experiments are being built in Japan and in China, but they are not stressing the thorium cycle as much as the plant technology.

Similarly, the gas-cooled electricity generating plant under study in South Africa is not including thorium in the cycle. Only in India are thorium-fueled cores being stressed, but in other than graphite moderated reactors, to utilize the vast reserves of thorium in India.

3. Recent Re-Examinations of the Thorium Cycle

In recent years, the disposition of spent fuel from US reactors has become a much more important factor in affecting the economics as well as the desirability of nuclear power. There are many reasons for the growth of importance of this topic, the most obvious of which is the accumulation of spent fuel in storage pools and dry storage facilities of the nuclear plants and slippage of the deadline by which DOE will be ready to accept the spent fuel for final disposition. The increased cost of management of spent fuel storage and disposition, along with the increased pressure to reduce the overall cost of the nuclear fuel cycle to meet the market competition, has reinvigorated the interest in thorium fuel cycles.

Another reason for revisiting this topic is the ongoing debate on the proliferation resistance attributes of spent fuel from US reactors (which do not recycle) and from other nations (which generally do recycle the Pu and U into LWRs). The total amount of Pu in the spent fuel around the world far exceeds the amount being discarded from the weapons programs in the US and Russia. Hence, the potential for reduction of the amount of actinides in the commercial fuel cycles has attracted significant attention.

It is tempting on the surface to conclude that previous investigations of thorium fuel cycles in LWRs have answered all the important questions and there is no need for any new research effort. That is an overly simplified view since the conditions under which the investigations of the sixties and seventies did not address many of the conditions under which present day reactors have to operate in fissile content, achievable burnup, core temperatures and power densities. Hence, several new initiatives have been undertaken in recent years.

The seed-blanket concept was recently revived through the introduction of the Radkowsky Thorium Reactor (RTR) concept with a once-through fuel cycle utilizing thorium [8]. It is being investigated by a collaboration between the Radkowsky Thorium Power Corporation and the Kurchatov Institute in Moscow with technical support from Brookhaven National Laboratory. Naturally, the collaboration is focused initially on the suitability of the RTR concept for application as a whole core in a Russian VVER light water reactor within the next few years. Nonproliferative, environmental and economic benefits are envisioned due to the elimination of the soluble neutron absorber systems and simplification of the reactor control and waste management needs. Besides making use of less than 20% uranium enrichments in the seed and blanket, the design calls for the use of metallic uranium-zirconium fuel in the seed and irradiating it for three years but leaving the ThO₂ (and denaturant UO₂) blanket fuel rods in core for about ten years. The reduced plutonium production rates and the implied higher probability of a fizzle yield of a weapon due to the mix of isotopes are given in Tables 4 and 5.

Kasten provided an excellent review of the RTR concept in comparison to the PWR [9]. He emphasized that the main attributes of much higher proliferation resistance due to the lower production of Pu isotopes should be considered against an already accepted spent fuel standard for proliferation resistance. Furthermore, the cost advantage claimed depends heavily on the assumed cost of fabrication of thorium fuel elements and on cost of spent fuel transportation adopted for the RTR. Kasten pointed out the need for further demonstration of the following:

- The practicality of exposing the blanket fuel and cladding to 10 years of operations and high burnup (100MWD/kg).
- The safety of the metallic seed with an average power in the seed at about 140% of the PWR and reaching the high burnups of about 150MWD/kg.
- The effect on the plant availability of extensive fuel shuffling of seed-blanket units and the removal of seed units from the assemblies.

Lee and Cho [10] also reported on analytical comparison of a thorium based seed blanket (RTR-like) square lattice PWR reactor to a conventional uranium fueled PWR.

The summary of the higher actinide production rates is given in Table 6. Their findings agree in principle that the plutonium production rate could be about 20% of that of the U cycle at a burnup of over 85 MWD/kg.

Another look at the mixed thorium-uranium dioxide fuel cycle was recently taken by Herring and MacDonald [11]. The initial uranium-235 enrichment was also limited to 20%, with 25% and 35% UO₂ initial content. That was found to allow the fuel to attain 72MWD/kg and 100MWD/kg respectively, thus producing a more economic fuel cycle by 13% and 25% even if the cost of fabrication of the mixed oxide was \$100/kg greater than the pure UO₂ case. Additionally, the Pu production was found to be about 22% of the pure UO₂ case, with a much higher content of Pu-238 thus leading to 5 times greater decay heat production than for conventional fuel and 40 times higher than weapons grade Pu. Their findings are reported in Tables 7 and 8. Note that the nonproliferation advantages of the thorium cycle are evident even when compared to the U cycle at a comparable level of burnup (i.e., at a burnup higher than the current practice in LWRs).

Several groups have recently been interested in thorium based fuel for accelerator assisted nuclear power systems. Furakawa (1990), suggested the breeding of U-233 in accelerator enhanced molten salt systems which would then be used to fuel uranium burners [12]. Rubbia introduced a thorium fueled accelerator driven system for power generation to reduce the radiotoxic waste problems [13]. Los Alamos also suggested the coupling of an actinide burning accelerator assisted system with Th fueled LWRs [14].

A newly established program at MIT is examining the potential advantages of both the seed-blanket concept and the mixed oxide concept. One MIT project, in collaboration with BNL, will examine the adequacy of the RTR arrangement from neutronic and thermal design perspectives both within the VVER triangular lattice arrangement and for a more conventional square lattice. Another MIT project, in collaboration with INEEL, will examine the neutronic and fuel design implications for high burnup of the homogeneous mixed-oxide fuel.

4. Neutronic Performance R & D Considerations

In principle, the seed blanket arrangement does provide the designer with two added degrees of freedom: the ability to better optimize the fuel to hydrogen atomic ratio to enhance the desired neutronic performance and the ability to change the burnup limits of the seed and blanket independently. Both of these will be examined in the MIT work to systematically assess the effects.

A particularly interesting observation from the neutronics point of view is that in cores with a higher conversion ratio, such as projected for the thorium cores, there is a higher potential for errors in predicting the burnup limits. This is shown in a reactivity-burnup plot, where the slope of the decreasing function is smaller in Th bearing cores, so that the impact of an error in the reactivity value is more pronounced (Fig. 4). Hence, neutron leakage effects on the calculations of single assemblies are more important in Th reactors. In fact, the underestimated burnup potential in the Indian Point I first core may be a consequence of the need for more precise calculations of the neutron balance.

On another point, in his review of the physics of absorption in Th and U at various neutron energies, Kasten points out the earlier findings that in a highly packed core (such as in LWRs) the effects of lumping of the fissile material are more negative, hence the LWR thorium cycle may not in principle be as reactive as the uranium cycle due to differences in the absorption resonance integrals (although it might be advantageous in the gas cooled reactor). As pointed out by Crowther in Ref. 15, however, the conversion ratio trend is not monotonically dependent on the Th to U ratio in the fuel, and the maximum conversion ratio is attained by a 50/50 fuel mixture (Fig. 5). In his very early evaluation [16], Crowther found that at a given water to fuel ratio, the increase in Th-232 to U-238 ratio increases the reactivity coefficients. Hence, if it is desired to maintain those coefficients constant, the ratio of water to fuel has to increase. This may decrease the achievable burnup.

Given that most of the earlier observations were obtained for fuel burnups much less than is currently practiced in LWRs, it is likely that the new investigation will find

significant differences between the Th and U fuels where none were found before. This is true of the neutronic behavior but also of other findings. For example, it has been reported that less fission gas release was found in the Shippingport ThO₂ fuel than in UO₂ fuel [17]. However, the lower temperatures of that cladding and fuel leave one wondering if the same can be true at the higher temperatures experienced in today's reactors. Thus, there is a need to re-examine the applicability of the earlier findings to the conditions of the current LWR technology.

5. Thorium Based Fuels R & D Considerations

5.1 Thorium Fuel Element Lessons

Only limited information can be found in literature on the thorium fuel experience of the Indian Point reactor. B&W was the original reactor supplier, and the reactor plant started in January 1963 was closed in October 1974. However, no systematic reporting on the operating conditions and fuel pin conditions exist in literature. It is known that the core reactivity was not as high as predicted by the methods used at the time, and this is often ascribed to the high absorption of stainless steel.

On the other hand the LWBR project provided documentation of its findings in many reports (for example Ref. 17 and 18]. However, some caution should be exercised when translating the LWBR experience into today's conditions. The success of the program depended on conditions that vary in many cases from current LWR conditions.

Here is a capsule description of the LWBR fuel design and experience.

- reactor core regions include: an axially movable seed, a standard blanket; a power flattening blanket, a radial reflector blanket, and thoria reflectors (top & bottom).
- fill gas: helium at atmospheric pressure.

- fuel material – binary fuel pellets [$^{233}\text{UO}_2$ (1.2 to 5.2 w/o) mixed with ThO_2]; thorium fuel pellets (ThO_2) -- each fuel rod contained one or more pellet types.
- coolant chemistry - no boric acid.
- 204 plant swing load cycles – [$>90\%$] to [35%-60%] for [4 hrs to 8 hrs] then return to [$>90\%$]; power changes performed subject to maneuvering constraints.
- operation from 21-Sep-77 to 01-Oct-82 {1836 calendar days}; during that time, 1210 effective full power days (for full power = 236.5 MWt) were produced.
- as-built clad outside diameters (mm) were: seed = 7.78; standard blanket = 14.52; power-flattening blanket = 13.40.
- seed cladding was subject to a re-crystallized anneal (RXA); blanket cladding was cold worked and given a stress relief anneal (SRA).
- rods were placed in a triangular array.
- the peak linear heat rates (kW/m) in destructively examined (DE) fuel rods are: seed = 22.0; blanket = 28.5; reflector = 11.2.
- the peak burnups (MWD/kg) in DE fuel rods are: seed = 53.4; blanket = 25.2; reflector = 4.1.
- best estimate EOL burnup values (MWD/kg) were [peak, average for peak rod]: seed [53.4, 29.8]; standard blanket [23.2, 13.2]; power flattening blanket [25.2, 14.7]; reflector [4.5, 2.2] -- note that these imply large axial peaking factors in all “limiting rods.”

- average reactor coolant temperatures (°C) were [277 to 266]; saturation temperatures (°C) for the RCS were [335 to 328] – compared to the Westinghouse 18-month reference core analyzed by Delgado in Ref. 19 [310] & [345].

5.2 Fuel Performance Models

In examining the fuel performance, there is a need for a model that can accommodate the conditions of operations of the fuel. In particular 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 if it is to last in the core for a much longer time than experienced in LWR conditions and tests to date. There are models for both of these phenomena that are used in analyzing the lifetime uranium fuel conditions in present day LWRs. These models will have to be modified to account for both the thorium physical properties and for the accompanying thermal conditions. The MIT effort will focus on the FLA codes and the FRAPCON-3, presently available at MIT. A short description of the FLA code package is given here.

FLA – FLA is a designation for an MIT package of computer codes for Fuel Lifetime Analysis. Two codes are now available in this package. They were developed by Luis Garcia-Delgado [19]:

- FLA01 -- Code for Cladding Corrosion (Apps H & I of Ref. 19) – The FLA01 code provides a cladding corrosion model and is taken mostly from the work of Forsberg et al. [20]. The code is presently designed for modeling the behavior of low-tin Zircaloy-4 and a representative PWR coolant chemistry. The technique has two advantages: that it is non-proprietary and that it is based on measured oxide thicknesses.

Input (for some specified position in one of the fuel rods): cladding outside diameter and dimensions associated with adjacent coolant subchannels; and the following variables as a function of time at the selected position—linear heat rate, mass velocity, pressure and bulk coolant temperature.

Output: oxide thickness as a function of time.

- FLA02 – Code for Fission Gas Release (Apps J & K of Ref. 19) – The FLA02 code provides a fission gas release (FGR) model taken from work by Weisman et al. [21]. The Weisman model employs parameters developed from experiments using UO₂ fuel rods. For using the model in FLA02, a temperature distribution (temperature versus [r & z] in a single rod) must be supplied as a function of time.

The temperature information is obtained in FLA02 from internal tabulations of fuel centerline temperature and fuel outside surface temperature versus linear heat rate and burnup. The tabulations are obtained from several sources (especially Maki—Ref. 22). Temperatures at other radial positions are found by a parabolic interpolation. The code is based on thermal conduction through cracked UO₂ fuel. Representative clad surface temperatures and temperature-rises through clad and gap are employed. This technique also has the non-proprietary advantage but is less well founded on fuel rod measurements.

Input (for some single rod): linear heat rate and burnup as a function of axial position and time.

Output: fission gas release percentage versus time.

Because metallic fuels have also been proposed for the seed in the more recent studies, a considerable departure from the existing practice, a new model for metallic fuel behavior will be needed. Metallic fuel has been used in research and naval reactors as well as in the EBR-II. In fact, EBR-II data includes some thorium-in-zirconium fuel irradiations in a fast spectrum. There is some expectation from the irradiation experience in EBR-II, that long lived fuels can be developed. However, the EBR-II fuel was relatively spongy and was sodium bonded. Hence, its suitability for the thorium-uranium fuel in a thermal reactor is not to be taken as proven. The effects of water chemistry on the fuel differ from the above mentioned reactors, and will require careful examination.

6. Thermal-hydraulics and Safety R &D Considerations

For the homogeneous mixed oxide case, the design of the assembly is essentially that of present day LWRs. In that case, the thermal performance is likely to be similar to present day reactors. One important difference might come about from the higher burnup conditions and higher number of fissions for the same operating power. This could lead to somewhat higher decay heat levels in the thorium-based fuels. However, at the same pin power, the thorium-uranium fuel will have a higher conductivity that should reduce the stored energy in the pin which might compensate for the higher decay heat during any of the transients.

For the heterogeneous (or seed-blanket) assembly, there are several thermal design issues that need attention. On the positive side, use of metallic fuel in the high energy density region will reduce the thermal energy stored in the fuel and enhance the safety response of the fuel. On the negative side, the seed is much more highly loaded which will increase the power density in the metallic fuel. The design parameters to be investigated should include those proposed for the Radkowski Thorium Fuel (RTF) concept, which is the design of the RTR to fit in present day LWRs.

6.1 High seed power density

Maximum local power density in the seed part of the assembly is 1700 W/cm^3 [23 and 24], which translates to a core-maximum linear heat rate of about 55kW/m. This value is about 1.3 times higher than that of the typical PWR fuel thus challenging the critical heat flux margins. Since the current trend in advanced LWRs is to increase safety margins to make the design more robust and give more time to the operator to cope with a transient, it will be a difficult task to combine the high seed power density with the expected safety margins in advanced reactor designs

6.2 Significant differences between seed and blanket power densities and geometries

There is a large difference between seed and blanket power densities. While the seed power density is 30% higher than for the standard PWR lattice, the blanket power density is only 0.7 of that of a standard PWR. To provide adequate cooling for both the seed and blanket fuel and to avoid large coolant temperature asymmetries between the seed and blanket subchannels, proper flow distribution among the seed and blanket subchannels must be ensured. The easiest solution that is compatible with neutron physics requirements is to employ a tighter-pitch lattice within the blanket fuel. The increased pressure drop in this region results in a smaller flow rate per subchannel. However, higher flow resistance in the blanket subchannels will promote cross-flow from these subchannels to open channels in the seed fuel, in particular at the blanket/seed interface. This phenomenon could result in the extensive depletion of coolant from the blanket subchannels and degradation of cooling of blanket fuel pins. If the seed is completely surrounded by a can wall, this concern is alleviated.

A number of subchannel codes have been developed and successfully used for the analyses of local conditions and critical heat flux margins in a PWR fuel lattice. These codes will provide a good base for thermal hydraulic analysis of the proposed seed and blanket assembly; however, they employ some input data or correlations that will have to be verified, re-evaluated or possibly newly developed to ensure their validity for the subject seed-blanket bundles. In particular, the mixing coefficients, which take into account the turbulent interactions between the individual subchannels, must be known in order to calculate local fluid conditions along each subchannel with adequate accuracy. In the seed and blanket assembly, these coefficients may be a strong function of subchannel position, in particular in the vicinity of the seed/blanket interface, and are not known a priori since most of the experiments have been performed for regular PWR lattices. Before resorting to experiments to obtain these coefficients for the seed and blanket assemblies, a literature search for the available data and state of the art of turbulent mixing will be required. Also, it is to be noted that considerable progress in distributed parameter analysis has been made in the last decade, so it may be worthwhile to look into available computer packages for the full 3-D fluid flow analysis that would allow assessments of the mixing coefficients with reasonable accuracy.

Another uncertainty in the seed and blanket unit lies in the critical heat flux (CHF) correlations. Most of the correlations developed so far are based on a standard PWR lattice and have limited range of validity. The key parameters affecting DNB are pressure, heated length, inlet subcooling, mass flux, local quality and hydraulic diameter. Operating pressure, fuel active length, and inlet subcooling are the same as for the PWR. Also, if the design achieves flow distribution among seed and blanket channels that matches their power densities, the axial profile of local quality should significantly differ from that of the standard PWR fuel. The hydraulic diameters for the proposed seed and blanket fuel are 17.5mm and 9.1mm, respectively, which should fall within the validity range of most correlations (for example the W-3 correlation is valid for heated diameters ranging from 5 to 17.8mm). Therefore, it is expected that the critical heat flux analysis with subchannel codes using available CHF correlations should provide reasonable DNBR values for the RTF design feasibility evaluation as long as the local fluid conditions are calculated correctly. For the licensing purposes of the RTF bundle, the development of an accurate correlation based on an experimental database will most likely be necessary*.

6.3 Tight lattice option for the blanket

As mentioned earlier, the results of Russian studies [25] indicate benefits of a tighter lattice with low moderator over fuel volume ratio in the blanket. If implemented, these changes would significantly affect thermal hydraulics parameters requiring more research and design developments, in particular in the area of mixing between subchannels, critical heat flux and reflooding the tight lattice in a LOCA. Fortunately, research on high-conversion PWRs is applicable also for the tight-lattice design of the blanket part of the RTF bundle. The impact of the heterogeneity in the seed-blanket concept on the results of the tight lattices remains unknown.

6.4 Flooding during LOCA

An important question raised by the new RTF bundle concept is its coolability during a LOCA. Specifically, during the reflooding phase, it is important to deliver coolant to the hot spot in the shortest time possible to prevent cladding temperature rise

* CHF tests on the VVERT SBU fuel assemblies are planned in Russia.

above the specified limit. The hot spot in the RTF bundle is located in the seed part of the assembly with relatively loose pitch with P/D larger by about 25% than that of a standard PWR lattice. The more open lattice in this bundle region is beneficial for flooding due to its lower flow resistance. On the other hand, higher power density will require shorter quench times. In regarding the blanket region, its low power density motivates less attention. Nevertheless the concern arises that the higher flow resistance of the blanket may result in a large diversion of the coolant inventory from the blanket towards the open seed lattice causing insufficient cooling of some fuel rods within the blanket. The experiments [26] for tight PWR lattices for high-conversion reactors showed a significant increase of the quench time in these geometries. Thus, reflooding of the blanket assemblies need to be examined.

6.5 More intensive decay heat generation

Another important factor that needs to be considered is slightly higher decay heat generation rate in the U/Th blanket than in the uranium-plutonium driver. This difference in decay heat derives from two circumstances. First, as a result of extended burnup the amount of fission products in the fuel will be higher increasing the decay power at the end of life. Secondly, the U-232 isotope exhibits high decay heat generation rate (see for example [27]). While this is regarded as a positive factor from the proliferation resistance viewpoint, it increases cooling requirements. Early after plant shutdown, the contribution of the decay heat from U-232 will be very small in comparison to decay power from the fission products (less than 1%). However, during long shutdown periods, decay heat share from U-232 becomes more important. For example, two months after shutdown, decay power share from U-232 will be approximately 10% if one assumes that the fuel contains 1% U-232. Should the content of U-232 increase to 10%, the decay heat generated by U-232 would be comparable to decay heat from the fission products. Since U-232 content is expected to be 1 to 2%, these peculiarities of the RTF should be considered in the shutdown cooling and during the cooling of the discharged fuel. The decay heat aspects also apply to uniformly mixed fuels using thorium and U-233.

6.6 Fuel centerline temperature reduction

Maximum fuel centerline temperature was evaluated to be 570 °C [23]. This is a relatively high temperature for the metallic fuel considering the high thermal conductivity of this fuel type. High fuel temperature is a result of high seed power density. To increase margins, alternative fuel design options will be sought and assessed to reduce fuel centerline temperature to the targeted 500 °C or less.

7. Thorium Ex-Core Considerations and Waste R&D Considerations

An overview of the ex-core thorium fuel cycle conditions relative to the all uranium cycle is given in Table 9. The front end of thorium cycles is likely to pose no challenges requiring further research or development. Because ThO₂ is the highest oxidation state of thorium, the physical changes due to oxidation in the Th-U oxide mixture will depend on the oxidation effects of uranium. However, the existing data on oxygen solubility in unirradiated UO₂ - ThO₂ solid solutions indicates favorable behavior compared to UO₂ [17]. Data on irradiated solid solutions is needed.

The behavior of the spent fuel and the resulting waste in thorium-based fuel cycles have been investigated in the past. While much has been accomplished, a review of the literature indicates the repository behavior of Th requires clarification. Primarily, some thermodynamic data on basic oxide, hydroxide, and carbonate species is lacking. In addition, the behavior of Th intrinsic and pseudocolloids is not well understood. The existing Th speciation and thermodynamic data, including information on colloids, is briefly presented. The behavior of natural Th and its estimated behavior in the repository near field is discussed.

7.1 Waste Characteristics

Previous work on Th fuel indicates some advantages over U based fuel. As expected, the actinide waste produced from Th fuel is much less than from U fuel [28]. Actinide waste production and radiotoxicity for storage times up to 10⁶ years were

calculated. Thorium with highly enriched ^{235}U , including U recycle, produced the lowest amount of Pu with the least radiotoxicity. The comparative radiotoxicity reduction was time dependent, with a maximum of 50 in the time frame examined. The chemical aspects of Th based nuclear fuel for reprocessing were also studied [29]. Favorable results from the reduction of higher actinides were found.

The waste resulting from Th fuel has been studied in relation to direct fuel disposal, formation of tailored ceramics, and reprocessing. As in the Th fuel studies, the long term radiotoxicity of Th based fuel is limited due to the lack of higher actinides [30]. Generally, Th based fuel produces fewer longer lived actinides than U fuel in comparable systems and enrichments. Treatment of the Th fuel for incorporation into stable ceramic phases has been performed [31]. The stable phases formed are robust and based on incorporation into ZrO_2 or formation of orthorombic U_3O_8 . If ceramics are foreseen as the host form for Th waste, this work can provide a basis for potential host phases. If reprocessing is expected in the future, the THOREX or electrochemical processes can be used [17, 32]. THOREX is a solvent exchange process based upon PUREX. The unique chemical feature of Th regarding dissolution time and production of waste has been studied. Grinding, solvent flow, chemical, and other engineering aspects of Th fuel dissolution have also been examined

7.2 Natural Thorium

Over 40 stable Th bearing minerals have been identified in the environment [33]. One of the most common Th minerals is monazite, a phosphate mineral. Thorianite is the Th phase used in fuel, ThO_2 . Oxides, hydroxide, silicates, aluminosilicates, and titanates of Th have also been observed. In the natural Th minerals, formation of other elemental phases is very common. In addition to common cations (Na^+ , K^+ , Ca^{2+}), Th containing minerals may include Ce, La, Fe, Zr, and U.

Since Th is a naturally occurring element, analog studies have been extensively performed [34-45]. The natural analog studies have variable results and conclusion, which can be attributed to the variety of geochemical sites examined. The sorption of Th to other minerals was observed in a number of studies [34,41,43]. This observation

indicates Th migration from a repository should be minimal, and certainly less than oxidized U. Examination of the sorbed Th showed the formation of secondary mineral phases and co-precipitates [35,36,42,44,45]. This observation indicates the initial dissolution of Th and the formation of new, insoluble minerals in the aqueous phase. This implies the geochemistry of the repository will be important in overall retention of Th. A number of studies showed migration and leaching of Th. In these instances, the presence of colloids was a leading factor in the migration of Th. This observation has powerful implications for the behavior of Th at Yucca Mountain. Recent work has shown the presence of Pu colloids at the Nevada Test Site, which shares geochemistry with the aquifer at Yucca Mountain [46]. If Pu transport is facilitated by colloids, it is reasonable to expect similar behavior with Th. In addition to colloids and secondary mineral phases, geochemical speciation studies show the presence of carbonate and oxyhydroxide species [37,39,40,45]. These studies also indicate the need for evaluating the thermodynamic data for the Th carbonate and oxyhydroxide species.

7.3 Repository Behavior

The behavior of Th in a hypothetical repository near field was evaluated [47]. The hydrolysis of Th and U and their subsequent chemisorption on Hanford basalt were studied using a variety of techniques, including x-ray photoelectron and IR spectroscopy. Data obtained indicates mixed complexes of U and Th to be on the basalt surface, the complexes being radionuclide oxides, hydrated oxides (hydroxides), and carbonates. The finding of mixed complexes is a common observation in complicated, environmental systems. The absorption of Th to the mineral matrix should greatly increase retardation.

A great deal of Th thermodynamic data is available. However, species that can be present in the environment, particularly colloids, are not sufficiently quantified. Most of the existing data pertains to hydroxide and carbonate quantification and speciation [48-57]. Solubility experiments have been performed, but thermodynamic data to describe the solubility is not given [51,58-60]. The sorption of Th to some surfaces has been examined [61]. However, surfaces which may be encountered in the Yucca Mountain repository environment have not received attention.

References

1. M.W. Rosenthal et al., "A Comparative Evaluation of Advanced Reactors." ORNL 3686, January 1965.
2. "An Evaluation of Advanced Converter Reactors." WASH-1087, US Atomic Energy Commission, April 1969.
3. W.C. Schick, Jr. et al., "Proof of Breeding In the Light Water Breeder Reactor." WAPD-1612, Bettis Atomic Power Laboratory, September 1987.
4. N.L. Shapiro et al., "Assessment of Thorium Fuel Cycles in Pressurized Water Reactors." EPRI NP-359, prepared for EPRI by C-E Power Systems, February 1977.
5. S. Glasstone and A. Sesonke, "Nuclear Reactor Engineering." 3rd Edition, Van Nostrand, 1981.
6. INFCE, "Advanced Nuclear Fuel Cycle Evaluation." Report of Group 8 of International Nuclear Fuel Cycle Evaluation, IAEA, 1980.
7. M. Lung and O. Gremm, "Perspectives on the Thorium Fuel Cycle." *Nucl.Eng.Des.* **180**, 130-146, 1998.
8. A. Galperin, P. Reichert and A. Radkowsky, "Thorium Fuel for Light Water Reactors-Reducing Proliferation Potential of Nuclear Power Fuel Cycle." In *Science and Global Security* **6**, 1997.
9. P. R. Kasten, A review of Ref. 8, Private Communications, November 1997.
10. K. T.Lee and N. Z. Cho, "Performance Analysis of a Thorium-Fueled Reactor with a Seed-Blanket Assembly Configuration." *Trans. ANS*, **77**, November 1997.
11. J. S. Herring and P. E. McDonald, "Characteristics of A Mixed Thorium-Uranium Dioxide High Burnup Fuel." *INEEL/EXT-99-00094*, November 1998.
12. K. Furukawa, A.Lecocq ,Y.Kato and K.Mitachi,"Thorium Molten Salt Nuclear Energy Synergetics." *J. Nucl. Sci. Technol.*, **27** (12), 1990.
13. C. Rubia, G. Carminati et al, "The Energy Amplifier." A presentation at CERN, 1994. See also C. Rubbia, S. Buono, et al., "A Realistic Plutonium Elimination Scheme with Fast Energy Amplifier and Thorium-Plutonium Fuel," CERN,1995.
14. D.E. Beller and W.C. Sailor, "A Closed ThUOX Fuel Cycle for LWRs with ADTT (ATW) Backend for the 21st Century." *LA-UR-98-4186*, October 1998.
15. R. L. Crowther, "The Mixed Th-232-U-238 Fuel Cycle." *Trans. ANS*, **2**, (2), 1959.
16. R. L. Crowther. "Resonance Capture in Mixtures of U-238 and Th-232." *Trans. ANS*, **4**, (2), 1961.

17. J. Belle and R. M. Berman, editors, "Thorium Dioxide: Properties and Nuclear Applications." *DOE/NE-0060*, Naval Reactors Office, USDOE, August 1984.
18. G.L.Olson, R. K. McCardell and D. Illum, "Fuel Summary Report: Shippingport Light Water Breeder Reactor." *INEEL/EXT-98-00799*, August 1998.
19. L. Garcia-Delgado, M.J. Driscoll, J.E. Meyer, and N.E. Todreas, "Design of an Economically Optimum PWR Reload Core for a 36-Month Cycle." MIT Nuclear Engineering Department Report *MIT-NFC-TR-013*, June 1998.
20. K. Forsberg, M. Limback, and A.R. Massih, "A Model for Uniform Zircaloy Corrosion in Pressurized Water Reactors." *Nucl. Eng. & Des.* **154**, 157-168, 1995.
21. J. Weisman, P.E. MacDonald, A.I. Miller, and H. Ferrari, "Fission Gas Release from UO₂ Fuel Rods with Time Varying Power Histories." *Trans ANS* **12**, 1969.
22. J.T. Maki, "Thermal Effects of Fuel Pellet Cracking and Relocation." SM Thesis, MIT Nuclear Engineering Department, July 1979.
23. A. Galperin and A. Radkowsky, "A Competitive Thorium Fuel Cycle for Pressurized Water Reactors." *Proc. of the Ninth Intl. Conf. on Emerg. Nuc. Ener. Systems (ECENES'98)*, pp. 637-646, Herzliya, Israel, June 1998.
24. A. Radkowsky and A. Galperin, "The Nonproliferative Light Water Thorium Reactor : A New Approach to Light Water Reactor Core Technology." *Nuclear Technology*, **124**, December 1998.
25. M.F. Troyanov et al., "Some Research and Development of Thorium Fuel Cycle in Russia." *Proc. of the Ninth Intl. Conf. on Emerg. Nuc. Ener. Systems (ECENES'98)*, pp. 681-688, Herzliya, Israel, June 1998.
26. J.M. Veteau, A. Dignonnet and R. Derauz, "Reflooding of Tight Lattice Bundles." *Nuclear Technology*, **107**, July 1994.
27. G. Koulikov, A. Chmelev, V. Glebov, and V. Apse, "Equilibrium Th-U fuel Cycle of LWRs with Elements of Protection Against Fissile Materials Proliferation." *Proc. of the Ninth Intl. Conf. on Emerg. Nuc. Ener. Systems (ECENES'98)*, pp. 483-490, Herzliya, Israel, June 1998.
28. V.A. Wichers, "Evaluation of Thorium Based Fuel." *Actinide Waste*. ECN-R-95-007, Netherlands Energy Res. Foundation, 1995.
29. R.J.M. Konings, P.J.A.M. Blankenvorde, E.H.P Cordfunke., and K. Bakker, "Evaluation of Thorium Based Fuel." *Chemical Aspects*. ECN-R-95-007, Netherlands Energy Res. Foundation, 1995.
30. M.V. Dingankar, V. Kalyanasundaram, and M. Srinivasan, "A Reassessment of Long-Lived Actinide Waste Hazard Potential from Thorium-233-Uranium Fueled Reactors." *Waste Mgt.* **12(4)**, 359, 1992.

31. J.G. Pepin, and G.J. McCarthy, "Phase Relations in Crystalline Ceramic Nuclear Waste Forms in the System Uranium Oxide Cerium Dioxide-Zirconium Dioxide-Thorium Dioxide at 1200°C in Air." *DOE/ET/41900-3*, 1981.
32. B. Wojtech, "Dissolution of Thorium Containing nuclear Fuel with Reduced Waste Production." *Chem. Eng. Tech.* **45** (16), 1022, 1973.
33. <http://web.wt.net/~daba/mineral/chem/chem-Th.html>
34. L. Perez del Villar, M. Pelayo, et al., "Mineralogical and Geochemical Evidence of the Migration /Retention Processes of U and Th in Fracture Fillings from the E. Berrocal Granitic Site (Spain)." *J. Contam. Hydrol.* **26**, 45, 1997.
35. M.T. Menager, C. Menet, et al., "Dispersion Of Uranium, Thorium and REE by Water-Rock Interaction Around an Intragranitic Uranium-Vein." Jalerys Mine, Morvan, France. *Appl. Geochem. Suppl.* **1**, 239, 1992.
36. Y. Pan, M.E. Fleet, N.D. MacRae, "Late Alteration in Titanite (CaTiSiO₅): Redistribution and Remobilization of Rare Earth Elements and Implications for Uranium/Lead and Thorium/Lead Geochronology and Nuclear Waste Disposal." *Geochim. Cosmochim. Acta* **57**, 355, 1993.
37. W.R. Alexander, R. Dayal, K. Eagleson, et al. "A Natural Analogue of High pH Cement Pore Waters from the Maqarin Area of Northern Jordan. II: Results of Predictive Geochemical Calculations." *J. Geochem. Explor.* **46**, 133, 1992.
38. P. Vilks, H.G. Miller, D.C. Doern, "Natural Colloids and Suspended Particles in the Whiteshell Research Area, Manitoba, Canada, and Their Potential Effect on Radiocolloid Formation." *Appl. Geochem.* **6**, 565, 1991.
39. T.E. Payne, T.D. Waite, "Uranium and Thorium Migration in Koongarra Groundwater - Implications to Nuclear Waste Disposal." *Chem. Aust.* **58**, 281, 1991.
40. I.G. McKinley, A. Bath, et al., "Results of the Oman Analog Study." *Radiochim. Acta* **44-45**, 311, 1988.
41. M. Gascoyne, J.J. Cramer, "History of Actinide and Minor Element Mobility in an Archean Granitic Batholith in Manitoba, Canada." *Appl. Geochem.* **2**, 37, 1987.
42. P.L. Airey, "Radionuclide Mi**57**, 331, 1992.
43. H.A. Wollenberg, D.G. Brookins, et al., "Uranium, Thorium and Trace Elements in Geologic Occurrences as Analogs of Nuclear Waste Repository Conditions." *NUREG/CP-0052*, 1984.
44. A. Martinez-Aguirre, M. Garcia-Leon, M. Ivanovich, "U and Th Speciation in River Sediments." *Sci. Total Environ.* **173**, 203, 1995.

45. J. Bruno, L. Duro, "Application of Predictive Geochemical Modeling of a Natural System: The El Berrocal Project." *Estud. Geol.* **50**, 411, 1994.
46. A.B. Kersting, D.W. Erfund, et al., "Migration of Plutonium in Groundwater at the Nevada Test Site." *Nature* **397**, 56, 1999.
47. D.L. Perry, "Near Field Chemical Speciation: The Reaction of Uranium and Thorium with Hanford Basalt at Elevated pH." *NUREG/CP-0052*, 1984.
48. N.J. Hess, A.R. Felmy, et al., "Characterization of Th Carbonate Solutions Using XAS and Implications for Thermodynamic Modeling." *MRS Symp.* **465**, 729, 1997.
49. R. Fabriol, G. Ouzounian, and A. Genter, A, "Modeling the Chemical Behavior of Radionuclides in Waste Disposal Site in A Granitic Environment - A Chemical Thermodynamic Approach." *MRS Symp.* **212**, 809, 1991.
50. M. Schweingruber, "Solubility and Speciation Calculations of Uranium, Plutonium, Neptunium, And Thorium In Natural Groundwaters - Theory, Thermodynamic Data and Preliminary Applications." *EIR-Bericht* 449, 1981.
51. H. Bischoff, U. Kramer-Schnabel, et al, "Determination of Formation Constants, Solubilities, and Ionic Speciation of Uranium, Plutonium, and Thorium Aqueous and Saturated Salt Solutions and Modelling Safety Aspects of ILW Repositories." *Mat. Res. Soc. Symp. Proc.* **257**, 331, 1992.
52. I. Grenthe, I. and B. Lagerman, "Studies on Metal Carbonate Equilibria. 23. Complex Formation in the Th(IV)-H₂O-CO₂ (g) System." *Acta Chem. Scan.* **45**, 231, 1991.
53. J. Bruno, I. Casas, and B. Lagerman, "Studies on Metal Carbonate Equilibria." 19. Complex Formation in the Th(IV)-H₂O-CO₂ (g) System. *Inor. Chim Acta* **140**, 299, 1987.
54. J.P. Jolivet, Y. Thomas, and B. Tavel, "Infrared Spectra of Cerium and Thorium Pentacarbonate Complexes." *J. Mole. Struct.* **79**, 403, 1982.
55. J. Dervin, and J. Faucherre, "Étude des Carbonates Complexes de Thorium et de Cérium. II- Constitution des Complexes en Solution." *Bull. Chim. France.* 2926, 1973.
56. J. Dervin, J. and Faucherre, "Étude des Carbonates Complexes de Thorium et de Cérium. I- Solubilité et Nature des Ions Complexes en Solution." *Bull. Chim. France.* 2930, 1973.
57. I. Engkvist, Y. Albinsson, "Hydrolysis Studies of Thorium Using Solvent Extraction Technique. Radiochim." *Acta.* **58/59**, 109, 1992.

58. B. Wierczinski, S. Helfer, M. Ochs, G. Skarnemark, "Solubility Measurements and Sorption Studies of Thorium in Cement Pore Water." *J. Alloys. Compd.* **271**, 272, 1998.
59. J. Barbier, R. Fabriol, G. Ouzounian, "Modeling the Chemical Environment Near a Waste Disposal Site in a Granitic Host-Rock. Influence of Factors Controlling the Solubility of Radionuclides." *MRS. Symp.* **257**, 599, 1992.
60. F.T. Ewart, J.L. Smith-Briggs, H.P. Thomason, S.J. Williams, "The Solubility of Actinides in a Cementitious Near-Field Environment." *Waste Mgmt.* **12**, 241, 1992.
61. D.M. Giaquinta, L. Soderholm, S.E. Yuchs, S. E., Wasserman, S. R., "Hydrolysis of Uranium and Thorium in Surface-Modified Bentonite under Hydrothermal Conditions." *J. Alloys Compd.* **249**, 145, 1997.

Figure 1. The fission yield of various Fuel Isotopes [Ref. 7]

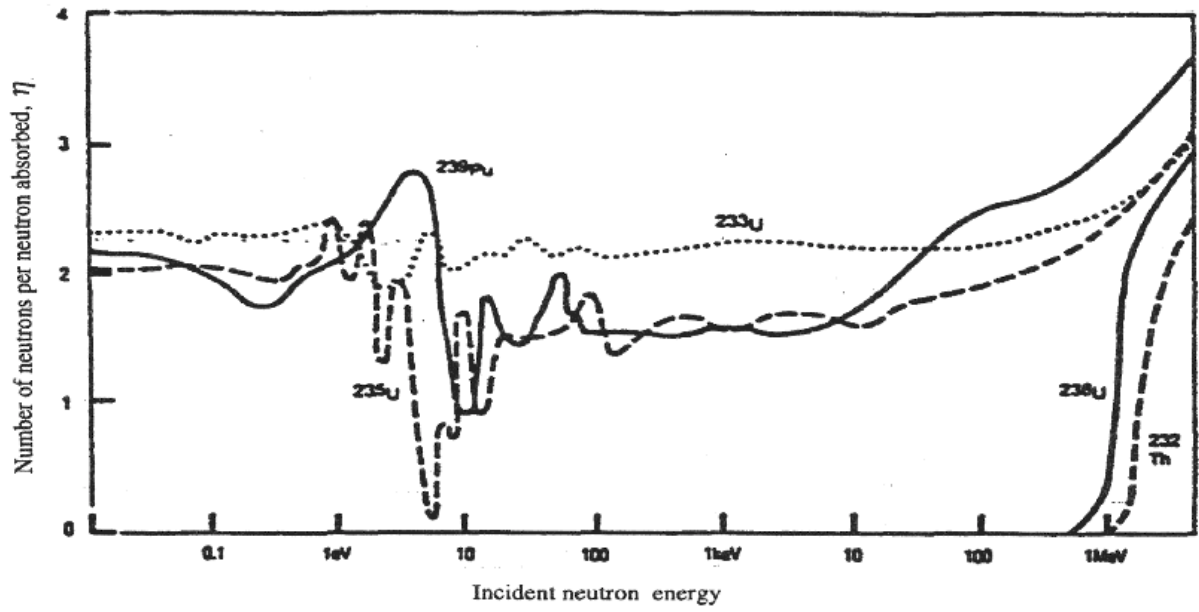


Figure 2. Average Annual ORE Demand vs. Equilibrium Cycle ORE Demand for 2 GWe Plant (75% Capacity Factor and 0.2% Tails) [Ref. 4]

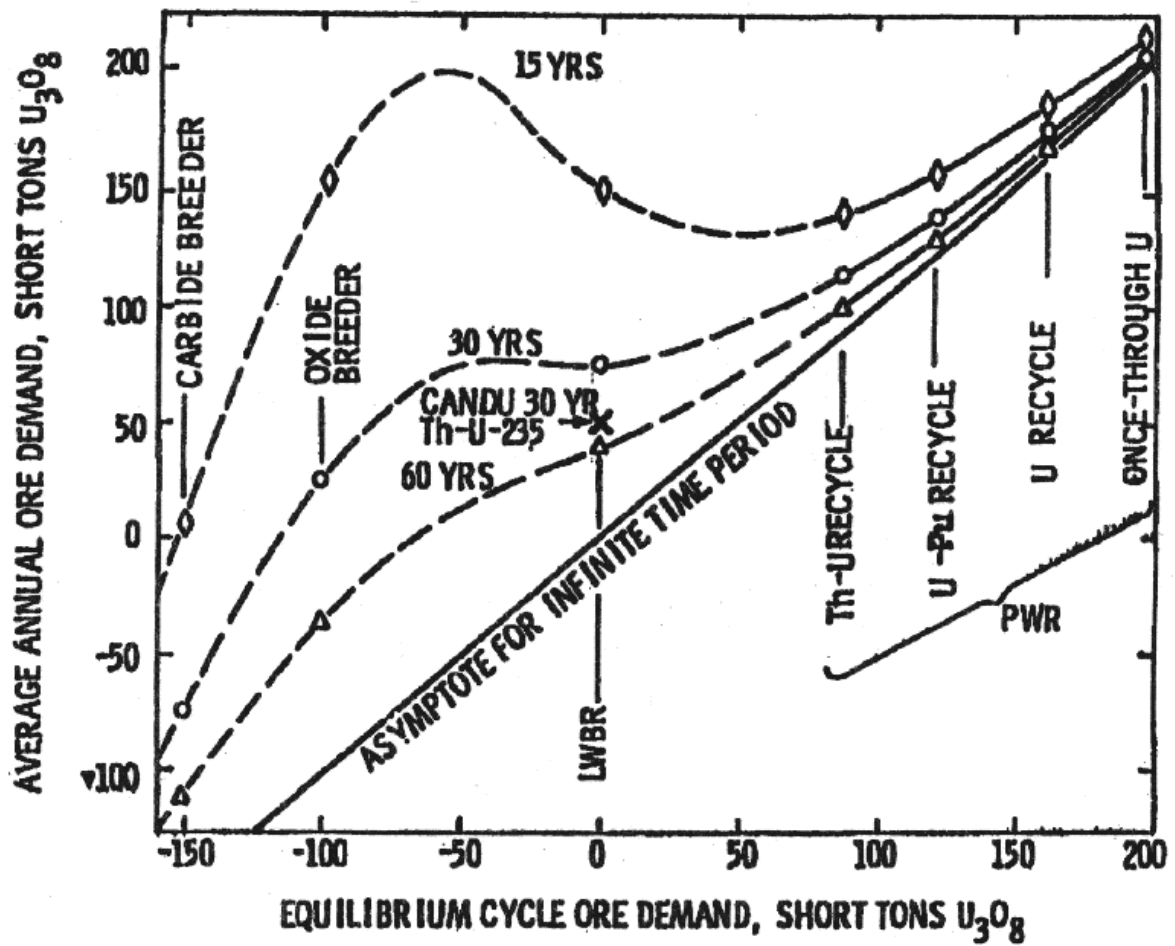


Figure 3. ORE Demand Components for 2 GWe Plant
 (75% Capacity Factor and 0.2% Tails) [Ref. 4]

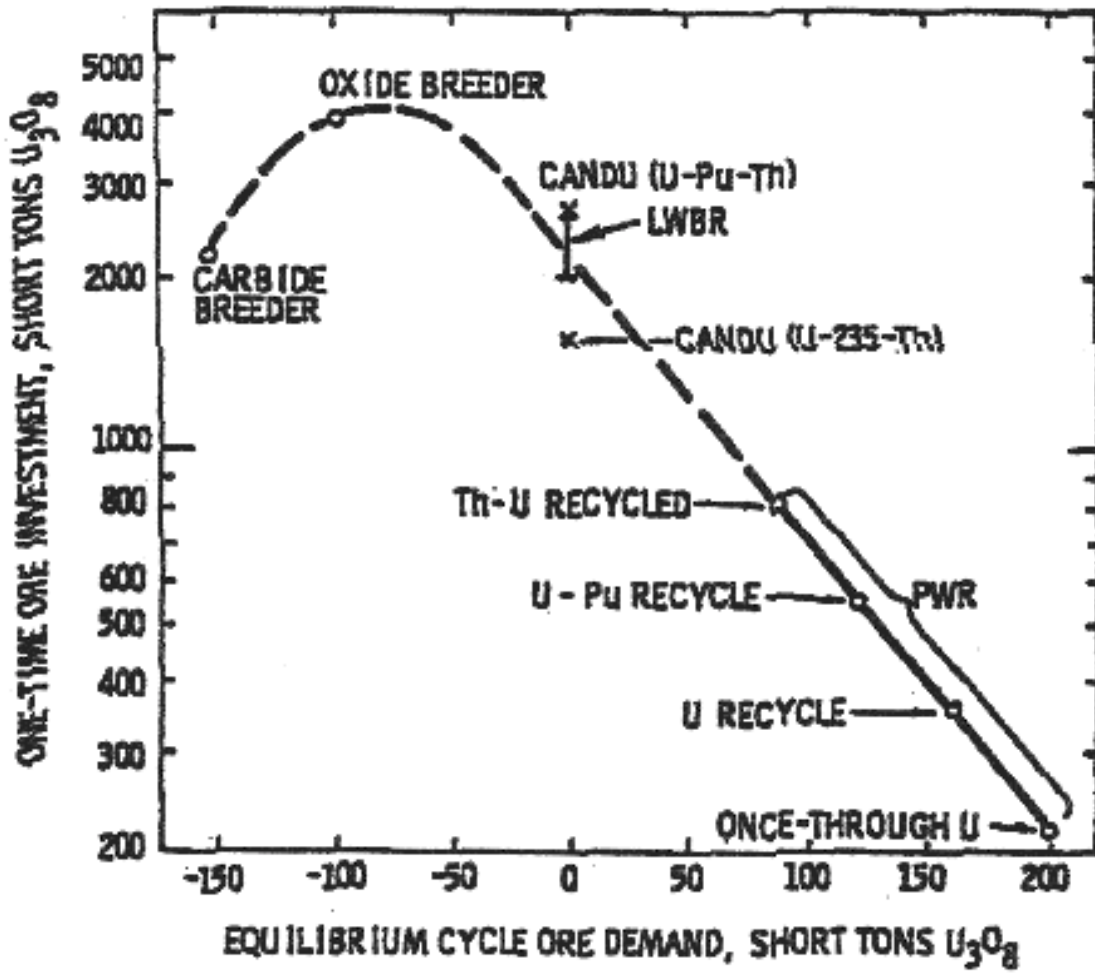


Figure 4. Effect of Leakage on Thorium Assessment

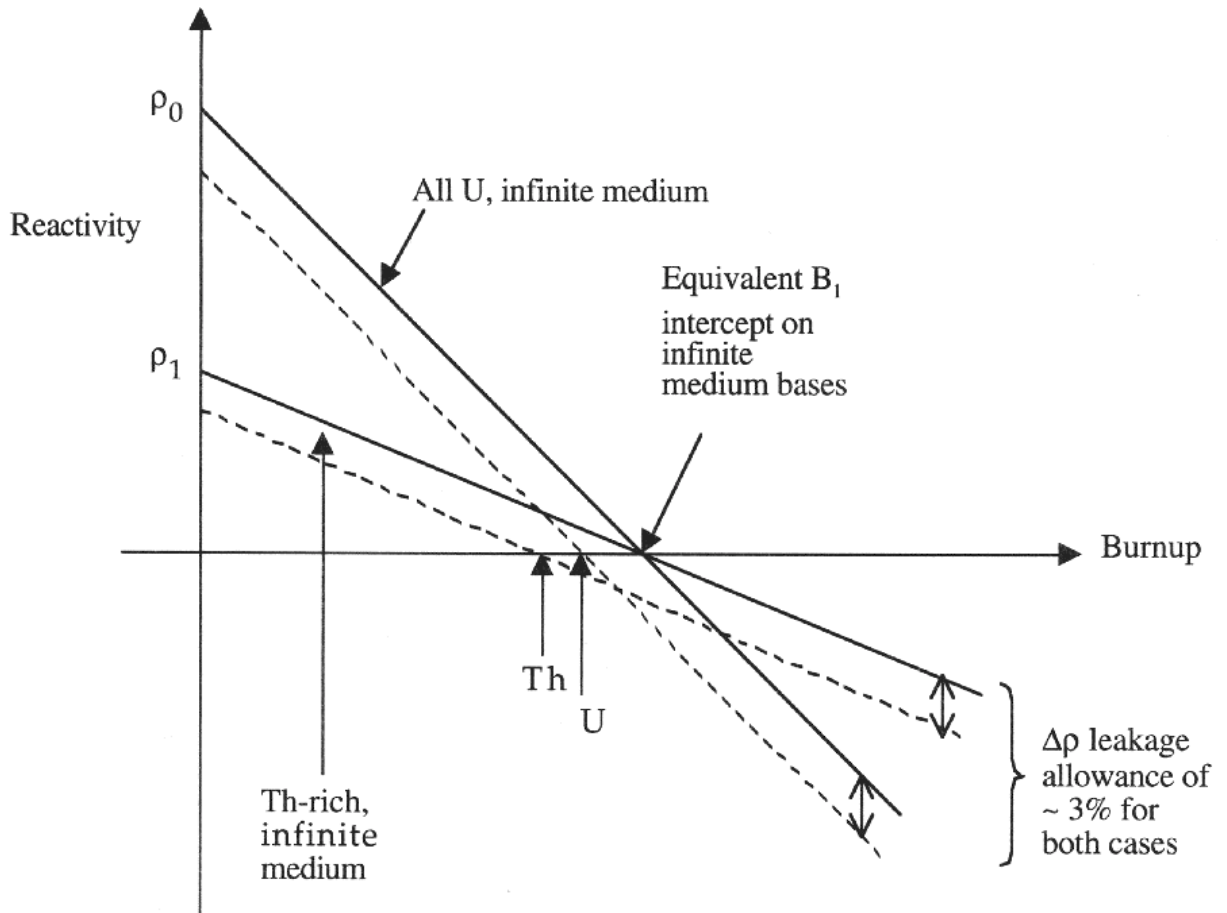
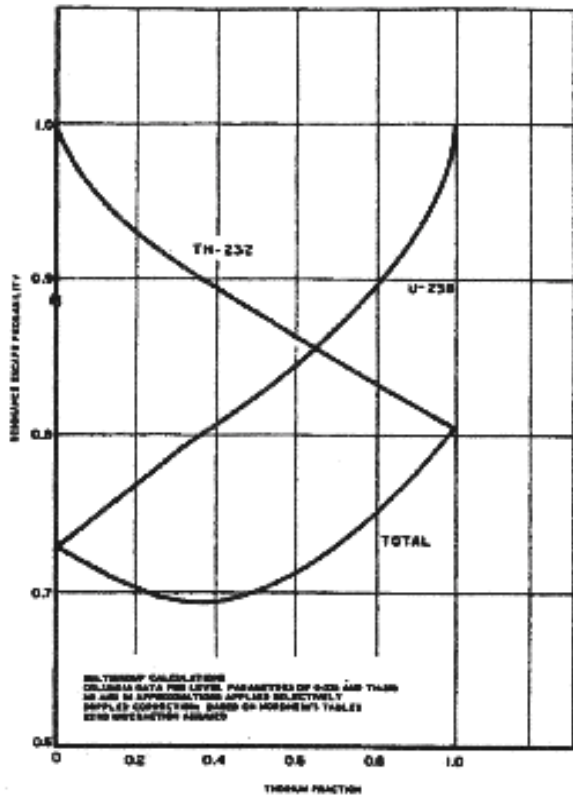
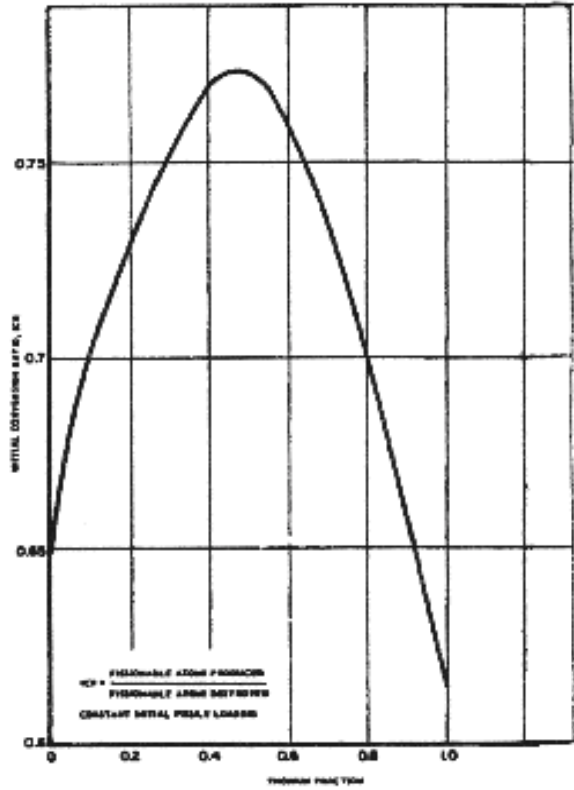


Figure 5. Neutronic Characteristics of U – Th Mixtures [Ref. 16]



Resonance Capture in Mixed U-238-Th-232



Effect of Thorium Fraction on Conversion

Table 1. Fissile Neutronic Properties

Parameter	U-233	U-235	Pu-239	Pu-241
Thermal ^(a) σ_a	364	405	1045	1121
(barns) σ_f	332	346	695	842
$\alpha = \sigma_c / \sigma_f$	0.096	0.171	0.504	0.331
η_{th}	2.26	2.08	1.91	2.23
Epithermal Resonance Integral (barns) (∞ Dilution)				
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
η_{epi}	2.10	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	0.0050

Table 2. Fertile Neutronic Properties

Parameter	Th-232	U-238	U-234	Pu-240
Thermal ^(a) σ_a	4.62	1.73	63	203
Epithermal (∞ Dilution) RI_a	85.6	278	660	8500
Shielded ^(b)	17	24		

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

(b) Oxide, 1/E spectrum, isolated rod @ 0.49 cm²/g

(c) Integrated over fission spectrum

All cross-section values in barns

Table 3. Characteristics of Thorium Experience in Large LWRs

	Elk River	Indian Point I	Shippingport (LWBR)
Reactor Type	BWR	PWR	PWR
Reactor Power	28MWe (58MWth)*	270MWe (585MWth)*	70MWe (237MWth)
Operation Dates	1962-1968	1962-1965	1977-1982
Fuel Assembly	Square 5 x 5	Square 14 x 14	Hexagonal
Assembly Radial Configuration	Homogeneous	Homogeneous	Seed region and Blanket Region
Fuel Composition	Mixed ThO ₂ -UO ₂	Mixed ThO ₂ -UO ₂ and ThO ₂	Mixed ThO ₂ -UO ₂ and ThO ₂
Core Axial Zoning	None	ThO ₂ Reflector	Extensive
Core Radial Zoning	two zones	three zones	four zones
U Enrichment	92% U-235	93% U-235	98% U-233
UO ₂ Concentration (mole %)	Mostly 4.36% Some 5.2%	0.0 up to 9.1	0.0 up to 5.0%
Cladding Material	Stainless 304	Stainless 304	Zircaloy 4
Thickness (mm)	0.51	0.51	0.56 Seed 0.71 Blanket 1.06 Reflector
Max/Ave Temp °C	318/299	293/-	
Fuel: Length(m)	1.52	1.9	2.66 7.8 Seed
Pellet Diameter(mm)	10.35	6.6	14.5 Blanket 21.1 Reflector
Burnup(MWD/kg) :			60 in Seed 30 in Blanket For pin average divide by 2
Max	8.5	32	
Ave.		14.8	
Peak Linear Power (kW/ft)			22 Seed 29 Blanket
Heat Flux: Peak (kW/m ²)	987	1,700	

*Elk River also had 14 MWth of coal superheat for a total thermal power of 72MWth. Indian Point had an oil fired superheater also. †From L.R. Weisser and G. Schileo, "Fabrication of Thorium Fuel Elements", an AEC monograph published by the American Nuclear Society, 1968; J. Belle and R.M. Berman, "Thorium Dioxide: Properties and Nuclear Applications", DOE/NE-0060.1984; and G.L. Olson. R.K. McCardell and D. Illum, "Fuel Summary Report: Shippingport Light Water Breeder Reactor", INEEL/EXT 98-00799, August 1998.

Table 4. Plutonium Production Rates and Associated Plutonium Compositions for the RTR and a PWR. (Weapon grade composition given for comparison.)*

	RTR (Th/U) Seed & Blanket	PWR (U)	Weapons Grade (for comparison)
Plutonium Isotope	Fraction of Isotope in Plutonium	Fraction of Isotope in Plutonium	Fraction of Isotope in Plutonium
Pu-232	0.0784	0.010	0.00012
Pu-239	0.4445	0.590	0.938
Pu-240	0.2067	0.210	0.058
Pu-241	0.1530	0.140	0.0035
Pu-242	0.1171	0.050	0.00022
Total Fraction	1.00	1.00	1.00
Total Plutonium Production Rate, kg/GWe-yr	48.4	-250	-
Sum of ²³⁹ Pu and ²⁴¹ Pu Fractions	0.598	0.730	0.942

*Ref. 8

Table 5. Probability of a "Nominal" and a "Fizzle" Yield versus Plutonium Grade*

Yield	Weapons Grade Pu	PWR Grade Pu	RTR-Seed Grade Pu	RTR-Blanket Grade Pu
"Nominal"	0.68	0.07	0.006	0.0002
"Fizzle"	0.06	0.36	0.55	0.74

*Ref. 8

Table 6. Fuel Compositions of RTR-Assembly and ABB/CE-Assembly [Ref. 7]

Initial Fuel Composition				
Nuclide	RTR-Seed	RTR-Blanket	RTR-Total	CE-Type
Th-232	--	24.48 ^a	24.48	--
U-235	3.34 [20]	0.42 [14]	3.76 [19]	5.39 [3.3]
U-238	13.33	2.59	15.92	158.08
Discharged Fuel Composition				
Nuclide	RTR-Seed	RTR-Blanket	RTR-Total	CE-Type
Th-232	--	21651.34 ^b	21651.34	--
U-232	--	3.39	3.39	--
U-233	--	387.74 [12.8]	387.74 [2.3]	--
U-235	664.03 [4.9] ^c	44.01 [1.4]	708.04 [4.3]	1331.52 [08]
U-238	12797.37	2066.07	14863.44	153008
Np-237	25.50	9.77	35.27	60.736
Pu-238	3.63 (5) ^c	8.32 (11)	11.95 (7)	20.79 (1)
Pu-239	7.20 (4)	28.72 (38)	35.92 (39)	881.84 (56)
Pu-240	87.25 (27)	12.20 (16)	99.45 (24)	388.36 (24)
Pu-241	53.52 (15)	11.08 (15)	64.60 (15)	216.66 (14)
Pu-242	30.69 (13)	15.57 (20)	46.26 (15)	80.01 (5)
Am-241	0.44	0.44	0.88	5.31
Am-243	2.72	7.74	10.46	15.12
Cm-242	0.28	0.23	0.51	1.97
Cm-244	0.62	9.01	9.63	4.62
Total Mass ^d	211.85	103.08	314.93	1675.42

^aAverage mass (kg/yr)

^bAverage mass (g/yr)

^cWeight % of U [] and Pu in ()

^dTotal mass of transuranic nuclides

Table 7. Plutonium Production in U and Mixed Th-U Cycles*

	U 4.5 yr, 45 MWD/kg	U 8% enr, 6 yr, 72 MWD/kg	Th-U 6 yr, 72 MWD/kg	Th-U 10 yr, 100 MWD/kg
Burnup (MWD/kg)	45	72	72	100
Production g/kg ihm				
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 per 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

*Ref. 11

Table 8. Spontaneous Neutron and Decay Heat*

Comparison of Spontaneous Neutron and Heat Production						
	Super Grade	Weapons Grade	U 4.5 yr, 45 MWD/kg	U 8% enr, 6 yr, 72 MWD/kg	Th-U 6 yr, 72 MWD/kg	Th-U 10 yr, 100 MWD/kg
Composition						
Pu-238	0.0%	0.012%	2.3%	4.4%	10.8%	13.9%
Pu-239	98.0%	93.80%	55.8%	54.6%	38.9%	36.4%
Pu-240	2.0%	5.80%	21.2%	19.6%	19.8%	19.4%
Pu-241	0.0%	0.35%	14.9%	15.4%	14.9%	14.0%
Pu-242	0.0%	0.022%	5.8%	5.9%	15.6%	16.3%
Spontaneous Neutron Production						
(n/kg-s)	1.82E+04	5.35E+04	3.93E+05	3.93E+05	7.26E+05	8.16E+05
relative	1.0	2.9	19.3	21.6	39.9	44.8
Decay Heat						
(W/kg)	2.0	2.3	16.1	27.8	63.4	80.7
relative	1.0	1.1	8.1	13.9	31.7	40.4

*Ref. 11

Table 9. Overview of the Ex-Core Thorium Fuel Cycle Relative to Its Uranium Counterpart

ASPECT	DIFFERENCE VS. URANIUM
Front End	
Mining	<ul style="list-style-type: none"> • Thorium is perhaps 3 times more abundant but much less is mined. Best resources are monazite sands in India and Brazil. • Because uranium is really mined for its U-235, a once-through cycle needs only about 1/10th as much Thorium. • U-free Th preferred because of absence of Th-230. • Tailings less of a problem because Rn-220 has a much shorter half-life than Ru-222.
Enrichment	<ul style="list-style-type: none"> • Must provide as U-235 or Plutonium (could be from dismantling weapons). • Recycled U-233 contains <u>U-232</u>, U-234.
Fabrication	<ul style="list-style-type: none"> • Typically as ThO₂ using processes similar to UO₂ and PuO₂, (which are incorporated to provide fissile enrichment).
Back End	
Storage and Transportation	<ul style="list-style-type: none"> • Pa-233 decay (T_{1/2}-27 days) creates more U-233 over first several months. • Similar fission product decay heat and gamma emission.
Direct Disposal	<ul style="list-style-type: none"> • ThO₂ is stable in oxidizing environment (unlike UO₂ which forms U₃O₈). • Factor ~ 10 lower concentration of radiotoxic higher actinides.
Reprocessing	<ul style="list-style-type: none"> • Solvent extraction (THOREX) similar to PUREX but same equipment has about half the processing rate, hence ~ 30% more expensive.
Refabrication	<ul style="list-style-type: none"> • Need to shield against hard gammas from U-232 decay chain (Bi-212, Tl-208), hence even more expensive than recycled U or Pu. • Preferable to delay recycle of Th for ~ 15 years to decay Th-288; one year suffices for Th-234.
Safeguards	<ul style="list-style-type: none"> • Must denature to ≤ 12% U-233 in U-238. • U-232 chain gammas complicate handling.