Th/U-233 Multi-recycle in PWRs

Fuel Cycle Research & Development

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EXECUTIVE SUMMARY

The use of thorium in current or advanced light water reactors (LWRs) has been of interest in recent years. These interests have been associated with the need to increase nuclear fuel resources and the perceived non-proliferation advantages of the utilization of thorium in the fuel cycle. Various options have been considered for the use of thorium in the LWR fuel cycle including: (1) its use in a once-through fuel cycle to replace non-fissile uranium or to extend fuel burnup due to its attractive fertile material conversion, (2) its use for fissile plutonium burning in limited recycle cores, and (3) its advantage in limiting the transuranic elements to be disposed off in a repository (if only Th/U-233 fuel is used). The possibility for thorium utilization in multi-recycle system has also been considered by various researchers, primarily because of the potential for near breeders with Th/U-233 in the thermal energy range.

The objective of this project is to evaluate the potential of the Th/U-233 fuel multi-recycle in current LWRs, with focus this year on pressurized water reactors (PWRs). In this work, approaches for ensuring a sustainable multi-recycle without the need for external source of makeup fissile material have been investigated. The intent is to achieve a design that allows existing PWRs to be used with minimal modifications. In all cases including homogeneous and heterogeneous assembly designs, the assembly pitch is kept consistent with that of current PWRs (21.5 cm used). Due to difficulties encountered with keeping the PWR design intact, the potential modifications (other than assembly pitch) that would be needed for PWRs to ensure a sustainable multi-recycle system have been investigated and characterized. Additionally, the implications of the use of thorium on the LWR fuel cycle are discussed.

Investigations have been conducted to assess the feasibility of both homogeneous and heterogeneous PWR assemblies to achieve Th/U-233 fuel multi-recycle. The conclusions from the study are:

1. A practical sustainable fuel cycle required by this study cannot be achieved with a homogeneous PWR assembly within the parameter space of initial U-233 content and reasonable moderator to fuel volume ratio (MR) obtained by varying the fuel pin size.
2. A 17-by-17 heterogeneous assembly design achieved the following attributes and partially met the requirements defined in this study for sustainable Th/U-233 multi-recycle:
    a. A derated core power of 1000 MWt.
    b. A 2.7-year cycle length with sustainable fissile inventory (assuming 3-batch fuel management).
    c. Discharge burnup of 18 GWd/t, which is significantly lower than the burnup of current PWRs.
    d. The Th/U-233 system requires hard neutron spectra to support high conversion of Th-232. The neutron spectra of both the seed and the blanket region are harder compared to the standard thermal systems.
    e. Compared to the used nuclear fuel (UNF) of the standard PWR uranium-dioxide (UOX) fuel system, the high level waste (HLW) of the Th/U-233 multi-recycle system has higher normalized radiotoxicity and decay heat at discharge (due to lower burnup), but those become lower 20 years and 50 years after discharge, respectively, due to the recycle of 99.9% of Th and U isotopes in the nuclear system and the low production of transuranic elements.
    f. Handling of the UNF of the Th/U-233 system should generate less concern than for the UOX system because of the lower neutron source and photon source levels. The lower neutron source is due to the low production of TRU isotopes.

g. Cycle by cycle assembly model has to be run in the future to ascertain if the same cycle length can be maintained in the succeeding recycles with the discharge fissile inventory derived from previous cycles.

3. A sensitivity study has been conducted, which suggests that:

   a. A full-core model is necessary to estimate the neutron leakage and evaluate the performance of core level heterogeneity.

   b. A detailed thermal-hydraulics study should be carried out in future studies on the 17-by-17 heterogeneous assembly design for Th/U-233 multirecycle.

   c. Considerations should be given to adopting hydride blanket fuel forms in future studies.

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ACRONYMS

ANL	Argonne National Laboratory
BNL	Brookhaven National Laboratory
BOC	Beginning of Cycle
BOL	Beginning of Life
BWR	Boiling Water Reactor
EOC	End of Cycle
EOL	End of Life
FCRD	Fuel Cycle Research and Development
FIR	Fissile Inventory Ratio
FP	Fission Products
GWd/t	Giga-Watt-day per metric ton (a unit of fuel burnup)
HLW	High Level Waste
HM	Heavy metal
HTR	High Temperature Reactor
IAEA	International Atomic Energy Agency
INL	Idaho National Laboratory
LWR	Light-Water-cooled Reactor
LWBR	Light-water Breeder Reactor
MA	Minor Actinides
MOX	Mixed Oxide
MR	moderator to fuel volume ratio
MSR	Molten Salt Reactor
MWt or MWe	Mega-Watt thermal or electric (unit of power)
OECD/NEA	Organization for Economic Co-operation and Development, Nuclear Energy Agency
ORNL	Oak Ridge National Laboratory
P&T	Partitioning and transmutation
PIE	Post Irradiation Examination
PWR	Pressurized Water Reactor
TRU	Transuranic Elements, e.g., Pu, Np, Am, Cm
UNF	Used nuclear fuel
UOX	Uranium-dioxide
USDOE	United States Department of Energy

SYSTEMS ANALYSIS CAMPAIGN Th/U-233 MULTIRECYCLE IN PRESSURIZED WATER REACTORS

1. INTRODUCTION

The use of thorium in current or advanced light water reactors (LWRs) has been of interest in recent years. These interests have been associated with the need to increase nuclear fuel resources and the perceived non-proliferation advantages of the utilization of thorium in the fuel cycle. Various options have been considered for the use of thorium in the LWR fuel cycle including: (1) its use in a once-through fuel cycle to replace non-fissile uranium or to extend fuel burnup due to its attractive fertile material conversion, (2) its use for fissile plutonium burning in limited recycle cores, and (3) its advantage in limiting the transuranic elements to be disposed off in a repository (if only Th/U-233 fuel is used). The possibility for thorium utilization in multirecycle system has also been considered by various researchers, primarily because of the potential for near breeders with Th/U-233 in the thermal energy range.

The objective of this project is to evaluate the potential of the Th/U-233 fuel multirecycle in current LWRs, with focus this year on pressurized water reactors (PWRs). In this work, approaches for ensuring a sustainable multirecycle without the need for external source of makeup fissile material have been investigated. The intent is to achieve a design that allows existing PWRs to be used with minimal modifications. In all cases including homogeneous and heterogeneous assembly designs, the assembly pitch is kept consistent with that of the current PWRs (21.5 cm used). Because of design difficulties associated with using the same geometry and dimensions as a PWR core, the potential modifications (other than assembly pitch) that would be needed for PWRs to ensure a sustainable multirecycle system have been investigated and characterized. Additionally, the implications of the use of thorium on the LWR fuel cycle are discussed.

In Section 2, background information on studies evaluating the use of thorium in the fuel cycle is provided, but focusing on Th/U-233 multirecycle. Recent studies done internationally and in the U.S. are briefly summarized. Additionally, the previous U.S. thorium breeder experiment in the Shippingport reactor is briefly discussed.

The objective of this work and the reactor design issues associated with multirecycle of Th/U-233 are discussed in Section 3. The approaches required to achieve a sustainable system are discussed and evaluated. Homogeneous assembly modeling results are presented in this section.

In Section 4, a 17-by-17 heterogeneous assembly design has been selected and evaluated, based on its positive attributes for sustainable Th/U-233 multirecycle. A feasibility study is briefly discussed at the end of this section followed by recommendations for future activities.

Section 5 discusses the attributes of the 17-by-17 heterogeneous assembly design. The material mass flow data and fuel cycle impact data are reported in this section.

Discussions on the fuel cycle implications of thorium fuel utilization are provided in Section 6. This includes information on fuel sources, fuel manufacturing, fuel reprocessing, and re-fabrication.

The conclusions of the study are provided in Section 7.

2. BACKGROUND

This study is focused on Th/U-233 multirecycle in LWRs and as such the background information provided here is for that approach. It should be noted that thorium has been considered or demonstrated for various fuel cycles in the past and a good source of background information on such systems is an IAEA document produced in 2005 [IAEA 2005].

2.1 Studies on Thorium Utilization in Limited Recycle LWR System

2.1.1 Comparison of Th/Pu and MOX Fuels for PWRs

A recent study considered the use of Thorium-plutonium (Th/Pu) oxide fuels as an evolutionary way to simultaneously reduce plutonium volumes and capture energy from the material [Björk 2009]. While that particular work is not directly related to the current project, it is included here to relate thorium as fertile material to U-238 in order to indicate expected differences in performance. In that work the neutronics properties of Th/Pu fuel and U/Pu mixed oxide (MOX) fuels with varying Pu isotope vectors were compared. For the supporting studies, burnup simulations using a lattice code were performed for a regular MOX PWR fuel assembly and for a thorium-plutonium PWR fuel assembly of the same geometry. The plutonium content of the two fuel types is chosen so that the same total energy release per fuel assembly is achieved. It was found that this requirement necessitated higher plutonium content in the Th/Pu case, because of the higher Th capture cross section than U-238. For the two systems, reactivity coefficients and kinetics parameters were also calculated. The study indicated that the MOX and Th/Pu fuels have fairly similar neutronics properties in existing PWRs. The Th/Pu fuel was additionally found to offer an advantage over MOX fuel with regards to coolant void reactivity and plutonium consumption. It was concluded that introducing Th/Pu-fuel would improve these factors without imposing any major hurdles from a reactor physics point of view.

2.2 Approaches for Achieving High Conversion in Water Cooled Systems

2.2.1 Feasible Ways to Achieve High Conversion in Thorium-Fueled CANDU and PWR Reactors

A recent study evaluated approaches to achieving high conversion in Th fueled CANDU reactors and PWRs [Guillemin 2009]. The study focused on Th-fueled CANDU reactors and PWRs as the third and last component of a reactor fleet. Plutonium extracted from spent fuel of a standard CANDU or PWR is used to support the conversion of thorium in the second component (fast reactor (FR) or CANDU) to uranium (mainly U-233) which feeds a Th/U multirecycling third component (CANDU and LWR). Breeding and multirecycling of U-233 in the third component CANDU reactor is optimized by modifying the core. The study indicated that slight moderator to fuel volume ratio (MR) variations have no significant impact on conversion and that heterogeneous bundles with fertile and fissile zones allow savings of about 7% of U-233 initial inventory for the same burnup. However, for all the examined geometric modifications in Th/U fueled CANDU, the fissile inventory ratio (FIR) at end of cycle is observed to be only dependent on the final burnup. The FIR is defined as the ratio of fissile mass at the end of cycle (EOC) to the fissile mass at the beginning of cycle (BOC). Breeding was achieved for short burnups of 6-7 GWd/t. In the multirecycling regime in CANDU reactors, higher burnup (~15-20 GWd/t at slightly lower conversion) can be sustained by the addition of a small quantity of uranium from the second component of the reactor fleet at each recycle.

For the third component PWR case, studies indicated that variations in either the moderator to fuel volume ratio or the use of heterogeneous seed/blanket assemblies do little to improve conversion. It was found that at equivalent burnup the Th/U conversion obtained by standard modifications is smaller in the PWR than in the CANDU reactor. A harder neutron spectrum such as a fast spectrum, achievable by extreme under-moderation, was found necessary to achieve breeding in existing PWR vessel and core.

Clearly the lower values of moderator ratio would result in difficulties associated with heat extraction and thermal-hydraulics. To overcome these difficulties, the authors studied a so-called spectrum shift control technique with a standard moderator to fuel volume ratio using heavy water coolant and managing reactivity by progressive addition of H_2O in the coolant. Another reactivity management technique based on fertile material withdrawal using mechanical means was also considered. With both the spectrum shift control and the fertile material withdrawal techniques, no breeding was achieved in the system (although higher conversion ratios were obtained).

2.2.2 Investigation of Feasibility of Thorium Breeder Reactor in BWRs

The breeding of U-233 for power generation in a boiling water reactor (BWR) has been investigated for sustainable development [Funahashi 2010]. In the study of Funahashi et al., the moderator to fuel number density ratio was varied for different discharge burnups to investigate its effect on U-233 breeding in the BWR system. This is implemented by changing the void fraction and the fuel pellet diameter. The effect of U-233 content was also evaluated. The study found that the conversion ratio is always less than 1.0 with the typical BWR 8x8 fuel assembly and the reasonable design variations that were done. However, tightening the fuel lattice (reducing the moderator area) and broadening the fuel pellet was found as the effective way to increase the conversion ratio. This would entail a modification of the assemblies, requiring more detailed system design.

2.2.3 Investigation of Fuel Assembly Design Options for High Conversion Thorium Fuel Cycle in PWRs

The possibility of achieving a self-sustainable Th/U-233 fuel cycle that can be adopted in the current generation of Pressurized Water Reactors (PWRs) has been explored by Shwageraus and others [Shwageraus 2009]. The study made the assumption that the use of movable fuel for reactivity control, as employed in the Shippingport LWR breeder program, is not an option nowadays. Based on this assumption, it then outlined some fuel design strategies to achieve (or to approach as closely as possible) a sustainable fuel cycle. It was noted that a high conversion ratio can be achieved through the use of advanced materials, heterogeneous reactor core configuration, optimization of fuel composition and other innovative features. However, none of these modifications should compromise the reactor safety under normal or potential accident conditions. Major design tradeoffs in the core design were discussed. Some of the strategies discussed for optimization of fuel assembly design include:

- Transition from homogeneous to heterogeneous “seed-blanket” fuel assembly. In order to improve the conversion ratio, it is necessary to increase capture reaction rates in the fertile material (Th). This goal can be achieved by reducing the competition for neutron absorption between U-233 and Th through utilization of heterogeneous “seed and blanket” (SB) fuel assembly geometry.
- Optimization of the seed and blanket region dimensions. Near optimal dimensions of the seed and blanket regions are required to be on the order of one neutron migration length for each region respectively.
- Increasing dimensions of blanket fuel pin. A larger fuel pin reduces the hydrogen to heavy metal (H/HM) ratio, hardens the blanket neutron spectrum and, thus, increases resonance absorption in thorium.
- Introduction of heterogeneous duplex seed pin. Although the blanket region is considered as the breeder zone in the heterogeneous assembly, it is also possible to take advantage of the Th in the seed region by creating heterogeneous duplex seed. The duplex seed consists of the inner zone that contains most of the fissile material and the outer seed ring that contains only fertile material (Th).

Results presented in that study suggest that seed-blanket designs are capable of achieving net breeding at least for a portion of the fuel cycle. However, only SB with duplex seed seems to be practical in the context of current industry experience of having at least 1000 days fuel residence time (which roughly

corresponds to an annual fuel cycle). In order to show basic feasibility of self sustainable fuel cycle with SB fuel, the following considerations have to be taken into account:

- Concentration of fissile material in the seed region will cause large power imbalance between seed and blanket regions. This would almost inevitably require at least some reduction in the core power density (Shippingport light water breeder reactor had core average power density considerably lower than that of the current PWRs). Detailed thermal-hydraulic analysis will be required to determine the acceptable power density level.
- Reactor operation at lower power would increase the fuel cycle length, which can be designed to match the peak of the conversion ratio. However, operating a reactor at lower power would reduce the economic competitiveness of the concept.
- Flat reactivity letdown curve with close to zero reactivity swing will require elaborate optimization of the core loading pattern.

It was noted in the study that not much consideration was given to thermal design and safety related issues. A major concern is the concentration of fissile material, primarily in the seed region, which causes a large power peak, challenging feasibility of the concept from a thermal-hydraulic point of view.

2.3 Recent Th Multirecycle Evaluation Under FCRD Program

In FY 2009, BNL [Raitses 2009] performed a study on thorium based fuel cycle options in PWRs. The study noted that most of the recent U.S. work on the utilization of thorium in a PWR has been limited to once-through cycle applications consistent with current U.S. practice. The BNL study was an initial examination of options for the utilization of thorium-based fuels if reprocessing of used nuclear fuel (UNF) is considered to make U-233 and/or LWR transuranic elements (TRU) available to serve as the source of initial fissile material required in place of LEU. Studies were performed to assess the potential performance of these options in a reference current PWR (1000 MWe Westinghouse design with 17x17 fuel assemblies) under the constraint that minimal or no modifications would be required to the assembly and reactor designs to facilitate implementation. These studies addressed several “recycle” scenarios, but all were limited to homogeneous fuel loading. The three cases were:

- Scenario-1: “Once-through equilibrium” Th/U-233 oxide assuming availability of an infinite source of pure U-233.
- Scenario-2: “Once-through equilibrium” Th/TRU oxide assuming an infinite source of TRU (Np, Pu, Am) based on used nuclear fuel (UNF) produced by an advanced light water reactor (ALWR) with a discharge burnup of 50 GWd/T, and cooled for 5-years prior to separating the TRU from the remaining UNF, followed by a 2-yr period for fabrication and transport prior to insertion into the reactor.
- Scenario-3: “Self-Re-cycle” Th/U/TRU oxide starting with the TRU-Th as in scenario 2 and then recycling the discharged U and TRU (Np, Pu, Am), with additional makeup as needed from ALWR discharge. [Raitses 2009]

The study evaluated initial heavy metal (HM) loading, mass flow (amount of HM charged/discharged each cycle), isotopic composition of discharged fuel, reactivity coefficients for chosen stage, and radioactivity, and heating rate of discharged fuels.

The initial scoping studies suggest that sufficient U-233 cannot be generated in a conventional PWR using homogeneous fuel to make it self-sustaining on a pure Th/U-233, and enriched LEU or plutonium from legacy UNF would be required to support fuel multirecycle. The BNL report also indicated that the use of U-232 presents significant proliferation risks, unless the U-233 is “denatured” by adding sufficient uranium to satisfy the “non-proliferation limit”. It was indicated that one barrier to misuse is the gamma

source from U-232 and its daughters. This barrier can however be overcome by remote handling and shielding.

2.4 LWR Breeder Demonstration at Shippingport Plant

The Shippingport Atomic Power Station was a test-bed for the design, construction, and operation of future commercial nuclear power plants and provided the foundation for today's PWR industry. Shippingport was a facility in which several different designs of light water cooled and moderated reactor cores were successfully operated [Clayton 1993]. A LWR Th/U-233 breeder core was installed in the Shippingport reactor to demonstrate the feasibility of breeding in LWRs, at a time when the U.S. Government was investigating approaches to expand the nation's energy resources through advanced technologies. The light water breeder reactor (LWBR) became critical in Shippingport in 1977 and operated till 1982 [Clayton 1993].

The LWBR core contained 12 hexagonal-shaped modules arranged in a symmetric array surrounded by 15 reflector modules. Each of the hexagonal fuel modules was composed of a central movable seed assembly surrounded by a stationary blanket assembly. The movable seed assemblies were used instead of control rods or soluble poisons to control reactivity. This minimized parasitic neutron capture and made breeding possible in the core. [Clayton 1993] A boron poison system was used for backup shutdown. Fuel was in the form of oxide ceramic pellets sealed within Zircaloy-4 tubes, similar to the fuel used today in LWRs. The fuel pellets in the seed and blanket assemblies were composed of the mixed oxides of uranium-233 and natural thorium (primarily Th-232) [Clayton 1993]**.** Depending on the core region, different mixtures of Th/U-233 fuels were used. In the seed region, 6% U-233 fuel was used, while in the blanket 3% U-233 fuel was used. The reflector region, surrounding the power-producing seed-blanket modules, contained fertile thoria fuel that absorbed escaping neutrons. This zone provided additional breeding of fissile U-233 [Nero 1979, Clayton 1993]. The core also employed axial blanket zones (ThO_2 fuel) at the tops and bottoms of the seed and blanket fuel rods. Figure 2.1 shows the radial layout of the Shippingport LWR breeder. The reactor axial section is presented in Figure 2.2, and shows the mechanism developed for fuel movement.

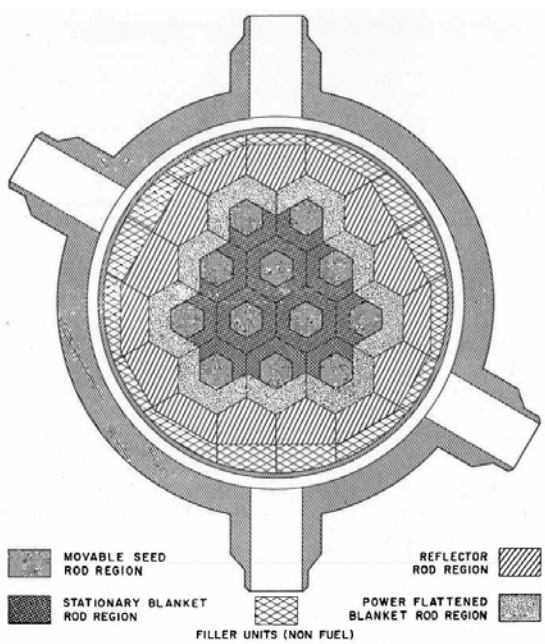


Figure 2.1. Radial Layout of Shippingport LWR Breeder Core [Nero 1979].

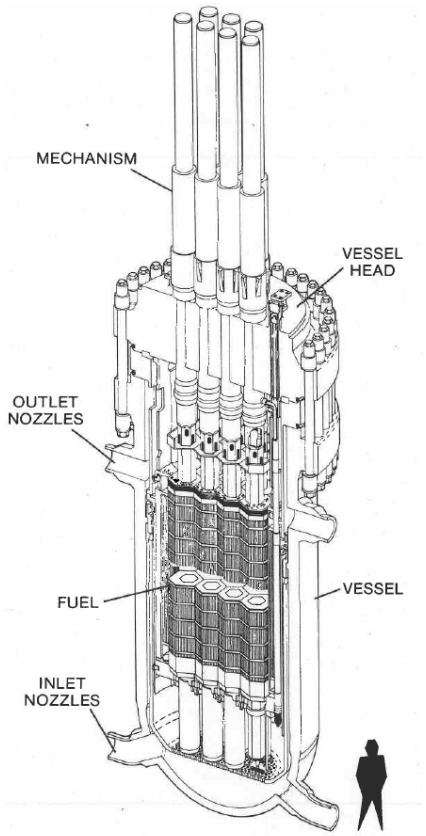


Figure 2.2. Axial Section of Shippingport LWR Breeder Core [Nero 1979].

The breeding ratio during the Shippingport LWR breeder demonstration was just slightly higher than 1.0. In any case, the successful operation of the Shippingport LWBR demonstrated the feasibility of using the thorium-uranium fuel cycle in a light-water environment. It shows that Th/U-233 cores can be designed and built, operated in existing LWR plants to produce electricity, and can breed enough fissile fuel to overcome modest losses in reprocessing and re-fabrication [Clayton 1993]. For the United States in particular, this means that the domestic supply of thorium can become an important energy source. Clayton also indicated that this resource can provide about 50 times as much energy as the domestic supply of uranium used in current LWRs. He noted that the light-water breeder thus has an energy potential that could meet the entire electrical needs of the United States for centuries. [Clayton 1993].

In Table 2.1, important data on the operation of the U.S. Shippingport LWBR reactor are reported. The data indicate that the operation of the Shippingport reactor does not meet the requirement of this study mainly due to the much lower discharge burnup compared to that of the current PWRs. However, the successful operation of the Shippingport LWBR reactor demonstrated the feasibility of Th/U-233 sustainable recycle in LWRs, and the data have provided important insights to this study.

It is important to note the much lower specific power density of the LWBR (5.6 W/g) compared to the current PWRs (~34 W/g). As a result, the discharge burnup of LWBR (6.8 GWd/t), even with a long operation length of ~5 years, is significantly lower than that of the current PWRs (~50 GWd/t). There are two major reasons for the difference between the operation capacity of the Shippingport LWBR and current PWRs: (1) smaller core dimension of LWBR (~1 m active diameter core versus ~3.4 m active diameter core (current PWRs)) limiting the core power level and (2) heavy core loading (core mass over core volume ratio is ~5 times as high as that of the current PWRs) results in lower core power density and therefore low discharge burnup.

The Shippingport LWBR data also suggest that the objective of this study is difficult to achieve. With core level heterogeneity and movable seed to control parasitic neutron loss, the Shippingport LWBR achieved a discharge burnup of 6.8 GWd/t (designed to be about 3.5 GWd/t). This indicates that it might be difficult to achieve high discharge burnup with a sustainable Th/U-233 system. Unfortunately, this conjecture could not be ascertained because the Shippingport LWBR did not stop operating due to reactivity limitations, but due to government termination of the program.

The unique aspect of the U.S. Shippingport LWBR is the core level heterogeneity, which helped reduce parasitic neutron loss and resulted in a small leakage of about 0.8%. This small leakage was made possible by the use of axial and radial ThO_2 blanket regions. Similar idea can well be implemented into the scope of the current study. However, due to time constraints, a full-core design and model has not been developed and would be part of future activities. The importance of a full-core model will also be discussed in the section on heterogeneous assembly designs (Section 4).

Table 2.1. Design and Operation Parameters for U.S. Shippingport LWBR

Parameter	Value
Core power, MWt [a)]	236.6
Discharge Burnup (GWd/t)	6.8
Capacity factor, %	68%
Number of batches	1
Designed cycle length (Effective Full Power Hours, EFPH)	>15,000
Designed FIR at EOL	1.012
Designed total fissile loading at BOL (Kg) [a)]	501.04
Designed total Th-232 loading at BOL (Kg) [a)]	42056.2
Designed total HM loading at BOL (Kg)	42557.24
Specific power density, W/g	5.6
Operation cycle length (EFPH) [b)]	29047
Operation FIR at EOL [b)]	1.0139

a) Values given in Reference [WAPD 1979]
b) Values given in Reference [Clayton 1993]

* BOL and EOL are beginning and end of life, respectively.

3. OBJECTIVE AND PRELIMINARY RESULTS

3.1 Objective

The objective of the current work is to investigate the feasibility of achieving self-sustainable Th/U233 fuel multirecycle in current generation of PWRs. This translates to a core design with a breeding ratio of about 1 (or higher) not requiring an external source of fissile material. The study will investigate fuel assembly designs and conditions under which a self-sustainable fuel cycle will be achieved. Both heterogeneous and homogeneous assembly designs will be investigated, though it is quite unlikely that the later can be used for the purpose. Major design tradeoffs in the core design will be considered. The effort this year will be devoted to core physics studies, but core safety issues will be investigated through comparison of safety parameters to those of typical PWRs. For a workable core design, core mass flow will be evaluated for use in comparing the fuel cycle metrics to those of other systems.

3.2 Attributes of a Workable System

The previous studies that have been done on a sustainable Th/U-233 multirecycle system suggest the following design characteristics:

- Fuel cycle is based on Th/U-233 fuel instead of low enriched uranium fuel. The original source of U-233 is assumed to exist. Subsequent cores have to derive their U-233 from preceding cores.
- A much harder spectrum is required, implying tighter pitch or reduced moderation (different reactor design). An under-moderated assembly with a much higher fuel to water ratio than used in current LWRs might be required. This reduces parasitic neutron capture in water, but might also result in thermal-hydraulic problems.
- Spectral shifting system might be necessary, thus requiring PWRs controlled by moveable fuel or fertile rods. However, this is undesirable as it deviates from the current PWR operational approaches; there might also be some regulatory hurdles to be overcome. The study will evaluate the design space typical of PWRs.
- It is quite likely that only a heterogeneous seed and blanket assembly design can be used to achieve a workable design that meets the project objective. A heterogeneous assembly or even heterogeneous core is problematic from a power distribution viewpoint. It appears that both the seed and the blanket would need to have fissile material.
- Approaches for reducing neutron absorption in non-fuel materials in order to achieve high conversion must be considered.
- The use of an under-moderated system might however result in a need to decrease the system power and power density, with the potential reduction in the fuel burnup. All of which would have economic penalties.
- The use of the spectral shift approach involving the substitution of light water with heavy water could help, but this will not be considered in the current study as it implies a significant perturbation of the current LWR system. Additionally, a heavy water plant would be required in the fuel cycle, with associated cost penalty.
- Increasing power density or flux would help as a means of increasing neutrons available for breeding. However, this might not be possible if the power density needs to be decreased to meet core cooling requirements.
- Enhancing neutron absorption in thorium to increase conversion. This necessitates the use of blanket zones.

Additionally, a workable design must consider the implications of the following items on licensing and safety:

- Use of a tight lattice unlike current LWRs.
- Core thermal margins.
- Provision of accident prevention.
- Acceptability of movable fuel as means of reactivity control.
- Feedback coefficients.
- Control stability and adequacy.
- Nuclear stability.

3.3 Project Plan

A systematic evaluation of a sustainable Th/U-233 multi-recycle in PWRs is planned. This work will include:

- Various sensitivity studies of PWR homogeneous and heterogeneous assemblies configurations to scope the feasibility range for high Th conversion to U-233.
- A "point" design will then be developed. This point design would be the basis of information for deriving other fuel cycle data. The viable design must have a conversion ratio greater than 1.0 and workable reactivity and safety parameters, with the void reactivity coefficient being negative. A cycle length greater than at least one year, and 1.5 years is desired. Reactivity control should not involve movable fuel, although this option might be evaluated if no other ones are plausible. LWR design safety margins should also be observed in the final design (a detailed LWR design is not planned as part of this year's effort).
- Specific results from this work will include a sustainable Th/U-233 assembly design, system cycle length and discharge burnup, nuclide mass data for charged and discharged fuel and post irradiation cooling mass data, decay heat and radiotoxicity, with quantities normalized to the energy generation.

3.4 Sensitivity Studies for PWR with Homogeneous Assembly

3.4.1 Introduction

Initial sensitivity studies have been done using a design with PWR fuel assembly dimensions for the multi-recycle of Th/U-233 fuel. The WIMS-9 [WIMS] code was used for the modeling effort. The current design requirements imposed are:

- Fuel cycle length of 1.5 years ideally, but 1 year is acceptable.
- A sustainable fuel cycle (breeding must be achieved so that no external fissile material is needed within the fuel cycle). Clearly, for the transition or startup cycle using thorium (Th) fuels, an external source of fissile material would be required.

In a first set of evaluations, a homogeneous fuel assembly model (where all fuel pins have the same material composition and geometry) is used to investigate if such PWR designs will be able to meet the fuel cycle requirements. The parameter space considered in this study currently involves moderator to fuel volume ratio (MR) and initial U-233 content of the Th based fuel. Moderator to fuel volume ratio is defined as the fraction of moderator volume to fuel volume. Preliminary results on the sensitivity to different initial uranium contents and fuel pin sizes are presented in the following subsections.

The core and fuel design characteristics are summarized in Table 3.1.

Table 3.1. Design Parameters for Homogeneous Th/U-233 Fuel Cycle Analysis

Parameter	Value		
Assembly geometry [a)]	17×17		
Core power, MWt [a)]	3000		
Number of assemblies [a)]	193		
Capacity factor, %	90		
Number of batches	3		
Active core height, cm [a)]	366		
Number of fuel rods per assembly [a)]	264		
Assembly gap, cm [a)]	0.08		
Fuel pitch, cm [a)]	1.25		
Fuel pellet density, g/cm^3	9.5		
Cladding thickness, cm [a)]	0.057		
Pellet cladding gap, cm [a)]	0.008		
Cladding material	Zr-4		
Cladding density, g/cm^3	6.5		
Guide tube inner radius, cm	0.5715		
Guide tube outer radius, cm	0.6120		
Coolant average density, g/cm^3	0.7116		
Moderator to fuel volume ratio (MR)	2.0	0.9	0.7
Fuel pellet radius, cm	0.410	0.500	0.533
Heavy metal per assembly, kg	425.0	621.0	720.3
Specific power density, W/g	36.6	25.0	21.6

a) Values given in Reference [Duderstadt 1976]

3.4.2 Standard PWR Performance Characteristics with Th/U-233 Fuel

Fuel assembly level data have been calculated for the infinite core multiplication factor (k-inf) and the Fissile Inventory Ratio (FIR). The FIR is currently being used in this study as an indicator of conversion ratio. Nuclei considered as fissile in the FIR calculations are U-233, U-235, Pu-239, Pu-241 and Pa-233 (which is counted as potential U-233 due to its short half-life: ~27 days). It should be noted that the fuel cycle does not sustain itself when FIR is less than 1 or equal to 1. This is because other fissile materials (e.g. U-235) in the thermal spectrum are not as effective as U-233 in sustaining the fission chain reaction. In this case, FIR > 1 is an approximate indicator of whether external source of U-233 is needed. As the homogeneous results introduced in the following all show that homogeneous 17-by-17 assembly design does not meet the requirement of sustainable Th/U-233 fuel cycle, FIR was used as an indicator of the conversion ratio of U-233. In the later sections on heterogeneous assembly designs, FIR233 is defined to exactly provide an indicator of U-233 breeding.

Figure 3.1 shows the results for various initial U-233 contents in the Th/U-233 oxide fuel with a standard PWR assembly design (Westinghouse, MR value 2).

Results for three initial U-233 contents are compared with respect to the k-inf: 2 wt%, 3 wt% and 4 wt%. It can be seen that the k-inf is sensitive to the initial U-233 content as expected. With an assumed leakage of 3% (equivalent 0.03 in k value), the k-inf for the 4 wt%-U-233 fuel takes about 2.6 years to fall below criticality (with final burnup ~33GWd/t); this would be equivalent to a discharged burnup of 50 GWd/t with cycle length of 1.4 years for a three-batch core. With the 3 wt% U-233 fuels, this period is shortened to about 1.5 years. The 2 wt%-U-233 fuel is impractical from a reactor operation viewpoint as k-inf immediately becomes subcritical.

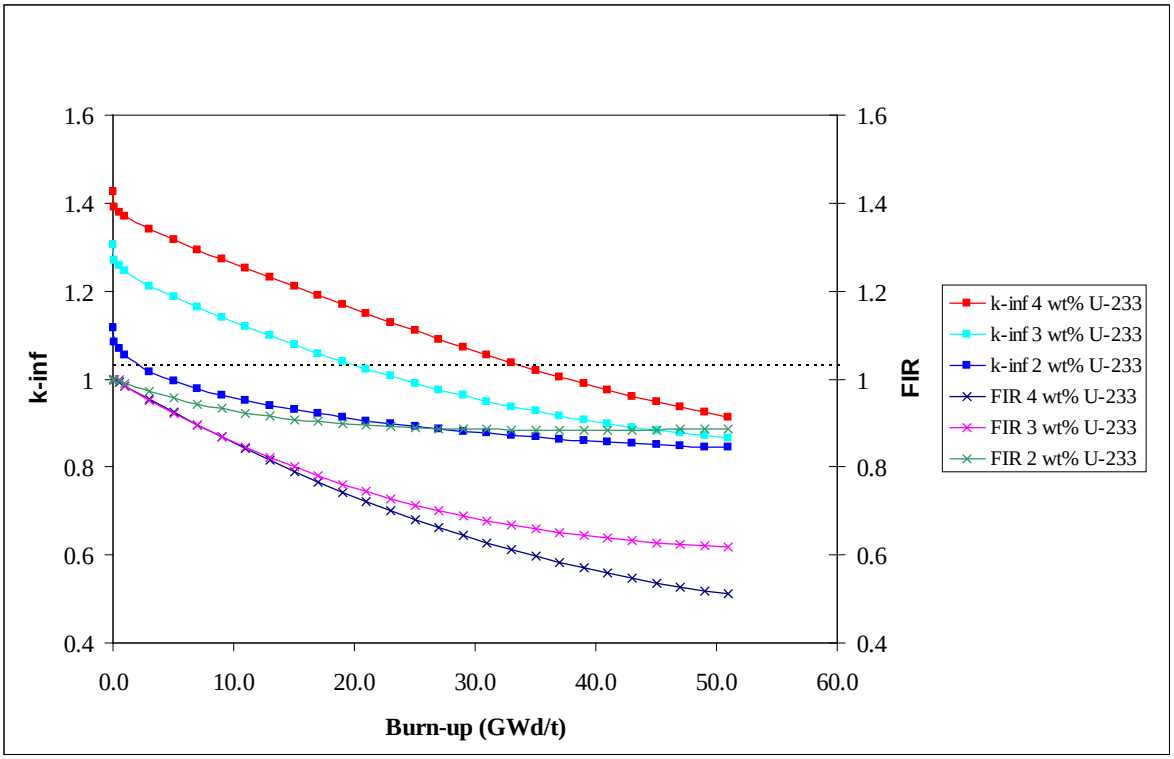


Figure 3.1. k-inf and FIR versus Burnup with MR of 2.0.

Only the 4 wt% U-233 fuel design would meet the greater than 1-year cycle length requirement for a three-batch fuel management scheme. The cycle lengths for 3 wt% U-233 fuels cannot support more than a one-batch core. All the fuels evaluated however do not meet the breeding requirement for the multi-recycling concept.

With the FIR curves shown in Figure 3.1, the FIR can be seen to drop consistently in all cases. For the 4 wt% U-233 fuel, the FIR drops to below 0.6 by the end of 3rd year. For the 3 wt% and 2 wt% fuels, the FIR drops to about 0.7 and 0.9 by the end of 1.5 years. *It is evident that decreasing the initial U-233 content can help to improve the FIR.* However, such a decrease in the initial U-233 content sacrifices k-inf rendering reactor operations impractical. Conversely, higher initial fissile content helps increase k-inf as more fissile materials are available for fission. But, it decreases the FIR because more fissile materials are burned than produced through the breeding process.

3.4.3 Trends with Variation of Moderator to Fuel Volume Ratio

Figure 3.2 shows results of k-inf and FIR for cases with various initial U-233 fuel contents with a lower MR of 0.9 made possible by enlarging the fuel pin size. Similar trend can be seen compared to the standard MR case. However, the initial k-inf decreases because the reduced MR leads to a decrease in the fission cross section of U-233.To have a sustainable Th fuel cycle, the design has to simultaneously meet the requirements of criticality (k-inf) and breeding condition (FIR). To this end, the current cases investigated could not provide a sustainable fuel cycle. However, it is interesting to notice that the FIR in the case of 2 wt% U-233 increases after about a year and becomes above unity after about 3 years. This implies that a better breeding condition is achieved by reducing MR with enlarged fuel pin size.

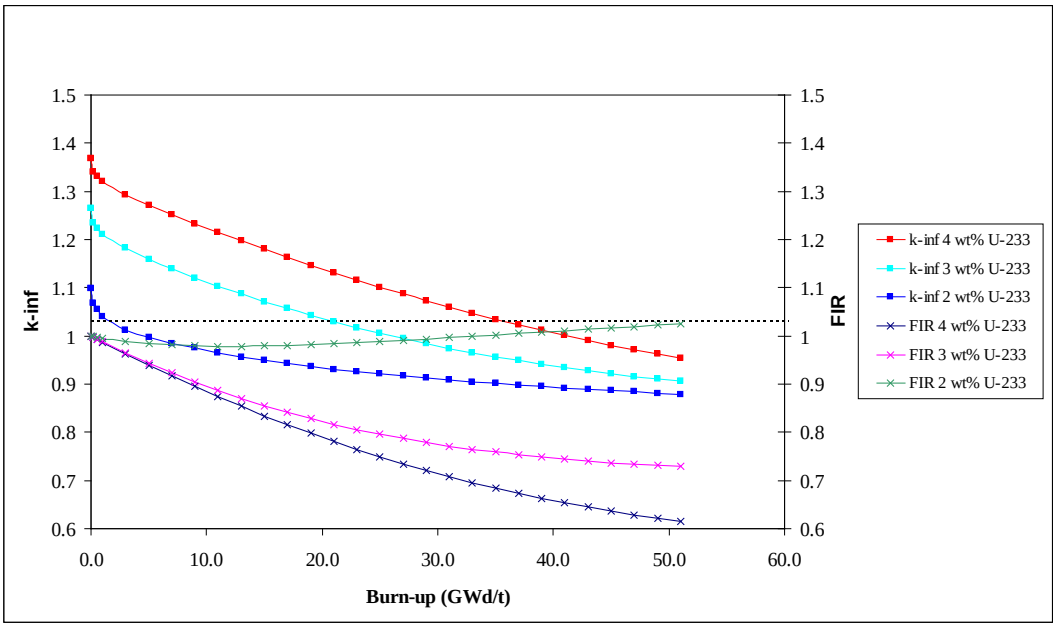


Figure 3.2. k-inf and FIR versus Burnup with MR of 0.9.

Based on these observations, additional sensitivity calculations were performed by reducing the moderator to fuel volume ratio further to 0.7 and results are provided in Figure 3.3. Similar trend can be seen for the k-inf curves compared to the standard MR case. The initial k-inf values are however lower for this new set of results. It should be noted that the FIR in the case of 2% initial U-233 content increases from beginning of irradiation and stays above unity at all time. This implies that a breeding condition can be achieved from the fissile inventory perspective. It is also apparent that such a breeding condition is accompanied by a k-inf curve dropping below unity instantaneously.

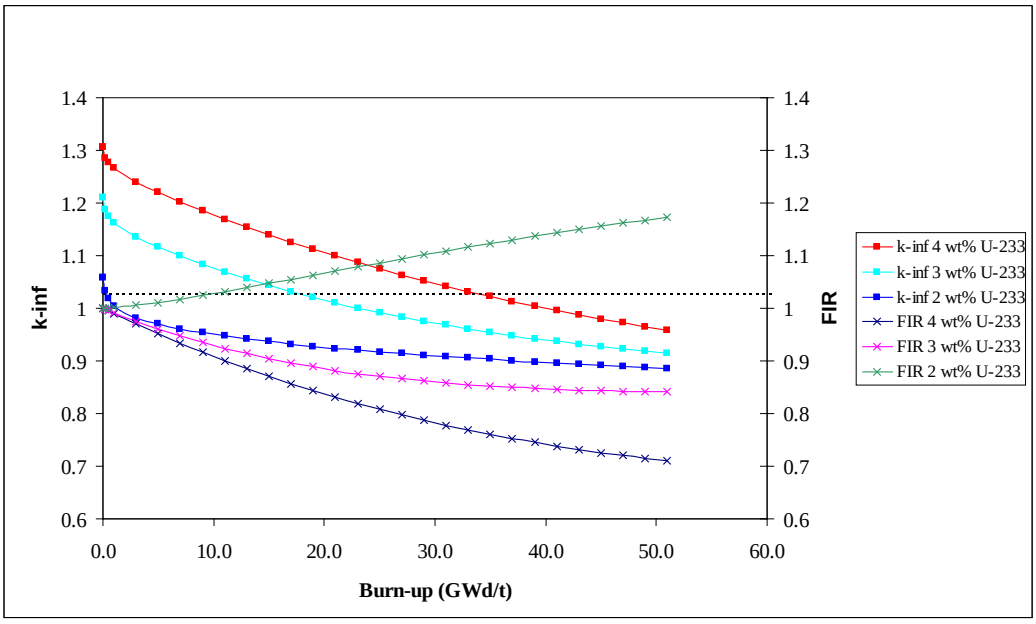


Figure 3.3. k-inf and FIR versus Burnup with MR of 0.7.

3.4.4 Trends with Fixed Initial U-233 Content

To further evaluate the effect of pin size (variation in MR), both k-inf and the FIR are compared fixing the initial U-233 content. The comparisons are provided in Figures 3.4, 3.5, and 3.6 for the 4 wt%, 3% and 2% fuels, respectively.

For the higher initial U-233 content (4 wt%), decreasing MR from 2.0 to 0.9 improves the critical burnup slightly, but further decreasing the MR from 0.9 to 0.7 decreases the critical burnup. A similar trend can be found with 3 wt% initial U-233 content case. For the 2 wt% initial U-233 content case, the fuel cycle becomes impractical from a criticality point of view. The observed trends imply that the influence of moderator to fuel volume ratio on k-inf is small within the parameter space considered here. However, the influence of MR on FIR is apparent. In the whole range of initial U-233 content considered here, the FIR is seen to improve consistently with decreasing MR with the improvement being most significant in the 2 wt% initial U-233 content case.

In order to explain the trends of the Th/U-233 depletion in a homogeneous fuel assembly, the normalized neutron spectra and the relative reaction rates are compared in Figure 3.7 and Table 3.2, respectively, for the 3 wt% U-233 fuel. The results obtained from 172-group WIMS-9 calculations are used for the spectral analyses. The neutron spectrum is hardened as the moderator volume reduces, and the harder spectrum improves the FIR because Th-232 has a relatively high absorption cross section in the epithermal energy range. The absorption (most reaction is fission) rate of U-233 decreases as the moderator to fuel volume ratio decreases to 2.0 to 0.7, while the absorption rate of Th-232 increases. As a result, the initial k-inf decreases and the FIR increases.

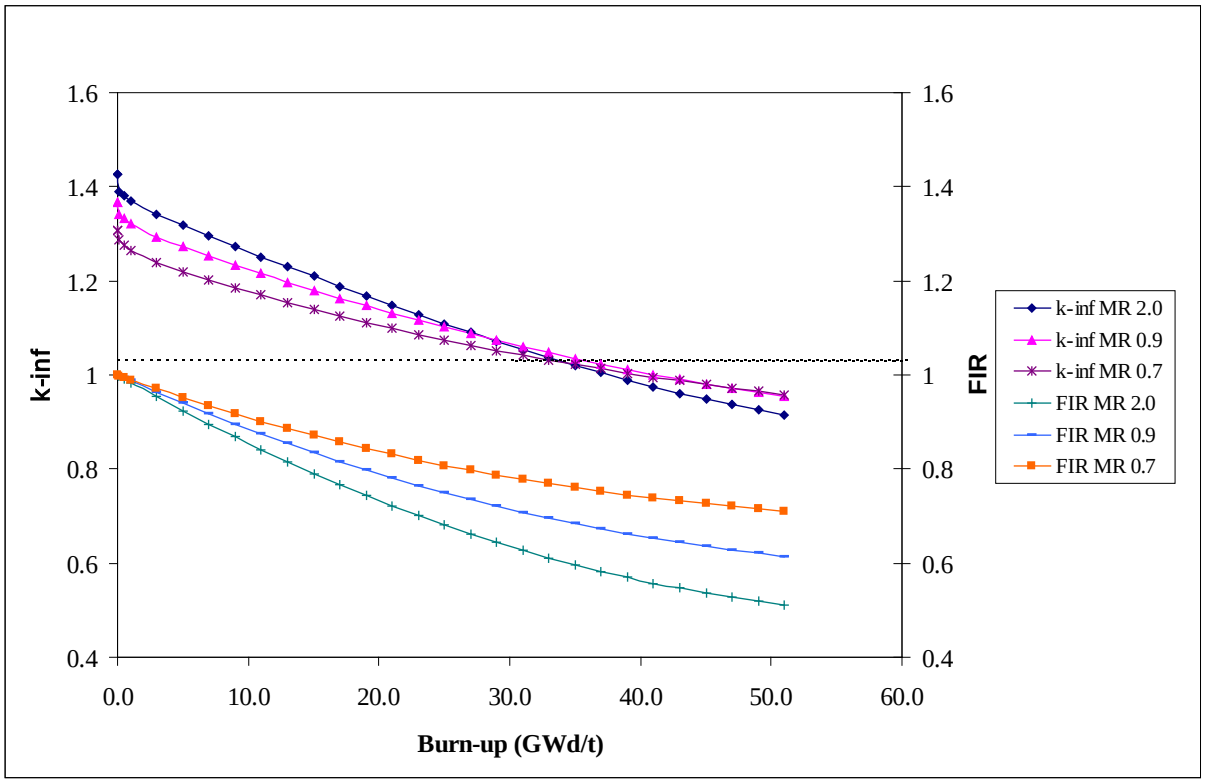


Figure 3.4. k-inf and FIR versus Burnup with 4 wt% initial U-233.

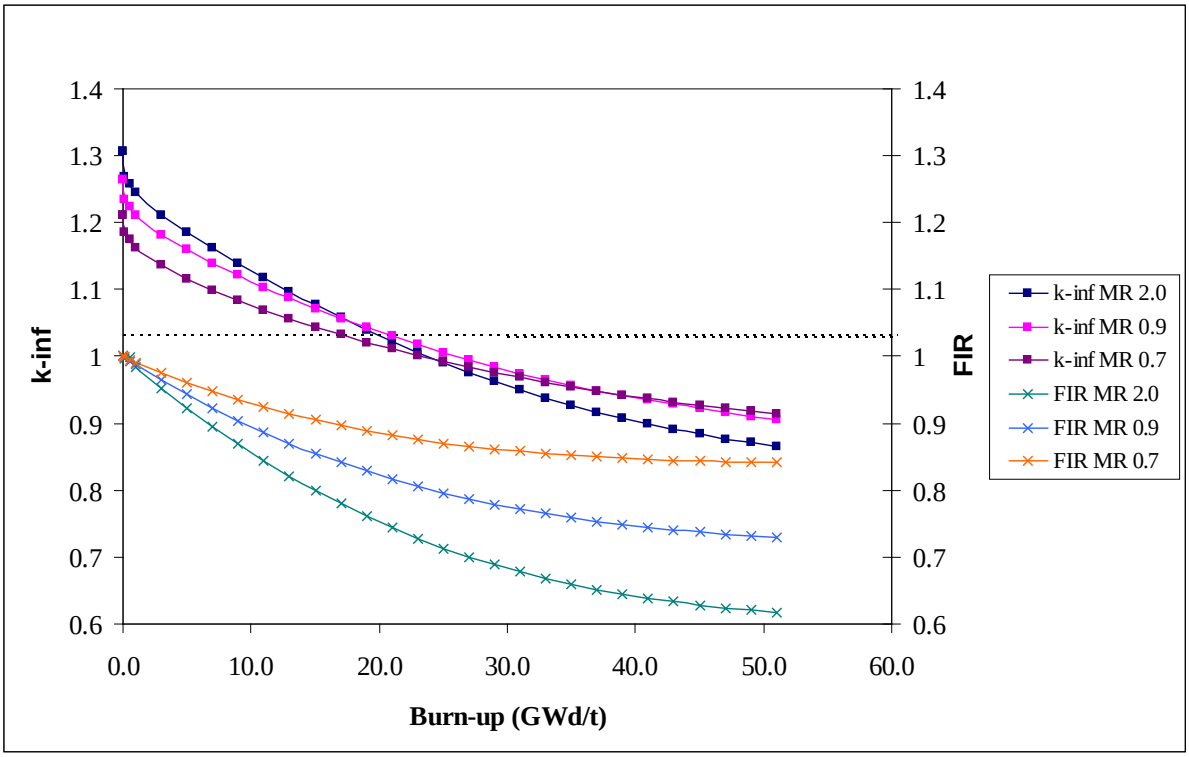


Figure 3.5. k-inf and FIR versus Burnup with 3 wt% initial U-233.

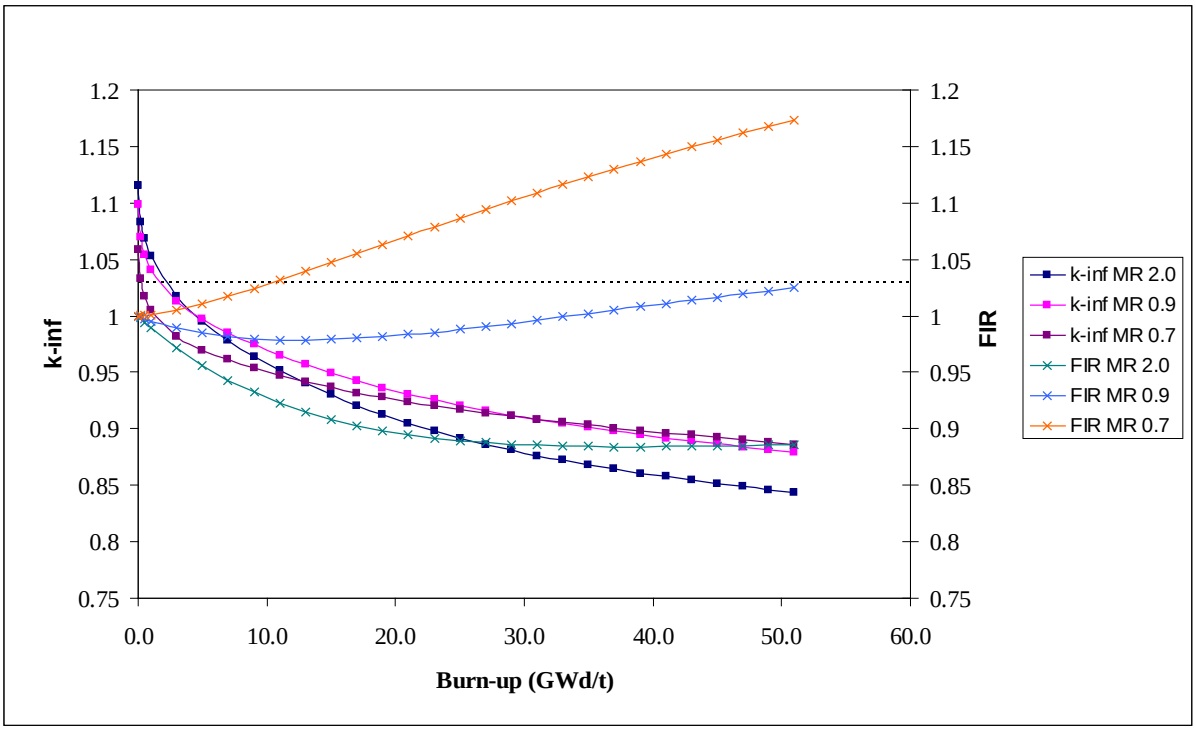


Figure 3.6. k-inf and FIR versus Burnup with 2 wt% initial U-233.

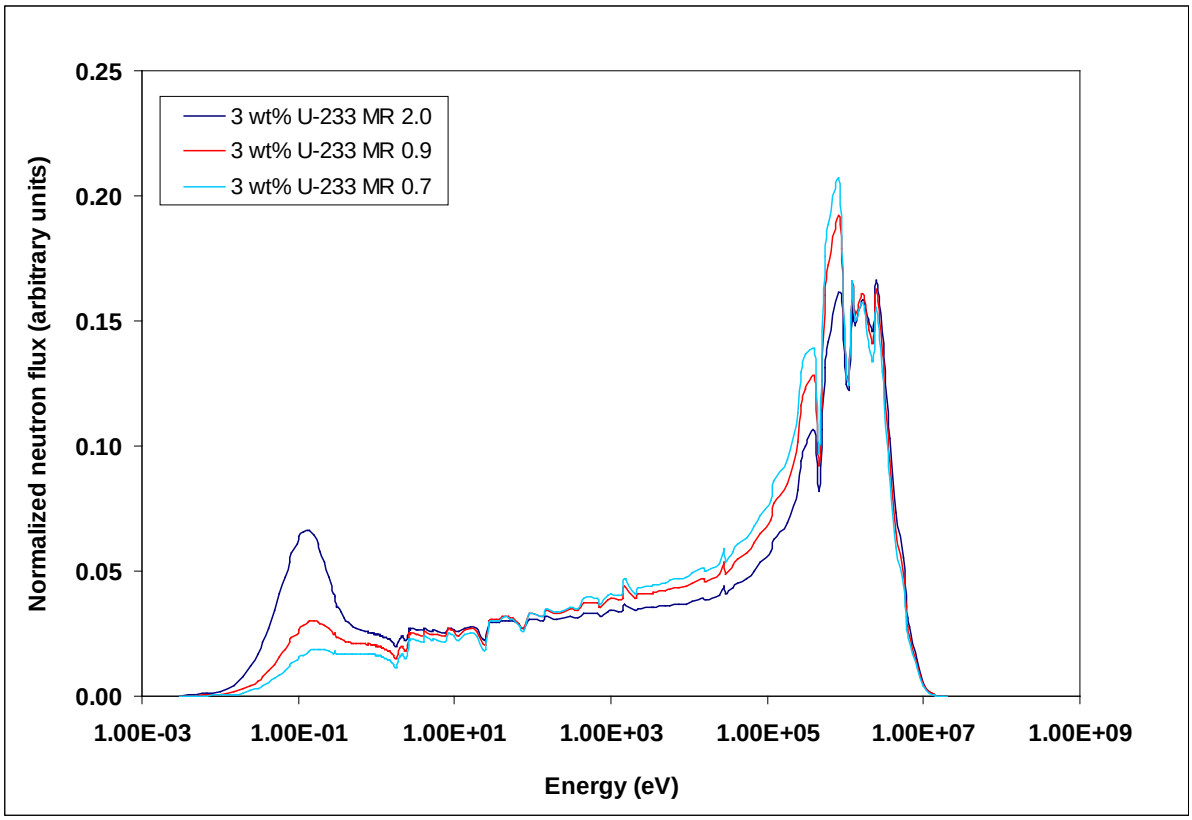


Figure 3.7. Normalized Neutron Spectra with 3 wt% Initial U-233.

Table 3.2. Relative Reaction Rates for 3 wt% Initial U-233

	Moderator to Fuel Volume Ratio 2.0		Moderator to Fuel Volume Ratio 0.7	
	Capture	Absorption	Capture	Absorption
Th-232	68.6%	33.4%	79.4%	42.1%
U-233	13.0%	57.8%	12.3%	53.6%
All other isotopes	18.4%	8.8%	8.3%	4.3%

3.4.5 Homogeneous Assembly Study Summary

The Th/U-233 fuel cycle performance parameters (such as k-inf, cycle length, and FIR) in a homogeneous PWR assembly are dependent on the U-233 content and MR. Increasing the initial U-233 content helps to improve k-inf but decreases the FIR. Decreasing the MR helps to improve the FIR but decreases k-inf. The current investigation led to the conclusion that a sustainable multi-recycle with Th/U-233 oxide fuel cannot be achieved with a homogeneous PWR assembly design within the parameter space of U-233 content (2- 4 %) and MR (2.0 – 0.7) achieved by increasing the fuel pin size. The estimated discharge burnup of a self-sustainable assembly design from the homogeneous scenario is less than 5 GWd/t, which is too small. It is necessary, therefore, that a heterogeneous assembly model be considered to obtain a feasible point design to realize a self-sustainable Th/U-233 fuel multi-recycle scheme. A heterogeneous assembly design similar to the seed-blanket design by Radkowsky would improve the FIR without sacrificing criticality as much [Galperin 1997]. These types of heterogeneous assembly designs have been considered in the following section.

4. SENSITIVITY STUDY: METHODS AND RESULTS

4.1 Introduction

In order to derive an optimum assembly design to best achieve the goal of self-sustainable Th/U-233 fuel cycle, various heterogeneous assembly designs were studies. During the process of the study, the k-inf curves of the heterogeneous assembly designs were found to be highly non-linear. As a result, the linear reactivity model is not directly applicable. A simplified point reactivity model was used to estimate the discharge burnup and cycle length with WIMS-9 assembly calculations. This model is used repeatedly during the part of this work where heterogeneous assembly designs were investigated.

A systematic investigation of the parameter space was also carried out in order to observe the trend leading to the final selection of the optimum heterogeneous assembly design. This investigation is essentially a sensitivity study in the parameter space of the scope of this work. The detailed method of this sensitivity study and its results are also covered in the following sub-sections.

4.2 Simplified Point Reactivity Model

A simplified point reactivity model (assuming flat power sharing between batches) was used to estimate the discharge burnup and the cycle length of the Th/U-233 fuel cycle. The following scheme describes the steps of the simplified point reactivity model:

1. An initial guess on the discharge burnup is made;
2. Based on the initial guess of the discharge burnup, the 3-batch fuel management scheme cycle length is determined as a third of the discharge burnup;
3. A numerical averaging then follows to obtain the average value of the k-inf at the end of the 1st, 2nd and 3rd batches in order to estimate the excess reactivity at the end of cycle. A deduction of neutron leakage/losses is necessary to determine if the excess reactivity is large enough (the neutron leakage/losses was assumed to be 3.0% in this study);
4. If the numerical average of k-inf, after deducting 0.03, is no less than unity, the discharge burnup is accepted and a higher discharge burnup is used for the next trial. If the numerical average, after deducting 0.03, is less than unity, the discharge burnup is rejected and a lower discharge burnup is used for the next trial;
5. The trials are finished and the discharge burnup and associated cycle length are determined when a higher discharge trial value fails and a lower discharge burnup value is accepted.

With this model, the discharge burnup and the cycle length of the heterogeneous assembly design were estimated. This model is believed to be more appropriate than directly applying the result of the linear reactivity model in this scenario where the reactivity let down curve exhibits strong non-linearity. It should be noted that due to the complex shape of the reactivity letdown curve, it was also confirmed that there was sufficient excess reactivity to overcome neutron leakage/losses at any time for the duration of the cycle. This was done by evaluating the average value of the k-inf of the 1st, 2nd, and 3rd batches at multiple points within the cycle.

4.3 Method for Sensitivity Study

A systematic investigation of the parameter space was used in this study to derive the optimum heterogeneous assembly design. The parameters considered in this study are the following: seed unit and blanket unit fuel pin sizes, initial U-233 content in the seed (blanket fuel contains ThO_2 only) and seed unit over blanket unit (SU/BU) ratio. Some alternative fuel design options considered are hydride fuel form for the blanket fuel and reduced moderator density.

A range of SU/BU ratios have been investigated by altering the assembly designs. Assembly designs studied include Radkowsky's seed and blanket assembly design, modified CORAIL checker board assembly design (both are 17-by-17 heterogeneous designs with 21.5 cm assembly pitch), 16-by-16 heterogeneous assembly design, etc. The range of SU/BU ratio is 1.0 to 0.25. All designs investigated include guide tubes so that there is adequate reactivity control.

It was found that the requirements of the Th/U-233 multi-recycle are better met with lower SU/BU ratio. The lower the SU/BU ratio, the more blanket units are available to breed U-233 as the blanket units are composed of pure ThO_2 fuel materials. Therefore, lowering SU/BU ratio helps with the breeding ratio which is reflected by the FIR. However, lowering SU/BU ratio necessitates derated core power. At BOL, the power of the reactor is concentrated in the seed units due to the initial distribution of fissile material (fissile U-233 in seed units only). The core power has to be derated such that the linear power density of the Th/U-233 system is similar to that of the current PWRs at BOL.

Derating the core power lowers the specific power density and therefore lowers the neutron flux of the reactor system. It was also found in this study that the specific power density has a direct impact on the depletion behavior of the reactor fuel. As a design with derated power of 1000 MWt is selected as the point design for this study, the impact of specific power density on fuel depletion behavior deserves some discussion. This discussion is provided in the next sub-section (Section 4.4).

Initial U-233 content were adjusted to generate the most ideal k-inf and FIR combination such that the assembly design is optimized to achieve the highest discharge burnup possible. However, as the SU/BU ratio becomes smaller, the necessary initial content of U-233 in the seed unit regions becomes higher. The optimum point design keeps the initial U-233 content at 7.8 wt%. This is well under the limit that would raise proliferation concerns.

Additionally, the seed and blanket unit pin sizes influence the performance of the assembly. The radius of the seed unit fuel pin was perturbed over a range of 0.36 cm to 0.55 cm and the radius of the blanket unit fuel pin was perturbed over a range of 0.50 cm to 0.66 cm (0.66 cm for a 16-by-16 assembly design). The blanket unit pin size is in general larger than that of the seed unit because large quantity of ThO_2 is needed to breed sufficient U-233 so that the FIR will hold above 1.0. This is consistent with the experience of the U.S. Shippingport LWBR in which large fuel pins were deemed necessary for the blanket units. It was found that the largest blanket unit fuel pin size allows superior performance of the system in meeting the fissile breeding condition.

It was also found that both k-inf and FIR are sensitive to the seed unit fuel pin size, which determines the MR in the seed region while the assembly pitch is fixed at 21.5 cm. In order to achieve high conversion of Th-232, the MR in both the seed region and the blanket region are kept low so that hard neutron spectra are obtained in both regions. Neutron spectra that are harder than those of the regular thermal systems were found to be crucial for achieving sustainability in the Th/U-233 fuel cycle in this study.

<u>A 17-by-17 heterogeneous design is selected as the optimum point assembly design for this study as a result of the parameter space search described above.</u> The SU/BU ratio was chosen to be 0.5 for this design. Reactor core power is derated to 1000 MWt in order to maintain the same linear power density as the current PWRs at BOL. Twenty-five guide tube positions are available in the assembly. More detailed description and performance data of this 17-by-17 assembly design are presented in the next Section 5.

It is necessary at this time to revisit the reason why heterogeneity helps to meet the requirement of the Th/U-233 multi-recycle. In this work, it was found that large quantity of pure ThO_2 fuel is necessary to achieve a high Th-232 conversion. Once U-233 was added in the blanket region, the FIR decreased as a result of increased fission versus capture. It follows that separating seed and blanket units can better assist the breeding of U-233 and other fissile materials and sustain the recycle scenario. This is consistent with the experience of the Shippingport LWBR and the finding in this work that smaller SU/BU ratio is better for Th-232 conversion.

4.4 Impact of Power Density on Fuel Depletion

In order to maintain a linear power density similar to that of the current PWRs at BOL, the power has to be derated for the heterogeneous assembly design (because of the reduced number of fuel pins in the assembly compared to current PWRs). A derated power results in a lower specific power density. It was found that the lower specific power density enables higher discharge burnup mainly due to an increase in reactivity.

The impact of specific power density on the depletion behavior was investigated by focusing on two physical parameters: reactivity penalty of Xe-135 and production of U-233. All results in this subsection are from WIMS-9 calculations and are presented in the WIMS-9 output units simply to show the comparisons of reaction rates and isotopic number densities.

Figure 4.1 shows the k-inf curves with a 16-by-16 heterogeneous assembly design for two different specific power densities of 19.81 MW/t and 4.0 MW/t. The 16-by-16 heterogeneous design is used here only as an illustration of the impact of specific power density on fuel depletion behavior. The initial values of the k-inf are identical, but the values are different after Xe equilibrium state, which is reached within a very short burnup (<0.2 GWd/t).

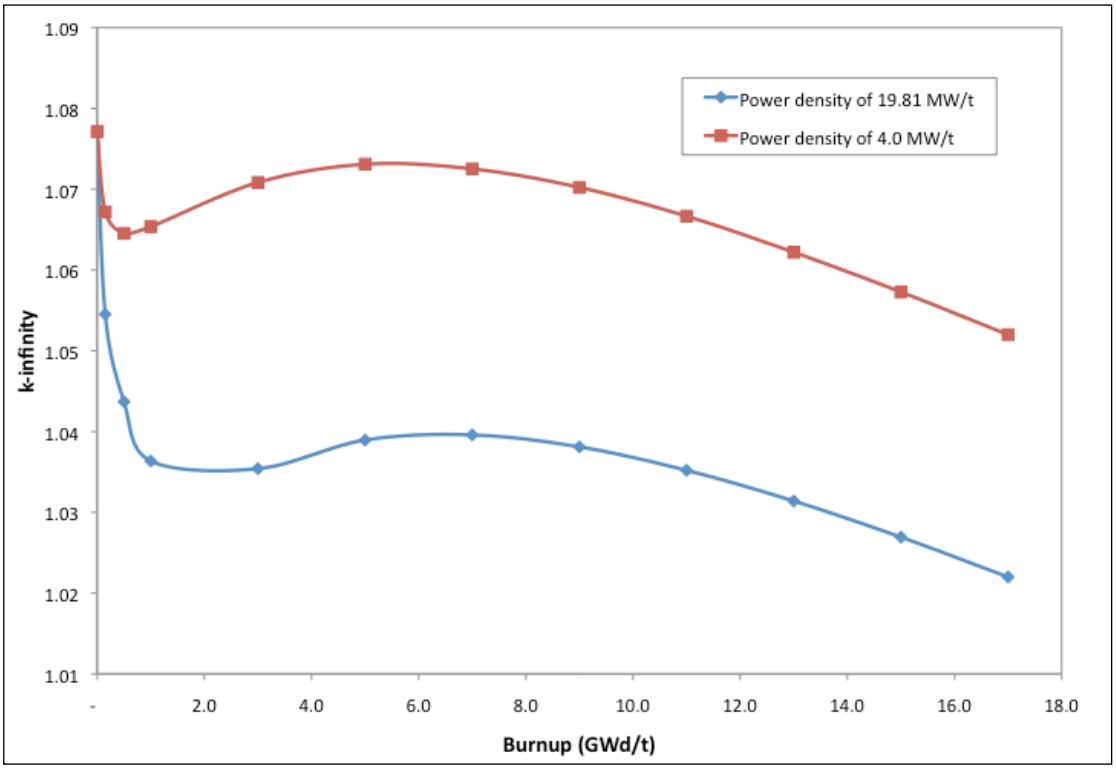


Figure 4.1. Comparison of k-inf.

The reaction rates at the Xe equilibrium state for the two cases are compared in Table 4.1. The total flux decreases as the specific power density decreases. As a result, all reaction rates decrease proportionally to the decrease of flux level except for the Xe-135 absorption rate. As shown in Table 4.1, the reaction rates of the lower power density (4 MW/t) system are about 20% of those of the higher power density (19.8 MW/t) system, but the absorption rate of the Xe-135 for the lower power density system is about 6 % of that of the higher power density system. The reduced reaction rate of Xe-135 is mainly due to the lower Xe-135 number density level at the equilibrium state (shown in the following Table 4.2). It is noted that the Xe-135 number density is dependent on the flux level at a relatively low specific power density. As

Xe-135 has a very large absorption cross section (effective cross section is about $5x10^4$ barns in PWRs), the lower number density of Xe-135 for the lower power density system leads to lower neutron absorption and therefore higher k-inf.

Table 4.1. Reaction Rates at Xe Equilibrium State (0.5 GWd/t Burnup)

	Power density of 19.8 MW/t		Power density of 4.0 MW/t	
	Absorption	υfission	Absorption	υfission
Fissile	1.21E+13	2.65E+13	2.43E+12	5.36E+12
Fertile	1.32E+13	8.27E+11	2.60E+12	1.64E+11
Structure	6.18E+11		1.22E+11	
Xe-135	2.44E+11		1.47E+10	
Sm-149	4.89E+10		1.07E+10	
Other FP	4.45E+10		9.50E+09	
Total	2.62E+13	2.74E+13	5.188E+12	5.52E+12
K-inf	1.04370		1.06454	
Total flux	4.24E+14		8.56E+13	

Table 4.2 shows the evolution of the isotopic number densities of Xe-135 and U-233 with respect to the burnup. The Xe-135 number density of the higher power density system is about a factor of 3.5 higher than that of the lower power density system. Also, the U-233 number density of the lower power density system is about 2% higher than that of the high power density system. This is due to the higher decay fraction of Pa-233. At lower specific power density, lower neutron flux level results in less absorption reactions of Pa-233. As a result, a higher fraction of Pa-233 decays to U-233 (with a short decay half life of 26.7 days) leading to a slightly higher U-233 production and contributing to a higher multiplication factor as well.

Table 4.2. Comparison of Number Density of Xe-135 and U-233

Burnup, GWD/t	Xe-135		U-233	
	19.8 MW/t	4.0 MWd/t	19.8 MW/t	4.0 MWd/t
0.0	0	0	3.75E-04	3.75E-04
0.2	5.27E-09	1.53E-09	3.73E-04	3.74E-04
0.5	5.28E-09	1.54E-09	3.70E-04	3.73E-04
1.0	5.28E-09	1.54E-09	3.67E-04	3.73E-04
3.0	5.32E-09	1.55E-09	3.65E-04	3.74E-04
5.0	5.35E-09	1.55E-09	3.66E-04	3.74E-04
7.0	5.37E-09	1.55E-09	3.66E-04	3.73E-04
9.0	5.39E-09	1.55E-09	3.65E-04	3.72E-04
11.0	5.41E-09	1.55E-09	3.64E-04	3.71E-04
13.0	5.43E-09	1.55E-09	3.63E-04	3.69E-04
15.0	5.45E-09	1.55E-09	3.62E-04	3.68E-04
17.0	5.47E-09	1.55E-09	3.60E-04	3.66E-04

4.5 Sensitivity Study Results

Although various assembly designs have been investigated only a few of the results are selected for discussion in this section. This is because the other designs have not met the requirements of the Th/U-233 multi-recycle and it would be not informative to show the details of the many assembly designs studied in this work which failed in meeting the design criteria. The following Figure 4.2 contains the key results of the sensitivity studies performed in this investigation, which are generated for the 17-by-17

heterogeneous assembly designs. Therefore, this result in particular reflects a boundary of the capability of Th/U-233 multi-recycle in PWR assembly designs as the 17-by-17 heterogeneous assembly design was determined to be the optimum point assembly design.

The linear fit (with least square of 0.997) suggests that for every 1% decrease in leakage of reactivity, there can be a gain of 4.21 GWd/t discharge burnup, which corresponds to about 0.63 years increase in cycle length (with a fixed specific power density of 6.73 W/g). If a 1.5% leakage is assumed, the discharge burnup can be as high as ~24 GWd/t and the associated cycle length is ~3.6 years (with 3-batch fuel management).

This result suggests that a core level model is necessary to capture appropriately the core leakage. It might also be possible to use core level heterogeneity to enhance the core performance (increase the discharge burnup) by lowering the leakage.

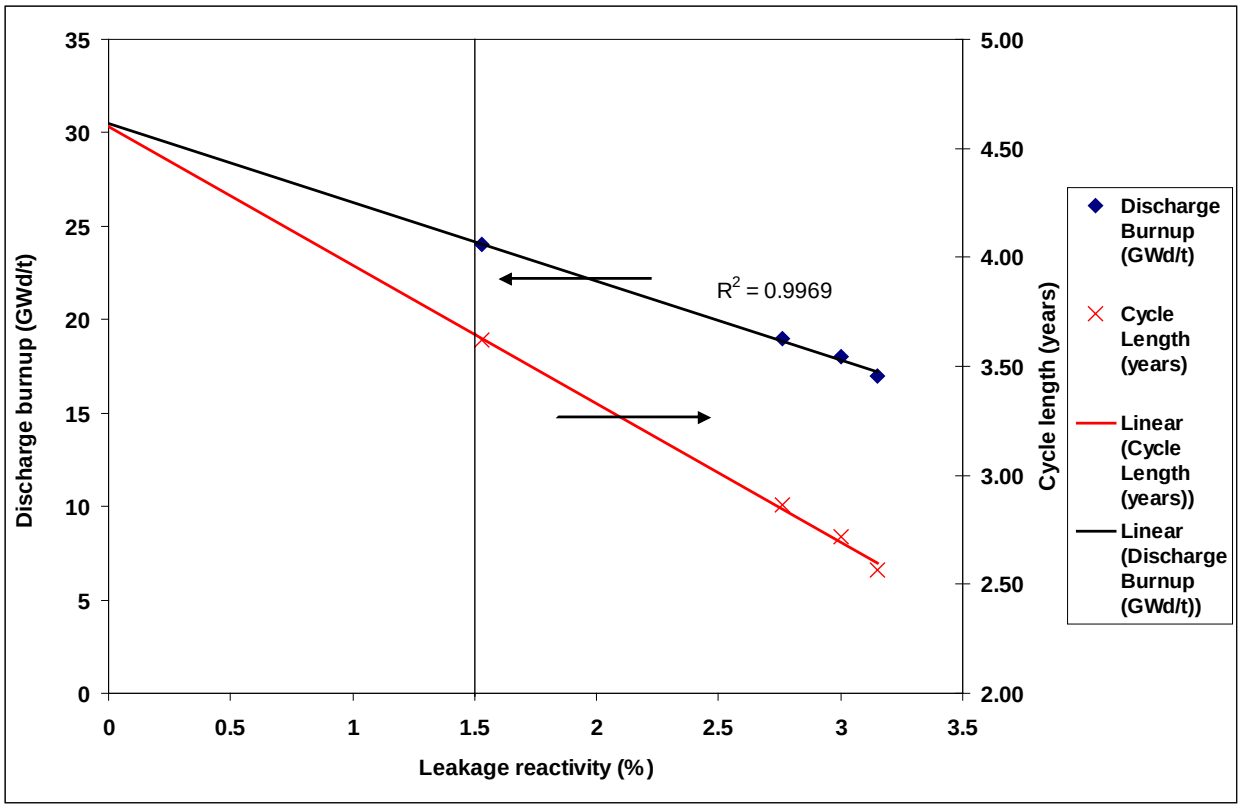


Figure 4.2. Key Sensitivity Study Result: Discharge Burnup versus Leakage Reactivity.

In the following Figure 4.3, another set of data points with a different seed unit MR value of 1.33 are added. This new data set demonstrates the limitation of the system design in meeting k-inf and FIR requirements when MR value is increased in order to ease thermal-hydraulic concerns. Compared to the optimum 17-by-17 heterogeneous assembly design, only the seed unit fuel pin size was reduced to increase the MR in that region.

These new set of data and the Figure 4.3 suggest that in order to help resolve possible thermal-hydraulic issues by reducing the pin diameters, either the discharge burnup has to be reduced (the data point of 3% leakage, 10 GWd/t discharge burnup) or the core leakage be lower than 3.0%. This indicates an inherent competition between moderation (when too low, causes thermal-hydraulic problems) and achievable discharge burnup assuming a 3-batch fuel management scheme.

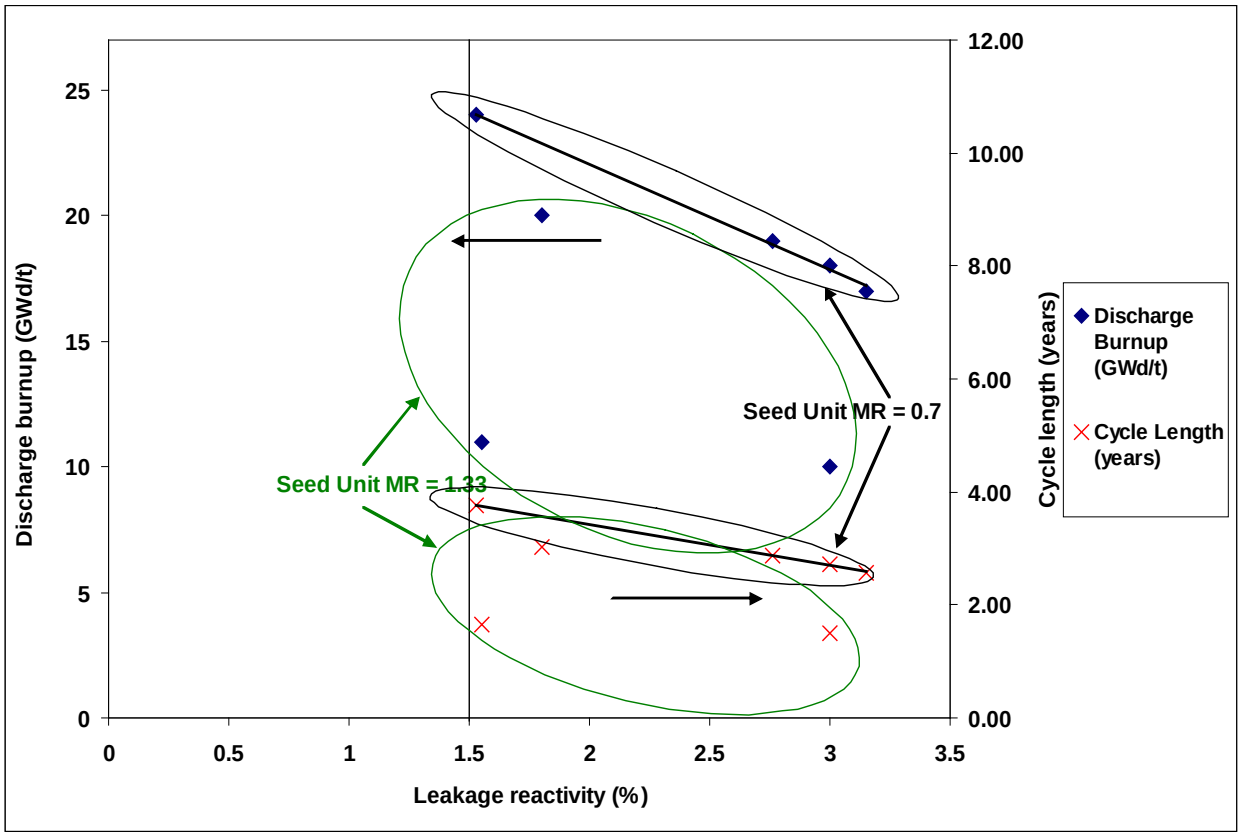


Figure 4.3. Sensitivity Study Result (More MR values Included): Discharge Burnup versus Leakage Reactivity.

4.6 Recommendations from Sensitivity Study

The following Table 4.3 summarizes the fuel design strategy options considered in this study and gives recommendations on future research activities. Initial scoping studies on some other fuel design strategies have also been conducted in this investigation to provide alternative perspectives. Major advantages and disadvantages are discussed briefly, based on which recommendations are drawn. The main activity recommended is based on the evaluation of the 17-by-17 heterogeneous assembly design. A detailed thermal-hydraulics study is necessary to demonstrate the feasibility of the system in removing the heat efficiently. A full-core model is also needed to estimate the leakage. Radial and axial blanket may be added to further reduce the leakage and improve the performance of the system.

Other fuel design options have also been considered. These include reducing MR by decreasing moderator (water) density. This is done manually in the WIMS-9 input, the temperature effects on the cross sections are not considered. (In reactor operation, temperature has to be increased to reduce moderator density). The fuel design options also include changing the fuel form of the blanket fuel from oxide fuel to hydride fuel (ThH_2). It was found that reducing water density from 0.713 g/cm^3 to 0.5 g/cm^3 produces similar effect as reducing moderator to fuel volume ratio from 1.33 to 0.75. This is because reducing the moderator density in fact reduces the amount of moderator even though the volume does not change. Future activities are not recommended on the reduced water density option due to the multiple difficulties that have to be overcome in order to achieve some small gain on the discharge burnup of the system.

The hydride fuel form option looks promising in that it extends the discharge burnup to a higher value (e.g., 25 GWd/t was obtained for a data point). This study produced similar results to the results from

Daphna Volaski [Daphna 2009] with similar conditions. However, reactivity booster is needed to start the core and its feasibility needs to be investigated. Such a study on the feasibility of reactivity booster falls out the scope of the current study.

Table 4.3. Recommendations for Future Activities

Fuel cycle options considered	Advantages	Disadvantages	Recommendations
Standard oxide fuel form + standard water density (17-by-17 heterogeneous assembly design)	1. Well known fuel form. 2. Minimum modifications on assembly design.	1. MR in seed region is low (thermal-hydraulic properties need to be investigated).	1. Conduct detailed thermal-hydraulics study to identify possible TH issues. 2. Full-core modeling to estimate reactivity leakage and also evaluate core heterogeneity impact.
Standard oxide fuel form + reduced moderator density	An improved discharge burnup by about 3-4 GWd/t.	1. Heat removal problems from thermal-hydraulics concerns. 2. Higher core temperature will impact on cross-sections, requiring very detailed coupling study of neutronics and TH.	Not considered in the future activities.
Hydride blanket fuel form + standard water density	An improved discharge burnup to about 25 GWd/t, cycle length of about 1.2 years [Daphna 2009].	1. No practical experience with fuel form fabrication, handling, etc. 2. Requires reactivity booster at the beginning of the fuel cycle.	1. Evaluate the feasibility of the booster. 2. Test ideas without implementing duplex fuel to avoid proliferation concerns.

5. STUDIES FOR PWR WITH HETEROGENEOUS ASSEMBLY

5.1 Introduction

A 17-by-17 heterogeneous design is selected to best meet the objectives of this study (see section 4.3). The fuel cycle information and mass flow data, decay heat, toxicity, etc. are given for this specific assembly design in this section. The advantages and disadvantages of this design are discussed. The PWR fuel cycle with 50 GWd/t discharge burnup was considered as the reference fuel cycle option in this study [Kim 2010].

5.2 Calculation Method

In this section, fuel cycle impact data for the 17-by-17 heterogeneous assembly design are compiled. These include masses, radiotoxicity, decay heat, neutron source strength and photon source strength. ORIGEN-2 [ORIGEN2] calculations were used to generate the pertinent mass flow data. A two-step process was utilized: the assembly lattice calculation was done using the WIMS-9 code to generate one-group cross sections and fluxes (along with the other core parameters), and the ORIGEN-2 depletion calculations were done using the one-group fluxes and the effective cross sections (which override base one-group cross-sections on ORIGEN-2 libraries). It needs to be pointed out that separate ORIGEN-2 calculations were run with the seed unit mass isotopic data and the blanket unit mass isotopic data. The results for the radiotoxicity, decay heat, neutron source strength and photon source strength are then obtained by averaging the results from both units according to their heavy metal weight ratio in the core.

The ORIGEN-2 calculations were performed with the following one-group base libraries:

- PWRDU3TH.LIB - ThO_2-enriched with denatured U-233 for the Th/U-233 systems.
- PWRUE.LIB - U-235 enriched UO_2, 50 GWd/t discharge burnup for the standard UOX system.

5.3 The 17-by-17 Heterogeneous Assembly Design

Design parameters for the 17-by-17 heterogeneous design are provided in Table 5.1. There are three attributes of this design that deserve some discussions.

Firstly, the core contains higher amount of fuel material than the current PWRs. The core power has to be derated in order to match the linear power density of the current PWRs at BOL. As a result, the specific power density of this design is significantly lower than that of the PWRs. With the experience of the U.S. Shippingport LWBR, the lower specific power density is expected (the U.S. Shippingport LWBR had ~6W/g specific power density).

Secondly, the moderator-to-fuel volume ratio values are low in both the seed and the blanket regions. This raises concerns for thermal-hydraulic properties of this specific assembly design. The low MR values are results of large seed and blanket fuel pins and short distances between fuel pins. Detailed thermal-hydraulics analysis is planned in future activities as it is crucial to address reactor operation issues with this assembly design.

Finally, the normalized neutron spectra of both the seed region and the blanket region were examined. The spectra are harder than the spectra of standard thermal systems both at the BOL and at the EOL (for the Th/U-233 system). This confirms the assumption that in order to achieve high conversion of Th-232, a hard neutron spectrum is necessary.

Figures 5.1 and 5.2 show the schematic drawing of the 17-by-17 heterogeneous assembly design and the k-inf and FIR versus burnup, respectively. The solid lines in Figure 5.2 represent the 1.0 cutoff for the FIR (when FIR falls below the cutoff, 1.0, the reactor is no longer breeding) and the leakage cutoff, which is 1.03 (assuming 3% neutron leakage/losses for the PWR, consistent with the value used for investigation of the homogeneous fuel assembly designs).

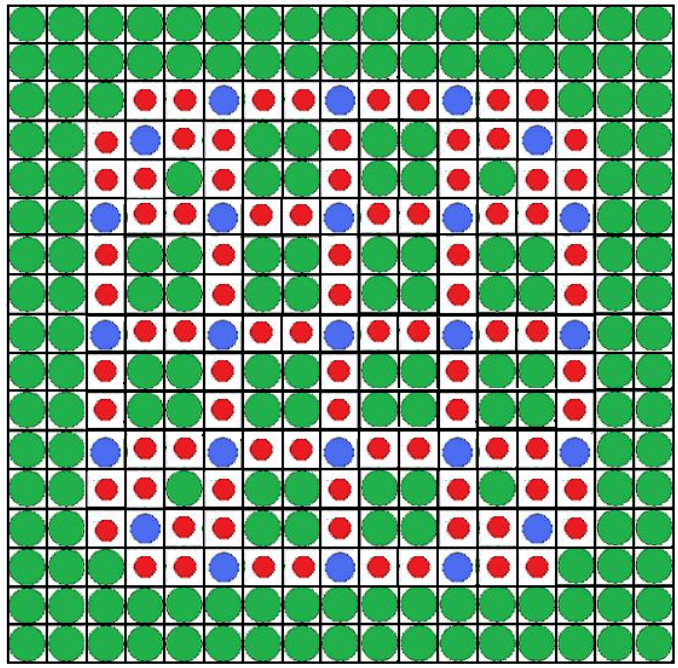
Both the assembly pitch and the fuel pitch are similar to those of current PWRs (21.5 cm and 1.26 cm respectively) in order to reduce the modifications of the assembly to a minimum level. The k-inf, in this design, is kept above the 1.03 threshold close to the start-up of the core in order to avoid reactor operation difficulties. It is important to realize that the dip in the k-inf curve would impact on the reactor operation if k-inf fell below the reactivity leakage threshold.

The estimated discharge burnup, assuming a 3-batch fuel management scheme, is about 18 GWd/t. This corresponds to a fuel cycle length of about 990 days (assuming a 0.9 capacity factor).

Even though the FIR is constantly above the threshold 1.0 before the fuel reaches 18 GWd/t burnup, it is not clear that a subsequent reactor cycle (second and subsequent passes) fueled with the discharge fuel will sustain the same 18 GWd/t burnup level. It is reasonable to assume that all Pa-233, with a relatively short decay half-life (27 days) compared to the cooling and reprocessing time, decay to U-233. However, the U-235 produced as the result of capture may not be as efficient as U-233 in fission reactions and therefore the fissile inventory becomes less efficient overall in fission reactions leading to less reactivity. In order to test if this 17-by-17 heterogeneous assembly design can hold the 18 GWd/t burnup with its own discharge state fuel on a cycle by cycle basis, at least one more recycle calculation has to be run with the WIMS-9 assembly model. This was not done in the current study but is planned in future activities.

Table 5.1. Design Parameters for 17-by-17 Heterogeneous Th/U-233 Fuel Cycle Analysis

Parameter	Value	
Assembly geometry	17x17	
Core power, MWt	1000	
Number of assemblies	193	
Capacity factor, %	90	
Number of batches	3	
Active core height, cm	366	
Number of seed fuel rods per assembly	88	
Number of blanket fuel rods per assembly	176	
Assembly gap, cm	0.04	
Fuel pitch, cm	1.26	
Assembly pitch, cm (including water gap)	21.5	
Seed fuel composition	$(ThU\text{-}233)O_2$ (7.8wt% U-233)	
Blanket fuel composition	ThO_2	
Seed fuel pellet density, g/cm^3	9.6	
Blanket fuel pellet density, g/cm^3	9.5	
Cladding thickness, cm	0.057	
Cladding material	Zr-4	
Cladding density, g/cm^3	6.5	
Guide tube inner radius, cm	0.57	
Guide tube outer radius, cm	0.61	
Coolant average density, g/cm^3	0.71	
Moderator to fuel volume ratio (MR)	0.7 (Seed region)	0.3 (Blanket region)
Fuel pellet radius, cm	0.51 (Seed region)	0.57 (Blanket region)
Heavy metal per assembly, kg	770.33 (221.46 kg in seed and 548.86 kg in blanket)	
Specific power density, W/g	6.73	



Green-Blanket Units
Red-Seed Units
Blue-Guide Tubes

Figure 5.1. Schematic Drawing of 17-by-17 Heterogeneous Assembly Design.

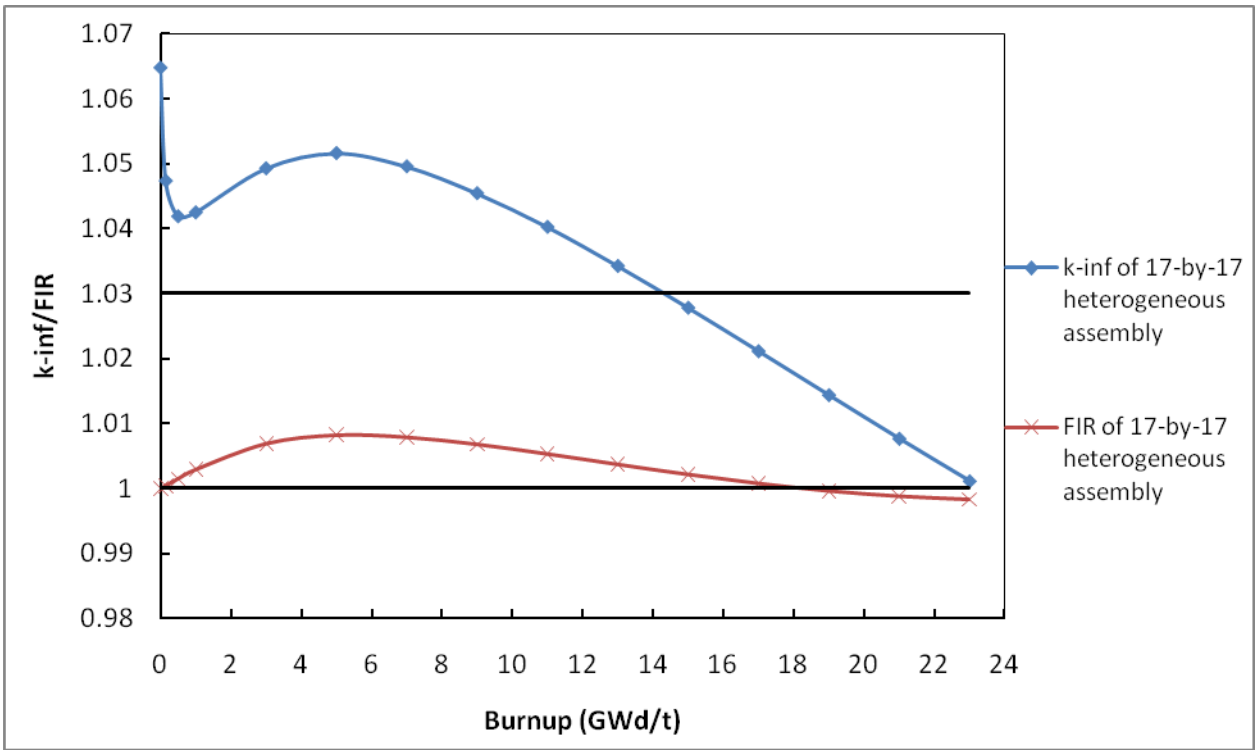


Figure 5.2. k-inf and FIR versus Burnup.

5.3.1 Mass Flow Data for PWR with 17-by-17 Heterogeneous Assembly Design

Table 5.2 presents the WIM-9 assembly level fuel cycle mass flow data for the Th/U-233 system. It is noted that the production of plutonium (Pu) and other minor actinides (MA) is extremely low, indicating a probable benefit of implementing the Th fuel cycle.

The results of Table 5.2 show that the initial fuel has 100% U-233/total-U and a fissile content of 2.2% in the total heavy metal. For the discharge fuel, the fissile content is 2.3%, but the U-233 fraction in the uranium is 0.90. The separation of this fissile uranium from the thorium-based fuel would constitute an increase in the proliferation risk. So appropriate safeguards and physical protection must be in place for the Th/U-233 sustainable fuel cycle. Alternatively, natural U-238 could be used to denature the fuel. This introduces the possibility that the design goals for a sustainable Th/U-233 fuel cycle might not be met, due to introduction of actinides that are neutron poisons and the fact that other fissile nuclides are less efficient sources of fission neutrons than U-233 in the thermal spectrum. It would also result in the production of Pu-239 and other higher actinides, which would offset any advantages of not having these nuclides in the fuel cycle (particularly in the UNF).

Table 5.2. Mass Flow Data for 17-by-17 Heterogeneous Th/U-233 Fuel (kg/assembly)

Nuclide	Charge	Discharge (18 GWd/t discharge burnup)
Th-232	753.16	737.02
Pa-233	0	0.22
U-233	17.17	16.70
U-234	0	1.64
U-235	0	0.25
U-236	0	1.53E-02
U-237	0	1.30E-05
U-238	0	1.83E-06
Np-237	0	8.67E-04
Np-239	0	4.68E-10
Pu-238	0	1.17E-04
Pu-239	0	5.77E-06
Pu-240	0	5.16E-07
Pu-241	0	2.60E-07
Pu-242	0	1.65E-08
Am-241	0	9.60E-09
Am-242m	0	1.07E-10
Am-243	0	1.34E-09
Cm-242	0	7.00E-10
Cm-243	0	7.41E-12
Cm-244	0	1.27E-10
Cm-245	0	5.02E-12
Total	770.33	755.86

*Fission product mass at discharge is 14.4 kg/assembly

Table 5.3 compares the isotopic discharge masses that were obtained from the ORIGEN-2 and the WIMS-9 calculations for the Th/U-233 system. The results are normalized to initial heavy metal metric ton. The differences between the ORIGEN-2 and WIMS-9 results (Diff) for the major isotopes including Th-232 and U-233 are no more than one tenths of a percent. The differences are as high as 5% for higher mass isotopes. However, the ORIGEN-2 results can be utilized for the fuel cycle analysis as the differences for the major nuclides are less than 1 percent.

Table 5.3. Comparison of Isotopic Masses at Discharge (g/t initial heavy metal)

Nuclide	Th/U-233 (18 GWd/t discharge burnup)		
	ORIGEN-2	WIMS-9	Diff (%)
FP	18840.33	18732.23	0.6
Th-232	956672.68	956764.47	0.0
Pa-233	289.49	287.94	0.5
U-232	43.42	n/a	n/a
U-233	21648.69	21678.75	-0.1
U-234	2151.37	2134.97	0.8
U-235	321.54	329.91	-2.5
U-236	19.20	19.90	-3.5
U-237	1.63E-02	1.68E-02	-3.3
U-238	2.27E-03	2.38E-03	-4.5
Np-237	1.07	1.13	-4.7
Pu-238	1.45E-01	1.52E-01	-5.1
Pu-239	7.19E-03	7.49E-03	-4.0
Pu-240	6.44E-04	6.70E-04	-3.9
Pu-241	3.24E-04	3.37E-04	-3.9
Pu-242	2.06E-05	2.14E-05	-3.6

5.3.2 Radiotoxicity

A variety of measures are available for quantifying the radiotoxicity of used nuclear fuel. In this study, the radiotoxicity data has been calculated using the ingestion dose coefficients obtained from the ICRP 72 database [ICRP 1996] by Idaho National Laboratory (INL). The UNF radiotoxicity values for the standard UOX system (the reference PWR system) and Th/U-233 system were estimated up to 10 million years after discharge and compared. For the purpose of comparison, the UNF radiotoxicity is normalized to unit electricity generated in one year (Sv/GWe-yr) and then normalized again to the radiotoxicity of the natural uranium ore that is needed to produce the LEU fuel for the reference standard PWR UOX fuel cycle.

The following Figure 5.3 shows the normalized radiotoxicity for the standard UOX system (50GWd/t discharge burnup) broken down by the leading contributors to the radiotoxicity. It can be seen that from 10 years after discharge to about 100 years after discharge, fission products (FP) and TRU are the main contributors to the total radiotoxicity. The contribution from the fission products to the radiotoxicity decreases sharply between 100 and 1000 years due to their natural decay. The TRU remains the main contributor to the radiotoxicity up to about a few hundred thousand years. It can also be observed that contribution from uranium isotopes is consistently low throughout the time range investigated.

Figure 5.4 shows the UNF normalized radiotoxicity for the Th/U-233 system, again broken down to indicate the leading contributors. This figure contains data for all the nuclear elements in the UNF (i.e. no assumption on which elements are removed during fuel separations) and are provided in this form for illustration. (For comparison to the UOX system in Figure 5.5, the actual elements and separations losses assumed to be going to the repository are included.) It can be seen that from 10 years to about 100 years after discharge, fission products (FP) and uranium isotopes are the main contributors to the total radiotoxicity. For these cases U-232 is the main contributor within the uranium isotopes. The FP contribution to the radiotoxicity decreases sharply between 100 and 1000 years. Contribution from the uranium isotopes then decreases and U-233 decays gradually into Th-229, which becomes the main contributor from 1000 years to about a million years. The TRU, different from the earlier scenario of standard UOX fuel cycle, contributes very small amount to the total radiotoxicity. This is because in the Th/U-233 cycle, very small amount of TRU are produced. It also needs to be pointed out that although Po-210, decay daughter of U-234, is present in the system only in very small amount, its contribution cannot be ignored due to its very high ingestion coefficient.

Figure 5.5 compares the total high level waste (HLW) radiotoxicity of the standard UOX fuel system and the Th/U-233 fuel system (18 GWd/t discharge burnup). For the UOX system, the UNF is assumed sent to the repository as HLW. In the Th/U-233 system, only the nuclear elements not recycled are assumed to be sent as HLW to the repository. These include the fission products, all non-uranium and non-thorium heavy elements, and 0.1% of the discharged uranium and thorium mass, which is assumed as system-loss during fuel separations. It is assumed that separation is done 5 years after discharge. The results indicate that in the first 20 years after discharge, the normalized radiotoxicity of the Th/U-233 system is greater than that of the UOX system, because its lower burnup results in more used nuclear fuel per energy generation.

After approximately 50 years, due to the natural decay of the fission products and the removal of 99.9% of Th and U isotopes in separation, the normalized radiotoxicity of the Th/U-233 system decreases sharply. The radiotoxicity of the standard UOX system drops below natural uranium level in about 200,000 years, while the radiotoxicity of the Th/U-233 system drops below natural uranium level in about 10,000 years. This trend for the Th/U-233 system versus the UOX system (along with that of the decay heat in Figure 5.8) is not dissimilar from that observed for fuel cycle systems in which the fuel is continuously recycled, irrespective of the fuel type or neutron spectrum. In such systems, it is the degree of fuel separation that proves limiting to performance improvements that can be made.

It can be concluded from the radiotoxicity viewpoint that the Th/U-233 system has similar performance compared to the standard UOX system in the short term after separation (which is 5 years after discharge) when fission product contributions dominate. However, in the mid and long term (beyond 50 years after discharge), the Th/U-233 system has much lower radiotoxicity levels due to the recycle of 99.9% of the Th and U isotopes.

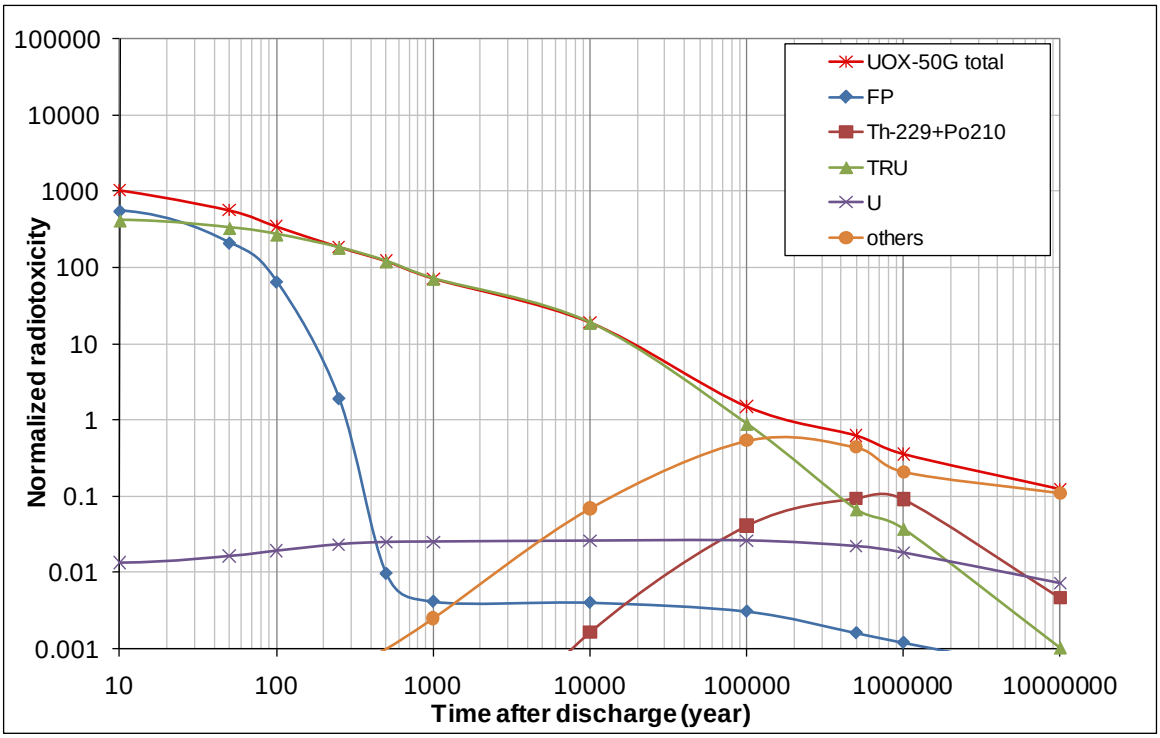


Figure 5.3. Normalized Radiotoxicity for Standard UOX Fuel Cycle (50 GWd/t Discharge Burnup).

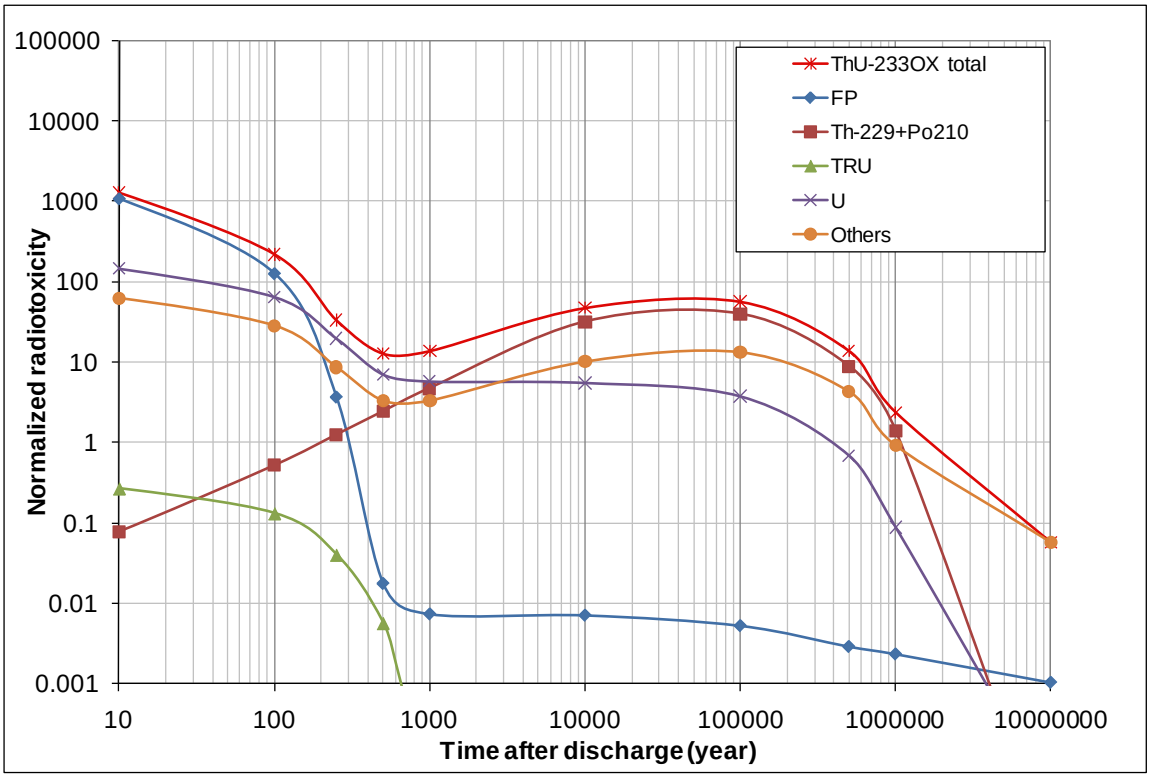


Figure 5.4. Normalized Radiotoxicity for 17-by-17 Heterogeneous Assembly Design (18 GWd/t Discharge Burnup).

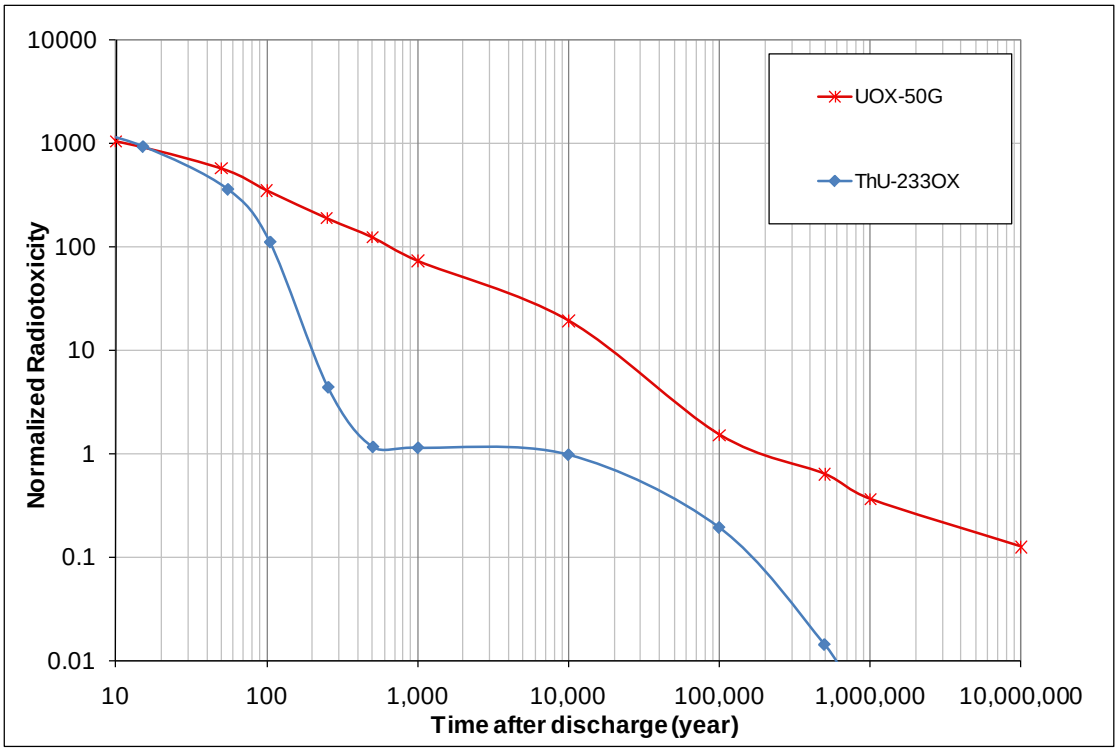


Figure 5.5. Comparison of Normalized Radiotoxicity Values.

Table 5.4. Comparison of Leading Contributors to Radiotoxicity (Normalized Value)

UOX-50G System				Th/U-233 System			
5 years after discharge		1,000 years after discharge		5 years after discharge		1,005 years after discharge	
Total	1.28E+03	Total	7.21E+01	Total	1.44E+03	Total	1.13E+00
Sr-90	28.5%	Am-241	50.7%	Sr-90	51.0%	Ac-227	56.5%
Cs-137	19.7%	Pu-240	29.6%	Cs-137	24.6%	Pa-231	36.5%
Pu-238	13.5%	Pu-239	17.9%	Cs-134	15.7%	Ra-223	5.1%
Cs-134	11.4%	Am-243	1.4%	Y-90	4.9%	Th-227	0.4%
Pu-241	7.5%	Pu-242	0.2%	Ce-144	1.6%	U-233	0.4%
Cm-244	7.3%	Pu-238	0.1%	Ra-224	1.1%	Th-229	0.4%

For completeness, Table 5.4 provides a comparison of the leading contributors to the total normalized radiotoxicity levels of the UNF of the standard UOX system and the Th/U-233 system. The total normalized radiotoxicity and percentage contributions of six leading contributors are reported for 5 years and 1,000 years after discharge, respectively (for Th/U-233, it is 1005 years after discharge).

At about 1,000 years after discharge, the normalized radiotoxicity of the Th/U-233 system is much smaller than that of the standard UOX system. As contribution from the fission products decreases with decay, the leading contributors to the radiotoxicity for the Th/U-233 system become Th, U and the decay daughters of U isotopes. Therefore, the removal of 99.9% of Th and U isotopes at separation caused the value of radiotoxicity of the Th/U-233 system to be smaller.

5.3.3 Decay Heat

The decay heat of both the standard UOX system and the Th/U-233 system have been assessed. Figure 5.6 shows the decay heat of the standard UOX system (50 GWd/t discharge burnup) broken down by the leading contributors to the decay heat. The decay heat is normalized to unit electricity generated in one year (W/GWe-yr). As for the UNF decay heat, it can be observed that from 10 years after discharge to about 100 years after discharge the FP and TRU are the main contributors to the total decay heat. The FP contribution to the decay heat decreases sharply between 100 and 500 years due to their natural decay. Contribution from TRU remains the primary contributor to the decay heat up to about 200,000 years. It can also be observed that contribution from uranium isotopes is consistently low throughout the time range.

Figure 5.7 shows the normalized decay heat for the Th/U-233 system, again broken down to indicate the leading contributors. This figure includes all the nuclear elements in the UNF, without consideration for the elements removed during fuel separations (provided in this form for illustrative purposes). From 10 years after discharge to about 100 years after discharge, FP is the main contributor to the total decay heat. The contribution of decay heat from FP decreases sharply between 100 and 300 years due to their natural decay. Contribution from the uranium isotopes decreases and U-233 decays gradually into Th-229, which becomes the main contributor from 1000 years to about a million years.

The TRU, different from the earlier scenario of standard UOX fuel cycle, contributes very small amount to the total decay heat. This is because in the Th/U-233 cycle, very small amount of TRU are produced. It is noted that other isotopes are main contributors of the decay heat as well for the whole time range covered. The contribution is mainly from decay daughters of Th and U isotopes, for instance, Ac-225, which is a beta decay daughter of Ra-225 (beta decay half life: 14.9 days), the alpha decay daughter of Th-229 (alpha decay half life: 7340 years).

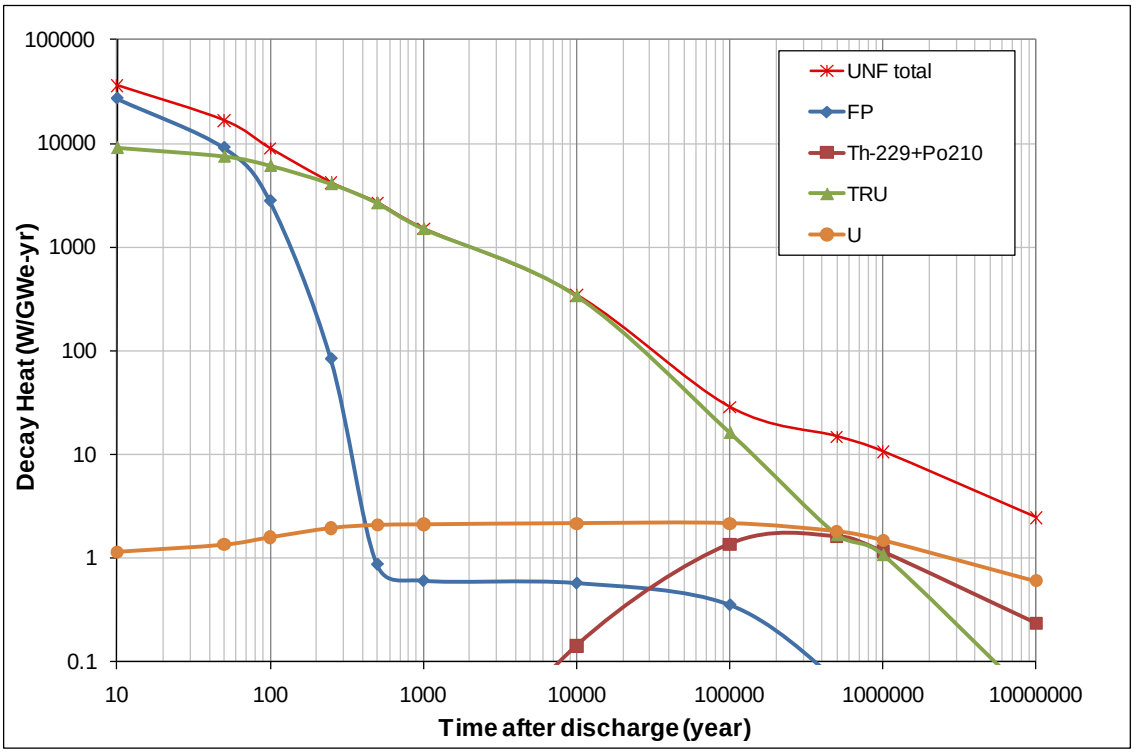


Figure 5.6. Decay Heat for Standard UOX Fuel Cycle (50 GWd/t Discharge Burnup).

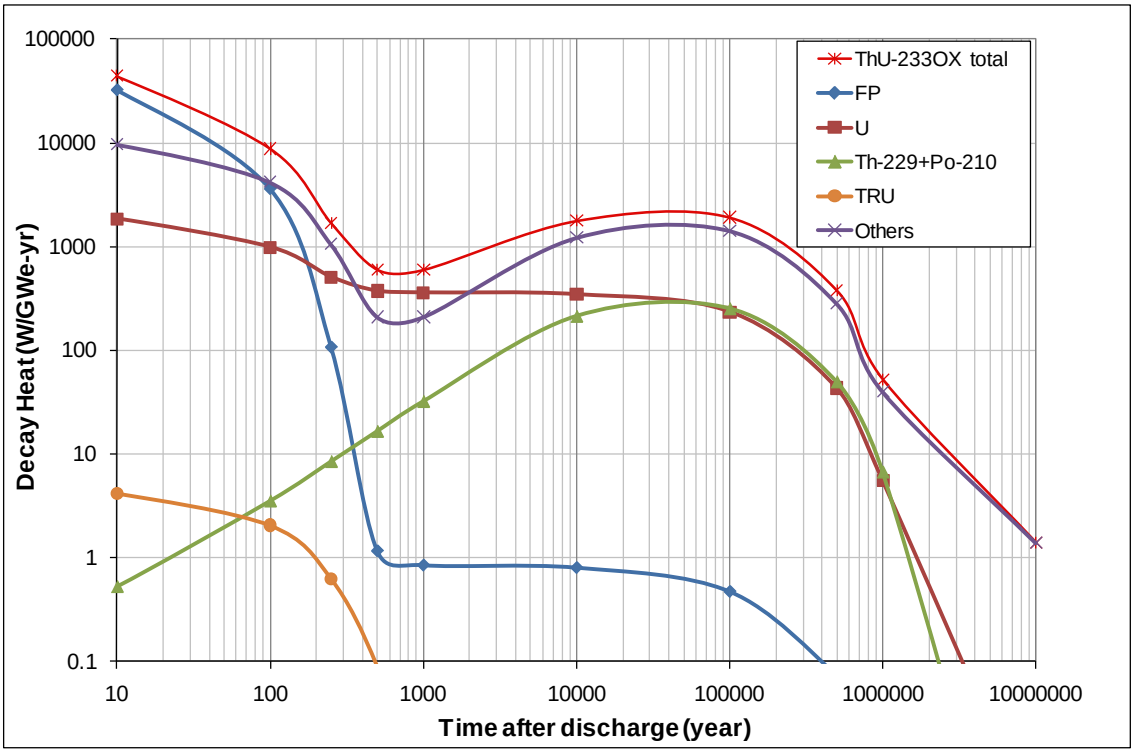


Figure 5.7. Decay Heat for 17-by-17 Heterogeneous Assembly Design (18 GWd/t Discharge Burnup).

Figure 5.8 presents the calculation result for normalized HLW decay heat of the Th/U-233 system and compares it to the standard UOX fuel cycle data as well. The time scale is in the unit of years after discharge and it is assumed that the separation of the Th/U-233 system UNF is done 5 years after discharge. For the UOX system, the UNF is assumed sent to the repository as HLW. In the Th/U-233 system, only the nuclear elements not recycled are assumed to be sent as HLW to the repository.

Similar to the trend of radiotoxicity, it is observed that compared to the standard UOX system the HLW decay heat of the Th/U-233 system is higher following UNF separations, because the lower burnup of the system results in more fission products per unit energy generation in one year. After approximately 50 years, the decay heat of the Th/U-233 system decreases sharply due to natural decay of the fission products, which are the main decay heat contributors in the short term. The trend after 50 years is due to the recycle of 99.9% of the Th and U isotopes and the low production of transuranic elements in the Th/U-233 system.

Table 5.5 provides comparison of the leading contributors to the HLW decay heat values for the standard UOX system and the Th/U-233 system. The decay heat and percentage contributions of six leading contributors are reported at fuel separation (i.e., 5 years) and 1,000 years after discharge respectively. The units of the reported decay heat are W/GWe-yr.

At about 1,000 years after discharge, the decay heat of the Th/U-233 system is much smaller than that of the standard UOX system. After the fission products decay out, the contributors to the decay heat for the Th/U-233 system become the decay daughters of U isotopes. Therefore, the removal of 99.9% of Th and U isotopes at separation caused the value of decay heat of the Th/U-233 system to be significantly smaller.

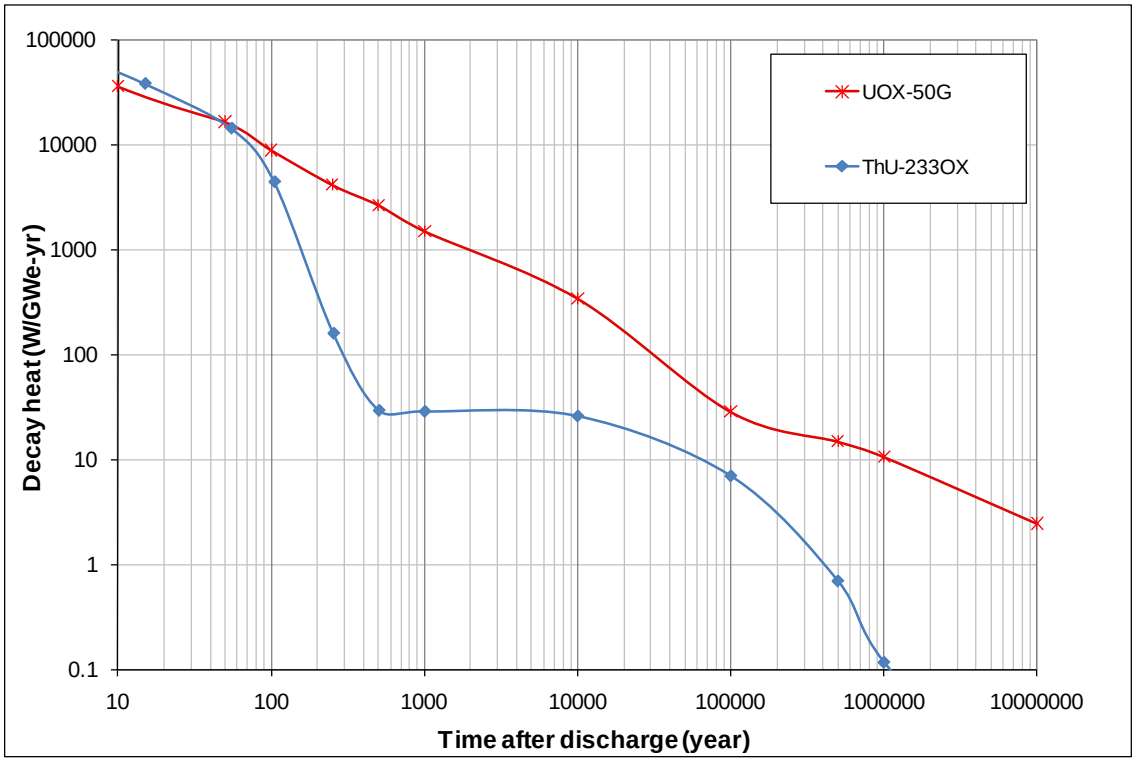


Figure 5.8. Comparison of Decay Heat Values.

Table 5.5. Comparison of Leading Contributors to Decay Heat (W/GWe-yr)

UOX-50G System				Th/U-233 System			
5 years after discharge		1,000 years after discharge		5 years after discharge		1,005 years after discharge	
Total	5.69E+04	Total	1.51E+03	Total	7.39E+04	Total	2.87E+01
Cs-134	20%	Am-241	58%	Y-90	30%	Po-215	18%
Y-90	18%	Pu-240	25%	Ba-137m	20%	Rn-219	17%
Ba-137m	18%	Pu-239	15%	Cs-134	16%	Bi-211	16%
Rh-106	8%	Am-243	2%	Sr-90	6%	Th-227	14%
Cm-244	7%	Pu-242	0.13%	Cs-137	6%	Ra-223	14%
Pu-238	6%	U-234	0.12%	Pr-144	5%	Pa-231	12%

5.3.4 Neutron and Photon Sources

In order to provide a measure of the UNF handling difficulty, the neutron and photon sources have been evaluated. The neutron source data combines (α,n) and spontaneous neutrons from the UNF.

Figure 5.9 is a comparison of the neutron source strength on a neutron emitted per second per UNF heavy metal mass basis between the standard UOX system (50 GWd/t burnup) and the Th/U-233 system. The data is post-irradiation and pre-fuel separations. Pre-fuel separations cooling time up to 100 years has been considered in generating this figure. It is observed that the total neutron source strength of the standard UOX system at discharge state is about 2 orders of magnitude higher than that of the Th/U-233

system. This, according to data in Table 5.6, is due to the much larger quantity of TRU isotopes produced by the UOX fuel cycle. The table shows that the TRU isotopes are the major contributors to the neutron source strength for the standard UOX system at discharge. Curium isotopes contribution to the neutron source dominates. As TRU isotopes are strong neutron emitters, the Th/U-233 system has the advantages of reduced total neutron source strength compared to the standard UOX system. It should be noted that the total neutron source of the Th/U-233 system is a weighted average of the seed and blanket unit according to their heavy metal weight ratio in the core. The comparison indicates that the Th/U-233 system might require less neutron shielding during fuel handling after discharge compared to the standard UOX system.

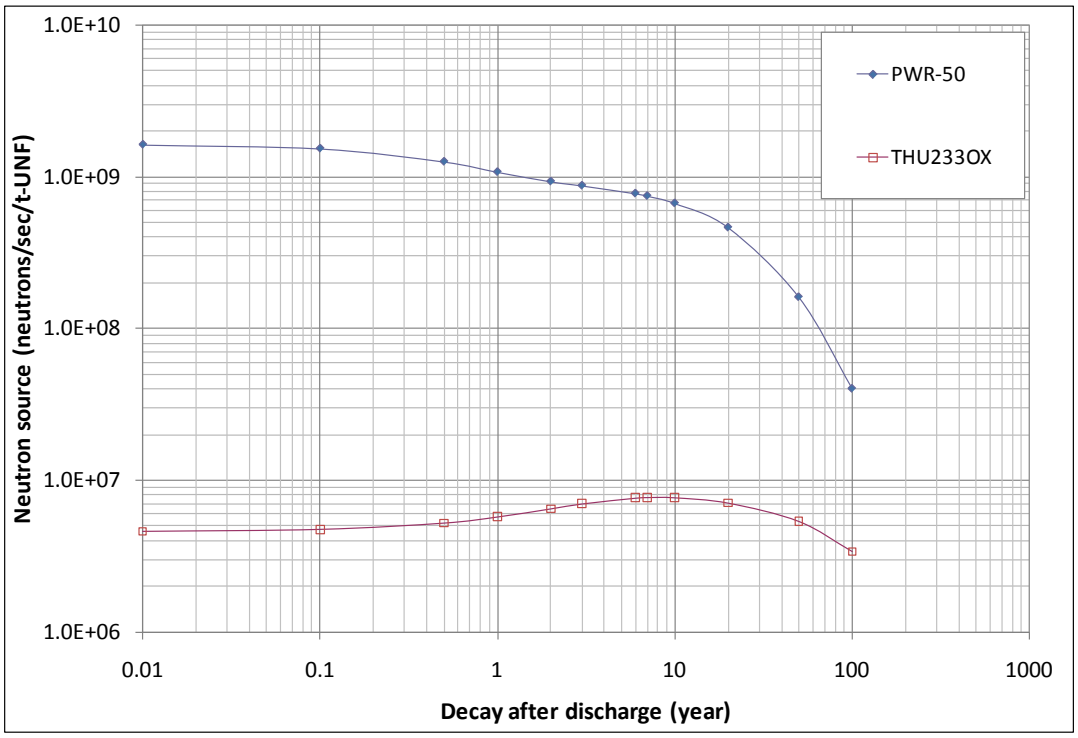


Figure 5.9. Comparison of Neutron Source Values.

Table 5.6. Comparison of Leading Contributors to Neutron Source at Discharge

UOX-50G (n/s/t-UNF)		Th/U-233 System (n/s/t-UNF)	
Total	1.63E+09	Total	4.61E+06
Cm-244	9.51E+08	Po-212	8.62E+05
Cm-242	6.57E+08	Po-216	8.28E+05
Cm-246	6.39E+06	U-232	7.90E+05
Pu-238	5.65E+06	Rn-220	6.77E+05
Pu-240	3.18E+06	Ra-224	4.96E+05
Cf-252	1.52E+06	Th-228	4.18E+05
Pu-242	1.48E+06	Bi-212	4.07E+05
Pu-239	2.76E+05	U-233	1.21E+05
Am-241	1.97E+05	U-234	7.47E+03

Figure 5.10 compares the total photon energy between the UNF of the two systems. The data is for pre-fuel separations cooling time up to 100 years. It can be seen that after discharge, the total photon energy of the Th/U-233 system is lower than that of the UOX system, due primarily to the higher specific power of the latter system. At the discharge, the short-lived fission products are dominant contributors and the photon source strength is proportional to the neutron flux level (or power density) during the irradiation in the core. At 50 years after discharge, Ba137m ($t_{1/2}$ = 2.6 min) is the dominant nuclide (contributes more than 80% of the total photon energy), which is the daughter of Cs-137 ($t_{1/2}$ = 30.1 year). The ORIGEN-2 results indicate that the UOX system produces more Cs-137 (about factor of 2) than the Th/U-233 system at 50 years after discharge. Table 5.7 compares the leading photon sources in the UNF of the two systems at discharge. It is noted that the total photon source of the Th/U233 system is also a weighted average of the seed and blanket unit according to their heavy metal weight ratio in the core. From a total photon source strength viewpoint, the Th/U-233 system can be seen to raise less concern for fuel handling after discharge compared to the standard UOX system.

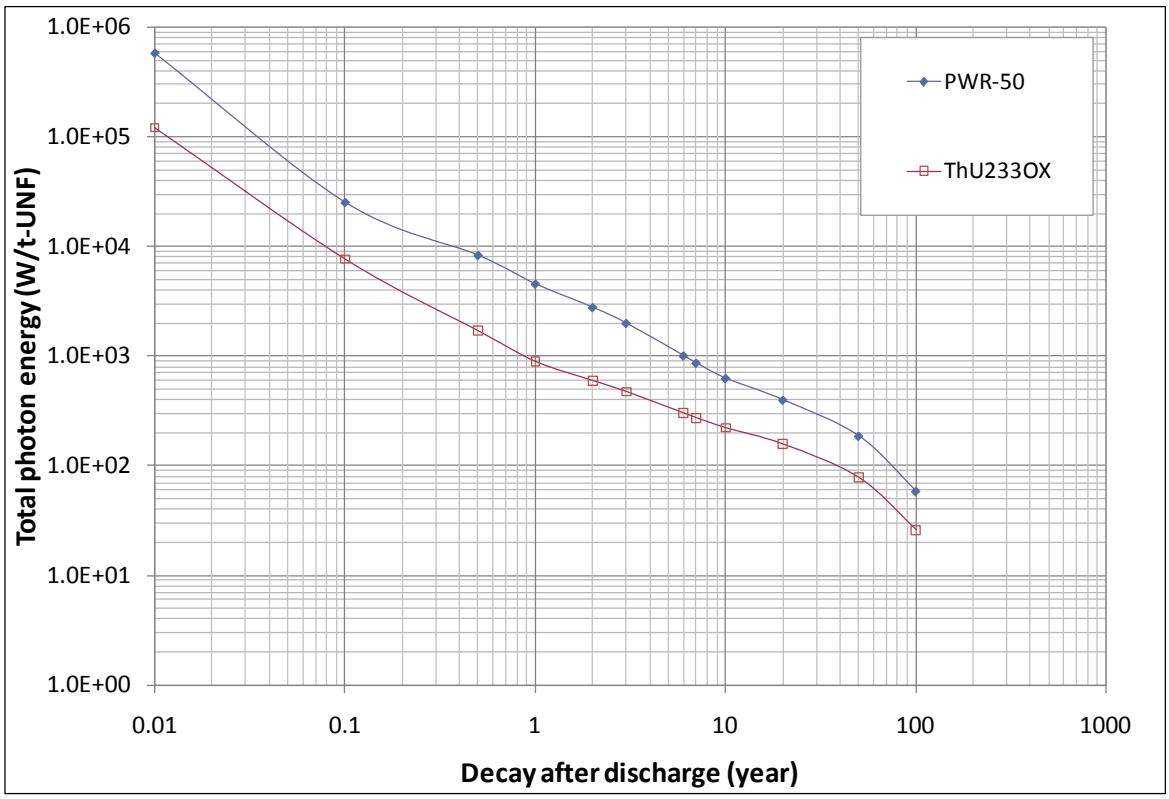


Figure 5.10. Comparison of Total Photon Energy Values.

Figure 5.11 shows the comparison between photon energy spectra for the UOX and Th/U-233 systems. It can be seen that the Th/U-233 system produces smaller population of photon on a per second per mass basis at the whole energy range compared to the standard UOX system (50 GWd/t discharge burnup). The mean photon energy for the Th/U-233 system is 0.381 MeV; this is an average weighted according to the heavy metal weight ratio of the seed and blanket regions in the core. The mean photon energy of the standard UOX system is 0.334 MeV, lower than that of the Th/U-233 system.

Table 5.7. Comparison of Leading Contributors to Photon Source (W/t-UNF) at Discharge

UOX-50G (W/t-UNF)		Th/U233 System (W/t-UNF)	
Total	5.82E+05	Total	1.21E+05
I-134	3.32E+04	La-142	9.30E+03
Cs-138	2.47E+04	La-140	8.25E+03
La-140	2.37E+04	I-134	8.15E+03
La-142	2.34E+04	Cs-138	7.09E+03
Np-239	2.23E+04	Pa-233	7.03E+03
Tc-104	1.87E+04	Rb-89	6.60E+03
I-132	1.84E+04	Rb-90	5.94E+03
I-135	1.53E+04	Kr-89	5.74E+03
Mo-101	1.36E+04	I-132	3.48E+03
I-136	1.11E+04	Kr-88	3.00E+03

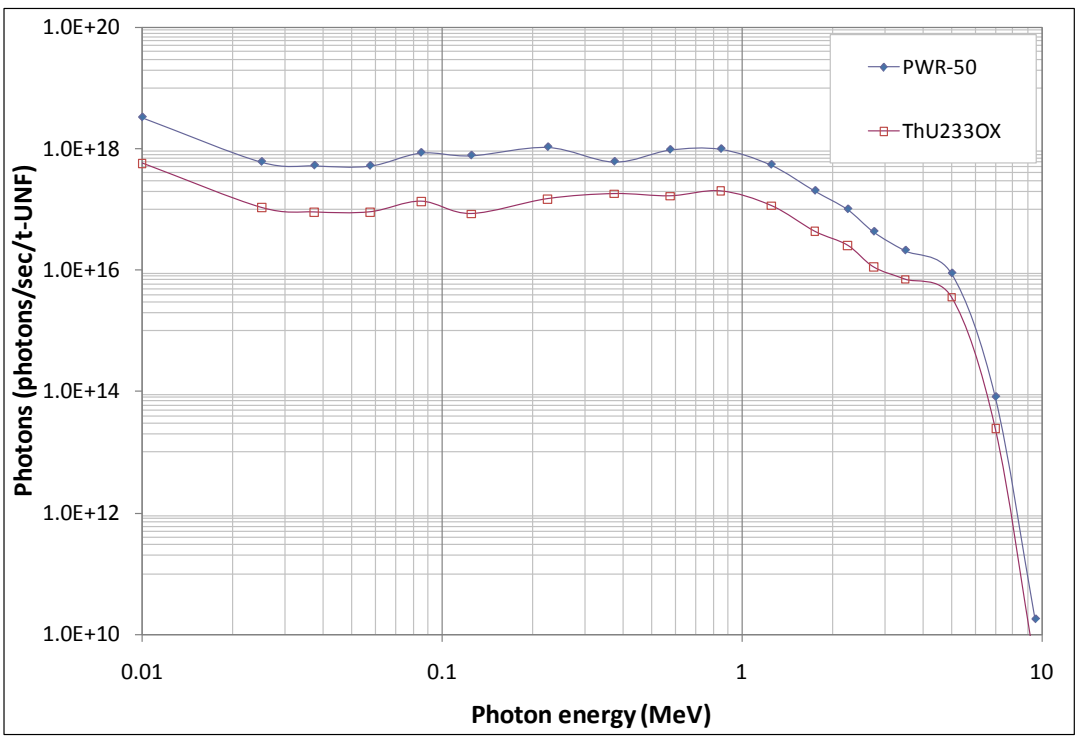


Figure 5.11. Comparison of Photon Energy Spectra at Discharge.

5.4 Comparison of Fuel Cycle Performance Data

A summary of fuel cycle performance data is provided in Table 5.8, which contains data for both the UOX and Th/U-233 systems. This data should provide indication of the fuel cycle impact of the Th/U-233 fuel with 17-by-17 heterogeneous assembly design in PWRs. Again, it is clear that the Th/U-233 systems produce much less quantity of TRU than the UOX system. Compared to the standard

UOX system, the Th/U-233 system has lower discharge burnup (18 GWd/MT versus 50 GWd/MT). One unique advantage of course is that the Th/U-233 systems are running on natural thorium resources only and requires no natural uranium resources. However, the lower burnup of the Th/U-233 systems also causes more heavy metal to be discharged for the same amount of electricity generation as the standard UOX system. A higher burnup would reduce the amount of material needing to be separated in the system.

Table 5.8. Fuel Cycle Impact for Th/U-233 Fuel with 17-by-17 Heterogeneous Design

Fuel Cycles	PWR-50	Th/U-233
Description		
Fuel	UOX	$(Th+U233)O_2$
Mining/Milling	Yes	Yes
Enrichment	Yes	Yes
Reactor Power, MW-thermal	3000	1000
Reactor Power, MW-electric	1000	333
Specific Power, W/g	33.7	6.73
Discharge Burnup, GWd/MT	50	18.0
Core Power Density, MW/m^3	91.88	30.63
Fuel Residence Time, years	4.52	8.13
Reactor Lifetime, years	60	60
Reactor Capacity Factor	0.9	0.9
Reactor	PWR	PWR
Used Fuel Processing	No	Yes
Used Fuel Disposal	Yes	No
Reactor Core Heavy Metal Data (per batch)		
Charge Uranium, MT	29.68	1.10
Discharge Uranium, MT	27.74	1.21 [a]
Charge Thorium, MT	-	48.45
Discharge Thorium, MT	-	47.42
Charge Fissile Fraction, %	4.21	2.23
Discharge Fissile Fraction, %	1.61	2.3
Performance per GWe		
Natural uranium MT/yr	166.07	0.0
Natural thorium MT/yr	0	1.11
Uranium enrichment/content	4.21% LEU	2.23% U233
LEU MTIHM/yr	19.71	0.0
Net TRU production MT/yr	0.26	6E-5
TRU to Waste MT/yr	0.26	6E-5
UNF Mass to Disposal MTIHM/yr	19.71	0.0
UNF Volume m^3/yr	-	-
HLW Mass to Disposal MT/yr	19.71	1.11 (FP of 1.05,Th/U loss of 0.06, and TRU)
DepU Mass to Storage MT/yr	146.36	0.00

[a]Total uranium mass including Pa-233.

6. FUEL CYCLE IMPACTS OF THORIUM AND U-233 UTILIZATION

6.1 Resources

Some estimates have suggested that thorium (Th) is between three to five times more abundant than uranium in the earth's crust and accessible via known mining techniques that might be simple and cost effective [IAEA 2005; Thorium-wiki]. Consequently Th has been considered as a fuel for supporting or replacing the uranium that is commonly used for commercial nuclear fuel. One country that appears to be moving ahead with thorium utilization is India that has started testing/demonstrating the use of thorium in the fuel cycle.

Thorium was initially considered as an alternative fuel for nuclear systems in the early days of the nuclear industry when it was thought that uranium resource was limited. Significant progress was made in the U.S. on the use of Th in those days, even cumulating in a LWR breeder demonstration program at the Shippingport Plant in the late 1970s to early 1980. As more uranium was mined and the uranium industry matured, the concern for uranium resources declined and less attention was given to the use of thorium and consequently thorium has not been exploited commercially.

While knowledge on the distribution of Th resources is quite poor due to low exploration of the material, recent IAEA, OECD and U.S. Geological Survey estimates suggest that Australia, Brazil and India possess large reserves of thorium [Thorium-wiki]. It is noted that the sources show large variations in reasonable assured reserved estimates for countries, particularly for Australia and Brazil.

Such estimates have also shown that the U.S. has a significant quantity of the material, ranking as high as third or fourth in the estimated reserve lists. In 2008, Senator Harry Reid (D-Nevada) and Senator Orrin Hatch (R-Utah) introduced the Thorium Energy Independence and Security Act of 2008, which would mandate a U.S. Department of Energy initiative to examine the commercial use of thorium in US reactors. The bill, however, did not reach a full Senate vote [Thorium-wiki].

6.2 Physics

There are two primary nuclear fuel resources in the earth crust: uranium and thorium. While uranium has a fissile component, U-235, thorium does not. In fact, the threshold neutron energy for fission in thorium in typical nuclear system is quite high and the probability for fission interaction even at those high energies is quite low. Consequently, a thorium fueled system needs an external source of fissile material to be useful in nuclear systems. Despite this fact, the Th-232 and U-233 fertile and fissile combination is attractive for LWRs. The U-233 can be produced from the absorption of a neutron in Th-232, the only isotope in natural thorium. The absorption cross section of Th-232 is nearly 3 times that of U-238 in a thermal spectrum; hence thorium provides a higher conversion to fissile material following neutron absorption. It is however inferior to U-238 in the fast spectrum. On the other hand, U-233 has an average number of neutron produced per neutron absorbed (eta) value greater than 2.0 in the thermal energy range, better than U-235 and Pu-239 over that range. Also in LWRs, higher fuel enrichment is required for Th-fueled core; this might be offset/ balanced over the irradiation cycle by the higher eta of U-233 compared to U-235.

6.3 Previous Thorium Utilization

Several experimental and prototype power reactors have been successfully operated in the 1950s to the 1980s using thorium fuel in high temperature reactors (HTRs), LWRs, and molten salt reactors (MSRs). Fuel forms considered have however been different for these systems [IAEA 2005]. For example, $(Th,U)O_2$ and $(Th,U)C_2$ fuels have been considered in high temperature reactors (HTR,) $(Th,U)O_2$ in LWRs, and $Li_7/BeF_2/ThF_4$ (Flibe) fuel in molten salt reactor (MSBR).

Thorium was successfully used in the U.S. LWRs such as BORAX (BWR), Elk River in the 1960s and the 1st core of Indian Point PWR. These LWRs used mixed enriched uranium-Th fuel. Uranium-233 and thorium fuel was used in the seed-blanket concept for thermal breeding at the Shippingport Atomic Power Station (1957 to 1982).

The thorium utilization experiments, demonstration, and commercial application in the HTRs were done in the U.S. and Europe in the 1950s to 1980s [IAEA 2005].

Thorium was also used in the molten-salt reactor experiment (MSR) from 1964 to 1969. The past MSR efforts centered at the Oak Ridge National Laboratory in the 1960s. An experimental reactor was built based on the technology to study the feasibility of such an approach. The effort culminated in the molten salt reactor experiment that used Th-232 as the fertile material and U-233 as the fissile fuel. This reactor was operated successfully for about five years and the operation was discontinued in 1976. The molten salt system is now being considered under the Generation IV nuclear systems program.

India is strongly considering the use of thorium for a sustainable energy system across three sets of reactor systems: pressurized heavy water reactors, fast reactors, and advanced heavy water reactors (AHWRs). The intent is to breed fissile U-233 and then use the material in a sustainable nuclear infrastructure. Thorium has been used in the blanket zone of the India FBTR system and the country is developing the 200 MWe AHWR-300 [IAEA 2005]. Other countries are also considering similar symbiotic nuclear system (e.g., Russia) [OECD 2009].

Thorium utilization in fast reactors has also been considered internationally. Studies comparing the breeding efficiencies of systems indicate that breeding is more favorable in the fast neutron spectrum than in the thermal spectrum [IAEA 2005].

6.4 Current Utilization Considerations in the U.S.

A U.S. company, the Lightbridge Company (McLean, VA), has been working to develop thorium-derived nuclear power. Lightbridge is considering using existing technology with the seed-blanket concept. The seed will be located in the assembly center and would be made of U-Zr metallic fuel. The blankets will contain Th-U oxide and will be in the assembly outer region. The company is scheduled to insert three test fuel rod assemblies into commercial power LWRs in Russia in the 2012-13 timeframe [Jacoby 2009]. The perceived non-proliferation advantage of a once-through thorium fuel cycle has also been a reason for Lightbridge's interest. It is being claimed by some of the advocates that at no point in the thorium fuel cycle can fuel or waste products be converted into nuclear bomb materials [Jacoby 2009]. This claim needs to be tempered since U-233 is a fissile material. The potential for reducing the radiotoxicity of spent fuel is another reason, as the use of thorium would greatly reduce the transuranic elements that are considered problematic for nuclear repositories. In this regard, it is noted that the daughter products from Th/U-233 irradiation have radioactivity (radiotoxicity) in the long entombment time-scale that have to be well characterized.

Thorium Energy Alliance, a small U.S.-based advocacy group, is promoting the use of thorium for power production [Jacoby 2009]. This group is advocating the use of the molten salt reactor design due to perceived safety advantages in reactor operation at low pressure and in density decrease with temperature which reduces core reactivity. Another perceived advantage by the group is the reduced proliferation risk with Th/U utilization in the MSR, as a result of the hard gamma rays from the associated U-232 and its decay daughters.

It is recognized by the groups that enormous investment of time, effort, and money is required before any new type of nuclear reactor could be licensed.

6.5 Benefits and Challenges of Th Utilization in the Fuel Cycle

6.5.1 Benefits

A detailed evaluation of Th fuel cycle is an ongoing activity at the IAEA. A 2005 IAEA report [IAEA 2005] has enumerated the potential benefits of thorium utilization. These benefits include (culled from the IAEA report):

- Thorium is 3 to 4 times more abundant than uranium, widely distributed in nature as an easily exploitable resource in many countries and has not been exploited commercially so far. Thorium fuels, therefore, complement uranium fuels and ensure long term sustainability of nuclear power.
- Thorium fuel cycle is an attractive way to produce long term nuclear energy with low radiotoxicity waste. In addition, the transition to thorium could be done through the incineration of weapons grade plutonium (WPu) or civilian plutonium.
- The absorption cross-section for thermal neutrons of Th-232 (7.4 barns) is nearly three times that of U-238 (2.7 barns). Hence, a higher conversion (to U-233) is possible with Th-232 than with U-238 (to Pu-239). Thus, thorium is a better 'fertile' material than U-238 in thermal reactors but thorium is inferior to depleted uranium as a 'fertile' material in fast reactor.
- For the 'fissile' U-233 nuclei, the number of neutrons liberated per neutron absorbed (represented as η) is greater than 2.0 over a wide range of thermal neutron spectrum, unlike U-235 and Pu-239. Thus, contrary to U-238/Pu-239 cycle in which breeding can be obtained only with fast neutron spectra, the Th-232/U-233 fuel cycle can operate with fast, epithermal or thermal spectra.
- Thorium dioxide is chemically more stable and has higher radiation resistance than uranium dioxide. The fission product release rate for ThO_2–based fuels is one order of magnitude lower than that of UO_2. ThO_2 has favorable thermophysical properties because of the higher thermal conductivity and lower coefficient of thermal expansion compared to UO_2. Thus, ThO_2–based fuels are expected to have better in–pile performance than that of UO_2 and UO_2–based mixed oxide.
- ThO_2 is relatively inert and does not oxidize, unlike UO_2 which oxidizes easily to U_3O_8 and UO_3. Hence, long term interim storage and permanent disposal in repository of spent ThO_2–based fuel are simpler without the problem of oxidation.
- Thorium based fuels and fuel cycles have intrinsic proliferation-resistance due to the formation of U-232 via (n,2n) reactions with Th-232, Pa-233 and U-233. The half-life of U-232 is only 73.6 years and the daughter products have very short half-life and some like Bi-212 and Tl-208 emit strong gamma radiations. From the same consideration, U-232 could be utilized as an attractive carrier of highly enriched uranium (HEU) and weapons grade plutonium (WPu) to avoid their proliferation for non-peaceful purpose.
- For incineration of WPu or civilian Pu in 'once-through' cycle, (Th, Pu)O_2 fuel is more attractive, as compared to (U, Pu)O_2, since plutonium is not bred in the former and the U-232 formed after the once-through cycle in the spent fuel ensures proliferation resistance.
- In Th-232/U-233 fuel cycle, much lesser quantity of plutonium and long-lived Minor Actinides (MA: Np, Am and Cm) are formed as compared to the U-238/Pu-239 fuel cycle, thereby minimizing the radiotoxicity associated in spent fuel. However, in the back end of Th-232/U-233 fuel cycle, there are other radionuclides such as Pa-231, Th-229 and U-230, which may have long term radiological impact.

6.5.2 Challenges

The challenges associated with Th fuel utilization in the fuel cycle were also discussed by the IAEA document [IAEA 2005]. The challenges include:

- The melting point of ThO_2 (3,350°C) is much higher compared to that of UO_2 (2,800°C). Hence, a much higher sintering temperature (>2,000°C) is required to produce high density ThO_2 and ThO_2–based mixed oxide fuels. Admixing of 'sintering aid' (CaO, MgO, Nb_2O_5, etc) is required for achieving the desired pellet density at lower temperature.
- ThO_2 and ThO_2–based mixed oxide fuels are relatively inert and, unlike UO_2 and (U, Pu)O_2 fuels, do not dissolve easily in concentrated nitric acid. Addition of small quantities of HF in concentrated HNO_3 is essential, which cause corrosion of stainless steel equipment and pipings in reprocessing plants. The corrosion problem is mitigated with addition of aluminum nitrate. Boiling THOREX solution [13 M HNO_3+0.05 M HF+0.1 M $Al(NO_3)_3$] at ~393 K and long dissolution period are required for ThO_2–based fuels.
- The irradiated Th or Th–based fuels contain significant amount of U-232, which has a half-life of only 73.6 years and is associated with strong gamma emitting daughter products, Bi-212 and Tl-208 with very short half-life. As a result, there is significant buildup of radiation dose with storage of spent Th–based fuel or separated U-233, necessitating remote and automated reprocessing and re-fabrication in heavily shielded hot cells and increase in the cost of fuel cycle activities.
- In the conversion chain of Th-232 to U-233, Pa-233 is formed as an intermediate, which has a relatively longer half-life (~27 days) as compared to Np-239 (2.35 days) in the uranium fuel cycle thereby requiring longer cooling time of at least one year for completing the decay of Pa-233 to U-233. Normally, Pa is passed into the fission product waste in the THOREX process, which could have long term radiological impact. It is essential to separate Pa from the spent fuel solution prior to solvent extraction process for separation of U-233 and thorium.
- The three stream process of separation of uranium, plutonium and thorium from spent (Th, Pu)O2 fuel, though viable, is yet to be developed.
- The database and experience of thorium fuels and thorium fuel cycles are very limited, as compared to UO_2 and (U, Pu)O_2 fuels, and need to be augmented before large investments are made for commercial utilization of thorium fuels and fuel cycles.

6.6 Transitioning to Th/U-233 Fuel Cycle

Some considerations have been given to the overall systems requirements to enable the transition from the current LWR fuel cycle using uranium fuel to a sustainable Th/U-233 multi-recycle system. In order to transition to a Th/U-233 fuel cycle to support a nuclear infrastructure similar to that currently in the U.S. (about 100 GWe per year), a lot of U-233 has to be created. There are some legacy U-233 materials from the weapons program that could be considered for starting up some of the initial Th/U-233 cores. However the bulk of the U-233 would have to be created by the conversion of Th-232. One option is to have thorium blanket assemblies in the LWR core periphery and to produce U-233 for later use (pre-breeder). This approach should result in the least uranium enrichment penalty to the LWR core. Issues with the approach would include the need for time to produce sufficient U-233 to fuel the same core (or even 1/3 fractional core) and for development of the thorium assembly plant to support thorium utilization in the LWR fuel cycle. This could be very long, given the position of the Th blanket and the low neutron flux in such positions. Safety issues with power generation in the thorium blanket fuel must also be addressed. The production of U-233 (primarily) in total U (fissile content) is a proliferation concern even with the presence of U-232. One could consider de-naturing the blanket, but the goal of minimizing transuranic elements would be undermined.

Another approach for creating U-233 is to have Th mixed with enriched uranium fuel in the core. As thorium is a stronger absorber than uranium, a measurable enrichment penalty is expected for the initial state. The buildup of U-233 will mitigate this penalty with irradiation. Calculations by BNL [Raitses 2009] suggest that about the same fissile content might be required for this system and a comparable typical PWR. Even in this thorium core, the time for breeding sufficient U-233 to support subsequent

cores is important. Furthermore, assuming that Th is the base fuel (e.g. for resource extension) in a mixed ThO_2 and enriched UO_2 fuel, the uranium component must have a high enrichment. For example if 5% fissile/HM is required, then for a Th/U proportion of 75%/25%, the uranium component enrichment must be about 20%: the acceptable LEU limit. High uranium enrichment is problematic, because no commercial facility currently supports enrichment above 5%.

Just from physics considerations, the fissile fraction of the fuel (e.g. U-235 fraction) must be limited for fissile breeding, but must be sufficiently high for core criticality. This suggests that the uranium fraction has to be greater than 25%. Of course, alternatively, the enriched uranium could be the major component (with reasonable U-235 fraction), but this defeats the purpose of using thorium and would further reduce U-233 production.

If weapons-grade uranium (highly enriched uranium in U-235 or even U-233) is available for use, then reasonable proportion of uranium could be used (about 5%) in the HM. The same goes for weapons-grade plutonium. However, the fuel would have to be produced at a highly secured location, with appropriate safeguards applied. It would however be difficult for this kind of fuel cycle to support a nuclear growth as this growth implies providing significantly more fissile material.

The proliferation risk considerations with Th/U-233 fuel should not be minimized. To transition to the Th/U fuel cycle, fuel would have to be reprocessed. During fuels separations and fabrication, thorium and uranium (U-233 mostly) will likely be in different streams at some point. For reduced proliferation risk, the U-233 (and other uranium nuclides) could be co-extracted with some thorium and the remaining thorium could be in a different stream. The two streams could then be mixed in appropriate proportions.

7. CONCLUSIONS

This report contains current information that has been accumulated/developed for the study on sustainable Th/U-233 multi-recycle in PWRs. Investigations have been conducted to assess the feasibility of both homogeneous and heterogeneous PWR assemblies to achieve Th/U-233 fuel multi-recycle. The conclusions are summarized as the followings.

1. A practical sustainable fuel cycle cannot be achieved with a homogeneous PWR assembly within the parameter space of initial U-233 content and reasonable moderator to fuel volume ratio (MR) obtained by varying the fuel pin size.
2. A 17-by-17 heterogeneous assembly design achieved the following attributes and partially met the requirements defined in this study for sustainable Th/U-233 multi-recycle:
    a. A derated core power of 1000 MWt.
    b. A 2.7-year cycle length with sustainable fissile inventory.
    c. Discharge burnup of 18 GWd/t, which is significantly lower than the burnup of current PWRs.
    d. The Th/U-233 system requires hard neutron spectra to support high conversion of Th-232. The neutron spectra of both the seed and the blanket region are harder compared to the standard thermal systems.
    e. Compared to the UNF of the standard UOX system, the HLW of the Th/U-233 multi-recycle system has higher normalized radiotoxicity and decay heat at discharge (due to lower burnup), but those become lower 20 years and 50 years after discharge, respectively, due to the recycle of 99.9% of Th and U isotopes in the nuclear system and the low production of transuranic elements.
    f. Handling of the UNF of the Th/U-233 system should generate less concern than for the UOX system, because of the lower neutron source and photon source levels. The lower neutron source is due to the low production of TRU isotopes.
    g. Cycle by cycle assembly model has to be run in the future to ascertain if the same cycle length can be maintained in the succeeding recycles with the discharge fissile inventory derived from previous cycles.
3. A sensitivity study has been conducted, which suggests that:
    a. A full-core model is necessary to estimate the neutron leakage and evaluate the performance of core level heterogeneity.
    b. A detailed thermal-hydraulics study should be carried out in future studies on the 17-by-17 heterogeneous assembly design for Th/U-233 multi-recycle.
    c. Considerations should be given to adopting hydride blanket fuel forms in future studies.

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