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NUREG/CR-7176
ORNL/TM-2013/543

Safety and Regulatory Issues of the Thorium Fuel Cycle

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Manuscript Completed: December 2013
Date Published: February 2014

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ABSTRACT

Thorium has been widely considered an alternative to uranium fuel because of its relatively large natural abundance and its ability to breed fissile fuel (^{233}U) from natural thorium (^{232}Th). Possible scenarios for using thorium in the nuclear fuel cycle include use in different nuclear reactor types (light water, high temperature gas cooled, fast spectrum sodium, molten salt, etc.), advanced accelerator-driven systems, or even fission-fusion hybrid systems. The most likely near-term application of thorium in the United States is in currently operating light water reactors (LWRs). This use is primarily based on concepts that mix thorium with uranium ($UO_2 + ThO_2$), add fertile thorium (ThO_2) fuel pins to LWR fuel assemblies, or use mixed plutonium and thorium ($PuO_2 + ThO_2$) fuel assemblies.

The addition of thorium to currently operating LWRs would result in a number of different phenomenological impacts on the nuclear fuel. Thorium and its irradiation products have nuclear characteristics that are different from those of uranium. In addition, ThO_2, alone or mixed with UO_2 fuel, leads to different chemical and physical properties of the fuel. These aspects are key to reactor safety-related issues.

The primary objectives of this report are to summarize historical, current, and proposed uses of thorium in nuclear reactors; provide some important properties of thorium fuel; perform qualitative and quantitative evaluations of both in-reactor and out-of-reactor safety issues and requirements specific to a thorium-based fuel cycle for current LWR reactor designs; and identify key knowledge gaps and technical issues that need to be addressed for the licensing of thorium LWR fuel in the United States.

An evaluation of in-reactor safety issues was performed based on nuclear physics fundamentals and available experimental data and study results. The *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition* (NUREG-0800) has been reviewed to identify specific items that would be impacted by changing the fuel to a form that contains thorium. Quantitative analyses were performed using the SCALE code system to compare key performance parameters of both $(Th,U)O_2$ and $(Th,Pu)O_2$ fuels against UO_2 and MOX fuels in LWRs. The reactivity coefficients, assembly power (between surrounding UO_2 assemblies and the assembly of interest), and single-assembly controlled lattice reactivities are compared for beginning, middle, and end of life.

The SCALE fuel assembly models from the in-reactor analyses were also used in ORIGEN calculations for low, normal, and high discharge burnup values to evaluate out-of-reactor characteristics of spent thorium fuel. Calculations were performed for all four fuel types to compare the depleted fuel isotopics, decay heat, radiological source terms, and gamma spectra.

Based on these evaluations, potential impacts on safety requirements and identification of knowledge gaps with regard to once-through LWR thorium fuel cycles were identified. Recommendations for additional analysis or research to develop a technical basis for the licensing of thorium fuel are summarized in phenomena identification and ranking tables (PIRTs). The PIRTs provide an assessment of how the changes in the phenomena could be addressed (e.g., additional analysis, new data required, experimental validation, etc.).

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LIST OF ACRONYMS AND UNITS

10 CFR	Title 10 of the *Code of Federal Regulations*
AEC	U.S. Atomic Energy Commission
AHWR	advanced heavy water reactor
ALARA	as low as reasonable achievable
AOO	anticipated operating occurrence
ARE	Aircraft Reactor Experiment
ATBR	Advanced Thorium Breeder Reactor
ATWS	anticipated transient without scram
AWBA	advanced water breeder applications
B&W	Babcock and Wilcox, Inc.
BAPL	Bettis Atomic Power Laboratory
BDBA	beyond design basis accident
BISO	bistructural isotropic
BOL	beginning of life
BWR	boiling water reactor
CE	continuous energy
CHF	critical heat flux
CPR	critical power ratio
DBA	design basis accident
DMSR	denatured molten salt reactor
DNBR	departure from nucleate boiling ratio
DOE	U.S. Department of Energy
DRE	DRAGON Reactor Experiment
EBR	experimental breeder reactor
EDS	externally driven system
EFPD	effective full power day
EOL	end of life
FBTR	Fast Breeder Test Reactor
FSV	Fort Saint Vrain
GDC	general design criteria, from Appendix A of 10 CFR
GT-MHR	gas-turbine modular high-temperature reactor
GWd	gigawatt day
HEU	high-enriched uranium
HFP	hot full power
HTGR	high-temperature gas-cooled reactor
HWR	heavy water reactor
IAEA	International Atomic Energy Agency
IFBA	integral fuel burnable absorber
IHECSBE	*International Handbook of Evaluated Criticality Safety Benchmark Experiments*
INCFE	International Fuel Cycle Evaluation Conference
ISFSI	independent spent fuel storage installation
k_{eff}	effective multiplication factor
KI	Kurchatov Institute
LAR	lifetime-averaged reactivity
LCE	laboratory critical experiment
LEU	low-enriched uranium
LOCA	loss of coolant accident
LWBR	light water breeder reactor

LWR	light water reactor
MHTGR	modular high-temperature gas-cooled reactor
MOL	middle of life
MOX	mixed oxide
MSBR	molten salt breeder reactor
MSR	molten salt reactor
MSRE	Molten Salt Reactor Experiment
MTC	moderate temperature coefficient
MTHM	metric ton (1000 kg) of heavy metal (U, Th, Pu)
NBOL	near beginning of life
NEA	Nuclear Energy Agency
NGNP	next generation nuclear plant
NRC	U.S. Nuclear Regulatory Commission
NUREG	publication of the U.S. Nuclear Regulatory Commission
OECD	Organisation for Economic Co-operation and Development
ORNL	Oak Ridge National Laboratory
PCI	pellet-clad interaction
PCMI	pellet-clad mechanical interaction
PB	Peach Bottom
PBMR	Pebble Bed Modular Reactor
pcm	percent mille (1.0E-5)
PFBR	prototype fast breeder reactor
PHWR	pressurized heavy water reactor
PIE	post-irradiation examination
PIRT	phenomena identification ranking table
PRA	probabilistic risk assessment
PSC	Public Service Company of Colorado
PUREX	Plutonium Uranium EXtraction
PWR	pressurized water reactor
PyC	pyrocarbon
R&D	research and development
RG	regulatory guide
RTC	reactivity coefficient
RTPI	Radowsky Plutonium Thorium Incinerator
SRP	standard review plan
SSET	self-sustaining equilibrium thorium
THOREX	THORium EXtraction
TMSR	thorium molten salt reactor
TREAT	Transient Reactor Test Facility
TRISO	tristructural isotropic
TRU	transuranic
UOX	uranium oxide (UO_2)
WABA	wet annular burnable absorber

1 INTRODUCTION

Thorium (Th) has been widely considered an alternative to uranium (U) fuel because of its natural abundance and its ability to breed fissile fuel (^{233}U) from natural thorium (^{232}Th). Possible scenarios for using thorium in the nuclear fuel cycle include use in different nuclear reactor types (light water, high temperature gas cooled, fast spectrum sodium, molten salt, etc.), advanced accelerator-driven systems, or even fission-fusion hybrid systems. Different concepts also exist for recycling and reusing the fissile isotopes produced during irradiation of thorium fuel and the fertile isotopes that remain in the fuel.

The primary objectives of this report are to summarize historical, current, and proposed uses of thorium in nuclear reactors; provide some important properties of thorium fuel; perform qualitative evaluations of both in-reactor and out-of-reactor safety issues and requirements specific to a thorium-based fuel cycle for current light water reactor (LWR) designs; and identify key knowledge gaps and technical issues that need to be addressed for the licensing of thorium LWR fuel in the United States.

The most likely near-term application of thorium in the United States is in currently operating LWRs, which is the focus of this report. This use is primarily based on concepts that mix thorium with uranium (UO_2 + ThO_2) fuel pins that are then added to typical LWR fuel assemblies. In addition, Thor Energy and Westinghouse have considered potential plans for testing mixed plutonium and thorium (PuO_2 + ThO_2) lead assemblies [1]. In general for these near-term applications, it has been assumed that the fuel cladding and assembly design will remain identical to that of the currently operating LWRs.

The addition of thorium to currently operating LWRs would result in a number of different phenomenological impacts in the nuclear fuel. Thorium and its irradiation products have nuclear characteristics different from those for uranium. ThO_2 fuel, alone or mixed with UO_2 fuel, leads to different chemical and physical properties of the fuel. These aspects are key to reactor safety-related issues, because they impact in-core safety parameters (e.g., power peaking, control rod worths, reactivity coefficients, and critical boron concentrations). In addition, characteristics of the spent fuel are impacted for storage and transportation (e.g., depleted fuel isotopics, decay heat, and radiological source terms).

Section 2 of this report provides background information, including the motivation for using thorium fuel and potential thorium fuel cycle options. The historical uses of thorium fuel in nuclear reactors, both in the United States and internationally, are examined, and currently proposed thorium fuel applications are discussed as well.

The remainder of the report focuses on comparison of the potential impacts of thorium-based fuels versus UO_2 and mixed oxide (UO_2 + PuO_2, also referred to as MOX) fuels in current LWRs. Section 3 discusses key properties of thorium fuel and how they may impact fuel behavior in and out of the reactor.

Sections 4 and 5 examine in-reactor aspects of safety and regulatory issues arising from thorium fuels in a once-through LWR fuel cycle. Section 4 provides a qualitative evaluation of in-core reactor safety by reviewing key sections of the *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition* (NUREG-0800) [2]. The review identifies specific items that would be impacted by changing the fuel to a form that contains

thorium. Section 5 contains quantitative analyses of thorium-based fuels in a once-through LWR fuel cycle. Although there are no currently existing thorium fuel or core designs formally available for review, there are a number of option groups available in the open literature. To demonstrate the different phenomena for a range of thorium fuel types, analyses have been performed using SCALE [3] to calculate key performance parameters. The performances of both $(Th,U)O_2$ and $(Th,Pu)O_2$ are compared against UO_2 and mixed oxide (UO_2 + PuO_2, or MOX) fuels because thorium fuels are currently being studied in Europe and the United States for both resource utilization and plutonium management benefits.

Section 6 discusses analyses of spent thorium-based fuels after discharge from the reactor to estimate important out-of-reactor behaviors and trends. The results for depleted fuel isotopics, decay heat, and radiological source terms are evaluated to confirm and illustrate the magnitude of the impact that thorium fuels have on safety parameters of interest.

Section 7 summarizes potential impacts on safety requirements and identification of knowledge gaps that require additional analysis or research to develop a technical basis for the licensing of thorium. The text discusses key phenomena and various needs associated with those phenomena for licensing thorium-based fuel, the safety areas that are impacted (e.g., neutronic design, fuel performance, source terms, etc.), and the level of importance that these phenomena and differences may carry. These issues are summarized in phenomena identification and ranking tables (PIRTs), which provide a high-level assessment of how the changes in the phenomena could be addressed (e.g., additional analysis, new data required, experimental validation, etc.).

2 HISTORICAL, CURRENT, AND PROPOSED USES OF THORIUM-BASED FUELS

From the very early days of nuclear energy, thorium was considered as an alternative or complement to uranium. Extensive nuclear energy research and development (R&D) programs were under way in the United States in the late 1950s and in Europe (Germany and the United Kingdom) in the early to mid-1960s that considered both uranium and thorium fuels. Indeed, at the International Fuel Cycle Evaluation Conference (INFCE) [4] of 1978, thorium was given almost equal importance as uranium.

The main driver for considering thorium in the early days was resource utilization and addressing the potential shortage of uranium if predictions of large nuclear growth were to be realized. However, the expansion in nuclear energy never materialized and consideration to recycle and reuse the ^{233}U from the thorium fuel cycle or plutonium from the uranium fuel cycle diminished – and with it interest in the thorium fuel cycle also diminished. However, in more recent years, new drivers and considerations such as proliferation resistance and waste management have seen a resurgence of interest in thorium fuels and fuel cycles, as well as the potential to improve resource utilization as a second expansion of nuclear energy.

In the United States, several demonstrations of the use of thorium in nuclear reactors have been performed. Irradiations in LWRs included the Elk River boiling water reactor (BWR) and the Indian Point and Shippingport pressurized water reactors (PWRs). Shippingport also demonstrated the breeding potential of using thorium in an LWR as part of the Light Water Breeder Reactor (LWBR) program. The use of thorium was also widespread in high-temperature gas-cooled reactors (HTGRs) including Peach Bottom 1 and Fort St. Vrain. A testing program used ^{233}U in the Molten Salt Reactor Experiment (MSRE) at Oak Ridge National Laboratory (ORNL).

Worldwide, Germany and the United Kingdom used thorium in their high-temperature gas reactors (AVR and THTR in Germany, DRAGON in the U.K.). India has been developing thorium fuel cycle technology for several decades since they possess extensive reserves of thorium yet lack significant uranium resources. This is being done as part of India's three-stage fuel cycle program, which involves using heavy water reactors (HWRs), then fast reactors, and eventually advanced heavy water thorium reactors. The Indian nuclear establishment has also been pursuing the possibility of moving directly to thorium HWRs such as the Advanced Heavy Water Reactor (AHWR). Canada has also considered the use of thorium in CANDU reactors.

Nevertheless, although significant experience has been gained with thorium-based fuels since the 1950s in test and demonstration reactors, there is no industrial-scale experience. Almost all of the world's nuclear reactors today rely on uranium fuels, and the rest rely on the uranium-plutonium fuel cycle in the form of MOX fuels. A number of countries are considering the use of thorium, with the most significant national program under way in India. A few organizations have been formed to pursue commercial interest in the use of thorium, including Lightbridge LLC in the United States and Thor Energy in Norway. These companies have demonstrated an interest in developing thorium fuel that can be substituted for uranium fuel in current operating LWRs.

2.1 MOTIVATION FOR USE OF THORIUM-BASED FUELS

The drivers and motivation for considering thorium include the following: resource availability, sustainability in thermal reactors, radiotoxicity of spent nuclear fuel, and proliferation resistance.

2.1.1 Resource Availability

Thorium is three to four times more abundant than uranium in the earth's crust (average abundance of thorium in the earth's crust is 9.6 ppm compared to uranium at 2.7 ppm) [5]. However, the concentration of these deposits needs to be taken into account when evaluating how economic it is to mine thorium. A recent report [6] states that there is a world total of 5.4 million metric tons of thorium (Table 2.1) for reasonably assured and inferred resources (recoverable at a cost of $80/kg), which is comparable with the amount of known economically recoverable uranium; however, this is an estimate based on uranium and rare earth resources, because there is no international standard classification for thorium resources and thorium is not currently a primary exploration target. Caution should therefore be used when using resource estimates for thorium.

Table 2.1. Thorium reserves by country [6]

Country	Tons	% of World Total
India	846,000	16
Turkey	744,000	14
Brazil	606,000	11
Australia	521,000	10
USA	434,000	8
Egypt	380,000	7
Norway	320,000	6
Venezuela	300,000	6
Canada	172,000	3
Russia	155,000	3
South Africa	148,000	3
China	100,000	2
Greenland	86,000	2
Finland	60,000	1
Sweden	50,000	1
Kazakhstan	50,000	1
Other countries	413,000	8
World total	**5,385,000**	

2.1.2 Sustainability in Thermal Reactors

As explained later in Section 3, the high conversion ratio for thorium in thermal reactors means that, with reprocessing, thorium could be self-sustainable (conversion ratio of 1.0 or greater) as a fuel for LWRs. This sustainability differs from the uranium-plutonium fuel cycle, which can only be self-sustaining in fast reactors. However, the use of thorium requires that an initial fissile fuel inventory also be available (either enriched uranium, plutonium, or ^{233}U from operation of other reactors) to breed the ^{233}U from thorium.

2.1.3 Radiotoxicity of Spent Nuclear Fuel

When considering the radiotoxicity of spent nuclear fuel, timescales are vitally important due to the radioactive decay of the material. Typically for ^{233}U/^{232}Th fuel, the timing can be considered over three phases.

i. *100 to 200 years*: the medium-lived fission products dominate the radiotoxicity of the spent fuel.

   The fission product yields for ^{233}U are similar to uranium and plutonium fuels, and so the overall radiotoxicity is similar over this time period.

ii. *200 to 1,000 years*: the transuranics (primarily Np, Pu, Am, and Cm) dominate the radiotoxicity of the spent fuel.

   In ^{233}U/^{232}Th spent fuel, because of the lower mass number of the fuel and hence longer neutron capture routes to the higher actinides, there are much lower quantities of plutonium and the longer-lived minor actinides compared with the uranium-plutonium (U-Pu) fuel cycle, typically several orders of magnitude less than the equivalent U-Pu fuel. Overall, this reduction in higher actinides results in radiotoxicities in this time frame that are a factor of 10 less for ^{233}U /^{232}Th fuels compared with U-Pu fuels.

iii. *>1,000 years*: long-lived ^{233}U daughter products dominate on these timescales.

   For ^{233}U /^{232}Th fuels, the radiotoxicity is dominated by the daughter products of ^{233}U. For example, at very long timescales (20,000 to 1,000,000 years), isotopes such as ^{231}Pa and ^{229}Th dominate. The dominance of these nuclides results in the radiotoxicity of the ^{233}U /^{232}Th fuels being approximately twice that of the equivalent U-Pu fuels on these extended timescales, although this can vary depending on the specific scenario chosen.

2.1.4 Proliferation Resistance

A specific role considered for thorium fuels is for burning excess fissile materials as part of an open fuel cycle [7] with highly enriched uranium (HEU) or plutonium being used as the fissile material. These materials blend well with thorium because they are chemically similar and can form robust compounds. The higher burnups proposed for thorium fuel (utilizing its material

properties) would ensure sufficient ^{233}U production to extend cycle lengths/burnup and aid in the destruction of the driver fissile material.

Using thorium as the matrix rather than uranium limits the plutonium production, but the use of closed thorium-based fuel cycles with the separation of uranium (which includes a substantial ^{233}U fraction of the total uranium) would require the same level of safeguards and security as other fissile materials. The International Atomic Energy Agency (IAEA) [8] and the US Nuclear Regulatory Commission NRC [9] classify ^{233}U with plutonium and HEU as Category I materials. However, the route by which ^{233}U is produced results in additional intrinsic radiation barriers that improve the proliferation resistance of the material due to the heat and high-energy gamma emissions. Denaturing of the ^{233}U by adding ^{238}U would also assist in lowering the proliferation risk, but this approach would offset the benefits of reduced plutonium production.

The need for a fissile-containing driver region of thorium-based fuels, using either more enriched uranium (typically enriched to 5–20 wt% ^{235}U) or plutonium, is itself not without proliferation concerns.

2.2 POTENTIAL USE OF THORIUM IN VARIOUS SCENARIOS

Thorium continues to be of interest for future alternative fuel cycles because use of thorium may offer waste disposal benefits as well as a resource for conversion into reactor fuel. Thorium fuel can be used in both converter (i.e., self-sustaining) and breeder reactors in nearly all neutron spectra while U-Pu breeder reactors are only possible with a fast spectrum. Thorium has been considered for use as a resource extender in most thermal reactors, as a thermal reactor breeder, and for various uses in fast neutron systems. A summary of once-through and recycle thorium concepts is provided below.

2.2.1 Once-Through Thorium Fuel Cycle Options

Thorium has been proposed for use in once-through fuel cycles as a means of extending uranium resources and reducing the production of transuranic species. In once-through fuel cycles, the discharged fuel is directly disposed of without any recycling or reprocessing and therefore contains the residual of the original fissile material as well as the generated ^{233}U fissile material. Primary differences in implementation between the various once-through options include the type of reactor, the geometric configuration of the fuel, and the source of fissile materials. The sections below describe representative implementations of once-through thorium fuel cycle options.

2.2.1.1 Once-Through in LWRs with Enriched Uranium

Using thorium in a once-through fuel cycle in a reactor requires enriched uranium or separated plutonium in the fuel to ensure criticality and desired irradiation cycle length for power production. Most experience with this technology involves the use of thorium in LWRs. For use in LWRs, the uranium fuel must be more highly enriched than in a conventional uranium oxide (UOX) fueled system (e.g., 10%–20% ^{235}U instead of <5% in a PWR with standard uranium fuel) due to the neutron-absorbing characteristics of thorium. These changes would require

significant modifications to the existing fresh fuel infrastructure (e.g., fabrication, shipping casks, etc.) to address the higher enrichments and the related criticality issues, in addition to any issues associated directly with the use of thorium. The "seed-blanket" approach has been studied [10] as the thorium-based alternative to the once-through UOX fuel cycle in PWRs. This system utilizes fuel assemblies that are the same size as conventional PWR assemblies and can be retrofit into existing commercial PWRs without requiring significant modifications to the reactors or onsite equipment (e.g., fuel-handling tools). The initial loading in the blanket contains thorium and a small amount of low enriched uranium (LEU) to produce power. During irradiation, sufficient ^{233}U is bred in the thorium so that it can contribute to power production. The LEU is also used to dilute the bred ^{233}U so that the fissile concentrations in the uranium are always below the HEU limit.

2.2.1.2 Once-Through in Heavy Water Reactors with Enriched Uranium

Several concepts for the use of thorium have been proposed for HWRs, including both the once-through and recycle options. The use of thorium for once-through operation is intended to reduce the consumption of mined uranium. Most HWR designs are based on the pressure tube concept, which allows for independent fueling of each channel and thus enables different residence times for different fuels. This can be important because the thorium fuels benefit from being irradiated longer than the driver fuel. Irradiation concepts could therefore involve a mixed channel operation with thorium and driver fuel bundles occupying different channels, or a more homogenous approach could have thorium and driver fuel elements in the same bundle if breeding ^{233}U was less important. In both cases, the use of slightly enriched uranium (<2%) is required due to the increased neutron absorption from the inclusion of thorium in the fuel. Discharged thorium fuel would be stored and eventually sent to disposal. Thorium has also been introduced into pressurized heavy water reactors (PHWRs) in India to flatten the power distribution [11].

2.2.1.3 Once-Through in High-Temperature Gas-Cooled Reactors with Enriched Uranium

High-temperature reactors have been strongly associated with the use of thorium as a means to reduce the need for uranium. Several HTGRs have operated with thorium fertile material (see Section 2.3.1.3), and thorium use has been considered in concepts such as General Atomic's Gas-Turbine Modular High-Temperature Reactor (GT-MHR) [12]. These fertile particles can be intermixed with fuel particles in the same fuel compact or pebble material or in separate fertile compacts or pebbles.

2.2.1.4 Once-Through in Thermal Reactors for Plutonium Disposition

Plutonium stockpiles from excess weapons program materials and separated civilian material have grown to large levels over the past few decades. This has helped create interest in using plutonium as the fissile material in thorium fuel cycle systems. Additionally, thorium has been proposed as an alternative fuel matrix material to replace uranium to avoid the production of additional plutonium from the conversion of ^{238}U. In principle, these systems are not truly once-through given the reprocessing of the spent fuel to recover the plutonium creating the stockpiles; however, for purposes of the thorium fuel cycle assessment, the irradiated Pu/Th fuel materials are not recycled, operating essentially as a once-through use of the fuel. One example concept for using plutonium and thorium is the use of a seed-blanket approach in LWRs involving

separated driver and blanket materials. Alternatively, plutonium can be intermixed with the thorium in a homogenous fashion such that there is little, if any, uranium in the system.

2.2.1.5 Once-Through in Molten Salt Reactors

Molten salt reactors (MSRs) were originally developed as thermal spectrum breeder reactors with recycling, primarily for the purposes of increased resource sustainability. The denatured MSR (DMSR) concept was developed [13] as an alternative to the molten salt breeder reactor (MSBR) in order to reduce proliferation risk by eliminating the online chemical processing system and operating the reactor as a once-through system. The DMSR would be initially loaded with a large quantity of thorium to serve as a fertile material along with some LEU (19.75% ^{235}U enrichment). During the operation of the reactor, fresh LEU is added but no additional thorium is fed to the system. This has the twin benefits of eliminating the need for online reprocessing and ensuring that there is sufficient ^{238}U in the salt to dilute the ^{233}U, maintaining the ^{233}U in a "denatured" state. All of the materials are contained in molten fluoride salt. The volatile gaseous fission products are removed from the salt, while all other actinides and fission products are not separated. DMSR concepts have a graphite moderator and a molten salt coolant, which would be circulated through the core and heat exchangers. At the end of reactor lifetime, the entire accumulated inventory would be disposed as waste.

2.2.1.6 Once-Through in Fast Reactor Systems

While fast reactors are typically not considered for once-through fuel cycles, it is possible to operate traditional fast reactor concepts (such as sodium-cooled reactors) in a once-through mode on enriched uranium with thorium as a fertile blanket material. Fast reactor "breed and burn" concepts are being proposed that do not involve recycling the fuel. Examples based on the uranium fuel cycle include the CANDLE [14], Traveling Wave Reactor [15], and the Fast Mixed Spectrum Reactor [16]. Given the ability to obtain a high conversion ratio (i.e., breeding) in fast spectrum systems, thorium once-through concepts can also be implemented. These systems involve an initial starting charge of enriched uranium (nominally 10 wt% ^{235}U) and operate by breeding fissile ^{233}U from the fertile thorium material and burning it in place. Some of the concepts envision operating without fuel shuffling or refueling during the full reactor lifetime, while other concepts would involve shuffling of the fuel bundles. In all of these systems, the discharged fuel would be expected to contain a relatively large quantity of fissile materials that would be disposed of directly.

2.2.1.7 Once-Through in Externally Driven Systems

Externally Driven Systems (EDSs) are concepts that use an external source of neutrons to offset some or all of the need for a fissile startup charge and overcome neutron losses due to fission product absorption as irradiation proceeds. These systems could utilize thorium as a fertile material by itself and potentially eliminate both enrichment and reprocessing in the fuel cycle. EDSs are not critical reactors; the neutron chain reaction is not self-sustaining. A large particle accelerator or fusion system typically acts a source of high-energy neutrons that converts thorium to ^{233}U, which then fissions to produce energy. Subcritical operation could enable very high burnup levels in the fission fuel, but energy economics and materials performance issues may limit this potential.

2.2.2 ^{233}U/Thorium Recycle Fuel Cycle Options

As a fertile material, thorium generally requires recycling to increase its resource utilization. Reactor systems supporting thorium fuel cycles can operate as converters or breeders. Converters require additional fissile material to operate, but breeders would eventually be self-sufficient on thorium/^{233}U. The nuclear properties of the thorium/^{233}U system support breeding in thermal reactors as well as in fast reactors. In recycle systems, the uranium that has been created (primarily ^{233}U) is separated from the thorium and recycled back into a reactor system; systems can self-recycle ^{233}U back into the system where it was created or fissile-producing breeder systems can provide fuel for other reactors. This separated uranium typically has a high ^{233}U fissile content and therefore represents a proliferation risk. The production of ^{232}U along with the ^{233}U provides a strong radiation field that both hinders recycle and possibly offers some proliferation protection. The standard process for recycling thorium is a modified version of the PUREX process called "THOREX" [17], which was developed and demonstrated to be effective although it has not reached industrial maturity. Alternative approaches are available based on fluoride volatility processes and electrochemical refining. The following sections contain descriptions of systems that involve recycle of generated ^{233}U.

2.2.2.1 Recycle in Light Water Reactors

In the recycle of thorium LWR discharge fuel, there are two options: ^{233}U/Th or Pu/Th fuels. Nuclear physics characteristics (e.g., neutron cross section ratios and the number of neutrons produced per fission) of thorium fuels suggest that sufficient ^{233}U cannot be generated in a conventional PWR to make it self-sustaining on a pure ^{233}U/Th cycle and thus enriched uranium or plutonium would be required in the fuel. If the objective is a complete avoidance of the need for natural uranium and uranium enrichment, sufficient quantities of ^{233}U need to be produced in breeder reactors. The Bettis Atomic Power Laboratory (BAPL) under the Advanced Water Breeder Applications (AWBA) program performed an extensive study of a potential ^{233}U/Th economy in PWRs. This study [18] envisioned "pre-breeder" reactors based on the Shippingport LWBR to create ^{233}U stockpiles. Several studies examining the use of Pu/Th fuel in a PWR to more efficiently burn the plutonium from conventional UOX spent fuel confirmed a significant increase in net plutonium consumption relative to MOX. There is still significant production of transuranic waste (TRU), but less than is produced with conventional MOX fuel.

2.2.2.2 Recycle in Heavy Water Reactors

Thorium can be used in an HWR with recycle to create a self-sustaining fuel cycle. A potential approach to recycle in CANDU reactors called the Self-Sustaining Equilibrium Thorium (SSET) cycle [19] would generate sufficient ^{233}U to replace fuel that is consumed during operation, but fissile material from elsewhere would be needed for startup. The SSET approach requires reprocessing of the fuel to recycle the ^{233}U. The optimal discharge burnup for such a fuel cycle is low (~13 GWd/MTHM) to maximize the ^{233}U content, which may not be economically attractive due to the high mass flow rates that would be required. The Indian nuclear program is heavily based on the use and breeding of ^{233}U in AHWRs [20–25]. Startup fissile material would likely be plutonium or enriched uranium.

2.2.2.3 Recycle in High-Temperature Gas-Cooled Reactors

The recycle of ^{233}U within a gas-cooled reactor has been widely considered. Irradiated thorium fuel in the form of coated particles can be extracted from the fuel compacts or pebbles and recycled based on similar processes proposed for uranium/plutonium recycle. The coated particles would likely be cracked with the thorium kernel materials being leached out of the particles, which would be more difficult for thorium particles in comparison to uranium. This leached material can then be separated using the THOREX process to obtain the ^{233}U used for the development of new fuel. Fuel fabrication processes will require R&D, as the current approaches are not performed with radioactive fuels that produce heat. Such self-recycle systems would be expected to have very high conversion ratios with the potential to be close to breeding.

2.2.2.4 Recycle in Molten Salt Breeder Reactors

From the 1950s through the mid-1970s, ORNL developed a MSBR concept [26] with a focus primarily on resource sustainability. The MSBR was designed to operate as a thermal breeder on the thorium-uranium cycle with high breeding ratios. The use of a molten salt fuel optimizes the overall breeding ratio by allowing continuous processing of the fuel to remove fission product poisons, which lower the breeding ratio, and separate ^{233}Pa for storage in regions of low flux outside the core to minimize parasitic neutron capture and thus enhance ^{233}U production. On-line fuel processing must be performed onsite. Alternative designs with lower conversion ratios (less than unity) have been considered to reduce the complexity of the online fuel processing and perhaps enable batch processing of the salt that can be performed offsite. Thorium-fueled MSRs have also been investigated in a fast reactor configuration as TRU and minor actinide burners [27, 28].

2.2.2.5 Recycle in Fast Reactor Systems

Recent examination of thorium-based fuels in fast spectrum systems has been relatively limited; most of the available information dates back to early studies. Thorium blankets can be used to breed ^{233}U analogous to breeding plutonium in a conventional fast breeder reactor. The reactor spectrum may have to be tailored to maximize ^{233}U production because a softer spectrum is more desirable for this application. Alternatively, thorium can be used as the matrix material in place of uranium in homogeneous or driver/target transmutation to limit production of new TRU. If additional uranium is added to address the proliferation risk from ^{233}U, this will negatively affect the rate at which TRU can be transmuted.

2.2.2.6 Recycle in Externally Driven Systems

In addition to using EDSs for once-through fuel cycles as discussed above, it is also possible to recycle the irradiated materials. In this case, while the system may avoid the need for fissile material for startup, it will require some form of fuel processing.

2.2.2.7 Recycle in Multi-Reactor Systems

Several options involving multi-reactor systems have been proposed for the use of thorium fuel [5, 11, 20–25]. There are some advantages to having a fuel cycle in which some reactors are dedicated to the production of fissile material that is used to fuel other reactor types. Typical second-stage reactor types have a thermal spectrum where the fissile material requirements are

low and ^{233}U represents good reactor fuel. One example of this fuel cycle is the Indian three-stage program where the second stage involves breeding of ^{233}U in blankets of fast reactors for subsequent use in HWRs. An alternative implementation considered is the use of "pre-breeders," consisting of either thermal or fast reactors that would provide fissile material to start up ^{233}U-fueled reactors with thorium and provide make-up fissile material. Large EDSs, such as so-called "fission-suppressed ^{233}U breeder" fusion systems [29], could also produce fissile materials for subsequent use in thermal-spectrum reactors. In principle, a fuel cycle based on this approach could eliminate fuel enrichment by generating fissile materials in the EDS but would still require reprocessing to recover fissile materials and fabricate new fuel.

2.3 HISTORICAL USE OF THORIUM IN NUCLEAR REACTORS

2.3.1 United States Experience with Thorium

2.3.1.1 Pressurized Water Reactors

Indian Point Unit 1 was a 275 MW_e PWR owned and operated by Consolidated Edison from 1962 to 1974 [30]. The first core of unit 1 contained mixed ThO_2-UO_2 (HEU) fuel. The initial core from Indian Point 1 was reprocessed at the West Valley Reprocessing Plant in West Valley, NY. Of the initial core, 1.1 metric tons of uranium was recovered in liquid form, which contained 7 wt% ^{233}U and 58 wt% ^{235}U, indicating a rather significant amount of thorium-to-uranium breeding in the reactor [31].

A portion of the Indian Point U-Th fuel was shipped to Babcock and Wilcox (B&W) for post-irradiation examination (PIE) [30]. Fuel specimens whose burnup ranged from 3.0 to 30.0 GWd/MTHM were examined. From a fuels standpoint, the U-Th fuel performed quite well. Fission gas release was less than 2.0%, which is similar to what was observed for standard UO_2 fuel at the time. In addition, distortion and swelling of the fuel were minimal, regardless of burnup. The conclusions of the PIE of the Indian Point 1 Cycle 1 fuel were that the ThO_2-UO_2 fuel was more resistant to thermal cracking, hour-glassing, and irradiation-induced swelling than UO_2 fuels operated under similar conditions.

In addition to Indian Point, significant thorium experience was gained in the Shippingport LWBR [32, 33]. The Shippingport LWBR was a 25 MW_e PWR designed by BAPL and operated by Duquesne Light Company from 1977 to 1982. The Shippingport reactor was originally developed to serve as a test reactor for naval and commercial power generation purposes. The reactor was designed to utilize different reactor cores. The third and final core, installed in 1977, is considered the LWBR and was developed to prove the concept of a pressurized water breeder reactor.

The LWBR core consisted of four primary fuel elements (seed, standard blanket, power-flattening blanket, and reflector blanket), as seen in Figure 2.1. The 12 seed elements were moveable and used for reactivity control. The seed elements contained an axially varying fraction of UO_2 (98.23 wt% ^{233}U) and ThO_2. The blanket and power-flattening blanket elements contained a higher fraction of ThO_2 and were arranged in an annular fashion, through which the

seed blankets were inserted to achieve reactor control. The reflector blankets contained only ThO_2. The fuel pellets in the reactor contained 1–5 wt% UO_2 in a ThO_2 matrix whose UO_2 weight fraction varied as a function of the particular fuel element being manufactured and the axial location in that fuel element. Each region of the core was optimized to increase neutron absorption in thorium. Comparisons of the fuel pre- and post-irradiation indicated an overall breeding ratio of 1.0139 [10]. During the LWBR experience, various shortcomings were identified such as the need for lower power density, higher initial enrichment in the seed fuel, a moveable seed region, more difficulty in reprocessing U/Th versus U/Pu, and fuel reprocessing and fabrication difficulties associated with additional shielding requirements [10].

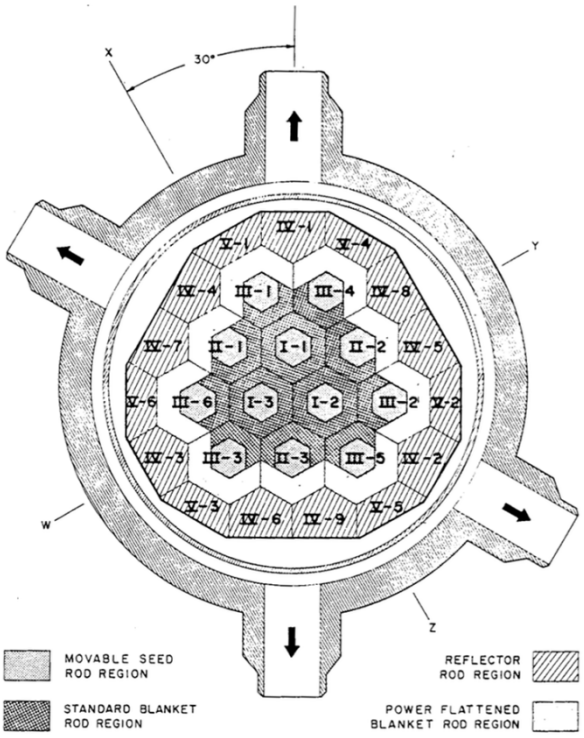


Figure 2.1. LWBR 2D cross section with element region identification [33].

2.3.1.2 Boiling Water Reactors

There have been two BWRs that utilized thorium in the United States. The first, BORAX-IV, was a 20 MW_t BWR that operated from 1956 to June 1958 at Argonne National Laboratory in 1956 and was used to test high-thermal-capacity fuel elements made from uranium and thorium ceramics. During operation, BORAX-IV demonstrated the feasibility of uranium-thorium oxide fuel elements while producing a measurable amount of ^{233}U. In addition, some transient tests and reactivity coefficient measurements were performed in BORAX-IV that might be applicable to near-term U/Th systems [34].

The 24 MW_e Elk River Reactor, a BWR built by Allis-Chalmers Manufacturing Company under contract to the U.S. Atomic Energy Commission (AEC) for the Rural Cooperative Power Company, operated from 1964 to 1968 [35]. The fuel was a mixture of ThO_2 and 93.5% enriched UO_2 clad in SS304L doped with 600 ppm boron. The reactor contained mostly 4.3 wt% UO_2 assemblies but also contained a number of 5.2 wt% UO_2 assemblies. A prototype fuel element

irradiated in the experimental breeder reactor (EBR) prior to operation of the Elk River Reactor showed good fuel performance up to 1000 MWd/MTHM with insignificant dimensional changes, little cracking, and good fission gas retention characteristics [36].

2.3.1.3 High-Temperature Gas-Cooled Reactors

HTGRs, which are typically graphite moderated, are well tailored to the utilization of thorium due to the high fuel burnup enabled by higher tolerance of the fuel to irradiation damage. Peach Bottom Unit 1 (PB1) [37] was a 40 MW_e HTGR demonstration plant that operated between 1966 and 1974 for 1,348 effective full power days (EFPDs). During operation, two graphite fuel cores were irradiated. The fuel used in PB1 was a mix of uranium and thorium carbides uniformly dispersed as coated particles in a graphite matrix that was compacted to form an annular fuel.

Fuel kernels in Core 1 were coated with a single pyrolytic carbon coating of ~55 μm. The outer diameters of the coated fuel particles ranged from 210–595 μm, and their overall packing fraction in the compact did not exceed 30%. Fuel compacts with varying uranium/thorium ratios were utilized. Overall, Core 1 operated for only half of its designed lifetime due to fuel failures. The single pyrolytic carbon layer was susceptible to fast-neutron-induced dimensional changes, damage due to fission product recoil, and gaseous fission product release from the particle. The problems with the single pyrolytic carbon layer led to significant radial expansion of the fuel compacts, which in turn interacted with the fuel elements, causing many to fail.

The fuel particle design was changed for Core 2, with an additional carbon buffer layer added between the fuel kernel and the pyrolytic carbon layer. The fuel element in the internal portion of the core contained equal parts of thorium and uranium, while the outer ring of fuel elements contained fertile particles that had an 18/5 thorium-to-uranium ratio. This change allowed Core 2 to operate for the length of its designed life of 900 EFPDs. The largely positive operational experience with the PB1 HTGR paved the way for a larger commercial HTGR at Fort Saint Vrain.

Fort Saint Vrain (FSV) [38] was a 330 MW_e commercial nuclear power plant owned and operated by Public Service Company of Colorado (PSC) that achieved first criticality in January 1974 and shut down in August 1989. The FSV reactor was a helium-cooled, graphite-moderated reactor that utilized a uranium-thorium fuel cycle. The FSV reactor employed many similar design features as PB1 but had many key differences: higher power (330 MW_e vs. 40 MW_e), a concrete reactor vessel (steel was used in PB1), and helium circulation powered by a steam-driven turbine. In 1989, the plant underwent a shutdown to repair a stuck control rod. During this shutdown, numerous cracks were discovered in several steam generators that were deemed too costly to repair.

FSV utilized TRISO (tri-structural isotropic) fuel particles that contained a central kernel of either HEU mixed with thorium (fissile particles) or pure thorium (fertile particles).The kernel was surrounded by concentric shells of a low-density pyrocarbon (PyC) buffer, dense inner PyC layer, silicon carbide layer, and dense outer PyC layer to form particles. These fuel particles were then added to a filler material (graphite) and compacted into rodlets that were loaded into a prismatic graphite block, as shown in Figure 2.2.

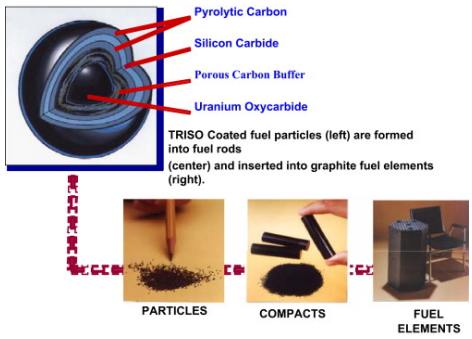


Figure 2.2. TRISO fuel particles, fuel compacts, and hexagonal fuel elements.

The FSV HTGR had a rather erratic operating history, but most problems were not related to the fuel composition or fuel form. Rather, most incidents were caused by ingress of water from the secondary system. The water ingress issues were related to the design and operation of the system and were unrelated to the use of thorium in the fuel.

2.3.1.4 Molten Salt Reactors

The Molten Salt Reactor Experiment [26] was an 8 MW_t experimental reactor operated at ORNL from 1962 to 1969. A smaller test reactor, the Aircraft Reactor Experiment (ARE), was built at Oak Ridge prior to the MSRE and was used primarily to investigate the nuclear stability of a circulating molten salt fuel system. During the ARE program, the attractiveness of molten salt reactors for civilian uses was recognized, which led to the development and construction of the Molten Salt Reactor Experiment (MSRE). The reactor was designed and constructed from 1960−1965, achieved first criticality in 1965, and sustained full-power operation beginning in December 1966. Continuous operation of the MSRE for a 6-month period brought a successful close to the first phase of operation. For the second phase, a chemical processing facility was connected to the reactor and was used to remove the original uranium fuel, which was replaced with ^{233}U fluoride, making MSRE the first reactor to operate using ^{233}U. Further research into chemical processes indicated that a single-fluid breeder reactor, which contains both fissile (^{233}U, ^{235}U, or ^{239}Pu) and fertile fuel (Th) in the same fluid, would be feasible and would have a breeding ratio of 1.05−1.07. Further research on MSRs was limited due to the decision to focus on fast breeder reactor development.

2.3.2 International Experience with Thorium

2.3.2.1 Germany

Germany gained significant experience through the HTGR pebble bed reactor projects AVR and THTR. In addition, Germany operated a BWR test reactor at Ligen that utilized thorium.

The first gas-cooled reactor project, AVR, was a 46 MW_t pebble bed reactor located at Jülich Research Centre that operated from 1967 to 1988 and utilized a number of different types of fuel pebbles [39]. Thorium and HEU carbide (ThC_2 and UC_2) bistructural isotropic (BISO) particles were utilized as well as thorium and HEU oxide (UO_2 and ThO_2) BISO particles. After AVR had been shut down, additional information regarding significant contamination in the AVR coolant loop became available [40]. The AVR primary coolant loop was heavily contaminated with

metallic fission products (mainly ^{90}Sr and ^{137}Cs) that were released as a result of peak fuel temperatures in the reactor that were much higher than expected and that were mobilized by fine graphite dust from fuel pebbles [40].

The THTR-300 was a 300 MW_e pebble bed HTGR constructed in Hamm-Uentrop, Germany, that operated from 1985 to 1989 [41]. THTR-300 utilized 93.0 wt% enriched UO_2 mixed with ThO_2 (10.635 ThO_2/UO_2 weight ratio) in the central fuel kernel of TRISO fuel particles that were then mixed with graphite filler and formed into fuel pebbles. THTR-300 had no significant operating problems and was shut down mainly due to uncertainties in fuel fabrication availability, uncertain spent fuel storage issues, and an impending operating license renewal. These financial uncertainties led to unsuccessful negotiations between the operating partners regarding continued funding.

Germany also operated a BWR called the Lingen Nuclear Power Plant that utilized Pu/Th test fuel, though very little information on the power plant could be identified.

2.3.2.2 United Kingdom

The 21.5 MW_t DRAGON Reactor Experiment (DRE) was an experimental reactor for the Organisation of Economic Co-operation and Development (OECD) High-Temperature Reactor project constructed in Winfrith, United Kingdom (UK) that operated from 1964 to 1975 [42]. Although constructed in the UK, the reactor was an international cooperation built under the OECD/Nuclear Energy Agency (NEA). The reactor was constructed mainly to conduct irradiation testing of fuels and fuel elements for high-temperature reactors. The core consisted of 37 hexagonal fuel elements that contained seven fuel rods arranged on a hexagonal pitch. The outer fuel rods were highly enriched UO_2 TRISO particles, while the inner fuel rods typically contained a mix of TRISO particles fueled with low-enriched UO_2, ThO_2, or PuO_2.

2.3.2.3 India

Over the past 25 years, India has amassed significant experience utilizing thorium in nuclear reactors due to the country's significant thorium reserves, rather small uranium reserves, and significant increase in electricity demand. In the 1950s, India formulated a three-stage nuclear power plan, which is still largely in place today. The first stage of the plan was to operate a number of PHWRs to generate plutonium. In the second stage, plutonium from the first stage would be used in fast breeder reactors with fertile thorium material. Fissile materials generated in the fast breeder reactors would then be used in the third-stage thorium-based PHWRs, which would lead to a fuel cycle based entirely on thorium breeding and subsequent burning of ^{233}U. A visualization of India's three-stage nuclear program can be found in Figure 2.3.

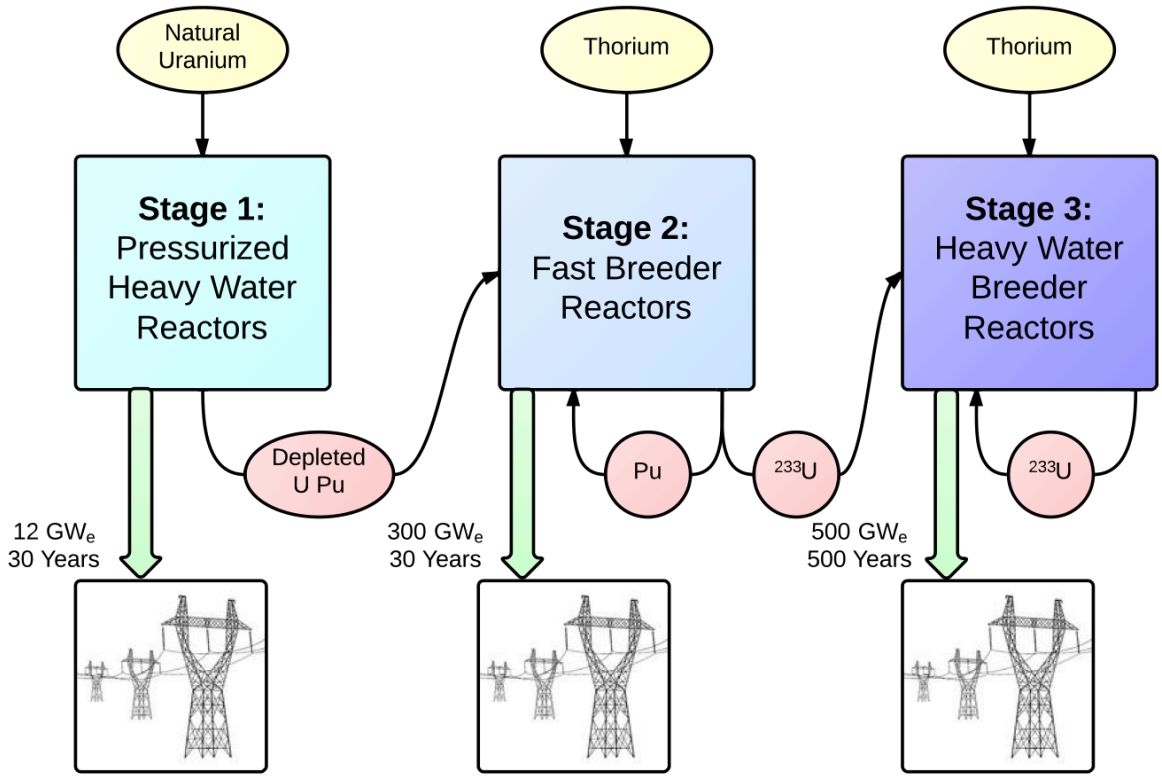


Figure 2.3. India's three-stage nuclear power program.

Test Reactors

The 40 MW_t CIRUS reactor located near Mumbai, India, operated as a materials irradiation test reactor from 1960 to 2010. The reactor was designed and constructed as a cooperative effort between India and Canada, and the heavy water moderator was provided by the United States. The CIRUS reactor operated on natural uranium fuel clad with aluminum and was used to perform many different material irradiation tests including ThO_2/UO_2 and ThO_2/PuO_2 irradiation in support of India's thorium program [20].

The DHRUVA reactor, whose design is similar to the CIRUS reactor, is a 100 MW_t heavy-water-moderated reactor that uses natural uranium fuel clad in aluminum. Thorium fuel has been irradiated as a seven-pin cluster containing ThO_2. Irradiation testing of thorium fuels is ongoing at DHRUVA, and more than 10 MTHM of ThO_2 pellets have been irradiated in the CIRUS and DHRUVA reactors [5].

The 30 kW_t KAMINI reactor, located in Kalpakam, India, is a water-moderated, BeO-reflected material test reactor that is fueled with ^{233}U-aluminum alloy [21]. Although KAMINI does not utilize thorium, post-irradiated thorium from other reactors in India has been reprocessed into the ^{233}U fuel used in this reactor.

Pressurized Heavy Water Reactors

The first stage in India's nuclear power plan is to use PHWRs to generate plutonium for a number of second-stage fast breeder reactors. India has constructed a number of these first-stage nuclear power plants of nearly identical design – 220 MW_e PHWRs that utilize natural UO_2 fuel with ThO_2 or depleted UO_2 fuel bundles to achieve power flattening in the initial core. These reactors are installed at a number of sites across India [11, 22].

Fast Breeder Test Reactors

The second stage in India's nuclear power plan is to utilize fast breeder reactors fueled with plutonium from the currently operating HWRs. India has constructed a Fast Breeder Test Reactor (FBTR) prior to large-scale deployment of the fast reactor concepts [23, 24]. The experience gained from the FBTR gave India confidence to begin construction of the Prototype Fast Breeder Reactor (PFBR) to be used as a final prototype reactor prior to large-scale deployment of fast breeder reactors [25]. The 40 MW_t sodium-cooled FBTR utilizes MOX fuel pins arranged in a hexagonal array that contain 30% PuO_2 and 70% UO_2 (85 wt% enriched). In addition, the FBTR has a large blanket region composed of assemblies filled with ThO_2.

2.4 CURRENT INTERESTS IN THORIUM

2.4.1 Light Water Reactors

2.4.1.1 Thor Energy

Thor Energy was established in 2006 to evaluate the feasibility of exploiting Norway's thorium deposits for energy production as the country's large export and indigenous energy sources, oil and gas, were being depleted. This initiative was prompted by a Norwegian government report [43] that concluded that the 170,000 metric tons of thorium estimated to be in Norway has a potential energy content more than 100 times greater than all of the oil extracted by Norway. Based on the current level of activity and investment by Thor Energy and its partners, it appears that Thor Energy's program of work is the most active of all current LWR-thorium research under way today.

The initial work completed by Thor Energy was a 2-year thorium fuel cycle feasibility study and identification of the most suitable reactors for utilizing thorium fuels. This feasibility study was completed in collaboration with industrial partners, including the Swedish utility Vattenfall and the KTH Royal Institute of Technology, and with support from experts in India. The report focused on whether the thorium fuel cycle could address three main arguments against uranium-fueled reactors: risk and consequences of severe accidents; proliferation issues related to plutonium production and inventories; and back-end fuel cycle issues such as spent fuel management and long-term geologic disposal issues including radiotoxicity.

The final report [44] was comprehensive in its assessment and included topics ranging from front-end issues (e.g., thorium mining and fuel assembly design and fabrication), reactor operation issues (e.g., reactor physics, reactor safety, and fresh and spent fuel assembly

handling), back-end issues (e.g., spent fuel storage, possible separations/recycling of spent fuel, and waste handling and storage), proliferation risks, licensing-related issues, and economic performance of the system.

After considering LWRs, HWRs, the Pebble Bed Modular Reactor (PBMR), and the Indian Advanced Thorium Breeder Reactor (ATBR), the focus of the subsequent work has been on thorium-MOX (plutonium) fuels as the nearest-term deployment option for thorium utilization. This is part of a three-step roadmap based on LWR technology.

	Main Challenge
1. Th-Pu once through in current LWRs	Fuel technology to be developed
2. Th-Pu and reuse of ^{233}U in current LWRs	Reprocessing and fuel fabrication
3. Breeding Th/^{233}U in advanced LWRs	Reactor core modification

The most recent and significant activity is the funding of a 5-year thorium irradiation project to be completed in the Halden test reactor in Norway. Partners in this substantial activity include Westinghouse, the UK National Nuclear Laboratory (NNL), Norway's Institute for Energy Technology (ITE), the European Commission's Institute for Transuranic Elements (ITU), South Africa's Steenskampskraal Thorium Ltd, the Finnish utility Fortum, and the French chemical company Rhodia, a rare earth mining company that owns thorium [45].

In particular, this experiment is intended to generate data on thermo-physical properties of thorium fuels. UO_2, standard U-Pu MOX, and Th-Pu MOX fuels will be irradiated and various rods removed during the irradiation to assess the impact of burnup on the key phenomena.

Using both online measurements and PIE (destructive and nondestructive), measurements will include centerline temperatures, pellet stack elongation, clad elongation, rod internal pressure, fission gas analysis, microscopy (grain structure, etc.), thermal conductivity, and micro-hardness. The behaviors being characterized from these measurements include temperature and thermal property changes, fission gas release, mechanical interactions and chemical interactions (e.g., stress corrosion cracking).

The Halden experiment is part of Thor Energy's overall development program to underpin future licensing requirements that include physical data acquisition, modeling tools development, reactor compatibility assessment, and advanced fuel and core design. Computational tool development efforts include modeling the fabrication and irradiation performance of Th-Pu fuels that are being developed in collaboration with Los Alamos National Laboratory, working with the University of Tokyo and Central Research Institute of Electric Power Industry (CRIEPI) in Japan on atomistic modeling of $(Th,Pu)O_2$, and a European Union Framework 6 project on the analysis of irradiated Th-Pu fuel pins. Advanced fuel and core designs efforts include Th-Pu BWR fuel bundle designs and development of a high-conversion thorium-fueled BWR in collaboration with the Massachusetts Institute of Technology (MIT) and Chalmers University.

2.4.1.2 Lightbridge

Lightbridge Corporation was formed in 1992 (then known as "Thorium Power Ltd") to develop nuclear fuel designs developed by Dr. Alvin Radkowsky, a former head of the U.S. Naval

Nuclear Propulsion Program. The development of the fuel designs and growth of the organization led to a collaboration with the Kurchatov Institute (KI) in Russia in 1994, where the initial fuel designs underwent further development and testing, including irradiation trials of the fuels in a research reactor in Russia. This collaboration subsequently led to the signing of PIE agreements with KI in 2008. In 2009, Thorium Power was renamed as Lightbridge Corporation and entered into an initial collaboration agreement with AREVA on thorium-based fuels.

Lightbridge is currently developing two types of advanced fuels for LWRs: (i) an evolutionary metallic fuel [46] with improved heat transfer and fuel properties and (ii) a thorium-uranium dioxide fuel [47]. The two designs do share some similarities.

The thorium-based fuel uses a seed and blanket assembly design and utilizes a once-through fuel cycle. The central section of the assembly is the seed, and this can be separated from the outer blanket section that contains the thorium-uranium rods. Both the seed and blanket rods contain LEU and require a higher enrichment than conventional LWR UO_2 fuel in order to provide the lifetime flux and multiplication factor for breeding ^{233}U. The blanket region uses Th-U oxide fuel rods, whereas the seed region uses uranium-zirconium metal fuel rods. The ability to separate the two regions is required so that the blanket region can remain in the core longer than the seed in order to breed sufficient ^{233}U and maximize the power generated from the thorium ore. The in-growth of ^{233}U is relatively slow, and to see sufficient fissions from ^{233}U, long irradiation times are required for the blanket. The seed is likely to remain in the core for three cycles (consistent with current LWR designs), whereas the blanket is required to remain for at least twice as many cycles. It is estimated that this approach will reduce uranium ore requirements by approximately 10% compared with standard UO_2 fuel.

The power share between the seed and blanket results in higher power in the seed region, which necessitates the use of metallic fuel. This fuel form has been extensively used in the reactors that power the Russian icebreakers, and the collaboration with Russia has provided access to the large database of irradiation experience with that fuel.

The fuel assembly (seed and blanket combined) is designed to fit within the existing outer envelope of conventional LWR fuel so that no core modifications would be required; however, refueling equipment would need modifications to allow movement and transfer of the blanket and seed portions of the design.

In addition to the UO_2 variant of the Lightbridge design, there are two further designs focused on: reactor-grade and weapons-grade plutonium disposition missions. A specific design for VVER reactors (Russian PWRs) was developed for the disposition of weapons-grade plutonium in the early to mid-2000s, known as the Radkowsky Thorium Plutonium Incinerator [47]. The seed in this case was weapons-grade plutonium in the form a star-shaped (three-lobed), helical assembly with 108 fuel rods per assembly. The driver fuel was plutonium-zirconium metal (Cermet). The outer blanket assembly had the same outer envelope and design as a regular VVER-1000 fuel assembly and contained 228 blanket rods with a mixture of ThO_2-UO_2.

The most recent developments of Lightbridge fuel designs appear focused on the metallic fuel designs, targeted at plant uprates, with little apparent recent development in the thorium seed-blanket concept.

2.4.2 Heavy Water Reactors

As discussed above, there has been historical interest in the use of thorium in HWRs, including CANDU and PHWR. While the primary activity is associated with the Indian nuclear development program, there has been recent collaboration between Canada and China on the use of thorium in CANDU reactors.

2.4.2.1 Indian Nuclear Energy Development Program

While the actual current use of thorium in India is not substantial, there continues to be significant development on the country's three-stage fuel cycle with thorium use focused on PHWRs, including a recently developed 300 MWe Advanced PWHR concept. Research infrastructure is being developed to further advance the development and use of thorium.

2.4.2.2 Canada-China Collaboration

Additional recent activity in the use of thorium in HWRs is in CANDU reactors with a collaboration between Canada and China. More specifically, a collaboration agreement between the Third Qinshan Nuclear Power Company, the Chinese North Nuclear Fuel Corporation, and the Nuclear Power Institute of China is supporting the development of the use of thorium in CANDU reactors [48]. The collaboration with Canada had previously been through Atomic Energy of Canada, Ltd and more recently with Candu Energy, Inc. The collaborative work has been arranged in a series of phases, with the first phases starting in 2007. The first phase focused on testing of CANDU-6 thorium oxide fuels, developing laboratory capabilities, and establishing a test line for fuel fabrication. The second phase, starting in 2009, was focused on conducting research on thorium fuel manufacturing including the fabrication of two thorium oxide CANDU fuel bundles. A third collaboration phase was announced in 2012 to consider the development of an "Advanced Fuel CANDU Reactor," which is optimized for the use of thorium (as well as recycled uranium).

2.4.3 High-Temperature Gas-Cooled Reactors

Despite significant historical development efforts, there is currently limited consideration of the use of thorium in HTGRs. Recent development activity by Steenkampskraal Thorium Limited leverages experience in the development of the PBMR to develop the TH-100 reactor, which is a 100 MW_t pebble bed reactor that can utilize thorium [49]. While still early in the development phase, an initial design is being actively pursued.

2.4.4 Molten Salt Reactors

While dormant for several decades, interest in the use of thorium in MSRs has significantly expanded recently, with a major new project in China sponsored by the Chinese Academy of Sciences being authorized in 2012. Through the Shanghai Institute of Applied Physics, a

Thorium Molten Salt Reactor (TMSR) [50] project has been established and research and development initiated with a goal to have a near-term 2 MW experimental reactor being developed in the 2017−2020 time frame. This would be followed at a later date with the design and development of a 100 MW reactor. The TMSR center is currently performing initial concept development activities for two reactor types, the traditional MSR and a concept with solid fuel using fluoride salt as a coolant, and is also developing experimental facilities including flow loops and materials characterization equipment.

3 MATERIAL PROPERTIES OF THORIUM-BASED FUELS

It is important to consider some of the key properties of the thorium-based fuel that lead to consideration of its use before evaluation of in-reactor and out-of-reactor issues related to a once-through LWR thorium fuel cycle.

3.1 FERTILE VERSUS FISSILE

Like uranium, thorium is also naturally occurring. However, unlike uranium, natural thorium contains only one isotope, ^{232}Th, with a very long half-life (1.41×10^{10} years). Natural uranium contains ~ 0.72 wt% ^{235}U that is fissile (fissions at all neutron energies), ~ 99.27 wt% ^{238}U that is fertile (capture of a neutron leads to production of a fissile isotope, ^{239}Pu), and a trace amount of ^{234}U. As illustrated in Figure 3.1, ^{232}Th behaves similarly to ^{238}U, as its capture of a neutron leads to the production of a fissile isotope, ^{233}U in this case. In order for thorium to be a resource for nuclear fuel, a fissile material (e.g. ^{235}U, ^{239}Pu, or ^{233}U) has to be used as a "driver" or a "seed" to generate additional neutrons to sustain a fission chain reaction and neutrons to be used for breeding fissile material.

Two fuel cycle options exist for thorium fuels:

(i) a once-through fuel cycle where the fertile thorium remains in the reactor for its lifetime, long enough to capture enough neutrons to produce sufficient fissile ^{233}U and offset a small amount of the uranium needed, and

(ii) a closed fuel cycle, where the ^{233}U is chemically separated from the spent fuel and recycled to make new fuel, based on the fissile ^{233}U. It is therefore possible to establish a self-sustaining fuel cycle with thorium in the same way it is with uranium-plutonium.

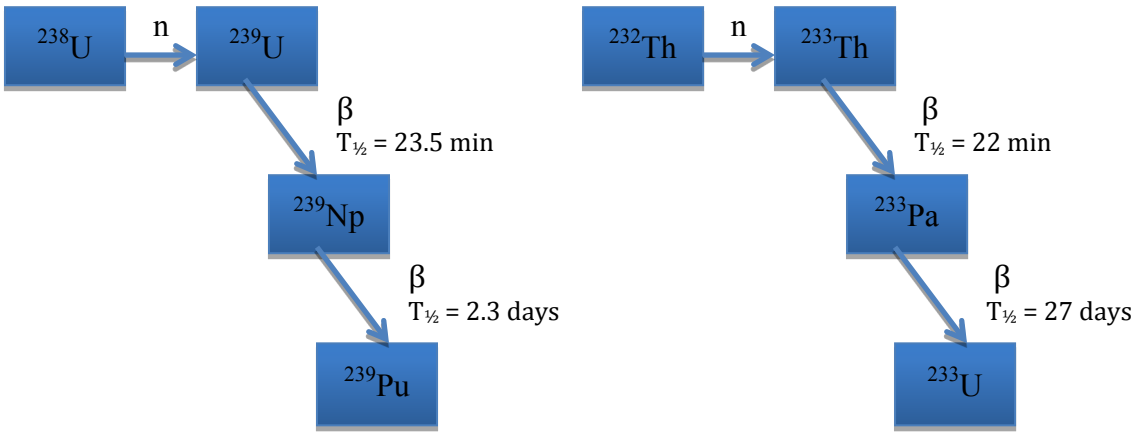


Figure 3.1. Capture and decay chains for ^{232}Th and ^{238}U.

One of the key benefits of the thorium fuel cycle option, however, is that it enables a higher conversion ratio to be achieved in a thermal reactor than is possible with the U-Pu fuel cycle. The conversion ratio quantifies the rate at which fertile nuclides (i.e., ^{232}Th in the thorium cycle and ^{238}U in the U-Pu fuel cycle) are converted to fissile nuclides:

$$Conversion\ Ratio\ (CR) = \frac{Rate\ of\ production\ of\ fissile\ istotopes}{Rate\ of\ consumption\ of\ fissile\ isotopes}.$$

A conversion ratio of 1.0 means that for every fissile atom consumed, a replacement fissile atom is created via fertile capture; this is the minimum requirement for a breeding cycle and is referred to as "breaking even." In a thermal reactor, the U-Pu cycle gives a conversion ratio of ~0.6–0.7, and therefore a breeding cycle (conversion ratio greater than 1.0) is not possible. The absorption cross section of thermal neutrons for ^{232}Th (7.4 barns) is almost three times that of ^{238}U (2.7 barns) based on JEF 3.1 [51]. This makes thorium a better fertile material than uranium in a thermal spectrum – the converse is true in a fast reactor.

A higher conversion ratio is possible for the thorium cycle, because for thermal neutrons ^{233}U has a neutron fission yield per neutron absorbed that is greater than 2.0 over most of the thermal energy range. This primarily results because ^{233}U has a larger thermal neutron fission-to-capture probability than either ^{235}U or ^{239}Pu. The average number of fission neutrons produced per neutron absorption (called "eta") is typically 2.3 for ^{233}U in a PWR, compared with 2.1 for ^{235}U and ^{239}Pu for thermal neutrons [51]. When accounting for losses due to parasitic absorption, this increased eta for ^{233}U can allow breeding in a thermal spectrum when care is taken to maximize the neutron economy of the system.

3.2 CRITICAL MASS

As stated above, thorium is not fissile; however, ^{233}U is an excellent fissile material. Facilities such as fuel manufacturing or reprocessing plants in operation today may not be suitable for storing or processing ^{233}U without introducing more stringent restrictions on processing and throughput.

3.3 THORIUM CAPTURE DECAY CHAINS

Thorium (n,2n) reactions result in production of ^{232}U. As shown in Figure 3.2, the decay chain and daughters of ^{232}U include hard gamma emitters (in particular ^{208}Tl and ^{212}Bi) that need to be considered in the fuel fabrication stages of recycled thorium fuel where remote handling and shielding protection for operators become an issue. Depending on the original ^{232}U concentration, it takes several days or weeks after separation for the concentration of gamma-ray-emitting daughter products to build up and require additional shielding, potentially providing a "window" of time in which handling may be easier. This issue is examined in the comparison of spent fuel radiological source terms in Section 6.

In addition, the relatively long half-life of ^{233}Pa can result in an increase in reactivity sometime after reactor shutdown as the ^{233}U concentration increases due to the ^{233}Pa β-decay. This has to be built into the reactor design and safety analyses.

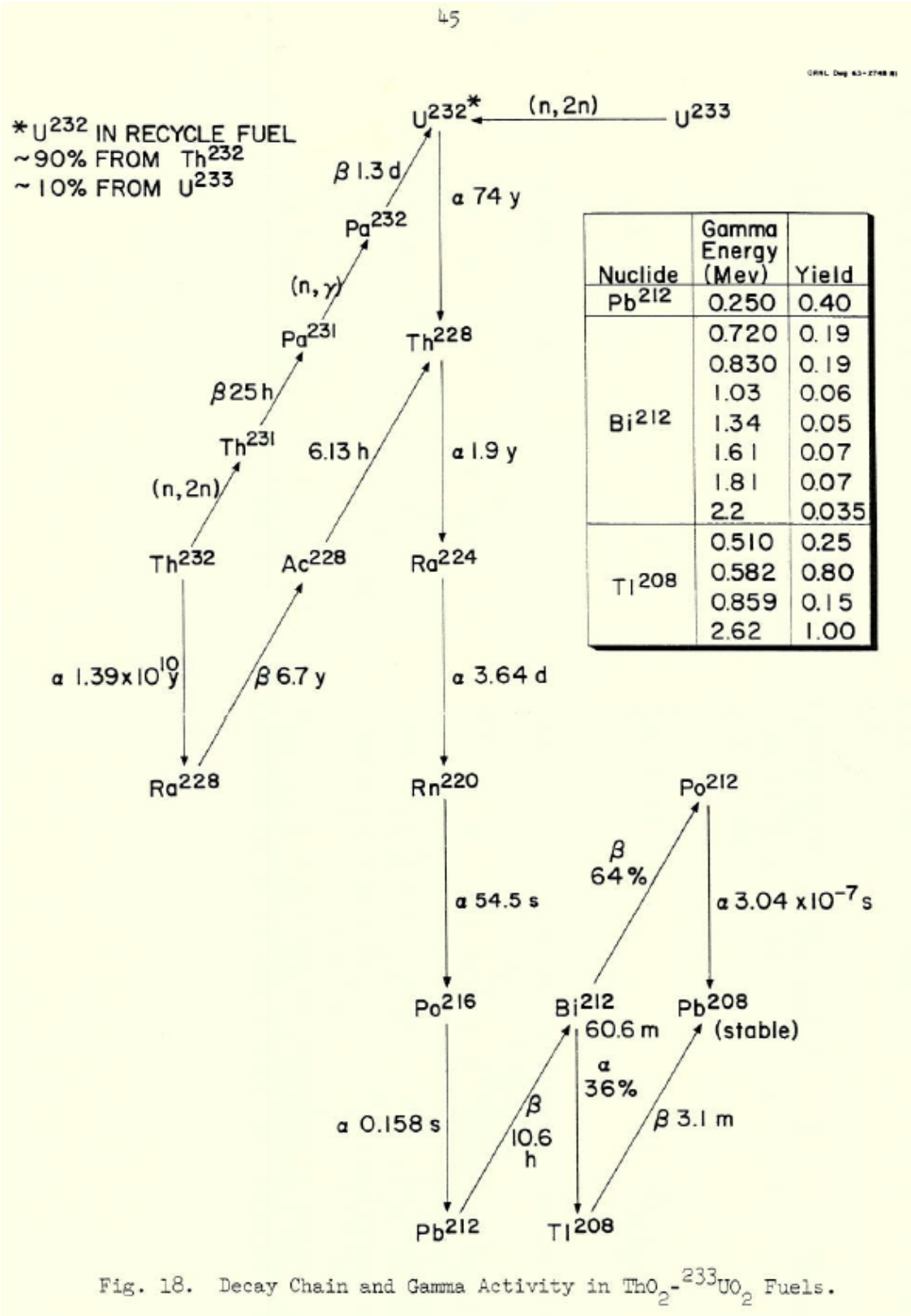


Figure 3.2. ^{232}U generation and decay diagram (from ORNL DWG 63-2748 R1).

3.4 MATERIAL PROPERTIES

Thorium dioxide (ThO_2) is chemically more stable and has improved performance under irradiation compared to uranium dioxide (UO_2). With thermal and mechanical properties similar to UO_2 and plutonium dioxide (PuO_2), ThO_2 is very compatible as a mixture with other fissile/fertile fuel materials. In particular, the melting point and thermal conductivity of ThO_2 is higher than these other fuels, therefore providing more thermal margin in accident conditions. In addition, a lower thermal expansion and greater ability to retain fission products further enhances it capabilities as a fuel, particularly for high-burnup fuels, such as in HTGRs.

However, the higher melting point means that a much higher sintering temperature is required for fuel manufacture. Its inert nature as a compound also makes thorium's chemical dissolution and separation more difficult, as would be required for reprocessing.

It should also be noted that the ThO_2 would likely need to be mixed with UO_2 or PuO_2, which would provide the fissile content of the fuel. Depending on the makeup of the blended composition, the characteristics of the other fuel materials could affect or dominate the overall fuel properties.

Section 4 of Reference 5 provides a good summary of some representative material property values, part of which is reproduced in Table 3.1. Section 6 of Reference 52 provides additional detailed materials properties correlations for UO_2, ThO_2, and ThO_2/PuO_2 mixtures.

It should be noted that many material properties vary as a function of temperature, fluence, burnup, or other variables. Also, a mixture of materials (e.g., ThO_2 and PuO_2) mixtures does not often exhibit a simple volume- or mass-weighted average of constituent properties.

Table 3.1. Comparison of key physical properties of UO_2, PuO_2, and THO_2 fuels, adapted from Table 6 of Reference 5

Property	**UO_2**	**PUO_2**	**ThO_2**
Crystal structure	FCC (CaF2 type)	FCC (CaF2 type)	FCC (CaF2 type)
Melting point, K	~3123	~2623	~3643
Theoretical density, g/cm^3 at 298 K	10.96	11.46	10.00
Thermal conductivity, $Wm^{-1}K^{-1}$ 773 K 1773K	 4.80 2.40	 4.48 1.97	 6.20 2.40
Coefficient of thermal expansion (K^{-4})	10×10^{-6} (298–1223 K)	11.4×10^{-6} (298–1223 K)	9.67×10^{-6} (298–1223 K)

4 QUALITATIVE EVALUATION OF THORIUM LICENSING IN LWRS

The Standard Review Plan (SRP) (NUREG-0800) [2] was written to provide guidance to NRC staff in performing various licensing and permit reviews, ensure quality and uniformity of reviews, and improve communication between the NRC, licensees, and the public. In order to provide useful information regarding the reactor and safety issues of using thorium in commercial nuclear power plants in the United States, the SRP was used in this report to guide the discussion of major issues associated with the addition of thorium to the nuclear fuel design. Detailed reviews were performed of those chapters in the SRP where a substantial impact was expected, and cursory reviews of chapters were performed for chapters where the impact is expected to be minimal. This section of the report discusses the chapters of the SRP that could be appreciably impacted by the use of thorium fuel in the current fleet of LWRs.

4.1 REACTOR (NUREG-0800, CHAPTER 4)

Several sections in Chapter 4 are of particular importance when considering the use of thorium-based fuels, in particular Sections 4.2, 4.3, and 4.4 (Fuel System Design, Nuclear Design, and Thermal and Hydraulic Design, respectively). Each of these sections has been evaluated due to the relevance of these sections to reactor and safety analysis when a different fuel is considered. Other chapters are related to these sections and therefore can also be impacted by the type of fuel.

4.1.1 Fuel System Design (NUREG-0800, Section 4.2)

Section 4.2 of the SRP focuses on the nuclear fuel system (fuel rods, pellets, springs, cladding, etc.) as it relates to safety during normal operating conditions and anticipated operating occurrences. The fuel system design has far-reaching impacts on other parts of the reactor system, and is of primary concern in reference to radiological consequences. There are a number of phenomena associated with thorium fuel that are, or could be, different when compared to typical UO_2 fuel including melting temperature, fission gas release, decay heat, and reactivity coefficients.

NUREG-0800 Section 4.2.I.3 summarizes the needs associated with new fuel materials in the following paragraph.

> New fuel designs, new operating limits (e.g., rod burnup and power), and the introduction of new materials to the fuel system require a review to verify that existing design-basis limits, analytical models, and evaluation methods remain applicable for the specific design for normal operation, [anticipated operating occurrences] AOOs, and postulated accidents. The review also evaluates operating experience, direct experimental comparisons, detailed mathematical analyses (including fuel performance codes), and other information.

As discussed in the previous section, thorium has been used in a wide variety of reactors as part of previous reactor and thorium fuel cycle development programs. The most relevant experience is the use of thorium in commercial LWRs including Indian Point 1 and Shippingport in support

of the LWBR program. These applications and their associated development programs (as well as the use of thorium in other reactor types) could potentially provide useful experimental information. However, the use of thorium in these systems is substantially different from the currently proposed uses in LWRs by Thor Energy and Lightbridge. Indian Point 1 used HEU with thorium, and the Shippingport core design was based on a tight-pitch seed-blanket configuration to maximize breeding. The available experience in the use of these thorium-containing fuels is limited in comparison to the experience base with uranium-based fuels, and therefore additional thorium-based fuel irradiation test data will be required. Experience with the behavior of the fuel materials in these reactors, depending upon data availability, may be useful.

Historically, there has been very little irradiation testing of thorium-based fuels, as opposed to uranium-based fuels for which a significant amount of experiential knowledge exists. The limited experimental data and irradiation experience lead to a considerable lack of data that is readily available for UO_2 and would typically be used by fuel performance, thermal-hydraulic, and other codes. Much of the data needed for thorium fuel will likely need to be generated through irradiation testing or other experiments, as similar data for UOX or MOX is unlikely to be applicable. In addition, new phenomena may require new methods development, similar to those developed to support MOX fuel designs. Additionally, transient fuel testing, which helps establish operating margins and fuel failure characteristics, would be needed for thorium fuels but would be difficult to attain as the Transient Reactor Test Facility (TREAT), where previous transient testing was performed, is no longer operational. However, international test facilities such as CABRI (France) or NSRR (Japan) may be able to provide needed transient testing for new fuels.

Data for nuclear codes (radiation transport, depletion, and decay) such as cross sections and decay data are available but have not been as extensively validated as corresponding data for uranium. This is especially important not only for ^{232}Th and ^{233}U but also the irradiation products of thorium, which include hard gamma emitters important to radioactive waste management (SRP Chapter 11) and radiation protection (SRP Chapter 12). Furthermore, the data typically used to validate the nuclear data, software, and methods, such as critical experiments, are lacking for thorium.

A key issue with using a new fuel will be the uncertainties that are assigned to particular operating parameters. Because of a lack of experiential knowledge of using thorium in modern LWRs, the uncertainties will likely be greater than those of using UO_2 fuels. These large uncertainties could impact operating margins in the reactor. The licensee may need to seek methods (experiments, validation, etc.) to reduce these uncertainties.

In the event of severe accidents, the radiological dose associated with radionuclide release from reactors operating using a certain fuel type becomes highly important. The retention of fission gases is an important characteristic of a nuclear fuel. Current research suggests that retention of fission gasses, along with the higher melting temperature, actually make thorium-based fuels more favorable than uranium-based fuels, but these characteristics will need to be evaluated during licensing and validated through experiments. In addition to the retention of fission gasses, the specific fission products and actinides produced during irradiation of thorium fuel vary from typical uranium-based fuels. The difference in the radionuclide inventories for thorium-based

fuels, their associated radiotoxicity, and decay heat characteristics will have an impact on the reactor and associated systems.

It is also likely that some regulatory guides and NUREG-series reports that are key to license review would need to reviewed, and possibly updated, for applicability to thorium-based fuels.

4.1.2 Nuclear Design (NUREG-0800, Section 4.3)

Section 4.3 focuses on the key nuclear properties of the nuclear fuel and nuclear core design including power distributions, reactivity coefficients, controls systems, etc. Utilization of thorium in the fuel will have a noteworthy impact on the nuclear design. The following statement from NUREG-0800 summarizes Section 4.3.

> The review of the nuclear design of the fuel assemblies, control systems, and reactor core is carried out to aid in confirming that fuel design limits will not be exceeded during normal operation or anticipated operational transients and that the effects of postulated reactivity accidents will not cause significant damage to the reactor coolant pressure boundary or impair the capability to cool the core and to assure conformance with the requirements….

Included in the licensee calculations are core axial and radial power distributions, as well as within-pin power distributions. The core axial and radial power distributions are important for determining peak power locations that may represent the most limiting conditions for the fuel. Core physics analysis will need to be performed by the licensee, and the introduction of thorium fuels may result in different power distributions from those typically observed for uranium-fueled LWRs. For example, when operating at the same power level, the introduction of fertile fuel as blankets, as considered in some thorium core designs, requires fissile driver fuel to operate at a higher power at the start of the cycle, which will increase the pin and assembly peaking factors in the reactor. This increased power peaking must be considered in determining operational and safety margins within which the reactor can be operated safely.

The within-pin power distributions are critically important to fuel performance, and the differences between uranium- and thorium-based fuels will need to be considered. The within-pin power distributions for thorium fuel, depending on the fuel design, may be different from those for uranium fuel. The increase in fissile isotopes that would develop along the radial edge of the fuel pellet could lead to a greater power in that portion of the fuel, similar to what is observed for uranium fuel. The difference between the rim effects for uranium and thorium fuels would need to be considered. Fuel densification and other thorium fuel irradiation properties could have an impact on these distributions, which would necessitate further detailed analysis. The fuel performance, transient analysis, and other fuel characteristics will need to be reanalyzed to ensure that applicable limits in general design criteria from Appendix A of 10 CFR (GDC 10) [53] are met: “GDC 10 requires that acceptable fuel design limits be specified that are not to be exceeded during normal operation, including the effects of anticipated operational occurrences.”

The long-term irradiation characteristics of thorium fuel will be different from those of uranium fuel. With the presence of fertile thorium fuel, the depletion characteristics will change as different fission products accumulate in the system, and as fissile ^{233}U is produced. This could have an impact on the shutdown margin at different times during the irradiation period. The

buildup of fissile ^{233}U in thorium-fueled systems is different from the buildup of ^{239}Pu in uranium-fueled systems. In reactors with a high conversion ratio (near break-even breeding), the reactivity of the system could remain flat or even increase as a function of irradiation. The different fission products and actinides that will accumulate in the system will also potentially change the reactivity worth of burnable absorbers, soluble poisons, controls rods, and other reactivity control mechanisms.

The addition of thorium to the nuclear design will also impact characteristics after the reactor is shut down. The ^{233}U produced from the decay of ^{233}Pa, which has a half-life of 27 days, could impact temporary and intermediate fuel storage (in the spent fuel pool), as well as refueling operations. The refueling procedure or fuel storage options would need to account for the increase in fissile material that accumulates within the first months after discharge from the reactor.

Due to the different irradiation characteristics (depleted fuel isotopes, fuel performance, etc.), the primary reactivity coefficients will differ from those for typical uranium fuel. Reactivity coefficients are calculated and compared in Section 5 of this report. The change in reactivity coefficients will require an evaluation of reactor transients covered in Chapter 15 of the SRP. Due to the buildup of ^{233}U and the placement of possible blanket breeder and fissile fuel assemblies and fuel pins, power oscillation and core stability could be significantly different from those for typical uranium fuel systems. Little data exist regarding core stability for thorium fuel systems, but all basic design criteria in GDC 12 regarding power oscillations and stability will need to be met: “GDC 12 requires that the reactor core and the associated coolant, control, and protection systems be designed to ensure that power oscillations that result in conditions exceeding specified acceptable fuel design limits are not possible, or can be reliably and readily detected and suppressed.”

4.1.3 Thermal and Hydraulic Design (NUREG-0800, Section 4.4)

Section 4.4 of the SRP covers the thermal and hydraulic design aspects of the nuclear reactor, and typically refers to the reactor core and how it relates to maintaining fuel integrity during normal operating and during anticipated operating occurrences. This review has assumed that the reactors and fuel assemblies using thorium are of very similar design to current commercially operating reactors, so the thermal and hydraulic design of the plant is likely to remain largely unaffected by the addition of thorium to the fuel. However, some thorium designs have proposed a tight-pitch lattice that would significantly impact the hydraulic behavior for the fuel assemblies and would need further evaluation.

Although the fuel material has virtually no impact on critical heat flux (CHF) and critical power ratio (CPR), changes to the assembly design geometry have the potential to change the CHF or CPR. If there is a change, from I.1 of Section 4.4, the following will be needed: independent computer calculations to substantiate vendor analyses, correlations with experimental data to verify processes, and independent comparisons and correlations of data from experimental programs. Also important are the uncertainties in the CHF and CPR correlations, which could need to be updated for thorium fuels. Issues such as fuel densification and rod bowing will be different for thorium fuel systems, and these issues will need to be accounted for in the uncertainties due to the minimal experience with thorium fuels.

4.2 RADIOACTIVE WASTE MANAGEMENT (NUREG-0800, CHAPTER 11)

This chapter considers the sources of radioactivity, the form in which they appear (solid, liquid, or gas), the volumes and concentrations, and how they are managed. Because the use of thorium will generate a different radioactive inventory (fission products, transuranics and associated daughter products), the impact on the acceptance criteria has to be thoroughly evaluated. This includes consideration of the variation over time and, in particular, the impact of key gamma-emitting daughter products of ^{232}U, namely, ^{212}Pb, ^{212}Bi, and particularly ^{208}Tl, which emits 2.6 MeV gamma rays. Fission product yields, including changes in tritium yields, and impact on liquid effluents will need to be evaluated. Furthermore, the ability of the existing equipment (e.g., filters) to cope with any increase or change in source terms and volumes will also need reevaluation.

Without the level of operating experience that exists for uranium fuel, the ability to demonstrate the annual average dose in restricted and unrestricted areas for thorium fuels will prove difficult, and so a more conservative approach may have to be taken.

4.3 RADIATION PROTECTION (NUREG-0800, CHAPTER 12)

The focus of the review process in Chapter 12 is to ensure that the occupational radiation exposure is to satisfy the as low as reasonably achievable (ALARA) criterion. 10 CFR Part 50 Appendix I provides the design objectives and limiting conditions to meet ALARA, including numerical values to be achieved. To achieve ALARA, a review of the radiation sources, radiation protection design features, and the protection program is required. As with Chapter 11, the source term of the fuel (including fission products, transuranics and associated daughter products) is the notable difference for thorium fuels. Again, the nuclear data, calculation tools, methodologies and uncertainties will need to be reviewed and reevaluated.

As noted above, there is very little operating experience with thorium fuels. Therefore, the ability to draw upon operating experience for this fuel type and its associated source term is limited. Therefore, there is likely to be greater emphasis on the uncertainties and ability to predict the inventories more accurately, building in conservatisms where appropriate.

The protection program related to 10 CFR 19.12 (instruction to workers) is also important to ensure that any changes in operations due to the use of thorium fuel are communicated and shared with employees, including any specific precautions or procedures to minimize exposure to radiation (e.g., additional high-energy gammas, additional necessary shielding, and ingrowth of gamma-emitting materials such as ^{208}Tl with time).

4.4 TRANSIENT AND ACCIDENT ANALYSIS (NUREG-0800, CHAPTER 15)

Chapter 15 focuses on the analysis of reactor safety implications of transient and accident conditions including anticipated operational occurrences (AOOs), design basis accidents/events (DBAs/DBEs), and some beyond design basis accidents/events (BDBAs/BDBEs) such as an

Anticipated Transient Without Scram (ATWS). Given the direct impact of fuel design changes on these safety analyses, satisfying the requirements in nearly all sections of Chapter 15 are expected to be impacted by the use of thorium fuel to some extent, although some sections will be impacted more extensively than others. A more detailed analysis, examining Chapter 15 and all associated regulatory guides, would be needed in the future for any licensee applying to use thorium fuel.

Assessments of fuel integrity or failure in this chapter depend greatly upon factors directly impacted by the use of thorium fuel such as power and temperature distributions within fuel assemblies and fuel pins, fuel swelling during irradiation, fission gas release from the fuel kernel into the rod plenum, and delayed neutron and reactor kinetics parameters. Underlying phenomena for these factors include thermo-mechanical properties of thorium fuels (e.g., heat transfer coefficients and melting temperature), microstructure evolution of thorium fuel pellets, fission product production and decay, and thermo-chemical diffusivity of various fission product species in thorium fuel. In addition, factors such as neutron cross-section data and decay heat generation data indirectly impact the analyses and have potentially large consequences.

Several different paths will need to be pursed in order to address the impacts of thorium fuel on Chapter 15. Some issues will require new analyses using available data, such as power distribution calculations. New experimental data will be needed for some physical phenomena in order to address issues like fission gas release and fuel swelling models. The uncertainty analyses required for Chapter 15 will necessitate quantification of various types of uncertainties, including neutron cross sections and material properties, and possibly reducing the uncertainties in these parameters in order to be able to satisfy acceptance criteria. Finally, more basic issues involving model and code validation will require new separate effects and integral tests including possible transient testing of thorium fuels in a facility such as the TREAT, which is no longer operational. In addition, some computer codes used for transient analysis may need extensive modification of existing capabilities or the addition of new capabilities to model the phenomena associated with thorium fuels; a similar situation to this occurred in the past regarding the need for transient analysis for the utilization of MOX fuels in LWRs.

Several representative issues identified in Section 15 provide illustrative examples and highlight some key findings. First, at a very basic level, a requirement stated in Section 15.0 indicates that "Fuel cladding integrity will be maintained if the minimum DNBR remains above the 95/95 DNBR limit for PWRs." This requirement for the determination of fuel failure, along with the other acceptance criterion "The calculated maximum fuel element cladding temperature shall not exceed 2200 degrees F" for a loss of coolant accident (LOCA) response, is dependent on the fuel design and the fuel thermal response, which will be directly affected by the use of thorium. These acceptance criteria form the basis for the reviews of all the Chapter 15 accidents and transients. A second example is that Appendix K of 10 CFR 50 sets the information requirements for the analysis of a LOCA. Several of the Appendix K requirements (e.g., power and temperature distributions and internal pin pressure) directly depend upon the fuel. Appendix K and Regulatory Guide 1.157 provide the foundation for acceptance of LOCA analyses that are reviewed in Chapter 15.6.5.

4.5 SEVERE ACCIDENTS (NUREG-0800, CHAPTER 19)

Chapter 19 focuses on probabilistic risk assessments (PRAs) and severe accident analyses. As with Chapter 15, a thorough reevaluation of Chapter 19 would be required for any applicant to use thorium fuels in existing LWRs due to the changes introduced by a different fuel material. The primary impacts of thorium fuels on the areas covered by Chapter 19 appear to be in the underlying regulatory guides and other documents rather than in the SRP itself; a preliminary assessment of the sections of Chapter 19 revealed no significant issues with any existing wording being applied to thorium fuels, but rather identified a broad and fundamental impact on the analyses being reviewed. Additional analyses would be needed, experimental data would be required, computer codes would require modification and validation, and uncertainty analyses would need to be performed.

Similar to Chapter 15, the analyses covered by Chapter 19 deal with transient and accident conditions; however, they extend beyond situations where fuel must remain intact and maintain acceptable failure levels in severe accidents where bulk fuel failures and core degradation are expected. Fundamental properties such as volumetric swelling, heat transfer coefficients, and the diffusivity and release of fission products in thorium fuels remain important, as do factors such as reactor kinetics parameters. Due to the nature of the accidents, however, other fundamental aspects of thorium fuels and behaviors of the integral system take on new and added importance. Fuel melting temperature, fission gas release, radiological source terms, and information regarding temperature- and composition-dependent eutectics that form during core degradation scenarios are all important to severe accident analysis, and each of them would likely require new experimental data for the development and validation of reliable models for the analysis of thorium fuels. Experimental data for fuel melt using thorium fuels, temperature- and burnup-dependent data for fission product inventory and fission gas release, fuel swelling data, and integral testing in transient and accident conditions (similar to Chapter 15 but with more extreme conditions) would all be needed. Uncertainty analyses covered by Chapter 19 would require quantifying the uncertainties in existing data and possibly new work to reduce some of these uncertainties to more reasonable or desirable levels. Computer codes used for severe accident analysis, such as MAAP [54] or MELCOR [55], would require data for their underlying models and subsequent verification and validation work to ensure they can reliably and accurately predict and model the evolution of severe accident scenarios for LWRs using thorium fuels and thus qualify them for such use. These codes are primarily used for the analysis of uranium-fueled LWRs, although some code versions are currently qualified for the analysis of MOX fuels as well.

In addition to the severe accident analyses in this chapter, an emphasis is placed upon the role and validity of PRAs. Adequate justification of the use and validity of PRAs with thorium fuels would likely require new work to quantify existing uncertainties associated with thorium-specific parameters relevant to transient and accident analyses (e.g., thermo-mechanical properties and neutron cross sections) as well as to generate probability distributions for various parameters required for PRA analysis.

5 QUANTITATIVE ASSESSMENT OF THORIUM FUEL IN LWRS

In order to gain some basic understanding of the operating characteristics of thorium-based LWR fuels, a large number of calculations were performed using SCALE 6.1 [3] with the 238-group ENDF/B-VII library for a typical LWR reactor fuel assembly using four different fuel compositions. As a basis for comparison, results were generated for uranium oxide (UO_2, or UOX) and mixed oxide (UO_2 + PuO_2, or MOX). Equivalent fuel compositions were then generated for uranium/thorium oxide (UO_2 + ThO_2, or U-Th) and plutonium/thorium oxide (PuO_2 + ThO_2, or Pu-Th). Here, the term "equivalent" is defined to be approximately equal in terms of lifetime-averaged reactivity (LAR) and, therefore, energy generated. Using the equivalent fuel compositions and the reference UOX and MOX fuel compositions, a number of different analyses were performed that included reactivity coefficients (fuel temperature, moderator temperature, boron), pin power peaking factors, assembly power sharing, boron letdown, and controlled lattice reactivity.

The results in this section represent a preliminary analysis of thorium-based fuels by way of comparison with conventional fuels and are not intended to serve as a basis for licensing reviews. Note that the results in this section should be viewed as indicative of likely behavior for these fuel types rather than a definitive statement of how they will perform due to the fact that the calculations performed for this work were assembly lattice calculations rather than full-core analyses. Assembly design optimization, core loading pattern optimization, three-dimensional effects including heterogeneity of fuel assembly burnup levels and heavy metal composition (e.g., UOX assemblies next to thorium-based fuel assemblies with differing ratios in the quantity of each type of assembly), and various other processes could impact the results. Nevertheless, the results in this section provide useful insight into some of the general trends and issues associated with comparison of the four fuel types studied.

5.1 FUEL DESIGN ASSUMPTIONS

Some basic assumptions required to perform the needed analyses are presented in this section. Additional assumptions that were required for the various analyses performed are given in the applicable sections.

Fuel Composition

The chosen fuel compositions represent a small set of the options available for thorium usage in LWR fuel. Fuel compositions were chosen based on likely near-term application of thorium in LWR reactor fuel. Other options exist, including mixed UO_2, PuO_2, and ThO_2 fuels; non-oxide fuel forms including metallic fuels; and plutonium isotopic vector variations (reactor grade or weapons grade). The full extent of the fuel options could not be covered under the scope of this study, but similar analyses could be performed for these other thorium options should their applications become more likely.

Fuel Assembly

A typical Westinghouse 17×17 fuel assembly design was used that is common in operating PWRs in the United States. Other fuel assembly types are commonly used in operating reactors, and the results for those other assembly types could vary from results generated in this study. Other reactor types, such as BWRs, could have significantly different characteristics with respect to thorium due to variations in neutron flux spectra.

For this preliminary study, it was assumed that all fuel pins in the lattice were of a single fuel type (i.e., all fuel pins in the lattice contain only one of the four previously determined fuel compositions). Mixing of fuel pin types in a single lattice was not analyzed but could be a feasible design choice for thorium-bearing LWR fuel. It could be preferential to have thorium located along the edges of the fuel assembly, near guide tubes, or in some other location in the lattice for use in a seed-blanket configuration. In addition, no burnable absorbers, for example, integral fuel burnable absorbers (IFBAs), wet annular burnable absorbers (WABAs), or gadolinia, were used in the assemblies, although they are necessary in modern PWRs. The goal of the analysis was not to generate an optimal fuel design or core loading pattern using thorium but to determine preliminary operating characteristics of potential fuel types.

Modeling

The majority of the analyses in the following sections were performed using single-assembly two-dimensional (2D) models that assume the fuel assembly lies in an infinite array of identical fuel assemblies that are also infinite in the axial direction. However, an actual reactor experiences leakage, variation in assembly characteristics, and other heterogeneous effects. Single-assembly modeling assumes that fuel assemblies are located next to other identical fuel assemblies. However, fuel assemblies in an operating reactor are placed adjacent to assemblies with differing burnups (i.e., operating histories) and initial fuel loadings, which lead to variations in flux, power, neutron flux spectra, and other parameters. It would also be feasible to construct a core of mixed fuel assembly types (e.g., UOX + U-Th fuel assemblies) with varying numbers of different assemblies. In order to evaluate these options, full-core modeling would be required. Some basic calculations (2×2 colorset calculations) have been performed in order to determine likely power sharing factors for a mixed core, but a full evaluation using a 3D core simulator has been deferred to future work.

5.2 ASSEMBLY DESIGN AND DETERMINATION OF EQUIVALENT FUEL COMPOSITIONS

For the analysis, a Westinghouse 17×17 fuel assembly design with publicly available dimensions [56] was chosen. A ¼-assembly model with uniform fuel grading (all fuel pins have the same initial fuel type and content) was then constructed in SCALE/TRITON, as illustrated in Figure 5.1. The large circles represent guide tubes. The different colors for fuel rods indicate groupings of fuel rods for depletion analysis; the groupings account for variation in the neutron flux and flux spectrum across the assembly and are based on the rod positions within the assembly.

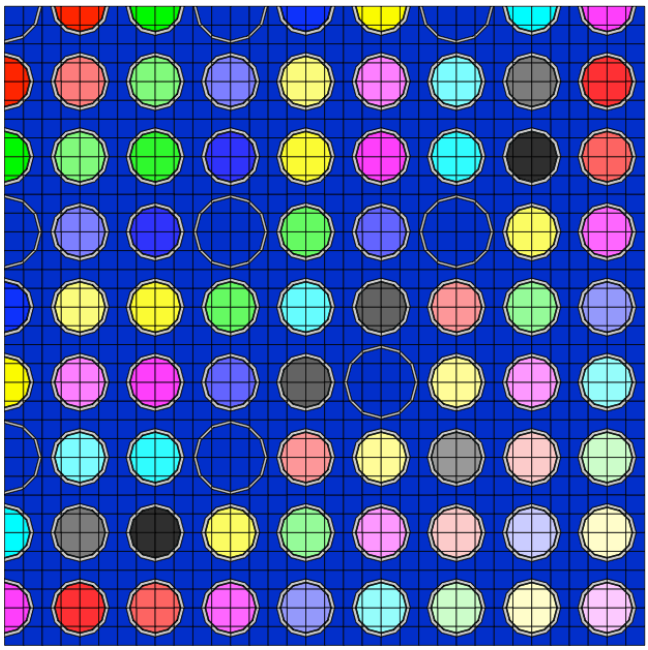


Figure 5.1. Quarter-assembly model of a Westinghouse 17×17 fuel assembly.

Prior to performing calculations, four fuel compositions were chosen for analysis that represent near-term fuel candidates to be considered in the United States: UOX, MOX, U-Th, and Pu-Th, respectively. A literature survey was conducted in an attempt to determine equivalent fuel compositions, in terms of LAR and energy generated, for the four fuel types. One reference provided true equivalent fuel compositions for UOX (4.0 wt% enriched) and MOX (8.0 wt% reactor-grade PuO_2) determined by full-core cycle analysis [57]. The 4.0 wt% enriched UOX fuel was used as a basis for comparison with other fuel types throughout this study. Other literature suggested relevant fuel compositions for U-Th and Pu-Th [58, 59], but it was unclear that the fuel compositions used in those studies were truly equivalent to 4.0 wt% UOX. True equivalent fuel compositions are typically determined by performing in-core fuel management and core design analysis with a full-core model. A simpler approach, the LAR method, was applied with an end-of-life (EOL) target of 48.5 GWd/MTHM in order to generate U-Th and Pu-Th fuel compositions equivalent to the UOX and MOX fuel compositions determined from the literature survey. In the LAR method, the reactivity of the model is integrated between beginning of life (BOL) and EOL and then divided by the total burnup to give an "average" reactivity over the life of the fuel lattice. The fuel composition is adjusted until the LAR of a certain assembly matches the UOX reference solution.

For the Pu-Th fuel, it was assumed that the plutonium isotopic vector in PuO_2 was identical to that used in the MOX fuel (reactor-grade plutonium). The PuO_2 and ThO_2 ratios were adjusted until the LAR matched that of UO_2, which yielded 9.0 wt% PuO_2 with a balance of ThO_2 for Pu-Th fuel.

For the U-Th fuel, determination of the equivalent compositions was slightly more complicated. In the U-Th fuel, there were two degrees of freedom: the fraction of UO_2 and ThO_2 and the ^{235}U enrichment. Excessively large fractions of UO_2 were required for low ^{235}U enrichments due to the increased capture of neutrons in thorium as compared to ^{238}U. For this reason, fuel

enrichment was increased to the maximum allowable for LEU (20.0 wt% ^{235}U) and the fraction of UO_2 and ThO_2 was adjusted until an acceptable fuel composition was found. The resulting fuel contained 26.0 wt% UO_2 and 74.0 wt% ThO_2. The detailed fuel composition and final isotopic vector for each fuel type is listed in Table 5.1.

Eigenvalue trajectories for the four equivalent fuel compositions can be found in Figure 5.2 with the fuel types noted in the plot legend. Note that the UOX infinite multiplication factor (k-inf) at EOL is lower than that for the other three fuel types, because of the higher rate of conversion in the other fuel types: creation of ^{233}U in the thorium fuels and ^{239}Pu in the MOX and Pu-Th fuels. The trajectories of the thorium fuel types are flatter, with lower BOL k-inf and higher EOL k-inf values. This characteristic could be advantageous for several reasons. Lower BOL k-inf would require lower soluble boron hold-down at beginning of cycle (BOC), could reduce rod worth requirements or burnable poison (BP) loading, and possibly make the moderator temperature coefficient (MTC) at BOC more negative (depending on the exact impact on soluble boron).

Table 5.1. Detailed fuel composition information

		UOX		MOX		Pu-Th		U-Th	
		Wt% of Isotope	*Fraction of All Heavy Metal*	*Wt% of Isotope*	*Fraction of All Heavy Metal*	*Wt% of Isotope*	*Fraction of All Heavy Metal*	*Wt% of Isotope*	*Fraction of All Heavy Metal*
UO_2	**^{234}U**	0.035%	**100%**	0.002%	**92%**			0.199%	**26%**
	^{235}U	4.000%		0.300%				20.000%	
	^{236}U	0.018%		0.001%				0.092%	
	^{238}U	95.947%		99.697%				79.709%	
PuO_2	**^{238}Pu**			2.460%	**8%**	2.460%	**9%**		
	^{239}Pu			54.690%		54.690%			
	^{240}Pu			26.160%		26.160%			
	^{241}Pu			9.510%		9.510%			
	^{242}Pu			7.180%		7.180%			
ThO_2	**^{232}Th**			0.000%		100.000%	**91%**	100.000%	**74%**

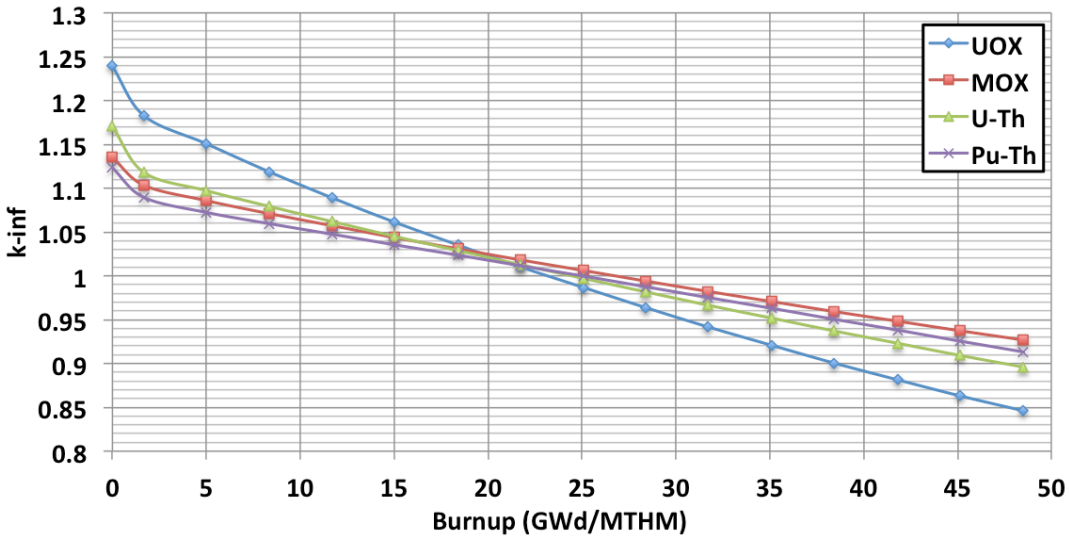


Figure 5.2. Eigenvalue trajectories for equivalent fuel compositions.

5.3 REACTIVITY COEFFICIENT CALCULATION MATRIX

Using the previously generated equivalent fuel compositions and associated isotopics as a function of depletion, SCALE/TRITON steady-state input files were generated at burnup values corresponding to BOL, near beginning of life (NBOL), middle of life (MOL), and EOL – 0.0, 1.7, 25.1, and 48.5 GWd/MTHM, respectively (16 total input files, four for each fuel type).

A matrix of state conditions was developed that would be used for the reactivity coefficient calculations. Chosen fuel temperatures span 300–2400 K and allow direct comparisons to corresponding continuous-energy KENO calculations (i.e., chosen temperatures are available on the continuous-energy cross-section libraries). The span of fuel temperatures is expected to cover nominal, startup, shutdown, and transient conditions. Four moderator temperatures with corresponding moderator densities between 566 K and 614 K were used, which span nominal operating conditions. Lower moderator temperatures (e.g., 300 K) were considered but have been excluded from the case matrix, as the low temperatures occur only during outage, startup, and shutdown. Showing low moderator temperatures on the same plot as the operating temperatures (566–614 K) compresses the operating temperature range to a point where it is difficult to determine the overall trends. Four soluble boron concentrations ranging from 0–2400 ppm were chosen to span possible operating conditions from BOC to EOC. In order to determine controlled lattice reactivity, calculations were performed for both "control rods in" and "control rods out" configurations (B_4C control rods were assumed). A summary of the state conditions for the reactivity coefficient case matrix is as follows:

Fuel temperature (T_f, K): 300, 900, 1200, 1500, 2100, and 2400;

Moderator temperature (T_m, K): 566 (0.7426 g/cm^3), 583 (0.7073 g/cm^3), 600 (0.6641 g/cm^3), and 614 (0.6160 g/cm^3);

Soluble boron concentration (C_b, ppm): 0, 600, 1200, and 2400;

Control state (CR): out, in.

Completion of the reactivity coefficient matrix resulted in more than 2000 SCALE/TRITON input files and corresponding calculations, which were completed using SCALE 6.1.2. Relevant data, including infinite multiplication factors and pin power distributions (peaking factors), were extracted from the SCALE/TRITON output files. The infinite multiplication factor data were used to generate reactivity coefficient plots over the range of conditions tested. A second-order polynomial was fitted to the eigenvalue data in order to smooth minor variations in the results that otherwise led to erroneous variations in the reactivity coefficient data. The polynomials were then used to generate the reactivity coefficient data and associated plots.

In this section, and throughout the document, differences in reactivity have been calculated in pcm (percent mille or 1.0E-5) using the following equation:

$$\Delta\rho = \frac{k_1 - k_2}{k_1 k_2}.$$

Reactivity coefficients (RTCs) have been calculated as $\Delta\rho$ divided by the change in the parameter of interest (P), as follows:

$$RTC = \frac{\Delta\rho}{P_1 - P_2}.$$

5.3.1 Normal PWR Operating Conditions (T_f = 900 K, T_m = 583 K, and C_b = 600 ppm)

This section presents reactivity coefficients for typical PWR conditions: T_f = 900 K, T_m = 583 K, and C_b = 600 ppm. In Figure 5.3, the moderator temperature and boron concentration were held constant, while the fuel temperature was varied to obtain the Doppler coefficient.

The Doppler coefficients (Figure 5.3) for MOX and thorium fuels are typically more negative than that of UOX for the selected state points. In the Pu-Th and U-Th fuels, the Doppler coefficients are considerably more negative than those for UOX and MOX. This characteristic is likely due to the presence of two large temperature-sensitive capture resonances in ^{232}Th at ~20 and ~50 eV, as seen in the ^{232}Th neutron capture cross section shown in Figure 5.4. The Doppler coefficient for MOX is more negative than that of UOX due to the presence of strong thermal capture resonances in ^{239}Pu, ^{240}Pu, and ^{242}Pu. The BOL flux spectra for the four fuel types can be found in Figure 5.5. In addition to showing the impacts that large low-energy thorium and plutonium resonances have on the flux spectra, this figure indicates significant spectral hardening effects resulting from the use of thorium and plutonium. UOX has the highest thermal neutron flux peak of the four fuels. Replacing some of the uranium in the fuel with thorium, as occurs when going from UOX to U-Th fuel, results in a nearly 45% reduction of the thermal neutron flux. Both plutonium-bearing fuels (MOX and Pu-Th) exhibit nearly identical thermal neutron flux levels that are about 85% lower than UOX and about 70% lower than U-Th. The lower thermal flux levels of MOX and Pu-Th fuels are accompanied by corresponding increases in fast flux levels. At fuel temperatures above 2000 K, the Doppler coefficients shown in Figure 5.3 are similar for all fuel compositions because the harder spectra at high temperatures remove the impact of the low-energy capture resonances in plutonium and thorium. The Doppler coefficients of all four fuel types become more negative as the fuel burnup level increases.

Figure 5.6 displays the calculated neutron flux spectra for U-Th and Pu-Th fuels at BOL and EOL. With increasing burnup, the flux spectra for both U-Th and Pu-Th soften; this differs from UOX fuel, where the neutron flux spectrum is known to harden with increasing burnup due to the depletion of ^{235}U and the buildup of ^{239}Pu. Figure 5.6 also shows that Pu-Th maintains a lower thermal neutron flux than U-Th for all burnup levels considered and that the neutron spectra of both fuels notably soften during irradiation.

For the results shown in Figure 5.7, the fuel temperature and boron concentration were held constant to obtain the moderator temperature coefficient (MTC). The MTCs for plutonium- and thorium-bearing fuels are more negative than that for UOX for the selected state points at BOL and NBOL. An increase in moderator temperature leads to a decrease in moderator density, and thus a hardening of the neutron spectra. The harder neutron spectra result in more absorption in the resonance region and more negative MTCs. Due to the large low-energy capture resonances in plutonium, the MTCs for fuel types containing plutonium (MOX and Pu-Th) are more negative at lower burnups. However, the Pu-Th and U-Th spectra in Figure 5.6 soften with increasing burnup. As a result, the MTCs for Pu-Th and U-Th become less negative than that for UOX at EOL. The MTC for MOX is more negative than that for UOX throughout the tested burnup range due to the low-energy capture resonances. All four fuel types exhibit increasingly negative MTCs as burnup increases, though not by the same amount.

In all of the analyses described above, boron concentrations were assumed to be the same in each case (600 ppm); however, in a real core or in a full core analysis, the critical boron concentration will vary depending upon the fuel type and burnup. Note that the MTC will be much less negative for higher boron concentrations. When the moderator temperature increases, the decrease in the moderator density reduces the soluble boron density. The decreased boron density results in less boron absorption, which increases the fuel reactivity and partially offsets the negative reactivity due to the reduction in the moderator density.

The boron reactivity results shown in Figure 5.8 were obtained by holding the fuel and moderator temperatures constant while varying the boron concentration in the coolant to obtain the boron concentration coefficient of reactivity (i.e., boron worth). For all cases, the boron reactivity coefficients remain fairly constant as the boron concentration increases, though uranium fuels show a somewhat parabolic shift above ~1200 ppm wherein the boron worth becomes less negative as boron concentration increases. The boron worth is significantly smaller for MOX and Pu-Th fuel because boron is primarily a thermal neutron absorber and plutonium fuels exhibit substantially reduced thermal neutron fluxes as discussed previously. All four fuels exhibit increasing (more negative) boron worth as a function of increasing burnup.

The data in the reactivity coefficient plots have been summarized in Table 5.2. In general, the Doppler coefficients and MTCs of thorium fuels are more negative than those of UOX fuel, which should increase safety margins. Conversely, the smaller boron worth for Pu-Th fuel could result in the reduced effectiveness of boron-based absorbers and may require modifications (such as enriched boron) to maintain effectiveness.

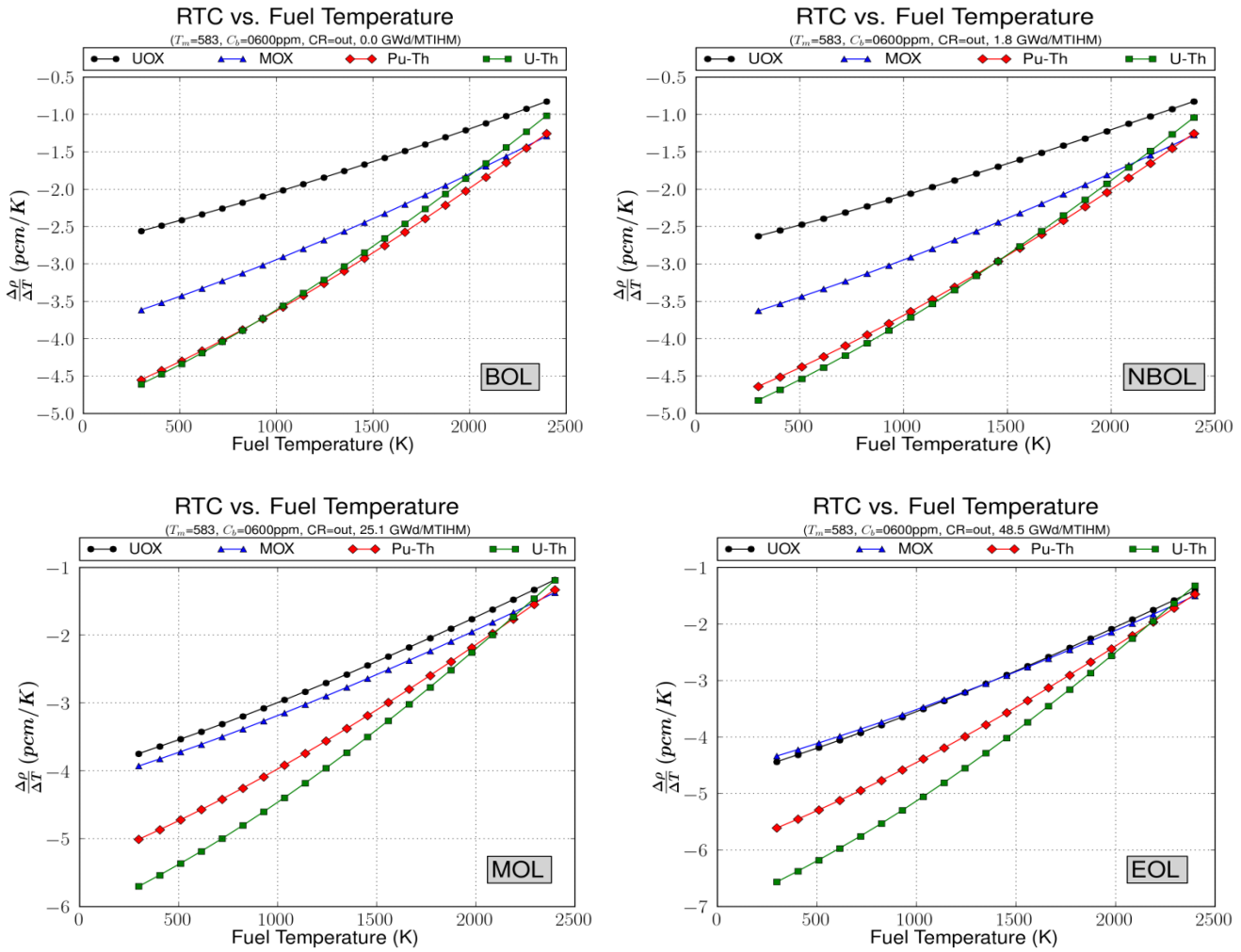


Figure 5.3. Doppler coefficients of reactivity for BOL, NBOL, MOL, and EOL at T_m= 583 K and C_b = 600 ppm.

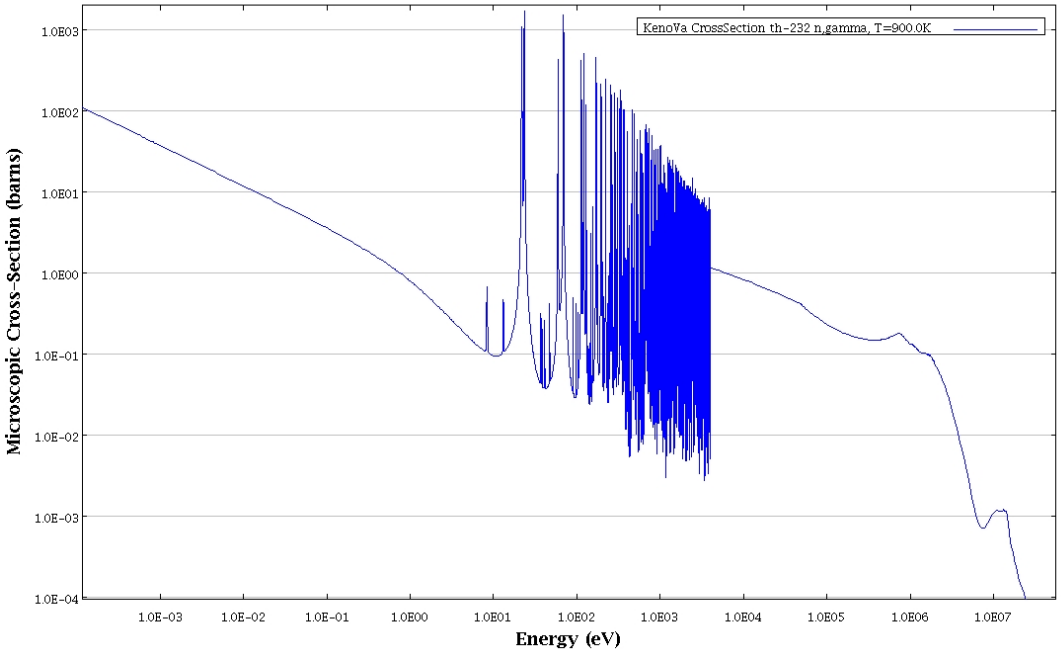


Figure 5.4. Neutron capture cross section for ^{232}Th, as reported by SCALE/TRITON.

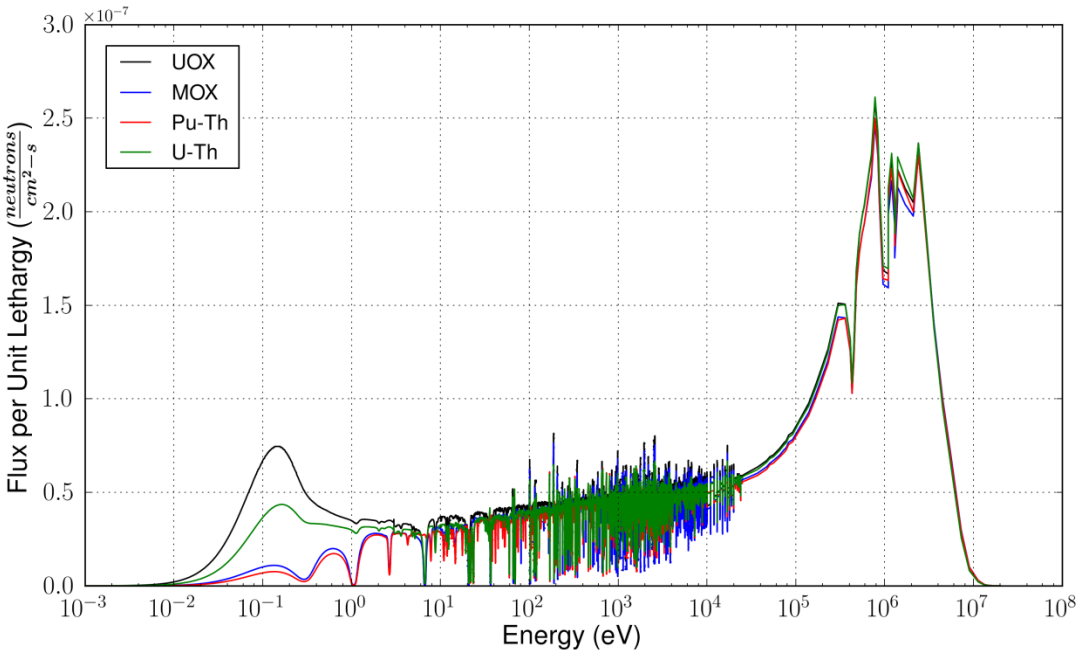


Figure 5.5. Flux spectra for UOX, MOX, Pu-Th, and U-Th pin cells at BOL.

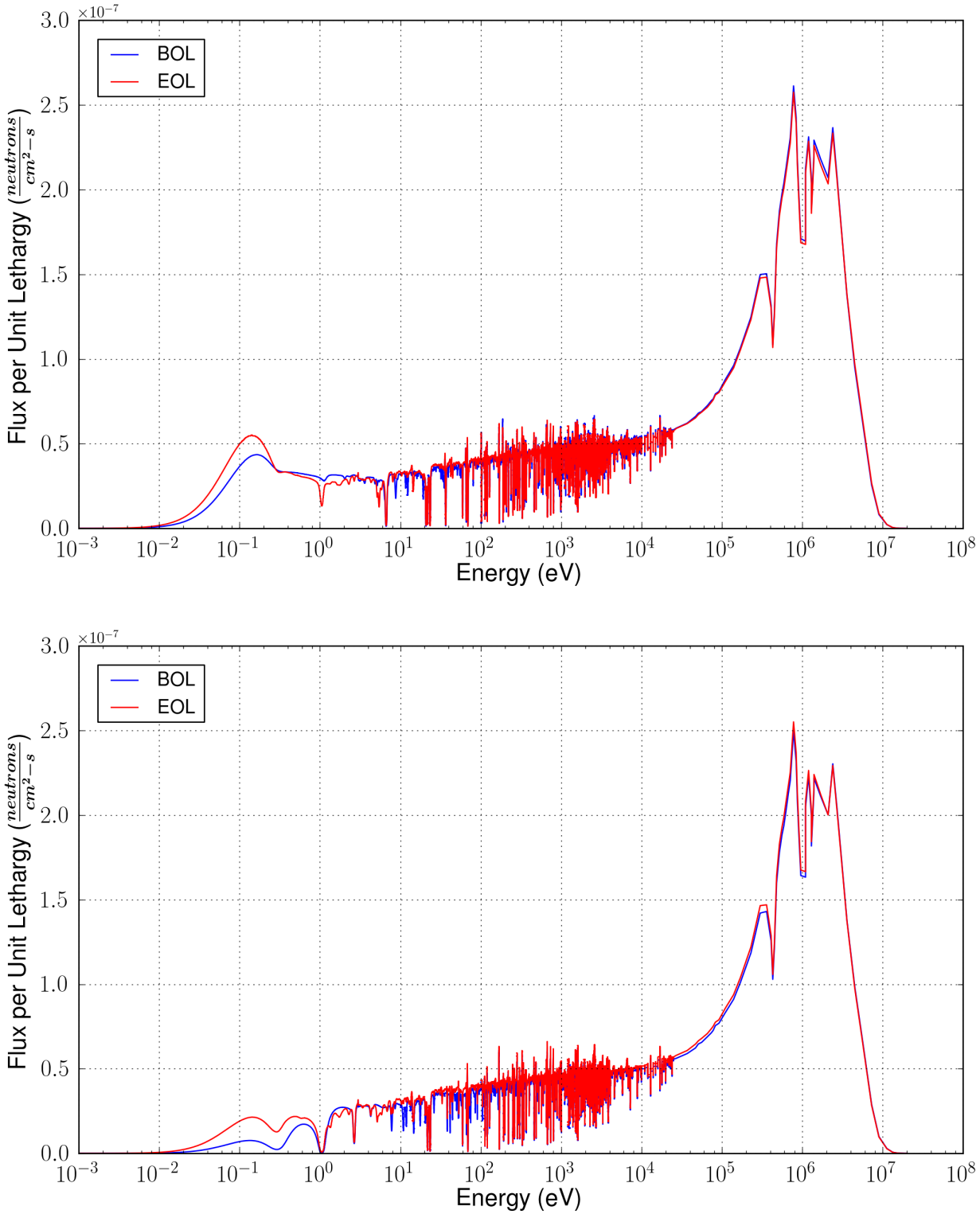


Figure 5.6. Neutron spectra for U-Th (top) and Pu-Th (bottom) for BOL and EOL.

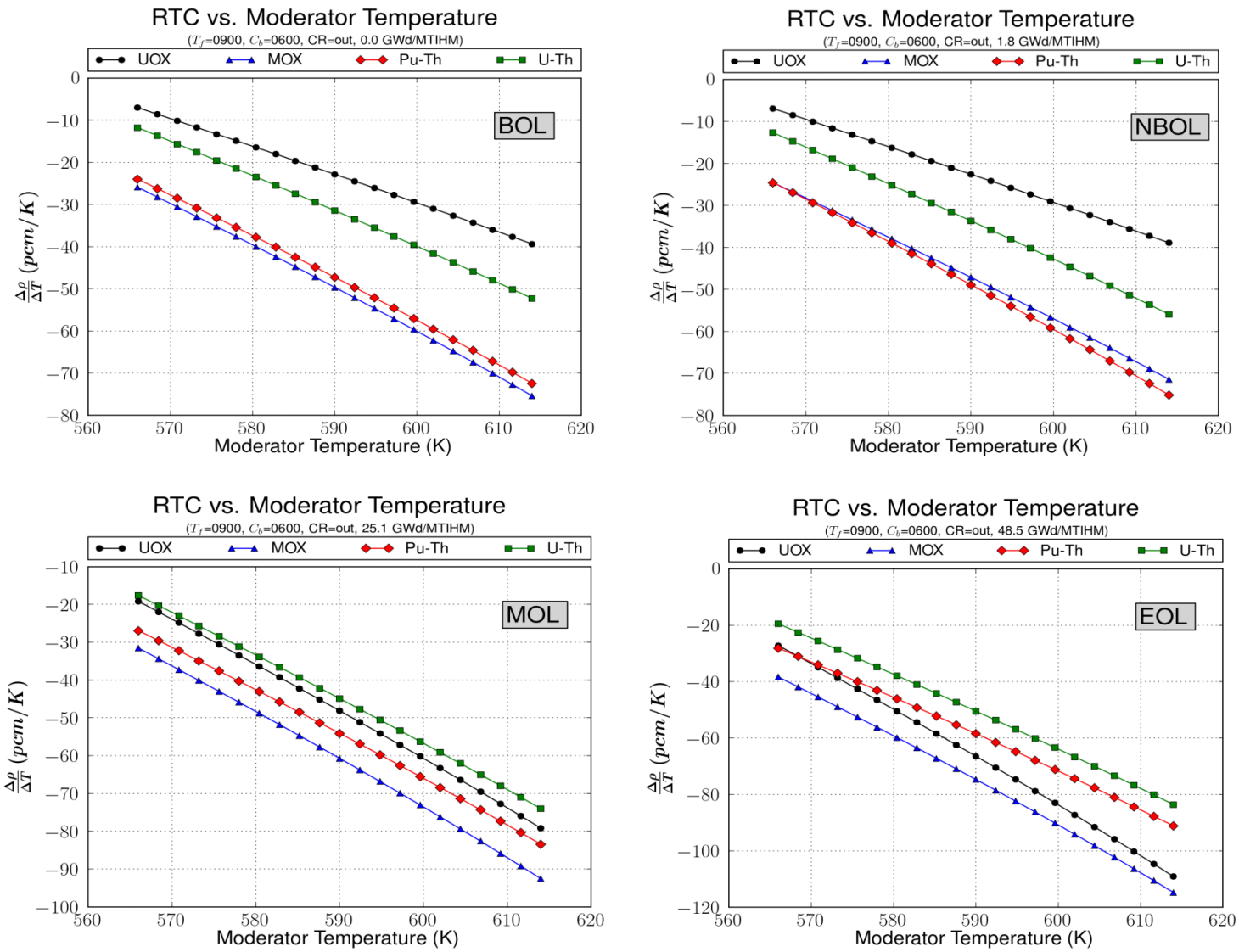


Figure 5.7. Moderator temperature coefficients of reactivity for BOL, NBOL, MOL, and EOL at T_f = 900 K and C_b = 600 ppm.

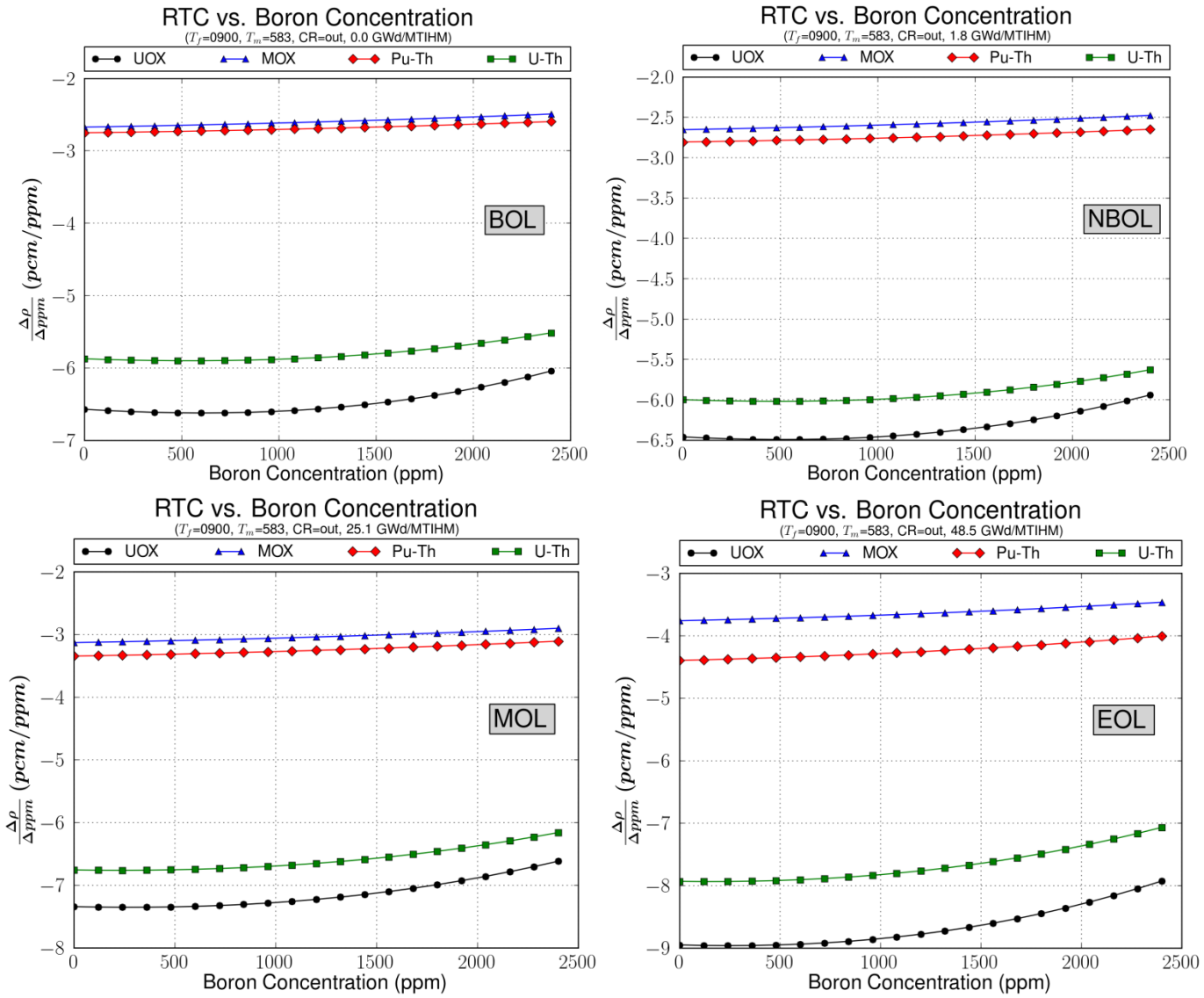


Figure 5.8. Boron coefficient of reactivity for BOL, NBOL, MOL, and EOL at T_f = 900 K and T_m = 583 K.

Table 5.2. Average reactivity coefficients over typical PWR conditions

	Average Fuel Temp. Coeff., $300 \leq T_f \leq 2400$ (pcm/K)			
	BOL	**NBOL**	**MOL**	**EOL**
UOX	-1.74	-1.77	-2.54	-3.02
MOX	-2.53	-2.53	-2.73	-3.02
Pu-Th	-3.03	-3.08	-3.31	-3.71
U-Th	-2.96	-3.09	-3.64	-4.18
	Average Moderator Temp. Coeff., $566 \leq T_m \leq 614$ (pcm/K)			
	BOL	**NBOL**	**MOL**	**EOL**
UOX	-23.01	-22.74	-48.50	-67.10
MOX	-50.04	-47.46	-61.19	-75.40
Pu-Th	-47.62	-49.25	-54.49	-58.91
U-Th	-31.68	-33.94	-45.24	-50.91
	Average Boron Coeff., $0 \leq C_b \leq 2400$ (pcm/ppm)			
	BOL	**NBOL**	**MOL**	**EOL**
UOX	-6.48	-6.35	-7.15	-8.67
MOX	-2.60	-2.58	-3.04	-3.64
Pu-Th	-2.69	-2.74	-3.25	-4.24
U-Th	-5.81	-5.92	-6.59	-7.68

5.3.2 High Fuel Temperature Operating Conditions ($T_f = 2400$ K, $T_m = 583$ K, and $C_b = 600$ ppm)

The reactivity coefficients for very high fuel temperatures are nearly identical to those for nominal operation conditions. The MTCs for elevated fuel temperatures have been plotted in Figure 5.9. All cases show a slight decrease in the moderator temperature coefficient (more negative) with an increase in fuel temperature, indicating that the sensitivity to the moderator density is greater at higher fuel temperatures. This is expected due to broadening of capture resonances in the fuel. The MTCs for thorium-based fuels are more negative than those of UOX, except at EOL. The MTCs for all fuel types at EOL are strongly negative, so there is no safety concern there.

The boron reactivity coefficients for elevated temperatures are also very similar to those for normal operating fuel temperatures (see Figure 5.10) – the boron reactivity coefficients are nearly constant as a function of boron concentration, and UOX has a more negative boron reactivity coefficient than the other tested fuel types.

Again, it is worth noting that some of these lattice-calculated values are unrealistic since critical boron concentrations at EOL will be approaching zero, and not 600 ppm. The actual values can only be determined in a full core analysis.

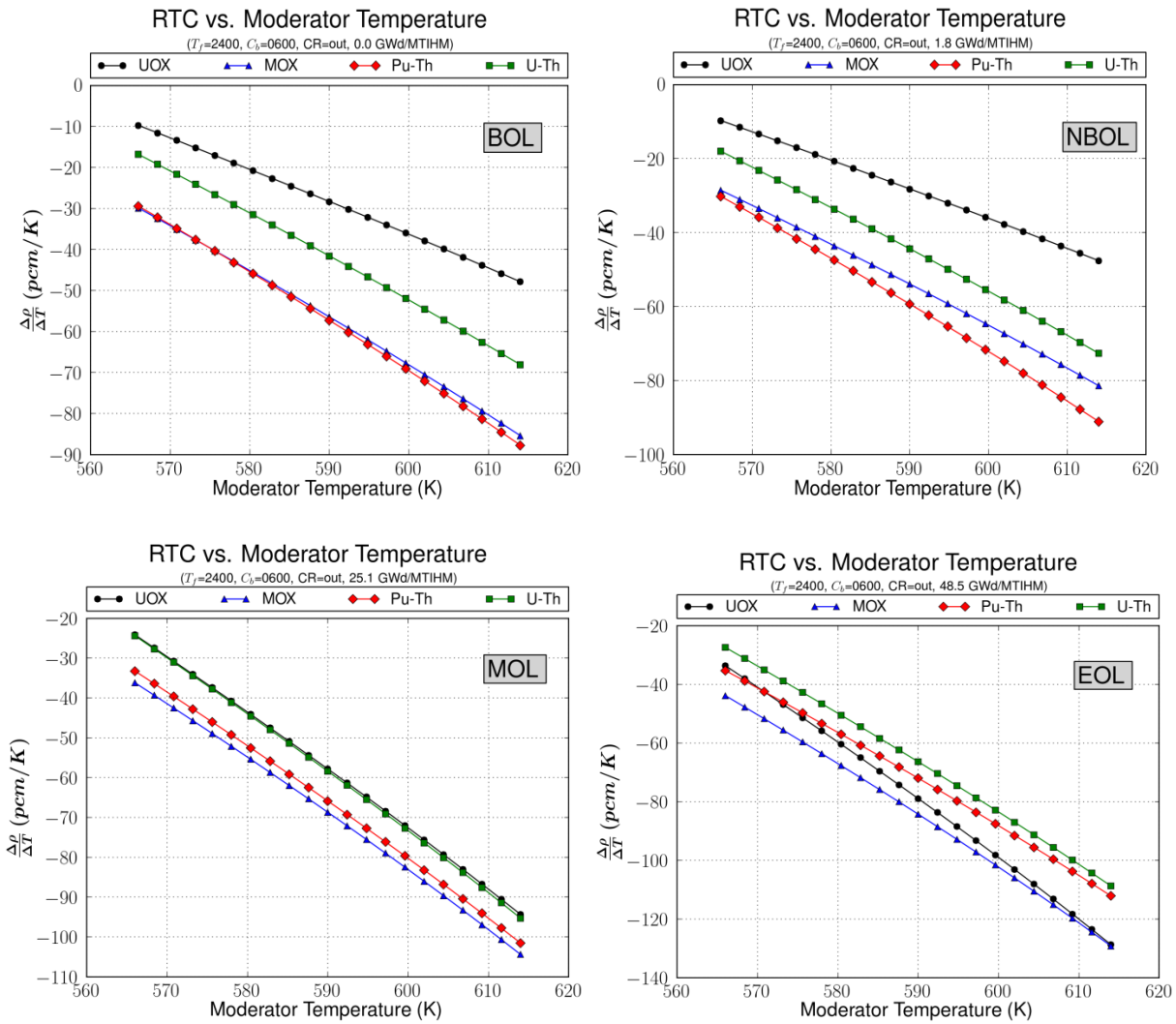


Figure 5.9. Moderator temperature coefficients of reactivity for BOL, NBOL, MOL, and EOL at Tf=2400 K and C_b=600 ppm.

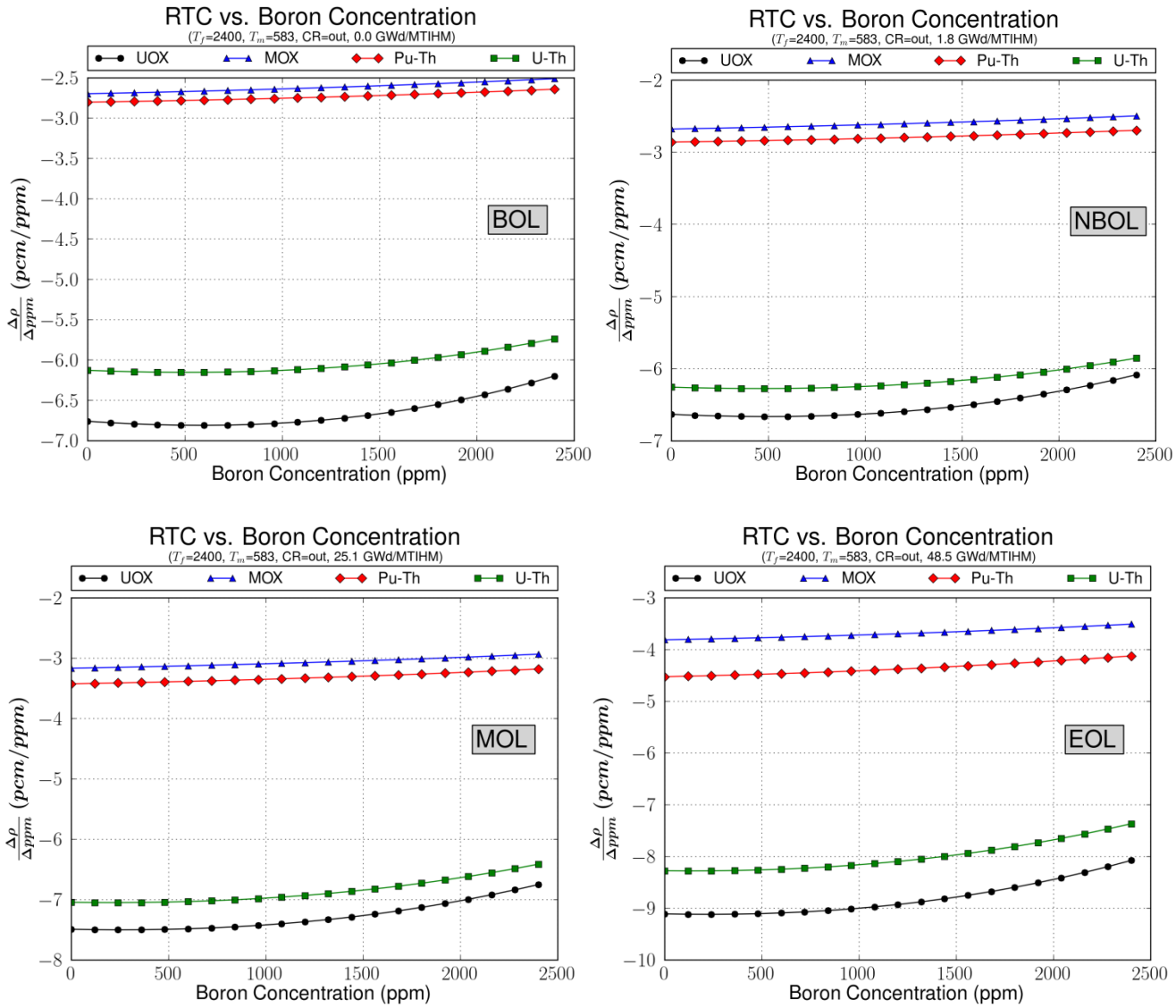


Figure 5.10. Boron coefficient of reactivity for BOL, NBOL, MOL, and EOL at T_f =2400 K and T_m=583 K.

5.3.3 High Boron Concentration Operating Conditions (T_f=900 K, T_m=583 K, and C_b=2400 ppm)

Like high-fuel-temperature reactivity coefficients, high-boron reactivity coefficients are similar to those for normal operating conditions. The Doppler coefficient of reactivity for high boron concentrations is plotted in Figure 5.11. In general, the Doppler coefficient is more negative for high boron concentrations. As with normal operation conditions, the Doppler coefficient for fuels containing thorium is larger in magnitude than that for fuel not containing thorium.

The MTC for higher boron concentrations, however, does show differences when compared to normal operating conditions. The MTCs for the high-boron conditions are plotted in Figure 5.12. The MTC is positive for UOX at BOL and NBOL. At MOL and EOL for UOX, the coefficient is positive for low moderator temperatures and negative for high moderator temperatures. Similar behavior is observed for U-Th fuel, with positive coefficients for low moderator temperatures and negative coefficients for high moderator temperatures throughout the assembly life. However, for the fuel types containing plutonium, the MTC is negative throughout the assembly life. The difference in behavior is due to the sensitivity of the UOX and U-Th fuel to boron coupled with the reduction in boron that occurs with the change in moderator density with increasing moderator temperature. The reduction in boron, caused by the reduction in moderator density, has a stronger positive reactivity impact than the negative reactivity impact due to the reduction in moderator density, resulting in a positive MTC for the UOX and U-Th cases. However, since these critical boron concentrations are much higher than those seen in PWR operations at hot full power (HFP), this trend is not of concern under normal operations.

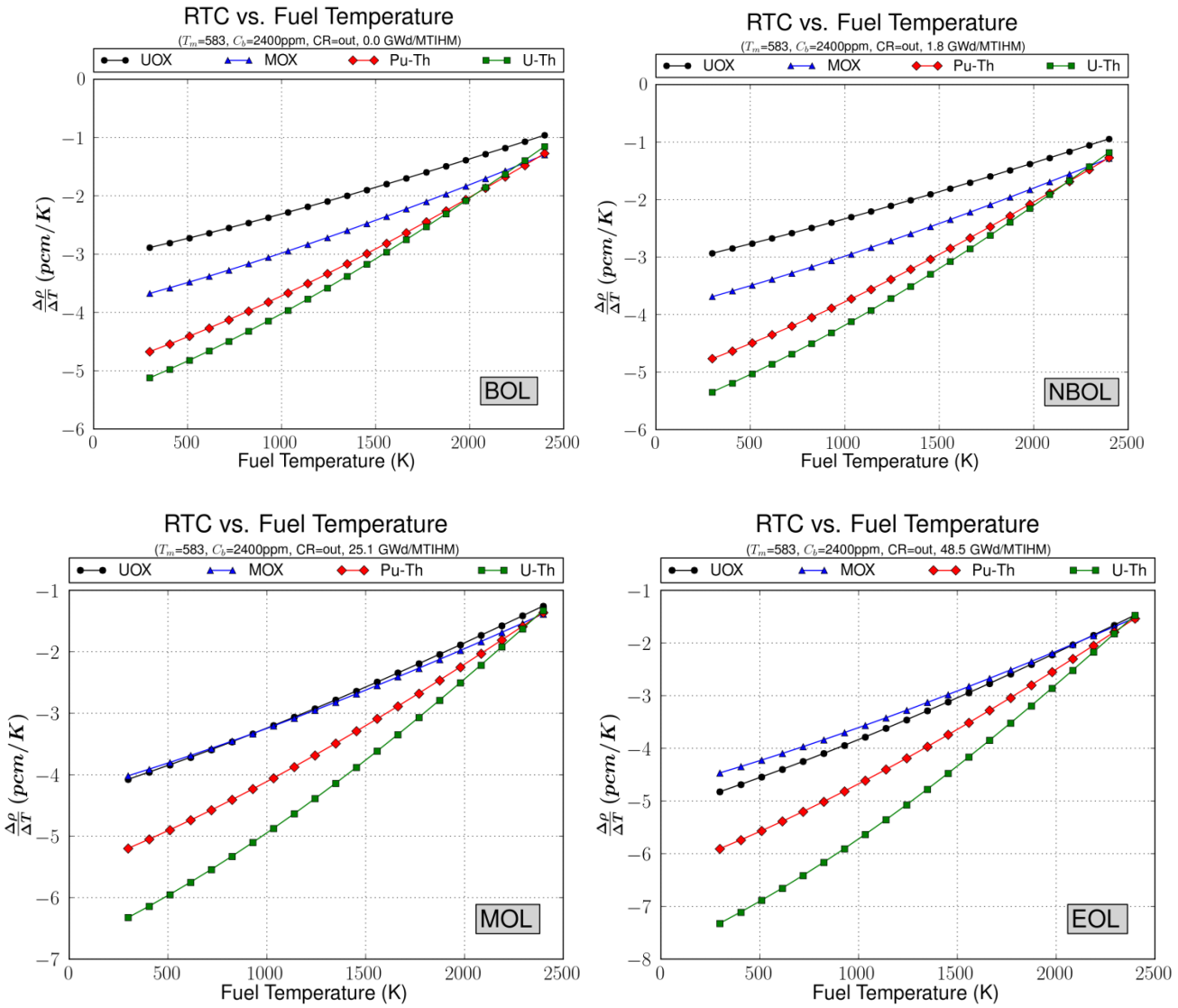


Figure 5.11. Doppler coefficients of reactivity for BOL, NBOL, MOL, and EOL at T_m = 583 K and C_b = 2400 ppm.

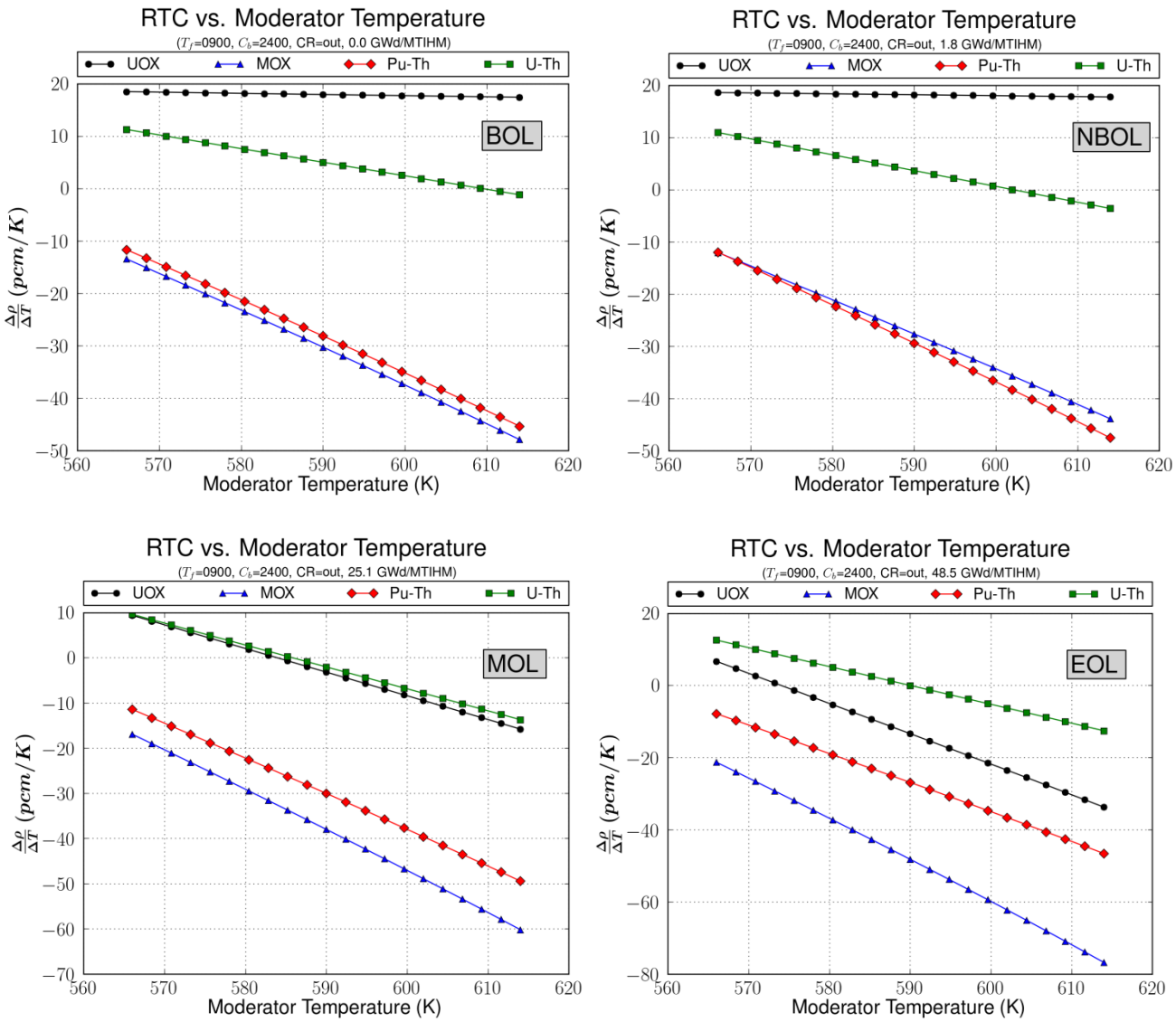


Figure 5.12. Moderator temperature coefficients of reactivity for BOL, NBOL, MOL, and EOL at T_f=900 K and C_b=2400 ppm.

5.4 CONTROL ROD LATTICE REACTIVITY

The single assembly models constructed for the reactivity coefficient test matrix were used to calculate control rod reactivities in a single-assembly lattice. Note that control rod worth is typically calculated in a core simulator, where the maximum rod worth for a certain core configuration can be obtained. In lieu of performing full core analyses, a subset of the input files used in the reactivity coefficient test matrix was modified to insert B_4C control rods into the guide tubes [56]. These input files were then used to calculate the infinite multiplication factors, which were compared to the data generated for the unrodded conditions simulated in the reactivity coefficient test matrix (i.e., rods-in compared with rods-out). These calculations of infinite lattices at low burnups (k-inf >> 1) and high burnups (k-inf <<1) produce results that would differ from full core rod worth calculations where fuel assemblies with varying enrichments and burnups are loaded, but they do provide an estimate of the control rod reactivity differences in the four fuel types compared in this report.

The controlled lattice reactivity as a function of burnup is plotted in Figure 5.13 for typical PWR operating conditions. For UOX, there is an increase in the reactivity worth of the control rods from BOL to EOL of more than 14,000 pcm; the increase in reactivity worth for MOX, Pu-Th, and U-Th is significantly less (~7,000–9,000 pcm). Comparing UOX and U-Th, the average reactivity worth over the life of the assembly is similar (~45,000 pcm), but the addition of thorium to the fuel tends to slightly increase the reactivity worth at BOL and decrease the reactivity worth at EOL. The shape of the reactivity worth curves for MOX and Pu-Th are very similar, with the Pu-Th fuel having a slightly greater reactivity worth as a function of burnup. Thus, the introduction of thorium itself does not appear to be the primary impact on control rod reactivity. The differences are primarily associated with the use of plutonium vs enriched uranium.

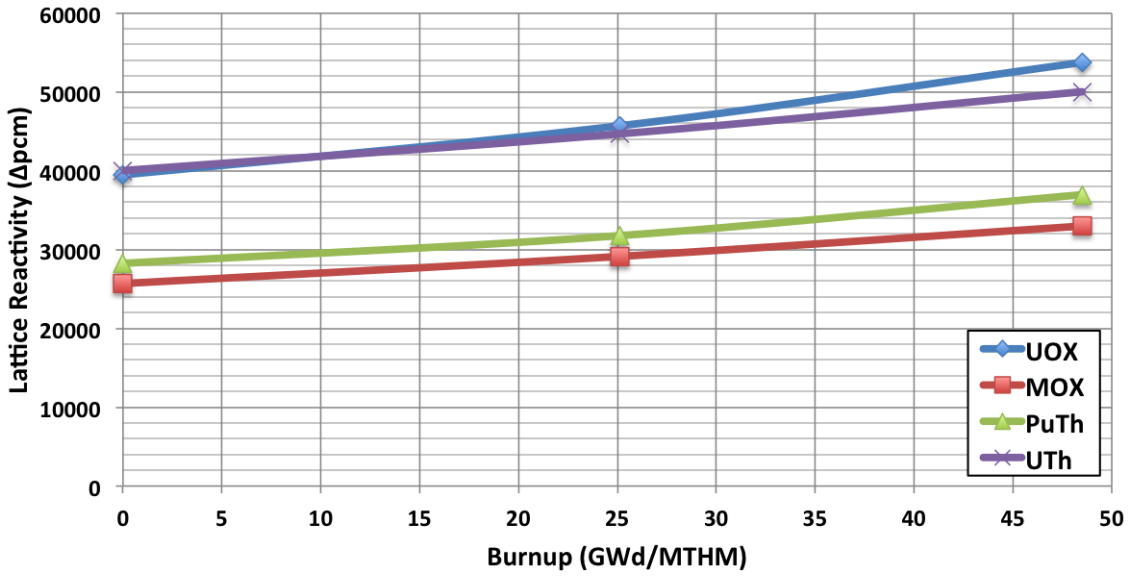


Figure 5.13. Control rod lattice reactivity vs. burnup at normal operating conditions.

Figure 5.14 shows the controlled lattice reactivity as a function of moderator temperature for BOL, MOL, and EOL. The previous trend of increasing reactivity worth as a function of burnup still holds, as well as the trend that values for the U-Th fuel composition are slightly higher than UOX at BOL and lower at EOL. In all cases, the reactivity worth increases as a function of increasing moderator temperature (and decreasing density), but the trend is strongest in UOX and U-Th fuels.

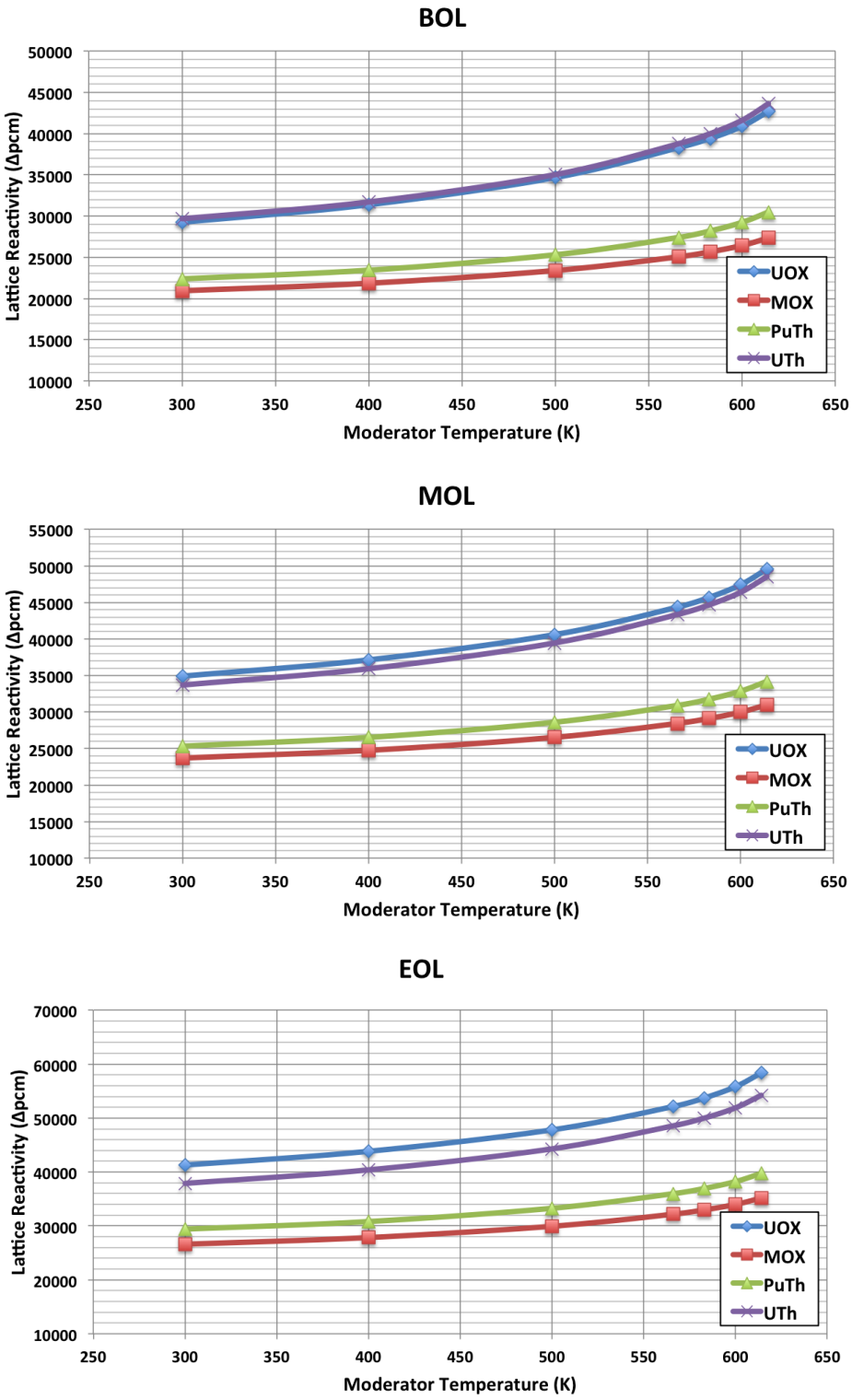


Figure 5.14. Control rod lattice reactivity vs. moderator temperature at BOL (top), MOL (middle), and EOL (bottom).

There is little correlation between controlled lattice reactivity and changing fuel temperature and changing boron concentration, so these plots have been omitted. The trend of increasing reactivity worth as a function of burnup is still observed for fuel temperature and boron concentration, as well as the trend that the increase in control rod reactivity as a function of increasing burnup is larger for the fuels that contain uranium (UOX and U-Th).

The B_4C controlled lattice reactivity studies have been summarized in Table 5.3. The reactivity worths have been averaged over typical operating conditions for PWRs. Table 5.3 also reflects the previous observation that the reactivity worths of fuels containing uranium generally increase at a greater rate as a function of burnup. Fuel materials containing plutonium have lower values over all tested conditions due to the hardened flux spectrum associated with plutonium-based fuel.

Table 5.3. Summary of single-assembly control rod lattice reactivity for B_4C rods

	Reactivity Worth (pcm)			
	UOX	**MOX**	**Pu-Th**	**U-Th**
BOL	40175	26044	28724	41159
MOL	46569	29545	32311	45542
EOL	54793	33473	37590	50973

In order to determine the corresponding controlled lattice reactivities for Ag-In-Cd (AIC) control rods, the B_4C used in previous calculations was replaced with AIC (80 wt% Ag, 15 wt% In, 5 wt% Cd). The results for typical PWR conditions are compared to unrodded conditions to generate lattice reactivity worths for AIC control rods. The AIC results are summarized in Table 5.4. The control rod reactivities for AIC rods are lower than those for B_4C rods, but the relative differences between traditional fuels (UOX and MOX) and thorium fuel types (Pu-Th and U-Th) are very similar to the cases with B_4C rods. As with B_4C control rods, the AIC reactivity worth for U-Th fuel type is slightly higher than for UOX at BOL, but lower at EOL. The results for MOX and Pu-Th AIC rods show trends similar to those for B_4C control rods – a nearly constant difference is observed as a function of burnup with the Pu-Th fuel having a slightly greater reactivity worth at each point.

Table 5.4. Summary of single-assembly controlled lattice reactivities for AIC rods

	Reactivity Worth (pcm)			
	UOX	**MOX**	**Pu-Th**	**U-Th**
BOL	29084	16811	18187	29154
MOL	33220	19207	20451	31777
EOL	39453	22129	24297	35667

5.5 CRITICAL BORON CONCENTRATION

In order to generate the critical boron concentration, a number of assumptions were required. Typically, critical boron concentration is calculated by the core simulator based on a current core design that contains a mix of fuel assemblies, each with a unique total burnup and irradiation history. Most PWRs operate on an 18-month, three-batch fuel cycle leading to a core that is roughly one-third fresh fuel, one-third once-burnt fuel, and one-third twice-burnt fuel.

The critical boron concentration for a reflected single-assembly model was estimated as a function of burnup. Using the previously calculated boron reactivity coefficients, a reasonable first approximation of the reflected single-assembly critical boron concentration was made. Using the calculated k_{inf} values for the first approximation of the boron concentration and the deviation from critical (k_{inf} = 1.0) as a function of burnup, an updated single-assembly critical boron concentration could be generated. The procedure is represented by the following equations:

$$\text{while } |k_{\text{inf},i}(t) - 1.0| \geq 0.00010;$$

$$\begin{cases} \Delta C_{b,i}(t) = \frac{k_{\text{inf},i}(t) - 1.0}{W_b} \\ C_{b,i+1}(t) = C_{b,i}(t) + \Delta C_{b,i}(t) \end{cases}$$

where

$k_{\text{inf},i}(t)$ is the calculated infinite multiplication factor at as a function of time for attempt *i;*
$C_{b,i}(t)$ is the boron concentration as a function of time for attempt *i*;
W_b is the boron worth averaged over all conditions; and
$C_{b,i+1}(t)$ is the boron concentration as a function of time for the next attempt.

This procedure was then iterated until the single-assembly critical boron concentration was converged such that $1.0 - k_{inf} \leq 10$ pcm from 0 to 45 GWd/MTHM.

Using the reflected single-assembly critical boron concentration, a linear trend between burnup and critical boron concentration was then assumed. The reactor was assumed to contain one-third fresh fuel (0 GWd/MTHM at BOC), one-third once-burnt fuel (15 GWd/MTHM at BOC), and one-third twice-burnt fuel (30 GWd/MTHM at BOC), leading to a discharge burnup of 45 GWd/MTHM for the twice-burnt batch of fuel. The assumption that the EOC boron must be zero was then imposed, and the boron letdown curve was shifted accordingly. The final approximated critical boron letdown curves can be found in Figure 5.15.

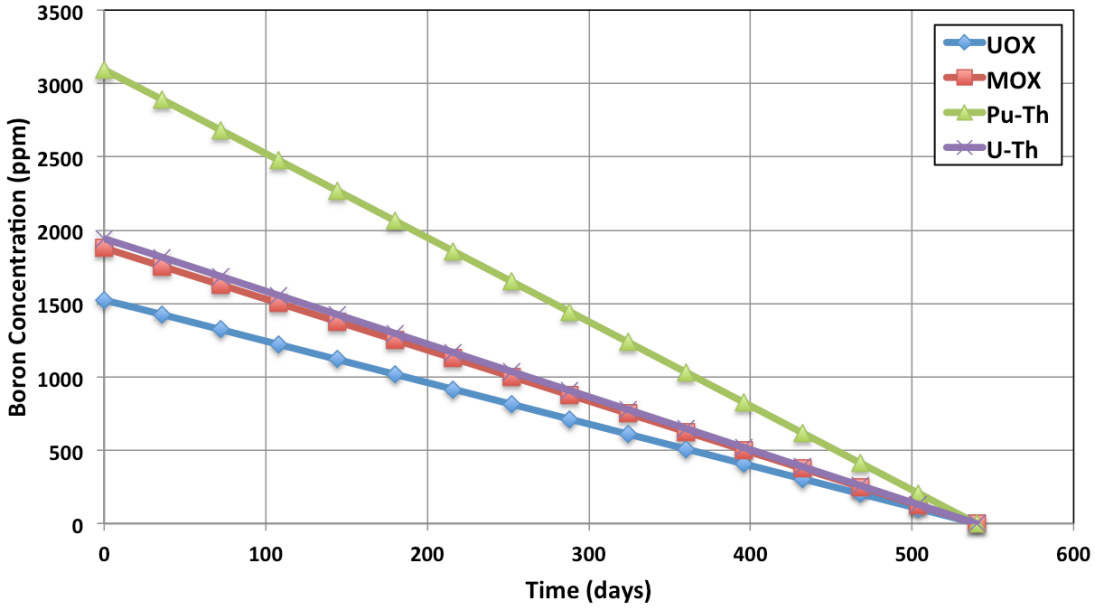


Figure 5.15. Approximate critical boron letdown curve for a full core of UOX, MOX, Pu-Th, or U-Th fuel assemblies.

The UOX BOC critical boron is ~1500 ppm, which is on the order of typical BOC critical boron concentrations in PWRs. From Figure 5.15, it is clear that adding thorium to a fuel type results in lower critical boron concentrations. The critical boron concentration for Pu-Th fuel is ~1700 ppm at BOC, which is ~200 ppm less than that for MOX. The approximate critical boron concentration at BOC for U-Th fuel is ~1200 ppm, which is ~300 ppm less than that for typical UOX fuel. These results are as expected from previous calculations of the eigenvalue as a function of burnup (Figure 5.2) and the boron worth for the different fuel types calculated in the reactivity coefficient analysis (Figure 5.8); U-Th fuel has a flatter eigenvalue trajectory and similar boron worth compared to UOX, so the critical boron concentration should be lower than UOX. Due to the reduced boron worth for MOX and Pu-Th fuels compared to the boron worth of UOX at BOL, observed differences in the critical boron concentrations are credible. Note that these boron letdown curves are merely first-order estimations. Full core analysis of realistic loading patterns that use burnable poisons to reduce excess reactivity and local power peaking is needed to more accurately assess the required critical boron concentrations.

5.6 PIN POWER DISTRIBUTIONS

For all cases in the reactivity coefficient test matrix, the pin power distributions were extracted and used to determine the fuel pin peak power. A sample pin power distribution for T_f=900 K, T_m=583 K, and C_b=600 ppm can be found in Figure 5.16 for BOL, MOL and EOL. As can be observed in Figure 5.11, the lattices containing plutonium result in higher power peaking near the guide tubes due to the increased thermal fission cross section of ^{239}Pu compared to that of ^{235}U. In the thorium lattices, U-Th and Pu-Th, the peaking factors are slightly higher than for corresponding non-thorium lattices (UOX and MOX). In all cases the power peaking decreases as a function of burnup.

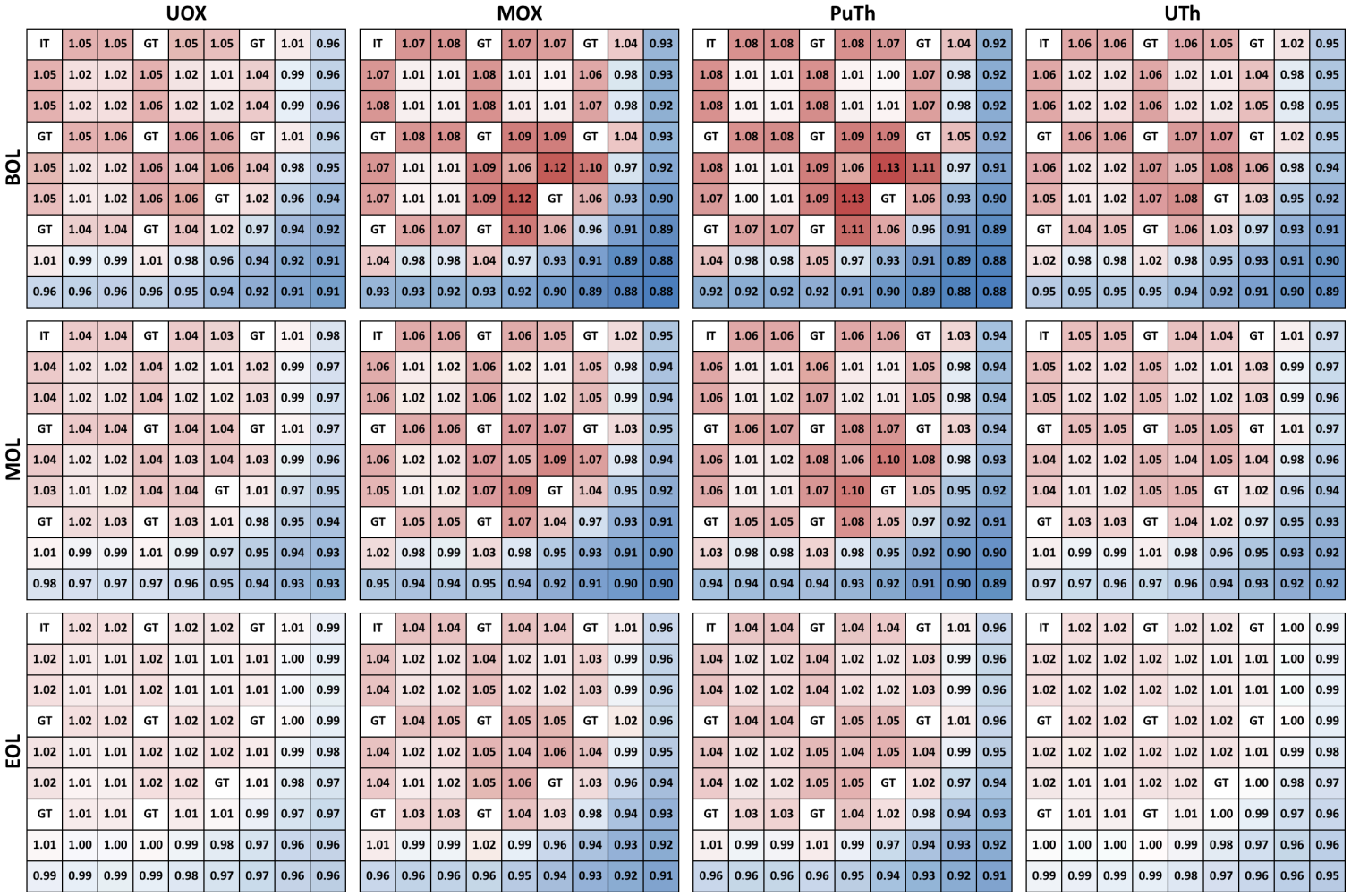


Figure 5.16. Pin power distribution for BOL, MOL, and EOL at T_f=900 K, T_m=583 K, and C_b=600 ppm.

The average maximum pin peaking factors are 1.052 for UOX, 1.097 for MOX, and 1.094 for Pu-Th and 1.062 for U-Th. Due to the difficulty in viewing the large amounts of pin power data, the fuel pin peak power data for all cases studied here have been plotted using a histogram style plot in Figure 5.17. The data in this figure have been broken into two subcategories: fuel composition (left) and time in life (right). As was observed in the pin power distributions for the normal PWR operating conditions, generally, the fuels containing thorium result in approximately the same or slightly higher maximum pin peaking factors than their non-thorium counterparts. The peaking is more dependent on whether the fuel contains plutonium rather than thorium. The right-hand plot shows that the pin peaking factors decrease as a function of increasing burnup for all fuel types.

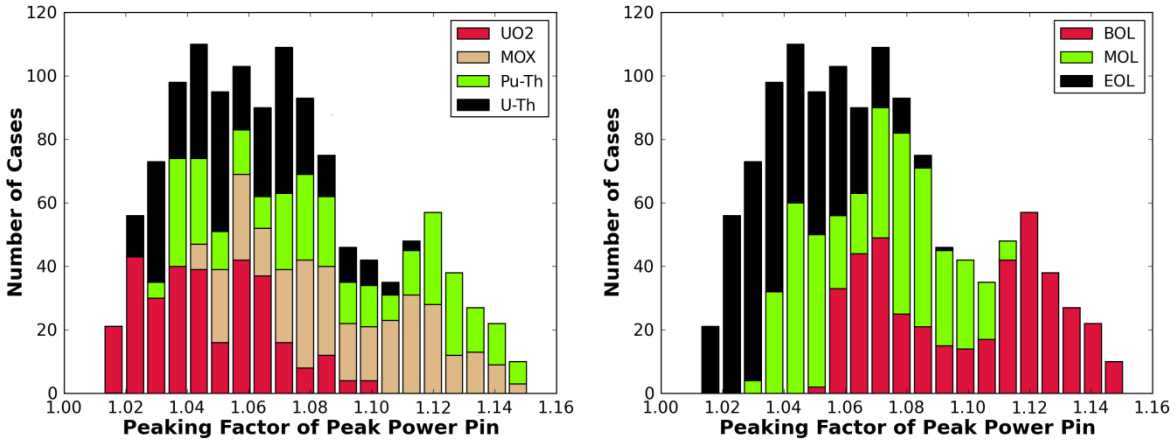


Figure 5.17. Histogram of fuel pin peaking factor data.

5.7 FUEL ASSEMBLY POWER SHARING

In lieu of full core calculations, SCALE/TRITON input files that represent a 2×2 ¼-assembly layout were generated. In these cases, the fuel is loaded in a checkerboard pattern with two UOX fuel assemblies located diagonally across from each other in the northwest (NW) and southeast (SE) corners. The other two locations were both filled with another fuel type (UOX, MOX, U-Th, or Pu-Th). UOX-UOX cases are provided as a reference. An example figure of the 2×2 layout, along with a plot of the fast and thermal flux distribution in the model, can be found in Figure 5.18. BOL UOX assemblies are in the NW and SE corners (shaded yellow), while MOL UOX assemblies are in the northeast (NE) and southwest (SW) corners (shaded fuchsia). The colors in the flux distributions in Figure 5.18 range from red (highest flux) to blue (lowest flux).

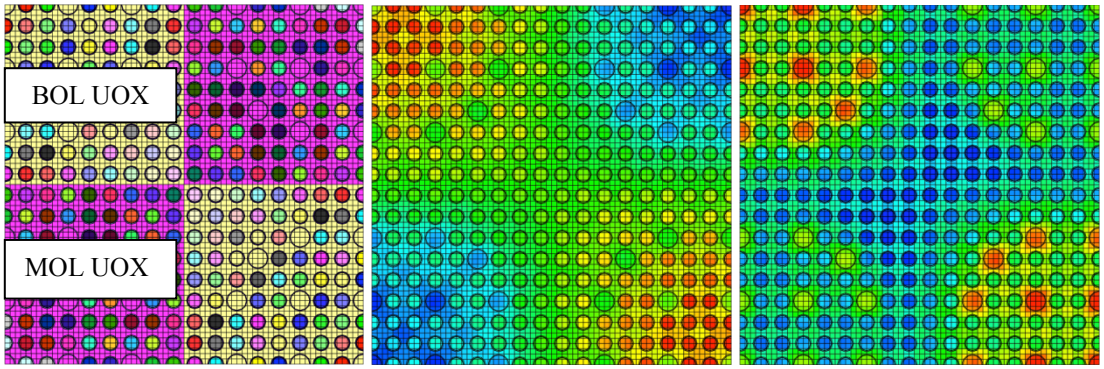


Figure 5.18. SCALE/TRITON representation of the 2×2 model (left), fast flux distribution (middle), and thermal flux distribution (right).

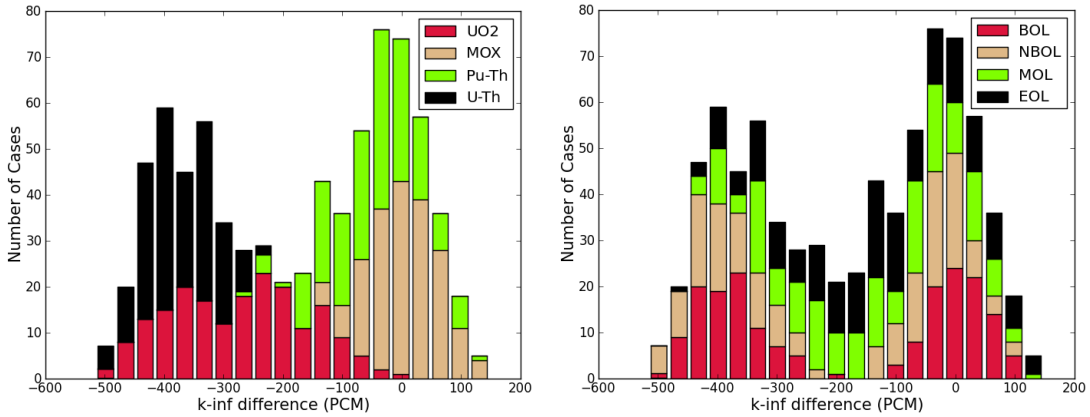
The fuel assembly power sharing and the maximum peaking factor have been extracted from the 2×2 assembly results and provided in Table 5.5. The table presents the data as the power fraction in the assembly of interest (row headings of Table 5.5) followed by the max peaking factor in square brackets. The data in the table facilitate comparisons of a core containing only UOX with cores consisting of UOX/MOX, UOX/Pu-Th, and UOX/U-Th. From the data in Table 5.5, it can be concluded that for nearly every case, the pin power peaking (noted in brackets in the table) is greater when MOX or Pu-Th fuel assemblies are present. In some cases, the fuel assembly power sharing is actually flatter for the MOX and Pu-Th cases, but the peaking factor of the peak power pin is almost always greater. In order to mitigate these impacts, it is likely that some sort of burnable absorbers or reduced plutonium loading would be needed in order to reduce power peaking. For the UOX/U-Th lattices, the fuel assembly power sharing and pin peaking are similar to the values observed for UOX/UOX models.

It should be noted that the true effect of the power sharing and the resulting power peaking (of assemblies and pins) could only be determined once a viable core design has been produced and full core analyses completed.

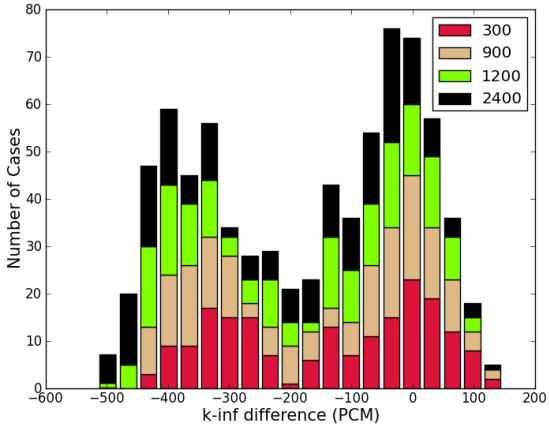
Table 5.5. Assembly power sharing and highest power peaking factor

	UOX BOL	UOX MOL	UOX EOL
UOX BOL	0.50 [1.06]	0.56 [1.22]	0.61 [1.36]
UOX MOL	0.44 [1.22]	0.50 [1.04]	0.55 [1.17]
UOX EOL	0.39 [1.36]	0.45 [1.17]	0.50 [1.02]
MOX BOL	0.51 [1.19]	0.57 [1.26]	0.62 [1.35]
MOX MOL	0.48 [1.25]	0.54 [1.21]	0.59 [1.32]
MOX EOL	0.46 [1.32]	0.51 [1.17]	0.57 [1.27]
Pu-Th BOL	0.50 [1.21]	0.55 [1.23]	0.61 [1.32]
Pu-Th MOL	0.47 [1.27]	0.53 [1.18]	0.58 [1.28]
Pu-Th EOL	0.44 [1.33]	0.50 [1.13]	0.55 [1.20]
U-Th BOL	0.49 [1.10]	0.55 [1.19]	0.60 [1.31]
U-Th MOL	0.46 [1.20]	0.51 [1.08]	0.57 [1.19]
U-Th EOL	0.42 [1.29]	0.48 [1.11]	0.53 [1.09]

In addition to SCALE/TRITON calculations, SCALE/KENO continuous-energy (CE) calculations (using the SCALE 6.2 beta 1) were performed for cases in the reactivity coefficient matrix. A subset of 192 KENO-CE results were generated for each fuel type using the same isotopic concentrations as the 2D TRITON calculations to determine if any systematic biases exist in the eigenvalue predictions obtained using SCALE/TRITON. The subset of calculations covers normal PWR operating conditions at BOL, NBOL, MOL, and EOL. The differences from KENO-CE (in pcm) are plotted using a stacked-histogram plot in Figure 5.19. The number of cases for each fuel type is equal. The data have been plotted using different subcategories: fuel type, burnup, and temperature. The height of each bar is the total number of cases that fall into a certain bin, and the different colors represent the number of cases in each subcategory.



(a) fuel type subcategory (b) fuel burnup subcategory



(c) fuel temperature subcategory (with temperatures in Kelvin)

Figure 5.19. Differences in eigenvalue between SCALE/TRITON and SCALE/KENO-CE for fuel type, fuel burnup and fuel temperature subcategories.

In Figure 5.19(a), it can be observed that the negative bias cases are typically for the uranium-fueled cases, while cases containing plutonium have biases that cluster around 0 pcm. Figure 5.19(b) shows that at BOL and NBOL the bias is double peaked. However, as the fuel is depleted, the double peaked shape of the bias can no longer be observed for MOL and EOL cases due to increasing similarity in the depleted fuel compositions. Figure 5.19(c) shows that there is little systematic bias associated with changing fuel temperature. The biases shown in Figure 5.19 are consistent with biases observed in previous analyses and are not expected to be problematic if SCALE were to be used for safety analyses of the selected fuel types. Additionally, these biases are expected to decrease significantly with the release of SCALE 6.2, which contains an improved 252-group cross section library that has shown excellent results in early tests.

5.8 SUMMARY OF THORIUM-BASED FUEL EVALUATION IN LWRS

LWR 2D lattice analyses were performed using a Westinghouse 17×17 fuel assembly design to gain a basic understanding of the neutronic behavior of thorium-based fuels (U-Th and Pu-Th) versus UOX and MOX fuels in current LWRs. The analyses included reactivity coefficients; controlled lattice reactivity to estimate rod worth; estimation of critical boron concentrations; and 2D power peaking (fuel pins and assemblies).

The Doppler and moderator temperature coefficients were generally more negative than those for UOX fuel, except for MTCs at EOL. For all burnups, the values were sufficiently negative to cause no safety concerns for reactor operation.

For boron reactivity, the U-Th boron worth was approximately 1 pcm/ppm less negative than that for UOX fuel. This difference is probably small enough to have minimal impact on LWR safety analyses, but further investigation would be needed to address that issue. The boron worth for the plutonium-based fuels (MOX and Pu-Th) was significantly less than UOX and would certainly require additional safety analyses.

The reactivities for B_4C and AIC control rods in a 17×17 assembly were very similar for UOX and U-Th fuel types at BOL, MOL, and EOL. For MOX and Pu-Th fuel types, the control rod lattice reactivities were similar to each other but much less than those for UOX fuel. This finding is similar to the soluble boron reactivity, the results of which are discussed in the previous paragraph. In both categories, thermal absorbers (soluble or solid) that are external to the fuel have less reactivity in the presence of plutonium-based fuels.

The estimated critical boron letdown curves showed that less soluble boron is needed for reactivity hold-down with thorium-based fuel types as compared to their counterparts (i.e., U-Th versus UOX, Pu-Th versus MOX). The reduced worth of soluble boron with plutonium-based fuels led to estimated critical boron concentrations for the MOX and Pu-Th fuel types that were several hundred ppm higher at BOC than for UOX and U-Th, respectively.

The peak pin power distributions for each fuel assembly type without any burnable poisons present showed that the peak pin powers in the U-Th fuel assembly were similar or slightly higher than those in the UOX assembly at all burnups. Similarly, the Pu-Th peak pin powers were similar or slightly higher than those in the MOX assembly. The plutonium-based fuel assemblies had peaking factors that were several percent higher than the UOX and U-Th fuel. These differences should be manageable with the use of burnable poison loadings in the fuel assemblies.

The fuel assembly relative power sharing, when placing two different fuel assembly types diagonally in a 2×2 fuel assembly lattice, showed that the power sharing factors between UOX and U-Th fuel assemblies were very similar for all burnup combinations. Once again, the behaviors of the MOX and Pu-Th fuels were very similar to each other when combined with UOX fuel assemblies. The assembly power sharing factors for MOX/UOX and Pu-Th/UOX

combinations were generally similar or flatter than UOX-UOX combinations, but the MOX and Pu-Th fuel assemblies still had higher pin peaking factors in most cases.

This initial study indicates that the use of U-Th fuel assemblies in LWRs is expected to have only a minor impact on lattice and core analyses. The behaviors of MOX and Pu-Th fuel assemblies appear to be similar to each other, but they show larger differences when compared to UOX fuel assemblies.

As noted previously, the assembly lattice calculations in this section represent a preliminary analysis of thorium-based fuels by way of comparison with conventional fuels and are not intended to serve as a basis for licensing reviews. The results presented indicate the likely behavior for these fuel types rather than provide a definitive statement of how they will perform. Full-core analyses would be needed to address assembly design optimization, core loading pattern optimization, three-dimensional effects including heterogeneity of fuel assembly burnup levels and heavy metal composition, and various other processes that could impact the results. Nevertheless, the results provide useful insight into some of the general trends and issues associated with comparison of the four fuel types studied and help identify possible safety and regulatory issues related to thorium fuel cycles using LWRs.

6 OUT-OF REACTOR CHARACTERISTICS OF THORIUM FUEL

The analyses performed in the previous section showed that for a once-through cycle with typical discharge burnups, the initial ^{235}U enrichment would need to be greater than the 5% limit currently used in LWRs and potentially could approach the 20% LEU limit (see Table 5.1) in order to provide sufficient fissile content to maintain a suitable power level throughout normal operation.

With regard to out-of-reactor issues prior to irradiation (handling, fuel manufacture, storage, etc.), the primary safety concern would be criticality safety limits for uranium enrichment beyond 5 wt%. The addition of thorium would likely have little impact prior to irradiation, because it is not fissile or highly radioactive in its natural state.

Thus, processes in the front-end of the fuel cycle would not be greatly impacted other than the issue of criticality safety limits for uranium enrichment beyond 5 wt%. However, issues in the back-end of the fuel cycle could arise for thorium-based fuels due to different decay heat and radiotoxicity characteristics associated with ^{232}Th irradiation. In the previous section, UO_2 (UOX) and MOX fuel were generally used as a basis of comparison for thorium-bearing U+Th and Pu+Th fuels, and that practice is continued here.

In order to generate the results for this section, the same input models used in the in-reactor analysis were used in the spent fuel analysis. The output files from the depletion cases were post-processed, and where applicable, additional decay calculations were performed using SCALE/ORIGEN in order to generate data as a function of time after discharge from the reactor. This analysis has been divided into three separate subsections, corresponding to depleted fuel isotopics, decay heat, and radiation source terms.

6.1 DEPLETED FUEL ISOTOPICS

In order to more effectively utilize thorium in reactors to breed fissile ^{233}U, there is a possibility that a utility would keep the fuel in the reactor for a much longer time than is typical for current U.S. LWRs. For this reason, the in-reactor depletion was extended to 100 GWd/MTHM in order to breed a sufficient amount of ^{233}U. Previously (e.g., Shippingport), thorium LWRs consisted of a "breeder-blanket" approach where only the blanket remains in the core for extended burnups. However, for simplicity in this analysis, all fuel is assumed to remain in the reactor for the entirety of the five selected burnup points: 25.1, 45.1, 60.2, 81.2, and 100.3 GWd/MTHM. These first two burnup points of 25.1 and 45.1 GWd/MTHM correspond to low and typical assembly-average discharge burnups observed in PWRs for UOX fuel. The higher burnup points are generally much higher than are typically observed in PWRs and should cover the range of fuel burnups that might be observed if an assembly was left in the reactor for more than three operating cycles. Note that this analysis does not consider the numerous fuel performance issues that could arise from such high burnup fuel, but does assume that the fuel could reach these high burnup levels in the reactor.

Depleted fuel masses (g/MTHM) for selected nuclides can be found in Table 6.1. Some of the data in Table 6.1 have been extracted and plotted as well. A plot of the ^{233}U production can be found in Figure 6.1. UOX and MOX fuels have been omitted from this plot because the ^{233}U production is virtually zero in these fuel types as no fertile ^{232}Th is initially present in the fuel. Figure 6.1 shows that the Pu-Th produces a greater amount of ^{233}U than the U-Th fuel, due to the increased fertile load present in the Pu-Th fuel and the harder neutron spectrum of the Pu-Th case. The Pu-Th case requires a lower fissile content (9 wt% plutonium using reactor-grade plutonium for Pu-Th case versus 26 wt% uranium using 20 wt% enriched uranium for the U-Th case) due to the effectiveness of the plutonium (high fission cross section and greater neutrons per fission), resulting in a higher fertile load (ThO_2) for Pu-Th.

Figure 6.1 also shows that there are diminishing returns in terms of ^{233}U production above 60 GWd/MTHM. The ^{233}U mass increases quickly up to 60 GWd/MTHM, then increases more slowly with increasing burnup due to increasing ^{233}U fissions as ^{235}U and Pu are depleted. It should be noted that this increase in the ^{233}U content is not sufficient to maintain constant reactivity of the fuel lattice; fissile material is being consumed faster than it is produced.

The total plutonium mass (^{239}Pu, ^{240}Pu, ^{241}Pu, and ^{242}Pu) is plotted in Figures 6.2 and 6.3. The data for the four fuel types have been split into two different plots corresponding to fuel that initially contains plutonium (MOX and Pu-Th, Figure 6.3) and fuel that does not initially contain plutonium (UOX and U-Th, Figure 6.3). From a plutonium disposition perspective, Figure 6.2 shows that Pu-Th fuel has a lower plutonium inventory than MOX fuel. This is due to the absence of ^{238}U in the Pu-Th fuel, which in the MOX case, breeds additional plutonium during irradiation. Figure 6.3 shows the buildup of plutonium for UOX and U-Th fuels as a function of burnup. The buildup of plutonium is much lower in the U-Th case due to the lower fraction of uranium of ^{238}U present in this case (higher ^{235}U enrichment and smaller fraction of uranium in the fuel).

Table 6.1. Selected actinide isotope masses (g/MTHM) for UOX, MOX, Pu-Th, and U-Th fuel types for various discharge burnup values

	25.1 GWd/MTHM				45.1 GWd/MTHM			
	UOX	**MOX**	**Pu-Th**	**U-Th**	**UOX**	**MOX**	**Pu-Th**	**U-Th**
Th-232	3.39E-04	1.22E-05	9.09E+05	7.39E+05	3.77E-03	1.68E-04	8.97E+05	7.27E+05
U-233	3.74E-04	4.06E-05	2.88E+02	2.96E+02	3.62E-03	6.53E-04	8.53E+03	7.79E+03
U-234	3.39E+02	2.08E+01	8.66E+00	5.15E+02	2.45E+02	3.60E+01	5.05E+02	9.67E+02
U-235	3.80E+04	2.70E+03	2.28E-01	5.00E+04	1.82E+04	1.96E+03	5.38E+01	2.71E+04
U-236	5.59E+02	2.81E+01	2.91E-01	6.51E+02	4.10E+03	2.07E+02	5.39E+00	4.89E+03
U-238	9.58E+05	9.16E+05	1.42E-03	2.07E+05	9.42E+05	9.02E+05	2.16E-02	2.02E+05
Pu-238	9.58E+05	9.58E+05	9.58E+05	9.58E+05	9.42E+05	9.42E+05	9.42E+05	9.42E+05
Pu-239	7.85E+02	4.27E+04	4.73E+04	2.89E+02	5.66E+03	3.04E+04	2.36E+04	2.14E+03
Pu-240	2.35E+01	2.10E+04	2.36E+04	7.28E+00	1.61E+03	2.02E+04	2.17E+04	4.98E+02
Pu-241	1.85E+00	7.86E+03	8.85E+03	5.33E-01	9.57E+02	1.00E+04	1.08E+04	3.89E+02
Pu-242	1.86E-02	5.74E+03	6.48E+03	4.36E-03	1.93E+02	6.00E+03	6.81E+03	6.69E+01
Up-237	6.49E+00	5.93E+00	8.04E-03	3.59E+00	3.09E+02	1.11E+02	1.20E+00	2.74E+02
Am-241	2.57E-03	4.35E+01	4.90E+01	7.36E-04	2.07E+01	5.52E+02	6.11E+02	8.42E+00
Am-243	1.34E-04	9.55E+01	1.06E+02	2.94E-05	2.78E+01	1.13E+03	1.25E+03	8.13E+00

	60.2 GWd/MTHM				80.2 GWd/MTHM			
	UOX	**MOX**	**Pu-Th**	**U-Th**	**UOX**	**MOX**	**Pu-Th**	**U-Th**
Th-232	5.39E-03	2.78E-04	8.86E+05	7.15E+05	6.04E-03	3.45E-04	8.78E+05	7.06E+05
U-233	4.29E-03	1.17E-03	1.34E+04	1.12E+04	4.13E-03	1.52E-03	1.58E+04	1.26E+04
U-234	1.78E+02	4.45E+01	1.23E+03	1.73E+03	1.38E+02	4.95E+01	1.92E+03	2.39E+03
U-235	8.70E+03	1.43E+03	2.02E+02	1.50E+04	4.63E+03	1.11E+03	3.78E+02	9.23E+03
U-236	5.44E+03	3.13E+02	1.70E+01	6.81E+03	5.75E+03	3.67E+02	3.68E+01	7.50E+03
U-238	9.26E+05	8.88E+05	4.22E-02	1.97E+05	9.13E+05	8.78E+05	6.54E-02	1.93E+05
Pu-238	9.26E+05	9.26E+05	9.26E+05	9.26E+05	9.13E+05	9.13E+05	9.13E+05	9.13E+05
Pu-239	6.35E+03	2.31E+04	1.05E+04	2.35E+03	6.41E+03	1.92E+04	4.76E+03	2.34E+03
Pu-240	2.77E+03	1.84E+04	1.76E+04	7.68E+02	3.31E+03	1.67E+04	1.34E+04	8.62E+02
Pu-241	1.76E+03	1.03E+04	1.02E+04	7.35E+02	2.08E+03	9.88E+03	8.62E+03	8.52E+02
Pu-242	7.39E+02	6.59E+03	7.61E+03	2.71E+02	1.26E+03	7.12E+03	8.35E+03	4.61E+02
Np-237	6.45E+02	1.89E+02	3.38E+00	6.07E+02	8.54E+02	2.39E+02	5.83E+00	8.35E+02
Am-241	5.55E+01	7.91E+02	8.14E+02	2.50E+01	7.13E+01	8.49E+02	7.73E+02	3.29E+01
Am-243	1.88E+02	1.74E+03	1.92E+03	6.41E+01	3.95E+02	2.09E+03	2.33E+03	1.45E+02

	100.3 GWd/MTHM			
	UOX	**MOX**	**Pu-Th**	**U-Th**
Th-232	6.39E-03	4.20E-04	8.63E+05	6.92E+05
U-233	3.65E-03	1.96E-03	1.75E+04	1.35E+04
U-234	9.68E+01	5.56E+01	3.04E+03	3.37E+03
U-235	1.73E+03	7.29E+02	7.00E+02	4.62E+03
U-236	5.56E+03	4.12E+02	9.54E+01	7.75E+03
U-238	8.91E+05	8.62E+05	1.33E-01	1.86E+05
Pu-238	8.91E+05	8.91E+05	8.91E+05	8.91E+05
Pu-239	6.43E+03	1.53E+04	1.20E+03	2.30E+03
Pu-240	3.72E+03	1.40E+04	7.04E+03	9.18E+02
Pu-241	2.29E+03	8.71E+03	5.51E+03	9.01E+02
Pu-242	1.97E+03	7.87E+03	9.14E+03	7.10E+02
Np-237	1.04E+03	2.93E+02	1.19E+01	1.08E+03
Am-241	7.99E+01	7.93E+02	5.22E+02	3.61E+01
Am-243	7.34E+02	2.50E+03	2.80E+03	2.91E+02

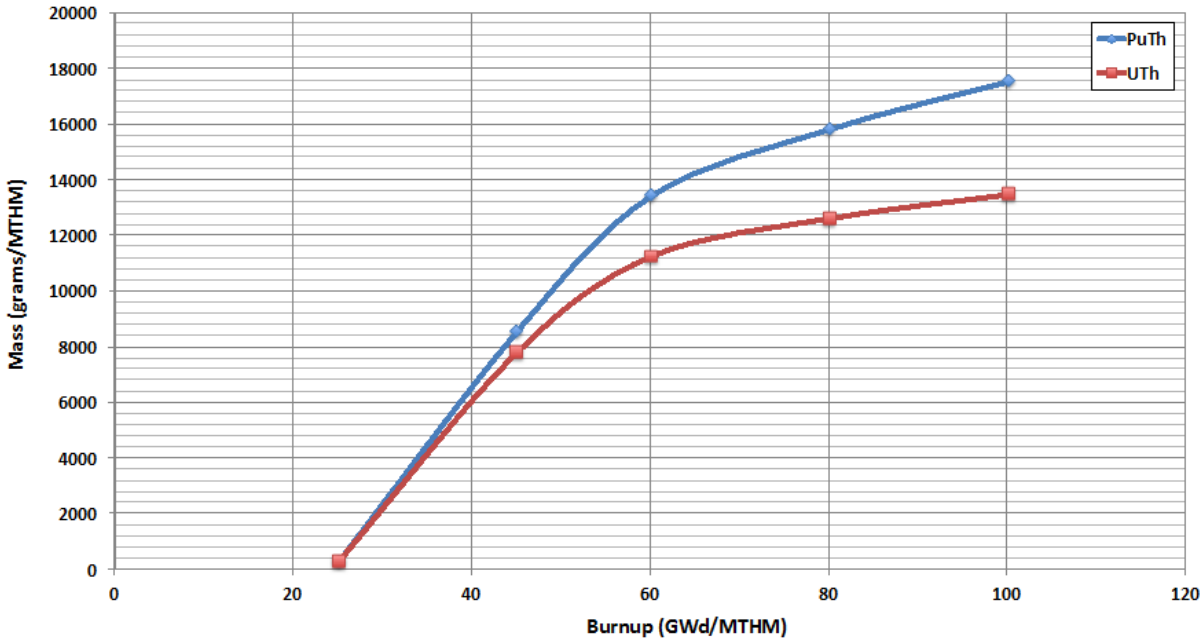


Figure 6.1. ^{233}U production as a function of fuel burnup.

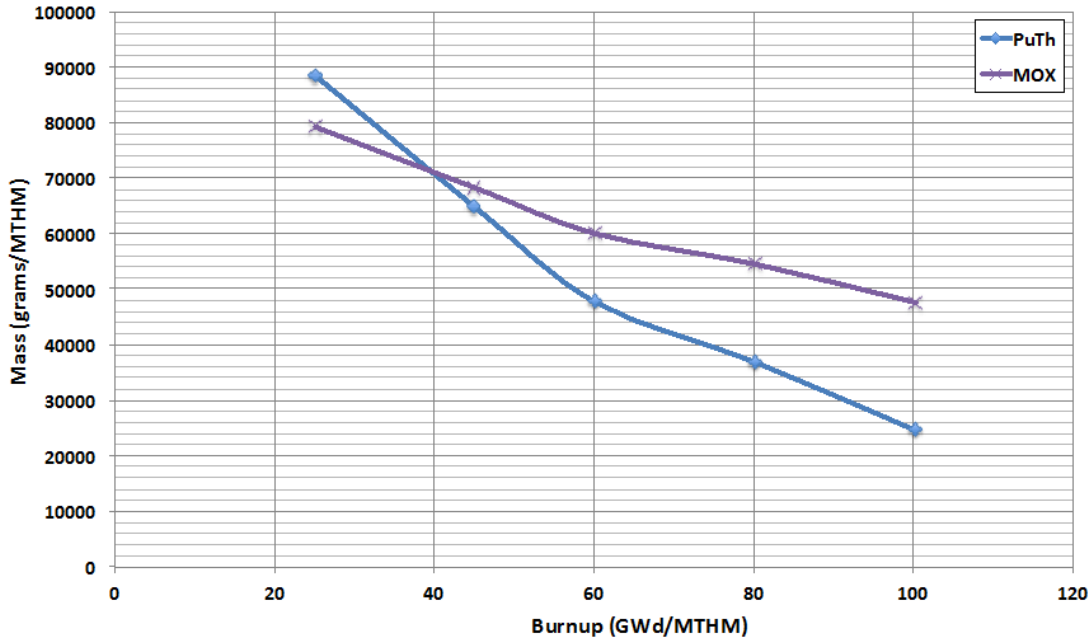


Figure 6.2. Total plutonium mass as a function of fuel burnup for fuel types that initially contain plutonium.

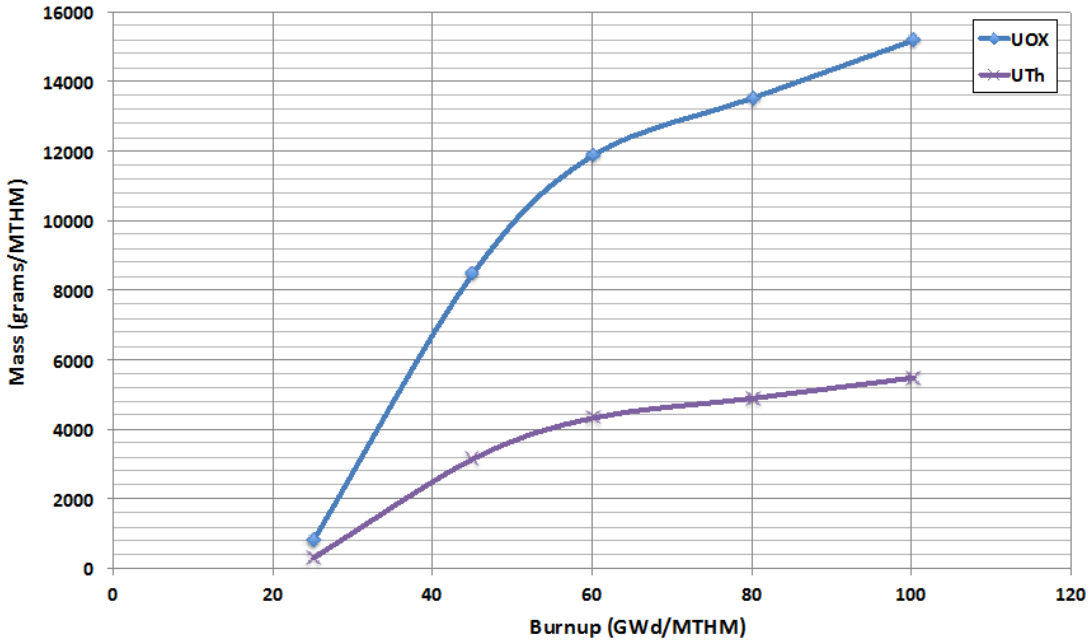


Figure 6.3. Total plutonium mass as a function of fuel burnup for fuel types that initially do not contain plutonium.

6.2 DECAY HEAT

Decay heat characteristics play a significant role in both in-reactor and out-of-reactor safety analyses, so deviation from the prototypic UOX decay heat curves could have notable implications in accidents, as well as medium- to long-term storage.

The isotopic concentrations shown in Table 6.1 revealed that the increase in ^{233}U concentration slows after 60 GWd/MTHM. This result and the unknown fuel performance characteristics of very high-burnup LWR fuel make it unlikely that the fuel would be used at very high burnups. In the remaining out-of-reactor analyses, 60 GWd/MTHM has been used as the maximum burnup. Burnup values of 25.1, 45.1, and 60.2 GWd/MTHM have been used and are referred hereafter as low, typical (or normal), and high discharge burnups, respectively.

The total decay heat values for UOX, MOX, Pu-Th, and U-Th up to 1000 years of decay time have been plotted in Figures 6.4 – 6.6. Figure 6.4 corresponds to low discharge burnup fuel (25 GWd/MTHM), Figure 6.5 corresponds to typical discharge burnup fuel (45 GWd/MTHM), and Figure 6.6 corresponds to high discharge burnup fuel (60 GWd/MTHM).

Each of the plots shows similar trends. The decay heats for all fuels at times immediately after shutdown are of the same order. In general, fuels that initially contain uranium (UOX and U-Th) follow the same general curve, while the fuels that initially contain plutonium (MOX and Pu-Th) follow a higher curve. As expected, the decay heat increases as a function of increasing fuel burnup. Interestingly, all four fuel types have similar decay heat values for decay times up to 0.1 year. This decay time range has been expanded in Figure 6.7 for the 45 GWd/MTHM

discharge burnup, which shows that the thorium-based fuels have slightly higher values up to 0.1 year.

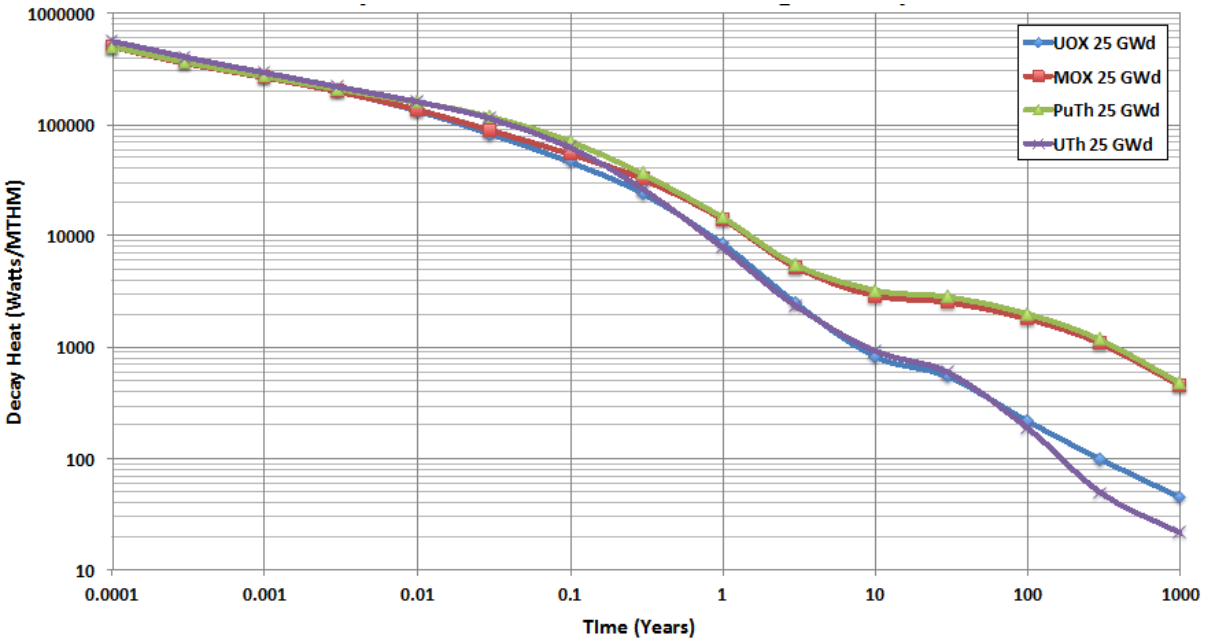


Figure 6.4. Decay heat as a function of decay time for low-burnup fuels.

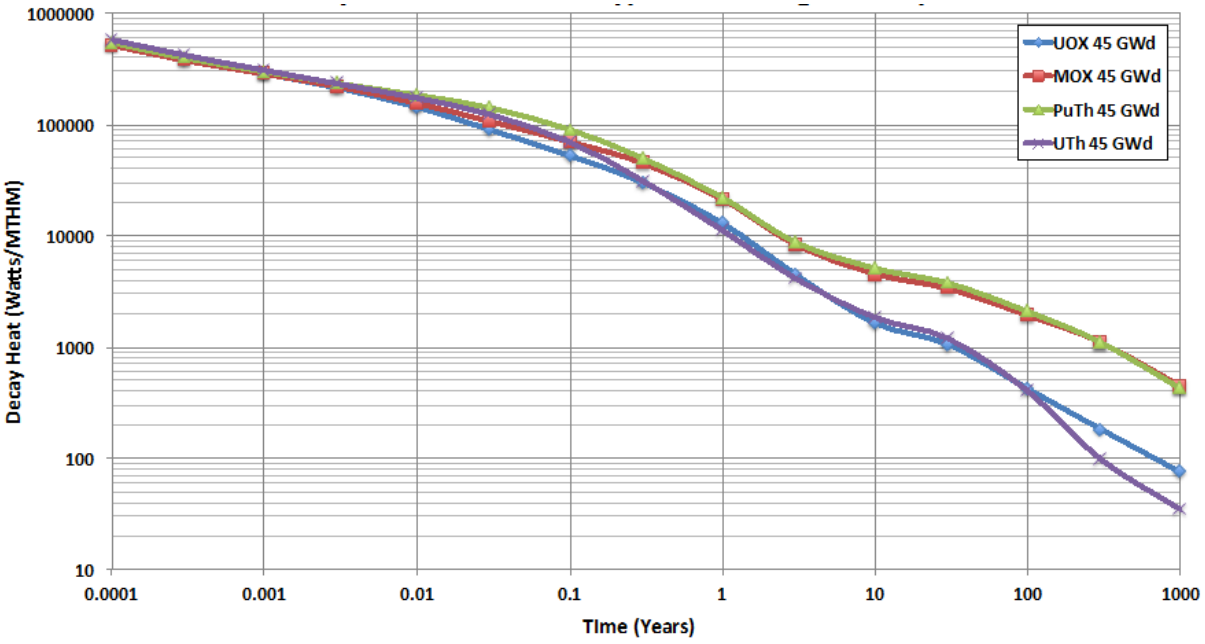


Figure 6.5. Decay heat as a function of decay time for typical burnup fuels.

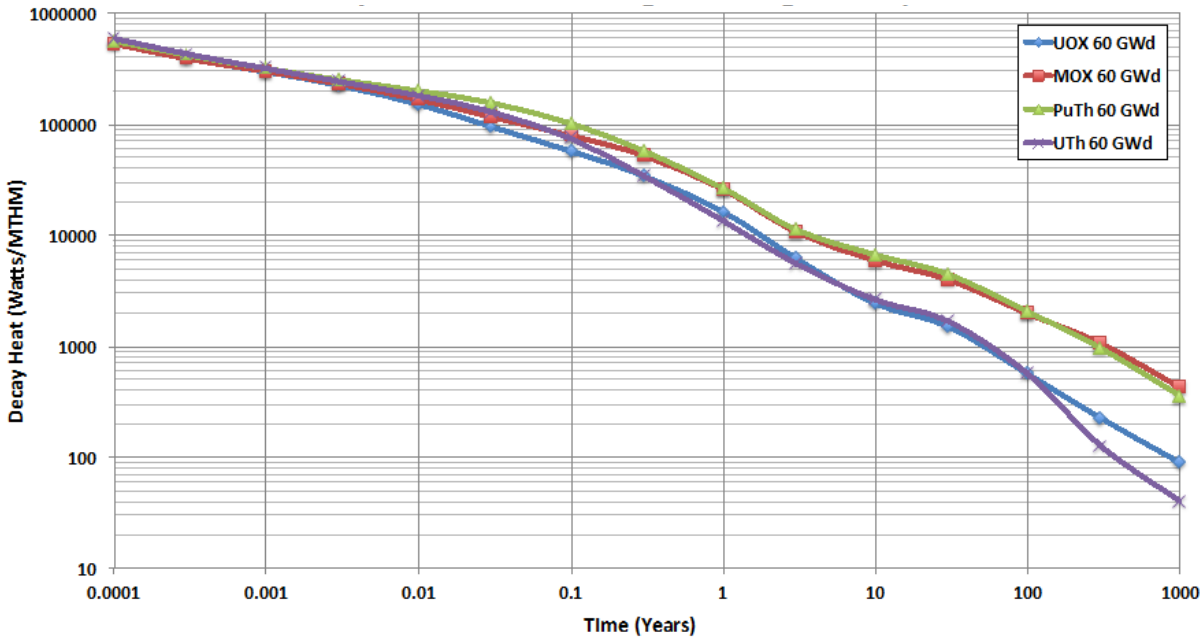


Figure 6.6. Decay heat as a function of decay time for high-burnup fuels.

The increase in decay heat up to 0.1 year could have implications on severe accidents, which can evolve over a number of days, and could also impact fuel handling and core reloading maneuvers, as well as storage in the spent fuel pool.

In order to further understand the main nuclides causing the differences up to 0.1 year, the total decay heat and top five nuclides contributing to the total decay heat at 0.1 year have been plotted in Figures 6.8–6.11 (UOX, MOX, U-Th, and Pu-Th, respectively) as function of decay time to 1000 years. Figures 6.10 and 6.11 show that ^{233}Pa, a daughter of ^{232}Th neutron capture, is the largest nuclide contributor to decay heat for the first 0.1 year for thorium fuel types. Likewise, ^{242}Cm is the largest contributor for the first year for MOX fuel (Figure 6.9). In the case of Pu-Th fuel, the decay heat contribution of ^{233}Pa and ^{242}Cm is nearly equal after 0.1 year of decay.

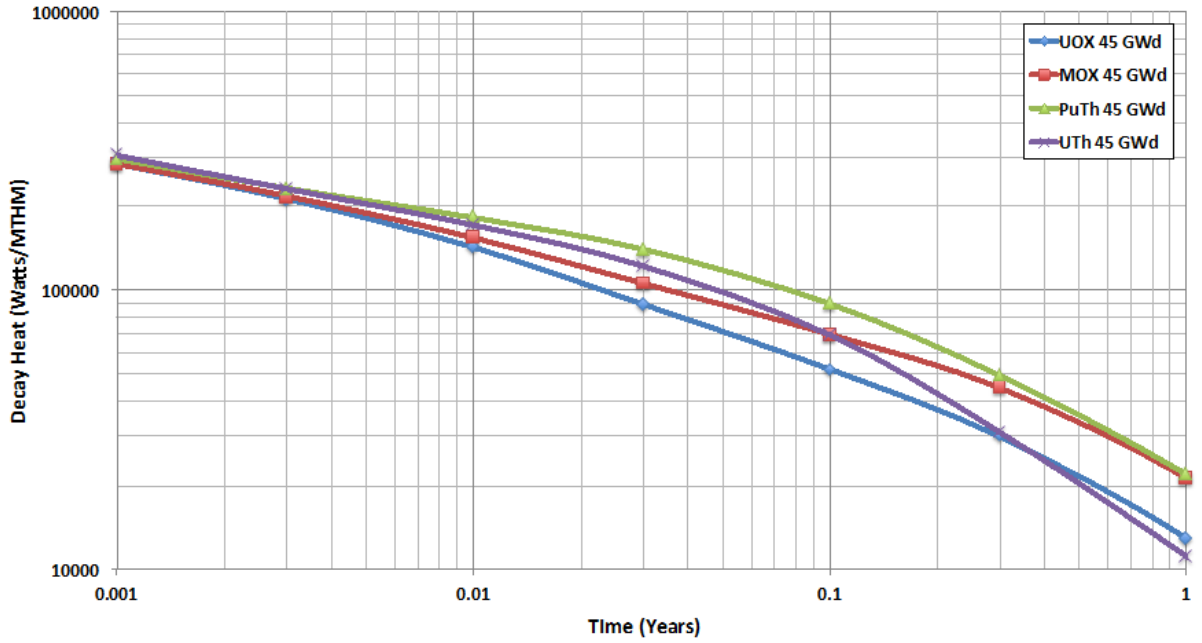


Figure 6 7. Decay heat between 1 hour and 1 year after discharge for typical burnup fuels.

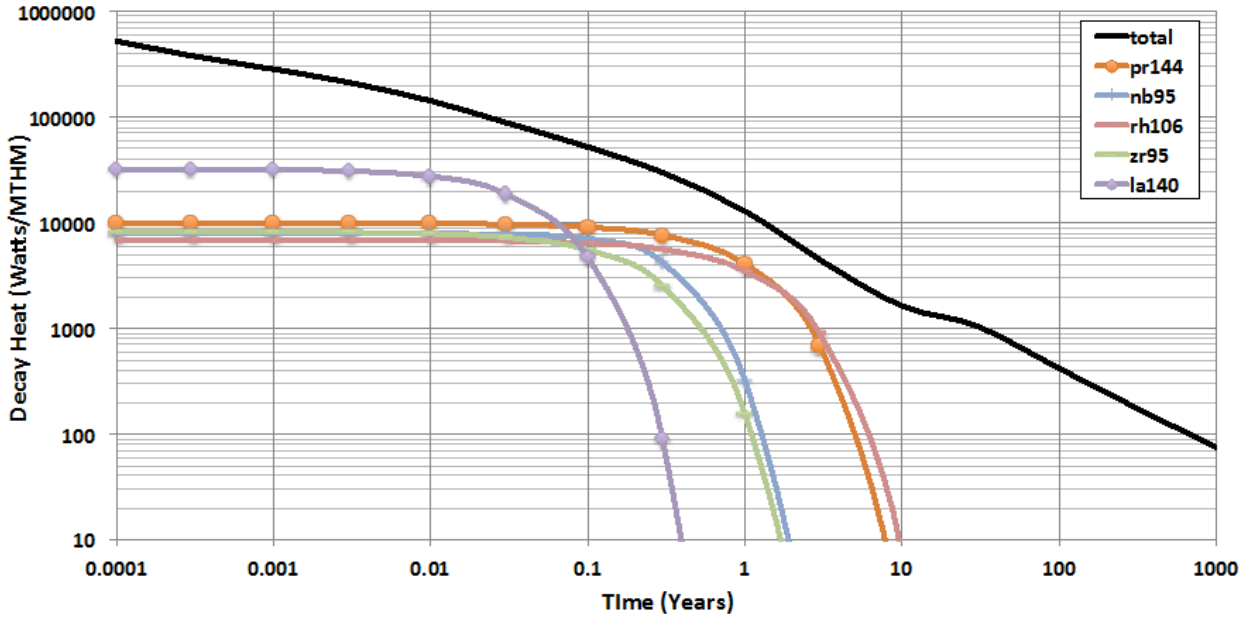


Figure 6.8. Decay heat for top five (at 30 days decay) contributing nuclides in UOX fuel for decay times to 1000 years.

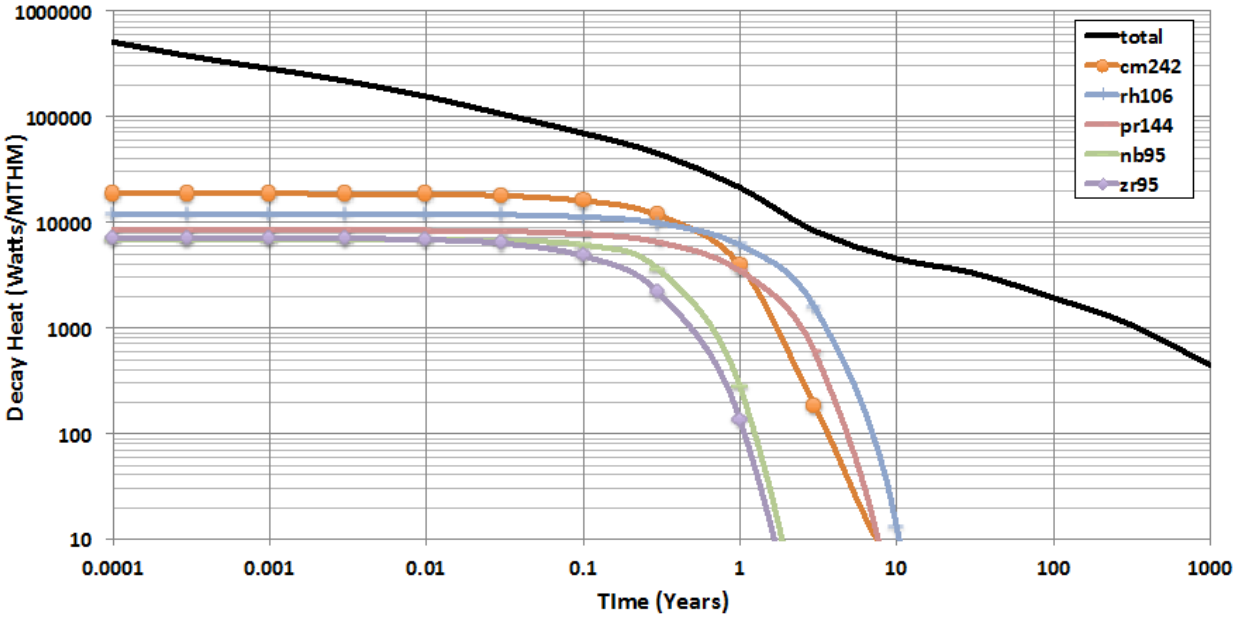


Figure 6.9. Decay heat isotopic components for top five (at 30 days decay) contributing nuclides in MOX fuel for decay times to 1000 years.

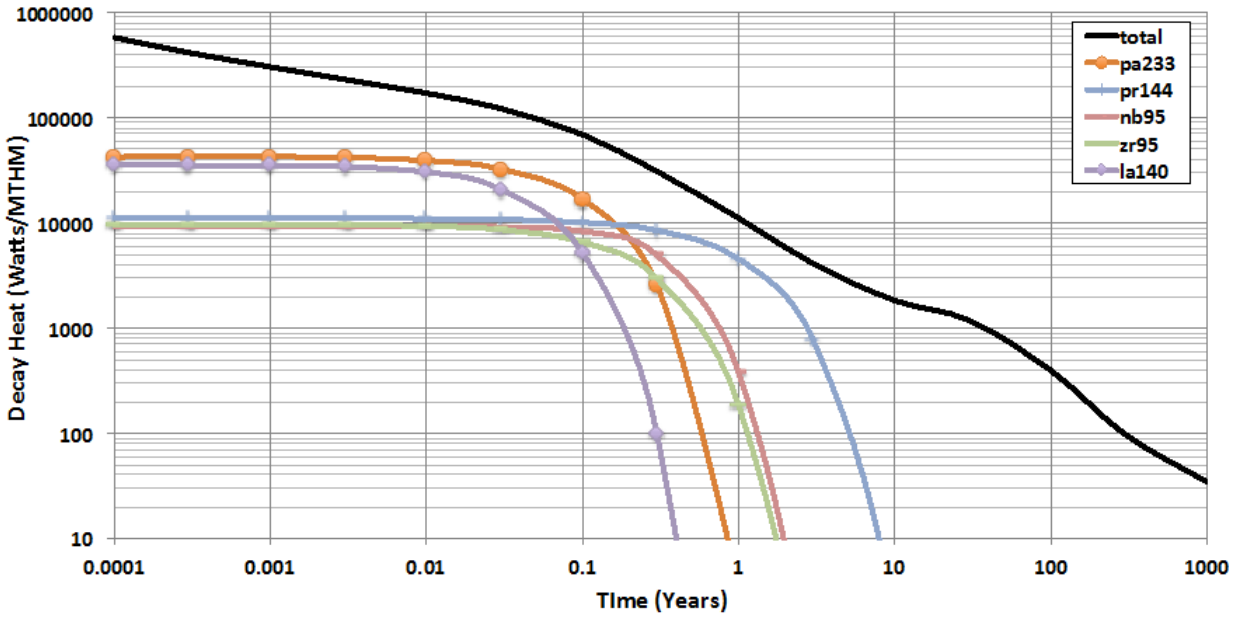


Figure 6.10. Decay heat isotopic components for top five (at 30 days decay) contributing nuclides in U-Th fuel for decay times to 1000 years.

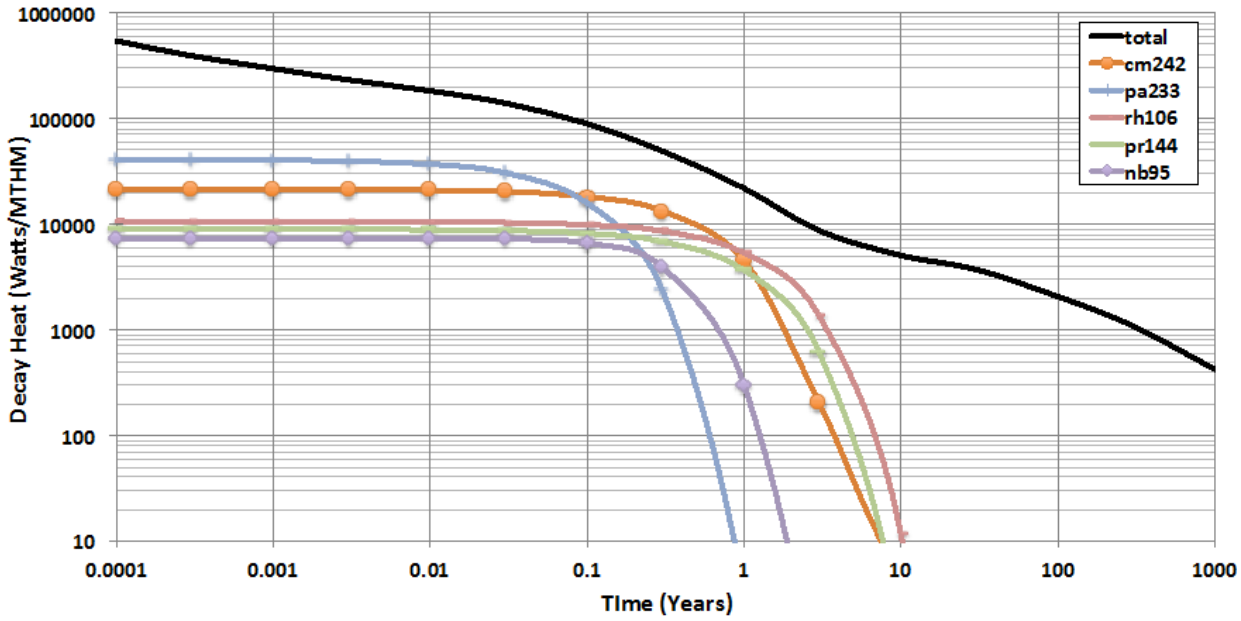


Figure 6.11. Decay heat isotopic components for top five (at 30 days decay) contributing nuclides in Pu-Th fuel for decay times to 1000 years.

The decay heat for extended decay times (100 to 1 million years) is plotted in Figure 6.12. After very long decay times (between 30,000 and 1 million years), the decay heat for thorium fuel is higher than that of UOX or MOX. Particularly interesting is the trend of the U-Th fuel at these timescales, because its decay heat actually increases slightly between 3,000 and 30,000 years. However, the total decay heat after such long decay times is very small – on the order of tens of watts per MTHM.

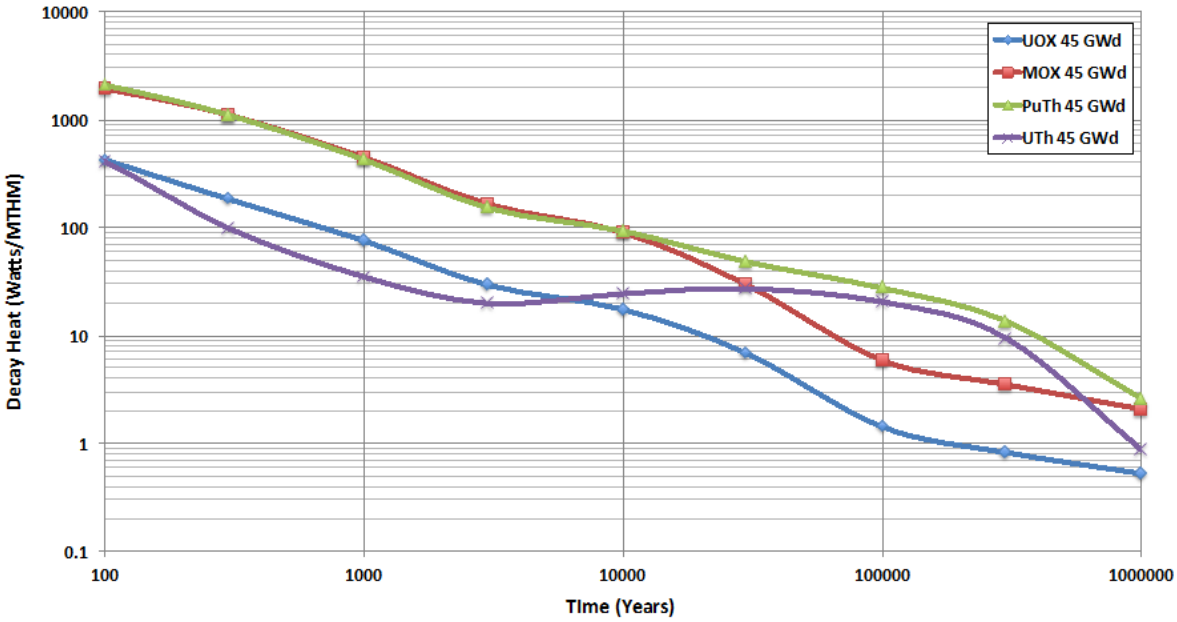


Figure 6.12. Decay heat between 100 and 1 million years after shutdown.

The decay heat for each fuel type is summarized in Table 6.2, where the top four rows (unshaded) provide the decay heat for each fuel type in kW/MTHM and the bottom three rows (shaded in gray) provide the decay heat as a ratio to typical UOX fuel. The data in the table show that immediately after shutdown (1 hour), the decay heat for the plutonium-based fuel types (MOX and Pu-Th) is slightly lower than the uranium-based fuel types (UOX and U-Th). The U-Th fuel decay heat is 25% higher than for UOX at 1 month due to ^{233}Pa but is less than or similar to UOX for longer decay times. The plutonium-based fuels have considerably larger decay heat values for 10 years or longer.

Table 6.2. Summary of decay heat for the four fuel types

	1 hr.	1 d.	3 d.	1 wk.	1 mo.	1 yr.	3 yr.	10 yr.	100 yr.
UOX	517.4	222.2	160.7	118.2	61.60	12.93	4.457	1.631	0.419
MOX	506.3	229.1	171.4	132.3	78.81	21.24	8.247	4.513	1.913
Pu-Th	511.9	232.7	187.8	156.6	99.27	22.56	9.072	5.563	2.719
U-Th	547.6	227.4	174.6	138.4	76.88	10.81	3.868	1.713	0.362
MOX/UOX	0.98	1.03	1.07	1.12	1.28	1.64	1.85	2.77	4.56
Pu-Th/UOX	0.99	1.05	1.17	1.32	1.61	1.74	2.04	3.41	6.48
U-Th/UOX	1.06	1.02	1.09	1.17	1.25	0.84	0.87	1.05	0.86

6.3 SOURCE TERMS

Calculation of radiological source terms enables consequence analysis of loss of fuel or containment integrity leading to radiological release. Differences in fuel composition and operating conditions can lead to different isotopics, and therefore, radiological source terms. In this section, the radiological source terms have been calculated for UOX, MOX, Pu-Th, and U-Th fuels at typical discharge burnup. The radiological source terms for UOX, MOX, Pu-Th, and U-Th are given in Tables 6.3, 6.4, 6.5, and 6.6, respectively. These radiological source terms are also shown in Figure 6.13 for all four fuel types. The first 69 isotopes in the tables were the isotopes used in a previous LWR spent fuel source term study (ranked from highest to lowest activities at 3 years), and the next 10 (shaded gray) were the next 10 highest ranking for that particular fuel type (also ranked from highest to lowest activities at 3 years). Note that the 10 shaded isotopes vary between fuel types. The activities in Tables 6.3–6.6 are given in units of Bq/MTHM.

At short decay times (less than 1 year), the thorium-based fuels have significantly higher source terms due to the ^{233}Pa decay. At 0.01 year (~4 days), the total source term for U-Th fuel is ~30% greater than for UOX fuel, and at 0.1 year (~1 month), the U-Th source term is ~55% greater. From 1 year to 100 years, the U-Th and UOX source terms are similar. However, at 300 to 500 years, the U-Th source term drops to almost half the UOX value as the contributions from the Th/^{233}U decay chains die away. The total source term for Pu-Th fuel is similar to the U-Th value for less than 1 year. For decay times of 1 year or greater, the Pu-Th source term is similar to the MOX source term because the plutonium source term dominates. At 300 to 500 years, the source terms for the plutonium-based fuels are ~6 to 10 times greater than the UOX and U-Th fuels, respectively.

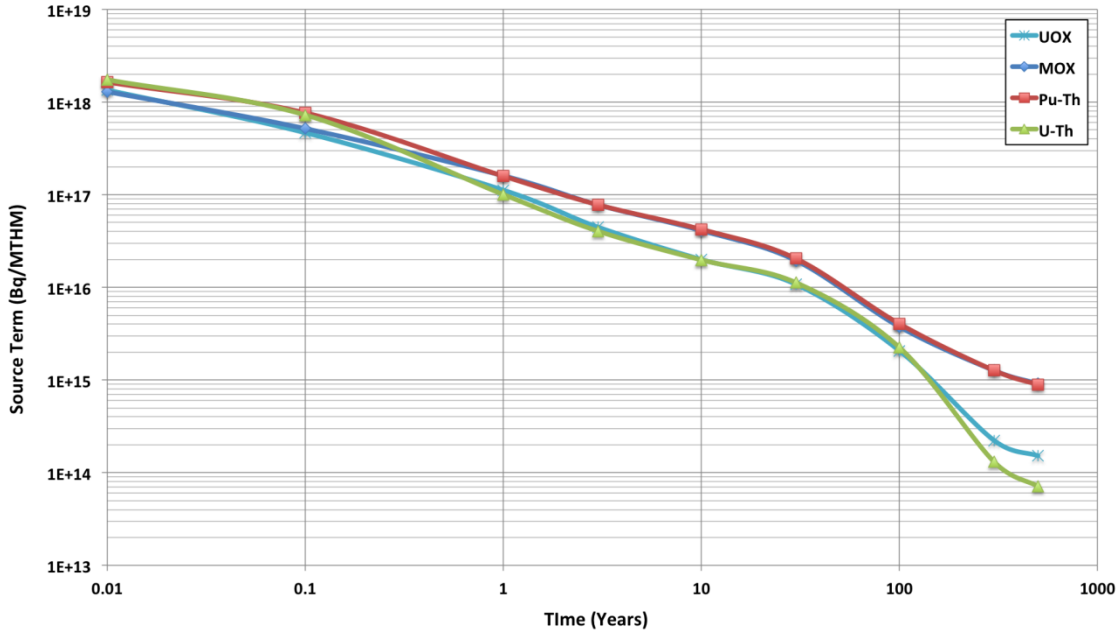


Figure 6.13. Source term as a function of decay time for UOX, MOX, Pu-Th, and U-Th fuels.

Table 6.3. Radiological source terms (Bq/MTHM) for the UOX fuel

Decay time (yrs)	0.01	0.1	1	3	10	30	100	300	500
pu241	6.75E+15	6.72E+15	6.44E+15	5.84E+15	4.16E+15	1.58E+15	5.29E+13	3.39E+10	3.02E+10
cs137	5.25E+15	5.24E+15	5.13E+15	4.90E+15	4.17E+15	2.63E+15	5.24E+14	5.22E+12	5.20E+10
ba137m	4.98E+15	4.96E+15	4.86E+15	4.64E+15	3.95E+15	2.49E+15	4.97E+14	4.95E+12	4.93E+10
cs134	9.60E+15	9.31E+15	6.88E+15	3.52E+15	3.36E+14	4.08E+11	2.55E+01	0.00E+00	0.00E+00
ce144	4.97E+16	4.59E+16	2.06E+16	3.49E+15	6.93E+12	1.32E+05	0.00E+00	0.00E+00	0.00E+00
pr144	4.97E+16	4.59E+16	2.06E+16	3.49E+15	6.93E+12	1.32E+05	0.00E+00	0.00E+00	0.00E+00
ru106	2.64E+16	2.48E+16	1.35E+16	3.44E+15	2.93E+13	3.56E+07	7.04E-14	0.00E+00	0.00E+00
rh106	2.64E+16	2.48E+16	1.35E+16	3.44E+15	2.93E+13	3.56E+07	7.04E-14	0.00E+00	0.00E+00
y90	3.68E+15	3.61E+15	3.54E+15	3.37E+15	2.85E+15	1.76E+15	3.26E+14	2.64E+12	2.14E+10
sr90	3.62E+15	3.61E+15	3.54E+15	3.37E+15	2.85E+15	1.76E+15	3.26E+14	2.64E+12	2.14E+10
kr85	4.84E+14	4.81E+14	4.54E+14	3.99E+14	2.54E+14	7.00E+13	7.69E+11	1.94E+06	4.90E+00
cm244	2.23E+14	2.23E+14	2.15E+14	1.99E+14	1.52E+14	7.09E+13	4.86E+12	2.30E+09	1.09E+06
pu238	1.68E+14	1.70E+14	1.77E+14	1.77E+14	1.68E+14	1.43E+14	8.25E+13	1.71E+13	3.54E+12
am241	7.15E+12	8.12E+12	1.76E+13	3.72E+13	9.20E+13	1.73E+14	2.01E+14	1.47E+14	1.07E+14
pr144m	4.75E+14	4.38E+14	1.97E+14	3.33E+13	6.62E+10	1.26E+03	0.00E+00	0.00E+00	0.00E+00
cm242	2.59E+15	2.25E+15	5.56E+14	2.51E+13	2.57E+11	2.33E+11	1.65E+11	6.18E+10	2.31E+10
pu240	2.33E+13	2.33E+13	2.33E+13	2.33E+13	2.34E+13	2.36E+13	2.36E+13	2.31E+13	2.27E+13
pu239	1.47E+13	1.48E+13	1.48E+13	1.48E+13	1.48E+13	1.48E+13	1.48E+13	1.47E+13	1.46E+13
np239	2.96E+17	2.01E+13	1.39E+12	1.39E+12	1.39E+12	1.38E+12	1.37E+12	1.35E+12	1.32E+12
nb95	6.07E+16	5.46E+16	2.50E+15	9.39E+11	8.95E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zr95	5.81E+16	4.07E+16	1.16E+15	4.26E+11	4.06E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00
te127m	3.10E+14	2.62E+14	3.24E+13	3.11E+11	2.70E+04	1.80E-16	0.00E+00	0.00E+00	0.00E+00
te127	2.24E+15	2.62E+14	3.17E+13	3.04E+11	2.64E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00
y91	3.95E+16	2.68E+16	5.45E+14	9.50E+10	6.64E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sr89	2.93E+16	1.87E+16	2.05E+14	9.13E+09	5.34E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ru103	6.16E+16	3.45E+16	1.04E+14	2.58E+08	6.31E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rh103m	6.09E+16	3.41E+16	1.03E+14	2.56E+08	6.24E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ce141	5.69E+16	2.82E+16	2.55E+13	4.38E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
te129m	1.52E+15	7.72E+14	8.76E+11	2.50E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
te129	9.60E+14	4.87E+14	5.53E+11	1.57E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
co60	6.08E-01	6.00E-01	5.33E-01	4.10E-01	1.63E-01	1.18E-02	1.18E-06	2.62E-10	2.62E-10
rb86	1.05E+14	3.09E+13	1.51E+08	2.37E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
co58	7.48E-03	5.42E-03	2.17E-04	1.71E-07	2.35E-18	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pr143	4.91E+16	9.35E+15	4.76E+08	2.96E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
la140	6.05E+16	1.03E+16	1.79E+08	1.02E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ba140	5.34E+16	8.94E+15	1.55E+08	8.85E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cs136	1.91E+15	3.39E+14	1.02E+07	1.99E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nd147	1.95E+16	2.45E+15	2.38E+06	2.22E-14	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
kr85m	1.13E+10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
kr87	3.01E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
kr88	1.12E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
xe133	5.59E+16	7.46E+14	9.93E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
xe135	2.57E+14	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
xe135m	1.24E+12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rb88	1.25E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ba139	6.62E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sr91	7.22E+13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sr92	8.02E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i131	2.93E+16	1.73E+15	8.04E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i132	2.57E+16	2.10E+13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i133	4.33E+15	1.65E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i134	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i135	7.20E+12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
te131m	1.25E+15	8.99E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
te132	2.50E+16	2.03E+13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
te131	3.28E+14	2.36E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rh105	9.36E+15	1.80E+09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ru105	5.74E+10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
mo99	2.82E+16	7.09E+12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
tc99m	2.73E+16	6.86E+12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nb97	1.68E+15	1.19E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nb97m	1.59E+15	1.05E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ce143	8.96E+15	5.80E+08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zr97	1.67E+15	1.10E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
la141	1.23E+10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Table 6.3. Continued

Decay time (yrs)	0.01	0.1	1	3	10	30	100	300	500
la142	2.74E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
y92	6.56E+09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
y93	1.33E+14	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
y91m	4.65E+13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pm147	7.46E+15	7.48E+15	5.92E+15	3.49E+15	5.49E+14	2.78E+12	2.58E+04	0.00E+00	0.00E+00
eu154	3.80E+14	3.77E+14	3.51E+14	2.99E+14	1.70E+14	3.39E+13	1.20E+11	1.20E+04	1.20E-03
s325	4.59E+14	4.51E+14	3.60E+14	2.18E+14	3.75E+13	2.46E+11	5.65E+03	0.00E+00	0.00E+00
te125m	1.01E+14	1.02E+14	8.81E+13	5.34E+13	9.19E+12	6.03E+10	1.38E+03	0.00E+00	0.00E+00
h3	2.65E+13	2.64E+13	2.51E+13	2.24E+13	1.51E+13	4.91E+12	9.56E+10	1.24E+06	1.61E+01
sm151	1.49E+13	1.49E+13	1.48E+13	1.46E+13	1.38E+13	1.18E+13	6.90E+12	1.48E+12	3.17E+11
ag110m	2.22E+14	2.03E+14	8.15E+13	1.07E+13	8.89E+09	1.39E+01	0.00E+00	0.00E+00	0.00E+00
am243	1.39E+12	1.39E+12	1.39E+12	1.39E+12	1.39E+12	1.38E+12	1.37E+12	1.35E+12	1.32E+12
sn121m	1.29E+12	1.29E+12	1.27E+12	1.23E+12	1.10E+12	8.02E+11	2.66E+11	1.13E+10	4.80E+08
cm243	1.04E+12	1.04E+12	1.02E+12	9.71E+11	8.22E+11	5.10E+11	9.63E+10	8.21E+08	7.01E+06
Total	1.34E+18	4.66E+17	1.12E+17	4.47E+16	1.99E+16	1.08E+16	12.07E+15	2.23E+14	1.52E+14

Table 6.4. Radiological source terms (Bq/MTHM) for MOX fuel

Decay time (yrs)	0.01	0.1	1	3	10	30	100	300	500
pu241	3.96E+16	3.94E+16	3.77E+16	3.42E+16	2.44E+16	9.24E+15	3.10E+14	7.69E+11	7.38E+11
ru106	4.60E+16	4.33E+16	2.35E+16	6.01E+15	5.11E+13	6.21E+07	1.23E-13	0.00E+00	0.00E+00
rh106	4.60E+16	4.33E+16	2.35E+16	6.01E+15	5.11E+13	6.21E+07	1.23E-13	0.00E+00	0.00E+00
cs137	5.33E+15	5.32E+15	5.21E+15	4.97E+15	4.23E+15	2.67E+15	5.32E+14	5.30E+12	5.28E+10
ba137m	5.05E+15	5.03E+15	4.93E+15	4.71E+15	4.01E+15	2.53E+15	5.04E+14	5.02E+12	5.00E+10
cs134	8.25E+15	8.00E+15	5.92E+15	3.02E+15	2.89E+14	3.50E+11	2.19E+01	0.00E+00	0.00E+00
ce144	4.22E+16	3.90E+16	1.75E+16	2.96E+15	5.89E+12	1.13E+05	0.00E+00	0.00E+00	0.00E+00
pr144	4.23E+16	3.90E+16	1.75E+16	2.96E+15	5.89E+12	1.13E+05	0.00E+00	0.00E+00	0.00E+00
cm244	2.84E+15	2.83E+15	2.73E+15	2.53E+15	1.94E+15	9.00E+14	6.18E+13	2.92E+10	1.38E+07
sr90	1.81E+15	1.81E+15	1.77E+15	1.69E+15	1.43E+15	8.81E+14	1.63E+14	1.32E+12	1.07E+10
y90	1.82E+15	1.81E+15	1.77E+15	1.69E+15	1.43E+15	8.81E+14	1.63E+14	1.32E+12	1.07E+10
pu238	1.03E+15	1.04E+15	1.10E+15	1.10E+15	1.04E+15	8.90E+14	5.14E+14	1.07E+14	2.26E+13
am241	1.01E+14	1.07E+14	1.62E+14	2.77E+14	5.97E+14	1.07E+15	1.23E+15	8.98E+14	6.52E+14
kr85	2.70E+14	2.69E+14	2.54E+14	2.23E+14	1.42E+14	3.91E+13	4.30E+11	1.08E+06	2.74E+00
cm242	1.85E+16	1.61E+16	3.97E+15	1.83E+14	5.16E+12	4.68E+12	3.32E+12	1.24E+12	4.64E+11
pu240	1.55E+14	1.55E+14	1.55E+14	1.55E+14	1.57E+14	1.59E+14	1.61E+14	1.57E+14	1.54E+14
pu239	5.31E+13	5.32E+13	5.32E+13	5.32E+13	5.32E+13	5.32E+13	5.31E+13	5.29E+13	5.26E+13
pr144m	4.03E+14	3.72E+14	1.67E+14	2.83E+13	5.62E+10	1.07E+03	0.00E+00	0.00E+00	0.00E+00
np239	2.28E+17	2.72E+13	1.28E+13	1.28E+13	1.28E+13	1.28E+13	1.27E+13	1.25E+13	1.23E+13
nb95	5.21E+16	4.70E+16	2.16E+15	8.11E+11	7.73E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00
te127m	5.47E+14	4.55E+14	5.63E+13	5.41E+11	4.69E+04	3.13E-16	0.00E+00	0.00E+00	0.00E+00
te127	2.73E+15	4.52E+14	5.52E+13	5.30E+11	4.59E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zr95	5.02E+16	3.52E+16	1.00E+15	3.68E+11	3.51E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00
y91	2.71E+16	1.84E+16	3.74E+14	6.51E+10	4.55E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sr89	1.86E+16	1.19E+16	1.30E+14	5.79E+09	3.39E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ru103	7.24E+16	4.05E+16	1.22E+14	3.04E+08	7.42E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rh103m	7.17E+16	4.01E+16	1.21E+14	3.01E+08	7.34E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ce141	5.41E+16	2.68E+16	2.42E+13	4.16E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
te129m	1.88E+15	9.56E+14	1.08E+12	3.09E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
te129	1.19E+15	6.03E+14	6.84E+11	1.95E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
co60	3.69E-01	3.65E-01	3.24E-01	2.49E-01	9.91E-02	7.14E-03	7.18E-07	5.95E-11	5.94E-11
rb86	5.29E+13	1.56E+13	7.59E+07	1.19E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
co58	9.66E-03	7.00E-03	2.81E-04	2.21E-07	3.04E-18	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pr143	4.48E+16	8.53E+15	4.34E+08	2.70E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
la140	5.80E+16	9.91E+15	1.72E+08	9.80E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ba140	5.14E+16	8.60E+15	1.50E+08	8.51E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cs136	2.88E+15	5.09E+14	1.54E+07	2.99E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nd147	1.93E+16	2.42E+15	2.35E+06	2.19E-14	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
kr85m	7.98E+09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
kr87	2.03E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
kr88	7.40E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
xe133	5.56E+16	7.42E+14	9.88E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
xe135	2.93E+14	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
xe135m	1.25E+12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rb88	8.26E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ba139	6.43E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sr91	5.10E+13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sr92	6.07E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i131	3.02E+16	1.78E+15	8.31E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i132	2.61E+16	2.13E+13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i133	4.30E+15	1.64E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i134	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i135	7.26E+12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
te131m	1.34E+15	9.60E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
te132	2.53E+16	2.06E+13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
te131	3.50E+14	2.52E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rh105	1.27E+16	2.43E+09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ru105	7.47E+10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
mo99	2.78E+16	6.99E+12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
tc99m	2.69E+16	6.76E+12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nb97	1.58E+15	1.12E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nb97m	1.50E+15	9.86E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ce143	8.18E+15	5.29E+08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zr97	1.57E+15	1.04E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Table 6.4. Continued

Decay time (yrs)	0.01	0.1	1	3	10	30	100	300	500
la141	1.17E+10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
la142	2.57E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
y92	4.96E+09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
y93	1.08E+14	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
y91m	3.28E+13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pm147	7.87E+15	7.87E+15	6.23E+15	3.67E+15	5.77E+14	2.93E+12	2.71E+04	0.00E+00	0.00E+00
eu154	6.74E+14	6.70E+14	6.23E+14	5.30E+14	3.02E+14	6.01E+13	2.13E+11	2.13E+04	2.13E-03
s325	6.25E+14	6.14E+14	4.90E+14	2.97E+14	5.11E+13	3.35E+11	7.69E+03	0.00E+00	0.00E+00
te125m	1.40E+14	1.41E+14	1.20E+14	7.26E+13	1.25E+13	8.21E+10	1.88E+03	0.00E+00	0.00E+00
sm151	4.23E+13	4.24E+13	4.21E+13	4.14E+13	3.92E+13	3.36E+13	1.96E+13	4.20E+12	9.01E+11
h3	3.02E+13	3.01E+13	2.86E+13	2.56E+13	1.72E+13	5.59E+12	1.09E+11	1.41E+06	1.83E+01
ag110m	4.87E+14	4.45E+14	1.79E+14	2.35E+13	1.95E+10	3.05E+01	0.00E+00	0.00E+00	0.00E+00
am243	1.28E+13	1.28E+13	1.28E+13	1.28E+13	1.28E+13	1.28E+13	1.27E+13	1.25E+13	1.23E+13
cm243	9.91E+12	9.89E+12	9.68E+12	9.23E+12	7.81E+12	4.85E+12	9.16E+11	7.81E+09	6.66E+07
am242m	6.58E+12	6.58E+12	6.55E+12	6.49E+12	6.27E+12	5.68E+12	4.03E+12	1.51E+12	5.64E+11
Total	1.30E+18	5.17E+17	1.60E+17	7.77E+16	4.09E+16	1.95E+16	3.75E+15	1.27E+15	9.13E+14

Table 6.5. Radiological source terms (Bq/MTHM) for Pu-Th fuel

Decay time (yrs)	0.01	0.1	1	3	10	30	100	300	500
pu241	3.91E+16	3.90E+16	3.73E+16	3.39E+16	2.41E+16	9.14E+15	3.07E+14	8.07E+11	7.76E+11
ru106	4.05E+16	3.81E+16	2.07E+16	5.29E+15	4.50E+13	5.47E+07	1.08E-13	0.00E+00	0.00E+00
rh106	4.05E+16	3.81E+16	2.07E+16	5.29E+15	4.50E+13	5.47E+07	1.08E-13	0.00E+00	0.00E+00
cs137	5.42E+15	5.41E+15	5.30E+15	5.06E+15	4.31E+15	2.72E+15	5.41E+14	5.39E+12	5.37E+10
ba137m	5.13E+15	5.12E+15	5.02E+15	4.79E+15	4.08E+15	2.57E+15	5.12E+14	5.10E+12	5.08E+10
ce144	4.48E+16	4.13E+16	1.86E+16	3.14E+15	6.24E+12	1.19E+05	0.00E+00	0.00E+00	0.00E+00
pr144	4.48E+16	4.13E+16	1.86E+16	3.14E+15	6.24E+12	1.19E+05	0.00E+00	0.00E+00	0.00E+00
cs134	8.54E+15	8.29E+15	6.13E+15	3.13E+15	2.99E+14	3.63E+11	2.27E+01	0.00E+00	0.00E+00
cm244	2.99E+15	2.98E+15	2.88E+15	2.67E+15	2.04E+15	9.48E+14	6.50E+13	3.08E+10	1.46E+07
y90	2.35E+15	2.33E+15	2.28E+15	2.18E+15	1.84E+15	1.14E+15	2.11E+14	1.71E+12	1.38E+10
sr90	2.34E+15	2.33E+15	2.28E+15	2.18E+15	1.84E+15	1.14E+15	2.11E+14	1.71E+12	1.38E+10
pu238	1.12E+15	1.14E+15	1.20E+15	1.20E+15	1.14E+15	9.72E+14	5.61E+14	1.17E+14	2.46E+13
kr85	4.22E+14	4.20E+14	3.96E+14	3.48E+14	2.22E+14	6.11E+13	6.71E+11	1.69E+06	4.27E+00
am241	1.04E+14	1.10E+14	1.64E+14	2.78E+14	5.95E+14	1.06E+15	1.22E+15	8.91E+14	6.47E+14
cm242	2.11E+16	1.84E+16	4.54E+15	2.08E+14	5.16E+12	4.67E+12	3.31E+12	1.24E+12	4.63E+11
pu240	1.48E+14	1.48E+14	1.48E+14	1.49E+14	1.50E+14	1.53E+14	1.54E+14	1.51E+14	1.48E+14
pr144m	4.28E+14	3.95E+14	1.77E+14	3.00E+13	5.96E+10	1.14E+03	0.00E+00	0.00E+00	0.00E+00
pu239	2.41E+13	2.41E+13	2.41E+13	2.41E+13	2.41E+13	2.41E+13	2.41E+13	2.40E+13	2.39E+13
np239	1.44E+13	1.42E+13	1.42E+13	1.42E+13	1.42E+13	1.42E+13	1.41E+13	1.38E+13	1.36E+13
nb95	5.64E+16	5.09E+16	2.35E+15	8.82E+11	8.41E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00
te127m	5.65E+14	4.71E+14	5.82E+13	5.59E+11	4.85E+04	3.23E-16	0.00E+00	0.00E+00	0.00E+00
te127	2.90E+15	4.68E+14	5.70E+13	5.48E+11	4.75E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zr95	5.46E+16	3.82E+16	1.09E+15	4.00E+11	3.81E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00
y91	3.83E+16	2.59E+16	5.27E+14	9.19E+10	6.43E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sr89	3.22E+16	2.05E+16	2.26E+14	1.00E+10	5.86E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ru103	5.70E+16	3.19E+16	9.60E+13	2.39E+08	5.84E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rh103m	5.64E+16	3.16E+16	9.49E+13	2.37E+08	5.78E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ce141	5.86E+16	2.91E+16	2.63E+13	4.51E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
te129m	2.03E+15	1.03E+15	1.17E+12	3.32E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
te129	1.28E+15	6.49E+14	7.36E+11	2.10E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
co60	3.83E-01	3.78E-01	3.36E-01	2.58E-01	1.03E-01	7.41E-03	7.45E-07	6.43E-11	6.43E-11
rb86	7.19E+13	2.12E+13	1.03E+08	1.62E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
co58	9.54E-03	6.91E-03	2.77E-04	2.19E-07	3.00E-18	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pr143	5.00E+16	9.52E+15	4.85E+08	3.02E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
la140	6.22E+16	1.06E+16	1.85E+08	1.05E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ba140	5.51E+16	9.23E+15	1.60E+08	9.13E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cs136	2.81E+15	4.97E+14	1.50E+07	2.92E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nd147	1.86E+16	2.34E+15	2.27E+06	2.12E-14	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
kr85m	1.55E+10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
kr87	3.87E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
kr88	1.41E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
xe133	5.34E+16	7.13E+14	9.50E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
xe135	2.72E+14	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
xe135m	1.18E+12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rb88	1.58E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ba139	6.76E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sr91	7.38E+13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sr92	8.01E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i131	2.99E+16	1.77E+15	8.24E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i132	2.55E+16	2.08E+13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i133	4.13E+15	1.57E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i134	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i135	6.84E+12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
te131m	1.52E+15	1.09E+08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
te132	2.48E+16	2.02E+13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
te131	3.98E+14	2.86E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rh105	9.62E+15	1.85E+09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ru105	5.68E+10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
mo99	2.61E+16	6.55E+12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
tc99m	2.52E+16	6.33E+12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nb97	1.58E+15	1.12E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nb97m	1.50E+15	9.89E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ce143	9.13E+15	5.91E+08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zr97	1.58E+15	1.04E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Table 6.5. Continued

Decay time (yrs)	0.01	0.1	1	3	10	30	100	300	500
la141	1.27E+10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
la142	2.90E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
y92	6.56E+09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
y93	1.33E+14	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
y91m	4.75E+13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pm147	7.24E+15	7.25E+15	5.74E+15	3.38E+15	5.32E+14	2.70E+12	2.50E+04	0.00E+00	0.00E+00
eu154	6.07E+14	6.03E+14	5.61E+14	4.77E+14	2.72E+14	5.42E+13	1.92E+11	1.92E+04	1.92E-03
s325	5.98E+14	5.88E+14	4.69E+14	2.84E+14	4.89E+13	3.21E+11	7.36E+03	0.00E+00	0.00E+00
te125m	1.34E+14	1.35E+14	1.15E+14	6.95E+13	1.20E+13	7.86E+10	1.80E+03	0.00E+00	0.00E+00
u232	5.30E+13	5.30E+13	5.25E+13	5.15E+13	4.80E+13	3.92E+13	1.94E+13	2.59E+12	3.47E+11
ra224	1.36E+13	1.49E+13	2.56E+13	3.92E+13	4.83E+13	4.04E+13	2.00E+13	2.67E+12	3.60E+11
p412	1.36E+13	1.49E+13	2.56E+13	3.92E+13	4.83E+13	4.04E+13	2.00E+13	2.67E+12	3.60E+11
po216	1.36E+13	1.49E+13	2.56E+13	3.92E+13	4.83E+13	4.04E+13	2.00E+13	2.67E+12	3.60E+11
bi212	1.36E+13	1.49E+13	2.56E+13	3.92E+13	4.83E+13	4.04E+13	2.00E+13	2.67E+12	3.60E+11
rn220	1.36E+13	1.49E+13	2.56E+13	3.92E+13	4.83E+13	4.04E+13	2.00E+13	2.67E+12	3.60E+11
Table	1.63E+18	7.64E+17	1.59E+17	7.78E+16	4.22E+16	2.04E+16	4.03E+15	1.27E+15	8.90E+14

Table 6.6. Radiological source terms (Bq/MTHM) for U-Th fuel

Decay time (yrs)	0.01	0.1	1	3	10	30	100	300	500
cs137	5.35E+15	5.34E+15	5.23E+15	5.00E+15	4.25E+15	2.68E+15	5.34E+14	5.32E+12	5.30E+10
ba137m	5.07E+15	5.06E+15	4.95E+15	4.73E+15	4.03E+15	2.54E+15	5.06E+14	5.04E+12	5.02E+10
y90	4.92E+15	4.86E+15	4.75E+15	4.53E+15	3.82E+15	2.36E+15	4.38E+14	3.55E+12	2.88E+10
sr90	4.86E+15	4.85E+15	4.75E+15	4.53E+15	3.82E+15	2.36E+15	4.38E+14	3.55E+12	2.87E+10
ce144	5.56E+16	5.13E+16	2.31E+16	3.90E+15	7.75E+12	1.48E+05	0.00E+00	0.00E+00	0.00E+00
pr144	5.56E+16	5.13E+16	2.31E+16	3.90E+15	7.75E+12	1.48E+05	0.00E+00	0.00E+00	0.00E+00
cs134	8.58E+15	8.33E+15	6.16E+15	3.15E+15	3.00E+14	3.65E+11	2.28E+01	0.00E+00	0.00E+00
pu241	2.82E+15	2.81E+15	2.69E+15	2.44E+15	1.74E+15	6.59E+14	2.21E+13	9.34E+09	7.86E+09
ru106	1.10E+16	1.04E+16	5.62E+15	1.44E+15	1.22E+13	1.49E+07	2.94E-14	0.00E+00	0.00E+00
rh106	1.10E+16	1.04E+16	5.62E+15	1.44E+15	1.22E+13	1.49E+07	2.94E-14	0.00E+00	0.00E+00
kr85	7.46E+14	7.42E+14	7.00E+14	6.15E+14	3.92E+14	1.08E+14	1.19E+12	2.99E+06	7.55E+00
pu238	1.27E+14	1.28E+14	1.31E+14	1.30E+14	1.23E+14	1.05E+14	6.03E+13	1.25E+13	2.58E+12
cm244	5.92E+13	5.90E+13	5.70E+13	5.28E+13	4.04E+13	1.88E+13	1.29E+12	6.10E+08	2.89E+05
pr144m	5.31E+14	4.90E+14	2.20E+14	3.72E+13	7.40E+10	1.41E+03	0.00E+00	0.00E+00	0.00E+00
am241	3.21E+12	3.62E+12	7.57E+12	1.58E+13	3.86E+13	7.24E+13	8.41E+13	6.16E+13	4.47E+13
cm242	1.00E+15	8.72E+14	2.15E+14	9.75E+12	1.21E+11	1.09E+11	7.74E+10	2.90E+10	1.08E+10
pu240	6.45E+12	6.45E+12	6.45E+12	6.46E+12	6.49E+12	6.54E+12	6.54E+12	6.41E+12	6.27E+12
pu239	5.45E+12	5.48E+12	5.48E+12	5.48E+12	5.48E+12	5.47E+12	5.46E+12	5.43E+12	5.41E+12
nb95	7.14E+16	6.44E+16	2.96E+15	1.11E+12	1.06E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zr95	6.87E+16	4.81E+16	1.37E+15	5.04E+11	4.80E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00
np239	9.61E+16	6.53E+12	4.74E+11	4.74E+11	4.74E+11	4.73E+11	4.70E+11	4.61E+11	4.52E+11
te127m	3.73E+14	3.13E+14	3.88E+13	3.72E+11	3.23E+04	2.15E-16	0.00E+00	0.00E+00	0.00E+00
te127	2.50E+15	3.13E+14	3.80E+13	3.64E+11	3.16E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00
y91	6.11E+16	4.14E+16	8.42E+14	1.47E+11	1.03E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sr89	5.30E+16	3.38E+16	3.71E+14	1.65E+10	9.65E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ru103	3.66E+16	2.05E+16	6.17E+13	1.54E+08	3.75E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rh103m	3.62E+16	2.03E+16	6.10E+13	1.52E+08	3.71E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ce141	6.49E+16	3.22E+16	2.91E+13	5.00E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
te129m	1.75E+15	8.90E+14	1.01E+12	2.88E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
te129	1.11E+15	5.62E+14	6.37E+11	1.82E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
co60	5.43E-01	5.36E-01	4.76E-01	3.66E-01	1.46E-01	1.05E-02	1.06E-06	1.97E-10	1.97E-10
rb86	1.31E+14	3.85E+13	1.88E+08	2.95E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
co58	7.59E-03	5.51E-03	2.21E-04	1.74E-07	2.39E-18	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pr143	5.83E+16	1.11E+16	5.65E+08	3.52E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
la140	6.72E+16	1.15E+16	1.99E+08	1.13E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ba140	5.94E+16	9.95E+15	1.73E+08	9.85E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cs136	1.94E+15	3.44E+14	1.04E+07	2.02E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nd147	1.89E+16	2.38E+15	2.31E+06	2.16E-14	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
kr85m	2.28E+10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
kr87	5.95E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
kr88	2.21E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
xe133	5.39E+16	7.20E+14	9.59E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
xe135	2.44E+14	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
xe135m	1.14E+12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rb88	2.47E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ba139	7.15E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sr91	1.15E+14	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
sr92	1.18E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i131	2.93E+16	1.73E+15	8.07E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i132	2.54E+16	2.07E+13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i133	4.18E+15	1.59E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i134	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
i135	6.66E+12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
te131m	1.53E+15	1.10E+08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
te132	2.47E+16	2.01E+13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
te131	4.02E+14	2.89E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rh105	4.16E+15	7.97E+08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ru105	2.51E+10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
mo99	2.66E+16	6.68E+12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
tc99m	2.57E+16	6.46E+12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nb97	1.77E+15	1.25E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
nb97m	1.67E+15	1.10E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ce143	1.07E+16	6.90E+08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
zr97	1.76E+15	1.16E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Table 6.6. Continued

Decay time (yrs)	0.01	0.1	1	3	10	30	100	300	500
la141	1.41E+10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
la142	3.28E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
y92	9.65E+09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
y93	1.83E+14	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
y91m	7.40E+13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
pm147	7.90E+15	7.90E+15	6.25E+15	3.68E+15	5.79E+14	2.94E+12	2.72E+04	0.00E+00	0.00E+00
eu154	2.80E+14	2.78E+14	2.58E+14	2.20E+14	1.25E+14	2.49E+13	8.84E+10	8.83E+03	8.82E-04
s325	4.23E+14	4.17E+14	3.33E+14	2.01E+14	3.46E+13	2.28E+11	5.22E+03	0.00E+00	0.00E+00
te125m	9.20E+13	9.37E+13	8.13E+13	4.93E+13	8.48E+12	5.57E+10	1.28E+03	0.00E+00	0.00E+00
u232	4.32E+13	4.32E+13	4.28E+13	4.20E+13	3.91E+13	3.20E+13	1.58E+13	2.11E+12	2.83E+11
ra224	1.12E+13	1.22E+13	2.09E+13	3.20E+13	3.94E+13	3.29E+13	1.63E+13	2.18E+12	2.94E+11
p412	1.12E+13	1.22E+13	2.09E+13	3.20E+13	3.94E+13	3.29E+13	1.63E+13	2.18E+12	2.94E+11
bi212	1.12E+13	1.22E+13	2.09E+13	3.20E+13	3.94E+13	3.29E+13	1.63E+13	2.18E+12	2.94E+11
po216	1.12E+13	1.22E+13	2.09E+13	3.20E+13	3.94E+13	3.29E+13	1.63E+13	2.18E+12	2.94E+11
rn220	1.12E+13	1.22E+13	2.09E+13	3.20E+13	3.94E+13	3.29E+13	1.63E+13	2.18E+12	2.94E+11
Table	1.74E+18	7.25E+17	1.00E+17	4.05E+16	1.97E+16	1.12E+16	2.24E+15	1.31E+14	7.13E+13

6.4 GAMMA SPECTRA

It is well known that the decay chain of ^{232}U contains substantial gamma emitters, the principal of those being ^{208}Tl (see Figure 5.19): ^{232}U decays to ^{228}Th with a half-life of 68.9 years, after which ^{228}Th decays to ^{224}Ra with a half-life of 1.9 years. The remaining decay chain is rather short lived and leads to ^{208}Tl, which emits four principal gamma rays; the highest yield is the 2.62 MeV gamma, which would cause major shielding and handling concerns. In addition to ^{208}Tl, ^{212}Bi is also a gamma emitter with a number of medium- to high-energy gammas.

In order to illustrate the impact of the ^{232}U decay chain, the gamma spectra for the four fuel types have been plotted in Figures 6.14–6.17 using the 47-group Bugle structure in SCALE 6.1.2. The gamma spectra were generated from the 45 GWd/MTHM typical burnup results, and the gamma spectra have been plotted at four decay times: 0.1 year, 1 year, 10 years, and 100 years of decay in Figures 6.14–6.17, respectively. In Figures 6.18 and 6.19, the gamma spectra for only the U-Th fuel have been plotted for decay times of 30 and 100 years with the principal ^{208}Tl and ^{212}Bi gammas marked with vertical arrows. The height of each arrow corresponds to the intensity of the isotope for that particular group. Corresponding plots of 0.1 and 1 year have been omitted as the gamma spectra at these times are still dominated by fission products and thallium and bismuth do not yet play an important role.

After 0.1 year (~30 days) of decay, the gamma spectra for all fuels are similar, with the thorium-bearing fuels having some higher intensity but low energy gammas. After 1 year of decay, the 2.62 MeV gamma from ^{208}Tl begins to have an impact. After 30 and 100 years of decay, the impact of the 2.62 MeV gamma is very apparent, resulting in a very visible spike in the gamma spectra for Pu-Th and U-Th at this energy. These results indicate that additional shielding will be required for intermediate- and long-term handling and storage of thorium-bearing fuels.

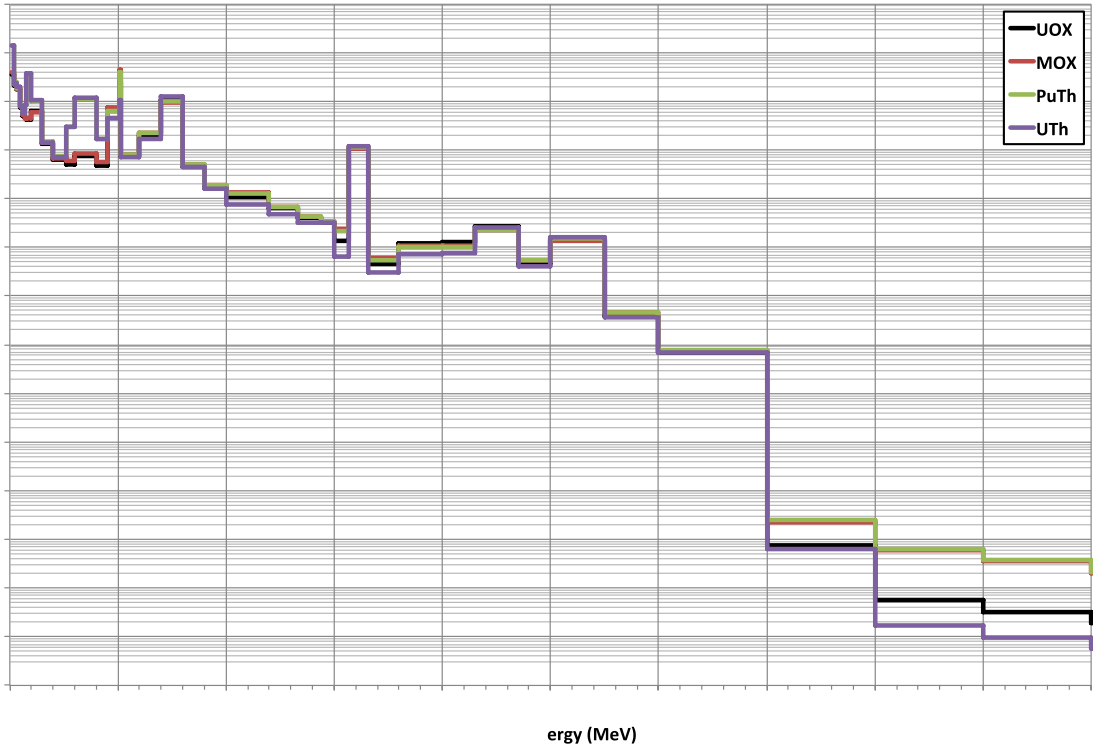


Figure 6.14. Gamma-ray spectra for the four fuel types after 0.1 year of decay.

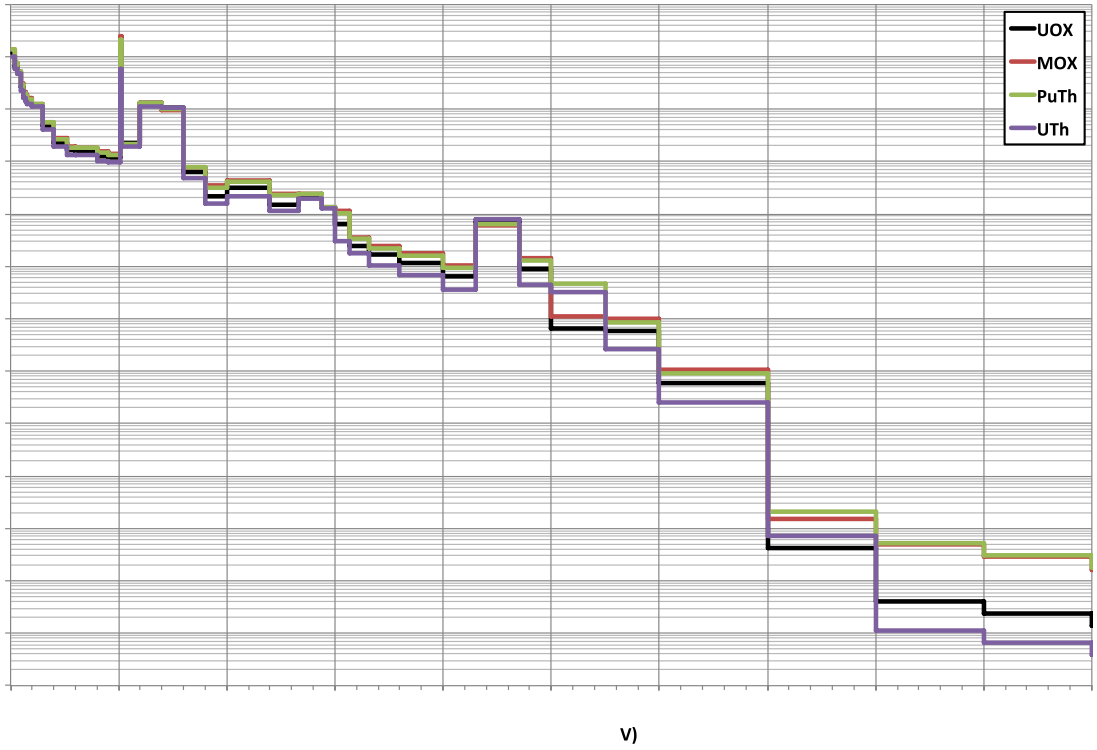


Figure 6.15. Gamma-ray spectra for the four fuel types after 1 year of decay.

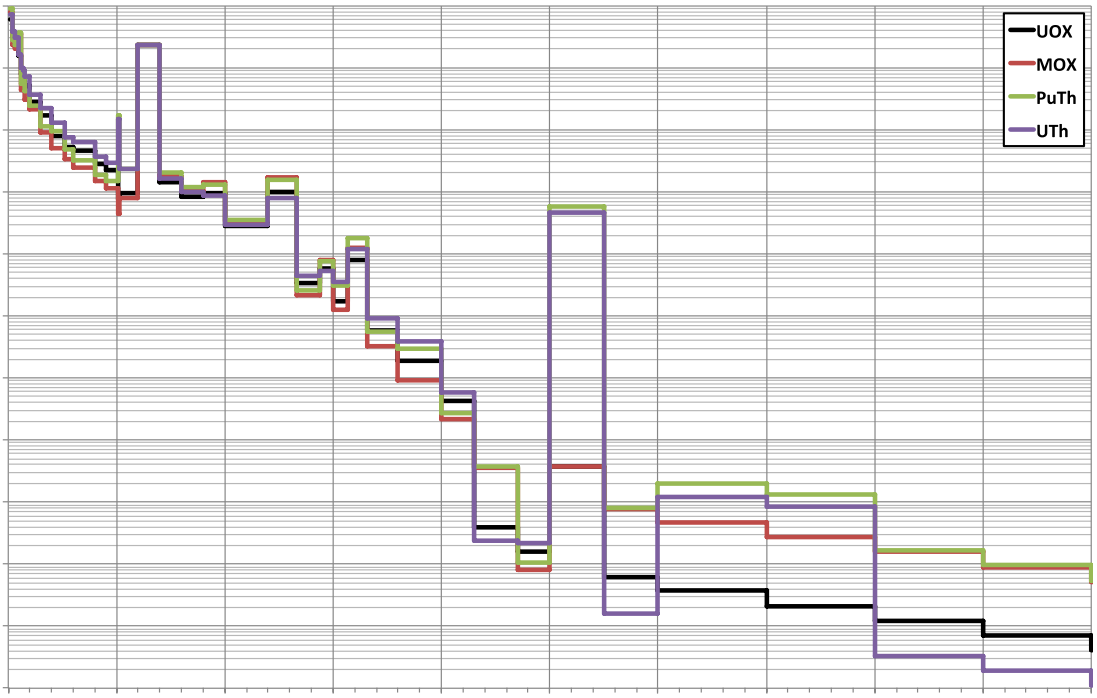


Figure 6.16. Gamma-ray spectra for the four fuel types after 10 years of decay.

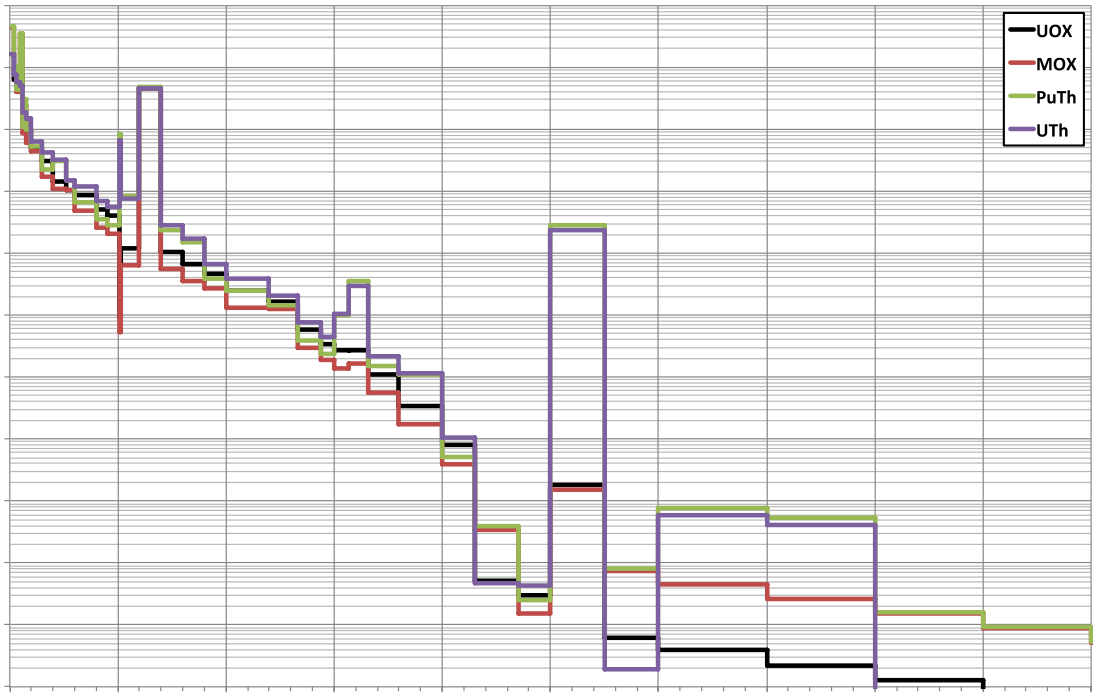


Figure 6.17. Gamma-ray spectra for the four fuel types after 100 years of decay.

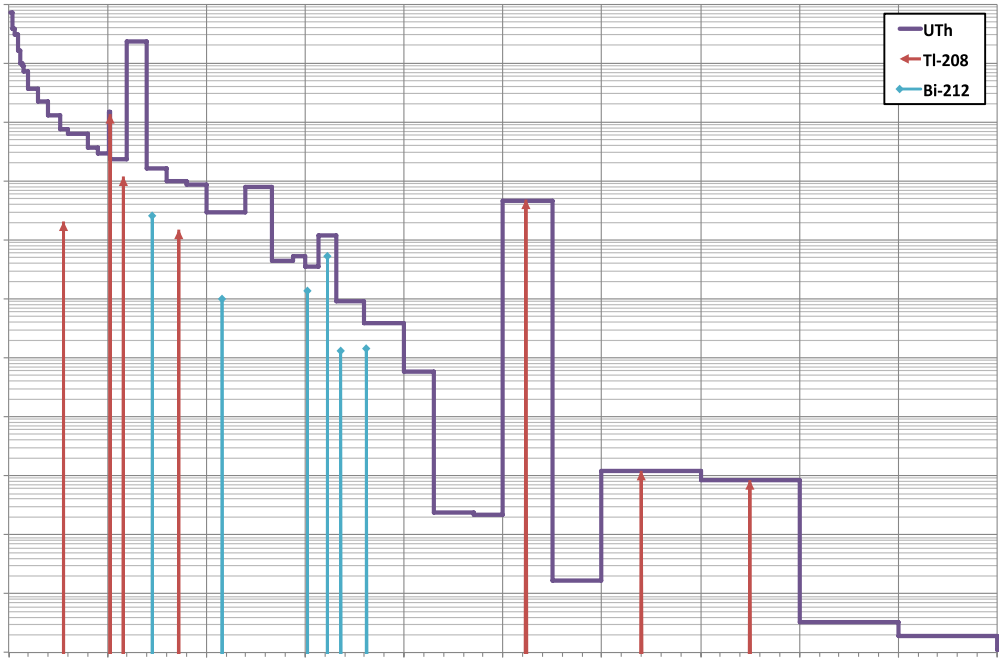


Figure 6.18. Gamma-ray spectrum for U-Th fuel with identification of major ^{232}U-decay chain gamma emitters after 30 years of decay.

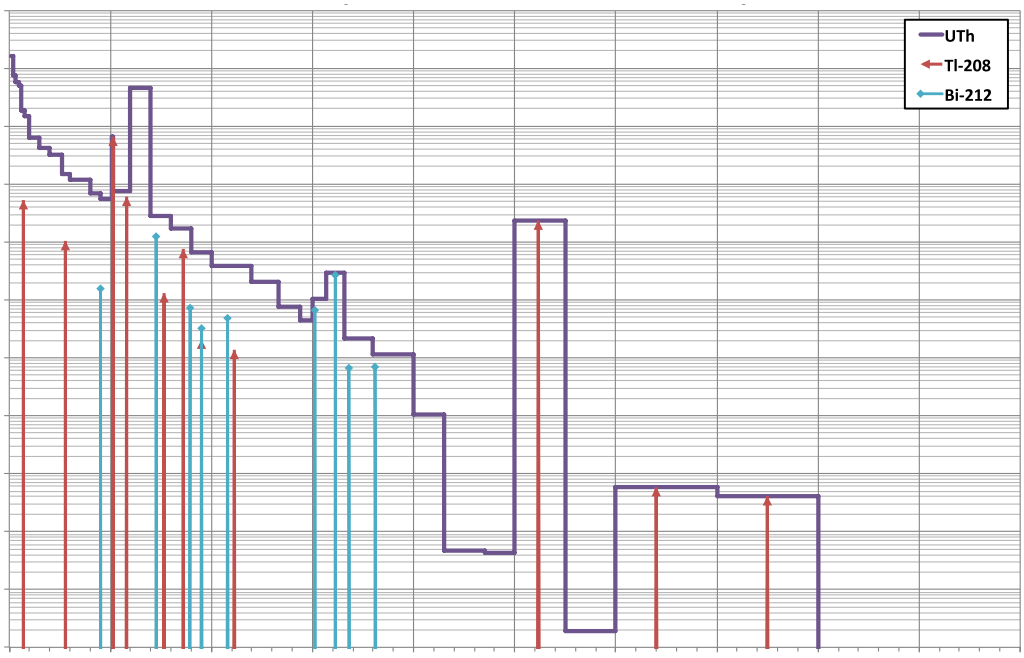


Figure 6.19. Gamma-ray spectrum for U-Th fuel with identification of major ^{232}U-decay chain gamma emitters after 100 years of decay.

6.5 SUMMARY OF THORIUM-BASED FUEL OUT-OF-REACTOR EVALUATION

Using the SCALE 2D fuel assembly models from the in-reactor analyses, ORIGEN calculations were performed for low-, normal-, and high-discharge burnup values. The calculations were performed for all four fuel types to compare the depleted fuel isotopics, decay heat, radiological source terms, and gamma spectra.

The masses of ^{233}U and Pu versus burnup were compared to determine how much of these fissile materials were bred or depleted for each fuel type. The Pu-Th fuel produces a greater amount of ^{233}U than the U-Th fuel due to the increased fertile load present in the Pu-Th fuel and the harder neutron spectrum of the Pu-Th fuel. The U-Th and Pu-Th fuel types have less plutonium at discharge than UOX and MOX, respectively, which is desirable for nonproliferation. The reason that the U-Th and Pu-Th fuel types have less plutonium at discharge is because they initially have less ^{238}U for breeding ^{239}Pu. A large portion of uranium is replaced by thorium in the initial fuel loading for the thorium-based fuels.

Immediately after shutdown (1 hour), the decay heat for the plutonium-based fuel types (MOX and Pu-Th) is slightly lower than the uranium-based fuel types (UOX and U-Th). The U-Th fuel decay heat is 25% higher than that for UOX at 1 month due to ^{232}Pa, but is less than or similar to UOX for longer decay times. The largest nuclide contributor to decay heat for the first 0.1 year (~30 days) for thorium-based fuel types is ^{233}Pa, a daughter of ^{232}Th neutron capture. Likewise, ^{242}Cm is the largest contributor for the first year for MOX fuel. In the case of Pu-Th fuel, the decay heat contribution of ^{233}Pa and ^{242}Cm is nearly equal after 0.1 year of decay. The plutonium-based fuels have considerably larger decay heat values for 10 years or longer. In general, fuels that initially contain uranium (UOX and U-Th) follow the same general curve, while the fuels that initially contain plutonium (MOX and Pu-Th) follow a higher curve.

At short decay times (less than 1 year), the thorium-based fuels have significantly higher source terms (up to 55%) due to the ^{233}Pa decay. From 1 year to 100 years, the U-Th and UOX source terms are similar. However, at 300 to 500 years, the U-Th source term drops to almost half the UOX value as the contributions from the Th/^{233}U decay chains die away. The total source term for Pu-Th fuel is similar to the U-Th value for less than 1 year. For decay times of 1 year or greater, the Pu-Th source term is similar to the MOX source term, because the plutonium source term dominates. At 300 to 500 years, the source terms for the plutonium-based fuels are ~6 to 10 times greater than the UOX and U-Th fuels, respectively.

The decay chain of ^{232}U contains substantial gamma emitters, the principal of those being ^{208}Tl, which emits four principal gamma rays; the highest yield is the 2.62 MeV gamma, which would cause major shielding and handling concerns. In addition to ^{208}Tl, ^{212}Bi is also a gamma emitter with a number of medium- to high-energy gammas. At 0.1 year of decay, the gamma spectra for all fuels are similar, with the thorium-bearing fuels having some higher intensity but low energy gammas. At 1 year of decay, the 2.62 MeV gamma from ^{208}Tl begins to have an impact. At 10 and 100 years of decay, the impact of the 2.62 MeV gamma results in a very visible spike in the gamma spectra for Pu-Th and U-Th. These results indicate that additional shielding will be required for intermediate- and long-term handling and storage of thorium-based fuels.

7 SUMMARY AND CONCLUSIONS

The primary objectives of this report were to summarize historical, current, and proposed uses of thorium in nuclear reactors; provide some important properties of thorium fuel; perform qualitative and quantitative evaluations of both in-reactor and out-or-reactor safety issues and requirements specific to a thorium-based fuel cycle for current LWR reactor designs; and identify key knowledge gaps and technical issues that need to be addressed for the licensing of thorium LWR fuel in the United States.

The in-reactor and out-of-reactor evaluations included both qualitative and quantitative assessments based on knowledge of the current safety basis for LWRs. These evaluations were conducted by performing a review of *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition* (NUREG-0800) and calculations using the SCALE code system.

For the qualitative assessment, each of the 19 chapters of NUREG-0800 was reviewed in order to identify key properties, phenomena, or issues caused by the use of thorium that would require alternative assessment or consideration. To capture and reflect the findings, for each of the chapters in which issues were identified, a table has been prepared (see Appendix A) that contains the chapter section and page number; excerpt of the text from NUREG-0800 that contained the issue affected by thorium; a brief explanation as to how thorium affects the safety basis, and an indication of the area impacted by the issue. During the review, it was clear that there were common phenomena that affected multiple areas across NUREG-0800, and so this latter item was included as a useful summary to identify common themes, gaps in knowledge, and classifications of requirements for future improvements to develop the technical basis for licensing thorium fuels. This assessment is not intended to be an exhaustive list of all technical or regulatory issues that would arise when considering use of thorium fuels, but the assessment does provide a clear indication of the topic areas that need to be addressed going forward.

For the quantitative assessment, a large number of calculations using the SCALE code system were completed to examine in-reactor behavior of thorium-based fuel. The models were based on a 17×17 Westinghouse PWR fuel assembly, typical of those in operation in the United States today. As a basis for comparison, results for uranium oxide (UO_2, or UOX) and mixed oxide ($UO_2 + PuO_2$, or MOX) were generated. Equivalent fuel compositions for uranium/thorium oxide ($UO_2 + ThO_2$) and plutonium/thorium oxide ($PuO_2 + ThO_2$) were then generated. Using the fuel compositions, a number of different analyses that included calculation of reactivity coefficients (fuel temperature, moderator temperature, boron), pin power peaking factors, assembly power sharing, boron letdown, and controlled lattice reactivity were performed.

Regarding out-of-reactor issues, for a once-through cycle with typical discharge burnups, the initial ^{235}U enrichment for U-Th fuels would need to be greater than the 5% limit currently used in LWRs and potentially could approach the 20% LEU limit in order to provide sufficient fissile content to maintain a suitable power level throughout normal operation. Prior to irradiation (handling, fuel manufacture, storage, etc.), the primary safety concern would be criticality safety limits for uranium enrichment beyond 5%. The addition of thorium to the fuel would likely have little impact prior to irradiation, because it is not fissile or highly radioactive in its natural state.

Thus, processes in the front-end of the fuel cycle would not be greatly impacted. However, issues in the back-end of the fuel cycle could arise for thorium-based fuels due to different decay heat and radiotoxicity characteristics associated with ^{232}Th irradiation. Depleted fuel isotopics, decay heat, and radiological source terms for thorium-based fuels have been calculated and compared to UOX and MOX fuel. For licensing applications, uncertainties would need to be assessed for spent fuel composition calculations used in setting source terms for radioactive material inventory, radiation source term and decay heat calculations, and for setting burned fuel compositions for criticality calculations.

There are several key phenomena associated with thorium and ThO_2 that differ notably from typical UO_2 fuel. Several of these phenomena are related to the nuclear properties of thorium versus uranium or plutonium, and several are chemical or material properties of ThO_2 or mixed thorium oxides ($ThO_2 + UO_2/PuO_2$) versus UO_2 or MOX fuels.

Key fundamental nuclear data include

- fission neutron yield (nu-bar) data,
- decay chains,
- cross sections,
- gamma data, and
- fission product yields

These fundamental nuclear properties have impacts on a number of key areas related to reactor and safety analyses, including steady state and transient performance, fuel handling and management (fresh and irradiated), reactor operations, and waste management.
The uncertainties on these data and the resulting impact on key safety parameters need to be fully evaluated. For example, a review of the literature reveals the branching ratios (emission probabilities) for characteristic gamma rays from the decay of ^{233}U are known only to an accuracy of 10% or poorer.

Key chemical or material properties include

- thermal conductivity,
- thermal expansion,
- chemical stability,
- melting temperature (and sintering temperature),
- grain size, and
- ceramic compatibility.

The changes in these parameters with irradiation are also key to understanding the overall performance of the fuel. Additionally, several more phenomena are interrelated between nuclear, chemical, material, and other physical phenomena. These key interrelated phenomena include

- decay heat,
- reactivity feedback coefficients,
- fission gas retention/release,
- fuel densification, swelling, and creep,
- fuel microstructure evolution under irradiation,

- shutdown margin,
- criticality,
- radiotoxicity, and
- core stability.

In order to fully address the requirements of NUREG-0800, these fundamental properties and phenomena need to be measured and thoroughly analyzed and characterized. Because thorium fuel is not as fully characterized as uranium or plutonium fuel, there are several challenges that need to be considered and addressed prior to the deployment of thorium fuels. Most of these challenges relate to the lack of experimental data and experiential knowledge using thorium. The following subsections discuss specific issues related to the need for measured or experimental data to be able to adequately address licensing of thorium fuels in LWRs.

7.1 AVAILABILITY OF MEASURED DATA

Computer simulation of measured data can and should be used to establish the relationships between calculated results and reality. Such studies are often used to validate computational methods, thereby demonstrating the adequacy of the computational method used and adding confidence in the results. Very little measured data are available for use in validating some key computational results that are needed for safety analysis of UO_2-ThO_2 fuel systems.

7.2 VALIDATION OF BURNED FUEL COMPOSITION CALCULATIONS

The ability to accurately calculate the composition of burned nuclear fuel is foundational. The radioactive material inventory (1) is used in offsite dose calculations, (2) is used as the source term for radiation shielding calculations, (3) provides the decay heat source term in fuel temperature calculations, and (4) determines the composition of the fuel used in calculation of k_{eff} used for determining subcritical margin at various points throughout irradiation and storage. More than 100 burned LWR UO_2 fuel samples have been at least partially characterized [60], but none of these burned fuel samples initially contained ThO_2. Consequently, there are little data available to quantify the accuracy of fuel composition calculations for irradiated U-Th or Pu-Th fuel systems.

7.3 VALIDATION OF DECAY HEAT CALCULATIONS

A large majority of the heat emitted from a fission event is released promptly in the form of the kinetic energy of the fission fragments. Following this initial prompt burst of energy, the radioactive fission products continue to decay, producing additional energy that is typically referred to as decay heat. The time dependence and amount of decay heat are important for safety analysis calculations performed to show that the fuel would be adequately cooled.

NUREG/CR-6972 [61] describes measured decay heat data and validation calculations performed using the SCALE computer code package. This report does include limited results for decay heat from ^{233}U fission and ^{232}Th fast fission. Further study is needed to evaluate the

adequacy of decay heat calculations for burned UO_2-ThO_2 fuel systems over the time ranges of interest.

7.4 VALIDATION OF CRITICALITY CALCULATIONS

Criticality calculations are validated by simulating laboratory critical experiments (LCEs) that are similar to the safety analysis models of interest. If the LCEs and safety analysis models have the same materials experiencing similar energy-dependent neutron fluxes, then both systems should have the same bias. If the LCEs or safety analysis models have materials that are in one and not the other, or the neutron spectra are different, the bias indicated by the LCEs may not be applicable to the safety analysis models. The SCALE sensitivity/uncertainty quantification capabilities can be used to help assess biases and uncertainties for applications that do not exactly match the LCE conditions.

In addition to criticality calculations, LCEs are needed to validate calculations and nuclear data for power distributions, absorber worth, reactivity coefficients, kinetics parameters, and other key reactor data.

Validation of unirradiated fuel criticality calculations

The 2012 version of the IHECSBE [62] includes descriptions for 1494 LEU critical configurations. Out of these, 1308 have enrichments no greater than 5 wt % ^{235}U, 186 configurations have enrichments between 5 and 10 wt %, and none have enrichments above 10 wt %. A total of 10 configurations from the LEU-COMP-THERM-060 evaluation include thorium. This LCE is a graphite-moderated lattice of LEU and is not adequately similar to water-moderated LEU to be useful for validation purposes.

Critical experiments should be used to validate a broad range of materials and conditions. For unirradiated material, validation should cover normal and abnormal conditions associated with the presence of LEU enriched up to 20 wt % ^{235}U for the processes and configurations associated with enrichment, conversion, fabrication, and transport of new fuel.

Critical experiments with fluorine and LEU enriched up to 20 wt % ^{235}U are needed to validate criticality calculations for UF_6 transport and uranyl fluoride solutions in fabrication processes. Critical experiments with LEU and thorium are needed to validate unirradiated fuel criticality calculations.

Additionally, it may be desirable to take credit for the presence of ^{232}Th in the ThO_2 to be added to the fuel. Because the thermal neutron capture cross section for ^{232}Th is greater than the capture cross section for ^{238}U, replacing some of the enriched UO_2 with ThO_2 could significantly reduce the reactivity of the fuel material. Depending upon where in the process the ThO_2 is added, critical experiments with both thorium and elevated uranium enrichments may be needed for some of these operations.

Validation of irradiated fuel criticality calculations

The primary reason for adding thorium to the fuel is to generate ^{233}U, which is then used as fuel in fission reactions. Currently, there is only a single set of critical experiments [63] that has uranium and plutonium isotopic vectors that are similar to what has been measured in highly burned UO_2 fuel that had initial enrichments below 5 wt% ^{235}U. These experiments do not include any thorium, ^{233}U, or uranium enriched to higher levels. While further study is needed, it is very unlikely that any LCEs currently exist that are appropriate for validating criticality calculations involving a mixture of actinides similar to what is expected from UO_2+ThO_2 fuel burned to levels commonly seen in existing commercial nuclear power plants.

The lack of LCE data similar to burned UO_2+ThO_2 fuel will need to be addressed either by performing such experiments or by quantifying the amount of additional safety margin needed to cover the potential k_{eff} biases introduced by the presence of unvalidated nuclide mixtures in the spent fuel models.

7.5 FUEL PERFORMANCE DATA NEEDS

There currently exist very little data on fuel performance of ThO_2 or ThO_2 mixed fuels. These data have historically been generated through irradiation testing in test reactors in the United States and abroad. Considerable irradiation testing data exist for UO_2 and for MOX fuels, and similar data would likely need to be generated in order to obtain key relations used for fuel performance codes. Semiempirical methods could be used to generate some of this data, but validation of the methods would still need to be supported by experimental results. In order to validate fuel performance codes for transients and severe accidents, detailed fuel transient data would be needed. Previously, the majority of the data was generated at the TREAT (Transient Reactor Test Facility), which is no longer operational.

7.6 PHENOMENA IDENTIFICATION AND RANKING

PIRTs have been generated that outline major characteristics and phenomena of thorium fuels related to their in-reactor use in LWRs as shown in Tables 7.1−7.4. These phenomena have been grouped into four categories: physical properties, nuclear data, fuel performance, and reactor safety. In addition, front-end and back-end issues are summarized in Tables 7.5 and 7.6, respectively.

These phenomena were given a high, medium, or low (H/M/L) ranking of importance, and the current level of knowledge was assessed as unknown, partially known, or known (U/P/K). The phenomena are color coded by category for clarity, and the rankings are color coded to visually identify the phenomena that are considered higher priorities.

The needs to increase the level of knowledge can generally be grouped into three main categories of gaps and future requirements as follows.

- **New data required,** for example, nuclear data (cross sections, gamma emission probabilities), fuel material properties

- **New analysis required,** for example, computational tools reevaluated to assess impact of new data, safety analysis reevaluated
- **New experiments required,** for example, critical experiments for code validation, material test reactor irradiations for demonstration of prototypic fuel, destructive radiochemical assay experiments for validation of isotopic fuel compositions in spent fuel

Significant needs exist in all categories to adequately address licensing of thorium fuels as outlined in the PIRTs. Despite the number of gaps identified, the process and review have indicated that by the use of exceptions (similar to what has been done for MOX fuel), thorium potentially could be licensed under the current regulations without additional rule making.

Table 7.1. PIRT for physical properties of thorium fuel in LWRs

Phenomenon or Characteristic	**Importance** (H=high, M=medium, L=low)	**Knowledge** (U=unknown, P=partially known, K=known)	**Commentary**	**Summary for Category**
Thermal Conductivity	H	P	Thermal conductivity for thorium oxides is fairly well known, but the properties when mixed with PuO_2 and UO_2 at varying fractions are not well studied.	Physical properties of ThO_2 have been studied historically though various research programs and ThO_2 has been used in LWRs previously. The limited research that is available is positive in that the physical properties of ThO_2 are generally preferable for use in LWRs. However, much less is known about the resulting physics properties of PuO_2/ThO_2 and UO_2/ThO_2 mixtures. Very little data are available for alternative fuel forms such as thorium nitrides, or other potential fuel candidates.
Thermal Expansion	H	P	Similar to thermal conductivity, thermal expansion characteristics for thorium oxides are fairly well known, but the properties when mixed with PuO_2 and UO_2 at varying fractions have not been well studied.	
Chemical Stability	H	P	Preliminary indications are that ThO_2 and thorium mixed oxides are chemically more stable than PuO_2 and UO_2, but further research is needed under irradiation conditions.	
Melting Temperature	H	P	The melting temperature of ThO_2 is greater than that for UO_2 and PuO_2, but the properties of mixed oxides of Pu/Th or U/Th are not well known.	
Heat Capacity	H	P	The heat capacity of ThO_2 is lower than that for UO_2 and PuO_2, and early research shows that in general, the heat capacity of mixed oxides of thorium are lower than the parent oxides; however, there is little research in this area.	

Table 7.2. PIRT for nuclear data of thorium fuel in LWRs

Phenomenon or Characteristic	**Importance** (H=high, M=medium, L=low)	**Knowledge** (U=unknown, P=partially known, K=known)	**Commentary**	**Summary for Category**
Cross Sections	H	P	The estimated uncertainties in the basic Th-232 and U-233 cross section data are on the same order as those for uranium and plutonium isotopes of interest; however, the relevance or impact of these uncertainties is not clearly known/understood due to limited availability of relevant benchmark data.	The nuclear data needed for licensing evaluations does exist, but the uncertainty in the data is not well known due to the lack of relevant experiments and other experiential knowledge that exists for typical UOX and MOX systems.
Decay Chains	H	P	Decay chains and fission products are reasonably well known, although further analysis would be needed in order to determine the quality of the data and if the quality is sufficient for licensing purposes.	
Delayed Neutrons	H	K	The delayed neutron characteristics, important for reactor control, transients, and decay, are reasonably well known.	
Gamma Emission Data	M	P	There are significant gamma emitters in the decay products of Th-232 and U-233. It is important to understand the impact of these data for storage of used fuel and shuffling partially used fuel.	
Fission Yields	H	P	Accurate fission yields, which are important for both reactor operation and spent fuel applications, are reasonably well known, although less experiential knowledge exists for U-233 than for other fissile nuclides (U-235 and Pu-239).	

Table 7.3. PIRT for fuel performance of thorium fuel in LWRs

Phenomenon or Characteristic	**Importance** (H=high, M=medium, L=low)	**Knowledge** (U=unknown, P=partially known, K=known)	**Commentary**	**Summary for Category**
Fission Gas Production/ Retention/ Release	H	U	Preliminary research suggests that fission gas production is higher in ThO_2 than UO_2 or MOX, but fission gas retention in ThO_2 fuel is better than in UOX and MOX. However, little is known regarding the release of fission gases from mixed oxides of thorium, especially those with high fractions of UO_2 and PuO_2.	Most research suggests that fuel performance characteristics of ThO_2 are similar to or slightly better than those of UO_2, PuO_2, and MOX. However, the research is rather limited, especially for fuel compositions of mixed thorium oxides including $(U,Th)O_2$ and $(Pu,Th)O_2$. It is likely that fuel irradiation experiments would be needed to generate and validate fuel performance data and codes. There exists some experiential knowledge of ThO_2 fuels from early experiments and in reactor operation; however, uncertainties are large compared to UO_2, especially for high-burnup and transient conditions.
Dimensional Changes – Swelling/ Densification/ Creep	H	U	Preliminary research suggests that dimensional changes of ThO_2 fuel under irradiation are lower (better) than those of UOX and MOX, but little is known regarding the irradiation-induced dimensional changes of mixed oxides of thorium.	
Microstructure Evolution	M	U	As with fission gas release and dimensional changes, early research suggests that ThO_2 has a more stable microstructure than UO_2, PuO_2, and MOX, but the research is limited and little is known regarding mixed oxides of thorium.	
Fuel/Clad Interaction	H	P	Some PCI and PCMI fuel performance data for ThO_2 exists due to previous operating experience, but more detailed data would be needed for a specific fuel and reactor design.	
Fuel Bowing	H	P	Some fuel bowing data for ThO_2 exists due to previous operating experience, but more detailed data would be needed for a specific fuel and reactor design.	
Fuel Performance Models	H	U	Some preliminary research has been performed for ThO_2 fuel performance, but fuel performance models (FRAPCON) would need to be developed for thorium fuel or mixed oxides of thorium. These models would need to be validated against measured data, which do not currently exist.	

Table 7.4. PIRT for reactor safety of thorium fuel in LWRs

Phenomenon or Characteristic	Importance (H=high, M=medium, L=low)	Knowledge (U=unknown, P=partially known, K=known)	Commentary	Summary for Category
Reactivity Coefficients	H	P	The preliminary analyses performed in this work for $(U,Th)O_2$ and $(Pu,Th)O_2$ indicate that addition of thorium to a typical W17 assembly would tend to decrease (more negative) fuel temperature and moderator temperature reactivity coefficients.	In this work, a preliminary study of various reactor safety aspects was performed. The results show that basic reactor safety characteristics are similar to those of current UOX and MOX fuel, being slight better in some areas and slightly worse in others. Further study should be initiated when realistic fuel design information becomes available as changes in fuel composition, fuel lattice design, burnable poisons, etc., impact the reactor safety aspects. Whole core analyses are required to confirm the results presented in the ORNL lattice analysis, especially due to the fact that the use of thorium fuels is likely to be in a mixed core, combined with other fissile materials (whether uranium or plutonium). The mixed core will change the power share and the importance of each of the fuel types and in turn will determine the maximum and likely percentage of the core that can be loaded as thorium fuels. Whole core assessments to determine the burnable poison loadings, core fractions, and critical boron concentrations will affect the overall reactivity coefficients and control rod worths.
Control Rod Worth	H	P	Addition of thorium tends to increase control rod worth at BOL but to decrease it as a function of burnup, when compared to UOX. Full-core calculations are needed to estimate the true control rod worth.	
Critical Boron	M	P	The worth of boron tends to decrease with the addition of thorium, which could lead to a reduction in the effectiveness of safety systems that utilize boron injection. Full-core calculations are needed to estimate the true critical boron concentrations.	
Power Peaking	M	P	Pin power peaking increases slightly with the addition of thorium; however, the lattice designs were not optimized using radial enrichment grading or use of burnable absorbers. Results for optimized lattices designs could have peaking factors similar to those for typical UOX or MOX fuel assemblies. Full-core calculations are needed to estimate the true power peaking in the reactor.	
Fuel Inventory	H	P	The fuel isotopic inventory is highly dependent on the assembly and reactor design, as well as operating conditions, burnup, etc. The ability to predict the isotopic inventory has far-reaching impacts on reactor safety, spent fuel issues, as well as other areas. There exists very little data (destructive assay) to validate computational tools used in predicting the fuel inventory and associated characteristics.	

Table 7.5. PIRT for front-end fuel cycle issues of thorium fuel in LWRs

Phenomenon or Characteristic	**Importance** (H=high, M=medium, L=low)	**Knowledge** (U=unknown, P=partially known, K=known)	**Commentary**	**Summary for Category**
Fuel Enrichment	M	P	If U-Th fuel form is commercialized, it is likely that initial uranium enrichment would need to be greater than 5–10 wt% ^{235}U, which is the maximum of currently licensed enrichment plants in the United States. Analysis by ORNL used 20 wt% enriched fuel in order to provide sufficient fissile content. If greater than 10% enrichment is required, a current plant would need to be back fitted and relicensed, a new plant would need to be constructed, or fuel would need to be obtained via a foreign supplier. In addition, increased enrichment could result in issues relating to fuel fabrication, transport, and handling such as criticality safety, physical protection, etc.	The issues associated with thorium in nuclear fuel prior to irradiation are not a concern due to thorium, which is not radioactive or fissile prior to irradiations. However, changes in the nuclear fuel due to the addition of thorium, such as increased uranium enrichment, or use of plutonium to provide the fissile content of the fuel, could lead to significant issues associated with fuel fabrication, transport, handling, criticality safety and physical protection.
Sintering Temperature	L	K	Research has shown that the sintering temperature of ThO_2 is greater than that of UO_2 and PuO_2. This could lead to different equipment needs in order to fabricate ThO_2 fuel.	
Plutonium Issues	H	P	Currently, there is no fully licensed and functional capability in the United States to manufacture MOX fuel, which would also apply to Pu-Th fuel. However, the issues for MOX fuel have been researched in detail in previous NRC publications. In addition, there has been an application for the MOX Fuel Fabrication Facility. This facility, if successfully constructed, would likely be able to handle Pu-Th fuel in addition to typical MOX fuel. Similar to increased enrichment of uranium, fuel containing plutonium could result in issues related to fuel fabrication, transport, and handling such as criticality safety, physical protection, etc.	

Table 7.6. PIRT for back-end fuel cycle issues of thorium fuel in LWRs

Phenomenon or Characteristic	**Importance** (H=high, M=medium, L=low)	**Knowledge** (U=unknown, P=partially known, K=Known)	**Commentary**	**Summary for Category**
Fuel Inventory	H	U	The fuel isotopic inventory is highly dependent on the assembly and reactor design, as well as operating conditions, burnup, etc. The ability to predict the isotopic inventory has far-reaching impacts on reactor safety, spent fuel issues, as well as other areas. Very little destructive assay data exist to validate computational tools used in predicting the fuel inventory and associated characteristics. The depleted fuel inventory for thorium is very different from that for uranium or plutonium, and validation of the predicted isotopics would be needed. Accurate prediction of ^{232}Th, ^{233}Pa, ^{232}U and ^{233}U have significant impacts on reactor safety, decay heat, and gamma spectra. There is also significant decay of ^{233}Pa to ^{233}U after reactor shutdown, which could lead to spent fuel pool criticality issues.	Back end issues associated with thorium are mainly due to the different isotopics that are generated with thorium irradiation. These isotopics, such as ^{232}U and ^{233}Pa, are important to gamma spectra and decay heat. Because ^{233}Pa decays to fissile ^{233}U with a half-life of ~27 days, ^{233}Pa generated during irradiation will become ^{233}U within the first year of decay. This generation of additional fissile material in ^{233}U after shutdown could lead to criticality concerns.
Decay Heat	H	P	Calculations performed for this report showed that the decay heat for thorium-bearing fuel assemblies is higher than those without thorium for the first year of decay. A significant contributor to the additional heat is decay of ^{233}Pa to ^{233}U, so accurate prediction of ^{233}Pa production is important.	
Gamma Emission	H	P	There are significant gamma emitters in the decay products of ^{232}U, primarily ^{208}Tl, which emits a 2.62 MeV gamma. This high-energy gamma becomes very apparent in the gamma spectra after ~10 years of decay, and would likely require additional shielding compared to what is currently used in commercial nuclear power plants.	

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APPENDIX A

APPENDIX A

The detailed chapter reviews of NUREG-0800 have been captured in the tables that follow. Sections 4.2, 4.3, and 4.4 were reviewed individually due to their importance in reactor and safety analysis, and their dependencies on other chapters. Some chapters are not expected to be impacted; therefore, these chapters do not have a corresponding table.

Chapters 1–3, "Introduction and Interfaces," "Sites Characteristics and Site Parameters," and "Design of Structures, Components, Equipment, and Systems" are expected to be minimally impacted by addition of thorium.

Table A.1. Section 4.2, Fuel System Design

Section	Page/Reference	Relevant Text from NUREG	Issue	Area of Impact
4.2 Fuel System Design	Page 4.2-2	New fuel designs, new operating limits (e.g., rod burnup and power), and the introduction of new materials to the fuel system require a review to verify that existing design-basis limits, analytical models, and evaluation methods remain applicable for the specific design for normal operation, AOOs, and postulated accidents. The review also evaluates operating experience, direct experimental comparisons, detailed mathematical analyses (including fuel performance codes), and other information.	This implies that a significant amount of new information would need to be provided for Th fuels: fuel performance codes would need new direct experimental comparison data at the least and could need new methods development related to thorium fuel.	Nuclear Data Core analysis methods Transient analysis methods Fuel performance analysis methods Severe accident analysis methods Kinetic parameters
4.2 Fuel System Design	Page 4.2-3	The available radioactive fission product inventory in fuel rods (i.e., the gap inventory expressed as a release fraction) is provided to those organizations that estimate the radiological consequences of plant releases.	Fission product inventory would need to be recalculated and fission products from thorium and U-233 fuel might need to be reanalyzed. New, more accurate data for U-233 fission products could be needed, as well as consideration for hard gammas, or other key contributors to radiological consequences.	Nuclear data – cross sections, decay data Source term – radiological Fission gas release

Table A.1. Continued

Section	Page/Reference	Relevant Text from NUREG	Issue	Area of Impact
4.2 Fuel System Design	Page 4.2-4	Review of the postulated fuel failures resulting from overheating of cladding, overheating of fuel pellets, excessive fuel enthalpy, pellet/cladding interaction (PCI), and bursting under Chapter 15.	All applicable transients would need to be reanalyzed with updated data for thorium fuel.	Fission gas release Thermal conductivity Fuel swelling Fuel creep Melting point Core and transient analysis methods Fuel performance methods
4.2 Fuel System Design	Page 4.2-4	Review of the estimates of radiological dose consequences under Chapter 15.	Radiological doses would be different due to the update in fuel type.	Source term – radiological Fission gas release
4.2 Fuel System Design	Page 4.2-7	Fuel and burnable poison rod internal gas pressures should remain below the nominal system pressure during normal operation or other limits	Any calculations with internal gas pressure would need to be reanalyzed with updated data and/or methods.	Fuel performance methods Fission gas release Fuel swelling
4.2 Fuel System Design	Page 4.2-8	Acceptable moisture levels for Zircaloy-clad uranium oxide fuel should be no greater than 20 micrograms per gram	This number is specific to uranium-based fuel and is not necessarily applicable to thorium-based fuels.	Material impurity limits
4.2 Fuel System Design	Page 4.2-8	If axial gaps in the fuel pellet column result from densification, the cladding has the potential to collapse into a gap.	Fuel densification estimates would need to be updated with new data/methods for thorium-based fuel.	Fuel performance methods Fuel densification / swelling Thermal conductivity
4.2 Fuel System Design	Page 4.2-9	Traditional practice has also assumed that failure will occur if centerline melting takes place. This analysis should be performed for the maximum linear heat generation rate anywhere in the core, including all hot spots and hot channel factors, and should account for the effects of burnup and composition on the melting point.	Fuel performance calculations would need to be performed with updated fuel performance data/methods and updated determination of the maximum linear heat generation rate.	Fuel melting point Thermal conductivity Fuel performance methods

Table A.1. Continued

Section	Page/Reference	Relevant Text from NUREG	Issue	Area of Impact
4.2 Fuel System Design	Page 4.2-9	The sudden increase in fuel enthalpy from a reactivity initiated accident (RIA) below fuel melting can result in fuel failure due to pellet/cladding mechanical interaction (PCMI)	New data/analysis would be needed to understand the impact of sudden increases in enthalpy.	Thermal conductivity Fuel swelling Fuel creep Melting point Core and transient analysis methods Fuel performance methods Kinetic parameters
4.2 Fuel System Design	Page 4.2-9	PCI is generally caused by stress-corrosion cracking due to fission product (iodine) embrittlement of the cladding, while PCMI is primarily a stress-driven failure. The design basis for PCI and PCMI can only be generally stated.	Increased iodine production is possible with the new fuel; new analysis/data would be needed.	Thermal conductivity Fuel swelling Fuel creep Core and transient analysis methods Fuel performance methods
4.2 Fuel System Design	Page 4.2-10	Evidence exists that gaseous swelling and fuel thermal expansion is responsible for cladding strains at high burnup levels and perhaps at even moderate burnups. Therefore, PCI or PCMI analyses of cladding strain for AOO transients and accidents should apply approved fuel thermal expansion and gaseous fuel swelling models, as well as irradiated cladding properties.	Fuel thermal expansion and gaseous swelling models/data need to be updated for the new fuel form.	Thermal conductivity Fuel swelling Fuel creep Core and transient analysis methods Fuel performance methods
4.2 Fuel System Design	Page 4.2-10	The ECCS evaluation model should include a calculation of the swelling and rupture of the cladding resulting from the temperature distribution in the cladding and from pressure differences between the inside and outside of the cladding.	TREAT test reactor performed many of the experiment analyses for fuel failures, but it is no longer operating.	Thermal conductivity Fuel swelling Fuel creep Fission gas release Core and transient analysis methods Fuel performance methods Data uncertainty

Table A.1. Continued

Section	Page/Reference	Relevant Text from NUREG	Issue	Area of Impact
4.2 Fuel System Design	Page 4.2-10	Regulatory Guide (RG) 1.157 provides guidelines for performing a realistic (i.e., best estimate) model to calculate the degree of cladding swelling and rupture.	RG 1.157 would need to be reanalyzed for applicability to thorium-based fuels.	Fuel performance methods Fission gas release Fuel swelling
4.2 Fuel System Design	Page 4.2-11	In severe RIAs, such as rod ejection in a PWR or rod drop in a BWR, the large and rapid deposition of energy in the fuel can result in melting, fragmentation, and dispersal of fuel. The mechanical action associated with fuel dispersal can be sufficient to destroy the cladding and the rod-bundle geometry of the fuel and produce pressure pulses in the primary system.	New analysis/data would be needed in order to evaluate such accidents for thorium-based fuel.	Thermal conductivity Melting temperature Core and transient analysis methods Kinetic parameters
4.2 Fuel System Design	Page 4.2-14	In-reactor testing of design features and lead-assembly irradiation of whole assemblies of a new design should be reviewed. The maximum burnup or fluence experience associated with such tests should also be reviewed and considered in relation to the specified maximum burnup or fluence limit for the new design.	New lead test assemblies and other irradiation data would be needed for thorium fuel assemblies.	Data uncertainty Fuels testing Fuel performance methods
4.2 Fuel System Design	Page 4.2-15	Alternatively, an ECCS evaluation model may be developed in conformance with the acceptable features of Appendix K to 10 CFR Part 50	Appendix K would need to be analyzed for applicability to thorium fuel.	Regulatory information
4.2 Fuel System Design	Page 4.2-15	Phenomenological models that should be reviewed include the following:	Nearly all of the phenomenological models noted following this piece of text would need to be updated for thorium fuel: radial power distribution, fuel and cladding temperature distribution	Nuclear Data Core analysis methods Transient analysis methods Fuel performance analysis methods Severe accident analysis methods Kinetic parameters

Table A.1. Continued

Section	Page/Reference	Relevant Text from NUREG	Issue	Area of Impact
4.2 Fuel System Design	Page 4.2-16	Because of the strong interaction between these models, overall code behavior should be checked against data (standard problems or benchmarks) and the NRC audit codes.	Fuel performance code review, similar to previous table entry, would be needed to ensure applicability to thorium fuel.	Lack of data Fuel performance analysis methods Core and transient analysis methods
4.2 Fuel System Design	Page 4.2-16	In addition to its effect on fuel temperatures (discussed above), densification affects (1) core power distributions (power spiking - see SRP Section 4.3), (2) the fuel linear heat generation rate (LHGR) - see SRP Section 4.4, and (3) the potential for cladding collapse. NUREG-0085 and RG 1.126 discuss densification magnitudes for power spike and LHGR analyses.	Calculations for these items would need to be redone, and RG 1.126 would need to be analyzed for applicability to thorium fuels.	Fission gas release Thermal conductivity Fuel swelling Fuel creep Melting point Core and transient analysis methods Fuel performance methods
4.2 Fuel System Design	Page 4.2-17	The memorandum on Request for Revised Rod Bowing Topical Reports, May 30, 1978, includes guidance for the analysis of fuel rod bowing…. At this writing, the causes of fuel rod bowing are not well understood and mechanistic analyses of rod bowing have not been approved.	Fuel rod bowing documents would need to be reviewed to determine whether they are still valid for thorium fuel.	Rod bowing Fuel creep fuel swelling Core and transient analysis methods Fuel performance methods
4.2 Fuel System Design	Page 4.2-17	The thermal performance code for calculating temperatures discussed in item 3.C.i above should be used to calculate fuel rod pressures in conformance with the fuel damage criteria of item 1.A.vi in Subsection II. This calculation should account for uncertainties in the estimated rod powers, code models, and fuel rod fabrication.	Uncertainties in rod powers and code models due to data uncertainties could be large due to the lack of data for thorium fuels.	Nuclear data needs Data uncertainties Fission gas release Fuel swelling Core and transient analysis methods Fuel performance methods

Table A.1. Continued

Section	Page/Reference	Relevant Text from NUREG	Issue	Area of Impact
4.2 Fuel System Design	Page 4.2-18	Fission Product Inventory - lots of regulatory guides, also: The NRC has plans to issue new guidelines for gap inventory (fission product release) from these accidents.	The regulatory guides (RGs) list on page 4.2-18 under fission product inventory would need to be analyzed, as there would be differences for thorium-based fuel. New guide might consider thorium, MOX as fuel.	Fission gas release Regulatory information Core analysis methods
4.2 Fuel System Design	Page 4.2-20	RG 1.25, RG 1.183, and RG 1.196 provide acceptable assumptions that may be used to evaluate the radiological consequences associated with a fuel-handling accident at a fuel handling and storage facility at reactor sites.	RGs would need to be analyzed for applicability to thorium fuels. In addition, it is known that the source term from thorium fuel would be much different from that for standard UO_2 fuels.	Regulatory information
4.2 Fuel System Design	Page 4.2-20	RG 1.77, RG 1.183, and RG 1.195 identify acceptable analytical methods and assumptions that may be used to evaluate the consequences of a rod ejection accident in PWRs.	RGs would need to be analyzed for applicability for thorium fuels.	Regulatory information
4.2 Fuel System Design	Page 4.2-22	These reports should also cite the applicable RGs (RG 1.3, RG 1.4, RG 1.25, RG 1.60, RG 1.77, RG 1.126, RG 1.157, and RG 1.183). Deviation from these guides or positions should be explained.	RGs would need to be analyzed for applicability for thorium fuels.	Regulatory information
4.2 Fuel System Design	Page 4.2-24	References	NUREG/CR reports and RGs would need to be reviewed for thorium fuels to confirm applicability. Some of the documents could need to be updated for thorium fuel.	Regulatory information
4.2 Fuel System Design	Page 4.2-33	APPENDIX B: INTERIM ACCEPTANCE CRITERIA AND GUIDANCE FOR THE REACTIVITY INITIATED ACCIDENTS	Much of the information/correlations in Appendix B are focused on uranium fuels. The appendix would likely need to be updated/reevaluated for thorium fuel.	Regulatory information

Table A.2. Section 4.3, Nuclear Design

Section	Page/Reference	Relevant Text from NUREG	Issue	Area of Impact
4.3 Nuclear Design	Page 4.3-2	The presentation of the core power distributions as axial, radial, and local distributions and peaking factors to be used in the transient and accident analyses. As discussed in Regulatory Guide (RG 1.206), power distributions within fuel pins are also required. These within-pin power distributions are important for pressurized-water reactor (PWR) and boiling-water reactor (BWR) applications as they affect isotopic buildup/burnup.	Within-pin power distribution would change with thorium fuel – new analysis needed. Power along the edge of the fuel pin could increase, as the production of U-233 from thorium is likely to be higher along the fuel edge.	Nuclear Data Core analysis methods Transient analysis methods Fuel performance analysis methods Severe accident analysis methods
4.3 Nuclear Design	Page 4.3-2	The effects of phenomena such as fuel densification should be included in these distributions and factors.	Fuel densification would change with the change in fuel type.	Nuclear data Fuel performance analysis methods Core analysis methods
4.3 Nuclear Design	Page 4.3-2	The translation of the design power distributions into operating power distributions, including…operating procedures and measurements, and necessary limits on these operations.	Changes in power distributions along with higher uncertainties may impact operating procedures and operating limits.	Data uncertainties Core and transient analysis methods Kinetics parameters
4.3 Nuclear Design	Page 4.3-2	The applicant's presentation of calculated nominal values for the reactivity coefficients, such as the moderator coefficient, which involves primarily effects from density changes and takes the form of temperature, void, or density coefficients; the Doppler coefficient; and power coefficients. The range of reactor states to be covered includes the entire operating range from cold shutdown through full power and the extremes reached in transient and accident analyses. The applicant should provide information on reactivity coefficients in the form of curves covering the full applicable range of the variables.	Reactivity coefficient would change as a result of thorium fuel.	Core and transient analysis methods Kinetics parameters Reactivity coefficients

Table A.2. Continued

Section	Page/Reference	Relevant Text from NUREG	Issue	Area of Impact
4.3 Nuclear Design	Page 4.3-3	The applicant's presentation of uncertainty analyses for nominal values, including the magnitude of the uncertainty and the justification of the magnitude by examination of the accuracy of the methods used in calculations (safety analysis report (SAR) Section 4.3.3), and comparison where possible with reactor experiments. For comparisons to experiments, it is important for the applicant to show that the experiments are applicable and relevant.	Uncertainty analysis would be very important for thorium fuels, as there are much less experimental data for thorium than for uranium. In addition, comparison to relevant experiments would be difficult due to the lack of experiments.	Data uncertainties Core and transient analysis methods
4.3 Nuclear Design	Page 4.3-3	The control requirements and provisions for control necessary to compensate for long-term reactivity changes of the core. These reactivity changes occur because of depletion of the fissile material in the fuel, depletion of burnable poison in some of the fuel rods, and buildup of fission products and transuranic isotopes.	Very different characteristics than typical UO_2 fuel. For a system whose breeding ratio is approaching 1.0, there would be very little change in reactivity and could result in an increase in reactivity over time. In addition, this increase would occur during shutdown as the buildup of U-233 from Pa-233 decay.	Core analysis methods
4.3 Nuclear Design	Page 4.3-3	The adequacy of the control systems to assure that the reactor can be returned to and maintained in the cold shutdown condition at any time during operation. The applicant shall discuss shutdown margins (SDM). Shutdown margins need to be demonstrated by the applicant throughout the fuel cycle.	The shutdown margin and evolution of the shutdown margin as a function of depletion is likely to change with thorium fuel.	Core analysis methods
4.3 Nuclear Design	Page 4.3-4	Descriptions and curves of maximum rates of reactivity increase associated with rod withdrawals, experimental confirmation of rod worths or other factors justifying the reactivity increase rates used in control rod accident analyses, and equipment, administrative procedures, and alarms which may be employed to restrict potential rod worths should be included.	Without currently operating thorium-fueled reactors, critical experiment data for validation of control rod worths would be needed.	Core and transient analysis methods

Table A.2. Continued

Section	Page/Reference	Relevant Text from NUREG	Issue	Area of Impact
4.3 Nuclear Design	Page 4.3-4	The area of criticality of the reactor during refueling. Discussions and tables giving values of keff for single assemblies and groups of adjacent fuel assemblies up to the number required for criticality, assuming the assemblies are dry and also immersed in water, are reviewed. The applicant needs to describe the basis for assuming that the maximum stated keff would not be exceeded.	This could be more important for Th/U-233 breeding systems due to the build-up of U-233 from Pa-233 decay (half-life of 27 days). This could have an impact on refueling operations.	Core analysis methods
4.3 Nuclear Design	Page 4.3-4	The areas concerning stability. These are: A. As per Section C.1.4.3.2.7 in RG 1.206, phenomena and reactor aspects that influence the stability of the nuclear reactor will be discussed by the applicant. B. Calculations and considerations given to xenon-induced spatial oscillations. C. Potential stability issues due to other phenomena or conditions, as presented by the applicant. D. Verification of the analytical methods for comparison with measured data.	All issues related to stability and oscillation would need to analyzed and verified through experiments. U-233 burn-in and the process and arrangement of fuel/targets may have a large impact on stability and power distributions.	Core and transient analysis methods Data uncertainties
4.3 Nuclear Design	Page 4.3-5	B. The database and/or nuclear data libraries used for neutron cross-section data and other nuclear parameters, including delayed neutron and photoneutron data and other relevant data. C. Verification of the analytical methods for comparison with measured data.	Nuclear libraries might need to be updated if data are sparse. Also, uncertainties would play a major role due to the scarcity of the data. Measured data would need to support analytical methods.	Nuclear data Data uncertainties Core analysis methods
4.3 Nuclear Design	Page 4.3-8	The design limits for power densities (and thus for peaking factors) during normal operation should be such that acceptable fuel design limits are not exceeded during anticipated transients and that other limits, such as the 1204C (2200F) peak cladding temperature allowed for loss-of-coolant accidents (LOCAs), are not exceeded during design-basis accidents.	The peak cladding temperature may no longer be applicable for thorium fuels.	Core and transient analysis methods Melting temperature Thermal conductivity

Table A.2. Continued

Section	Page/Reference	Relevant Text from NUREG	Issue	Area of Impact
4.3 Nuclear Design	Page 4.3-8	The acceptance criteria in the area of power distribution are that the information presented should satisfactorily demonstrate that: A. A reasonable probability exists that the proposed design limits can be met within the expected operational range of the reactor, taking into account the analytical methods and data for the design calculations; uncertainty analyses and experimental comparisons presented for the design calculations; the sufficiency of design cases calculated covering times in cycle, rod positions, load-follow transients, etc.; and special problems such as power spikes due to densification, possible asymmetries, and misaligned rods.	This entire set of requirements affecting power distribution would need to be demonstrated analytically and by experiment for thorium fuel.	Core and transient analysis methods Data uncertainties Fuel performance methods Fuel densification Fuel creep Fuel swelling
4.3 Nuclear Design	Page 4.3-9	Acceptance criteria for power spike models can be found in a NUREG report on fuel densification, and are discussed in Regulatory Guide (RG) 1.126.	It is unlikely that this NUREG (NUREG-0085, "The Analysis of Fuel Densification," July 1976) and RG are applicable for thorium fuels.	Fuel densification Regulatory information
4.3 Nuclear Design	Page 4.3-9	The judgment to be made under this SRP section is whether the reactivity coefficients have been assigned suitably conservative values by the applicant. The basis for that judgment includes the use to be made of a coefficient, i.e., the analyses in which it is important; the state of the art for calculation of the coefficient; the uncertainty associated with such calculations, experimental checks of the coefficient in operating reactors; and any required checks of the coefficient in the startup program of the reactor under review.	Some of these latter checks and estimates of uncertainty mentioned here would need to be explicitly demonstrated for a thorium reactor, since no experience exists from operating reactors. This further emphasizes the need for critical experiments for thorium systems.	Reactivity coefficients Kinetic parameters Core and transient analysis methods Data uncertainties

Table A.2. Continued

Section	Page/Reference	Relevant Text from NUREG	Issue	Area of Impact
4.3 Nuclear Design	Page 4.3-10	The reactor core's nuclear design is one of several key design aspects that ensure fuel design limits will not be exceeded during normal operations. Compliance with GDC 10 significantly reduces the likelihood of fuel failures occurring during normal operations, including anticipated operational occurrences, thereby minimizing the possible release of fission products to the environment.	Fuel design limits are the key to compliance with GDC 10; thorium fuels data are a major input requirement for meeting the review criteria for SRP sections 4.2 and 4.3.	Nuclear Data Core analysis methods Transient analysis methods Fuel performance analysis methods Severe accident analysis methods
4.3 Nuclear Design	Page 4.3-12	The review examines the calculation of effective delayed neutron fraction (βeff) and prompt neutron lifetime (*l**) and verifies that appropriate values are used in the reactivity accidents reviewed under SRP Sections 15.4.8 and 15.4.9. Regulatory Guide 1.77 provides guidance for calculating effective delayed neutron fraction and prompt neutron	These nuclear parameters would need to be reevaluated for thorium fuels, and RG 1.77 would need to be examined to ensure it is applicable to thorium fuels.	Kinetic parameters Core and transient analysis methods
4.3 Nuclear Design	Page 4.3-13	Some vendor codes do not use reactivity coefficients. When they are used, the reviewer determines from the applicant's presentations that suitably conservative reactivity coefficients have been developed for use in reactor analyses such as those for control requirements, stability, and transients and accidents.	It is unclear whether vendor codes can properly handle thorium due to lack of previous testing and validation data.	Kinetic parameters Reactivity coefficients Core and transient analysis methods
4.3 Nuclear Design	Page 4.3-15	The source of the neutron cross-sections used in fast and thermal spectrum calculations is described in sufficient detail so that the reviewer can confirm that the cross-sections are comparable to those in the current ENDF/B data files (i.e., ENDF/B-VII) and other sources of nuclear data, such as JENDL and JEFF3, etc. If modifications and normalization of the cross-section data have been made, the bases used must be determined to be acceptable.	More detail in cross-section data could be needed for thorium fuel, or relevant experiments to further validate the data.	Nuclear data Core and transient analysis methods

Table A.2. Continued

Section	Page/Reference	Relevant Text from NUREG	Issue	Area of Impact
4.3 Nuclear Design	Page 4.3-16	The procedures used to generate problem-dependent cross-section sets are given in sufficient detail so that the reviewer can establish that they reflect the state of the art. The reviewer confirms that the methods used for the following calculations are of acceptable accuracy: the fast neutron spectrum calculation; the computation of the uranium-238 resonance integral and correlation with experimental data; the computation of resonance integrals for other isotopes as appropriate (for example, plutonium-240); calculation of the Dancoff correction factor for a given fuel lattice; the thermal neutron spectrum calculation; the lattice cell calculations, including fuel rods, control assemblies, lumped burnable poison rods, fuel assemblies, and groups of fuel assemblies, and calculations of fuel and burnable poison depletion and buildup of fission products and transuranium isotopes.	Specifically, calculation of the U-238 resonance integral may not apply as there may be much less U-238 in the system as compared to a typical uranium-fueled system.	Nuclear data Data uncertainties Core and transient analysis methods
4.3 Nuclear Design	Page 4.3-16	The gross spatial flux calculations that are used in the nuclear design are discussed in sufficient detail so that the reviewer can confirm that the following items are adequate to produce results of acceptable accuracy: the method of calculation (e.g., diffusion theory, Sn transport theory, Monte Carlo, synthesis); the number of energy groups used; the number of spatial dimensions (1, 2, or 3) used; the number of spatial mesh intervals, when applicable; and the type of boundary conditions used, when applicable.	The group structure may no longer be applicable, as typical group structures are optimized for U-235/U-238/Pu-239 resonances. Difference resonances would likely necessitate an update to the group structure. The addition of thorium, especially in designs that have separate breeder blanket regions, could see large variations in flux that result in impacts to the accuracy of various calculations.	Nuclear data Data uncertainties Core and transient analysis methods

Table A.2. Continued

Section	Page/Reference	Relevant Text from NUREG	Issue	Area of Impact
4.3 Nuclear Design	Page 4.3-16	Verification of the database, computer codes, and analysis procedures has been made by comparing calculated results with measurements obtained from critical experiments and operating reactors. The reviewer ascertains that the comparisons cover an adequate range for each item and that the conclusions of the applicant are acceptable.	Critical experiment measurements for thorium are limited; new critical experiments for comparison would likely be needed.	Nuclear data Data uncertainties Core and transient analysis methods

Table A.3. Section 4.4, Thermal and Hydraulic Design

Section	Page/Reference	Relevant Text from NUREG	Issue	Area of Impact
4.4 Thermal and Hydraulic Design	Page 4.4-1	The review of new prototype plants, new critical heat flux (CHF) or critical power ratio (CPR) correlations, and new analysis methods require additional independent audit analyses. The required analyses may be in the following form: A. Independent computer calculations to substantiate reactor vendor analyses. B. Reduction and correlations of experimental data to verify processes or phenomena which are applied to reactor design. C. Independent comparisons and correlations of data from experimental programs. These reviews also include analyses of experimental techniques, test repeatability, and data reduction methods.	Critical heat flux and critical power ratio would likely change, and as such, A, B, and C would apply.	Fission gas release Thermal conductivity Fuel swelling Fuel creep Melting point
4.4 Thermal and Hydraulic Design	Page 4.4-2	The review evaluates the uncertainty analysis methodology and the uncertainties of variables and correlations such as CHF and CPR. The review also evaluates the uncertainties associated with the combination of variables.	These correlations could change and would need to have updated uncertainties assigned.	Fission gas release Thermal conductivity Fuel swelling Fuel creep Melting point Data uncertainties
4.4 Thermal and Hydraulic Design	Page 4.4-4	(GDC) 10, as it related to whether the design of the reactor core includes appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded….	Margins may decrease for thorium fuels due to lack of operational experience and larger uncertainties. In addition, SAFDLs may change for thorium fuels.	Fission gas release Thermal conductivity Fuel swelling Fuel creep Melting point Data uncertainties
4.4 Thermal and Hydraulic Design	Page 4.4-5	Uncertainties in the values of process parameters (e.g., reactor power, nuclear and engineering hot channel factors), core design parameters…should be treated with at least a 95-percent probability at the 95-percent confidence level.	Uncertainties would likely be greater for thorium fuels with respect to hot channel factors, power distributions, and core design parameters.	Fission gas release Thermal conductivity Fuel swelling Fuel creep Melting point Data uncertainties

Table A.3. Continued

Section	Page/Reference	Relevant Text from NUREG	Issue	Area of Impact
4.4 Thermal and Hydraulic Design	Page 4.4-5	Problems affecting DNBR or CPR limits, such as fuel densification or rod bowing, are accounted for by an appropriate design penalty which is determined experimentally or analytically. Sub-channel hydraulic analysis codes, such as those described in "TEMP-Thermal Enthalpy Mixing Program," BAW-10021.	The departure from nucleate boiling ratio (DNBR) and critical power ratio (CPR) limits would likely change, and those data/codes would need to be updated in order to perform evaluations. Fuel densification is not known for thorium fuels, and rod bowing may differ as well.	Fission gas release Thermal conductivity Fuel swelling Fuel creep Melting point Data uncertainties
4.4 Thermal and Hydraulic Design	Page 4.4-7	GDC 10 requires that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of AOOs.	GDC 10 would need to be analyzed for applicability to thorium fuels.	Regulatory information
4.4 Thermal and Hydraulic Design	Page 4.4-8	The reviewer must understand currently acceptable thermal and hydraulic design practice for the reactor type under review. This understanding can be most readily gained from (1) topical reports describing CHF correlations, system hydraulic models and tests, and core subchannel analysis methods, (2) standard texts and other technical literature which establish the methodology and the nomenclature of this technology, and (3) documents that summarize current staff positions concerning acceptable design methods.	Topical reports for CHF, standard texts, and current staff positions may not exist or be applicable for thorium systems.	Fission gas release Thermal conductivity Fuel swelling Fuel creep Melting point Data uncertainties / needs Regulatory information

Table A.3. Continued

Section	Page/Reference	Relevant Text from NUREG	Issue	Area of Impact
4.4 Thermal and Hydraulic Design	Page 4.4-9	The reviewer establishes that the thermal-hydraulic design and its characterization by minimum critical heat flux ratio (MCHFR) or DNBR have been accomplished and are presented in a manner that accounts for all possible reactor operating states as determined from operating maps.	MCHFR and DNBR need to be analyzed to ensure applicability to thorium fuel.	Fission gas release Thermal conductivity Fuel swelling Fuel creep Melting point Data uncertainties/needs Regulatory information

Chapter 5, "Reactor Coolant System and Connected Systems," is expected to be minimally impacted by addition of thorium.

Table A.4. Chapter 6, Engineered Safety Systems

Section	Page/Reference	Relevant Text from NUREG	Issue	Area of Impact
6.2.1 Containment Functional Design	Page 6.2.1-1	The containment design basis includes the effects of stored energy in the reactor coolant system, decay energy….	Refers to heat source, which would be affected by thorium	Source term – heat
6.2.2 Containment Heat Removal Systems	6.2.2-2	Potential effects include debris screen blockage, failure of pump seals, and other downstream components, and debris fouling of nuclear fuel	Fuel design, including geometry could affect this	Mechanical Design
6.2.2 Containment Heat Removal Systems	6.2.2-2	Review of fission product control features of containment heat removal systems is performed under SRP Section 6.5.2	May require reevaluation of the fission product inventory, including type of material, for example, gaseous, solid, etc.	Source term – heat
6.5.1 ESF Atmosphere Cleanup Systems	6.5.1-2	The environmental design criteria, the design pressure and pressure differential, relative humidity, maximum and minimum temperature, and radiation source term	Source term would be different for thorium fuels, so this would require reevaluation.	Source term – radiological
6.5.2 Containment Spray as a Fission Product Cleanup System	6.5.2-1	Sections of the SAR related to accident analyses, dose calculations, and fission product removal and control are reviewed to establish whether the applicant claims fission product scrubbing….	Source term would be different for thorium fuels, so this would require reevaluation.	Source term – radiological
6.5.3 Fission Product Control Systems and Structures	6.5.3-1	(a) provide a basis for developing the mathematical model for design basis loss-of-coolant accident (LOCA) dose computations….	Source term would be different for thorium fuels, so this would require reevaluation.	Source term – radiological
6.5.5 Pressure Suppression Pool as a Fission Product Cleanup System	6.5.5-1	Sections of the applicant's safety analysis report (SAR) related to accident analyses, accident dose calculations, and fission product….	Source term would be different for thorium fuels, so this would require reevaluation.	Source term – radiological Source term – heat

Table A.5. Chapter 6, Instrumentation and Control

Section	Page/Reference	Relevant Text from NUREG	Issue	Area of Impact
Appendix 7.1-A Acceptance Criteria and Guidelines for Instrumentation and Controls Systems Important to Safety	Appendix 7.1-A-12	Evaluation of I&C system contributions to design margin for reactor core and coolant systems should be a part of the review of the adequacy of I&C protective and control functions	Use of new fuel such as thorium coupled with less operating experience than for UO_2 fuels would require review.	Source term – heat Core analysis methods Reactivity coefficients
Appendix 7.1-A Acceptance Criteria and Guidelines for Instrumentation and Controls Systems Important to Safety	Appendix 7.1-A-20	GDC 25 Protection System Requirements for Reactivity Control Malfunctions	Fuel design limits would be affected by thorium fuels, including uncertainty. The effects on safety are covered in Chapter 15.	Core analysis methods Reactivity coefficients
Appendix 7.1-A Acceptance Criteria and Guidelines for Instrumentation and Controls Systems Important to Safety	Appendix 7.1-A-20	GDC 28, "Reactivity Limits"	Reactivity feedback coefficients and associated uncertainties would have to be reevaluated for thorium fuels.	Core analysis methods Reactivity coefficients
7.2 Reactor Trip System	7.2-5	RTSs should incorporate multiple means for responding to each event discussed in the SAR Chapter 15, "Transient and Accident Analyses."	Cross reference to the transient and accident analyses, which would be impacted by use of thorium fuels.	Core and transient analysis methods Reactivity coefficients Kinetics parameters

Table A.5. Continued

Section	Page/Reference	Relevant Text from NUREG	Issue	Area of Impact
7.4 Safe Shutdown Systems	7.4-2	Typical functions required for safe shutdown are: • Reactivity control • Reactor coolant makeup • Reactor pressure control • Decay heat removal • Suppression Pool Cooling (BWR)	Several items in this list would be affected by the use of thorium fuels (reactivity control, decay heat removal) and would need reevaluation.	Core analysis methods Transient analysis methods Source term – heat Reactivity coefficients Kinetics parameters
BTP 7-12 Guidance on Establishing and Maintaining Instrument Setpoints	BTP 7-12-2	...requires identification of the analytical limit associated with each variable. Clause 6.8.1 requires that allowances for uncertainties between the analytical limit and device setpoint…	Safety parameters, margins, and uncertainties would be affected by use of thorium fuels and would need to be reevaluated.	Core analysis methods Transient analysis methods Source term – heat Reactivity coefficients Kinetics parameters

Chapter 8, “Electric Power,” is not expected to be impacted by addition of thorium.

Table A.6. Chapter 9, Auxiliary Systems

Section	Page/Reference	Relevant Text from NUREG	Issue	Area of Impact
9.1.1 Criticality Safety of Fresh and Spent Fuel Storage and Handling	9.1.1-2	1. Fuel assembly design to verify that appropriate fuel assembly data were used	The use of thorium fuel would affect the fuel assembly design information contained in the nominal base case assessment. Although likely to be same mechanical design, carrier enrichment and fuel compositions would need to be reevaluated.	Mechanical design changes QA/QC of fuel production route and verification of material
9.1.1 Criticality Safety of Fresh and Spent Fuel Storage and Handling		3. Evaluation of performance effectiveness of the neutron absorbing materials in the fresh and spent fuel racks.	The fresh and spent thorium fuel would have different effects on the worth of the absorbing materials and as such would need to be reevaluated.	Criticality analysis methods, Depletion analysis methods, Reactivity coefficients
9.1.1 Criticality Safety of Fresh and Spent Fuel Storage and Handling		4. Computational methods and related data to verify that acceptable computational methods and data were used.	Methods for evaluation of fresh and irradiated thorium fuels would need to be reevaluated.	Criticality analysis methods, Depletion analysis methods, Nuclear data
9.1.1 Criticality Safety of Fresh and Spent Fuel Storage and Handling		5. Computational method validation to verify that the validation study is thorough and uses benchmark critical experiments that are similar to the normal-conditions and abnormal-conditions models and to verify that the neutron distribution coefficient (Keff) bias and bias uncertainty values are conservatively determined.	Validation data for thorium fuels would need to be reviewed. Critical experiments for these fuels are more limited than uranium fuels and therefore further validation studies would be required. This includes the need to reevaluate the nuclear data used by the analysis tools.	Criticality analysis methods, Depletion analysis methods, Nuclear data, Validation of codes, methods and nuclear data
9.1.1 Criticality Safety of Fresh and Spent Fuel Storage and Handling		7. Normal-conditions models to verify that normal conditions are modeled conservatively and that all modeling approximations and assumptions are appropriate.	With the in-growth of U-233 from the decay of Pa-233, thorium introduces need for additional analysis conservatisms.	Criticality analysis methods Depletion analysis methods, Nuclear data, Validation of codes, methods and nuclear data

Table A.6. Continued

Section	Page/Reference	Relevant Text from NUREG	Issue	Area of Impact
9.1.2 New and Spent Fuel Storage	9.1.2-1	1. The quantity of new and spent fuel to be stored	The reload scheme for thorium fuels is likely to be notably different from standard UO_2 fuels in order to allow sufficient neutron capture in thorium to produce U-233. In the event that a seed-blanket design is adopted, then there would be a different storage time for each of these two components.	Criticality analysis methods, Depletion analysis methods, Nuclear data
9.1.2 New and Spent Fuel Storage	9.1.2-2	4. The effectiveness of natural coolant circulation through the spent fuel storage racks and the ability of new fuel racks to drain fluids if the new fuel storage facility is intended for dry storage or to be flooded if the new fuel storage facility is intended for wet storage.	The decay heat of the irradiated thorium fuel would be different from UO_2 and would vary differently with time. Therefore a reevaluation would be required.	Source term – heat
9.1.2 New and Spent Fuel Storage		7. The ability to provide both radiological shielding for personnel by maintaining adequate water levels in the spent fuel pool and adequate shielding for the new fuel if recycled fuel is used.	Thorium fuels have a different source term, including specific hard gamma emissions. Therefore, a reassessment of the shielding would be required.	Source term – radiological
9.1.4 Light Load Handling System (Related to Refueling)	9.1.4-2	The primary organization reviews the light load handling system (LLHS) consisting of all components and equipment for handling new fuel from the receiving station to loading spent fuel into the shipping cask…	Spent thorium fuel would have a different source term for heat and a harder gamma spectrum from the daughter products of decay. Therefore, shielding assessments would be required to ensure sufficient shielding and protection for the workforce. In addition, any additional shielding needs would result in greater weight of shielding material, and therefore all lifting operations would need to be reviewed.	Source term – radiological

Chapter 10, "Steam and Power Conversion System," is not expected to be impacted by the addition of thorium.

Table A.7. Chapter 11, Radioactive Waste Management

Section	Page/Reference	Relevant Text from NUREG	Issue	Area of Impact
11.1 Source Terms	11.1-1	….sources of radioactivity that are input to the radioactive waste management systems for treatment of liquid and gaseous wastes	Would require reevaluation of the fission product inventory, including type of material, for example, gaseous, solid, etc.	Source term – heat Source term – radiological
	11.1-1	The staff's review of the radioactive source terms includes consideration of parameters used to determine the concentration of each isotope in the reactor coolant; fraction of fission product activity released to the coolant;	Would require reevaluation of the fission product inventory, including type of material, for example, gaseous, solid, etc.	Source term – heat Source term – isotopes Source term – radiological
	11.1-2	A-D	Would require reevaluation of the fission product inventory, including type of material, for example, gaseous, solid, etc.	Source term - heat Source term – radiological Source term – isotopics
	11.1-3	10 CFR Part 20, as it relates to determining the operational source term that is used in calculations associated with potential radioactivity in effluents to unrestricted areas	Would require reevaluation of the fission product inventory, including type of material, for example, gaseous, solid, etc.	Source term - heat Source term – radiological Source term – isotopics
		10 CFR Part 50, Appendix I, as it relates to determining the operational source term that is used in calculations associated with potential radioactivity in effluents considered in the context of numerical guides for design objectives and limiting conditions for operation to meet the ALARA criterion for radioactive material in LWR effluents	Would require reevaluation of the fission product inventory, including type of material, for example, gaseous, solid, etc.	Source term - heat Source term – radiological
		General Design Criterion 60 (GDC) as it relates to determining the operational source term that is used in calculations associated with potential radioactivity in effluents to unrestricted areas, such that a nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents provided during normal reactor operation, including anticipated operational occurrences	Would require reevaluation of the fission product inventory, including type of material, for example, gaseous, solid, etc.	Source term - heat Source term – radiological Source term – isotopics
		THESE OBSERVATIONS COVER THE REMAINDER OF THIS SECTION		

Table A.7. Continued

Section	Page/Reference	Relevant Text from NUREG	Issue	Area of Impact
11.2 Liquid Waste Management System	11.2-1	The review of the LWMS includes the design, design objectives, design criteria, methods of treatment, expected releases, and calculation methods and principal parameters used in calculating effluent source terms and releases of radioactive materials in liquid effluents, **THESE OBSERVATIONS COVER THE REMAINDER OF THIS SECTION**	Would require reevaluation of the fission product inventory, including type of material, for example, gaseous, solid, etc.	Source term - heat Source term – radiological Source term – isotopics
11.3 Gaseous Waste Management System	11.3-2	Equipment and ventilation system design capacities, expected flows, source terms and radionuclide concentrations, expected decontamination factors or removal efficiencies for radionuclides, and holdup or decay time. **THESE OBSERVATIONS COVER THE REMAINDER OF THIS SECTION**	Would require reevaluation of the fission product inventory, including type of material, for example, gaseous, solid, etc.	Source term - heat Source term – radiological Source term – isotopics
11.4 Solid Waste Management System	11.4-5	If not included in the review of SRP Sections 11.2 and 11.3, an evaluation of source terms and dose calculations is conducted to assess	Would require reevaluation of the fission product inventory, including type of material, for example, gaseous, solid, etc.	Source term - heat Source term – radiological Source term – isotopics
11.5 Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems	N/A	Covered by previous sections		Source term - heat Source term – radiological Source term – isotopics
BTP 11-5 Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure	11-5-3	For a PWR: 1 percent of the operating fission product inventory in the core… For a BWR: A fission product release rate consistent with the noble gas release ... of 100 μCi/s per MWt….	Would require reevaluation of the fission product inventory, including type of material, for example, gaseous, solid, etc.	Source term - heat Source term – radiological Source term – isotopics

Table A.7. Continued

Section	Page/Reference	Relevant Text from NUREG	Issue	Area of Impact
BTP 11-6 Postulated Radioactive Releases Due to Liquid-containing Tank Failures	11-6-3	The radionuclides selected for the radioactive source term and total inventory should include those that have the highest potential exposure consequences…including long-lived fission and activation products and environmentally mobile radionuclides.	Would require reevaluation of the fission product inventory, including type of material, for example, gaseous, solid, etc.	Source term - heat Source term – radiological Source term – isotopics

Table A.8. Chapter 12, Radiation Protection

Section	Page/Reference	Relevant Text from NUREG	Issue	Area of Impact
12.1 Assuring that Occupational Radiation Exposures Are As Low As Is Reasonably Achievable	12.1-1	"...to assuring that occupational radiation exposure (ORE) will be as low as is reasonably achievable (ALARA)"	Thorium fuels would have a different source term.	Source term – radiological
	12.1-2	Information describing how the applicant has used operating experience from past designs and from operating plants to develop improved radiation protection design	Thorium fuels would have a difference source term. In addition, there is lack of operating experience with thorium fuels and additional dose from hard gammas.	Source term – radiological Shielding code validation
	12.1-3	Acceptance criteria 1 and 2	Thorium fuels would have a difference source term. In addition, there is lack of operating experience with thorium fuels and additional dose from hard gammas.	Source term – radiological Shielding code validation
12.2 Radiation Sources	12.2-1	...as it relates to radiation sources in normal operations, anticipated operational occurrences (AOOs), and accident conditions used as the bases for determining the radiation protection design features …	Thorium fuels would have a difference source term. In addition, there is lack of operating experience with thorium fuels and additional dose from hard gammas.	Source term – radiological Shielding code validation
	12.2-2	The staff also reviews sources of radioactive material used as the bases for determining the design features needed to comply with the requirements…	Thorium fuels would have a different source term.	Source term – radiological Source term – isotopics
		The specific areas of review are as follows (1 and 2).	Thorium fuels would have a different source term.	Source term – radiological
	12.2-3	4.2 FUEL SYSTEM DESIGN - as it relates to the bases for determining the radioactive content of: irradiated fuel, including fuel burn up, fuel enrichment, fuel bundle materials; irradiated control rods materials; and fuel power density. **OTHER ITEMS ALSO MAY BE AFFECTED IF DESIGN CHANGES ARE REQUIRED, FOR EXAMPLE, CONTROL RODS, CORE INTERNALS, etc.**	Thorium fuels would have a different source term, enrichment, burnup, etc.	Core analysis methods Source term – heat Source term – radiological Source term – isotopics

Table A.8. Continued

Section	Page/Reference	Relevant Text from NUREG	Issue	Area of Impact
	12.2-4	9.1.2 NEW AND SPENT FUEL STORAGE	Thorium fuels would have a different source term, enrichment, burnup, etc.	Core analysis methods Source term – heat Source term – radiological Source term – isotopics
		9.1.3 SPENT FUEL POOL COOLING AND CLEANUP SYSTEM	Thorium fuels would have a different source term, enrichment, burnup, etc.	Core analysis methods Source term – heat Source term – radiological Source term – isotopics
	12.2-5	11 RADIOACTIVE WASTE MANAGEMENT	Thorium fuels would have a different source term.	Source term – heat Source term – radiological Source term – isotopics
	12.2-9	Shielding and ventilation design fission product source terms will be acceptable if developed using these bases….	Thorium fuels would have a different source term.	Source term - heat Source term – radiological Source term – isotopics
	12.2-9	The source term used…should consider isotopic concentrations associated with operation at the TSs allowed limits for primary-to-secondary leakage….	Thorium fuels would have a different source term.	Source term – heat Source term – radiological Source term – isotopics
	12.2-12	Neutron and prompt gamma source terms are based on reactor core physical calculations and operating experience from reactors of similar design.	Thorium fuels would have a different source term. In addition, there is lack of operating experience with thorium fuels.	Core analysis methods Source term – heat Source term – radiological Source term – isotopics
12.3-12.4 Radiation Protection Design Features	12.3-12.4-18	The applicant's shielding design is acceptable…if assumptions regarding source terms, cross sections…are realistic.	Thorium fuels would have a different source term with increased uncertainties. In addition, cross sections for thorium fuel cycle have higher uncertainties.	Source term – heat Source term – radiological Source term – isotopics
		GENERAL COMMENT – NUMEROUS AREAS AFFECTED BY SOURCE TERM AND SO WILL REQUIRE THOROUGH REASSESSMENT		

Chapters 13 and 14, "Conduct of Operations" and "Initial Test Program and ITAAC-Design Certification," are expected to be minimally impacted by addition of thorium.

Table A.9. Chapter 15, Transient and Accident Analysis

Section	Page/Reference	Relevant Text from NUREG	Issue	Area of Impact
15.0 Introduction - Transient and Accident Analyses	15.0-2	The following are some examples of AOOs in pressurized-water reactor (PWR) and boiling-water reactor (BWR) designs	Transient events are typically affected by reactivity coefficients (e.g., moderator temperature coefficient, boron coefficient) or control rod worth (e.g., rod ejection, dropped rod). Thorium fuels would affect these coefficients, and therefore a large number of the transient responses. Thorough review would be needed. Decay heat is also a key driver, and this would be affected by a change in source term for thorium fuels.	Core analysis methods Kinetics parameters Source term – decay heat Transient analysis methods
	15.0-5	Fuel cladding integrity shall be maintained by ensuring that the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 DNBR limit for PWRs and that the critical power ratio (CPR) remains above the minimum critical power ratio (MCPR) safety limit for BWRs	DNBR determined by fuel performance, which would be affected by change to thorium (e.g., fuel center line temperatures, power vs. time etc.)	Fission gas release Thermal conductivity Fuel swelling Fuel creep Melting point
	15.0-6	Condition III events	Source term would change for thorium fuels. Therefore, these events would need to be reevaluated.	Source term – decay heat Source term – radiological
	15.0-7	For loss-of-coolant accidents (LOCAs)…	Items (i), (ii), (iv), and (v) are all affected by thorium, either due to fuel performance (e.g., cladding temperature) or decay heat.	Fission gas release Thermal conductivity Fuel swelling Fuel creep Melting point Source term – decay heat Core and transient analysis methods
		e.g., core power, core inlet temperature, reactor system pressure, core flow, axial and radial power distribution, fuel and moderator temperature coefficient, void coefficient, reactor kinetics parameters, available shutdown rod worth, and control rod insertion characteristics.	Fuel design and type affect these parameters.	Core analysis methods Transient analysis methods Kinetics parameters

Table A.9. Continued

Section	Page/Reference	Relevant Text from NUREG	Issue	Area of Impact
		Range of values for plant parameters is representative of fuel exposure or core reload, and that the range is sufficiently broad to cover the predicted fuel cycle ranges, to the extent practicable, based on the fuel design	Fuel design and type affect these parameters.	Core analysis methods Transient analysis methods Kinetics parameters
	15.0-8	The reviewer also ensures that the application specifies the permitted fluctuations and uncertainties associated with reactor system parameters and assumes the appropriate conditions, within the operating band, as initial conditions for transient analysis.	Lack of data for Th fuels would likely result in larger uncertainties.	Core analysis methods Transient analysis methods Kinetics parameters
	15.0-9	Evaluation Model	There is limited validation for Th fuels as well as lack of specific models for thorium fuels in range of required tools. Development of these tools, validation, and subsequent evaluation would be required.	Core analysis methods Transient analysis methods Kinetics parameters
	15.0-10	Results	Several of the results from the analysis would be dependent on the fuel type and would therefore need reassessment, for example, fluxes, temperatures.	Core analysis methods Transient analysis methods Kinetics parameters
	15.0-12	GDC 25, 26, 27 and 28	All of these would be impacted by the use of Th fuels and would need to be reevaluated.	Core analysis methods Transient analysis methods Kinetics parameters
15.0.1 Radiological Consequence Analyses Using Alternative Source Terms	15.0.1-2	An AST is characterized by radionuclide composition and magnitude, chemical and physical form of the radionuclides, and the timing of the release of these radionuclides.	Source terms are governed by the fuel type, and therefore use of thorium would affect these analyses.	Source term – decay heat Source term – radiological Source term – isotopics

	15.0.1-4	A review of the core inventory determined by the licensee to ensure that it is consistent with the current licensing basis rated thermal power, enrichment, and burnup.	The use of thorium fuels would change the basis for enrichment, burnup, and the core inventory. Therefore, a reassessment is required.	Core analysis methods Source term – decay heat Source term – radiological Source term – isotopics
15.0.2 Review of Transient and Accident Analysis Method	15.0.2-2	The purpose of the review is to verify that the evaluation model is adequate to simulate the accident under consideration. This includes methods to estimate the uncertainty in the calculation, as…	All codes and models used in evaluation of transients would be affected by the use of thorium, whether from the perspective of source term, reactivity coefficients, decay heat or rod worths. Therefore, this chapter would be affected throughout, including documentation, evaluations model, code assessment, and uncertainty analysis.	Core analysis methods Transient analysis methods Kinetics parameters
15.0.3 Design Basis Accident Radiological Consequences of Analyses for Advanced Light Water Reactors	15.0.3-1	…source terms…	Thorium fuels would have different source terms than UO_2 fuels, and so a reevaluation would be required.	Source term – decay heat Source term – radiological Source term – isotopics
		Several other references to the same issue. Not repeated here for brevity.		
15.1.1 - 15.1.4 Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve	15.1.1-15.1.4-1	A heat removal rate in excess of the heat generation rate in the core, causes a decrease in moderator temperature which increases core reactivity and can lead to a power level increase and a decrease in shutdown margin.	Reactivity feedback coefficients, decay heat, fuel temperatures, and control rod worth would all be affected by thorium fuels. Therefore, the entire chapter would need review and reassessment for thorium fuels. Includes analysis tools and methodology.	Core analysis methods Transient analysis methods Kinetics parameters
15.1.5 Steam System Piping Failures Inside and Outside of Containment (PWR)	15.1.5-1	The negative moderator temperature coefficient and the cooldown of the reactor system cause an increase in core reactivity. The core reactivity increase may cause a loss of reactor core shutdown margin and a resulting increase in reactor power.	Reactivity feedback coefficients, decay heat, fuel temperatures and control rod worth would all be affected by thorium fuels. Therefore, the entire chapter would need review and reassessment for thorium fuels. Includes analysis tools and methodology.	Core analysis methods Transient analysis methods Kinetics parameters

15.2.1 - 15.2.5 Loss of External Load; Turbine Trip; Loss of Condenser Vacuum; Closure of Main Steam Isolation Valve (BWR); and Steam Pressure Regulator Failure (Closed)	15.2.1-15.2.5-2	For a BWR without select rod insert, reactor scram occurs. For a PWR, there is also a sudden reduction in steam flow causing the pressure and temperature in the shell side of the steam generator to increase. The latter effect, in turn, results in an increase of reactor coolant temperature, a decrease in coolant density.	Reactivity feedback coefficients, decay heat, fuel temperatures, and control rod worth would all be affected by thorium fuels. Therefore, the entire chapter would need review and reassessment for thorium fuels. Includes analysis tools and methodology.	Core analysis methods Transient analysis methods Kinetics parameters
15.2.6 Loss of Nonemergency AC Power to the Station Auxiliaries	15.2.6-1	Reactor coolant system is isolated, and the pressure and temperature of the coolant increase	Reactivity feedback coefficients, decay heat, fuel temperatures and control rod worth would all be affected by thorium fuels. Therefore, the entire chapter would need review and reassessment for thorium fuels, including analysis tools and methodology.	Core analysis methods Transient analysis methods Kinetics parameters
15.2.7 Loss of Normal Feedwater Flow	15.2.7-1	For both PWRs and BWRs, fission product decay heat must be transferred from the reactor coolant system following a loss of normal feedwater flow	The total decay heat and variation with time would be affected by thorium fuels, and therefore the impact would need to be investigated further. The values used in the analytical model would have to be adapted.	Source term – decay heat
	15.2.7-2	The analytical methods are reviewed by the organization responsible for reactor systems to ascertain whether the mathematical modeling and computer codes have been previously reviewed and accepted by the staff. If a referenced analytical method has not been previously reviewed, the reactor systems reviewer requests initiation of a generic evaluation of the new analytical model by the organization responsible for methods and code review	New models and code updates would be required.	Core analysis methods Transient analysis methods
15.2.8 Feedwater System Pipe Breaks Inside and Outside Containment (PWR)	15.2.8-2	The parameters of importance for these transients include:	A number of the listed parameters are affected by thorium fuel through the cycle, for example, hot and average heat flux and total core reactivity. Therefore, a review of the effect on the parameters and the analysis is required.	Core analysis methods Transient analysis methods
		The parameter values in the analytical model, the initial conditions of the core, and all nuclear design parameters are reviewed. This review includes:	A number of the listed parameters are affected by thorium fuel through the cycle, for example, Doppler coefficients, kinetics, and control rod worth. Therefore, a review of the effect on the parameters and the analysis would be required.	Core analysis methods Transient analysis methods Kinetics parameters
		SEVERAL OTHER REFERENCES TO THE SAME ISSUE. NOT REPEATED HERE FOR BREVITY.		

Chapter 16, “Technical Specifications,” has multiple areas of impact, but in each area, the impact has been covered in a different chapter or section.

Chapters 17 and 18, “Quality Assurance” and “Human Factors Engineering,” are not expected to be impacted by the addition of thorium.

The review of Chapter 19, “Severe Accidents,” revealed no dramatic issues with any existing wording being applied to thorium fuels but rather identified a broad and fundamental impact on the analyses being reviewed. Additional analyses would be needed, experimental data would be required, computer codes would require modification and validation, and uncertainty analyses would need to be performed.

NRC FORM 335
(12-2010)
NRCMD 3.7

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER
(Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG/CR-7176
ORNL/TM-2013/543

2. TITLE AND SUBTITLE

Safety and Regulatory Issues of the Thorium Fuel Cycle

3. DATE REPORT PUBLISHED

MONTH	YEAR
February	2014

4. FIN OR GRANT NUMBER

NRC JCN V6299

5. AUTHOR(S)

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6. TYPE OF REPORT

Technical

7. PERIOD COVERED (Inclusive Dates)

8. PERFORMING ORGANIZATION - NAME AND ADDRESS (If NRC, provide Division, Office or Region, U. S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)

Oak Ridge National Laboratory
Managed by UT-Battelle, LLC
Oak Ridge, TN 37831-6170

9. SPONSORING ORGANIZATION - NAME AND ADDRESS (If NRC, type "Same as above", if contractor, provide NRC Division, Office or Region, U. S. Nuclear Regulatory Commission, and mailing address.)

Division of Systems Analysis
Office of Nuclear Regulatory Research
U. S. Nuclear Regulatory Commission
Washington, DC 20555–0001

10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

The primary objectives of this report are to summarize historical, current, and proposed uses of thorium in nuclear reactors; provide some important properties of thorium fuel; perform qualitative and quantitative evaluations of both in-reactor and out-of-reactor safety issues and requirements specific to a thorium-based fuel cycle for current LWR reactor designs; and identify key knowledge gaps and technical issues that need to be addressed for the licensing of thorium LWR fuel in the U.S.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

Thorium, SCALE, LWR, fuel cycle

13. AVAILABILITY STATEMENT

unlimited

14. SECURITY CLASSIFICATION

(This Page)

unclassified

(This Report)

unclassified

15. NUMBER OF PAGES

16. PRICE

Federal Recycling Program

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, DC 20555-0001

OFFICIAL BUSINESS