Sustainable Thorium Nuclear Fuel Cycles: A Comparison of Intermediate and Fast Neutron Spectrum Systems

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Abstract

This paper presents analyses of possible reactor representations of a nuclear fuel cycle with continuous recycling of thorium and produced uranium (mostly U-233) with thorium-only feed. The analysis was performed in the context of a U.S. Department of Energy effort to develop a compendium of informative nuclear fuel cycle performance data. The objective of this paper is to determine whether intermediate spectrum systems, having a majority of fission events occurring with incident neutron energies between 1 eV and 10^5 eV, perform as well as fast spectrum systems in this fuel cycle. The intermediate spectrum options analyzed include tight lattice heavy or light water-cooled

reactors, continuously refueled molten salt reactors, and a sodium-cooled reactor with hydride fuel. All options were modeled in reactor physics codes to calculate their lattice physics, spectrum characteristics, and fuel compositions over time. Based on these results, detailed metrics were calculated to compare the fuel cycle performance. These metrics include waste management and resource utilization, and are binned to accommodate uncertainties. The performance of the intermediate systems for this self-sustaining thorium fuel cycle was similar to a representative fast spectrum system. However, the number of fission neutrons emitted per neutron absorbed limits performance in intermediate spectrum systems.

Introduction

The U.S. Department of Energy (U. S. DOE) Fuel Cycle Technologies (FCT) Program Office has recently completed an Evaluation and Screening (E&S) of nuclear fuel cycle options (Wigeland et al., 2014). A major objective of this effort was to identify fuel cycle options that offer significant benefits relative to the current once-through commercial nuclear fuel cycle in the U.S. with respect to nine specified high-level criteria, such as waste management and resource utilization, to identify areas for investments in research and development. This endeavor is a key pillar of the U.S. DOE FCT effort to enable development and deployment of sustainable fuel cycles that improve performance and ensure long-term viability in significant areas such as waste management and resource utilization. As part of this initiative, a wide variety of physics and inventory analyses have been developed and documented for explicit nuclear fuel cycles. Using these

analysis examples, the physics data and performance metrics were calculated for each evaluation group assuming a fully deployed and developed fleet (equilibrium state) that generates a fixed amount of energy per year. These comprehensive analyses and accompanying supporting documentation are assembled in a standardized catalog framework (Idaho National Laboratory 2014) that enables exploration of the performance of many different fuel cycles in a consistent fashion relative to specified metrics/criteria. These nuclear fuel cycle studies were subject to a systematic peer review within a team of collaborators at multiple national laboratories within the DOE complex. Specifically, engineers and scientists at Argonne National Laboratory (ANL), Brookhaven National Laboratory (BNL), Idaho National Laboratory (INL), Lawrence Livermore National Laboratory (LLNL), and Oak Ridge National Laboratory (ORNL) performed and reviewed the physics analyses.

One of the assumptions made during the E&S was that the performance of fuel cycles with intermediate spectrum reactor systems approaches that of fast spectrum systems and therefore they were included with fast spectrum systems. The present paper aims to evaluate the validity of that inclusion for the performance of intermediate spectrum systems, when compared to fast spectrum systems, in one distinct fuel cycle option. In addition, the paper is also motivated by the possibility that some intermediate spectrum systems, in this context defined by the fission reaction spectra of the reactor system, could provide benefits similar or comparable to fast spectrum systems. Therefore, intermediate spectrum systems could be an attractive implementing technology to achieve the performance benefits identified in the E&S. It should be noted that the use of a

particular technology does not necessarily imply an "intermediate spectrum". In this study, an intermediate spectrum system was defined as a reactor system with a majority (more than 50%) of fissions occurring between 1 eV and 10^5 eV. The objective of this paper is to compare the performance of intermediate spectrum critical reactors and fast spectrum critical reactors for a specific fuel cycle with continuous recycling of U/Th and only Th fertile feed. The intermediate reactor systems include a tight lattice pressurized heavy water reactor (PWR-D_2O), a tight lattice boiling light water reactor (RBWR-Th), a continuously refueled molten salt reactor (MSR), and a sodium-cooled reactor with hydride fuel (SFR-ThH). The reference fast spectrum system is a sodium-cooled fast reactor with metal thorium fuel (Wigeland et al., 2014). The systems in this paper are example configurations, which have been studied to inform on the impact of fission reactions in the intermediate energy regime. These systems are intended to inform on some aspects of fuel cycle performance in a technology neutral way.

Physics background

Thorium-based fuel cycles have been proposed for over half a century due primarily to the relative abundance of thorium compared to uranium. In a continuous recycling mode, thorium-fed systems exhibit the potential for breeding in thermal, intermediate, and fast neutron energy space due to η (eta), the reproduction factor (number of fission neutrons emitted per neutron absorbed), being greater than 2.0. Therefore ^{233}U/Th fueled systems offer the possibility of a self-sustaining system with thermal, intermediate, or fast fission reaction spectra. Specifically, the η values for ^{233}U are higher than those of the other

fissile nuclides over the majority of the intermediate energy regime. In contrast, the intermediate ^{239}Pu η is greater than 2.0 only above 10^4 eV and between 1 eV and 5 eV. This is shown in Figure 1. However, for all nuclides, η is significantly greater in the fast energy range where it increases exponentially.

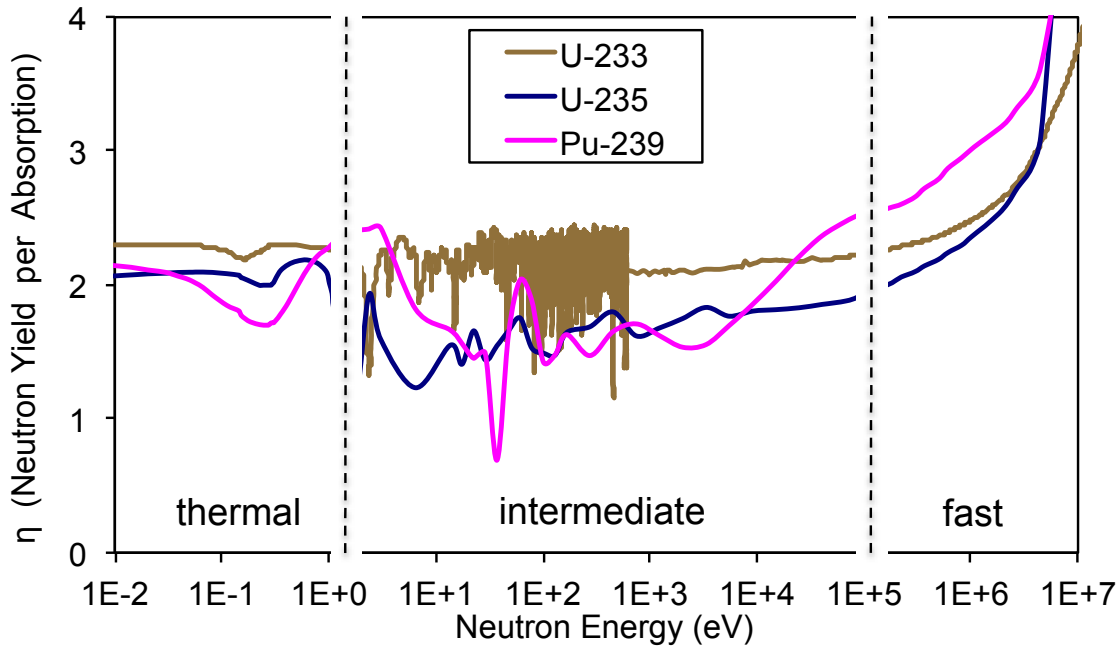


Figure 1. Neutrons produced from fission per neutron absorbed in fuel (η)

Natural thorium does not contain any fissile isotope and is a purely fertile material: the likelihood of fission events is very low for neutron energies lower than ~1 MeV. Therefore, use of thorium fuel can be envisioned with an external source of fissile material or in a fuel cycle with continuous recycle of U3 and Th. In this context, the acronym "U3" refers to the entire uranium vector produced by irradiating thorium; this uranium vector contains large fractions of ^{233}U but also contains some ^{232}U, ^{234}U, ^{235}U, and other isotopes.

The neutron energy spectrum in a nuclear reactor is complex. It depends on the reactor geometry and constituent materials (fuel, cladding, coolant, moderator, etc.), the coupled conservation relationships in the system (neutron transport, heat conduction, flow field), and the underlying nuclear data and constitutive properties. Condensing this highly complex behavior with a continuum of neutron energies spanning many orders of magnitude into a few spectral groups necessitates significant qualitative engineering judgment to define the quantitative boundaries. Wigeland et al. (2014) defines the two main classes, which are termed "thermal" and "fast", based on the predominant energy of the neutrons causing fission in the system. Figure 2 shows the typical flux spectra of thermal and fast systems along with a system with an "intermediate" spectrum that lies between the two. While quite similar to the thermal neutron spectrum, the typical intermediate neutron spectrum is missing the peak in the thermal neutron energy range. The definitions of thermal, intermediate, and fast energy ranges assume group boundaries at 1 eV and 10^5 eV. In this paper (as in the broader E&S study) "thermal", "intermediate", and "fast" systems are defined based on the energy range in which the majority (>50%) of the fissions occur.

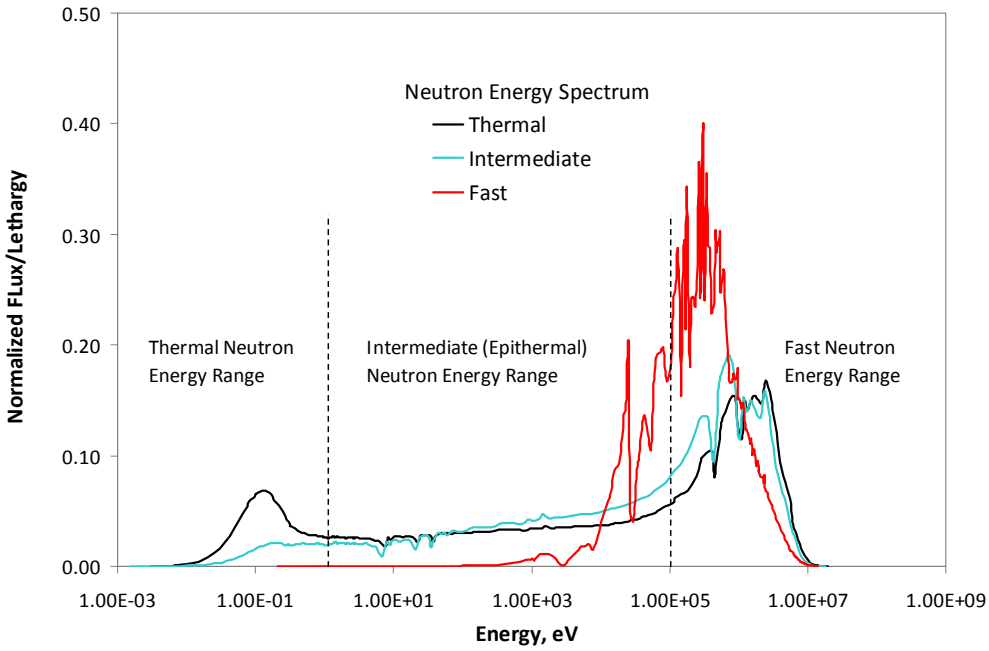


Figure 2. Examples of Neutron Energy Spectra for Thermal, Intermediate, and Fast Neutron Fission Systems

The fuel cycle of interest is a single-stage self-sustaining system with fertile ^{232}Th natural resource feed and continuous recycle of recovered uranium (U3, representing the entire recovered uranium vector). No other fissile material is utilized within the stage, no uranium natural resource is used, and no uranium enrichment is required. Several fuel forms were considered in the study including oxide fuel, hydride fuel, and molten salt fuel for the intermediate spectrum reactors and metallic fuel for the fast reactor. The nuclear power plant/transmutation (NPPT, or reactor) is an intermediate spectrum reactor or a fast reactor. To meet the objective of continuous recycle without an external source of fissile material, it is necessary to analyze a configuration that is self-sustaining based on the burnup dependent U3 inventory ratio of the configuration. Another important quantity is the fissile inventory ratio (FIR), which is defined as the time-dependent mass

of ^{233}U, ^{233}Pa, and ^{235}U divided by the initial mass of ^{233}U and ^{235}U. The role of ^{233}Pa is very important in thorium based fuel cycles as it accumulates in the fuel due to its 27 day half-life. Thus, due to beta decay of ^{233}Pa into ^{233}U, the amount of ^{233}U will continue to increase in the discharged fuel during the cooling period. Due to the assumptions for loss rates in reprocessing (1.0%) and fabrication (0.2%), which were fixed boundary conditions for the E&S study (Wigeland et al., 2014), a minimum uranium inventory ratio of about 1.012 is required. The material flow diagram for the analyzed system is shown in Figure 3. A cooling time of 5 years before reprocessing and a fabrication time of 2 years were assumed for every reactor system except for the MSR.

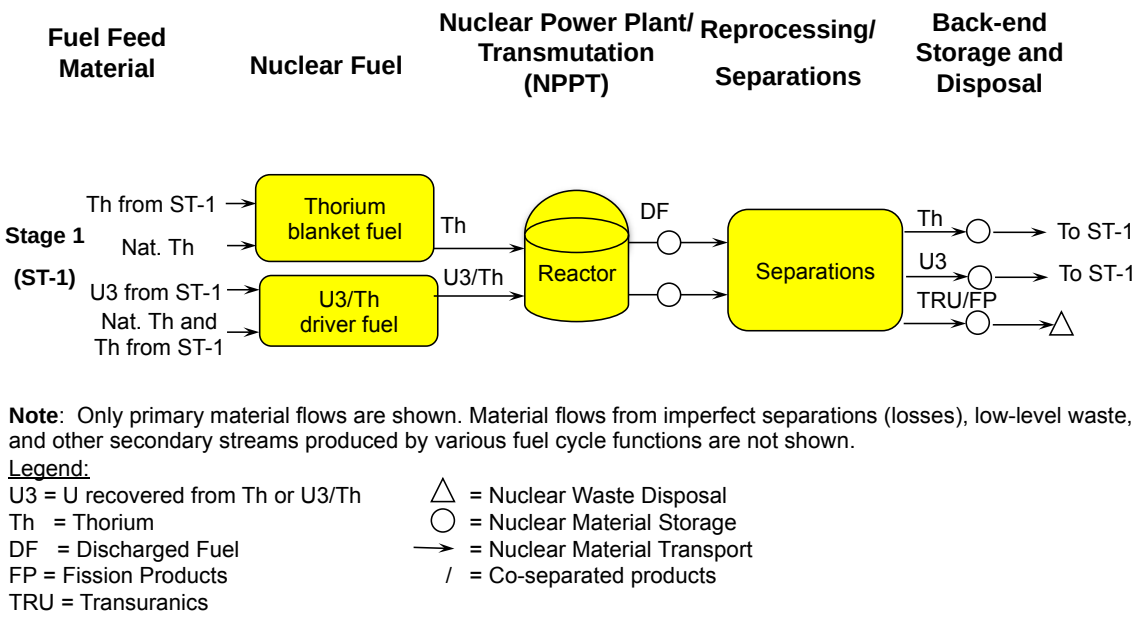


Figure 3. Material flow diagram for this nuclear fuel cycle

Modeling tools and approaches used in this study

A variety of modeling approaches were utilized for the reactor physics calculations in this study. All analyses used cross sections from the ENDF/B-VII.0 library (Chadwick 2006). In general, a stochastic or deterministic neutron transport tool was used along with a depletion solver to calculate the burnup, transmutation, and decay of isotopes in the fuel cycle. The PWR-D_2O and RBWR-Th configurations were studied using the Serpent tool (Leppänen et al., 2014), a continuous energy Monte Carlo tool developed primarily for reactor physics analysis and generation of few-group constants in complex reactor geometries. Serpent has been benchmarked to a wide variety of reactor physics problems. For the PWR-D_2O configuration, a script was used to automatically calculate the inventories after discharge and assist with cycle-by-cycle calculations to approach equilibrium conditions.

The molten salt cooled configurations were analyzed using SCALE/TRITON from the SCALE 6.1 package (Oak Ridge National Laboratory, 2011) with a cross-section library based on ENDF/B-VII.0. This required development of detailed calculation methodology and scripting to analyze the online feed and removal of materials at element-dependent rates; additional details of the methodology and calculation approach are presented in Powers et al. (2014).

The sodium-cooled reactor configuration with hydride fuel was analyzed using the MocDown tool (Seifried et al., 2013), a two-tier solver that couples MCNP (Los Alamos

National Laboratory 2005) with ORIGEN (Croff 1980) and automates an efficient iterative search for the equilibrium composition based on a prescribed fuel cycle scheme. The MocDown tool was modified to accommodate the specific fuel cycle schemes for this work. In addition, these results were verified with a simple depletion calculation with MCODE (Xu et al., 2006), which also couples MCNP with ORIGEN.

Each analysis was performed at or near equilibrium conditions to reflect the equilibrium state of the fully developed and deployed fuel cycle. The resultant charge and discharge masses were normalized per unit electricity generation. The standard assumptions of 5-years for spent fuel cooling and 2-years for fuel fabrication were utilized for the solid fuel cases (Wigeland et al., 2014). Each of the options explored in this paper was shown to be self-sustaining while accounting for losses in separations and fabrication with continuous recycle of U3 and only Th feed. The mass flow data was normalized via the equation,

$$\overline{M}_{charge}^{fuel}\left(\frac{t}{GWe \cdot yr}\right) = \frac{P_{th}(MW)}{BU(GWd/t)P_{el}(MW)} \cdot \frac{365.25d}{1yr} \tag{1}$$

where P_{th} is the reactor thermal power in megawatts (MW), BU is the fuel average discharge burnup in gigawatt-days per ton of heavy metal (GWd/t), and P_{el} is the reactor electric power in MW. Detailed mass flow data are not presented in this paper, but the methodology used to analyze the data is consistent with that in Wigeland et al. (2014). Further details on the MSR systems are presented in Powers et al. (2014). The radioactivity, radiotoxicity, and decay heat information for each configuration in this fuel cycle were calculated according to the methodology in Wigeland et al. (2014) and Stauff

et al. (2014). To be technologically neutral and to focus on physics differences the parameters were normalized using a uniform thermal efficiency of 33%.

The objective of our study, as chartered by DOE, was to develop a library of information about the relative performance of various nuclear fuel cycles in a technology neutral way. The study informs on differences in fuel cycle performance based on fundamental physics characteristics. Some fast reactor technologies have higher outlet temperatures and therefore higher thermal efficiency, but that is a design dependent feature. A fast reactor could also be implemented using a technology with a low thermal efficiency. High thermal efficiency is possible with thermal reactors, for example high temperature gas cooled reactors with graphite moderator or a molten salt cooled reactor. The objective of the study was to decouple the fundamental physics characteristics from the technology implementation to the extent possible. Assuming a uniform thermal efficiency enables an "apples-to-apples" comparison of fuel cycle performance, whereas assuming higher outlet temperature technologies for fast reactors and lower outlet temperature for thermal reactors will result in a technology-driven (as opposed to physics-driven) bias towards fast reactors.

Reference sodium-cooled fast spectrum system

The thorium-fueled sodium-cooled fast reactor (SFR-Th) system used as a reference for comparing with the intermediate spectrum systems is based on the SuperPRISM reactor designed by General Electric (Dubberley et al., 2000). Although the SFR-Th core is

fueled with metallic thorium, the SuperPRISM oxide core configuration is modified as shown in Figure 4 in order to achieve a conversion ratio of ~1.012 and maintain the ^{233}U to heavy metal ratio within reason. The core is made of 192 driver assemblies and 103 blanket assemblies. The total core power is 1000 MW_t and corresponds to a specific power density of 26 W/gHM. For this core, the simulations were performed using the REBUS-3/DIF3D code suite (Toppel, 1983 and Derstine, 1984) using ENDF/B-VII.0 libraries.

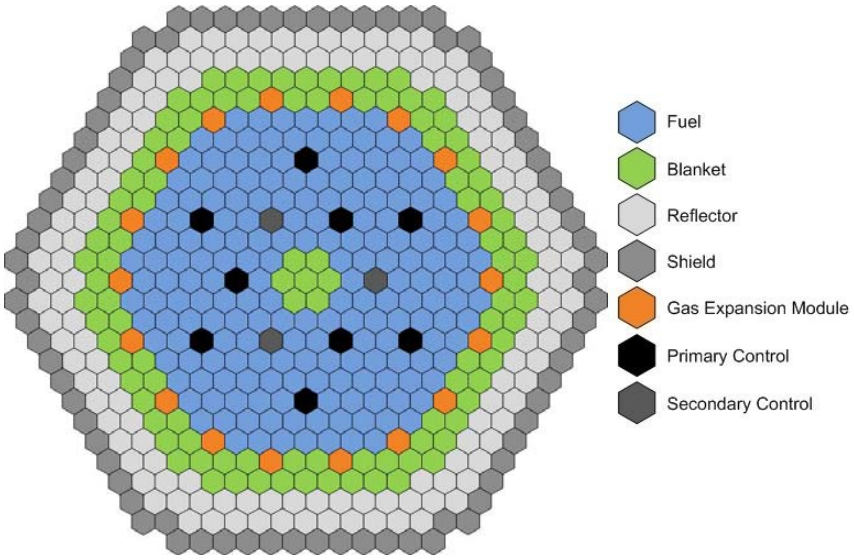


Figure 4. SFR-Th core configuration

Both the blanket and fuel assemblies remain in the core for three consecutive cycles and are discharged with an average burnup of 57.8 GWd/t and 1.6 GWd/t, respectively. At equilibrium, the charged driver fuel contains approximately 18.9wt% of trans-thorium elements, which includes 13.9wt% of ^{233}U. The principal performance characteristics are provided in Table 1 for the SFR-Th core.

Table 1. Core Characteristic for SFR-Th

Characteristic	Value
Thermal power, MWt	1000
Cycle length, EFPD	310.3
Number of batches	3
Fuel inventory, tHM	38.5
- driver	25.6
- blanket	12.9
Charge fuel mass per batch, t	12.8
Average discharge burnup, GWd/t	24.2
- driver	57.8
- blanket	1.6
Peak power density, W/cm^3	359

Analysis of water-cooled configurations

The question of a water-cooled reactor with self-sustaining U3/Th fuel has been explored in a number of studies, including the Light Water Breeder Reactor (Freeman et al., 1989), various light water moderated parametric reactor physics studies (Permana et al., 2007; Permana et al. 2008; Shwageraus et al., 2008; Yun et al., 2010; Lindley et al., 2014), reduced moderation boiling water reactors (Ganda et al., 2012; Shaposhnik et al., 2013; Lindley et al., 2014), and heavy water cooled pressurized water reactors (Takaki and Mardiansah, 2012), among many other studies. It is outside the scope of this paper to provide a comprehensive literature review or to develop such a reactor system. However this, and other previous work, has helped to motivate the present study. It is noted that the designs in the literature are not subject to the approach and constraints applied in the

present study, which constitutes an independent and technology neutral assessment of theoretical fuel cycle performance. The examples explored in the present paper are all breakeven fuel cycles (when accounting for 1.0% losses in separations and 0.2% losses in fabrication), with > 50% fission reactions initiated by neutrons with energies between 1 eV to 10^5 eV. Additional constraints on the example systems are presented in the E&S report (Wigeland et al., 2014).

It is important to note that the trade-off between breeding and criticality is a key challenge in the development of a self-sustaining U3/Th water reactor with an intermediate neutron energy spectrum. In this paper, two water-cooled heterogeneous $^{232}Th/^{233}U$ seed and blanket assembly configurations are analyzed: (1) an example D_2O-cooled tight lattice pitch PWR (PWR-D_2O) with some conceptual similarities to systems proposed by Hibi et al. (2001) and Takaki et al. (2012) and (2) an example Reduced moderation Boiling Water Reactor-Thorium (RBWR-Th) similar to that proposed by Ganda et al. (2012) and Shaposhnik et al. (2013). This system is similar to the Hitachi RBWR concept (Takeda et al., 2007).

Heavy water cooled PWR configuration

The PWR-D_2O is a heavy water-cooled PWR with similarities to designs from Mitsubishi/JAERI (Hibi et al., 2001) and the University of Tokyo (Takaki and Mardiansah, 2012). On a conceptual level the Mitsubishi/JAERI design was taken as an example with a discharge burnup of 40 GWd/t. However, the example configuration

analyzed in the present work is not directly representative of either the Hibi et al. (2001) or Takaki and Mardiansah (2012) designs. The active core height assumed in these calculations is 200 cm. A radial slice of the analyzed geometry is shown in Figure 5. Reflective boundary conditions were applied to all the radial surfaces while vacuum boundary conditions were applied to the axial surfaces. Scoping calculations via SERPENT indicated that the system is self-sustaining.

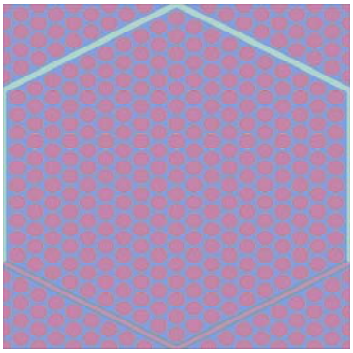


Figure 5. Radial slice of D_2O-cooled PWR analyzed

The PWR-D_2O example meets the definition of an intermediate spectrum reactor, with roughly 75% of the fissions occurring between 1 eV and 10^5 eV. The isotopic convergence of the recovered uranium vector after multiple recycling calculations is shown in Figure 6. The radially reflected multiplication factor at near-equilibrium is shown in Figure 7.

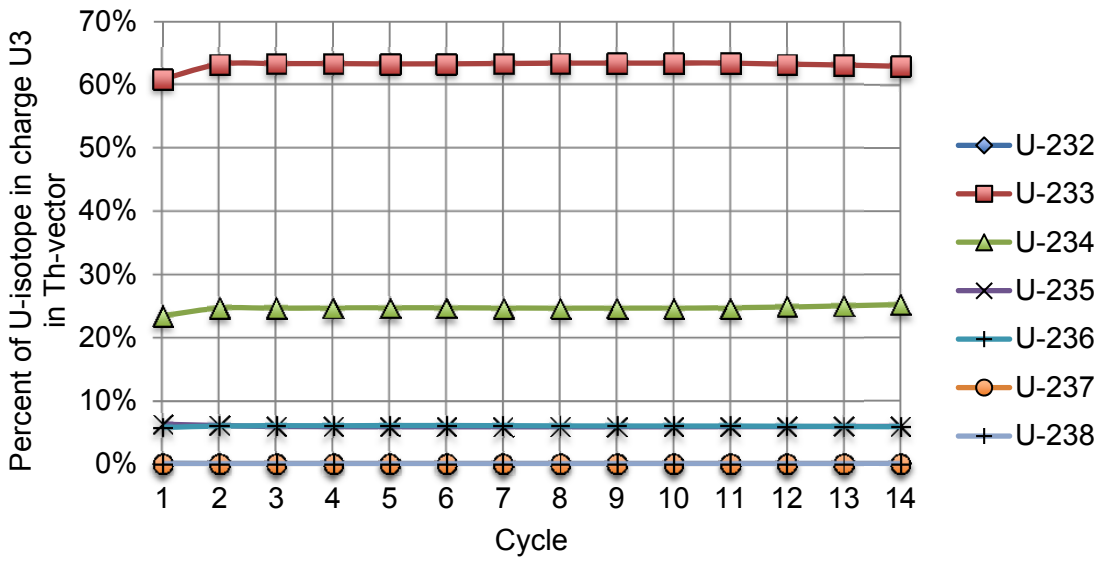


Figure 6. Convergence of U3 isotopic vector as a function of recycle for D_2O-cooled PWR

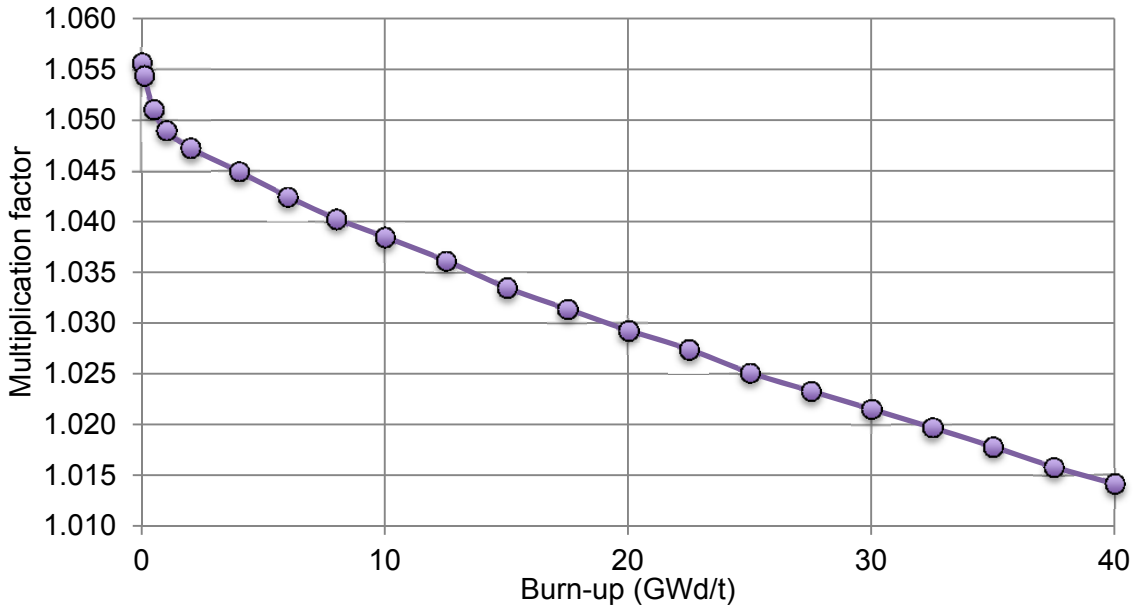


Figure 7. Radially reflected multiplication factor for D_2O-Cooled PWR at near equilibrium

The equilibrium cycle mass flows in the fuel cycle were generated assuming 5 years for cooling and 2 years for fabrication. A 40 GWd/t discharge burnup was targeted with assumptions of a 3-batch fuel management scheme and a radial leakage and absorption in reflectors/core structures of 2.5% of reactivity. The uranium inventory ratio at a near equilibrium condition, after accounting for losses and decay, is ~1.01. Therefore, the system is a very slight breeder at near equilibrium, but is reasonably close to self-sustaining. The assumed reactor parameters are shown in Table 2 and fuel parameters in Table 3.

It is noted that the assumed fuel management scheme and leakage fraction will have an impact on achievable burnup. Potential sensitivities related to the number of fuel batches and neutron leakage were explored in Brown and Todosow (2015) for similar systems with U/Th fuel. With a few design modifications, a higher fuel burnup could be achieved. For example, if the number of fuel batches were increased to 5 and the radial leakage and absorption in reflectors/core structures were on the order of 0.8% (representative of a large "pancake" core), then the discharge burnup of the configuration can reach 74 GWd/t.

Table 2. Reactor Parameters for D_2O-Cooled PWR

Parameter	Stage Number
Stage	1
Core Configuration	Tight pitch hex
Core Thermal Power, MWt	1200*
Net Thermal Efficiency, %	33

* Value is assumed based on 313 assemblies with an assembly power of 3.84 MWt

Table 3. Fuel Parameters for D_2O-Cooled PWR

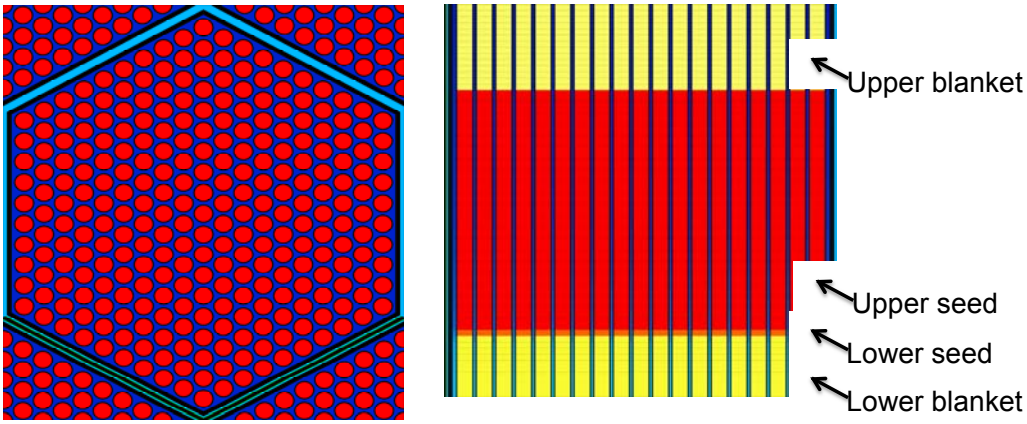
Parameter		Fuel Type	
Purpose		Driver (Th)	Blanket (Th)
Chemical Form		Oxide	Oxide
Physical Form		Pin Bundle - Ductless	Pin Bundle - Ductless
Average Discharge Burnup, GWd/t		45.1	2.2
Fuel Composition	Initial Nuclear Material(s)	U3/ThOX	ThOX
	(U-235+ U-233)/Total U	68.9%	0.0%
	Th/Total heavy metal	85.9%	100.0%
	TRU/Total HM	0.0%	0.0%
Fuel Residence Time in Reactor, EFPY		8.2	8.2

Light water cooled BWR configuration

Similar versions of the RBWR-Th design from 2012 (Ganda et al., 2012) and 2014 (Gorman et al., 2014) were analyzed independently via Serpent and MocDown at BNL and ANL, respectively. Except for the initial fuel loading, the RBWR-Th is charged with only thoria (ThO_2) and sends only fission products to waste, recycling all actinides except for losses that end up in the waste streams. The design is a variant of the RBWR-AC core proposed by Hitachi, in which the UO_2 and MOX fuels are arranged in a hexagonal tight-lattice, has a high outlet void fraction, axially segregates seed and blanket regions, and fits within an Advanced Boiling Water Reactor (ABWR) pressure vessel. The RBWR-Th

shares these characteristics but replaces depleted UO_2 with ThO_2 as the primary fertile fuel, eliminates the internal blanket while axially elongating the seed region, and eliminates absorbers from the axial reflectors.

However, it is notable that due to the strongly negative void reactivity coefficient, the RBWR-Th with pure thorium resource feed may experience issues with shutdown margin; design studies are presently ongoing to resolve this issue (Zhang et al., 2013; Shaposhnik et al., 2014). The radial and axial geometry of the RBWR-Th modeled by BNL is shown in Figure 8. The FIR and radially-reflected multiplication factors calculated for this example system are shown in Figure 9. Table 4 and Table 5 describe the RBWR-Th core parameters and pin geometry used in the ANL MocDown model. Table 6 shows the physics parameters and incident fission neutron spectrum calculated by ANL and BNL for the two similar RBWR-Th configurations. Note the agreement in the energy distribution of the incident neutrons, which shows that the two examples meet the intermediate spectrum definition (~70% of fissions in the 1 eV to 100 keV range). There is a significant difference between the two example systems analyzed here. The example analyzed at ANL is similar to a recent iteration of the RBWR-Th design (Gorman et al. 2014). However, the example analyzed at BNL is similar to an earlier iteration of the design (Ganda et al., 2012) and would require innovative approaches, such as those suggested by Shaposhnik et al. (2014), to meet shutdown margin constraints. Shutdown margin is the key driver in the difference in the two example systems and also in the achievable discharge burnup. The objective of presenting these results is to illustrate the impact of important design considerations on potential fuel cycle parameters.



(a) (b)

Figure 8. Radial (a) and axial (b) slice of RBWR-Th assembly

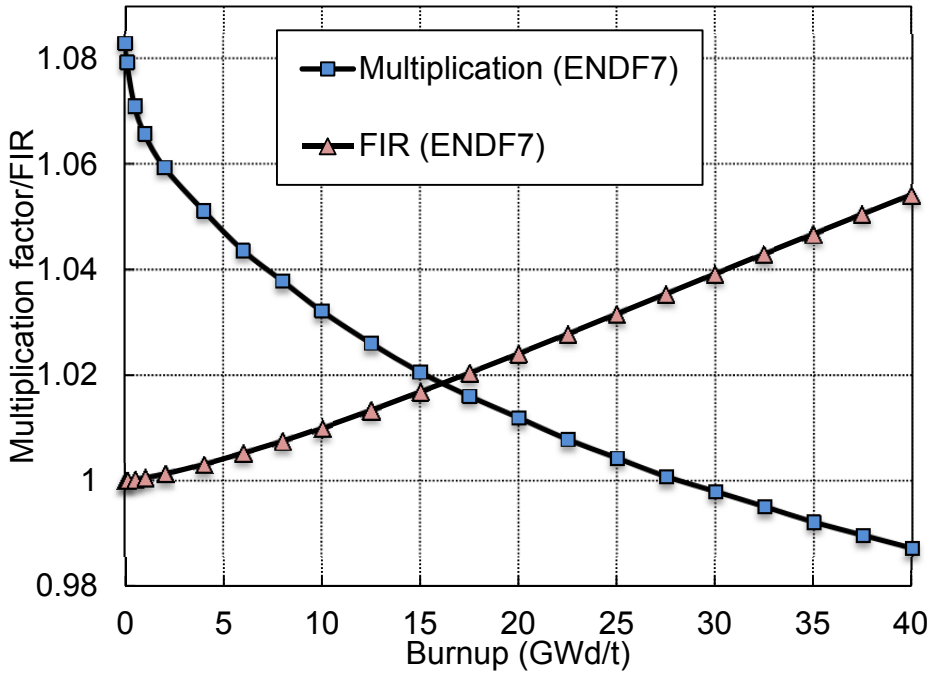


Figure 9. Radially reflected multiplication factor for RBWR

Table 4 RBWR-Th core design parameters (Gorman *et al.*, 2014)

Thermal power, MWt	3926
Core mass, tHM	361
Fuel residence time, EFPD	2300
Number of batches	5
Cycle length, EFPD	460
Average discharge burnup, GWd/t	23.3
Pin power, MWt	0.0188
Primary coolant pressure, MPa	7.25
Coolant inlet temperature, ºC	282.56
Exit quality	0.277

Table 5 RBWR-Th seed and blanket pin geometry (Gorman *et al.*, 2014)

Axial height (blanket/seed/blanket), cm	40/300/40
Cladding inner diameter, cm	0.885
Cladding outer diameter, cm	1.005
Pin pitch, cm	1.135
Fuel smear density, g/cm^3	8.99
Heavy metal smear density, gHM/cm^3	7.90
Fuel cross section temperature, K	900
Structure cross section temperature, K	600

Table 6 Calculated RBWR-Th physics parameters

Case	ANL	BNL
Fissions in E < 1 eV range, (at MOL)	8%	4%
Fissions in 1 eV – 100 keV range, (at MOL)	70%	70%
Fissions in 100 keV < E range, (at MOL)	22%	26%
Core Avg. Discharge Burnup (B_d) [GWd/t]	23.3	40.0
Fissile Inventory Ratio*	1.00	1.05

* mass of discharged $^{233}U + ^{235}U + ^{233}Pa$ / mass of charged $^{233}U + ^{235}U$

Homogeneous molten salt cooled configurations

A relevant alternative technology is a MSR system. The analysis approach consisted in starting with a thermal-spectrum MSR configuration and decreasing the moderator content until an intermediate spectrum was obtained. This can be thought of as

approaching an intermediate spectrum by reducing moderation in a thermal-spectrum design as opposed to adding moderator to a fast-spectrum design.

MSR analysis requires modeling several material feed and removal functions: direct discard of fuel salt, salt treatment (passive) processes, separations (active) processes that extract species such as rare earth element fission products, and fresh feed of thorium. These processes are applied sequentially in this work in the following order: (1) salt discard; (2) uranium and protactinium separations; (3) salt treatment and separations; and (4) thorium addition.

Salt treatment and separations calculations were performed using effective cycle times from the Molten Salt Breeder Reactor (MSBR) program and batch removal calculations, as detailed in Powers et al. (2014). The MSBR program defined a cycle time as the amount of time required to remove 100% of a given element from a fuel salt. Cycle times were converted to removal fractions for this work, with a removal fraction of 1.0 used when the depletion time step length matched or exceeded the cycle time for an element.

The methods used to model MSR material flows in this work approximate continuous flow processes as a series of batch processing steps due to limitations in the current depletion codes. While this batch processing approximation introduces some level of error, short depletion steps minimize the magnitude of the errors introduced. Based on the results of a study described in Powers et al. (2014) that examined the net effect that depletion step length had on modeling material flow processes and depletion, 3-day

depletion steps were used for all MSR calculations unless otherwise noted in order to allow 100% removal of protactinium during a single depletion step.

Sensitivity to the ^{233}U radiative capture cross section evaluation is an issue that was recently identified in the literature for molten salt reactors (Aufiero et al., 2013). This issue is especially relevant for intermediate spectrum MSR systems. This sensitivity is reviewed here, because it significantly impacts the neutron economy in these systems and also because the ENDF-BVII.0 cross section evaluation used in the present paper yields higher radiative capture rates in ^{233}U than the JEFF-3.1 evaluation used in many recent studies (Aufiero et al., 2013). The differences in this cross section between several standard evaluations are shown in Figure 10.

The ^{233}U cross section in ENDF/VII.0 has new evaluations in the fast energy region and resonance region (Chadwick et al., 2006). In the fast energy range, the capture cross section was renormalized using the available experimental data. The primary neutron capture data in the range (10 keV – 0.5 MeV) is some sparse legacy data from Hopkins and Diven (1962). The ENDF/VII.0 evaluation in this energy range shows significantly improved agreement with legacy measurements. There is additional legacy measured capture cross section data in this energy regime (Spivak et al., 1957), which shows good agreement with the data in Hopkins and Diven (1962). The ENDF/B-VII.0 evaluation shows improved agreement, when compared to ENDF/B-VI.8 or JEFF-3.1, with the available measured neutron capture cross section data in the energy range (10 keV – 0.5 MeV). It should be noted that the JEFF-3.1 ^{233}U evaluation is based on JENDL 3.3 and

ENDF/B-VI.4. The data from Hopkins and Diven (1962) and Spivak et al. (1957) are legacy data. Cross section measurement, data reduction, and uncertainty quantification techniques have matured considerably since these legacy measurements were made. Additionally, the data is sparse over orders of magnitude in energy. It is noteworthy that JEFF-3.2 has adopted the ENDF/B-VII evaluation. However, there is still a lack of recent experimental data over a very large energy space.

It is noted that the performance of the MSR systems in the present work was calculated using ENDF/B-VII.0 evaluations. Therefore, the calculated radiative capture in ^{233}U will be higher than studies that utilize other cross section sets, and in particular studies that rely on JEFF cross sections, such as Aufiero et al. (2013).

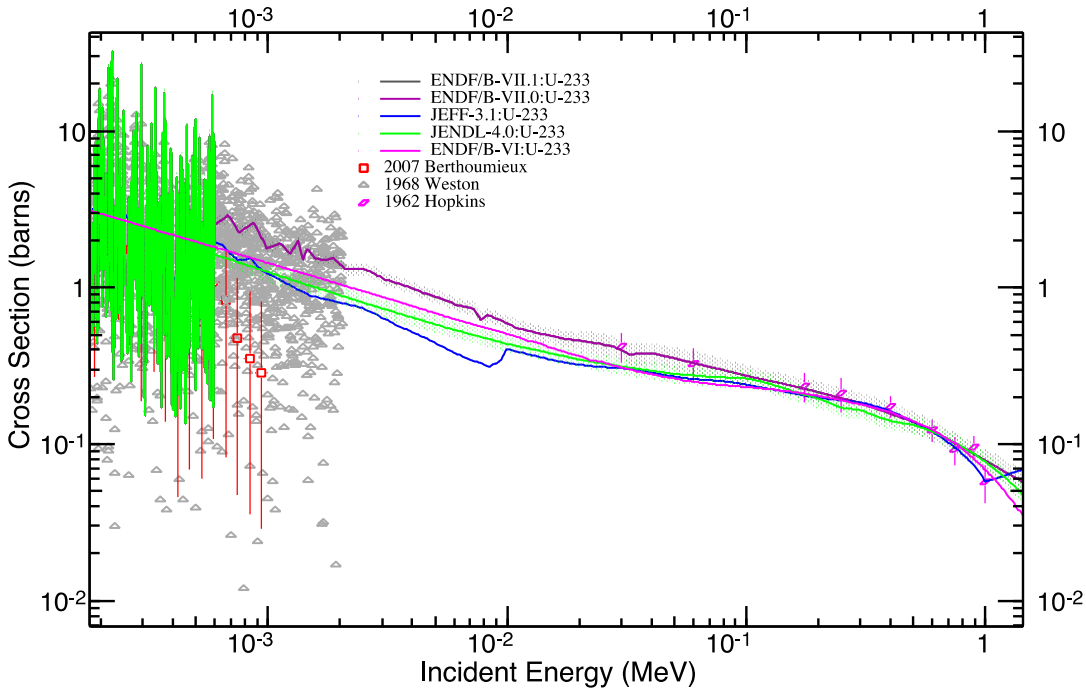


Figure 10. ^{233}U radiative capture cross section and measurements

The unit cell from the MSR design used for a thermal MSR system provided a starting point (Powers et al., 2014). The central fuel salt flow channel in this unit cell was then expanded, increasing the relative volume fraction of the unit cell occupied by fuel salt and decreasing the moderator to fuel ratio, thereby hardening the neutron spectrum. Figure 11 provides an illustrative example of this process by showing the MSBR Zone I cell design (13 vol% fuel salt) and the MSBR Zone II cell design (37 vol% fuel salt), as described in Bettis and Robertson (1970), and an effective unit cell with a flow channel containing the volume-weighted average (~20 vol%) of the two zones.

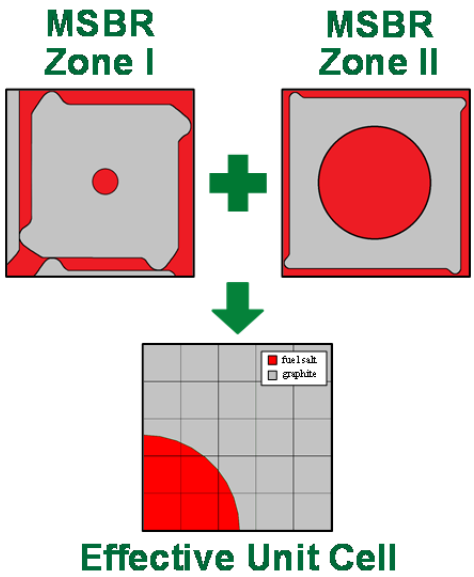


Figure 11. MSBR Zone I and Zone II designs and a core volume-weighted Effective Unit Cell (Red: Fuel Salt, Grey: Graphite)

Parametric analyses were performed varying the fuel salt volume fraction to identify an example intermediate-spectrum MSR system that could also maintain self-sustaining criticality at equilibrium. The calculations used to identify the example system assume a

specific fuel salt (FLiBe) and fission product extraction rate, with additional details provided in Powers et al. (2014). The lattice geometry is variable. These results are not intended to provide universal insight on the reactor physics feasibility of MSR concepts. The results are analysis examples that are representative of specific possible approaches to an MSR system. The results were achieved by hardening the spectrum in a thermal MSR design with the goal of increasing the fraction of fission events in the intermediate energy range.

Figure 12 shows the fraction of fissions occurring in the intermediate energy region, which varies almost linearly with the fuel salt volume fraction in the MSR unit cell. These results indicate that a fuel salt volume fraction of just below 37% would be the minimum to ensure over 50% intermediate fissions. Figure 13 informs on the feasibility of these unit cell designs by showing the equilibrium infinite multiplication factor as a function of fuel salt volume fraction in the MSR unit cell; an equilibrium infinite multiplication factor of over 1.02 is required to account for neutron leakage. Design improvements and optimizations could provide small increases to the multiplication factors shown in Figure 13, so all of the unit cell designs in the range of 13% to 42% fuel salt volume fraction may be feasible, but additional calculations would be needed to explicitly support these engineering judgments. Fuel salt volume fractions above 42% would become difficult to design for critical operation for this specific set of lattice and salt configurations, though the range could perhaps be extended to 45–50% in this system with some effort. Changing the salt from FLiBe (2 LiF + BeF_2) salt to another carrier salt (e.g., LiF or chloride salts) could also prove beneficial but would need to be investigated,

as these alternate carrier salts would push the system into the fast spectrum at large fuel salt volume fractions. These salts have been considered for various fast MSR concepts since the 1950s (Briant and Weinberg 1957; Forsberg 2007).

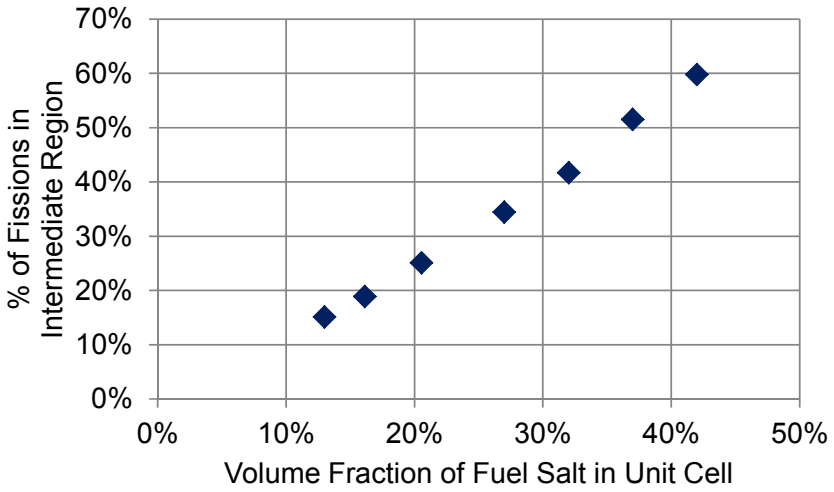


Figure 12. Percentage of fissions caused by neutrons in the intermediate energy region as a function of volume fraction of fuel salt in an MSR unit cell

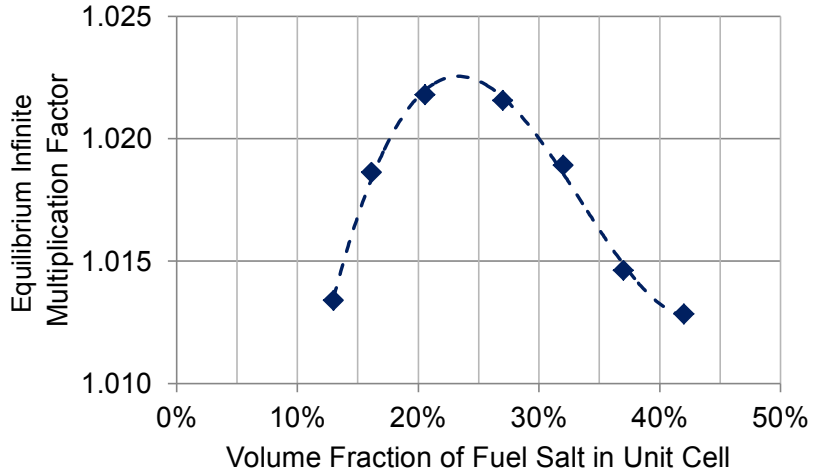


Figure 13. Equilibrium infinite multiplication factor as a function of volume fraction of fuel salt in an MSR unit cell

Based upon the results shown in Figure 12 and Figure 13, two unit cells appear to be of particular interest: one with 37 vol% fuel salt that produces 51% intermediate fissions and one with 42 vol% fuel salt that produces 60% intermediate fissions. It is worth noting that the 37 vol% fuel salt unit cell happens to be the same as in the MSBR Zone II unit cell, which was intentionally designed to have a harder neutron energy spectrum to increase the breeding ratio in that core region. Depletion studies indicate that both unit cell designs require about the same thorium feed material flow rate, so their resource utilization should be fairly similar.

Sodium-cooled reactor with UThH fuel configuration

There is ongoing work at the University of California at Berkeley to develop a sodium-cooled reactor technology that continuously recycles U3 using a hydride fuel form (Zhang and Greenspan 2014). In this paper, the design is referred to as the sodium-cooled fast reactor with hydride fuel, or the SFR-ThH. However, it is notable that the fission reaction spectrum is actually intermediate under the definition used in this study. A major benefit of this configuration is the ability to achieve a negligible sodium void worth for a relatively large SFR core (~3000 MWt). Specifically, using U3 instead of TRU as the main fissile material greatly reduces the void coefficient of reactivity due to ^{233}U's more gradual increase in the reproduction factor, η, upon spectrum hardening. In addition, using hydride fuel to moderate the neutrons softens the spectrum so that most of the fission reactions occur in the intermediate spectrum in which the η slope is more gradual.

The design parameters, pin geometry, and material compositions are shown in Table 7 and Table 8, respectively. The charge fuel composition was obtained after performing multiple recycle calculations via MocDown until equilibrium was reached. This composition was then used as input in a radially reflected pin model to calculate the discharge burnup (no cooling or reprocessing losses assumed) and the corresponding U3 inventory ratio. A depletion calculation was performed to produce the radially-reflected multiplication factor versus burnup curve shown in Figure 14. Assuming 0.5% radial leakage, which is within the stated 0.5-1% (Zhang and Greenspan 2014) and a 5-batch refueling scheme, the achievable discharge burnup is 40 GWd/t according to the linear reactivity model (Driscoll et al., 1990). The percent of fissions between 1 eV and 10^5 eV is 66%, indicating that this reactor design is representative of an intermediate spectrum based on the definition in the study. The uranium and transuranic inventory ratio of the design was 1.0 accounting for 0.1% fabrication losses and decay. The standard loss rates in Wigeland et al. (2014) were not used, but with a small adjustment in U3 in Th fraction or hydrogen to heavy metal ratio these losses could be accounted for and the physics performance would be very similar.

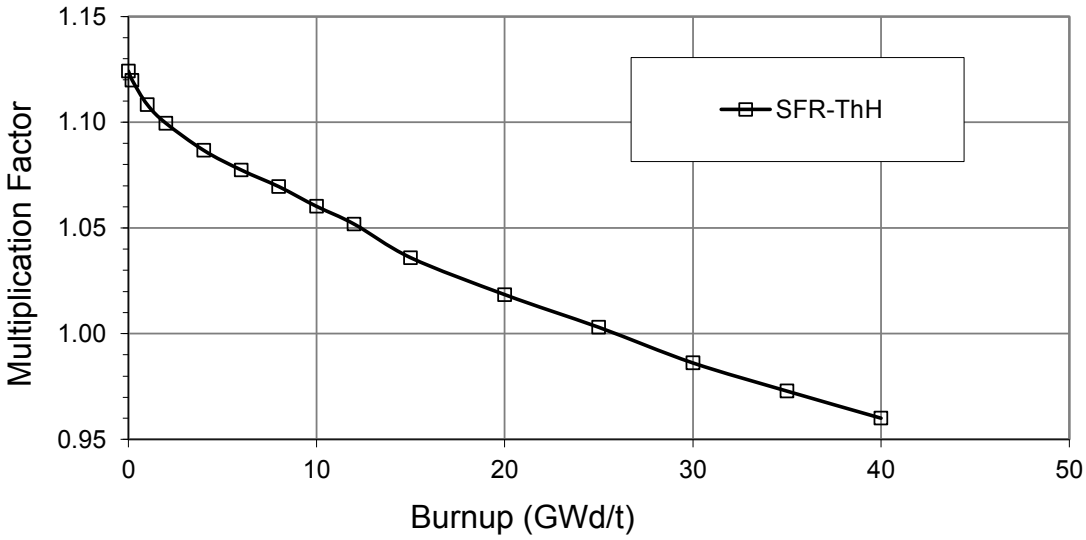


Figure 14. Radially reflected multiplication factor for SFR-ThH

Table 7. SFR-ThH core design parameters

Parameter	Value
Thermal power, MWt	3000
Core mass, tHM	109
Fuel residence time, EFPD	1450
Number of batches	5
Cycle length, EFPD	290
Average discharge burnup, GWd/t	40
Pin power, MWt	0.0093
Coolant density, g/cm3	0.851

Table 8. SFR-ThH seed and blanket pin geometry

Dimension	Value

Axial height (blanket/seed/blanket), cm	60/80/60
Cladding inner diameter, cm	0.6324
Cladding outer diameter, cm	0.744
Pin pitch, cm	0.82584
Fuel pellet smear density, g/cm^3	8.55
Heavy metal smear density, gHM/cm^3	8.53
Fuel stoichiometry	$UThH_{0.5}$
Blanket stoichiometry	$ThH_{0.5}$

Comparison of analyses for different reactor representations

The different intermediate spectrum designs using Th-based fuel investigated in this study were selected to represent a diverse sampling of this unique design space. These options came from recent literature or were designed specifically for this study with many approximations and a limited number of constraints. These designs use different fuel forms, coolants, and moderators to achieve the continuous recycling of U3/Th while fitting the definition of intermediate spectrum. The fraction of fissions in each energy range is shown in Figure 15 for each of these four designs.

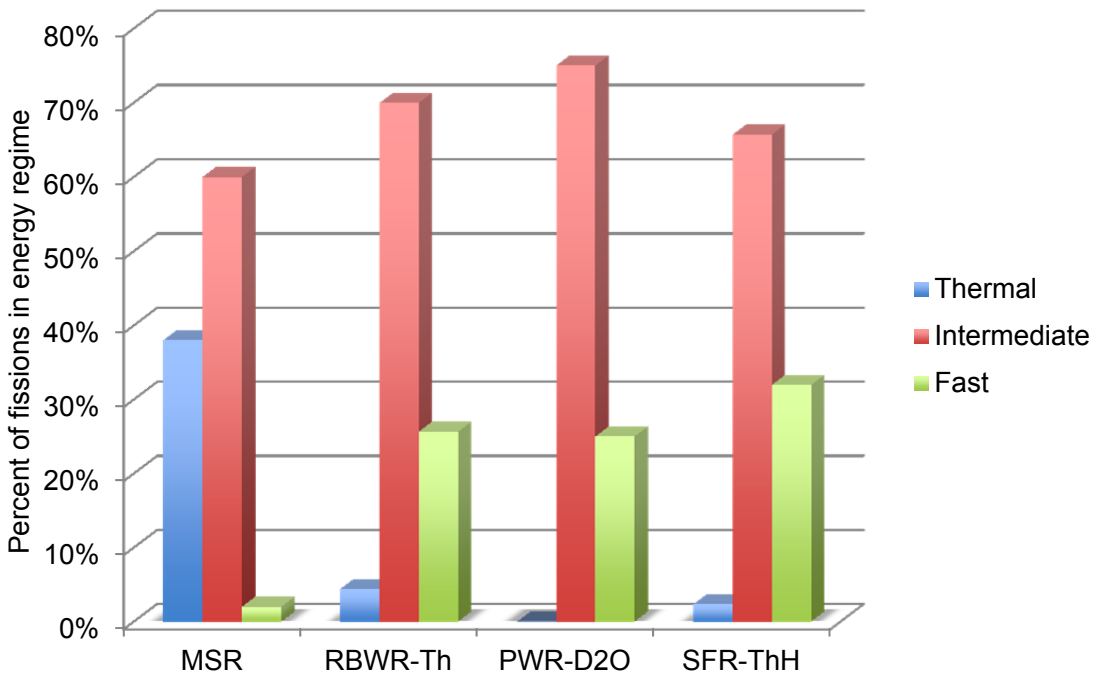


Figure 15. Percent of fissions in thermal, intermediate, and fast energies for several reactor systems

The flux and fission reaction spectra are shown in Figure 16 and Figure 17, respectively for the PWR-D_2O, RBWR-Th, MSR, and SFR-ThH. The values were plotted over the same set of equal-lethargy 238 energy bins. The main observation from Figure 15 is that the three solid fuel designs have significant amounts of fast fissions and little to no thermal fissions, while the MSR has almost 40% of its fissions caused by thermal neutrons. This reflects how an intermediate spectrum design can be achieved by hardening the spectrum of a thermal reactor (as in the MSR) or softening the spectrum of a fast reactor (as in SFR-ThH).

These observations yield the following conclusions about the intermediate systems analyzed in this study. The solid fuelled systems tend to be "high intermediate" systems, meaning that although the intermediate spectrum drives the fission events, the systems also exhibit a significant percent of fast fissions. In contrast, the liquid fueled systems tend to be "low intermediate" systems, because although the majority of fissions occur in the intermediate energy range the systems also exhibit a significant percent of thermal fissions. This conclusion is limited to the example systems analyzed in this particular study. The case with the highest percent of intermediate fissions, the PWR-D_2O (75%), also has the lowest percent of thermal fissions (0.3%). However, all cases with more than 50% intermediate fissions are representative of an intermediate spectrum technology realization, in the context of this study.

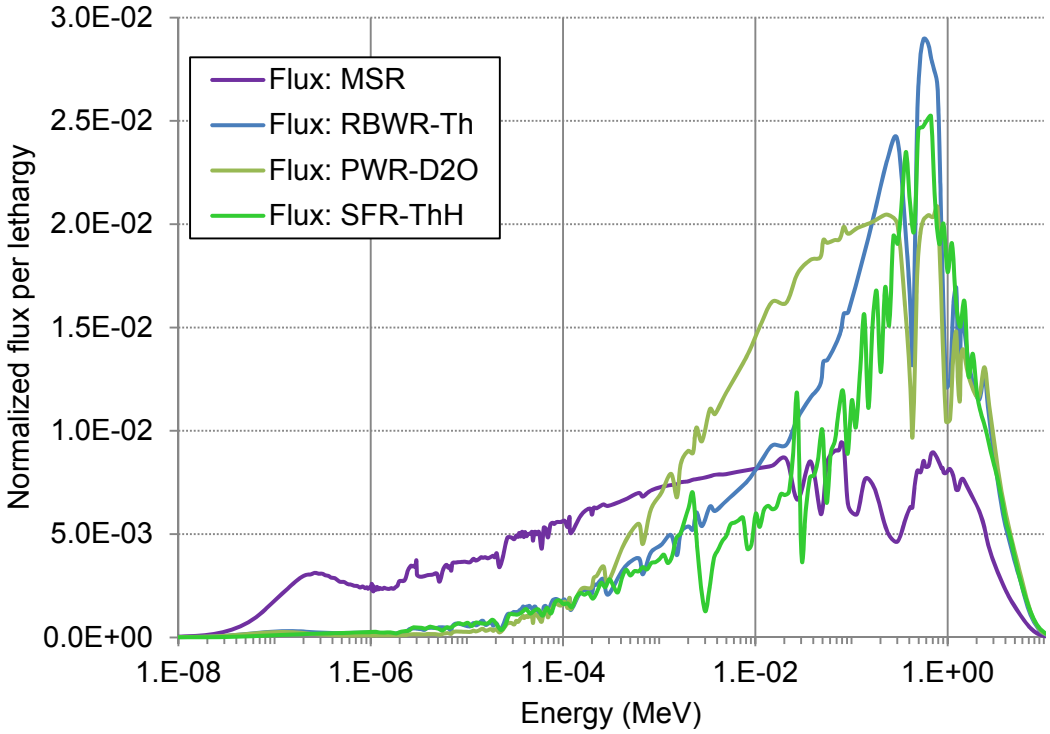


Figure 16. Flux spectra comparison for intermediate spectrum systems

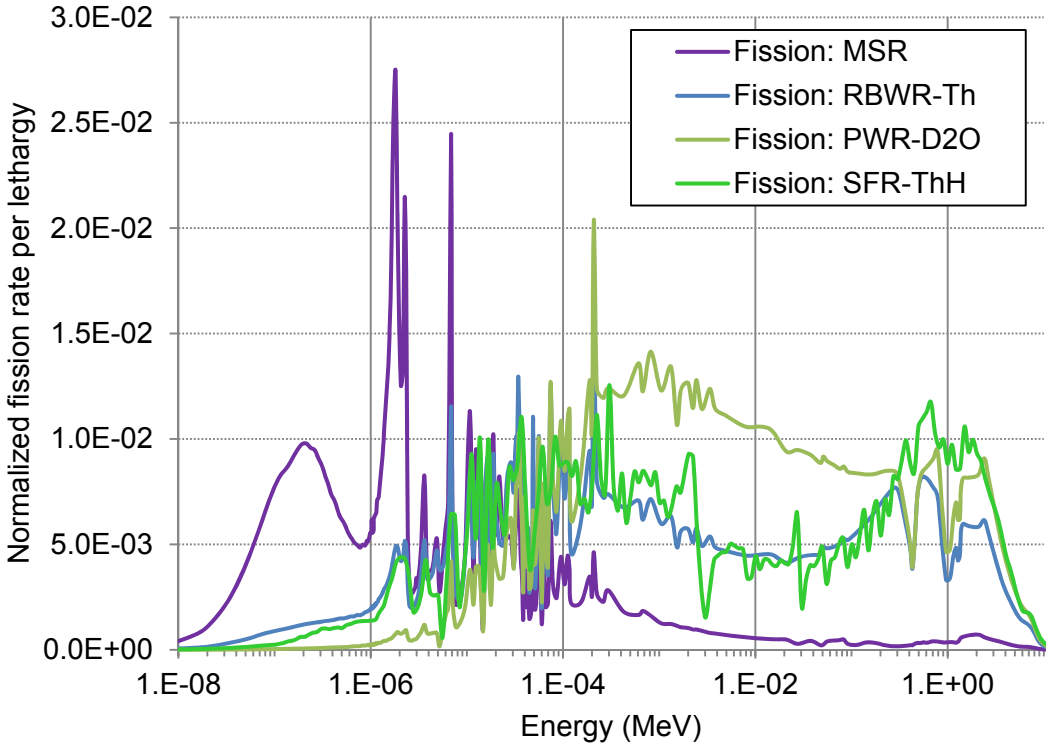


Figure 17. Fission reaction spectra comparison for intermediate spectrum systems

Other than the MSR case, which used graphite moderator, relatively light moderators are utilized in these example configurations. However, it is noted that in spite of the difference in moderator mass, a fission reaction spectrum similar to that of the MSR can be obtained with a hydrogen moderator. This assertion is justified by considering a system that does not quite meet the definition of intermediate spectrum in this study: a PWR with light water moderator in a breakeven U3/Th fuel cycle.

Scoping calculations were performed regarding a PWR configuration with the objective of achieving a self-sustaining system with continuous recycle of recovered uranium in

thorium and only fertile thorium resource feed. The objectives were to determine whether the system was self-sustaining and whether it can be considered an intermediate spectrum system. This PWR is a heterogeneous $^{232}Th/^{233}U$ assembly that fits within the form factor of a 17 x 17 Westinghouse PWR assembly (Yun et al., 2010). This geometry is shown in Figure 18. Some noted caveats include the fact that the blanket fuel rods are in direct contact, which would leave no room for conventional grid spacers and may lead to closed thermal hydraulic channels. Additionally there is no fissile content in the blanket fuel rods, which means that initially almost all of the fission events will occur in the driver rods at zero burnup, the specific power is significantly de-rated versus a standard PWR, due to the high heavy metal loading and power generated in the seed, and the thermal hydraulic feasibility of the configuration has not been studied and may present concerns for viability.

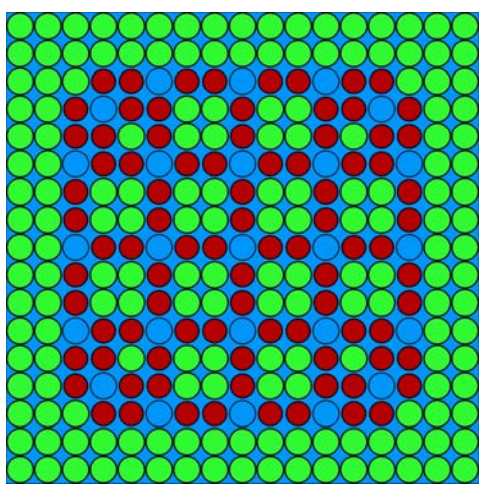


Figure 18. PWR assembly configuration analyzed (seed rods are red, blanket rods are green)

The key questions regarding this light water cooled PWR include: (1) Is the configuration self-sustaining with a degraded uranium vector at equilibrium? (2) Is this

PWR a thermal spectrum or an intermediate spectrum reactor technology? It is noted that none of these questions was answered definitively in previous studies.

A study was performed to assess the sensitivity of this configuration to several fissile fractions of a degraded U3 vector. The intention is to explore the criticality and the sustainability. An example degraded U3 vector was assumed for these scoping comparisons. The vector is shown in Table 9. The vector was used with the assembly model shown in Figure 18 with several driver zone U3 fractions. The percent of U3 in Th was assumed to be 8.8%, 12.8%, and 14.5% respectively. The cases straddle the transition from critical to subcritical and also from self-sustaining to a system requiring an external resource feed. The infinite multiplication factor and total uranium inventory ratio are shown in Figure 19 and Figure 20, respectively. It is evident that the 12.8% U3 case is the closest to the required inventory ratio and criticality constraints, although the uranium vector is continuing to degrade and the system is not in an equilibrium state. Consequently, this 12.8% configuration will be further detailed. It is noted that the optimized fissile content of this system at equilibrium would be different.

Table 9. U3-vector utilized for degraded vector sensitivity studies

Isotope	Mass percent in U
^{232}U	0.05%
^{233}U	63.03%
^{234}U	25.12%
^{235}U	5.92%
^{236}U	5.88%
^{237}U	0.00%
^{238}U	0.00%

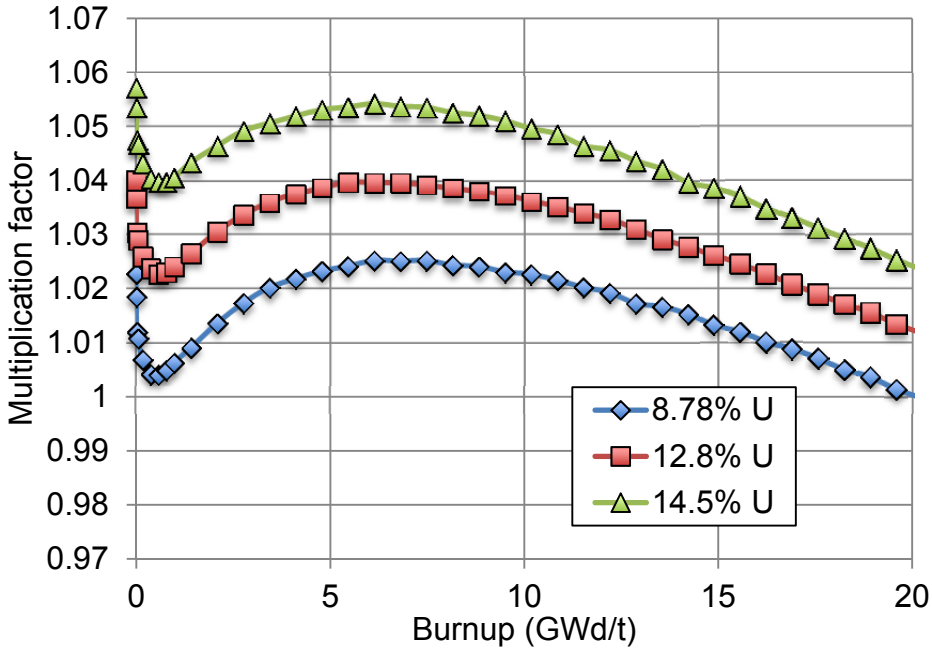


Figure 19. Infinite multiplication factor as a function of U3 content in the driver zone

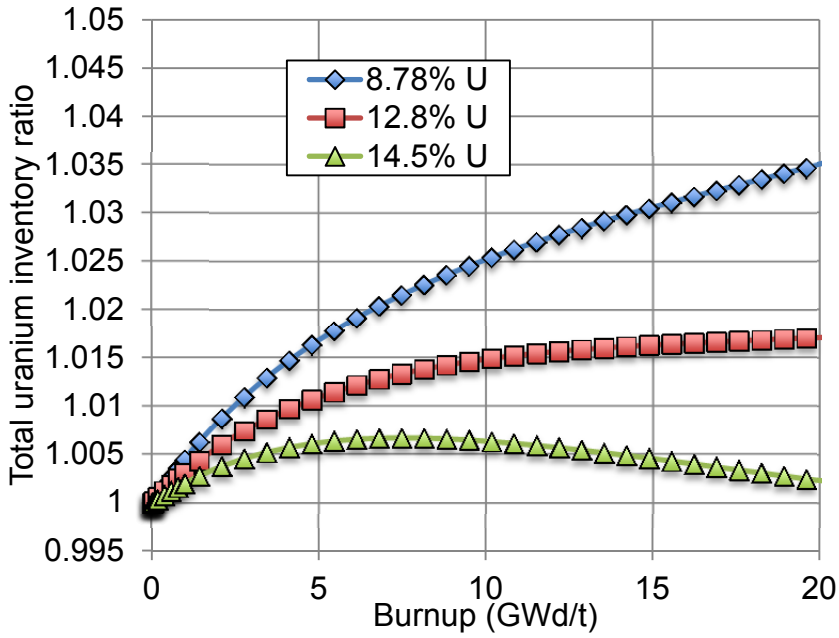


Figure 20. Total uranium inventory ratio as a function of U3 content in the driver zone

The percent of fissions in each energy regime is shown in Figure 21 for the 12.8% configuration. This answers the question regarding the spectral characteristics of this configuration. The fact that a majority of fissions do not occur in any energy regime

indicates that this geometry is neither representative of a thermal spectrum or intermediate spectrum, by the definition used in this study. Instead, this is a "mixed" configuration with shared characteristics of both thermal and intermediate spectra.

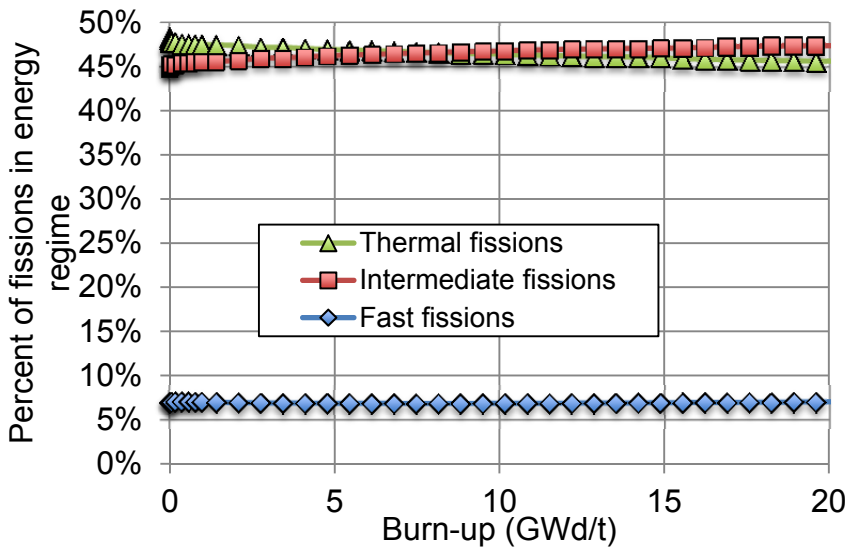


Figure 21. PWR configuration: percent of fissions in thermal, intermediate, and fast energy regime

The reactivity behavior of this system is highly non-linear. Several simplified options exist to calculate fuel management parameters, including a piecewise linear reactivity model and a non-linear reactivity model. To calculate the cycle length and other associated parameters a quadratic reactivity model was utilized. The stability of non-linear reactivity models, and in particular quadratic reactivity models has been extensively studied (Parks and Lewins, 1987). Given that the reactivity trace of this particular configuration is almost parabolic, the 2nd order generalized reactivity model was selected for application here (Driscoll et al., 1990). The reactivity as a function of burnup and associated curve-fit are shown in Figure 22. The reactivity assumes 3% neutron leakage, which yields a discharge burnup of approximately 17 GWd/t for a three-batch core. This discharge burnup is indicated with an arrow on the plot.

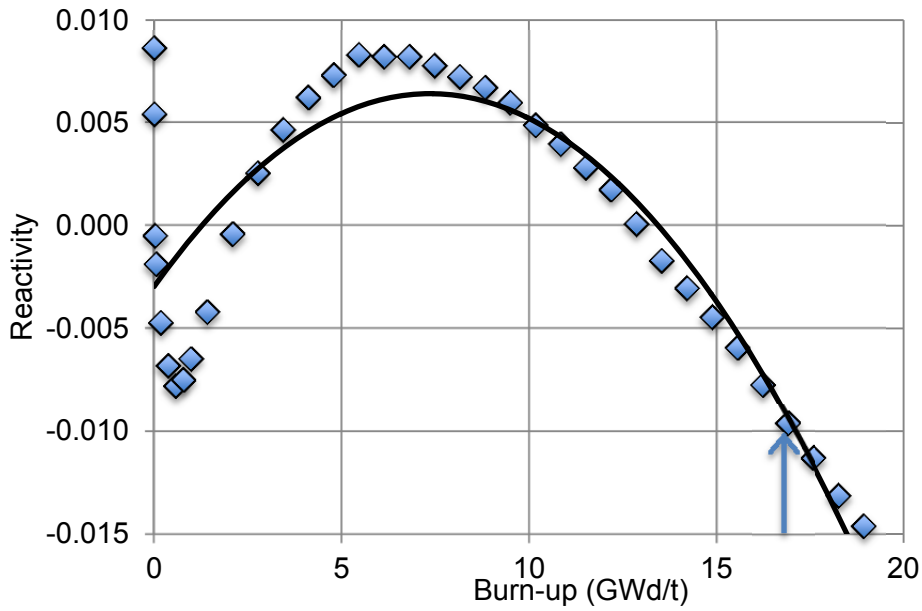


Figure 22. Reactivity as a function of burnup assuming 3% neutron leakage, a 3-batch cycle, and a quadratic reactivity model. Discharge burnup is indicated with the blue arrow.

When averaged over 17 GWd/t burnup the percent of thermal fissions is 47%, intermediate fissions is 46%, and fast fissions is 7%. The uranium and transuranic inventory ratio, after accounting for losses and decay, is 1.010. Therefore this system is actually expected to be a slight breeder, although recycle to near equilibrium conditions was not performed for this case. To summarize the key findings for this realization:

- The system is neither thermal nor intermediate by the definition in this study. This should be considered a "mixed" thermal and intermediate spectrum case.
- Although the analyses indicate that the system may be self-sustaining at equilibrium, there is still uncertainty whether this is really the case. The design may not be viable and there are demonstrated issues with the ^{233}U capture cross section in this energy regime.
- Even if self-sustaining, the system would have low burnup (~17 GWd/t) and very low power density due to de-rating.

- In addition, it was found that this PWR has a very similar fission reaction spectrum when compared to an example MSR, as shown in Figure 23. The fraction of fissions in the intermediate energy regime is 46% for the example PWR and 51% for the example MSR. This finding supports the assertion that an intermediate spectrum design can be achieved by hardening the spectrum of a thermal reactor. In addition, it shows how two breakeven U3/Th systems can have similar fission reaction spectra even with significantly differences in moderator mass.

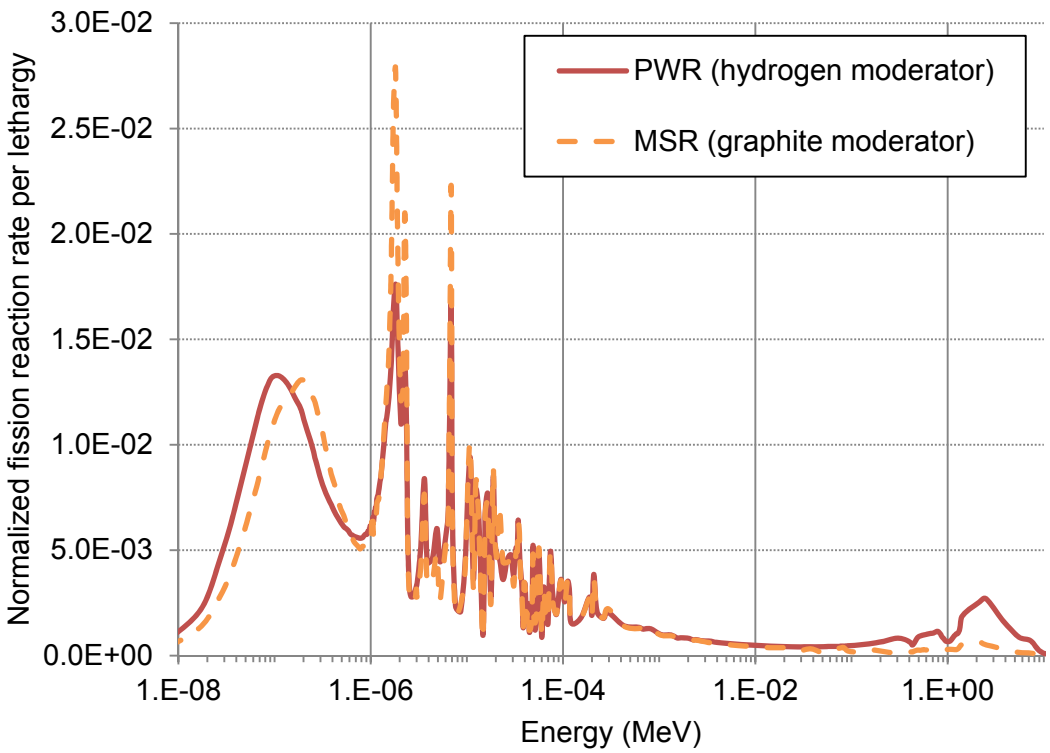


Figure 23. Fission reaction spectra comparison for breakeven U3/Th systems with hydrogen and graphite moderator

Comparison of intermediate spectrum performance with fast spectrum performance

The over-arching goal of the effort is to compare performance of intermediate spectrum systems with fast spectrum systems for this fuel cycle. In the FCT study, fuel cycles were assessed with respect to high-level criteria (Wigeland et al., 2014), to inform on promising options for investment in research and development. These high-level criteria are associated with key quantitative metrics that are calculated based on the physics analyses and mass flows developed for the analysis examples for each fuel cycle option. Each criterion informs on either the benefits or challenges of the given fuel cycle option. Note that only the metrics pertaining to benefit criteria (such as nuclear waste management, environmental impact, resource utilizations) were calculated for the intermediate spectrum designs in this study and compared to the corresponding metrics values for the fast spectrum analysis example (SFR-Th). To account for uncertainties and differences in calculation approaches and designs, each calculated metric is assigned a letter score (A, B, C, *etc.*) based on a binned approach. That is, if the calculated metric values for two different designs fall within the same bin range, then they exhibit the same performance for that metric. More detailed descriptions of the metrics, including their development, rationale, bin boundaries, and calculation methodology can be found in Wigeland et al. (2014). Detailed descriptions of the metrics are presented in Appendix C of the E&S report, including parameter values and metrics calculation approaches (Wigeland et al., 2014). Development of these metrics was integral to a multi-year multi-laboratory study and reproduction of the metric calculation approach is outside the scope of this paper. Interested readers are referred to Appendix C of the E&S report.

In this paper, three specific intermediate spectrum reactor representations, the PWR-D_2O, RBWR-Th, and SFR-ThH, are compared to the reference fast system, an SFR with metallic fuel (SFR-Th), for the benefit metrics. Comparison of the MSR system metrics is not in the scope of this paper. However, MSR systems were analysis examples for several options in the E&S. Interested readers are referred to Wigeland et al. (2014) and Powers et al. (2014). The mass flows at equilibrium are based on the detailed physics analyses described in the paper and are normalized to the electricity generated in GWe-yr. The calculated metric values and binned scores for these criteria are shown in Table 10. Key acronyms used in this table include Spent Nuclear Fuel (SNF), High Level Waste (HLW), Depleted Uranium (DU), Recovered Uranium (RU), Recovered Thorium (RTh), and Low Level Waste (LLW).

The metrics values of the systems compared in Table 10 are all in the same bins, except for the Activity of HLW at 100,000 year after discharge for the D_2O-cooled PWR. In fact, the value of the activity for this intermediate system is right above the bin boundary for this metrics (23×10^{-4} MCi/GWe-Yr) and it is very likely that optimization in the design of this system should drive down this value into the bin C.

The variation in average discharge burnup, which is directly linked to the total annual mass flow of fuel, is an important factor impacting the values of the environmental and resource utilization metrics, the mass of HLW and the volume of LLW. The total mass

flow is linked through the losses to the mass of High Level Waste and to the mass of natural Thorium feed required.

As seen from the results, the benefit metrics performance of the analyzed intermediate spectrum systems in this fuel cycle is very similar to that of the fast spectrum systems. These conclusions may not be applied universally, but the examples analyzed in this study represent a diverse set of technology-specific designs. It is important to note that these specific intermediate spectrum designs are large cores with low neutron leakage, whereas the fast spectrum example in Table 10 is a relatively small reactor core with higher leakage, which consequently relies on radial breeding blankets to achieve self-sustainability. Those radial blanket assemblies are typically discharged with a low burnup, driving down the average discharge burnup of the SFR, which is one of the most impactful parameters affecting metrics performance since it represents the amount of energy released per mass of fuel. Less leaky designs (larger cores) will enable higher burnup and lower mass flows per energy generated, so they will inherently score higher for all metrics relating to resource utilization and waste management. So it is possible that with a more optimized fast reactor design, the binned values may be improved for some of the metrics. It is important to note that burnup and breeding performance of intermediate spectrum systems is primarily determined by the value of η, which is highest at fast energies for all fissile nuclei.

Table 10. Comparison of performance based on several high-level criteria and associated metrics

Criterion	Value and units	SFR-Th Fast Spectrum	PWR-D_2O Intermed. Spectrum	RBWR-Th Intermed. Spectrum	SFR-ThH Intermed. Spectrum
Burnup	GWd/t	23.2	40.0	23.3	40.0
Fissile Inventory Ratio	After 1.0% losses in separations and 0.2% in fabrication	1.00	1.01	1.00	1.00
Nuclear Waste Management	Mass of SNF+HLW disposed, t/GWe-yr	1.56/A	1.45/A	1.45/A	1.41/A
	Activity of SNF+HLW (@100 years), MCi/GWe-yr	1.16/C	1.43/C	1.28/C	1.39/C
	Activity of SNF+HLW (@100,000 years), 10^{-4} MCi/GWe-Yr	22.7/C	23.5/D	21.5/C	12.7/C
	Mass of DU+RU+RTh disposed, t/GWe-yr	0/A	0/A	0/A	0/A
	Volume of LLW, m^3/GWe-yr	1118/D	838/D	1075/D	730/D
Environmental Impact	Land use per energy generated, km^2/GWe-yr	0.086/A	0.084/A	0.084/A	0.084/A
	Water use per energy generated, ML/GWe-yr	23746/B	23728/B	23746/B	23728/B
	Carbon emission - CO_2 released per energy generated, kt CO_2/GWe-yr	48.4/B	34.0/B	48.5/B	33.9/B
	Radiological exposure, Sv/GWe-yr	B	B	B	B
Resource Utilization	Natural Uranium required per energy generated, t/GWe-yr	0/A	0/A	0/A	0/A
	Natural Thorium required per energy generated, t/GWe-yr	1.65/A	1.52/A	1.52/A	1.48/A

As mentioned, the direct comparisons between the fast spectrum analysis example (SFR with metallic fuel) and the intermediate spectrum examples should be considered

carefully before making universal conclusions regarding the tradeoffs between intermediate and fast spectrum designs. However, this study showed that both fast and intermediate spectrum designs for the U3/Th fuel cycle generally yield similar performance in terms of the metrics calculated.

To isolate the effect of the spectrum on the reactor's breeding (and burnup) performance, a scoping study was performed on the SFR-ThH pin model. The reference 0.5 hydrogen-to-heavy metal ratio of the hydride fuel was adjusted to 0.25 and 0.75 while maintaining the roughly same discharge burnup (~40 GWd/t). This was done by simultaneously changing the initial ^{233}U content of the SFR-ThH fuel to roughly match this desired burnup level. The results shown in Table 9 clearly indicate that harder spectra for U3/Th fuels produce improved breeding. If another scoping study were performed with the breeding ratio fixed, then the burnup would be expected to increase as the spectrum hardens: ^{233}U, like all other fissile materials, has much higher η values in the fast energy range. The results from this parametric study indicate that the harder the spectrum, the higher the breeding ratio and/or burnup. Therefore, from a physics point of view, with neutron leakage being equal, the intermediate spectrum systems should offer no breeding or burnup advantages compared to fast spectrum systems.

Table 11. Breeding performance summary for SFR-ThH with various hydrogen to heavy metal ratios

Hydrogen-to-Heavy Metal Ratio	% Fission in E < 1eV (MOL)	% Fission in 1eV - 10^5 eV (MOL)	% Fission in 10^5 eV < E (MOL)	Core Average Discharge Burnup (B_d) [GWd/t]	Fissile Inventory Ratio
0.25	0.53%	55.59%	43.88%	40.02	1.12
0.5	2.34%	65.70%	31.96%	40.00	0.99
0.75	4.88%	69.52%	25.61%	41.08	0.90

Although the burnup and breeding performance is dictated by η, there may be discrete advantages for the intermediate spectrum systems from the perspective of parameters like void coefficient. In this sense, the intermediate spectrum may enable a very low leakage sodium-cooled reactor design because the void worth is reduced. As noted in Zhang and Greenspan (2014), this is due to the relatively constant η of ^{233}U throughout the intermediate energy regime (lack of positive slope). However, it is noted that it is possible to design fast spectrum thorium reactors with low or negligible void worth (Fiorina et al., 2013). In addition, ^{233}U η is greater than 2.0 over many neutron energies, which means that a self-sustaining system is possible with a wide variety of technology options.

Summary

Physics analyses were performed for a diverse set of technology options for an intermediate spectrum system deployed in a particular fuel cycle: continuous recycling of U3/Th in critical reactors with pure fertile thorium feed. The objective was to determine

whether using an intermediate spectrum critical reactor instead of a fast spectrum critical reactor could achieve similar performance in this fuel cycle.

This effort included an examination of a broad spectrum of intermediate spectrum reactor technologies in this fuel cycle, and highlights the technology neutral approach of the overall FCT effort. The study showed that there are two classes of intermediate-spectrum systems considered in this study: those that have significant thermal- spectrum contributions to the overall fissions (with a MSR example in this study) and the other types that have significant fast-spectrum contributions (with examples using water, liquid metal coolant, or a heavier moderator being considered in this study). The analyzed examples with intermediate spectrum reactor systems showed very similar performance, in terms of the calculated metrics, as the reference fuel cycle using an example sodium-cooled fast reactor design with metallic fuel.

The burnup and breeding performance of intermediate spectrum systems in this fuel cycle will be limited by the lower η at intermediate energies versus fast energies. This was confirmed by the parametric scoping calculations that isolated the spectral effects. It is important to note that a comparison between different core designs (a small fast reactor core with high leakage versus large cores with low leakage for the some of the intermediate spectrum technology options) should not be used to make universal conclusions regarding the tradeoffs between fast and intermediate spectra. However, the results of this study indicate that the overall fuel cycle performance is expected to be

similar for fast and intermediate spectrum critical reactors if deployed in this fuel cycle at equilibrium.

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