Fuel Cycle Analysis of Once-Through Nuclear Systems

Fuel Cycle Research & Development

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EXECUTIVE SUMMARY

The fuel cycle performance characteristics of once-through nuclear systems have been evaluated and are presented in this report. The systems include medium burnup (50 GWd/t) and high burnup (100 GWd/t) PWRs, the CANDLE reactor of the Tokyo Institute of Technology, the sustainable sodium-cooled fast reactor (SSFR) by ANL, the fast mixed spectrum reactor (FMSR) by BNL, the ultra-long life fast reactor (ULFR) by ANL, the General Atomics Energy Multiplier Module (EM^2), and the traveling wave reactor (TWR) of TerraPower. Besides the PWRs, the other once-through nuclear systems are fast spectrum systems that have been proposed for achieving extremely long fuel residence time and high uranium utilization. To meet the intended goals, the fast spectrum systems have adopted a design concept that is quite different from that employed for LWR. A breed and burnup concept with propagating burn zone has been utilized with low power density, multi-batch fuel management scheme (in some cases), etc.

The fuel cycle performance parameters of these systems have been compared to those of the medium burnup PWR that has been considered as the reference system in this study. For consistent comparison, most of the fuel cycle parameters have been normalized to the electricity generation in one year (i.e., per GWe-yr). Results indicate that the reference PWR system discharges a used nuclear fuel (UNF) quantity of ~20 metric ton per GWe-year. On the other hand, the fast spectrum systems discharge 3 – 9 ton of UNF per GWe-year depending on the design choices. However, because of the higher breeding ratios of the fast spectrum systems, their plutonium production rate per GWe-year is higher than that of the reference LWR system.

Compared to the reference LWR system, the decay heat levels of the UNFs of the once-through fast spectrum systems are lower. At discharge, the heating level of the fast spectrum systems is 1- 4 MW/GWe-year, which is about a factor of 10 – 40 times smaller than that of the reference PWR. The UNF radiotoxicity has been evaluated using the ingestion dose coefficient specified by ICRP 72. At ten years after discharge, the radiotoxicity values of the once-through fast–spectrum-system UNFs are about a factor of 2 – 5 lower than for the reference system, because of the lower power densities in the once-through fast spectrum systems. At this time point, the fission products dominate the hazard, but the hazard associated with the shorter-lived fission products decreases quickly. The contribution from the actinides becomes dominant after 100 years. Subsequently, after about 1,000-100,000 years, the UNF radiotoxicity values of the once-through fast–spectrum-system are higher because of the contribution of the plutonium isotopes. It takes ~200,000 years for the PWR UNF radiotoxicity to become lower than that of the natural uranium material used in making the enriched uranium fuel for the system. On the other hand, it takes less or comparable time before the radiotoxicity values of once-through fast–spectrum-system UNF fall below the level of natural uranium ore: ~120,000 years for CANDLE, SSFR, and FMSR, and ~200,000 years for ULFR, EM^2, and TWR.

As a measure of the UNF handling difficulty, the neutron and photon source levels per unit mass of the UNF were also evaluated. The once-through fast spectrum systems have a lower minor actinide production rate compared to the LWR system. Consequently, the neutron sources of the fast spectrum system are about a factor of 2 – 8 smaller at discharge. Similarly, the fast spectrum systems have a lower photon source rate by a factor of 3 – 9 at the discharge state. However, the high Cs-137 production rate of the fast systems results in a higher photon source level after 10 years.

The uranium utilization values for the systems have also been compared in this study. For the PWR systems, the uranium utilization is less than 1% regardless of the burnup. For the once-through fast spectrum systems, the uranium utilization could be increased to ~30%, depending on the core design choices. However, some technical design issues would have to be resolved in order for these core concepts to be practical.

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Acronyms

ACZ	Active Control Zone
ANL	Argonne National Laboratory
ARC	Advanced Reactor Concept
BNL	Brookhaven National Laboratory
BOC	Beginning of Cycle
BOL	Beginning of Life
BWR	Boiling Water Reactor
CANDLE	Constant Axial shape of Neutron flux, nuclide density and power shape During Life of Energy production
DU	Depleted Uranium
EM^2	Energy Multiplier Module
EOC	End of Cycle
EOL	End of Life
FCRD	Fuel Cycle Research and Development
FCMI	Fuel-Cladding Mechanical Interaction
FCZ	Fixed Control Zone
FMSR	Fast Mixed Spectrum Reactor
FP	Fission Product
GA	General Atomics
HLW	High-level waste
HM	Heavy metal
HTR	High Temperature Reactor
HWR	Heavy water-cooled reactor
IAEA	International Atomic Energy Agency
LEU	Low Enriched Uranium
LFR	Lead-Cooled Fast Reactor
LWR	Light-Water-Cooled Reactor
MA	Minor Actinides
MOX	Mixed Oxide
MSR	Molten Salt Reactor
NU	Natural Uranium
PWR	Pressurized Water Reactor
SFR	Sodium-cooled Fast Reactor
SNF	Spent Nuclear Fuel
SSFR	Sustainable Sodium-cooled Fast Reactor
SSTAR	Small, Sealed, Transportable, Autonomous Reactor
TRU	Transuranic elements, e.g., Pu, Np, Am, Cm
TWR	Traveling Wave Reactor
ULFR	Ultra Long Life Fast Reactor
UNF	Used Nuclear Fuel
USDOE	United States Department of Energy
USNRC	United States Nuclear Regulatory Commission

SYSTEM ANALYSIS CAMPAIGN FUEL CYCLE ANALYSIS OF ONCE-THROUGH NUCLEAR SYSTEMS

1. INTRODUCTION

Once-through fuel cycle systems are commercially used for the generation of nuclear power, with little exception. The bulk of these once-through systems have been water-cooled reactors (light-water and heavy water reactors, LWRs and HWRs). Some gas-cooled reactors are used in the United Kingdom. The commercial power systems that are exceptions use limited recycle (currently one recycle) of transuranic elements, primarily plutonium, as done in Europe and nearing deployment in Japan. For most of these once-through fuel cycles, the ultimate storage of the used (spent) nuclear fuel (UNF, SNF) will be in a geologic repository.

Besides the commercial nuclear plants, new once-through concepts are being proposed for various objectives under international advanced nuclear fuel cycle studies and by industrial and venture capital groups. Some of the objectives for these systems include:

1. Long life core for remote use or foreign export and to support proliferation risk reduction goals: In these systems the intent is to achieve very long core-life with no refueling and limited or no access to the fuel. Most of these systems are fast spectrum systems and have been designed with the intent to improve plant economics, minimize nuclear waste, enhance system safety, and reduce proliferation risk. Some of these designs are being developed under Generation IV International Forum activities and have generally not used fuel blankets and have limited the fissile content of the fuel to less than 20% for the purpose on meeting international nonproliferation objectives. In general, the systems attempt to use transuranic elements (TRU) produced in current commercial nuclear power plants as this is seen as a way to minimize the amount of the problematic radio-nuclides that have to be stored in a repository. In this case, however, the reprocessing of the commercial LWR UNF to produce the initial fuel will be necessary. For this reason, some of the systems plan to use low enriched uranium (LEU) fuels. Examples of systems in this class include the small modular reactors being considered internationally; e.g. 4S [Tsuboi 2009], Hyperion Power Module [Deal 2010], ARC-100 [Wade 2010], and SSTAR [Smith 2008].
2. Systems for Resource Utilization: In recent years, interest has developed in the use of advanced nuclear designs for the effective utilization of fuel resources. Systems under this class have generally utilized the breed and burn concept in which fissile material is bred and used *in situ* in the reactor core. Due to the favorable breeding that is possible with fast neutrons, these systems have tended to be fast spectrum systems. In the once-through concepts (as opposed to the traditional multirecycle approach typically considered for fast reactors), an ignition (or starter) zone contains driver fuel which is fissile material. This zone is designed to last a long time period to allow the breeding of sufficient fissile material in the adjoining blanket zone. The blanket zone is initially made of fertile depleted uranium fuel. This zone could also be made of fertile thorium fuel or recovered uranium from fuel reprocessing or natural uranium. However, given the bulk of depleted uranium and the potentially large inventory of recovered uranium, it is unlikely that the use of thorium is required in the near term in the U.S. Following the breeding of plutonium or fissile U-233 in the blanket, this zone or assembly then carries a larger fraction of the power generation in the reactor. These systems tend to also have a long cycle length (or core life) and they could be with or without fuel shuffling. When fuel is shuffled, the incoming fuel is generally depleted uranium (or thorium) fuel. In any case, fuel is burned once and then discharged. Examples of systems in this class include the CANDLE concept [Sekimoto 2001], the traveling wave reactor (TWR) concept of TerraPower [Ellis 2010], the

ultra-long life fast reactor (ULFR) by ANL [Kim 2010], and the BNL fast mixed spectrum reactor (FMSR) concept [Fisher 1979].

3. Thermal systems for resource extension: These systems were primarily considered during the INFCE/NASAP evaluations [NASAP 1979] and include various LWR designs for increasing resource utilization (both uranium and thorium). This class would include the Radkowsky seed-blanket concept. Also included in this class are the thermal reactor systems being considered for deployment as small modular reactors, such as IRIS [Carelli 2004], mPower [mPower], and NuScale [NuScale] that are all water cooled reactors.

The purpose of this work is to provide relevant systems and fuel cycle information for some of these once-through fuel cycle systems. In this report, the intent is on providing information on most of the systems from open sources and from scoping studies recently done within the program. As there is insufficient fuel cycle information on the first class of systems, they are not discussed in this report. This lack of information is because most of these have tended to be commercial systems that are still being developed. This is also true for the thermal reactors IRIS, mPower, and NuScale in the third class. Consequently, this work is focused on the middle categories of systems, titled systems for resource extension.

The omission of the systems under class 1 can also be justified on the ground of lack of significant impact to the fuel cycle. These systems have generally been designed to allow fairly near-term deployment. Consequently, even though the systems might have a long core life, they generally achieve it by derating the core power density and using currently known fuel designs. As a result, the fuel burnup is similar to those of typical systems. These new designs do not add much to the fuel cycle beyond known trends for the once-thorough and full recycle nuclear systems. Regarding the INFCE/NASAP thermal systems, only marginal gains (only percent increases not magnitude increases) were indicated at the time when LWR fuel burnup was still quite low in the 1970s. Some of the design changes that were suggested at the time have been implemented into existing LWRs to raise nuclear fuel burnup in the United States.

The computation methods used in this study are summarized in Section 2. Information and scoping results for the once-through nuclear systems are provided in Section 3. The fuel cycle performance parameters of the various systems are compared in Section 4. Section 5 contains the conclusions for this work.

2. COMPUTATION METHODS

The primary objective of this work is to evaluate and compare core performance data and fuel cycle parameters, such as heavy metal masses and used nuclear fuel (UNF) heating and radiation hazards, for the once-through nuclear systems. Various once-through nuclear systems are considered in this study. The Pressurized Water Reactor (PWR) system is considered representative of the commercialized or to be commercialized systems, because the bulk of the commercial systems worldwide are Light-Water Reactors (LWRs) with minor variations. The PWR is a thermal burner system (i.e., the conversion ratio is less than unity), while the newly proposed once-through systems, with the objectives of long core life and high uranium utilization, are fast spectrum breeder systems. Thus, both LWR and fast reactor computation tools are required to evaluate the core performance parameters in this study.

For the LWR systems, the core performance parameters were determined using the linear reactivity model [Driscoll 1990]. Previous studies have indicated that the linear reactivity model coupled with a unit assembly tool can be used to capture the major core performance parameters such as reactivity change versus burnup and average discharge burnup, provided an appropriate neutron leakage/loss fraction is used. In this study, the assembly calculations were performed using the WIMS9 code [WIMS 2005], and the uranium enrichment requirements were determined to achieve the target discharge burnup with the assumed neutron leakage/loss fraction of ~3.5%.

For the fast spectrum systems, the core performance parameters were obtained from whole-core analysis using the REBUS-3 code system [Toppel 1983]. In the REBUS-3 calculations, the fuel composition and the material properties were appropriately adjusted by the smeared density, the thermal expansion and irradiation induced swelling. It is noted that several design criteria such as the peak fast fluence, peak burnup, power peaking factor, etc are relaxed to far beyond the current design limits developed from past irradiation experience. Thus, the study assumes that advanced core design technologies and materials will be available when the proposed once-through nuclear systems are deployed.

The fuel cycle parameters were obtained from ORIGEN-2 [Croff 1980] calculations. The ORIGEN-2 code is widely used for calculating the transmutation of nuclides. It solves the fuel depletion evolution using one-group cross sections or the decay equations for a large number of actinides, fission products, and activation products. However, the ORIGEN-2 code with its cross section libraries cannot be used directly for fuel cycle analysis because the ORIGEN-2 core package does not have the right cross sections for the once-through nuclear systems considered in this study. To solve this issue, a two-step process is used. First, whole-core or lattice calculations are performed using REBUS-3 or WIMS9 code to generate effective one-group cross sections, along with other reactor physics parameters. Subsequently, the ORIGEN-2 depletion calculations are performed with the effective cross sections replacing the original cross sections in the ORIGEN-2 library. In this process, effective cross sections for most of the important actinides are replaced, but those of the fission products are not replaced because the neutronics code (REBUS-3 or WIM9) does not trace all fission products. The accuracy of this coupling procedure was tested and the results are compared in Section 4.

3. ONCE-THROUGH FUEL CYCLE SYSTEMS

3.1 Light Water Reactor

In a once-through fuel cycle, nuclear fuel is used for generating energy in a nuclear system, and then used nuclear fuel (UNF) is removed, stored for some period of time, and subsequently disposed off in a geologic repository for long-term isolation from the biosphere. Thus, the nuclear fuel is used once in the reactor and sent to isolation without further reprocessing. In the United States, Light Water Reactors (LWRs) using enriched UO_2 fuels are widely employed for generating electricity based on the once-through fuel cycle strategy.

The LWR systems that are commercially operated now employ fuels that have an average discharge burnup less than 50 GWd/t. However, the LWR systems have evolved to the third generation, so-called to Advanced Light Water Reactor (ALWR), by incorporating technical progress based on lessons learnt through past reactor operation experience. They are currently offered to potential customers and are under construction in some countries. Since future ALWR designs target high fuel burnup, two PWR systems with medium burnup (50 GWd/t) and high burnup (100 GWd/t) are considered as representative once-through LWR systems in this study.

Table 3.1 shows the design parameters of the PWRs needed to meet the average fuel burnup targets of 50 GWd/t and 100 GWd/t. Mass flow data for each fuel batch is summarized in Table 3.2. In these tables, PWR-50 and PWR-100 indicate the medium burnup and high burnup PWRs, respectively. The PWR-50 system requires low enriched uranium (LEU) fuel with a U-235 content of 4.21 %, while the PWR-100 system requires 8.5% LEU fuel to achieve the higher average burnup of 100 GWd/t. Data in Table 3.2 indicate that about 5.1 % and 10.3% of the initial heavy metal is destroyed by fission in the PWR-50 and PWR-100 systems, respectively.

Table 3.1. Primary Design Parameters of PWR System

Parameters	PWR-50	PWR-100
Reactor power, MWt/MWe	3000 / 1000	3000 / 1000
Target burnup, GWD/t	50	100
Specific power density, MW/t	33.7	33.7
Number of batches	3	3
Cycle length per batch, year	1.5	3.0
Enrichment, %	4.2	8.5

Table 3.2. Mass Flow of PWRs (ton per batch)

	PWR-50		PWR-100	
	Charge	Discharge	Charge	Discharge
Fissile	1.25	0.46	2.52	0.61
U	29.67	27.77	29.67	26.04
Pu	0.00	0.35	0.00	0.51
MA	0.00	0.03	0.00	0.08
Total heavy metal	29.67	28.15	29.67	26.62
U-235/U, %	4.21	0.82	8.50	1.16
Fissile fraction, %	4.21	1.64	8.50	2.31

3.2 Ultra Long Life Fast Reactor

The ultra-long life sodium-cooled fast reactor (ULFR) concept was developed aiming for reactor operation without refueling over a long reactor lifetime [Kim 2010]. The average discharge burnup of typical sodium-cooled fast reactor designs is limited under the current fuel irradiation experience (~10% to 20%). For this ULFR study it is assumed that current and future advances in fuel development and technology would lead to high burnup fuels (average discharge burnup greater than 30%). Thus, design criteria such as fast fluence limit of the cladding material and fuel cumulative damage factor were not allowed to overly constrain fuel burnup in this study.

Zirconium-based alloy is commonly used as metallic fuel in fast reactor designs due to its excellent compatibility with steel-based cladding materials [Hofman 1997], dimensional stability, high heavy metal loading, and good neutron economy. For the ULFR however, molybdenum-based alloy was selected in order to increase the heavy metal loading in the core. Previous irradiation tests [Smith 1957] indicate that the fuel swelling of molybdenum-based metallic fuel is acceptable and its thermal properties are similar to those of zirconium-based metallic fuel. It is noted that the original ULFR core design was developed using a ternary metallic fuel of U-Pu-10Mo, but that design has been revised in this study to use a binary fuel of U-10Mo in order to derive a core concept consistent with once-through fuel cycle operation.

The primary design parameters of the ULFR core concept with a power rating of 3000 MWt are provided in Table 3.3 and the radial core layout is depicted in Figure 3.1.

Table 3.3. Design Parameters for ULFR

Parameter	Value	
Power, MWt	3000	
Specific power density, MW/t	9.4	
Capacity factor, %	90	
Enrichment (inner/middle/outer/blanket), %	9.0 / 11.0 / 13.0 / 0.25	
	Driver and internal blanket	Radial Blanket
- Number of pins	127	169
- Assembly pitch, cm	17.50	17.50
- Overall duct height, cm	535.0	535.0
- Fuel form	U-10Mo	U-10Mo
- Fuel density, g/cm^3	17.2	17.2
- Smeared density, % TD	75	85
- Lower blanket length, cm	50	-
- Active core height, cm	175	275
- Upper blanket length, cm	50	-
- Pin diameter, cm	1.33	1.14
- Pin pitch-to-diameter ratio	1.08	1.10
Volume fraction, %		
- Fuel	41.8	45.0
- Bond	13.9	7.9
- Structure	17.7	19.3
- Coolant	26.5	27.8

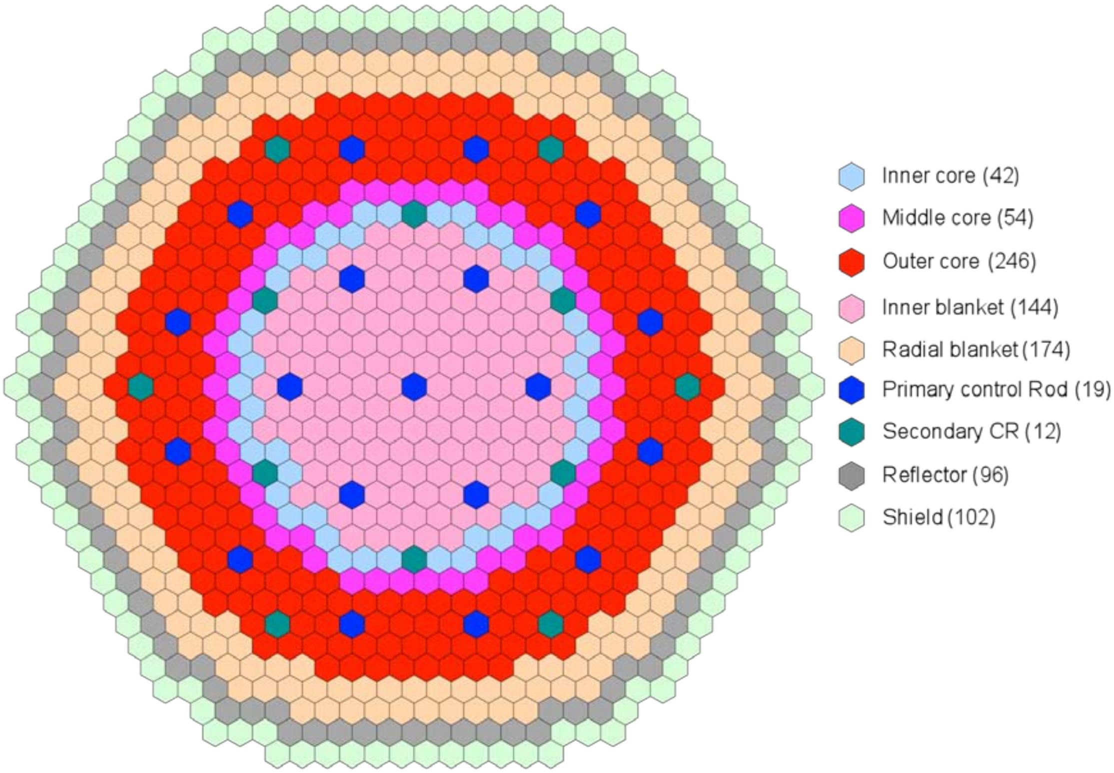

Figure 3.1. Radial Core Layout of ULFR.

The core consists of 342 driver assemblies, 144 internal blanket assemblies, and 174 radial blanket assemblies. The ULFR core has an annular core layout. All internal blanket assemblies are located at the core center, and are surrounded sequentially by driver assemblies and radial blankets. A sensitivity study [Kim 2010] had indicated that this annular core layout can maintain criticality longer than a core layout that has a scattered distribution of the internal blankets by propagating the burn zone. In order to achieve inward power (burnup) propagation, different uranium enrichments have been used for the driver fuels, varied along the core radial direction. The enrichments of the inner, middle, and outer core zones are 9%, 11%, and 13%, respectively. Depleted uranium fuel with U-235 content of 0.25% is loaded into the internal, axial and radial blanket core zones.

The assembly design parameters were determined such that the reactor can maintain criticality for more than 50 years without refueling and the peak excess reactivity can be controlled by a limited number of control assemblies. The active core height is 175 cm and the lower and upper axial blanket have a height of 50 cm. The fuel volume fraction of the driver and internal blanket assemblies is 41.8%, while it is 45.0% for the radial blanket assembly. The smeared density is assumed to be 75% for the driver and internal blanket fuels, and 85% for the radial blanket fuel to allow free fuel swelling. A further reduction of the smeared density has not been considered in this study although a significantly high burnup is expected.

The total heavy metal loading at the beginning of core life (BOL) is 320 metric ton, which results in a specific power density of 9.4 MW/t. Although the core can maintain criticality for the reactor lifetime, the reactor capacity factor was assumed to be 90% (allowing for reactor maintenance time). The ULFR core concept adopted a fission gas vented fuel in order to reduce the height of the assembly duct. The overall duct height of the ULFR assemblies is 535 cm.

The core performance parameters for the ULFR are provided in Table 3.4 and the evolution of the core multiplication factor and breeding ratio with burnup are plotted in Figure 3.2. The core multiplication factor increases initially as the breeding ratio increases, but decreases gradually as the fuel is burned. The core multiplication factor increases again after 35 years because the primary burn zone has moved to the core central region, where the neutron importance is high. Subsequently, the core multiplication factor decreases again as the breeding ratio decreases (i.e., fissile material decreases). Due to this indicated behavior (characteristics of an annular core design), the ULFR core can maintain criticality for 54 years with a capacity factor of 90%.

Table 3.4. Core Performance Parameters of ULFR-3000

Parameter	Value
Thermal power, MWt	3000
Cycle length, year	54
Number of batches	1 (no refueling)
Initial heavy metal inventory (driver/blanket), ton	103 / 217
Discharge burnup (driver/ blanket), GWd/t	316 / 95
Peak excess reactivity, %Δk	3.87
Peak discharge fast fluence, 10^{23} neutrons/cm^2	22.1
Average power density (driver/blanket), W/cm^3	109.1 / 33.1
Overall breeding ratio	~1.12
Power sharing of driver fuels (BOL/EOL), %	94 / 27

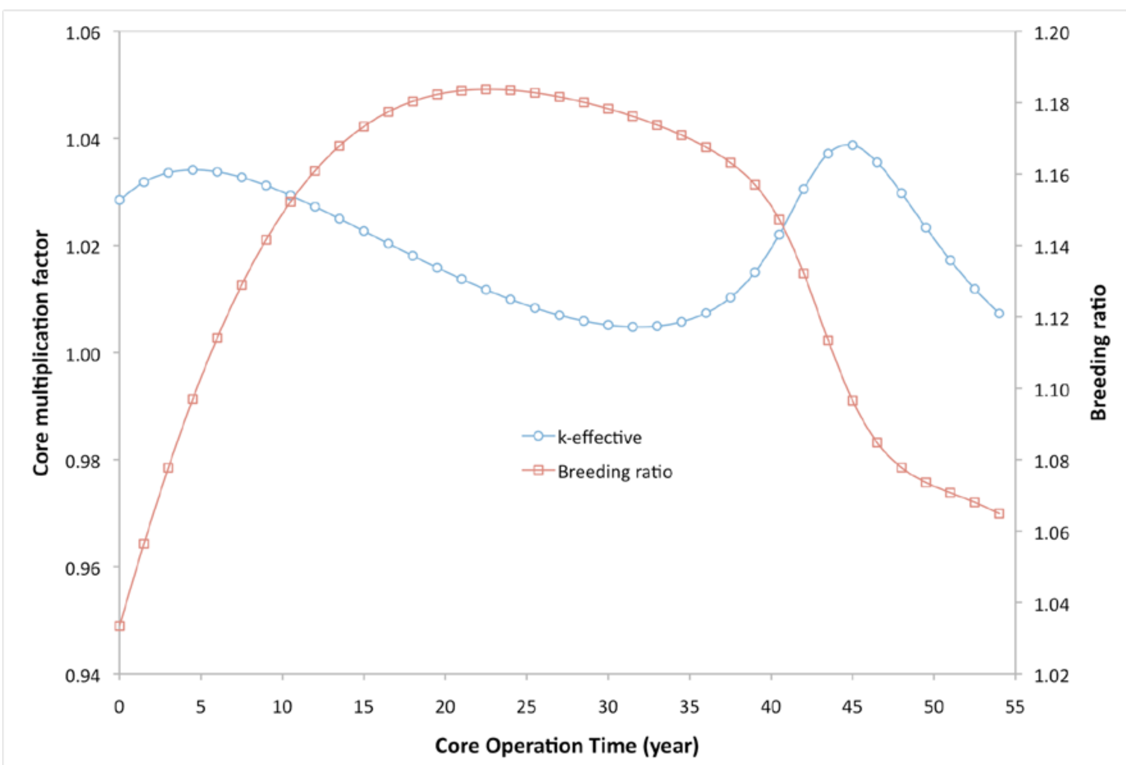

Figure 3.2. Core Multiplication Factor and Breeding Ratio of ULFR.

The overall average burnup of the ULFR discharge fuel is 167 GWd/t. The average discharge burnup of the driver fuel is 316 GWd/t while that of the blanket fuel is 95 GWd/t. The peak fast fluence is ~22×10^{23} n/cm^2, which is about 5 times higher than the current experience. The power sharing of the driver fuel is 94 % at BOL, but decreases to 27% at the end of life, due to the movement of the primary burn zone during the reactor operation.

The main mass flow data for the ULFR are presented in Table 3.5. The ULFR requires 103 ton of 12.2% LEU (average), which requires ~2600 ton of natural uranium and burns about 56.2 ton (17.6% burnup) heavy metal nuclides by fission. Thus, the uranium utilization of the ULFR core concept is about 2.2%.

Several benefits have been attributed to the ULFR; these include capital and operational cost reductions, low proliferation risk, and effectively holding LWR spent fuel without disposal until technologies for a closed nuclear fuel cycle are developed and deployed. Only reactor physics calculations have been done for this concept, and as such not all these claims have been substantiated. Consequently, any future work on the concept will include safety analysis, development of the advanced core cooling methods, and comparative cost analysis (which are not planned for this FY 2010 study).

Table 3.5. Mass Flow Data for ULFR (tons)

	Charge			Discharge		
	Driver	Blanket	Total	Driver	Blanket	Total
Fissile	12.5	0.6	13.1	4.8	12.0	17.0
U	102.6	217.0	319.6	59.5	179.6	239.1
Pu	-	-	-	8.5	15.1	23.6
MA	-	-	-	0.5	0.2	0.7
Total heavy metal	102.6	217.0	319.6	68.5	194.9	263.4
U-235/U, %	12.2	0.25	4.1	0.2	0.02	0.05
Fissile fraction, %	12.2	0.25	4.1	7.1	6.2	6.4

3.3 CANDLE Reactor Concept

The CANDLE (Constant Axial shape of Neutron flux, nuclide density and power shape During Life of Energy production) reactor concept has been considered for very high uranium fuel utilization. The concept was proposed by researchers at the Tokyo Institute of Technology [Sekimoto 2001]. Scoping analysis has been done for this concept within the current work and the results from the analysis are presented here. The reactor design typically has a *starter zone* (at core bottom) and a very tall axial *depletion zone*. The starter zone is used for initial power generation and for the ignition of power generation in the depletion zone. This is accomplished by the use of leaking neutrons to breed fissile material in the depletion zone. This breeding is followed by significant power generation by the *derived* fissile material with continuing core operation.

In this concept, the core active burn-zone moves axially with time. There are various design issues to be resolved before this concept can be considered feasible for further development and deployment. These pertain to (1) the very high fuel burnup that are possible with the design and for which no workable fuel design exists at the current time; (2) the potential difficulties with reactor control due to the very tall core; (3) the feasibility of cooling the core active zone (pressure drop consideration), which is also associated with the long length core.

Figure 3.3 depicts the CANDLE reactor concept that has been modeled in this work. While the original CANDLE [Sekimoto 2001] concept adopted a lead-bismuth cooled fast system with metallic fuel, an alternative sodium-cooled concept has been recently introduced [Sekimoto 2010]. Consequently, the scoping calculation of the CANDLE core concept performed in this study assumed a sodium-cooled fast spectrum system with U-Zr binary metallic fuel. The height of the starter zone is 120 cm and it is designed to have different enriched LEU fuels axially to enhance the axial propagation of the burn-zone: 13%, 7% and 3% from the core bottom with the lengths of 80 cm, 20 cm, and 20 cm, respectively; average enrichment is 10.3%. A depletion zone of height 6.8 m, which contains U-Zr binary metallic fuel with depleted uranium (DU), is located above the starter. The core has a diameter of 4.0 m and is surrounded by a 50 cm thick radial reflector made of depleted uranium. For simplicity, all fuels have the same fuel, coolant, and structure volume fractions of 37.5%, 30.0% and 20%, respectively.

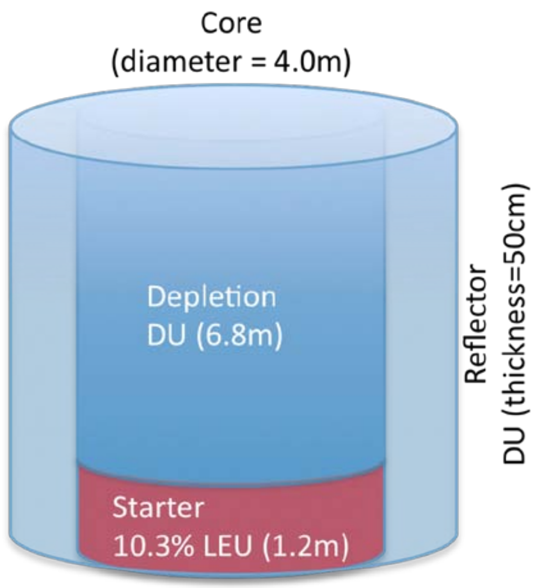

Figure 3.3. Conceptual Drawing of CANDLE Reactor.

The CANDLE reactor considered here has a power rating of 3000 MWt. The core performance parameters are provided in Table 3.6. The time evolution of the core multiplication factor and the axial position of the peak power density are plotted in Figure 3.4 as a function of reactor operation time. Similarly, the propagations of power density, burnup of starter and axial blanket zones (the radial blanket is not included), and the fissile mass are displayed in Figures 3.5, 3.6, and 3.7, respectively.

Table 3.6. Core Performance Parameter of CANDLE

Parameter	Value
Thermal power, MWt	3000
Core height including depletion zone, m	8.0
Cycle length with full power, year	200
Enrichment of starter fuel, %	10.3
Specific power density, MW/t	3.7
HM inventory of initial core, ton	820
Average burnup (starter/depletion/reflector), GWd/t	362 / 396 / 25
Peak excess reactivity, %Δk	3.2
Peak discharge fast fluence, 10^{23} neutrons/cm^2	41.9
Overall breeding ratio	1.20

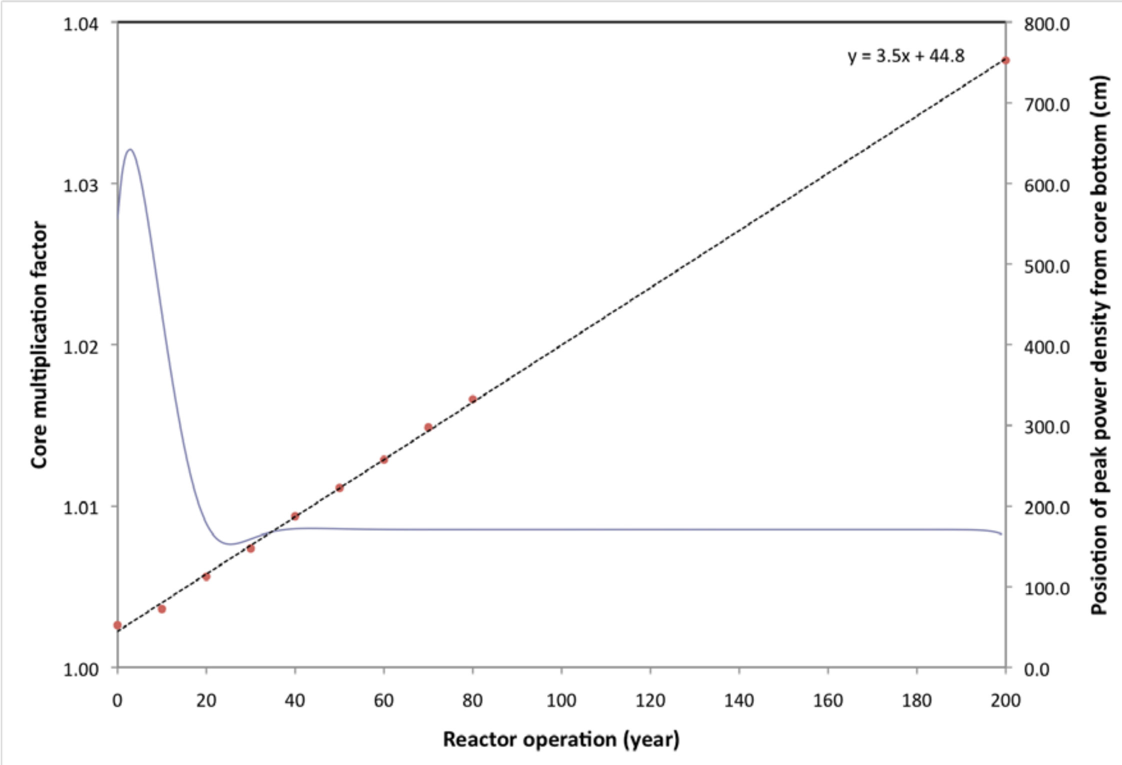

Figure 3.4. Core Multiplication Factor and Peak Power Density Position of CANDLE.

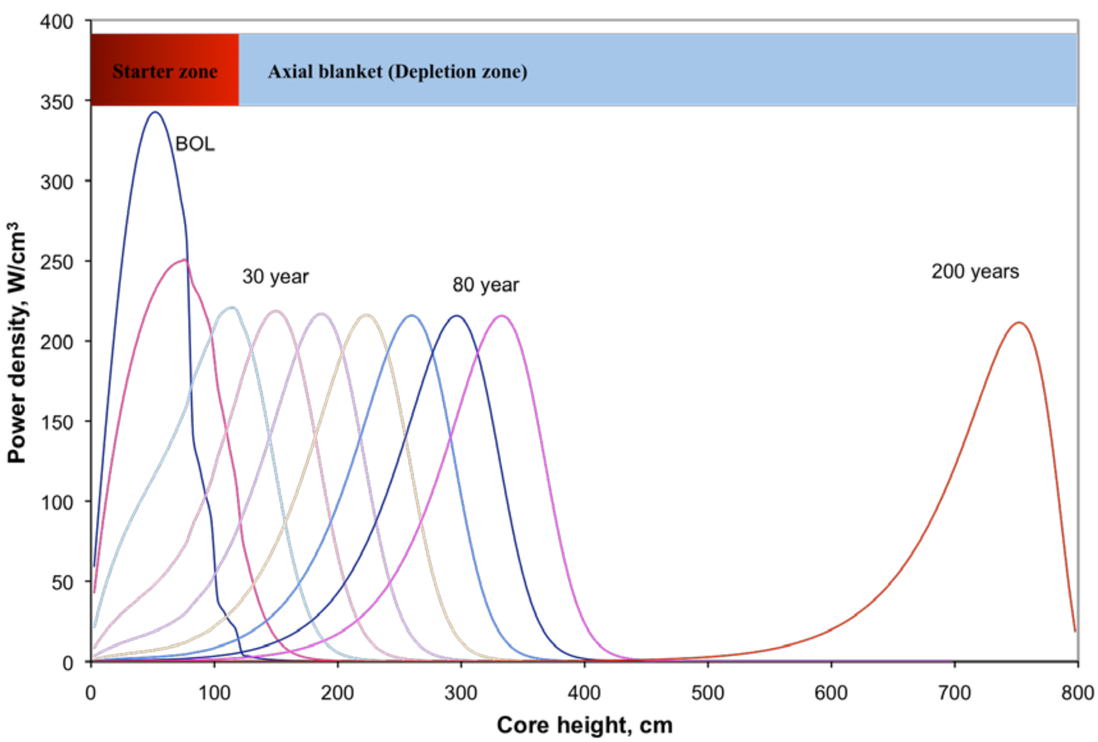

Figure 3.5. Propagation of Power Density Profile in CANDLE.

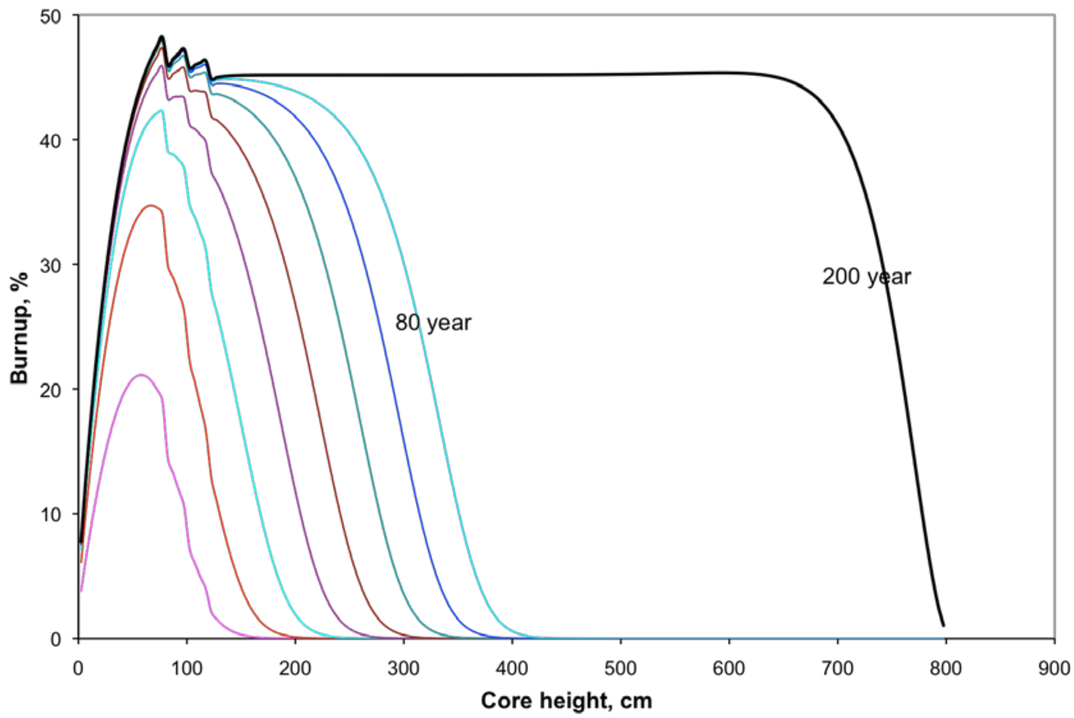

Figure 3.6. Burnup Profile in CANDLE.

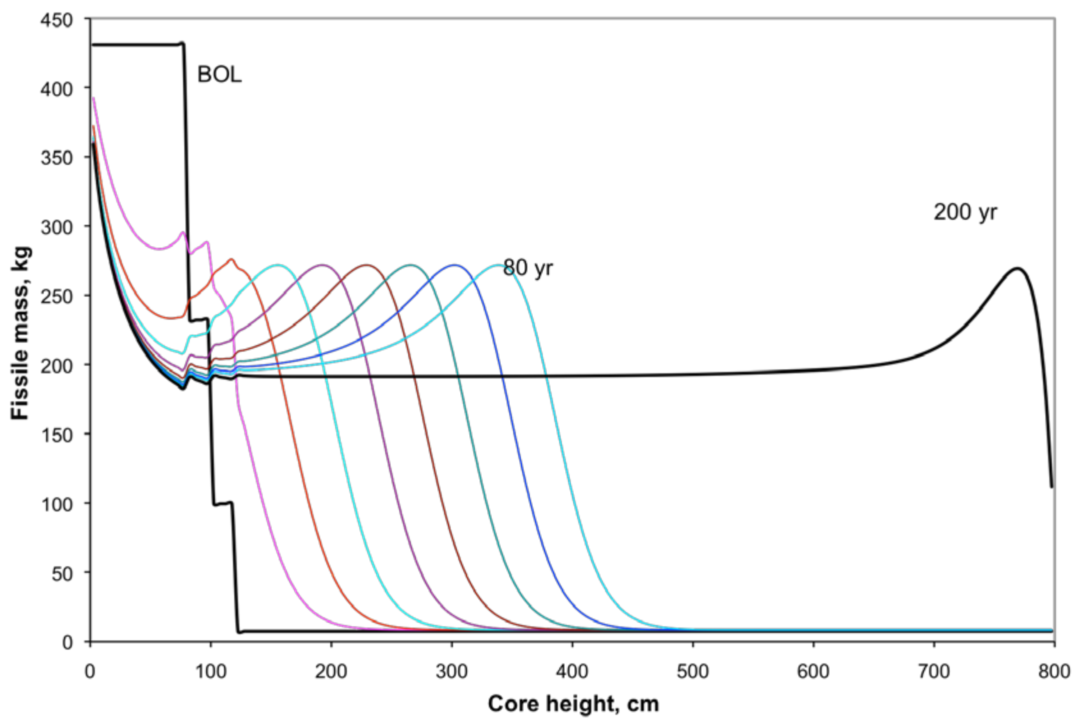

Figure 3.7. Fissile Mass Profile in CANDLE.

Initially, the core multiplication factor increases slightly and decreases as U-235 is burned. However, the core multiplication factor increases again and stabilizes as Pu-239 is bred in the depletion zone. As can be seen in Figure 3.5, most of the core power is initially generated in the starter zone. With increase in the irradiation time, the power density of the starter zone decreases due to the poisoning effect of fission product accumulation. Consequently, the burn zone moves to the neighboring depletion zone. This burn zone propagates to the top of the core. The average burnup of the depletion (axial blanket) zone is ~40%, and the overall average burnup including the radial blanket is about 25%.

The calculated main mass flow data of the CANDLE concept is summarized in Table 3.7. The CANDLE core requires 79.3 ton of LEU fuel for the starter zone, which implies ~1700 ton of natural uranium assuming 0.25% U-235 in depleted uranium. The CANDLE reactor burns about 25% heavy metal by fission, which is 203 metric ton. Thus, the overall uranium utilization of the CANDLE reactor is ~12%.

Table 3.7. Mass Flow for CANDLE System (tons)

	BOL				200 years			
	Starter	Depletion	Blanket	Total	Starter	Depletion	Blanket	Total
Fissile	8.2	1.1	0.7	10.1	5.3	26.4	15.2	46.4
U	79.6	447.6	296.5	823.7	42.6	244.1	274.6	561.3
Pu	-	-	-	-	6.7	36.3	15.3	57.7
MA	-	-	-	-	0.5	1.6	0.1	2.1
Total heavy metal	79.6	447.6	296.5	823.7	49.3	281.9	289.9	621.1
U-235/U, %	10.3	0.25	0.25	1.2	1.2	0.1	0.2	0.2
Fissile fraction, %	10.3	0.25	0.25	1.2	10.7	9.4	5.1	7.5

The burn-zone propagation speed is dependent on the core power level. For instance, the propagation speed is about 3.5 cm per year with the power level of 3,000 MWt (see Figure 3.4), but it increases to 9.3 cm per year for a core power level of 9,000 MWt. The reactor operation time (i.e., how long the core can maintain criticality) is dependent on the height of the depletion zone. In Figure 3.4, the core maintains criticality for 200 years and becomes subcritical when the neutron leakage dominates neutron balance as the burn zone approaches the top of the reactor.

In principle, the CANDLE concept can be used to burn all the depleted uranium resulting from the creation of the initial enriched uranium fuel by increasing the core height because the core can maintain criticality as long as the depleted uranium is available. Possible operation times are shown in Table 3.8.

Table 3.8. Operation Time of CANDLE Cores

Power (MWt)	Height (m)	Operation (year)	Burnup (%)
3000	3.0	59	20
3000	6.0	148	24
3000	8.0	200	25
9000	[a] 15.3	142	28

a) Maximum core height that could be designed using all depleted uranium resulting from making LEU fuel for the starter zone.

In Table 3.8, the height indicates the sum of the starter and depletion zone lengths. For all cases, the height of the starter zone is fixed as 1.2 m and that of the depletion zone is changed. Using all depleted uranium arising from making the LEU fuel of the starter zone, the maximum core height would be

15.3 m. For this case, the reactor can maintain criticality for 142 years with a reactor power rating of 9000 MWt, which is about 426 years with a core power rating of 3000 MWt.

3.4 Fast Mixed Spectrum Reactor

Various concepts have been considered for achieving very high fuel burnup and utilization. An example is the CANDLE concept discussed in Section 3.3 that involves no fuel refueling. Other concepts however attempt to use traditional fuel management approaches for achieving high utilization. An example of this is the fast mixed spectrum reactor (FMSR) design developed by the Brookhaven National Laboratory in the 1970s. A 3000 MWt FMSR was proposed to offer excellent non-proliferation characteristics and to achieve good utilization of uranium resources [Fischer 1979]. The design is in a class of breed and burn systems in which traditional assemblies are used and fuel is charged and discharged.

For the FMSR however, fissile (driver) and blanket fuels are charged into the fast and thermal core zones, respectively. Then fissile material is bred in the thermal zone during core operation. As sufficient fuel is bred and after the driver fuel assembly has reached its discharge burnup, the core is shuffled. During the shuffling process, the burned driver fuel assemblies are discharged, and the bred fuel assemblies are shuffled into the fast core zone. Then fresh depleted uranium blanket assemblies are charged into the thermal core zone. The core is then restarted. It is noted that during the breeding phase, a sufficient irradiation time of the blanket fuel in the thermal zone is required to ensure favorable breeding before it is moved into the fast core zone. In addition, in order to minimize the reactivity change at each reloading phase, the number of discharge fuels is limited and as a result the number of batches increases significantly in the FMSR concept. These directions tend to increase the fast neutron fluence on the pin for a given burnup.

The evaluation of the FMSR system performed by BNL assumed an initial equilibrium state that was used for the analysis. Due to the need to ascertain that such an equilibrium state could be reached, this work performed an evaluation of the concept. The results of that analysis are presented in the following.

The FMSR core concept proposed by BNL is separated into fast and thermal core zones using Beryllium (Be) moderator. The primary purpose of the moderator claimed in [Fischer 1979] is for power flattening and reactivity management, and minimization of the fluence. In order to quantify the impact of the Be moderator, the core performance characteristics of a solid core, in which the moderator between the fast and thermal zones was removed, were compared with the original FMSR core concept. The results indicate that the impacts of the Be moderator on the core performance characteristics are minimal. In addition, the sustainability of the FMSR core concept by BNL could not be reproduced using the proposed cycle length of one year because the fuel residence time was insufficient for the system to breed an adequate amount of plutonium from the depleted uranium. Thus, scoping fuel cycle analysis was performed using the solid FMSR core concept without Be moderator and a cycle length of 1.5 years was found to be required.

Figure 3.8 shows the conceptual drawing of the FMSR and Table 3.9 contains the design parameters. The core has 408 fuel assemblies: 240 assemblies in the fast zone and 168 assemblies in the thermal zone. The fast and thermal zones are represented by driver and blanket zones in the solid core concept, respectively. In the axial direction, the driver fuel is divided into three zones: lower axial blanket, active core and upper axial blanket. The active core height is 160 cm and the thickness of each axial blanket is 40 cm. The fuel form is assumed to be U-Zr binary metallic fuel. In the BNL study, the fuel volume fraction was assumed to be 39% for gas-cooled system and 50% for sodium-cooled system. In this scoping analysis, the sodium cooled core only was evaluated by reducing the fuel volume fraction to 43% because the proposed value is unrealistically high for a sodium-cooled fast reactor.

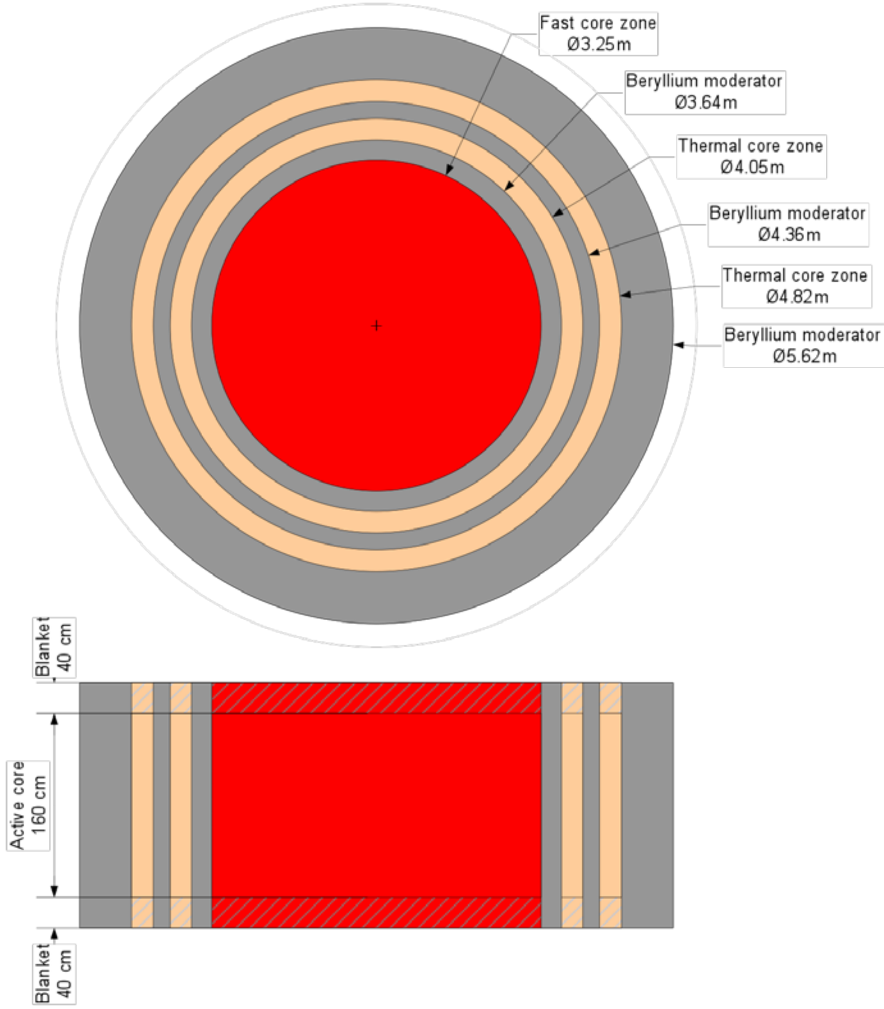

Figure 3.8 Conceptual Drawing of FMSR.

Table 3.9. Design Parameters of FMSR

Parameter	Value
Thermal power, MWt	3000
Number of assemblies (fast/blanket)	240 /168
Number of batches	34
Cycle length, year	1.5
Capacity factor, %	90
Fuel volume fraction, %	43
Fuel form	U-Zr alloy

A 34-batch fuel management scheme was adopted for the FMSR by dividing the core into 34 radial subzones: 20 subzones in the fast core zone and 14 subzones in the blanket zone. The core cycle length was assumed to be 1.5 year with 90% capacity factor. Thus, the fuel resides in the core for 51 years, from charge to discharge. Each zone contains 12 fuel assemblies, and as a result 12 fuel assemblies are replaced at the end of each cycle. To maintain criticality, about 9.2% enriched uranium is loaded into the fast core zone for the first cycle. Depleted uranium assemblies are loaded into the core in subsequent cycles. A fresh depleted uranium assembly is initially loaded into the outermost ring of the thermal core zone, and then it is gradually moved to the inner ring with reload cores. The fuel resides in the thermal core zone for 14 cycles to breed sufficient plutonium. After 14 cycles, the fuel is shuffled into the fast core zone and resides an additional 20 cycles. In the fast core zone, the fuel moves from the innermost ring to the outermost ring of the fast core zone.

Table 3.10 provides the core performance parameters and Figure 3.9 shows the evolution of primary core physics parameters over 60 cycles. Figure 3.9 indicates that it takes 82.5 years (which is 55 cycles) before the core approaches an equilibrium state. At the equilibrium state, the core multiplication factor (uncontrolled) is about 1.025, the breeding ratio is 1.29, and the average burnup of the discharged fuel is 256 GWd/t. During the transition period, the core multiplication factor increases significantly.

Table 3.10. Core Performance Parameters of FMSR

Parameter	Value
Thermal power, MWt	3000
Core height including axial blanket, cm	240
Cycle length, year	1.5
Average U enrichment of fast zone fuel, %	9.2
Specific power density, MW/t	15.7
HM inventory of initial core, ton	19.1
Discharge burnup, GWd/t	256
Excess reactivity at equilibrium state, %Δk	2.5
Overall breeding ratio	1.29

Table 3.11 contains the assembly-wise mass flow data for the depleted uranium fuel over its residence in the core. By the time the depleted uranium fuel is moved into the fast core (driver) zone, its Pu content is 3.3% and its peak fast fluence is 1.2×10^{23} neutrons/cm^2. After the fuel is discharged its burnup is ~260 GWd/t and its peak fast fluence is 19.8×10^{23} neutrons/cm^2. Typical design fluence for a sodium-cooled reactor using HT-9 cladding is 4×10^{23} neutrons/cm^2. Clearly, a fuel that can reach such a high fluence level would be required before this design becomes practical. The BNL report indicates that the peak fluence of the FMSR is 8.0×10^{23} neutrons/cm^2 and the average discharge burnup is 110 GWd/t with the one-year cycle length used in that study. However, these values could not be reproduced in this study: the discharge burnup is shorter than the estimated ball-park number using the specific power density and fuel residence time (i.e., ~160 GWd/t = 16MW/t × 34 years × 80% × 365 days/year), and much shorter than the minimum burnup (~20%) that is required to breed sufficient plutonium for sustainability [Heidet 2010].

The equilibrium cycle of the FMSR core can maintained criticality by feeding in depleted uranium. Thus, the uranium utilization of the FMSR core approaches its discharge burnup of ~27%. The BNL report on the FMSR indicated some design issues needing to be resolved in the areas of reactor physics, thermal-hydraulics, fuels and materials, before the design can be considered feasible [Fischer 1979].

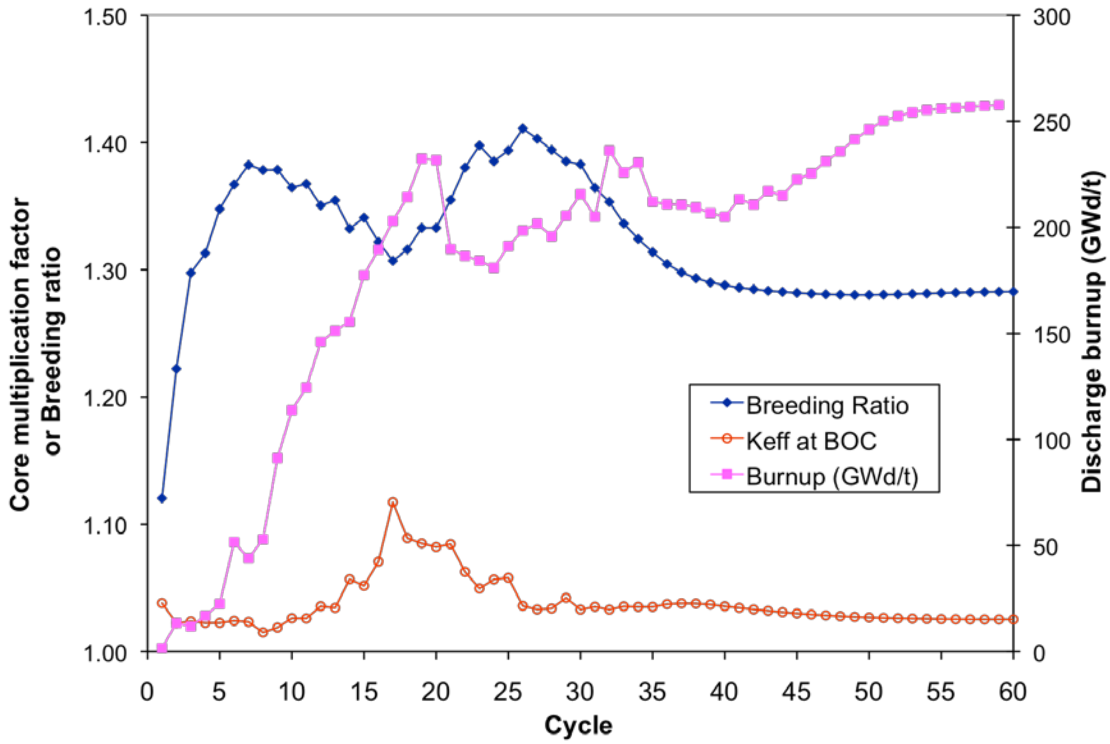

Figure 3.9 Trends of Core Physics Parameters of FMSR.

Table 3.11. Mass Flow per Assembly in FMSR at Equilibrium State (kg)

	Charge	When fuel shuffled into fast core	Discharge
Fissile	1.2	15.9	34.5
U	469.2	450.0	296.4
Pu	0.0	15.5	44.1
MA	0.0	0.0	0.8
Total HM	469.2	465.5	341.3
U235/U, %	0.3	0.2	0.0
Fissile/HM, %	0.0	3.4	10.1
Pu/HM, %	0.0	3.3	12.9
Peak fast fluence, 10^{23} neutrons/cm^2	0.0	1.2	19.8

3.5 Sustainable Sodium Cooled Fast Reactor

As an alternative to the FMSR core concept, the Sustainable Sodium-cooled Fast Reactor (SSFR) core concept has been developed. The primary purpose of the SSFR is to develop a sustainable sodium-cooled fast reactor using depleted uranium (DU) feed only; as in the FMSR design. Sustainability implies a core k-effective maintained at constant critical value for as long as required. Since the fissile content of depleted uranium is insufficient to make the fast reactor core critical, the core requires fissile material initially. The core however becomes sustainable eventually due to the utilization of bred plutonium. For establishing a sustainable fuel cycle, the depleted uranium should be irradiated for a certain period until sufficient plutonium is bred. The original idea of the FMSR is to charge depleted uranium into the thermal core zone for breeding plutonium and shuffle it into the fast core zone as sufficient plutonium is bred. However, the SSFR adopts a conventional sodium-cooled fast reactor concept. It does not have a thermal zone, as the FMSR. In addition, since a cylindrical core geometry was used in the FMSR feasibility study by BNL [Fischer 1979] and hence a multi-batch fuel movement was not modeled explicitly, the evaluation for the SSFR has been done using the hexagonal-Z core configuration for more detailed evaluation. The results of the two systems are however expected to be similar.

The primary design parameters and the core radial layout are provided in Table 3.12 and Figure 3.10, respectively. The core has a power rating of 3000 MWt. It consists of 408 driver assemblies. For power flattening, the first core is divided into four zones: inner, middle, outer core and depletion zones with uranium enrichments of 9.0, 11.0, 14.0, and 0.25%, respectively. The active core height is 120 cm and there are upper and lower axial blankets with 40 cm thickness each. The fuel volume fraction is about 45%. The fuel assembly has 127 fuel pins and a pitch of 20 cm.

Table 3.12. Design Parameter of SSFR

Parameter	Value
Power, MWt	3000
Specific power density, MW/t	16.9
Capacity factor. %	90
Initial enrichment (IC/MC/OC/DU), %	9.0 / 11.0 / 14.0 / 0.25
Assembly design parameters	
- Number of pins	127
- Assembly pitch, cm	20.0
- Overall duct height, cm	436.0
- Fuel form	U-10Zr
- Fuel density, g/cm^3	15.6
- Smeared density, % TD	75
- Lower blanket length, cm	40
- Active core height, cm	120
- Upper blanket length, cm	40
- Pin diameter, cm	1.55
- Pin pitch-to-diameter ratio	1.07
Volume fraction, %	
- Fuel	44.7
- Bond	14.9
- Structure	15.7
- Coolant	24.7

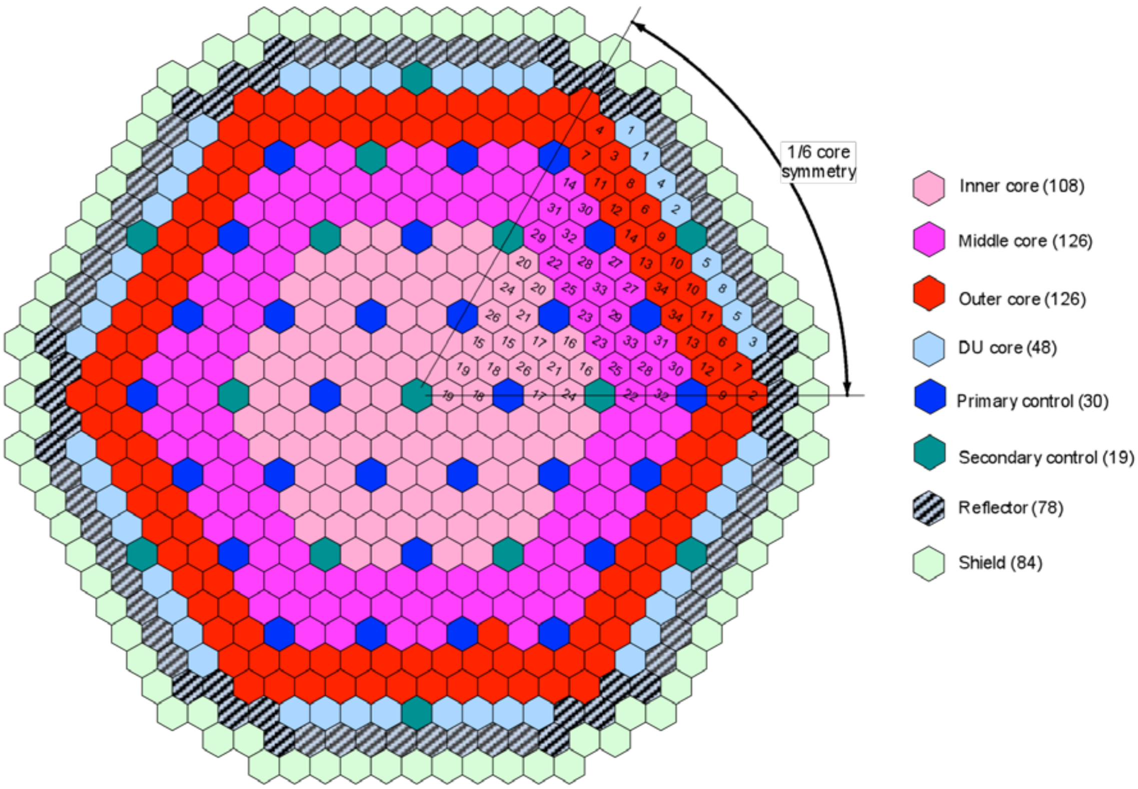

Figure 3.10. Radial Core Layout of SSFR.

The fuel management scheme (including number of batches, cycle length, etc.) was determined from a sensitivity study. The sensitivity study indicated that the depleted uranium fuel should be irradiated more than a lower burnup value in order to breed sufficient plutonium. There is also an upper burnup value as the core cannot be critical when the reactivity penalty from the accumulation of fission products becomes significant. For sustainability, the discharge burnup of the fuel assembly (originally depleted uranium) must be within the range of 20 – 28%. The SSFR core adopts a 34-batch fuel management scheme with a 1.5-year cycle length. Since there are 408 fuel assemblies including the radial blanket, 12 used fuel assemblies are replaced by fresh depleted uranium fuel assemblies at the beginning of each cycle. The sequential fuel movements in a one-sixth core for 34 batches are indicated in Figure 3.10.

The core performance parameters are provided in Table 3.13 and the time evolution of the core multiplication factor is plotted in Figure 3.11 for 100 cycles (which is 150 years). The difference between the BOC and EOC values indicate the reactivity change over each cycle. The core multiplication factor approaches an equilibrium value after a transition period (after 60 cycles, the core multiplication factor is stable). Figure 3.12 displays the average burnup profile as a function of fuel residence time. As indicated in Figure 3.10, the fresh fuel (i.e., depleted uranium fuel) is charged into the core periphery where the power density is relatively small. Thus, the burnup is small for the first 14 cycles. However, as the fuel is shuffled into the active core zones, the burnup increases and becomes 280 GWd/t (peak fast fluence of 26.7×10^{23} n/cm^2) when the fuel is discharged after 34 cycles.

The assembly-wise mass flow for the SSFR is provided in Table 3.14. At the equilibrium cycle, the SSFR core is sustainable with depleted uranium. Thus, the uranium utilization of the SSFR is equivalent to its discharge burnup of ~29%.

Table 3.13. Core Performance Parameter of SSFR

Parameter	Value
Thermal power, MWt	3000
Cycle length, year	1.5
Number of batches	34
Initial heavy metal inventory, ton	177.6
Discharge burnup, GWd/t	276.6
Peak excess reactivity at equilibrium cycle, %Δk	3.1
Peak discharge fast fluence, 10^{23} neutrons/cm^2	26.7
Average power density, W/cm^3	96.2
Overall breeding ratio	1.26

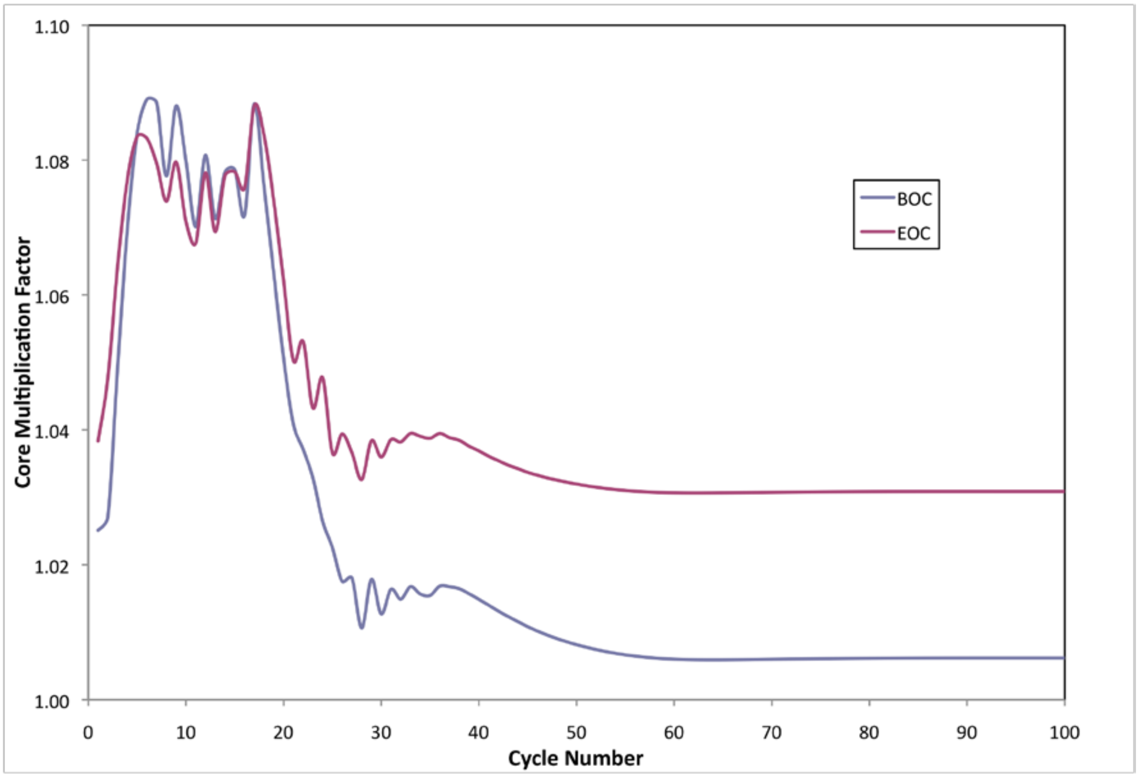

Figure 3.11. Core Multiplication Factor of SSFR.

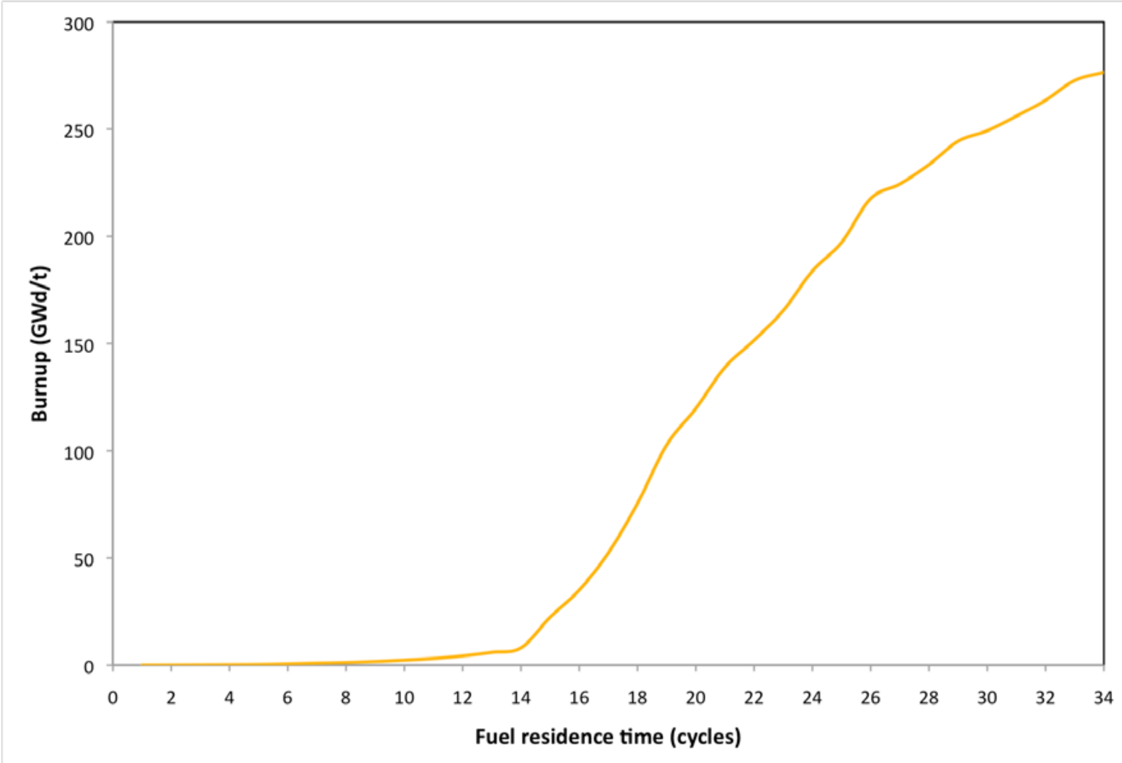

Figure 3.12. Burnup Profile vs. Fuel Residence Cycles in SSFR.

Table 3.14. Mass Flow per Assembly of SSFR at Equilibrium Cycle (kg)

	Charge	Discharge
Fissile	1.1	31.7
U	435.4	266.0
Pu	0.0	41.4
MA	0.0	0.8
Total HM	435.4	308.1
U235/U, %	0.25	0.02
Fissile/HM, %	~0.0	10.3
Pu/HM, %	0.0	13.4

3.6 TerraPower Traveling Wave Reactor Concept

The Traveling Wave Reactor (TWR) concept being developed by TerraPower is intended to provide a technology pathway for fast reactors that do not require reprocessing facilities and a system that offers a high fuel utilization [Ellis 2010]. The system adopts the breed and burn concept in a fast reactor, relying in the use of depleted uranium fuel to generate a significant fraction of the system power. The system will have no external fuel refueling but will allow internal fuel shuffling. Similarly to all breed and burn concepts, the initial core of the TWR requires some amount of fissile fuel, which is currently assumed to be enriched uranium fuel. Since the intent is a regime with no fuel reprocessing, the use of transuranic elements derived from LWR used nuclear fuel is not an option. TerraPower speculates that the TWR should be able to achieve a uranium utilization that is 40 times greater than that of current LWRs.

The current version of the TWR design is based on elements of sodium-cooled fast reactor technology that have been tested in a large number of one-of-a-kind reactors over the years. Conceptually, the core consists of hexagonal fuel assemblies containing enriched uranium fuel or depleted uranium fuel. The core arrangement is such that the breed and burn wave does not move, but is "stationary". This stationary wave is achieved by periodically moving fuel material in and out of the breed and burn zones (shuffling). Metallic fuel is considered for the design because it offers high heavy metal loading and excellent neutron economy. Zirconium is used for alloying the metallic fuel to improve the dimensional stability of the fuel during irradiation and to inhibit low-temperature eutectic and corrosion damage of the cladding.

TerraPower is now considering the "repurposing" (or re-cladding or reconditioning) of the fuel following use. This is to allow the high burnup in a given pass through the core to be increased to a much higher value (about 50%). This repurposing could involve a simplified reprocessing step. In the paper by Ellis et al., the proposed approach is melt refining [Ellis 2010]. This is a new twist to the TWR concept. The evolving reactor design and associated fuel cycle for the TWR is however to be expected.

The activities on the TWR design are proprietary to TerraPower LLC. Information is provided here to inform USDOE efforts on assessing advanced reactor concepts. Figure 3.13 shows the radial core layout at the beginning of life (BOL) for a core with a power rating of 3000 MWt. The core consists of two zones: active control zone (ACZ) and fixed control zone (FCZ). Power is mostly produced in the active zone, while the fixed control zone is used for internal fuel storage. The active core zone contains 360 igniter assemblies and 78 feed assemblies, while the fixed control zone contains 672 feed assemblies. There are 25 control assemblies in the active core zone and 66 control assemblies in the fixed control zone.

The active and feed fuel assemblies contain __ fuel pins arranged in a triangular pitch array. The assembly pitch at the fabrication stage is __ cm. The fuel pin diameter and cladding thickness are __ mm and ___ mm, respectively. Fuel pins are made of sealed cladding containing a lower shield, a binary metallic fuel and a gas plenum from the bottom of the pin. The fuel pin is helically wrapped with wire to maintain pin spacing so that coolant can flow freely through the pin bundle. The heights of the heated fuel pin and fission gas plenum are ___ cm and ___ cm, respectively. The overall assembly height is __ cm including the lower shield of _ cm. Sodium is used as the initial thermal bond between the fuel column and the cladding. At the fabrication state, the resulting volume fractions of fuel, structure, bond sodium and coolant sodium are ~34, ~21, ~14 and ~31 %, respectively.

A U-Zr binary metallic fuel was adopted in the TWR core concept. The initial smeared density is chosen to be _% to allow for the release of fission gas from the metallic fuel. The Zr weight fraction in the binary fuel is assumed to be _%, and ~14 % enriched uranium is used for the igniter fuel while depleted uranium is used for the feed fuel.

The initial core is designed to ensure criticality with a small amount of excess reactivity. The excess reactivity of this breed-and-burn core increases monotonically and the movable control assemblies are used for its control. As noted, fuel shuffling is planned for the TWR. The spent fuel assemblies are replaced by fertile assemblies that originally reside in the fixed control zone. Over the reactor life, it might be necessary to move the spent fuel assemblies more than once to achieve adequate breeding ratio and to minimize damage to fuel assemblies. As in typical breed and burn concepts, the core life depends on the number of assemblies available for fuel shuffling.

(Removed due to proprietary data)

Figure 3.13. Core Layout of Initial TWR Core Concept.

The core performance parameters and the mass flow data for the TWR are presented in Tables 3.15 and 3.16, respectively. Due to the lack of the detailed shuffling schemes from BOL to EOL, the analysis was performed by using the BOL and EOL data that have been obtained by TerraPower. In this calculation, the fuel repurposing (or reconditioning) was not considered. The core is able to operate for 16 cycles with a cycle length of 800 days. Thus, the reactor lifetime is about 38 years with a capacity factor of 93%. About 9.8% of the fuel heavy metal is destroyed after the 38-year operation.

It is re-iterated that the design data for the TWR is changing at the current time, because the system is actively being designed. The data provided in this study is meant to give an indication of the design performance and is not to be taken as the final design data for the TWR.

Table 3.15. Core Performance Parameter of TWR

Parameter	Value
Thermal power, MWt	3000
Core height including axial blanket, cm	
Reactor operation, year	38
Uranium enrichment of ignition assemblies, %	~14
Capacity factor, %	93
Specific power density, MW/t	7.5
Number of cycles	16
Cycle length, days	800
Average discharge burnup, %	9.8

Table 3.16. Mass Flow of TWR (tons)

	Charge	Discharge
Fissile	10.0	20.6
U	399.3	336.7
Pu	0.0	22.9
MA	0.0	0.5
Total heavy metal	399.3	360.1
U-235/U, %	2.5	0.3
Fissile fraction, %	2.5	5.7

3.7 Energy Multiplier Module Design Of General Atomics

General Atomics (GA) has been developing the Energy Multiplier Module (EM^2) concept as a reprocessing-free approach to the nuclear fuel cycle that improves fuel utilization and incorporates both depleted uranium and used nuclear fuel wastes into the fuel cycle without reprocessing [Schleicher 2009]. The breed and burn reactor concept is used to obtain high fuel burnup (about 3 to 5 times that of operating LWRs). It is planned to have a reactor design that enables factory-built, modular plants for improved economics.

According to GA open literature, the current design is a helium gas-cooled fast reactor using ceramic materials. The system consists of a starter section and a depleted (DU) and/or used nuclear fuel (UNF) conversion section. Initially, power is generated in the starter section and excess neutrons are used for converting the fertile material into fissile fuel over the reactor life, which is targeted to be greater than 30 years. A porous mono-carbide fuel that maintains good thermal conductivity while providing space for solid fission product accumulation is being contemplated. The core outlet temperature is 850^{o}C. The reactor is to be coupled to a Brayton cycle with thermal efficiency of ~50% according to GA [Schleicher 2009]. The fuel cladding and internal core structures are constructed of high density composite; work would be required to qualify the performance of the material under the high temperature and irradiation fields of the EM^2 core. The fuel form and supporting structural elements contain channels that permit continuous venting of fission product gases, which is needed to avoid excessive pressure buildup during long, uninterrupted periods of operation. This is an item requiring further qualification to ensure that it is acceptable to the USNRC.

The gas-cooled fast reactor is designed to have attractive safety characteristics, namely negative temperature coefficient over the full operating temperature range and a negligible void coefficient [Schleicher 2009]. The reactor is also designed to have low excess reactivity, thus enabling effective shutdown and reactivity control mechanisms to be embedded in the reflector, and also reducing reactor vessel height. The small size of the reactor allows it and the power conversion system to be sited completely underground. This underground sitting and the lack of refueling have been attributed to decreasing the proliferation risk of the nuclear system. [Schleicher 2009]

The reference module size is 500 MWt. GA indicated that the excess reactivity is less than 1% for the full life of the reactor. Additionally, GA stated that all power profiles and component temperatures have been found to be acceptable. It was also stated that the fuel burnup achieved is identical to the total mass of the starter, the implication being that the waste at end of life is nearly the same as the waste component of initial fueling. As a consequence, there would be no further growth of the nation's nuclear waste inventory. GA claims that in the proposed second generation designs, the waste stream will contract and SNF storage could be essentially eliminated, and given the substantive inventories of DU and SNF, it is conceivable that these reactors could provide the entire U.S. energy supply for over 500 years without mining and enriching new uranium fuel. [Schleicher 2009]

The literature indicates that GA is planning to re-use burned EM^2 fuel in subsequent cores. For spent nuclear fuel to be re-used in the EM^2, the fuel has to be processed somehow. For proliferation risk reduction, GA does not intend to use full separations, but is promoting a yet undemonstrated manufacturing process that is a variant on the DUPIC process. The primary motivation is to exclude wet-chemistry or chemical separation. Only cladding removal and volatile fission product removal by heating is proposed prior to introduction of the material into the fuel fabrication process - solid fission products are left in place. A very similar process is also being considered to modify the bred fuel at the end-of-life (which contains significant fissile material) to serve as the starter material for a subsequent generation of reactors. Clearly if the concept includes the reprocessing or reconditioning the fuel, then it is no longer a once-through fuel cycle concept. At the time of the original selection of concepts to evaluate, it was though that the EM^2 is a once-through fuel cycle system due to its the long core life. There had also been statements to the effect that the long life would allow the U.S. sufficient time to decide what to do with spent fuel, implying maybe the fuel would be disposed off. As a result, our analysis for this concept has assumed that it is a once-through fuel cycle system.

At the current time, only limited work has been done on this design and consequently only ball-park estimates of economics has been performed by GA. The company noted that such estimates indicate that the system shows considerable promise. GA also makes the argument that while higher fuel utilization is a significant benefit to life cycle cost, the improvement in the area of financial risk owing to capitalization of the entire fuel load may be even more important. The comparatively low unit power and compact size are advantageous in today's capital-constrained world [Schleicher 2009]. These features also facilitate transitioning to a more efficient model for constructing nuclear plants. Both the reactor vessel and

Brayton power conversion vessel/internals can be manufactured in the U.S. and shipped by commercial truck transport. The company projects a construction time of 2-3 years, following a mature supply system and licensing process [Schleicher 2009].

According to GA, the non-proliferation attributes for the system are:

- Eliminates need for long-term storage of spent fuel containing significant Pu.
- Eliminates need for conventional reprocessing (isotopic separation) for the long-term future.
- Eliminates need for U-235 enrichment (for 2nd generation EM^2 units).
- Reactor core inaccessible without special remote handling equipment.
- Low excess reactivity- core cannot be easily reconfigured for material insertion/extraction

The technology needs and research challenges identified by GA include:

- Transport and thermal-chemical behavior of fuel and fission products over decades
- Projecting properties of SiC composites under high neutron fluences and high temperatures
- Efficient and effective separation of fission products from reactor discharge
- Defining and establishing manufacturing base to realize the cost effective fabrication of modular reactors like EM^2.

The design parameters and selective performance parameters for the $EM^2$are provided in Table 3.17. The information provided here for the EM^2 is the property of GA and is used to inform this USDOE study. The core power rating is 500 MWt and the core can maintain criticality for more than 30 years without refueling. The coolant exit temperature is 850°C, which results in a thermal efficiency of 47.6%, according to GA. The EM^2 reactor has a right-cylindrical shape with an approximate m diameter and _ m height including reflector and shield. The active core consists of starter, fertile and converter zones surrounded by Beryllium oxide reflector, graphite reflector and boron carbide shield. Axially, the LEU fuel regions (starter and fertile) are sandwiched between three converter layers. The starter and fertile regions contain _% and _% LEU fuels, respectively, and the converter contains depleted uranium fuel. As a result, the core average uranium enrichment is 6.1%.

The mass flow for the EM^2 is provided in Table 3.18. The average fuel fissile fraction increases to 8.4% at the EOL due to the high breeding ratio, and the overall burnup is 136 GWd/t.

Table 3.17. Design and Core Performance Parameters of EM^2

Parameter	Value
Thermal Power, MW	500
Cycle length without refueling, year	> 30
Vessel diameter, m	5.0
Coolant material	He
Coolant outlet temperature, °C	850
Specific power density, MW/t	11.8
HM inventory of initial core, ton	42.5
Core average burnup, GWd/t	136
Overall breeding ratio	1.1
Peak fast fluence, 10^{23} neutrons/cm^2	7.3

Table 3.18. Mass Flow of EM^2 (assumes no recycling)

	Charge (kg)	Discharge (kg)
Fissile	2.6	3.1
U	42.5	33.0
Pu	0.0	3.5
MA	0.0	0.1
Total heavy metal	42.5	36.6
U235/U, %	6.1	1.1
Fissile fraction, %	6.1	8.4

4. FUEL CYCLE PERFORMANCE PARAMETERS

Fuel cycle performance parameters of the different once-through nuclear systems evaluated in this study are compared in this section. In Section 4.1, the system core performance parameters are presented. Comparisons of used nuclear fuel (UNF) characteristics, decay heat, and radiotoxicity, are provided in Sections 4.2, 4.3, and 4.4, respectively. Neutron and photon source rates calculated for the systems are presented in Section 4.5. A comparison of the uranium utilization values is given in Section 4.6.

4.1 Computation Bases

Table 4.1 provides a summary of the design and core performance parameters that were used in the fuel cycle performance analysis of the once-through nuclear systems. The PWR-50 and PWR-100 cases represent the PWR fuel cycles with fuel discharge burnups of 50 GWd/t and 100 GWd/t, respectively. In the following comparison of cases, the PWR-50 case is considered the reference fuel cycle option. In Table 4.1 the average uranium enrichment of the feed fuels represent the U-235 mass fraction per total heavy metal mass in the core including the axial and radial blankets. Thus, these values are smaller than the enrichments of the driver fuels that are required for igniting (driving) the reactors. Similarly, the average burnup is the homogenized value over all fuels.

Except for the SSFR and FMSR systems, the once-through fast spectrum systems employ a one-batch fuel management scheme in which the whole nuclear fuels are charged and discharged at the beginning of life (BOL) and end of life (EOL), respectively. Consequently, for consistent comparison to the multi-batch nuclear systems, the annual UNF production rate is evaluated by dividing the heavy metal inventory by the reactor lifetime. It is noted that the TWR is considered a one-batch system although fuel shuffling occurs 15 times; the fuel shuffling is between the active core zone and fixed core zone, and the UNF discharge occurs at EOL. The SSFR and FMSR systems utilize a multi-batch fuel management scheme as in the once-through LWR system, but for these fast spectrum systems, LEU fuel is only required for the initial core because the core is sustainable by feeding depleted uranium (DU) fuel in subsequent cycles. For these two fast spectrum systems, the burnup values that are indicated are those for the equilibrium state.

Except for EM^2 and TWR systems, the thermal efficiencies of the LWR and fast spectrum systems were assumed to be 33.3% and 40%, respectively. The thermal efficiencies of the EM^2 and TWR were reported as 47.6 % and 37.8%, respectively. The thermal efficiency of the EM^2 is higher than those of the other systems because of the high operating temperature obtainable using gas coolant and graphite.

Compared to LWR systems, the once-though fast spectrum systems are significantly derated. However, due to their long fuel residence times, the average burnup of the fast spectrum systems are mostly higher than those of the LWR systems. The burnups of the CANDLE and equilibrium SSFR and FMSR reactors are about 250 – 270 GWd/t and the burnups of ULFR, EM^2 and TWR are 166, 136 and 93 GWd/t, respectively.

As a thumb rule, the burning of one gram of fissile material produces thermal energy of about one MWt-day, which is equivalent to the destruction of approximately a ton of heavy metal by fission per year in a 3000 MWt reactor. Except for the EM^2, the power rating of the once-through nuclear systems is designed to be 3000 MWt, indicating a heavy metal consumption of about one metric ton per year. For the EM^2 system, the heavy metal fission rate per year is about one-sixth of the values for other systems, due to its smaller power rating.

Table 4.1. Summary of Design and Core Performance Parameters of Once-Through Nuclear Systems

	PWR-50	PWR-100	CANDLE	SSFR	FMSR	ULFR	EM^2	TWR
Reactor Power, MW-thermal	3000	3000	3000	3000	3000	3000	500	3000
Reactor Power, MW-electric	1000	1000	1200	1200	1200	1200	238	1135
Thermal efficiency, %	33.3	33.3	40.0	40.0	40.0	40.0	47.6	37.8
Reactor Capacity Factor, %	90	90	90	90	90	90	90	93
Neutron spectrum	thermal	thermal	fast	fast	fast	fast	fast	fast
Fuel form	UOX	UOX	U-Zr	U-Zr	U-Zr	U-Mo	UC	U-Zr
Uranium enrichment, %	4.21	8.5	1.2	[a] 6.2/0.25	[a] 3.8/0.25	4.1	6.1	2.5
Tail uranium enrichment, %	0.25	0.25	0.25	0.25	0.25	0.25	0.35	0.30
Number of batches	3	3	1	34	34	1	1	1
Average burnup, GWd/t	50	100	258	277	257	166	136	93
Specific power density, MW/t	33.7	33.7	3.7	16.9	15.7	9.4	11.8	7.5
Cycle length per batch, year	1.5	3.0	[b] 200.0	1.5	1.5	54.0	37.0	38.0
HM inventory, ton	89.0	89.0	823.7	177.6	191.4	319.6	42.5	399.3
HM charge per batch, ton	29.7	29.7	823.7	5.22	5.6	319.6	42.5	399.2
HM discharge per batch, ton	28.1	26.6	621.1	3.7	4.1	263.4	36.6	360.1
HM fission, ton/year	1.03	1.02	1.03	1.02	1.02	1.04	0.16	1.03

a) First core and charge fuel of equilibrium core
b) Reactor operation time with 8 m core active height.

The fuel cycle performance parameters such as mass flow, decay heat, radiotoxicity, neutron and photon source levels, and uranium utilization were evaluated using the ORIGEN-2 code. As indicated in Section 2, a two-step process was utilized for the fuel cycle analysis. First, whole-core or lattice calculation is performed using the REBUS-3 or WIMS9 code for generating effective one-group cross sections, along with other core parameters. Subsequently, ORIGEN-2 depletion calculations are performed overriding the existing cross sections from the ORIGEN-2 library with the effective cross sections.

In this process, region- and burnup-dependent effective one-group cross sections are generated because the neutron spectrum varies significantly in the once-though fast spectrum systems employing the breed and burnup concept. For instance, in order to correctly account for the spectrum change from charge to discharge in the SSFR core, the effective cross sections were generated by tracing the fuel movement for 34 cycles at the equilibrium cycle and providing the data to the ORIGEN-2 depletion calculations. Similarly, the region- and burnup-dependent flux levels were also obtained by tracing the fuel movements. Consequently, the ORIGEN-2 code performs the depletion calculation with the right one-group cross sections and flux levels. Analogue approaches were applied for the other once-through fast spectrum systems.

The effective cross sections for most of the actinides are replaced, but the fission product cross sections from the ORIGEN-2 library are used. In this study, *fftfc.lib* and *pwrue.lib* (which were originally prepared for FFTF fast reactor and PWR analysis, respectively) were the sources of cross sections for the *remaining* actinides and fission products of the fast-spectrum and LWR systems, respectively.

Since the ORIGEN-2 calculations used effective one-group cross sections obtained from the core (or assembly) neutronics calculations, the actinide masses from the two calculations should be similar. Table 4.2 provides a comparison of the isotopic discharge masses obtained from the neutronics-code and ORIGEN-2 calculations for the PWR-50 and ULFR cases. The results are normalized to the initial heavy metal metric ton (IHMMT). While only two systems (PWR and ULFR) were selectively compared in this table, similar trends were observed for the other systems.

Table 4.2. Comparison of Isotopic Masses at Discharge (g/IHHMT).

	PWR-50			ULFR		
Nuclide	WIMS9	ORIGEN-2	Diff. (%)	REBUS-3	ORIGEN-2	Diff. (%)
U-235	7696.7	7416.0	-3.6	2,438.2	2,515.0	3.2
U-236	5475.2	5523.0	0.9	4,034.4	4,060.0	0.6
U-238	922340.0	921700.0	-0.1	741,679.3	739,500.0	-0.3
Np-237	636.1	636.3	0.0	1,490.3	1,491.0	0.0
Pu-238	297.2	297.9	0.2	772.1	773.9	0.2
Pu-239	6132.0	6089.0	-0.7	61,142.6	61,290.0	0.2
Pu-240	2814.2	2933.0	4.2	10,977.0	11,130.0	1.4
Pu-241	1779.3	1794.0	0.8	757.1	811.3	7.2
Pu-242	853.2	873.8	2.4	194.5	199.0	2.3
Am-241	60.7	59.9	-1.3	485.9	454.4	-6.5
Am-242m	0.8	0.8	-2.2	22.8	20.1	-11.7
Am-243	195.9	196.8	0.5	23.6	23.9	1.5
Cm-242	25.4	25.3	-0.5	6.9	6.9	0.0
Cm-243	0.7	0.7	0.7	0.2	0.3	9.0
Cm-244	83.7	84.8	1.2	5.5	5.9	6.4
CM-245	5.5	5.6	1.3	0.8	0.8	5.5

The maximum difference (Diff.) observed for the PWR-50 case is less than 5%, with most differences generally less than 1.5%. Differences as high as ~12% are observed for the higher actinides of the ULFR case, despite the fact that region- and burnup-dependent cross sections from the REBUS-3 calculations were provided to the ORIGEN-2 calculations. This indicates that the fuel movement over the 34 cycles is not properly captured in the ORIGEN-2 model. However, the ORIGEN-2 result with the effective cross sections for each once-through nuclear system can be utilized for the fuel cycle analysis because the errors of the major nuclides such as U-238 and Pu-239 are less than one percent and the total heavy metal mass (or the fission product mass) is well conserved.

4.2 Discharge UNF Masses and Characteristics

Table 4.3 provides the discharged UNF masses for each once-through nuclear system, including the plutonium isotopic vectors (compositions). The results were obtained from the ORIGEN-2 calculations and the values are normalized to unit electricity generation for one year (t/GWe-yr) for consistent comparison. It is noted that values of the multi-batch systems (such as PWR, SSFR and FMSR) are those for the equilibrium cycle.

As indicated in Table 4.1, the specific power densities of the once-through fast spectrum systems are significantly derated, implying higher heavy metal inventories than for the LWR system. However, the high burnup with a long fuel residence time for these systems results in lower UNF mass per unit electricity generation (UNF production rate). Figure 4.1 shows the trend of the normalized UNF production rate versus average burnup. Generally, the normalized UNF production rate is inversely proportional to the average burnup; the minor variations were caused by the different thermal efficiency and capacity factor. The PWR-50 system has the highest UNF mass (~20 t/GWe-yr), while the SSFR has the lowest UNF mass (~3t/GWe-yr). The UNF production rate of the TerraPower TWR is comparable to that of the PWR-100.

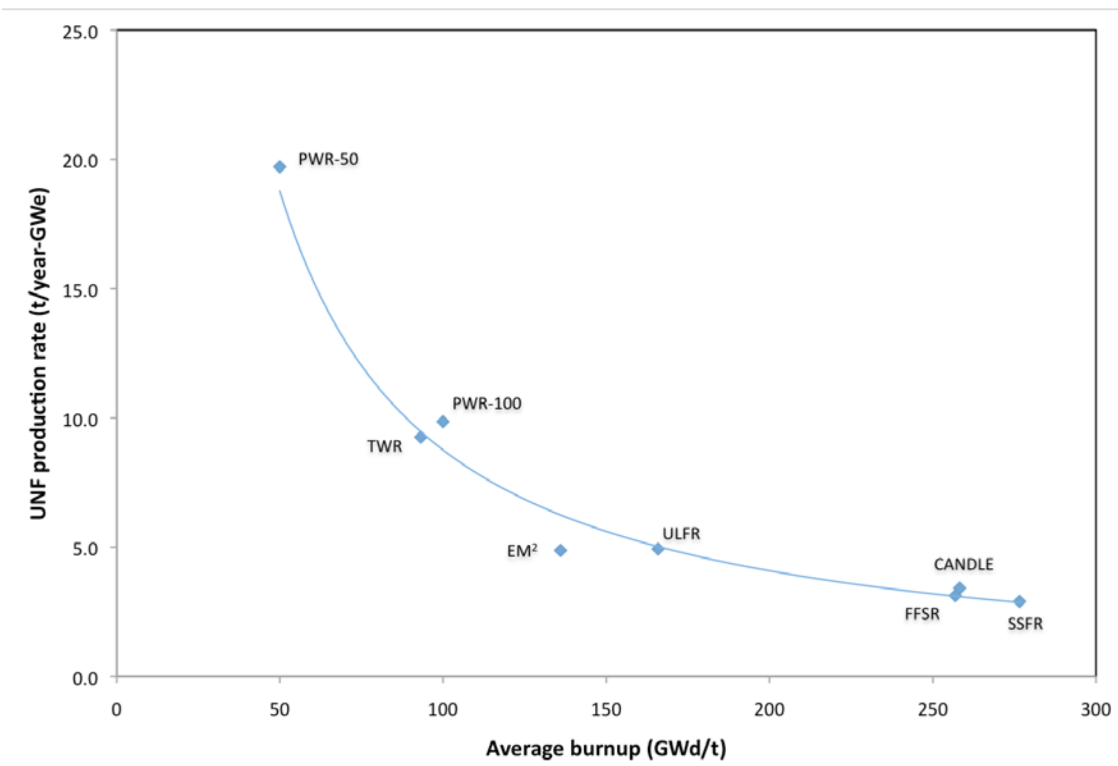

Figure 4.1. Comparison of Discharge UNF Masses.

Table 4.3. Normalized UNF Production Rates and Plutonium Isotopic Vectors (at Discharge State)

	PWR-50	PWR-100	CANDLE	SSFR	FMSR	ULFR	EM^2	TWR
Normalized UNF production rate (t/GWe-yr)								
Total used nuclear fuel	19.71	9.86	3.42	2.90	3.13	4.93	4.87	9.26
Heavy metal	18.68	8.83	2.58	2.05	2.28	4.06	4.20	8.35
Uranium	18.43	8.64	2.33	1.77	1.98	3.68	3.79	7.81
Plutonium	0.24	0.17	0.24	0.28	0.29	0.37	0.39	0.53
Minor actinides	0.02	0.03	0.01	0.01	0.01	0.01	0.01	0.01
Fission products	1.03	1.02	0.84	0.85	0.85	0.87	0.68	0.91
Plutonium isotopic vector (%)								
Pu-238 ($T_{1/2}$ = 87.7 yr)	2.5	6.8	0.6	0.6	0.6	1.0	1.1	0.9
Pu-239 ($T_{1/2} = 2.4\times10^4$ yr)	51.6	46.3	76.8	74.3	76.2	82.6	77.5	84.9
Pu-240 ($T_{1/2}$ = 6565 yr)	23.7	22.2	21.1	22.3	20.8	15.0	19.1	13.3
Pu-241 ($T_{1/2}$ = 14.35 yr)	14.9	15.6	0.7	2.1	1.9	1.1	1.9	0.9
Pu-242 ($T_{1/2} = 3.7\times10^5$ yr)	7.2	9.1	0.8	0.6	0.5	0.3	0.3	0.2

The plutonium production rates of the once-through fast spectrum systems are comparable or higher than that of the PWR-50 system even though the total UNF production rate is smaller. In particular, the used fuel Pu-239 content of the fast spectrum systems is higher than that of the LWR system due to the higher conversion of uranium (breeding ratio) in those systems. As a result, the discharged used nuclear fuels of the fast spectrum systems have higher decay heat and radiotoxicity values than those for the PWR-50 system during the time frame of 1.0×10^3 - 1.0×10^5 years.

As expected, the minor actinide (MA) production rate of the once-through fast spectrum systems is smaller than that of the PWR-50 system although the Pu production rate is higher. This is mainly due to the larger fission-to-capture ratio of the plutonium isotopes in the fast spectrum environment.

The fission product (FP) production rates per unit *thermal* energy generation of the once-through nuclear systems are comparable as long as the thermal power ratings are similar (see HM fission rate in Table 4.1). However, the FP production rate per unit *electricity* generation (t/GWe-yr) is dependent on the thermal efficiency of each once-through nuclear system. Since the thermal efficiencies of the fast spectrum systems are higher than that of the LWR system, the FP production rate is lower for fast spectrum systems. In particular, due to its high thermal efficiency, the EM^2 produces the smallest quantity of fission product for a given electricity generation rate.

4.3 Decay Heat

The UNF decay heat levels for the once-through nuclear system have been evaluated. The decay heat values at the discharge state are provided in Table 4.4, including the values of the specific power density and the normalized UNF production rate per unit electricity generation. In this table, two decay heat values are provided: the decay heat per one metric ton of the UNF (MW/t) that was obtained from the ORIGEN-2 calculations and the normalized decay heat per unit electricity generation (MW/GWe-yr).

The leading contributors to the decay heat are provided in Table 4.5 for the discharge state and at 10,000 years after discharge. At the discharge state, very short-lived nuclides such as U-239, Np-239, I-134, etc are the leading contributors. These nuclides quickly saturate during core irradiation and are proportional to the neutron flux level. They however quickly decay-out following neutron irradiation and discharge. Since the neutron flux level is proportional to the power density, the decay heat level of the UNF at the discharge state is roughly proportional to the power density of each once-through nuclear system.

Table 4.4 shows that the decay heat per unit UNF mass of the PWR-100 is comparable to that of the PWR-50 because they have the same specific power density, but its specific decay heat (W/t-UNF) is half because the UNF production rate per unit electricity generation is about half that of the PWR-50. As aforementioned, the fast spectrum systems are derated. Consequently, their decay heat per unit UNF mass is smaller than that of the PWR reference system. In addition, the smaller normalized UNF production rate of the fast spectrum systems causes the lower normalized decay heat. The UNF discharged from the CANDLE system has the smallest normalized decay heat due to its smallest power density. Generally, the normalized decay heats of the fast spectrum systems are 1- 4 MW/GWe-year, which are about 10 – 40 times smaller than that of PWR-50.

The normalized decay heat curves of the once-through nuclear systems after 10-year post irradiation cooling are plotted in Figure 4.2, and the major contributors on the normalized decay heat curves of the PWR-50 and SSFR systems are plotted in Figures 4.3 and 4.4, respectively. The trends for the other fast spectrum systems are similar to that of the SSFR.

Table 4.4. Comparison of UNF Decay Heat at Discharge

Parameter	PWR-50	PWR-100	CANDLE	SSFR	FMSR	ULFR	EM^2	TWR
Specific power density, MW/t	33.70	33.70	3.66	16.89	15.67	9.39	11.76	7.51
UNF production rate, t/GWe-yr	19.71	9.86	3.42	2.90	3.13	4.93	4.87	9.26
Decay heat per unit UNF mass, MW/t	1.99	2.00	0.24	0.76	0.74	0.63	0.68	0.43
Normalized decay heat per unit electricity generation, MW/GWe-yr	39.14	19.74	0.83	2.20	2.30	3.11	3.32	4.02

Table 4.5. Decay Heat of Leading Contributors (W/t-UNF)

PWR-50		PWR-100		CANDLE		SSFR		FMSR		ULFR		EM^2		TWR	
At Discharge															
Total	$2.0x10^6$	Total	$2.0x10^6$	Total	$0.2x10^6$	Total	$0.8x10^6$	Total	$0.7x10^6$	Total	$0.6x10^6$	Total	$0.7x10^6$	Total	$0.4x10^6$
U239	55,420	U239	50,780	U239	9,199	U239	23,000	U239	22,810	U239	21,890	U239	23,780	U239	15,800
Np239	49,700	Np239	45,540	Np239	8,261	Np239	20,650	Np239	20,480	Np239	19,650	Np239	21,350	Np239	14,190
I134	38,540	I134	38,300	I134	4,434	Tc104	14,140	Tc104	13,740	I134	11,720	I134	12,500	I134	8,239
Cs138	34,500	Cs138	34,340	Tc104	4,408	I134	14,090	I134	13,710	Tc104	11,580	Cs138	11,830	Tc104	8,041
Cs140	33,510	Cs140	33,340	Cs138	4,218	Cs138	13,540	Cs138	13,160	Cs138	11,180	Tc104	11,560	Cs138	7,836
Nb102	31,150	Nb102	31,010	Nb102	4,065	Nb102	12,910	Nb102	12,560	Nb102	10,740	Nb102	11,350	Nb102	7,462
Y 96	30,180	Y 96	29,960	Cs140	3,788	Cs140	12,020	Cs140	11,690	Cs140	10,020	Cs140	10,770	Cs140	7,047
La142	29,550	Tc104	29,300	La142	3,345	La142	10,610	La142	10,330	La142	8,852	Y 96	9,579	La142	6,241
Tc104	29,330	La142	29,020	Y 96	3,291	Y 96	10,390	Y 96	10,120	Y 96	8,730	La142	9,485	Y 96	6,156
La140	27,100	La140	27,970	La140	3,071	La140	9,831	La140	9,519	La140	7,994	La140	8,410	La140	5,472
10,000 Years After Discharge															
Total	17	Total	24	Total	114	Total	154	Total	152	Total	116	Total	128	Total	89
Pu239	9.07	Pu239	11.79	Pu239	76.62	Pu239	101.40	Pu239	103.10	Pu239	88.14	Pu239	89.21	Pu239	69.64
Pu240	7.42	Pu240	10.29	Pu240	35.99	Pu240	52.07	Pu240	48.13	Pu240	27.38	Pu240	37.61	Pu240	18.64
Am243	0.49	Am243	1.22	Am243	0.24	Am243	0.20	Am243	0.16	U234	0.16	Am243	0.20	U234	0.08
Pu242	0.10	U234	0.21	U234	0.14	Sb126m	0.17	Sb126m	0.15	Sb126m	0.10	U234	0.18	Sb126m	0.05
U234	0.09	Pu242	0.18	Sb126m	0.14	U234	0.12	U234	0.11	Am243	0.06	Np237	0.09	Np237	0.04
Np237	0.05	Np237	0.09	Pu242	0.07	Np237	0.08	Np237	0.07	Np237	0.06	Sb126m	0.08	Am243	0.02
Np239	0.04	Np239	0.09	Np237	0.06	Pu242	0.07	Pu242	0.06	Tc 99	0.03	U236	0.03	Tc 99	0.02
Sb126m	0.01	Am241	0.09	Tc 99	0.04	Tc 99	0.05	Tc 99	0.04	Pu242	0.02	Pu242	0.03	U236	0.01
Am241	0.01	Cm245	0.09	Sb126	0.03	Sb126	0.03	Sb126	0.03	Sb126	0.02	Tc 99	0.03	Sb126	0.01
Cm245	0.01	Sb126m	0.03	U236	0.02	U236	0.02	U236	0.02	U236	0.02	Po214	0.02	Pu242	0.01

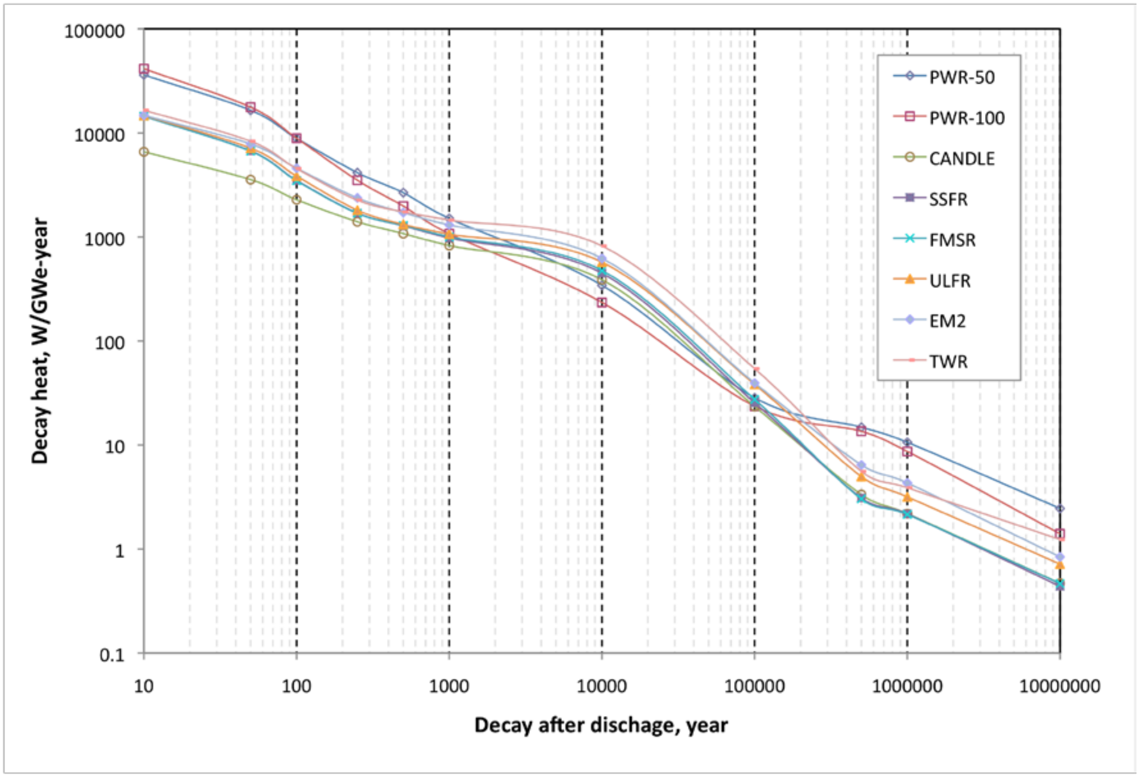

Figure 4.2. Normalized Decay Heat per Unit Electricity Generation.

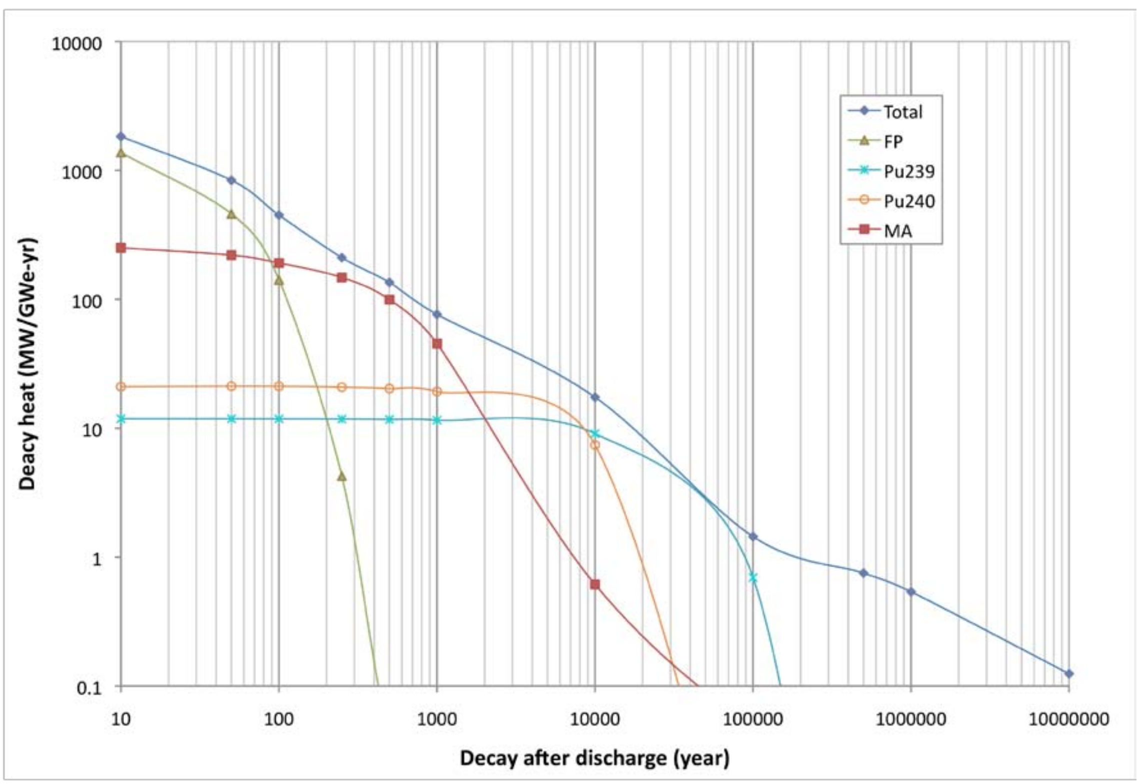

Figure 4.3. Normalized Decay Heat of PWR-50.

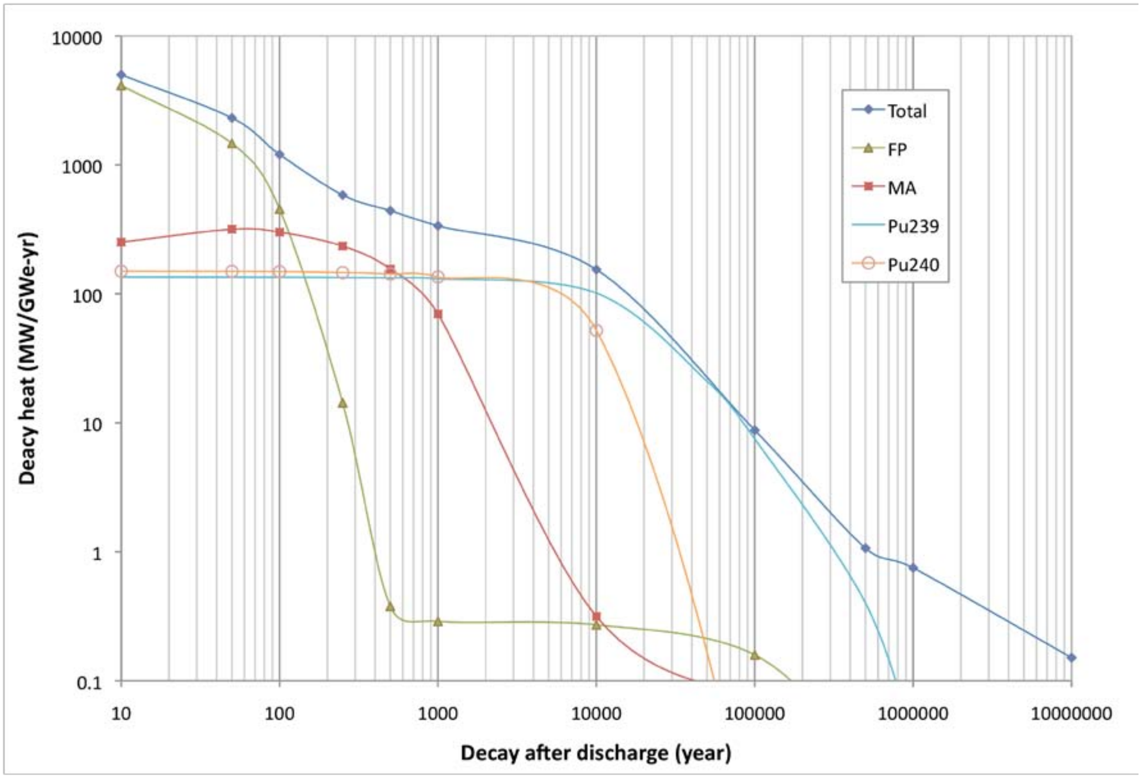

Figure 4.4. Normalized Decay Heat of SSFR.

Figure 4.2 indicates that the normalized decay heat of the PWR system is higher than that of the fast spectrum systems from discharge to 1,000 years. However, the normalized decay heat level of the fast spectrum system UNF is higher than that of the PWR system during the time frame of 1,000 – 100,000 years. This is mainly due to the higher Pu production rate in the fast spectrum systems. Table 4.5 shows that the very short-lived nuclides are leading contributors on the decay heat at the discharge state, but these nuclides decay out quickly and plutonium isotopes (in particular, Pu-239 and Pu-240) become the dominant contributors thereafter.

4.4 Radiotoxicity

A variety of measures are available for quantifying the radiotoxicity of the used nuclear fuel. In this study, the ingestion dose coefficients obtained from the ICRP 72 database [ICRP 1996] by INL were utilized. It is noted that the cancer dose measures have been used in previous fuel cycle studies performed by ANL [Kim 2005]. The two dose conversion sets were compared in this study, before evaluating the UNF radiotoxicity values of the once-through nuclear systems. The total number of nuclides in the INL set is 665 (non-zero value only), while it is 737 in the ANL set, and the dose conversion factor (Sv/Ci or Sv/kg) of each nuclide provided by INL is ~20% lower than the corresponding value of the ANL set.

The radiotoxicity of the UNF discharged from each once-through nuclear system was estimated up to 10 million years after discharge. All radiotoxicity values were normalized to unit electricity generation in a year (Sv/GWe-yr) and then normalized again to the radiotoxicity of the natural uranium ore that is needed to produce the LEU fuel for the reference once-through fuel cycle (PWR-50. The required natural uranium for the PWR-50 per unit electricity generation can be calculated using

$$M^{NU}\left(t/_{GWe\cdot yr}\right)=\frac{M^{HM}\left(t/_{batch\cdot GWe}\right)}{L\left(yr/_{batch}\right)}\times f_{NU}^{LEU},$$

where

M^{NU} = natural uranium mass to generate 1-GW electricity for one year,
M^{HM} = heavy metal mass per batch to support 1-GWe power rating reactor,
L = cycle length per batch,
f_{NU}^{LEU} = required natural uranium mass per unit LEU mass in the enrichment process.

Using the primary design parameters provided in Table 4.1, the required natural uranium for the PWR-50 is 166 t/GWe-yr.

Figure 4.5 shows the normalized radiotoxicity values of the UNF discharged from the once-through nuclear systems. The values are broken-down by major contributors in Figures 4.6, 4.7, and 4.8 for PWR-50, SSFR, and EM^2, respectively. The trends of other fast spectrum systems are similar to that of the SSFR. The radiotoxicity values of the leading contributors, which were selected by sorting the radiotoxicity values at 10, 10,000, and 1.0x10^6 years, are compared in Table 4.6. Since the trend of FMSR is similar to SSFR, the FMSR results are not provided in Table 4.6.

Generally, the trends of the normalized radiotoxicity are similar to those of the decay heat ones. At ten years after discharge, the UNF radiotoxicity values of the fast spectrum systems are about a factor of 2 – 5 lower than that of the reference LWR UNF (see the total values in Table 4.6). At this point, the fission products dominate the hazard, but the radiation hazard associated with the shorter-lived fission products quickly decreases and the contribution from the actinides becomes dominant after 100 years.

About thousand years after discharge, the UNF radiotoxicity values of the fast spectrum systems are higher than that of the PWR-50 UNF because of the contribution of the plutonium isotopes. Table 4.6 indicates that Pu-239 and Pu-240 are the dominant contributors to the radiotoxicity at ten thousand years after discharge. As indicated in Table 4.3, the plutonium production rates (in particular, Pu-239) of the fast spectrum systems are higher than that of the thermal systems although the total UNF production rates are lower.

Table 4.6 shows that the radiotoxicity of the plutonium isotopes is no longer leading one million years after discharge and it is the daughters of the uranium isotopes (such as Th-229, Po-210, etc) that are the dominant contributors. Since the UNF production rate of the LWR system is higher than that of the fast spectrum systems, the radiotoxicity of the LWR system is higher again after 200,000 years.

For the LWR systems, it takes ~200,000 years before the radiotoxicity of the UNF falls below the level of the natural uranium ore (which was estimated to be ~250,000 years with ANL dose coefficients set). Although the UNF radiotoxicity values of the fast spectrum systems is higher than that of the LWR system during the time-frame of 1,000 – 100,000 years, it takes less or comparable time for the UNF radiotoxicity of the fast spectrum systems to fall below the level of the natural uranium ore: ~120,000 years for CANDLE, SSFR, and FMSR, and ~200,000 years for ULFR, EM^2, and TWR.

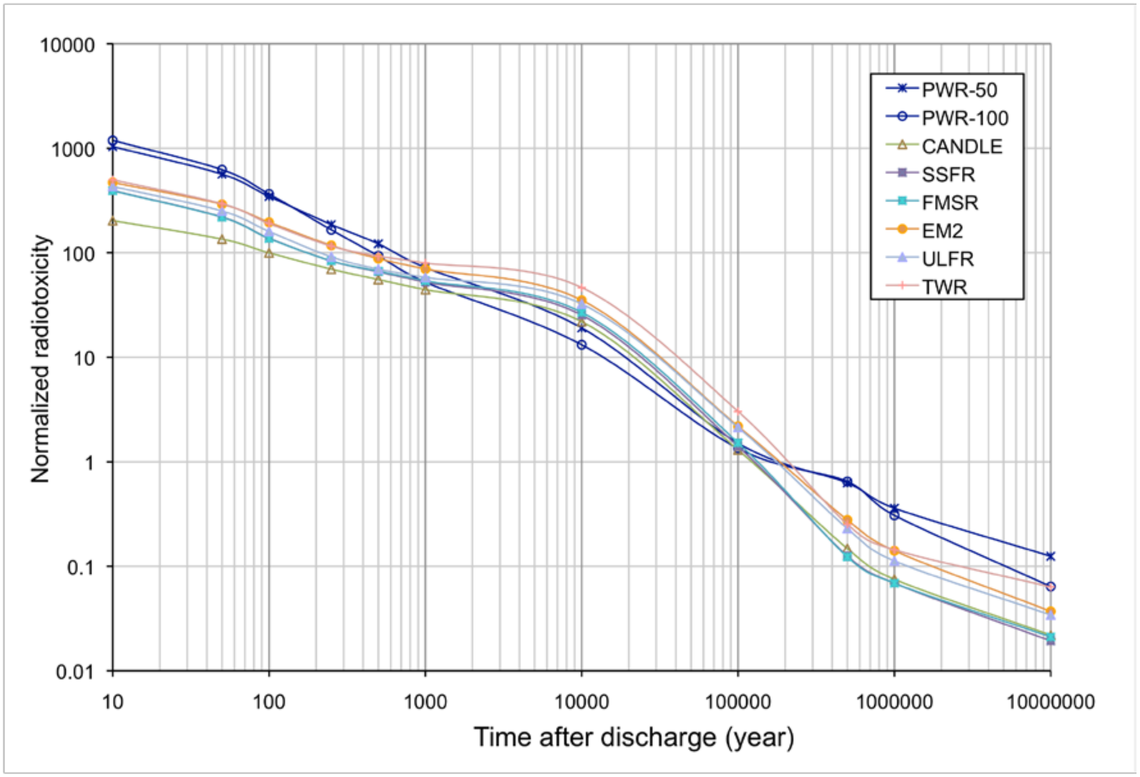

Figure 4.5. Comparison of Normalized Radiotoxicity.

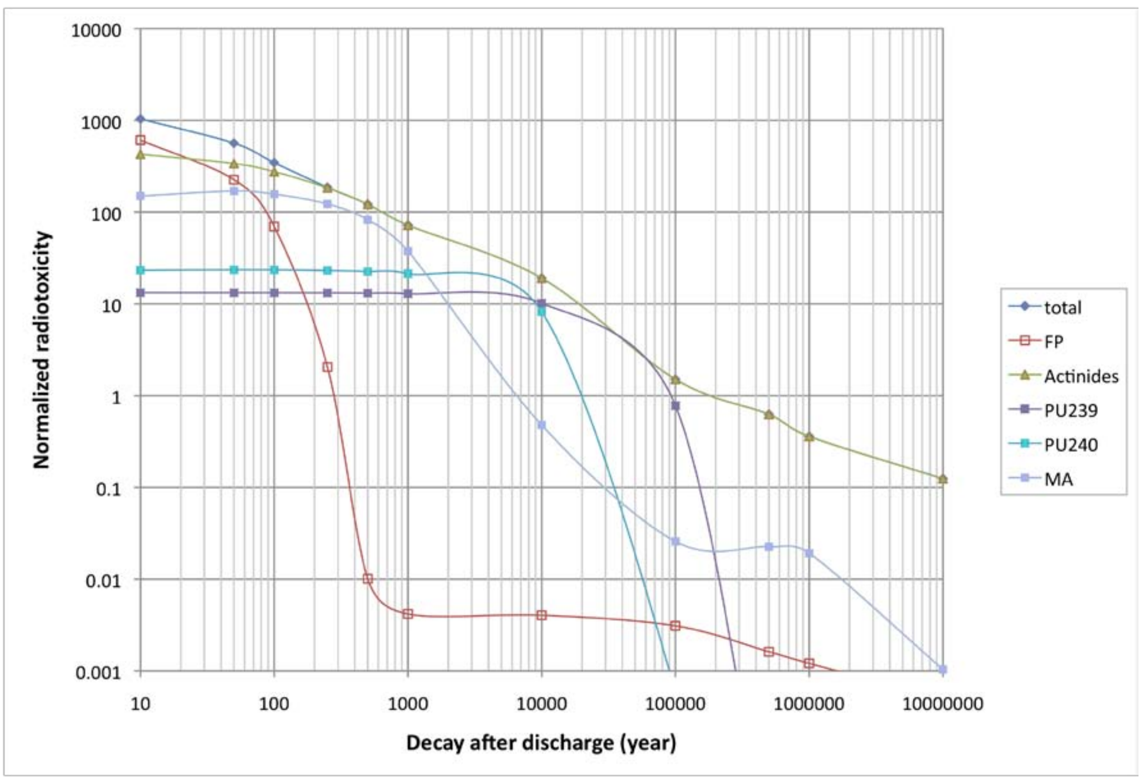

Figure 4.6. Breakdown of PWR-50 Radiotoxicity.

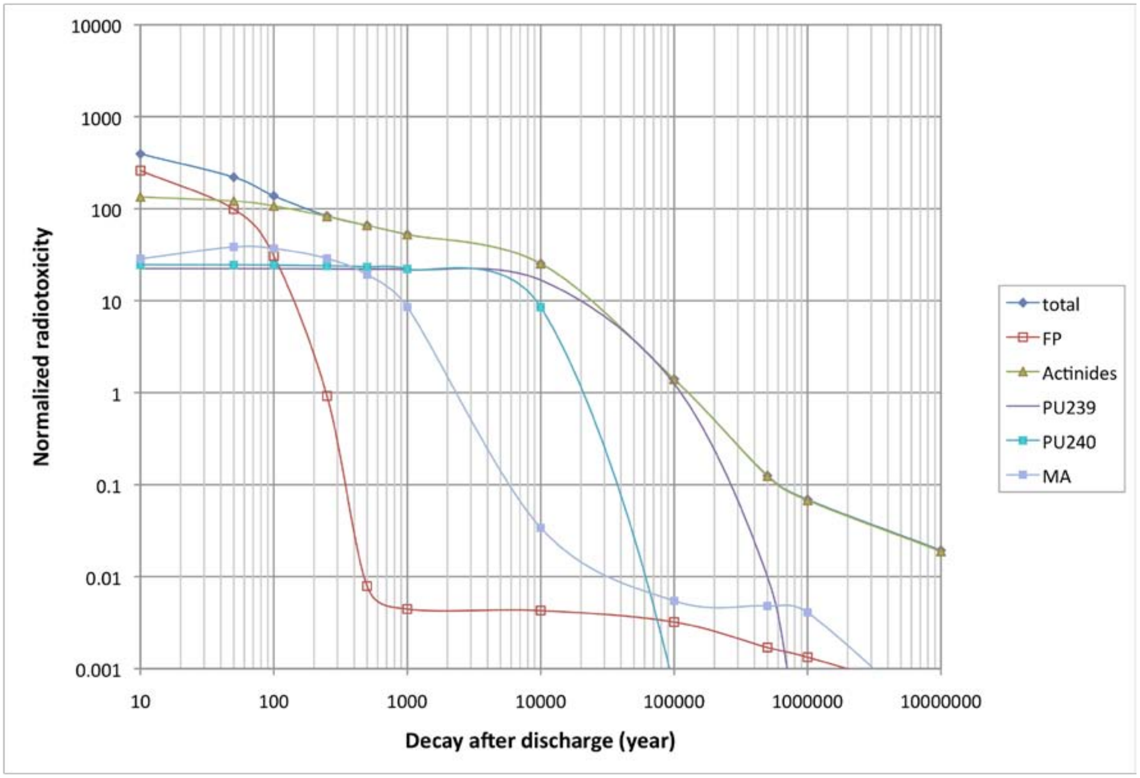

Figure 4.7. Breakdown of SSFR Radiotoxicity.

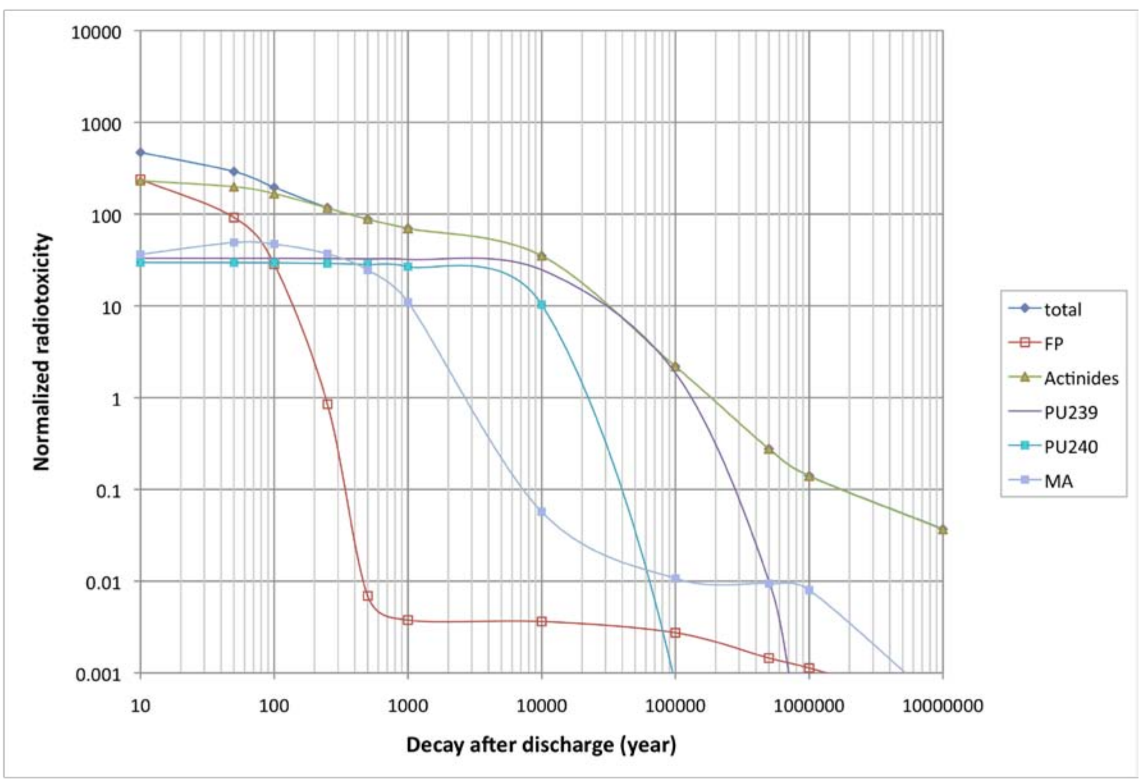

Figure 4.8. Breakdown of EM^2 Radiotoxicity.

Table 4.6. Comparison of Leading Contributors to UNF Radiotoxicity

	PWR-50		PWR-100		CANDLE		SSFR		ULFR		EM^2		TWR	
Year	Isotope	Hazard	Isotope	Hazard	Isotope	Hazard	Isotope	Hazard	Isotope	Hazard	Isotope	Hazard	Isotope	Hazard
	Sr 90	323.7	Sr 90	314.8	Cs137	47.0	Cs137	133.1	Cs137	118.8	Sr 90	119.3	Cs137	139.1
	Cs137	223.8	Pu238	305.6	Sr 90	37.5	Sr 90	108.3	Sr 90	113.2	Pu238	116.1	Sr 90	137.9
	Pu238	165.4	Cs137	212.1	Pu238	36.6	Pu238	46.5	Pu238	98.4	Cs137	104.6	Pu238	97.2
	Cm244	77.5	Cm244	156.4	Am241	28.9	Pu240	24.5	Pu239	33.0	Pu239	32.9	Pu239	48.9
	Pu241	75.5	Pu241	56.0	Pu240	20.0	Am241	23.6	Pu240	21.9	Am241	29.9	Pu240	28.0
10	Am241	70.3	Am241	54.3	Pu239	19.8	Pu239	22.3	Am241	17.9	Pu240	29.7	Am241	19.1
	Y 90	31.2	Cs134	30.7	Y 90	3.6	Pu241	12.4	Y 90	10.9	Pu241	16.2	Y 90	13.3
	Cs134	27.1	Y 90	30.4	Pu241	3.3	Y 90	10.4	Pu241	8.6	Y 90	11.5	Pu241	10.7
	Pu240	23.3	Pu240	15.6	Am242m	2.0	Cs134	5.7	Cs134	4.8	Cm244	4.6	Cs134	1.9
	Pu239	13.3	Pu239	8.5	Cs134	1.8	Cm244	3.1	Cm244	1.3	Cs134	3.1	Am242m	1.1
	Total	1037.3	Total	1190.6	Total	203.2	Total	393.6	Total	431.6	Total	470.9	Total	499.1
	Pu239	10.2	Pu239	6.6	Pu239	14.9	Pu239	16.7	Pu239	24.7	Pu239	24.7	Pu239	36.7
10,000	Pu240	8.2	Pu240	5.7	Pu240	6.9	Pu240	8.5	Pu240	7.6	Pu240	10.3	Pu240	9.7
	Am243	0.4	Am243	0.5	Am243	0.04	Am243	0.03	Po210	0.02	Am243	0.04	Po210	0.01
	Total	19.1	Total	13.2	Total	21.9	Total	25.3	Total	32.4	Total	35.2	Total	46.5
	Th229	0.1	Th229	0.1	Th229	0.02	Th229	0.02	Po210	0.03	Po210	0.04	Po210	0.04
1,000,000	Po210	0.1	Po210	0.1	Po210	0.02	Po210	0.01	Th229	0.03	Th229	0.03	Th229	0.03
	Total	0.4	Total	0.3	Total	0.1	Total	0.1	Total	0.1	Total	0.1	Total	0.1

4.5 Neutron and Photon Sources

As a measure of UNF handling difficulty, the neutron and photon source rates have been evaluated. The combined neutron sources from (α,n) and spontaneous neutrons per unit UNF mass for 50 years from core discharge are compared in Table 4.7 and the leading contributors of each once-through nuclear system at the discharge are provided in Table 4.8. In order to account for the (α,n) reaction in the light nuclides, the structural materials (cladding, duct, etc) were included in the ORIGEN-2 calculations.

Table 4.7. Neutron Sources per Unit UNF Mass (neutrons/sec/t-UNF)

System	0.01 year	1 years	2 years	10 years	20 years	50 years
PWR-50	1.63E+09	1.08E+09	9.31E+08	6.70E+08	4.64E+08	1.61E+08
PWR-100	5.81E+09	4.51E+09	4.10E+09	2.94E+09	2.03E+09	7.07E+08
CANDLE	2.32E+08	1.62E+08	1.44E+08	1.14E+08	9.00E+07	5.45E+07
SSFR	7.17E+08	3.85E+08	3.07E+08	2.26E+08	1.71E+08	8.74E+07
FMSR	5.87E+08	3.07E+08	2.42E+08	1.79E+08	1.36E+08	7.32E+07
ULFR	2.77E+08	1.35E+08	1.02E+08	7.81E+07	6.33E+07	4.02E+07
EM^2	5.88E+08	3.30E+08	2.69E+08	1.99E+08	1.49E+08	7.48E+07
TWR	1.47E+08	6.21E+07	4.34E+07	3.36E+07	2.91E+07	2.16E+07

Table 4.8. Leading Contributors to Neutron Source at Discharge (neutrons/sec/t-UNF)

PWR-50		PWR-100		CANDLE		SSFR	
Total	1.64×10^9	Total	5.47×10^9	Total	2.32×10^8	Total	7.17×10^8
Cm244	57.9%	Cm244	69.9%	Cm244	45.9%	Cm242	57.6%
Cm242	40.9%	Cm242	25.8%	Cm242	37.7%	Cm244	36.1%
Cm246	0.4%	Cf252	2.5%	Pu240	6.8%	Pu240	3.2%
Pu238	0.3%	Cm246	1.1%	Pu238	3.4%	Pu238	1.6%
Pu240	0.2%	Pu238	0.4%	Cm246	2.4%	Cm246	0.5%
FMSR		ULFR		EM^2		TWR	
Total	5.87×10^8	Total	2.77×10^8	Total	5.88×10^8	Total	1.47×10^8
Cm242	59.5%	Cm242	64.7%	Cm242	54.2%	Cm242	73.4%
Cm244	33.5%	Cm244	23.7%	Cm244	38.8%	Cm244	13.6%
Pu240	3.6%	Pu238	5.3%	Pu238	3.0%	Pu240	5.6%
Pu238	1.8%	Pu240	4.3%	Pu240	2.8%	Pu238	5.2%
Pu239	0.6%	Pu239	1.0%	Pu239	0.5%	Pu239	1.5%

The spontaneous neutrons are the dominant contributors to the neutron sources (about 90% for all systems). Since the contribution of curium isotopes is predominant (> 90%), the neutron source level at the discharge stage is dependent on the minor actinide mass. As indicated in Table 4.3, the once-through fast spectrum system has a lower minor actinide production rate than the LWR system. As a result, the neutron sources of the once-through fast spectrum systems are smaller than those of the LWR system.

The photon sources are compared in Figure 4.9 and the photon energy spectra are provided in Figure 4.10. The leading contributors to the PWR-50 and SSFR systems at discharge and 10 years after discharge are provided in Table 4.9. The photon source is displayed as the total photon energy per unit UNF mass (W/t-UNF, which can be changed to MeV/sec/t-UNF). The calculated mean photon energy is ~0.3 MeV regardless of the system types.

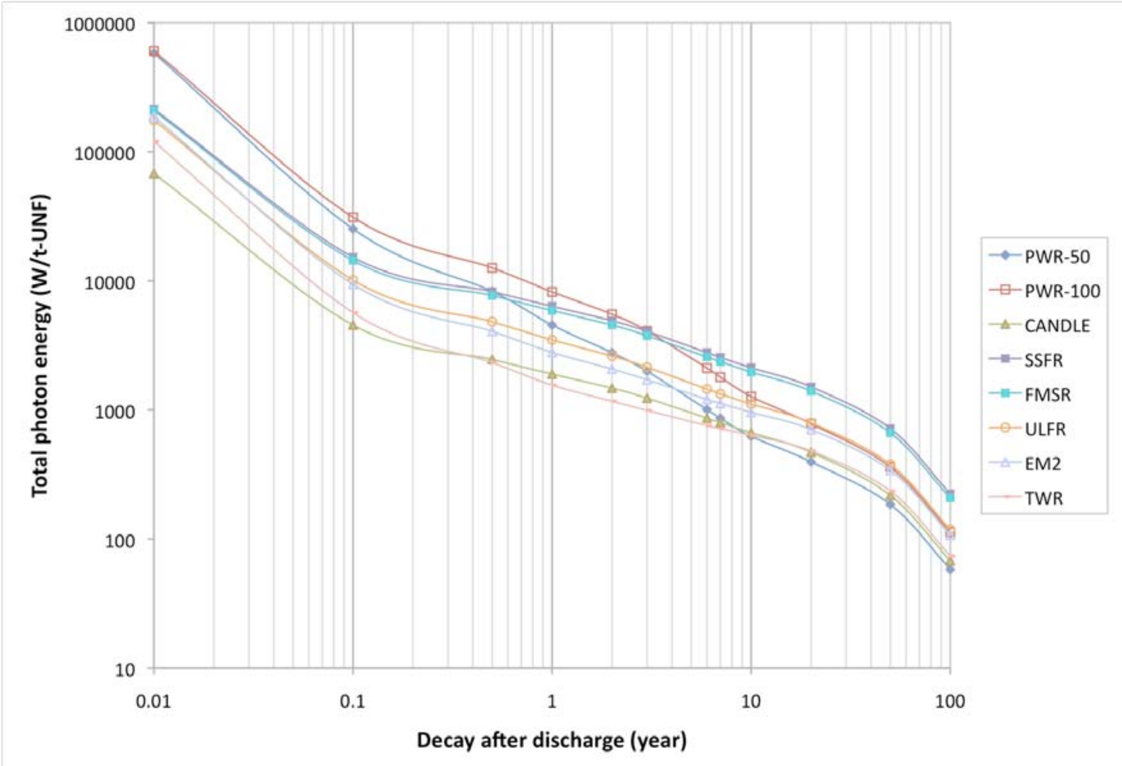

Figure 4.9. Comparison of Photon Source per Unit UNF Mass.

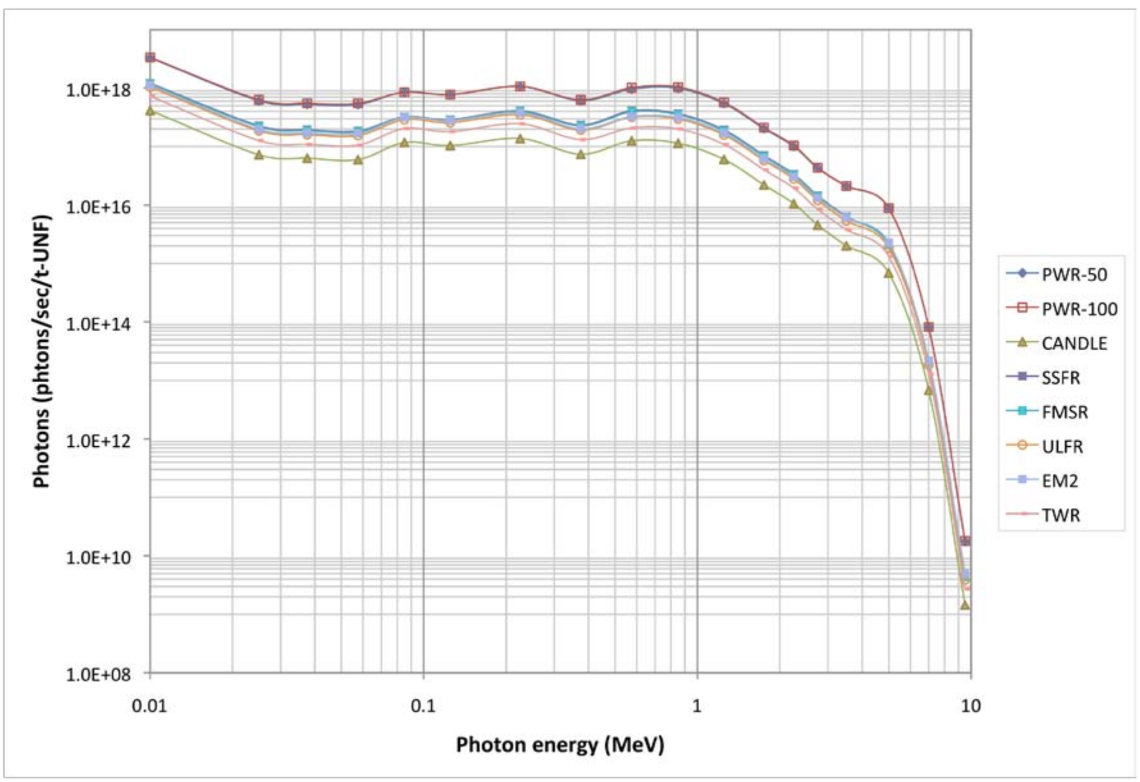

Figure 4.10. Comparison of Photon Spectra.

Table 4.9. Leading Contributors on Photon Source

PWR-50				SSFR			
Discharge		10 year		Discharge		10 Years	
Total	[a] 544,568	Total	620	Total	[a] 214,032	Total	2,121
I134	6.9 %	Ba137m	67.3 %	I134	6.4 %	Ba137m	80.0 %
Cs138	5.1 %	Cs134	15.4 %	Cs138	5.1 %	Eu154	8.2 %
La140	4.9 %	Eu154	9.0 %	Np239	4.9 %	Cs134	6.5 %
La142	4.9 %	Y 90	6.0 %	Tc104	4.8 %	Y 90	3.5 %
Np239	4.6 %			La140	4.5 %		
Tc104	3.9 %			La142	4.4 %		
I132	3.8 %			I132	4.1 %		
I135	3.2 %			Mo101	3.4 %		
Mo101	2.8 %			I135	3.1 %		
I136	2.3 %			Cs134	2.2 %		

a) Unit is Watt per unit ton of UNF mass.

At the discharge state, the short-lived fission products are predominant contributors and hence the photon source is proportional to the neutron flux level (or power density) during irradiation in the core. As a result, the PWR systems have the highest photon source and the CANDLE has the lowest photon source per unit UNF mass. At 10 years after discharge, Ba-137^{m} ($T_{1/2}$ = 2.6 min) is the predominant nuclide, which is the daughter of Cs-137 ($T_{1/2}$ = 30.1 year). The ORIGEN-2 results indicate that the SSFR produces more Cs-137 (factor of 4) than the PWR-50. As a result, the photon sources of the SSFR are higher at 10 years.

4.6 Uranium Utilization

The *natural uranium* (NU) utilization is defined as ratio of the heavy-metal mass burned by fission to the total mass of the natural uranium used in making the LEU fuel,

$$U(\%) = \frac{\Delta M^{HM}(t)}{M^{NU}(t)} \times 100,$$

where ΔM^{HM} = heavy metal mass burned by fission,
M^{NU} = natural uranium mass to obtain LEU mass.

This definition is applicable to the one-batch fast spectrum systems or the LWR systems because both burnt heavy metal and required natural uranium masses are clearly determined. However, it is difficult to apply the definition to the multi-batch fast spectrum systems because they do not need enriched uranium except for the initial core. Thus, the uranium utilization of the multi-batch fast spectrum system is defined as the ratio of the burned heavy metal mass to the provided depleted uranium mass, which is equivalent to the average burnup at the equilibrium cycle.

Table 4.10 provides the uranium utilization evaluated for the nuclear systems. For the PWR-50 system, the required natural uranium mass per unit electricity generation is 166 t/GWe-yr, and the fission rate is 1.03 t/GWe-yr. As a result, the uranium utilization of the PWR-50 is 0.6%, which is less than 1%. A pertinent question is whether high fuel burnup can be used to increase uranium utilization in LWRs. Because of the higher enrichment required to extend the burnup by a factor of two for the PWR-100 system, the uranium utilization is comparable to that of the PWR-50 system, which again is less than 1%. Thus increasing the burnup to the 100 GWd/t does not help in increasing the uranium utilization.

Table 4.10. Uranium Utilization of Once-Through Nuclear Systems

	PWR-50	PWR-100	CANDLE	SSFR	FMSR	ULFR	EM^2	TWR
Uranium enrichment, %	4.21	8.5	1.2	0.25	0.25	4.1	6.1	2.5
Tail uranium enrichment, %	0.25	0.25	0.25	0.25	0.25	0.25	0.35	0.30
HM charge per batch, t/batch	29.7	29.7	823.7	5.22	5.6	319.6	42.5	399.2
Cycle length, yr/batch	1.5	3.0						
Required NU, t/GWe-yr	166.1	173.0	7.10	0.0	0.0	40.2	75.9	48.3
HM fission, t/GWe-yr	1.03	1.04	1.03	1.02	1.02	1.04	0.16	1.03
Burnup, %	5.2	10.2	24.6	29.4	27.2	17.6	13.9	9.8
Uranium utilization, %	0.6	0.6	[a] 12.1	[b] < 29.4	[b] < 27.2	2.2	0.9	1.9

a) This value was obtained for the active core height of 8.0 m, but it could be increased to 24.7% when the core is designed using all depleted uranium resulting from making LEU fuel for the starter zone.

b) Value would be lower if the transition cycle(s) are included.

For the CANDLE system, the uranium utilization is dependent on the reactor operation time. As shown in Table 3.7, the reactor operation time is dependent on the active core height and the uranium utilization could be 24.5% when the active core height is consistent with using all depleted uranium that is generated from making LEU of the start zone. For an active core height of 8.0 m, the uranium utilization is 12.5 %. For other one-batch fast spectrum system (ULFR, EM^2, and TWR), the uranium utilization is less than 2.2%, which is much smaller than that of the CANDLE reactor because the reactor cannot maintain criticality for a long time like the CANDLE system and because not enough depleted uranium is burned in the designs. *The EM^2 and TWR designers have indicated that the spent fuel elements from these designs could be used to start-up subsequent reactors. If such designs are possible (i.e., with no fuel performance and reactivity constraints), then theoretically higher utilization can be obtained*.

Since the SSFR and FMSR systems are sustainable by feeding depleted uranium, the enriched uranium is not needed after the first cycle. At the equilibrium cycle, the uranium utilization is equivalent to the average burnup of 27 – 29%. However, it is emphasized here that it takes a long time to achieve such a high uranium utilization because the system slowly converges to the equilibrium cycle with the 34-batch fuel management scheme.

5. CONCLUSIONS

The fuel cycle performance parameters of once-through nuclear systems have been evaluated in this study. The considered once-through nuclear systems are PWRs with medium (50 GWd/t) and high (100 GWd/r) burnups, the CANDLE reactor of the Tokyo Institute of Technology, the sustainable sodium-cooled fast reactor (SSFR) by ANL, the fast mixed spectrum reactor (FMSR) by BNL, the ultra-long life fast reactor (ULFR) by ANL, the General Atomics Energy Multiplier Module (EM^2), and the traveling wave reactor (TWR) of TerraPower. Besides the PWRs, the other once-through nuclear systems are sodium-cooled or gas-cooled fast spectrum systems that have been proposed for achieving extremely long fuel residence time and high uranium utilization. To meet the intended goals, the fast spectrum systems have adopted the breed and burn concept with propagating burn zone. The CANDLE, ULFR, EM^2 and TWR use a one-batch fuel management scheme. In contrast, the SSFR (or FMSR) uses a multi-batch (more than 30 batches) fuel management scheme to make the core sustainable by feeding depleted uranium.

The fuel cycle performance parameters of the once-through nuclear systems have been compared to those of the medium burnup PWR that has been considered as the reference system in this study. Compared to this reference PWR, the once-through fast spectrum systems require enriched uranium fuel only for the initial cycle in order to start the reactor. The power density of each once-through fast reactor systems is significantly derated to extend the fuel residence time. For consistent comparison, the fuel cycle performance parameters such as the used nuclear fuel (UNF) mass, decay heat, and radiotoxicity are normalized to the unit electricity generation in one year (i.e., per GWe-yr). However, the fuel handling measures such as neutron and photon sources are normalized per unit mass of UNF.

Compared to the reference PWR system, the power densities of the fast nuclear systems are significantly derated, which results in higher heavy metal inventories. However, the high fuel burnup with a long fuel residence time in these systems results in lower used nuclear fuel (UNF) production rate. The reference PWR system produces UNF of ~20 metric ton per GWe-year, while the once-through fast spectrum systems produce 3 – 9 ton per GWe-year depending on the burnup. The plutonium production rates of the once-through fast spectrum systems are however higher than that of the reference PWR system. In particular, the fast spectrum systems produce more Pu-239 due to their higher breeding ratio.

Compared to the reference PWR system, the decay heat levels of the once-through fast-spectrum-system UNF are lower due to their lower UNF production rate and lower power density. At the discharge state, the decay heat levels of the fast spectrum systems are 1- 4 MW/GWe-year, which is about 10 – 40 times lower than that of the reference PWR. However, the decay heat level becomes higher than that of the reference PWR during the time frame of 1,000 – 100,000 years mainly due to the higher Pu (in particular, Pu-239 and Pu-240) production rate in the fast spectrum systems.

The radiotoxicity of the UNF was calculated using the ingestion dose coefficient specified by ICRP 72. Generally, the overall trend of the UNF radiotoxicity is similar to that of the decay heat. At ten years after discharge, the radiotoxicity of the fast spectrum systems is about a factor of 2 – 5 lower than that of the reference PWR system. At this point, the fission products dominate the hazard, but the hazard associated with the shorter-lived fission products decreases quickly. Subsequently, the contribution from the actinides becomes dominant after 100 years. About thousand years after discharge, the UNF radiotoxicity of the fast spectrum systems becomes higher than that of the PWR system because of the contribution of the plutonium isotopes. It takes ~200,000 years for the PWR UNF radiotoxicity to become lower than that of the natural uranium material used in making the enriched uranium fuel for the system. On the other hand, it takes less or comparable time for the UNF of the once-through fast–spectrum-system: ~120,000 years for CANDLE, SSFR, and FMSR, and ~200,000 years for ULFR, EM^2, and TWR.

As a measure of the UNF handling difficulty, the neutron and photon sources were evaluated. Since the contribution of curium isotopes is predominant (> 90%), the neutron source level at the discharge state is dependent on the minor actinide mass. The fast spectrum systems have lower minor actinide production rates compared to the PWR system. As a result, the neutron source levels of the fast spectrum system UNF are about a factor of 2 – 8 smaller than that of the reference PWR at the discharge state. Similarly, the once-through fast spectrum systems produce fewer photon sources at the discharge state (factor of 3 – 9 lower). However, the high Cs-137 production rate in the UNF of the fast spectrum systems results in a higher photon source level after 10 years.

For the PWR systems, the natural uranium utilization is less than 1% regardless of the burnup. For the once-through fast spectrum systems, the natural uranium (or depleted uranium) utilization could be increased to ~30%, depending on the core design choices. However, some technical design issues should be resolved before these core concepts are considered practical: ultra high burnup fuel is required, extremely long reactor lifetime is necessary for effective breeding and burning, the system core size is large, and there are potential difficulties with power and reactivity control, etc.

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