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Heterogeneous Recycling in Fast Reactors

Fuel Cycle

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Heterogeneous recycling in fast reactors

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Abstract

Due to perceived proliferation risks, current US fast reactor designs have avoided the use of uranium blankets. While reducing the amount of plutonium produced, this omission also restrains the reactor design space and has several disadvantages over blanketed cores. This study investigated many blanket options that would satisfy the proliferation concern while minimizing negative fuel cycle impact. To do so, a multi-variable metric was developed that combines 6 attributes: proliferation resistance, fuel fabrication, radiotoxicity, breeding gain, reactivity penalty and transportation. The final version of the metric consisted of using a yes or no decision on the proliferation criteria proposed by Bathke (for technologically advanced nations). The remaining 5 attributes are scaled between 0 and 1 with assigned weights for each. For our analysis, a 2400MWth sodium cooled core was considered. One row of blanket was added radially. Metal fuel composed of depleted uranium, zirconium and Np/Pu from light water reactor used fuel was used for the driver.

It was determined that to meet the prescribed proliferation resistance criteria, a minimum of 4% MA (by volume) was needed in the blanket assemblies. However, increasing the amount of MA past 4% became detrimental to the combination of the other 5 attributes, mainly impacting the radiotoxicity, fuel fabrication and transportation. The addition of moderation by itself did not provide any means of dissipating proliferation issues. In the cases studied, it was determined that $ZrH_{1.6}$ and BeO were the most promising moderating materials. They both provided some reduction in required MA concentration but at the expense of the radiotoxicity of the end product. Using our defined metric, it was determined that moderation provided no immediate benefit. It should also be noted that the homogeneous or heterogeneous addition of moderators has minimal impact on such scoping studies.

Separation of the Cm/Bk/Cf vector from the Am was also studied. The blankets were composed of Am while the remaining Cm/Bk/Cf was left to decay in storage. The metric was then applied to the combined streams for all attributes except proliferation. The separated case performed worst in all cases examined. Also, as expected, varying the uranium composition vector from natural (NU), depleted (DU) and recycled (RU) had very little impact on our metric, thus the choice of uranium vector would be mostly left to cost and initial fabrication considerations. It should however be noted that the k-infinity at beginning-of-life was obviously higher for the recycled and natural cases. Looking at the reactivity over the first cycle indicates that NU provides an additional ~40pcm over DU while RU provides ~60pcm, which could provide 30 and 45 extra days of operation, respectively, or a reduction in driver core enrichment for a given cycle length.

A post-processing GUI was also developed. The GUI provides easy evaluation of many of the parameters needed for the metrics presented. It also allows processing of 1 and 2 stream isotopic information in the VISION input format. This post-processing tool will facilitate future fast reactor analysis performed using the ERANOS software.

Table of Contents

1 Introduction 8
1.1 Objectives 9
2 Reference Core 11
2.1 ERANOS and MCNP Comparison 16
2.2 Modeling assumptions in this study 19
2.2.1 Geometry study of the blanket 21
3 Figure-of-Merit 27
3.1 Proliferation metric 27
3.1.1 Charlton 28
3.1.2 Bathke 32
3.2 Fuel Fabrication 34
3.3 Radiotoxicity 36
3.4 Breeding Gain 38
3.5 Reactivity Penalty 39
3.6 Transportation 40
3.7 Global metrics 41
3.7.1 Figure of Merit 1 (FOM_1) 41
3.7.2 Figure of Merit 2 (FOM_2) 42
3.7.3 Figure of Merit 3 (FOM_3) 42
3.7.4 Figure of Merit 4 (FOM_4) 43
3.7.5 Figure of Merit 5 (FOM_5) 43
3.8 Summary 43
4 Blanket Studies 45
4.1 Minor Actinide content 45
4.2 Moderation study 49
4.2.1 MCNP Verification 57
4.3 Separation study 58

4.4 Natural, Depleted or Recycled Uranium study 60
4.5 Near equilibrium cases 62
4.6 Summary 63
5 Sensitivity Study of FOM 64
5.1 FOM 1 Sensitivity Analysis 64
5.2 FOM_2/FOM_5 Sensitivity Analysis 65
5.3 Summary 68
6 Conclusion 70
7 References 72
8 Appendix I: Python GUI 75
9 Appendix II: Fuel Fabrication Metric 78
10 Appendix III: VISION Post-Processing Requirements 80
11 Appendix IV: Material Properties 83

List of Tables

Table 2-1: Main Characteristics of the Core 11
Table 2-2: Fuel Description 12
Table 2-3: Fuel Isotopic Composition 13
Table 2-4: Axial Layout of the Core 13
Table 2-5: Radial Reflector 14
Table 2-6 Radial Shield 14
Table 2-7: Assembly Description 14
Table 2-8:Control Rod Sizes 14
Table 2-9: Fuel Composition of Modified Core 15
Table 2-10: Fuel Isotopic Composition of Modified Core 15
Table 2-11: Driver Fuel Minor Actinide Vector 20
Table 2-12: Blanket Minor Actinide Vector 21
Table 2-13: Pin Size Study 22
Table 2-14: Pin Study Results (R6-R10) 22
Table 2-15: Pin Study Results (R11-R15) 23
Table 2-16: Blanket Composition 23
Table 2-17: Main Results for 4 Blanket Options 25
Table 2-18: Ratio of Selected Isotopes 26
Table 3-1: Utility Function (u_1) for DOE Attractiveness Level (x_1) 28
Table 3-2: Significant Quantity (SQ) Definition 31
Table 3-3: Weights of FOM_1 32
Table 3-4: Bathke's Attractiveness Criterion 34
Table 3-5: Weights for FOM_3 43
Table 3-6: Summary table 44
Table 4-1: FOM_1 for different MA contents (Note: based on Charlton's normalized weights) ... 46
Table 4-2: Utility function values $u_1 - u_6$ 46
Table 4-3: FOM_2 for different MA contents 46
Table 4-4: x_7 -x_{12} values 47
Table 4-5: FOM_3 for different MA contents 47
Table 4-6: FOM_4 for different MA contents 48
Table 4-7: FOM_5 for different MA contents 48
Table 4-8: Results for Selected Materials – 4%MA, 10% Moderator 51
Table 4-9: Results for Selected Materials – 4%MA, 20% Moderator 51
Table 4-10: Results for Selected Materials – 4%MA, 30% Moderator 51
Table 4-11: FOM_1 for different Moderator content (4%MA) 52
Table 4-12: $u(x_1)$ - $u(x_6)$ values 53
Table 4-13: FOM_2 for different Moderator content (4%MA) 53
Table 4-14: x_7 -x_{12} values 54

Table 4-15: FOM_3 for different Moderator content (4%MA) 54
Table 4-16: FOM_4 for different Moderator content (4%MA) 55
Table 4-17: FOM_5 for different Moderator content (4%MA) 55
Table 4-18: FOM_4 – Minor Actinide and $ZrH_{1.6}$ variations 56
Table 4-19: FOM_5– Minor Actinide and $ZrH_{1.6}$ variations 56
Table 4-20: Ratio of Selected Isotopes 57
Table 4-21: FOM_2 for Am only case (9 cycles) 58
Table 4-22: FOM_2 for All Am case (9 cycles) 58
Table 4-23: x_7 -x_{12} values for AM only cases 59
Table 4-24: x_7 -x_{12} values for All MA cases 59
Table 4-25: FOM_4 for Am only case (9 cycles) 59
Table 4-26: FOM_4 for All Am case (9 cycles) 59
Table 4-27: FOM_5 for Am only case (9 cycles) 60
Table 4-28: FOM_5 for All Am case (9 cycles) 60
Table 4-29: Uranium vectors 60
Table 4-30: Results for NU, DU and RU – 4%MA 60
Table 4-31: Results for NU, DU and RU (MA – only) – 4% MA 61
Table 4-32: FOM_2 for DU, NU and RU (1 cycle) 61
Table 4-33: Reactivity (pcm) as a function of time 61
Table 4-34: FOM_4 for DU, NU and RU (9 cycles) 62
Table 4-35: Blanket analysis for 9, 18 and 27 cycles 62
Table 5-1: Weights of FOM_1 64
Table 5-2: FOM_1 for different MA contents (Note: based on Charlton's normalized weights) ... 64
Table 5-3: Weights of FOM_1 with doubled u_4 65
Table 5-4: FOM_1 for different MA contents (u_4 weight doubled) 65
Table 5-5: Weights for FOM_2 66
Table 5-6: FOM_2 for different MA contents 66
Table 5-7: Weights for FOM_2 (Increased weight for w_{11}) 66
Table 5-8: FOM_2 for different MA contents (increase in weight of u_{11}) 66
Table 5-9: FOM_2 for different Moderator content (4%MA) 67
Table 5-10: Weights for FOM_2 (Increased weight for w_{10}) 67
Table 5-11: FOM_2 for different Moderator content (4%MA) (increase in weight of u_{10}) 67

List of Figures

Figure 2-1: Radial Core Layout (Grey = Driver Fuel, Purple = Reflector, Blue = Shield) 12
Figure 2-2: Eigenvalue comparison between ERANOS and MCNP 17
Figure 2-3: Radial (top) and Axial (bottom) core flux profiles at BOC 18
Figure 2-4: Version 1 layout 24
Figure 2-5: Version 2 layout 25
Figure 3-1: Heating Rate from Pu Attribute 29
Figure 3-2: Weight Fraction of Even Pu Isotopes Attribute 30
Figure 3-3: Fuel Fabrication (u_7) as a function of neutron emission rate (x_7) 36
Figure 3-4: Utility function u_8 - Radiotoxicity by inhalation 37
Figure 3-5: Utility function u_9 - Radiotoxicity by ingestion 38
Figure 3-6: Breeding Gain (u_{10}) 39
Figure 3-7: Reactivity Penalty (u_{11}) 40
Figure 3-8: Transportation (u_{12}) 41
Figure 4-1: Pu-239 fission cross-section (red) and radiative capture (green) 49
Figure 8-1: GUI Initial Interface 76
Figure 8-2: Processed data in GUI 77

1 Introduction

Recent sodium fast reactor designs have avoided the use of depleted uranium blankets over concerns of creating weapons grade plutonium. Such an omission may reduce proliferation risks but it also restrains the reactor design space considerably. This proposal will analyze various blanket and transmutation target configurations that could broaden the design space while still addressing the non-proliferation issues. The proposed work will clarify the benefits of using a depleted uranium blanket for the sodium fast reactor and will evaluate the use of this blanket for the transmutation of minor actinides. The blanket designs will be assessed based on the transmutation efficiency of key isotopes and also on mitigation of associated proliferation risks.

This work will attempt to define valuable proliferation resistance and transmutation efficiency metrics that will facilitate cross-comparisons between designs. This study will also evaluate the core performance of the sodium fast reactor under different scenarios in which depleted uranium blankets are modified to include minor actinides with or without moderators (e.g. BeO, MgO, B_4C, hydrides, …). This will be done in an effort to increase the sustainability of the reactor and increase its power density while still offering a proliferation resistant design with the capability of burning minor actinide waste produced from light water reactors (LWRs). The possibility and issues of using recycled (as opposed to depleted) uranium in the blankets will also be analyzed. The various designs will compare MA transmutation efficiency, plutonium breeding characteristics, proliferation risk, shutdown margins and reactivity coefficients with a current reference sodium fast reactor design employing homogeneous recycling. The work will also evaluate the out-of-core accumulation and/or burn-down rates of MAs and plutonium isotopes on a cycle-by-cycle basis. This cycle-by-cycle information will be produced in a format readily usable by the fuel cycle systems analysis code, VISION, for assessment of the sustainability of the deployment scenarios.

In order to compensate for the absence of a fertile blanket, current breeder designs emphasize low coolant volume fraction and large core diameter in an effort to reduce the core leakage, enhancing neutron economy and thus increasing the conversion ratio. Such a design reduces the core coolant, which penalizes heat removal capability and decreases the achievable specific power (MWth/kgHM). Decreasing specific power to respect thermal-hydraulic considerations negatively impacts fuel cycle costs. Additionally, larger cores, which are common for break-even breeding, can have a very unstable radial power profile, meaning that small perturbations can cause large local changes. Thus the consequences of not having a blanket have a significant impact on safety, sustainability and economics of the design that this proposal will try to remedy by adding an acceptable blanket.

Additionally, current analysis methodologies have always focused on a single optimization parameter such as transmutation of minor actinides, proliferation …. This study proposes a

combine approach that deals with many issues simultaneously. The addition of blankets must be weighted by all its advantages and disadvantages to provide a proper assessment.

1.1 Objectives

The study will look into the transmutation efficiency and proliferation resistance of the various heterogeneous recycling strategies listed below. In all the proposed recycling studies, the mass flow of minor actinides will be provided from LWR spent fuel at an exposure level of 51 GWd/MTHM.

1) Heterogeneous moderating material addition will be studied by replacing some blanket fuel pins by moderator. This softens the spectrum, enhancing the breeding gain, but diminishing fuel content. It also has an impact on the transmutation of certain isotopes by increasing capture in fissile isotopes and also influences the proliferation issues by denaturing the plutonium vector. Candidates for moderating materials include zirconium hydride and beryllium oxide. Heterogeneous moderated designs sometimes lead to peaking issues that will be assessed in the simulations.
2) Homogeneous moderated material addition will also be studied through zirconium hydride-bearing fuels. As mentioned above, the softer spectrum can create advantageous situations for both transmutation and proliferation purposes. However, the space occupied by moderating pins in the heterogeneous design reduces the number of pins remaining that can be loaded with minor actinides. This situation could potentially lead to a scenario where a high percentage of the minor actinides are destroyed during the course of an irradiation; but the overall throughput of minor actinide mass is less than if no moderating pins were used. As an alternative to the separate moderating pins, it is proposed that the moderator be directly incorporated into the minor actinide target pin in the form of the carrying matrix (i.e, inert matrix). The unique attribute of this approach is that target pins are not displaced by moderator pins. Instead, the moderation is incorporated in the form of actinide metal hydride fuels (e.g., MA-Zr-H). The work proposed by this task is to evaluate the optimal zirconium-hydrogen content in MA-Zr-H compositions that gives the maximum percent reduction (i.e., transmutation efficiency) and highest throughput (i.e., minor actinide consumption).
3) Minor actinide inclusion in the blankets will be studied as a way to increase transmutation efficiency and also as a means to reduce the weapons-usability of the resulting plutonium. Studies will include comparisons of different recycling scenarios and their impact on transmutation efficiency and proliferation resistance.
4) Minor actinide concentration will also be varied to identify the optimal scenario which can transmute the minor actinides efficiently while still providing an adequate breeding gain, a secure plutonium vector and improved safety coefficients.

5) The impact of using recycled uranium instead of depleted uranium in the blankets will also be addressed in this study. The slightly higher fissile content of recycled uranium will increase the power output at the BOL, which may have a small impact on transmutation efficiency. The interest in using recycled uranium stems from the possibility of keeping some of the spent uranium in the separation streams of the LWR spent fuel.
6) A combined set of metrics will be proposed to facilitate the blanket analysis. These metrics will combine proliferation resistance, fuel fabrication, transportation, toxicity, reactivity penalty and minor actinide transmutation.

2 Reference Core

This section gives a detailed description of the core models that are reproduced in ERANOS and MCNP and used in this benchmarking. Most of the core description is taken from (Stauff, 2008). Two MCNP input files were generated. The file *Core* contains the MCNP input for the original SFR core design and the file *1lay* contains the MCNP input for the original core designed modified by the author as described later in this report.

Core Design

The main characteristics of the core are described in Table 2.1. It is a 2400 MWt core using a metallic fuel. This design has been obtained using ABR1000 as the initial reference design. It is a self-sustaining core with a conversion ratio of one without using a Breeding Blanket.

Table 2-1: Main Characteristics of the Core

Thermal power	**2400 MW**
Number of fuel assemblies	360
Fuel pins per assembly	271
Number of control assemblies	19

The core has 3 different fuel zones: Inner, Middle and Outer core, as shown in Figure 2.1. These zones have different fuel compositions but the assemblies are structurally the same. Around this driver fuel is added 150 assemblies of Reflector (2 rows) and 102 assemblies of Shielding (1 row).

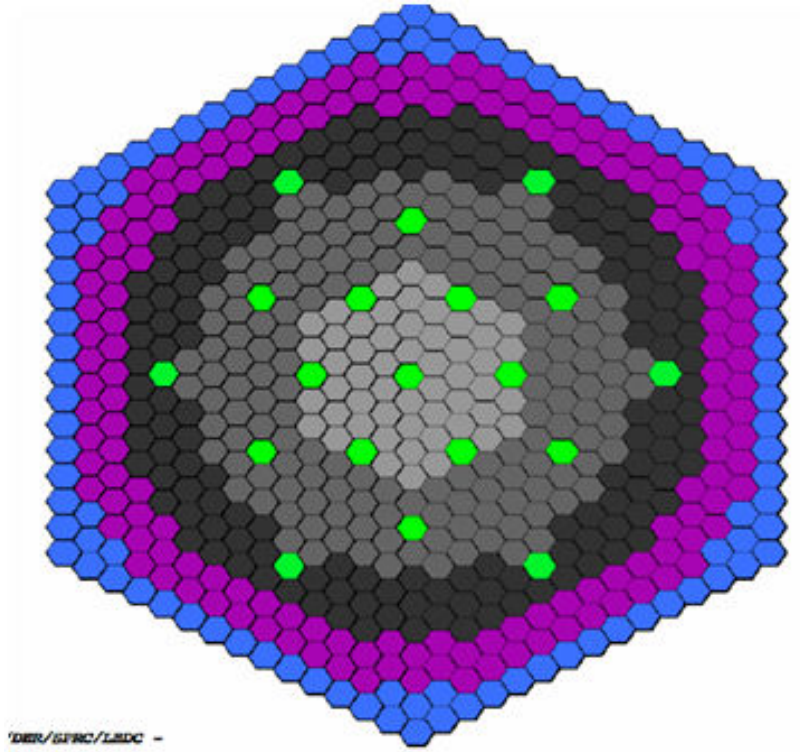

Figure 2-1: Radial Core Layout (Grey = Driver Fuel, Purple = Reflector, Blue = Shield)

Table 2.2 presents the fuel composition of the original core.

Table 2-2: Fuel Description

				Weight Percentages [%]		
	MCNP Material Number	**Number of Assemblies**	**Density [g/cm^3]**	**Uranium**	**Transuranics**	**Zirconium**
Inner Core	14	54	13.700519000	67.004700	11.995200	21.00
Middle Core	12	156	14.711031869	71.245521	12.754479	16.00
Outer Core	10	150	16.139520100	76.334487	13.665513	10.00

Table 2.3 presents the isotopic composition of the fuels described in Table 2.2.

Table 2-3: Fuel Isotopic Composition

Uranium	Weight Percent [%]	Transuranics	Weight Percent [%]	Zirconium	Weight Percent [%]
U-235	0.711	Np-237	6.6627788	Zr-90	51.45
U-238	99.299	Pu-238	2.74092	Zr-91	11.22
		Pu-239	48.801869	Zr-92	17.15
		Pu-240	23.063897	Zr-94	17.38
		Pu-241	6.952595	Zr-96	2.8
		Pu-242	5.080743		
		Am-241	4.679631		
		Am-243	1.470741		
		Cm-243	5.017E-3		
		Cm-244	4.97653E-1		
		Cm-245	3.8127E-2		
		Cm-246	6.02E-3		

Table 2.4 gives the axial layout of the core and the description of the materials compositions as modeled in the homogeneous ERANOS model.

Table 2-4: Axial Layout of the Core

	Height [cm]	Fuel [%]	Na [%]	HT9 [%]	B4C [%]
Lower Grid	20	0	28.6	71.5	0
Lower Shielding	40	0	28.6	25.7	45.7
Lower Reflector	40	0	28.6	71.5	0
Inner Fuel	102	34.29	11.43 (bond) and 28.54	25.73	0
Middle Fuel	102	34.29	11.43 (bond) and 28.54	25.73	0
Outer Fuel	102	34.29	11.43 (bond) and 28.54	25.73	0
					Helium [%]
Gas Plenum	120	0	28.6	25.7	45.7
Upper Assembly Shield	20	0	28.6	25.7	45.7
Radial Reflector	342	0	8.5	91.5	0
Radial Shield	342		21.0	34.2	44.8
Sodium Holes	202	0	29.0 (bond) and 38.30	32.7	0
Rods	140	0	38.3	32.7	29.0

Tables 2.5 through 2.8 present the relevant data for the heterogeneous ERANOS core model. All of the MCNP5 models in these tables were heterogeneous.

Table 2-5: Radial Reflector

	Outer Radius [cm]
HT-9	0.984037
Pitch of the Pins	1.96807
Number of the Pins	61
Number of the Rings	6
Duct	No

Table 2-6 Radial Shield

	Outer Radius [cm]
B4C20	2.143
HT-9	2.6756
Pitch of the Pins	5.35137
Number of the Pins	7
Number of the Rings	2
Duct	No

Table 2-7: Assembly Description

Fuel	Lower Assembly Reflector	Lower Assembly Shielding	Upper Gas Plenum	Lower Grid Plate Shielding	Upper Shielding	Outer Radius [cm]
Fuel	HT-9	B4C20	Helium	HT-9	B4C20	0.301487636
Na	HT-9	HT-9	HT-9	HT-9	HT-9	0.357361
HT-9						0.40400
pitch of the pins						0.875762
number of pins						271
number of rings						10
Duct						Yes

Table 2-8:Control Rod Sizes

Rod	Hole	Outer Radius [cm]
B4C20	Na	0.2772369
Na		0.4022317
HT-9		0.4378810
pitch of the pins		0.875762
number of pins		271
number of rings		10
Duct		Yes

The original SFR core was modified in MCNP to replace the first reflector row with a blanket whose characteristics are described in Tables 2.9 and 2.10.

Tables 2.9 and 2.10 present the parameters of the core modified from the original design and should be compared with Tables 2.2 and 2.3 respectively.

Table 2-9: Fuel Composition of Modified Core

				Weight Percentages [%]		
	MCNP Material	**Number of Assemblies**	**Density [g/cm^3]**	**Uranium**	**Transuranics**	**Zirconium**
Inner Core	14	54	13.70	67.15	11.85	21.00
Middle Core	12	156	14.71	71.40	12.60	16.00
Outer Core	10	150	16.14	76.50	13.50	10.00
Blanket*	110	72	16.14	75.00	15.00	10.00

* Separate isotope weight percentages for transuranic isotopes are given in Table 2.10.

Table 2-10: Fuel Isotopic Composition of Modified Core

Uranium	**Weight Percentage [%]**	**Transuranics**	**Weight Percentage in Fuel [%]**	**Weight Percentage in Blanket [%]**	**Zirconium**	**Weight Pecentage [%]**
U-235	0.3	Np-237	5.631		Zr90	51.45
U-238	99.7	Pu-238	2.777		Zr91	11.22
		Pu-239	51.01		Zr92	17.15
		Pu-240	22.95		Zr94	17.38
		Pu-241	11.07		Zr96	2.80
		Pu-242	6.557			
		Am-241		57.497		
		Am-242m		0.1945		
		Am-243		29.715		
		Cm-242		1.89×10^{-3}		
		Cm-243		0.0918		
		Cm-244		11.39		
		Cm-245		0.989		
		Cm-246		0.1157		
		Cm-247		2.04×10^{-3}		
		Cm-248		1.55×10^{-4}		
		Cf-249		2.53×10^{-6}		
		Cf-250		7.33×10^{-7}		
		Cf-251		4.18×10^{-7}		
		Cf-252		6.90×10^{-8}		

2.1 ERANOS and MCNP Comparison

The modified core was modeled using ERANOS2.2. Software information is provided below:

Software Name: ERANOS
Software Version: 2.2
Code Developer: CEA
Computer manufacturer name and model: Dell and HP Workstations
Name and version of operating system: Ubuntu 11.04 (64-bit)
Libraries or interfaces required: JEFF3.1 Library provided with software

The model was then independently verified using the MCNP software listed below:

Software Name: MCNP
Software Version: 5 v1.40
Code Developer: LANL
Computer manufacturer name and model: PSSC
Name and version of operating system: CentOS6
Libraries or interfaces required: Intel Compiler, MPICH2, JEFF3.1 ACE format library from NEA

The MCNP model was depleted using the MCODE2.2 and ORIGEN2.2 software with the following information:

Software Name: MCODE
Software Version: 2.2
Code Developer: MIT
Computer manufacturer name and model: PSSC
Name and version of operating system: CentOS6
Libraries or interfaces required: NA

Software Name: ORIGEN
Software Version: 2.2
Code Developer: ORNL
Computer manufacturer name and model: PSSC
Name and version of operating system: CentOS6
Libraries or interfaces required: NA

It was found that to correctly predict the k-eff evolution with ERANOS, the driver fuel region needed to be treated heterogeneously for cross-section generation and the core needs to be solved using the homogeneous transport solver VARIANT. Figure 2.2 presents the results. ERANOS was run using both a homogeneous fuel model and a heterogeneous fuel model.

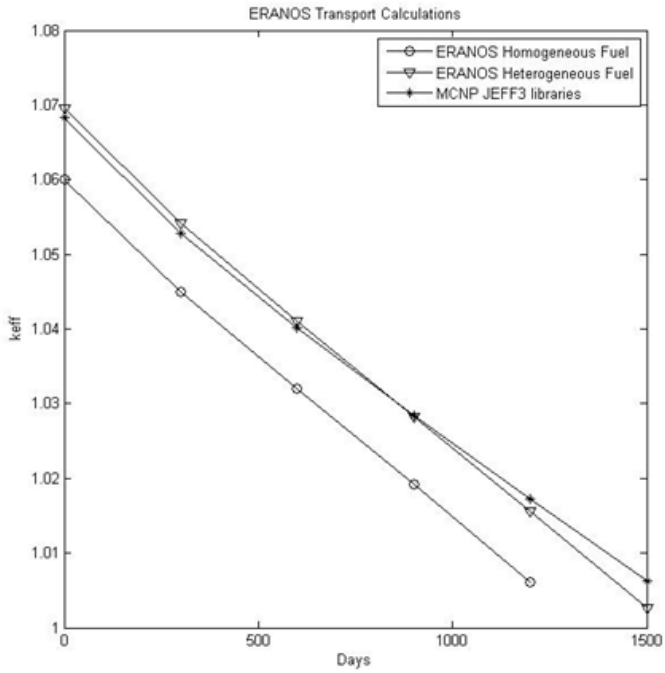

Figure 2-2: Eigenvalue comparison between ERANOS and MCNP

Qualitatively it is easy to see that the ERANOS transport calculation with heterogeneous fuel compares very well with the MCNP model. Quantitatively, the ERANOS heterogeneous model agrees with the MCNP model with an average of 0.14 % relative error on k_{eff} and the ERANOS homogeneous model shows an average of 0.87 % relative error on k_{eff} compared to the MCNP model. From this, it was concluded that only the ERANOS transport theory model with lattice calculations that treat heterogeneously modeled fuel would be used for future analysis of the optimal scenarios. Scoping analysis can still be done with homogeneous modeling as long as one is aware of this reactivity trend.

The axial and radial core flux profiles were calculated and compared for the ERANOS model and the MCNP model. Figure 2.3 presents the comparison inside the fuel region of the core. The flux was normalized by the core power.

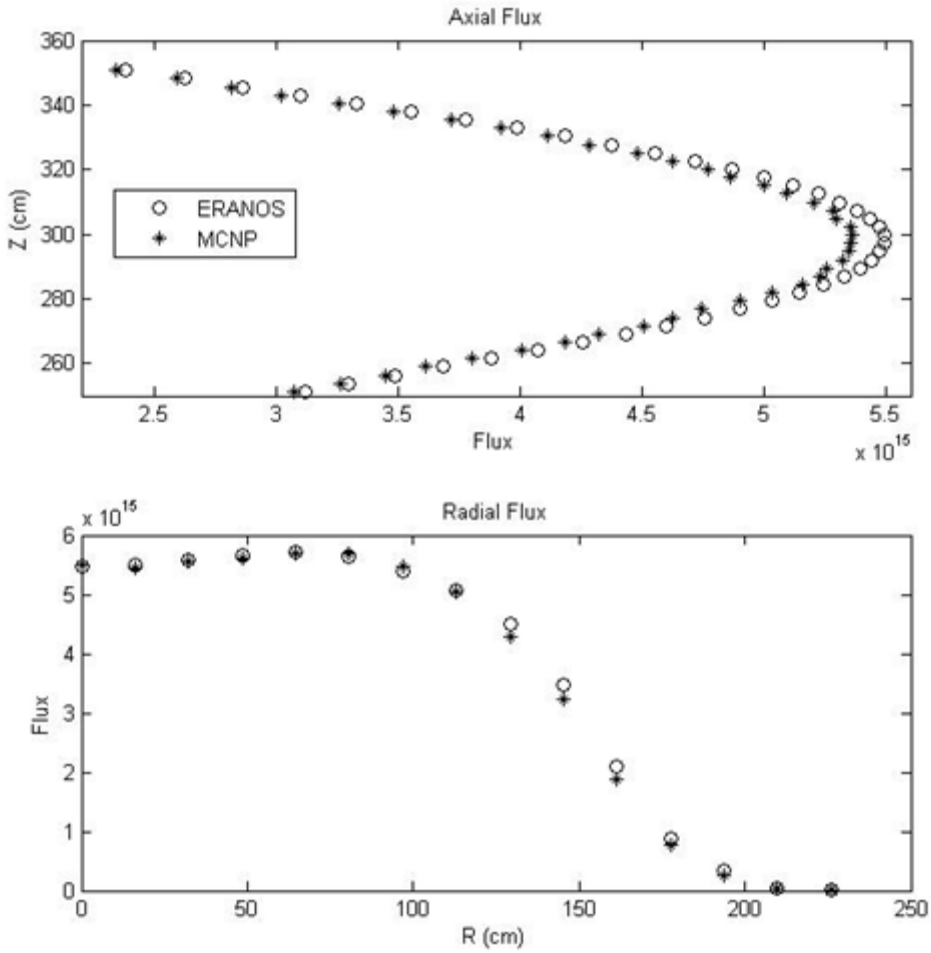

Figure 2-3: Radial (top) and Axial (bottom) core flux profiles at BOC

Figure 2.3 shows that the core flux profiles reported by ERANOS agree with the MCNP reported flux profiles to a high degree of accuracy. In fact, it was found that in the fuel region of the reactor the flux shapes agreed on average to within 2.08 % axially and 1.32 % radially. Good agreement on isotopic composition was also found. This analysis is further discussed in Section 4.2.1.

2.2 Modeling assumptions in this study

Additional modeling assumptions made in this study are summarized below:

- Cycle information
    - A three batch core with three enrichment zones was modeled in ERANOS.
    - Material was shuffled at the end of each cycle with the discharge fuel being placed out of the core for one cycle for reprocessing.
    - Cycle length was set at 500 effective full power days and a capacity factor of 85% was used.
    - Initial enrichment of the reference case was adjusted in the three zones such that the reactivity at the end of cycle would be 0 (+/- 100 pcm)
    - Entire core is decayed during the reactor downtime
    - Stored fuel (i.e. fuel being reprocessed) is also decayed during downtime and storage time.
    - Cycle length was set such that the average fluence of the driver fuel over three cycles would stay below $4x10^{23}$ neutrons/cm^2.
    - Single pass blanket assemblies remained in the core for 9 cycles to remain below the same fluence limit. The blanket residence time is thus between 14-15 years.
    - Driver fuel is composed of Pu and Np from LWR used fuel at 51 GWd/MTHM. Metal fuel was used with Zr diluent. (see Table 2-11 for actinide composition)

Blanket assemblies are composed of the remaining actinides of the LWR used fuel, a uranium vector (either DU, NU or RU) and a Zr diluent (see Table 2-12 for actinide composition)

    - A 75% smearing was assumed to evaluate the fuel density for metals and 85% for blanket assemblies that included oxide moderators.
    - Driver fuel reprocessing removes all fission products and reloads all actinides. Additional Pu and Np from LWR used nuclear fuel is added if needed to compensate for loss of reactivity in fuel.
    - Reference case is composed of a single row of blanket with no minor actinides and no moderation (i.e. metal fuel with 90% depleted uranium and 10% Zr).

- Fission product treatment
    - In the MCNP comparison, all fission products were assumed to stay in the fuel.
    - ERANOS has the additional option to re-normalize the fission products to allow volatiles to escape to the plenum. This approximation was made in all analysis (except MCNP verification) to get a better estimate of cycle reactivity.
    - Volatile fission products where then added to the exit stream in the post-processing.

- Modeling assumptions
    - All driver assemblies were modeled heterogeneously in ECCO since it is needed to accurately predict the cycle lengths.
    - Blankets were modeled homogeneously since they have a small impact on cycle reactivity. Additional details of the homogeneous representation are provided in section 2.2.1.

- Post-Processing
    - The post-processing code collects all discharge actinides and fission products throughout the run.
    - Fission products are expanded according to the distribution of reference (Tommasi, 2006).
    - Charge and discharge data can be provided as either a single stream (i.e. driver + blankets) or for two streams (i.e. separate blanket and driver data).
    - Isotopes are manipulated according to the specifications of VISION. Currently, provides 81 pseudo-isotopes with prescribed weights. Format is completely adjustable for future expansion/contraction needs.

Tables 2.11 and 2.12 present the driver fuel and blanket actinide vectors. These were calculated using TRITON/NEWT for a typical PWR lattice with 51GWd/MTHM burnup and 5 years of cooling.

Table 2-11: Driver Fuel Minor Actinide Vector

	Composition (%)
Np237	5.63
Pu238	2.78
Pu239	51.01
Pu240	22.95
Pu241	11.07
Pu242	6.56

Table 2-12: Blanket Minor Actinide Vector

	Composition (%)
Am241	57.49
Am242m	0.19
Am243	29.71
Cm242	1.90×10^{-3}
Cm243	9.18×10^{-2}
Cm244	11.39
Cm245	9.89×10^{-1}
Cm246	1.16×10^{-1}
Cm247	2.04×10^{-3}
Cm248	1.55×10^{-4}
Cf249	2.53×10^{-6}
Cf250	7.33×10^{-7}
Cf251	4.18×10^{-7}
Cf252	6.90×10^{-8}

2.2.1 Geometry study of the blanket

Many studies have gone further than simple isotopic analysis and provided detailed blanket designs to transmute minor actinides [Refs]. Before going further in our analysis, it was important to investigate the need for detailed modeling of the blanket assemblies and the impact that various designs will have on the compositions. This section will study various homogeneous and heterogeneous designs, as well as compare the homogeneous and heterogeneous modeling capabilities within ERANOS.

<u>Pin Size</u>

This section evaluates the effect pin size has on core performance. Ten cases are designed with different pin pitch. To maintain the same material composition in the blanket region as described in the core description, the radius of the fuel pin is adjusted correspondingly. As listed in Table 2.13, the smaller the pin pitch, the smaller the fuel radius, and thus more pins are placed in each assembly. In this parametric study, the only difference between these 10 models is the pin size (and hence number of pins per assembly). Material compositions in the blanket material are identical: 29.67 vol% $ZrH_{1.6}$, 10 vol% MA, and 60.33 vol% Uranium. The 10 models are simulated using ERANOS for 650 days once-through cycle.

Table 2-13: Pin Size Study

Model Name	# of Rings	Pin Count	Pin Pitch (cm)	Fuel Radius (cm)
R6	6	91	1.45	0.52
R7	7	127	1.24	0.44
R8	8	169	1.12	0.38
R9	9	217	0.95	0.34
R10	10	271	0.89	0.30
R11	11	331	0.80	0.28
R12	12	397	0.73	0.25
R13	13	469	0.68	0.23
R14	14	547	0.63	0.21
R15	15	631	0.58	0.20

The results are summarized in Table 2.14 and Table 2.15. After 650 days of burn-up, little difference is observed among the 10 models. A fatter pin design is favored for manufacturing reasons.

Table 2-14: Pin Study Results (R6-R10)

Model Name	R6	R7	R8	R9	R10
Heating Rate (W/kg)	233.83	233.81	233.97	233.69	233.87
Photon Heating Rate (W/kg)	12.48	12.52	12.58	12.56	12.62
Neutron Production Rate (n/s.kg)	1.30E+008	1.30E+008	1.30E+008	1.30E+008	1.30E+008
Total MA Mass(kg)	677.4	677.4	677.4	677.5	677.5
Pu-238 (kg)	31.46	31.44	31.43	31.40	31.40
Pu-239 (kg)	96.79	97.13	97.63	97.49	97.97
Fissile Pu (kg)	97.89	98.22	98.72	98.58	99.05
Total Pu (kg)	148.22	148.53	149.05	148.85	149.36
Fissile Pu/Total Pu	0.660	0.661	0.662	0.662	0.663
Reactivity (pcm)	1040.8	1043.9	1051.9	1044.3	1053.7

Table 2-15: Pin Study Results (R11-R15)

Model Name	R11	R12	R13	R14	R15
Heating Rate (W/kg)	233.82	233.78	233.79	233.76	233.72
Photon Heating Rate (W/kg)	12.63	12.64	12.66	12.66	12.66
Neutron Production Rate (n/s.kg)	1.29E+008	1.29E+008	1.29E+008	1.29E+008	1.29E+008
Total MA Mass (kg)	677.5	677.5	677.5	677.5	677.5
Pu-238 (kg)	31.39	31.38	31.37	31.36	31.36
Pu-239 (kg)	98.03	98.12	98.29	98.36	98.3337
Fissile Pu (kg)	99.12	99.20	99.37	99.44	99.41
Total Pu (kg)	149.41	149.48	149.64	149.70	149.67
Fissile Pu/Total Pu	0.663	0.664	0.664	0.664	0.664
Reactivity (pcm)	1052.6	1052.6	1055.1	1055.3	1053.6

Heterogeneous Assembly

This section presents four blanket designs (designated as Version A, Version B, Version 1, and Version 2) that are of same material composition, but different geometrical arrangement of isotopes. Burnup calculations of 1300 days with transport theory solver are performed for each model.

Version A

Version A assumes that the entire blanket assembly is homogenized. The blanket fuel composition is described in Table 2.16, which is then homogenized with all other structures described previously.

Table 2-16: Blanket Composition

Blanket Material Composition (vol %)	
ZrH1.6	29.67
Minor Actinides	10.00
Uranium	60.33

Version B

Version B is a 6 ring heterogeneous representation of the blanket design with pin dimensions specified in Table 2.15. Comparing version A and B will help determine the need for explicitly modeling the blanket, as was done for the fuel in the model verification.

Version 1 and 2

Version 1 and 2 are heterogeneous pin loadings based on Version B. All 3 versions have identical material composition and pin dimension. The only difference between Version B and

the numbered versions is that in Versions 1 and 2, the moderator is located in separate pins rather than being mixed into all pins as in Version B. Both Versions 1 and 2 have 27 separate pins of ZrH1.6 and 64 pins loaded with MA and Uranium. The number of pins was determined such that ZrH1.6 represents 29.67% of all available fuel volume, just as in Version B. In order to examine the effect of varying the positions of the ZrH1.6 pins, Version 1 and 2 represent two different arrangements of the heterogeneous pin loadings. Detailed assemblies map are presented below in Figures 2.4 and 2.5, with the lighter pins being ZrH1.6.

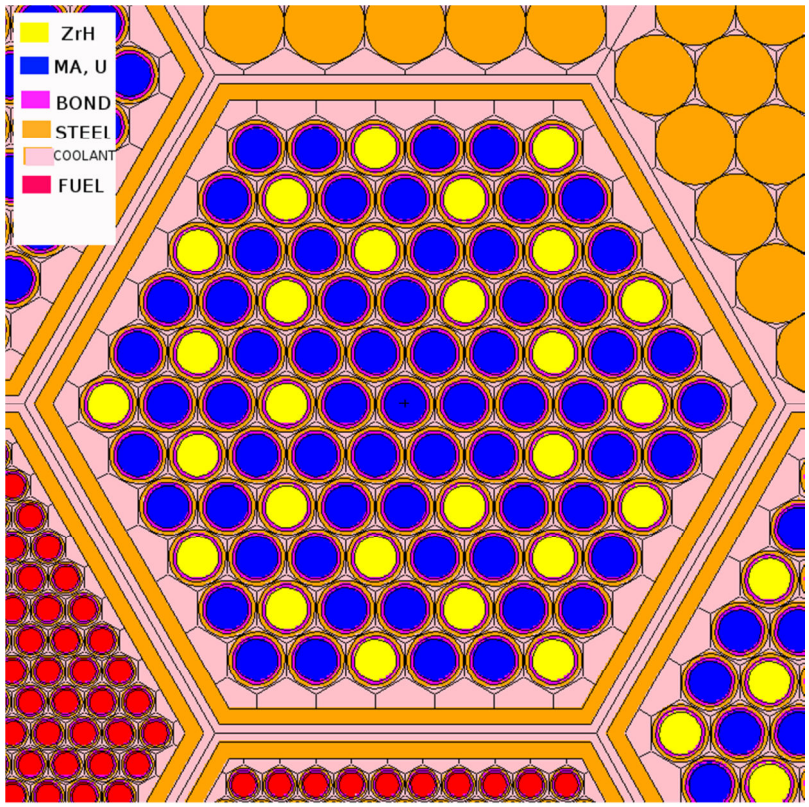

Figure 2-4: Version 1 layout

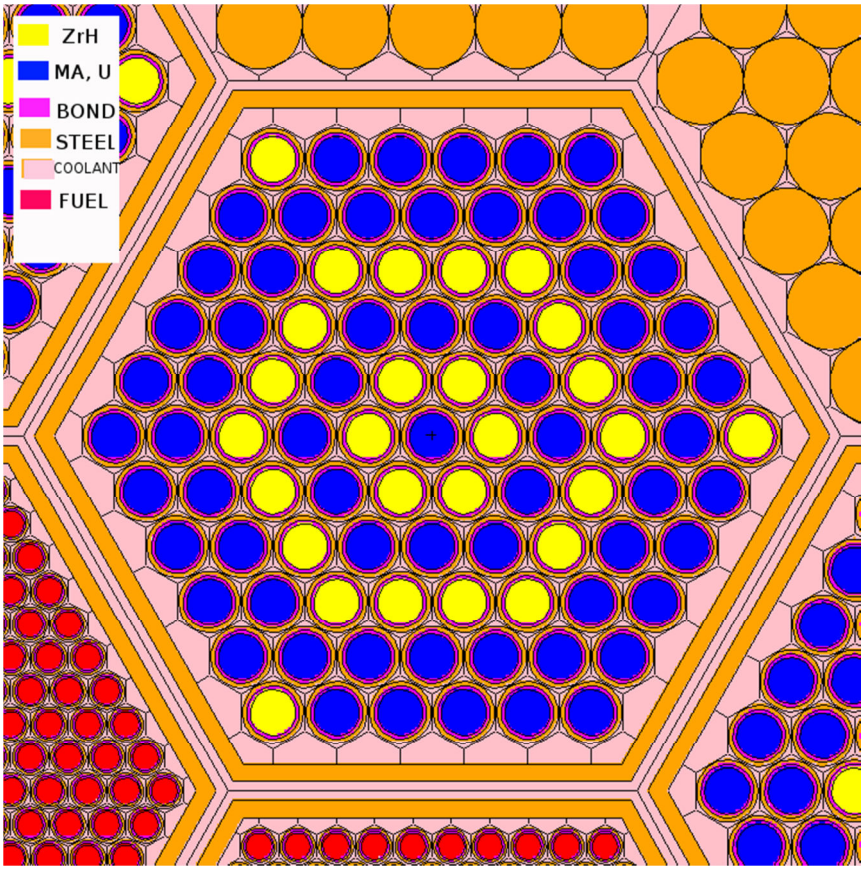

Figure 2-5: Version 2 layout

Results for all 4 options are presented in table 2.17.

Table 2-17: Main Results for 4 Blanket Options

	Version A	Version B	Version 1	Version 2
Heating rate (W/kg)	2.35×10^{2}	2.35×10^{2}	2.34×10^{2}	2.34×10^{2}
Photon Heating rate (W/kg)	1.47×10^{1}	1.42×10^{1}	1.25×10^{1}	1.24×10^{1}
Neutron Production rate (N/s/kg)	1.42×10^{8}	1.43×10^{8}	1.46×10^{8}	1.46×10^{8}
MA Transmutation (kg)	1.39×10^{2}	1.40×10^{2}	1.47×10^{2}	1.47×10^{2}
Pu239 / Total Pu	0.617	0.608	0.570	0.567
Pu238 / Total Pu	0.223	0.230	0.262	0.264
Change in reactivity (pcm)	-1090	-1130	-1080	-1080

Results from Version A and B are very similar, indicating that there is not a significant benefit in using the explicit geometry in the lattice calculation stage of the ERANOS model. Version 1 and 2 also present similar results thus making the pin layout fairly inconsequential on the overall blanket performance. Comparing the results of version B and version 1 (or 2) presents some noticeable differences. For instance, the transmutation efficiency of version 1 is roughly 5% larger and the Pu238 content represents a significantly larger proportion of the total Pu. However, version 1 also offers a smaller breeding gain.

In conclusion, this study indicates that various heterogeneous pin arrangements could provide some small tradeoffs that might warrant more in depth analysis. However, for the purpose of scoping studies, a homogeneous model such as version A provides a sufficient level of details.

MCNP Verification

MCNP analysis was performed for Version B and Version 1 of the blanket designs. In Version B, the $ZrH_{1.6}$ moderator (30% of fuel volume) is distributed evenly in all the pins. Version 1 is a design with solid ZrH pins dispersed amongst fuel pins. The reactivity versus time was not affected by the target design, thus mass comparisons of selected isotopes were looked at instead.

Table 2-18: Ratio of Selected Isotopes

	ERANOS	MCNP	Relative Error (%)
Version B			
Am241	0.601	0.593	1.30
Am243	0.646	0.634	1.88
Cm244	1.390	1.451	-4.19
Version 1			
Am241	0.582	0.620	-6.09
Am243	0.647	0.681	-5.03
Cm244	1.397	1.426	-2.04

Results indicate a reasonably good agreement between the ERANOS models and MCNP models. This agrees with the conclusion provided by the ERANOS modeling that blankets in the reference core can be appropriately modeled using the simple fully homogeneous model for mass balance scoping analysis. Final blanket designs would require an heterogeneous analysis for power peaking, manufacturing and handling capabilities, but such a detailed analysis is not required for the fuel cycle scoping work.

3 Figure-of-Merit

Many studies on nuclear reactors and fuel cycle focus only on one particular aspect. For example, proliferation papers focus often solely on proliferation resistance methodologies whereas transmutation studies focus only on transmutation efficiency. The aim of this work is to provide a multi-attribute assessment methodology.

Two proliferation resistance approaches are proposed in this section. The first will be based on an approach developed by Charlton (Charlton, 2007). In his paper, he developed a proliferation resistance methodology based on multi-attribute utility analysis for the evaluation of various fuel cycles. His method for the proliferation attribute takes several metrics into consideration to assess the overall effectiveness of fuel cycle facilities. Proliferation specific criteria from his study will be leveraged and combined with additional criteria defined for our specific purposes.

The second approach will use the same formal concept of multi-attribute analysis but will confer a yes or no decision on the proliferation attribute using the definition provided by Bathke (Bathke, 2009). This approach will also be coupled with additional criteria for the purpose of this study.

The focus of this study thus becomes defining additional attributes such as fabrication, minor actinide transmutation (or production), fissile material production, and reactivity penalty for the assessment of heterogeneous recycling in fast reactors.

A metric is assigned to each attribute which is defined by a utility function that varies between 0 and 1. The normalization of the utility function is based on the minimum and maximum values observed in all runs. Each metric is also assigned a weight that represents its importance in the overall multi-attribute figure of merit. There are three different weight assessment methods: direct importance weights, trade-off weights, and swing weights (Fischer, 2002). For this study we will use direct importance weights. According to this weight assessment method, numerical values are assigned with the sole purpose of signifying their relative importance. As a starting point for the FOM, all attributes will be assigned equivalent weights. It is understood that this method may be inaccurate or even biased. Biases arise due to an error in numerical judgment. Therefore, it is imperative that weights be assessed carefully to ensure that the results of the evaluation are consistent with the expected preferences. Thus, sensitivity analysis of weights will be performed in Chapter 5 to determine the overall impact on the FOM with changes in weights. In this Chapter, a summary of the Charlton and Bathke proliferation metrics will be presented, followed by the definition of the additional metrics used in the study.

3.1 Proliferation metric

Many studies have been performed in the past to quantify the proliferation resistance of materials and fuel cycles (Ford). In this study, the metrics by Charlton and Bathke will be leveraged and are summarized below.

3.1.1 Charlton

Charlton proposed 12 metrics to evaluate the proliferation resistance of fuel cycle processing plants. The first 6 of these are directly related to the proliferation quality of the material. These metrics include Department of Energy attractiveness level, heating rate from plutonium, weight fraction of even plutonium isotopes, concentration of fissile materials, radiation dose rate, and the size and the weight of a fuel assembly. The latter 6 deal mainly with material accountability and physical processes, and are thus excluded from this study. The metrics of interest are summarized below:

DO*E Attractiveness Level (u_1)*

DOE attractiveness level is usually considered as one of the most important aspects in the proliferation resistance. This is the first metric for the proliferation attribute in Charlton's paper. His utility function which measures the quality of material in process for this metric will be used in this study. Unattractive materials are less likely to be stolen or diverted by a proliferator. Thus, materials with lower quantities have been assigned higher utility function values and vice-versa. The utility function table for this metric is as follows:

Table 3-1: Utility Function (u_1) for DOE Attractiveness Level (x_1)

Attractiveness	**Category**			
	I	II	III	IV
B	0.00	0.05	0.10	0.15
C	0.15	0.25	0.35	0.45
D	NA	0.40	0.65	0.90
E	NA	NA	NA	1.00

This attribute is fairly straightforward and can be obtained from the definitions from DOE M474.1-1 (Department of Energy, 2000). In our case, the attractiveness will always fall in attractiveness level C and its level within C will be determined based on the total amount of Pu in the assembly.

Heating Rate from Plutonium (u_2)

This is the second metric for the proliferation attribute by Charlton. Any material with a high heating rate makes assembling an explosive device more difficult (National Academy of Sciences, 1994). Charlton captures this fact in this utility function where x_2 is the heating rate (W/kg) from the plutonium in the material and x_{2max} is the maximum possible heating rate set to 570 W/kg, the heating rate of pure Pu-238. The utility function for this metric is as follows:

$$u_2(x_2) = 1 - \exp\left[-3\left(\frac{x_2}{x_{2,\max}}\right)^{0.8}\right]$$

Equation 3-1

The heat rate from plutonium needed for evaluating x_2 is calculated using data from ORIGEN2.2 (Croff, 1983). The post-processing utility, EPT, described in appendix takes the plutonium vector for any fuel or blanket region, multiplies each isotope by its heat rate generation and normalizes by the mass in the region. Figure 3-1 illustrates this metric.

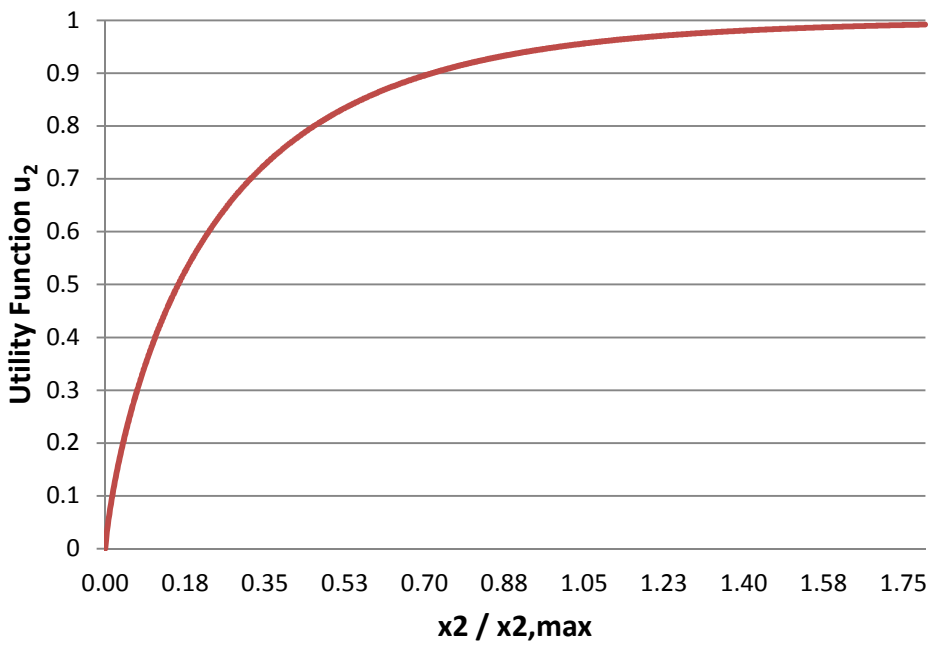

Figure 3-1: Heating Rate from Pu Attribute

Weight Fraction of Even Pu Isotopes (u_3)

The high rate of spontaneous fission in Pu-240 can easily cause pre-detonation and/or fizzle in a nuclear explosive (Charlton, 2007). Also, the presence of even plutonium isotopes significantly complicates the construction of nuclear explosives. Therefore, Charlton has constructed a utility function that rewards the presence of even plutonium isotopes in the material. A highly desirable case is when the quantity of plutonium is identically zero. A utility function value of 1 is assigned in this case. The utility function is defined by Equation 3-2.

$$u_3(x_3) = 1 - \exp\left[-3.5\,(x_3)^{1.8}\right]$$

Equation 3-2

The x_3 variable is also computed by the EPT processing code and is defined by Equation 3-3. The code sums up the mass of even plutonium isotopes simulated, namely Pu-238, Pu-240 and Pu-242, and divides by the total plutonium mass. The utility function is shown in Figure 3-2.

$$x3 = \frac{mass\ of\ even\ Pu\ isotopes}{mass\ of\ all\ Pu\ isotopes}$$

Equation 3-3

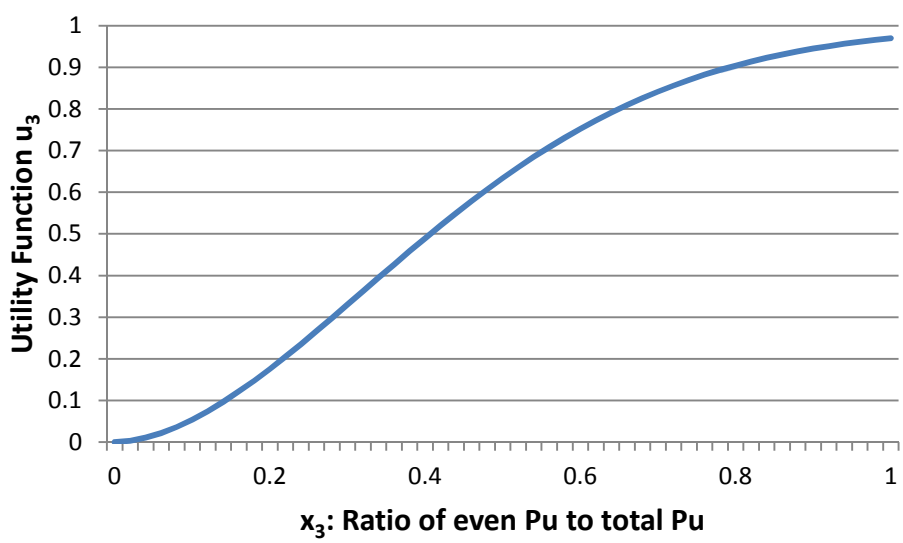

Figure 3-2: Weight Fraction of Even Pu Isotopes Attribute

Concentration (u_4)

The concentration metric considers the concentration of fissile material in the processing step. It is assumed that higher concentration of material with lower mass or volume will be preferable to a proliferator to steal or divert. The input for this metric is the number of SQs (significant quantity) of material per ton based on the International Atomic Energy Agency's (IAEA) definition (Charlton, 2007). The utility function and the plot for this metric are as follows where x_4 is the concentration for the material (SQs per ton) and x_{4max} is the maximum possible concentration calculated assuming the material was pure plutonium metal.

$$u_4(x_4) = \begin{cases} 1 & if\ x_4 < 0.01 \\ 1 - \exp\left[-2.5\left(\dfrac{x_4}{x_{4,\max}}\right)\right] & f\ x_4 \geq 0.01 \end{cases}$$

Equation 3-4

Significant quantity in our case, as defined in a report by Cochran and Pain (Cochran, 1995), corresponds to:

Table 3-2: Significant Quantity (SQ) Definition

1 SQ	8 kg of Pu
1 SQ	25 kg of Np
1 SQ	75 kg of LEU

Radiation Dose (u_5)

The utility function for this metric is based on acute biological effects of whole body radiation dose to the proliferator. Any material with high dose rate would require special and expensive equipment for remote handling purposes. Also, it can impose health risks to the proliferator. Therefore, materials with high radiation dose rate are assigned higher utility function values. Above a threshold of 600 rem/hrSQ^{-1} (dose rate per significant quantity) it is assumed that there is no increase in proliferation resistance as death is certain in all cases. The utility function for the radiation dose is given by Equation 3-5:

$$u_5(x_5) = \begin{cases} 0 & if\ x_5 \leq 0.2 \\ 0.0520833x_5 - 0.010416 & if\ 0.2 < x_5 \leq 5 \\ 0.0035714x_5 + 0.232143 & if\ 5 < x_5 \leq 75 \\ 0.00095238x_5 + 0.428571 & if\ 75 < x_5 \leq 600 \\ 1 & if\ x_5 > 600 \end{cases}$$

Equation 3-5

The utility function for this attribute requires the dose rate in rem /(hr.SQ) for the unshielded material at a distance of 1 m. This is approximated using the data by Bayes (2009). He had evaluated the dose rate for major actinides from recycling in light water reactor assemblies. Shielding and geometry effects would certainly yield different results but not significant enough for our purposes of comparison between similar cases.

Size/Weight (u_6)

The utility function for the size/weight of a single unit in process (e.g., a fuel assembly) is a binary function. This metric captures the fact that it would be difficult for the proliferator to steal or divert any material that is large and heavy. An input of "yes" (for volume > 0.0566m^3 or mass > 90.72kg) corresponds to a utility function value of one and an input value of "no" corresponds to zero.

$$u_6(x_6) = \begin{cases} 1 & if\ x_6 = yes \\ 0 & if\ x_6 = no \end{cases}$$

Equation 3-6

This simple attribute takes into account the difficulty of moving assemblies of large size or weight which can be determine easily by the dimensions of the assembly and its density. In our case, this utility function always takes a value of 1.

Weights

The six attributes presented above were part of a subset of many attributes for evaluating the proliferation resistance of a process. Since we are only addressing the material in this study, the assigned weights need to be renormalized. The weights given for each attribute by Charlton and their renormalization are given in Table 3-3.

Table 3-3: Weights of FOM_1

Attribute	Charlton weights	Renormalized weights
u_1	0.10	0.2222
u_2	0.05	0.1111
u_3	0.06	0.1333
u_4	0.10	0.2222
u_5	0.08	0.1778
u_6	0.06	0.1333

3.1.2 Bathke

Bathke defines a figure of merit (FOM) to quantify the attractiveness of nuclear materials for both proliferation states and sub-national groups. His FOM is based on the Department of Energy's calculation of "attractiveness level" for its nuclear materials in nuclear facilities (Bathke, 2009). Primary factors considered in material's attractiveness include bare critical mass, internal heat generation, and radiation dose rate (Bathke, 2008). As mentioned in Section 3.1.1, material attractiveness is one of the most important factors in proliferation resistance considerations. The definition of proliferation resistance used in this study is "that characteristic

of a nuclear energy system that impedes the diversion or undeclared production of nuclear material, or misuse of technology, by States in order to acquire nuclear weapons or other nuclear explosive devices" (International Atomic Energy Agency, 2002). Bathke defines two FOMs. The first one is applicable to technically advanced proliferation nation or a sub-national group for whom any nuclear yield is acceptable. Bathke's second FOM applies to a relatively unadvanced proliferation nation. Since his first FOM is more conservative, we will use it to evaluate material attractiveness for proliferation resistance and all safeguards and physical protection issues. The FOM is defined by Equation 3-7.

$$FOM = 1 - \log_{10}\left(\frac{M}{800} + \frac{Mh}{4500} + \frac{M}{50}\left[\frac{D}{500}\right]^{\frac{1}{\log_{10} 2}}\right)$$

Equation 3-7

where M is the bare critical mass of the metal in kg, h is the heat content in W/kg, and D is the dose rate of 0.2*M evaluated at 1 m from the surface in rad/h.

The critical mass is evaluated using the reaction rates computed from the ECCO lattice calculations in the data preparation sequence of the ERANOS simulations and simple critical sphere buckling. An extra ECCO calculation is performed at the end-of-cycle for the fuel material in the blanket (i.e. no clad and no sodium) to get the reactions rates of the fuel material. It was identified that our critical mass values were larger than expected. Simple ERANOS models of pure U-235 and Pu-239 were analyzed and revealed some inconsistency in determining a proper diffusion coefficient for the simple geometry. We had initially estimated the diffusion coefficient as $1/3\Sigma_t$ which only works for isotropic materials. Critical mass measures were thus adjusted by modifying the diffusion coefficient provided by ERANOS. MCNP was used to compute the critical radius of a few typical configurations and a conservative diffusion coefficient value was implemented in the post-processing code. The dose calculations are estimated using data calculated at INL (Bays et al). Even though this data is case specific, it should provide an order of magnitude estimate of the dose. This study provides dose in Sv/(kg.hr) for the major actinides for a given configuration. Neglecting the contribution from fission products in terms of dose is considered a conservative assumption when discussing proliferation impact. The decay heat rate and the neutron emission rate will be obtained from ORIGEN2.2 (Croff, 1983) data for the key isotopes. The evaluation of these parameters is automated in the EPT code.

This FOM corresponds to nuclear material's attractiveness in the following manner when applied to metals or alloys and attractiveness values are provided in Table 3-4. It should be noted that the FOM represents the degree of preference of materials for weapons proliferation.

Table 3-4: Bathke's Attractiveness Criterion

FOM	Weapons Utility	Attractiveness	Attractiveness Level
>2	Preferred	High	B
1-2	Attractive	Medium	C
0-1	Unattractive	Low	D
<0	Unattractive	Very Low	E

According to Bathke, the FOM must be less than 1 to make the material unattractive for use in a nuclear device. We will go even further in our study to make sure that we only consider the highest level of proliferation resistance and therefore only consider FOM values that are less than 0 which Bathke correlates with material "very low" in attractiveness. A binary "yes" or "no" decision will be conferred on the Bathke proliferation attribute in order to decide which cases are acceptable from the proliferation point of view for our study. Any FOM with values less than 0 will be automatically assigned a "yes" decision meaning that the material is as proliferation resistant as possible and is acceptable from the proliferation point of view in this study. We will disregard any case with FOM greater than 0 and assign it a "no" decision in order to meet our most conservative criteria. It should be noted that negative FOM values do not refer to "proliferation-proof" material. Table 3-4 only presents the level of preference of particular nuclear material for use in a nuclear weapon or explosive device. By choosing the minimum FOM that corresponds to the least preferable material, we are only increasing the proliferation resistance to the highest level possible. A sensitivity study was also performed with the dose value. Setting the dose to 0 has very little impact on the FOM value since in our case it is mainly driven by the heat rate.

3.2 Fuel Fabrication

The major factors that complicate the handling of spent nuclear fuel are neutron emission, photon emission, and decay heat. All three of these factors usually go hand-in-hand, thus our metric will be focused solely on neutron emission for simplicity. Neutron emission from spent nuclear fuel impacts transport operations and also puts severe constrains on the fuel manufacturing process. The neutron emission metric will be associated with the dose received by a worker. Therefore, high neutron emission rate is highly undesirable. A relationship between neutron emission rate and an effective dose equivalent to a worker working 8 hours a day, 260 days a year, at an average distance of 1 meter from a fast neutron source of an irradiated blanket fuel assembly was established using the neutron rate and the fluence-to-dose conversion factor (Knoll, 2000). It was determined conservatively that a neutron emission rate of 1×10^4 neutrons/s/kg, without shielding, would approximately correspond to the maximum annual dose limit of occupational radiation exposure to adults working with radioactive material set by the Nuclear Regulatory Commission (NRC) of 50 mSv per year (United States Nuclear Regulatory

Commission). Therefore, a utility function value of 1 is assigned to neutron rates below 1×10^4 neutrons/s/kg since such fuel would require little to no shielding of the workers.

Radiation shielded containment chambers such as glove box and hot cells, used for handling radioactive materials, usually have viewing lead glass windows with a thickness of 4 inches to 16 inches to facilitate operation and maintenance. Cost-effective manipulators in hot cells are limited to 12 inches of lead glass windows (International Atomic Energy Agency, 2004). It was determined that this thickness would shield approximately a neutron emission rate of up to 1×10^6 neutrons/s/kg while making sure that the maximum amount of occupational radiation exposure to the worker is limited to 50 mSv per year. Hence, the utility function value for a neutron rate between 1×10^4 neutrons/s/kg and 1×10^6 neutrons/s/kg was selected to linearly decrease to account for the increasing shielding costs.

Neutron rate beyond 1×10^6 neutrons/s/kg requires even thicker shielding, specially designed stronger and better manipulators, complex handling operations, and frequent maintenance resulting in hefty costs. Also, it was determined that any thickness beyond 60 inches could be considered impractical due to visibility issues (Rob, 2011). This maximum thickness would shield approximately 1×10^{14} neutrons/s/kg such that a yearly worker remains below the annual dose limit. Therefore, the utility function with a steep negative slope was assigned for neutron rates between 1×10^6 neutrons/s/kg and 1×10^{14} neutrons/s/kg in order to take into consideration the increasing shielding as well as associated manipulator's costs. Neutron rate beyond 1×10^{14} neutrons/s/kg has a sharp exponential drop. The detailed calculations are included in the appendix. The utility function is defined by Equation 3-8.

Neutron Emission Rate (u_7)

$$u_7(x_7) = \begin{cases} 1 & if\ x_7 \le 10^4 \\ -3E-07x_7 + 0.8 & if\ 10^4 < x_7 \le 10^6 \\ -4E-15x_7 + 0.5 & if\ 10^6 < x_7 \le 10^{14} \\ \exp[-2E-14x_7] & if\ x_7 > 10^{14} \end{cases}$$

Equation 3-8

The x_7 metric is defined in neutrons per kg per second and its utility function can be seen in Figure 3-3.

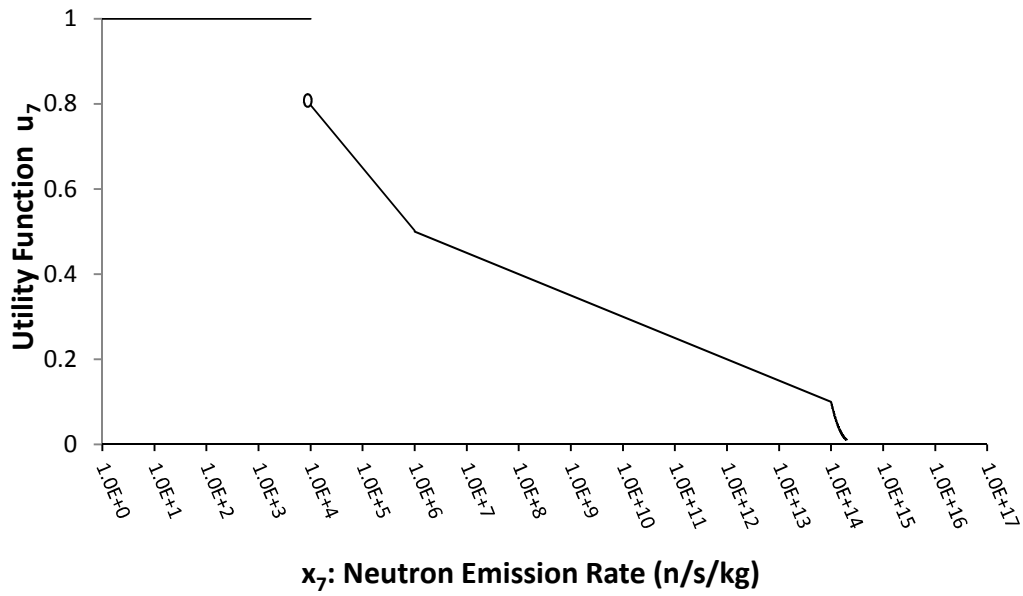

Figure 3-3: Fuel Fabrication (u_7) as a function of neutron emission rate (x_7)

3.3 Radiotoxicity

Using the ICRP 72 data on toxicity (International Commission on Radiological Protection, 1972) and the isotope masses from the reference core, several Am and Cm isotopes were identified as very toxic for both inhalation and ingestion. Two utility functions have been developed based on the isotopes in the blankets and their contribution to toxicity.

Radiotoxicity Inhalation (u_8)

Utility function u_8 is for the inhalation toxicity of the material present in the blankets and is defined in Equation 3-9. Metric x_8 is defined as the sum of all contributions in Sv/kg.

$$u_8(x_8) = \begin{cases} 1 & if\ x_8 < 1.0E8 \\ -1E-10x_8 + 1.0101 & if\ 1.0E8 \le x_8 \le 1.0E10 \\ 0 & if\ x_8 > 1.0E10 \end{cases}$$

Equation 3-9

Figure 3-4 presents the utility function with respect to x_8.

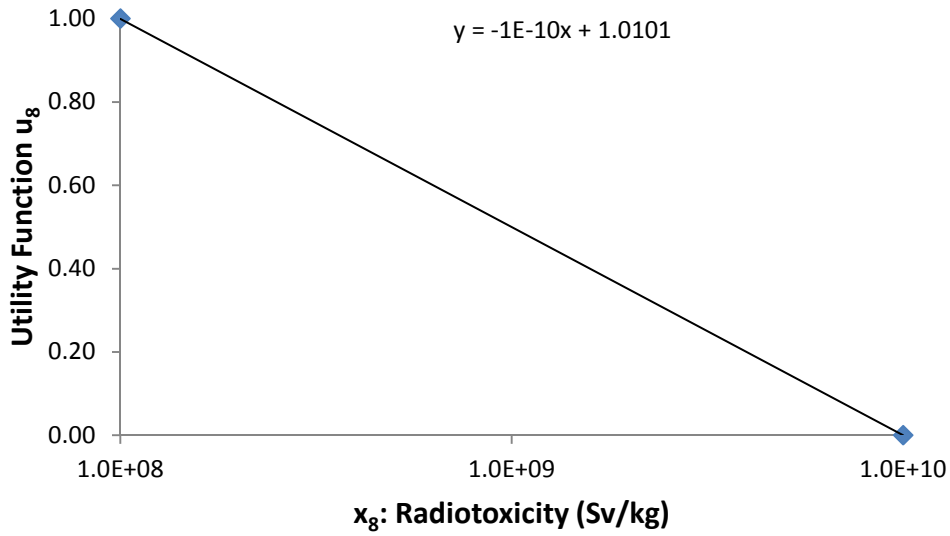

Figure 3-4: Utility function u_8 - Radiotoxicity by inhalation

Radiotoxicity Ingestion (u_9)

Utility function u_9 is very similar to the previous, except that it deals with the radiotoxicity by ingestion. It is defined by Equation 3-10. Metric x_9 is defined as the sum of all contributions in Sv/kg.

$$u_9(x_9) = \begin{cases} 1 & if\ x_9 < 1.0E5 \\ -1E-8x_9 + 1.001 & if\ 1.0E5 \le x_9 \le 1.0E10 \\ 0 & if\ x_9 > 1.0E10 \end{cases}$$

Equation 3-10

Figure 3-5 shows the utility function u_9 as a function of x_9.

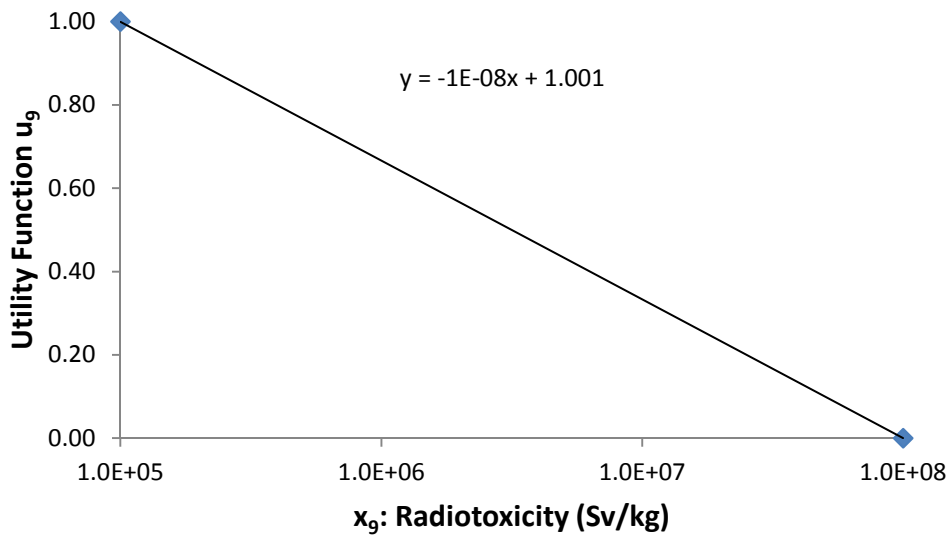

Figure 3-5: Utility function u_9 - Radiotoxicity by ingestion

3.4 Breeding Gain

The production of fissile plutonium is highly undesirable from a non-proliferation point of view. One of Charlton's six proliferation metrics explicitly favors higher weight fraction of even plutonium isotopes. However, fissile plutonium also plays an important role in achieving larger breeding gains which ultimately reduces supply constraints for operating fast reactors and provides fuel for startup cores. A metric is defined for fissile plutonium production. This metric assesses the conversion of U-238 to Pu-239. The utility function rewards higher fissile production rate and higher breeding gains to maintain the sustainability of the core. It only looks at the material in the blankets.

Fissile Plutonium Production (u_{10})

This utility function is defined as a linear function using a maximum value of 1 and a minimum value close to zero. The metric x_{10} is defined as the ratio of the Pu-239 mass at the end of life to the mass of U-238 at the beginning of life.

$$u_{10}(x_{10}) = \begin{cases} 0 & \text{if } x_{10} < 0.01 \\ 1.0101x_{10} - 0.0101 & \text{if } 0.01 \le x_{10} \le 1.00 \\ 1 & \text{if } x_{10} > 1.00 \end{cases}$$

Equation 3-11

Figure 3-6 presents the utility function u_{10}.

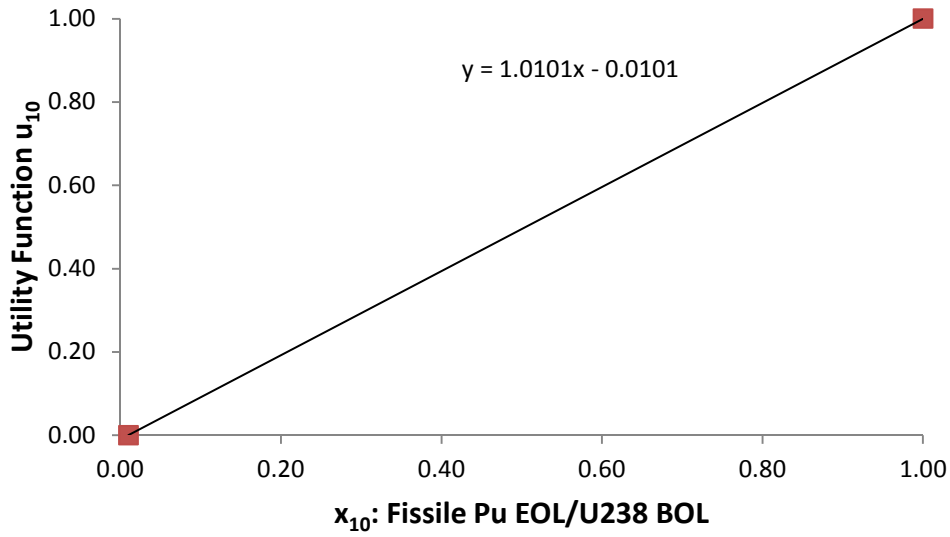

Figure 3-6: Breeding Gain (u_{10})

3.5 Reactivity Penalty

The introduction of minor actinides and moderators has a significant impact on the core performance. Thus, we introduce a metric to assess the reactivity penalty produced by these additions. This metric captures the loss and the gain of neutron economy by analyzing the value of k-infinity at the end of cycle. In a general sense, the infinite multiplication factor, k-infinity, is the ratio of neutrons produced in one generation to the neutrons lost in preceding generation through absorption. More neutrons mean more possibilities for breeding, transmutation of minor actinides, and overall increase in power density. Therefore the utility function for this metric rewards higher k-infinity values with higher utility function values.

Reactivity Penalty (u_{11})

This utility function is a linear function between typical k-infinity values that we observed in multiple runs. The metric x_{11} is the k-infinity of the blanket region at the beginning-of-life (BOL).

$$u_{11}(x_{11}) = \begin{cases} 0 & \text{if } x_{11} < 0.10 \\ 2.5x_{11} - 0.25 & \text{if } 0.1 \le x_{11} \le 0.5 \\ 1 & \text{if } x_{11} > 0.5 \end{cases}$$

Equation 3-12

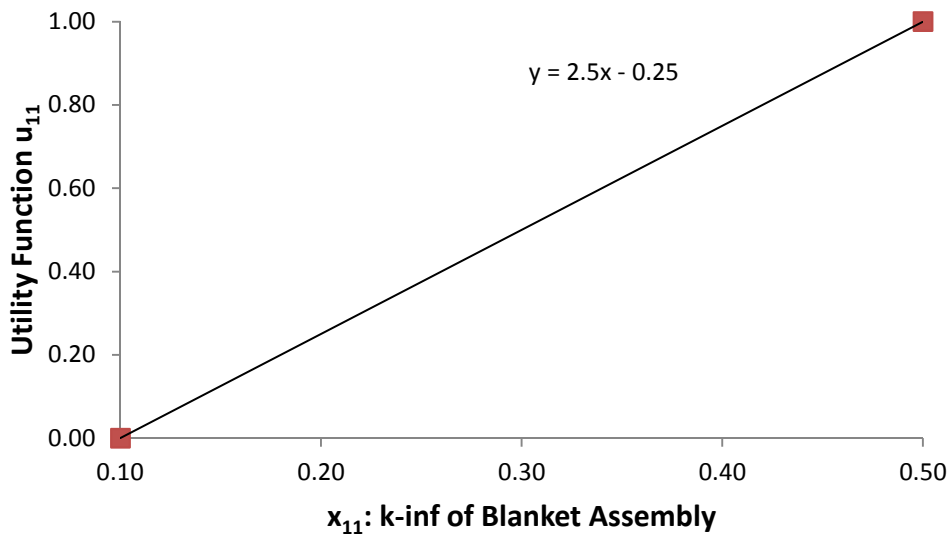

Figure 3-7: Reactivity Penalty (u_{11})

3.6 Transportation

Heat rate, neutron emission and photon rate can complicate the handling of spent fuel quite considerably. Instead of assessing the impact of each individual component, it was assumed that heat rate was a representative measure since it affects cooling and is directly proportional to neutron and photon rates.

Heat Rate (u_{12})

A utility function based on the heat rate is constructed in a way such that higher heat rates are assigned with lower utility function values and vice-versa. The metric x_{12} is defined as the heat rate of the material in W/kg.

$$u_{12}(x_{12}) = \begin{cases} 1 - 0.0005x_{12} & if\ 0 \le x_{12} \le 2000 \\ 0 & if\ x_{12} > 2000 \end{cases}$$

Equation 3-13

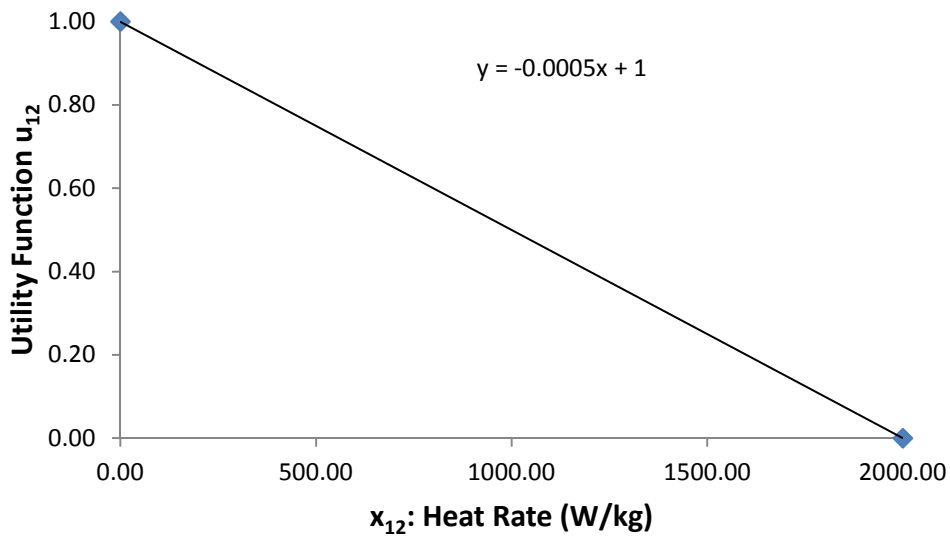

Figure 3-8: Transportation (u_{12})

3.7 Global metrics

Most assessment methodologies focus only on one attribute. In this study, we will construct a methodology that includes multiple attributes such as proliferation resistance, fuel fabrication, transportation, fissile material production, and reactivity penalty for the assessment of various strategies in heterogeneous recycling in fast reactors. Metrics which are assigned to attributes with associated weights will give us an opportunity to determine the importance of each attribute in our figure of merit. Therefore, the user will have the ability to find the optimal strategy depending on the assigned weights. We wish to find the best scenario for heterogeneous recycling which transmutes the maximum amount of minor actinides, produces enough fissile plutonium for the sustainability of the core, considers fuel handling and transportation constraints and the core performance while still being proliferation resistant. In doing so, a sensitivity analysis will also be performed. Finally, the multi-attribute figure of merit will provide a solid ground for comprehensive comparisons.

3.7.1 Figure of Merit 1 (FOM_1)

This figure of merit will take into account only the first six metrics from Charlton's proliferation resistance methodology with renormalized weights of Table 3-3. The idea of FOM_1 is to study the effects of minor actinide inclusion and moderation on proliferation resistance.

$$FOM_1 = \sum_{i=1}^{6} w_i x_i$$

Equation 3-14

3.7.2 Figure of Merit 2 (FOM_2)

This figure of merit will take into account the six additional attributes defined in this study that cover fuel fabrication, radiotoxicity, and transportation constraints as well as fissile material production and reactivity penalty. The goal is to access how minor actinide inclusion and moderation affect the above mentioned attributes. Equal weights are given to all attributes except for the two radiotoxicity measures that are combined as one.

$$FOM_2 = \sum_{i=7}^{12} w_i x_i$$

Equation 3-15

3.7.3 Figure of Merit 3 (FOM_3)

This multi-attribute FOM will include Charlton's six metrics for proliferation resistance attribute along with fuel fabrication, transportation, fissile material production, and reactivity penalty attributes. Thus, FOM_3 is the combination of FOM_1 and FOM_2. The sensitivity analysis of the weights will be performed in Chapter 5.

$$FOM_3 = \sum_{i=1}^{12} w_i x_i$$

For the reference case, weights are assigned according to Table 3-5.

Table 3-5: Weights for FOM_3

	Weights
w_1	0.0370
w_2	0.0185
w_3	0.0222
w_4	0.0370
w_5	0.0296
w_6	0.0222
w_7	0.1667
w_8	0.0833
w_9	0.0833
w_{10}	0.1667
w_{11}	0.1667
w_{12}	0.1667

These weights represent equal weighting of the 6 major attributes with the proliferation attribute subdivided by the Charlton weights and the radiotoxicity attribute with two equal components.

3.7.4 Figure of Merit 4 (FOM_4)

The figure of merit will be directly obtained from the post-processing tool that assigns the material a figure of merit based on Bathke's material attractiveness definition.

3.7.5 Figure of Merit 5 (FOM_5)

This FOM will include a yes or no decision for the proliferation resistance attribute based on Bathke's definition. Only cases that are assigned with "yes" decision for proliferation resistance attribute will then be considered for the multi-attribute FOM that includes fuel fabrication, transportation, fissile material production, and reactivity penalty attributes. Initially equal weights will be assigned for this FOM with both radiotoxicity measures combined as one.

3.8 Summary

In this section we defined a series of utility functions and figures of merit that address proliferation, fuel fabrication, transportation, radiotoxicity and reactivity penalty. The functions will be used to approximate the impact of different modifications that will be made to the blankets to find an optimal configuration. None of these will truly provide a universal answer to the global question of proliferation and performance, but it will allow us to compare many cases using a prescribed evaluation.

The utility functions defined in sections 3.3 to 3.6 are based on observed results and cannot be applied directly to different core configurations or material compositions. The general idea behind still stands, but the actual values could be adjusted to be representative of a wider set of problems.

It is envisioned that initial analysis will identify preferred figure of merits and that a down-selection will be performed. Additionally, sensitivity analysis on the weights will be performed in Chapter 5. Table 3-6 summarizes the proposed figures of merit.

Table 3-6: Summary table

Metric	Description	Metrics Included
x_1	DOE Attractiveness Level (Charlton)	—
x_2	Heating Rate from Pu (Charlton)	—
x_3	Weight Fraction Even Pu Isotopes (Charlton)	—
x_4	Concentration of Fissile Material (Charlton)	—
x_5	Dose Rate (Charlton)	—
x_6	Size/Weight of Assembly (Charlton)	—
x_7	Neutron Emission Rate	—
x_8	Radiotoxicity (Inhalation)	—
x_9	Radiotoxicity (Ingestion)	—
x_{10}	Breeding Gain	—
x_{11}	Reactivity Penalty	—
x_{12}	Heat Rate (Transportation)	—
Bathke FOM	Bathke's FOM for Prolif. Attractiveness	—
FOM_1	—	x_1- x_6
FOM_2	—	x_7- x_{12}
FOM_3	—	x_1- x_{12}
FOM_4	—	Bathke FOM
FOM_5	—	Bathke FOM + x_7- x_{12}

4 Blanket Studies

A fertile blanket has been a matter of great concern in sodium fast reactors because of the potential for high quality weapons grade plutonium production. A fast reactor's surplus neutrons have historically been used for the breeding of fissile material and are currently being investigated for the destruction of spent nuclear fuel actinides produced from light water reactors (Bays et al, 2008). Beside producing more power and flattening the power density profile of the core, one of the most important advantages of a blanket is in achieving positive breeding gain. While an absence of a traditional fertile blanket may reduce proliferation risks, it adversely affects breeding gain, fissile inventory, reactivity, achievable specific power, and the distribution of power density. Therefore, it has a substantial impact on the sustainability of the core and fuel cycle costs. This chapter will look at ways to reduce the proliferation risk of blankets while having minimal impact on the operation of the reactor. The addition of minor actinides and moderators will be analyzed to mitigate the proliferation risk. This study will also clarify the benefits of a uranium (natural, depleted, and recycled) blanket and its importance in the sustainability of the core while still taking proliferation into consideration.

4.1 Minor Actinide content

A previous study at MIT concluded that blanket designs for the resulting bred plutonium to be less attractive than 51GWd/MTHM light water reactor spent fuel requires minor actinide addition (Stauff, 2010). This study also concluded that it was not possible to build a SFR employing a "secure" blanket without minor actinide input from light water reactors. Therefore, minor actinides will be included in the blankets for different recycling scenarios. Concentration of minor actinides will be varied (up to 10% volume). Depleted uranium blanket assemblies with minor actinides will be assessed on a cycle-by-cycle basis to find the optimal minor actinide content based on various attributes defined in chapter 3. The available volume of the blanket region with one row is 1.6572×10^6 cm^3. The MA content of 10% by volume refers to 723.72 kg of MA. Am241 for example consists of 57.5% by mass of total MA. Therefore the MA 10% by volume refers to 423.72 kg of Am241. In all cases, blanket assemblies were left in the core for 9 cycles with 500 days in each cycle. The data presented corresponds to the end-of-life (EOL) at cycle 9. The cases have a blanket region of depleted uranium (80-88% volume), MA (Am and beyond) (2-10% volume), and zirconium (10% volume). The initial composition of the actinide vector is provided by a LWR spent fuel vector at 51GWd/MTHM with details provided in Chapter 2.

This section will evaluate the minor actinide addition in the context of the five figures of merit presented in Chapter 3. Table 4-1 shows the impact of minor actinide addition using FOM1 with varying minor actinide volume fraction in the blanket assemblies from 1% to 10%.

Table 4-1: FOM_1 for different MA contents (Note: based on Charlton's normalized weights)

	1% MA	2%MA	3%MA	4%MA	5%MA	6%MA	7%MA	8%MA	10%MA
FOM_1	0.356	0.370	0.382	0.393	0.404	0.415	0.425	0.435	0.454

These results indicate that higher minor actinide contents are favorable from a proliferation standpoint. This figure-of-merit includes Department of Energy attractiveness level (u_1), heating rate from plutonium(u_2), weight fraction of even plutonium isotopes (u_3), concentration of fissile materials (u_4), radiation dose rate (u_5), and the size and the weight of a fuel assembly (u_6). Table 4-2 presents the individual contributions from each of the six utility functions making up FOM_1.

Table 4-2: Utility function values $u_1 - u_6$

	1%MA	2%MA	3%MA	4%MA	5%MA	6%MA	7%MA	8%MA	10%MA
u_1	0.15	0.15	0.15	0.15	0.15	0.15	0.15	0.15	0.15
u_2	0.120	0.176	0.223	0.264	0.301	0.333	0.363	0.389	0.436
u_3	0.0745	0.0963	0.120	0.144	0.169	0.195	0.221	0.247	0.298
u_4	0.749	0.749	0.749	0.748	0.748	0.748	0.748	0.748	0.748
u_5	0.00918	0.0284	0.0477	0.0672	0.0867	0.106	0.126	0.146	0.186
u_6	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0

Addition of minor actinides had no effect on two of these metrics: Department of Energy attractiveness level (u_1), and the size and the weight of a fuel assembly (u_6). Minimal impact was seen in the concentration of fissile materials (u_4) which is based on the notion of significant quantities as defined by the IAEA. Therefore, the increase in proliferation resistance is mostly due to the heating rate from plutonium, weight fraction of even plutonium isotopes and radiation dose rate. Since we are interested in more than just proliferation resistance, it is interesting to also look at our other set of attributes, mainly fuel fabrication, fuel transportation, toxicity, fissile Pu production, and reactivity. Table 4-3 summarizes the results for FOM_2 with greater details provided in Table 4-4.

Table 4-3: FOM_2 for different MA contents

	1% MA	2%MA	3%MA	4%MA	5%MA	6%MA	7%MA	8%MA	10%MA
FOM_2	0.437	0.437	0.436	0.435	0.434	0.432	0.431	0.429	0.426

Table 4-4: x_7 -x_{12} values

	1%MA	2%MA	3%MA	4%MA	5%MA	6%MA	7%MA	8%MA	10%MA
x_7	8.8E+06	1.7E+07	2.6E+07	3.5E+07	4.4E+07	5.2E+07	6.1E+07	7.0E+07	8.7E+07
x_8	1.4E+07	1.5E+07	1.5E+07	1.6E+07	1.7E+07	1.8E+07	1.8E+07	1.9E+07	2.1E+07
x_9	2.6E+08	4.8E+08	6.9E+08	9.1E+08	1.1E+09	1.3E+09	1.6E+09	1.8E+09	2.2E+09
x_{10}	0.0673	0.0672	0.0671	0.0670	0.0670	0.0668	0.0668	0.0667	0.0666
x_{11}	0.217	0.229	0.240	0.251	0.261	0.272	0.282	0.292	0.312
x_{12}	1178	1212	1246	1280	1316	1351	1388	1424	1500

x_7: Neutron emission rate (neutrons/kg/s)
x_8: Radiotoxicity – ingestion (Sv/kg)
x_9: Radiotoxicity – inhalation (Sv/kg)
x_{10}: Ratio of Pu-239 at EOC to U-238 at BOC
x_{11}: k_{inf} of blanket at EOC
x_{12}: Heat rate (W/kg)

From table 4.3 we can see a small reverse trend to that of table 4.1. The radiotoxicity, neutron emission rate, heat rate increase significantly as we increase the MA content of the blanket which has a negative impact (closer to 0) on the FOM. However, the k_{inf} increases with more MA which has a positive impact on the performance of the core. The mass based breeding ratio (x_{10}) is fairly constant for all cases with a slight decrease as we add MA. When combining the results of both tables to form FOM_3, we can immediately notice the impact of the weights that were given to each criterion. Table 4-5 shows the results of FOM_3 with varying MA content.

Table 4-5: FOM_3 for different MA contents

	1% MA	2%MA	3%MA	4%MA	5%MA	6%MA	7%MA	8%MA	10%MA
FOM_3	0.424	0.425	0.427	0.428	0.429	0.429	0.430	0.430	0.430

The subjective distribution of weight suggests that one can get any desired results from this FOM by weighting preferred parameters accordingly. As a starting point for the FOM, all 6 major attributes have been assigned equivalent weights. It is understood that this method may be inaccurate or even biased. Biases arise due to an error in numerical judgment. Therefore, it is imperative that weights be assessed carefully to ensure that the results of the evaluation are consistent with the expected preferences. Thus, sensitivity analysis of weights will be performed in Chapter 5 to determine the overall impact on the FOM with changes in weights.

The proliferation resistance metrics along with reactivity penalty metric outweigh other attributes such as radiotoxicity, fissile plutonium production, and heat rate to favor the addition of minor actinides. However, from the transportation, fuel fabrication, and sustainability of the core standpoint, less minor actinide inclusion is desired as seen in Table 4-3. It should also be noted that the previous study at MIT (Stauff, 2010) had concluded that blanket designs for the resulting

bred plutonium to be less attractive than 51MWD/kg light water reactor spent fuel requires minor actinide doping of at least 2% according to Saito's proliferation criteria (Stauff, 2010). Therefore, we must find an optimal minor actinide content (by volume) to be included in the blankets that produces enough plutonium for the sustainability of the core, considers fuel handling constraints and the core performance while still meeting this proliferation requirement.

An alternative approach proposed in Chapter 3 is to consider the proliferation to be a "yes or no" decision, and then compromise between the other attributes to find a suitable blanket composition. FOM_4 proposes using the Bathke criteria and setting the safe limit to any value less than zero. Table 4-6 presents the performance of the blanket assemblies with varying MA content with respect to the Bathke proliferation metric

Table 4-6: FOM_4 for different MA contents

	1% MA	2%MA	3%MA	4%MA	5%MA	6%MA	7%MA	8%MA	10%MA
FOM_4	0.281	0.133	0.0339	-0.0384	-0.0938	-0.138	-0.173	-0.202	-0.245

These results indicate that our proliferation metric is met once we have at least 4% minor actinides in our reference blanket assemblies. Additional minor actinides will improve the proliferation resistance, but as shown in Table 4-3 will also hinder other performance criteria. For the analysis in this study, FOM_2 is better suited since it will allow us to optimize our blanket design without having to deal with conflicting attributes.

According to the Bathke criteria, FOM must be less than 1 to make the material unattractive for use in a nuclear device. We will go even further in our study to make sure that we only consider the highest level of proliferation resistance and therefore only consider FOM values that are less than 0 which Bathke correlates with material "very low" in attractiveness. A binary "yes" or "no" decision will be conferred on the Bathke proliferation attribute in order to decide which cases are acceptable from the proliferation point of view for our study. Any FOM with values less than 0 will be automatically assigned a "yes" decision meaning that the material is considered proliferation resistant for purposes of this study. We will disregard any case with FOM greater than 0 and assign it a "no" decision in order to meet our most conservative criteria. It should be noted that negative FOM values do not refer to "proliferation-proof" material. By choosing the minimum FOM category that corresponds to the least preferable material, we are only increasing the proliferation resistance to the highest level possible.

Table 4-7: FOM_5 for different MA contents

	1% MA	2%MA	3%MA	4%MA	5%MA	6%MA	7%MA	8%MA	10%MA
FOM_5	-	-	-	0.435	0.434	0.423	0.431	0.429	0.426

It is evident from Table 4-6 that a minimum of 4% minor actinides (by volume) is required for the desired level of proliferation resistance. From Table 4-7 it is also evident that 4%MA is preferable even when fuel fabrication, transportation, fissile material production, and reactivity penalty attributes are taken into consideration.

4.2 Moderation study

Beside the inclusion of minor actinides in the SFR blanket, moderator addition has also been studied as a possibility to breed less attractive plutonium. A recent study [Reference] had concluded that the combination of minor actinide addition and moderator could result in a plutonium vector less attractive than 50MWD/kg light water reactor spent fuel and could also reduce the amount of minor actinides needed. . Nonetheless, moderator addition in itself was not sufficient to make the plutonium vector less attractive. Moderators act by softening the spectrum and encouraging interactions in a range where the fission-to-capture ratio is lower and the system is more inclined to produce Pu-240. This is illustrated in Figure 4-1. By favoring the production of Pu-240 over Pu-239 due to the spectral shift, moderators significantly affect the quality of the bred plutonium from the proliferation standpoint. However, without any initial actinides present in the fuel, moderators alone are not enough of a deterrent since they offer very little in terms of dose and neutron source. Adding used LWR fuel provides not only the aforementioned attributes but also provides a source of Am for the production of Pu-238 that also complicates weapons fabrication.

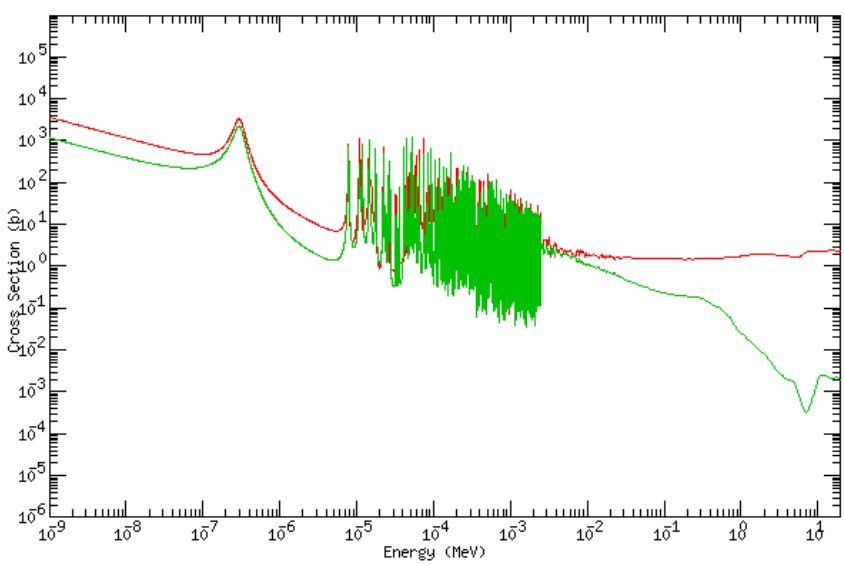

Figure 4-1: Pu-239 fission cross-section (red) and radiative capture (green)

Numerous moderating materials were identified from the literature to be included in the blanket . The list of moderators include boron carbide (B_4C), beryllium fluoride (BeF_2), lithium fluoride (LiF_2), beryllium oxide (BeO), magnesium oxide (MgO), calcium hydride (CaH_2), yttrium hydride (YH_2), and zirconium hydride ($ZrH_{1.6}$). Over the years, these materials have been considered for fast reactors for various reasons.

Tucek and Wider (2007) introduced moderators such as boron carbide (B_4C) and calcium hydride (CaH_2) in several dedicated pins within each fuel subassembly to compare the effectiveness of these moderators to improve Doppler fuel temperature reactivity feedback and the coolant temperature reactivity coefficient in SFRs. Fortescue (2009) investigated beryllium fluoride (BeF_2) as a moderator in an experimental breeder reactor. Both beryllium fluoride (BeF_2) and lithium fluoride (LiF_2) are used in thorium reactor designs (Morse, 2011). Roushe and Simnad (1978) used magnesium oxide (MgO) moderator in a reactor incorporating means for preventing molten fuel from breaching the containment vessel. Schmidt and Croixmarie tested moderators such as calcium hydride (CaH_2) and boron carbide (B_4C) in order to accelerate the process of transmutation significantly (Scdmidt et al). Pillion and Grandjean have argued that in terms of moderators, yttrium hydride has been preferred to calcium or zirconium hydride because of the best compromise between thermal stability and hydrogen content (Pillion et al). Many of these materials have already been tested in a radioactive environment.

Our objective is to find the best material that will provide a combination of minimizing radiotoxicity, produce enough plutonium vector for adequate breeding gain and sustainability of the core, and at the same time produce as little heat and spontaneous neutrons as possible for fuel handling and transportation considerations. A recent study (Massie, 2010) proposed an optimization strategy to select which material would provide the greatest benefit (if included in the blanket) for long term isotope disposal looking more specifically at heat rate. That study determined that $ZrH_{1.6}$ at 30-40% by volume provided considerable benefits to the long term decay heat.

All of these moderators were run for a single cycle at different moderator concentrations while the minor actinide concentration was held constant at 4% by volume. All moderator properties that we used can be found in Appendix. Additional minor actinides content were also tested but had no impact on the observed trends.

Tables 4-8 through 4-10 summarize our findings for different moderator volumes.

Table 4-8: Results for Selected Materials – 4%MA, 10% Moderator

	4%MARef	10% $ZrH_{1.6}$	10% YH_2	10%CaH_2	10%LiH
Fissile Pu EOL/Total Pu EOL	0.905	0.875	0.876	0.880	0.875
MA transmutation (%)	-3.65	-5.21	-5.20	-4.96	-5.15
Heat Rate (W/kg)	1.112E2	1.482E2	1.488E2	1.451E2	1.492E2
Photon Heat Rate (W/kg)	2.680E1	3.358E1	3.375E1	3.314E1	3.367E1
Neutron Production Rate (N/s/kg)	3.664E7	4.248E7	4.265E7	4.228E7	4.319E7
Fissile Pu EOL/U238 BOL	0.0108	0.0127	0.0127	0.0125	0.0126
k-infinity BOL	0.251	0.189	0.191	0.199	0.182
	10%BeO	10%MgO	10%BeF_2	10% LiF	10%B_4C
Fissile Pu EOL/Total Pu EOL	0.904	0.905	0.903	0.905	0.906
MA transmutation (%)	-3.81	-3.72	-3.83	-3.68	-3.16
Heat Rate (W/kg)	1.181E2	1.153E2	1.192E2	1.139E2	9.673E1
Photon Heat Rate (W/kg)	2.830E1	2.772E1	2.855E1	2.730E1	2.309E1
Neutron Production Rate (N/s/kg)	3.789E7	3.743E7	3.814E7	3.747E7	3.561E7
Fissile Pu EOL/U238 BOL	0.011	0.0111	0.0114	0.0110	0.00928
k-infinity BOL	0.228	0.238	0.226	0.227	0.188

Table 4-9: Results for Selected Materials – 4%MA, 20% Moderator

	4%MARef	20% $ZrH_{1.6}$	20% YH_2	20%CaH_2	20%LiH
Fissile Pu EOL/Total Pu EOL	0.905	0.837	0.838	0.845	0.842
MA transmutation (%)	-3.65	-7.09	-7.08	-6.61	-6.56
Heat Rate (W/kg)	1.112E2	1.903E2	1.926E2	1.866E2	1.853E2
Photon Heat Rate (W/kg)	2.680E1	3.989E1	4.046E1	3.954E1	3.863E1
Neutron Production Rate (N/s/kg)	3.664E7	5.121E7	5.175E7	5.135E7	5.182E7
Fissile Pu EOL/U238 BOL	0.0108	0.0143	0.0143	0.0141	0.0138
k-infinity BOL	0.251	0.162	0.165	0.174	0.145
	20%BeO	20%MgO	20%BeF_2	20% LiF	20%B_4C
Fissile Pu EOL/Total Pu EOL	0.892	0.895	0.892	0.895	0.897
MA transmutation (%)	-4.15	-3.95	-4.20	-3.84	-2.92
Heat Rate (W/kg)	1.292E2	1.224E2	1.318E2	1.195E2	8.863E1
Photon Heat Rate (W/kg)	2.959E1	2.825E1	3.015E1	2.735E1	2.012E1
Neutron Production Rate (N/s/kg)	4.183E7	4.071E7	4.242E7	4.076E7	3.724E7
Fissile Pu EOL/U238 BOL	0.0125	0.0119	0.0126	0.0116	0.00843
k-infinity BOL	0.208	0.226	0.204	0.206	0.147

Table 4-10: Results for Selected Materials – 4%MA, 30% Moderator

	4%MARef	30% $ZrH_{1.6}$	30% YH_2	30%CaH_2	30%LiH
Fissile Pu EOL/Total Pu EOL	0.905	0.802	0.802	0.810	0.813
MA transmutation (%)	-3.65	-9.02	-9.07	-8.38	-7.64
Heat Rate (W/kg)	1.112E2	2.336E2	2.396E2	2.334E2	2.160E2
Photon Heat Rate (W/kg)	2.680E1	4.601E1	4.728E1	4.635E1	4.192E1
Neutron Production Rate (N/s/kg)	3.664E7	6.037E7	6.171E7	6.176E7	6.035E7
Fissile Pu EOL/U238 BOL	0.0108	0.0158	0.0159	0.0156	0.0147
k-infinity BOL	0.251	0.146	0.149	0.157	0.126
	30%BeO	30%MgO	30%BeF_2	30% LiF	30%B_4C
Fissile Pu EOL/Total Pu EOL	0.878	0.883	0.877	0.884	0.884
MA transmutation (%)	-4.56	-4.20	-4.64	-4.01	-2.74
Heat Rate (W/kg)	1.426E2	1.306E2	1.472E2	1.255E2	8.271E1
Photon Heat Rate (W/kg)	3.098E1	2.868E1	3.193E1	2.725E1	1.772E1
Neutron Production Rate (N/s/kg)	4.659E7	4.457E7	4.765E7	4.459E7	3.936E7
Fissile Pu EOL/U238 BOL	0.0137	0.0127	0.0139	0.0123	0.00775
k-infinity BOL	0.189	0.215	0.185	0.188	0.117

From these results we can make a few observations in identifying suitable moderators. The first thing we notice is that most of the hydrides behave quite similarly. They all have a high moderating power, thus making them all excellent moderators. Out of the materials selected, they provide the best transmutation properties, mainly through the higher absorption rate in Am-241. However, this causes higher photon and neutron rates since we are creating more higher actinides. Hydrides also provide safer plutonium than the reference case since it lowers the fissile concentration despite producing more fissile plutonium per initial U238 loading than any other case. There is significant experience with hydrides, specifically $ZrH_{1.6}$, in irradiation environments and in transmutation analysis. Since the trends of hydride concentrations are so similar, only one such moderator, $ZrH_{1.6}$, will be followed in the rest of the study. Even though the maximum temperature that it can withstand is ~800C, this has shown to be acceptable for blanket assemblies (Bays, 2011).

One negative consequence of the hydrides is the reactivity penalty that we can infer from the k-infinity of the material at BOL. The consequence of this is an increase in required driver fuel enrichment for a given cycle length. Most of the other moderators behave very similarly by providing less of a reactivity penalty at the BOL but lesser gains in transmutation efficiency and breeding potential. Out of the non-hydrides analyzed, BeO was selected since it was representative of most oxides and fluorides and due to its current use in irradiation environments.

From the previous results, a list of selected cases will be further analyzed using the metrics definitions of Chapter 3 over 9 cycles (DPA limit of blankets). Tables 4-11 through 4-19 show the impact of moderator addition using the five FOMs. Cases are presented where the moderator volume fraction in the blanket assemblies is varied from 5% to 25% for both $ZrH_{1.6}$ and BeO. The minor actinide content is kept constant at 4% by volume.

Table 4-11: FOM_1 for different Moderator content (4%MA)

	5% $ZrH_{1.6}$	10% $ZrH_{1.6}$	15% $ZrH_{1.6}$	20% $ZrH_{1.6}$	25% $ZrH_{1.6}$
FOM	0.390	0.393	0.397	0.401	0.405

	5% BeO	10%BeO	20% BeO	25% BeO
FOM	0.389	0.392	0.400	0.404

Table 4-11 indicates that increasing moderation provides greater proliferation resistance for both $ZrH_{1.6}$ and BeO, and that moderators behave quite similarly with regards to FOM_1 for different moderator contents. Individual components of this FOM are listed in Table 4-12.

Table 4-12: $u(x_1)$ - $u(x_6)$ values

	5% $ZrH_{1.6}$	10% $ZrH_{1.6}$	15% $ZrH_{1.6}$	20% $ZrH_{1.6}$	25% $ZrH_{1.6}$
u_1	0.15	0.15	0.15	0.15	0.15
u_2	0.257	0.265	0.273	0.282	0.292
u_3	0.140	0.149	0.159	0.169	0.181
u_4	0.742	0.746	0.751	0.756	0.762
u_5	0.062	0.064	0.066	0.069	0.072
u_6	1.0	1.0	1.0	1.0	1.0

	5% BeO	10% BeO	20% BeO	25% BeO
u_1	0.15	0.15	0.15	0.15
u_2	0.255	0.261	0.273	0.279
u_3	0.141	0.153	0.180	0.196
u_4	0.739	0.740	0.745	0.748
u_5	0.063	0.067	0.075	0.081
u_6	1.0	1.0	1.0	1.0

From Table 4-12, one can observe that moderation increases attributes u_2 to u_5 which deal with the Pu heating rate, the even Pu fraction, the concentration of fissile material and the radiation dose, respectively. Table 4-13 presents the FOM_2 results.

Table 4-13: FOM_2 for different Moderator content (4%MA)

	5% $ZrH_{1.6}$	10% $ZrH_{1.6}$	15% $ZrH_{1.6}$	20% $ZrH_{1.6}$	25% $ZrH_{1.6}$
FOM	0.410	0.397	0.389	0.383	0.379

	5% BeO	10%BeO	20% BeO	25% BeO
FOM	0.422	0.413	0.394	0.386

Once again, the trend of FOM_2 is the reverse of that of FOM_1 due to the counter-acting attributes. It should also be noted that BeO performs better than the $ZrH_{1.6}$. Details are provided in Table 4-14.

Table 4-14: x_7 -x_{12} values

	5% $ZrH_{1.6}$	10% $ZrH_{1.6}$	15% $ZrH_{1.6}$	20% $ZrH_{1.6}$	25% $ZrH_{1.6}$
x_7	3.40E+07	3.50E+07	3.61E+07	3.72E+07	3.84E+07
x_8	1.66E+07	1.68E+07	1.69E+07	1.69E+07	1.69E+07
x_9	8.87E+08	9.09E+08	9.31E+08	9.55E+08	9.80E+08
x_{10}	0.0668	0.0678	0.0687	0.0696	0.0706
x_{11}	0.211	0.188	0.173	0.162	0.153
x_{12}	1329	1338	1342	1343	1341

	5% BeO	10% BeO	20% BeO	25% BeO
x_7	3.46E+07	3.62E+07	3.96E+07	4.15E+07
x_8	1.68E+07	1.73E+07	1.82E+07	1.87E+07
x_9	9.01E+08	9.38E+08	1.02E+09	1.06E+09
x_{10}	0.0678	0.0699	0.074	0.0762
x_{11}	0.239	0.228	0.207	0.197
x_{12}	1344	1377	1445	1481

Adding moderation, for the same amount of MA, increases the heat rate (x_{12}), radiotoxicity (x_8, x_9) and neutron emission rate (x_7). Cross-sections tend to increase as neutrons are moderated, thus the added $ZrH_{1.6}$ promotes more absorption moving nuclides up the actinide chain. The added moderation also has a negative impact on the k_{inf} of the blanket (x_{11}). Many of the actinides (especially U-238) have threshold fission reactions, thus the added moderation reduces the neutron energy below the fission cutoff. The mass breeding ratio (x_{10}) increases slightly since the slower neutrons favor absorption in U-238. With the current weight scheme, the overall effect is to favor a lower moderation content with regard to FOM_3. This conclusion would only be changed if we considerably favor the breeding ratio gain.

The combination of all 12 attributes to form FOM_3 is presented in Table 4-15 using equal weighting of the 6 major attributes as described in Table 3-5.

Table 4-15: FOM_3 for different Moderator content (4%MA)

	5% $ZrH_{1.6}$	10% $ZrH_{1.6}$	15% $ZrH_{1.6}$	20% $ZrH_{1.6}$	25% $ZrH_{1.6}$
FOM_3	0.406	0.397	0.390	0.386	0.383

	5% BeO	10%BeO	20% BeO	25% BeO
FOM_3	0.417	0.410	0.396	0.389

Once again this metric is of little value and can be biased in any preferred direction. The proposed approach is thus to use FOM_4 for proliferation as shown in Table 4-16.

Table 4-16: FOM_4 for different Moderator content (4%MA)

	5% $ZrH_{1.6}$	10% $ZrH_{1.6}$	15% $ZrH_{1.6}$	20% $ZrH_{1.6}$	25% $ZrH_{1.6}$
FOM_4	0.00348	0.000935	-0.00456	-0.0127	-0.0235

	5% BeO	10%BeO	20% BeO	25% BeO
FOM_4	-0.047268	-0.095772	-0.185600	-0.226060

The results in table 4.16 indicate that adding $ZrH_{1.6}$ moderation to the 4% MA will at first have a negative impact on the proliferation metric but FOM_4 eventually returns negative since increased moderation increases the absorption in U238 to create more minor actinides. The small addition of moderation changes the isotopic composition enough to reduce the critical mass from 1786kg (no moderation) to 1672kg (5% moderation). Added moderation has little impact on the critical mass (1717kg at 25% moderation), but does increase dose rate and heat rate which are both important factors in the Bathke criteria. The BeO on the other hand improves the proliferation for any moderation level. The combined FOM is presented in Table 4-17.

Table 4-17: FOM_5 for different Moderator content (4%MA)

	5% $ZrH_{1.6}$	10% $ZrH_{1.6}$	15% $ZrH_{1.6}$	20% $ZrH_{1.6}$	25% $ZrH_{1.6}$
FOM_5	-	-	0.389	0.383	0.379

	5% BeO	10%BeO	20% BeO	25% BeO
FOM_5	0.422	0.413	0.394	0.386

Tables 4-18 and 4-19 summarize the MA and moderator content variations for FOM_4 and FOM_5 for $ZrH_{1.6}$.

Table 4-18: FOM_4 – Minor Actinide and $ZrH_{1.6}$ variations

	1% MARef	1%MA5%$ZrH_{1.6}$	1%MA10% $ZrH_{1.6}$	1%MA15%$ZrH_{1.6}$	1%MA25%$ZrH_{1.6}$
FOM_4	0.281	0.324	0.321	0.315	0.294

	2% MARef	2%MA5%$ZrH_{1.6}$	2%MA10% $ZrH_{1.6}$	2%MA15%$ZrH_{1.6}$	2%MA25%$ZrH_{1.6}$
FOM_4	0.133	0.177	0.174	0.169	0.149

	3% MARef	3%MA5%$ZrH_{1.6}$	3%MA10% $ZrH_{1.6}$	3%MA15%$ZrH_{1.6}$	3%MA25%$ZrH_{1.6}$
FOM_4	0.0339	0.0771	0.0745	0.0688	0.0495

	4% MARef	4%MA5%$ZrH_{1.6}$	4%MA10% $ZrH_{1.6}$	4%MA15%$ZrH_{1.6}$	4%MA25%$ZrH_{1.6}$
FOM_4	-0.0384	0.00348	0.000935	-0.00456	-0.0235

	5% MARef	5%MA5%$ZrH_{1.6}$	5%MA10% $ZrH_{1.6}$	5%MA15%$ZrH_{1.6}$	5%MA25%$ZrH_{1.6}$
FOM_4	-0.0938	-0.0535	-0.0559	-0.0628	-0.0793

	6% MARef	6%MA5%$ZrH_{1.6}$	6%MA10% $ZrH_{1.6}$	6%MA15%$ZrH_{1.6}$	6%MA25%$ZrH_{1.6}$
FOM_4	-0.138	-0.0987	-0.101	-0.106	-0.123

	7% MARef	7%MA5%$ZrH_{1.6}$	7%MA10% $ZrH_{1.6}$	7%MA15%$ZrH_{1.6}$	7%MA25%$ZrH_{1.6}$
FOM_4	-0.173	-0.135	-0.137	-0.148	-0.158

Table 4-19: FOM_5– Minor Actinide and $ZrH_{1.6}$ variations

	4% MARef	4%MA5%$ZrH_{1.6}$	4%MA10% $ZrH_{1.6}$	4%MA15%$ZrH_{1.6}$	4%MA25%$ZrH_{1.6}$
FOM_5	0.435	-	-	0.389	0.379

	5% MARef	5%MA5%$ZrH_{1.6}$	5%MA10% $ZrH_{1.6}$	5%MA15%$ZrH_{1.6}$	5%MA25%$ZrH_{1.6}$
FOM_5	0.433	0.407	0.394	0.386	0.374

These results indicate that $ZrH_{1.6}$ moderation does not provide added benefit to the addition of MA and a somewhat similar trend can be observed with BeO (FOM5 with 4%.MARef is still higher than any value in Table 4.15). However, we will still carry forward some analysis with moderation to verify that these trends hold in other situations. Additionally, FOM_1 and FOM_3 will no further be used since they provide little added value in differentiating between cases. It should be noted that we could use FOM_1 as a "yes or no" binary decision like done in FOM_4. From the results in table 4.1 and 4.11, an equivalence between FOM_1 and FOM_4 would be at around FOM_1 equal to 0.393 (any value above this would be considered proliferation resistant).

4.2.1 MCNP Verification

MCNP analysis was performed with varying levels of moderation in the blankets to assess the accuracy limits of ERANOS. A fresh core was depleted for 1500 days with 1 row of blankets which contains 10% minor actinides (Am, Cm, Bk, Cf) and varying levels of $ZrH_{1.6}$ (0%, 10%, 30% and 50%) and depleted uranium. In all cases the reactivity versus time behaved in the same way with eigenvalue differences in the 100pcm range for the first 1200 days and slightly larger at 1500 days (300pcm). These results are to be expected since the blankets contribute very little to the overall neutron balance and the discrepancies over longer lifetime can be partially attributed to the lumped fission product representation. For further comparison, the ratio of selected individual isotopes between end-of-cycle (EOC) and beginning-of-cycle (BOC) were compared. The selected isotopes were Am241, Am243 and Cm244 since they represent the largest initial minor actinide mass at BOC. Table 4-20 illustrates our findings.

Table 4-20: Ratio of Selected Isotopes

	ERANOS	MCNP	Relative Error (%)
0% $ZrH_{1.6}$			
Am241	0.789	0.792	-0.49
Am243	0.822	0.825	-0.44
Cm244	1.124	1.120	0.35
10% $ZrH_{1.6}$			
Am241	0.716	0.718	-0.30
Am243	0.751	0.753	-0.20
Cm244	1.245	1.249	-0.27
30% $ZrH_{1.6}$			
Am241	0.600	0.593	1.04
Am243	0.643	0.634	1.44
Cm244	1.412	1.451	-2.67
50% $ZrH_{1.6}$			
Am241	0.506	0.409	23.89
Am243	0.568	0.493	15.33
Cm244	1.514	1.706	-11.25

From the results we see very good agreement in the mass ratios up to 30% ZrH in the targets with a substantial deterioration when going to 50%. This deterioration of accuracy with well-moderated blankets must be considered when optimal cases are selected. A full core MCNP

model over multiple cycles is an impractical proposition, but the differences reported here will be considered when evaluating the trends of our study.

4.3 Separation study

The separation of Cm, Bk and Cf was performed for the reference case and a few down-selected cases from the moderation and minor actinide loading study of sections 4.1 and 4.2. Based on previous analysis using the FOM_4 Bathke proliferation metric, it was determined that a minimum of 4% MA was necessary to meet our threshold. A case with 4% MA was run as well as a case with 4% MA and 5% $ZrH_{1.6}$ moderator. Tables 4.21 to 4.28 show the main results for these cases. It should be noted that the radiotoxicity, neutron production, photon heat rate and heat rate are taken at the end of 9 cycles when all MA are re-combined. It should also be noted that the total amount of minor actinide (for a given MA volume) is fixed between the separated and un-separated cases. This means that the un-separated case will have the same mass of Am as the separated case. It also means that the freed volume was compensated for by additional depleted uranium. The out of core decay is performed alongside the core depletions ensuring that both the out of core material and in-core material are decayed/transmuted over the same total time.

Due to the added complexity of the fuel cycle scheme the analysis between the Am only case and the case with the full MA vector in the blanket, the results of section 4.3 are only for the minor actinides at discharge. Thus, the results of the reference case with 4% MA will differ between this section and previous ones, however the comparison between Am only and all MA remains valid.

Table 4-21: FOM_2 for Am only case (9 cycles)

	4% MARef	4%MA5%$ZrH_{1.6}$
FOM_2	0.556	0.539

Table 4-22: FOM_2 for All Am case (9 cycles)

	4% MARef	4%MA5%$ZrH_{1.6}$
FOM_2	0.569	0.550

When comparing the results between Table 4-21 and Table 4-22, we notice that the separated case (4.19) has a lower FOM than the case with all minor actinides. More detailed results are presented in Tables 4-23 and 4-24.

Table 4-23: x_7 -x_{12} values for AM only cases

	4% MARef	4%MA5%ZrH$_{1.6}$
x_7	6.04E+07	5.72E+07
x_8	3.05E+06	2.89E+06
x_9	1.46E+09	1.38E+09
x_{10}	0.0668	0.0667
x_{11}	0.230	0.193
x_{12}	38.7	36.3

Table 4-24: x_7 -x_{12} values for All MA cases

	4% MARef	4%MA5%ZrH$_{1.6}$
x_7	5.18E+07	4.89E+07
x_8	2.74E+06	2.59E+06
x_9	1.31E+09	1.24E+09
x_{10}	0.0671	0.0669
x_{11}	0.251	0.211
x_{12}	36.5	34.1

Tables 4-23 and 4-24 show that the separated cases provide no benefit with regard to x_7-x_{12} when combining all minor actinides. The separated cases produce more neutrons, are more toxic and produce more heat. They also produce less fissile material.

Tables 4-25 and 4-26 show the FOM$_4$ Bathke proliferation metric for the separated and un-separated cases for the blanket material at the end of 9 cycles. It should be noted that the proliferation metric was calculated without recombining the streams. Thus, proliferation is measured on the actual assembly.

Table 4-25: FOM$_4$ for Am only case (9 cycles)

	4% MARef	4%MA5%ZrH$_{1.6}$
FOM	-0.0250	0.0184

Table 4-26: FOM$_4$ for All Am case (9 cycles)

	4% MARef	4%MA5%ZrH$_{1.6}$
FOM	-0.0384	0.00348

One thing to note from Tables 4-25 and 4-26 is that separating the MA and adding moderation actually improves the proliferation metric.

Table 4-27: FOM_5 for Am only case (9 cycles)

	4% MARef	4%MA5%$ZrH_{1.6}$
FOM	0.556	-

Table 4-28: FOM_5 for All Am case (9 cycles)

	4% MARef	4%MA5%$ZrH_{1.6}$
FOM	0.569	-

4.4 Natural, Depleted or Recycled Uranium study

All previous analyses were performed using depleted uranium with a tails enrichment of 0.3%. A few selected cases were re-run using natural uranium and recycled uranium from an LWR (51GWd/MTHM burnup). Table 4-29 presents the uranium vectors used.

Table 4-29: Uranium vectors

	Depleted U (%)	Natural U (%)	Recycled U (%)
U-232	-	-	Negligible
U-233	-	-	Negligible
U-234	-	-	0.0204
U-235	0.3	0.72	0.8768
U-236	-	-	0.6444
U-238	99.7	99.28	98.4594

Table 4-30 presents important parameters for 4% minor actinide loading with no moderator for one 500-day cycle.

Table 4-30: Results for NU, DU and RU – 4%MA

	DU	NU	RU
Radiotoxicity ingestion (Sv/kg)	2.57E+06	2.62E+06	2.63E+06
Radiotoxicity inhalation (Sv/kg)	8.56E+08	8.56E+08	8.55E+08
Neutron Production Rate (N/s/kg)	3.66E+07	3.66E+07	3.66E+07
Photon Heat Rate (W/kg)	26.8	28.4	28.9
Heat Rate (W/kg)	111.2	115.0	116.1
Fissile Pu EOL/U238 BOL	0.0108	0.0108	0.0108
k-infinity BOL	0.251	0.307	0.329

From this table we notice no significant change in radiotoxicity, neutron production and breeding gain. However, we do notice small variations in heat rate and photon heat rate as well as the BOL k-infinity. These changes are all correlated since the highest fissile content at BOL is that of RU followed by NU and DU. Thus a smaller reactivity penalty will lead to more fission events in the fissile material increasing the heat load generated by fission products. This can be confirmed by examining at the heat rate produced by minor actinides only presented in Table 4-31.

Table 4-31: Results for NU, DU and RU (MA – only) – 4% MA

	DU	NU	RU
Photon Heat Rate (W/kg)	26.8	28.4	28.9
Photon Heat Rate – MA only (W/kg)	12.2	12.2	12.1
Heat Rate (W/kg)	111.2	115.0	116.1
Heat Rate – MA only (W/kg)	73.9	74.0	73.7

These results indicate that the significant changes observed in heat rate are attributed to the fission products in the blanket and not the minor actinides. The implication of these results indicate that DU, NU and RU will perform quite similarly in a blanket assembly, but RU and NU would provide a bit more initial power to the core. Consequently the heat rate would be slightly higher in the first cycle, but mostly from the much shorter lived fission products. The long term implications of using DU, NU or RU indicate no real preference and would be left to cost considerations and ease of fabrication. This is reflected in Table 4-32 where FOM_2 is larger for RU due to its higher k_{inf} at beginning-of-cycle.

Table 4-32: FOM_2 for DU, NU and RU (1 cycle)

	DU	NU	RU
FOM_2	0.554	0.582	0.592

This significant increase in k-infinity at the beginning-of-life can be seen more readily in the reactivity profiles. Table 4-33 provides the reactivity change over the first cycle for a given driver core enrichment.

Table 4-33: Reactivity (pcm) as a function of time

	0 Days	125 Days	250 Days	375 Days	500 Days
DU	1210	996	897	760	583
NU	1254	1038	939	802	625
RU	1270	1054	954	816	640

From this table we can see that natural uranium maintains its reactivity level roughly 40pcm higher than the depleted case and that the recycled uranium is about 60 pcm higher. This added reactivity can be translated into days by dividing the reactivity swing by the total cycle length.

This indicates that the natural uranium case could run for an extra 30 days and the recycled uranium case 45 days longer than the depleted uranium case. These gains could also be translated to a reduction in driver core enrichment for a fixed cycle length.

Another aspect to analyze is the impact on proliferation as measured by FOM4 in Table 4-34 over 9 cycles. For all three cases, the Bathke FOM is satisfied.

Table 4-34: FOM_4 for DU, NU and RU (9 cycles)

	DU	NU	RU
FOM_4	-0.038397	-0.01943	-0.015524

4.5 Near equilibrium cases

All of the results presented so far were performed on the basis of a startup core in which blanket assemblies remained for the maximum achievable DPA. A single blanket pass implied approximately 15 years of residence time. In this section we will look at the impact on the blankets as we approach driver fuel equilibrium by inserting a fresh blanket for an additional 9 and 18 cycles. Only the 4% MA case with no separation was analyzed and results are presented in Table 4.33.

Table 4-35: Blanket analysis for 9, 18 and 27 cycles

	9 cycles	18 cycles	27 cycles
x_7	3.5E+07	3.5E+07	3.5E+07
x_8	1.6E+07	1.6E+07	1.6E+07
x_9	9.1E+08	9.1E+08	9.1E+08
x_{10}	0.0670	0.0665	0.0660
x_{11}	0.251	0.251	0.251
x_{12}	1280	1279	1244
FOM_4	-0.0384	-0.0352	-0.049

x_7: Neutron emission rate (neutrons/kg/s)
x_8: Radiotoxicity – ingestion (Sv/kg)
x_9: Radiotoxicity – inhalation (Sv/kg)
x_{10}: Ratio of Pu-239 at EOC to U-238 at BOC
x_{11}: k_{inf} of blank et at EOC
x_{12}: Heat rate (W/kg)

From these results we can notice very small differences in the blanket performance. We note a small decrease in the Pu-239 production (x_{10}) and a small decrease in heat rate (x_{12}). The

proliferation metric is also changed by a very small amount because of the change in the plutonium vector. Small changes in the driver fuel composition as we near equilibrium slightly change the spectrum that the blankets are seeing, but as can be seen in Table 4.33, the effects of these changes are minimal.

4.6 Summary

In this section, it was determined that to meet the prescribed proliferation resistance criteria, a minimum of 4% MA (by volume) was needed in the blanket assemblies. However, increasing the amount of MA past 4% became detrimental to the combination of the other 5 attributes, mainly impacting the radiotoxicity, fuel fabrication and transportation.

The addition of moderation by itself did not provide any means of ameliorating proliferation issues, but the combination of BeO and minor actinides shows a possible reduction in the amount of minor actinides needed. Using our defined metric, it was determined that moderation provided no immediate benefit. Other studies have shown however, that for long term concerns, the situation might change (Massie, 2012).

Separation of the Cm/Bk/Cf vector from the Am was also studied and also shown to give no improvement over our reference case where all minor actinides are recycled simultaneously. Also, as expected, varying the uranium composition vector among natural, depleted and recycled had very little impact on our metric, thus the choice of uranium vector would be mostly left to cost and initial fabrication considerations.

5 Sensitivity Study of FOM

In this section, we will analyze the sensitivity of the weights on the results reported in the previous section. There are three different weight assessment methods: direct importance weights, trade-off weights, and swing weights (Fischer, 2002). For this study we have used direct importance weights. Numerical values have been assigned with the sole purpose of signifying their relative importance.

5.1 FOM 1 Sensitivity Analysis

For the FOM_1, the six proliferation resistance attributes have been assigned the same weight as in Charlton's paper. However, they have been renormalized since we have only included six of his attributes that address material properties. As mentioned earlier in this study, the following table summarizes the weights assigned for each Charlton attribute.

Table 5-1: Weights of FOM_1

Attribute	Charlton weights	Renormalized weights
u_1	0.10	0.2222
u_2	0.05	0.1111
u_3	0.06	0.1333
u_4	0.10	0.2222
u_5	0.08	0.1778
u_6	0.06	0.1333

Using the renormalized weights, FOM_1 for different MA contents is as follows.

Table 5-2: FOM_1 for different MA contents (Note: based on Charlton's normalized weights)

	1% MA	2%MA	3%MA	4%MA	5%MA	6%MA	7%MA	8%MA	10%MA
FOM	0.356	0.370	0.382	0.393	0.404	0.415	0.425	0.435	0.454

It is obvious that more minor actinide concentration is preferred from the proliferation resistance perspective. The utility function value (u(x)) for Department of Energy attractiveness level (u_1) and the size and the weight of a fuel assembly (u_6) remain the same for all minor actinide content. The utility function value for heating rate from plutonium (u_2), weight fraction of even plutonium isotopes (u_3), and radiation dose rate (u_5) increase with minor actinide addition. However, the utility function value for concentration of fissile materials (u_4) decreases slightly with minor actinide addition. We would like to see how the overall FOM_1 would change if u_4's Charlton weight were to be doubled. The table below lists the new weights.

Table 5-3: Weights of FOM_1 with doubled u_4

Attribute	New weights	Renormalized weights
u_1	0.10	0.1818
u_2	0.05	0.0909
u_3	0.06	0.1090
u_4	0.20	0.3636
u_5	0.08	0.1454
u_6	0.06	0.1090

Even after doubling the Charlton weight for concentration of fissile materials (u_4), we see the same result as earlier that is the increase in minor actinide content is preferable from the proliferation standpoint. The new FOM_1 with doubled Charlton weight of u_4 is presented in the table below.

Table 5-4: FOM_1 for different MA contents (u_4 weight doubled)

	1% MA	2%MA	3%MA	4%MA	5%MA	6%MA	7%MA	8%MA	10%MA
FOM_1	0.428	0.439	0.448	0.457	0.466	0.475	0.483	0.491	0.507

Therefore, for FOM_1, it does not make a significant difference whether the Charlton weights of individual attributes are changed or not. Regardless of the variation in their weights, the same pattern occurs that favors the addition of greater amounts of minor actinides.

In the case of moderation, the utility function value (u(x)) for Department of Energy attractiveness level (u_1) and the size and the weight of a fuel assembly (u_6) remain the same for all moderator content whereas the utility function value for heating rate from plutonium (u_2), weight fraction of even plutonium isotopes (u_3), concentration of fissile materials (u_4), and radiation dose rate (u_5) all increase with moderator addition. Therefore variation in their assigned weights is not expected to make any significant change on the overall FOM_1.

5.2 FOM_2/FOM_5 Sensitivity Analysis

This figure of merit takes into account fuel fabrication (u_7), radiotoxicity ((u_8) and $u_{(9)}$), fissile material production (u_{10}), reactivity penalty (u_{11}) and transportation constraints (u_{12}). In this study, equal weights were given to all attributes except for the two radiotoxicity measures (ingestion and inhalation) which were combined as one.

The following table lists the weights assigned to each attribute.

Table 5-5: Weights for FOM_2

	Weights
w_7	0.2
w_8	0.1
w_9	0.1
w_{10}	0.2
w_{11}	0.2
w_{12}	0.2

Table for FOM_2 for different MA contents based on the weights from above

Table 5-6: FOM_2 for different MA contents

	1% MA	2%MA	3%MA	4%MA	5%MA	6%MA	7%MA	8%MA	10%MA
FOM_2	0.437	0.437	0.436	0.435	0.434	0.432	0.431	0.429	0.426

As we increase the MA content of the blanket, the neutron emission rate, radiotoxicity, and heat rate increase significantly which has a negative impact on the FOM. However, the kinf increases with more MA which has a positive impact on the performance of the core and the FOM. The mass based breeding ratio is basically the same for all cases with a slight decrease with increase in MA content. Overall, we observe that neutron emission rate, radiotoxicity, and heat rate dominate the FOM in order to favor less MA inclusion in the blanket. Therefore, we will increase the weight of k_{inf} (u_{11}) by reducing the weight of mass breeding ratio (u_{10}) to see how it affects the overall FOM.

The table lists the new weights assigned to each attribute.

Table 5-7: Weights for FOM_2 (Increased weight for w_{11})

	Weights
w_7	0.2
w_8	0.1
w_9	0.1
w_{10}	0.1
w_{11}	0.3
w_{12}	0.2

The following table provides FOM_2 values for different MA contents when the weight of k_{inf} is increased to 0.3 from 0.2 by reducing the weight of mass breeding ratio to 0.1 from 0.2.

Table 5-8: FOM_2 for different MA contents (increase in weight of u_{11})

	1% MA	2%MA	3%MA	4%MA	5%MA	6%MA	7%MA	8%MA	10%MA
FOM_2	0.461	0.463	0.465	0.467	0.468	0.470	0.471	0.472	0.473

Now the trend reverses to favor ever so slightly the addition of MA in the blanket if more weight is assigned to k_{inf}. Therefore, it is clear how sensitive this FOM is to the assigned weights.

In the study of moderated cases, less moderation was preferred as shown in the table below when equal weights were given to all attributes except for the two radiotoxicity measures (ingestion and inhalation) which were combined as one.

Table 5-9: FOM_2 for different Moderator content (4%MA)

	5% $ZrH_{1.6}$	10% $ZrH_{1.6}$	15% $ZrH_{1.6}$	20% $ZrH_{1.6}$	25% $ZrH_{1.6}$
FOM_2	0.410	0.397	0.389	0.383	0.379

As the moderation is increased, neutron emission rate, radiotoxicity and heat rate increase as well which impacts the FOM negatively. While k_{inf} also decreases with increase in moderation which negatively affects the FOM, the mass breeding ratio increases with increase in moderation. This is the only attribute that positively affects the FOM. Therefore, we will increase the weight of mass breeding ratio significantly by reducing the weight of neutron emission rate, k_{inf} and heat rate to see the effect of moderation on FOM_2.

Table 5-10: Weights for FOM_2 (Increased weight for w_{10})

	Weights
w_7	0.1
w_8	0.1
w_9	0.1
w_{10}	0.5
w_{11}	0.1
w_{12}	0.1

The following table provides FOM_2 values for different moderator contents when the weight of mass breeding ratio is increased to 0.5 from 0.2 by reducing the weight of each neutron emission rate, k_{inf}, and heat rate to 0.1 from 0.2.

Table 5-11: FOM_2 for different Moderator content (4%MA) (increase in weight of u_{10})

	5% $ZrH_{1.6}$	10% $ZrH_{1.6}$	15% $ZrH_{1.6}$	20% $ZrH_{1.6}$	25% $ZrH_{1.6}$
FOM_2	0.316	0.310	0.306	0.303	0.301

In this case, even when the weight of mass breeding ratio is significantly increased, we still see the same pattern of preference for less moderation. The increase in neutron emission rate, radiotoxicity, and heat rate along with decrease in k_{inf} with increase in moderation

overwhelmingly favors less moderation even when the weight of mass breeding ratio is significantly increased.

5.3 Summary

When the sensitivity analysis was performed on FOM_1 that includes Charlton's six proliferation attributes, there was no significant change in the result. Except for the utility function value for concentration of fissile materials (u_4) which decreases with minor actinide addition, the utility function values for heating rate from plutonium (u_2), weight fraction of even plutonium isotopes (u_3), and radiation dose rate (u_5) increase with minor actinide addition. The utility function value (u(x)) for Department of Energy attractiveness level (u_1) and the size and the weight of a fuel assembly (u_6) remain the same for all minor actinide content. Therefore, only u_4 could impact our conclusion and even when its initial assigned weight was doubled, the outcome remained the same. That is: more actinide content is preferable from a proliferation standpoint based on Charlton's definition. Regardless of how the weights are assigned to Charlton's six attributes, the fact that more minor actinide inclusion is better from proliferation standpoint remains the same. For the moderated cases, the utility function value (u(x)) for Department of Energy attractiveness level (u_1) and the size and the weight of a fuel assembly (u_6) remain the same whereas the utility function value for heating rate from plutonium (u_2), weight fraction of even plutonium isotopes (u_3), concentration of fissile materials (u_4), and radiation dose rate (u_5) increase with moderator addition. Therefore, change in their weights will not impact the observed trends on FOM_1.

The results of chapter 4 show that less minor actinide and moderation is favorable for FOM_2 which takes into account fuel fabrication (u_7), radiotoxicity ((u_8) and $u(_9)$), fissile material production (u_{10}), reactivity penalty (u_{11}) and transportation constraints (u_{12}). With more minor actinide content, neutron emission rate, radiotoxicity, and heat rate significantly increase which is not desirable from the core performance, fuel fabrication and transportation perspectives. However, k_{inf} increases with increase in MA content which is absolutely desirable for the performance of the core. The fissile material production (mass breeding ratio) slightly decreases with increase in MA content. Using an equal weighing scheme, the neutron emission rate, radiotoxicity, and heat rate dominated FOM_2 over k_{inf} to prefer less minor actinide. When the sensitivity analysis was performed by increasing the weight of k_{inf} while reducing the weight of fissile material production (mass breeding ratio), a slight preference for MA inclusion was observed. This shows some sensitivity in the conclusions directly linked to the importance of the reactivity penalty. Further analysis would be needed to quantify the true cost of such reactivity penalty in order to justify having more MA than necessary to satisfy the proliferation metric.

Less moderation was preferred in the moderated cases when equal weights were assigned. With an increase in moderation, increase in neutron emission rate, radiotoxicity and heat rate along with decrease in k_{inf} was observed which negatively impacted FOM_2. The only attribute that positively contributed to FOM_2 was the mass breeding ratio. Therefore, for sensitivity analysis,

the weight of the mass breeding ratio was significantly increased (to 50% of total) by reducing the weight of neutron emission rate, k_{inf} and heat rate. Nonetheless, no change was noticed in the previous conclusion of preference for less moderation. Thus, unless breeding is the only viable criteria, moderation provides no benefit.

6 Conclusion

Due to perceived proliferation risks, current fast reactor designs have avoided the use of uranium blankets. While reducing the amount of plutonium produced, omission of fertile blankets also restrains the reactor design space and its advantages over blanket-free cores. The advantages of fertile blankets include: (Barre, 1977):

1. Provide large breeding gains, thus reducing supply constraints for operating fast reactors and the possibility of providing fissile material for startup cores.
2. Provide neutron reflection to the core, thus reducing the fissile inventory required in the driver fuel for criticality.
3. Flatten the power profile of the core.
4. Contribute to shielding against neutrons and photons.
5. Produce some power.

This study investigated many blanket options that would mitigate proliferation concern while minimizing negative fuel cycle impact. To do so, a multi-variable metric was developed that combines 6 attributes: proliferation resistance, fuel fabrication, radiotoxicity, breeding gain, reactivity penalty and transportation. The final version of the metric consists of using a yes or no decision on the proliferation criteria proposed by Bathke (for advanced technological nations). The remaining 5 attributes are scaled between 0 and 1, and are currently equally weighted. The weights can be determined based on the perspective/preference of each stakeholder. Sensitivity studies have shown that conclusions may change only in some extreme scenarios. The fuel transportation attribute was estimated by looking at the neutron emission of a blanket assembly and the required amount of shielding needed to protect workers. The radiotoxicity attribute was defined using the ICRP72 values for each isotope with limits set by the minimum and maximum values observed in our cases. The breeding gain was evaluated by a ratio of fissile Pu-239 at end-of-life to initial U-238 in the blankets. The reactivity penalty was simply related to the k-infinity of the blanket material at end-of-life. The transportation attribute was estimated using the heat rate of the fuel which is a limiting factor for transportation without active cooling. These definitions are by no means exhaustive, nor unique, but provide a simple way to compare the many cases at hand. Using a similar approach, a broader set of attributes could be defined for studies comparing different recycling strategies.

For our analysis, a 2400MWth sodium cooled core was considered. One row of blanket was added radially. Metal fuel composed of depleted uranium, zirconium and Np/Pu from light water reactor used fuel was used for the driver. The following options were considered for our blanket design:

1) Minor actinide addition (Am/Cm/Bk/Cf from light water reactor used fuel)
2) Addition of moderator (Hydrides, Oxides, Carbides, …)

- Homogenously mixed with the fuel
- Heterogeneously mixed in the assembly

3) Am-only in the blanket assemblies
4) Uranium vector composition (natural, depleted, recycled)

It was determined that to meet the prescribed proliferation resistance criteria, a minimum of 4% MA (by volume) was needed in the blanket assemblies. However, increasing the amount of MA past 4% became detrimental to the combination of the other 5 attributes, mainly impacting the radiotoxicity, fuel fabrication and transportation.

The addition of moderation by itself did not provide any means of dissipating proliferation issues. Additionally, in combination with minor actinide addition, the spectral shift in the blanket did provide some volume fraction for which 4% MA was not sufficient. In the cases studied, it was determined that $ZrH_{1.6}$ and BeO were most promising. They both provided some reduction in MA concentration but at the expense of the increased radiotoxicity of the end product. Using our defined metric, it was determined that moderation provided no immediate benefit. Other studies have shown however, that for long term concerns, the situation might change (Massie, 2012). It should also be noted that the homogeneous or heterogeneous addition of moderators has minimal impact on such scoping studies.

Separation of the Cm/Bk/Cf vector from the Am was also studied. The blankets were composed of Am while the remaining Cm/Bk/Cf was left to decay in storage. The metric was then applied to the combined streams for all attributes except proliferation. The separated case performed worst in all cases examined.

Also, as expected, varying the uranium composition vector from natural, depleted and recycled had very little impact on our metric, thus the choice of uranium vector would be mostly left to cost and initial fabrication considerations. It should however be noted that the k-infinity at beginning-of-life was obviously higher for the recycled and natural cases. Looking at the reactivity over the first cycle indicates that NU provides an additional ~40pcm over DU while RU provides ~60pcm, which could provide respectively 30 and 45 extra days of operation or a reduction in driver core enrichment.

Sensitivity of the weights was analyzed for FOM_2 which indicated that only in extreme cases would the conclusion be affected. The data trends indicate that only if the reactivity penalty was considered of utmost importance would it be favorable to increase the MA content past the proliferation threshold. Additionally, moderation would be favored if the breeding gain were the only parameter being considered.

For this project, a post-processing GUI was also developed. The GUI provides easy evaluation of many of the parameters needed for the metrics presented. It also allows processing of 1 and 2 stream isotopic information in the VISION input format.

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8 Appendix I: Python GUI

The ERANOS Processing Tool (EPT), developed in Python, was created to parse the large output files of the multiple runs. In doing so, verification can be performed on the code itself which provides much greater consistency than parsing every single file individually. Version control was provided using git and was stored at www.github.com.

Installation of the GUI requires Python version 2.6 or higher which can be found at http://www.python.org/ and the PyQT library.

The code is composed of 8 python files and requires 1 input file. All other functions are provided by the GUI.

- EPT.pyw: Main driver for the GUI, defines all the options available from dropdown menus.
- Eranos.py: Main driver for the processing code. Reads mass information for all cycles. The code will parse the file using user-defined keywords and others found in the ERANOS output.
    - Driver fuel must be defined as FUEL1, FUEL2, …
    - Blanket assemblies must be defined as BLANK
- Parameters.py: Contains all isotope parameters to compute decay heat, photon heat, neutron emission rate, external dose and radiotoxicity. The values were taken from an INL spreadsheet developed by Steve Piet and the data was taken from references [Ref]. This file also contains the pseudo-fission products distribution provided by CEA for both the sfp and fp pseudo-isotopes. [Ref]
- Isotope.py: Defines an isotope class for element names, mass, charge number, ORIGEN ID number, …
- Material.py: Defines a material class that computes all parameters related to the fuel and blanket materials.
- Vision.py: Processes the list of isotopes in the prescribed vision format. This requires the input file isoprocess.txt which describes how certain isotopes must be contained. The format of this file is identical to the one developed at INL for LWR analysis using TRITON [Ref].

The post-processing tool looks through an output data file from ERANOS and finds the mass balance data for all fuel materials and a blanket material (if present) for each time-step at each cycle. If a material with a lumped fission product is encountered, the lumped fission product is split into individual nuclides based on data provided by CEA. Lumped fission products were created in the ERANOS library to reduce the computational burden of tracking many fission products. Validation studies were performed on their use and are documented in Tommasi (2006). A graphical user interface was created to make the task of post-processing easier on the

user. The user can select an ERANOS output data file from the "File" menu and the GUI will call the correct routine to load the data from the file, notifying the user if any error is encountered. Figure A.1 presents the initial interface of the GUI.

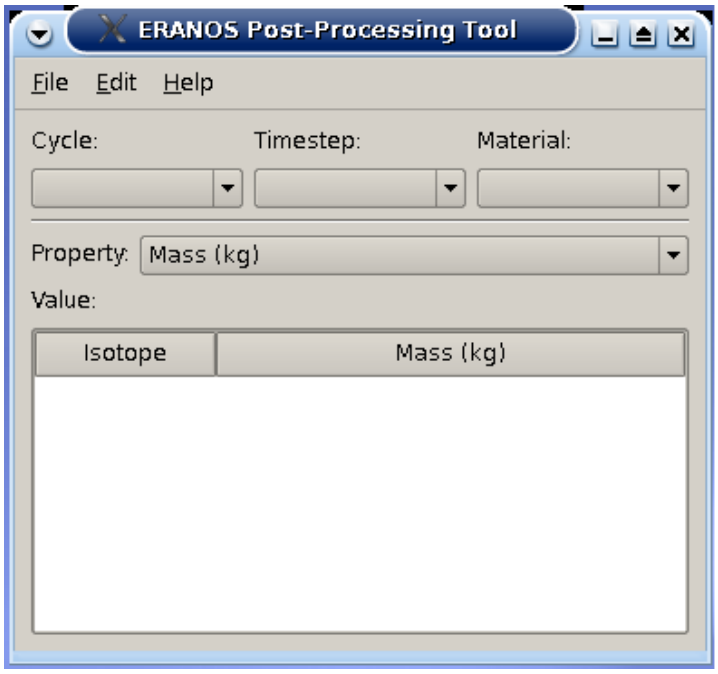

Figure 8-1: GUI Initial Interface

The user can then view the isotopic data directly from the GUI or alternatively write it to a text file. From the GUI, the user can view the mass, volume, heat rate (W/kg), photon heat rate (W/kg), the neutron production rate (N/s/kg), the external dose rate (Sv/hr/kg at 1 m), radiotoxicity for ingestion and inhalation (Sv/kg), ratio of fissile mass to total actinide mass, DPA, burnup, and for select materials, the critical mass of a bare sphere (kg). The first five utility functions from *Charlton et al.* (DOE attractiveness, heating rates, weight fraction of even Pu isotopes, concentration, and dose rates) and the two figure-of-merits from *Bathke et al.* can also be displayed from the GUI or written to a text file. Figure A.2 illustrates some of these capabilities. Additionally, utility function u_7, u_8, u_9 and u_{12} can also be displayed.

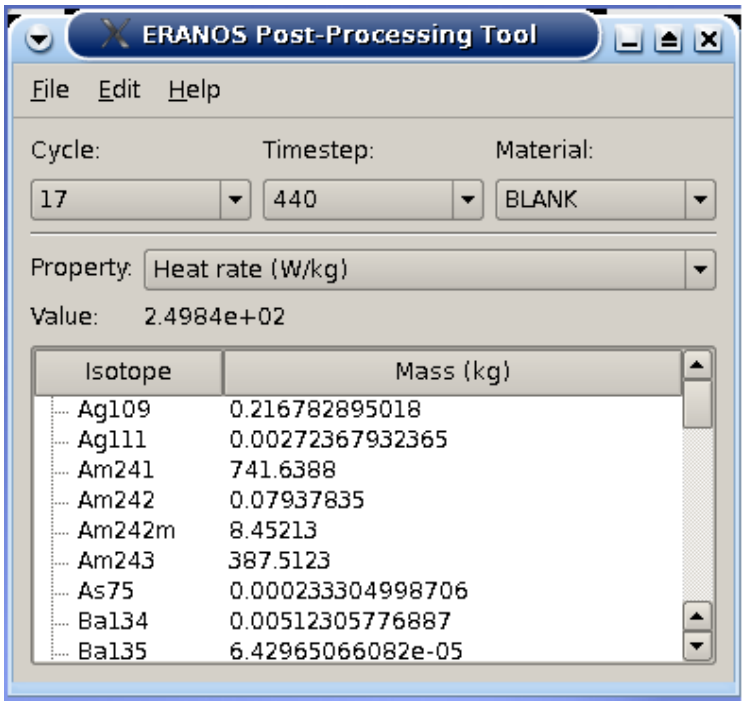

Figure 8-2: Processed data in GUI

A capability for automated creation of VISION input files was added into the graphical user interface. This option creates a text file with both 1 stream (combined fuel and blanket) and 2 stream mass flows (separated fuel and blanket). These options are selected from the File menu.

Verification of the post-processor was performed internally by manually extracting the cycle mass flows and performing the processing in a spreadsheet. The separation of isotopes was also performed accordingly by using the yields of the lumped fission products provided by CEA.

9 Appendix II: Fuel Fabrication Metric

Integration of flux over time of exposure gives fluence, Φ. Since there is almost no attenuation in air, we can estimate fluence to be $\Phi = N / 4\pi d^2$ where d is the distance from the source and N is the total integrated number of neutrons. Effective dose equivalent, HE, is correlated to flux and is given by the relation $H_E = h_E \Phi$ where h_E is the fluence-to-dose factor and is evaluated for different models using photon-neutron-electron transport calculations [10]. Using the neutron rate and fluence-to-dose conversion factor h_E , we estimate the effective dose equivalent to an individual who spends 8 hours a day, 260 days a year, working at an average distance of 1 meter from a fast neutron source of an irradiated blanket fuel assembly. Even though biological effects of an exposure to ionization radiation are not absolute physical quantities that can be measured with high precision, effective dose equivalent calculation provides a fair estimation of the potential effects of exposure to radiation.

Maximum Neutron Rate - Unshielded

Let neutron production rate = *x* neutrons/s/kg

Mass of the irradiated nuclear fuel in the blanket region = 13,039 kg

Number of blanket assemblies = 60

Mass of each assembly = 13,039 kg / 60 = 217.32 kg per assembly

Total neutron fluence per assembly = 217.32 kg * *x* n/s/kg * 8 hr/day * 3600 s/hr * 260days/yr = 1.63E9*x* neutrons

Assuming isotropic scattering, $\Phi = N / 4\pi d^2$

$= 1.63E9x \text{ neutrons} / 4\pi(100\text{cm})^2$

$= 12971.13x \text{ neutrons/cm}^2$

Average energy for fast neutron source = 2 Mev

Considering ROT (uniform rotation of body about axis), $h_E = 2E\text{-}10 \text{ Sv.cm}^2$

According to the Nuclear Regulatory Commission (NRC) [10], the maximum amount of radiation exposure to individual members of the public is limited to 1 mSv (millisievert) per year and the maximum amount of occupational radiation exposure to adults working with radioactive material is limited to 50 mSv per year. We estimate the neutron rate, *x*, that would correspond to an effective dose equivalent of 50 mSv per year.

$$H_E = h_E\ \Phi = 2E\text{-}10\ Sv.cm^2 * 12971.13x\ neutrons/cm^2$$

For H_E of 50 mSv per year per assembly, x = 2×10^4 neutrons/s/kg (approximation)

A conservative value of 1×10^4 neutrons/s/kg was set as our lower limit. This provides some margin to the occupational dose which is needed since the calculation doesn't account for the dose from other particles.

Maximum Neutron Rate – Shielded by 12 inches of lead glass

We also wish to know the maximum neutron rate that can be shielded from a lead glass window of approximate thickness of 12 inches with a density of 5.2 g/cm^3 while making sure that workers are exposed to radiation levels of no more than 50 mSv / yr.

Using the definition of macroscopic cross-section and the limit of $2x10^4$ neutrons/s/kg limit at the edge of the glass, we can determine that the maximum neutron emission rate has to be lower than $2x10^6$ neutrons/s/kg.

Maximum Neutron Rate – Shielded by 60 inches of lead glass

With similar assumptions as above, the maximum neutron emission rate has to be lower than $1.7x10^{14}$ neutrons/s/kg.

10 Appendix III: VISION Post-Processing Requirements

VISION currently collapses the array of isotopes into a more manageable set of 81 pseudo-isotopes that keeps important nuclides and condenses short-lived ones. To make the post-processing flexible, the number of isotopes and their definition can be varied by modifying the "isoprocess.txt" file. The first step consists at defining the total number of isotopes to keep track of by using the "NOISOTOPES" variable. Here are some general guidelines in defining isotopes:

Simple isotope

If you wish to define an isotope by itself:

ISOTOPE001 1 ra228

This indicates that isotope 001 will be composed of one isotope using the data from Ra-228.

Sum of multiple isotopes

To sum multiple isotopes:

```
ISOTOPE002 8 pb208 rn220 po216 po212 bi212 pb212 tl208 ra224
```

This creates an isotope names Pb-208 by summing the 8 isotopes listed.

Weighted sum of multiple isotopes

Weights can also be assigned to each isotope to represent multiple decay branches:

```
ISOTOPE003 2 cm242 am242
WEIGHTS003 1 0.827 /
```

This indicates that Cm-242 will be the sum of all Cm-242 and 82.7% of Am-242.

Renaming isotopes

In some cases, it is preferable to rename a sum of many isotopes for simplicity:

```
ISOTOPE004 20 kr80 kr81m kr82 kr83 kr83m kr84 kr85m kr86 kr87 kr88 kr89 kr90 kr91 kr92 kr93 kr94
kr95 kr96 kr97 kr98 krstable
```

The sum of 20 isotopes will be named Kr-stable.

The sample input file that was used in this analysis follows.

```
NOISOTOPES 81

ISOTOPE001 3 he4 np236 pu236
WEIGHTS001 1 0.01695 0.01695 /
ISOTOPE002 8 pb208 rn220 po216 po212 bi212 pb212 tl208 ra224
ISOTOPE003 1 ra228
ISOTOPE004 2 th228 ac228
ISOTOPE005 2 th232 pa232
ISOTOPE006 9 bi209 ac225 ra225 fr221 at217 po213 bi213 pb209 tl209
ISOTOPE007 2 th229 th226
ISOTOPE008 3 pb206 bi210 po210
ISOTOPE009 8 pb210 u230 th226 ra222 rn222 po218 bi214 pb214
ISOTOPE010 1 ra226
ISOTOPE011 2 th230 pa230
ISOTOPE012 10 pb207 th227 fr223 ra223 rn219 po215 po211 bi211 pb211 tl207
ISOTOPE013 1 ac227
ISOTOPE014 3 pa231 u231 th231
ISOTOPE015 4 u232 np236 np236m pu236
WEIGHTS015 1 0.98305 0.98305 0.98305 /
ISOTOPE016 3 u233 th233 pa233
ISOTOPE017 4 u234 pa234 pa234m th234
ISOTOPE018 1 u235
ISOTOPE019 1 u236
ISOTOPE020 1 u238
ISOTOPE021 2 np237 u237
ISOTOPE022 2 pu238 np238
ISOTOPE023 3 pu239 np239 u239
ISOTOPE024 4 pu240 np240 np240m u240
ISOTOPE025 1 pu241
ISOTOPE026 2 pu242 am242
WEIGHTS026 1 0.173 /
ISOTOPE027 1 pu244
ISOTOPE028 1 am241
ISOTOPE029 1 am242m
ISOTOPE030 2 am243 pu243
ISOTOPE031 2 cm242 am242
WEIGHTS031 1 0.827 /
ISOTOPE032 1 cm243
ISOTOPE033 3 cm244 am244 cf248
ISOTOPE034 3 cm245 am245 pu245
ISOTOPE035 1 cm246
ISOTOPE036 1 cm247
ISOTOPE037 1 cm248
ISOTOPE038 1 cm250
ISOTOPE039 5 cf249 es253 cm249 bk249 cf253
ISOTOPE040 3 cf250 bk250 cf254
ISOTOPE041 4 cf251 bk251 es255 fm255
ISOTOPE042 1 cf252
ISOTOPE043 1 h3
ISOTOPE044 1 c14
ISOTOPE045 1 kr81
ISOTOPE046 1 kr85
ISOTOPE047 20 kr80 kr81m kr82 kr83 kr83m kr84 kr85m kr86 kr87 kr88 kr89 kr90 kr91 kr92 kr93 kr94
kr95 kr96 kr97 kr98 krstable
ISOTOPE048 1 sr90
```

ISOTOPE049 19 sr86 sr87 sr87m sr88 sr89 sr91 sr92 sr93 sr94 sr95 sr96 sr97 sr98 sr99 sr100 sr101 sr102 sr103 sr104 srstable
ISOTOPE050 1 tc99
ISOTOPE051 21 tc99m tc100 tc101 tc102 tc102m tc103 tc104 tc105 tc106 tc107 tc108 tc109 tc110 tc111 tc112 tc113 tc114 tc115 tc116 tc117 tc118 tcstable
ISOTOPE052 1 i129
ISOTOPE053 22 i127 i128 i130 i130m i131 i132 i133 i133m i134 i134m i135 i136 i136m i137 i138 i139 i140 i141 i142 i143 i144 i145 istable
ISOTOPE054 1 cs134
ISOTOPE055 1 cs135
ISOTOPE056 2 cs137 ba137m csba137
ISOTOPE057 18 cs133 cs134m cs135m cs136 cs138 cs138m cs139 cs140 cs141 cs142 cs143 cs144 cs145 cs146 cs147 cs148 cs149 cs150 csstable
ISOTOPE058 3 ce144 pr144 pr144m
WEIGHTS058 1 0.985 0.015 /
ISOTOPE059 1 pm147
ISOTOPE060 1 sm146
ISOTOPE061 1 sm147
ISOTOPE062 1 sm151
ISOTOPE063 1 eu154
ISOTOPE064 1 eu155
ISOTOPE065 1 ho166m
ISOTOPE066 159 LA
ISOTOPE067 1 se79
ISOTOPE068 2 zr93 nb93m zrnb93
WEIGHTS068 1 0.95 /
ISOTOPE069 3 zr95 nb95 nb95m zrnb95
WEIGHTS069 1 0.991 0.009 /
ISOTOPE070 2 rh106 ru106 rhru106
ISOTOPE071 1 pd107
ISOTOPE072 1 cd113m
ISOTOPE073 2 sb125 te125m sbte125
WEIGHTS073 1 0.23 /
ISOTOPE074 3 sn126 sb126m sb126 snsb126
WEIGHTS074 1 0.14 1 /
ISOTOPE075 826 FP
ISOTOPE076 1 fe55
ISOTOPE077 1 co57
ISOTOPE078 1 co58
ISOTOPE079 1 co60
ISOTOPE080 1 ni59
ISOTOPE081 1 ni63

11 Appendix IV: Material Properties

$ZrH_{1.6}$

```
SIMPLE_MATERIAL  'MODERATOR'  FERTILE
  WEIGHT_PERCENTAGE 5.32

ELEMENT CIP 100.0000
'Zr90' 50.55
'Zr91' 11.02
'Zr92' 16.85
'Zr94' 17.08
'Zr96' 2.75
'H1'   1.74
 EXPANSION (EXP_FUEL)
```

YH_2

```
SIMPLE_MATERIAL  'MODERATOR'  FERTILE
  WEIGHT_PERCENTAGE 4.085

ELEMENT CIP 100.0000
'Y89'      97.78
'H1'       2.22
 EXPANSION (EXP_FUEL)
```

LiH

```
SIMPLE_MATERIAL  'MODERATOR'  FERTILE
  WEIGHT_PERCENTAGE 0.779

ELEMENT CIP 100.0000
'Li6'      6.5489
'Li7'      80.769
'H1'       12.679
 EXPANSION (EXP_FUEL)
```

CaH_2

```
SIMPLE_MATERIAL  'MODERATOR'  FERTILE
  WEIGHT_PERCENTAGE 1.615

ELEMENT CIP 100.0000
'Ca40'     92.30
'Ca42'     0.62
'Ca43'     0.13
'Ca44'     1.99
'H1'       4.79
 EXPANSION (EXP_FUEL)
```

BeF_2

```
SIMPLE_MATERIAL  'MODERATOR'  FERTILE
  WEIGHT_PERCENTAGE 1.8867

ELEMENT CIP 100.0000
'Be9'      80.83
'F19'      19.17
 EXPANSION (EXP_FUEL)
```

B_4C

```
SIMPLE_MATERIAL  'MODERATOR'  FERTILE
  WEIGHT_PERCENTAGE 2.394

ELEMENT CIP 100.0000
'B10'      15.57
'B11'      62.69
'C0'       21.74
 EXPANSION (EXP_FUEL)
```

BeO

```
SIMPLE_MATERIAL  'MODERATOR'  FERTILE
  WEIGHT_PERCENTAGE 2.8595

ELEMENT CIP 100.0000
'Be9'      36.03
'O16'      63.97
 EXPANSION (EXP_FUEL)
```

MgO

```
SIMPLE_MATERIAL  'MODERATOR'  FERTILE
  WEIGHT_PERCENTAGE 3.401

ELEMENT CIP 100.0000
'Mg24'     47.67
'Mg25'     6.03
'Mg26'     6.64
'O16'      39.69
 EXPANSION (EXP_FUEL)
```