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Computational Neutronics Methods and Transmutation Performance Analyses for Fast Reactors

R. Ferrer
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Idaho National Laboratory
Idaho Falls, Idaho 83415

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ABSTRACT

The once-through fuel cycle strategy in the United States for the past six decades has resulted in an accumulation of Light Water Reactor (LWR) Spent Nuclear Fuel (SNF). This SNF contains considerable amounts of transuranic (TRU) elements that limit the heat and dose capacity of the current planned repository strategy. A possible way of maximizing the utilization of the repository is to separate the TRU from the LWR SNF through a process such as UREX+1a, and convert it into fuel for a fast-spectrum Advanced Burner Reactor (ABR). The key advantage in this scenario is the assumption that recycling of TRU in the ABR (through pyroprocessing or some other approach), along with a low capture-to-fission probability in the fast reactor's high-energy neutron spectrum, can effectively decrease the decay heat and toxicity of the waste being sent to the repository. The decay heat and toxicity reduction can thus minimize the need for multiple repositories. This report summarizes the work performed by the fuel cycle analysis group at the Idaho National Laboratory (INL) to establish the specific technical capability for performing fast reactor fuel cycle analysis and its application to a high-priority ABR concept. The high-priority ABR conceptual design selected is a metallic-fueled, 1000 MWth SuperPRISM (S-PRISM)-based ABR with a conversion ratio of 0.5. Results from the analysis showed excellent agreement with reference values. The independent model was subsequently used to study the effects of excluding curium from the transuranic (TRU) external feed coming from the LWR SNF and recycling the curium produced by the fast reactor itself through pyroprocessing. Current studies to be published this year focus on analyzing the effects of different separation strategies as well as heterogeneous TRU target systems.

DISCLAIMER

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GLOSSARY, ACRONYMS, AND ABBREVIATIONS

ABR	Advanced Burner Reactor
ANL	Argonne National Laboratory
DOE	Department of Energy
GNEP	Global Nuclear Energy Partnership
INL	Idaho National Laboratory
LWR	Light Water Reactor
MA	Minor Actinides
MC^2-2	Code System for Calculating Fast Neutron Spectra and Multigroup Cross-Sections
ORNL	Oak Ridge National Laboratory
REBUS-3	Code System for Analysis of Fast Reactor Fuel Cycles
RSICC	Radiation Safety Information Computational Center
S-PRISM	SuperPRISM Advanced Fast Reactor
SNF	Spent Nuclear Fuel
TRU	Transuranics
UREX	Uranium Extraction Process

1. Introduction

In support of the Global Nuclear Energy Partnership (GNEP), the fuel cycle analysis group at the Idaho National Laboratory (INL) has established a specific technical capability to perform fast reactor physics analysis and has completed an initial application of this capability to a high-priority ABR prototype. This goal was achieved by utilizing two widely used fast reactor analysis code packages, MC^2-2 [1] and REBUS-3 [2]. An overview of the methods and code packages used is presented in Section 2.0. The Argonne National Laboratory (ANL) report ANL-AFCI-177 [3] served as source for reference designs and computational results used for verification of the independent INL calculations.

The SuperPRISM (S-PRISM)-based [4] 1000 MWth ABR with a conversion ratio of 0.5 was selected as a high-priority prototype design for this exercise. The fuel was assumed to be composed of U-TRU-Zr metallic alloy. This choice of design represented a modest compromise between the needs of the program to burn transuranics (TRU) and the material irradiation experience from past test programs [5-7]. These needs are competing factors, since the amount of TRU that can be burned in an ABR is bounded by the limits on the integrity of the fuel and supporting structure. Detailed information, such as the assembly design, fuel composition, external fuel cycle modeling, and burnup limit is presented in Section 3.0.

Three sets of data summarizing the results of the above design computations are compared in Section 4.0. The first set of results was obtained from the ANL-AFCI-177 report. The second set was obtained by re-executing computations reported in the ANL-AFCI-177 report at INL using our versions of MC^2-2 and REBUS-3 with identical code input data as was used to produce the results in the report. Finally, the results from our own independent model form the third set of data. The results reveal that while differences exist between these sets, they are minor and can be explained by differences in the assumed fundamental data, fuel cycle enrichment search limits and slightly different model geometries. Important neutronics and fuel cycle parameters are identified and assessed from the point of view of their impact on the design, safety, and overall performance of the system as a result of the work reported here.

Once the basic models had been established and verified, a study concerning the external feed source was performed and the results from this study are also presented in this report. The study involved the exclusion of curium from the TRU external feed coming from LWR SNF and recycling only the curium produced by the fast reactor itself, while maintaining the material and geometric properties of the reactor design constant. The results from this study are presented in Section 5.0.

Additional studies recently completed in connection with the initial work reported here involve varying the TRU elements sent to the burner reactor by assuming different UREX processes as the reactor's external feed. Parallel to these studies, the fuel cycle analysis group has also developed a heterogeneous ABR design with TRU targets and is currently studying the optimization of such designs. These studies are not included in this report due to space and scope limitations, but will form the bulk of a second report to be published later this year.

Finally, Section 6.0 presents some conclusions and an outline of needs for future work. The identified needs include developing two or multiple-tier recycling capabilities between thermal and fast spectrum reactors and expanding the capability for fast reactor fuel cycle analysis by tracking higher mass minor actinides (MA) beyond Cm-246 (which have been previously disregarded in fast reactor recycle studies and may have a significant effect on the fuel discharge decay heat and toxicity).

2. Computational Models and Code Packages

The code packages most commonly used in the U.S. for fast reactor cross-section generation and fuel cycle analysis are MC^2-2 and REBUS-3, respectively. The MC^2-2 package was used to generate region-dependent, 33 energy-group cross-sections at operating temperatures based on ENDF/B-V.2 collapsed libraries. It should be noted that there are two additional code packages (SDX and DB2), which are typically used in concert with MC^2-2 and REBUS-3, but are not publicly available, and thus were not used in the calculations performed by INL. The first of these, SDX, accounts for the spatial effect of heterogeneity introduced by pins in an assembly (which is modeled initially in MC^2-2 as a homogenized region). The second, DB2, takes the fission energy spectrum from areas on the periphery of the core and collapses the cross-sections for the neighboring control rod, reflector, and shield regions based on the leakage spectrum. Our approach was to disregard the spatial heterogeneity introduced by pins in the assembly (The heterogeneity effects can be argued to be minimal since the average neutron mean-free-path in a fast reactor is roughly about the flat-to-flat dimension of a hexagonal assembly.). This assumption is supported by the good agreement of our results in reference to those published in ANL-AFCI-177. Additionally, a generic representative of the Pu-239 fission energy spectrum in MC^2-2 was used to collapse the reflector and shield region cross-sections.

The REBUS-3 nodal diffusion option in hexagonal-z geometry was used to perform the flux calculations. In our fuel cycle model individual fuel assemblies are homogenized into "like neutron spectrum" representative enrichment zones. Therefore, independent batches of fuel are tracked within the external fuel cycle but not explicitly spatially represented in the physics calculation. Furthermore, the constraints in the equilibrium calculations involved a search of the specific fresh fuel charge enrichment given a discharge burnup limit (18 atom percent). This was done by first estimating the initial fuel composition by assuming a certain approximate enrichment. The fuel cycle code searches for an enrichment that does not violate the maximum burnup limit for the uncontrolled core (the multiplication factor equal to 1.0 at end-of-life). An automated scripting system is used to re-calculate the cross-sections for each enrichment zone based on that zone's fuel inventory at equilibrium. This ensured that the group constants correspond to the equilibrium case (since the initial cross section set is based only on an estimate of the actual TRU enrichment). Finally, the corrected charge enrichment and equilibrium cycle length is reported. The peak fast flux was used to calculate the peak fast fluence and a limit of 4.0×10^{23} n/cm^2 for HT9 cladding, assumed to be the limiting fluence in the ANL-AFCI-177 report [5-7], was applied. The fuel cycle length was adjusted, based on this limit, and a final REBUS-3 calculation was performed. A schematic of this methodology is shown below in Fig. 1.

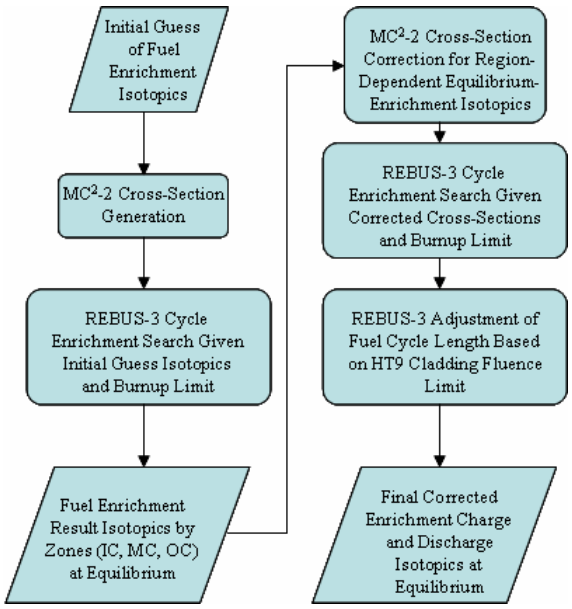

Figure 1. Schematic of Fast Reactor Fuel Cycle Methodology Performed at INL.

It is important to note that a number of different versions of MC^2-2 and REBUS have been developed over the years by ANL and that the only versions that were available to INL, obtained through the Radiation Safety Information Computation Center (RSICC), date to 1997 and are not the most recent versions. This is a source of some of the differences observed between the computational results reported here and the reference results reported in ANL-AFCI-177. One major difference between different versions of the REBUS code package concerns the estimation of the conversion ratio. The released RSICC version of REBUS-3 utilizes the conversion ratio defined as the amount of fissile material produced divided by the amount of fissile material destroyed. The origin of this definition comes from the fact that most fast reactors have been historically used for breeding. Since a premise of the GNEP program concerning the deployment of fast reactors involves the intentional destruction of TRU, the current sense of the conversion ratio is related to the total amount of TRU produced divided by the amount of TRU destroyed. Since not all of the TRU is composed of fissile isotopes, the computed conversion ratio with respect to breeding can be different in a burner system than the conversion ratio based on TRU production and destruction for the same system. The definition of the conversion ratio based on TRU consumption and production rates was implemented in the latest version of REBUS at ANL used to generate the results published in the ANL-AFCI-177 report. The approach that INL has taken in the present work is to estimate the TRU conversion ratio using the following formula,

$$CR = \frac{R_{HM} - R_{TRU}}{R_{HM}} = 1 - \frac{R_{TRU}}{R_{HM}},$$

where R_{HM} is the average rate of mass consumption between beginning-of-life (BOL) and end-of-life (EOL) of total Heavy Metal (HM) and R_{TRU} is the rate of mass consumption between BOL and EOL of TRU. This definition assumes that all the uranium consumed in the ABR produces TRU (i.e. assumes only neutron capture by U-238 and no fission) but gives a fair representation of TRU verses total heavy metal destruction. As expected, the estimated conversion ratio using this method yields a slightly higher conversion ratio than the results published in the ANL-AFCI-177 report. The definition used in the rest of this report is the TRU-based conversion ratio, unless noted otherwise.

3. Reference ABR Core Design Description

The strategy followed for reducing the conversion ratio of the ABR design from the original S-PRISM design was to reduce the fuel pin diameter in the fuel assemblies. This resulted in a reduction of the fuel volume fraction and a subsequent increase in the TRU enrichment. A higher TRU enrichment increased the TRU fission relative to the U-238 capture, thus reducing the conversion ratio. The volume fractions for the S-PRISM metallic-fueled design and the reference metallic-fueled ABR are summarized in Table 1 below. Note that a decrease in fuel volume fraction causes an increase in bond and coolant volume fraction.

The thermal conductivity of the fuel for the reference ABR degrades with the increased TRU enrichment. Thus a larger weight fraction of zirconium had to be added to the fuel in order to increase the high enriched fuel's thermal conductivity. Additionally, the TRU enrichment was tailored across the core in order to flatten the power distribution (enrichment splitting is 1.00, 1.25, and 1.50 in the inner, middle, and outer core regions, respectively).

Table 1. S-PRISM versus ABR Assembly Material Volume Fractions.

	S-PRISM CR=1.0	Metallic ABR CR=0.5
Fuel	28.30 %	22.08 %
Bond	9.43 %	7.36 %
Structure	25.70 %	26.41 %
Coolant	36.57 %	44.15 %

The fuel assembly design cold dimensions for the reference ABR are listed in Table 2 Material thermal expansion at operating conditions was not included in the INL model. While the S-PRISM assembly design has 271 pins per assembly, the assembly design for the ABR has 324 pins per assembly. The reduced thermal performance of higher TRU enriched fuel pins required a larger number of pins per assembly in order to reduce the average linear power to an acceptable limit [3]. The fuel pin dimensions are summarized below in Table 3

Table 2. ABR Fuel Assembly Design.

Assembly pitch, cm	16.142
Inter-assembly gap, cm	0.432
Duct outside flat-to-flat, cm	15.710
Duct material	HT9
Duct thickness	0.394
Fuel pins per assembly	324
Spacer type	Grid

Table 3. ABR Fuel Pin Design.

Bond	Na
Core Height, cm	101.06
Plenum Height, cm	191.14
Overall Pin Length, cm	407.04
Fuel smeared density, %	75
Fabrication density, %	100
Pin Diameter, cm	0.623
Pin-to-Pitch-to-diameter Ratio	1.293
Clad thickness, cm	0.0559
Wire wrap diameter, cm	N/A

The radial layout of the core consists of three driver fuel regions; the inner, middle, and outer core. A schematic of this layout is shown below in Fig. 2. The inner core consists of four rings containing a total of 42 assemblies with a charge TRU enrichment of 27.3%. The middle core consists of two rings containing a total of 66 assemblies with a charge TRU enrichment of 34.1%. Finally, the outer core consists of a single ring containing 36 assemblies with a charge TRU enrichment of 40.9%. Such an enrichment splitting allows flattening of the power distribution. The reflector and shield regions of the core correspond to the last three rings of the core. While the ultimate shutdown and primary control rods are shown in the schematic, these were modeled as fully withdrawn.

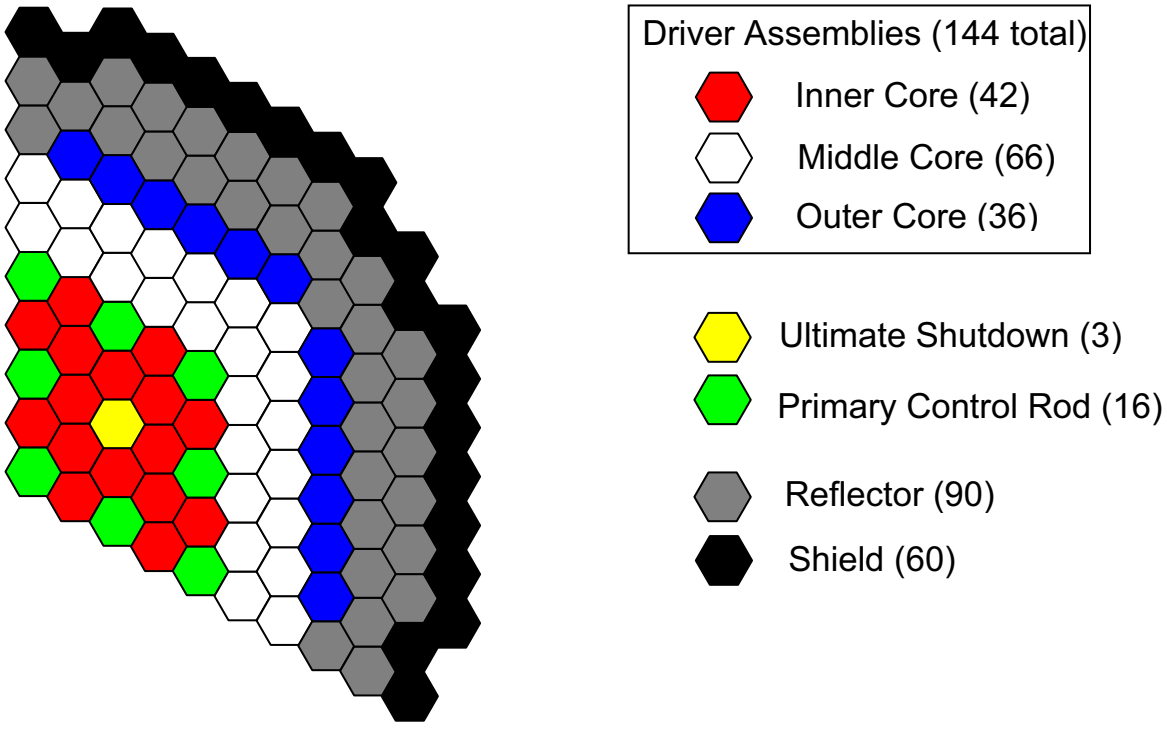

Figure 2. Core Layout of Advanced Burner Reactor (1/3 symmetry).

In this study, the TRU component of LWR SNF was assumed to be separated through the UREX+1a process and sent to the fuel fabrication and separation plant to be converted into ABR fuel. Depleted uranium from enrichment plants is assumed to be the only other external feed to provide fuel material to the fuel fabrication plant. The fuel fabrication plant feeds the ABR with the necessary charge isotopic vector. The discharged fuel from the reactor is reprocessed through pyroprocessing, and charged back into the ABR. The reduced waste from this fuel cycle is assumed to be sent to interim storage or a repository. A schematic of this fuel cycle is shown below in Fig. 3.

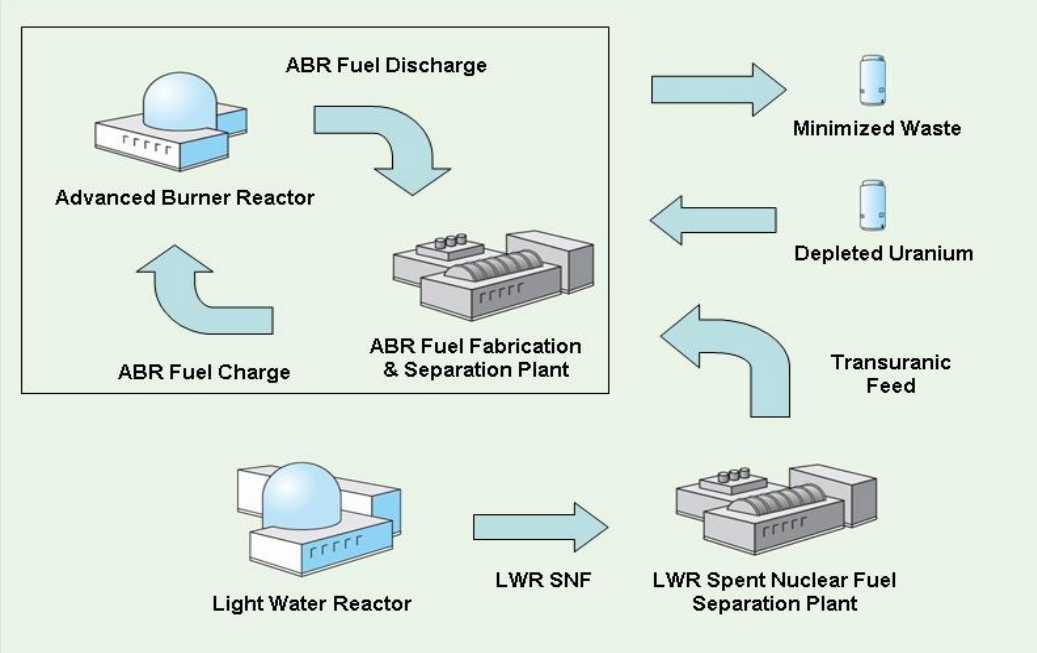

Figure 3. External Fuel Cycle Model of Advanced Burner Reactor.

The isotopic composition of the spent LWR fuel used in this analysis was taken from the ANL-AFCI-177 report and is listed in Table 4 These values were calculated using ORIGEN-RA [8] for a standard 17x17 PWR assembly loaded with enriched uranium and irradiated to 50 MWd/kg. It was assumed that the LWR SNF separation plant recovers 100% of the TRU isotopes and that an unlimited amount is available for the fuel fabrication plant. The fuel fabrication plant is also assumed to be able to recover all the discharged fuel from the ABR and draw the makeup isotopes from the depleted uranium and LWR SNF separation plant. The assumed discharge cooling time of the fuel is 297.32 days and the reprocessing time 148.66 days.

Table 4. Weight Percent Composition of External Feed Separated from LWR SNF using UREX+1a.

TRU Isotopes	w/o
Np-237	4.777
Pu-236	0.000
Pu-238	2.310
Pu-239	47.899
Pu-240	22.510
Pu-241	10.580
Pu-242	6.519
Am-241	3.356
Am-242m	0.006
Am-243	1.475
Cm-242	0.000
Cm-243	0.005
Cm-244	0.515
Cm-245	0.041
Cm-246	0.005

4. Comparison Between Reported, Reference, and Independent Models

As part of the verification process for INL fast reactor fuel cycle analysis capabilities established in this effort, it was first necessary to compare the results from models built internally with those reported by other groups. As explained in previous sections, the choice of the ANL-AFCI-177 report as a source of conceptual design information was a natural choice given its scope and reporting of analysis results. Furthermore, an input file for this ABR case was obtained from ANL and executed using our computational models and code packages. This additional resource reveals that results slightly vary between different versions of MC^2-2 and REBUS-3 given exactly the same input.

It is also important to note the input file received from ANL was for verification purposes and only was not used to create the INL model, which was created independently. The reasons are twofold; first, the effort to build an independent model began before the input was received and is based on our interpretation of ANL-AFCI-177, and secondly, the purpose of this exercise was to demonstrate the capability for verified independent modeling, so using the aforementioned input to build the independent model would have defeated this purpose.

The equilibrium charge enrichment for the three different regions of the ABR design is listed in Table 5 below.

Table 5. Charge Enrichment by Volume and Weight Percent for ABR concept.

	Inner Core	Middle Core	Outer Core
TRU enrichment, v/o	26.6	33.3	39.9
U, w/o	66.1	58.7	51.7
TRU, w/o	23.9	29.3	34.3
Zirc, w/o	10.0	12.0	14.0

The fuel cycle characteristics for the three cases discussed in this report are presented below in Table 6. As previously noted, slightly different methodologies were used to calculate the conversion ratio. As expected, the algorithm added to REBUS version used in the report by ANL is able to compute the TRU-based conversion ratio, thus reporting a value of 0.50. Using the conversion ratio formula presented in Section 2.0, we estimated the value of the conversion ratio by assuming that the U-238 undergoes only neutron capture interactions, with no fission. This yields to a slightly higher amount of TRU produced, and thus a slightly higher conversion ratio. The estimated conversion ratio for the INL model (0.56) is slightly higher than the one obtained from the results produced at INL using the ANL model input file (0.54) due to the fact that the TRU enrichment for the INL model (26.6%) is slightly lower than the reference model (27.1%). As explained in Section 3.0, the strategy followed to lower the conversion ratio is to increase the TRU enrichment, which explains the lower conversion ratio for the reference models.

During the first REBUS-3 enrichment search, the maximum burnup limit (18 atom percent) is surpassed earlier in the cycle than the predicted 221 Effective Full Power Days (EFPD) cycle length. This is due to the fact that the calculated fuel fabrication densities in the INL model are lower than in the reference case. The total initial amount of HM is lower for the INL model (9,419 kg) than in the reference model (9,448 kg), effectively increasing the burnup. In order to meet the maximum peak fast fluence limit of 4.00×10^{23} n/cm^2, the cycle length was shortened to 217 EFPD. The combination of a shorter cycle length and a lower HM charge mass results in a slightly lower average driver burnup (130.0 MWd/kg) in the INL model than in the reference model (131.9 MWd/kg). The fuel fabrication density, which is the

source of this difference, was pre-computed from assumed values before the reference input was received. These assumed densities are not reported in ANL-AFCI-177 and seem to use higher pure-metal density for the americium and curium than those used in our model, which are based on separate published values [9].

The TRU consumption rate, an important parameter in the determination of the support ratio (ratio of LWRs per ABR) was found to be lower in the INL model (166 kg/EFPY) than in the ANL model (172 kg/EFPY) and the report (174 kg/EFPY). This is a direct effect of the charge TRU enrichment, which follows the same trend across the three cases. As explained above, the lower TRU enrichment for the INL case is due to the burnup limit being reached at an earlier point in the cycle. This is due to the assumed lower fuel fabrication density, specifically the assumption of a lower pure-metal americium and curium density.

Table 6. Reported Fuel Cycle Parameters Between Reference and INL Models.

		ANL-AFCI-177	Reference Input*	INL
Conversion Ratio		0.50	0.54**	0.56**
Charge Enrichment, TRU/HM (v/f)	IC	27.3%	27.1%	26.6%
	MC	34.1%	33.9%	33.3%
	OC	40.9%	40.7%	39.9%
Fuel residence time, cycles	IC	6	6	6
	MC	6	6	6
	OC	7	7	7
Burnup (MWd/kg)	Ave. Driver	131.9	131.9	130.0
Peak Fast Fluence, 10**23 n/cm**2	IC	4.00	4.02	4.00
	MC	3.96	3.97	3.97
	OC	3.75	3.77	3.74
HM loading, kg		9,449	9,448	9,419
TRU loading, kg		3,084	3,062	3,007
Fissile Pu loading, kg		1,348	1,342	1,324
Cycle length, EFPD		221	221	217
TRU Consumption Rate, kg/EFPY		174	172	166
TRU Charge, kg/EFPY		894	887	885
HM Charge, kg/EFPY		2,683	2,681	2,721

* These results correspond to the input deck built and sent by the authors of ANL-AFCI-177 and executed at INL. Note that the execution of this case was not done using the scheme outlined in Section 2.0, Fig. 1. Only one MC^2-2 and one REBUS-3 executions were performed.

** These conversion ratios were calculated using the methodology outlined in Section 2.0.

While the data listed above is important from a reactor physics and fuel cycle perspective, the ultimate purpose of these calculations is to provide the mass flow of HM isotopes as they are consumed and produced in the system. The available capabilities in the released versions of MC^2-2 only allow the user to generate fast reactor cross-sections from a limited library of isotopes. In order to generate a full listing of the isotopes produced and destroyed in the fast reactor, the authors of the ANL-AFCI-177 report used the charge mass data at equilibrium for the ABR design as input for ORIGEN-RA [8] to perform a constant flux depletion, followed by subsequent decay calculation. The standard ORIGEN-S package in Scale5.1 [10] does not currently include a fast reactor flux library, but plans are underway to include a fast reactor library in the near future. A tabulated comparison of the mass charge data for the two models

is presented in Table 7. As expected, the amount of TRU in terms of grams per Metric Ton In Heavy Metal (g/MTIHM) is lower in the INL case when compared to the values reported in ANL-AFCI-177. The percent difference is shown on the last right hand side column with differences in the ranges between 2.0 and 6.0 %. The negative percentage is due to the lower TRU enrichment in the INL case.

Table 7. Charge Isotopic Data for Reference and INL Models.

	ANL-AFCI-177	Reference Input	INL	% Difference (INL vs. ANL-AFCI-177)
Plutonium (g/MTIHM)	295,310	293,376	288,908	-2.17%
Neptunium (g/MTIHM)	7,122	7,052	6,695	-6.00%
Americium (g/MTIHM)	21,261	21,015	20,203	-4.97%
Curium (g/MTIHM)	9,597	9,492	9,404	-2.01%
U-235 (g/MTIHM)	509	505	495	-2.78%
Pu-238 (g/MTIHM)	11,102	10,972	10,558	-4.90%
Pu-239 (g/MTIHM)	128,610	128,108	126,525	-1.62%
Pu-240 (g/MTIHM)	102,590	101,825	100,794	-1.75%
Pu-241 (g/MTIHM)	21,640	21,451	20,892	-3.46%
Am-241 (g/MTIHM)	10,541	10,413	9,968	-5.44%
U-235/U	0.08%	0.08%	0.07%	-
TRU/HM	33.3%	33.1%	32.5%	-
BOEC HM (kg)	9,449	9,448	9,419	-0.31%
BOEC TRU (kg)	3,084	3,062	3,007	-2.49%

In order to attempt a further, more detailed, comparison the mass balance or change in isotopic inventory from BOL to EOL was examined for these models. This data is not available in ANL-AFCI-177, so only mass balances for BOL and EOL for the other two data sets are compared. As expected, the BOL TRU charge in terms of total kilograms in the reactor is larger for the reference case than the INL case, due to a higher TRU enrichment. This higher TRU enrichment provides the reference model with a higher amount of fissile plutonium, which enables a longer cycle length. It also lowers the U-238 makeup, which leads to a reduction in the total parasitic capture and subsequent production of more TRU. Despite of this difference, the net consumption between the two models follows the same trend for all the tracked isotopes. Agreement in mass flows is essential for the purpose of providing correct charge and discharge mass rates for subsequent use in GNEP systems studies.

Table 8. Fuel Cycle Mass Balance for Reference and INL Models.

	Mass Balance (kg)					
	Reference Input			INL		
	BOL	EOL	NET	BOL	EOL	NET
U-234	10.8	10.6	-0.1	10.2	10.1	-0.1
U-235	4.3	4.1	-0.3	4.2	3.9	-0.3
U-236	5.4	5.4	0.0	5.3	5.3	0.0
U-238	6365.9	6243.2	-122.7	6392.5	6268.3	-124.2
Np-237	57.2	52.2	-5.0	54.1	49.4	-4.7
Pu-236	0.0	0.0	0.0	0.0	0.0	0.0
Pu-238	103.0	100.1	-2.9	98.7	96.0	-2.8
Pu-239	1153.2	1103.2	-49.9	1140.5	1093.3	-47.2
Pu-240	968.4	944.4	-24.0	957.2	934.4	-22.7
Pu-241	188.7	178.9	-9.7	183.9	174.8	-9.1
Pu-242	296.3	289.4	-6.9	287.1	280.6	-6.5
Am-241	92.0	87.1	-4.9	87.7	83.0	-4.7
Am-242m	6.6	6.6	-0.003	6.3	6.3	0.000
Am-243	97.0	95.5	-1.563	93.4	91.9	-1.479
Cm-242	4.5	5.2	0.675	4.4	5.1	0.660
Cm-243	0.4	0.4	-0.003	0.4	0.4	-0.003
Cm-244	67.7	67.7	-0.057	66.4	66.4	-0.036
Cm-245	17.5	17.4	-0.045	17.3	17.2	-0.042
Cm-246	9.6	9.6	-0.003	9.8	9.8	-0.003
Total HM (kg)	9448.5	9221.0	-227.5	9419.4	9196.1	-223.3
Total TRU (kg)	3062.1	2957.7	-104.3	3007.2	2908.5	-98.7

A final comparison of the models involved comparing the multiplication factor as a function of cycle length. The results from this comparison are shown below in Table 9. Since in the reference case the cycle length was longer (221 EFPD), the reported multiplication factors were used to linearly interpolate values for time steps equal to the time steps reported for the INL model cycle length (217 EFPD).

Table 9. Comparison of predicted multiplication factors from different models.

Time (Days)	Reference Input	INL	Milli-k difference
0	1.0297	1.0288	-0.900
54	1.0224	1.0217	-0.671
54	1.0223	1.0217	-0.661
108	1.0150	1.0146	-0.431
163	1.0077	1.0075	-0.210
217	1.0005	1.0006	0.025

5. Comparative Study of Separation Strategies

Once our model had been adequately verified relative to independent reference models, a number of studies were performed by perturbing the baseline design in order to study the effects on the system. While maintaining the ABR prototype design constant, we began perturbing the external feed coming from the LWR SNF to study the effects of using different separation processes in a single tier recycling scenario. Only one of these perturbations is incorporated in this interim report. A report summarizing the findings of all of the aforementioned studies, along with a study of TRU target design in the ABR, will be published later this year.

The UREX+4 separation process, as modeled in this study, assumes that the curium coming from the LWR SNF is separated from the rest of the TRU and disposed of. As is evident from Table 10 below, no curium is drawn from the external feed to fabricate the fuel. Since curium is only a small component relative to the rest of the TRU, the weight percentages of the rest of the external feed remains approximately the same as in the case of UREX+1a.

Table 10. Weight Percent Composition of External Feed from Separated LWR SNF.

Separation Process	UREX+4 (w/o)	UREX+1a (w/o)
Np-237	4.786	4.777
Pu-236	0.000	0.000
Pu-238	2.315	2.310
Pu-239	47.995	47.899
Pu-240	22.555	22.510
Pu-241	10.601	10.580
Pu-242	6.532	6.519
Am-241	3.363	3.356
Am-242m	0.006	0.006
Am-243	1.478	1.475
Cm-242	0.000	0.000
Cm-243	0.000	0.005
Cm-244	0.000	0.515
Cm-245	0.000	0.041
Cm-246	0.000	0.005

The reactor physics and fuel cycle parameters resulting from this perturbation are tabulated below in Table 11 It is evident from observing the parameters listed in the table that the exclusion of curium from the TRU causes very little differences in the fuel cycle characteristics. The conversion ratio, charge enrichment, peak fast fluence, and cycle length all remain virtually the same. The UREX+4 case is loaded with slightly more HM (9,426 kg) than the reference UREX+1a (9,419 kg). This causes the average burnup to be slightly lower for the UREX+4 case (129.8 MWd/kg) than in the reference case (130.0 MWd/kg) for equal power and cycle length. The higher HM loading in the UREX+4 case is due to the lower mass density of curium. Exclusion of curium from the TRU produces a denser fuel, thus the differences in HM and TRU loading in Table 11 will be expected.

Table 11. Fuel Cycle Parameters Between UREX+4 and UREX+1a Case for Metallic-Fueled CR=0.5 ABR.

		UREX+4	UREX+1a
Conversion Ratio		0.56	0.56
Charge Enrichment, TRU/HM (v/f)	IC	26.6%	26.6%
	MC	33.3%	33.3%
	OC	39.9%	39.9%
Fuel residence time, cycles	IC	6	6
	MC	6	6
	OC	7	7
Burnup (MWd/kg)	Ave. Driver	129.8	130.0
Peak Fast Fluence, 10**23 n/cm**2	IC	4.00	4.00
	MC	3.96	3.97
	OC	3.73	3.73
HM loading, kg		9,426	9,419
TRU loading, kg		3,009	3,007
Fissile Pu loading, kg		1,328	1,324
Cycle length, EFPD		217	217
TRU Consumption Rate, kg/EFPY		166	166
TRU Charge, kg/EFPY		886	885
HM Charge, kg/EFPY		2,724	2,722

Similar to the comparison presented in the previous section, the charge data at equilibrium for both of these cases is summarized in Table 12. The charge data in g/MTIHM follows the expected trend explained above: the UREX+4 case has a higher mass charge for the TRU and HM isotopes than the UREX+1a case.

It is important to note the mass balance tabulated in Table 13. The net consumption for the whole reactor in terms of kilograms between BOL and EOL for the listed isotopes is basically the same for both UREX+4 and UREX+1a, with the exception of curium. While the mass of HM at BOL and EOL is larger in the UREX+4 than in the UREX+1a case, the net consumption in kilograms is comparable between the two (with the exception of curium).

At equilibrium, in the case of UREX+4, more net kilograms of curium are produced (thus the positive number), than in the UREX+1a case. The Cm-242 isotope is always accumulated due to neutron capture by Am-241, which produces Am-242 (16.2 hour half-life) 80% of the time with subsequent decay to Cm-242. In the cases of Cm-243, Cm-244, and Cm-245, they are all found to be neither produced nor destroyed in the UREX+4, thus having no net mass increase or decrease. This leaves Cm-244 as the 'problem' curium isotope that, when not included in the TRU external feed for the fuel fabrication, ends up being produced at a rate of 0.459 kilograms between BOL and EOL.

For this specific ABR design and under the assumption that single tier recycling is desirable (and that separation of TRU elements is undesirable), it is evident that if the goal of the program is to reduce the amount of TRU (and specifically MA), then UREX+1a proves to be a more desirable separation strategy than UREX+4, since it effectively consumes more MA than the latter.

Table 12. Charge Isotopic Data for UREX+4 and UREX+1a Cases.

	UREX+4	UREX+1a
Plutonium (g/MTIHM)	289,622	288,908
Neptunium (g/MTIHM)	6,731	6,695
Americium (g/MTIHM)	20,298	20,203
Curium (g/MTIHM)	8,511	9,404
Pu-242 (g/MTIHM)	30,281	30,139
Pu-238 (g/MTIHM)	10,611	10,558
Pu-239 (g/MTIHM)	126,844	126,525
Pu-240 (g/MTIHM)	100,935	100,794
Pu-241 (g/MTIHM)	20,950	20,892
Am-241 (g/MTIHM)	10,012	9,968
Am-242m (g/MTIHM)	599	597
TRU/HM	32.5%	32.5%

Table 13. Fuel Cycle Mass Balance for UREX+4 and UREX+1a Cases.

	Mass Balance (kg)					
	UREX+4			UREX+1a		
	BOL	EOL	NET	BOL	EOL	NET
U-234	10.3	10.1	-0.1	10.2	10.1	-0.1
U-235	4.2	3.9	-0.3	4.2	3.9	-0.3
U-236	5.3	5.3	0.0	5.3	5.3	0.0
U-238	6397.2	6273.1	-124.1	6392.5	6268.3	-124.2
NP237	54.5	49.8	-4.7	54.1	49.4	-4.7
PU236	0.0	0.0	0.0	0.0	0.0	0.0
PU238	99.3	96.5	-2.8	98.7	96.0	-2.8
PU239	1143.7	1096.3	-47.5	1140.5	1093.3	-47.2
PU240	958.9	936.1	-22.8	957.2	934.4	-22.7
PU241	184.5	175.3	-9.2	183.9	174.8	-9.1
PU242	288.6	282.1	-6.5	287.1	280.6	-6.5
AM241	88.1	83.4	-4.7	87.7	83.0	-4.7
AM242	6.3	6.3	-0.003	6.3	6.3	0.000
AM243	93.9	92.4	-1.482	93.4	91.9	-1.479
CM242	4.4	5.1	0.666	4.4	5.1	0.660
CM243	0.4	0.4	0.003	0.4	0.4	-0.003
CM244	61.4	61.8	0.459	66.4	66.4	-0.036
CM245	15.8	15.8	0.000	17.3	17.2	-0.042
CM246	8.8	8.8	0.000	9.8	9.8	-0.003
Total HM (kg)	9425.8	9202.7	-223.1	9419.4	9196.1	-223.3
Total TRU (kg)	3008.8	2910.2	-98.6	3007.2	2908.5	-98.7

A standard way of quantifying the consumption of TRU or MA for an ABR design is to compute the consumption rate of a certain system during a calendar year. This enables the modeling of important global fuel cycle characteristics for different GNEP scenarios, such as the support ratio (LWRs per ABR), repository benefits or drawbacks from certain strategies, and the pertinent economics of the scenario. While the two cases studied here, that of using a single tier recycling through either UREX+4 or UREX+1a separation processes, have the same total TRU consumption rate (166 kg/EFPY) and the same approximate TRU conversion ratio (0.56), they have different MA consumption rates (16.51 kg/EFPY for UREX+4 and 17.35 kg/EFPY for UREX+1a). This difference can have a significant impact from the point of view of repository benefits, especially since in UREX+4 case there is a net increase in curium at a rate of 1.90 kg/EFPY. On the other hand, comparison of strategies via total TRU-inventory or consumption rates is only a first step. Metrics of waste management, proliferation resistance, and energy recovery depend on the composition by element and isotope; such a comparison is beyond the scope of this report.

Table 14. TRU Net Consumption (kg/EFPY) for UREX+4 and UREX+1a Cases.

	UREX+4	UREX+1a
NP237	-7.96	-7.92
PU236	0.00	0.00
PU238	-4.72	-4.70
PU239	-80.07	-79.67
PU240	-38.52	-38.37
PU241	-15.51	-15.42
PU242	-11.04	-10.98
AM241	-7.95	-7.91
AM242	-0.01	0.00
AM243	-2.50	-2.50
CM242	1.12	1.11
CM243	0.01	-0.01
CM244	0.77	-0.06
CM245	0.00	-0.07
CM246	0.00	-0.01
Total TRU	-166.4	-166.5

6. Conclusions and Future Work

The goal of the INL effort summarized in this report was to establish and apply fast reactor fuel cycle analysis capabilities to a high-priority ABR prototype, which was based on the S-PRISM 1000 MWth design. By performing this task, we are able to systematically assess key transmutation technologies relevant to GNEP in the fast reactor area and at the same time understand and identify present and future additional modeling needs.

Using two available fast reactor fuel cycle analysis packages, MC^2-2 and REBUS-3, we have successfully constructed a model based on data from the ANL-AFCI-177 report and we have verified the validity of the results against two references. Excellent agreement was found between each model and differences were thoroughly understood. The main differences between the models studied are due to a difference in assumed fuel fabrication atom density and slightly different dimensions (accounting for thermal expansion).

After verification of the independent ABR model constructed at INL was completed, a study was performed in which two different LWR SNF separation technologies were modeled and their impact on the ABR design was assessed. The results show that for this specific ABR design, if a single tier recycling is desirable, then the separation of curium is undesirable. If the goal of the program is to reduce the amount of TRU (and specifically MA), it more desirable to use the UREX+1a separation technology than UREX+4, since it effectively consumes more MA than the latter. This is only an initial conclusion, since the evaluation of strategies must also include metrics of waste management, proliferation resistance, and energy recovery, all of which are beyond the scope of this report.

This report covers one of a number of perturbations to the TRU external feed are currently under study at INL. By quantifying not only the TRU conversion ratio, but the specific mass consumption of HM, TRU, and MA, for each of these perturbations, the fuel cycle analysis can provide data to evaluate which design is better suited for certain fuel cycle and repository goals.

In parallel to these studies, INL has also developed an ABR design with MA targets in the periphery of the core. This design is currently being evaluated and, along with perturbation studies on the TRU external feed, will be subject of an in-depth report later this year.

In the process of performing these analyses, we have identified a number of additional analysis capabilities required for further performance evaluation. These include the capability for modeling two-tier or multiple tier, synergistic, thermal and fast spectrum MA burning. Another necessary capability, in order to fully assess the repository strategy, is to track higher mass MA (such as heavier curium isotopes, berkelium and californium, all of which may have significant impact on the decay heat and toxicity of the waste destined to the repository). It is also evident from repeated usage of the computer code packages employed in this work, that major improvements in productivity can be achieved by adding further automation and standardizing input in order to allow the analyst to focus on the reactor physics and fuel cycle strategies. This can also minimize the probability of user input error.

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