PHYSOR 2012 – Advances in Reactor Physics – Linking Research, Industry, and Education
Knoxville, Tennessee, USA, April 15-20, 2012, on CD-ROM, American Nuclear Society, LaGrange Park, IL (2012)

NEUTRONICS AND DEPLETION METHODS FOR PARAMETRIC STUDIES OF FLUORIDE-SALT-COOLED HIGH-TEMPERATURE REACTORS WITH SLAB FUEL GEOMETRY AND MULTI-BATCH FUEL MANAGEMENT SCHEMES[1]

Anselmo T. Cisneros[2]
University of California Department of Nuclear Engineering
Berkeley, CA 94720-1730, USA
tommycisneros@berkeley.edu

Dan Ilas
Oak Ridge National Laboratory
P.O. Box 2008, Oak Ridge, TN 37831-6170, USA
ilasd@ornl.gov

ABSTRACT

The Advanced High-Temperature Reactor (AHTR) is a 3400 MW_{th} fluoride-salt-cooled high-temperature reactor (FHR) that uses TRISO particle fuel compacted into slabs rather than spherical or cylindrical fuel compacts. Simplified methods are required for parametric design studies such that analyzing the entire feasible design space for an AHTR is tractable. These simplifications include fuel homogenization techniques to increase the speed of neutron transport calculations in depletion analysis and equilibrium depletion analysis methods to analyze systems with multi-batch fuel management schemes.

This paper presents three elements of significant novelty. First, the Reactivity-Equivalent Physical Transformation (RPT) methodology usually applied in systems with coated-particle fuel in cylindrical and spherical geometries has been extended to slab geometries. Secondly, based on this newly developed RPT method for slab geometries, a methodology that uses Monte Carlo depletion approaches was further developed to search for the maximum discharge burnup in a multi-batch system by iteratively estimating the beginning of equilibrium cycle (BOEC) composition and sampling different discharge burnups. This Iterative Equilibrium Depletion Search (IEDS) method fully defines an equilibrium fuel cycle ($k_{eff,}$ power, flux, and composition evolutions) but is computationally demanding, although feasible on single-processor workstations. Finally, an analytical method, the Non-Linear Reactivity Model, was developed by expanding the linear reactivity model to include an arbitrary number of higher order terms so that single-batch depletion results could be extrapolated to estimate the maximum discharge burnup and BOEC k_{eff} in systems with multi-batch fuel management schemes. Results from this method were benchmarked against equilibrium depletion analysis results using the IEDS.

Key Words: fluoride-salt-cooled high-temperature reactor (FHR), high-temperature reactor (HTR), Reactivity-Equivalent Physical Transformation (RPT), slab geometry, multi-batch depletion, Linear Reactivity Model (LRM), Non-Linear Reactivity Model (NLRM)

---

[1] This manuscript has been authored by UT-Battelle, LLC, under contract DE-AC05-00OR22725 with the U.S. Department of Energy. The United States Government retains and the publisher, by accepting the article for publication, acknowledges that the United States Government retains a non-exclusive, paid-up, irrevocable, world-wide license to publish or reproduce the published form of this manuscript, or allow others to do so, for United States Government purposes. The U.S. Department of Energy Nuclear Criticality Safety Program sponsored the preparation of this paper.

[2] Corresponding author

1. INTRODUCTION

Optimizing a fuel management scheme is a challenging and critical aspect of designing a nuclear reactor system. Though multi-batch fuel cycles add complexity to a nuclear fuel cycle, they can significantly extend the discharge burnup and thereby increase fuel utilization and economics in a nuclear system. Modeling these systems is challenging because of the energy self-shielding effects that make neutron transport challenging, the lack of functionality in standard depletion tools to model multi-batch depletion systems, and the computational intensity of iterative methods to search for the equilibrium fuel cycle. The primary objective of this paper is to present methodologies to perform parametric studies for fuel design and fuel management schemes for the Advanced High-Temperature Reactor (AHTR).

The AHTR is a fluoride-salt-cooled reactor that utilizes TRISO particle fuel compacted into slabs of fuel [1]. Cross-sectional views through the AHTR core, fuel assembly, and fuel plate are shown in Figure 1. These TRISO particles are compacted into slabs that surround an inert graphite moderator and are encased in a graphite sleeve to form a fuel plate. Eighteen fuel plates are held between a Y-shaped control blade guide and a hexagonal box to form fuel assemblies. In total, the AHTR contains 252 or 253 of these fuel assemblies (the central fuel assembly can be removed such that the remaining fuel elements can be divided into an even number of burnup batches). The exact dimensions of the baseline AHTR are described explicitly in Reference 1.

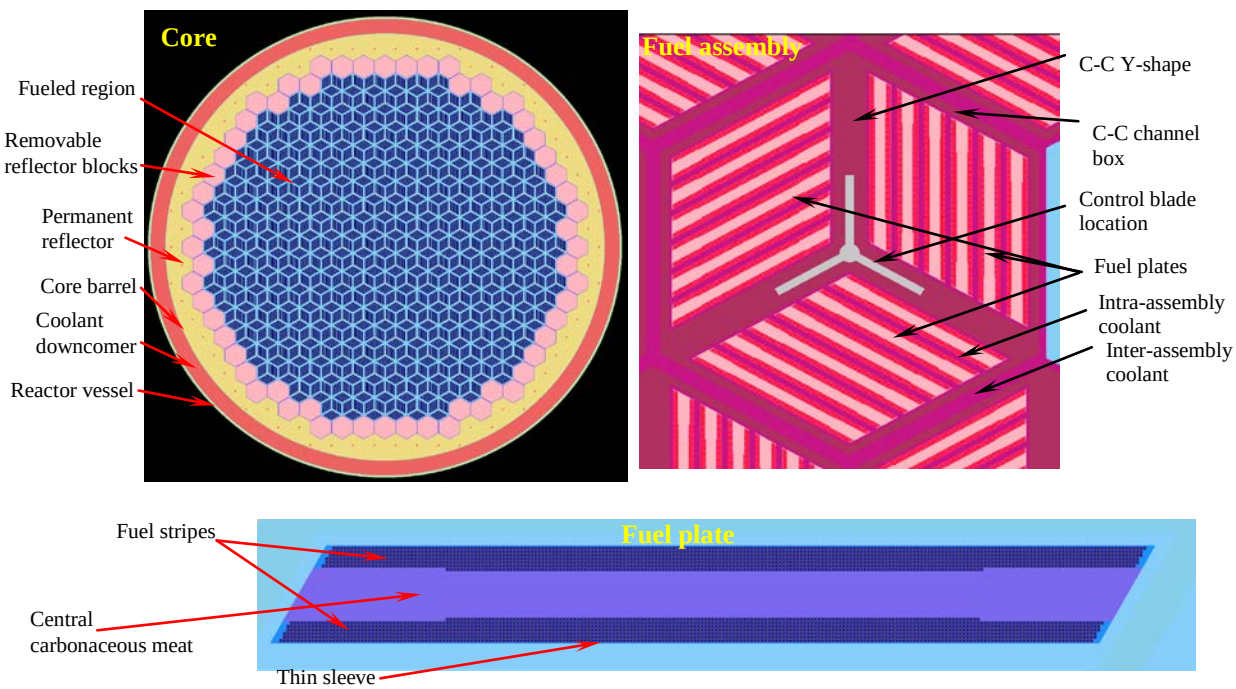

Figure 1. Schematic views of the AHTR core, fuel assembly, and fuel plate.

The required neutron transport for a parametric study would be arduous utilizing a neutronics model that modeled the fuel grains explicitly. Section 2 outlines the homogenization methodology developed to speed up the neutron transport computations required to estimate the maximum discharge burnup for specific combinations of fuel design and fuel management schemes – specifically the Reactivity-equivalent Physical Transform (RPT) [2] – and compares depletion results to those obtained with an explicit grain model of the AHTR. Section 3.1 presents a formalized methodology developed to iteratively search for the maximum burnup equilibrium cycle and describes its implementation into the SCALE package of nuclear system analysis tools with Monte Carlo neutron transport. Section 3.2 presents an analytical methodology to utilize single-batch depletion results to predict the maximum attainable burnup in a multi-batch fuel cycle. Though these equilibrium multi-batch depletion methods were developed to analyze the AHTR, they are presented as a means to analyze an arbitrary nuclear reactor with a multi-batch fuel management scheme.

2. REACTIVITY-EQUIVALENT PHYSICAL TRANSFORM

The RPT method is an efficient and effective way to model a coated-particle-fueled system [2] and can be implemented in most neutron transport codes without modifying the source code. The RPT method models the fuel in cylindrical and spherical compacts by collecting the fuel particles in a smaller RPT active region and then performing a volume-weighted material homogenization on the transformed active region.

The RPT method accounts for the increased resonance self-shielding of the particle fuel by increasing the density of actinides locally in the RPT homogenized region [2]. However, the extent to which this density must be increased must be calibrated against a high-fidelity simulation of the system that uses continuous energy and models fuel particles explicitly. This methodology has been implemented to study high-temperature gas-cooled reactors with spherical and cylindrical fuel compacts [2, 3], although no work has been performed with the RPT methodology to model systems with slab geometry or FHRs.

2.1. Implementation

Generating an RPT model of a particle fuel system in slab geometry is similar to the process in cylindrical or spherical geometries. The fuel particles are collected into smaller RPT active slabs and then particles and matrix in the RPT active region are homogenized by volume. These transforms can be performed for solid slab systems as well as annular slab systems. Additionally, layered slab systems can be approximated as solid slab systems by exchanging the position of the central moderator with the active region to further increase the self-shielding effects – the matrix and RPT active region conserve the same volume as the active region in the real system. Figure 2 presents the RPT homogenization process for a solid slab (top) and for a layered slab which is RPT-transformed into a layered (middle) or a solid (bottom) slab configuration.

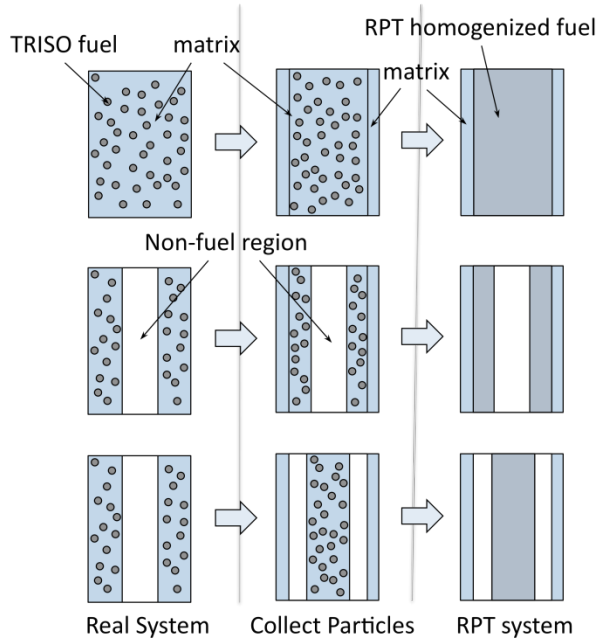

Figure 2. Volume homogenizations for RPT method in slab geometries. Top: solid slab; Middle: layered slab transformation; Bottom: solid slab approximation.

In addition to changing the geometry of system, one must generate an RPT homogenized concentration vector, N_{RPT}, for the active region. Let us define the RPT volume fraction (VF_{RPT}) as the ratio of volumes of the RPT active region to the active region in the real system and N_{fuel} as the composition vector of the coated fuel particles and matrix homogenized by volume. The RPT method populates the area vacated by the physical transform with matrix material, N_{matrix}, and homogenizes all the other material over a smaller RPT active region such that the masses of both the coated particles and the matrix are conserved between the real system and the RPT transform. Using the RPT volume fraction definition, the composition vector is obtained from conservation of mass as shown:

$$\begin{aligned} V_{active\ region} \cdot N_{fuel} &= V_{RPT} \cdot N_{RPT} + \left(V_{active\ region} - V_{RPT}\right) \cdot N_{matrix} \\ V_{active\ region} \cdot N_{fuel} &= V_{active\ region} VF_{RPT} \cdot N_{RPT} + V_{active\ region}\left(1 - VF_{RPT}\right) \cdot N_{matrix} \\ N_{fuel} - \left(1 - VF_{RPT}\right) \cdot N_{matrix} &= VF_{RPT} \cdot N_{RPT} \\ N_{RPT} &= \frac{1}{VF_{RPT}} N_{fuel} - \frac{\left(1 - VF_{RPT}\right)}{VF_{RPT}} N_{matrix} \end{aligned} \tag{1}$$

To select the extent of the volume reduction in the active region, the VF_{RPT} must be calibrated against a high-fidelity continuous-energy explicit grain model of the system that is to be modeled. To calibrate an RPT model, the k_{eff} of the system at the beginning of cycle (BOC) is first calculated with high fidelity using continuous-energy models with explicit grain representation to properly account for the double heterogeneity of the fuel. Then, a parametric

study at BOC is performed for the homogenized system to determine the VF_{RPT} that matches the k_{eff} of the RPT model with that of the high-fidelity model. Figure 3 presents the k_{eff} as a function of VF_{RPT} for a full-core model of the AHTR with the baseline fuel design for the layered slab and the solid slab approximation transformations; these multiplication constants have a statistical uncertainty within 100 pcm. The baseline AHTR TRISO fuel design used in this study was the AGR-2 fuel [4] with a packing fraction of 40% and an enrichment of 19.75wt% to maximize the refueling interval without introducing proliferation concerns.

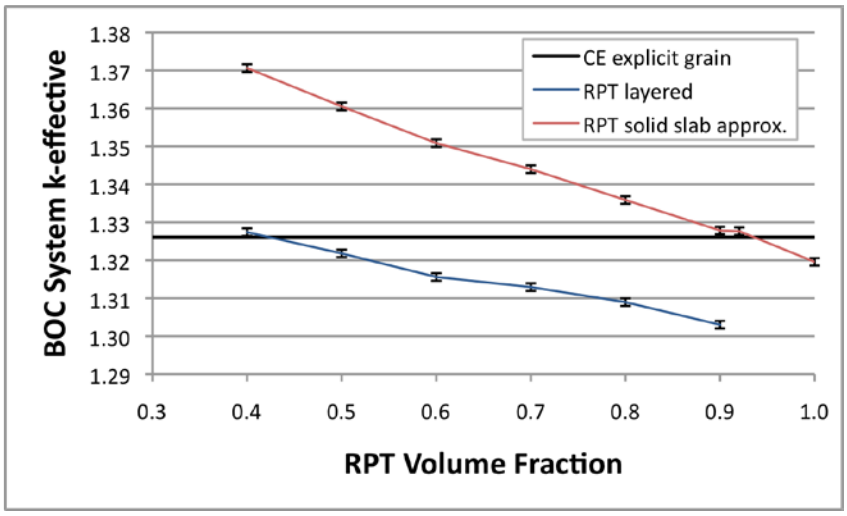

Figure 3. BOC *k*-effective sensitivity to RPT volume fraction.

The baseline fuel design has significant inherent self-shielding because of the high fuel-to-moderator ratio, and as a result, the double-heterogeneity effect only weakly increases the resonance self-shielding in this system. Systems with softer neutron spectra, either from additional moderation or because of lower enrichment, require more volume reduction to induce the correct amount of self-shielding. Notice in Figure 3 that the RPT annular-slab transform requires a greater volume reduction to match the k_{eff} of the benchmark. The solid slab approximation RPT transform is able to induce more self-shielding because more of the fuel is grouped together. This effect enables the solid slab approximation RPT transform to be used for modeling systems with softer neutron spectra where the annular slab RPT transform might fail.

2.2. Verification

To determine to what extent the RPT methodology can be utilized for depletion analysis of the AHTR, single-batch depletion results obtained using the layered slab RPT transform and solid slab approximation RPT transform were compared to results obtained using a multi-group explicit grain model of AHTR in conjunction with a Dancoff factor method for the baseline fuel design [5]. The verification efforts compared the predicted maximum discharge burnup, the k_{eff} evolution, and the evolution of a set of critical isotopes. Additionally, the reduction in computational intensity attributed to using these simplified models was quantified.

2.2.1. Neutronics model

The neutron transport calculations were performed using the 3-D Monte Carlo neutron transport code KENO-VI as an engine in the depletion calculations of the TRITON sequence of SCALE [6]. TRITON sequentially executes a series of modules from the SCALE package, beginning with the cross section processing modules (CRAWDAD, BONAMI, WORKER, CENTRM, PMC) to create problem-dependent self-shielded multi-group cross section libraries. The multi-group cross section libraries thus created are then used in the KENO-VI module to generate a transport solution (fluxes and k_{eff}) for the system, which is further post-processed by KMART. Finally, the fluxes from the transport calculations are used (after preparation by COUPLE) in the ORIGEN-S code to deplete the fuel. These steps are repeated for the next time steps until the time evolution of the system is completed. A multi-group explicit grain model of the AHTR was used as a high-fidelity benchmark model against which the RPT depletion results are compared. In this explicit grain benchmark, TRISO particles are modeled on a simple cubic lattice. The buffer layer and inner pyrolytic carbide layer are homogenized together as are the outer pyrolytic carbide layer and the graphite matrix; however, the fuel kernels and the silicon carbide layers are not altered. For the purpose of this analysis, the reactor was divided into two radial zones, with the central zone containing the innermost four rings of fuel assemblies, and three axial zones, with the central axial zone containing 3/5 of the total height of the fueled core. Each depletion step used 10,000 active neutron histories for the neutron transport calculation resulting in a statistical error of 100 pcm in the system k_{eff}.

2.2.2. Discharge burnup

The maximum discharge burnup for a reactor with a single-batch fuel management scheme is estimated by interpolating to the point where $k_{eff} = 1$. These results are presented for both models with both RPT transforms in Table I.

Table I. Estimated maximum burnup for single-batch AHTR with a single-batch fuel management scheme and baseline fuel design for various models

Model	Estimated discharge burnup (GWd/MT)	Error (%)
Explicit grain	79.1	-
Multi-group RPT layered slab	80.9	+2.4
Multi-group RPT solid slab approximation	81.7	+3.3

2.2.3. *k*-effective evolution

The k_{eff} evolution predicted by performing depletion analysis with various models was compared for the AHTR depletion with a single-batch fuel management scheme and the baseline fuel design using a full-core model, as shown in Figure 4; sufficient histories were used to limit statistical error within 100 pcm for each model.

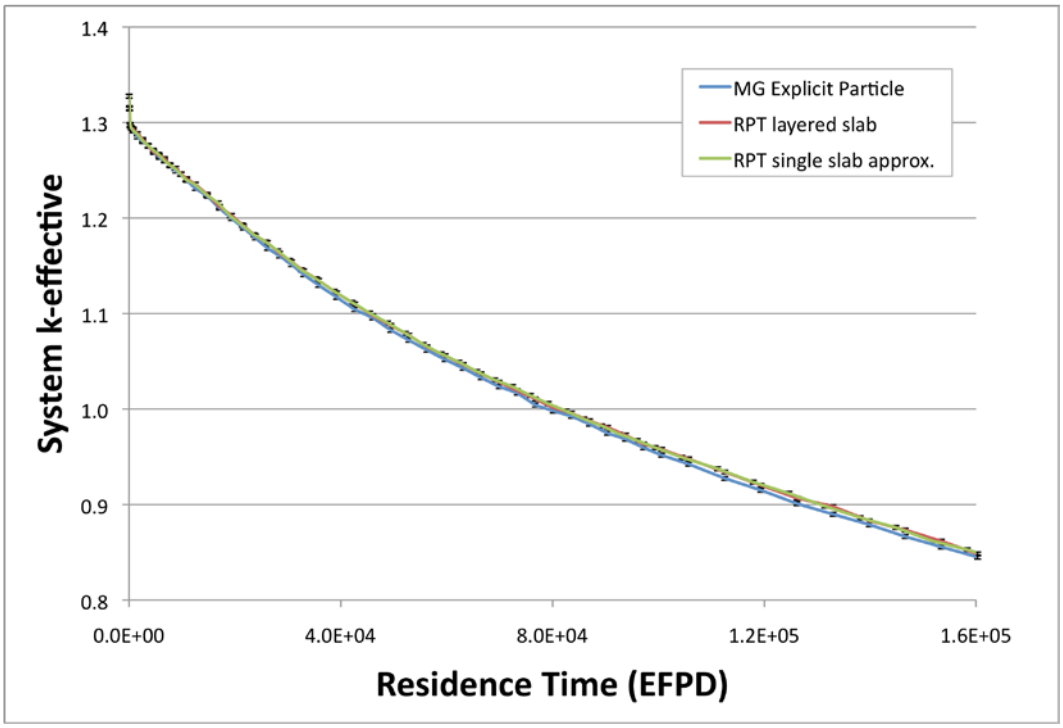

Figure 4. Comparison of predicted *k*-effective evolutions.

2.2.4. Composition evolution

To ensure that the maximum discharge burnup and k_{eff} evolution were predicted based on correct physics, the composition evolution for a set of important isotopes (^{235}U, ^{238}U, ^{239}Pu, and ^{240}Pu) were compared for the depletion zones described in Section 2.2.1. While these zones are rather coarse, they provide confirmatory comparisons. Increasing the number of depletion zones would add significantly to the computational time. The isotopic concentration evolutions for the depletion zone in the center radially and axially are presented in Figure 5.

These composition evolutions confirm that the evolutions for the RPT simplifications match the reference results.

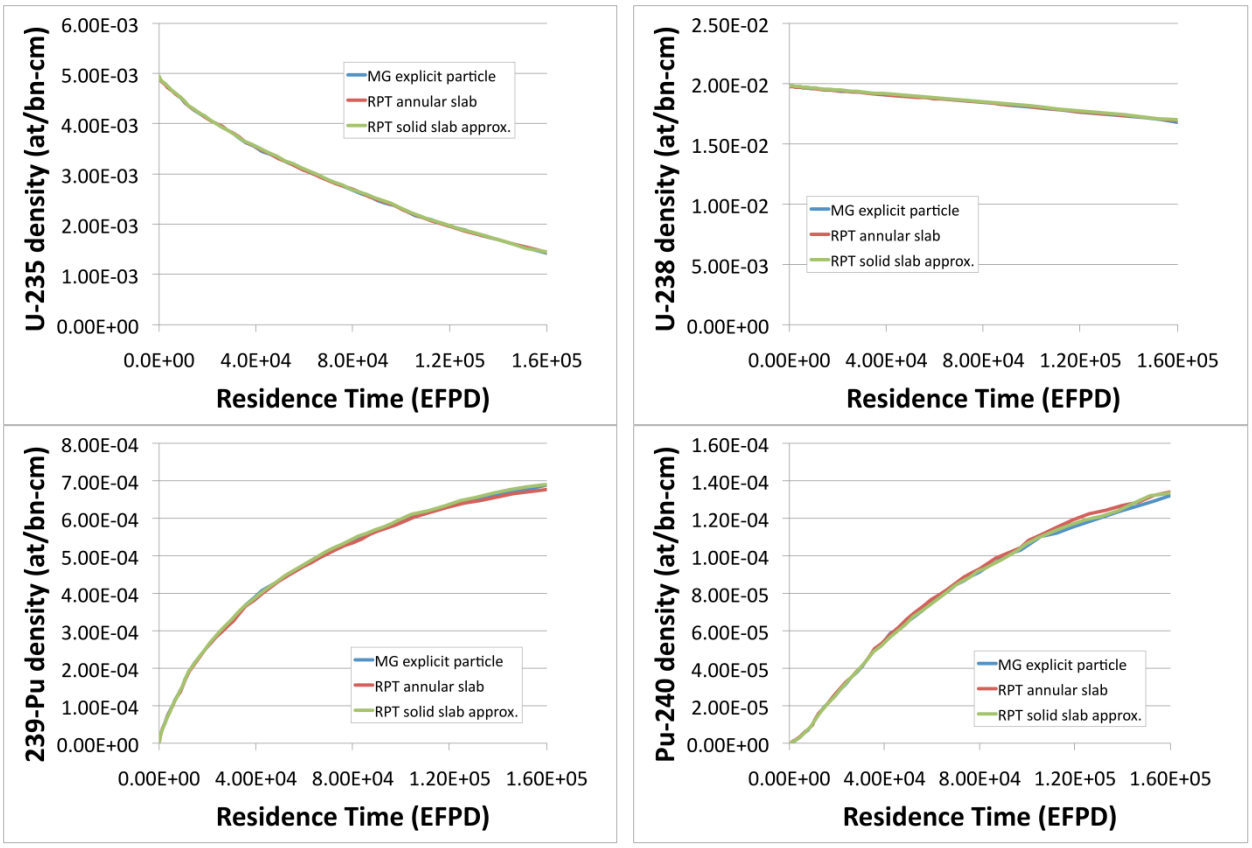

Figure 5. Isotopic concentration evolutions in AHTR baseline (central radial position, central axial position).

2.2.5. Reduction in computational requirements

The motivation behind implementing the RPT methodology for depletion analysis is to reduce the computation time required to perform depletion analysis. Therefore, significant reductions in the computational time are required to justify the error introduced by implementing the RPT geometric simplification. The time required to perform the depletion analysis of the AHTR with baseline fuel and a single-batch fuel cycle was used as a metric of computational requirements. The comparison of computational requirements is presented in Table II, where the comparisons for the RPT models are shown relative to the explicit grain model reference.

Table II. Computations requirements for depletion analysis

Model	Comp. time (days)	Speed up	Burnup error (%)
Explicit grain	17.2	-	-
Multi-group RPT annular slab	1.7	10.4	+2.3
Multi-group RPT solid slab approximation	0.8	20.3	+3.3

3. EQUILIBRIUM DEPLETION ANALYSIS METHODS

Two approaches were developed to determine the maximum discharge burnup for multi-batched fuel cycles: the Iterative Equilibrium Depletion Search (IEDS) and the Non-Linear Reactivity Model (NLRM). The IEDS methodology was developed to explicitly search for the maximum burnup of a multi-batch fuel cycle by imposing the maximum burnup condition [i.e., the k_{eff} at the end of equilibrium cycle (EOEC) should be 1.0] and searching for the beginning of equilibrium cycle (BOEC) composition for equilibrium cycles with different discharge burnups. Alternatively, an analytical methodology, the NLRM, was investigated to estimate maximum burnup based on the depletion results from single-batch calculation. This section describes both of these methodologies and compares the maximum burnup predicted by each of them.

3.1. Iterative Equilibrium Depletion Search

3.1.1. Iterative equilibrium analysis search methodology

The IEDS methodology utilizes a two-tiered solver to explicitly define the maximum burnup equilibrium cycle. The first tier is a search for the equilibrium fuel cycle that simultaneously determines the BOEC composition, as well as the evolutions of power, flux, and k_{eff} for a cycle with a given discharge burnup. The second tier searches for the maximum discharge burnup by systematically sampling EOEC k_{eff} as a function of burnup until a burnup is identified that meets the maximum discharge burnup condition. The IEDS algorithm is presented below in Figure 6.

In the multi-batch systems, the BOEC composition is burned for a single refueling interval, the oldest (most depleted) batch of fuel is discharged, the remaining fuel is shuffled, and fresh fuel is added to the core. The change to the fuel composition, N, over the irradiation campaign can be modeled by a differential equation.

$$\frac{dN}{dt} = -A(batch,t)N \quad . \tag{2}$$

The depletion matrix, A, contains the depletion parameters including decay constants, cross sections, and flux distributions as a function of batch and time in the equilibrium cycle.

This differential equation can be integrated from BOEC to EOEC to determine the EOEC composition,

$$\int_{BOEC}^{EOEC} \frac{dN}{N} = \int_{BOEC}^{EOEC} -A(batch,t)dt$$

$$N_{EOEC} = N_{BOEC} \cdot \exp\left(-\int_{BOEC}^{EOEC} A(batch,t)dt \right) \quad . \tag{3}$$

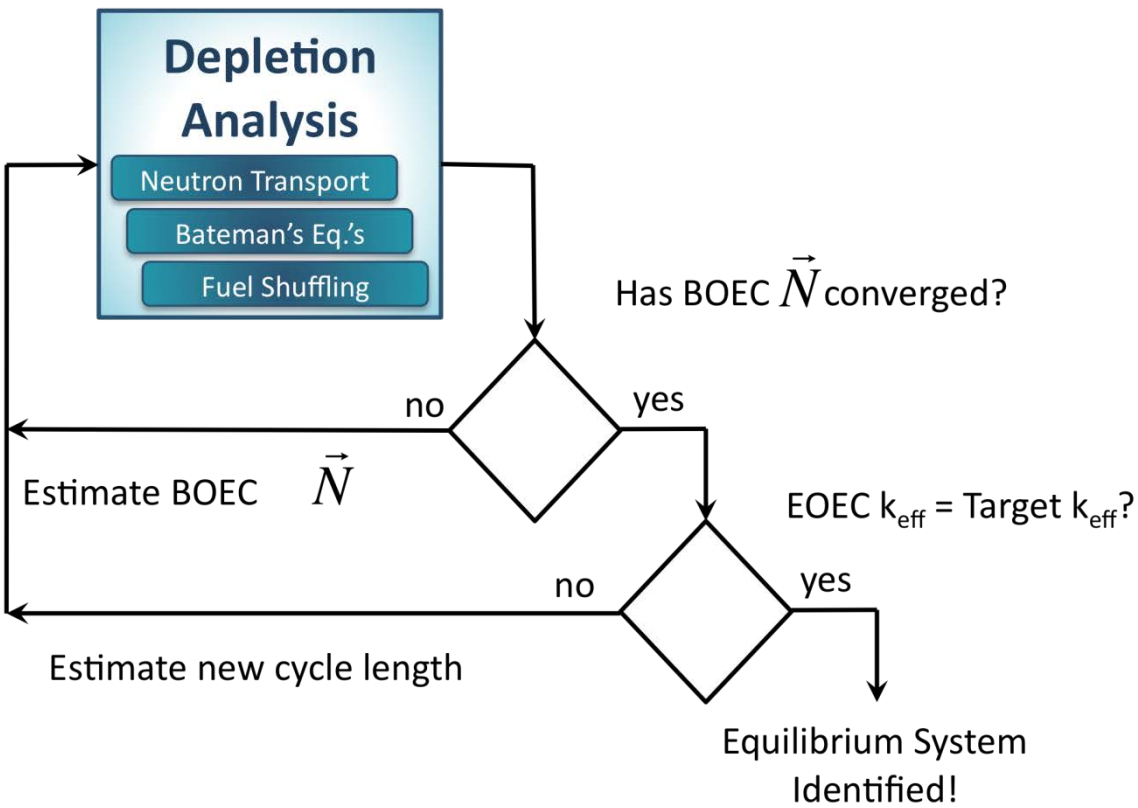

Figure 6. Algorithm for the Iterative Equilibrium Depletion Search.

The BOEC composition can then be estimated by shuffling the EOEC fuel and loading fresh fuel into the core as

$$N_{BOEC} = m_{shuffling} \cdot N_{EOEC} + m_{fresh\,fuel} \cdot N_{makeup} \tag{4}$$

Here, $m_{shuffling}$ is a matrix that advances each batch of fuel. Combining equations 3 and 4, the equilibrium cycle requirement can be defined as follows:

$$N_{BOEC} = m_{shuffling} \cdot N_{BOEC} \exp\left(- \int_{BOEC}^{EOEC} A(batch,t)dt \right) + m_{fresh\,fuel} \cdot N_{makeup} \tag{5}$$

Equation (5) is solved iteratively assuming a constant discharge burnup, starting with a guess for the BOEC composition (first tier in Figure 6). The concentrations of a set of critical isotopes and the EOEC k_{eff} are checked for convergence within an acceptable error after each iteration; the EOEC k_{eff} is used as a metric for the convergence of all the isotopes not in the set of important isotopes. Once these convergence criteria are met, the EOEC k_{eff} is passed to the second tier of the IEDS algorithm. This IEDS approach is similar to the methodology implemented in REBUS-3 and MRTAU [7, 8].

The goal of the second tier of the IEDS algorithm is to systematically sample the EOEC k_{eff} as a function of burnup to find the burnup that imposes the maximum burnup condition, EOEC k_{eff} =1.0, by initiating the search for equilibrium fuel cycle for various discharge burnups. Usually, EOEC k_{eff} is well behaved as a function of discharge burnup for the AHTR and can be approximated as a linear function around the maximum burnup. After the first two points of EOEC k_{eff} and burnup are generated, the maximum burnup can be determined by interpolating or extrapolating to a discharge burnup that imposes the EOEC k_{eff} to 1.0 and the depletion cycle is repeated. If the EOEC k_{eff} is not equal to 1.0, the procedure is repeated until the maximum burnup condition is satisfied.

3.1.2. Implementation with SCALE

The equilibrium cycle evaluation was performed using the TRITON module of SCALE. A driver program, SCALE Equilibrium Analysis Utility (SEAU), was developed to iteratively execute TRITON, shuffle EOEC fuel, insert fresh fuel, assess the convergence of the BOEC composition, and pass the EOEC k_{eff} to the next iteration loop once the BOEC composition has converged.

SEAU includes a module to automatically initiate the search for equilibrium cycle and interpolate or extrapolate once EOEC k_{eff} is sampled twice. The search is completed once an equilibrium fuel cycle is identified with an EOEC k_{eff} =1.0 within an acceptable error. These data points can be used to make a linear regression for EOEC k_{eff} as a function of discharge burnup and this function can be evaluated to estimate the maximum discharge burnup.

As stated earlier, there are two criteria for the convergences of the equilibrium fuel cycle: the EOEC k_{eff} and the concentrations of a set of important isotopes. For the equilibrium depletion analysis performed for this project, the EOEC k_{eff} is required to converge within 0.002 – though a more rigorous convergence criterion could be implemented if desired, at the expense of a longer computational time. The concentrations of the isotopes in the set of critical isotopes between estimates of the BOEC composition were required to converge within 1% for each isotope for every isotope in each depletion zone in the model. The set of important isotopes was limited to ^{235}U, ^{238}U, and ^{239}Pu. Both the precision and the set of isotopes can be easily modified by the user.

3.2. The Non-linear Reactivity Model

3.2.1. The non-linear reactivity methodology

Previous multi-batch fuel cycle parametric studies were performed using the Linear Reactivity Model (LRM)[9] to estimate the maximum attainable burnup of a system based on the depletion results of a single-batch system in light-water-reactor systems [10] and FHRs [11]. The LRM assumes that reactivity decreases linearly with burnup, b, and the aggregate reactivity, $\rho_{aggregate}$, is an average of the reactivities in each batch of fuel (i.e., burnup-dependent reactivity) [9].

$$\rho_{aggregate} = \frac{1}{n}\sum_{i=1}^{n}\rho(b_i)$$
$$\rho(b) = y^o + y'b \quad . \tag{6}$$

One can utilize a unit cell model if one takes leakage into account by subtracting a leakage reactivity from the weighted summation of reactivities. However, only full-core models are utilized in this study.

The accuracy of the LRM model can be increased and its applicability expanded to more systems when higher order polynomials are used to model the evolution of reactivity. Recent studies have expanded the LRM to include higher order terms [10, 12], but the details in these studies are not well defined. The maximum discharge burnup can be estimated using NLRM assuming an mth order polynomial and n equal-size reloads per fuel cycle, as shown below.

$$\rho(b) = y^o + y'b + \ldots + y^{(m)}b^m$$
$$\rho_{EOEC} = \frac{1}{n}\sum_{i=1}^{n}\rho(b_i^{EOEC})$$
$$\rho_{EOEC} = \frac{1}{n}\sum_{i=1}^{n}\rho\left(\frac{ib_{discharge}}{n}\right)$$
$$\rho_{EOEC} = \frac{1}{n}\sum_{i=1}^{n}y^o + y'\left(\frac{ib_{discharge}}{n}\right) + \ldots + y^{(m)}\left(\frac{ib_{discharge}}{n}\right)^m$$
$$\rho_{EOEC} = y^o + \frac{y'b_{discharge}}{n^2}\sum_{i=1}^{n}i + \ldots + \frac{y^{(m)}b_{discharge}^m}{n^{m+1}}\sum_{i=1}^{n}i^m$$
$$0 = y^o + \frac{y'b_{\max}}{n^2}\sum_{i=1}^{n}i + \ldots + \frac{y^{(m)}b_{\max}^m}{n^{m+1}}\sum_{i=1}^{n}i^m \quad . \tag{7}$$

Here, $y^o - y^{(m)}$ are coefficients of an mth order polynomial expansion of reactivity with respect to the burnup, b, and the summation index i is the batch number for a fuel management scheme with n batches.

This methodology gives a simple analytic expression for the maximum burnup in terms of the reactivity polynomial coefficients that can easily be evaluated algebraically for low-order models (first- to third-order models) and numerically for higher order models.

3.2.2. Benchmark of the NRLM for predicting the maximum discharge burnup

The order of polynomial expansion used to represent reactivity as a function of burnup will affect the accuracy of the maximum discharge burnup predicted by the NLRM. Figure 7 compares the reactivity evolution predicted by polynomials with different orders to the benchmark depletion results. The model used for Figure 7 utilizes a layered slab RPT homogenized model of the

plate-type fuel implemented into a full-core model of the AHTR; sufficient statistics were used to limit the statistical uncertainty in the system reactivity below 100 pcm.

To assess the accuracy of the NRLM for predicting maximum discharge burnup, a comparison with the results predicted by the IEDS was performed for the baseline model of the AHTR for fuel management schemes with different numbers of batches, as shown in Figure 8; these IEDS calculations assumed a full-core model of the AHTR utilizing the layered slab RPT homogenization. Furthermore, the accuracy of using interpolating polynomials of different orders for the single-batch reactivity was estimated.

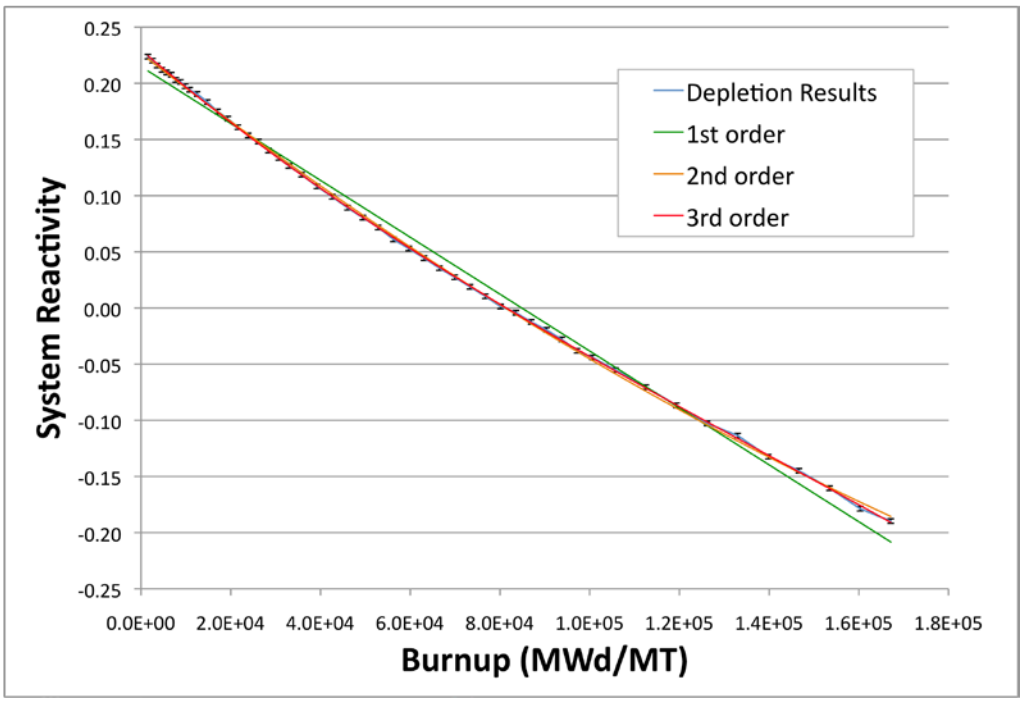

Figure 7. Comparison of Polynomial Expansions of Reactivity Evolution.

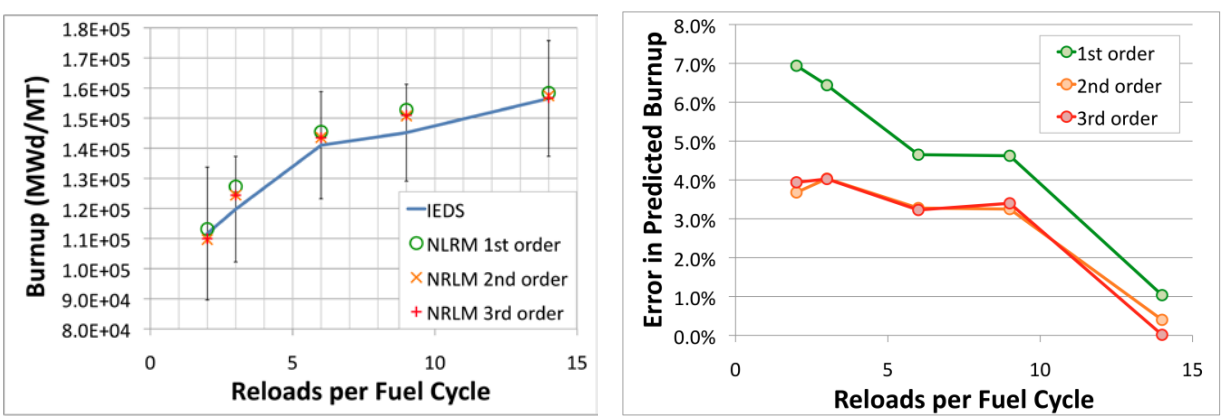

Figure 8. Comparison of maximum discharge burnup predicted by IEDS to NRLM (left) and associated error (right).

The second-order NRLM predicts the maximum discharge burnup within 4%, and this error is deemed acceptable for qualitatively sampling the design space for a multi-batch fuel cycle. Figure 7 shows that, for the reference AHTR core, there is no gain in accuracy associated with utilizing a third-order regression relative to a second-order representation of reactivity as a function of burnup. The large uncertainty in the IEDS is associated with the fact that only a limited number of data points were used to calculate the EOEC k_{eff} v. discharge burnup in the regression used to estimate the maximum burnup.

3.2.3. Computational speed up due to NRLM

The IEDS methodology requires iterative calculation of the BOEC composition vector depletion. At a minimum, the BOEC composition should be depleted (an iteration should be performed) once for every batch of fuel so that each batch of fuel in the model has the correct exposure time. Complicated fuel management schemes increase the number of depletion zones and, as a result, increase convergence difficulties. Conversely, the NRLM only requires a single depletion calculation that can be used to estimate the discharge burnup for fuel management schemes with an arbitrary number of batches.

Table III compares the computational requirements for the equilibrium depletion analysis for a few cases to highlight the significant reduction in computational requirements by using the NRLM methodology.

Table III. Computations requirements for depletion analysis

Equilibrium depletion method	Batches	Comp. time (days)	Burnup error (%)
Iterative Equilibrium Depletion Search	2	6.9	-
Iterative Equilibrium Depletion Search	6	20.7	-
Non-Linear Reactivity Model	2	1.0	3.7
Non-Linear Reactivity Model	6	1.0	3.3

4. CONCLUSIONS

The RPT method is a geometry simplification that is used to account for the double heterogeneity self-shielding effect observed in the neutron transport of nuclear systems with particle fuel. The RPT method has previously been implemented in cylindrical and spherical systems and can also be applied to slab geometry to model the plate-type fuel in the AHTR. These layered slab systems can be modeled with two different kinds of RPT transforms, a layered slab transform or a solid slab approximation transform. The solid slab approximation transformation allows more systems with softer spectra to be modeled than the layered slab RPT transform. The results between the depletion analysis using these RPT transforms and a multi-group explicit grain

model of the AHTR matched well with respect to predicted maximum discharge burnup, k_{eff} evolution, and the evolution of isotopic concentrations. Utilizing the RPT method for depletion analysis increases the speed of depletion analysis by 10–20 times. Therefore, the RPT can be utilized for depletion analysis in parametric studies if one accounts for the uncertainty introduced by utilizing the RPT method.

A methodology for assessing the maximum discharge burnup of the AHTR with a multi-batch fuel management scheme is presented. This methodology was implemented in SCALE by using the driver program SEAU.

The concepts of the linear reactivity model can be expanded to a broader set of systems by introducing higher order terms as in the NRLM. The NRLM can utilize the single-batch depletion analysis results to predict the maximum discharge burnup in a multi-batch fuel cycle using a second-order regression of reactivity as a function of burnup. The maximum burnup predicted by NRLM matches the results predicted by the IEDS with an accuracy of 4%. Therefore, this methodology can be used to quickly and effectively predict the maximum discharge burnup and refueling intervals, making parametric studies over large design spaces feasible for systems with multi-batch fuel management schemes.

ACKNOWLEDGMENTS

This material is based upon work supported under a Department of Energy Nuclear Energy University Programs Graduate Fellowship. Any opinions, findings, conclusions or recommendations expressed in this publication are those of the author(s) and do not necessarily reflect the views of the Department of Energy.

The work performed by the first author was supported by the Nuclear Engineering Science Laboratory Synthesis (NESLS) internship program at Oak Ridge National Laboratory. The AHTR concept is under development at ORNL as part of U.S. DOE's Advanced Reactor Concepts program. The authors acknowledge the suggestions of J.C. Gehin and B.J. Ade during the ORNL internal review process.

REFERENCES

1. D. Holcomb, D. Ilas, V. K. Varma, A. T. Cisneros, R. P. Kelly, J. C. Gehin, *Core and Refueling Design Studies for the Advanced High Temperature Reactor*, ORNL/TM-2011/365 (2011).
2. Y. Kim, M. Baek, "Elimination of Double-Heterogeneity through a Reactivity-Equivalent Physical Transformation," *Proc. of Global 2005*, Tsukuba, Japan Oct 9–13, Paper 548 (2005).
3. Y. Kim, F. Venneri, "Optimization of One-Pass Transuranium Deep Burn in a Modular Helium Reactor," *Nuclear Science and Engineering*, **160**[1], pp. 59–74 (2008).
4. J. Philips, C. Barnes, J. Hunn "Fabrication and Comparison of Fuels for Advanced Gas Reactor Irradiation Tests," *Proc. HTR 2010*, Progue, Czech Republic, October 18–20 (2010).

5. R.P. Kelly, D. Ilas, "Verification of a Depletion Method in SCALE for the Advanced High Temperature Reactor," *Proceedings of PHYSOR 2012 – Advances in Reactor Physics*, Knoxville, TN April 15-20, 2012.
6. *SCALE: A Comprehensive Modeling and Simulation Suite for Nuclear Safety Analysis and Design*, ORNL/TM-2005/39, Version 6.1, Oak Ridge National Laboratory, Oak Ridge, TN (2011). (Available from Radiation Safety Information Computational Center at Oak Ridge National Laboratory as CCC-785.)
7. B. J. Toppel, *A User's Guide to the REBUS-3 Fuel Cycle Analysis Capability*, ANL-83-2, Argonne National Laboratory (1983).
8. S. Bays, S. Piet, M. Pope, G. Youinou, A. Dumontier, D. Hawn, *Transmutation Dynamics: Impacts of Multi-Recycling on Fuel Cycle Performances*, INL/EXT-09-16857, Idaho National Laboratory (2009).
9. M. J. Driscoll et al., *The Linear Reactivity Model for Nuclear Fuel Management*, American Nuclear Society (1990).
10. O. H. Zabunoglu, "Determination of Enrichment of Recycle Uranium Fuels for Different Burnup Values," *Annals of Nuclear Energy*, **35**, 285–290 (2008).
11. T. K. Kim, T. A. Taiwo, W. S. Yang, *Preliminary Neutronic Studies for the Liquid-Salt-Cooled Very High Temperature Reactor (LS-VHTR)*, ANL-GenIV-052, Argonne National Laboratory (2005).
12. E. Friedman, S. Kolesnikov, E. Shwageraus, E. Galperin, *Dissolution, Reactor and Environmental Behavior of ZrO2-MgO Inert Fuel Matrix: Neutronic Evaluation of MgO-ZrO_2 Inert Fuels*, Department of Nuclear Engineering, Ben-Gurion University of the Negev, Final Report for the University of Nevada, Las Vegas (2005).