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Pebble Bed Reactors Design Optimization Methods and their Application to the Pebble
Bed Fluoride Salt Cooled High Temperature Reactor (PB-FHR)

By

Anselmo Tomas Cisneros Jr

A dissertation submitted in partial satisfaction

of the requirements for the degree of

Doctor of Philosophy

in

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in

the Graduate Division of the

University of California, Berkeley

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Fall 2013

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Bed Fluoride Salt Cooled High Temperature Reactor (PB-FHR)

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Anselmo Tomas Cisneros Jr

Abstract

Pebble Bed Reactors Design Optimization Methods and their Application to the Pebble Bed Fluoride Salt Cooled High Temperature Reactor (PB-FHR)

By

Anselmo Tomas Cisneros Jr

Doctor of Philosophy in Engineering – Nuclear Engineering

University of California, Berkeley

Professor Ehud Greenspan, Chair

The Fluoride salt cooled High temperature Reactor (FHR) is a class of advanced nuclear reactors that combine the robust coated particle fuel form from high temperature gas cooled reactors, direct reactor auxiliary cooling system (DRACS) passive decay removal of liquid metal fast reactors, and the transparent, high volumetric heat capacitance liquid fluoride salt working fluids – flibe (33%⁷Li₂F-67%BeF) – from molten salt reactors. This combination of fuel and coolant enables FHRs to operate in a high-temperature low-pressure design space that has beneficial safety and economic implications. In 2012, UC Berkeley was charged with developing a pre-conceptual design of a commercial prototype FHR – the Pebble Bed- Fluoride Salt Cooled High Temperature Reactor (PB-FHR) – as part of the Nuclear Energy University Programs' (NEUP) integrated research project. The Mark 1 design of the PB-FHR (Mk1 PB-FHR) is 236 MWt flibe cooled pebble bed nuclear heat source that drives an open-air Brayton combine-cycle power conversion system. The PB-FHR's pebble bed consists of a 19.8% enriched uranium fuel core surrounded by an inert graphite pebble reflector that shields the outer solid graphite reflector, core barrel and reactor vessel. The fuel reaches an average burnup of 178000 MWt-d/MT. The Mk1 PB-FHR exhibits strong negative temperature reactivity feedback from the fuel, graphite moderator and the flibe coolant but a small positive temperature reactivity feedback of the inner reflector and from the outer graphite pebble reflector.

A novel neutronics and depletion methodology – the multiple burnup state methodology was developed for an accurate and efficient search for the equilibrium composition of an arbitrary continuously refueled pebble bed reactor core. The Burnup Equilibrium Analysis Utility (BEAU) computer program was developed to implement this methodology. BEAU was successfully benchmarked against published results generated with existing equilibrium depletion codes VSOP and PEBBED for a high temperature gas cooled pebble bed reactor.

Three parametric studies were performed for exploring the design space of the PB-FHR -- to select a fuel design for the PB-FHR] to select a core configuration; and to optimize the PB-FHR design. These parametric studies investigated trends in the dependence of important reactor performance parameters such as burnup, temperature reactivity feedback, radiation damage, etc on the reactor design variables and attempted to understand the underlying reactor physics responsible for these trends.

A pebble fuel parametric study determined that pebble fuel should be designed with a carbon to heavy metal ratio (C/HM) less than 400 to maintain negative coolant temperature reactivity coefficients.

Seed and thorium blanket-, seed and inert pebble reflector- and seed only core configurations were investigated for annular FHR PBRs – the C/HM of the blanket pebbles and discharge burnup of the thorium blanket pebbles were additional design variable for core configurations with thorium blankets. Either a thorium blanket or graphite pebble reflector is required to shield the outer graphite reflector enough to extend its service lifetime to 60 EFPY. The fuel fabrication costs and long cycle lengths of the thorium blanket fuel limit the potential economic advantages of using a thorium blanket. Therefore, the seed and pebble reflector core configuration was adopted as the baseline core configuration.

Multi-objective optimization with respect to economics was performed for the PB-FHR accounting for safety and other physical design constraints derived from the high-level safety regulatory criteria. These physical constraints were applied along in a design tool, Nuclear Application Value Estimator, that evaluated a simplified cash flow economics model based on estimates of reactor performance parameters calculated using correlations based on the results of parametric design studies for a specific PB-FHR design and a set of economic assumptions about the electricity market to evaluate the economic implications of design decisions.

The optimal PB-FHR design – Mark 1 PB-FHR – is described along with a detailed summary of its performance characteristics including: the burnup, the burnup evolution, temperature reactivity coefficients, the power distribution, radiation damage distributions, control element worths, decay heat curves and tritium production rates. The Mk1 PB-FHR satisfies the PB-FHR safety criteria. The fuel, moderator (pebble core, pebble shell, graphite matrix, TRISO layers) and coolant have global negative temperature reactivity coefficients and the fuel temperatures are well within their limits.

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1 Introduction

1.1 Fluoride Salt Cooled High Temperature Reactors (FHRs)

FHR is an advanced reactor concept that combines the robust coated particle fuel form from high temperature gas cooled reactors, direct reactor auxiliary cooling system (DRACS) passive decay removal of liquid metal fast reactors, and the transparent, high volumetric heat capacitance liquid fluoride salt working fluids – flibe ($33\%{}^7\text{Li}_2\text{F}-67\%\text{BeF}$) – from molten salt reactors. This combination of fuel and coolant enables FHRs to operate in a high-temperature low-pressure design space that is very economically attractive. High temperatures increase thermal efficiency, which in turn increases electricity output for a given investment in capital, uranium resources and radioactive waste disposal. Moreover, high temperatures open up new processes heat markets for nuclear energy. Producing energy for the industrial sector would expand the market for nuclear energy by approximately 100%, increasing the opportunity for profits and reducing greenhouse-gas emissions. Operating at atmospheric pressures reduces capital costs by allowing the utilization of thin-walled reactor vessels – this also increases flexibility in terms of transportation and fabrication. Operating in the low-pressure regime also reduces the driving mechanism – advection by pressurized water or helium – to transport radioactive material from the core to the public.

1.1.1 TRISO Particle Fuel

TRISO particle fuel was originally developed for high temperature gas-cooled reactors (HTGRs) (Todreas & Kazimi, 1990). TRISO particle fuel compacts consist of small – 200 μm - 800 μm diameter – fuel kernels encased in a several structural layers – porous graphite buffer layer, an inner pyrolytic graphite layer, a silicon carbide layer, and an outer pyrolytic graphite layer – dispersed in an inert graphite matrix. Fuel kernel designs have been proposed utilizing low enriched uranium (LEU), highly enriched uranium (HEU), thorium or even plutonium nuclear fuels. The buffer layer accommodates fission recoils, fission gases and kernel swelling. Together the pyrolytic graphite and silicon carbide layers form a “mini-pressure vessel” around the fuel kernel to prevent the release of fission products. TRISO fuel can survive operating temperatures up to 1200 °C (Todreas & Kazimi, 1990) and accident conditions up to 1800 °C. (World Nuclear new, 2013)

TRISO particle fuel technology is being developed for the Next Generation Nuclear Plant by the Department of Energy. The Advanced Gas Reactor (AGR) Fuel program is performing fuel irradiation experiments in the Advanced Test Reactor (ATR) at Idaho National Laboratory to qualify TRISO particle fuel for nuclear power applications (Grover, Petti, & Davenport, 2013).

1.1.2 Liquid Salt Coolant

Liquid salts were originally used in fluid fueled systems – the Aircraft Reactor Experiment and the Molten Salt Reactor program. The Molten Salt Reactor Experiment experienced significant corrosion due to fission products in its ${}^7\text{LiF}-\text{BeF}_2-\text{ZrF}_4-\text{UF}_4$

primary loop. However, the intermediate loop with clean salt (similar to the operating mode of FHRs) experienced no detectable corrosion. Corrosion experiments performed at the University of Wisconsin – Madison show low corrosion rates in FHR surrogate systems composed of liquid fluoride salt, Stainless Steel and graphite (Zheng, Kelleher, Cao, Sridharan, & Anderson, 2013).

Liquid fluoride salts have excellent thermal hydraulic properties such as high volumetric heat capacity and a high boiling point – greater than 1400 °C (Williams, Toth, & Clarno, 2006) (Forsberg, Peterson, & Pickard, Molten-Salt-Cooled Advanced High-Temperature Reactor for Production of Hydrogen and Electricity, 2003). The high volumetric heat capacity enables FHRs to maintain a high average temperature – 650 °C – small temperature rise across the core – 100 °C. This mode of operation yields thermal efficiencies similar to HTGRs but FHRs operate within a temperature regime covered by the ASME boiler and pressure vessel code.

Flibe’s high boiling point simplifies both the safety case and thermal hydraulic design. Since the boiling point of flibe is less than the temperatures at which fuel failures begin – around 1600 – 1800 °C (Todreas & Kazimi, 1990) (Williams, Toth, & Clarno, 2006), the integrity of the fuel can be assumed so long as the fuel remains covered in salt. The thermal hydraulic design in FHRs is relatively simple because it utilizes a single-phase coolant.

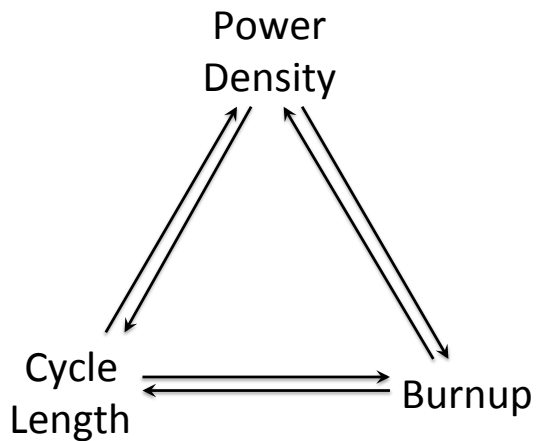
1.1.3 Direct Reactor Auxiliary Cooling System (DRACS)

The direct reactor auxiliary cooling system removes decay heat from the core passively in severe accidents. During an accident the flow in the primary loop – normally blocked from the DRACS loop – reverses to initiate the DRACS. A secondary natural circulation salt loop transports heat from the reactor to an external heat sink. Utilizing a DRACS is an efficient way to passively remove heat in a severe accident, enabling FHRs to operate at significantly higher total powers as well as power densities compared to HTGRs.

1.1.4 Fuel Form Comparison

The low volumetric fuel loadings in TRISO fuel systems result in undesirable trade offs between power density, cycle length and burnup. For example, heavy metal loading can be increased to allow longer cycle length or higher power density, but this hardens the neutron spectrum too much resulting in poor fuel utilization. In pebble bed reactors fuel is circulated online so the circulation rate can be increased without negatively effecting power density or burnup.

Fixed Fuel



Pebble Fuel

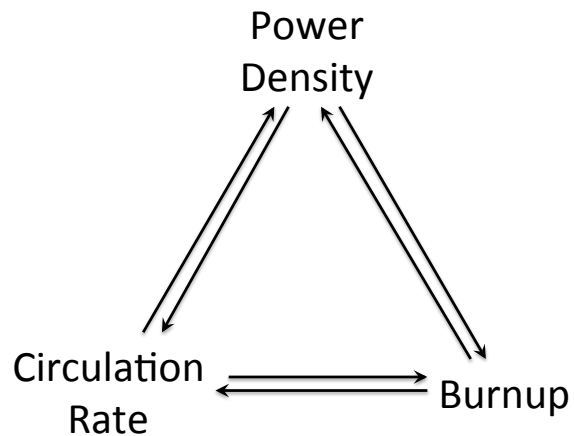


Figure 1-1 Trade offs between fuel cycle parameters; (left) fixed fuel; (right) pebble fuel

Because pebble bed reactors (PBRs) can add additional reactivity on the fly they can operate with low-excess reactivity. This enables high reactivity low burnup pebbles to drive subcritical high burnup pebbles, increasing neutron economy and burnup. Moreover, this mode of operation obviates reactivity insertion accidents caused by control rod ejection when operating in the equilibrium state.

Continuously recirculating pebbles through the core adds complexity and uncertainty to the operation of a nuclear reactor. The Arbeitsgemeinschaft Versuchsreaktor (AVR) experiment operated for over 21 year with no significant pebble handling problems (Gas-Cooled Reactor Safety and Accident Analysis, 1985) (Gottaut & Kruger, 1990). However, the Thorium High Temperature Reactor (THTR) experienced mechanical problems due to broken pebbles (Schwarz & Baumer, 1988) (Baumer, Kalinowski, & Rohler, 1990). These pebbles were likely damaged by control elements implemented directly surface of the pebble bed. The THTR also experienced discrepancies of pebble burnup distributions possibly from holdup of pebbles along the outer reflector due to temperature dependence of graphite friction coefficient in helium. The coefficient of friction in a FHR PBR is much smaller – around .2 – than in a HTGR PBR because fluoride salts act as lubricants for the graphite pebbles and FHRs operate with a smaller temperature rise (Caron & al, 2008). Additionally, inert graphite pebbles are circulated through the areas most susceptible to holdup further mitigating this issue. Therefore, these complications should be less severe for FHR PBRs.

1.2 FHR Design Evolution

In 2002 the Very High Temperature Reactor (VHTR) served as the basis for the original FHR reactor concept, the Liquid Salt Very High Temperature Reactor (LS-VHTR). The LS-VHTR used the prismatic fuel block design from the VHTR with smaller coolant channels (because of the higher heat capacitance) within the SPRISM core footprint (Kim, Taiwo,

& Yang, 2005). The LS-VHTR evolved into the Advanced High Temperature Reactor as the outlet temperatures transitioned from 1000 °C down to 700 °C (Peterson & Zhao, 2006). In 2006 Zwaan proposed using pebble fuel rather than prismatic fuel (de Zwaan, Boer, Lathouwers, & Kloosterman, 2007) and the FHR design evolution branched into two camps with UC Berkeley adopting pebble bed variants of the FHR and the a group at Oak Ridge National Laboratory (ORNL) continuing to develop fixed fuel variants of FHRs (Green, et al., 2010) (Varma, et al., 2012). UC Berkeley proposed a central station sized (2400 MWt) integral pebble bed AHTR (Integral PB-AHTR) adopting 6cm diameter fuel pebbles similar to those used in the pebble bed modular reactor (PBMR) in an unstructured cylindrical pebble bed (Griveau, Fardin, Zhao, & Peterson, 2007). Then UC Berkeley proposed several design advances for the modular PB-AHTR including: 3cm annular pebbles to reduce average fuel temperature, pebble channel assemblies that allow higher heavy metal loading in the pebbles to reduce spent fuel volume and reduce the coolant volume, a multiple channel configuration to flatten the power distribution, fixed reactivity control element channels (Bardet, Blandford, Fratoni, Niquille, Greenspan, & Peterson, 2008).

In 2009 the PB-AHTR evolved again to an annular configuration (Hong, 2009). In this annular configuration the coolant flows from a solid reflector assembly up and out radially reducing the pressure drop across the core dramatically (Cisneros, Scarlet, Laufer, Greenspan, & Peterson, 2012). Granular dynamics experiments at UC Berkeley confirmed with simulation and experiments that radial zones could be maintained in a fully packed pebble bed with converging and diverging zones – this breakthrough was integrated in the Annular PB-AHTR and explored concepts with thorium blankets or graphite pebble reflectors to shield the outer graphite reflector (Bickle, Laufer, Li, & Peterson, 2010) (Cisneros, Greenspan, & Peterson, Use of Thorium Blankets in a Pebble Bed Advanced High Temperature Reactor, 2010).

Schematics of earlier FHR PBRs are presented in Figure 1-2.

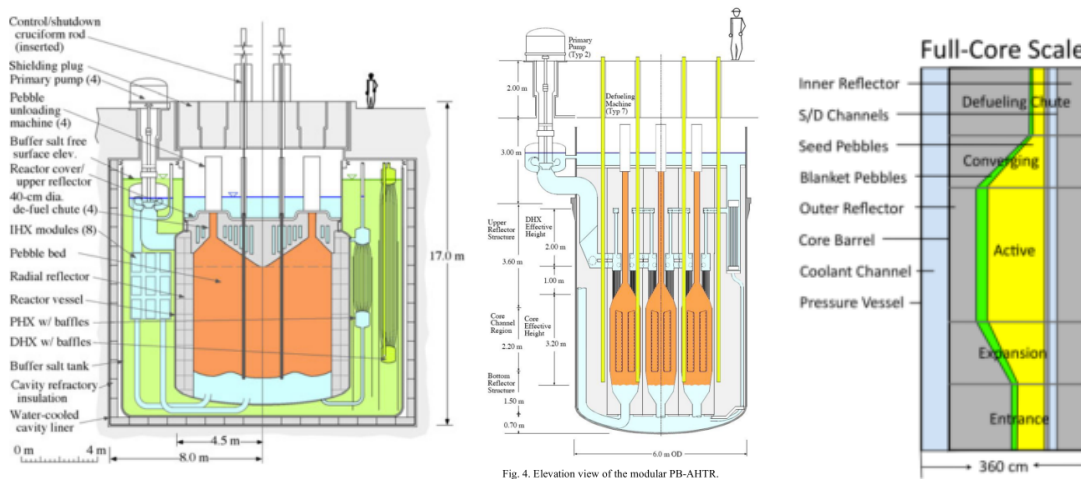


Fig. 4. Elevation view of the modular PB-AHTR.

Figure 1-2 The evolution of FHR Pebble bed reactors: (left) Integral PB-AHTR; (center) Modular PB-AHTR; (right) Annular PB-AHTR (Griveau, Fardin, Zhao, & Peterson, 2007) (Bardet, Blandford, Fratoni, Niquille, Greenspan, & Peterson, 2008) (Cisenros, Laufer, Scarlet, Zweibaum, & Seifried, 2012).

1.2.1 PB-FHR Concept

In 2012, UC Berkeley was charged with developing a commercial prototype FHR pre-conceptual design as part of the Nuclear Energy University Programs’ (NEUP) integrated research project. The strategic decision was made to limit the reactor vessel to 3.5m so it could be transported via rail and the power rating was downgraded to 236 MWt to match the power demand of the General Electric 7FB gas turbine (modified slightly to introduce external heating in one stage of reheat) – this smaller reactor would limit the risk associated with a first of a kind advanced nuclear reactor concept eventually leading to multi- Gigawatt evolutionary FHR designs such as the AHTR (Andreades, et al., 2014).

This Mark-1 (Mk1) PB-FHR shares many features with the earlier variants of the PB-AHTR. The Mk1 PB-FHR is an annular FHR PBR fueled with 19.9w% LEU fuel as shown in Figure 1-3. The PB-FHR’s pebble bed is segregated into an inner fueled region surrounded by an outer inter pebble reflector. This pebble reflector shields the outer structural components by moderating and attenuating neutrons. The pebbles are injected at the bottom of the core and float up through the pebble bed.

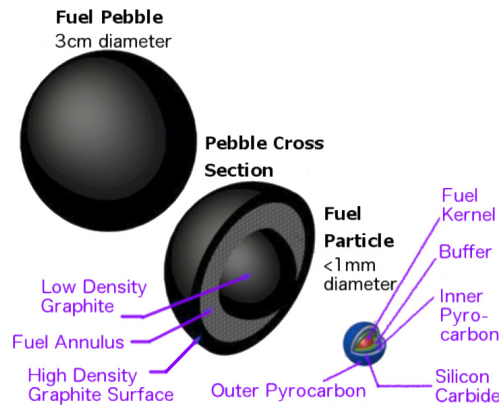


Figure 1-3 Detail of Mk1 PB-FHR annular pebble compact geometry

Coolant is injected radially from the inner reflector assembly and axially from the bottom of the pebble bed, moving up and out radially through the core until it is collected in the outlet plenum, pumped through the coiled tube air heaters (CTAHs) and finally returned to the core via the downcomer. These CTAHs transfer the nuclear heat to a nuclear air-Brayton combined cycle (NACC) power conversion system. This NACC gives the PB-FHR power plant the flexibility to produce peaking power with natural gas when electricity prices are high or sell nuclear energy as steam rather than electricity when electricity prices are low (Andreades, Lindsay, & Peterson, 2014) (Andreas, Scarlet, Dempsey, & Peterson, 2014)

The Mk1 PB-FHR is controlled primarily through the pebble recirculation rate or control rods inserted into channels in the central graphite reflector. Under accident conditions shutdown blades can be injected into the pebble bed. However these shutdown blades

move against the effective gravity, so there is free space below the core for the pebble bed to expand into, rather than exacerbating the gravity loads and gas blowing forces as in HTGR PBRs.

A high level schematic of the Mk1 PB-FHR nuclear power plant is presented in Figure 1-4 and a cross sectional view of the computer aided drafting model of the reactor core is displayed in Figure 1-5. Key Mk1 PB-FHR design parameters are presented in Table 1-1.

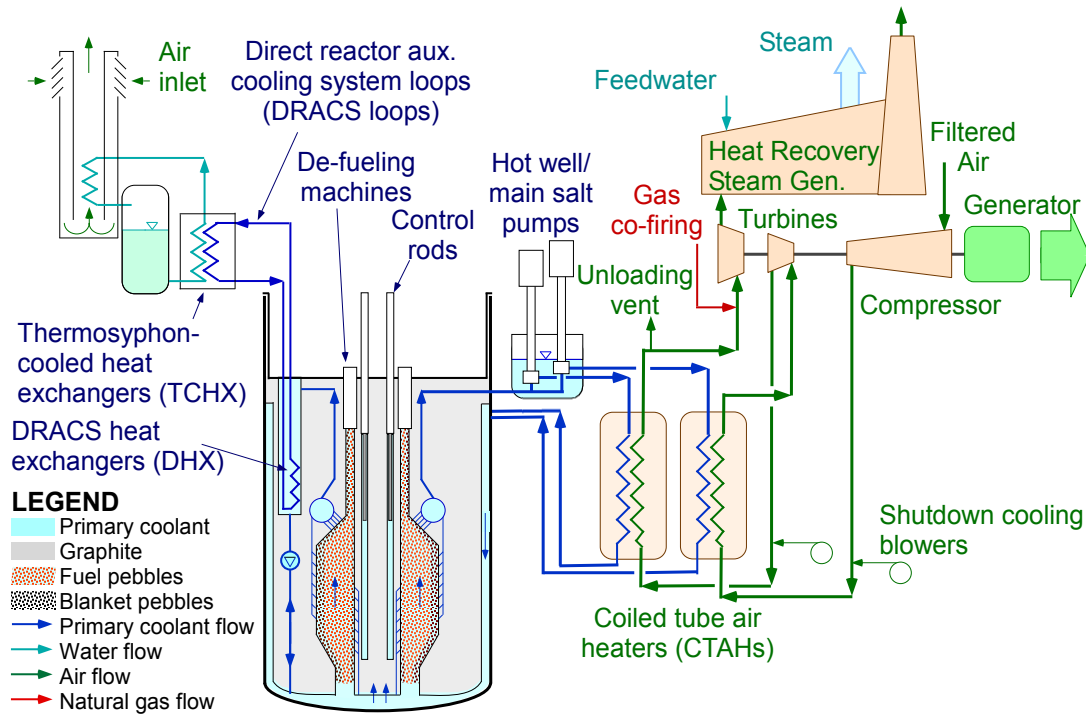


Figure 1-4 Schematic of the PB-FHR power plant (Andreas, Scarlat, Dempsey, & Peterson, 2014)

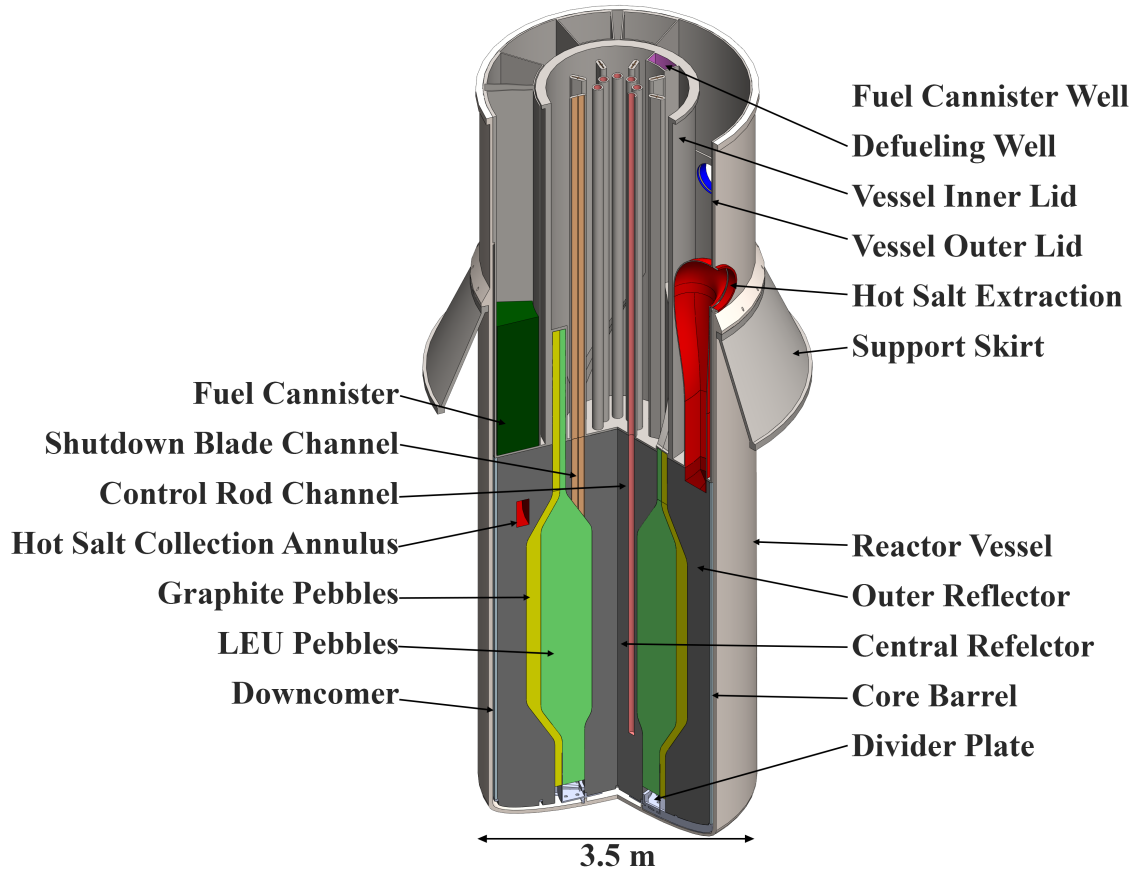


Figure 1-5 Diagram of the Mk1 core and reactor vessel (Krumweide, Scarlet, Choi, Phan, & Peterson, 2013)

Table 1-1 Key Mk1 PB-FHR design parameters (Andreades, et al., 2014)

Nuclear thermal power (MWt)	236
Core inlet temperature (°C)	600
Core outlet temperature (°C)	700
Primary coolant mass flow rate (kg/s)	976
CTAH outlet temperature (°C)	670
Base-load net electric power output (MWe)	100
Base-load thermal efficiency (%)	42.5
Co-firing net electrical power output (MWe)	241.8
Co-firing efficiency* (%)	66.4
Uranium Enrichment (w% ²³⁵ U)	19.9
Specific Fuel Loading (MWe/HM MT)	137
Specific Fissile Fuel Loading (MWe/fissile HM MT)	690
Burnup (MWd/MT)	177847
Pebble Recirculation Rate (Pebbles/day)	7185

* gas to peak power

1.3 Scope of Study

This manuscript presents the development and application of methodologies for the neutronics reactor design of the PB-FHR. This manuscript is subdivided into four main chapters.

Chapter 2 discusses the neutronics and depletion methodology utilized to analyze these FHR PBRs. The multiple burnup state methodology (MBSM) is proposed as a novel algorithm for performing equilibrium depletion analysis on an arbitrary continuously refueled pebble bed reactor. A program, Burnup Equilibrium Analysis Utility (BEAU), was developed to implement this methodology. This chapter serves as a reference manual for using BEAU. This tool was also benchmarked against published results generated with VSOP and PEBBED for a HTGR PBR.

Chapter 3 presents the results of three parametric studies exploring the design space of FHR PBRs, to develop a fuel design for the Mk1 PB-FHR, to select a core configuration and optimize the reactor core geometry. These parametric studies investigated trends in several reactor performance parameters such as burnup, temperature reactivity feedback, radiation damage, etc and attempted to understand the underlying reactor physics causing these trends.

Chapter 4 presented the detailed multi-object optimization of the PB-FHR core design. This optimization derived various physical constraints from the high-level safety regulatory criteria. These physical constraints were applied along in a design tool, Nuclear Application Value Estimator (NAVE), that constructed a simplified cash flow economics model of specific instances of the PB-FHR to assess the economic implications of design decisions using surrogate models developed based on the results of the full-core parametric study from Chapter 3 to explore the PB-FHR's design space.

Chapter 5 presents the neutronics results of the Mk1 PB-FHR including: the burnup, the burnup evolution, temperature reactivity coefficients, the power distribution, radiation damage distributions, control element worths, decay heat curves and tritium production rates.

Outside of these four main chapters, three appendices are included: a manual for a depletion coupling tool, mocup.py, and sample MCNP5 input decks for a FCC pebble unit cell model and the Mk1 PB-FHR.

2 Equilibrium Fuel Cycle Modeling Methodology

2.1 Introduction

This chapter presents the theoretical basis- and methodology used for the equilibrium depletion analysis of the PB-FHR. The majority of these analysis techniques apply to all continuously refueled pebble bed reactors (PBR).

The continuously refueled PBR mode of operation is reviewed to provide a basis for the depletion scheme. Several pebble bed depletion analysis methodologies are reviewed and a new methodology – the multiple burnup state method (MBSM) – is proposed and defined. The code Burnup Equilibrium Analysis Utility (BEAU), which couples a static, continuous-energy, Monte Carlo neutron transport with point depletion analysis for pebble bed equilibrium depletion, is presented as a means for implementing this new methodology. BEAU and the MBSM are benchmarked together against existing equilibrium depletion results from literature for a high temperature gas cooled reactor.

2.2 Continuously Refueling Pebble bed Reactor Operation

The methods used to model the equilibrium depletion analysis of a continuously refueled pebble bed reactor (PBR) are a reflection of its operation. Therefore, before the methods used to model these reactors are discussed it is instructive to review the operations of PBRs and how they differ from multi-batch or once through fuel cycles.

2.2.1 Pebble Motion

Pebbles move according to granular flow rather than being shuffled around the core as in multi-batch fuel cycles. Pebble circulation time is much slower than coolant recirculation time. Fuel pebbles are introduced to the core at either the bottom of the bed and float up axially to the top where the pebbles are removed as in FHR PBRs or they are introduced at the top of the core and descend to the bottom where the pebbles are removed, as in traditional HTGR PBRs. In either case, pebbles in a constant cross section region move in plug flow – that is they are relatively fixed in their radial position and only move axially. This plug flow behavior has been confirmed computationally and experimentally at UC Berkeley; see Figure 2-1. To first order, the ratio of the number of various pebble types is maintained as the pebbles move through regions with varying cross sectional area – for pebble bed models with constant pebble porosity – the measure of interstitial space between pebbles – this reduces to constant cross sectional area fractions, but more detailed models must account for varying porosity when specifying the boundaries between pebble zones (i.e. fuel type regions, depletion zones, etc.).

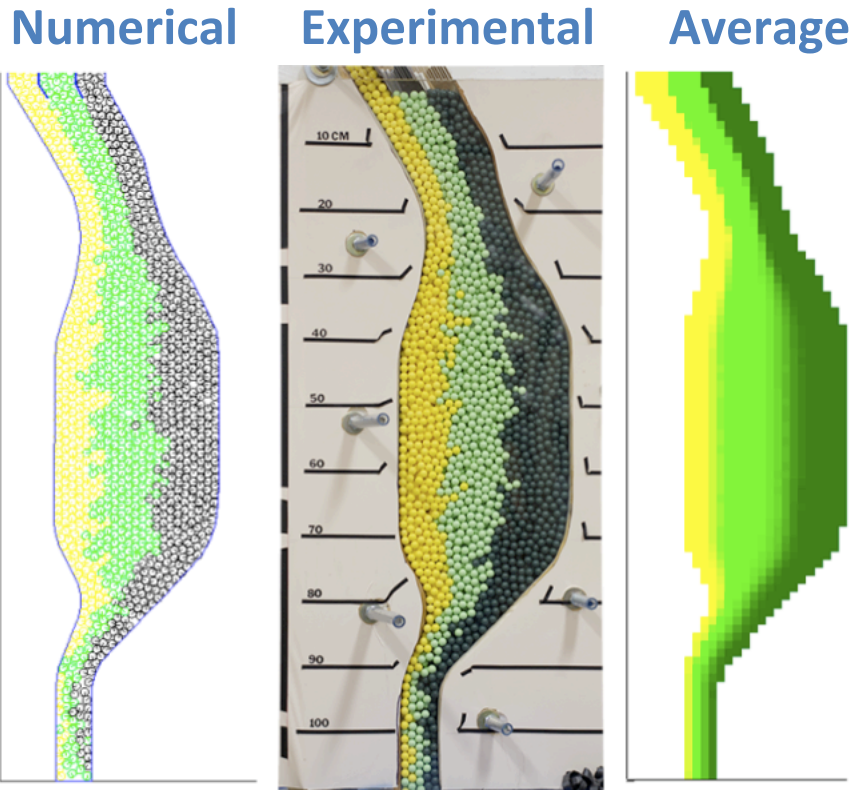


Figure 2-1. (left) granular flow numerical simulation and (right) granular flow experiment of a proposed FHR 16MWt test reactor (Bickle, Laufer, Li, & Peterson, 2010).

After an initial start up phase, all the boundary conditions of the PBR system remain constant: pebble insertion and removal rate, the temperature rise across the pebble bed and control element position. Eventually, an equilibrium state will be established – the goal of most equilibrium depletion analysis tools is to determine this equilibrium state.

Therefore, the reactor core behaves as a function of an aggregate fuel composition more so than any particular burnup because at any given point in the pebble bed there is a rich distribution of pebbles with various burnups throughout the core.

2.2.2 Reactivity Management

PBRs normally operate with limited excess reactivity –the limit being no excess reactivity. In an idealized multi-batch PBR fuel cycle lower burnup fuel pebbles are introduced to the core perfectly compensate for as pebbles are burned – however, after the pebbles are introduced into the core their identity is lost so pebbles are differentiated based on their burnup (rather than number of passes through the core), such that fuel pebbles below the burnup cut-off are reinserted into the core and fuel pebbles above this burnup cut-off are discharged from the core. Alternatively, a PBR could be operated in a Once through then out (OTTO) fuel cycle. In PBRs utilizing an OTTO fuel cycle, there is a stronger burnup gradient throughout the core and lower average burnup; however, a pebble sorting system is not required, there is less demand

on the pebble handling system and fuel composition throughout core is known with more certainty than in a multi-batch PBR.

2.3 Equilibrium Pebble bed Depletion Analysis Methodology

2.3.1 Assumptions

Based on the assumptions of pebble flow and reactivity management, the following modeling assumptions can be drawn:

1. A PBR can be represented by an equilibrium core with a composition or distribution of compositions developed based on the time average of composition of fuel moving through the control volume of a depletion zone.
2. Fuel pebbles progress in plug flow axially; therefore, fuel charged in a given (R,Z) node is discharged from the previous axial node in the same radial position.
3. PBRs maintain criticality primarily by enforcing a burnup limit of fuel pebbles removed from the core.

2.3.2 Existing Pebble bed Depletion Methodologies

Much research and development had invested in equilibrium pebble bed depletion analysis. The following literature review for equilibrium pebble bed depletion analysis focuses on the work of INL with PEBBED and of Dr. Fratoni at UC Berkeley (Terry, Gougar, & Ougouag) (Fratoni, Development and Application of Methodologies for the Neutron Design of the Pebble Bed Advanced High Temperature Reactor (PB-AHTR), 2008). Existing methods developed for the German PBMR program have been extensively reviewed in the development of PEBBED which performs similar to these legacy codes. Dr. Fratoni focused on coupling generic codes (MCNP5 and ORIGEN) to perform equilibrium pebble bed depletion analysis (X-5 Monte Carlo Team, 2003) (Croff, 1983).

2.3.2.1 Unit Cell Methodologies

Unit cell pebble bed equilibrium depletion analysis methods are a valuable tool for design parametric studies for PBRs (Fratoni, Greenspan, & Peterson, Neutronic and Depletion Analysis of the PB-AHTR, 2007). These methods enable a designer to feasibly explore a large design space without having to develop a detailed design for an entire full-core reactor. However, unit cell modeling results in distortion from not accounting for neutron leakage effects, neutron spectrum effects of external reflectors or spatial variation in the neutron flux intensity and spectrum.

2.3.2.1.1 Infinite Uniform Bed Method

The infinite uniform bed method (IUBM) assumes that for a well-mixed PBR there is an equal probability of finding pebbles at any burnup level at any position in the core and that the average flux to which pebbles are exposed is constant during the entire residence time of a pebble (Fratoni & Greenspan, Equilibrium Core Composition Search Methodologies for Pebble Bed Reactors, 2010). The IUBM uses a neutron transport model of a single pebble and depletes it at a constant neutron flux to develop a history of k_{∞} as a function of time. The maximum burnup is determined by solving for the residence time to satisfy the criticality criterion given in Equation 2-1.

$$\frac{1}{k_{\infty,core}} = \frac{\int_0^T \frac{P_{pebble}(t)}{k_{\infty,core}(t)} dt}{\int_0^T P_{pebble}(t) dt}$$

Equation 2-1

where $k_{\infty,core}$ is the target k_{∞} for the core (this target k_{∞} should account for neutron leakage), t is instantaneous time, T is the residence time that corresponds to the maximum burnup and P_{pebble} is the power per pebble at a given time.

Furthermore, the average flux is determined by iteratively estimating at the average flux until the average power per pebble satisfies Equation 2-2.

$$\frac{P_{core}}{N_{pebbles}} = \frac{Q_f \phi \Psi^{pebble}}{T} \int_0^T \Sigma_{fission}^{pebble}(t) dt$$

Equation 2-2

Where P_{core} is the power of a full PBR core, $N_{pebbles}$ is the number of pebbles in the PBR, Q_f is the average energy per fission event, ϕ is the total neutron flux, Ψ is the volume of the pebble, T is the residence time that corresponds to the maximum burnup, $\Sigma_{fission}^{pebble}$ is the time dependent 1-group macroscopic fission cross section for the pebble. The convergence algorithm for the IUBM is presented in Figure 2-2.

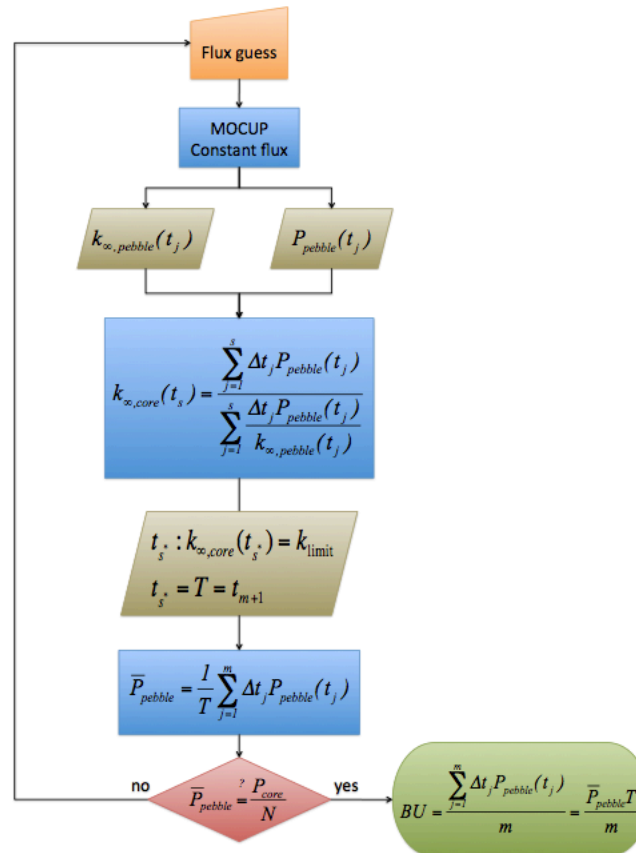


Figure 2-2. Flow chart for Infinity Uniform Bed Method

The advantage of the IUBM is that it converges fast. Furthermore, Fratoni states that the attainable burnup is not found to be sensitive to flux (Fratoni, Development and Application of Methodologies for the Neutron Design of the Pebble Bed Advanced High Temperature Reactor (PB-AHTR), 2008). Therefore, if one is only interested in the maximum attainable burnup of a system one can simply neglect the loop to solve for the average flux that imposes the correct power per pebble. However, this methodology neglects all spatial heterogeneities and the effect that neighboring pebbles of different burnups have on the burnup dependent flux spectrum.

2.3.2.1.2 Single Pebble in Equilibrium Bed Method

The single pebble in equilibrium (SPIEB) bed method is an expansion on the IUBM developed to account for effect of the on the burnup dependent flux spectrum (Fratoni & Greenspan, Equilibrium Core Composition Search Methodologies for Pebble Bed Reactors, 2010). Rather than using an infinitely reflected model of a single pebble at a specific burnup as in IUBM, the neutron transport model for SPIEB uses a large bed of pebbles at an assumed equilibrium composition as the boundary conditions outside of a test pebble. This equilibrium composition is the time-averaged composition of the pebble over the expected fuel cycle as shown in Equation 2-3.

$$\bar{N}^{bed} = \frac{1}{T} \int_0^T \bar{N}^{pebble}(t) dt$$

Equation 2-3

Where, N^{bed} is the equilibrium composition vector of a fuel pebble, T is the expected residence time of a pebble and N^{pebble} is the instantaneous composition vector of a fuel pebble, and t is time.

The search algorithm for the SPIEB searches for three parameters: the residence time, T , of the pebbles in the pebble bed, the equilibrium composition, N^{bed} , of the bed, and the average flux in the pebble bed. An initial estimate of the average neutron flux is made. A single pebble is depleted assuming this constant flux. The residence time of this pebble is selected to impose a target power such that Equation 2-4 is satisfied.

$$P_{pebble}^{target} = \frac{\int_0^T P_{pebble}(t) dt}{T}$$

Equation 2-4

For each depletion calculation, the equilibrium composition is updated according to Equation 2-3. This depletion sequence is performed iteratively until the N^{bed} converges. Once the N^{bed} converges, the k_{∞} is compared against the target k_{∞} . If the latest estimate of k_{∞} is outside of the acceptable error, the estimate of the neutron flux is updated and this search sequence begins again until k_{∞} has converged to the target value. A flow chart of this search sequence is presented in Figure 2-3.

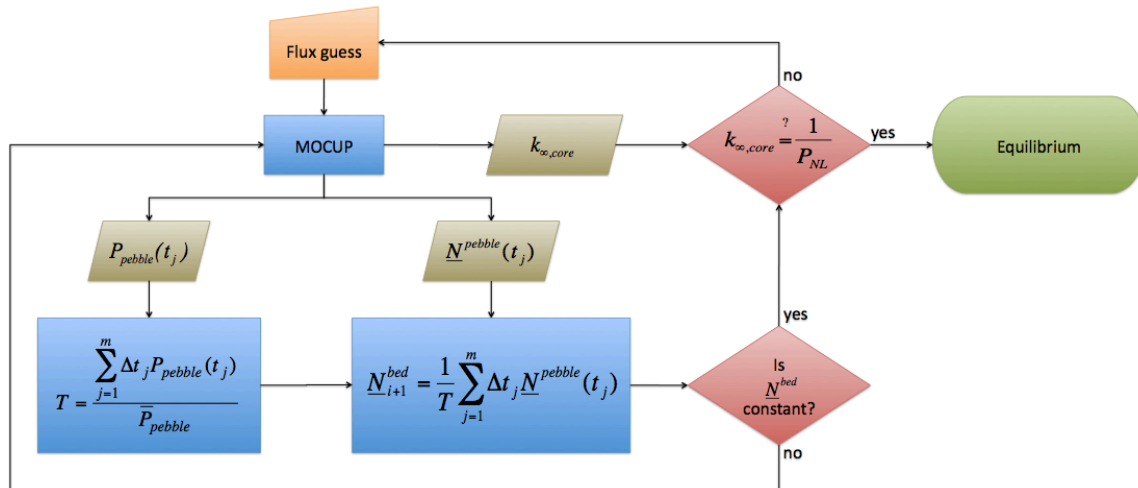


Figure 2-3. Flow chart for Single Pebble in Equilibrium Bed Method.

The SPIEB method is more computationally intense because the neutron transport model includes the additional bed pebbles. However, the results from SPIEBM diverge from those obtained using IUBM in system where the neutron spectrum changes significantly from the beginning of life (BOL) to end of life (EOL) as is the case for pebbles fueled with light water reactor (LWR) transuranics (i.e. deep burn pebbles).

2.3.2.1.3 Unit Cell Methodologies Opportunities for Improvement

Both the IUBM and SPIEM require many iterations of neutron transport to produce burnup dependent nuclear data when in theory, this nuclear data is more dependent on the aggregate composition of the core rather than a particular point in burnup. For equilibrium depletion methods to explore the design space it is desirable to on limit the number of neutron transport iterations required to generate accurate results.

2.3.2.2 Full Core Methodologies

As stated previously, in many cases using a full-core neutronics and depletion model obviates simplifications made in unit cell analysis or might be required to calculate full-core parameters. However, the computational demand of the equilibrium depletion calculation grows as the neutron transport model becomes more complicated (more depletion zones, more cells, more materials, more tallies, etc.) and also as the number of variables to converge increases. Effective full-core equilibrium depletion modeling must strike a balance between detail and computational speed.

2.3.2.2.1 Steady-State Asymptotic Fuel Loading Pattern Search Algorithm Method

Idaho National Laboratory has been developing the PBR fuel cycle code PEBBED code that converges directly to the equilibrium condition using the steady-state asymptotic fuel loading pattern search algorithm (SSAFLPSA) (Terry, Gougar, & Ougouag). This methodology uses a full-core, two-dimensional (R,Z) model of a PBR where the pebble bed is sub-divided in spectral zones with constant radial cross sections (Hudson, Ougouag, Rahnema, & Gougar, 2009). The burnup solver depletes fuel in these spectral zones under constant flux for a time period determined by the axial height of the zone

and the axial velocity of the fuel in each radial channel (Gougar, Reitsma, & Joubert, A Comparison of Pebble Mixing and Depletion Algorithms Used in Pebble-Bed Reactor Equilibrium Cycle Simulation, 2009). The assumed composition calculated by the burnup solver in each spectral zone is the average of the inlet and outlet composition (Gougar, Reitsma, & Joubert, A Comparison of Pebble Mixing and Depletion Algorithms Used in Pebble-Bed Reactor Equilibrium Cycle Simulation, 2009).

$$\bar{N}_{ave} = \frac{\bar{N}_{inlet} + \bar{N}_{outlet}}{2}$$

Equation 2-5

The SSAFLPSA iteratively solves for the flux distribution in the PBR and then uses that flux distribution in the burnup solver to update the fuel composition of the equilibrium PBR model (Hudson, Ougouag, Rahnema, & Gougar, 2009). These steps are iteratively performed until a self-consistent solution is reached. Later versions of PEBBED have an outer loop to update the multi-group nuclear data based on the converged equilibrium state (Hudson, Ougouag, Rahnema, & Gougar, 2009).

The SSAFLPSA has been integrated with thermal hydraulics and fuel performance codes and used to perform design studies of HTGRs for the Next Generation Nuclear Plant (NGNP) program (Boer & Ougouag, 2010).

This methodology does not search for the maximum burnup condition of a particular reactor design, but rather the equilibrium state assuming given number of passes through the bed. An additional outer loop that iterates on either pebble velocity or number of passes would be required to find the maximum burnup equilibrium state of a given PBR.

By homogenizing the equilibrium composition across pebbles of various passes, one loses the ability to calculate burnup specific cross sections for depletion analysis – though this is not a serious issue for low enriched uranium fueled systems because the spectrum remains relatively constant with increasing burnup (Fratoni & Greenspan, Equilibrium Core Composition Search Methodologies for Pebble Bed Reactors, 2010).

This method was implemented with deterministic methods in PEBBED to quickly solve for the flux distribution. Deterministic methods require several approximations to generate a system specific set of multi-group nuclear data (Gougar, Yoon, & Ougouag, Multiscale Analysis of Pebble Bed Reactors, 2010). The self-shielding effect in coated-particle fuel exacerbates the challenges associated with preparing multi-group nuclear data.

2.3.2.2.2 Full Core Method

The Full Core Method (FCM) was developed to perform 3-D modeling of PBRs (Fratoni & Greenspan, Equilibrium Core Composition Search Methodologies for Pebble Bed Reactors, 2010). Like the SSAFLPSA, the FCM assumes cross sections depend on position rather than composition and that fuel in radial channels progress axially. Equation 2-6

relates the composition of the equilibrium state in a spectral zone to the compositions of pebbles within this spectral zone.

$$\bar{N}_{ave} = \frac{\sum_{i=1}^{n(\bar{r})} \bar{N}_{pebble}(t_i)}{n(\bar{r})}$$

Equation 2-6

Where $n(r)$ is the number of times a pebble can be found at the position r , and t_i is a time at which the pebble can be found at the position r .

The search for the equilibrium state begins by assuming an arbitrary equilibrium composition distribution over the reactor and residence time. Neutron transport is performed to generate a set of system specific one-group cross sections for each spectral zone for depletion analysis and the power distribution. A burnup solver depletes fuel assuming flux and cross sections are constant for a given spectral zones. Branch depletion calculations are made for each possible path originating from a single entry plane. These branches are collapsed when they feed into a single spectral zone (i.e. if pebble channels feed into a collection plenum). A new equilibrium core composition distribution is estimated using Equation 2-6 assuming time-averaged compositions for each pebble in each spectral zone. Neutron transport and burnup analysis are performed iteratively until $k_{effective}$ converges to a constant value. An outer iteration loop varies the residence time until the core $k_{effective}$ becomes 1.0. Figure 2-4 presents a flow chart for this search algorithm.

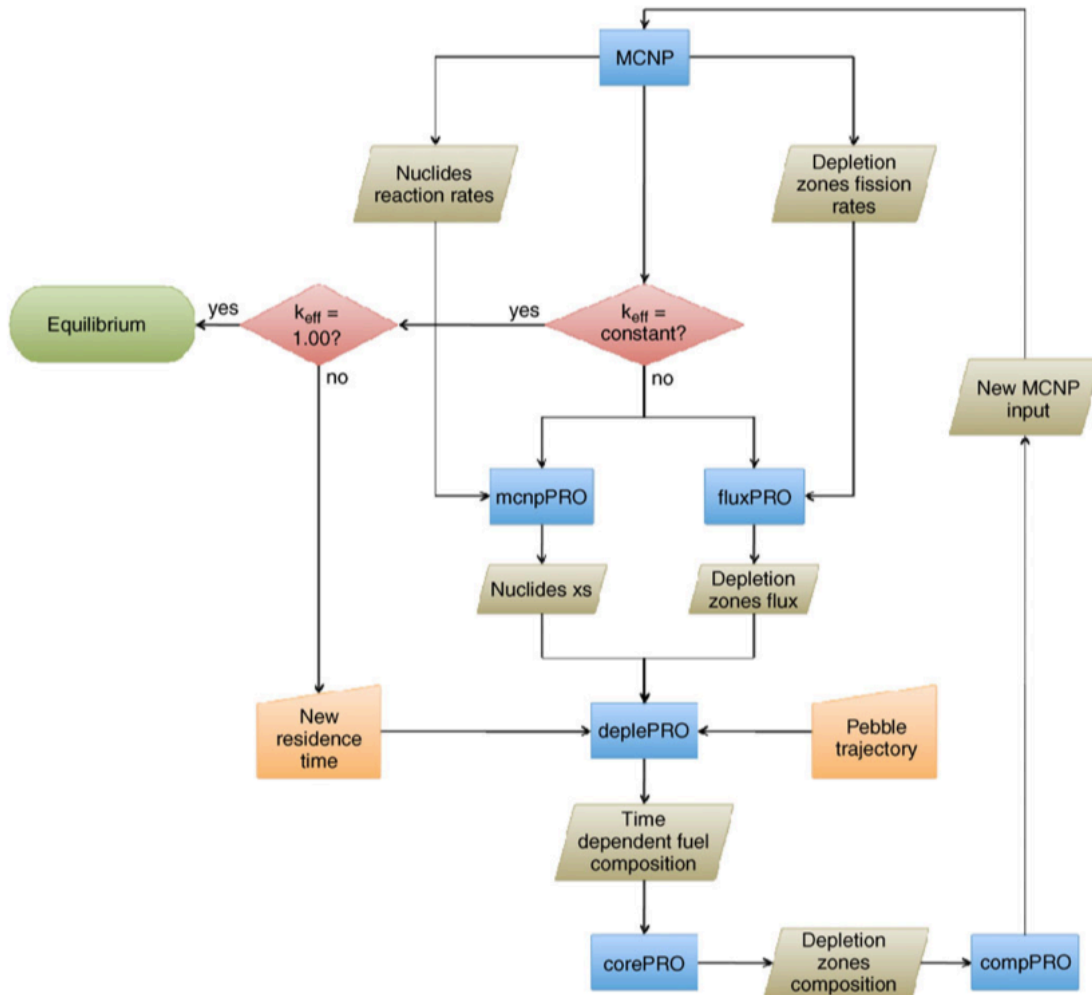


Figure 2-4. Flowchart for the full core method search algorithm.

The FCM has been implemented to find the maximum burnup state of the modular PB-AHTR using a full-core model, using MCNP, ORIGEN2 coupled with MOCUP.

The equilibrium state of a PBR includes a converged residence time and a converged equilibrium composition distribution. The challenge of converging to an equilibrium state increases as one increases the detail of the core model (number of depletion zones and isotopes explicitly followed). However, one should incorporate the equilibrium composition distribution into the convergence criteria to ensure that the $k_{\text{effective}}$ has converged because the equilibrium composition have converged.

As stated earlier by averaging the equilibrium composition across pebbles of various passes, one loses the ability to calculate burnup specific cross sections for depletion analysis.

2.3.2.2.3 Full Core Methodologies Opportunities for Improvement

There are opportunities for advancement in equilibrium pebble bed depletion methodology. Continuous-energy Monte-Carlo neutron transport should be utilized when investigating advanced reactor systems to ensure that cross sections for depletion are calculated making as few assumptions as possible. The implementation of these methodologies should include new Monte-Carlo codes optimized for reactor analysis like Serpent to broaden their applicability (Leppanen, 2007). To investigate systems with strongly burnup dependent (as in plutonium-fueled systems) neutron spectrums one should use burnup dependent cross sections (in each spatial region) for depletion analysis. Finally, these methodologies do not address systems with multiple fuel make up streams, (i.e. seed and blanket systems) where the total residence time of the pebbles (i.e. number of passes through the reactor) are loosely (almost independent) coupled.

2.3.3 Multiple Burnup State Method

The multiple burnup state method (MBSM) developed in this work is an alternative method to perform equilibrium pebble depletion analysis and can be implemented with unit cell models as well as more complicated full-core models.

The biggest difference between the MBSM and the FCM is that the MBSM explicitly models all the burnup states of pebbles in a given spectral zone rather than homogenizing these pebbles into a single representative pebble. These burnup states can be thought of as either the time averaged composition of a pebble during a portion of a pebbles residence time in a core (in unit cell models) or rather this portion could represent a circulation through the core (in full-core models); see Figure 2-5. In a unit cell model, this representation of a well-mixed pebble bed at equilibrium enables the user to calculate a set of burnup dependent cross sections (a set for each burnup state) thereby, obviating the need for multiple neutron transport calculations to calculate the cross sections at every burnup state (as in the IUBM and SPIEM) and a separate criticality calculation of a pebble composed of only the equilibrium state pebbles. In a full-core model, the MBSM representation enables the user to utilize burnup dependent depletion cross sections and obviates the assumption that a single material can represent pebbles at multiple burnup states.

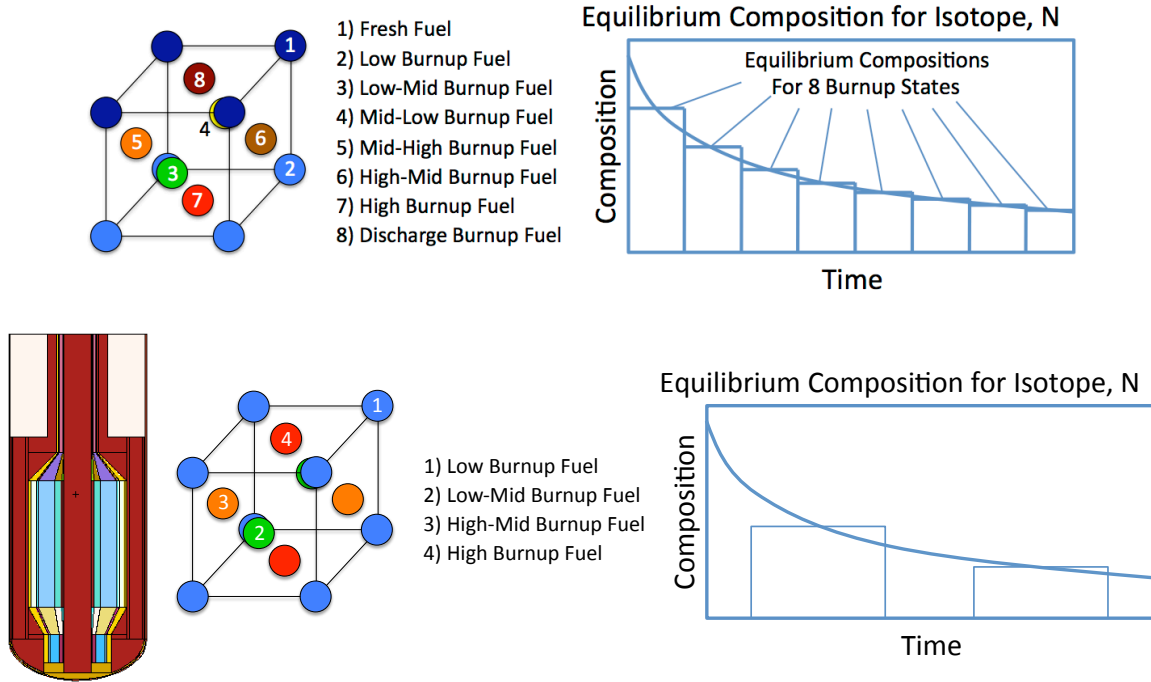


Figure 2-5 (top left) representation of a MBSM unit cell; (top right) fuel composition evolution and burnup state compositions; (bottom left) full core representation of MBSM unit cell; (bottom right) fuel composition evolution and burnup state compositions.

The burnup solver for the MBSM returns the average burnup for each fuel type (usually only one) and the aggregate power of the system operating at equilibrium for a given estimate of the equilibrium composition distribution in the PBR, fission neutron source and set of cycle lengths (each fuel type can have an independent cycle length) (Cisneros, Greenspan, & Peterson, Pebble Bed Reactor Depletion Analysis with Multiple Fuel Types, 2011). Neutron transport is performed to generate a full-core flux distribution and set of depletion cross sections for each burnup state. The fuel in each burnup state is depleted with these system-, spatial- and burnup-dependent cross sections under a constant flux and then advanced to the next burnup state. The flux used for depletion is calculated by scaling the total flux tally for a given burnup state by an estimate at the fission neutron source – thereby, preserving the shape of the neutron flux rather than the power distribution in the core.

$$\Phi_b^{depl} = S^{estimate} \frac{\int_{V_{j,k,b}} \int_0^\infty \psi(r_{j,b}, z_{k,b}, E) dE dV}{\int_{V_{core}} \int_0^\infty \bar{v}(E) \Sigma^{(n, fission)}(r, z, E) \psi(r, z, E) dE dV}$$

Equation 2-7

The fuel composition vector for each fuel type is advanced through each of its burnup states. In full-core models where the pebbles circulate through the core multiple times,

the fuel compositions from multiple depletion branches are collapsed into a single composition when it is reintroduced to the core.

$$\langle \bar{N}^{collapsed} \rangle = \frac{\sum_{p=i}^{n_{progression}} \bar{N}_p(T_p) \phi_p^{pebble} A_p(H)}{\sum_{p=i}^{n_{progression}} \phi_p^{pebble} A_p(H)}$$

Equation 2-8

The aggregate power of the system (unit cell or full-core) is taken as the sum of the time-averaged power over its residence time in each of the burnup states in the system as shown in Equation 2-9.

$$P_{system} = \sum_{b=1}^B \left(\int_{o}^{T_b} \int_{V_b} \bar{Q}^b \bar{N}(r_{j,b}, z_{j,b}, t_b) \sigma_b^{(n, fiss)} \Phi_b^{depl} dV dt_b \right)$$

Equation 2-9

The expected burnup for each fuel stream is taken as the sum of the volume-averaged burnups accumulated in each circulation through the reactor.

$$BU_{stream} = \sum_{c=1}^C \sum_{p=1}^{P_c} \left[\frac{\sum_{p=1}^{P_c} \frac{V_p}{V_p} \sum_{b=1}^{B_p} \left(\int_{o}^{T_b} \int_{V_b} \bar{Q}^b \bar{N}(r_{j,b}, z_{j,b}, t_b) \sigma_b^{(n, fiss)} \Phi_b^{depl} dV dt_b \right)}{\sum_{p=1}^{P_c} \int_{V_b} \rho(r_{j,b}, z_{j,b}) dV} \right]$$

Equation 2-10

The search for the maximum burnup equilibrium state in the MBSM utilizes iteration four loops. First for each fuel type, MBSM searches for the total cycle length to impose its target burnup assuming an estimated fission neutron source. The second loop searches for the fission neutron source intensity to impose the target power for the entire system. These first two iteration loops are performed implicitly with the same set of nuclear data without rerunning neutron transport. The third iteration loop converges on the equilibrium state of the core by rerunning neutron transport to update the neutron flux distribution and set of cross sections used for depletion. This loops uses a combination of system $k_{effective}$ and burnup state isotopic concentrations used for the equilibrium PBR model to test the convergence of the equilibrium state model of the PBR for an assumed set of target burnups for each of the fuel streams. Finally, the outer most iteration loop searches for the burnup of the driver fuel stream to impose a target $k_{effective}$ – usually 1.0. The burnup of the driver fuel type is used as an independent variable in this iteration loop. In seed and blanket systems, the seed would be considered the driver fuel stream because the $k_{effective}$ of the system is most sensitive to its burnup. Figure 2-6 shows a flow chart for the MBSM's search for the maximum burnup equilibrium state.

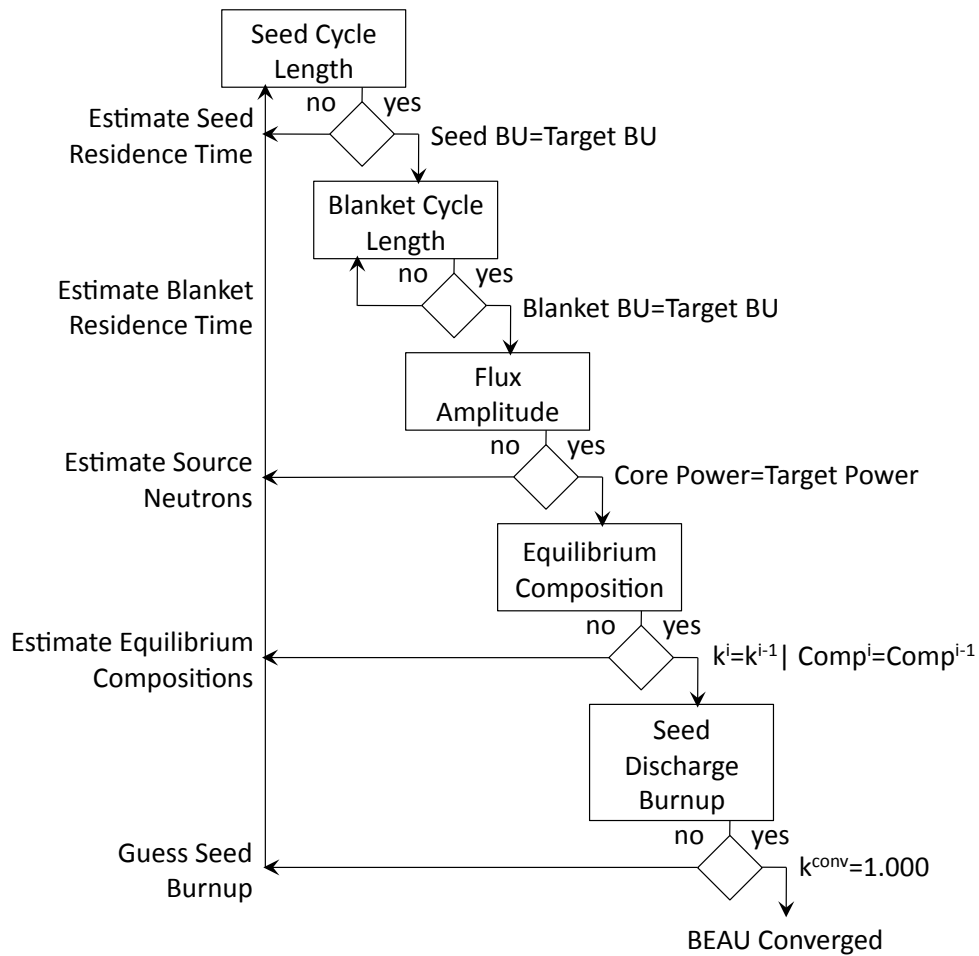
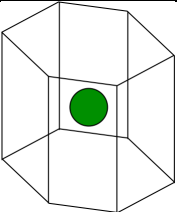
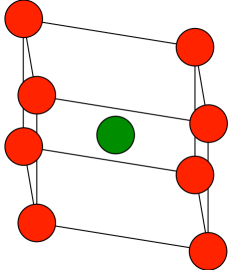
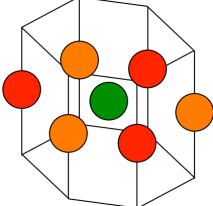
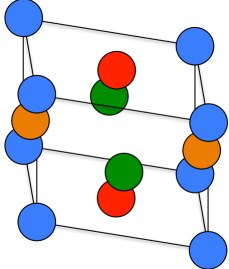
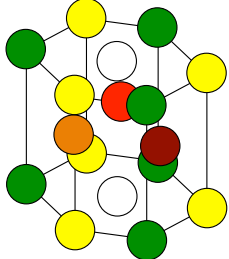


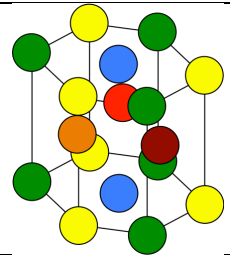
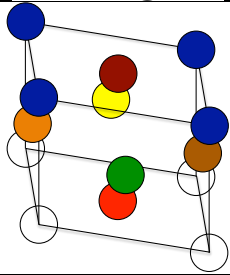
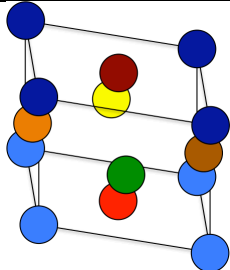
Figure 2-6 Flowchart for the multiple burnup state methods search algorithm.

A unit cell implementation of the MBSM can generate the full set of depletion cross sections in a single neutron transport calculation while accounting for the influence of neighboring pebbles with different burnups without making assumptions about transforming burnup dependent k_{∞} to a core averaged k_{∞} . A full-core implementation of the MBSM enables one to account for the burnup dependence of spatially sensitive depletion cross sections.

By utilizing more burnup states in a model, one increases the effort required for convergence. Furthermore, creating a MBSM repeated structure becomes challenging if it cannot be represented in a standard crystal structure. Table 2-1 lists some recommended crystal structures for the MBSM repeated structure unit cell with 1-8 burnup states. MBSM can be applied to more complicated ratios of burnup states by stochastically filling a pebble bed with a given ratio of pebbles of various burnup states – though setting up this kind of model would require significant effort. These recommended unit cell structures can represent systems with 1-8 unique burnup states with maximum packing fractions high enough to model randomly packed pebble beds.

Table 2-1. Unit cell structure for multiple burnup state pebble bed repeated structures

Number Burnup States	Crystal Structure	Maximum Pebble Packing Fraction (%)	Image
1	Simple Hexagonal (SH)*	60	
2	Body Centered Cubic (BCC)	68	
3	Hexagonal	60	
4	Face Centered Cubic (FCC)	74	
5	Hexagonal Close Pack with Voided Site (HCPVS)	62	

6	Hexagonal Close Pack (HCP)	74	
7	Face Centered Cubic with Single Voided Site (FCCVS)	65%	
8	FCC	74%	

* Any other unit cell can be used

2.4 Implementation of MBSM with Burnup Equilibrium Analysis Utility (BEAU)

To facilitate the utilization of this methodology and increase the repeatability of these calculations a code, Burnup Equilibrium Analysis Utility (BEAU), was developed to implement this methodology. BEAU was developed with the following set of functional requirements:

1. Accurately perform equilibrium pebble bed depletion analysis for PB-FHR design studies.
2. Provide an easy-to-use interface:
 - a. Set up depletion with a minimal number of requisite files.
 - b. Flexible with respect to defining simulation parameters (defining which isotopes to test convergence in which burnup states, power level, acceptable error in fuel cycle dependent parameters, depletion fidelity, control of numerical stability options, pebble circulation definition)
 - c. Intuitive fuel cycle definition (fuel advancement scheme and makeup fuel compositions).
 - d. Intuitive communication of equilibrium depletion results
3. Enable analysis of advanced recirculation schemes (burnup dependent recirculation, fertile blankets, etc.)
4. Utilize reputable general purpose nuclear codes such as: MCNP5, Serpent and ORIGEN2.

BEAU is an equilibrium pebble bed depletion analysis framework written in Python that couples continuous-energy Monte Carlo neutron transport (MCNP5 or Serpent) with ORIGEN2 (Python Software Foundation). Early versions of BEAU utilized the depletion coupling code MOCUP as an interface between MCNP and ORIGEN (Moore, Schnitzler, Wemple, Babcock, & Wessel, 1995). However, a new more flexible interface between continuous energy Monte Carlo neutron transport codes and ORIGEN was developed, `mocup.py`; see Appendix A. `Mocup.py` reads the output file from a neutron transport calculation with tallies to calculate depletion cross sections for either MCNP5 or Serpent and organizes the results into an object (`mpos` object) that holds all the information to perform depletion analysis (depletion cross sections, flux distributions, volumes, fuel temperatures, depletion material compositions, etc.). Additionally, `mocup.py` has tools to easily work with material composition vectors, such as: reading and writing material composition vectors from MCNP5 input decks, Serpent input decks, importing material or ORIGEN punch files, adding composition vectors, multiplying composition vectors by factors, etc. `Mocup.py` can be utilized as a stand-alone depletion framework and its user manual and a simple benchmarking study are attached as Appendix A. `Mocup.py` is also utilized as a driver for the multi-batch equilibrium depletion analysis code Fast BEAU.

This subsection presents the technical details of implementing the MBSM with BEAU. The first subsection discusses the structure of BEAU including how it conceptualizes and defines a PBR fuel cycle, initializes the depletion analysis of an equilibrium pebble bed fuel cycle, and numerically calculates various fuel cycle parameters. The second section discusses how BEAU searches for the maximum burnup equilibrium state of a PBR. The final subsection describes how to set up, execute and analysis results of an equilibrium depletion analysis calculation with BEAU.

2.4.1 Fuel Cycle Analysis in BEAU

This subsection describes how BEAU models PBR fuel cycle numerically. This section will describe the fuel cycle objects (burnup states, progressions, circulation, circulation progressions and equilibrium fuel cycles) BEAU utilizes to define a PBRs fuel cycle. Also, this subsection will present how BEAU translates high level fuel cycle parameters (cycle length, fission neutron source and burnup) to a low-level depletion analysis description and conversely, how BEAU translates detailed depletion results into high level results (burnup).

2.4.1.1 Fuel Cycle Objects

This section describes the fuel cycle objects used by BEAU to define a PBR fuel cycle. An example fuel cycle where the core is discretized into five axial regions, four radial regions and the fuel pebbles circulate through the core twice will be used as a representative example to present the definition, attributes and methods of these fuel cycle objects. Figure 2-7 presents the geometry of the PBR's discretization and a conceptualization of burnup states (rectangles) that represent this fuel cycle. Table 2-2 presents some fuel cycle parameters.

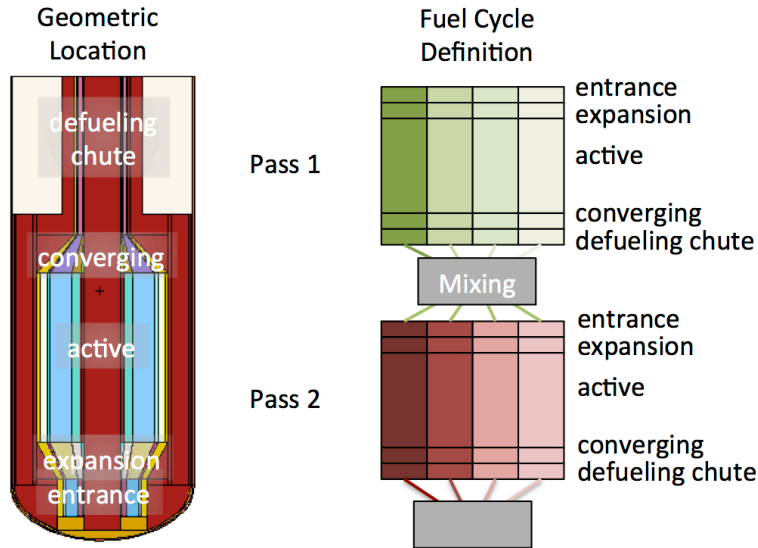


Figure 2-7 Representative PBR Fuel Cycle; (left) physical representation of the discretization (R, Z) of the burnup states; (right) conceptualization of the burnup states (smallest rectangles) that define a PBR fuel cycle.

Table 2-2. Parameters of the PB-FHR representative Fuel Cycle.

Fuel Cycle Parameters	Values
Power (MWt)	250
Power Density (MWt/m ³)	20
Number of Circulations	2
Number of Radial Channels	4
Number of Axial Nodes	5
Makeup Fuel	19.9w% ²³⁵ U LEU
Fuel Type	UC _{0.5} O _{1.5} Fuel Kernels in Annular Pebbles

The fuel cycle objects will be defined in increasing level of hierarchy from the low-level objects (burnup states) to high-level objects (equilibrium fuel cycle objects).

2.4.1.1.1 Burnup State

A burnup state represents the fuel in a given spatial (r_j, z_k) and burnup (b) bin. Figure 2-8 represents the conceptualization of the representative fuel cycle subdivided into burnup states. A burnup state contains all the attributes and methods to deplete a fuel from the point in burnup when it enters the control volume of a burnup state to when it exits. Additionally, it can return its time averaged power and time averaged material composition. A burnup state is the most fundamental fuel cycle object. Burnup states are represented by mocup.py's depletion objects. Table 2-3 presents a summary of the most important attributes and methods of a burnup state.

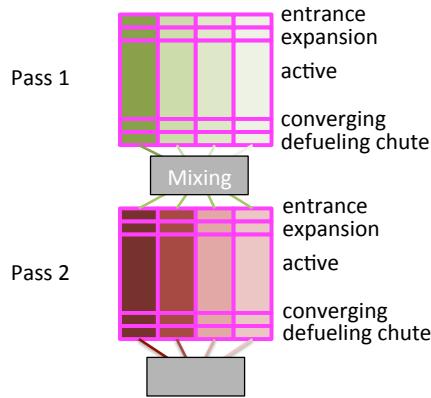


Figure 2-8. Conceptualization of the representative fuel cycle with each burnup state outlined in magenta.

Table 2-3. Summary of important attributes and methods of a burnup state

Attributes	
source	This is a list of the fission neutron source for each depletion timestep.
time	This is a list of the duration of each depletion timestep in terms of effective full power days, EFPD.
transport_module	This is a flag to define what continuous energy Monte Carlo neutron transport code is used in a specific calculation.
nct	“nuclide correspondence table”, This is a dictionary that maps MCNP5 or Serpent cross section libraries or isotopic identifiers in ZZAAM format.
mpo	This is a mocup.py object that contains all the information to perform a depletion analysis: cell title, fuel material title, material composition, volume, depletion cross sections, flux (normalized to the fission neutron source), ORIGEN calculated power, estimate of power density normalized per source neutron, and temperature of material.
Methods	
origenBEAU*	This method performs depletion analysis assuming constant depletion cross sections and constant neutron flux by calling ORIGEN2.
time_ave_mat	This method returns a dictionary with the time-averaged compositions of the burnup state indexed by the cell title of the burnup state after the burnup state has been depleted.

* this is a method in BEAU that accepts a depletion object

The origenBEAU method depletes the charge material for the entire residence time in a given burnup state. BEAU uses its own depletion method to call ORIGEN rather than the depletion method from mocup.py because many intermediate fuel compositions are must be reported to produce an equilibrium material rather than in traditional burnup analysis where either the end of step or middle of step compositions are sufficient. .

Each of these timesteps is divided into ten equal length sub-timesteps. The time-averaged powers are recorded for each timestep in the burnup state's mpo files and the isotopic composition vectors are stored in ORIGEN punch files in a directory called "moi_files" indexed by the burnup state's cell and depletion step number (moi.<burnup state cell>.<timestep>.pch).

The equations in the descriptions of the MBSM were all formulated continuously however results such as isotopic composition evolutions are calculated at discrete points in time. The isotopic composition vectors are recorded as the end of step (EOS) composition for each depletion step in addition to the beginning of step (BOS) composition required to initiate the depletion calculation. Therefore, the time-averaged composition for updating the neutron transport model can be calculated according to Equation 2-11.

$$\langle \bar{N}_{j,k,b}^i \rangle = \frac{\sum_{l=1}^{L_b} \Delta t_l \left(\frac{N_{j,k,b}(t_l) + N_{j,k,b}(t_{l-1})}{2} \right)}{\sum_{l=1}^L \Delta t_l}$$

Equation 2-11

2.4.1.1.2 Progression

A progression object represents a collection of burnup states in series. A progression can be thought of as a radial branch or channel because fuel in a given progression in theory does not mix with any parallel progressions due to plug flow. A progression object accepts an mpo object with the depletion information for each burnup state in the progress, and the current estimate of the cycle length for the progression and the fission neutron source and sets up the depletion analysis calculations for its burnup states and controls the depletion calculation process. Additionally, the progression object returns the power of its burnup states. Figure 2-9 represents the conceptualization of the representative fuel cycle subdivided into progressions. Table 2-4 presents a summary of the most important attributes and methods of a progression.

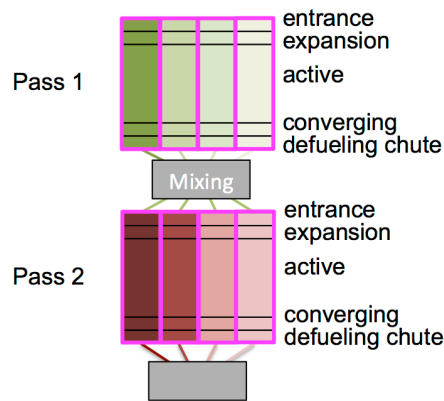


Figure 2-9. Conceptualization of the representative fuel cycle with each progression outlined in magenta.

Table 2-4. Summary of important attributes and methods of a progression

Attributes	
mpo	This single mpo file contains all the information required to deplete each burnup state in the progression.
depletions	This is a list that stores the burnup states as depletion objects
neutron_source	This is the latest estimate of the fission neutron source used to calculate the flux used for depletion analysis
cycle_length	This is the latest estimate of the residence time for the entire progression (i.e. the sum of the residence times of all the burnup states in the progression).
depl_parms	An object will all the settings for a depletion calculation: minimum number of burnup steps, initial length of burnup step in terms of (MWd/MT), minimum number of initial burnup steps, maximum burnup step.
pebble_flow	This is a dictionary where the pebble flux shape of each progression is mapped to its title.
cells	This is a list of all the cell titles of the burnup states of the progression
time	This is a dictionary that maps the residence time of specific burnup states to the cell title of each burnup state.
ave_powers	This is a dictionary that maps the time-averaged power of the burnup states to the cell title of each burnup state.
prog_power	This is the time-averaged power of the progression.
Methods	
populate	This method sets up the depletion analyses of each burnup state by initializing and populating a depletion object for each burnup state.
deplete	This method executes the depletion analysis of the burnup states sequentially.
power	This method calculated the time-averaged power of each burnup state and the entire progression and records them in the progressions ave_powers and prog_power attributes respectively.

Populating the depletion objects for each burnup state is one of the most important functions of a progression object. The cycle length of the progression is allocated between the burnup states based on the volume of the burnup states. Since the burnup states in a single progression are connected in series, the flow between these burnup states must be constant in a fully-packed pebble bed. Therefore, the volume of burnup state will dictate the fraction of the progression’s cycle length that is allocated to each burnup state; see Equation 2-12.

Equation 2-12
$$T_{j_b, k_b, b} = \frac{V_{j_b, k_b, b}}{V_{progression}} T^{progression}$$

BEAU needs to define the number- and duration of timesteps to deplete over in addition to the total residence time to completely define a burnup state. BEAU needs to use enough timesteps to be able to construct a representative material composition and representative power for each burnup state. Additionally, BEAU needs to be efficient about the number of timesteps in each depletion zones to minimize the computational requirements of the equilibrium search. BEAU determines the number- and duration of timesteps in a burnup state based on a set of depletion parameters in the depletion parameters object: minimum number of depletion steps, maximum burnup of initial depletion steps, minimum number of initial depletion steps and maximum length of a depletion step – each of these parameters can be adjusted by the user. Figure 2-10 shows the residence time of a burnup state subdivided into six timesteps. When depleting under constant flux, one cannot be certain of the burnup traversed over any particular time period therefore an estimate of the power is required. For the purposes of subdividing the residence time within a burnup state into timesteps, one can assume a constant power determined based on the multiplying the depletion flux by latest estimate of the time-averaged macroscopic cross sections as shown in Equation 2-13. BEAU then generates a set of timesteps (transformed from burnup steps) for each burnup states and records them in the depletion object’s time attribute.

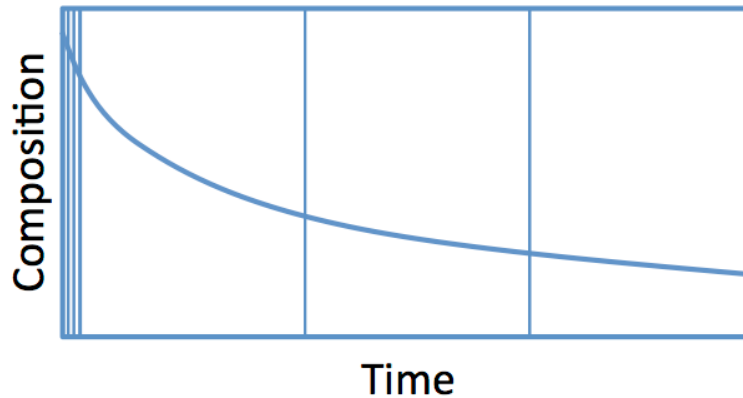


Figure 2-10. The residence time of a burnup state allocated into six depletion steps.

Equation 2-13

$$\bar{P}_{j,k,b}^{estimate} = S^{estimate} \frac{\int_{V_{j,k,b}} \int_0^{\infty} \psi(r_{j,b}, z_{k,b}, E) dE dV}{\int_{V_{core}} \int_0^{\infty} \bar{v}(E) \Sigma^{(n, fiss)}(r, z, E) \psi(r, z, E) dE dV} \sum_n^{isotopes} Q^n \sigma_{j,k,b}^{(n, fiss)} \langle {}^n \bar{N}_{j,k,b}^{i-1} \rangle$$

The deplete method depletes each burnup state in a progression in sequence. For all but the last burnup state, the EOS material composition vector is scaled by volumes to generate a charge material composition vector for the next burnup state as shown in Equation 2-14.

$$\bar{N}_{j,k,b}(0) = \frac{V_{j_b,k_b,b}}{V_{j_{b-1},k_{b-1},b-1}} \bar{N}_{j,k,b-1}(T_{b-1})$$

Equation 2-14

The power of the fuel progression is the sum of the powers in each on the constituent burnup states; see Equation 2-15.

$$P^{progression} = \sum_{burnup\ states,b}^{progression} \sum_{l=1}^{L_b} \frac{P_{j,k,b}(t_l) \Delta t_l}{T_b}$$

Equation 2-15

2.4.1.1.3 Circulation

A circulation represents a collection of progressions in parallel. A circulation can be thought of as a single pass through the pebble bed. Figure 2-11 represents the conceptualization of the representative fuel cycle subdivided into circulations. When one models recirculating PBRs, one can limit the number of burnup states by assuming a single homogenized fuel composition is used at the inlet of the PBR rather than creating branches for every history of radial branches fuel was circulated through. Figure 2-12 compares conceptualizations of a fuel cycle where the progressions are collapsed into a single composition after every pass through the representative fuel cycle and a fuel cycle with burnup states for each radial channel history. The FCM and SSAFLPSAM both make assumptions about mixing upon each pass to make the depletion and convergence of the depletion model tractable. Table 2-5 presents a summary of the most important attributes and methods of a circulation.

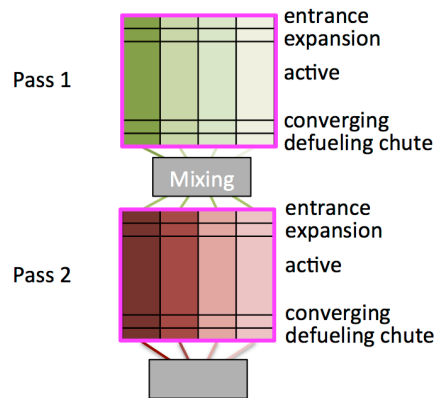


Figure 2-11. Conceptualization of the representative fuel cycle with each circulation outlined in magenta.

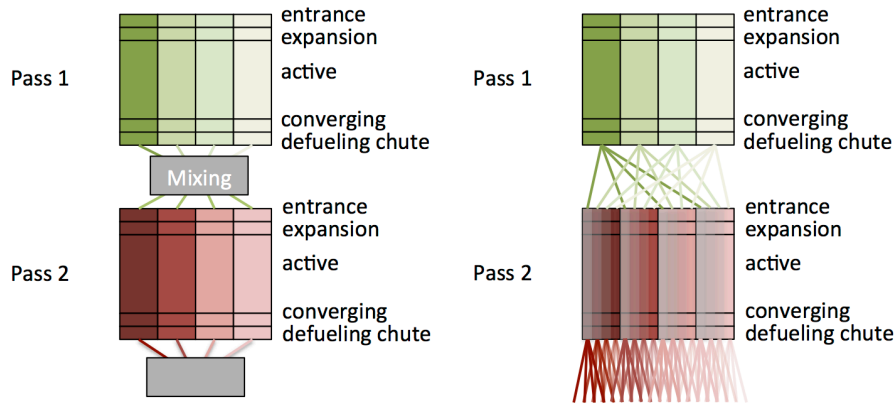


Figure 2-12. Comparison of fuel cycle conceptualizations; (left) assuming mixing after each circulation; (right) assuming burnup state for each history of radial progressions.

Table 2-5. Summary of important attributes and methods of a circulation

Attributes	
mpo	This single mpo file contains all the information required to deplete each burnup state in the circulation.
progressions	This is a dictionary that stores the progressions as depletion objects.
neutron_source	This is the latest estimate of the fission neutron source used to calculate the flux used for depletion analysis.
cycle_length	This is the latest estimate of the expected residence time for the entire progression (i.e. the pebble flow weighted average of the residence times of all the burnup states in the progression).
depl_parms	An object will all the settings for a depletion calculation: minimum number of burnup steps, initial length of burnup step in terms of (MWd/MT), minimum number of initial burnup steps, maximum burnup step.
pebble_flow	This is a dictionary where the pebble flux shape of each progression is mapped to its title.
ave_power	This is the time-averaged power of the circulation.
Methods	
populate	This method initializes and populates each progression in the circulation.
deplete	This method executes the depletion analysis of circulation.
mix	This method produces a pebble flow weighted average of the discharge composition of the circulation.
power	This method calculated the time-averaged power of the entire circulation and records them in the circulation's ave_power attribute.

Since the progressions are in parallel rather than series (as in progressions) the expected cycle length is an average cycle length rather than the sum of the residence times. Additionally, each progression within a circulation can have a different flow rate.

Therefore, the relative flow rate of pebbles with respect to other progressions in a given circulation is taken into account when estimating the cycle length in each of the constituent progressions by utilizing the Equation 2-16 as the relationship between the expected cycle length of a circulation and cycle lengths of its constituent progressions and assuming that the cycle length of each progression should be inversely proportional to the pebble flux at the outlet of the core and that the cross sectional area at the outlet of the core is proportional to the total volume of the fuel progression. Equation 2-17 is used to calculate the cycle lengths of the circulation's constituent progressions. Because BEAU searches for the maximum burnup state by perturbing the cycle lengths of each fuel stream, the relative values of the pebble flow distribution is used rather than an absolute pebble flow rate.

$$\langle T^{circulation} \rangle = \frac{\sum_{p=i}^{n_{progression}} T_p \phi_p^{pebble} A_p(H)}{\sum_{p=i}^{n_{progression}} \phi_p^{pebble} A_p(H)}$$

Equation 2-16

$$T_{p=j} = \langle T^{circulation} \rangle \frac{\sum_{p=i}^{n_{progression}} \phi_p^{pebble} V_p}{\phi_{p=j}^{pebble} \sum_{p=i}^{n_{progression}} V_p}$$

Equation 2-17

The volume used in BEAU is the aggregate volume of fuel kernels rather than the volume occupied by the fuel pebbles. This accounts for effects caused by spatially dependent porosity distributions (so long as the spatially dependent porosity distribution is accounted for in calculating the fuel volume). One can think of changes in pebble porosity as transforms on the control volume of the burnup states. Higher porosity or lower packing fractions reduce the amount of pebbles that can reside in a depletion zone, effectively reducing the volume as compared to a constant pebble porosity case. This volume reduction should be accounted for in the volume of fuel kernels.

After the neutron transport has been performed, the depletion calculations of progressions are not strongly mutually dependent – they are dependent in that their total power influences the search for the fission neutron source, but their compositions are not directly dependent as with fuel cycle objects that are in series. Therefore, no effort is made to deplete the progressions in any particular order.

The mix method uses Equation 2-8 to produce a collapsed composition. However, assuming a constant volume of progression to cross sectional area at the outlet ratio, Equation 2-8 can be reformulated based on volume of progressions, which is recorded in the mpo files of the fuel progressions, as shown in Equation 2-18.

$$\langle \bar{N}^{collapsed} \rangle = \frac{\sum_{p=i}^{n_{progression}} \bar{N}_p (T_p) \phi_p^{pebble} V_p}{\sum_{p=i}^{n_{progression}} \phi_p^{pebble} V_p}$$

Equation 2-18

The power of a circulation is the sum of the powers in each on the constituent progressions; see Equation 2-19.

$$P^{circulation} = \sum_{p=1}^{n_{progressions}} \bar{P}_p$$

Equation 2-19

2.4.1.1.4 Circulation Progression

A circulation progression object represents a collection of circulations in series from the beginning of cycle (BOC) to the end of cycle (EOC). All the fuel in the circulation progressions constituent burnup states were at some point the same BOC composition. Therefore, a circulation progression can be thought of as the fuel cycle of a single fuel type. Figure 2-13 represents the conceptualization of the representative fuel cycle subdivided into circulation progressions. In addition to initializing circulations objects and controlling the depletion of the circulations in series, the circulation progression must also calculate the burnup of the entire fuel cycle the fuel type. Burnup is calculated in the circulation progression object, because the lower-order fuel cycle objects do not necessarily have the BOC fuel composition needed to calculate burnup in terms of MWd/MT.

Table 2-6 presents a summary of the most important attributes and methods of a circulation progression.

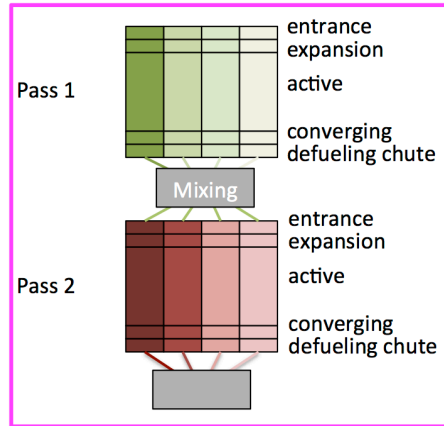


Figure 2-13. Conceptualization of the representative fuel cycle with each circulation progression outlined in magenta.

Table 2-6. Summary of important attributes and methods of a circulation progression

Attributes	
mpo	This single mpo file contains all the information required to deplete each burnup state in the circulation.
passes	This is a dictionary that stores the circulation objects.
circuits	This is a list that stores the sequence of the circulations.
neutron_source	This is the latest estimate of the fission neutron source used to calculate the flux used for depletion analysis.
makeup	This is a mocup material object that represents the BOC composition. This makeup fuel composition is given in units of moles/cm ³ .
cycle_length	This is the latest estimate of the expected residence time for the entire circulation progression.
target_burnup	This is the target burnup for the circulation progression in terms of MWd/MT
depl_parms	An object will all the settings for a depletion calculation: minimum number of burnup steps, initial length of burnup step in terms of MWd/MT, minimum number of initial burnup steps, maximum burnup step.
pebble_flow	This is a dictionary where the pebble flux shape of each progression is mapped to its title.
ave_power	This is the time-averaged power of the circulation progression.
Methods	
populate	This method initializes and populates each circulation in the circulation progression.
deplete	This method executes the depletion analysis of the circulations sequentially.
power	This method calculates the power of the entire circulation progression.
burnup	This method calculates the expected burnup of the entire circulation progression.
search_cycle_length*	This method iteratively searches for the cycle length of the circulation progression to impose the target burnup.

* The functionality of this method will be discussed in detail in Interpolation Type Searches.

The cycle length of the circulation progression is what one traditionally thinks of as the cycle length for a given fuel type. The cycle length for the circulations progression must be allocated amongst its constituent circulations analogous to how the cycle length of a progression is allocated amongst its constituent burnup states. The same flow of pebbles feeds each of the circulations in series and as a result the residence time in each constituent circulation is dependent on the circulation's volume as shown in Equation 2-20 and not dependent on relative pebble fluxes in any fuel progressions.

Equation 2-20 $\langle T^{circulation} \rangle = \frac{\Psi^{circulation}}{\Psi^{circulation\ progression}} T^{circulation\ progression}$

The circulation progression's deplete method depletes each circulation in sequence. Before the depletion begins BEAU generates a BOC ORIGEN composition files for each the first burnup state in each of the progressions in the first circulation. Additionally, after each circulation BEAU uses the circulation's mix method to generate a collapsed material composition for the next circulation. This collapsed material composition is used to generate ORIGEN compositions files for the first burnup state in each progression in the next circulation.

Calculating burnup becomes more complicated in fuel cycles that involve fuel histories that branch-off and collapse. In these complicated fuel cycles the concepts of series and parallel become very important to how burnup is calculated because burnup depends on a fuel's power production history. The expected burnup of a fuel object that has constituent lower-order fuel objects in series (progression, circulation progression) is simply the sum of the partial burnups of these lower-order fuel objects, because there is only one possible history. Conversely, the expected burnup of a fuel object that has constituent lower-order fuel objects in parallel (circulation) is the pebble-flow-weighted average of burnups of these lower-order fuel objects (similar to the expected cycle length), because the fuel could have taken multiple branches or histories during its depletion. Equation 2-21 - Equation 2-24 present the equations BEAU uses to calculate the burnups of burnup states (Equation 2-21), progressions (Equation 2-22), circulations (Equation 2-23) and circulation progressions (Equation 2-24) respectively.

Equation 2-21
$$BU_b = \frac{\bar{P}_b T_b}{\rho_{cp} \Psi_b}$$

Equation 2-22
$$BU_p = \sum_{b=1}^{n_{burnup\ states}} \frac{\bar{P}_b T_b}{\rho_{cp} \Psi_b}$$

Equation 2-23
$$BU_c = \frac{\sum_{p=1}^{n_{progressions}} \left(\sum_{b=1}^{n_{burnup\ states}} \frac{\bar{P}_b T_b}{\rho_{cp} \Psi_b} \right) \phi_p^{pebble} \Psi_p}{\sum_{p=1}^{n_{progressions}} \phi_p^{pebble} \Psi_p}$$

Equation 2-24
$$BU_{cp} = \sum_{c=1}^{n_{circulations}} \left[\frac{\sum_{p=1}^{n_{progressions}} \left(\sum_{b=1}^{n_{burnup\ states}} \frac{\bar{P}_b T_b}{\rho_{cp} \Psi_b} \right) \phi_p^{pebble} \Psi_p}{\sum_{p=1}^{n_{progressions}} \phi_p^{pebble} \Psi_p} \right]$$

The power of the circulation progression is the sum of the powers in each on the constituent circulations; see Equation 2-25.

$$P^{circulation\ progression} = \sum_{c=1}^{n_{circulation}} \bar{P}_c$$

Equation 2-25

2.4.1.1.5 Equilibrium Fuel Cycle

An equilibrium fuel cycle object represents a collection of circulation progressions in parallel. An equilibrium fuel cycle object is the highest-order fuel cycle object and can be conceptualized as the fuel cycle object that represents the entire equilibrium fuel cycle of a PBR. By utilizing a fuel cycle object higher than a circulation progression, one can model fuel cycles with multiple fuel types, such as a PBR with a seed and blanket type configuration. Figure 2-14 represents the conceptualization of the representative fuel cycle with an additional circulation progression of a fertile thorium blanket to demonstrate the application of an equilibrium fuel cycle object to a more complicated fuel cycle. In addition to initializing the circulation progressions objects, controlling the depletion of the circulation progressions in parallel and calculating the full-core power, the equilibrium fuel cycle object has methods to update the equilibrium neutron transport model of the PBR. Table 2-7 presents a summary of the most important attributes and methods of an equilibrium fuel cycle object.

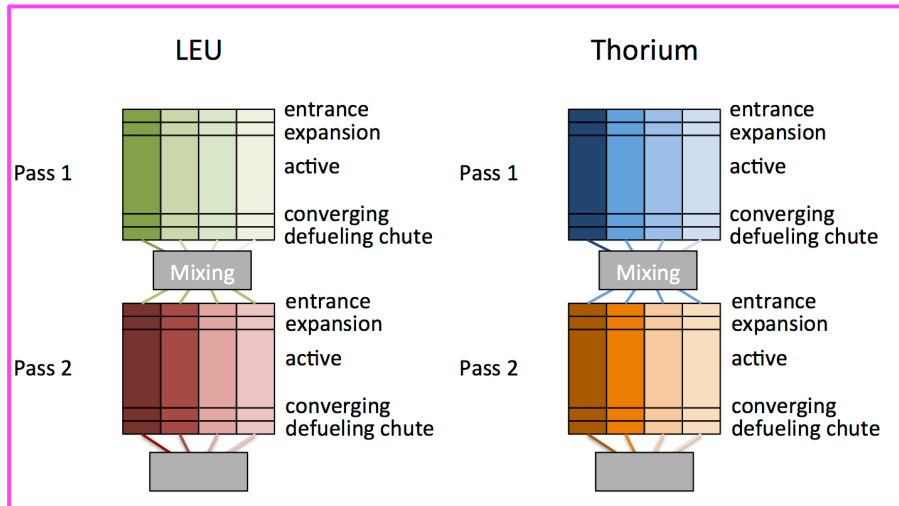


Figure 2-14. Conceptualization of the representative fuel cycle with an additional circulation progression as a single equilibrium fuel cycle object outlined in magenta.

Table 2-7. Summary of important attributes and methods of a circulation

Attributes	
mpo	This single mpo file contains all the information required to deplete each burnup state in the circulation.
circ_progs	This is a dictionary that stores the circulation progression objects.
circuits	This is a dictionary that stores lists of the sequence of the circulations mapped to the circulation progression title.

neutron_source	This is the latest estimate of the fission neutron source used to calculate the flux used for depletion analysis.
makeup	This is a dictionary that maps the mocup material object that represents the BOC composition for each circulation progression to their respective titles.
cycle_length	This is a dictionary that maps latest estimate of the expected residence time for each circulation progression to their respective titles.
target_k	This is the target $k_{\text{effective}}$ for the full-core PBR or k_{∞} for a unit cell model.
target_power	This is the target power for the full-core PBR.
target_burnup	This is a dictionary that maps the target burnup for the circulation progression for each circulation progression to their respective titles.
pebble_flow	This is a dictionary where the pebble flux shape of each progression is mapped to its title.
driver	This is the title of the circulation progression of the driver fuel type
ave_power	This is the time-averaged power of the full-core PBR.
AUSCE*	This is an AUTomatic Salt Concentration Equilibrator object that holds the attributes and methods to calculate the equilibrium ${}^6\text{Li}$ concentration and update the neutron transport input deck.
Methods	
populate	This method initializes and populates each circulation progression in the circulation progression.
power	This method calculates the power of the full-core PBR
search	This method searches for the cycle length to impose the target burnup by executing the search_cycle_length method for each circulation progression
deplete	This method depletes each circuit progression for the defined fission neutron source strength and cycle length definition. (note: this method was developed mostly for troubleshooting BEAU)
compBRO	This method updates the neutron transport input deck to reflect the latest estimate of time-averaged compositions for each burnup state.
search_source*	This method iteratively searches for the fission neutron source to impose the target power.
search_comp***	This method iteratively searches for the equilibrium state of the PBR for a fixed set of target burnups for each circulation progress.
search_burnup***	This method iteratively searches for the burnup of the driver circulation progression of the PBR that imposes the target $k_{\text{effective}}$
output**	This method makes an output file for a maximum burnup equilibrium fuel cycle search calculation.

* The functionality of this method will be discussed in the section of Search for Maximum Burnup Equilibrium Fuel Cycle.

** This method accepts an equilibrium fuel cycle object

The populate method initializes and populates each circulation progression in the equilibrium fuel cycle object. It should be noted that structure of each populate method in a fuel cycle object populates each of its lower-order fuel cycle object all the way down to the burnup state level.

The compBRO method updates neutron transport of the PBR at the equilibrium state. It uses Equation 2-18 to calculate the new estimates of the fuel compositions at equilibrium state. Numerical oscillations in the estimates of fuel compositions resulted in difficulty converging to an equilibrium model of a PBR for a set of target burnups. Therefore, BEAU applies a user-defined weighting factors to the new estimate of the fuel composition and the previous estimate of the fuel composition when updating the neutron transport model; see Equation 2-26. By default this weighting factor is set to 0.5.

Equation 2-26

$$\langle \bar{N}_{j,k,b}^{transport} \rangle = (1 - w) \langle \bar{N}_{j,k,b}^{i+1} \rangle + w \langle \bar{N}_{j,k,b}^i \rangle$$

2.4.1.2 Search for Maximum Burnup Equilibrium Fuel Cycle with BEAU

This subsection describes how BEAU searches for the PBR fuel cycle parameters (cycle lengths, fission neutron source, etc.) to determine the maximum burnup condition for a given PBR: the maximum burnup at which the PBR satisfies the user's criticality criterion at the target power. This section will describe each of BEAU's search modules (search cycle, search source, search comp and search burnup) and discuss their convergence criteria. Finally, this section presents FHR-specific methodology to search for the equilibrium ⁶Li concentration integrated into BEAU.

The function of the search module in BEAU is presented in Table 2-8 Search modules in BEAU.. Notice that there are only two search types. Search modules of the same search type have almost identical structures, which are discussed together in the next two sub-sections.

Table 2-8 Search modules in BEAU.

Search Module	Function	Search Type
search_cycle_length	This method searches for the cycle length of a given circulation that imposes the current estimate of the target burnup for a circulation progression assuming the current estimate of the fission neutron source.	Interpolation
search_source	This method searches for the fission neutron source the simultaneously imposes the target power and the set of target burnups for all the circulation progressions.	Interpolation
search_composition	This method searches for the converged equilibrium state that satisfies target power and target burnups requirements for all the circulation progressions.	Fixed Point
search_burnup	This method searches for the maximum burnup of the driver fuel stream to impose the criticality criterion for a converged equilibrium state which satisfying the target power and if applicable, other burnup criteria.	Interpolation
AUSCE	This method searches for the converged equilibrium concentration of ⁶ Li for the equilibrium state of a flibe-cooled PBR.	Fixed Point

2.4.1.2.1 Interpolation Type Searches

The interpolation type search modules – search_cycle_length, search_source, and search_burnup – involve perturbing an independent fuel cycle parameter – circulation progression cycle length, fission neutron source, driver burnup – to impose a target dependent fuel cycle parameter – burnup, power or model multiplication factor. Table 2-9 presents a summary of these interpolation type search modules.

Table 2-9. Summary of Interpolation Type Search Modules.

Module name	Independent Fuel Cycle Variable	Dependent Fuel Cycle Variable
search_cycle_length	Circulation progression cycle length	Circulation progression burnup
search_source	Fission neutron source	Total system power
search_burnup	Driver fuel burnup	Model multiplication factor

For the default mode of search, BEAU makes an educated guess at the independent fuel cycle parameter and perturbs this value by a small amount greater and less than this

educated guess. BEAU performs burnup analysis to calculate the corresponding dependent fuel cycle parameters to these perturbed values to develop a linear equation of the dependent value as a function of the independent values using a least square fit. After this initial perturbation, BEAU moves into a convergence loop where the linear regression is used to estimate the independent value that imposes the target dependent value.

The educated guess for the cycle length is made by using Equation 2-13 to estimate power in each burnup state and combining Equation 2-12, Equation 2-17, and Equation 2-20 to determine the fraction of the total residence time of the circulation progression allocated to each burnup state as shown in Equation 2-27. Substituting these equations into Equation 2-24, one can develop an expression for the estimated burnup as a function of circulation progression cycle length, Equation 2-28. The circulation progression object has a method, `normalized_burnup_estimate`, to perform these summations to return this estimate of burnup normalized to cycle length. The small perturbation to the estimated cycle length in the `search_cycle_length` module is controlled by a value in the depletion parameters object, `burnup_perturb`, which defines the perturbation in terms of the estimated cycle length; by default `burnup_perturb` is set to 20%.

$$T_{j_b, k_b, b} = \frac{\Psi_{j_b, k_b, b}}{\Psi_{progression}} \frac{\sum_{p=i}^{n_{progression}} \phi_p^{pebble} \Psi_p}{\phi_p^{pebble} \sum_{p=i}^{n_{progression}} \Psi_p} \frac{\Psi_{circulation}}{\Psi_{circulation\ progression}} T^{circulation\ progression}$$

Equation 2-27

Equation

2-28

$$BU_{cp} = \sum_{c=1}^{n_{circulations}} \left(\frac{\sum_{p=1}^{n_{progression}} \left(\frac{\sum_{b=1}^{n_{burnup\ states}} \left(\frac{\int_0^\infty \int_0^\infty \psi(r, z, E) dE dV}{\int_0^\infty \bar{v}(E) \Sigma^{(n, fission)} \psi(r, z, E) dE dV} \sum_{i=1}^{isotopes} Q^i \sigma_b^{(n, fission)} \frac{\Psi_{j_b, k_b, b}}{\Psi_{progression}} \frac{\sum_{p=i}^{n_{progression}} \phi_p^{pebble} \Psi_p}{\phi_p^{pebble} \sum_{p=i}^{n_{progression}} \Psi_p} \frac{\Psi_{circulation}}{\Psi_{circulation\ progression}} T^{circulation\ progression}}{\rho_{cp} \Psi_b} \right) \right) \phi_p^{pebble} \Psi_p}{\sum_{p=1}^{n_{progression}} \phi_p^{pebble} \Psi_p} \right)$$

The educated guess for the fission neutron source is calculated by estimating the fission rate assuming a reasonable value for the energy released per fission (200MeV) and multiplying by a reasonable guess for the average number of neutrons per fission (2.5)

as shown in Equation 2-29. The small perturbation to the estimated fission neutron source in the search_source module is controlled by a value in the depletion parameters object, source_perturb, which defines the perturbation in terms of the estimated fission neutron source; by default source_perturb is set to 0.20.

$$S = \frac{P^{target}}{\bar{Q}} \bar{\nu} \approx \frac{P^{target}}{200MeV} 2.5$$

Equation 2-29

The educated guess for the driver fuel's burnup is calculated based on the fissile content of the fuel. It is assumed a typical burner system should get roughly 10000 MWd/MT (~1% FIMA) for every 1% fissile in the makeup fuel composition as shown in Equation 2-30. The small perturbation to the estimated driver burnup in the search_burnup module is controlled by a value in the depletion parameters object, burnup_perturb, which defines the perturbation in terms of the estimated cycle length; by default burnup_perturb is set to 20%.

$$BU^{est} = 10000 \left(\frac{MWd}{MT} \right) \frac{\sum_i^{fissile} N_i}{\sum_i^{actinide} N_i}$$

Equation 2-30

Each dependent fuel cycle parameter has a user-controlled acceptable error stored in the depletion parameters object. If for a given estimate of an independent fuel cycle parameter, the resulting dependent fuel cycle parameter is within its acceptable error, either greater- or less than, the search loop stops. These acceptable errors are listed in Table 2-10.

Table 2-10. Acceptable Error Variables for Interpolation Type search modules.

Name	Definition	Search Module	Default Value
burnup_err	This is the acceptable error for the burnup in terms of the target burnup.	search_cycle_length	0.5%
power_err	This is the acceptable error for power in terms of the target power	search_source	0.5%
keff_err	This is the acceptable error for the models multiplication factor in absolute terms	search_burnup	0.1%

2.4.1.2.2 Fixed-Point Type Searches

Rather than search for independent fuel cycle parameters, fixed point type searches in BEAU aim to converge to a self-consistent equilibrium state. There are two of these

fixed-point search modules. Search_comp searches for the equilibrium state of a PBR for a set of target burnups and AUSCE searches for the equilibrium ⁶Li concentration in fluoride salt coolant for pebble bed variants of FHRs.

The search_comp module updates the equilibrium state with the compBRO module as explained previously and checks the convergence of the equilibrium state each time neutron transport is performed with the comp_convergence module. This module has three convergence criteria: number of iterations, the system multiplication factor and the convergence of important isotopes. A user-controlled minimum number of iterations must be made to reduce influence the new converged state on the initial estimate of the converged state, usually a BOL model or the converged equilibrium state at a different burnup. Then the system multiplication factor is used as a rough metric of convergence to an equilibrium state. Finally to ensure that the multiplication factor has converged because the equilibrium state has converged, BEAU checks a set of user-defined important isotopes defined in the depletion parameters object. The important isotopes dictionary contains a pairs of burnup states and lists of isotopes to be explicitly checked for convergence – note that each burnup state has an independent list of isotopes. The acceptable error for these isotopes is stored in a value called mat_err in the depletion parameters object. When all of these convergence criteria are satisfied the iteration loops is complete.

AUSCE updates all the material cards for materials containing coolant with its equilibrate method and checks for convergence of ⁶Li with its li6_convergence module. AUSCE has two convergence criteria: number of iterations and the ratio of ⁶Li to ⁹Be, “Li ratio”. The equilibrium ⁶Li is a function of ⁹Be $\sigma(n,\alpha)$, ⁶Li $\sigma(n,abs)$ and the concentration of ⁹Be as shown by Equation 2-31 (Fratoni, Greenspan, & Peterson, Neutronic and Depletion Analysis of the PB-AHTR, 2007).

$$N_{Li-6} = \frac{\sigma_{Be-9}^{(n,\alpha)}}{\sigma_{Li-6}^{(n,abs)}} N_{Be-9}$$

Equation 2-31

The first step in this search module involves checking for convergence. The convergence check first checks the number of iterations and then moves on to the convergence of the Li ratio. The acceptable error for the Li ratio is defined by a user-controlled parameter called mat_err in the AUSCE object. When all of these convergence criteria are satisfied a convergence flag is set to converged and the composition of the lithium is no longer updated for any of the remaining iterations in the equilibrium composition search loop.

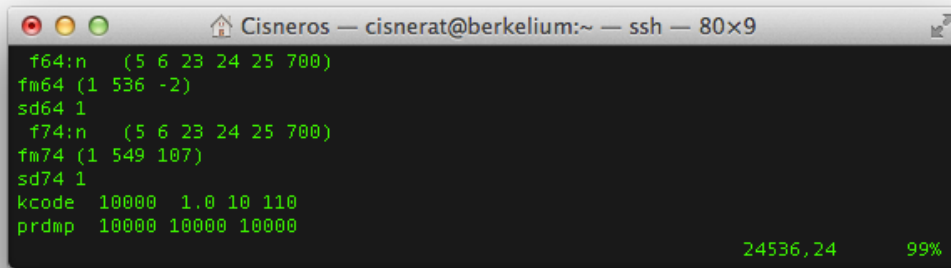
2.4.1.3 Executing BEAU

2.4.1.3.1 Neutron Transport Input Deck

The neutron transport input deck should be format to perform a standard depletion analysis with mocup.py as described in appendix A. For MCNP, this entails including tallies to generate depletion cross sections with the appropriate comment and ensuring

volumes are defined correctly for the aggregate volume of the fuel. For Serpent, the user must include a burn flag for the materials of each burnup state and as for MCNP, ensure the aggregate volume for each burnup state is defined in the input deck.

Additional tallies for lithium production and destruction rates must be added to the input deck to utilize the AUSCE method. By default tally 64 and 74 are identified as the tallies for the destruction of ${}^6\text{Li}$ as shown in Figure 2-15. In these tallies cells 5, 6, 23, 24, 25 and 700 are the only cells with flibe, materials 536 and 549 are single isotope materials for ${}^6\text{Li}$ and ${}^9\text{Be}$ respectively and reaction -2 and 107 are the mt reaction numbers for total neutron absorption and (n, α) respectively.



```
Cisneros — cisnerat@berkelium:~ — ssh — 80x9
f64:n (5 6 23 24 25 700)
fm64 (1 536 -2)
sd64 1
f74:n (5 6 23 24 25 700)
fm74 (1 549 107)
sd74 1
kcode 10000 1.0 10 110
prdmp 10000 10000 10000
24536,24 99%
```

Figure 2-15. Tallies for AUSCE

2.4.1.3.2 Equilibrium Fuel Cycle Definition

The BEAU equilibrium fuel cycle definition is a python script that initializes- and fully defines an equilibrium fuel cycle object and executes the search for the equilibrium burnup state. This sub-section presents the user-defined parameters that must be defined to perform a pebble bed equilibrium depletion analysis maximum burnup search, using BEAU. This equilibrium fuel cycle definition corresponds to the representative fuel cycle used to explain the fuel cycle objects; Figure 2-16 shows the representative fuel cycle with each fuel cycle objects labeled.

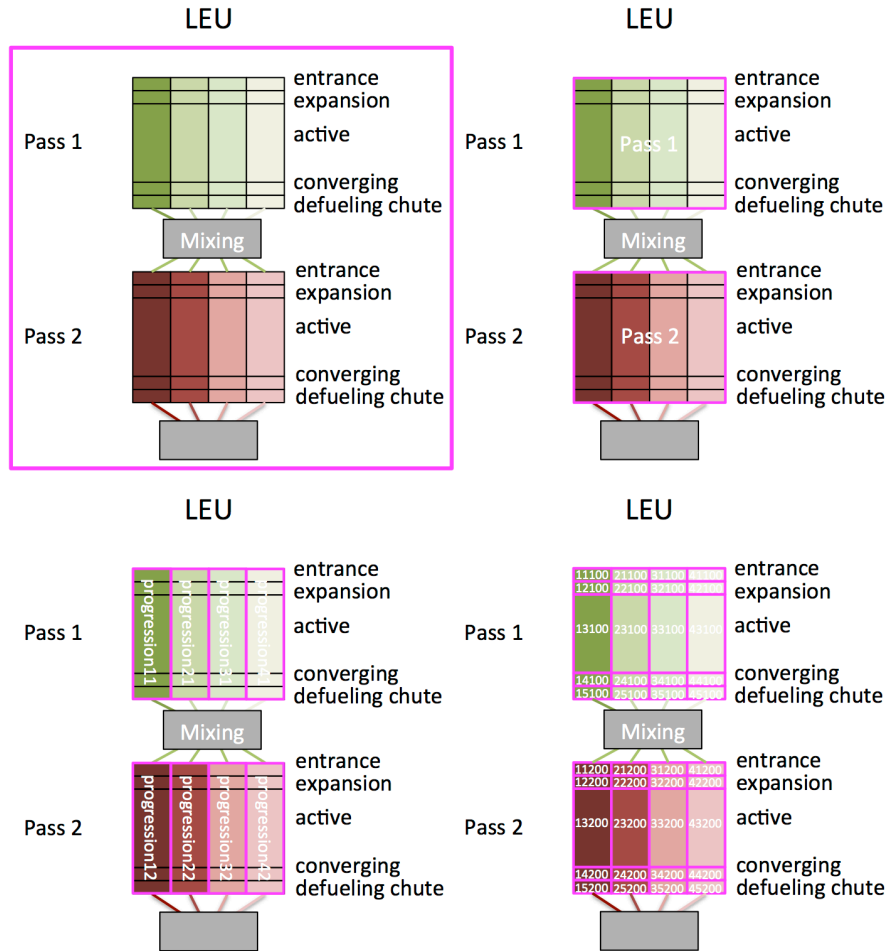
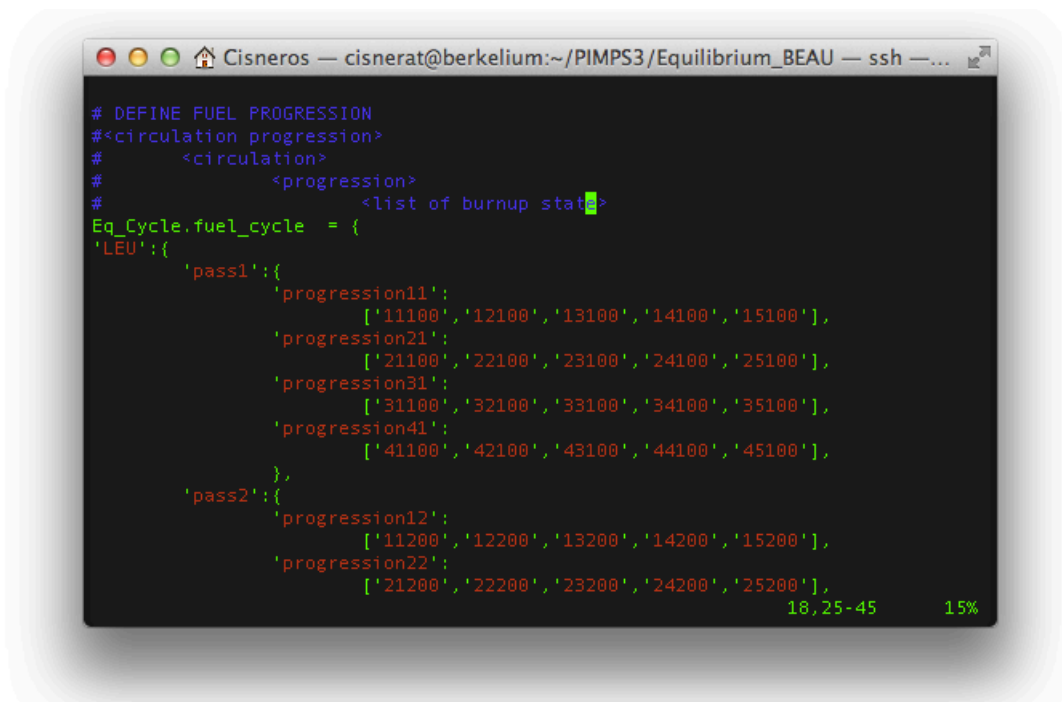


Figure 2-16. Representative fuel cycle with each fuel cycle object labeled; (upper left) representative fuel cycle subdivided into circulation progressions; (upper right) representative fuel cycle subdivided into circulations; (lower left) representative fuel cycle subdivided into progressions; (lower right) representative fuel cycle subdivided into burnup states.

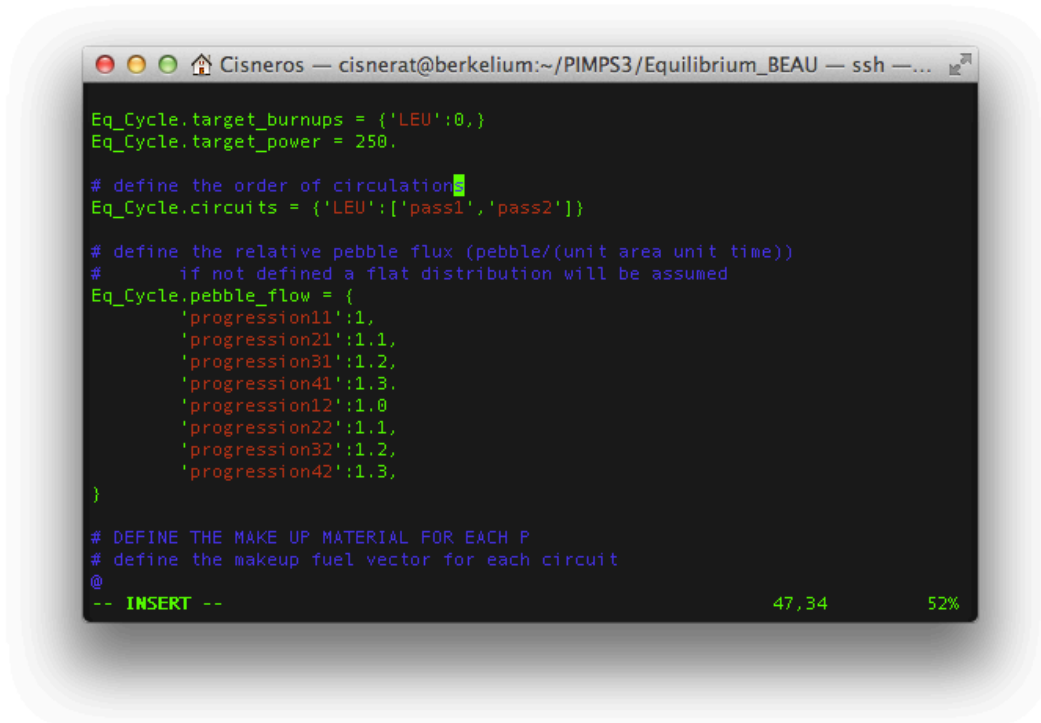
The burnup states are organized circulation progressions, circulations and progressions in the fuel_cycle dictionary as shown in Figure 2-17. The first level of this dictionary indexes the circulation progressions by their titles. The second level of this dictionary indexes the circulations by their titles. The third level of this dictionary maps a sequential list of burnup states indexed by their cell titles to the title of their progression.



```
# DEFINE FUEL PROGRESSION
#<circulation progression>
#   <circulation>
#       <progression>
#           <list of burnup stat>
Eq_Cycle.fuel_cycle = {
'LEU':{
  'pass1':{
    'progression11':
      ['11100','12100','13100','14100','15100'],
    'progression21':
      ['21100','22100','23100','24100','25100'],
    'progression31':
      ['31100','32100','33100','34100','35100'],
    'progression41':
      ['41100','42100','43100','44100','45100'],
  },
  'pass2':{
    'progression12':
      ['11200','12200','13200','14200','15200'],
    'progression22':
      ['21200','22200','23200','24200','25200'],
  }
}
18,25-45 15%
```

Figure 2-17. Equilibrium fuel cycle definition (1): organization of fuel cycle objects.

The order of the circulations are defined in the <equilibrium fuel cycle>.circuits dictionary as shown in Figure 2-18. This dictionary contains lists of the circulations in sequence to the title of their respective circulation progressions.



```
Cisneros — cisnerat@berkelium:~/PIMPS3/Equilibrium_BEAU — ssh —...
Eq_Cycle.target_burnups = {'LEU':0,}
Eq_Cycle.target_power = 250.

# define the order of circulation
Eq_Cycle.circuits = {'LEU':['pass1','pass2']}

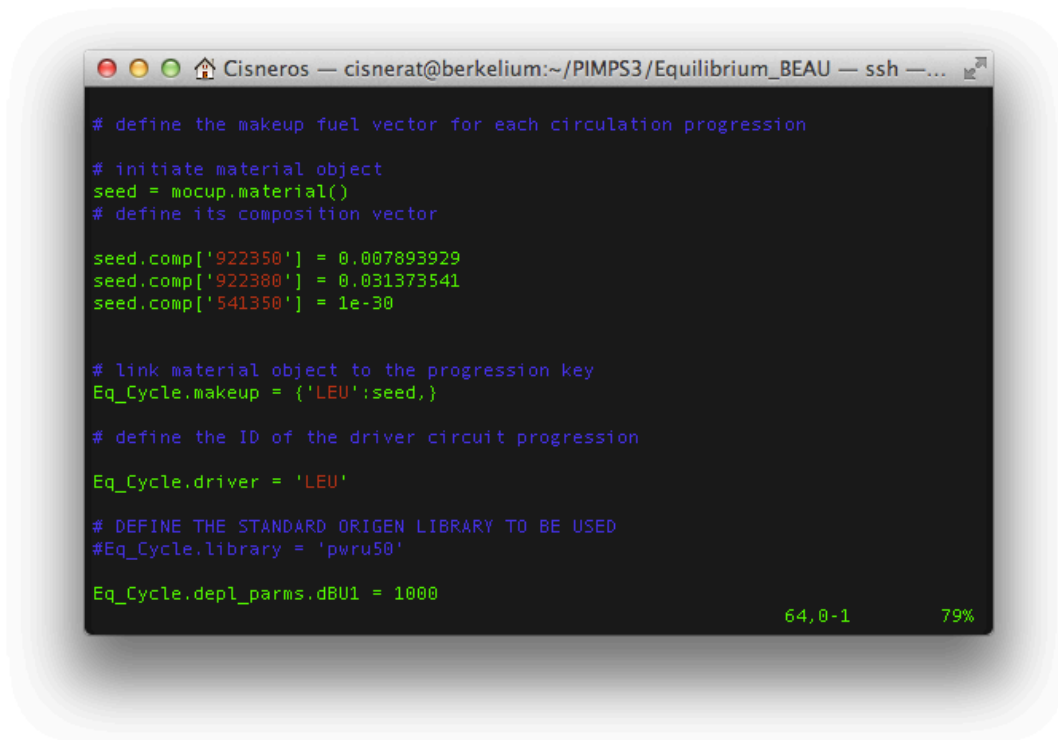
# define the relative pebble flux (pebble/(unit area unit time))
# if not defined a flat distribution will be assumed
Eq_Cycle.pebble_flow = {
    'progression11':1,
    'progression21':1.1,
    'progression31':1.2,
    'progression41':1.3,
    'progression12':1.0,
    'progression22':1.1,
    'progression32':1.2,
    'progression42':1.3,
}

# DEFINE THE MAKE UP MATERIAL FOR EACH P
# define the makeup fuel vector for each circuit
@
-- INSERT -- 47,34 52%
```

Figure 2-18. Equilibrium fuel cycle definition (2): target burnups, target power, circuits, and pebble flow definition.

Figure 2-18 also shows how the initial set of target burnups, the target power and the pebble flux distribution are defined. The <equilibrium fuel cycle>.target_burnups maps the target burnups of each circulation progression to their titles (this target burnup is set to 0 because BEAU will estimate the target burnup based on the fissile content of the makeup composition or an user-defined initial guess). The target power is defined in terms of MWt. The pebble flow distribution is defined in relative pebble flux (pebbles/(unit area * unit time)).

The user must also define the makeup composition vector for each circulation progression in <equilibrium fuel cycle>.makeup using a mocup.py material object and map them to each circulation progression in the makeup dictionary as shown in Figure 2-19. Additionally, the user must declare which circulation progression is the driver by setting <equilibrium fuel cycle>.driver to the title of the circulation progression of the driver fuel stream.



```
# define the makeup fuel vector for each circulation progression

# initiate material object
seed = mocup.material()
# define its composition vector

seed.comp['922350'] = 0.007893929
seed.comp['922380'] = 0.031373541
seed.comp['541350'] = 1e-30

# link material object to the progression key
Eq_Cycle.makeup = {'LEU':seed,}

# define the ID of the driver circuit progression

Eq_Cycle.driver = 'LEU'

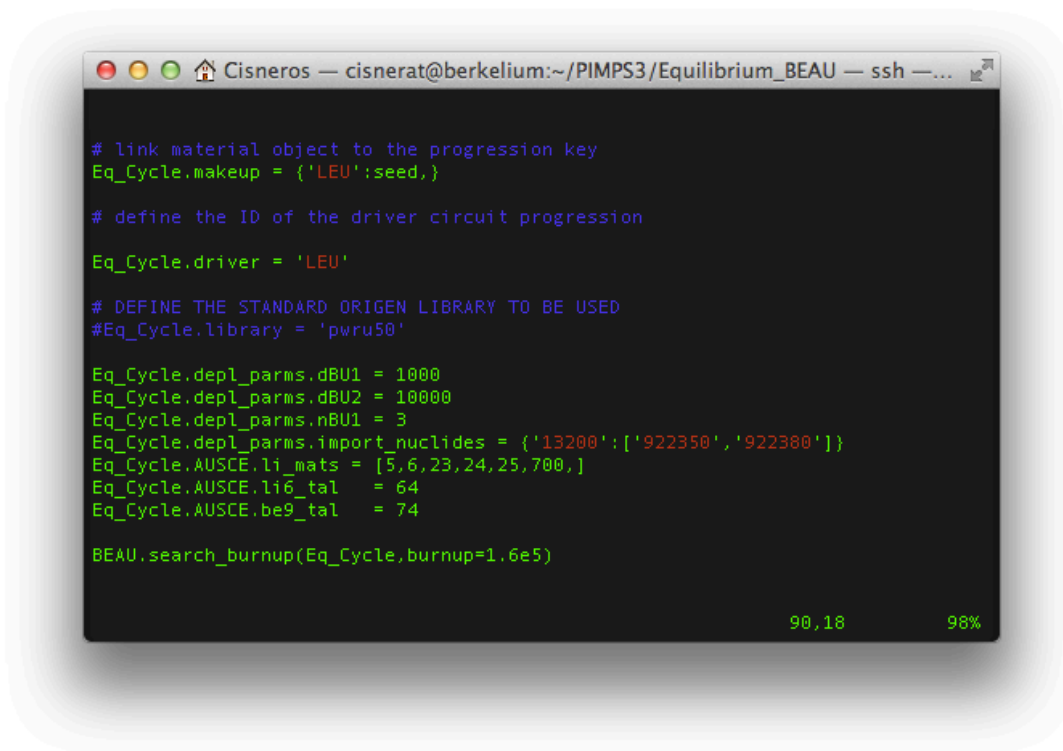
# DEFINE THE STANDARD ORIGEN LIBRARY TO BE USED
#Eq_Cycle.library = 'pwr50'

Eq_Cycle.depl_parms.dBU1 = 1000
```

64,0-1 79%

Figure 2-19. Equilibrium fuel cycle definition (3): definition of makeup fuel composition and identification of driver circulation progression.

The user can change any of the depletion parameters in the <equilibrium fuel cycle>.depl_parms depletion parameters object as shown in Figure 2-20. It is important to define the important nuclides to the <equilibrium fuel cycle>.depl_parms.import_nuclides dictionary as shown in Figure 2-20. This dictionary maps burnup states to a list of isotopes that are explicitly checked for convergence in the equilibrium composition search loop. The burnup states are identified by their cell titles and the isotopes are defined in ZZAAAM format. The user may define as many isotopes in as many burnup states as they want, however the list of isotopes defined in each burnup state are defined independently.



```
Cisneros — cisnerat@berkelium:~/PIMPS3/Equilibrium_BEAU — ssh — ...

# link material object to the progression key
Eq_Cycle.makeup = {'LEU':seed,}

# define the ID of the driver circuit progression
Eq_Cycle.driver = 'LEU'

# DEFINE THE STANDARD ORIGEN LIBRARY TO BE USED
#Eq_Cycle.library = 'pwru50'

Eq_Cycle.depl_parms.dBU1 = 1000
Eq_Cycle.depl_parms.dBU2 = 10000
Eq_Cycle.depl_parms.nBU1 = 3
Eq_Cycle.depl_parms.import_nuclides = {'13200':['922350','922380']}
Eq_Cycle.AUSCE.li_mats = [5,6,23,24,25,700,]
Eq_Cycle.AUSCE.li6_tal = 64
Eq_Cycle.AUSCE.be9_tal = 74

BEAU.search_burnup(Eq_Cycle,burnup=1.6e5)

90,18 98%
```

Figure 2-20. Equilibrium fuel cycle definition (4): depletion parameters, AUSCE variables and initiation of maximum burnup equilibrium depletion analysis search.

The user can also update parameters used in AUSCE for the equilibrium salt calculation. The `li_mats` lists defines which materials are updated by AUSCE. The `li6_tal` and `be9_tal` identify the tallies used to calculate the ${}^6\text{Li}(n,\text{abs})$ and ${}^9\text{Be}(n,\alpha)$, respectively.

Finally, the command `BEAU.search_burnup` initiates the maximum burnup equilibrium depletion analysis search for the defined equilibrium fuel cycle object. The user can define an initial estimate of the driver burnup; this burnup value is perturbed to develop a linear regression for the system multiplication factor as a function driver burnup.

2.4.1.3.3 BEAU Output Files

The equilibrium depletion analysis results from BEAU are presented in a formal output file, `output.py`, status update files in the `scratch` directory, ORIGEN output files in the `moi_files` directory, and intermediate neutron transport input and output files in the `backup` directory.

2.4.1.3.3.1 *output.py*

The `output.py` file is formal output file produced by BEAU. It is organized into four sections: fuel cycle description, the neutron transport input deck, the depletion parameters, and the fuel cycle search results. `Output.py` is a python file so it can be imported by external python scripts. `Output.py` is heavily documented with comments so the user can clearly understand each of the outputs.

2.4.1.3.3.2 *scratch directory*

The scratch directory presents the current status of the equilibrium depletion analysis search. Each of the search modules has two corresponding output files a condensed version that has a “.csv” ending and a more verbose output with results from each of the lower order search modules. The output files presented in Table 2-11.

Table 2-11. Scratch directory output files

Search Modules	Condensed Output	Verbose Output
search_cycle_length	cycle_length.csv	cycle_length.o
search_source	source.csv	source.o
search_comp	comp.csv	comp.o
search_burnup	burnup.csv	burnup.o

2.4.1.3.3.3 *moi_files directory*

The ORIGEN point depletion analysis results are recorded in the moi_files directory. For every depletion calculation, three types of files are recorded: punch files (.pch), ORIGEN input files (.inp) and ORIGEN output files (.out). The isotopic evolution for each burnup state is recorded in the ORIGEN punch files. Each ORIGEN punch contains the isotopic composition of the burnup state for the initial composition (moi.<burnup state>.0.pch) to the EOS of each timestep (moi.<burnup state>.i.pch). The ORIGEN input file(s) is recorded in the file titled moi.<burnup state>.inp. If multiple ORIGEN executions are required both ORIGEN input files are recorded sequentially in the same file. The ORIGEN output files(s) is recorded in the file titled moi.<burnup state>.out.

2.4.1.3.3.4 *backup directory*

The backup directory records each iteration of the equilibrium state’s neutron transport input deck and resulting output files in files labeled inp<iteration number> and outp<iteration number> respectively.

2.5 Benchmarking of MBSM

2.5.1 Full-core Analysis

Most of the knowledge base on continuously refueled pebble bed depletion analysis is related to the pebble bed high temperature gas reactors such as the Pebble bed Modular Reactor, PBMR. As stated earlier, INL has developed a deterministic code, PEBBED, to perform this kind of equilibrium analysis for the NGNP program and benchmarked it against VSOP developed by Julich Research Center for the German HTR program in the Paper: “A COMPARISON OF PEBBLE MIXING AND DEPLETION ALGORITHMS USED IN PEBBLE-BED REACTOR EQUILIBRIUM CYCLE SIMULATION” (Gougar, Reitsma, & Joubert, A Comparison of Pebble Mixing and Depletion Algorithms Used in Pebble-Bed Reactor Equilibrium Cycle Simulation, 2009). This sub-section will present the definition of a simple HTGR PBR benchmark and results of a study to reproduce the critical recirculation rates, $k_{\text{effective}}$ and discharge fuel composition at the predicted critical recirculation rates in this paper with the MBSM implemented by BEAU.

2.5.1.1 Benchmark Description

This benchmark core is a simplified version of the PBMR. It is a 10 m tall 3 m in diameter helium cooled high temperature reactor subdivided into 5 radial channels and 10 axial nodes. The fuel design is based on the fuel described in IAEA CRP benchmark model of the PBMR400 equilibrium core. Both VSOP and PEBBED used the same set of nuclear data generated for the PBMR400 equilibrium core so the temperatures taken from this benchmark were used in the Monte Carlo neutron transport model. The assumed dimensions of this benchmark are presented in Table 2-12 - Table 2-15.

Table 2-12. Benchmark TRISO Particle description

TRISO Particle Description	
Fuel Kernel Radius (μm)	250
Buffer thickness (μm)	95
iPyC thickness (μm)	40
SiC thickness (μm)	35
oPyC thickness (μm)	40
*Particle Packing Fraction (%)	9.34
Enrichment (w%)	9.6
Kernel material / density (g/cm^3)	UO ₂ / 10.4
Buffer material / density (g/cm^3)	porous graphite / 1.05
iPyC material / density (g/cm^3)	pyro graphite / 1.9
SiC material / density (g/cm^3)	silicon carbide / 3.18
oPyC material / density (g/cm^3)	pyro graphite / 1.9
Matrix material / density (g/cm^3)	graphite / 1.74
Kernel Temperature (°C)	876.08
**TRISO layers + matrix Temperature (°C)	857.79

* Packing Fraction was calculated to impose the 9g/Pebble Fuel Loading description

** TRISO layers + matrix temperature values was assumed to be the average temperature in the active region of the pebble.

Table 2-13. Benchmark fuel pebble description

Fuel Pebble Description	
Active core radius (cm)	2.5
Inert shell thickness (cm)	0.5
Pebble Packing Fraction (%)	61
Shell material / density (g/cm^3)	graphite / 1.74
Shell temperature (°C)	857.79
Average Helium temperature (°C)	766.74

Table 2-14. Benchmark full core description

Full Core Descriptions	
Power (MWt)	200
Active Radius (cm)	150
Active Height (cm)	1000
Thickness of Radial Reflectors (cm)	100
Thickness of Axial Reflectors (cm)	100
Reflector material / density (g/cm ³)	graphite / 1.74
Upper Reflector Temperature (°C)	488.1
Radial Reflector Temperature (°C)	766.74
Lower Reflector Temperature (°C)	997.3

Table 2-15. Benchmark radial channel boundaries

Radial Channel	1	2	3	4	5
Outer Radius (cm)	60	90	120	135	150

This benchmark description was used to develop a full-core model in MCNP5 for BEAU. The fuel kernels were modeled explicitly on a simple cubic lattice. However, the TRISO layers were homogenized together by volume with the graphite matrix to accelerate the neutron transport. The pebbles were modeled in repeated unit cells that fill each R,Z node. The number of burnup states in each unit cell correspond to the number of passes through the core so that there will be a corresponding burnup state for each (R,Z, pass) combination. Multi-pass fuel cycles with both six and seven passes were analyzed as part of the test matrix*.

The pebble unit cell for the seven pass fuel cycles utilizes a face-centered cubic configuration with the sites at lower corners vacant and the spacing between the pebbles increased to induce the 61% pebble packing fraction. This configuration has seven pebble halves that can be filled with coated particles that correspond to different burnup states. Thereby, this unit cell can explicitly model a burnup state for each pass through the reactor when it performs multi-pass depletion analysis. This seven-burnup state unit cell configuration is presented in Figure 2-21.

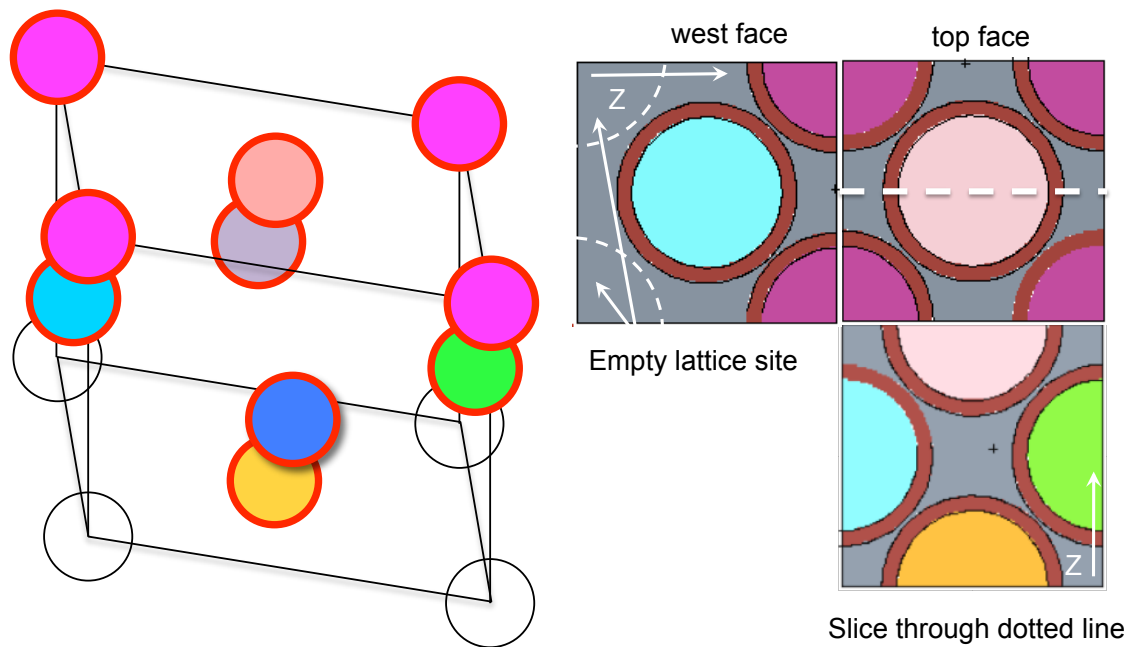


Figure 2-21. Pebble unit cell configuration – note each color (magenta, pink, lavender, aqua, blue, yellow and green) represents TRISO particles and matrix at different burnup states; (left) 3-dimensional sketch of seven burnup state configuration; (right) MCNP5 plot of slices through the seven burnup state pebble unit cell.

The pebble unit cell for a six pass and the once through fuel cycle utilizes a hexagonal close pack unit cell with the spacing between the pebbles increased to impose a pebble packing fraction of 61%. This hexagonal close packed unit cell is presented in Figure 2-22.

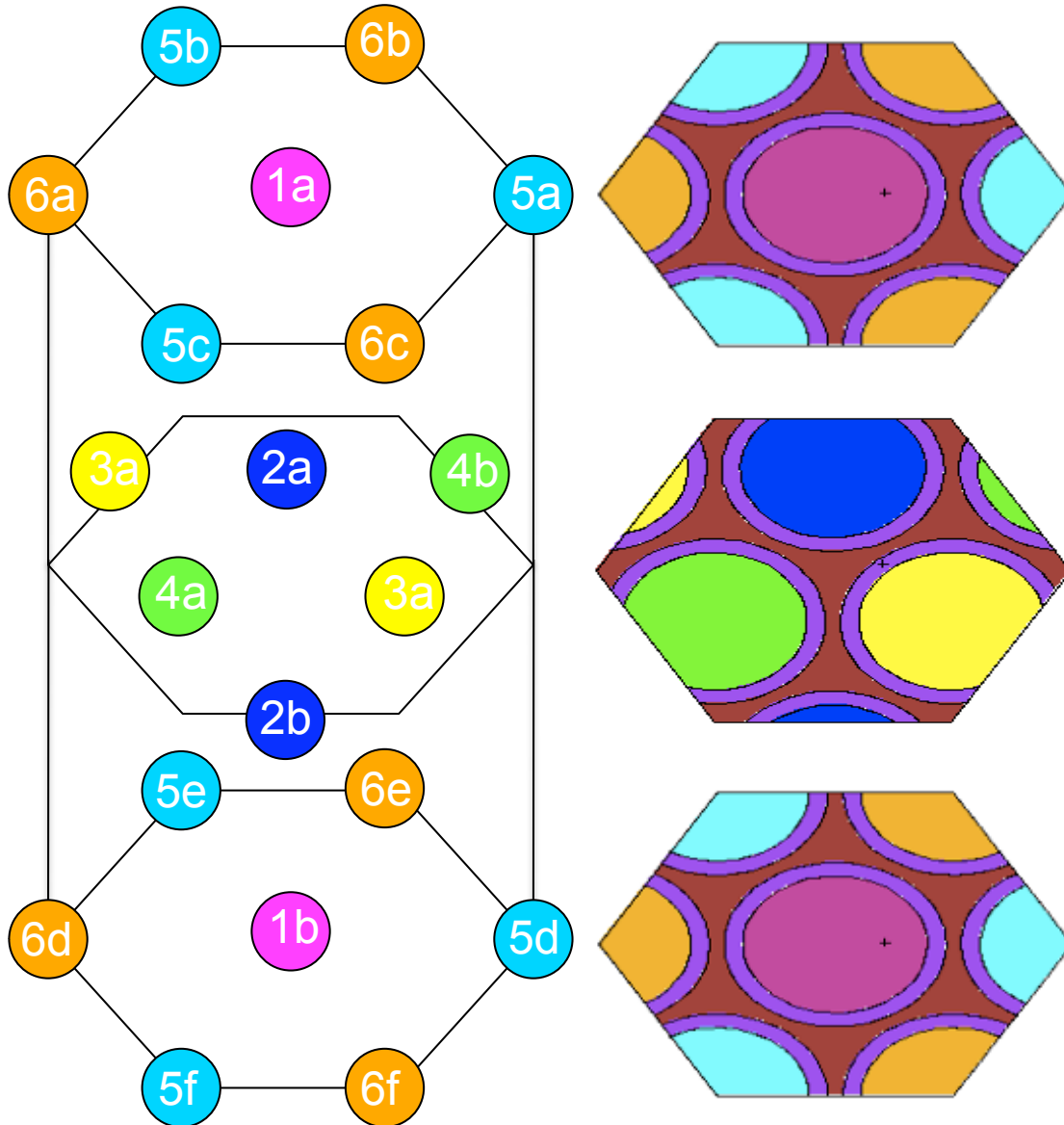


Figure 2-22. Pebble unit cell configuration – note each color (magenta, blue, yellow, green, aqua and orange) represents TRISO particles and matrix at different burnup states; (left) 3-dimensional sketch of HCP configuration; (right) MCNP5 plot of slices through the HCP pebble unit cell.

The pebble bed in the MCNP model is divided into 5 radial nodes and 10 axial nodes – 50 (R,Z nodes) each with several burnup states – according the benchmark description. Figure 2-23 presents the geometry of the MCNP5 model.

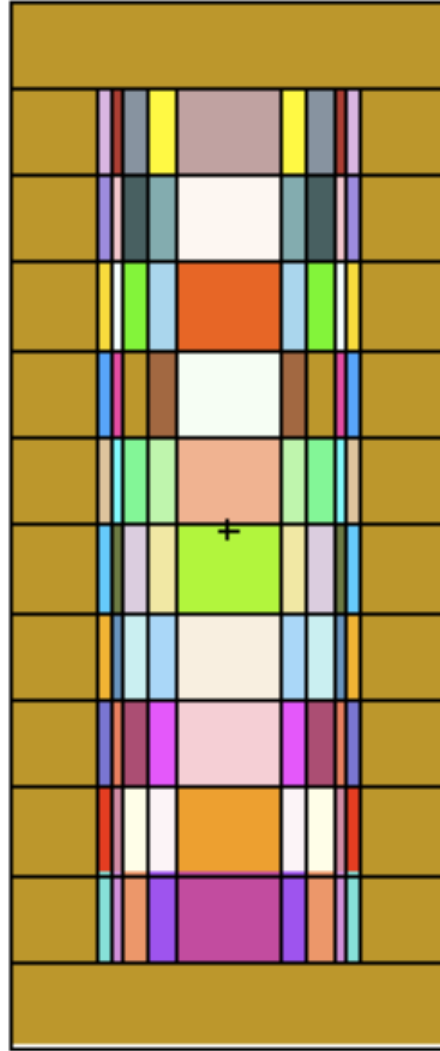


Figure 2-23. MCNP5 plot of the fullcore geometry – note that the fine detail of the pebble geometry have been homogenized to focus on the radial and axial meshing.

BEAU was utilized to model the equilibrium fuel cycle of the simplified PBMR. The target power (`<equilibrium fuel cycle object>.target_power`) of the system was set to 200 MWt, so the fission neutron source was solved to impose this target power. The radial pebble flow distribution was set to a flat distribution in the depletion benchmark, so the relative pebble flow rate in BEAU (`<equilibrium fuel cycle object>.pebble_flow`) was set to a flat distribution as well. The fuel cycle for the multipass case equilibrium depletion calculation consists of a single circulation progression, with either 6 or 7 circulations. Each of these circulations consists of 5 progressions that correspond to each of the radial channels. These progressions each have ten burnup states that correspond to each of the axial nodes. The fuel cycle for the once through fuel cycle cases only has a single circulation subdivided into progressions and burnup states as in the multipass cases. The BOC (i.e. makeup) composition is presented in Table 2-16.

Table 2-16. Beginning of cycle composition for benchmark HTGR

isotope	concentration in kernel (mol/cm ³)	concentration in kernel (at/bn-cm)	concentration in RZ node (at/bn-cm)
¹⁶ O	7.71*10 ⁻²	4.64*10 ⁻²	2.46*10 ⁻⁴
²³⁴ U	3.66*10 ⁻⁵	2.21*10 ⁻⁵	1.17*10 ⁻⁷
²³⁵ U	3.74*10 ⁻³	2.26*10 ⁻³	1.19*10 ⁻⁵
²³⁸ U	3.48*10 ⁻²	2.09*10 ⁻²	1.11*10 ⁻⁴

Since the discharge fuel composition is used as a metric of comparison between BEAU, VSOP and PEBBED it is essential that this composition is converged, therefore the concentrations of ²³⁵U, ²³⁸U, ²³⁹Pu, and ²⁴¹Pu are checked explicitly for convergence in the last burnup state before discharge – this is the closest value to the discharge fuel (the reported value from literature) that will be used to compare results.

BEAU utilizes depletion cross sections and flux distributions based on tallies in the MCNP5 model of the latest estimate of the equilibrium state of the core rather than using the same nuclear data as the deterministic references.

A test matrix was set up to benchmark BEAU against both VSOP and PEBBED for continuously refueled equilibrium depletion analysis in two modes of operation: once through then out (OTTO) and multi-pass depletion analysis.

Table 2-17. Test matrix for equilibrium depletion benchmarking

Case	Fuel Cycle	Test	Metric
O1	OTTO	Search for burnup to impose $k_{\text{effective}}=1$ for OTTO fuel cycle	Burnup (ie critical recirculation rate)
O2a	OTTO	Simulate Critical Recirculation Rate Predicted by VSOP (214 pebbles/day) for OTTO fuel cycle	Equilibrium $k_{\text{effective}}$ / Discharge Composition
O2b	OTTO	Simulate Critical Recirculation Rate Predicted by PEBBED (234 pebbles/day) for OTTO fuel cycle	Equilibrium $k_{\text{effective}}$ / Discharge Composition
M1a	7-Pass	Search for burnup to impose $k_{\text{effective}}=1$ for 7-pass fuel cycle (to compare to VSOP)	Burnup (ie critical recirculation rate)
M1b	6-Pass	Search for burnup to impose $k_{\text{effective}}=1$ for 7-pass fuel cycle (to compare to PEBBED)	Burnup (ie critical recirculation rate)
M2a	7-Pass	Simulate Critical Recirculation Rate Predicted by VSOP (1448 pebbles/day) for multi-pass fuel cycle	Equilibrium $k_{\text{effective}}$ / Discharge Composition
M2b	6-Pass	Simulate critical recirculation rate predicted by PEBBED (1280 pebbles/day) for multi-pass fuel cycle	Equilibrium $k_{\text{effective}}$ / Discharge Composition

2.5.1.2 Results Multiple Burnup State Method

BEAU returns the critical recirculation rate in terms of burnup rather than a recirculation rate. Therefore, this critical burnup is transformed to the critical recirculation rate according using Equation 2-32.

$$f_{core} = \frac{P_{core} \bar{m}_{passes}}{BU * M_o^{actinide}} \bar{n}_{pebbles}$$

Equation 2-32

Additionally, the discharge composition from BEAU is taken from the ORIGEN composition files in terms of molar mass of the actinides and fission products in the fuel kernels in a given burnup state (R,Z,B), whereas the discharge composition from VSOP and PEBBED are given in terms of atoms/bn-cm normalized to the volume of the pebbles and the coolant. Therefore, the discharge from each burnup state is transformed to the nodal composition according to the Equation 2-32.

$$\bar{N}_{node} = \frac{\bar{m}_{passes}}{V_{node}} \bar{N}_{kernel}$$

Equation 2-33

Three benchmark tests were performed to compare results for an OTTO fuel cycle: Case O1 where BEAU searches for the burnup it estimates will result in a critical core, Case O2a where BEAU tries to reproduce the results from VSOP at its predicted equilibrium burnup and Case O2b where BEAU tries to reproduce the results from PEBBED at its predicted equilibrium burnup.

2.5.1.2.1 Once Through Then Out (OTTO) Benchmark Results

The results for test O1 are presented in Table 2-18. BEAU predicts a similar burnup to PEBBED's results.

Table 2-18. Results comparison for test O1: search for critical recirculation rate for OTTO fuel cycle

Test O1: Search for Critical Recirculation Rate for OTTO fuel cycle			
Parameter	BEAU Value	VSOP Value	PEBBED Value
Equilibrium $k_{effective}$	1.001 ± .001	1	1
Burnup (MWd/MT)	94080 ± 500	104000	95000
Passes	1	1	1
Critical recirculation rate (pebble/ day)	236 ± 1	214	234

The results for test O2a are presented in Table 2-19. BEAU estimates a lower $k_{effective}$ for the critical recirculation rate, showing the same bias as the search for the critical recirculation results. The fissile isotope contents are significantly lower for BEAU than VSOP. Additionally, the nodal concentrations are within the same order of magnitude for the most important isotopes: ^{235}U , ^{238}U , ^{239}Pu and ^{241}Pu .

Table 2-19. Results comparison for test O2a: reproduce results from VSOP for OTTO fuel cycle

Case O2a: Reproduce VSOP OTTO critical fuel cycle results		
Parameter	BEAU Value	VSOP Value
Equilibrium $k_{\text{effective}}$	0.978±0.001	1
Burnup (MWd/MT)	104000±500	104000
Passes	1	1
Critical recirculation rate (pebble/ day)	214	214
Nodal concentration U-235 (at/bn-cm)	1.94E-06	2.21E-06
Nodal concentration U-238 (at/bn-cm)	1.04E-04	1.04E-04
Nodal concentration Pu-239 (at/bn-cm)	6.72E-07	7.01E-07
Nodal concentration Pu-241 (at/bn-cm)	2.95E-07	3.01E-07
Nodal concentration Am-244 (at/bn-cm)	4.23E-15	1.93E-14
Nodal concentration Xe-135 (at/bn-cm)	3.27E-13	2.14E-13
Nodal concentration Gd-155 (at/bn-cm)	8.29E-10	2.03E-09

The results for test O2b are presented in Table 2-20. The equilibrium $k_{\text{effective}}$ predicted in BEAU agrees much better with PEBBED than VSOP. Additionally, the nodal concentrations agree much better for the most important concentrations.

Table 2-20. Results comparison for test O2b: reproduce results from PEBBED for OTTO fuel cycle

Case O2b: Reproduce PEBBED OTTO critical fuel cycle results		
Parameter	BEAU Value	PEBBED Value
Equilibrium $k_{\text{effective}}$	0.998±0.001	1
Burnup (MWd/MT)	95000±500	95000
Passes	1	1
Critical recirculation rate (pebble/ day)	234	234
Nodal concentration U-235 (at/bn-cm)	2.42E-06	2.32E-06
Nodal concentration U-238 (at/bn-cm)	1.05E-04	1.05E-04
Nodal concentration Pu-239 (at/bn-cm)	7.03E-07	6.90E-07
Nodal concentration Pu-241 (at/bn-cm)	3.00E-07	2.99E-07
Nodal concentration Am-244 (at/bn-cm)	2.92E-15	1.03E-14
Nodal concentration Xe-135 (at/bn-cm)	3.64E-13	1.43E-13
Nodal concentration Gd-155 (at/bn-cm)	6.86E-10	2.25E-09

2.5.1.2.2 Multi-Pass Fuel Cycle (MEDUL) Benchmark Results

BEAU was also benchmarked against VSOP and PEBBED for multi-pass fuel cycles. The benchmark HTGR multi-batch fuel cycle presented benchmark paper is defined as a 6-batch fuel cycle. However, there is a significant discrepancy between the burnup algebraically predicted by the given benchmark parameters given in Table 2-14 for VSOP's estimated critical recirculation rate 1448 pebbles/day with Equation 2-31. The reported burnup of 1.09×10^5 MWd/MT at the reported critical recirculation rate at the

benchmarks power and fuel loading can be imposed if the pebbles were recirculated 7 times. Figure 2-24 presents the estimated burnup (using Equation 2-31) for the estimated recirculation rate estimated by VSOP (1448 pebbles / day) and PEBBED (1280 pebbles per day). One can see that the estimated burnup for 1448 pebbles per day is closer to the reported critical burnup for 7 passes than the reported 6 passes and that the estimated burnup for 1280 pebbles per day is close to this reported discharge burnup for 6 passes. Therefore, when benchmarking against VSOP for the multi-pass fuel cycle results as 7-pass fuel cycle will be assumed rather than the 6-pass fuel cycle stated in the paper. This discrepancy in the passes in the VSOP definition is independent of any depletion analysis results and explains the shift of the bias from under predicting the critical pebble recirculation rate for the OTTO fuel cycle to over predicting the recirculation for the multi-pass tests.

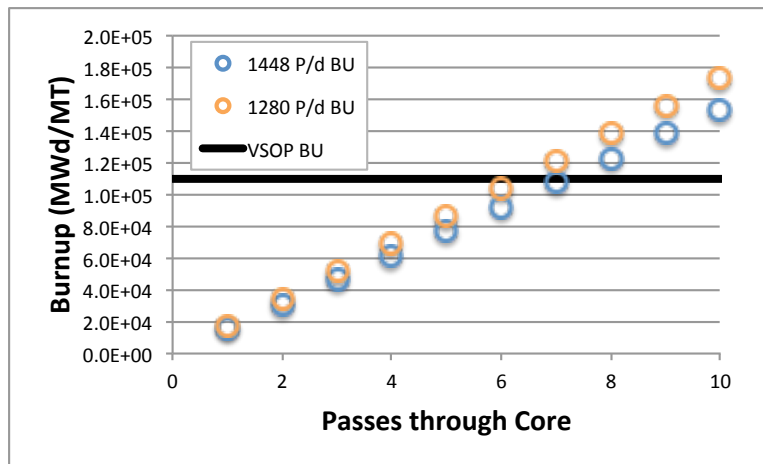


Figure 2-24. Burnup as a function of the number of passes through the core for the benchmark HTGR for various recirculation rates.

In test M1a, BEAU estimates the critical recirculation rate for a 7-pass fuel cycle variant of the benchmark HTGR. The results for test M1a are presented in Table 2-21. BEAU predicts a similar burnup to VSOP’s results.

Table 2-21. Results comparison for test M1a: search for critical recirculation rate for seven-pass fuel cycle

Case M1a: Search for critical recirculation rate for seven pass fuel cycle		
Parameter	BEAU Value	VSOP Value
Equilibrium $k_{\text{effective}}$	0.99907 ± .001	1
Burnup (MWd/MT)	106322 ± 500	107400
Passes	7	7
Critical recirculation rate (pebble/ day)	1464±7	1448

In test M1b, BEAU tries to estimate the critical recirculation rate for a 6-pass fuel cycle variant of the benchmark HTGR. The results for test M1b are presented in Table 2-21. BEAU predicts a similar burnup to PEBBED’s results.

Table 2-22. Results comparison for test M1a: search for critical recirculation rate for six-pass fuel cycle

Case M1b: Search for critical recirculation rate for six pass fuel cycle		
Parameter	BEAU Value	PEBBED Value
Equilibrium $k_{\text{effective}}$	1.0013 ± .001	1
Burnup (MWd/MT)	105659 ± 500	104200
Passes	6	6
Critical recirculation rate (pebble/ day)	1262±6	1280

In Test M2a, BEAU estimates the $k_{\text{effective}}$ for the critical recirculation rate predicted by VSOP for a 7-pass fuel cycle. The results for test M2a are presented in Table 2-23. The fissile isotope contents are slightly lower for BEAU than VSOP, consistent with the bias in the OTTO depletion results, though the nodal concentrations are within the same order of magnitude for the most important isotopes.

Table 2-23. Results comparison for test M2a: reproduce results from VSOP for six-pass fuel cycle

Case M2a: Reproduce VSOP seven pass critical fuel cycle results		
Parameter	BEAU Value	VSOP Value
Equilibrium $k_{\text{effective}}$	0.9958±.001	1
Burnup (MWd/MT)	107400±500	107400
Passes	7	7
Critical recirculation rate (pebble/ day)	1448	1448
Nodal concentration U-235 (at/bn-cm)	1.77E-06	1.64E-06
Nodal concentration U-238 (at/bn-cm)	1.04E-04	1.03E-04
Nodal concentration Pu-239 (at/bn-cm)	7.40E-07	7.50E-07
Nodal concentration Pu-241 (at/bn-cm)	3.60E-07	3.65E-07
Nodal concentration Am-244 (at/bn-cm)	7.52E-13	6.23E-12
Nodal concentration Xe-135 (at/bn-cm)	3.80E-11	2.93E-11
Nodal concentration Gd-155 (at/bn-cm)	8.39E-11	4.44E-10

In Test M2b, BEAU estimates a $k_{\text{effective}}$ for the critical recirculation rate predicted by PEBBED for a 6-pass fuel cycle. The results for test M2a are presented in Table 2-24. The concentrations estimated with BEAU are similar to the values reported from PEBBED for the most important isotopes.

Table 2-24. Results comparison for test M2b: reproduce results from PEBBED for six-pass fuel cycle

Case M2b: Reproduce PEBBED six pass critical fuel cycle results		
Parameter	BEAU Value	PEBBED Value
Equilibrium $k_{\text{effective}}$	1.0045±.001	1
Burnup (MWd/MT)	104200±500	104000
Passes	1	1
Critical recirculation rate (pebble/ day)	1280	1280
Nodal concentration U-235 (at/bn-cm)	1.91E-06	1.67E-06
Nodal concentration U-238 (at/bn-cm)	1.04E-04	1.04E-04
Nodal concentration Pu-239 (at/bn-cm)	7.43E-07	7.23E-07
Nodal concentration Pu-241 (at/bn-cm)	3.56E-07	3.33E-07
Nodal concentration Am-244 (at/bn-cm)	5.08E-13	4.17E-12
Nodal concentration Xe-135 (at/bn-cm)	3.22E-11	2.83E-11
Nodal concentration Gd-155 (at/bn-cm)	9.51E-11	3.42E-10

2.6 Conclusions

The MBSM methodology was developed as a new methodology to perform equilibrium depletion analysis for unit cell and full-core models of PBRs. BEAU was developed to implement MBSM and serve as a design and analysis tool for PBR neutronics and depletion analysis. BEAU was benchmarked against deterministic equilibrium depletion analysis codes and exhibited reasonable agreement on burnup, criticality and the isotopic concentration of important isotopes such as ^{235}U , ^{238}U and ^{239}Pu at discharge but significant disagreement for Am-244 and Gd-155.

3 PB-FHR Parametric Studies

3.1 Introduction

The goal of this manuscript is to develop a reactor core design for an Annular PBR FHR. A large number of design parameters are required to define a nuclear reactor core design, which translates into a vast design space within which the optimum core design is located.

Massively Parallel Pre-computation (MPP) is a design methodology in which wide array of possible systems, covering the design space of interest along several chosen “dimensions”, are investigated. These dimensions can be attributes such as fuel to moderator ratio, kernel thickness, blanket fuel burnup, radial thickness of the fueled zone in the active region of the pebble bed, radial thickness of the pebble reflector zone in the active region of the pebble bed, power density, etc. The MPP design sequence first performs detailed analysis to calculate several performance metrics (burnup, temperature reactivity coefficients, peak power and dose, etc.) for several design points throughout the design space. These results are used to develop correlations to estimate the performance metrics of new design points within the explored design space. These correlations can be integrated to develop a surrogate model with the functionality to assess the multi-objective fitness of design points to perform design optimization. Elements of the MPP were used throughout these parametric studies, but fully implemented into the pebble bed core design parametric study (Section 3.3.3 & Chapter 4).

Since the design space increases exponentially with the number of design dimensions, a modular approach was used to transform the many-dimensional Annular FHR PBR design space into three tractable few-dimensional design studies: a pebble fuel design study, a core configuration design study, and a pebble bed geometry design space.

This chapter implements the analysis methodology from Chapter 2 to perform the parametric studies for the reactor design of the PB-FHR. First the neutronics models and the frameworks developed to generate these models for each of the parametric studies are presented. Then each of results of the three modules of the Annular FHR PBR design study – pebble fuel design, core configuration and the pebble bed design – are presented and discussed. The pebble fuel design and core configuration design studies lead directly to baseline fuel designs and core configurations that were adopted for subsequent studies. The results, physics and analysis of the parametric studies for the PB-FHR pebble bed core design are presented in this chapter, but the pebble bed core multi-objective safety and economics optimization is very complicated and discussed separately in Chapter 4.

3.2 FHR Input-deck Maker for Parametric Studies (FIMPS)

FHR Input-deck Maker for Parametric Studies (FIMPS) was developed to repeatably- and easily create self-consistent input decks for neutronics analysis. Creating detailed neutronics input decks by hand is an arduous and error-prone process. Many attributes

of a model are interrelated, so changes to a single attribute result in multiple changes, creating opportunities for human errors. For example, changing the kernel diameter of a TRISO particle results in changing all the dimensions of the TRISO layers (usually defined by their thicknesses), the pitch of these (if packing fraction or alternatively carbon to heavy metal ratio (C/HM) is held constant), and increasing the kernel diameter changes the characteristic temperatures in the fuel kernel and the TRISO layers by changing the combination of heat flux out of the kernel, average power density and heat transfer resistance of the TRISO components. Therefore, FIMPS was created to transition from generating input decks manually to describing the geometry to FIMPS and empowering FIMPS to execute an established, robust methodology to define a consistent set of input data for a specific instances of a PB-FHR model.

FIMPS conceptualizes repeatable PB-FHR models as python objects with attributes and methods. Consider a few PB-FHR model objects: fuel pebble (pebble), a homogeneous pebble bed model of a 900 MWt annular pebble bed fluoride salt cooled advanced high temperature reactor (PB-AHTR) or a heterogeneous model of a small modular pebble bed fluoride salt cooled high temperature reactor (MPP PB-FHR). Each of these objects has a various fundamental attributes. For example, a pebble has attributes like shell radius, material of the pebble shell, temperature of the pebble shell, etc. Additionally, there are calculations that will be frequently performed for a given class of modeling object; these calculations are transformed into methods so they are executed easily and repeatably. Again using a pebble object as an example, they will have methods such as calculating the C/HM, calculating the average density of a pebble and iteratively calculating the C/HM and average density while perturbing the geometry attributes of the pebble in concert to impose a target combination of C/HM and average density.

Object oriented programming is an efficient way to develop neutronics models. In the parametric studies presented in this chapter, a single structure of an MCNP5 input deck for different instances of a model is recycled, changing only the values of the surface cards, temperature cards and/or material compositions – this skeleton input deck can be thought of another attribute of a model object. High-level overviews of the modeling objects utilized in these parametric studies are presented in the following subsections.

3.2.1 TRISO

The TRISO object has all the attributes and methods required to represent a PB-FHR TRISO particle. This object does not produce a model but is a fundamental component of other model objects.

3.2.1.1 Engineering Parameters

The TRISO particle object's geometry is defined in terms of radial thicknesses and the composition of its constituents are defined in terms of mocup.py material objects. In higher-level FIMPS models TRISO particles are modeled on simple cubic arrays, but initially these particles are defined in a spherical unit cell model. Typically, the extent of the matrix in these TRISO particles are defined in terms of packing fraction so the TRISO

object has methods to transform the radial dimensions of the matrix to a packing fraction and to define these radial dimensions in terms of a TRISO packing fraction.

3.2.1.2 Heat Transfer Model

FIMPS uses a unit cell heat transfer model based on the analytical approach presented by Stainsby, et al (Stainsby, Worsley, Greif, Dawson, Coddington, & Baker, 2010). The heat transfer model is decomposed into two scales: Pebble-scale and TRISO-scale. In this methodology the power distributions on these scales are decomposed so that they add up to the true distribution. The power distribution on the pebble scale is assumed to be uniform over the active region of the pebble and heterogeneous on the TRISO scale – positive in fuel kernel and negative in the TRISO layers and graphite matrix; as shown in Figure 3-1. Each component of the TRISO particle is assumed to have constant power density and constant thermal conductivity such that the heat conduction equation can be solved analytically for each component such that the temperature profile is continuous and there is zero stored energy in the TRISO scale – the heat transfer effects the temperature difference between components and the zero stored energy conditions determines these TRISO scale temperature relative to the pebble scale temperatures. Equation 3-1 determines the temperature distribution in the fuel kernel relative to the maximum temperature, T_0 , assuming no heat flux at the center of the kernel. Equation 3-2 determines the temperature distribution in an arbitrary layer of the TRISO particles with inner radius, r_i , and outer radius, r_{i+1} , and constant power flowing out of the inner and outer surfaces. This temperature profile can be superimposed onto the pebble-scale temperature distribution to estimate the actual temperature profile.

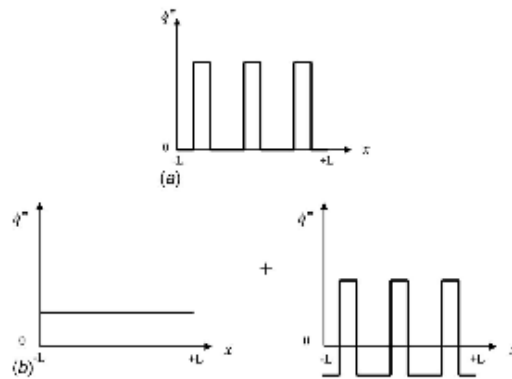


Figure 3-1. Power distribution in the FIMPS heat transfer model: (a) true power distribution; (b) decomposed power distribution one the pebble scale in the active region (left) and TRISO scale (right).

$$T(r_{kernel}) = T_0 - \frac{q}{6k}(r_{kernel})^2$$

Equation 3-1

$$T(r_{layer}) = T_i - \frac{qr_{i+1}^2}{6k} * \left(1 + \frac{(r_{layer})^2}{r_{i+1}^2} \right) + \frac{q}{3k}r_i(r_{layer}) + \frac{dT}{dr} \Big|_{r=r_i} ((r_{layer}) - r_i)$$

Equation 3-2

3.2.2 Pebble

An infinite pebble bed model is ideal for developing a foundation of understanding for the PB-FHR system. The pebble model was used to assess the implication of pebble fuel design choices independent of the pebble bed design. This model was utilized to perform the fuel design study.

3.2.2.1 Engineering Parameters

The annular pebble geometry used in the PB-FHR is defined by the radial dimensions of its layers – pebble core, active region, shell – and the composition of its constituents are defined in terms of mocup.py material objects for the pebble core and shell or a TRISO object for the active region. Usually, the pebble diameter is a fixed parameter so the pebble is defined in terms of the radius of the pebble, radial thickness of the shell and radial thickness of the pebble core. However, a more reactor physics based way to define pebbles is based on their C/HMs and their average densities. Equation 3-3 presents the calculation for C/HM in FIMPS, where N^C and N^{HM} are the homogenized carbon- and the actinide atomic densities respectively. Equation 3-4 presents the calculation for average density in FIMPS, where ρ is density.

$$C/HM = \frac{N_{shell}^C V_{shell} + N_{active}^C V_{active} + N_{core}^C V_{core}}{N_{active}^{HM} V_{active}}$$

Equation 3-3

$$\langle \rho \rangle = \rho_{shell} \frac{V_{shell}}{V_{pebble}} + \rho_{active} \frac{V_{active}}{V_{pebble}} + \rho_{core} \frac{V_{core}}{V_{pebble}}$$

Equation 3-4

The pebble object has methods to develop a pebble fuel designs that impose a set of C/HM and average density values by changing the core radius, porosity of the pebble core and the TRISO packing fraction in a way that minimizes the average temperature of the fuel by minimizing the thickness of the active layer within the pebble. If the target C/HM and density are within the bounds of C/HMs and densities that can be produced with the maximum TRISO packing fraction and the size of the active region is adjusted to impose the target C/HM as shown in Equation 3-5; then the density of the core is adjusted to impose the target density as shown in Equation 3-6. These steps are performed iteratively to generate a pebble design that simultaneously meets the C/HM- and average density criteria (design sequence 1). Alternatively, for high C/HM designs a lower TRISO packing fraction must be used to impose higher C/HMs and target densities. In these cases, the TRISO packing fraction is adjusted iteratively to impose C/HM at the minimum thickness of particles and the density of the pebble core is adjusted to impose the target density, and these two steps are iterated until the design is within the limits of the target C/HM and target density (design sequence 2). For combinations of target C/HM and target density that cannot be designed with either of these design sequences, a third design sequence is executed where the TRISO packing fraction is adjusted to impose the C/HM and the radius of the pebble core is adjusted to impose the target density iteratively until the design is within the target values for C/HM and density (design sequence 3).

Design sequences 1 and 2 produce pebble designs with the minimum average fuel temperature, though design sequence 3 should be close to the optimum because the initial guess of the fuel design is an unconverged pebble design from design sequence 1. The logic flowchart for selecting a design sequence is presented in Figure 3-2 and the three pebble design sequences are presented in Figure 3-3. Only pebble core densities between 0.5 and 1.74 g/cm³ are considered due to feasibility of fabrication. Only a single density of the graphite matrix is considered in the fuel design sequences. In reality, the porosity of the graphite matrix around the TRISO particles should increase (i.e. lower matrix density) with increasing TRISO packing fraction because not as much pressure can be applied because the TRISO particles are more likely to make contact and form stress chains resulting in fuel failure (Morris & Pappano, 2007); therefore, the matrix density corresponding to the maximum TRISO packing fraction should be assumed.

$$\frac{V_{active}}{V_{pebble}} = \frac{N_{shell}^C f + N_{core}^C \left(1 - \frac{V_{shell}}{V_{pebble}}\right)}{\left(C/HM * N_{active}^{HM} + N_{core}^C - N_{active}^C\right)}$$

Equation 3-5

$$\rho_{core} = \frac{\langle \rho \rangle V_{pebble} - \rho_{shell} V_{shell} - \rho_{active} V_{active}}{V_{core}}$$

Equation 3-6

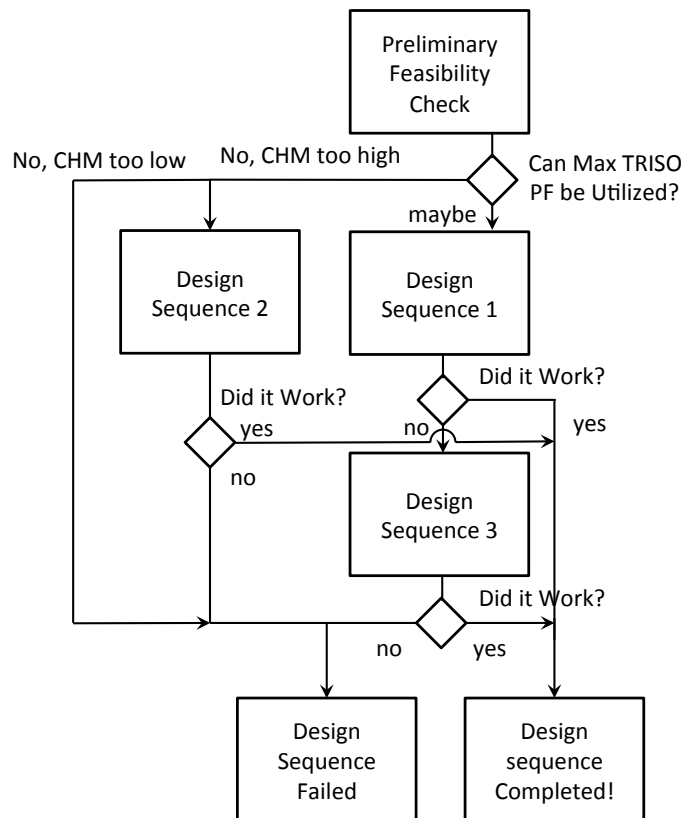


Figure 3-2. Flow chart for pebble design sequence selection

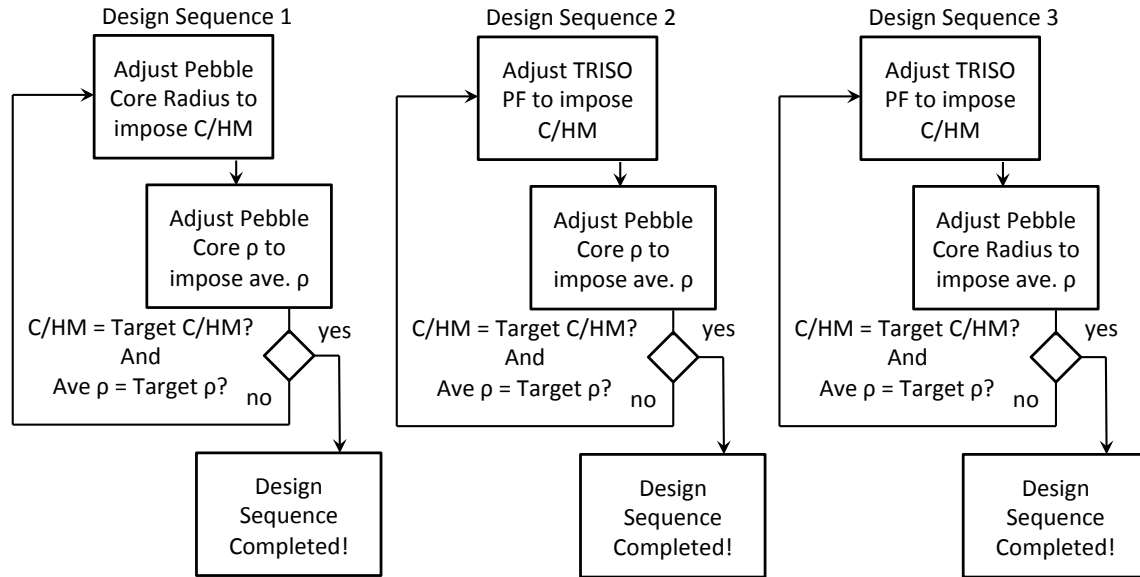


Figure 3-3. Flow chart for pebble design sequences

3.2.2.2 Heat Transfer Model

The temperature distribution in the pebble is based on a 1-D radial pebble unit cell model. Heat is assumed to be generated uniformly in the active region of the pebble. This heat flows through the shell and is convected into the bulk of coolant. The temperature rise from the bulk of the coolant to the outer radial surface of the active region is estimated using a thermal circuit. The Nusselt number for the convection thermal resistance element is estimated using Wakao's modified correlation, as shown in Equation 3-7 (Galvez, Wang, Margossian, Dionisio, & Peterson, 2010). The temperature distribution in the active region derived from the heat equation in spherical coordinates is presented in Equation 3-8. This temperature distribution is averaged by volume to produce the characteristic temperature to be used in the neutronics model.

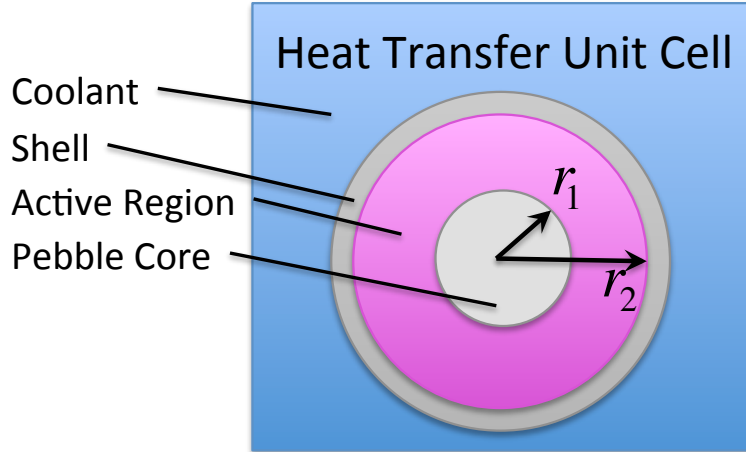


Figure 3-4. Pebble unit cell heat transfer geometry

Equation 3-7
$$Nu = 2 + 1.1Re^{.60}Pr^{.33}$$

$$T(r_{active}) = T_{s,2} + \frac{qr_2^2}{6k} * \left(1 - \frac{(r_{active})^2}{r_2^2}\right) + \frac{q}{3k} \left(\frac{r_1^3}{r_2} - \frac{r_1^3}{(r_{active})}\right)$$

Equation 3-8

3.2.2.3 Neutronics and Depletion Model

The pebble unit cell neutronics and depletion model uses a FCC unit cell with 8 burnup states simulated in MCNP5 as described in Table 2-1. The TRISO particles are modeled as spherical fuel kernels on a simple cubic lattice surrounded by a matrix material that is the volume weighted average of the TRISO layer materials and the graphite matrix; this methodology has been shown to produce results equivalent to models that explicitly model the TRISO particle layers (Fratonia, 2008). There is a lattice of TRISO particles with an independent fuel material composition for each burnup state. These lattices fill the active region cells in each of the 14 pebbles (The pebbles in the corners of the unit cell are 1/8th of a pebble so 4 corners are grouped into a single burnup state such that each burnup state occupies the volume of 1/2 of a pebble) in the FCC unit cell. Reflective boundaries are implemented on each of the surfaces of the unit cell such that the model represents an infinite pebble bed. This FCC unit cell behaves similar to a bed of pebbles that are well-mixed with respect to burnup because these 8 burnup states are located within close proximity to each other. BEAU is utilized to determine the equilibrium state of this system using the MBSM methodology described in Chapter 2.

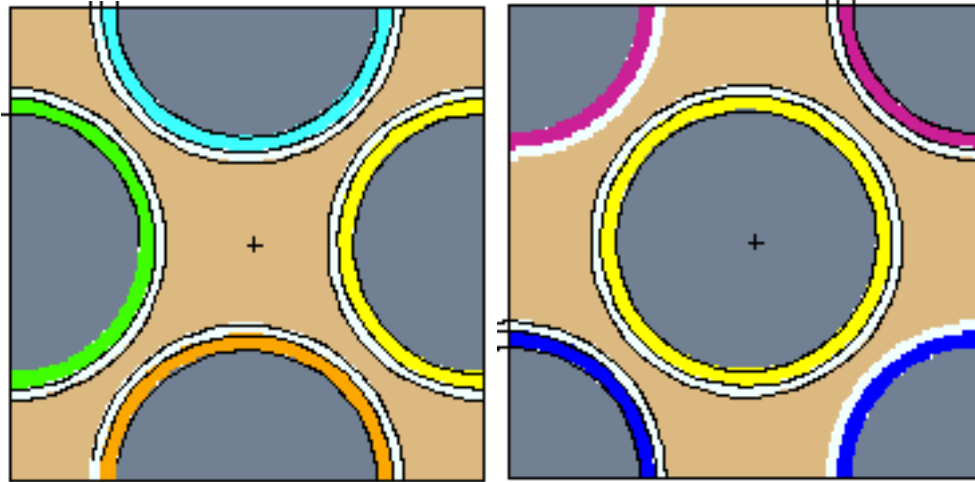


Figure 3-5. MCNP5 model of the pebble unit cell neutronics model: aqua, magenta, green, yellow, orange and blue are active regions of the fuel pebble; tan, white and gray are the flibe, shell and inert graphite pebble core respectively.

The pebble model has a simple fuel cycle representation in BEAU. This model utilizes a single progression with 8 burnup states necessitating a single equilibrium fuel cycle object, with a single circulation progression, with a single circulation, with a single progression as shown in Figure 3-6.

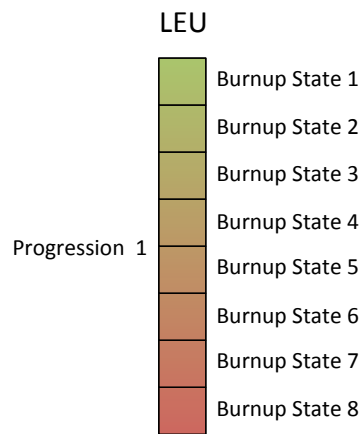


Figure 3-6. Fuel cycle object conceptualization of the pebble depletion model.

Only the evolution of the isotopic composition in the fuel was tracked explicitly, assuming the activation of graphite-based materials is negligible. The equilibrium concentration of ^6Li for each model was estimated on-the-fly using the AUSCE option in BEAU as described in Chapter 2.

The neutron spectrum in this unit cell model is harder than in a finite system because there is no moderation from the inner-, outer- and axial graphite reflectors. Moreover, since the target $k_{\text{effective}}$ is assumed to be 1.0, the burnup is over estimated because no leakage is considered. Therefore, results from this model should be used to understand

trends and tradeoffs with respect to fuel design, rather than assuming they apply to a finite system without distortions.

3.2.3 PB-AHTR Model

A full core model is required to understand full-core phenomena such as depletion analysis with multiple fuel types, control element worth, radiation damage to graphite reflectors, etc. A homogeneous (with respect to fuel compositions within the pebble bed) full-core model was used to assess the trade offs associated with different core configurations of an earlier 900 MWt variant of an Annular FHR PBR—the Annular PB-AHTR. This model was utilized to perform the core configuration selection design study.

3.2.3.1 Engineering Parameters

Several engineering parameters were used to define the geometry and operating conditions in the homogeneous full-core model of the PB-FHR. The fuel design of the seed and blanket pebbles were defined with the pebble fuel design objects described in the previous section; independent pebble objects were used to define the seed- and blanket pebbles respectively. The radial dimensions of the entrance-, active- and defueling chute regions could each be defined explicitly by the user; otherwise, the default radial dimensions were taken as the geometry described by R. Hong, et al (Hong, 2009). Likewise, the axial build of these same regions could each be defined explicitly by the user; otherwise, the default radial dimensions were taken as the same baseline geometry. The outer angle of the expansion and converging region were also adjustable. FIMPS inferred the total power of the system by multiplying the total volume of the expansion-, active- and converging regions in the seed and the blanket by an adjustable power density. FIMPS accepted these user defined parameters and worked out the detailed geometry such that all regions interface well and the area fractions of the seed and blanket are maintained at a fixed values established by the dimensions in the active regions of the pebble bed to be consistent with the physics of granular flow in packed beds as discussed in section 2.2.1.

3.2.3.2 Heat Transfer Model

FIMPS utilizes the previously described pebble unit cell heat transfer model as the foundation of the full-core heat transfer model. The power density and bulk coolant temperature are estimated from user-defined inputs. The power density in the seed and blanket are estimated using a user input blanket power fraction (fraction of power produced in the blanket of the system) as shown in equation 3-9 and equation 3-10 respectively; where, Q is taken as the full core power; f_{blanket} is taken as the blanket power fraction; ρC_p is taken as the volumetric heat capacity; and T_{inlet} and T_{outlet} are the temperatures at which coolant enters and exits the core respectively. The default blanket power fraction in FIMPS is 20%, but this can be updated based on simulations. The bulk coolant temperature for each region (seed and blanket) is determined based on the assumed power fraction produced in that region, the outlet temperature of the system and the temperature rise across the core. The flow conditions (i.e. core characteristic superficial velocity) of the system are estimated based the power of the core, the temperature rise, the core averaged heat capacity and an estimate of the

superficial area as shown in Equation 3-11; where $\phi_{\text{superficial}}$ is the angle between the horizontal and the surface normal to the coolant flow in the pebble bed and θ is the porosity of the pebble bed. These boundary conditions – region specific bulk temperatures, region specific power density and core average Reynolds number – are pushed to the pebble unit cell heat transfer model to estimate the average temperatures in each of the fuel types in the system. The temperature distribution from this characteristic pebble is applied for each the fuel in each burnup state as the distribution of power between the pebbles is unknown before the calculation is initiated.

$$T_{\text{bulk}}^{\text{seed}} = \frac{Q(1-f_{\text{blanket}})}{2\rho C_p} + T_{\text{inlet}}$$

Equation 3-9

$$T_{\text{bulk}}^{\text{blanket}} = T_{\text{outlet}} - \frac{Q(f_{\text{blanket}})}{2\rho C_p}$$

Equation 3-10

$$v_{\text{superficial}} = \frac{Q \cos(\phi_{\text{superficial}})}{\theta \pi \left((R_{\text{blanket}}^{\text{outer}})^2 - (R_{\text{seed}}^{\text{inner}})^2 \right) \rho C_p \Delta T}$$

Equation 3-11

3.2.3.3 Neutronics and Depletion Model

The PB-AHTR MCNP5 model is used to simulate the 900 MWt Annular PB-AHTR with a LEU seed and a Thorium blanket. In the active region of the pebble bed, a 90 cm radius solid graphite reflector is surrounded by a 150cm thick annular pebble bed filled with LEU seed (130 cm thick) and a thorium blanket (20 cm thick) – though seed only- and graphite pebble reflector variants are also investigated. The TRISO particles are represented as spherical fuel kernels on a simple cubic lattice embedded in the volume weighted homogenized material representing the TRISO layers and graphite matrix like in the pebble model. This universe of TRISO particles is implemented in the active regions of pebbles in a FCC unit cell. Only 4 burnup states are used for each fuel type – LEU or Thorium – such that pebbles of a given burnup state mate at the unit cell interfaces in this repeated structure. These 4 burnup state FCC unit cell repeated structures are implemented throughout the entire volume of the seed and blanket pebble bed respectively (i.e. within either region there is no axial, radial or azimuthal heterogeneity of the fuel composition). The extent of the PB-AHTR full-core model terminates outside the outer radial graphite reflector, above the defueling chute and below the free surface of the entrance region; vacuum boundary conditions are conservatively assumed outside these boundaries.

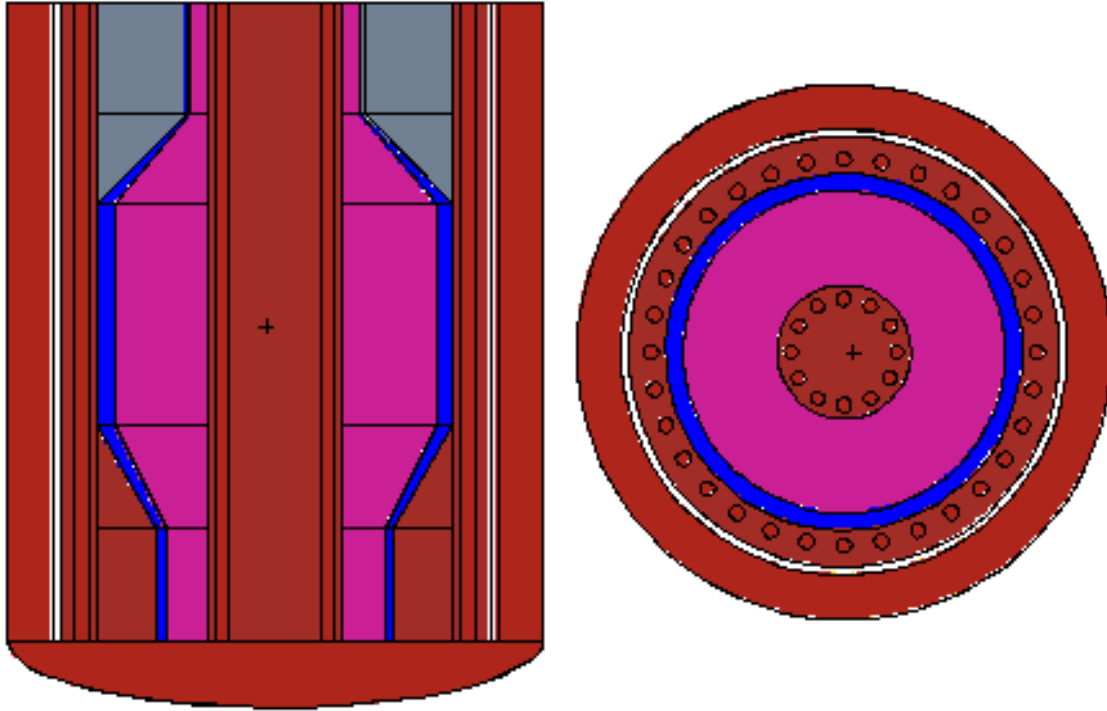


Figure 3-7. MCNP5 full-core model of the 900 MWt PB-AHTR homogenous: (left) R-Z axial build; (right) X-Y horizontal build in the active region.

The equilibrium fuel cycle object in this model requires two circulation progressions. These circulation progressions are relatively simple. Each burnup state for each region corresponds to a very simple circulation object. This circulation object contains a single progression, with a single burnup state as shown in Figure 3-8. These circulations do not correspond directly to pass through the core (especially for the thorium circulation progression, given the long cycle length) but rather to a range of burnups (i.e. the first arbitrary number of passes correspond to the first circulation object in the circulation progression object, etc.).

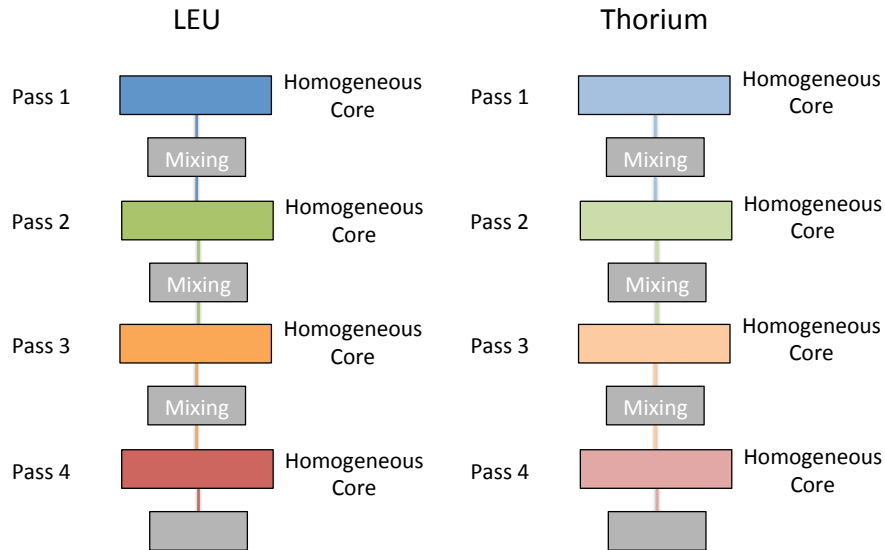


Figure 3-8. Fuel cycle object conceptualization of the pebble depletion model: (left) circulation progression for the LEU pebbles; (right) circulation progression for thorium pebbles

Only the evolution of the isotopic composition in the fuel kernels is tracked explicitly. The equilibrium concentration of ^6Li for each model was estimated using the AUSCE option in BEAU as described in Chapter 2.

This 900 MWt PB-AHTR full-core model neglects any axial- or radial variation in the neutron spectrum and flux intensity in the depletion analysis – leading to under predicting power in the lower section of the core and over predicting power in the top of the core. The defueling chute is challenging to cool so it is critical for the thermal hydraulic design to accurately estimate the power in the top of the core.

3.2.4 MPP PB-FHR Full-Core

The MPP PB-FHR full-core model was used to explore the trade-offs associated with adjusting the power density and geometric dimensions of the pebble bed of a 290 MWt PB-FHR, fully implementing the MPP design methodology. This model was developed to address all the deficiencies of the previous models by subdividing the PB-FHR into multiple axial- and radial zones. The scope of the pebble bed optimization focused around a simplified MPP PB-FHR with just a LEU seed surrounded by an inert graphite pebble reflector.

3.2.4.1 Engineering Parameters

The heterogeneous full-core model has several design options for the user to adjust. Like the homogeneous full-core model, the pebble design is defined using pebble objects. Porosity distributions are coarsely implemented with two pebble packing fractions – a lower packing fraction to be used near the wall and in the entrance-, expansion-, converging-, and defueling chute regions as well as next to the wall, and the pebble packing fraction in the bulk of pebble bed – away from the wall in the active region. The number of passes through the core is a user-defined parameter. The

number of passes through the core has implications into the multiple burnup state unit cell, volume cards and the structure of the fuel cycle definition. The number of control rods and control blades are adjustable parameters. Furthermore, the dimensions of these control elements – the control rod channel and relative width and thickness of the rod and the length, width and thickness of the control blades in terms of pebble radii – can be adjusted by the user. The radial thicknesses in active region – active thickness, thickness of the pebble bed and thickness of the solid graphite reflector are adjustable parameters. The radial thickness of defueling chute is defined in terms of the pebble diameters to prevent bridging: 5 pebble diameters by default, and the radial thickness of the entrance region is dictated by the thickness required to establish the graphite pebble reflector. This thickness is the minimum radial thickness such that in the entrance region, the pebble reflector is simultaneously the minimum radial thickness to establish a discrete radial zone and the same ratio of LEU and reflector pebbles in the active region is imposed in the entrance region – ratio of pebbles is used rather than area fraction because of the heterogeneous distribution of porosity in the active region. The height of the active region is set to impose the volume of the expansion-, active- and converging regions of the active region of the pebble bed such that the product of these volumes and the power density produce the total power of the core. The coolant inlet- and outlet channels are represented by cells with coolant and graphite structures homogenized. The depth into the inner- and outer graphite reflectors, axial extent of these coolant channels and volume fraction of coolant are all user adjustable parameters.

3.2.4.2 Heat Transfer Model

The heat transfer model for the heterogeneous full-core model is similar to that for the PB-AHTR full-core model with a few adjustments in assumptions. Rather than estimating the Reynolds number based on the specific design the user can supply this Reynolds number directly. However, the bulk temperature of the coolant is estimated based on the inlet and outlet temperatures across the core. The Reynolds number, bulk coolant temperature and power density are pushed to the pebble heat transfer model, which calculates the characteristic temperatures within the pebble used for all fuel pebbles in this model.

3.2.4.3 Neutronics and Depletion Model

The MPP PB-FHR full-core MCNP5 model is used to simulate the 290 MWt Annular PB-FHR with a LEU seed and an inert graphite pebble blanket. The pebble bed in this neutronics model is subdivided into 4 radial zones and 5 axial zones. The structure of the radial subdivision of the pebble bed is designed to reflect the distribution of the multi-group flux spectrum in the PB-FHR, as shown in Figure 3-9. The first radial zone captures the thermal flux peak due to the inner reflector and serves as a separate portion of the pebble bed to implement the lower pebble packing fraction due to the wall effects. The second radial zone captures the fast flux peak. The third radial zone corresponds to the bulk of the pebble bed. Finally, the fourth radial zone captures the thermal peak from the inert graphite pebble reflector. The five axial zones correspond

to the entrance-, expansion-, active-, converging- and defueling chute regions. In each of these (R,Z) nodes there is a burnup state that corresponds to each of the passes the pebbles make through the core – these burnup states are implemented into an FCC pebble unit cell.

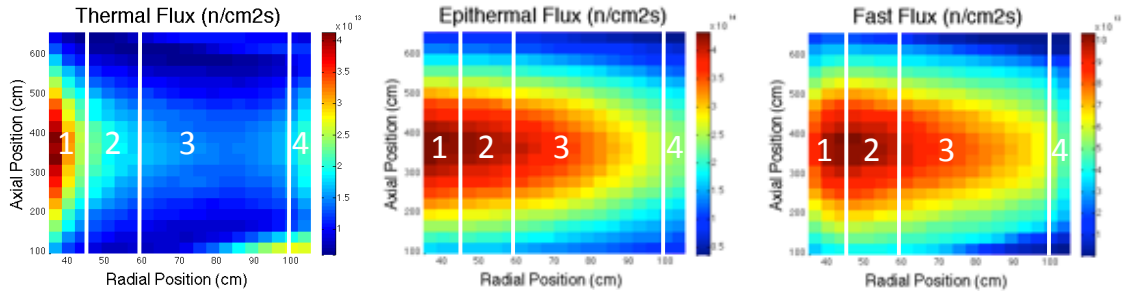


Figure 3-9. Multigroup flux distribution in the PB-FHR pebble bed with boundaries of four radial zones: (left) thermal flux distribution; (center) epithermal flux distribution; (right) fast flux distribution.

As stated in the engineering parameters section 3.2.4.1, the inlet and outlet channels are homogenized together with the graphite structural materials in a porous graphite cell at the interface between the solid graphite reflectors and the pebble bed in the neutronics model. Channels for control rods and control blades are explicitly modeled in the MCNP model with cells written, but commented out to analyze the worth of control elements. Tallies can be generated automatically for depletion-, neutron spectrum-, neutron economy-, or radiation damage distribution analysis. This MPP PB-FHR full-core model’s MCNP5 model is presented in Figure 3-10.

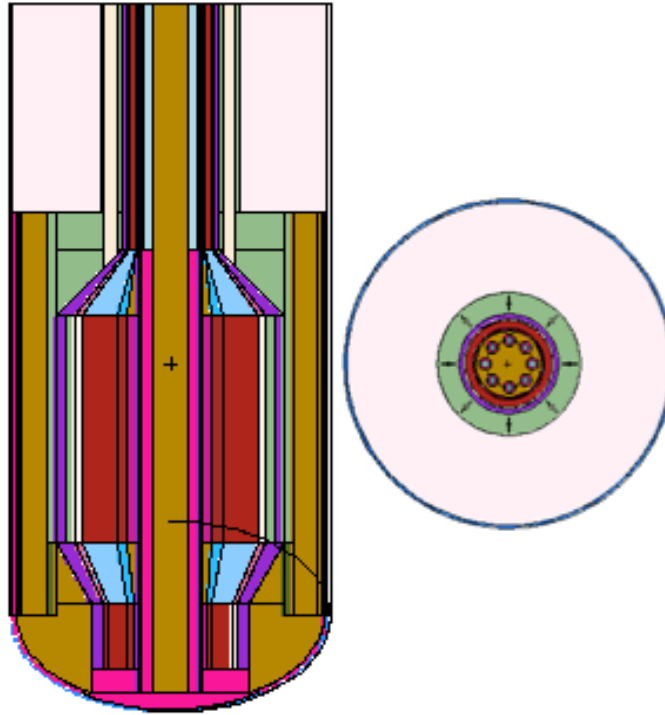


Figure 3-10. MCNP5 model of the MPP PB-FHR full-core neutronics model: (left) R-Z axial build; (right) X-Y horizontal build in the defueling chute.

The heterogeneous full-core model takes advantage of all of BEAUs depletion capabilities. Each of the four radial zones is a fuel progression for each circulation through the core. Between each pass, the discharge compositions from each progression in the circulation are collapsed into a single well-mixed composition before they are charged to the next circulation through the core. The conceptualization of the fuel cycle objects for the heterogeneous full-core model for a 8-pass fuel cycle is presented in Figure 3-11.

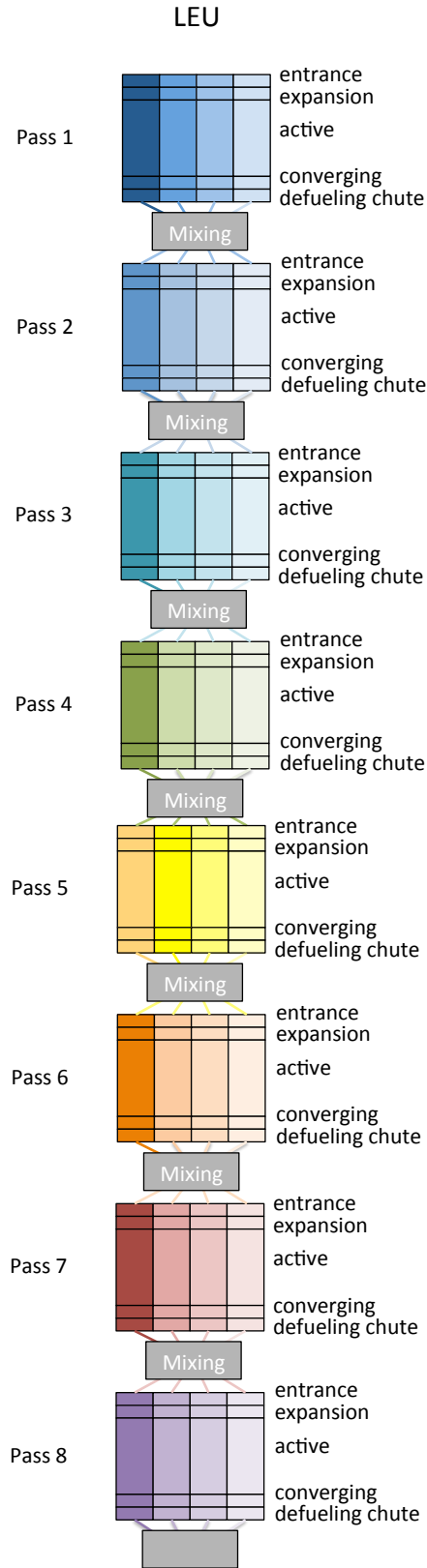


Figure 3-11. fuel cycle object conceptualization of the pebble depletion model.

3.3 Parametric Design Studies

As stated earlier, the PB-FHR design studies were broken down into three major studies: LEU pebble fuel design, core configuration and pebble bed core design. These design studies have been performed utilizing the neutronics models discussed in Sections 3.2. In each of the following modular design studies, a broad set of design points are analyzed to ensure the optimum design point is enveloped by the explored design space. The studies presented in the following subsections each include a description of their design space, design criteria, and results of their parametric study – though the design criteria- and optimization for the pebble bed design study are discussed separately in Chapter 4.

3.3.1 Fuel Design

The objective of this fuel design study is to select a baseline fuel design for the PB-FHR going forward. Because at the time of this first study, decisions about the core configuration and pebble bed geometry had not yet been made the pebble unit cell model was utilized for this study to decouple the fuel design studies from the other core design studies. The next few subsections discuss the design space-, design criteria-, and results for this fuel design study.

3.3.1.1 Design Space

The design space for the fuel design study encompassed C/HMs from 200 to 600 and TRISO kernel diameters from 250 μ m to 1000 μ m assuming 19.9% enriched low enriched uranium (LEU) fuel; all of the fuel designs were designed to have an average pebble density of 1.745 g/cm³ and an average power density of 16.2 MWt/m³. The pebble fuel designs, MCNP5 input decks and BEAU calculation directories were generated with FIMPS pebble object's design tools as discussed in Section 3.2.2. Fuel designs with low carbon to heavy metal ratios had significantly harder neutron spectrums because of the increased fuel loading. Additionally, these pebbles with high fuel loadings had lower fuel temperatures because the power per pebble was distributed amongst more fuel particles, reducing the power per particle and the fuel temperatures. Large kernel diameters concentrated the power produced per pebble into fewer TRISO particles resulting in higher average- and maximum temperatures. Also, the periphery of these large fuel kernels absorbed neutrons in the resonance energies, self-shielding the inner region of the kernel from these neutrons.

3.3.1.2 Design Criteria

An acceptable FHR pebble fuel design must satisfy the following design criteria related to safety and economics: the sign of the fuel and coolant temperature reactivity coefficients must be negative and furthermore; under an anticipated transients without scram (ATWS) accidents for the reactivity feedback mechanisms to establish a long term equilibrium state with only subcritical neutron fissioning, the maximum coolant temperature must be less than approximately 750 °C-800 °C to protect the structural metallic components in the reactor system (this is a preliminary design criterion until

coupled thermal-structural analysis defines time at temperature limits for the metallic structural components); the peak fuel operating temperature must be less than 1200 °C. Of the designs that satisfy these safety criteria the optimal design is the design point with the highest burnup (i.e. the lowest fuel cost).

3.3.1.3 Results

3.3.1.3.1 Fuel Temperature

The volume averaged- and maximum pebble fuel temperatures for the pebble unit cell are presented as a function of C/HM and kernel diameter in Figure 3-12. These values were calculated using the unit cell heat transfer module in FIMPS assuming the nominal power density (16.2 MWt/m³), flow conditions (Re ~1200) and bulk coolant temperature (650 °C). The average fuel temperature is used in estimating the ATWS response, while as the maximum fuel temperature serves as a proxy for fuel performance. These results are highly idealized, as they do not account for thermal property degradation, geometric- or burnup related power peaking.

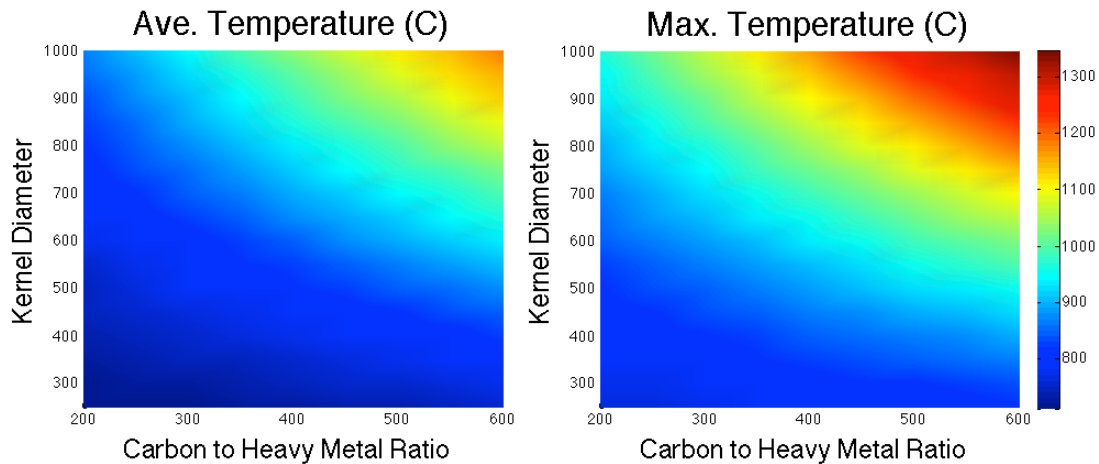


Figure 3-12. Fuel temperatures for fuel design space: (left) volume averaged fuel temperature; (right) maximum fuel temperature.

These trends behave as expected. Fuel designs with large kernels and high C/HM do exceed the maximum fuel temperature limit. However, the maximum temperatures for the vast majority of the design space satisfy the peak fuel temperature constraint with several hundred degrees of margin.

3.3.1.3.2 Burnup

The burnup was calculated for several design points using BEAU; this distribution of burnups is presented in Figure 3-13.

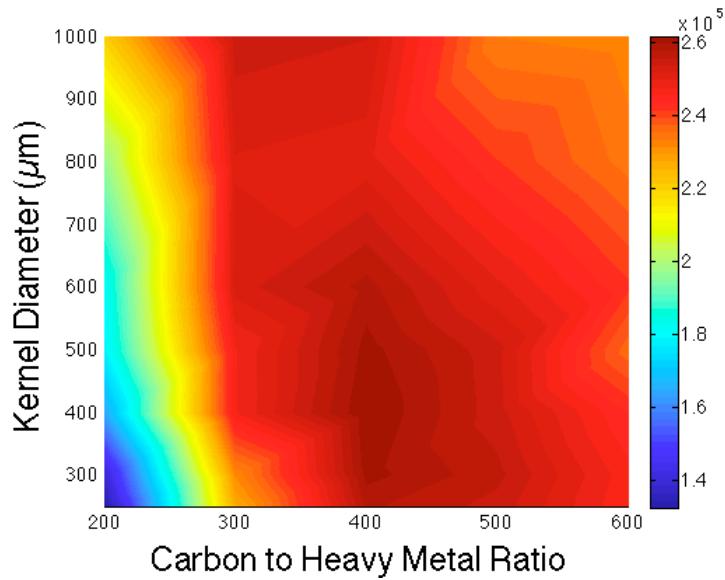


Figure 3-13. Distribution of burnups (MWt-d/MT) for fuel design space.

The multiplication factor initially increases with increasing C/HM as these fuel designs have softer spectrums that make fission more probable – however, for C/HM greater than 400 the system becomes over-moderated and more neutrons are absorbed in the coolant and structural materials. The multiplication factor increases with increasing kernel diameter as fewer neutrons are lost to resonance absorptions – again, reducing breeding. Ultimately, a balance of soft spectrums, neutron economy and modest breeding results in the highest burnup.

3.3.1.3.3 Temperature Reactivity Coefficients

The reactivity coefficients for the fuel design space are calculated by perturbing the temperature of one group of cells by changing the cross sections libraries in the material cards and updating the temperature card in the MCNP input file. For the fuel reactivity coefficient calculation, the temperatures in all the LEU fuel kernels cells are perturbed. For the coolant reactivity coefficient, the coolant density is perturbed in addition to the material- and temperature cards. The distributions of fuel- and coolant temperature reactivity coefficients are presented in Figure 3-14.

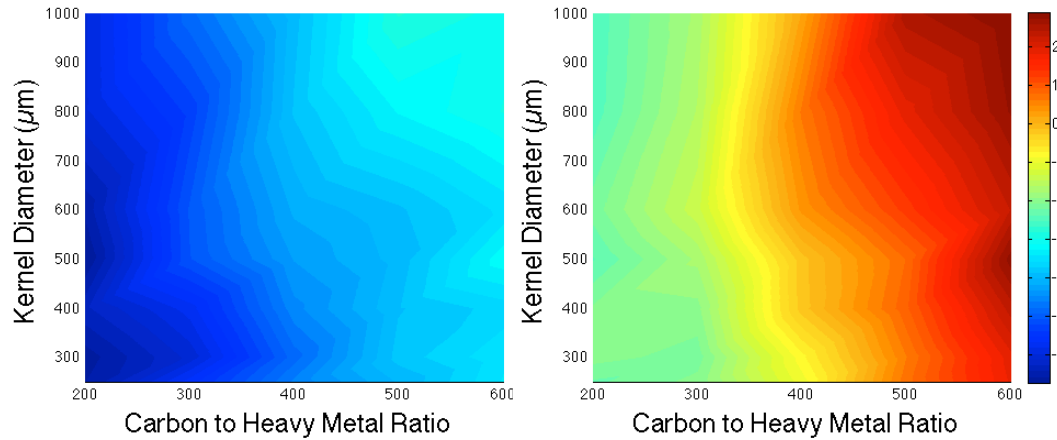


Figure 3-14. Temperature reactivity coefficients (pcm/K) in the fuel design space: (left) LEU fuel temperature reactivity coefficient; (right) coolant reactivity coefficient.

The negative fuel reactivity feedback is caused by increased absorption in resonance energies because of Doppler broadening. Therefore, this mechanism is expected to be strongest for systems with harder spectrums and higher burnups as more neutrons are in these resonance energies, as exhibited in Figure 3-14.

Three reactivity feedback mechanisms affect in the coolant reactivity feedback: degradation of the neutron moderating capability of the coolant, removal of neutron absorbing material in the coolant, reduced neutron reflection from the coolant. Only the first two mechanisms can be simulated in the pebble unit cell model. The negative moderation component of the reactivity feedback is stronger at low C/HM as these systems are severely under moderated. Moreover in these low C/HM fuel designs, more of the neutrons are absorbed in the fuel because of the high fuel loading and as a result they are less sensitive to the positive, “removal of neutron absorbing material in the coolant” reactivity feedback component.

All the fuel temperature reactivity coefficients are strongly negative at the equilibrium state. The coolant temperature reactivity coefficient transitions from negative to positive at a C/HM of approximately 400. The increases in the concentration of ${}^6\text{Li}$ in the coolant increase the positive, “removal of neutron absorbing material in the coolant” reactivity component; this will affect the balance coolant reactivity feedback. The transition from the initial- to the equilibrium concentration of ${}^6\text{Li}$ in the coolant is outside the scope of this design study, but will likely be an important part of the safety analysis for licensing of a FHR nuclear power plant.

3.3.1.3.4 Anticipated Transient Without Scram Response

During a hypothetical ATWS accident temperature rise in the coolant introduces negative reactivity shutting down critical neutron multiplication thereby cooling the fuel kernels introducing positive reactivity – the fuel and coolant will eventually equilibrate to a subcritical state with a hot shutdown temperature between the characteristic operating temperature of the fuel and the coolant. In a simplified model of an ATWS the

fuel and coolant will equilibrate to a produce a single characteristic core outlet temperature (noticing that the temperature difference between the fuel and the coolant needed to remove decay heat from the fuel to the coolant is small and can be neglected) as shown in Figure 3-15. In this simplified model it is assumed the positive reactivity insertion from reducing the temperature of fuel kernels in the core is offset by negative reactivity insertion from increasing the coolant temperature to equilibrate locally with the fuel temperature. The average ATWS hot shutdown temperature can be estimated algebraically with just the nominal characteristic temperatures of the fuel kernels and coolant along with the temperature reactivity coefficients for the fuel and the coolant; see Equation 3-12. The limiting temperature is not this hot shutdown temperature, but the hot shutdown core outlet temperature. This limiting core outlet temperature is estimated by assuming the temperature rise during passive cooling is 100K and assuming that the hot shutdown temperature is approximately half way between the inlet and outlet coolant temperature. The distribution of the hot shutdown coolant outlet temperatures for the fuel design space is presented in Figure 3-16.

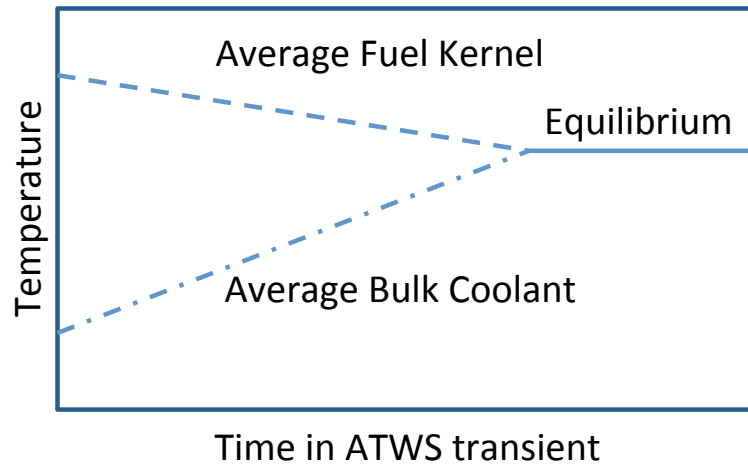


Figure 3-15. Temperature evolution of fuel kernel and coolant during a hypothetical ATWS accident –note the exact transition to equilibrium is not know so this transition is represented with broken lines.

$$T_{ATWS\ hot\ shutdown} \approx \frac{\frac{\partial \rho}{\partial T_{fuel}} T_{fuel}^o + \frac{\partial \rho}{\partial T_{coolant}} T_{coolant}^o}{\frac{\partial \rho}{\partial T_{fuel}} + \frac{\partial \rho}{\partial T_{coolant}}}$$

Equation 3-12

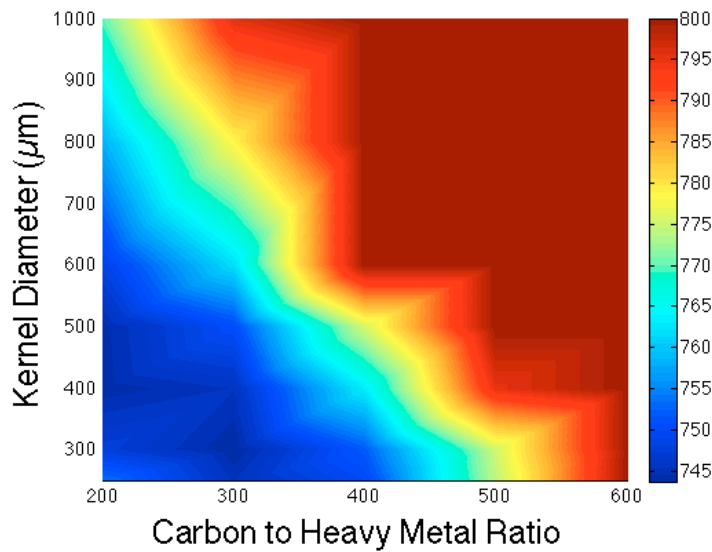


Figure 3-16. Maximum ATWS core coolant outlet temperature – temperatures above 800 °C are colored dark red so the reader can discern line where the fuel designs transition from acceptable to unacceptable with respect to ATWS response.

This hot shutdown temperature is manageable for fuel designs with high fuel loading (< 400 C/HM) and moderately sized fuel kernels (<600μm). For these fuel designs the low average fuel temperature limits the maximum core outlet coolant temperature. Moreover, the stronger negative coolant reactivity coefficients in this region of the design space increases the weight of the lower coolant temperature.

3.3.1.3.5 Baseline Pebble Design

The baseline pebble design was established as the design point with a C/HM of 300 and a kernel diameter of 400μm. The feasible design space is most limited by the ATWS response. The baseline LEU pebble design was selected as a design point close to the maximum burnup point (C/HM: 400, kernel diameter: 400μm), but a moderate distance within the boundary of this feasible design space, as shown in Figure 3-17. The engineering parameters for this fuel design are presented in Table 3-1.

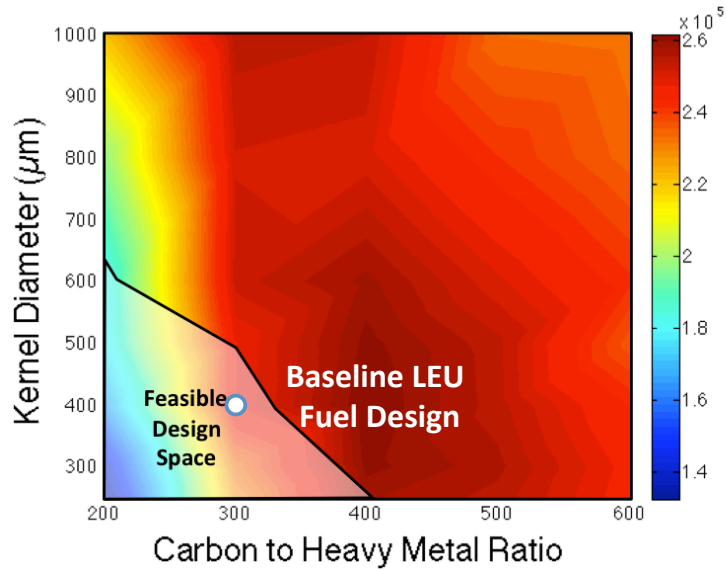


Figure 3-17. Baseline LEU fuel design point presented in the feasible fuel design space presented on top of the burnup distribution for the fuel design space.

Table 3-1 Dimensions of baseline fuel pebble.

Fuel Design Definition Parameters	
Carbon the heavy metal ratio	300
Fuel kernel diameter	400
Pebble parameters	
Pebble diameter (cm)	3.0000
Pebble shell thickness (cm)	0.1000
Radius of pebble core (cm)	1.25114
Density of pebble core (g/cm ³)	1.59368
Density of pebble shell (g/cm ³)	1.74000
TRISO parameters	
Thickness of buffer layer (μm)	100
Thickness of inner PyC layer (μm)	35
Thickness of SiC layer (μm)	35
Thickness of outer PyC layer (μm)	35
TRISO packing fraction (%)	40
Enrichment of LEU (w%)	19.9
Density of kernel UC _{0.5} O _{1.5} (g/cm ³)	10.5
Density of buffer layer (g/cm ³)	1.0
Density of PyC layers (g/cm ³)	1.87
Density of SiC layers (g/cm ³)	3.2

3.3.2 Core Configuration

The objective of this core configuration design study is to select a core configuration for the PB-FHR going forward. The next few subsections discuss the design space- (i.e. core configuration options), design criteria-, and results for this core configuration design study. These results are used to establish a baseline core configuration going forward.

3.3.2.1 Core Configuration Options

A key evolution of the annular pebble bed from the previous incarnations of FHR PBRs is the use of multiple pebble types with individual pebble fuel designs and residence times implemented as burnup limits. An alternative approach would be to incorporate a reflector composed of inert graphite pebbles immersed in the same flibe coolant as the active core as proposed for the LIFE engine (Kramer, Latkowski, Abbott, Boyd, Powers, & Seifried, 2009).

One of the key economic considerations of this class of pebble bed reactors is the power density of core and the service lifetime of the plant, possibly limited by radiation damage to structural components caused by high energy fission neutrons. This damage could be mitigated by absorbing epithermal neutrons in a thorium fueled fertile blanket and moderating and attenuating other high-energy neutrons in the flibe and structural graphite in the blanket. As an added benefit, a significant amount of low-cost energy could be generated from this fertile blanket because thorium is abundant widely considered an industrial waste. Alternatively, an inert graphite pebble reflector could be utilized to perform a similar function though without any power generation.

For this study, three core configurations were investigated: a seed only pebble bed, a LEU fueled pebble bed with a fertile thorium blanket, or a LEU fueled pebbled with an inert graphite pebble reflector. These core design options are presented in Figure 3-18. The C/HM of the thorium blanket pebbles and discharge burnup of these blanket pebbles are taken as design variables for the designs of the PB-AHTR with a thorium blanket.

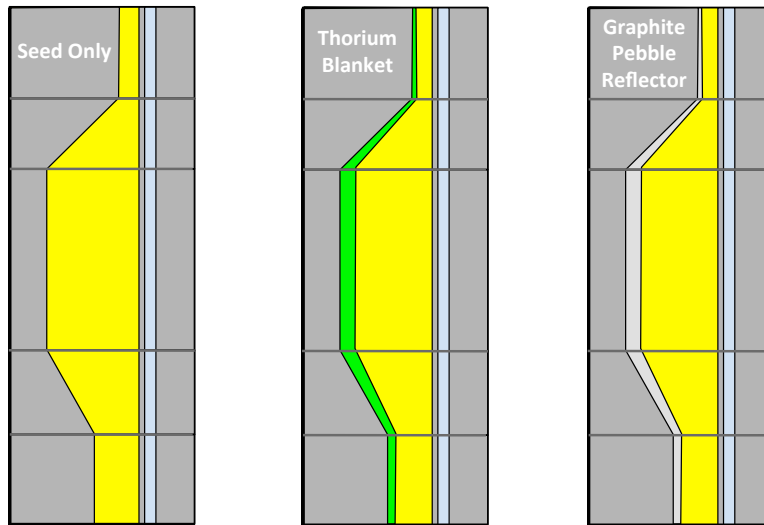


Figure 3-18. Potential PB-AHTR Core Configurations: (left) seed only; (center) thorium blanket; (right) graphite pebble reflector

The thorium blanket employs a once-through fuel cycle without reprocessing. It is assumed that reprocessing fuel from the coated particle fuel will be unacceptably expensive. Therefore, the blanket utilizes pure fertile thorium and breeds its own fissile material without enrichment or reprocessing.

To optimize the thorium blanket several combinations of carbon to heavy metal ratios and discharge burnup were investigated. In a seed and blanket system like the PB-AHTR, the burnup in the blanket is not limited by criticality, but rather any subcriticality in the blanket is compensated by the LEU seed fuel. The design space considered for this study covered carbon to heavy metal ratios from 100 to 400 and thorium pebble burnups from 100000 to 250000 MWd/MT – thorium can be recirculated enough times to impose any cycle length or burnup. The carbon to heavy metal ratio of the blanket fuel pebble was induced using the previously discussed pebble design algorithm, however a larger fuel particle was utilized to impose these high fuel loading designs to increase the volume fraction of the fuel kernel – the dimensions of this particle are presented in Table 3-2.

Table 3-2. Dimensions of thorium blanket TRISO particles

Thorium Fuel Kernel Diameter (μm)	600
Buffer layer thickness (μm)	64
Inner pyrocarbon layer thickness (μm)	26
Silicon carbide layer thickness (μm)	31
Inner pyrocarbon layer thickness (μm)	55

3.3.2.2 Design Criteria

The outer graphite reflector should last the life of plant, which might be 60 or more Effective Full Power Years (EFPY). For this study a radiation damage limit of 19.1 DPA on

the outer graphite reflector is implemented based on the “turn around” point for graphite at 700 °C – a discussion of this limit is presented in section 4.3.2.3.1. Either a thorium blanket or pebble reflector can significantly reduce radiation damage to the outer solid graphite reflector. However, this shielding should be quantified and compared to a simpler system without any radiation damage mitigation to understand the tradeoffs between reducing radiation damage verse the requirement of pebble sorting and radial zoning.

The same requirements in terms of negative temperature reactivity coefficients and gentle ATWS response (hot shutdown coolant outlet temperature <750 °C) must be applied to the full-core system. However, the effect of the blanket must be integrated into the ATWS response model for the cases with a blanket.

The fuel utilization of the selected design must be economically competitive with alternatives. The thorium blanket can reduce the fuel costs of the system by generating energy from fuel that is less expensive than the LEU seed pebbles. However, the subcritical blanket reduces the burnup from the LEU seed fuel and requires fuel fabrication. Therefore, the fuel utilization must be compared on a basis that reflects these factors.

3.3.2.3 Results

3.3.2.3.1 Radiation Damage

The service lifetime of the outer graphite reflector is estimated based on the peak radiation damage rate in outer graphite reflector. The peak radiation damage rate is calculated with a mesh tally in the first 1cm of the outer graphite reflector adjacent to the pebble bed with 20 axial nodes. The maximum DPA rate is taken as the limiting rate and the service lifetime of the outer graphite reflector is estimated as the DPA limit divided by this peak DPA rate. The service lifetime for outer reflector is presented in Table 3-3 for various core configurations.

Table 3-3. Service lifetime of outer graphite reflector

	Outer reflector service lifetime (EFPY)				
Seed only	39				
Blanket BU (MWd/MT)	100000	150000	200000	250000	Average
C/HM: 100	102	100	97	96	99
C/HM: 200	119	129	125	128	125
C/HM: 300	154	160	148	148	156
C/HM: 400	187	173	192	175	182
Graphite reflector	339				

Either a pebble reflector or thorium blanket is required to meet the outer reflector service lifetime design criteria. A graphite pebble reflector is more effective at shielding

the outer graphite reflector because it only moderates and attenuates neutrons and produces no high-energy neutrons.

3.3.2.3.2 Temperature Reactivity Coefficients

The full-core reactivity coefficients are calculated by perturbing the temperature of one group of cells by changing the cross sections libraries in the material cards and updating the temperature card in the MCNP input file. For the LEU fuel reactivity coefficient calculation, the temperature in all the LEU seed fuel kernels cells are perturbed. Likewise, for the blanket reactivity coefficient calculation, the temperature in all the thorium blanket fuel kernels cells are perturbed holding everything else constant. For the coolant reactivity coefficient, the coolant density is perturbed along with the temperature.

The LEU fuel reactivity coefficients are presented in

Table 3-4. The LEU fuel reactivity coefficients are relatively consistent because the fuel design is the same resulting in similar neutron spectra and similar burnups.

Table 3-4. LEU Fuel Temperature Reactivity Coefficients

	LEU Fuel Temperature Reactivity Coefficients (pcm/K)				
Seed Only	-4.7				
Blanket BU (MWd/MT)	100000	150000	200000	250000	Average
C/HM: 100	-4.7	-4.4	-4.5	-4.2	-4.4
C/HM: 200	-4.6	-4.8	-4.5	-3.9	-4.4
C/HM: 300	-4.5	-4.3	-4.5	-4.5	-4.5
C/HM: 400	-4.5	-4.5	-4.2	-4.5	-4.4
Graphite Reflector	-4.5				

The coolant reactivity coefficients are presented in Table 3-5. In the full-core model, all three mechanisms of the coolant reactivity feedback are modeled. The reduced reflection from the flibe is a complicated feedback mechanism that changes its effect for different core configurations. For systems with high fuel loading (i.e. low C/HM) the reduced reflection mechanism results in an increased flow of neutrons from the high reactivity seed into the low reactivity blanket where they are more likely to be absorbed in the resonance region than in the seed, thereby reducing the aggregate reactivity of the system (Cisneros, Greenspan, & Peterson, Use of Thorium Blankets in a Pebble Bed Advanced High Temperature Reactor, 2010). There is a continuum from high fuel loading blanket to high C/HM blanket to graphite reflector. As fuel loading in the blanket is reduced the interaction in the blanket shifts from neutron absorption to neutron moderation. In the case of the pebble reflector, increased communication between the seed and pebble reflector increases the moderating capability of the pebble reflector partially offsetting the loss of moderation from the flibe (Cisneros, Greenspan, & Peterson, Use of Thorium Blankets in a Pebble Bed Advanced High Temperature Reactor, 2010).

Table 3-5. Coolant Temperature Reactivity Coefficients

	Coolant Temperature Reactivity Coefficients (pcm/K)				
Seed Only	-0.6				
Blanket BU (MWd/MT)	100000	150000	200000	250000	Average
C/HM: 100	-0.7	-0.7	-0.7	-0.6	-0.7
C/HM: 200	-0.5	-0.5	-0.6	-0.5	-0.5
C/HM: 300	-0.5	-0.5	-0.5	-0.5	-0.5
C/HM: 400	-0.5	-0.5	-0.2	-0.3	-0.4
Graphite Reflector	-0.3				

The thorium blanket fuel reactivity coefficients are presented in Table 3-6. All of these reactivity coefficients are either negative or close to zero as the thorium fuel cross section resonances broaden with increasing fuel temperature like the LEU fuel. However, since less fissioning occurs in the blanket this Doppler broadening phenomenon has a smaller impact on the aggregate reactivity compared the LEU fuel.

Table 3-6. Blanket Fuel Temperature Reactivity Coefficients

	Blanket Temperature Reactivity Coefficients (pcm/K)				
Blanket BU (MWd/MT)	100000	150000	200000	250000	Average
C/HM: 100	-0.3	-0.1	-0.1	-0.3	-0.2
C/HM: 200	-0.2	-0.2	-0.1	-0.1	-0.1
C/HM: 300	-0.1	-0.3	-0.2	+0.0	-0.1
C/HM: 400	-0.2	-0.1	+0.1	+0.0	-0.1

These reactivity coefficients along with the fuel temperatures are integrated into a proxy for the transient response: ATWS coolant outlet temperature. Equation 3-13 integrates the influence of the blanket into this calculation. The initial fuel temperatures in the LEU- and thorium fuel are updated based on the power density in the seed and blanket respectively with the simplified unit cell heat transfer model. The calculated hot shutdown coolant outlet temperatures are presented in Table 3-7.

$$T_{ATWS \text{ hot shutdown}} \approx \frac{\frac{\partial \rho}{\partial T_{fuel}} T_{fuel}^o + \frac{\partial \rho}{\partial T_{blanket}} T_{blanket}^o + \frac{\partial \rho}{\partial T_{coolant}} T_{coolant}^o}{\frac{\partial \rho}{\partial T_{fuel}} + \frac{\partial \rho}{\partial T_{blanket}} + \frac{\partial \rho}{\partial T_{coolant}}}$$

Equation 3-13

Table 3-7. Hot shutdown coolant outlet temperature

	Hot Shutdown Coolant Outlet Temperature (°C)				
Seed Only	743				
Blanket BU (MWd/MT)	100000	150000	200000	250000	Average
C/HM: 100	747	745	746	747	746
C/HM: 200	750	750	750	748	749
C/HM: 300	751	751	751	751	751
C/HM: 400	752	752	754	753	753
Graphite Reflector	758				

The increase in the hot shutdown coolant outlet temperature in designs with a thorium blanket or graphite pebble reflector comes from increasing the power density in the seed to allocate volume for either the thorium blanket or inert pebble reflector. However, the ATWS response is fairly consistent for all the proposed designs.

3.3.2.3.3 Fuel Utilization

The fuel utilization metrics are calculated based on the discharge burnup of the LEU seed, thorium blanket and the ratio of the core’s power produced in each region. The discharge burnup of the LEU seed and the coarse power distribution are calculated with BEAU – the discharge burnup of the thorium blanket is an engineering parameter. Several fuel utilization metrics – seed burnup, effective burnup, fuel costs – are used to compare various design points and compare to literature.

Burnup is a fundamental measure of how effectively the fuel is being utilized. The burnup of the LEU seed is presented for each of the design points in for this study in Table 3-8. The LEU burnup decreases in systems with either fertile blankets or pebble reflectors as LEU fuel is replaced in the periphery of the pebble bed increasing the leakage from the LEU fueled region. The leakage is more pronounced for systems with high fuel loading (ie low C/HM).

Table 3-8. Homogeneous full-core LEU fuel burnup

	Burnup (MWd/MT)				
Seed Only	221000				
Blanket BU (MWd/MT)	100000	150000	200000	250000	Average
C/HM: 100	209000	210000	211000	219000	210000
C/HM: 200	212000	211000	212000	211000	211000
C/HM: 300	213000	214000	214000	213000	213000
C/HM: 400	214000	214000	213000	213000	214000
Graphite Reflector	216000				

Effective burnup was a fuel utilization metric developed to account for the additional energy produced in the thorium blanket (Cisneros, Greenspan, & Peterson, Use of

Thorium Blankets in a Pebble Bed Advanced High Temperature Reactor, 2010). The LEU seed is largely responsible for the energy produced in the thorium blanket – without the excess neutrons from the LEU seed, the thorium blanket would only produce heat from spontaneous fission and radioactive decay. Additionally, since thorium is sometimes considered a waste it is assumed that fertile thorium metal can be procured for free (or at least order of magnitude less than natural uranium) and it will not incur the costs of enrichment. Table 3-9.

Equation 3-14 presents the definition of effective burnup and Equation 3-15 presents a simple way to calculate the effective burnup in a pebble bed from equilibrium state parameters. The effective burnup is presented for each of the design points in for this study in Table 3-9.

$$\text{Equation 3-14} \quad \text{effective burnup} = \frac{\int_{\text{fuel cycle}} P_{\text{system}}(t) dt}{HM_{\text{seed}}}$$

$$\text{Equation 3-15} \quad \text{effective burnup} = \frac{P_{\text{system}}^{eq}}{P_{\text{seed}}^{eq}} \text{burnup}_{\text{seed}}$$

Table 3-9. Homogeneous full-core effective burnup

	Effective Burnup (MWd/MT)				
Seed Only	221000				
Blanket BU (MWd/MT)	100000	150000	200000	250000	Average
C/HM: 100	216000	228000	229000	228000	228000
C/HM: 200	224000	223000	225000	224000	224000
C/HM: 300	222000	223000	223000	222000	223000
C/HM: 400	221000	222000	221000	221000	221000
Graphite Reflector	216000				

These burnup related fuel utilization metrics are being used as proxies for the fuel costs of these design points, so fuel costs is the best metric for fuel utilization as a proxy for economics. Additionally, the effective burnup metric assumes that the thorium metal is free neglecting the fact that a significant portion of the fuel costs in TRISO fuels are associated with fuel fabrication. Significant energy per thorium pebble must be generated (i.e. high burnup) to justify fabricating the fuel. Figure 3-19 presents the blanket fuel costs normalized to electricity produced as a function of blanket discharge burnup.

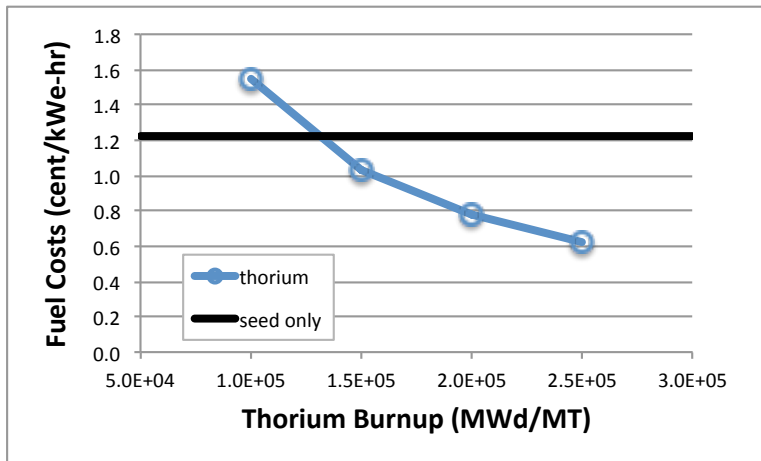


Figure 3-19. Normalized thorium pebble fuel costs as a function of burnup

The average fuel cost of the PB-AHTR is calculated by weighing the fuel costs of the LEU seed or thorium blanket by the fraction of the power produced in that region. The fuel cost is estimated using the fuel cost model presented in the Assessment of High Temperature Gas-Cooled Reactor (HTGR) Capital and Operating Costs (Gandrik, Wallace, Demick, Melancon, & Patterson, 2011) assuming a thermal efficiency of 41.4%. ;this model is described in detail in Chapter 4.

Table 3-10. Homogeneous full-core LEU fuel costs

Seed Only	Burnup (cent/kWe-hr)			
	1.22			
Blanket BU (MWd/MT)	100000	150000	200000	250000
C/HM: 100	1.31	1.26	1.23	1.23
C/HM: 200	1.28	1.26	1.24	1.24
C/HM: 300	1.27	1.25	1.24	1.24
C/HM: 400	1.27	1.25	1.24	1.24
Graphite Reflector	1.24			

The utilization of thorium fuel does not significantly affect the effective burnup or normalized fuel costs because so little power is produced in the thorium blanket (~7% of the total power). Increasing the size of the thorium blanket would either increase the capital costs of the nuclear power plant or displace volume of the seed region, thereby increasing the power density in that region and reducing the LEU fuel burnup reducing the effective burnup. Studies for a LWR transuranics (deep burn) fuel PB-AHTR with a thorium blanket indicate that minimizing the thickness of the blanket increases the effective burnup of the entire system as shown in Figure 3-20 – note the effective burnup is significantly higher than LEU fuelled systems because the initial fissile content of the deep burn fuel is ~50%.

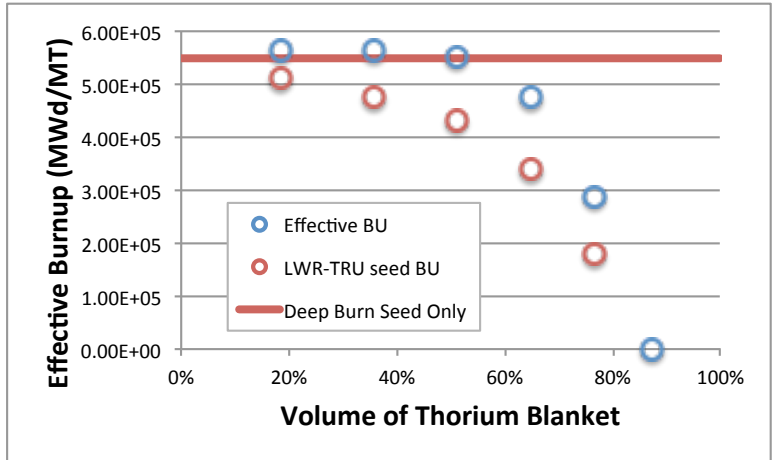


Figure 3-20. Effective burnup as a function of thorium blanket volume fraction for a deep burn variant of the PB-AHTR

The long time periods to reach an equilibrium state will be a key challenge associated with utilizing these fertile blankets in an FHR. At the low power density – as compared to LWR for Fast Reactors – in the blanket region of the core, the residence times required to breed up enough fissile material (≥ 200000 MWd/MT) to generate enough power in the blanket to improve the fuel costs economics relative to the seed only case are comparable to the lifetime of the nuclear power plant – 40-60 EFPY as shown in Figure 3-21.

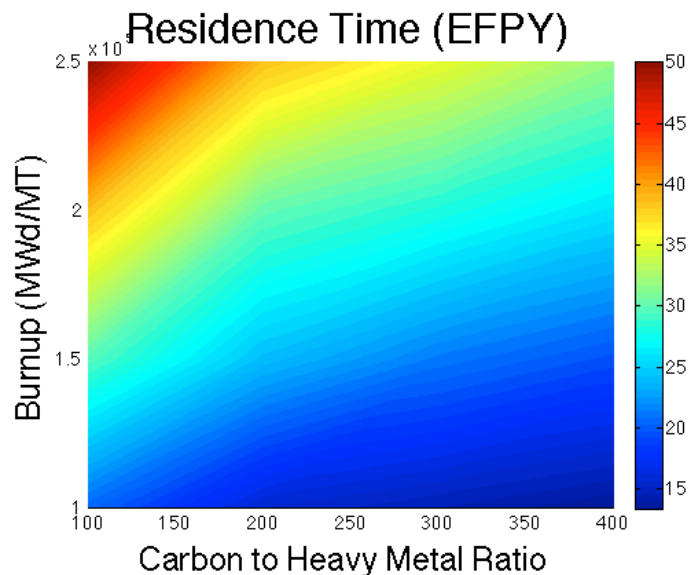


Figure 3-21. Residence time of the equilibrium state of the blanket fuel as a function of carbon to heavy metal ratio and discharge burnup in the PB-AHTR.

For the PB-AHTR to reach an equilibrium mode of operation, the fertile blanket fuel must be passed through multiple cores for almost a century or must be initially fueled with partially enriched starter fuel. Either of these options would require a significant

amount of effort for a very limit return in terms of increased fuel utilization or improved economics.

3.3.2.3.4 Baseline Core Configuration

The graphite pebble reflector core configuration was selected as the baseline core configuration for the PB-AHTR and PB-FHR. Either a pebble reflector or thorium blanket is required to shield the outer graphite reflector so it can survive the life of plant. In quantifying the benefit of utilizing a thorium blanket over pebble reflector, there is very little upside in terms of AWTs response – an additional 10 °C lower hot shutdown coolant outlet temperature – and reducing fuel costs – 0.01 cent/kWe-hr – for the additional effort of developing- and operation expenses of a second fuel type. Moreover, if the thorium fuel is developed the most economically attractive blanket fuel modes or operation are the most challenging to implement in terms of establishing the equilibrium state in the blanket due to the long residence times – 40-50 EFY.

3.3.3 Pebble bed Core Design

The final parametric study was performed to optimize the geometry of pebble bed assuming the baseline fuel design and baseline core configuration established in the previous two design studies. The results of this parametric study were interpreted in two ways: the implications on the high-level reactor physics parameters – neutron economy, flux spectrum, neutron leakage, etc. – were assessed as a function of engineering parameters. An alternative approach was to assess the sensitivity of performance parameters to the engineering parameters. These two perspectives together form a complete top-down and bottom-up understanding of the implications of design decisions in the PB-FHR.

3.3.3.1 Design Space

The design space for the pebble bed core parametric study included several design points that for a 290 MWt power level with a reactor vessel outer diameter of 3.5m (so the vessel is rail transportable). Within this envelop, the radial thickness of the LEU fuel zone of the active region of the pebble bed (*active thickness*), the radial thickness of pebble reflector zone of the active region of the pebble bed (*pebble reflector thickness*) and the *power density* – defined for this study as the power normalized the volume of the power production region of the pebble bed (expansion-, active- and converging regions)– are taken as the dimensions of the design space for this parametric study. A fourth parameter *active height* is also used to fully define a specific point design. The active height is height that the power production portion of the pebble bed would be if it were a regular annular cylinder rather than an annular cylinder with truncated annular cones above and below it. Increasing the radial thickness of the active region and/or the pebble reflector is accommodated by reducing the thickness of the outer graphite reflector. However, the radius of the inner graphite reflector was maintained for this study. Figure 3-22 shows how the engineering parameters affect the geometry of the MPP PB-FHR core.

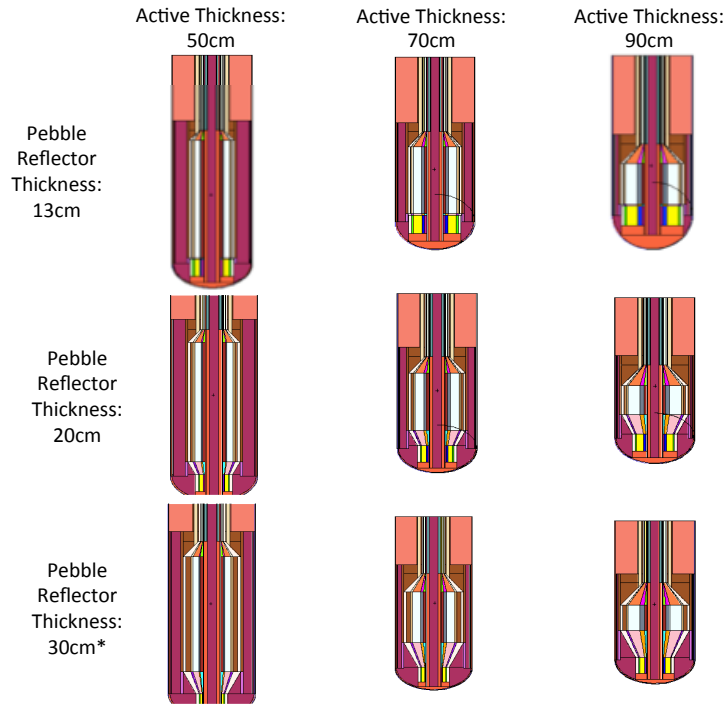


Figure 3-22. Full-core pebble bed geometries for the MPP PB-FHR for various combinations of active thicknesses and pebble reflector thicknesses for a 290 MWt MPP PB-FHR with a power density of 28 MWt/m³; note 90cm active thickness, the large pebble reflector case actual uses a pebble reflector thickness of 26cm rather than 30cm.

The center of the investigated design space was selected as a MPP PB-FHR design point with engineering parameters that correspond to an early version of the Mk 1 PB-FHR – Power Density: 23 MWt/m³, active thickness: 70 cm, pebble reflector thickness: 20 cm, active height: 3.33 m (Andreades, et al., 2014). There are only subtle differences between the pebble bed geometry between the MPP PB-FHR and the Mk 1 PB-FHR, therefore the results of this parametric study apply to the Mk 1 PB-FHR’s design space; an assessment of similitude between these two cores is investigated in Chapter 5 concluding the MPP PB-FHR parametric results can be applied to the Mk 1 PB-FHR.

Design points for this parameter study were analyzed at the maximum burnup equilibrium state determined by BEAU using the MBSM as discussed in Chapter 2.

3.3.3.2 Sensitivity to Engineering Parameters

The next sections assess how reactor physics parameters at the equilibrium state (i.e. BEAU determines case specific discharge burnups for each design point) are affected by changing the engineering parameters: power density, active thickness and the pebble reflector thickness.

3.3.3.2.1 Power Density

Power density is integrating into the neutronics model of the PB-FHR in two ways: First, the power density determines the active volume of the core and thereby the height of the active region; secondly, the power density affects the average fuel temperature in

the neutronics model. The dimensions of the design points used to assess the affects of changing the power density in the PB-FHR are presented in Table 3-11.

Table 3-11. Case definitions for design points in power density reactor physics implications study

Power Density (MWt/m ³)	23	28	33
Active Height (m)	4.04	3.33	2.83
Active Thickness (cm)	70	70	70
Pebble reflector thickness (cm)	20	20	20

The neutron economies for these design points are presented in Table 3-12. Notice that leakage increases with increasing power density. Increased power densities results in smaller, shorter pebbledbeds, thereby enabling more leakage in higher power density cores.

Table 3-12. Neutron economy for PB-FHR for various power densities

Power Density (MWd/m ³)	Fuel (%)	Moderator (%)	Flibe (%)	Leakage (%)
23	83.3	0.8	3.5	12.4
28	83.0	0.8	3.5	12.7
33	82.6	0.8	3.5	13.1

A FHR specific, continuous-energy multiplication factor formula was derived for additional neutron economy analysis of FHRs (Cisneros, Greenspan, & Peterson, Use of Thorium Blankets in a Pebble Bed Advanced High Temperature Reactor, 2010). $k_{\text{effective}}$ is the ratio of neutrons produced and neutrons lost. In FHRs, a non-negligible amount of neutrons are produced from (n,2n) reactions on the beryllium in the coolant; therefore, this source of neutrons is integrated into the FHR $k_{\text{effective}}$ formula. The reproduction factor, η , is defined as the ratio of neutrons produced from fissions in the fuel to neutrons absorbed in the fuel – here, taken as the fuel kernels. The neutron utilization factor, f , is defined as the ratio of neutrons absorbed in the fuel kernels to the neutrons absorbed anywhere else in the active region of the pebble bed – this includes all the components of the fuel pebbles and the flibe coolant. The non-leakage probability factor, L , is defined as the probability that a neutron born in the active region will be absorbed in the active region and is calculated as the ratio of absorptions in the active region of the pebble bed (not including the pebble reflector) to these absorptions plus the net leakage out of the pebble bed. The non-fission neutron generation factor, $F_{(n,Xn)}$, is the ratio of total neutrons generated (from fission and (n,Xn) reactions in the coolant) to fission born neutrons. When multiplied together many terms cancel out to leave this product as is the ratio of total amount of neutrons generated (fissions and (n,xn)) to the total amount of neutrons lost (absorptions and neutrons leaked out of the system). This FHR multiplication factor formula is presented in Equation 3-16.

Equation 3-16 $k_{effective}^{FHR} = \eta f L F_{(n, Xn)}$

The components of this FHR multiplication factor were calculated for design points with different power densities and presented in Figure 3-23.

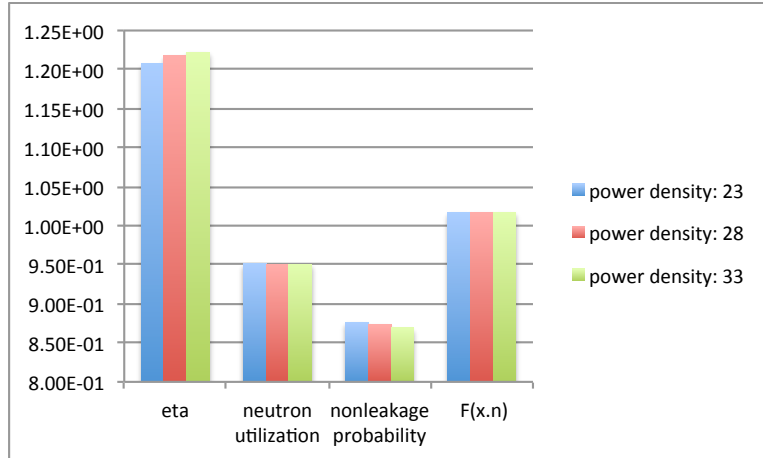


Figure 3-23. Continuous four factor components for the MPP PB-FHR for various power densities.

Increased power density results in an increase in the ^{135}Xe reactivity penalty (from operating at higher powers) as shown in Figure 3-24. This reactivity penalty increases a small amount – 70 pcm – across the design space.

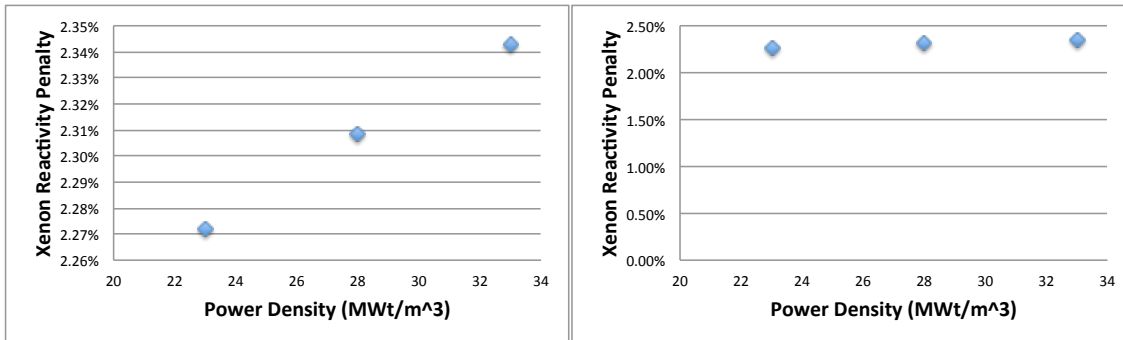


Figure 3-24. Xenon reactivity penalty as a function of power density.

The change reactivity associated with the power density dependent induced fuel temperature change can be estimated using the fuel temperature reactivity coefficient from the 28 MWt/m³ design point: -3.7 pcm/K. The average fuel temperatures for these design points and estimates of changes in reactivity from the 28 MWt/m³ design point are presented in Figure 3-25.

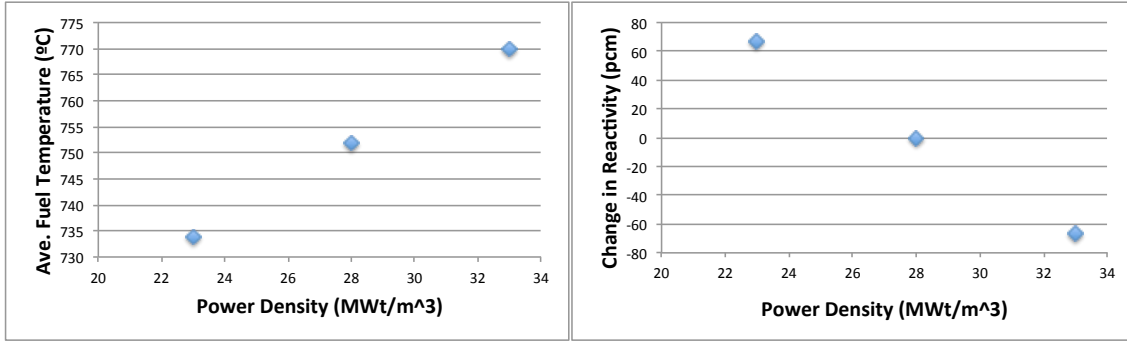


Figure 3-25. Temperature reactivity penalty as a function of power density.

To compensate for the increased leakage (i.e. reduced non-leakage probability factor)-, xenon-, and temperature- reactivity penalties in designs with increasing power density, the fuel must effectively increase the fissile content of the equilibrium state of core by discharging fuel at lower burnups, thereby increasing eta as shown in Figure 3-23. The average burnups for these design points is presented in Figure 3-26.

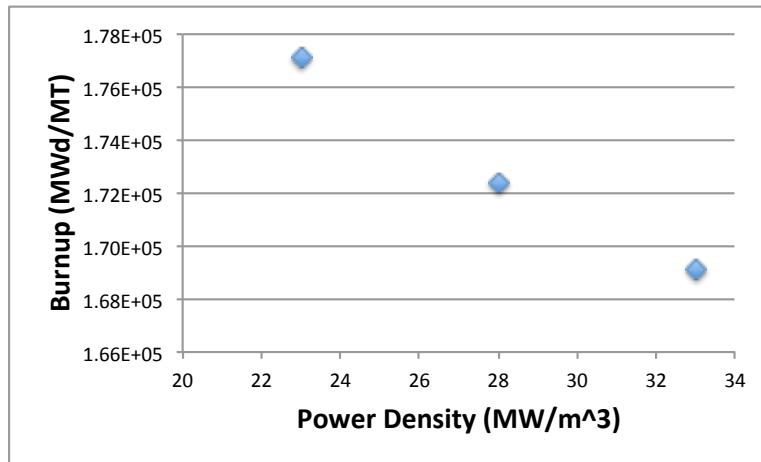


Figure 3-26. Burnup as a function of power density for the MPP PB-FHR for design points

3.3.3.2.2 Active Thickness

The active thickness is adjusted in the neutronics model by adjusting the radial dimensions of the pebble bed, the radial boundaries of the pebble reflector and outer graphite reflector and correspondingly adjusting the height of the active region to impose the design core power at a given power density. The engineering parameters of the design points to assess the affects of changing the active thickness in the PB-FHR are presented in Table 3-13.

Table 3-13. Case definitions for design points in power density reactor physics implications study

Active thickness (cm)	50	70	90
Active height (m)	5.44	3.33	2.27
Power density (MWt/m ³)	28	28	28
Pebble reflector thickness (cm)	20	20	20

The neutron economies for these design points are presented in Table 3-14 and the components of this FHR multiplication factor for these design points are presented in Figure 3-27.

Table 3-14. Neutron economy for PB-FHR for various power densities

Active Thickness (cm)	Fuel (%)	Moderator (%)	Flibe (%)	Leakage (%)
50	80.4	0.9	3.5	15.2
70	83.0	0.8	3.5	12.7
90	83.9	0.8	3.5	11.9

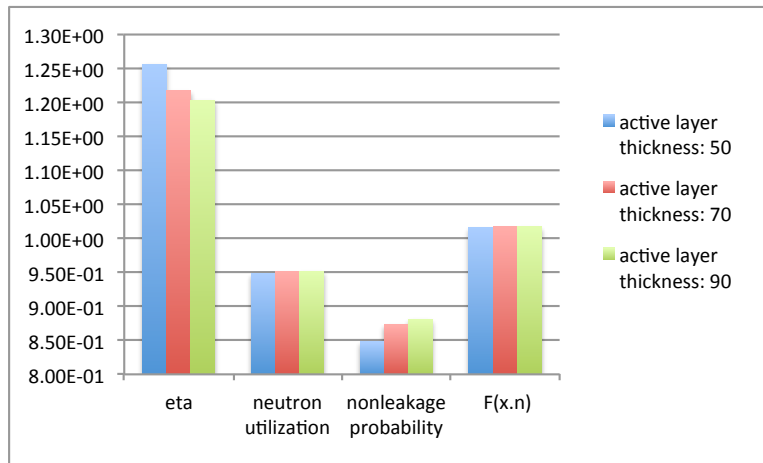


Figure 3-27. Continuous four factor components for the MPP PB-FHR for various active thicknesses.

There are competing factors that affect the trends in neutron leakage with respect to the active thickness. Large active thickness results in reduced radial leakage. However, for a fixed power level at a given power density, increases in the active region are accompanied by reductions in the effective height of the pebble bed resulting in increases in the axial leakage. Figure 17 decomposes the leakage into radial and axial components (top and bottom of defueling chute and entrance region as well as the conical surfaces of the expansion and converging region), establishing that the radial components of leakage dominate.

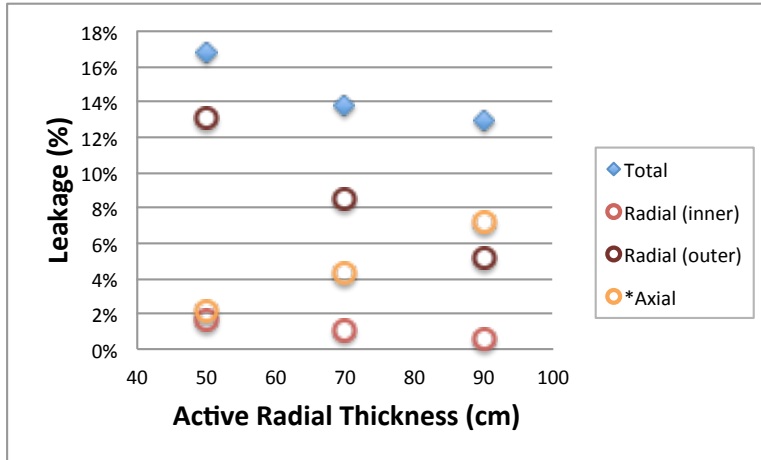


Figure 3-28. Total leakage and its components for various active radial thicknesses.

Designs with larger active thicknesses result in less moderation from the graphite reflectors. Figure 3-29 shows that increasing the active thickness hardens the neutron spectrum.

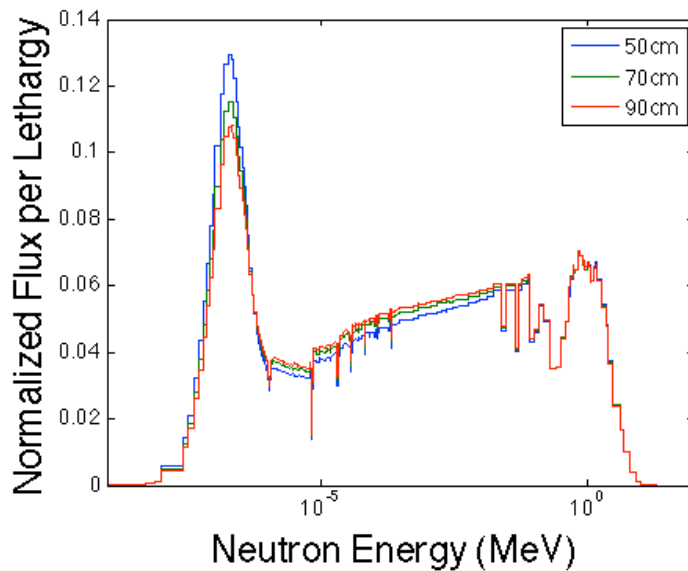


Figure 3-29. Neutron spectral comparison for PB-FHR design points with various active thicknesses.

The competing effects of reduced leakage and hardening spectrum result in non-monotonic trends in average burnup, as shown in Figure 3-30.

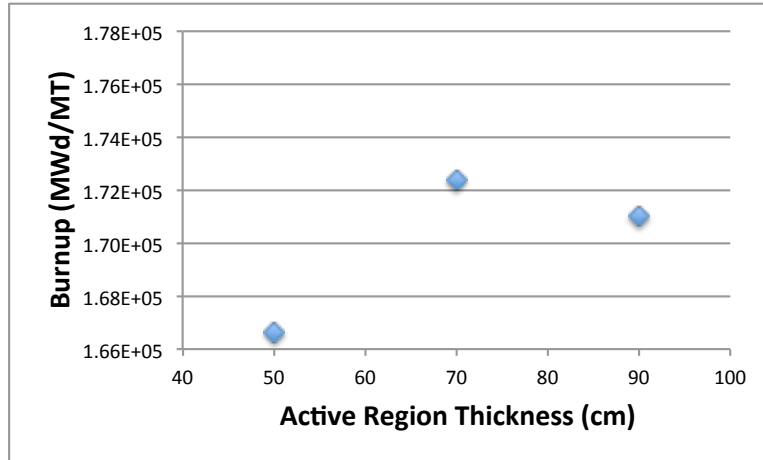


Figure 3-30. Burnup in the MPP PB-FHR for design points with various active thicknesses

3.3.3.2.3 Pebble Reflector Thickness

The pebble reflector thickness is implemented directly into the neutronics model. Increases in the pebble reflector thickness are accommodated by reducing the outer graphite reflectors inner radius. The engineering parameters of the design points to assess the affects of changing the pebble reflector thickness in the PB-FHR are presented in Table 3-15.

Table 3-15. Case definitions for design points in power density reactor physics implications study

Pebble reflector thickness (cm)	13	20	30
Power density (MWt/m ³)	28	28	28
Active thickness (cm)	70	70	70
Active height (m)	3.33	3.33	3.33

The neutron economies for these design points are presented in Table 3-16 and the components of this FHR multiplication factor for these design points are presented in Figure 3-31. Increasing the pebble reflector thickness increases the leakage from the pebble bed because the pebble reflector is a less effective reflector than the solid graphite reflector because the flibe absorbs more neutrons than graphite and increasing the pebble reflector thickness increases the volume of the pebble reflector at the expense of the outer solid reflector. Burnup decreases to compensate for the slight increase in leakage with increase pebble reflector thickness.

Table 3-16. Neutron economy for PB-FHR for various power densities

Pebble reflector thickness (cm)	Fuel (%)	Moderator (%)	Flibe (%)	Leakage (%)
13	83.6	0.8	3.5	12.2
20	83.0	0.8	3.5	12.7
30	82.6	0.8	3.5	13.1

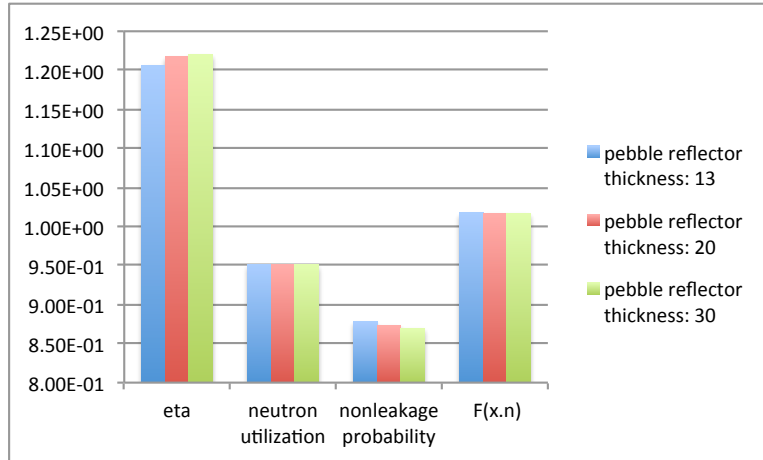


Figure 3-31. Continuous four factor components for the MPP PB-FHR for various pebble reflector thicknesses.

The pebble reflector thickness is important because it moderates and attenuates neutrons that would damage the outer solid graphite reflector that must last the life of plant. Figure 3-32 shows that increasing the pebble reflector increases the shielding of the outer graphite reflector – this should translate to increased shielding for the core barrel and reactor vessel as well. The DPA rates in Figure 3-32 are calculated with MCNP5 using DPA tallies (mt 444) assuming a displacement energy of 31 eV for the graphite.

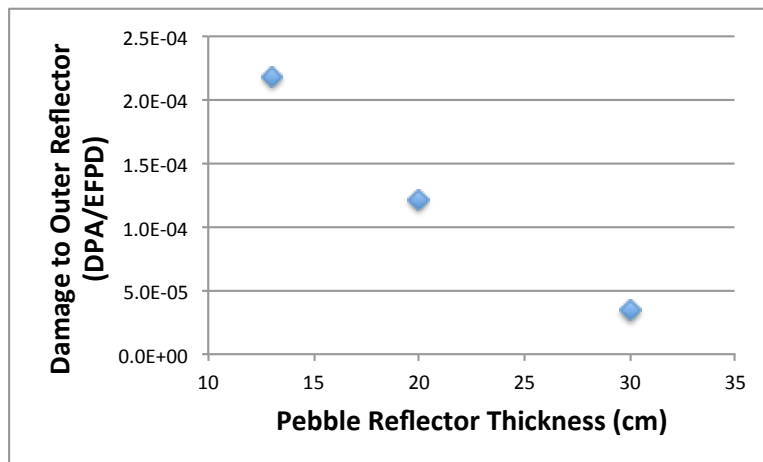


Figure 3-32. Damage to Outer reflector in the MPP PB-FHR for design points with various pebble reflector thicknesses

3.3.3.3 Sensitivity of Performance Metrics

An alternative way to interpret the results of the parametric study is to assess how the performance parameters of interest vary as a function of the set of engineering parameters. This study utilizes the Alternating Conditional Expectation (ACE) methodology to decompose the results of the parametric study into functional dependencies (optimized transforms) of the engineering parameters (i.e. predictor parameters) (Breiman & Friedman, 1985). This approach enables one to see how the

performance parameters are correlated to specific engineering parameters, enables one to make surrogate models based on these transforms and the relative magnitudes of the individual transformations provides insights to the relative sensitivity of performance parameters to specific engineering parameters

In the ACE methodology the sum of the predictor variable's transformations (ϕ_i) produce a transformed response variable (θ); this transformed response variable can be inverted into a physically meaningful value. Each transformation (for predictor- or response variables) is a set of values corresponding to each value of the predictor variable rather than an explicit algebraic function, but correlations for these transformations can be developed based on these values.

$$\theta(Y(x_0, x_1, \dots, x_n)) = \sum_{i=0}^n \phi(x_i)$$

Equation 3-17

$$\theta \approx f(Y) \therefore Y \approx f^{-1}(\theta)$$

Equation 3-18

These predictor variable transformations are estimated by first assuming arbitrary transformations and then updating the transformations for the predictor variables and then updating the transformation of the response variable. The predictor- and response- variable transformations are iteratively updated until the not explained variance no longer decreases as shown in Equation 3-19 - Equation 3-21.

$$\phi_i(x_i) = E \left[\theta(Y) - \sum_{j \neq i}^n \phi_j(x_j) \middle| x_i \right]$$

Equation 3-19

$$\theta(Y) = \frac{E \left[\sum_{i=0}^n \phi_i(x_i) \middle| Y \right]}{\left\| E \left[\sum_{i=0}^n \phi_i(x_i) \middle| Y \right] \right\|}$$

Equation 3-20

$$\sigma^2(\theta, \phi_0, \phi_1, \dots, \phi_n) = E \left[\theta(Y) - \sum_{i=1}^n \phi_i(x_i) \right]^2$$

Equation 3-21

A set of ACE transformations were estimated for each performance metric – average burnup, fuel temperature reactivity coefficient, coolant temperature reactivity coefficient, peak radiation damage rate in the inner graphite reflector, peak radiation damage rate in the core barrel, peak gas production rate in the core barrel, and peak power density in the active region (R, Z, pass) of the pebble bed – using the ACEPACK module of R. These ACE correlations were benchmarked by plotting values estimated from ACE correlations against the simulation results in Figure 3-33. The ACE correlations exhibit reasonable agreement, though for some metrics sensitive to Monte Carlo

statistics, the correlations do not agree as well due to stochastic error in the results that are passed into the ACE transformations.

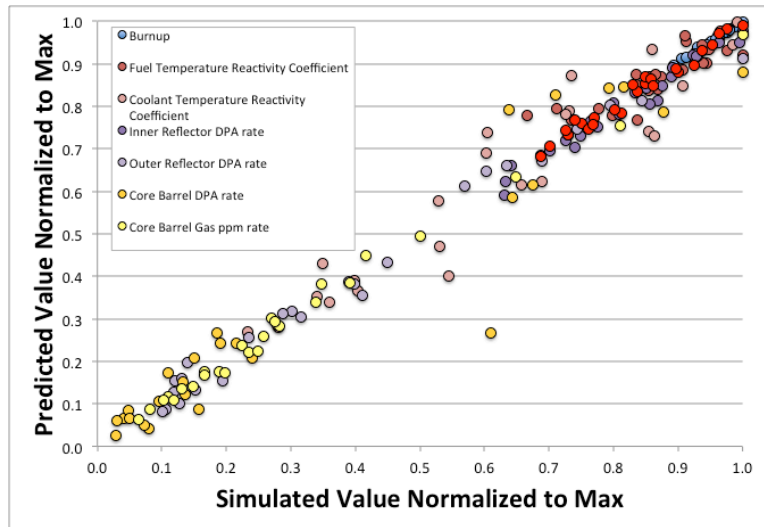


Figure 3-33. Values predicted with ACE correlations plotted against simulation results normalized to the maximum simulated result for each data set of reactor performance metric.

An added benefit of utilizing ACE transformations is that the transformations themselves give insights into how each predictor variable effects the response variable and the relative sensitivity of response variables to each predictor variable. For example, if the slope of a given predictor variable transformation is positive, then increasing that variable also increases the value of the specific response variable. The difference of the maximum and minima of each predictor variable transformation can be thought of as a proxy for relative sensitivity – this difference is larger for predictor variables that are more important to a specific response variable.

The following subsections present the ACE transformations for the performance parameters required to perform a economics optimization of the PB-FHR – this optimization is discussed in Chapter 4. Additionally, these transformations are used to estimate specific performance parameters as a function of pebble bed geometries for a give power density. Finally, the transformations are processed to give the range of the transformation to be used as a proxy for relative sensitivity of specific performance parameters to specific engineering parameters.

The design space was decomposed from power density, active thickness and pebble reflector thickness to power density, active thickness, pebble reflector thickness and active height. Including the active height parameter enables the design space to extrapolate the design space to different total power levels. However, the active thickness and active height are intimately correlated as shown in Figure 3-34. Therefore, when thinking about implications of changing the geometric parameters it is important to recognize that those changes are linked to corresponding changes in the

complimentary geometry parameters as well as the power density for a given total thermal power.

These two parameters can be considered together for cores with a single thermal power level and a single power density by presenting expected performance parameters for design points that satisfy these conditions on a pebble bed geometry plot, such as Figure 3-37.

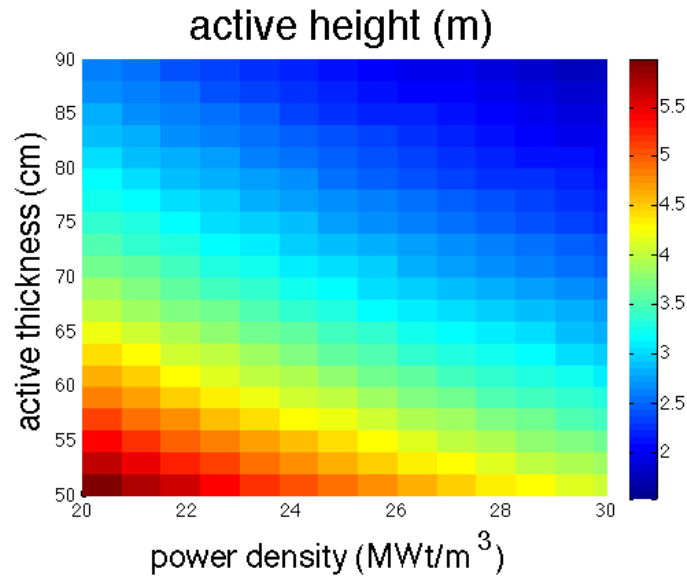


Figure 3-34. Active height for the PB-FHR as a function of active thickness and power density for 290 MWt PB-FHR.

3.3.3.3.1 Burnup

The burnup for each design point was estimated using BEAU. This average burnup is a function of the ability of the pebble bed to maintain criticality for factors that effect the neutron spectrum and neutron leakage, since these factors effect how much fissile inventory must be preserved to maintain criticality.

The ACE transformations of the engineering parameters for the burnup results for this parametric study are presented together in Figure 3-35.

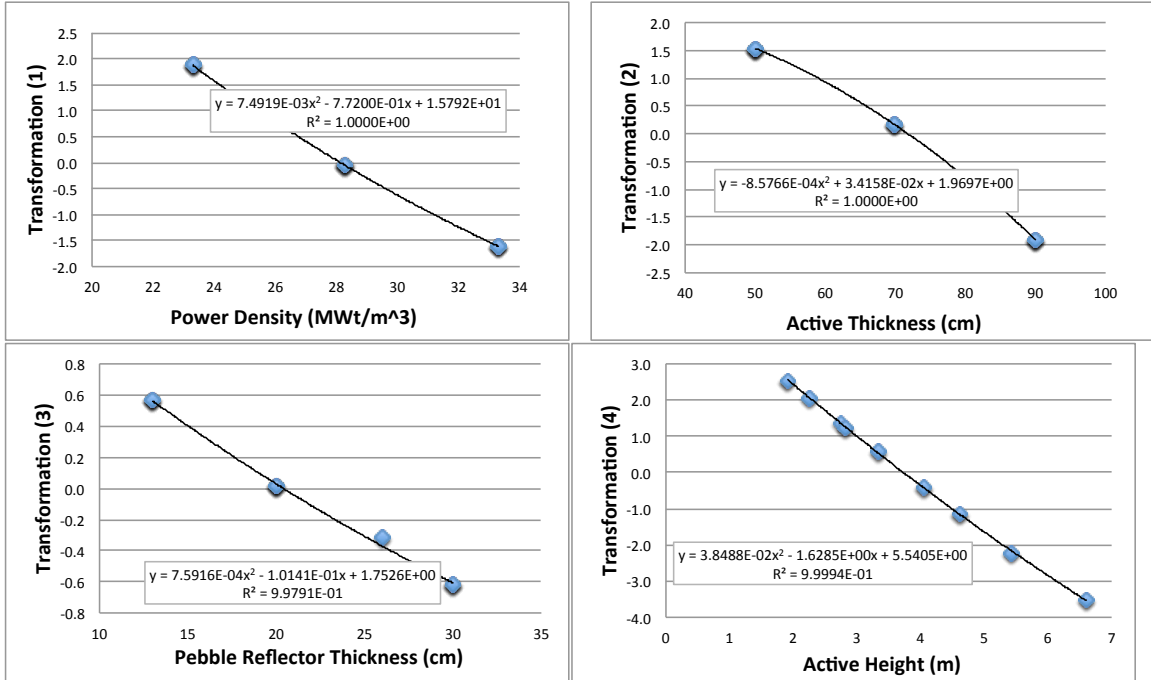


Figure 3-35. ACE predictor variable transformations for burnup: (upper left) power density transformation; (upper right) active thickness transformation (cm); (lower left) pebble reflector thickness transformation (cm); (lower right) active height transformation (m).

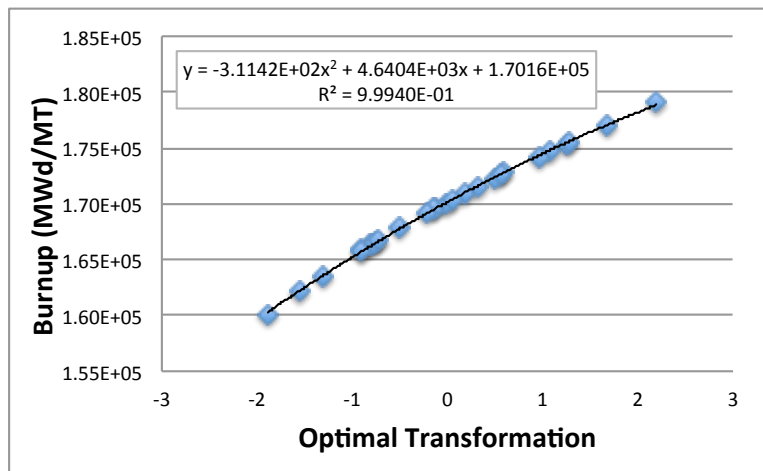


Figure 3-36. ACE burnup transformation to non-dimensional space.

Increasing power density increases the temperature of the fuel, thereby reducing burnup. Increasing the pebble reflector thickness reduces the burnup because the pebble reflector enhances leakage and hardens the neutron spectrum relative to the solid graphite reflector. The spectral component of burnup sensitivity causes the negative trend in burnup with increasing active thickness. However, the burnup penalty from increased leakage in thinner cores appears in the active height transformation due to the correlation of tall cores with thin cores as shown in Figure 3-34. The combination of these two transformations exhibit the parabolic trajectory across the active thickness dimensions shown in Figure 3-30, as shown in Figure 3-30 and Figure 3-37.

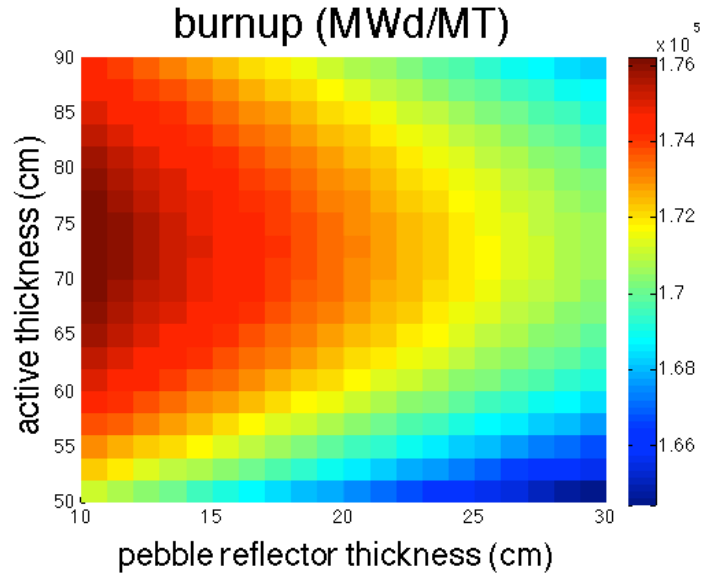


Figure 3-37. Burnup as a function of pebble bed geometry for a power density of 28 MWt/m³ an fixed thermal power of 290 MWt.

The burnup transformation distance traversed across the design space can be used as a proxy for relative sensitivity to engineering parameters. This range of transformations for burnup are presented in Figure 3-38. The burnup is most sensitive to the active height and least sensitive to the pebble reflector thickness.

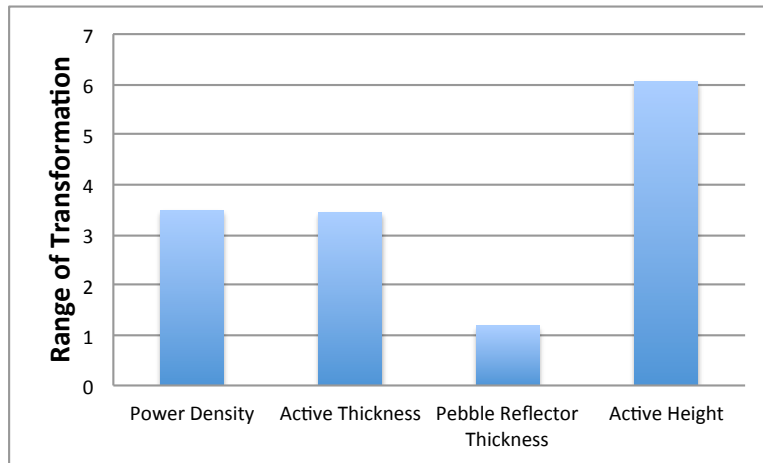


Figure 3-38. Magnitude of burnup sensitivity to engineering parameters

3.3.3.3.2 Fuel Reactivity Coefficient

At the equilibrium state of the PB-FHR, the temperature of the fuel was perturbed in the MCNP model by through adjusting the temperature cards and the cross section libraries of the fuel. The fuel temperature reactivity coefficient was calculated using Equation 3-22.

$$\alpha_{fuel} = \frac{\rho(T_{hot}^{fuel}) - \rho(T_{cold}^{fuel})}{T_{hot}^{fuel} - T_{cold}^{fuel}}$$

Equation 3-22

Negative fuel temperature reactivity feedback works by reducing the resonance escape probability because of broadened cross sections resonances. Therefore, this feedback mechanism is sensitive to the epithermal flux or put another way, design points with softer spectrums should have less negative fuel temperature reactivity coefficients. Moreover, the buildup of transuranics increases the magnitude of the negative temperature reactivity coefficients. Thus, strongly negative fuel reactivity coefficients are correlated with attributes that harden the neutron spectrum or increase the attainable burnup.

The ACE transformations of the engineering parameters for the fuel temperature reactivity coefficient results for this parametric study are presented together in Figure 3-39.

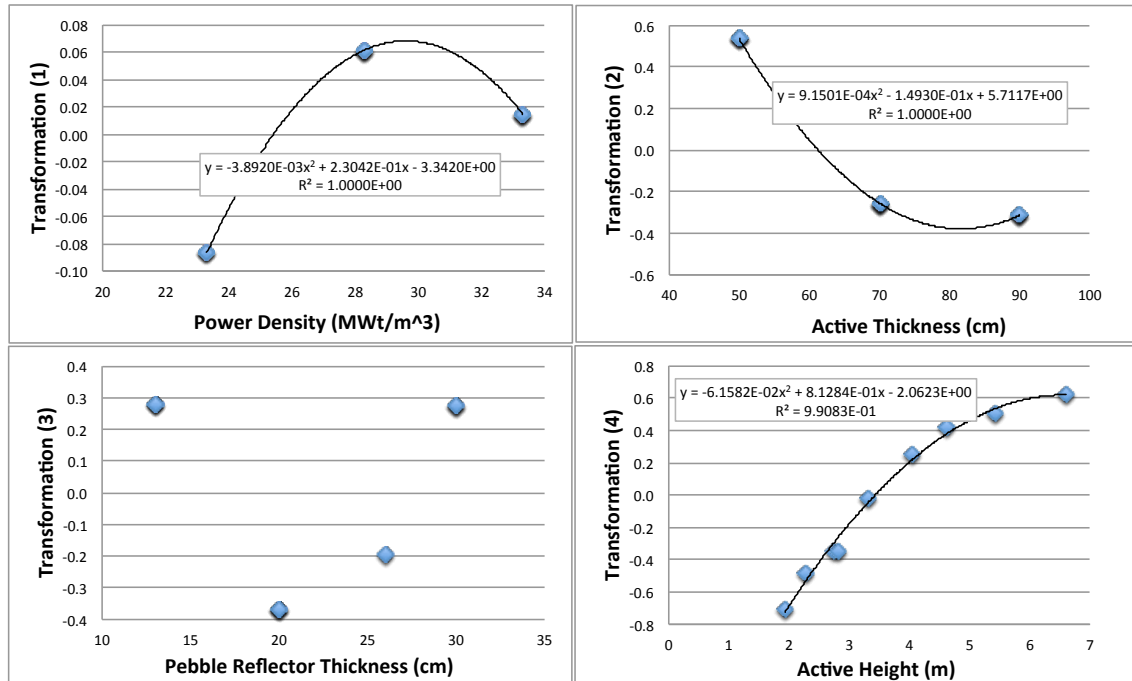


Figure 3-39. ACE predictor variable transformations for fuel temperature reactivity coefficient: (upper left) power density transformation; (upper right) active thickness transformation (cm); (lower left) pebble reflector thickness transformation (cm); (lower right) active height transformation (m).

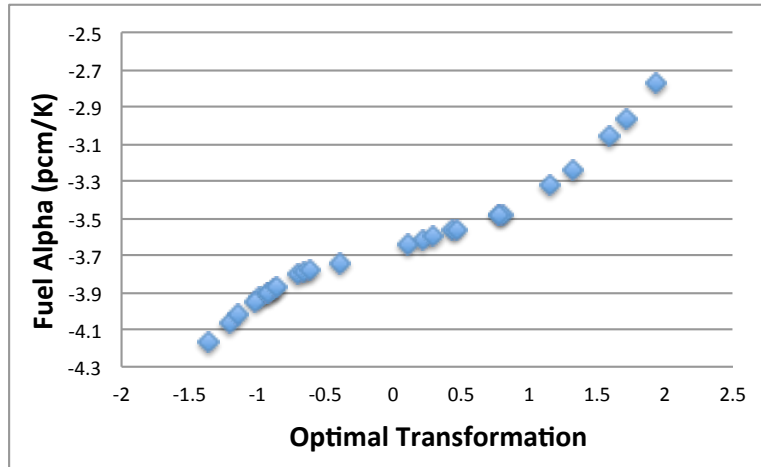


Figure 3-40. ACE fuel temperature reactivity coefficient transformation to non-dimensional space.

Increasing the active thickness hardens the neutron spectrum increasing the negativity of the fuel temperature reactivity coefficient while reducing burnup, which decreases the negativity of the fuel temperature reactivity coefficient; as a result the transformation for the active thickness has a parabolic shape. The pebble reflector thickness transformation has a parabolic shape also due to competing effects. Increased moderation from the solid reflector for designs with thin pebble reflectors softens the neutron spectrum reducing the negativity of the fuel temperature reactivity coefficient. Systems with intermediate pebble reflector thicknesses have harder spectrums and moderate burnups resulting in more negative fuel temperature reactivity coefficients. While systems with larger pebble reflector thicknesses do not burn as deeply and therefore these systems have less negative fuel temperature reactivity coefficients. Increasing the active height reduces the burnup thereby reducing the negativity of the fuel temperature reactivity coefficient. This trend is somewhat mitigated by the spectrum softening in very tall, cigar-shaped cores.

These transformations are used to estimate the fuel temperature reactivity coefficient as a function of pebble bed geometry on a constant thermal power, constant power density basis presented in Figure 3-41. All the fuel temperature reactivity coefficients are solidly negative in a relatively window between -2.7 and -4.2 pcm/K. However, these fuel temperature reactivity coefficients are not as negative as those for the infinite core in Section 3.3.1, which had a harder spectrum and higher burnup.

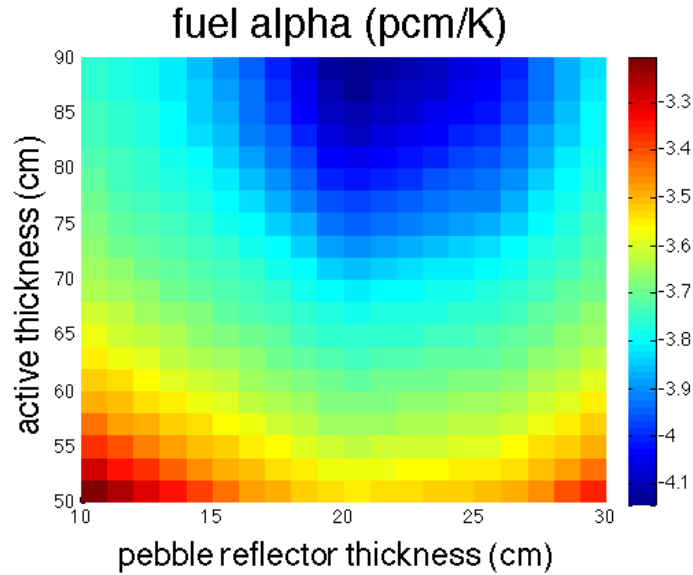


Figure 3-41. Fuel temperature reactivity coefficient as a function of pebble bed geometry for a power density of 28 MWt/m³

The range of transformations (i.e. relative sensitivity) of the fuel temperature reactivity coefficient to specific engineering parameters is presented in Figure 3-42.

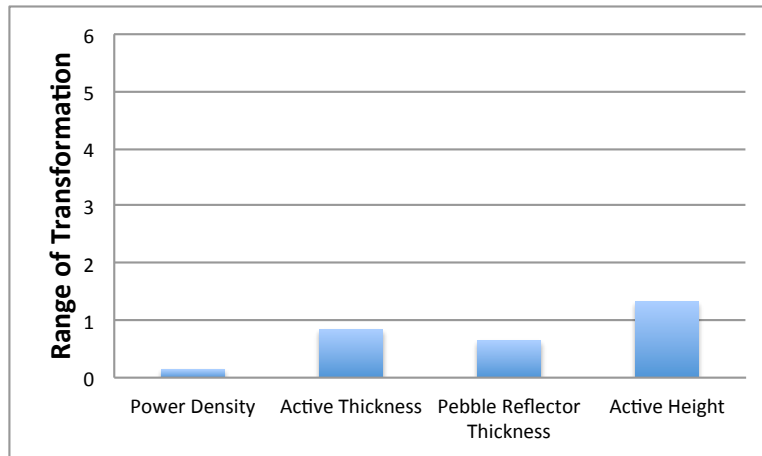


Figure 3-42. Magnitude of fuel temperature reactivity coefficient sensitivity to engineering parameters

3.3.3.3.3 Coolant Reactivity Coefficient

At the equilibrium state of the PB-FHR, the temperature of the coolant was perturbed in the MCNP model through adjusting the density of the coolant, the temperature cards and the cross section libraries of the coolant. The coolant temperature reactivity coefficient was calculated using Equation 3-23.

$$\alpha_{coolant} = \frac{\rho(T_{hot}^{coolant}) - \rho(T_{cold}^{coolant})}{T_{hot}^{coolant} - T_{cold}^{coolant}}$$

Equation 3-23

The ACE transformations of the engineering parameters for the coolant temperature reactivity coefficient results for this parametric study are presented together in Figure 3-43.

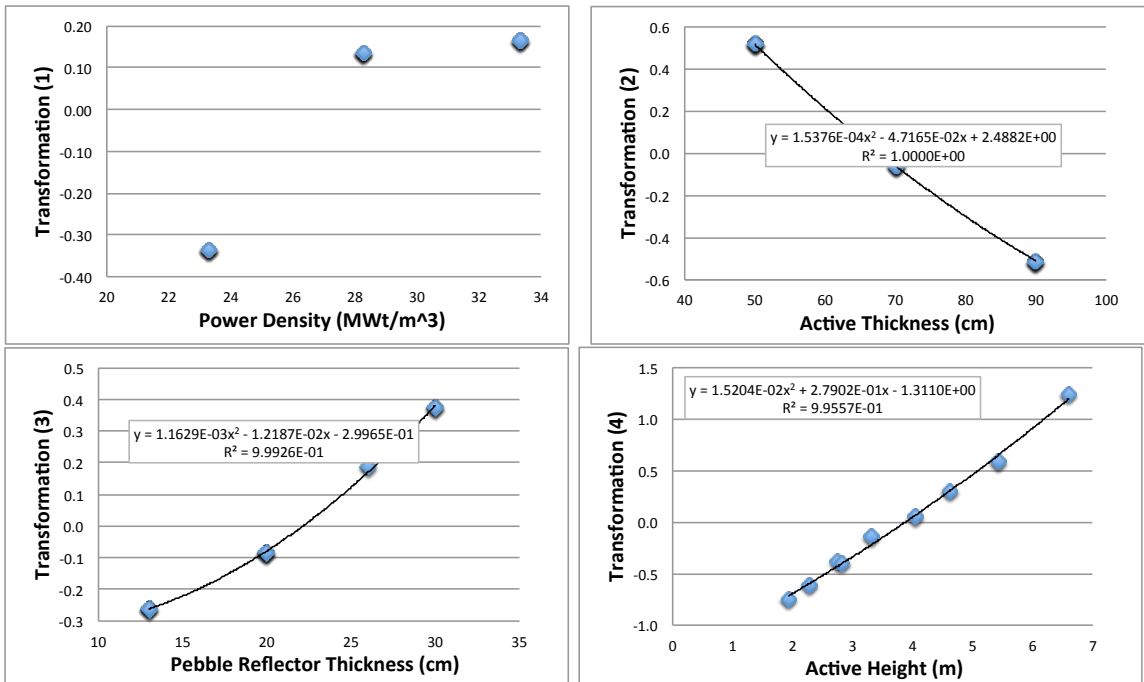


Figure 3-43. ACE predictor variable transformations for coolant temperature reactivity coefficient: (upper left) power density transformation; (upper right) active thickness transformation (cm); (lower left) pebble reflector thickness transformation (cm); (lower right) active height transformation (m).

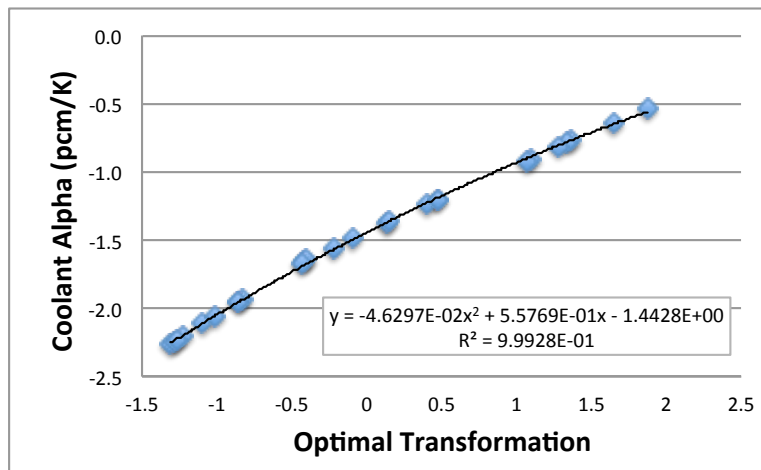


Figure 3-44. ACE coolant temperature reactivity coefficient transformation to non-dimensional space.

As stated earlier coolant reactivity feedback has three mechanisms initiated by coolant voiding: removing moderating atoms, removing neutron absorbers, and allowing neutrons to leak out of the pebble bed. Designs with large active thicknesses have harder neutron spectrums and are more reliant on the coolant for neutron moderation and are less sensitive to the thermal absorption in ⁶Li. Therefore, increasing active

thickness increases the negativity of the coolant temperature reactivity coefficient. This same mechanism causes the coolant temperature reactivity coefficient to become less negative as the height of the core increases and the neutron spectrum softens. Increasing the thickness of the pebble reflector reduces the sensitivity of the system to increased leakage due to coolant voiding and as a result the coolant temperature reactivity coefficient becomes less negative with increasing pebble reflector thickness. The power density has little effect of the coolant temperature reactivity coefficient.

These transformations are used to estimate the fuel temperature reactivity coefficient as a function of pebble bed geometry on a constant thermal power, constant power density basis presented in Figure 3-45. All the coolant temperature reactivity coefficients are also solidly negative. The increased contribution of the leakage feedback mechanism in the smaller 290 MWt PB-FHR could explain why the coolant temperature reactivity coefficients are much more negative for the 290 MWt core than those for the larger 900 MWt core.

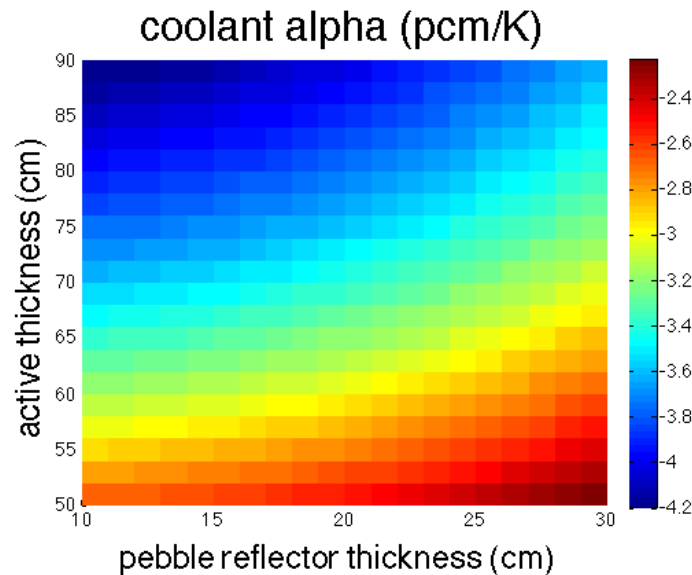


Figure 3-45. coolant temperature reactivity coefficient as a function of pebble bed geometry for a power density of 28 MWt/m³

The range of transformations, the proxy for relative sensitivity, of the fuel temperature reactivity coefficient to specific engineering parameters is presented in Figure 3-46.

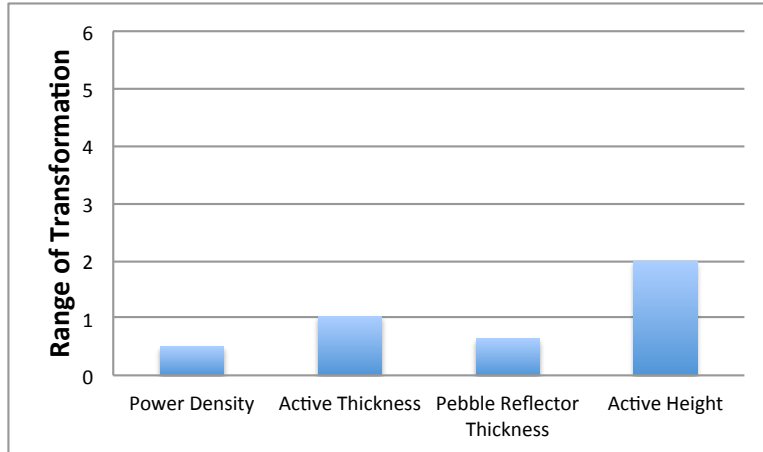


Figure 3-46. Magnitude of coolant temperature reactivity coefficient sensitivity to engineering parameters

3.3.3.3.4 Inner Reflector Radiation Damage

The peak damage rate in the inner graphite reflector is calculated at the equilibrium state of the PB-FHR. The damage rate is tallied with MCNP5 on a mesh structure (10 radial nodes and 20 axial nodes) over the first 10cm of the inner reflector adjacent to the active region of the pebble bed. The damage rate for each node is calculated and the maximum damage rate is recorded as the limiting inner reflector radiation damage rate for each point design.

The ACE transformations of the engineering parameters for the peak inner reflector damage rate results for this parametric study are presented together in Figure 3-47.

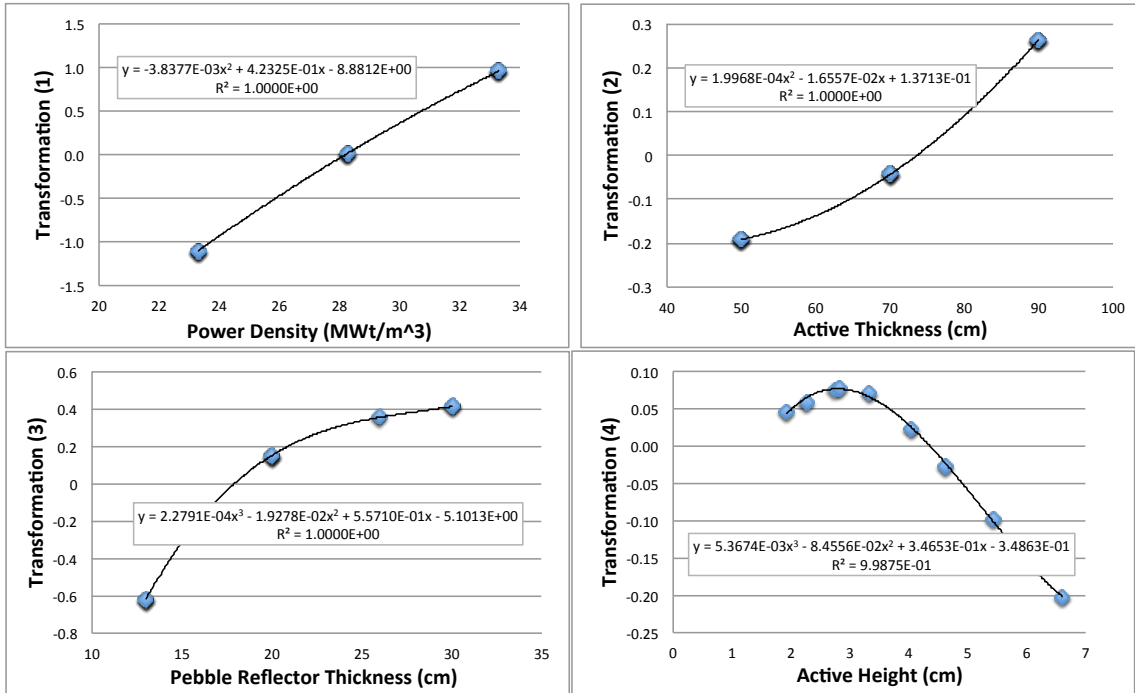


Figure 3-47. ACE predictor variable transformations for inner reflector radiation damage: (upper left) power density transformation; (upper right) active thickness transformation (cm); (lower left) pebble reflector thickness transformation (cm); (lower right) active height transformation (m).

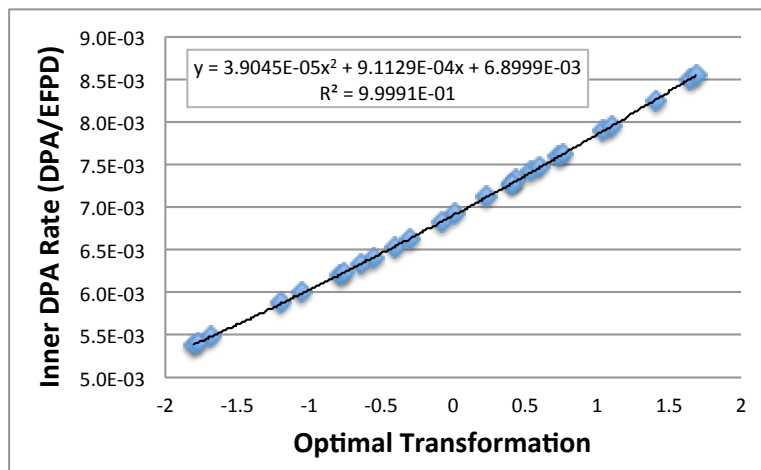


Figure 3-48. ACE inner reflector radiation damage rate transformation to non-dimensional space.

Increasing the power density increases the neutron flux and thereby directly increases the damage rate. Increasing the active region hardens the neutron spectrum thereby increasing the damage rate. Increasing pebble reflector thickness reduces the outer reflector moderation shifting the power distribution towards the inner graphite reflector thereby increasing the dose rate. The active height of the pebble bed has little influence on the inner reflector damage rate.

The transformations are used to estimate inner reflector dose rate as a function of pebble bed geometry on a constant thermal power, constant power density basis

presented in Figure 3-49. These DPA rates result in service lifetimes on the order of 10 EFPY.

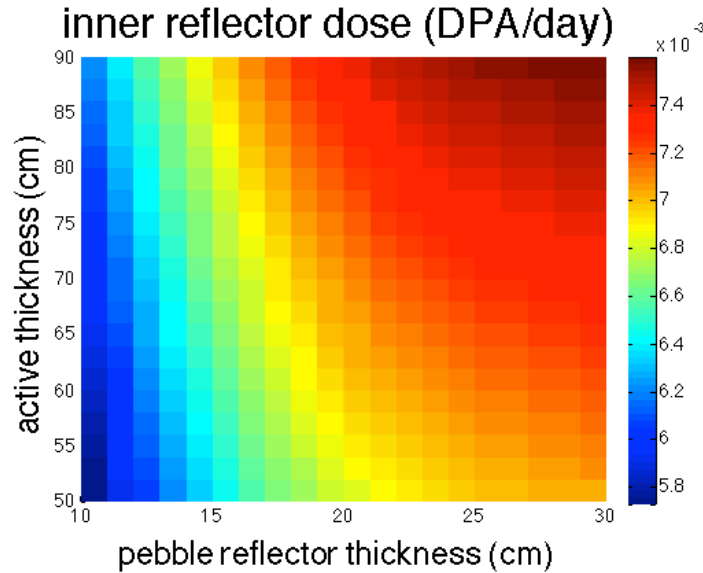


Figure 3-49. Inner reflector radiation damage rate as a function of pebble bed geometry for a power density of 28 MWt/m³

The range of transformations (i.e. relative sensitivity) of the fuel temperature reactivity coefficient to specific engineering parameters is presented in Figure 3-50.

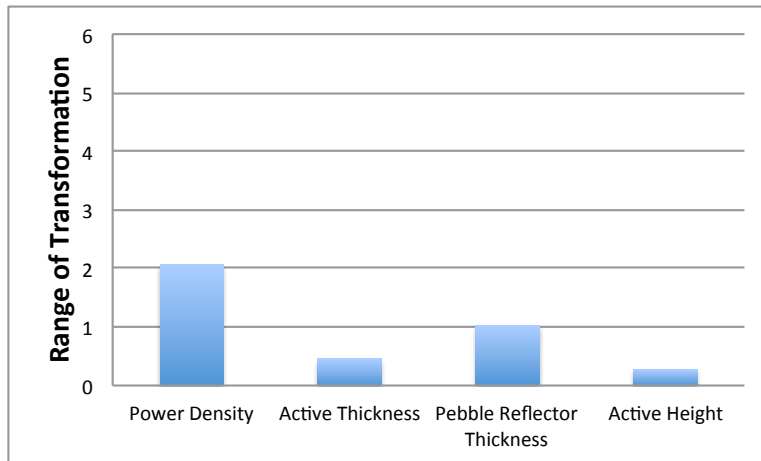


Figure 3-50. Magnitude of inner reflector radiation damage rate sensitivity to engineering parameters

3.3.3.3.5 Outer Reflector Radiation Damage

The peak damage rate in the outer graphite reflector is calculated like the damage rate in inner graphite reflector. The damage rate is tallied with MCNP5 on a similar mesh structure (10 radial nodes and 20 axial nodes) over the first 10cm of the outer reflector adjacent to the active region of the pebble bed – some designs are tallied over entire thickness if outer reflector if it is less than 10cm thick. The damage rate for each node is

calculated and the maximum damage rate is recorded as the limiting outer reflector radiation damage rate for each point design.

The ACE transformations of the engineering parameters for the peak outer reflector damage rate results for this parametric study are presented together in Figure 3-51.

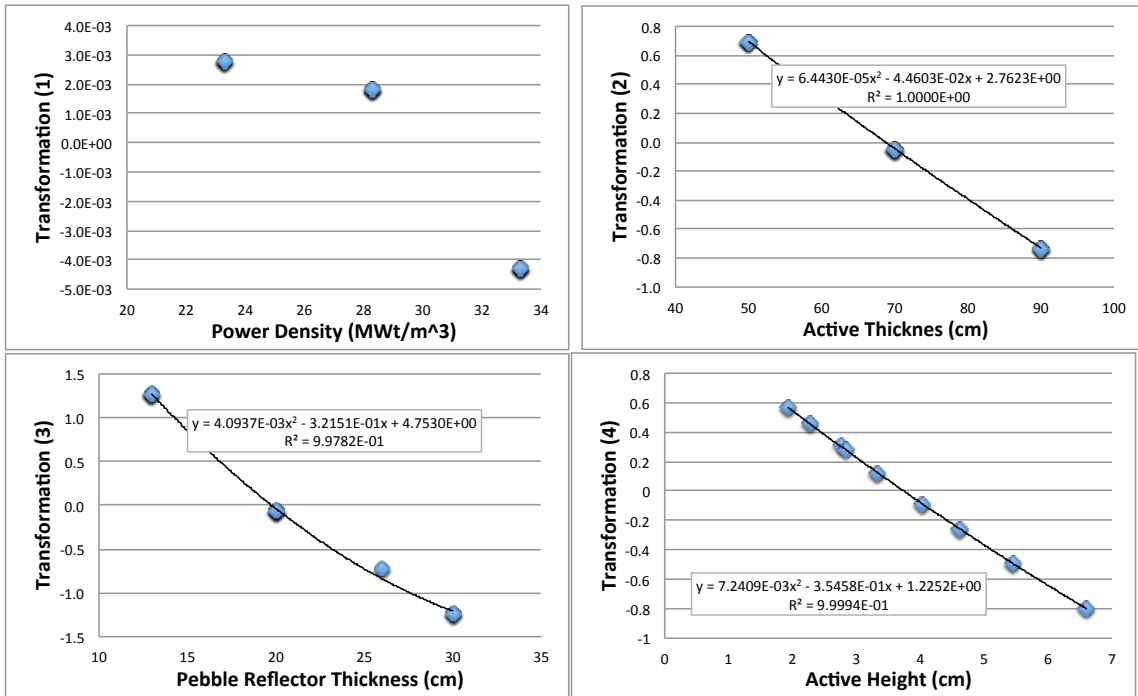


Figure 3-51. ACE predictor variable transformations for outer reflector radiation damage: (upper left) power density transformation; (upper right) active thickness transformation (cm); (lower left) pebble reflector thickness transformation (cm); (lower right) active height transformation (m).

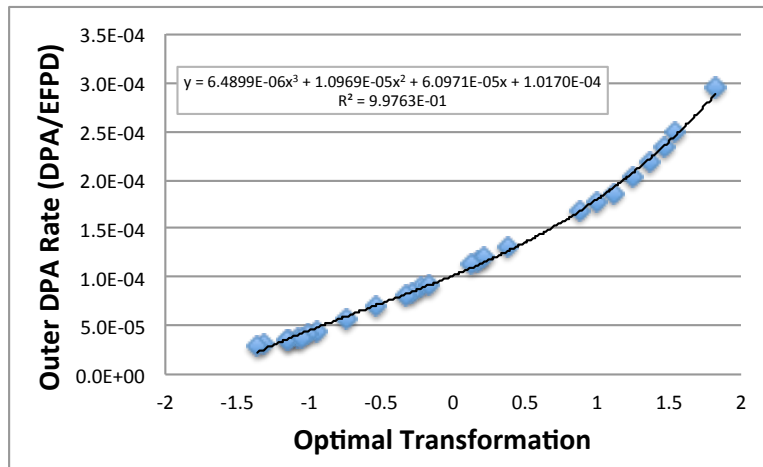


Figure 3-52. ACE outer reflector radiation damage rate transformation to non-dimensional space.

The pebble reflector thickness shields the outer graphite reflector and moves the solid reflector away from the high-energy neutron source. Increasing the active thickness reduces damage to the outer graphite reflector because the fission neutrons born in the

center of the core are shielded by the periphery of this active region. The trend in power density is counterintuitive because it seems relatively constant. However, the increase in the outer reflector damage rate with increasing power density dependency appears via the active height transformation, as short cores are correlated with high power densities; see Figure 3-53. The outer reflector radiation damage rate distributions for several power densities presented in this figure and show that increasing power density slightly increases the outer reflector dose rate.

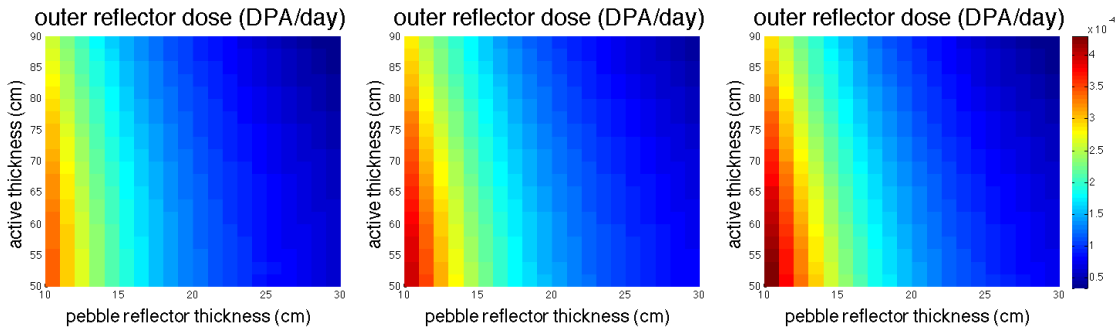


Figure 3-53. Outer reflector radiation damage rate as a function of pebble bed geometry for a thermal power of 290 MWt: (left) power density at 23 MWt/m³; (center) power density at 28 MWt/m³; (right) power density at 33 MWt/m³.

The range of transformations (i.e. relative sensitivity) of the fuel temperature reactivity coefficient to specific engineering parameters is presented in Figure 3-54. It is surprising to see the outer reflector radiation damage rate is not strongly dependent on the power density as is the case for the damage rate to the inner reflector.

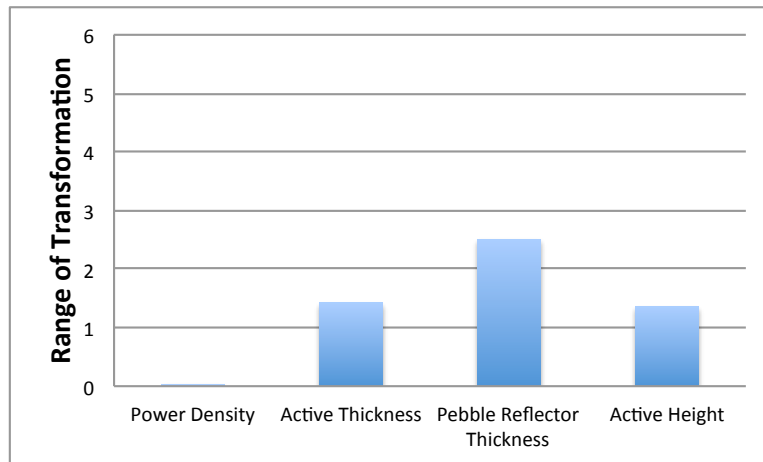


Figure 3-54. Magnitude of outer reflector radiation damage rate sensitivity to engineering parameters

3.3.3.3.6 Core Barrel Radiation Dose

The peak damage rate in the core barrel is calculated like the damage rates in graphite reflectors. The damage rate is tallied with MCNP5 on a similar mesh structure (10 radial nodes and 20 axial nodes) over the entire thickness of the core barrel in the axial extent of the active region of the pebble bed. The damage rate for each node is calculated and

the maximum damage rate is recorded as the limiting outer reflector radiation damage rate for each point design.

The ACE transformations of the engineering parameters for the peak core barrel radiation damage rate results for this parametric study are presented together in Figure 3-55.

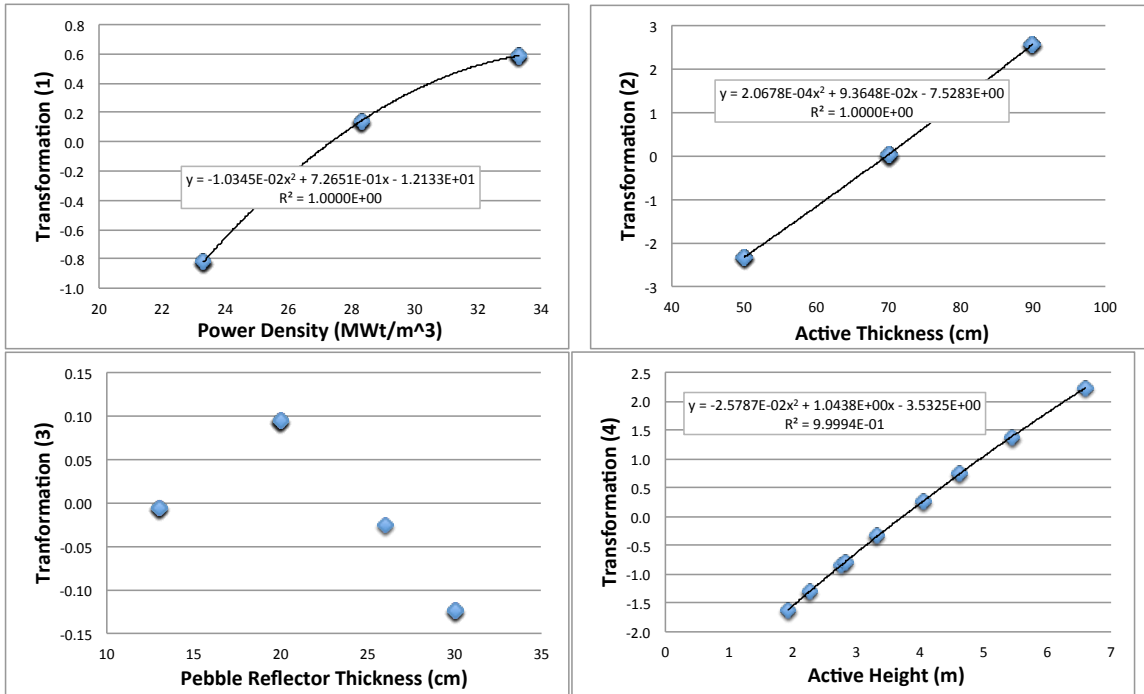


Figure 3-55. ACE predictor variable transformations for core barrel radiation damage: (upper left) power density transformation; (upper right) active thickness transformation (cm); (lower left) pebble reflector thickness transformation (cm); (lower right) active height transformation (m).

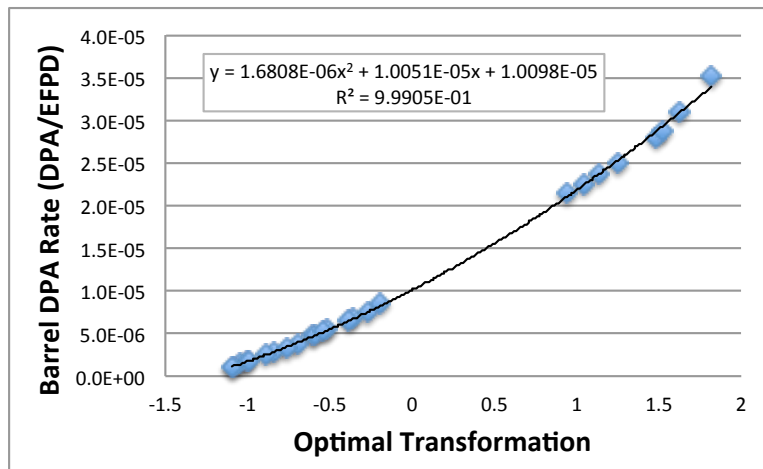


Figure 3-56. ACE core barrel radiation damage rate transformation to non-dimensional space.

Unlike the outer reflector, the position of the core barrel is fixed. The core barrel dose increases with increasing active thickness because this moves the source of high-energy

neutrons closer to the core barrel. The pebble reflector does not significantly effect the attenuation of high-energy neutrons because increasing the pebble reflector thickness is in effect swapping an excellent moderator – graphite – for flibe, another neutron moderator in the interstitial volume of the pebble reflector. Core barrel dose increases marginally with power density.

The transformations are used to estimate the core barrel damage rate as a function of pebble bed geometry on a constant thermal power, constant power density basis presented in Figure 3-57.

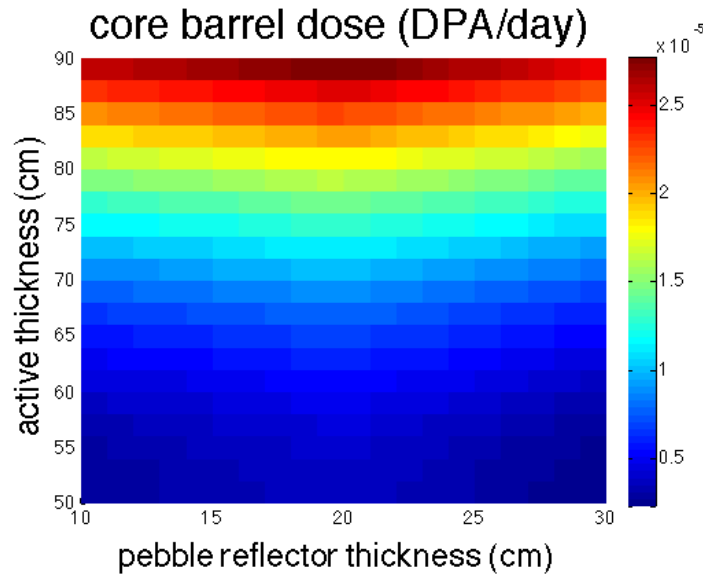


Figure 3-57. Core barrel radiation damage rate as a function of pebble bed geometry for a power density of 28 MWt/m³

The range of transformations, the proxy for relative sensitivity, of the fuel temperature reactivity coefficient to specific engineering parameters is presented in Figure 3-58.

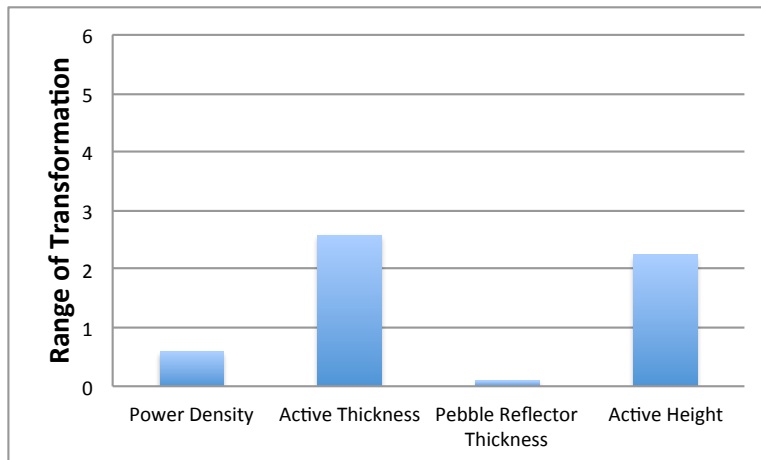


Figure 3-58. Magnitude of core barrel radiation damage rate sensitivity to engineering parameters

3.3.3.3.7 Core Barrel Gas Production

The peak gas production rate in the core barrel is calculated like the damage rates in graphite reflectors. The total gas production – the sum of (n,H)-, (n,D)-, (n,T)-, (n,³He)- and (n, α) reactions (i.e. mt reaction numbers 203, 204, 205, 206 and 207) – is tallied with MCNP5 on the same mesh structure used to tally the damage in the core barrel. The gas production rate for each node is calculated and the maximum gas production rate is recorded as the limiting outer reflector radiation damage rate for each point design.

The ACE transformations of the engineering parameters for the peak core barrel gas production rate results for this parametric study are presented together in Figure 3-59.

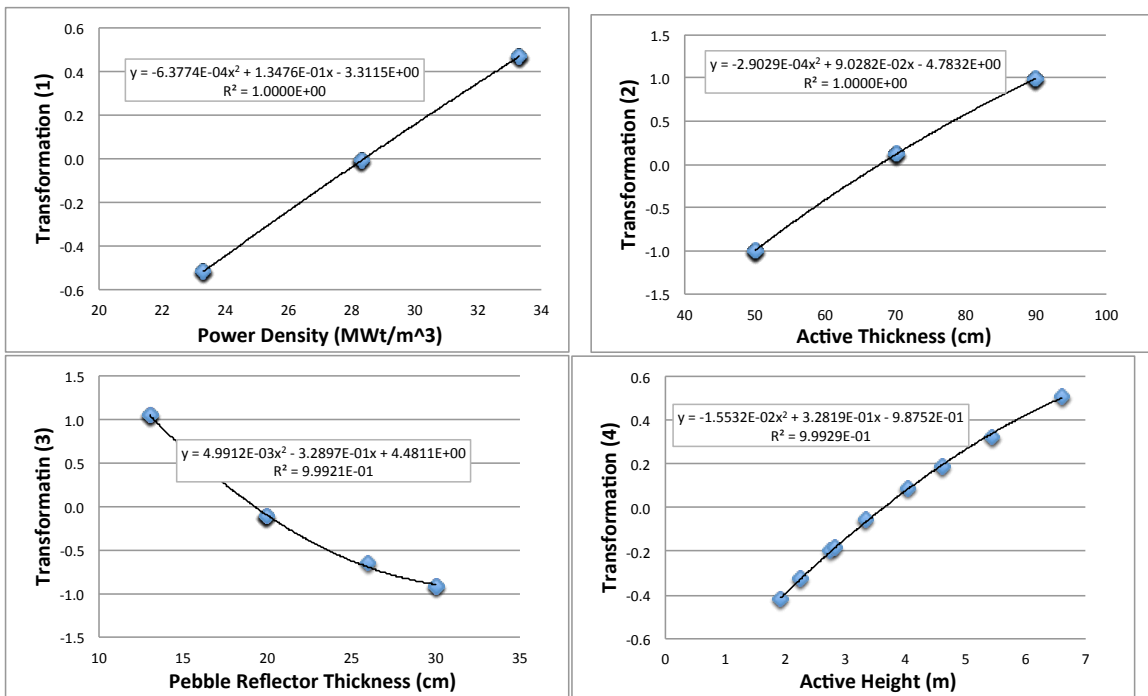


Figure 3-59. ACE predictor variable transformations for core barrel gas production: (upper left) power density transformation; (upper right) active thickness transformation (cm); (lower left) pebble reflector thickness transformation (cm); (lower right) active height transformation (m).

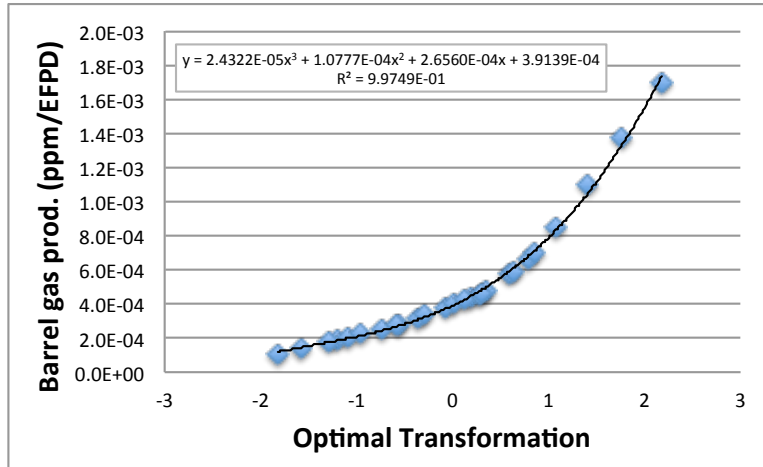


Figure 3-60. ACE core barrel gas production rate transformation to non-dimensional space.

Most of the trends are consistent between the peak core barrel radiation damage and gas production because at a fundamental level these are both measures of high energy neutron fluence. However, gas production reactions require the highest energy neutrons; basically, a fission-born neutron has to stream directly to the core barrel and then knock-off a small particle – H,D,T,³He or α . Flibe is more effective at shielding these high energy neutrons because of the high energy resonances in ¹⁹F and ⁹Be. Therefore, the gas production rate in the core barrel is sensitive to the thickness of the pebble reflector layer while the DPA rate is not.

The transformations are used to estimate the core barrel gas production rate as a function of pebble bed geometry on a constant thermal power, constant power density basis presented in Figure 3-61.

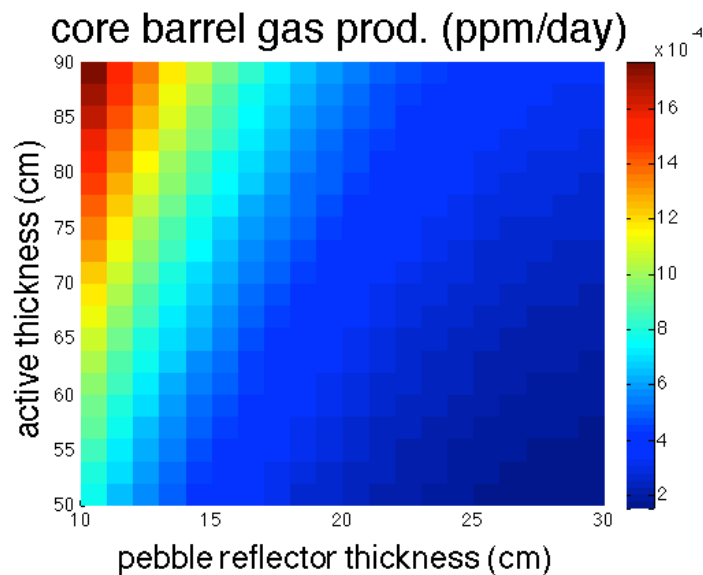


Figure 3-61. Core barrel gas production rate as a function of pebble bed geometry for a power density of 28 MWt/m³

The range of transformations, the proxy for relative sensitivity, of the fuel temperature reactivity coefficient to specific engineering parameters is presented in Figure 3-62.

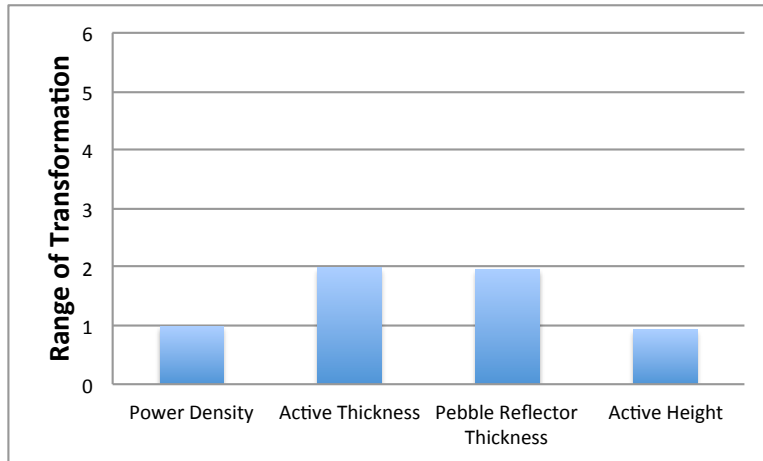


Figure 3-62. Magnitude of core barrel gas production rate sensitivity to engineering parameters

3.3.3.3.8 Peak Power Density

The peak power density is calculated at the equilibrium state of the PB-FHR. The 238-group flux distribution a complex mesh structure in the active region of the pebble bed: the pebble bed is subdivided into 20 axial nodes, each of the four radial depletion zones is subdivided into 10 radial nodes and the energy is divided into 238 energy groups. Corresponding 238-group macroscopic fission cross sections are calculated the first burnup state in the expansion region (one burnup state earlier in the first fuel progression); cross sections from the first fuel progression are used to capture the “burnup” peaking in addition to geometric peaking. The multi-group flux is crossed with the multi-group macroscopic cross sections and multiplied by an assumed fission energy of 200MeV resulting in a conservatively high local power density for each node in the mesh. The peak power density is the largest of these local low-burnup power densities. This peak power density will determine the maximum sustained fuel temperature.

The ACE transformations of the engineering parameters for the peak power density results for this parametric study are presented together in Figure 3-63.

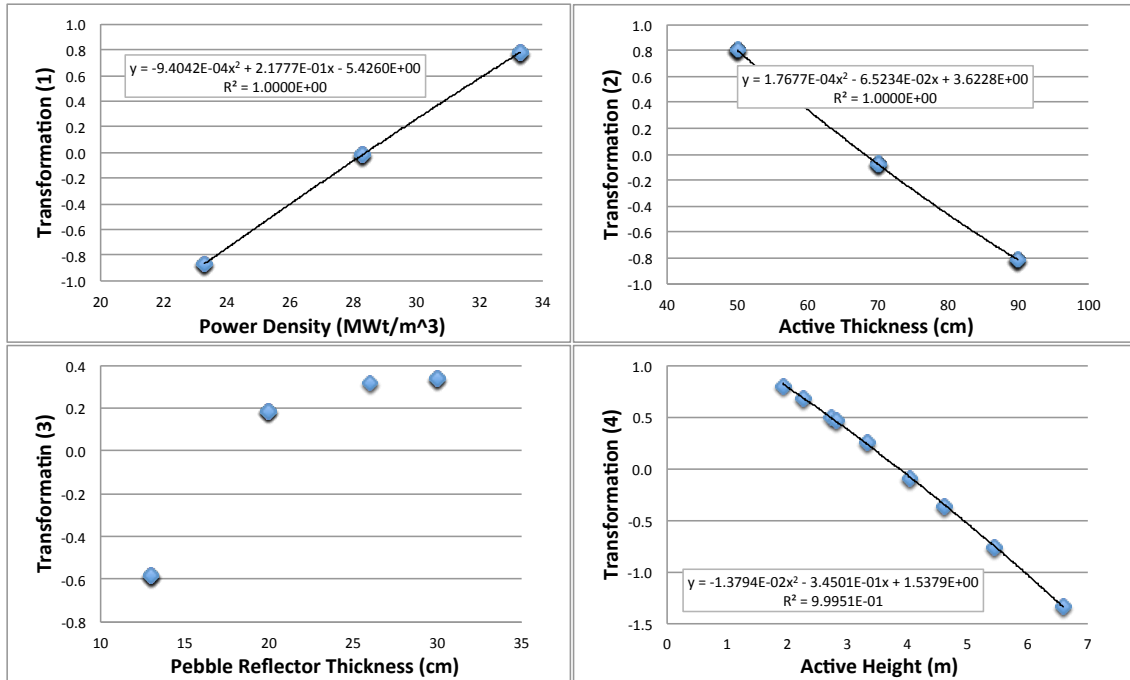


Figure 3-63. ACE predictor variable transformations for peak power density: (upper left) power density transformation; (upper right) active thickness transformation (cm); (lower left) pebble reflector thickness transformation (cm); (lower right) active height transformation (m).

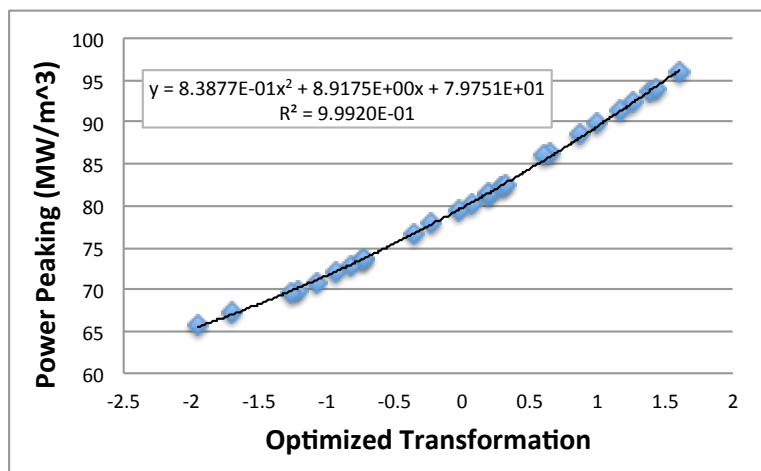


Figure 3-64. ACE peak power density transformation to non-dimensional space.

Increasing power density linearly increases the peak power. Increasing the active thickness moves the core volume away from the central reflector reducing the radial peaking. Increasing the volume of the outer pebble reflector reduces the power density near the periphery of the pebble bed preventing the peripheral power density peaking from balancing the power peaking in the pebble bed. The tall cores are correlated with higher volumes and thereby lower power density and thereby lower peak power; this correlation is reflected in the transformation of the active height.

The transformations are used to estimate the peak power densities as a function of pebble bed geometry on a constant thermal power, constant power density basis presented in Figure 3-65.

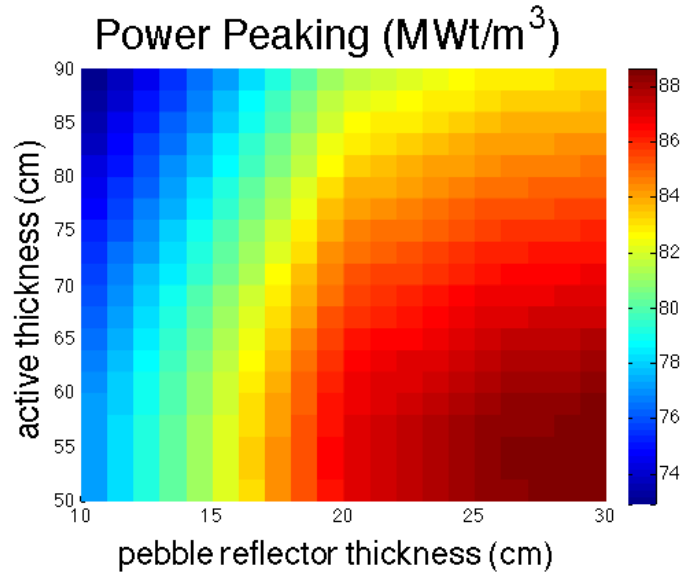


Figure 3-65. Peak power density as a function of pebble bed geometry for a power density of 28 MWt/m³

The range of transformations, the proxy for relative sensitivity, of the fuel temperature reactivity coefficient to specific engineering parameters is presented in Figure 3-66.

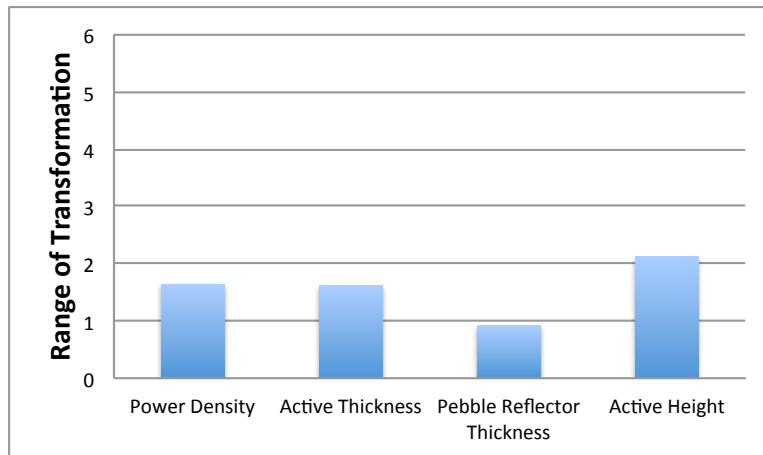


Figure 3-66. Magnitude of peak power density sensitivity to engineering parameters

3.4 Conclusions

A broad design space for FHR PBRs was explored using the equilibrium depletion methods developed in Chapter 2. Utilizing the design tool, FIMPS, to quickly and repeatably produce neutronics and depletion models for FHR PBR parametric design studies.

The pebble design study determined that the C/HM ratio must be selected carefully to impose negative coolant temperature reactivity coefficients – which strongly influences ATWS response. Based on the results from this pebble design study a baseline pebble fuel design was adopted.

Seed and thorium blanket-, seed and inert pebble reflector- and seed only core configurations were investigated for annular FHR PBRs. Either a thorium blanket or graphite pebble reflector is required to shield the outer graphite reflector enough to extend its service lifetime to 60 EFPY. The fuel fabrication costs and long cycle lengths of the thorium blanket fuel limit the potential economic advantages of using a thorium blanket. Therefore, the seed and pebble reflector core configuration was adopted as the baseline core configuration.

The design space of an SMR FHR PBR with the fuel design developed in section 3.3.1 and the seed and inert graphite pebble reflector core configuration developed in section 3.3.2 was explored in section 3.3.3, mapping the sensitivity of high level neutronic properties (flux spectrum, burnup, leakage, etc.) to geometry perturbations and developing correlations of reactor performance parameters (burnup, reactivity coefficients, radiation damage rates, peak power, etc.) as a function of the design parameters (power density, active thickness, pebble reflector thickness and active height) within the explored design space – these correlations are implemented in Chapter 4 to develop a surrogate model of the PB-FHR and optimize the PB-FHR simultaneously for safety and economics.

4 PB-FHR Core Safety and Economics Optimization

4.1 Introduction

The aim of this chapter is to optimize the PB-FHR's core design. There are multiple objectives to satisfy when designing nuclear reactors. No nuclear reactor will be built unless it presents a strong safety case. At the same time, no one will finance a nuclear reactor in a free market unless it is economically competitive with alternative energy generating options. Therefore, the optimized PB-FHR must simultaneously be designed to exhibit excellent safety characteristics and superior economics.

The first section of this chapter identifies a set of safety criteria to limit the design space to only instances of the PB-FHR that provide excellent safety. Then a simple cash flow model of a PB-FHR project is presented to assess the economic impacts of design decisions; this framework has been developed into a simulation tool – Nuclear Application Valuation Executable (NAVE). Then NAVE is used to explore the feasible design space of the PB-FHR and perform sensitivity analysis to several economic assumptions with a high level of uncertainty.

4.2 Safety Requirements

The philosophy around the safety and licensing basis for the FHR integrated research project is presented in the first FHR white paper. (Cisenros, Laufer, Scarlat, Zweibaum, & Seifried, 2012) This white paper describes a set of six FHR safety regulatory design criteria (RDC) from which all the safety-based functional requirements are derived:

1. Maintain control of radionuclides
2. Control heat generation (reactivity)
3. Control heat removal and addition
4. Control primary coolant inventory
5. Maintain core and reactor vessel geometry
6. Maintain reactor building structural integrity

The system function requirements presented in the white paper were reviewed in the subsequent sections to establish quantifiable physical design criteria or limitations for core design optimization. Tables identifying the functional requirements of specific systems and subsystem from this white paper are reproduced in this section. Moreover, these tables classify specific functional requirements by their relevance to regulatory design criteria (RDC), economics, investment protection (IP), and/or safeguards (SG) – the functions relative to RDCs are highlighted.

4.2.1 Fuel

Table 4-1 presents a summary of functional requirements for FHR fuel. Interfacing with the primary coolant is a function of convective heat transfer – corrosion and primarily outside the realm of the core design optimization. Providing barrier to radionuclides dictates that the TRISO particles fair well with respect to fuel performance. Detailed fuel performance analysis will be required to definitively assess the failure rate of these

particles in the PB-FHR. But without a fuel performance code, a peak fuel temperature limit of 1200 °C is assumed as a fuel performance constraint. The stability of the power level and shape will depend on the feedback mechanisms in the core. As transient analysis is beyond the scope of this work, negative fuel- and coolant reactivity coefficients are used as a proxy for the stability of power level and power shape.

Table 4-1. Summary of functional requirements for FHR fuel.

FUEL SUBSYSTEM FUNCTIONAL REQUIREMENTS	
1. supply heat for power conversion system	<i>economics</i>
2. be feasible to manufacture	<i>economics</i>
3. minimize energy output normalized fuel cycle costs	<i>economics</i>
4. interface with primary coolant system	<i>economics, RDC3</i>
5. interface with fuel handling system	<i>economics</i>
6. provide barrier to radionuclides generated in fuel kernel	<i>RDC 1</i>
7. have stable power level and power shape under anticipated occurrences	<i>RDC 2, RDC 1</i>
8. respond gently in transients events	<i>IP</i>
9. fuel enrichment	<i>SG, PS</i>

4.2.2 Primary Coolant

Table 4-2 presents a summary of functional requirements for the primary coolant. There are positive and negative components of the reactivity feedback from the coolant that can lead to systems with overall positive coolant temperature reactivity feedback. Therefore, to assure stable feedback it is important to intentionally design the core to have negative coolant temperature reactivity coefficients. The other safety related functional requirements relate to the thermal hydraulics design of the primary coolant system, which is outside the scope of this optimization.

Table 4-2. Summary of functional requirements for the primary coolant.

PRIMARY COOLANT FUNCTIONAL REQUIREMENTS	
1. transfer heat from fuel systems to the intermediate loop	<i>economics, IP, RDC 3</i>
2. interface with primary pump and intermediate heat exchanger	<i>economics, IP, RDC 3</i>
3. interface with fuel and fuel handling systems	<i>economics, IP, RDC 2</i>
4. interface with graphite reflector system	<i>economics</i>
5. interface with core barrel and reactor vessel system	<i>economics, IP, RDC3, 4, 5,</i>

4.2.3 Graphite Structures

Table 4-3 presents a summary of function requirements for the graphite structures. Functional requirements 2, 3, 4, 5, 6, 12, 13 can be satisfied by maintaining the structural integrity of the graphite components. Graphite initially shrinks, then swells in response to neutron irradiation. Structural analysis is the best way to assess whether or not structural integrity can be maintained, however detailed structural analysis is beyond the scope of this project. Therefore, the fast neutron fluence or displacements per atom (DPA) level that cause the graphite to reach the turning point – the radiation

dose where graphite returns to its original volume – will be used as a surrogate metric for structural integrity of the graphite structures. This measure is graphite temperature dependent and is commonly used in graphite moderated reactor design studies (Idaho National Laboratory, 2010).

Table 4-3. Summary of functional requirements for the graphite structures

GRAPHITE STRUCTURES FUNCTIONAL REQUIREMENTS	
1. reflect and moderate neutrons	<i>economics</i>
2. maintain core geometry	<i>economics, RDC 3</i>
3. provide flowpaths for primary coolant (pebble bed variant)	<i>economics, IP, RDC 3</i>
4. maintain control and shut down channel geometry	<i>economics, IP, RDC 5</i>
5. maintain channel for instrumentation	<i>economics, RDC 5</i>
6. shield core barrel and reactor vessel from neutrons	<i>economics, RDC 5</i>
7. interface with fuel handling system	<i>economics</i>
8. interface with primary coolant and intermediate heat exchanger	<i>economics, RDC 3</i>
9. interface with reactivity control and shutdown systems	<i>economics, RDC 5</i>
10. interface with core barrel and upper support systems	<i>economics, RDC 5</i>
11. interface with instrumentation system	<i>economics</i>
12. displace coolant volume	<i>economics</i>
13. provide thermal inertia	<i>IP, RDC 3</i>

4.2.4 Core Barrel and Downcomer

Table 4-4 presents a summary of functional requirements for the core barrel and downcomer. The functional requirements of the core barrel will be performed so long as it maintains its structural integrity. The effects of neutron irradiation of steels in this FHR temperature regime (600 °C) is not often studied so there is significant uncertainty around the radiation damage limits for the core barrel. High temperatures enable steels to self-heal faster and increase the ductility. Therefore, the high temperature radiation damage limits established for Fast Reactors are increased to a radiation damage limit of 10 DPA is used as the damage limit and 10ppm gas production are taken as proxies for structural integrity of this system.

Table 4-4. Summary of functional requirements for the core barrel

CORE BARREL AND DOWNCOMER REQUIREMENTS	
1. guide flow to lower plenum	<i>economics</i>
2. minimize by-pass flow through core barrel	<i>economics</i>
3. interface with primary coolant	<i>economics</i>
4. interface with graphite reflector	<i>economics</i>
5. maintain integrity for the life of plant	<i>economics</i>

4.2.5 Reactivity Control System

Table 4-5 presents a summary of functional requirements for the reactivity control system. Preliminary control worth studies indicate reactivity can be managed with control rods and shutdown blades. The control worth criteria are taken from a technology neutral interpretation of the general design criterion (GDC) 26 for nuclear power plants (Nuclear Regulatory Commission, 2011). GDC 26 including the requires two independent redundant reactivity control systems with high reliability under all anticipated normal and accident conditions including holding the reactor core subcritical under cold conditions. The limiting case for the reactivity control system considered for this study will be the maintaining subcriticality, long term under the cold zero power conditions. In this hypothetical cold zero power condition all the xenon has decayed, the reactor system is held at the freezing point of flibe – 460 °C – and the control element with the most control worth fails to engage (i.e. a single failure) (Williams, Toth, & Clarno, 2006). Reactivity worth predictions for the control elements for the Mk1 PB-FHR core design are presented in Section 5.3.4.

Table 4-5. Summary of functional requirements for the reactivity control system

REACTIVITY CONTROL SYSTEM FUNCTIONAL REQUIREMENTS	
1. initiate reactivity control automatically to maintain reactor components within design limits	<i>economics, IP, RDC 2</i>
2. maintain subcriticality at cold zero power	<i>economics, RDC 2</i>
3. sense accident conditions and engage reactivity control	<i>IP, RDC 2</i>
4. capability to ensure integrity of components during accidents	<i>IP, RDC 2</i>
REACTIVITY CONTROL SUBSYSTEM	
5. fine reactivity control during normal operation	<i>economics, IP, RDC 2</i>
6. reversibility	<i>economics</i>
7. fail into safe state	<i>IP, RDC 2</i>
RESERVE REACTIVITY CONTROL SUBSYSTEM	
8. control reactivity	<i>economics, IP, RDC 2</i>
9. not susceptible to common mode failure with shut-down system	<i>IP, RDC 2</i>
10. reversibility	<i>economics</i>
11. intrinsic core temperature feedback	<i>IP, RDC 2</i>

4.2.6 Direct Reactor Auxiliary Cooling System

Table 4-6 presents a summary of functional requirements for the direct reactor auxiliary cooling system (DRACS). An implicit functional requirement for the DRACS system is to maintain its own structural integrity, so it can perform its other functional requirements based on the temperature evolution during a severe accident. A 750 °C temperature limit during transients will be used as a proxy for the structural integrity during an accident (Cisneros, Scarlat, Laufer, Greenspan, & Peterson, 2012). Structures may be able to survive heating to yet higher temperatures, and this limit should be revised as additional analysis becomes available, but 750 °C is a conservatively low value and if achieved would provide high confidence in structural integrity of metallic components.

Coupled neutronics-thermal hydraulic-structural analysis modeling will be required to assess the actual conditions during an accident and assess the structural integrity of the DRACS system – though for this optimization study the simple hot shutdown temperature model will be used to estimate accident conditions and apply the temperature limit.

Table 4-6. Summary of functional requirements for the FHR DRACS system

DRACS SYSTEM FUNCTIONAL REQUIREMENTS	
DRACS HEAT EXCHANGER & DIODE	
1. transfer decay heat from primary coolant to DRACS coolant	<i>IP, RDC 3, 5</i>
2. maintain low LMTD to prevent freezing	<i>RDC 3</i>
DRACS PIPING & INSULATION/ELECTRICAL HEATING	
3. transfer heat from primary loop to ultimate heat sink during an accident	<i>RDC 3, 6</i>
4. prevent overcooling and freezing, recover from localized freezing	<i>RDC 3</i>
NATURAL DECAY HEAT EXCHANGER	
5. use ambient air as the ultimate heat sink for decay heat removal	<i>RDC 3</i>
6. control air flow to prevent overcooling	<i>RDC 3</i>
7. minimize heat loss under normal operating conditions	<i>economics</i>

4.2.7 Reactor Vessel

Table 4-7 presents a summary of functional requirements for the reactor vessel. Like the core barrel, the functional requirements of the reactor vessel will be satisfied so long as it maintains its structural integrity. Therefore, the steel reactor vessel shall utilize the same performance metrics as the core barrel.

Table 4-7. Summary of functional requirements for FHR reactor vessel

REACTOR CAVITY SYSTEM	
REACTOR VESSEL/GUARD VESSEL SUBSYSTEM	
1. contain primary coolant in primary integral loop	<i>economics, IP, RDC 5</i>
2. transfer structural loads from reactor to building	<i>Economics, RDC 5</i>
3. minimize heat losses from the reactor	<i>economics</i>
REACTOR CAVITY COOLING/INSULATION SUBSYSTEM	
4. maintain concrete structures within their design limits	<i>economics, IP, RDC 6</i>
5. minimize heat losses from the reactor	<i>economics</i>
ELECTRICAL HEATING SUBSYSTEM	
6. maintain salt in liquid phase	<i>economics, IP, RDC 5</i>
BUFFER SALT SUBSYSTEM (IF USED)	
7. provide excess salt	<i>IP, RDC 3, 5</i>
8. reduce stress on reactor pressure vessel	<i>IP RDC 3, 5</i>
9. interface with reactor pressure vessel	<i>IP RDC 3, 5</i>
CONCRETE WALLS SUBSYSTEM	
10. low-leakage containment boundary	<i>IP, RDC 1, 4, 6</i>
11. radiation shielding	<i>economics</i>
12. reliable heat sink for decay heat during BDBEs	<i>RDC 3</i>

4.2.8 Summary of Performance Criteria

Table 4-8 presents a summary of safety criteria identified in this section.

Table 4-8. Summary of safety criteria

Performance Metrics
Fuel temperature reactivity coefficient < 0
Coolant temperature reactivity coefficient < 0
Peak fuel temperature < 1200 °C
Equilibrium coolant outlet temperature < 750 °C
k-effective < 1.0 with single failure for cold zero power conditions without ¹³⁵ Xe
Peak dose to inner graphite reflector < turnaround point dose limit
Peak dose to outer graphite reflector < turnaround point dose limit
Peak dose to core barrel < 10 DPA with < 10ppm gas concentration
Peak dose to reactor vessel < 10 DPA with < 10ppm gas concentration

4.3 Economic Evaluation Model

The economic performance is an important component of the feasibility of the PB-FHR. Many of the variables calculated in neutronics analysis (burnup, cycle length, radiation damage rate, etc.) are only proxies for the economics (burnup, cycle length, radiation damage rate, etc.) and since many of these economics proxies are at odds with each other (like power density, cycle length, and burnup) and one is often forced to choose between pareto efficient design points without a basis to compare these metrics.

This study uses the economic results (net present value (NPV), levelized unit cost of electricity (LUEC) or internal rate of return (IRR)) of an entire self-consistent PB-FHR design scenario as optimization metric for the design of the PB-FHR. This section presents a simplified economics cash flow model used to calculate the PB-FHR economics results, the underlying assumption used to estimate the costs in the economics model, the implementation of this model and some scoping studies to develop a specific scenario for which to optimize the PB-FHR design.

4.3.1 Simplified Economics Model

The goal of the simplified economics model is to have a means of comparing to economic performance of different point designs of the PB-FHR. Therefore, this model must be able to translate changes in the physical design to changes in capital costs and the resulting changes in physical performance into changes in the variable cash flows of the PB-FHR project and length of the project.

The heart of the simplified economics model is a cash flow for a PB-FHR scenario as shown in Figure 4-1. Preconstruction costs are charged at the start of the project. The overnight costs (direct, indirect and contingency) are evenly spread out over the construction time. After construction, power production begins. The power-producing period (major period) of operation is sub-divided into several minor periods. In each of these minor periods the costs for nuclear fuel, fossil fuel and operation and maintenance (O&M) are accrued and as a result of operation, revenues from electricity (derived from both nuclear energy and natural gas) are collected. At the end of these minor periods the central reflector assembly needs to be replaced and no electricity can be generated during this nuclear outage. At the end of the major period decommissioning costs are incurred. The methodology used to estimate all of these costs is presented in the following subsections.

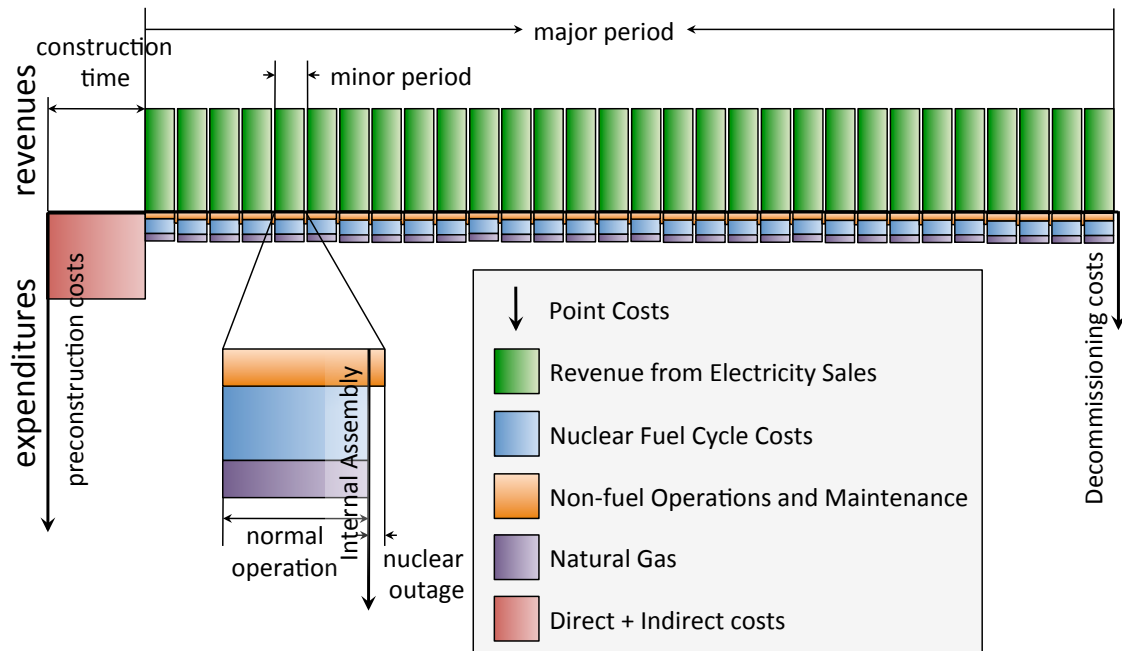


Figure 4-1. Simplified cash-flow economics model for PB-FHR project.

Many elements of this capital costs model vary as a function of the attributes (power, volume of components, burnup, etc.) of a specific PB-FHR reactor design. Therefore, the costs of the elements of the capital costs model are presented as costs normalized to specific design attributes. The next sections present the basis for estimating each of the elements of the simplified economics model where costs with high uncertainty elements are given both optimistic- and conservative normalized costs.

4.3.1.1 Capital Costs

A top-down economics model for the high temperature gas cooled reactor (HTGR) was used as a basis to estimate the capital costs of the PB-FHR because they share many characteristics: these reactors both operate at high temperatures (outlet temperatures 700-950 °C), their structural materials are composed of nuclear graphite and steel, they utilize Brayton cycles for their power conversion system and they both utilize coated particle fuel (Gandrik, Wallace, Demick, Melancon, & Patterson, 2011). Furthermore, since this technical evaluation by the Idaho National Laboratory (INL) is a complete assessment of the costs of the HTGR, following this model accounts for all of the costs of a nuclear power plant. These capital costs are broken down into four components: pre-construction costs, direct costs, indirect costs and contingency. The design optimization of the PB-FHR is targeted at a set of several Nth of a kind (NOAK) co-located PB-FHRs. One significant departure from the INL's capital cost model is the inclusion of flibe as a capital cost because its cost scales with the pebble bed's size and its costs is on the order of other explicitly estimated direct costs.

4.3.1.1.1 Preconstruction Costs

The preconstruction costs model is based on the model used in INL's models, which includes land, land rights, licensing and applications costs. The costs from INL's technical

evaluation are presented along with the preconstruction costs assumed in the simplified economics model are presented in Table 4-9.

Table 4-9. Preconstruction Costs.

2011 HTGR 350 MWt 4-pack reference		
Task	NOAK Single Unit	NOAK 4-Modules
Land and Land Rights (\$ million)	5	10
Licensing and Application Costs (\$ million)	71.5	81
Simplified Economics Model		
Cost	Single Unit	Add'l Modules
Pre-construction Costs (\$ million)	76.5	4.8
2013 Pre-construction Costs (\$ million)	81.2	5.1

4.3.1.1.2 Direct Costs

The estimates of direct costs for the components of the HTGR are used as a basis for estimating the corresponding costs in the PB-FHR. For cost elements that are well developed in the PB-FHR and well defined in the PB-FHR cost estimation are based on physical attributes (mass, volume, height ,etc.) of the specific PB-FHR design. Where less information about the designs are known the reference cost is scaled by power from the 4-pack of 350 MWt HTGR with an outlet temperature of 750 °C utilizing a Brayton power conversion cycle, as this case is the closest to a modular PB-FHR with an outlet temperature of 700 °C. In a few instances the reference cost value is taken directly. The costs from INL’s report are assessed in 2013 dollars assuming 3% inflation rate.

4.3.1.1.2.1 Power Conversion System

The cost of the PB-FHR power conversion system is estimated as the cost of a modified gas turbine and the associated heat recovery steam generator. The PB-FHR will utilize a modified General Electric 7FB gas turbine (Andreades, et al., 2014). This turbine is commercially available today and Thermoflex – a heat balance modeling and engineering program – integrated with PEACE – a program integrated with Thermoflex to evaluate the costs of turbo machinery components –lists the price of this turbine at \$48 million (Thermoflow). The normalized cost of the heat recovery steam generator was similarly estimated with Thermoflex and PEACE.

Table 4-10. Normalized Power Conversion System Costs.

GE 7FB turbine	
GE 7FB turbine (\$ million)	48
Heat Recovery Steam Generator (\$ million)	60
Steam Turbines Heat Recovery Steam Generator (\$ million)	30
PB-FHR electric power (MW-e)	242
2013 Normalized Power Conversion System Costs (\$ million/MW-e)	0.53

4.3.1.1.2.2 Reactor Initial Core

The cost of the reactor initial core is estimated by calculating the fuel cost per pebble and multiplying this value by the number of fuel pebbles in the PB-FHR. The details of the fuel costs are presented in Nuclear Fuel Costs sub-section in the variable costs section.

4.3.1.1.2.3 Reactor Vessel

The reactor vessel system is the reactor vessel together with the vessel support skirt. The cost of this system is assumed to scale with the mass of steel used in these components. The mass of the reactor vessel is based on the mass of the reactor vessel in up to the defueling chute (this component of the reactor vessel mass is a increases with the height of the specific reactor design), plus the mass of the vessel that extends above the pebble bed around the upper core structures and the reactor skirt (this component is independent of specific reactor design).

Table 4-11. Normalized Reactor Vessel Costs (Gandrik, Wallace, Demick, Melancon, & Patterson, 2011).

2011 600 MWt NGNP reference	
Parameter	Value
Reactor Vessel Weight (MT P91)	891
Head Weight (MT P91)	484
Cross Vessel Weight (MT P91)	16
Total (MT P91)	1391
Direct Reactor Vessel Costs (\$ million)	38.3
Normalized Costs (\$ million/MT)	0.028
2013 Normalized Costs (\$ million/MT)	0.030

The steel assumed in INL’s model, P91, is more expensive than the baseline metallic structural materials for the PB-FHR – 316 stainless steel; therefore, these costs are conservative.

4.3.1.1.2.4 Reactor Graphite Internals

The PB-FHR uses graphite structural materials in the inner- and outer solid graphite reflectors and to displace salt volume wherever possible. The cost of the reactor graphite internals is assumed to scale with the mass of graphite used in the inner and outer solid graphite reflectors – it is assumed that the mass of graphite before machining dictates costs, therefore the mass of graphite that once where in the vacant channels are included in the reactor graphite internals cost. The normalized costs of graphite is estimated based on Dominion’s estimate of reactor graphite internals for a 600 MWt HGTR and an estimate of the amount of graphite in this reactor. The mass of graphite in the core can be approximated as a solid cylinder of graphite as tall as the core and the two axial reflectors.

Table 4-12. Normalized Reactor Graphite Internals Costs (Idaho National Laboratory, 2007) (Kim & Venneri, 2008) (Gandrik, Wallace, Demick, Melancon, & Patterson, 2011).

2011 HTGR 600 MWt 4-pack reference	
Parameter	Value
Reactor Diameter (m)	6.83
Reactor Height (m)	10.30
Reactor Volume (m ³)	377
Graphite Mass (MT)	656
Graphite Cost (\$ million/4x600 MWt)	46.5
Normalized Cost (\$ million / MT)	0.0177
2013 Normalized Cost (\$ million / MT)	0.0188

4.3.1.1.2.5 *Flibe*

The PB-FHR utilizes the liquid fluoride salt – flibe – as its primary coolant. This coolant flows throughout the primary coolant loop as well as the DRACS loop. Currently it is assumed that the total salt in the system is approximately three times the volume of salt in the pebble bed.

The cost of flibe is highly uncertain because the lithium must be enriched to compositions with only ppm-a levels of ⁶Li. The capability to enrich lithium on an industrial scale is no longer available in the United States, although the Shanghai Institute of Applied Physics has recently demonstrated lithium enrichment using crown ethers. Oak Ridge National Laboratory made an economical assessment of the Advanced High Temperature Reactor – a large GW-e scale FHR – and estimated the costs of enriched flibe (Holcomb, Peretz, & Qualls, 2011). They used economic estimates from a 1971 review and determined the 2011 costs using the U.S. Bureau of Labor Statistic Producer Price Index; however, today laboratory quantities of enriched lithium can be purchased at \$2.5/g-⁷Li. These flibe costs are presented in Table 4-13 for the baseline price as well as an upper limit for flibe costs assuming today’s laboratory quantities costs.

Table 4-13. Normalized Flibe Costs.

2011 AHTR 3400 MWt reference	
Parameter	Value
Mass of Flibe in Primary Loop (MT)	2320
Primary Salt Costs (\$ million)	298
Normalized Cost (\$ million / MT)	0.128
Optimistic Normalized Cost (\$ million / MT)	0.136
High Primary Salt Costs (\$ million)	923
Conservative Normalized Cost (\$ million /MT)	0.398

4.3.1.1.2.6 Reactor Building

The PB-FHR uses a low leakage containment, created by the reactor cavity and adjacent cover gas handling systems volumes. This is enclosed inside a filtered confinement reactor building like the HTGR that provides defense in depth and aids in the control of radioactive and beryllium contamination. The reactor building for the HTGR is significantly more developed for the HTGR because of work done by General Atomics for the Gas Turbine Modular Helium Reactor and NGNP program (Idaho National Laboratory, 2007). The simplified economics model assumes that the reactor building costs will scale roughly as a proportion to building height, which in turn will scale with the height the reactor vessel (as the building height will need space to pull the reactor and internal assemblies in and out). It was assumed that the building dimensions for the 600 MWt HTGR were close to those reported in the NGNP pre-conceptual design. The costs from INLs technical evaluation were normalized to height of reactor vessel as shown in Table 4-14.

Table 4-14. Normalized Reactor Building Costs (Idaho National Laboratory, 2007) (Gandrik, Wallace, Demick, Melancon, & Patterson, 2011).

HTGR 600 MWt 4-pack reference	
Parameter	Value
Reactor Vessel Height (m)	32.9
Reactor Building Costs (\$ million)	29.5
Normalized Cost (\$ million / m)	0.897
2013 Normalized Cost (\$ million / m)	0.962

4.3.1.1.2.7 Heat Exchanger

The PB-FHR uses two coiled tube air heaters (CTAHs) to transfer heat from the primary loop to the compressed air to be expanded in the turbine. The costs of these heat exchangers are assumed to scale with the size of the heat exchanger. The reference cost of the HTGR heat exchangers are from the NGNPs 65 MWt process heat exchangers as described in the NGNP pre-conceptual design document. The mark 1 PB-FHR pre-conceptual design document gives preliminary dimensions of the PB-FHRs CTAHs. It should be noted that this is not a direct analogy because the NGNPs heat exchangers transfer heat between two high-pressure helium heat transport systems at temperatures up to 950 °C, whereas the CTAH transfers heat from low-pressure flibe to high-pressure air.

Table 4-15. Normalized Heat Exchanger Costs (Idaho National Laboratory, 2007) (Gandrik, Wallace, Demick, Melancon, & Patterson, 2011).

	600 MWt HTGR 1-pack	PB-FHR
Diameter (m)	3.81	3.6
Height (m)	16	12
Heat Exchanger Modules (#)	1	2
Heat Exchanger Volume (m ³)	182	244
Power (MWt)	65	236
Cost (\$ million)	19	-
Cost (\$ million / m ³)	0.104	0.104
Power Density (MWt/m ³)	0.356	0.966
Normalized Cost (\$ million / MWt)	0.292	0.108
2013 Normalized Cost (\$ million / MWt)	0.310	0.115

4.3.1.1.2.8 Heat Rejection System

The heat rejection system cools the water coming off of the steam bottoming turbine and includes circulating water pumps, piping, cooling towers, etc (The Economics Modeling Working Group of the Generation IV International Forum, 2007). The cost of the heat rejection system is assumed to scale with the thermal power of the nuclear reactor based on the cost of the corresponding system in the HTGR though this is overly conservative because the PB-FHR has a smaller heat rejection load due the steam bottoming cycle.

Table 4-16. Normalized Heat Rejection Costs (Gandrik, Wallace, Demick, Melancon, & Patterson, 2011).

2011 Reference (4-Pack 350 MWt HTGR)	
Power (MWt)	1400
Cost (\$ million/4x350MWt)	84.8
Normalized Cost (\$ million / MWt)	0.0605
2013 Normalized Cost (\$ million / MWt)	0.0642

4.3.1.1.2.9 Reactor Metallic Internals

The reactor metallic internals include the core support plate, divider plate between pebble types, the metallic lining for the hot salt collection plenum, the external-core pebble defueling chute, external core metallic channel, various wells etc. The cost of the reactor metallic internals is assumed to scale with the thermal power of the nuclear reactor based on the cost of the corresponding system in the HTGR.

Table 4-17. Normalized Reactor Metallic Internals Costs. (Gandrik, Wallace, Demick, Melancon, & Patterson, 2011)

2011 Reference (4-Pack 350 MWt HTGR)	
Power (MWt)	1400
Cost (\$ million/4x350MWt)	63.1
Normalized Cost (\$ million / MWt)	0.0451
2013 Normalized Cost (\$ million / MWt)	0.0478

4.3.1.1.2.10 Direct Reactor Auxiliary Cooling System

The PB-FHR utilizes a DRACS loop for cooling in severe accidents where as the HTGR uses a RCCS. However, since there is no directly analogous system in the HTGR and these systems perform the same function it is assumed that the costs of the DRACS in the PB-FHR scale with power in the same way the RCCS in the HTGR scale with power. This is a weak model, but when the design of the PB-FHR DRACS system is more developed one can develop a better cost-estimate of this system.

Table 4-18. Normalized Direct Reactor Auxiliary Cooling System Costs (Gandrik, Wallace, Demick, Melancon, & Patterson, 2011).

2011 HTGR 350 MWt 4-pack reference	
RCCS Cost (4-350 MWt) (\$ million)	38.8
Normalized DRACS Cost (\$ million / MWt)	0.0278
2013 Normalized DRACS Cost (\$ million / MWt)	0.295

4.3.1.1.2.11 Pebble Handling

The PB-FHR utilizes a pebble handling system that screens the pebbles by burnup and can utilize two-pressure hydraulics to transport pebbles between repositories for fresh and spent pebbles. Where as the prismatic HTGR uses a core refueling machine that handles much larger fuel blocks. However, since there is no directly analogous system in the HTGR and these systems perform the same function it is assumed that the costs of the pebble handling system in the PB-FHR scale will be similar to the costs of the core refueling machine in the HTGR, however it is not anticipated that the costs of these machines scale with power. This is a weak model, but when the design of the PB-FHR pebble handling system is more developed one can integrate a better cost-model of this system into this simplified economics model.

Table 4-19. Pebble Handling System Costs (Gandrik, Wallace, Demick, Melancon, & Patterson, 2011).

2011 HTGR 350 MWt 4-pack reference	
Core Refueling Equipment Cost (4-350 MWt) (\$ million)	121.5
Pebble Handling System (\$ million / unit)	30.4
2013 Pebble Handling System (\$ million / unit)	32.3

4.3.1.1.2.12 *Balance of Equipment*

The previously described 11 components of the direct costs (explicit costs) comprise approximately 80% of the direct costs – the remaining 20% of the direct costs are represented by an item called balance of equipment (BOE) (Gandrik, Wallace, Demick, Melancon, & Patterson, 2011). The estimate of the costs of the balance of equipment in the direct costs are calculated as shown in the Equation 4-1.

$$BOE = \sum^{11} (explicit\ cost) \frac{BOE\ percent}{100\% - BOE\ percent}$$

Equation 4-1

4.3.1.1.3 *Indirect Costs*

The indirect costs represent the cost of construction, associated overhead and engineering services. These construction costs are estimated as a fraction of the sum of the direct costs. INL’s estimate of the indirect cost’s relation to direct costs was based on General Atomics Preconceptual design for the HTGR and two sets of historical LWR costs (Gandrik, Wallace, Demick, Melancon, & Patterson, 2011). Since the components of the PB-FHR are developed to be rail-transportable and employ modern modular construction techniques, the simplified economics model assumes significantly reduced construction costs and field engineering costs in the indirect costs model for the PB-FHR; these cost ratios are presented in Table 4-20.

Table 4-20. Relative Indirect Costs.

(% of direct costs)	Average Reference Indirect Costs	Assumed PB-FHR Indirect Costs
Construction Services	20	10
Home Office & Engineering Services	16	8
Field Office & Engineering Services	10	5
Owner’s Costs	12	12
Total (Indirect Costs / Direct Costs)	57	43

In addition to these construction costs an additional \$20 million in indirect costs are charged for the final reactor design – this should not scale with reactor size and the design cost is assumed to only be charged only once per site.

4.3.1.1.4 *Contingency*

Since the costs of advanced reactors are highly speculative, a contingency factor of 20% is assumed (Gandrik, Wallace, Demick, Melancon, & Patterson, 2011). The contingency is calculated by multiplying the (preconstruction costs, direct costs and indirect costs) by this contingency factor.

4.3.1.1.5 Variable Cash Flows

The simplified economics model includes several costs that only accumulate once the PB-FHR starts generating electricity including Operations and Maintenance (O&M), Nuclear Fuel Costs, Fossil Fuel Costs, Electricity Revenue and Inner Reflector Assembly Replacement.

4.3.1.1.5.1 Operations and Maintenance

INL’s costs assessment of the HTGR presents the breakdown of annual O&M (Gandrik, Wallace, Demick, Melancon, & Patterson, 2011). The additional payroll cost required to operate additional reactor modules does is very small – therefore there is a strong economic incentive to co-locate several PB-FHRs at a single site. O&M costs are only slightly dependent on the power level of the reactor due to different supply costs and outage costs.

At this points it is really challenging to extrapolate O&M results from the HTGR to the PB-FHR due to uncertainty about the operation procedures and requirements for the PB-FHR. The design philosophy of the PB-FHR is to aggressively simplify operation – this should result in minimized O&M costs.

Table 4-21. Annual Operations and Maintenance for high temperature gas cooled reactors for Single and additional modules, for 350 MWt and 600 MWt.

Operations and Maintenance Costs (\$ millions)		Power (MWt)	
		350	600
Module	Single Unit	37.56	34.44
	Additional Unit	20.71	17.59

The O&M uses separate linear functions for O&M costs depending on whether the a module is a single/first reactor at a given sight or an additional module at a site, as shown in Equation 4-2.

$$O \& M (first) = 30.072 + 0.01248P (MWt)$$

$$O \& M (additional) = 13.222 + 0.01248P (MWt)$$

Equation 4-2

4.3.1.1.5.2 Nuclear Fuel Costs

The nuclear fuel costs in the simplified model based on the simple fuel cycle model presented in INL’s HTGR capital and operating cost assessment (Gandrik, Wallace, Demick, Melancon, & Patterson, 2011) (Shropshire, et al., 2007). This model requires one to calculate the mass of uranium ore, the mass of natural uranium required for enriching, the amount of uranium tails produced, and the mass of enriched uranium in the final fuel.

The nuclear fuel costs are used for two calculations: the initial reactor core and the annual fuel costs; see Equation 4-3.

$$Product(\text{initial core}) = N_{\text{pebbles}} * HM_{\text{pebble}}$$

$$Product(\text{annual fuel}) = \frac{P * 365.25 * CF_{\text{nuclear}}}{BU}$$

Equation 4-3

Where N_{pebbles} is the number of fuel pebbles in the PB-FHR pebble bed, HM_{pebbles} is the heavy metal loading per pebble, P is the thermal power in MWt, CF is the capacity factor and BU is the burnup in (MWt-d/MT). The natural uranium required is calculated based on the amount of the enriched product fuel required, product enrichment and concentration of ^{235}U in the tails as shown in Equation 4-4.

$$Feed = Product \left(\frac{w\%P - w\%T}{w\%F - w\%T} \right)$$

Equation 4-4 $Tails = Feed - Product$

Where $Feed$ is the natural uranium requirement, $Product$ is the enriched uranium requirement from Equation 4-3, $w\%P$ is the enrichment of the Product, $w\%T$ is the enrichment of the Tails, $w\%F$ is the enrichment of the natural uranium. The weight of yellowcake is estimated by multiplying the uranium feed rate by 1.18 the ratio of mass of yellowcake required to produce a unit mass of pure uranium.

The amount of separative work (SWU) required for enrichment is calculated as follows in Equation 4-5

$$SWU = Product * V(w\%P) + Tails * V(w\%T) - Feed * V(w\%F)$$

$$V(w\%) = (1 - 2 * w\%) * \ln \left(\frac{1 - w\%}{w\%} \right)$$

Equation 4-5

Once the required product, feed, tails, uranium ore and SWU requirements are determined these masses, SWU requirements can be multiplied by the commodity costs presented in Table 4-22.

Table 4-22. Fuel cycle component costs

Reference HTGR fuel costs	
Fuel Cycle Cost	Normalized Costs
Uranium Ore (U_3O_8) (\$/kg Nat U Ore)	112
Uranium Conversion (\$/kg Feed)	12
Uranium Enrichment (\$/SWU)	129
Tails Disposal (\$/kg Tails)	12
Fuel Fabrication (\$/kg Product)	11255
Spent Fuel Storage (\$/kg Product)	247
Spent Fuel Disposition (\$/kg Product)	3494

4.3.1.1.5.3 Natural Gas Costs

Since the turbine also burns natural gas the cost of this fuel is an important component of the simplified economics model.

The amount of natural gas required is estimated as follows:

$$Feed = P_{peaking}^{thermal} * CF_{fossil} * 365.25$$

It should be noted in the simplified economics model the natural gas heat source of the turbine has a different capacity factor than the nuclear heat source.

The energy information agency lists the average price of natural gas for electricity production on a month-by-month basis as shown in Figure 4-2 (U.W. Energy Information Administration, 2013). The average annual price estimate should be used to account for seasonal variation in demand, the latest estimate of average annual price of natural gas for this data set is \$ 4.20 / 1000ft³.

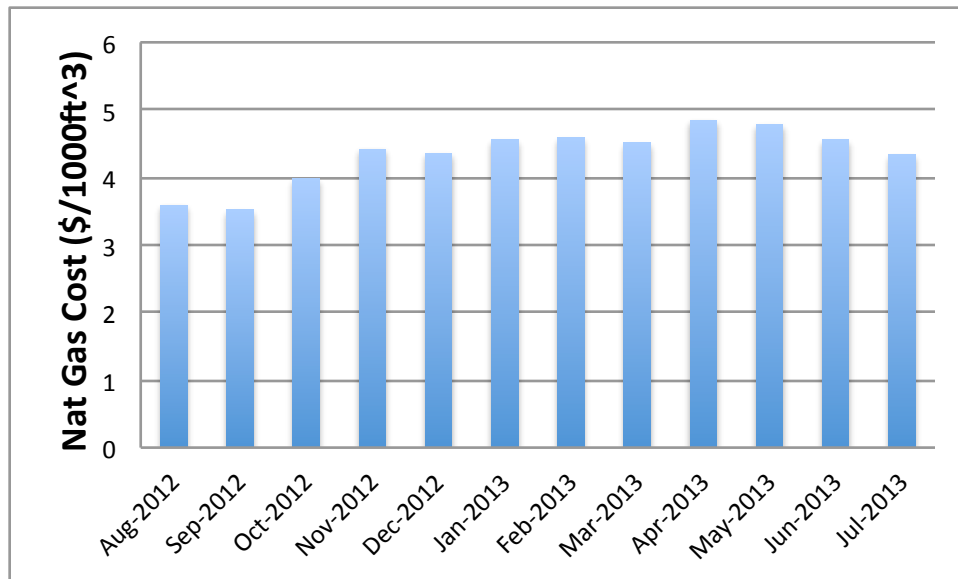


Figure 4-2. Monthly Average Costs of Natural Gas for electricity production.

4.3.1.1.5.4 Electricity Revenue

The simplified economics model uses a crude model to estimate the amount of revenue generated from electricity sales, but it still allows one to explore the trade-offs associated with producing peaking power and ancillary services. The simplified model divides the electricity revenue into three components: nuclear electricity revenue, fossil electricity revenue and ancillary services revenue.

The equations for revenue from nuclear electricity, fossil electricity and ancillary services are presented in Equation 4-6 - Equation 4-8 respectively:

Equation 4-6 $Revenue_{nuclear} = P_{nuclear}^{thermal} * \eta_{nuclear} * CF_{nuclear} * P_{base}$

Equation 4-7
$$Revenue_{fossil} = P_{fossil}^{thermal} * \eta_{fossil} CF_{fossil} * p_{peak}$$

Equation 4-8
$$Revenue_{ancillary} = P_{fossil}^{thermal} * \eta_{fossil} CF_{ancillary} * p_{ancillary}$$

Since there are separate capacity factors and prices for each revenue source one has the flexibility to explore trade studies around modes of operation with respect to offering the natural gas heat power generation capability for flexible power generation and collecting revenue for ancillary services or operating the fossil heat source to produce baseload electricity along with the nuclear heat source.

The assumed average base and peak electricity prices were estimated using historical data from the California Independent System Operator (CaISO); Hourly Locational Marginal Prices (LMP) and Ancillary Services (AS) Clearing Prices are publicly available through their Open Access Same-time Information System (OASIS) website (California Independent Service Operator, 2013). The hourly data from the beginning of October 2012 – the end of September 2013 was used to calculate the expected baseload price of electricity and the maximum average peaking price (MAPP) as a function of capacity factor. The expected baseline price is a simple average all the electricity prices over this time period. However, for a given peaking capacity factor, the MAPP the average of electricity prices in the highest percentile equal to the capacity factor. The hourly LMP data set is presented a long with the MAPP in Figure 4-3; additionally, a table of the peak price for several capacity factors is given in Table 4-23.

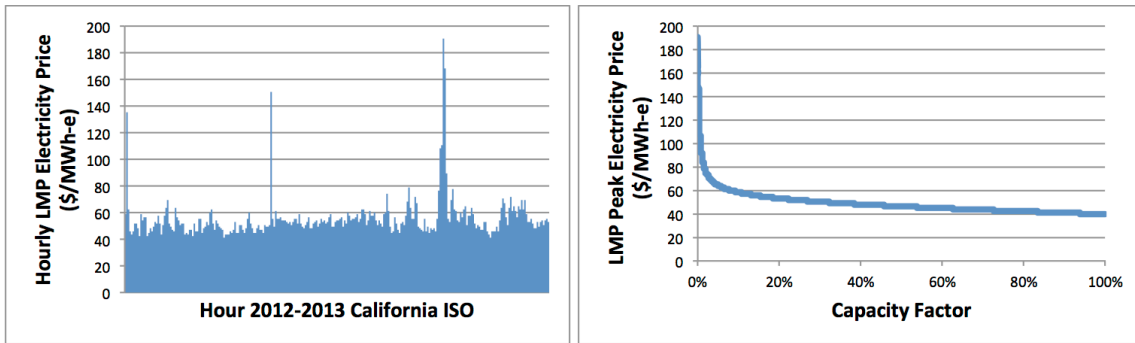


Figure 4-3. Electricity Locational Marginal Price (LMP) for California Independent System Operator Oct 2012 – Sept 2013 (left); maximum average peaking price (right)

Table 4-23. Reference Peak Electricity Prices as a function of Capacity Factor from 10/2012 -10/1013 California Independent System Operator Historical Data.

Reference Peak Electricity Prices California ISO	
Capacity Factor (%)	Max. Average Peaking Price (\$/MWh-e)
100%	39.54
90%	41.00
87%	41.39
80%	42.26
70%	43.52
60%	44.91
50%	46.44
40%	48.20
30%	50.38
20%	53.34
10%	58.42

Using these prices assumes that the introduction of the new generating capacity of the PB-FHR project does not effect the marginal prices of electricity and for the MAPP it assumes the operator knows exactly which hours will be the most profitable over a given timeframe. Therefore, utilizing these electricity prices will give optimistic economic results – especially, when using MAPP to estimate the on-peak electricity prices as a function of the peaking unit’s capacity factor.

The assumed average prices for different ancillary services were estimated from historical data from CaISO over the same time period (California Independent Service Operator, 2013). These average prices along with the hourly prices for regulation up services are presented in Figure 4-4.

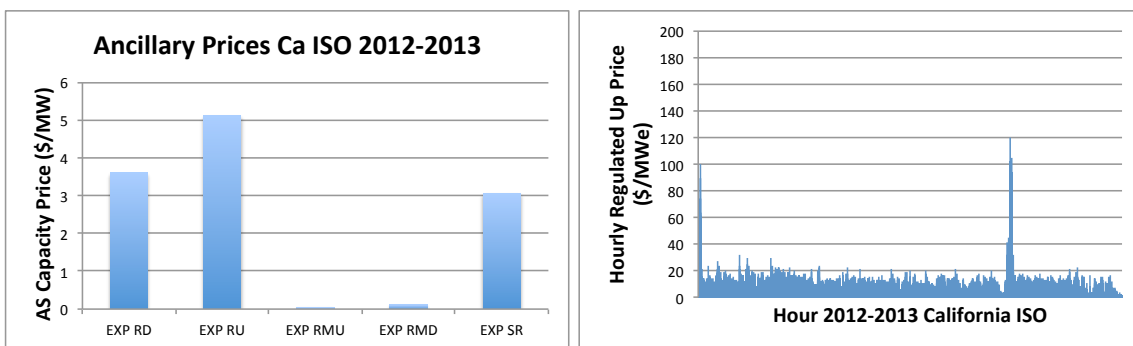


Figure 4-4. Ancillary services prices for California Independent System Operator 10/2012 – 10/2013: annual average price per ancillary service (left); Hourly regulated up ancillary service price (right)

4.3.1.1.5.5 Inner Reflector Assembly

The inner reflector assembly must be periodically replaced because the radiation damage causes geometric distortions to the inner reflector assembly, resulting in loss of

alignment in the channels for control elements or internal stress that compromise the integrity of this assembly.

The cost of replacing this assembly is estimated with three components: the cost of the inner assembly’s graphite, the cost of metallic internals and lost revenue due to down time. The cost of the graphite is estimated as the mass of the graphite volume of the inner reflector multiplied by the normalized reactor graphite internals cost, and the same estimate of the cost of the metallic internals in the capital costs estimate is used in the inner reflector assembly cost estimate. Furthermore, the replacement must occur during an outage period where no electricity is produced; this outage pushes electricity production into the future, thereby reducing the present value of those revenues.

4.3.1.1.6 Decommissioning

INL’s HTGR economics study estimated the decommissioning cost by adapting decommissioning costs from 1986 and utilizing labor-, energy cost-, and burial/disposal-adjustment factors. The decommissioning cost in this study was already given in normalized terms as shown in Table 4-24

Table 4-24. Normalized decommissioning costs (Gandrik, Wallace, Demick, Melancon, & Patterson, 2011).

Reference HTGR	
Normalized decommissioning costs (\$ million / MWt)	0.204780

4.3.2 Nuclear Application Value Estimator (NAVE)

The previously described simplified economics model is implemented in an object oriented Python framework – Nuclear Application Value Estimator (NAVE). This framework defines a scenario object as the combination of a set of economic assumptions (ie all the normalized costs, interest rates, construction times, capacity factors, electricity prices, etc.), and a specific PB-FHR design (power density, active thickness, thickness of pebble reflector, active height, etc.) that together can be utilized to populate all the cost elements in the simplified economics model. In this framework the set of economics assumptions and PB-FHR design are two objects with their own attributes and methods. Furthermore, all the elements of cash flow model (see Figure 4-1) are represented as a collection of either point cost- or rate cash flow objects.

4.3.2.1 Economic Assumptions Object

The economic assumption object holds all the economic assumptions used to populate the simplified economics model. Each assumption (object attribute) used in NAVE is defined in Table 4-25.

Table 4-25. Summary of variable economic assumptions for the PBFHR.

Financing Assumptions	
rate	Annual Interest Rate
Duration Assumptions	
construction_time	Construction time (years)

outage_time	Time to install a new reflector (years)
creep_lifetime	Maximum limit on the duration of the PB-FHR project (effective full power years)
SS_316_dose_limit	Radiation dose limit on the stainless steel core barrel (DPA)
SS_316_gas_limit	Gas concentration limit in the stainless steel core barrel (ppm)
Revenue Assumptions	
nuclear_CF	Capacity factor of the nuclear heat source
fossil_CF	Capacity factor of the fossil heat source
ancillary_CF	Capacity factor of providing ancillary service
base_electricity_price	Price of electricity produced from the nuclear heat source (\$/MWd-e)
peak_electricity_price	Price of electricity produced from the fossil heat source (\$/MWd-e)
ancillary_price	Price of ancillary services provided by the PB-FHR (\$/MWd-e)
nuclear_eta	Thermal efficiency of conversion of power from the nuclear heat source to electric power
fossil_eta	Thermal efficiency of conversion of power from the fossil heat source to electric power
peaking	The power level of the fossil heat source (MW-t)
Capital Costs Assumptions	
cost_preconstruction	Pre-construction costs (\$)
cost_steel	Cost of reactor vessel (\$/MT steel)
cost_graphite	Cost of reactor graphite internals (\$/MT graphite)
cost_flibe	Cost of flibe special material (\$/MT flibe)
cost_DRACS	Cost of DRACS (\$/MW-t)
cost_reactor_building	Cost of Reactor Building (\$/m reactor height)
cost_reactor_metallic	Cost of Reactor metallic internals (\$/MW-t)
cost_IHX	Cost of coiled tube air heater (\$/MW-t)
cost_PCS	Cost of the gas turbine and heat recover steam generator (\$/MW-t (nuclear))
cost_Pebble_Handling	Cost of the pebble handling system (\$/MW-t)
cost_heat_rejection	Cost of the heat rejection system (\$/MW-t)
BOE_fraction	Cost of balance of equipment (\$/\$ direct costs)
indirect_costs_fraction	Cost of indirect costs (construction services, home office and engineering services, field office and engineering services, owner's costs) (\$ /\$ direct costs)
contingency	Contingency (\$ / \$ preconstruction-, direct- and indirect-costs)
decommissioning	Decommissioning costs (\$/ MW-t)
Fuel Costs Assumptions	
cost_U3O8	Costs of uranium ore (\$/kg Uranium Ore)

cost_conversion	Costs of conversion of uranium ore (\$/kg Natural Uranium)
cost_SWU	Cost of separative work (\$/SWU)
natural_enrichment	Enrichment of Natural Uranium (w%)
tails_enrichment	Enrichment of Uranium Tails (w%)
cost_tails_disposal	Cost of Disposal of Depleted Uranium (\$/kg DU)
cost_fuel_fab	Cost of TRISO fuel fabrication (\$/kg LEU)
cost_fuel_storage	Cost of Interim Fuel Storage (\$/kg LEU)
cost_fuel_disposition	Cost of Fuel Disposition (\$/kg LEU)
cost_nat_gas	Cost of Natural Gas (\$/MWd-t (fossil))
Operations and Maintenance Assumptions	
OM_m	Fixed Component of O&M Costs (\$)
OM_b	Variable Component of O&M Costs (\$/MW-t)

4.3.2.2 Design Object

The design object holds all the information about the PB-FHR reactor design needed to estimate the costs of the PB-FHR project along with economics assumptions object. One can define an instance of the PB-FHR in design space with axis of power density, active thickness, pebble reflect thickness and active height; the geometric design variables are presented in Figure 4-5.

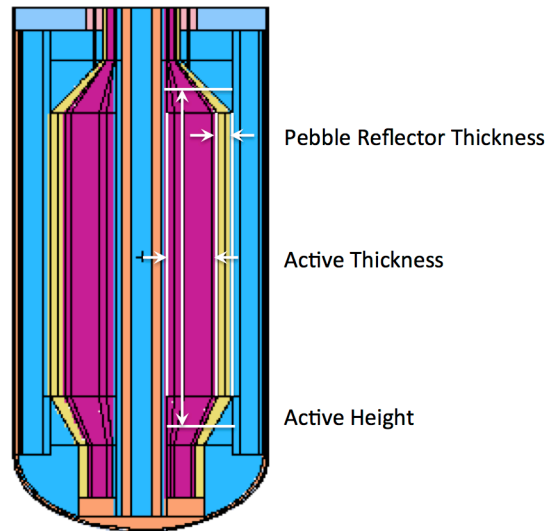


Figure 4-5. Geometry of the PB-FHR with geometric parameters labeled.

Power density (MWt/m^3) is defined as the power of the nuclear heat source normalized by the volume of the active pebble bed in the expansion-, active- and converging- regions – note this volume includes pebble structural components (shell, matrix, etc.) and the flibe coolant. The active thickness (cm) is the radial thickness of the pebble bed in the active region of the PB-FHR. The pebble reflector thickness is the radial thickness of the inert pebble reflector in the active region. The active height (m) represents the effective height of the PB-FHR if it were a simple annular cylinder rather than union of

several annular cylinders and truncated cones; see Equation 4-9. In effect, the active height of the core can be used to impose the power of the nuclear heat source for a set of power density and active thickness of a specific instance of the PB-FHR.

$$Power = Power\ Density * \pi \left(\left(R_{outer}^{active\ region} \right)^2 - \left(R_{inner}^{active\ region} \right)^2 \right) * Active\ Height$$

$$Active\ Height = \frac{Power}{Power\ Density * \pi \left(\left(R_{outer}^{active\ region} \right)^2 - \left(R_{inner}^{active\ region} \right)^2 \right)}$$

Equation 4-9

4.3.2.2.1 PB-FHR surrogate model

The design object has methods to estimate a set of design-specific performance metrics (burnup, fuel temperature reactivity coefficient, coolant temperature reactivity coefficient, inner graphite reflector peak DPA rate, outer graphite reflector peak dose rate, core barrel peak dose rate, core barrel peak gas production rate and the peak power density) based on the pre-computed results from Chapter 3 (Petroski, Cheatham, Hejzlar, Povirk, Schloss, & Whitmer, 2012). The results from parametric studies Chapter 3 were transformed into correlations of each of the four design-variables using the alternating conditional expectation (ACE) algorithm (Breiman & Friedman, 1985) (Touran, 2012).

The functional requirement of the design object is to predict reactor performance metrics (response variables) as a function of design-variables (predictor variables) based on a set of pre-computed results (measurements). The ACE algorithm is a simple methodology for estimating dependence of response variables to predictor variables that minimize the variance between the numerically predicted values and the measurements while making minimal assumptions of the functional forms of the transformations of these predictor variables. The implementation of the ACE algorithm is discussed in Section 3.3.3.3 of Chapter 3.

These transformations were implemented in NAVE as ACE objects. These objects have attributes describing the relationships between design parameters and optimal transformations as well as the relationship between the performance metric and its optimal transformation. Furthermore, these ACE objects have a method that returns the estimate of the performance metric at a given point in design space (ie complete combination of design-variables). Each of these transformation-relationships is defined as either a polynomial function to an arbitrary power or a set of points, which can be interpolated between. This study uses the convention that if a given transformation is well behaved, a polynomial function is used to approximate it in the ACE object, otherwise the ACE object will interpolate between the set of transformed points directly. Fortunately, the majority of transformations to which the performance metrics are most sensitive are well behaved. The methodology to estimate these performance metrics and physics behind the trends are presented in the chapter on the parametric study.

The mass of the reactor vessel, reactor internal graphite, and quantity of flibe in the pebble bed were calculated explicitly using the geometry engine in FIMPS. Additionally,

once the peak power in the core is estimated the corresponding maximum fuel temperature in the fuel cycle is calculated using the unit cell heat transfer model in Pebble object in the design specific FIMPS pebble bed object.

4.3.2.3 Implementation of Safety Performance Metrics

Because a scenario object in NAVE has access to all the safety related reactor performance metrics it can estimate all the safety related reactor performance metrics to enforce the safety requirements listed in Table 4-8.

4.3.2.3.1 Radiation Damage

The reactor lifetime is limited by the radiation damage limits of the outer structural components. The maximum lifetime with respect to each radiation damage limit: dose to outer graphite reflector, dose to core barrel, and gas-production in the core barrel. These radiation-limited lifetimes are compared to the creep lifetime (assumed to be 60 EFPY) and the most limiting lifetime is used as the major period in the simplified economics model. Once any one of these damage limits is reached either that component is assumed to no longer have structural integrity, however, these components are not designed to be replaced; therefore, the reactor can no longer operate within its safety basis.

The radiation damage limit of the inner graphite reflector imposes the length of the minor periods. Once this damage limit is reached the central graphite assembly along with much of the reactor metallic internals must be replaced over a nuclear outage at which point no electricity can be produced.

At low fluences the graphite preferentially expands into thermal expansion cracks in one crystal direct and shrinks in a perpendicular direction (Idaho National Laboratory, 2010). Eventually, incompatibility of crystal structures leads to the generation of new porosity, initially slowing the shrinking and then leading to swelling. This rate of dimensional distortion is dependent on the nominal operation temperature; see Figure 4-6.

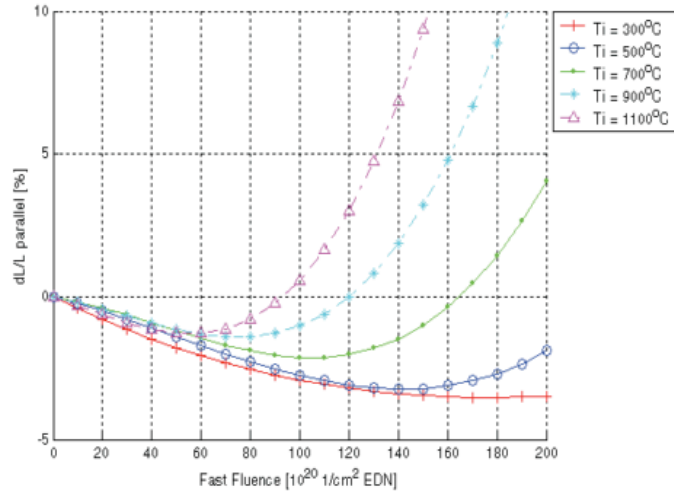


Figure B-3. Typical irradiation-induced dimensional changes in reactor graphite, parallel direction.

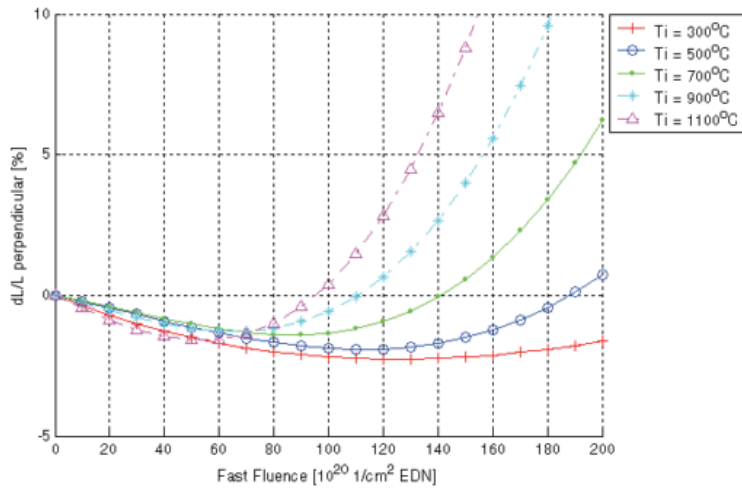


Figure B-4. Typical irradiation-induced dimensional changes in reactor graphite, perpendicular direction.

Figure 4-6. Irradiation-induced dimensional changes in reactor graphite: (top) parallel direction; (bottom) perpendicular direction (Idaho National Laboratory, 2010).

The crystal structure of graphite is an important component of the shrinkage and swelling in nuclear graphite. Well-ordered graphite can exhibit large dimensional changes because the growth and shrinkage of the graphite crystals are well aligned. Therefore, polygranular graphites are often used for nuclear applications.

The radiation damage for graphite is traditionally limited by the point at which graphite returns to its original volume the (turnaround point), because this volume expansion can lead to build up of internal stresses leading to structural failure (Idaho National Laboratory, 2010). However, this turnaround point is only a proxy for structural integrity. Figure 4-7 shows the turnaround points for the perpendicular direction (the most limiting direction) as a function of nominal operating temperature. Table 4-26 lists the

turnaround points for 600 and 700 °C – the nominal temperatures of the inner and outer graphite reflectors respectively.

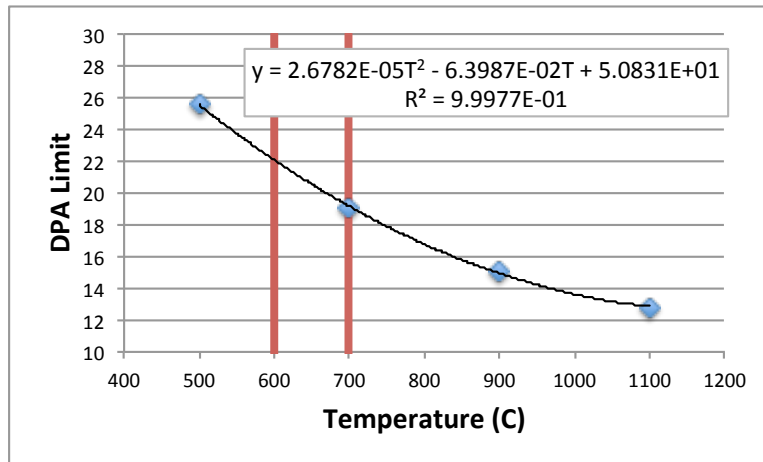


Figure 4-7. Turnaround point for nuclear graphite as a function of operating temperature.

Table 4-26. Important turnaround points for nuclear graphite.

Operating Temperature (°C)	Turnaround Point (DPA)
600	22.1
700	19.1

These radiation damage limits serve as proxies for the integrity of structural components in the NAVE framework. However, as the design of the PB-FHR matures detailed structural analysis must be performed to assess the impacts of radiation induced dimensional changes on the internal stresses within graphite structures and alignment of channels for reactivity control and instrumentation.

The high temperature of the steel (600 °C) allows the radiation damage due to atom displacements to largely self-heal. Therefore, a 10 DPA dose limit assumed for the steel structures in the PB-FHR – core barrel, reactor vessel, pebble divider plate, etc. In addition to the radiation damage dose limit, a 10ppm gas density is enforced. Gas bubbles increase brittleness in these steels. As the design of the PB-FHR matures, structural analysis must be performed to understand how the evolution of mechanical properties of the steel components lead to failure to develop a better technical basis for setting radiation damage limits for steel components.

4.3.2.3.2 Temperature Reactivity Coefficients

The fuel- and coolant temperature reactivity coefficients are calculated as part of NAVE’s surrogate model and used as proxies for reactor stability and intrinsic power feedback during accidents. In addition to the sign of the reactivity coefficient, the combination of these reactivity coefficients with the nominal characteristic fuel temperature and the characteristic coolant temperature can be used to estimate the asymptotic temperature that will eventually be established in the core following an

anticipated transient without scram (ATWS); see Equation 4-10 (Cisenros, Laufer, Scarlet, Zweibaum, & Seifried, 2012). The characteristic temperature of the coolant (the left term) is a hard limit on the outlet temperature during an accident as the heat removal (the right term) can be enhanced to some extent by optimizing the DRACS loop.

$$T_{eq}^{outlet} \sim \frac{T_{fuel}^{nominal} \frac{\partial \rho}{\partial T_{fuel}} + T_{coolant}^{nominal} \frac{\partial \rho}{\partial T_{coolant}}}{\frac{\partial \rho}{\partial T_{fuel}} + \frac{\partial \rho}{\partial T_{coolant}}} + \frac{\Delta T_{eq}^{core}}{2}$$

Equation 4-10

This temperature is a proxy for structural integrity of the reactor metallic internals during an anticipated transients without scram (ATWS) accident. More work is needed to understand the temperature limits of these metallic internals during a transient, but it is assumed that these components can survive temperatures up to 750 °C. A design point is rejected if either of the temperature reactivity coefficients is positive or the ATWS hot shutdown coolant outlet temperature is greater than the temperature limit.

4.3.2.3.3 Peak Fuel Temperature

Though there is significant margin between the average fuel temperature in the PB-FHR and the temperature limits for TRISO fuel (~1200 °C), there is significant power peaking in the PB-FHR due to the neutron flux shape, moderation from the inner graphite reflector and the distribution of reactivity in pebbles of different burnups (Todreas & Kazimi, 1990). The peak power density (with respect to radial and axial position and burnup) is estimated in the surrogate model and this peak power is used to conservatively determine the peak fuel temperature in the PB-FHR. A design point is rejected if the peak fuel temperature is greater than the temperature limit

4.3.2.4 Value Estimator

NAVE assesses the economic competitiveness of an instance of the PB-FHR scenario by populating the cash-flows in the simplified economics model and then calculating either the net present value, the levelized unit electricity cost (LUEC), or the internal rate of return (IRR) for that instance of a PB-FHR scenario. Each of these models works by defining cash flow objects and either summing the net present value of these cash flow objects or manipulating the collection of these cash flow objects.

4.3.2.4.1 Cash Flow Objects

The components of the simplified economics model are represented as a set of cash flow objects. The purpose of these cash flow objects is to discount the cash flows to the start of the project. The simplified economics model only requires two types of cash flows, point- and rate cash flows. Each of these cash flow objects has a method to return their net present value (NPV) and all the attributes necessary to evaluate their NPV. Furthermore, these cash flow objects have an attribute called account that lets one organize multiple cash flows together.

4.3.2.4.1.1 Point Cash Flows

A point cash flow is a flow of money that occurs at a single point in time. Its NPV is evaluated as shown in Equation 4-11. Where C_{point} is cost a fixed point in time, T , rate is the interest rate, and T_o is the reference time. NAVE uses years as its default unit of time and takes the time preconstruction costs are paid as the default reference time.

Equation 4-11
$$NPV_{point} = C_{point} * (1 + rate)^{-(T-T_o)}$$

4.3.2.4.1.2 Rate Cash Flows

A rate cash flow is flow of money that occurs a finite period from an initial time to a final time at a constant rate. Its NPV is evaluated as shown in Equation 4-12. Where C_{rate} is the rate of payment (this must be in the same units of time at the interest rate and reference times), $T_{initial}$ and T_{final} are the times at which the payment begins and ends respectively.

Equation 4-12
$$NPV_{rate} = C_{rate} * \left(\frac{(1 + rate)^{(T_{initial}-T_o)}}{\ln(1 + rate)} - \frac{(1 + rate)^{(T_{final}-T_o)}}{\ln(1 + rate)} \right)$$

4.3.2.4.2 Net Present Value

The net present value of the PB-FHR scenario is calculated by populating the capital- and variable costs as previously described. The preliminary cost is represented by a point cost at the start of the project, where as the other capital costs are approximated as rate cash flows accrued continuously over the construction period. The variables cash flows are distributed between several minor periods. The duration of these minor periods is limited by the radiation damage limit of the inner reflector assembly. The last minor period will likely be limited by a radiation damage limit of the outer structural components or the thermal creep lifetime of the reactor vessel. NAVE has a logic sequence to assess whether it makes fiscal sense to produce power over this period or just decommission the reactor. The net present value is calculated by simply summing the net present value of the collection of cash flow objects that represents each of the capital costs and variable cash flows. NAVE uses the convention that cash flows can be either negative or positive so that when these they are summed the result is the net present value. A summary of this calculational flow is presented in Figure 4-8.

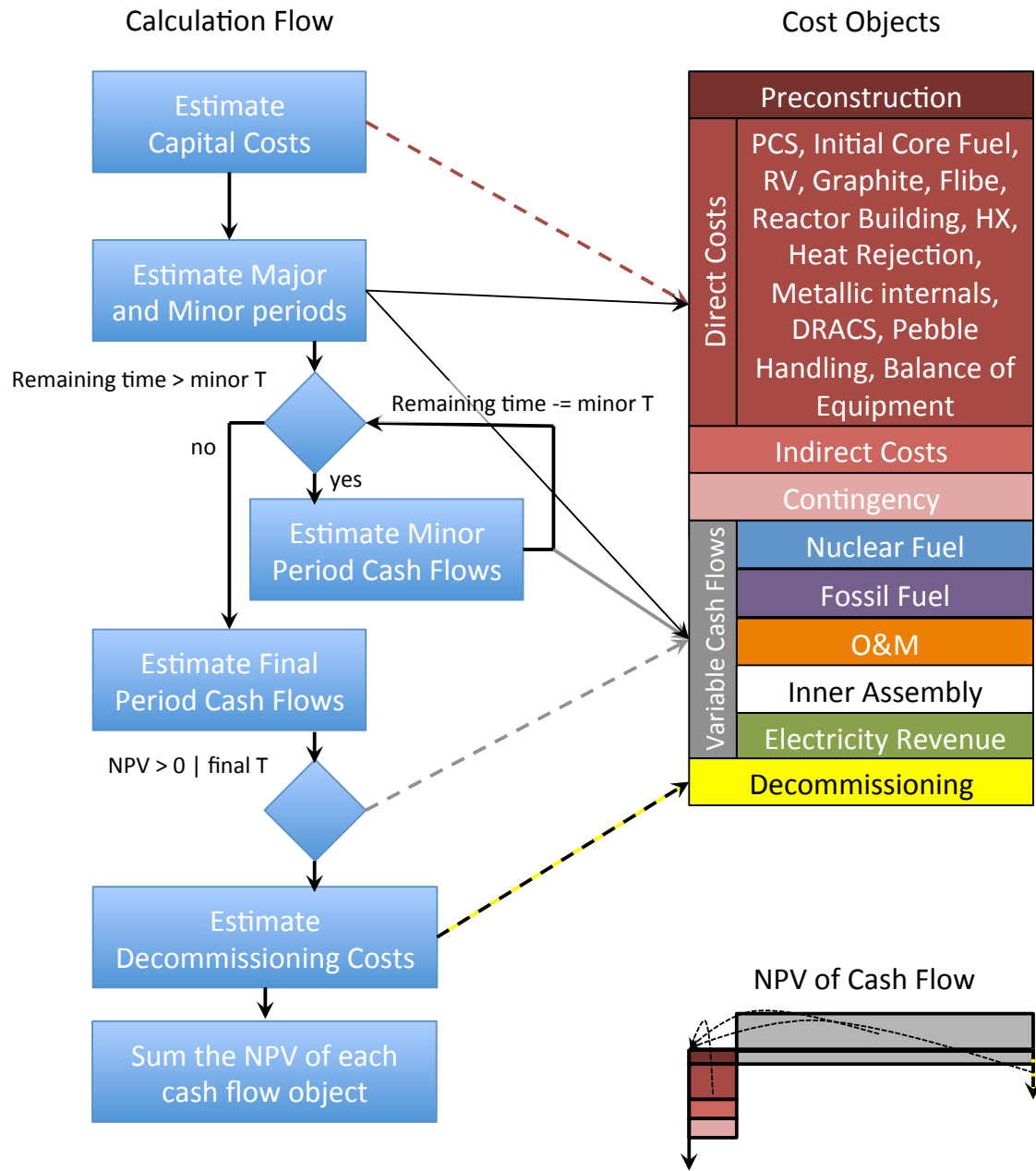


Figure 4-8. Calculational Flow of Net Present Value of a PB-FHR scenario.

4.3.2.4.3 Levelized Unit Cost of Electricity

The LEUC of a PB-FHR scenario calculation is based on the NPV calculation. However, initially the prices of base- and peak electricity are set to the same value and it is assumed that no revenue is collected from ancillary services. Next, all the NPV of the cash flow objects for a specific scenario are determined using their respective NPV methods. Then the NPV of all the electricity revenues are subtracted from the scenario NPV – this revenue, must be equal and opposite to the sum of all the other costs. If one assumes there is a multiplication factor by which the electricity revenues are multiplied,

such that the sum of the product of this factor and the electricity revenues and all the non-revenue costs of the PB-FHR scenario equal zero, the LEUC can be calculated by multiplying the assumed price of electricity by this value as shown in Equation 4-13.

$$\begin{aligned}
 NPV_{scenario} &= \sum_i^{elements} NPV_i \\
 C * NPV_{electricity\ revenue}^{initial} &= NPV_{electricity\ revenue} \Big| P_{electricity} = LUEC \\
 0 &= \sum_i^{elements} NPV_i \Big| P_{electricity} = LUEC \\
 0 &= NPV_{scenario} - NPV_{electricity\ revenue} + C * NPV_{electricity\ revenue} \\
 C &= 1 - \frac{NPV_{scenario}}{NPV_{electricity\ revenue}} \\
 LUEC &= C * P_{electricity}^{initial}
 \end{aligned}$$

Equation 4-13

4.3.2.4.4 Internal Rate of Return

The IRR is calculated by searching for the interest rate for the PB-FHR scenario resulting in a NPV of 0. By default, 16 iterations of the bisection search are performed resulting in a maximum error of 0.03% (absolute error on interest rate) in the IRR.

4.3.2.5 Characteristic Results

The baseline set of assumptions for the simplified economics are presented in Table 4-27. A low interest rate was assumed because nuclear reactors are more likely to be built in regions with low interest rates likely occur in regulated markets where utilities have a captive market, and because as a small reactor the capital placed at risk to build each BP-FHR module is small relative to conventional light water reactors. The electricity prices are taken from the latest estimate of annually averaged locational marginal price from the California ISO. Conservatively, the peak electricity price is assumed to be the same as the average price. A high capacity factor was assumed for the nuclear heat source and ancillary services, but it was assumed that the natural gas heat source operated at a lower capacity factor like a conventional peaker gas turbine. Table 4-28 presents the cost breakdown for each of the cash flow accounts for a first unit at a site and an additional unit. Furthermore, these characteristic costs and revenues are presented in Figure 4-9 to highlight the savings associated with co-location.

Table 4-27. Important assumptions for the characteristic results.

General Scenario Assumptions	
Interest Rate (%)	5
Average Base Electricity Price (\$/MW-e*hr)	39.54
Nuclear Heat Source Capacity Factor (%)	95
Average Peak Electricity Price (\$/MW-e*hr)	39.54
Fossil Heat Source Capacity Factor (%)	30
Ancillary Service Source Capacity Factor (%)	87
Mark 1 Assumptions	
Nuclear Electric Capacity (MW-e)	100
Fossil Electric Capacity (MW-e)	142
Total Power Plant Capacity (MW-e)	242
Nuclear Power Density (MWt/m ³)	23
Active Thickness (cm)	70
Pebble Reflector Thickness (cm)	20
Active Height (m)	3.33

Table 4-28. Cost breakdown for first unit and additional unit for baseline PB-FHR scenario.

		First Unit	Additional Unit*
Pre Construction Costs (\$ million)		-81.2	-3.2
Direct Costs	Power conversion (\$ million)	-116.6	-116.6
	Reactor initial core (\$ million)	-22.4	-22.4
	Reactor vessel (\$ million)	-1.8	-1.8
	Reactor internal graphite (\$ million)	-1.5	-1.5
	Flibe (\$ million)	-8.3	-8.3
	Reactor building (\$ million)	-11.5	-11.5
	Heat exchanger (\$ million)	-25.1	-25.1
	Heat rejection system (\$ million)	-14.1	-14.1
	Reactor metallic internals (\$ million)	-10.5	-10.5
	Direct Reactor Auxiliary Cooling System (\$ million)	-6.5	-6.5
	Pebble Handling (\$ million)	-30.0	-30.0
	Balance of Equipment (\$ million)	-62.0	-62.0
Indirect Costs (\$ million)		-152.0	-89.7
Contingency (\$ million)		-107.5	-74.6
Variable Costs	Operation & Maintenance (\$ million)	-586.7	-287.3
	Nuclear Fuel (\$ million)	-202.6	-202.6
	Fossil Fuel (\$ million)	-127.5	-127.5
	Inner Reflector (\$ million)	-17.1	-17.1
	Electricity Revenue (\$ million)	775.3	775.3
Decommissioning (\$ million)		-2.6	-2.6
Total Net Present Value (\$ million)		-806.7	-339.6
Levelized Unit Electricity Costs (¢/kWe-hr)		8.07	5.69

*The decision to produce power for a final period changes based on the number of modules so the variable costs change as a function of the number of modules, this chart was produced assuming the additional unit produced electricity in as many periods as the first unit.

** The IRR search as performed where electricity is only produced in the final period if it is profitable.

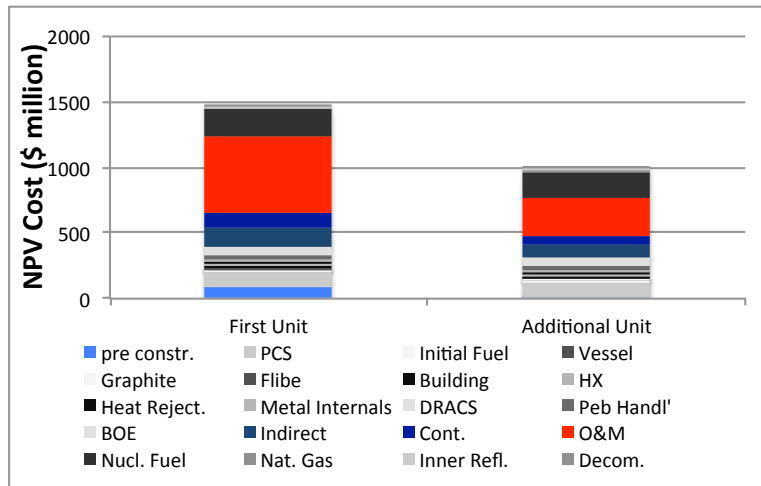


Figure 4-9. Capital and Variable Costs Comparison between the first and subsequent PB-FHR modules

4.3.2.6 Initial Scoping Studies

Based on the initial characteristic results one can see that there are significant opportunities for efficiency by siting multiple PB-FHR modules at a single site. Furthermore, the possibility of generating more revenue by operating the natural gas heat source of the PB-FHR more often to generate more electricity.

4.3.2.6.1 Economies of Co-location

From the economic cost assumptions and in Table 4-28 one can observe that operating multiple PB-FHRs at a single site can save costs on preconstruction costs, O&M costs and to an extent in indirect and contingency costs – these two costs might be considered artifacts of the structure of the top down model that takes into account the preconstruction costs when estimating the indirect costs and the contingency. Additional costs savings might occur due to learning during construction or the economies of modularity, however those factors were not explicitly considered as part of this study. Licensing makes up most of the preconstruction costs, and much of this licensing cost can be distributed between multiple PB-FHR modules. Likewise, much of the non-technical staff and overhead at a nuclear power plant can be shared between multiple PB-FHR modules. Figure 4-10 shows that as more PB-FHR modules are constructed at a single site, the project-averaged LUEC approaches the limiting value: the LUEC of an additional PB-FHR module, as shown in Figure 4-10.

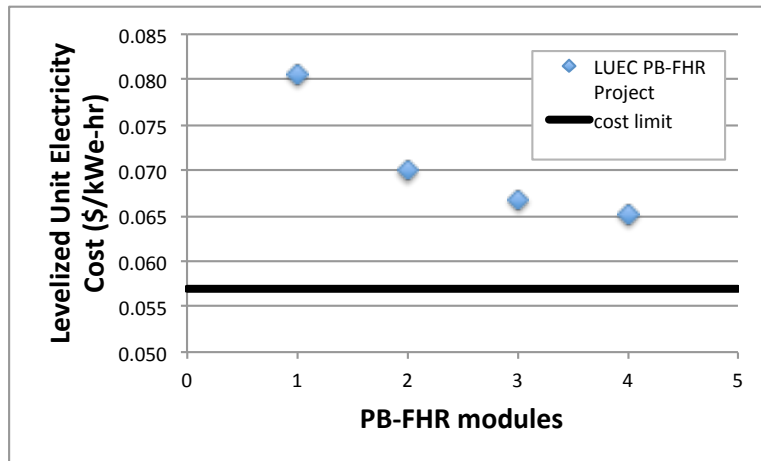


Figure 4-10. Levelized Unit Electricity Costs as a function of the number of baseline PB-FHR modules

4.3.2.6.2 Natural Gas Peaking Capacity Factor Optimization

Another key characteristic of a PB-FHR plant is how often the modules are co-firing natural gas. Because the PB-FHR produces electricity from the nuclear heat source, the combination of temperature and pressure in the second reheat stage of the PB-FHR’s gas turbine are above the auto-ignition conditions for natural gas, enabling the power conversion system to quickly produce peaking power by burning natural gas. The PB-FHR natural gas heat source can either generate as much electricity as possible, or alternatively, the peaking capacity can be used to produce a combination of ancillary services to the grid and produce as much power as possible when electricity prices are high.

A sensitivity study was performed to compare the economic impact of co-firing the PB-FHR for various fractions of potential power generating hours under different operation modes: maximum power production, frequency regulation, or spinning reserves. These operation modes are described in the subsequent subsections followed by estimates of the NPV of a PB-FHR project with 4-modules

4.3.2.6.2.1 Operation Mode A: Maximum production

The PB-FHR co-fires with the capacity factor of a gas turbine combine cycle plant 87%, but does not produce ancillary services because all the capacity is used for electricity production. Figure 4-11 presents the distribution of electric power produced from the nuclear heat source and the natural gas combustor for this case. In this figure, electric power is the vertical axis and time is horizontal axis so that area in this figure represents energy produced.

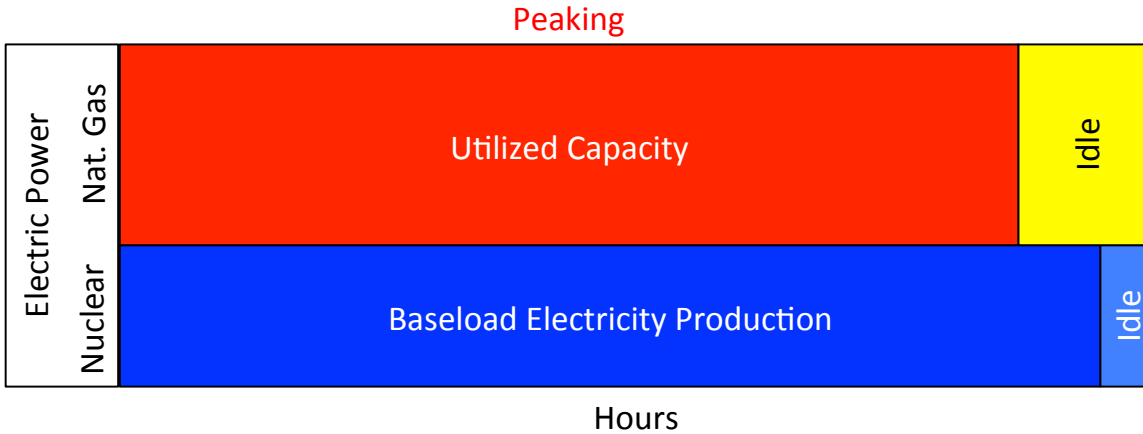


Figure 4-11. Allocation of electricity generation capacity for the Operation Mode A

Assume that the times of electricity production outages are unknown. Therefore, the PB-FHR owner collects revenue at the average annual electricity price from electricity produced in both the nuclear heat source and natural gas combustor.

4.3.2.6.2.2 Operation Mode B: regulation

The PB-FHR produces power for a fraction of the highest marginal price hours and the rest of the time regulation services are provided to the grid; in this scenario, the PB-FHR either co-fires at its maximum capacity or provides regulation services during the same number of hours as Operation Mode A co-fires at its maximum rate. When regulation services are provided to the grid, revenue is collected for this ancillary service (4.36 \$/MWe-hr – the average of regulation up and regulation down), electricity is generated at a capacity factor of ~50% as shown in Figure 4-12. Figure 4-13 presents the allocation of electric power produced from the nuclear heat source and the natural gas combustor for this case.

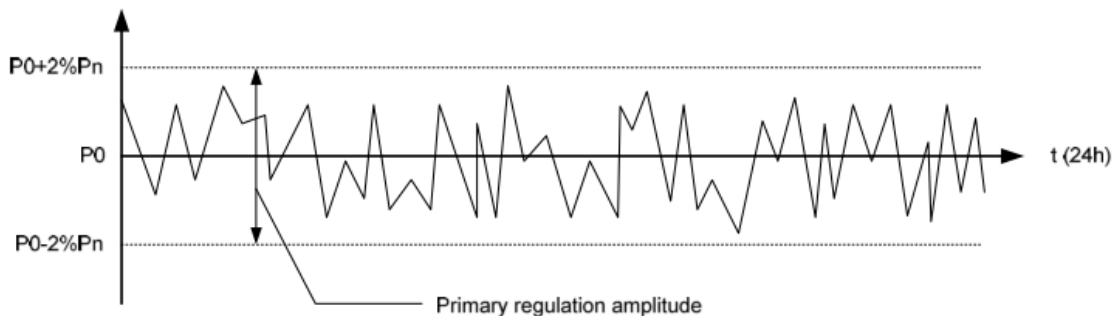


Figure 4-12. Operating margin of the frequency regulation (Bruynooghe, Eriksson, & Fulli, 2010)

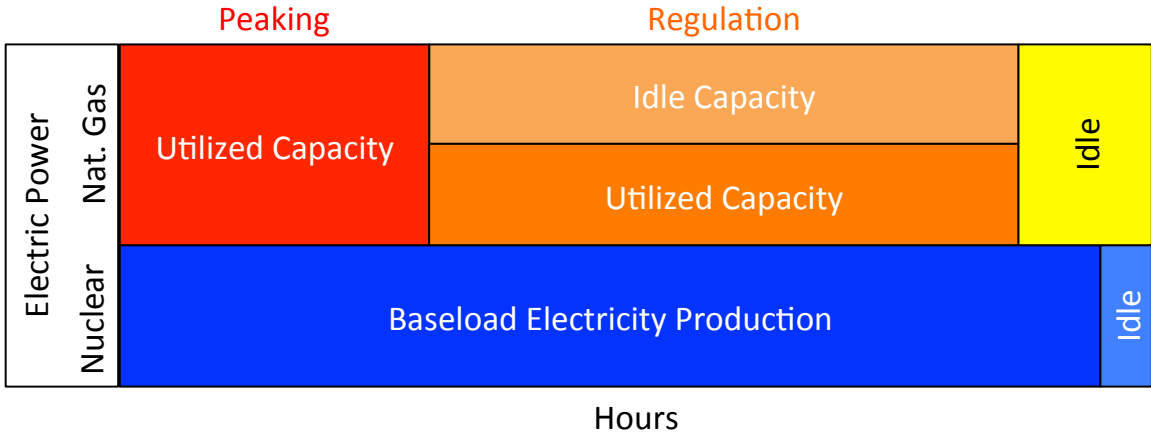


Figure 4-13. Allocation of electricity generation capacity for the Operation Mode B

Notice that the potential energy produced in case is subdivided into some peak electricity production and some grid regulation. In the regulation section only half of the electric energy production capacity is utilized. This operation mode is equivalent to operating with a reduced capacity factor. Figure 4-14 shows the transformation to the scenario modeled in NAVE.

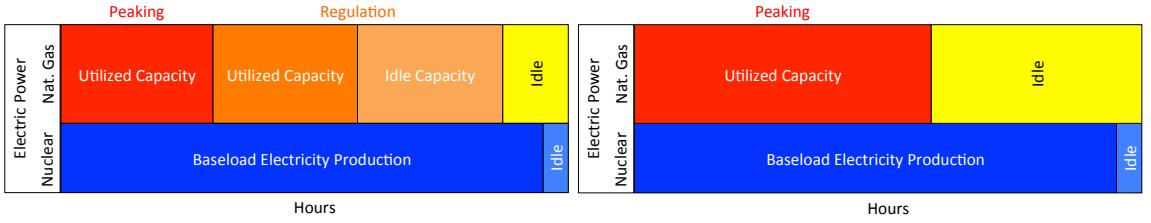


Figure 4-14. Allocation of electricity generation capacity for the operation mode B at 30% time co-firing at maximum power: (left) intermediate transformation; (right) NAVE conceptualization of operation mode B

It is assumed that the PB-FHR owner collects revenue at the average annual electricity price from electricity produced from the nuclear heat source in this scenario. However, the owner collects revenue from electricity produced in the peaking periods at a price averaged over the highest price hours, where as during the regulation period the price of electricity is averaged over lower priced hours. It is assumed that the times of electricity production outages during the regulation period are unknown so the price of electricity in this period is conservatively assumed to be the average price of all the hours outside of the peaking period. Figure 4-15 presents the hourly local marginal electricity price organized by highest price percentile and the electricity price averaged for the co-firing hours and averaged for the hours where the PB-FHR provides frequency regulation – these two electricity prices are weighted by the amount of energy produced in each mode to produce a new effective electricity price.

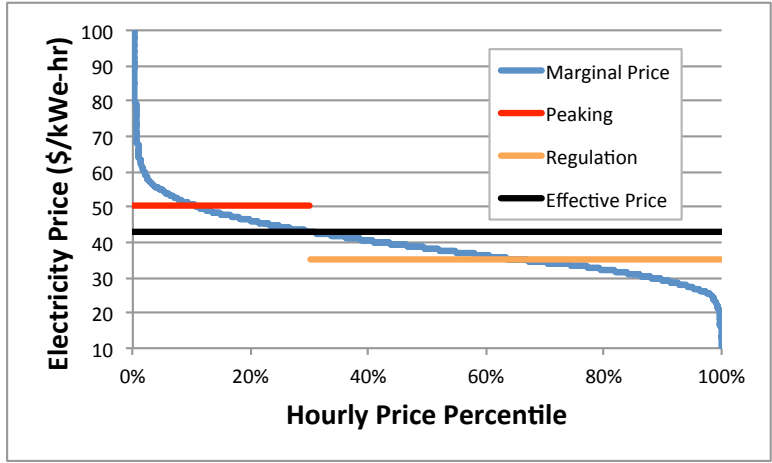


Figure 4-15. Electricity prices for operation mode B

4.3.2.6.2.3 Operation Mode C: spinning reserve

The PB-FHR produces power for a fraction of the highest marginal price hours and the rest of the time spinning reserve is provided to the grid. When spinning reserves are provided to the grid, revenue is collected for this ancillary service (3.06 \$/MWe-hr), and electricity is generated with a capacity factor lower than 50%, a capacity factor while providing spinning reserves is assumed to be 20%. Figure 4-16 presents the distribution of electric power produced from the nuclear heat source and the natural gas combustor for this case.

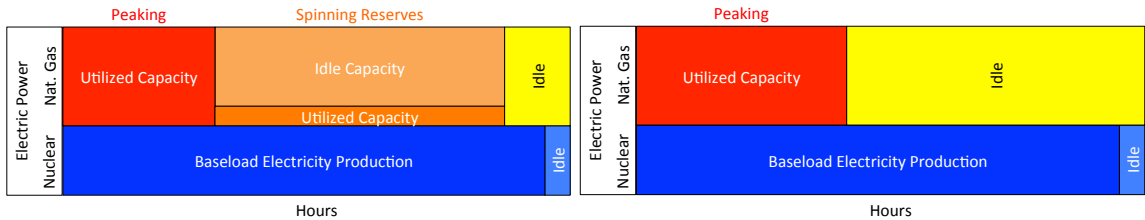


Figure 4-16. Allocation of electricity generation capacity for the operation mode A at 30% time co-firing at maximum power: (left) standard visualization; (right) NAVE conceptualization of operation mode C

As in operation mode B, assume that the PB-FHR owner collects revenue at the average annual electricity price from electricity produced from the nuclear heat source in this scenario and that the owner collects revenue from electricity produced in the peaking periods at a price averaged over the highest price hours, and during the spinning reserve period the price of electricity is averaged over lower priced hours for fewer hours. Figure 4-17 presents the hourly local marginal electricity price organized by highest price percentile and the electricity price averaged for the co-firing hours, averaged for the hours where the PB-FHR provides spinning reserve and the effective electricity price.

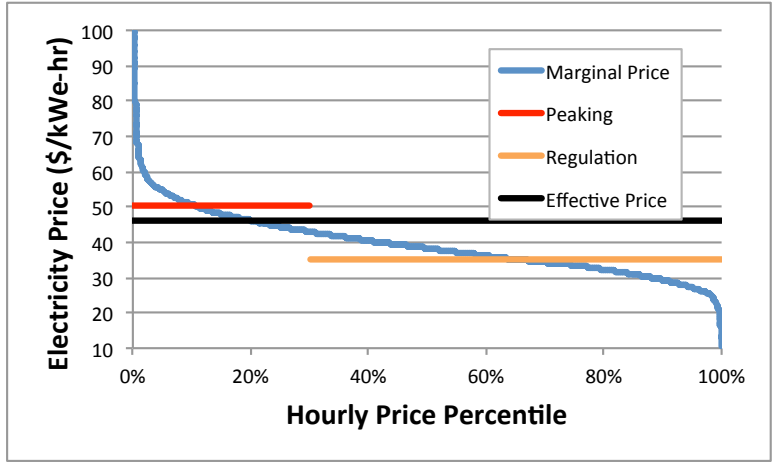


Figure 4-17. Electricity prices for operation mode C

4.3.2.6.2.4 Results

For operation modes B and C the net present value of the PB-FHR project was evaluated with the effective capacity factors and electricity prices for each scenario at with various ratios of times co-firing at maximum power output and providing ancillary services in Figure 4-18. There is no variation in co-firing time for scenario A so it is presented as a line.

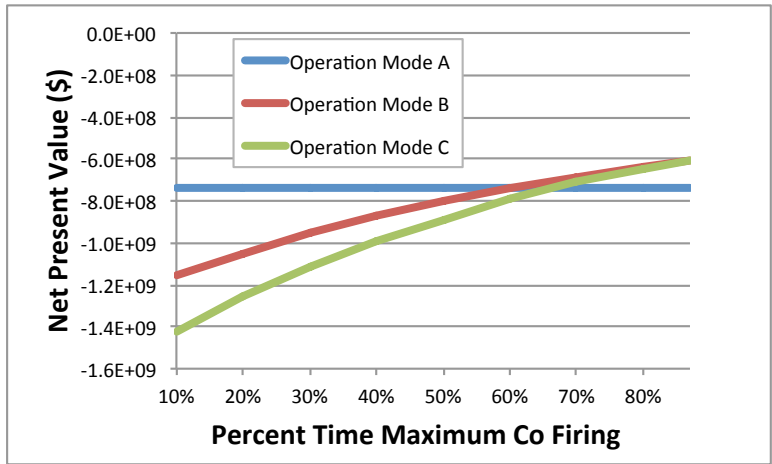


Figure 4-18. Net present values for each operation mode as a function of time spent co-firing at maximum power.

This study shows that for the assumed natural gas price (the 2012-2013 average US natural gas price – \$4.2/1000ft³) and electricity price distribution (2012-2013 California ISO electricity prices) the PB-FHR should co-fire at maximum power as much as possible for this set of historical data on the prices of electricity and ancillary services. For low fractional co-firing, the price of electricity is significantly higher than the annual average, but not much electricity is produced; these trade offs are presented in Figure 4-19. Low utilization might be partially offset by collecting revenue for ancillary service, but more net value is produced if the PB-FHR is co-fired longer. The assumption of perfect knowledge about deciding when to co-fire at maximum power or provide ancillary

services breaks down when more time is spent co-firing at full power and in reality scenarios B and C should converge to scenario A at 87% maximum co-firing leading to a higher effective electricity prices than the annual average electricity price as assumed in scenario C. No effects on the O&M were included in this study. Sensitivity to other economic assumptions are discussed in section 4.5.

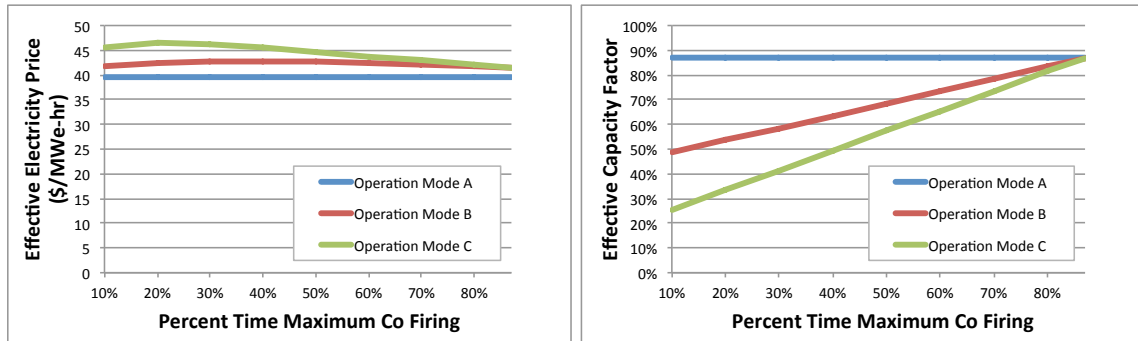


Figure 4-19. Economic attributes for different PB-FHR operational mode scenarios: (left) effective peaking electricity price; (right) effective capacity factor for co-firing.

4.3.2.7 Design Project

These two studies show that the economics of PB-FHR improve significantly when multiple modules are co-located at a single site and that the PB-FHR is most competitive when the gas turbine is operating as close as possible to maximum capacity. Therefore, a site with 4 co-located PB-FHRs that operates their natural gas heat source with a capacity factor of 87% while not competing in the ancillary services market.

4.4 PB-FHR Optimization

The PB-FHR scenario object was optimized using NAVE by holding the set of economic assumptions constant and minimizing the LUEC of the PB-FHR scenario with different design points. Each of these design points is a combination of active thickness, power density and active height to impose a nuclear power of 236MWt. The design space spanned power densities from 20-35 MWt/m³, active thicknesses from 50-90 cm, and pebble reflector thicknesses from 10-30 cm. Figure 4-20 presents the LUEC distributions for several different power densities; Figure 4-21 presents the optimized LUEC at each power density; Additionally, Figure 4-22 presents coarsely optimized dimensions of the pebble bed for different power densities.

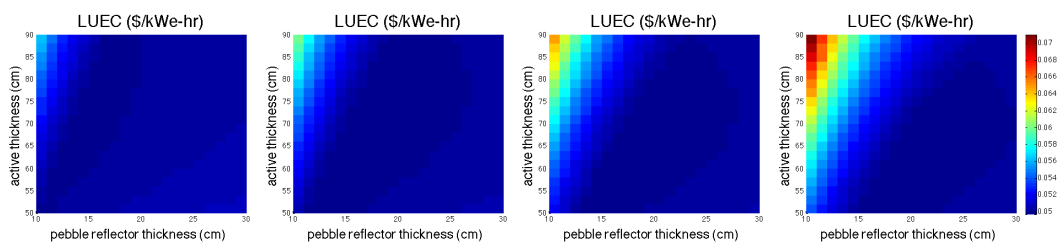


Figure 4-20. Levelized unit cost distributions for the PB-FHR projects as a function of power densities 20, 25, 30 and 35 MWt/m³ from left to right.

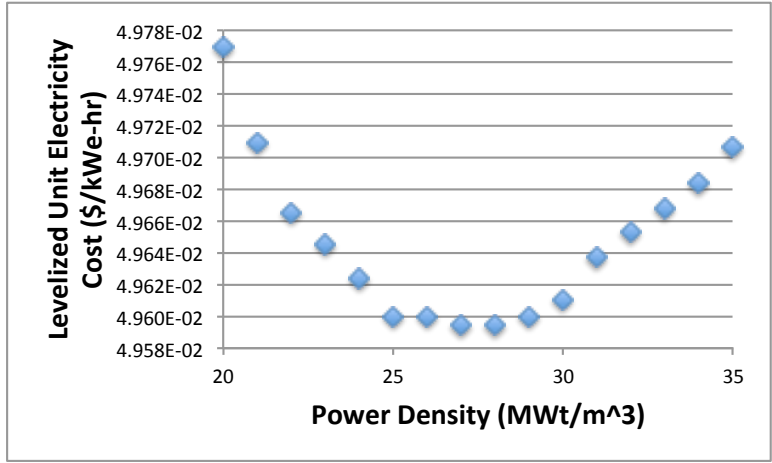


Figure 4-21. Levelized Unit Electricity Cost of PB-FHRs optimized for specific power densities.

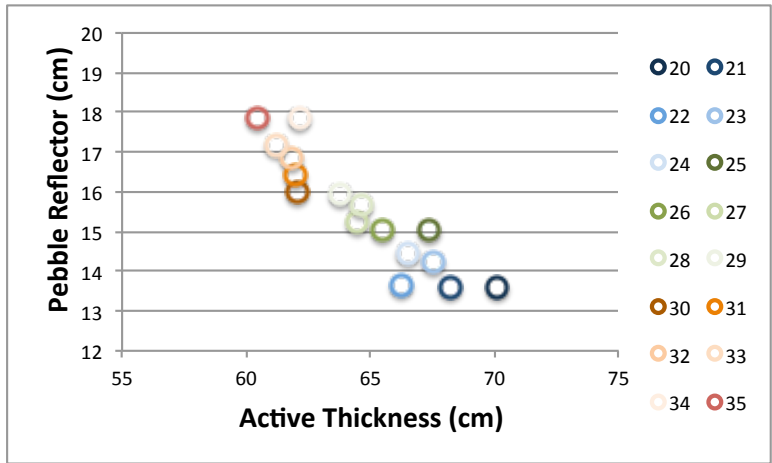


Figure 4-22. Optimized dimensions of pebble bed core for power densities – cool colors are low power densities.

There are several reactor design points that yield similar LUEC values (4.9-5.1 cents/kWe-hr). These LUEC distributions highlight the importance of appropriately sizing the inert pebble reflector for a specific power density. Given the significant uncertainties in cost models one cannot definitively define a single optimum design point for the PB-FHR. However this study has identified a rather large optimized design space in which the optimized PB-FHR is located.

The PB-FHR scenario with the lowest LUEC has a power density of 28 MW/m³. The capital costs and variable costs are well balanced for optimized cores with power densities between 25 MWt/m³ and 29 MWt/m³; the sensitivities of capital- and variable costs are discussed in the following sub-sections.

The optimized pebble bed dimensions change as a function of power density. Designs with low power densities optimize to larger radial thicknesses of the pebble bed with thin graphite pebble reflectors. As the power density increases larger pebble reflectors are required to protect the outer graphite reflectors. With these larger pebble reflectors,

smaller pebble reflectors are required to balance minimizing capital costs and fuel costs (ie maximizing burnup).

The LUEC distribution for the optimized power density is presented in Figure 4-23. The design variables that define the optimized design point and the corresponding economic performance parameters are summarized in Table 4-29.

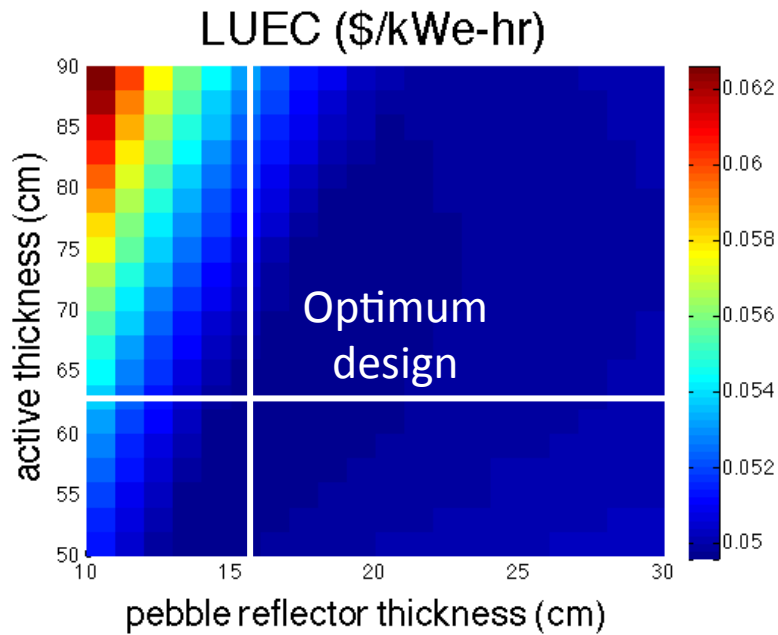


Figure 4-23. Levelized unit electricity cost of PB-FHR project as a function of PB-FHR core geometry for a power density of 28 MW/m³.

Table 4-29. Optimized PB-FHR parameters and economics performance parameters.

Optimized Design Description	
Design Parameter	Value
Power Density (MWt/m ³)	28
Active Thickness (cm)	63
Pebble Reflector Thickness (cm)	15.5
Active Height (m)	3.20
Economic Performance	
Internal Rate of Return (%)	2.07
Levelized Unit Electricity Cost (¢/kWe-hr)	4.96

This LUEC distribution is a function of several lower level integral economic metrics: capital costs, variable costs and the cycle length. Confidence in these high-level LUEC distributions requires the confirmation that the trends in the lower level economic metrics influence LUEC as expected; these trends are discussed in the subsequent sections.

4.4.1 Overnight Capital Costs

Managing capital costs is an important part of reactor optimization for the PB-FHR. Financing these capital costs accounts for over a third of the LUEC. Figure 4-24 presents the overnight capital costs normalized per kWe – including electricity produced from nuclear heat as well as natural gas heat – for several power densities.

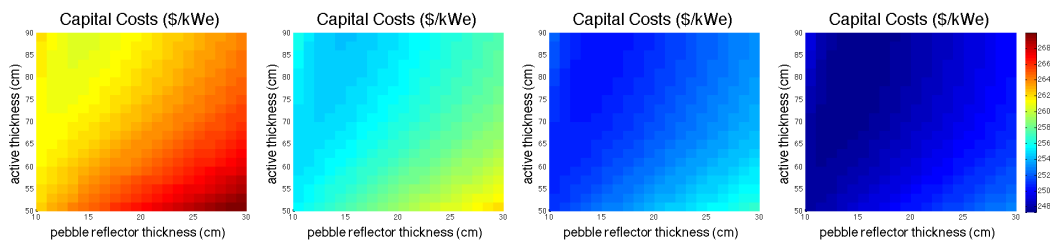


Figure 4-24. Overnight capital costs distributions for the PB-FHR projects as a function of power densities 20, 25, 30 and 35 MWt/m³ from left to right.

The overnight costs of the PB-FHR decrease with increasing power density as the amount of many structural components are reduced. This trend in capital costs agrees with the trend in LUEC from power densities of 20 – 24 MWt/m³.

Another dimension of these distributions is the pebble bed geometry dependency. Figure 4-25 compares the overnight capital costs as a function of pebble bed geometry to the LUEC of the PB-FHR scenario, both at a power density of 28 MWt/m³.

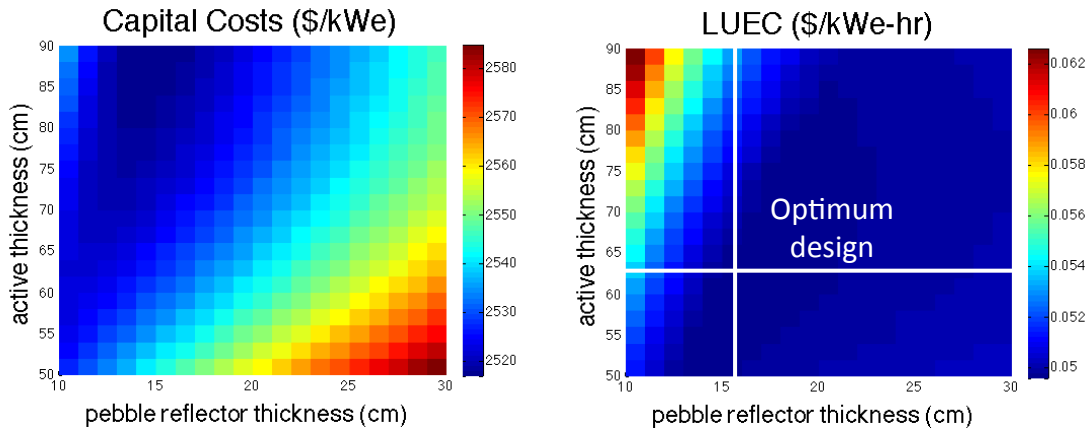


Figure 4-25. PB-FHR capital costs as a function of PB-FHR core geometry for a power density of 28 MW/m³ (left) and the Levelized unit electricity costs (right)

Short radially thick cores with thin pebble reflectors are most efficient with respect to structural materials as this maximizes the fraction of the pebble bed to structural graphite within the reactor vessel, minimizing the height and amount of flibe and structural materials.

Many of the components of capital cost models do not scale with the dimensions of the pebble bed: the power conversion system, heat exchanger, preconstruction costs, DRACS system, heat rejection system, reactor metallic, pebble handling system, etc. Therefore to understand the drivers of capital cost, it is best to focus ones attention on the cost components that are directly effected by the pebble bed design, group them by how they scale and determine the sensitivity the overnight costs to these scaling groups. Three scaling groups were selected: capital that scales with the volume of the pebble bed, the capital the scales right with the height the reactor vessel and the capital of the structural graphite. The direct overnight costs of these components are presented in Figure 4-26, whereas the components that populate these scaling groups are presented in Table 4-30.

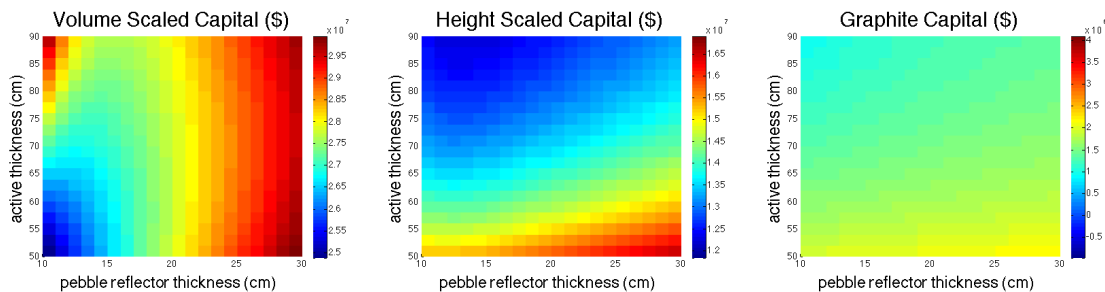


Figure 4-26. Overnight PB-FHR capital cost components as a function of PB-FHR core geometry for a power density of 28 MW/m³: Volume Scaled Capital Costs (left); Height Scaled Capital Costs (center); Graphite Capital Costs (right); the color scales span the same difference in overnight costs.

Table 4-30. Cost components in the three scaling groups.

PB-FHR Capital Scaling Groups	
Pebble bed	Reactor Initial Core
Volume Group	Flibe
Reactor Vessel	Reactor Vessel
Height Group	Reactor Building
Graphite	Reactor Graphite Internals

The pebble bed volume- and reactor vessel height scaled capital costs dominate with respect to the graphite scaled capital. One can see that the super position of the pebble bed volume- and the reactor vessel height scaled capital cost distributions roughly add up to the total overnight cost distribution.

The volume scaled capital costs are minimized for thin active regions with thin pebble reflectors. The flibe volume increases with increasing pebble reflector thickness, as the size of the pebble bed increases. Moreover, the geometry constraints result in the capital cost variation as a function of active thickness. Recall that the algorithm used to define the geometry of each instance of the PB-FHR holds the sum of the volumes of the expansion-, active- and converging regions constant for a given power density because it is assumed that most of the power in the PB-FHR is produced in these regions. However, the volume of the active region in the defueling chute is selected to impose a residence time for the fuel pebbles and geometry of the pebble entrance is defined such the fractions of pebble in the active and reflector regions are consistent with the active region and that the pebble reflector layer is at least 4 pebbles thick. These conditions are easily satisfied with a thin entrance region when the pebble reflector occupies a large fraction of the area as in cases with thin radial thicknesses. Increasing active thicknesses and pebble reflector thicknesses result in thicker pebble regions resulting in higher capital costs. The entrance geometry algorithm results in entrance regions with larger diameters than the active region and faults in the geometry definition for design points with pebble reflector radii less than 12 cm. This issue is exacerbated for design points with large active thicknesses because the area fraction of pebble bed is smaller.

The height scaled costs are minimized for PB-FHR design points with large active regions and thin pebble reflectors. Larger active thicknesses enable the pebble bed to accommodate more volume per unit height, minimizing the vertical build. Conversely, larger pebble reflectors necessitate taller expansion and converging regions, driving the vertical build.

The capital cost of graphite does not significantly impact the overall capital costs – this observation supports the design philosophy of replacing as much of the core or plenums with graphite as feasible.

4.4.2 Variable Costs

Variable costs are another important component of reactor optimization for the PB-FHR.

The variable costs include the revenue of sale of electricity produced from both the natural gas heat source and the nuclear heat source, the cost of nuclear fuel, the cost of natural gas and the costs of operations and maintenance. The net value of these cost components enable the PB-FHR pay back the capital costs generate profit.

Figure 4-27 presents the net annual variable costs for the PB-FHR project for several power densities.

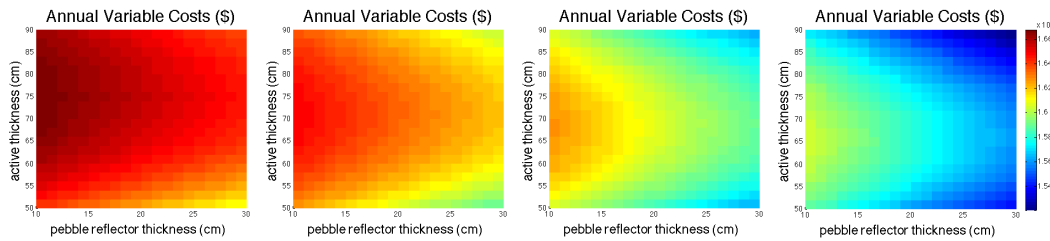


Figure 4-27. Net annual variable costs for the PB-FHR projects as a function of power densities 20, 25, 30 and 35 MWt/m³ from left to right.

The changes in variable costs as a function of the pebble bed design are dominated by the fuel costs. The fuel costs increase with increasing power density because increasing power density penalizes burnup.

Figure 4-28 compares the net sum of the variable costs as a function of pebble bed geometry to the LUEC of the PB-FHR scenario, both at a power density of 28 MWt/m³.

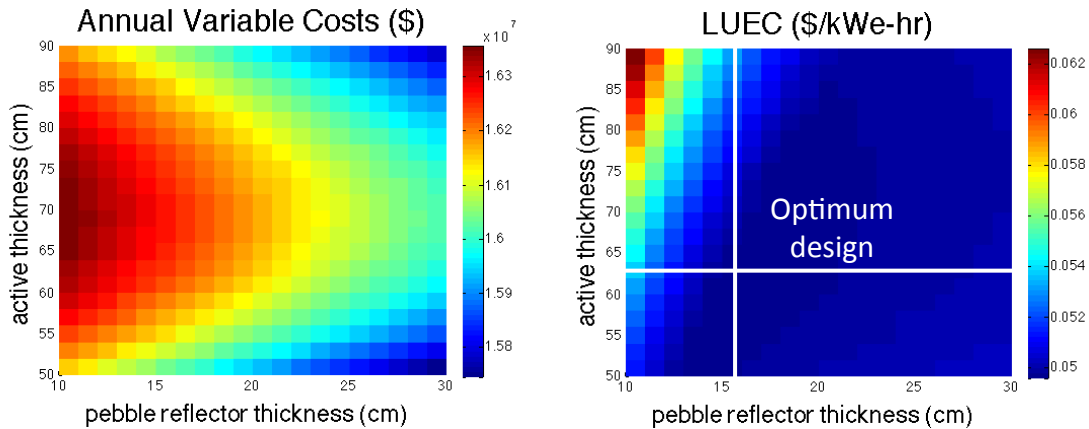


Figure 4-28. Sum of annual variable costs of PB-FHR capital costs as a function of PB-FHR core geometry for a power density of 28 MW/m³.

4.4.3 Reactor Module Lifetime

The service lifetime of the PB-FHR (ie the major period) is the final lower level integral economic metric. The reactor module lifetime dictates for how long the PB-FHR project can collect revenue. Figure 4-29 present the power production period of the PB-FHR in the cash flow model, limited either by thermal creep (assumed 60 years), radiation damage to the outer graphite reflector or core barrel for several power densities – note that the radiation damage of the reactor vessel is always less than for the core barrel so

if the reactor vessel will be below the radiation damage limit so long as the core barrel is below the same limit.

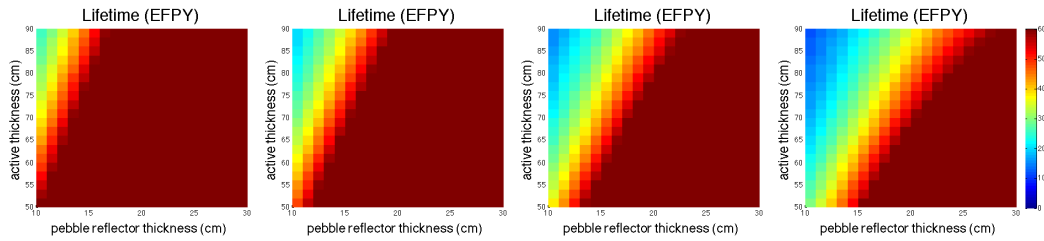


Figure 4-29. Service lifetime (time of decommissioning) for the PB-FHR projects as a function of power densities 20, 25, 30 and 35 MWt/m³ from left to right.

Figure 4-29 shows that there is a required thickness of the pebble reflector so that the PB-FHR can maintain integrity until the thermal creep limit. This thickness increases with increasing power density and increasing active thickness. A more shielding is required for PB-FHR point designs with larger active thicknesses because the neutron spectrum is harder than in point designs with smaller active thicknesses.

The distribution of project lifetimes as a function of pebble bed geometry is presented for the optimized power density with the limiting lifetime constraint – either thermal creep, radiation damage to the outer graphite reflector, radiation damage to the core barrel or buildup of gases in the core barrel – in Figure 4-30.

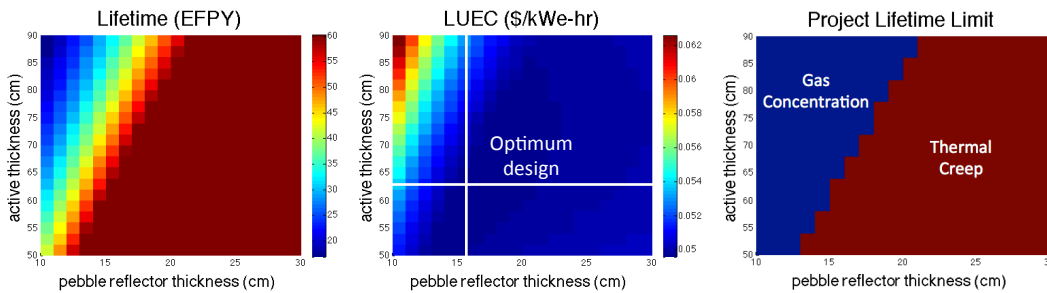


Figure 4-30. Operating Lifetime of PB-FHR projects as a function of PB-FHR core geometry for a power density of 28 MWt/m³: operating lifetime (left); levelized unit electricity cost distribution (center), lifetime limiting factor (right).

It is important to note that the optimized design points are all located on the interface where the service lifetime of the project transitions from being limited by power density to being limited by thermal creep. Additionally, the criterion for this limit is based on the gas concentration limit for the core barrel. Additional research should be performed to establish the gas concentration limit in the core barrel based on coupled neutronics (to define the radiation damage rate distribution and gamma heating in the core barrel), thermal (to estimate the temperature distribution) and structural analysis (thermal expansion and stress analysis with gas concentration dependent structural properties).

In addition to the lifetime of the entire project another economic consideration is the service lifetime of the inner reflector assembly (ie the minor period). When this assembly is replaced, a new assembly must be procured and the reactor cannot produce

power during this outage time. Furthermore, if this minor period is on the order of 10 EFPY the inspection of the reactor vessel can be performed during this outage. Moreover, this outage time pushes any additional revenue into the future making it less valuable. The alignment of the major period with the minor period for a specific point design is a value for minimizing the costs from replacing the central assembly. Figure 4-31 presents the service lifetime for the inner reflector assembly and final period as a function of pebble bed geometry for the optimized power density.

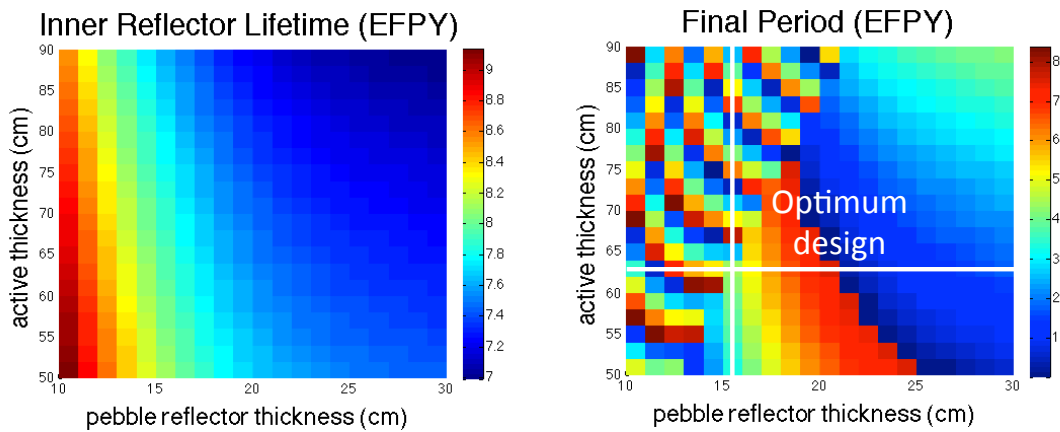


Figure 4-31. Operating Lifetime the inner reflector assembly of PB-FHR projects as a function of of PB-FHR core geometry for a power density of 28 MW/m³: operating lifetime (left); operating period of the final power producing period (right).

The final period is a measure of the alignment between the major and minor periods for a given PB-FHR design point. Large final periods imply these two periods are well aligned. When the service lifetime of the PB-FHR is limited by the dose to the outer graphite reflector the major period changes rapidly as a function of pebble bed geometry. In the design space where thermal creep limits the service lifetime, the major period is fixed at 60 EFPY and the final period changes slowly as the minor period changes. However, this does not appear to be a strong driver of economic performance as the optimized points are not located at the final period maxima.

4.4.4 Enforcement of Safety Criteria

In addition to the radiation damage constraints to which NAVE adheres, NAVE can be used to assess the safety metrics based on temperature reactivity coefficients and power density: fuel temperature reactivity coefficient, coolant temperature reactivity coefficient, peak fuel temperature and the ATWS coolant hot shutdown temperature.

The temperature reactivity coefficients are important safety metrics for the PB-FHR. Negative fuel- and coolant temperature reactivity coefficients are a fundamental attribute of the FHR safety basis. Figure 4-32 presents the temperature reactivity coefficients of the fuel and coolant for 28 MW/m³. Both of these temperature reactivity coefficients are negative for any pebble bed geometry in the assumed design space.

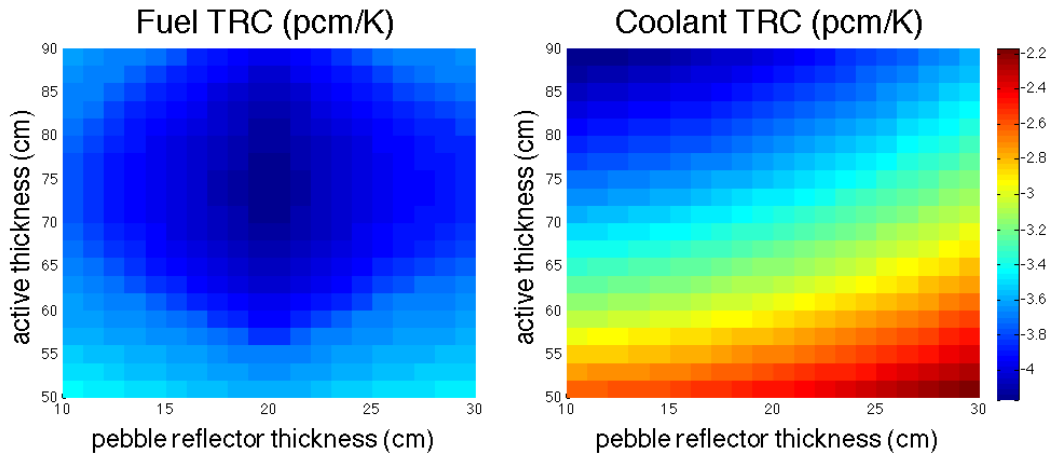


Figure 4-32. Temperature reactivity coefficients as a function of PB-FHR pebble bed geometry for a power density of 28 MWt/m³: (left) fuel kernel temperature reactivity coefficient; (right) coolant temperature reactivity coefficient – including density effect.

Low operating fuel temperature is another important safety metric. TRISO fuel particles should operate at temperatures below 1200 °C, however some low burnup pebbles will pass through the regions of high-intensity, soft-spectrum neutron flux next to the inner reflector, resulting in the peak power density (Todreas & Kazimi, 1990). The resulting peak fuel temperature was estimated by conservatively assuming the boundary conditions for a unit cell pebble heat transfer model were held at the outlet coolant temperature. These peak fuel temperatures are presented in Figure 4-33 – notice none of the peak fuel temperatures are above the temperature limit for TRISO particles.

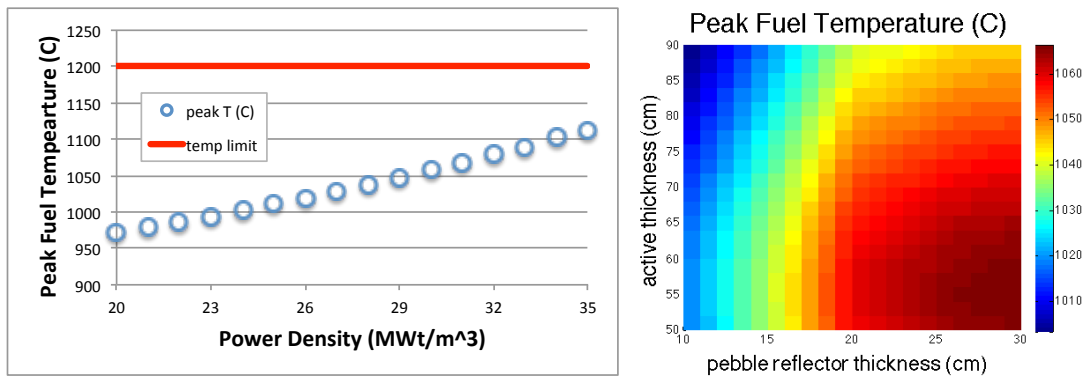


Figure 4-33. Peak fuel temperatures: (left) peak fuel temperature as a function of power density for the optimized geometry at each power density; (right) peak fuel temperature as a function of pebble bed geometry for a power density of 28 MWt/m³.

ATWS has been identified as the limiting case of the safety basis for the PB-FHR. The metric for safety for this transient is the time at temperature of the metallic internal structures and reactor vessel in this event. The long term temperature in an simplified ATWS model can be calculated based on the reactivity coefficients and initial characteristic temperatures of the fuel and the coolant (Cisenros, Laufer, Scarlet, Zweibaum, & Seifried, 2012). Estimates of the ATWS coolant hot shutdown temperatures are presented in Figure 4-34 – note that these are the characteristic

temperature of the coolant, not the temperature of the metallic components. The maximum temperature seen by the metallic structures can only be determined based on the design of the DRACS system (design work on the DRACS system is still on going). Coupled neutronics and thermal hydraulics are required to confirm the results of this simplified model.

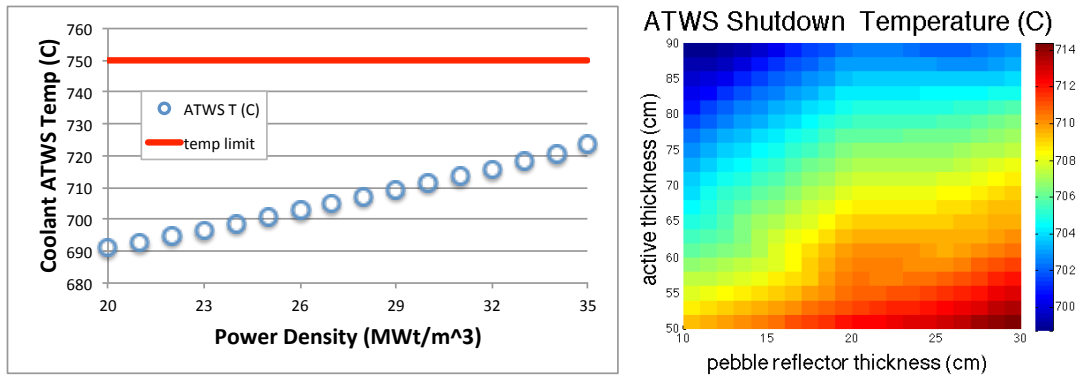


Figure 4-34. ATWS coolant hot shutdown temperatures: (left) ATWS coolant hot shutdown temperature as a function of power density for the optimized geometry at each power density; (right) ATWS coolant hot shutdown temperature as a function of pebble bed geometry for a power density of 28 MWt/m³.

The PB-FHR is robust in terms of its safety characteristics as it meets all of its safety objectives.

4.5 Sensitivity to High Uncertainty Cost Drivers

4.5.1 Natural Gas Costs

The cost of natural gas strongly influences the marginal cost of electricity. In many electricity markets the electricity prices are set by the marginal cost of fuel for natural gas turbine combined cycle plants. Therefore, the cost of natural gas feeds back into the profitability of the PB-FHR via the annual average cost of electricity.

The sensitivity of the NPV of a PB-FHR project to natural gas prices was assessed by using a high capacity factor (87%) gas turbine simple economics to estimate the annual average price of electricity for a gas turbine to break even (i.e. LUEC), then updating the new natural gas prices and setting electricity prices as this break even electricity price into the PB-FHR simplified economics model in NAVE and estimating the NPV of the project PB-FHR scenario corresponding to a given natural gas price. Values for the normalized capital costs and O&M costs for a gas turbine ('gas turbine combine cycle') were taken from the Annual Energy Outlook published by the Energy Information Administration (U.S. Energy Information Administration, 2013). The natural gas prices were varied from the latest annual average natural gas price from the 2013 natural gas price – 4.20 \$/1000ft³ – to the projected 2030 energy market in Europe (Commission of the European Communities, 2008). The interest rate for this study (gas turbine and PB-FHR cash flow models) was set to the IRR of the gas turbine combine cycle project providing base electricity assuming the 2013 natural gas prices and average electricity prices – 7.15% – assuming that this would be a reasonable rate of return a utility would

expect for a new electricity generation development. The NPV of the PB-FHR project at this interest rate as a function of wholesale natural gas prices and estimate natural gas price dependent average electricity prices are presented in Figure 4-35.

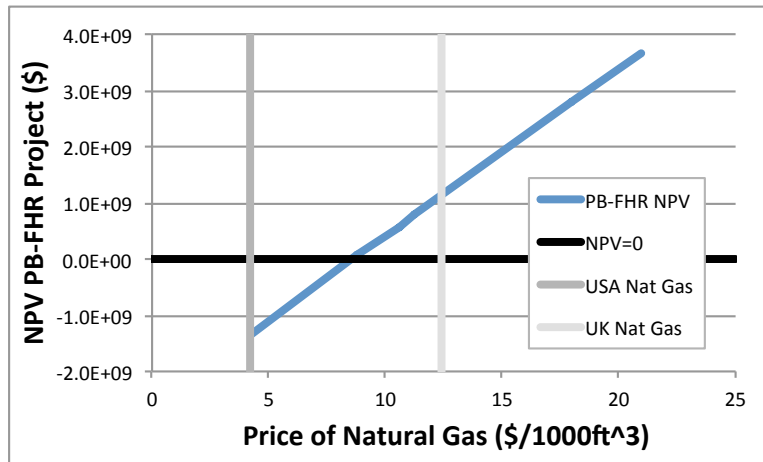


Figure 4-35. NPV of a PB-FHR project with updated average annual electricity prices as a function of the price of natural gas.

This study predicts that the PB-FHR becomes more economically attractive as the price of natural gas increases because the clearing price for electricity increases. At a natural gas price of 8.9 \$/1000ft³ the NPV of the PB-FHR project is zero; therefore, above this natural gas price a PB-FHR project has a better rate of return than a combine cycle gas turbine. The 2012 price of natural gas is above this threshold price for LNG imported to Japan, German, and the UK as shown in Figure 4-36.

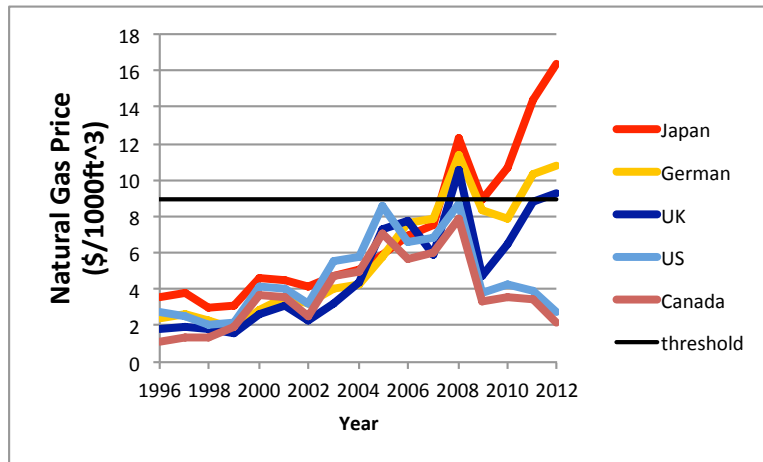


Figure 4-36. Historical International Natural Gas Prices (British Petroleum, 2013).

4.5.2 Ancillary Services

In addition to collecting revenue for electricity sales, the PB-FHR can provide ancillary services such as electricity spinning reserves and regulation. Spinning reserves on the California ISO is additional capacity on automatic generation control that is online and synchronized with the grid. Regulation corrects the mismatch between supply and

demand of electricity production (Bruynooghe, Eriksson, & Fulli, 2010). Electricity ISOs pay generators additional revenue to provide these ancillary services as shown in Figure 4-4. To perform this function the electricity generator must operate with spare capacity. This means that there is an opportunity cost associated with not producing electricity at the maximum rate. Moreover, load following results in reduced efficiency and increased O&M at the end of life – unfortunately, quantitative models for these effects are often not publically available. The degradation is mostly influenced by the number of cold starts, but also influenced by the number of warm and hot starts. The mode of operation of load following for the PB-FHR would look like a warm or hot start rather than a cold start, because the PB-FHR is always producing power from the nuclear heat source, mitigating these consequences of load following (Andreades, Lindsay, & Peterson, 2014) (Andreades, Lindsay, & Peterson, 2014) (Andreas, Scarlat, Dempsey, & Peterson, 2014). Quantifying the additional revenue from providing these services are dependent on identifying the optimum mode of operation based on the physical capability of the power conversion system, which is an area of on-going research and development. However, qualitatively ancillary services offer another avenue to gain revenue and provide value to the electric grid.

4.5.3 Resource Adequacy

California's wholesale energy market relies on a resource adequacy program to guaranty sufficient generating capacity (Department of Market Monitoring, 2013). This program requires load-serving entities to establish contracts for generation capacity to meet peak loads with margin for outages and other resource limitations. This contracted capacity must participate in both the day ahead- and the real time market through bids or self-schedules. This resource adequacy program provides an additional stream of revenue for the PB-FHR in addition to electricity generation.

Future developments in the grid make the PB-FHR attractive for this- and similar resource adequacy programs. As a large portion of California's generation capacity becomes intermittent with the increase of solar and wind capacity, the ramping capacity of the grid will become an increasingly important element of the resource adequacy program. Currently, the resource adequacy is contracted in single year contracts, but the California ISO is working to transition to multi-year contracts and are taking steps to develop a forward capacity market to incentivize investment in generating capacity.

Participating in this program adds a significant revenue stream for the PB-FHR project. This contribution is quantified as an add-on to the LUEC in Table 4-31 and as a fraction of the revenue from electricity generation in Table 4-32 for various historical resource adequacy prices and capacity factors (California Public Utilities Commission, 2013).

Table 4-31. Contribution of potential resource adequacy revenue to profit per kWe-hr.

Contribution to Profit (cent/kWe-hr)			Natural Gas Heat Source Capacity Factor (%)	
			87	30
Resource Adequacy Price (\$/kWe-month)	Median	2.2	0.33	0.53
	85 percentile	4.0	0.61	0.96
	Maximum	12.0	1.82	2.89

Table 4-32. Contribution of potential resource adequacy revenue to electricity revenue.

Fraction of Revenue (%)			Natural Gas Heat Source Capacity Factor (%)	
			87	30
Resource Adequacy Price (\$/kWe-month)	Median	2.2	7.5	11.5
	85 percentile	4.0	13.0	19.1
	Maximum	12.0	30.9	41.5

4.5.4 Process Heat

Providing industrial steam is a strategy for the PB-FHR to collect revenue when the price of electricity is low (Forsberg & Curtis, Meeting the Needs of a Nuclear-Renewable Electrical Grid with a Fluoride-Salt-Cooled High-Temperature Reactor Coupled to a Nuclear Air-Brayton Combined Cycle Power System, 2014). Steam produced from the nuclear heat source if it can be sold to industrial consumers for less than it would cost them to produce it themselves in natural gas boilers. Therefore, this market for nuclear heat effectively puts floor on the profits a PB-FHR can earn. Figure 4-37 presents the prices of electricity for two grids (California ISO and ERCOT) are compared to the effective electricity price for steam – the price of the owner could collect for selling the energy as steam rather than electricity – and manipulated to give the average price of energy for sold as electricity or sold as electricity and steam (when economically favorably).

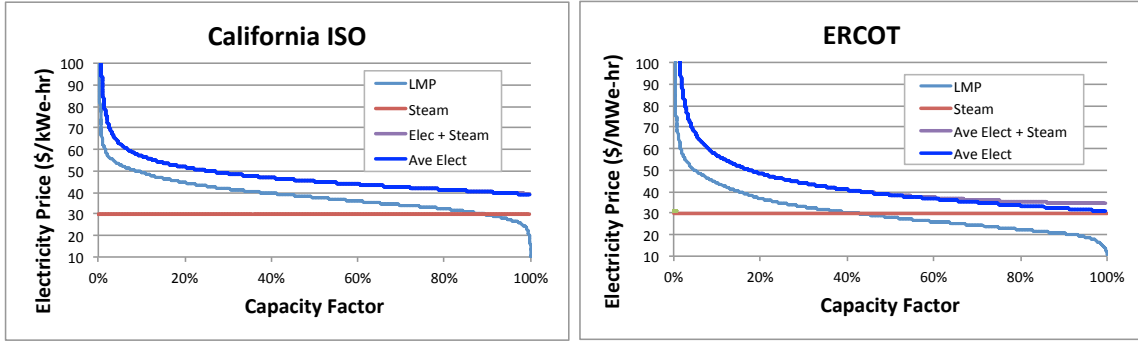


Figure 4-37. Electricity prices and effective price of steam organized into percentiles and presented as instantaneous prices, and average price of energy assuming either electricity sales only or electricity and steam sales: (left) California ISO – Comanche Peak NPP Oct 2012-Sept 2013); (right) ERCOT – HB_BUSAVG Oct 2012-Sept 2013.

$$\langle P \rangle = \frac{\sum_{i=1}^j P_{electricity}^i + \sum_{j=1}^k P_{steam}^j}{k} \quad \left| \quad P_{steam} \geq P_{electricity}^j \right.$$

Equation 4-14

The sale of process heat in the ERCOT market is more important because the electricity price on the ERCOT market is more frequently below the threshold limit to sell steam. Averaged over all the hours in the year the average effective price for electricity increases 0.03 and 0.34 cents/kWe-hr on the California ISO and ERCOT respectively.

4.5.5 Construction Costs

In addition to costs that directly affect the capital costs of the system, the small size of the PB-FHR has beneficial effects on the construction costs of the PB-FHR in terms of the effort required to assemble of power plant on site and in terms of the financial penalty of interest accrued before revenues can be generated (Rosner & Goldberg, 2011).

By limiting the geometry of all the PB-FHR components such that can be shipped by rail, these components can be factory fabricated reducing the effort required to assemble them on sight in terms of labor and in terms of the construction period. At the current stage of the reactor design it is hard to quantify exactly how much these costs can be reduced; however, simplification and volume minimization are key design philosophies in the PB-FHR project. Recall that the indirect costs of reactor include the construction costs and field office and engineering services. In the baseline economics model, it is assumed that these costs are halved. The sensitivity of LUEC to the fraction by which construction costs (scaled relative to a large nuclear power plant) and the construction time (the baseline construction time is 3 years) are presented in Figure 4-38.

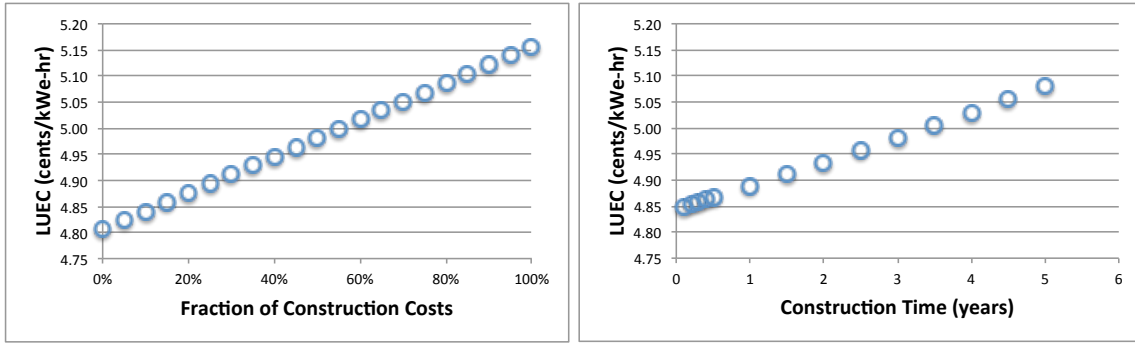


Figure 4-38. Sensitivity of levelized unit electricity costs to: (left) fraction of construction costs; (right) construction time.

These assumed small and modular reactor (SMR) construction advantages together result in cost savings of approximately 0.3 cents/kWe-hr.

4.5.6 Operations and Maintenance

Operations and maintenance has potentially negative implications on the economics of the PB-FHR because O&M normalized to power increases in SMRs relative to conventional gigawatt nuclear power plants (Carelli, Garrone, Locatelli, Mancini, Mycoff, & Ricotti, 2010). General Atomics’ O&M costs differed only by a small factor between the 350 MWt and 600 MWt power rating HTGRs – see Table 4-21; this results in O&M costs that are approximately 33% of the PB-FHRs total costs. Other studies of SMR costs estimate O&M should account for only 8-15% or 19% of the costs (Carelli, Garrone, Locatelli, Mancini, Mycoff, & Ricotti, 2010) (Shropshire, et al., 2007). Due the limited technology base with FHRs and the preliminary nature of the PB-FHR design status, it is not clear what the actual costs will be for O&M on a physical basis; however, a sensitivity study was performed to see how the LUEC responses to reductions in O&M; see Figure 4-39. Mitigating these O&M costs through design simplification to expected levels for other SMRs could significantly reduce the LUEC by a up to whole cent.

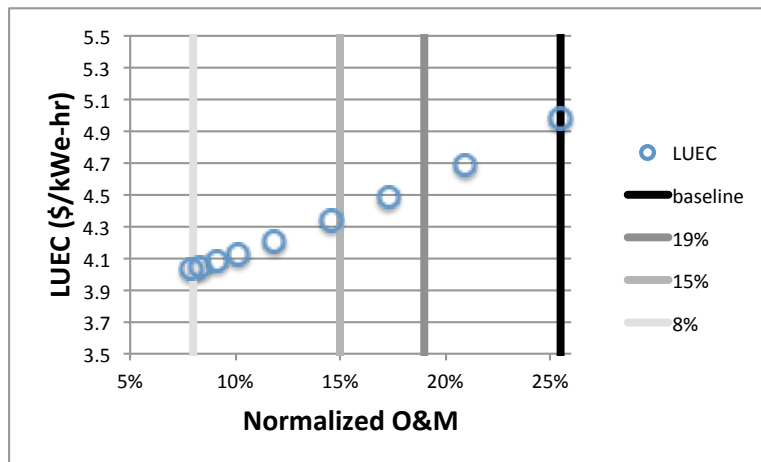


Figure 4-39. Sensitivity of levelized unit electricity costs to operations and maintenance costs

4.5.7 TRISO fabrication costs

Fuel fabrication costs for TRISO particle fuel are still speculative at best for a commercial scale facility. B&W has been tasked with standing up domestic TRISO particle fuel fabrication for cylindrical fuel compacts in the NNGP program (Phillips, Nagley, & Shaber, 2012). Institute of Nuclear and New Energy Technology (INET) at Tsinghua University is developing a commercial scale Pebble fuel fabrication facility for the HTR-PM project, but little information is available about the costs of this facility and the fuel it produces (Dong, 2011). In the past annular fuel pebbles have been fabricated and are technically feasible – however, most of the international community are either focused on cylindrical compacts or spherical pebbles for HTR, not for FHR (Wolf, Ballensiefen, & Frohling, 1975)s. Therefore, the experience basis for HTR TRISO fuel doesn't apply directly to the PB-FHR.

FHRs have a different safety basis that should reduce fuel costs relative to HTR fuel. HTRs rely on a combination of passive conduction and radiation heat transfer to remove decay heat in a severe accident – both of these mechanisms result in high fuel temperatures during transients. Further complicating the situation, is the fact that the high pressure in the coolant provides a release mechanism for any circulating radiation. To minimize this source term HTRs rely on the robust fuel form, requiring an expensive quality assurance program (Shropshire, et al., 2007). Liquid convection DRACS cooling during severe accidents in FHRs prevents the fuel temperatures from rising without active heating. Furthermore, since FHRs operate at low pressure there isn't a strong driving force to transport radiation from the core to the public – if the boundary is compromised only the gaseous radioactive isotopes have the potential for release. Both of these mechanisms increase the safety in FHRs; therefore, it is possible that FHRs can provide a comparable level of safety with lower fuel quality standards than those established by the HTR community – this can result in lower fuel costs.

TRISO fabrication costs are a significant cost in the PB-FHR project over its lifetime. Nuclear fuel costs account for 14% of all the NPV costs (in a conservative scenario with high natural gas utilization) of which fuel fabrication is 42%; therefore, fuel fabrication accounts for approximately 6% of the NPV of all the costs. Figure 4-40 presents the sensitivity of LUEC to TRISO fabrication costs.

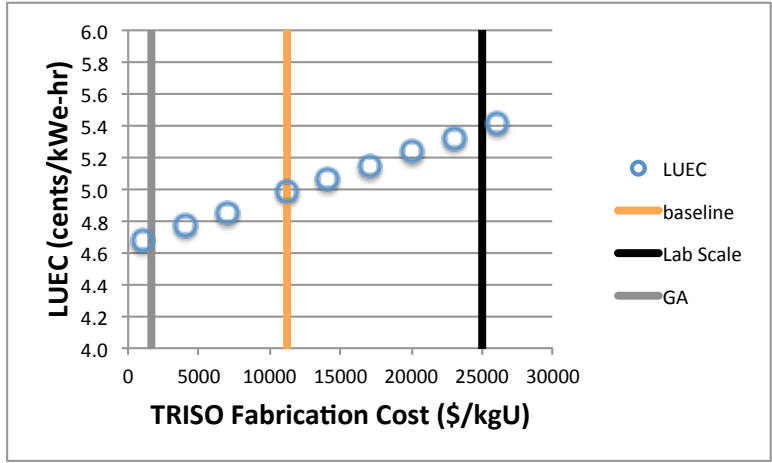


Figure 4-40. Sensitivity of levelized unit electricity costs to cost of TRISO fabrication.

It should be noted that due to the high burnup and excellent thermal efficiency of the gas turbine, the fuel costs of the PB-FHR (with the baseline assumed TRISO fabrication costs) are comparable to those of light water reactors and coal as shown in Figure 4-41.

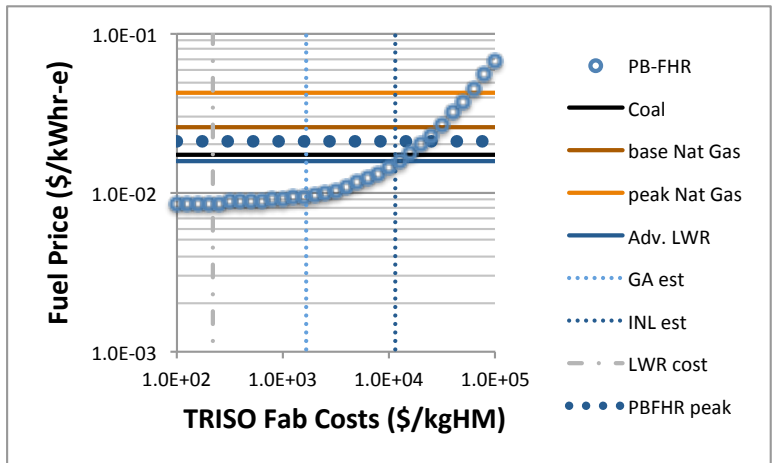


Figure 4-41. Comparison of fuel costs normalized to electricity produced for the PB-FHR and other energy sources.

4.5.8 Flibe costs

The costs of flibe are highly uncertain because there is not an established technical basis for handling flibe at an industrial level because of concerns around toxicity and there is no commercial scale lithium enrichment capability. Due to an efficient use of flibe its capital cost only constitutes about 0.5% of the NPV of the PB-FHR. Figure 4-42 shows that the LUEC of the PB-FHR is not very sensitive to the cost of flibe.

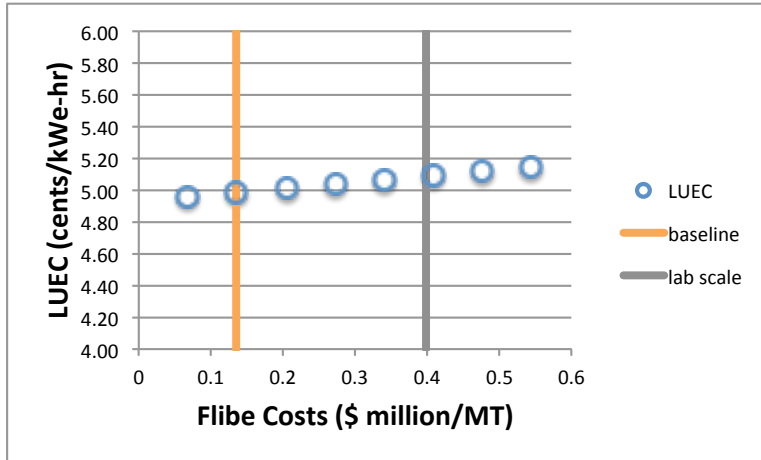


Figure 4-42. Sensitivity of leveled unit electricity costs to cost of enriched flibe.

4.5.9 Enrichment above 5w% LEU

The cost of uranium enrichment costs is another source of uncertainty. Uranium enrichment facilities in the United States are currently licensed to enrich up to 5w% ²³⁵U, whereas the PB-FHR was designed to use fuel enriched to 19.9w% to minimize fuel fabrication costs by maximizing burnup. Enrichment facilities would need to license their plant up to these higher enrichment levels – the Global Laser Enrichment facility and the American Centrifuge Plant are pursuing licenses to enrich uranium up to 8 and 10w% ²³⁵U respectively (United States Nuclear Regulatory Commission, 2012) (United States Nuclear Regulatory Commission, 2006). Unfortunately, preliminary neutronics scoping studies on lower enrichment options for the PB-FHR estimate burnups too low to be commercially viable – approximately 3000 MWd/MT with 8w% ²³⁵U enriched LEU. Procuring LEU enriched to these levels will likely involve a premium on top of the additional seperative work physically required to enrich uranium to 19.9w%. Enrichment comprises less of the costs of the PB-FHR than fuel fabrication – 22% of the fuel costs and 3% of the total NPV costs. Figure 4-42 shows that the sensitivity of the LUEC of the PB-FHR to the cost of enrichment.

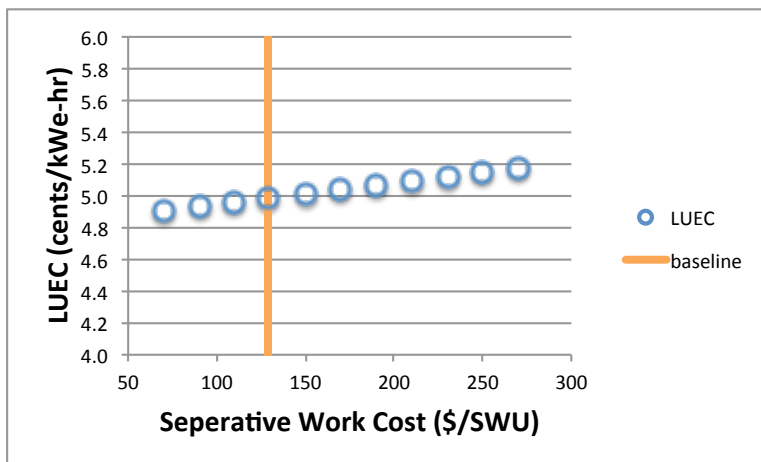


Figure 4-43. Sensitivity of leveled unit electricity costs to cost of enrichment.

4.6 Conclusions

Safety Design Criteria were derived from the System Functional Requirements for FHRs. A simplified cash flow model was developed to assess the economic implications of various design decisions. These safety design criteria, the simplified cash flow model and a surrogate model of performance parameters were integrated together in a flexible framework – NAVE – to optimize the PB-FHR.

This analysis suggests that the optimal design point (with respect to LUEC) within the PB-FHR design space lies between power densities of 25 and 29 MWt/m³ – balancing capital costs with effective fuel utilization. The concentration of gas accumulated in the metallic structural components of the PB-FHR begins to limit the service lifetime of the PB-FHR plant as power density increases – this can be mitigated by shielding from the inert graphite pebble reflector and shielding outside the solid reflector.

The economic performance in LEUC of the optimized design point was similar to that of the Mk1 PB-FHR and given the high uncertainties in the economics assumptions the Mk1 PB-FHR pebble bed core geometry was adopted as the baseline geometry.

The sensitivity of the economic performance of the PB-FHR to several high uncertainty cost drivers to guide future research. As the cost of natural gas electricity (due to scarcity or carbon taxes) increases the PB-FHR becomes more attractive. The flexibility of the PB-FHR's power conversion system should increase the revenue of the PB-FHR by providing ancillary services or using the process heat market to put a floor on the price of heat from the PB-FHR. The fuel fabrication costs of TRISO fuel strongly affect the LUEC and research should be performed to leverage the strong safety basis of FHRs into lower fuel costs for FHRs.

5 Mark 1 Design Characteristics

5.1 Introduction

Chapter 2 provided a framework for determining the equilibrium state of a continuously refueled fluoride salt cooled PB-FHR. This framework was used to perform design trade studies for the PB-FHR in Chapter 3 and when combined with a simplified economics model, the Mark 1 PB-FHR (Mk1 PB-FHR) was established as the baseline design for the PB-FHR because it performed similarly with respect to economics to an economically optimized PB-FHR. The focus of this Chapter is to present the detailed results from the neutronics and depletion analysis of the Mk1 PB-FHR (Andreades, et al., 2014).

5.2 Neutronics and Depletion Model

The design of the Mk1 PB-FHR is similar to the design assumed for the heterogeneous full-core parameter study. The design of the PB-FHR evolved over the summer of 2013 when UC Berkeley's FHR group began to perform power conversion system optimization, computer Aided Drafting (CAD) and physical arrangement – particularly in the design and implementation of the central reflector assembly. The thermal power of the nuclear heat source was reduced from 290 MWt to 236 MWt based on optimization of the power conversion system. Detailed design of the central reflector assembly was performed to integrate coolant channels into this assembly. Additionally, the defueling chute was moved out radially to accommodate the relocation of shutdown blades from outside the defueling chute to in the higher worth region inside the defueling chute. The entrance region was also moved out radially making it easier to impose the desired ratio of LEU- to inert graphite pebbles. Also, the primary heat exchanger was moved out of the reactor vessel. The full-core model was developed based on the design drawing presented in Figure 5-1 and Figure 5-2. The geometric dimensions defining the PB-FHR core are presented in Table 5-1. The geometric dimensions of the fuel pebble and TRISO particle are presented in Table 5-2. The geometric dimensions of the control elements are presented in Table 5-3. The average temperatures of the components of the PB-FHR are estimated using the same methodology presented in section 3.2.4.2; these temperatures are presented in Table 5-4. Finally a picture of the MCNP5 model is presented in Figure 5-3.

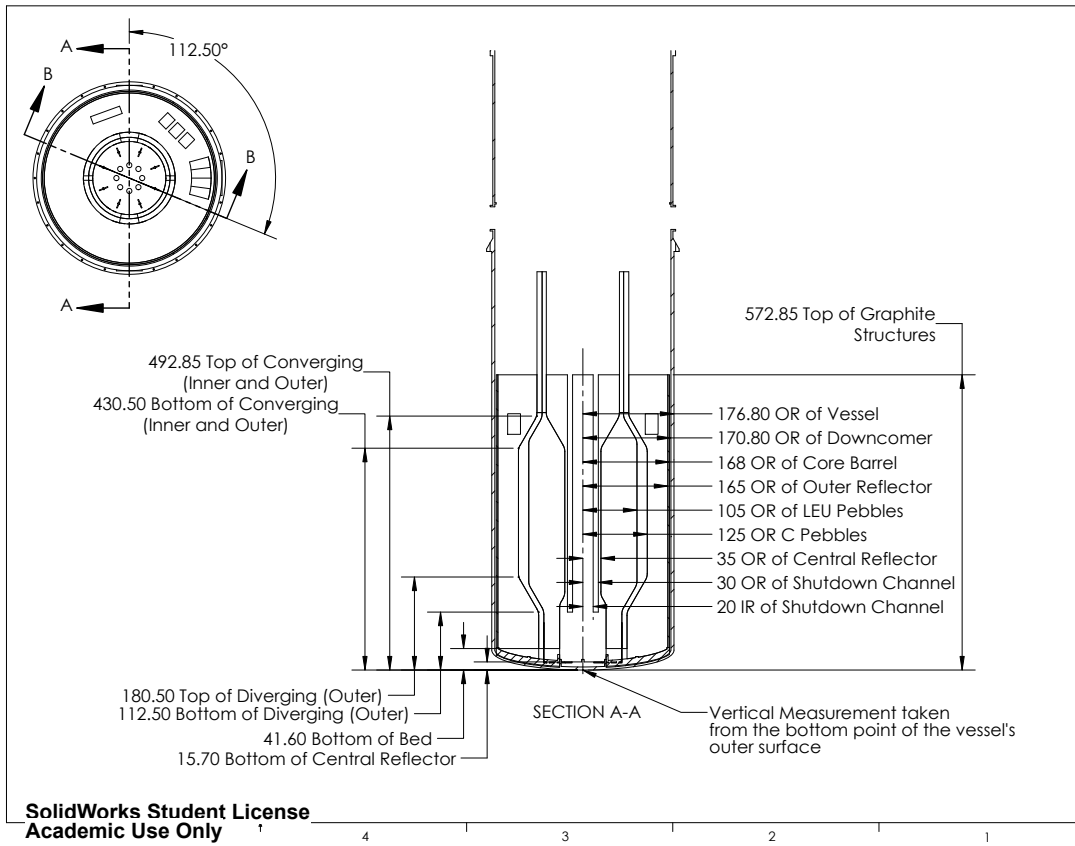


Figure 5-1. Design drawing of the Mk1 PB-FHR design: Section A-A

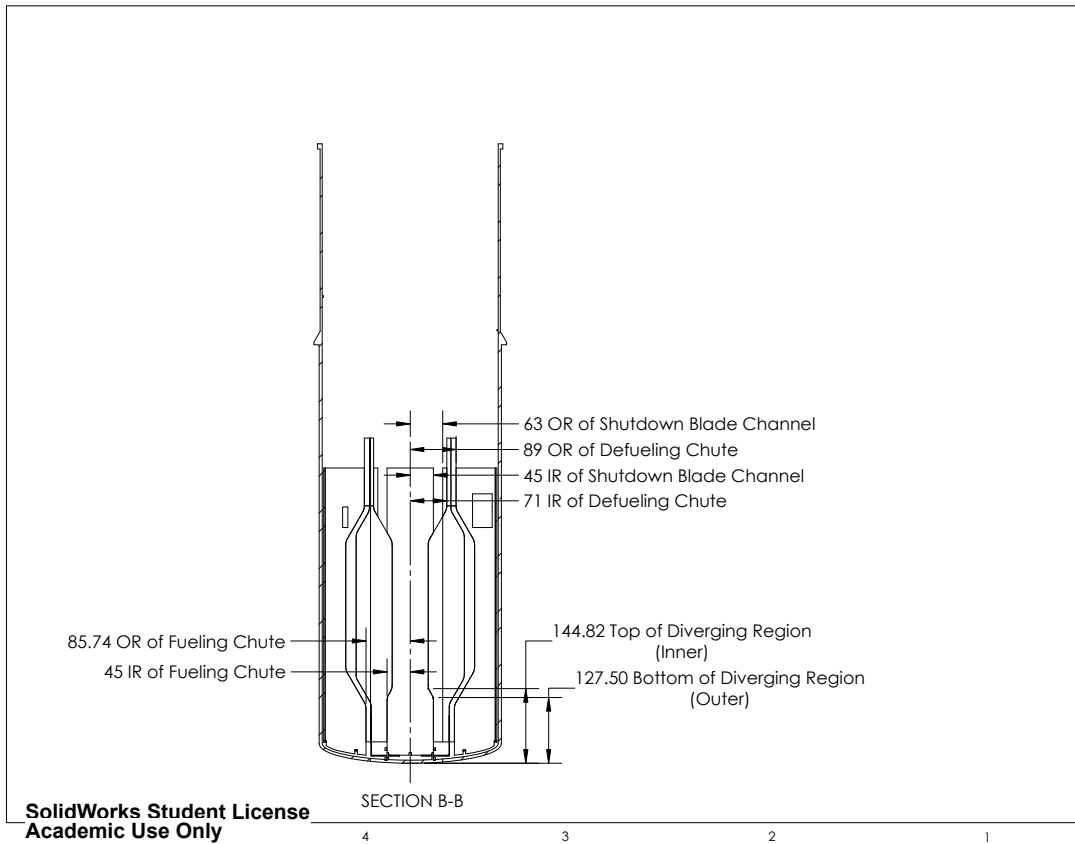


Figure 5-2. Design drawing of the Mk1 PB-FHR design: Section B-B

Table 5-1. Dimensions of Mark 1 PB-FHR pebble bed

Entrance Region	
Axial distance to bottom of the bed (cm)	41.60
Upper boundary of entrance region (inner) (cm)	127.50
Upper boundary of entrance region (outer) (cm)	112.50
Outer radius of central reflector (cm)	45.00
Radial interface of LEU- and reflector pebbles (cm)	75.41
Outer radius of pebble reflector (cm)	85.74
Expansion Region	
Angle of expansion (inner)(°)	30.00
Angle of expansion (outer)(°)	30.00
Active Region	
Lower boundary of active region (inner) (cm)	144.82
Lower boundary of active region (outer) (cm)	180.50
Upper boundary of active region (inner) (cm)	430.50
Outer radius of central reflector (cm)	35.00
Radial interface of wall effect- and bulk LEU pebbles (cm)	46.10
Radial interface of LEU- and reflector pebbles (cm)	105.00

Radial interface of wall effect- and bulk reflector pebbles (cm)	113.90
Outer radius of pebble reflector (cm)	125.00
Outer radius of solid graphite reflector (cm)	162.00
Outer radius of shield (cm)	164.70
Outer radius of core barrel (cm)	167.20
Outer radius of down comer (cm)	170.00
Outer radius of reactor vessel (cm)	175.00
Converging Region	
Angle of converging (inner) (°)	30.00
Angle of converging (outer) (°)	30.00
Defueling Chute Region	
Lower boundary of defueling chute region (cm)	492.85
Upper boundary of defueling chute region (cm)	572.85
Outer radius of central reflector (cm)	71.00
Radial interface of LEU- and reflector pebbles (cm)	83.79
Outer radius of pebble reflector (cm)	89.00
Inlet Plenum	
Thickness of reactor vessel at center line (cm)	5.00
Thickness of down comer at center line (cm)	9.09
Density of Components	
Density of graphite reflector (g/cm ³)	1.74
Density of SS-316 steel (core barrel / reactor vessel) (g/cm ³)	8.00
Density of flibe (600 °C) (g/cm ³)	1.9872
Density of flibe (650 °C) (g/cm ³)	1.9628
Density of flibe (700 °C) (g/cm ³)	1.9384
Density of Boron Carbide (g/cm ³)	2.4

Table 5-2. Dimensions of Mark 1 PB-FHR fuel pebble

Pebble Parameters	
Outer radius of inert pebble core (cm)	1.25114
Outer radius of active region (cm)	1.40000
Outer radius of pebble shell (cm)	1.50000
Pebble packing fraction (low PF) (%)	57
Pebble packing fraction (bulk) (%)	61
Density of inner pebble core (g/cm ³)	1.59368
Density of pebble shell (g/cm ³)	1.74
Carbon to heavy metal ratio (#)	300.8
Average density of pebble (g/cm ³)	1.745
TRISO Parameters	
Diameter of fuel kernel (μm)	400
Thickness of buffer layer (μm)	100
Thickness of inner PyC layer (μm)	35
Thickness of SiC layer (μm)	35
Thickness of outer PyC layer (μm)	35
TRISO packing fraction (%)	40
Enrichment of LEU (w%)	19.9
Density of kernel UC _{0.5} O _{1.5} (g/cm ³)	10.5
Density of buffer layer (g/cm ³)	1.0
Density of PyC layers (g/cm ³)	1.87
Density of SiC layers (g/cm ³)	3.2

Table 5-3. Dimensions of PB-FHR Mark 1 control elements.

Control Rods	
Number of Control Rods	8
Control rod channel diameter (cm)	10
Control rod width (cm)	8
Control rod thickness (cm)	2
Bottom axial position (withdrawn) (cm)	430.50
Bottom axial position (engaged) (cm)	112.50
Shutdown Blades	
Number of Shutdown Blades	8
Blade Length (cm)	18
Blade Width (cm)	4.5
Blade Thickness (cm)	1.5
Bottom axial position (withdrawn) (cm)	572.85
Bottom axial position (engaged) (cm)	180.50

Table 5-4. Temperatures in MCNP5 model of the Mark 1 PB-FHR

Fuel kernel (°C)	730
TRISO layers + matrix (°C)	700
Inner pebble core (°C)	710
Pebble shell (°C)	6840
Bulk coolant (°C)	650
Reflector pebble (°C)	700
Bulk coolant (pebble reflector) (°C)	700
Inlet/down comer Coolant (°C)	600
Outlet/ shutdown blade channel Coolant (°C)	700
Inner solid graphite (°C)	600
Inner porous graphite (°C)	600
Outer solid graphite (°C)	700
Outer porous graphite (°C)	700
Shield (°C)	600
Core Barrel (°C)	600
Reactor Vessel (°C)	600

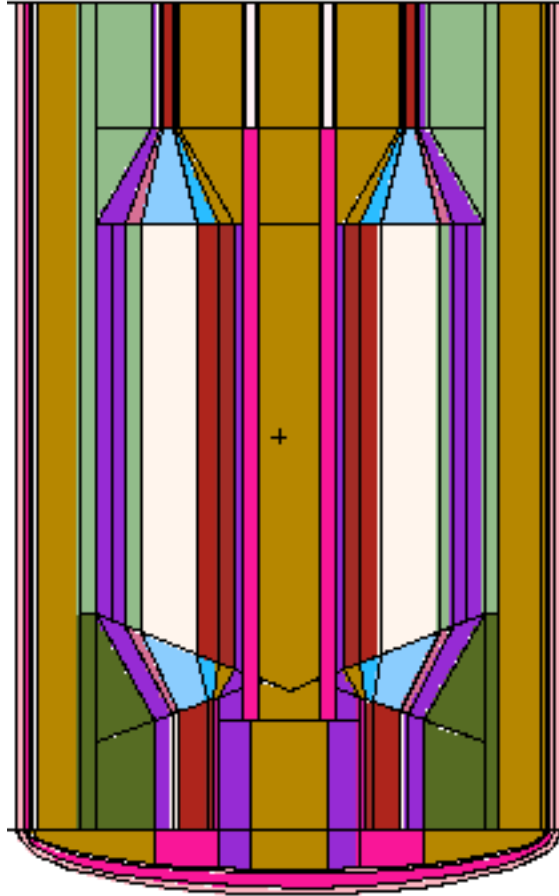


Figure 5-3. PB-FHR Mark 1 MCNP5 model

The inlet coolant channels were modeled by homogenizing the structural graphite with coolant assuming the coolant occupies a volume fraction of 40%; this region is referred to as the inner porous graphite reflector. This porous graphite region occupies the volume within 10 cm of the outer radius of the central graphite reflector assembly. The inner radius of the porous graphite cell established in the active region is maintained all the way down to the down comer as shown in purple in Figure 5-3. Similarly, another porous graphite region (60% graphite, 40% flibe) represents the coolant channels in the outer graphite reflector this region is referred to as the outer porous graphite reflector. This region occupies a volume 10 cm deep into the outer graphite reflector from the bottom of the active region to the top of the neutronics model; the outer radial dimension established in the active region is maintained all the way to the top of model as shown in mint in Figure 5-3.

The equilibrium depletion model is similar to the model used in the PB-FHR heterogeneous full-core parametric study defined in Section 3.2.4, only the power was changed from 290 MWt to 236 MWt. The pebble bed is subdivided into 20 (R,Z) zones with 8 burnup states (represented a unique pass through the core) in each of these zones. Since there is only a single fuel type, there is only one circulation progression object. Each of the 8 passes through the pebble bed corresponds to a pebble circulation

within the single circulation progression. Each radial segment corresponds to a fuel progression in each of these circulations. These fuel progressions are divided into 5 burnup states that correspond to the entrance-, expansion-, active-, converging- and defueling chute regions respectively. Figure 5-4 presents a conceptualization of the structure of this equilibrium depletion analysis model.

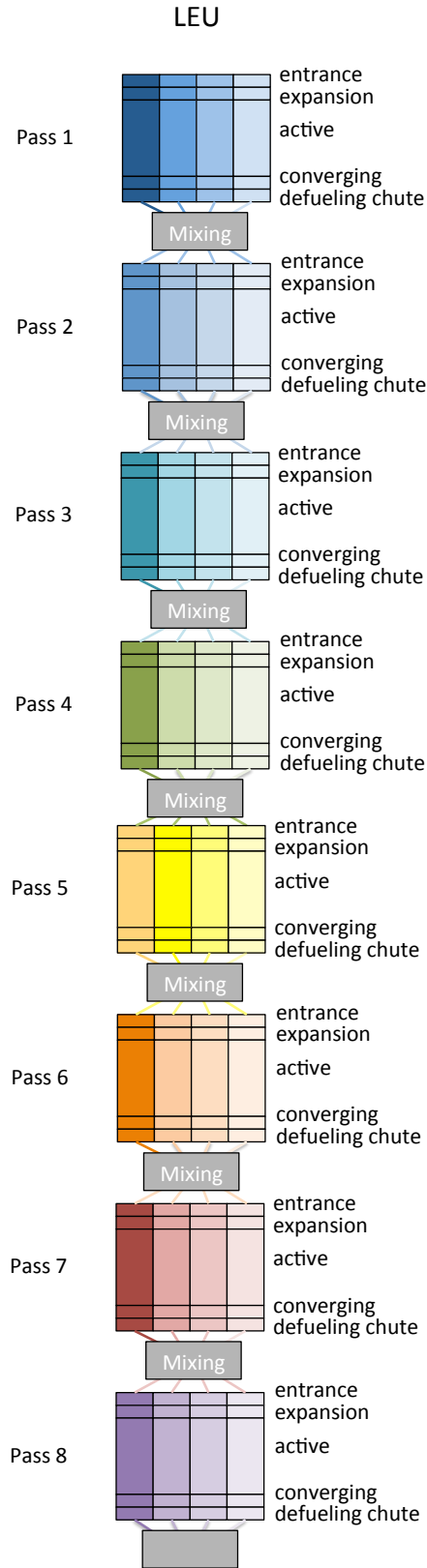


Figure 5-4. Equilibrium depletion scheme conceptualization for Mark 1 PB-FHR.

5.2.1 Heterogeneous Full-core model comparison

To accept the design analysis performed in Chapter 4 as applicable to the Mark 1 design one must quantify the differences between the the calculated values using the MPP full-core model (described in Chapter 2) and the calculated values of Mark 1 full-core model previously described.

To assess the representivity of the MPP full-core model to the Mark 1 full-core model a variety of neutronics parameters are compared including, BOL $k_{\text{effective}}$, average burnup, average fluence, neutron leakage probability, neutron flux spectrum, axial neutron flux distribution and axial power distribution. This set of parameters is not inclusive, but gives one a good sense of which- and to what extent these models are similar.

Several coarse neutronic and depletion parameters used to assess the representivity of MPP model and the Mark 1 model are presented in Table 5-5.

Table 5-5. Comparison of coarse full-core neutronics performance metrics for Mark 1 design to MPP PB-FHR neutronics model.

	Mk1 PB-FHR	MPP PB-FHR
BOL $k_{\text{effective}}$ (± 80 pcm)	1.30012	1.29398
Eq. Neutron Leakage (%)	14.0	17.4
Ave. Burnup (MWd/MT)	177948	174123

The neutron flux spectra averaged over all the fuel kernels in fuel pebbles are compared in Figure 5-5.

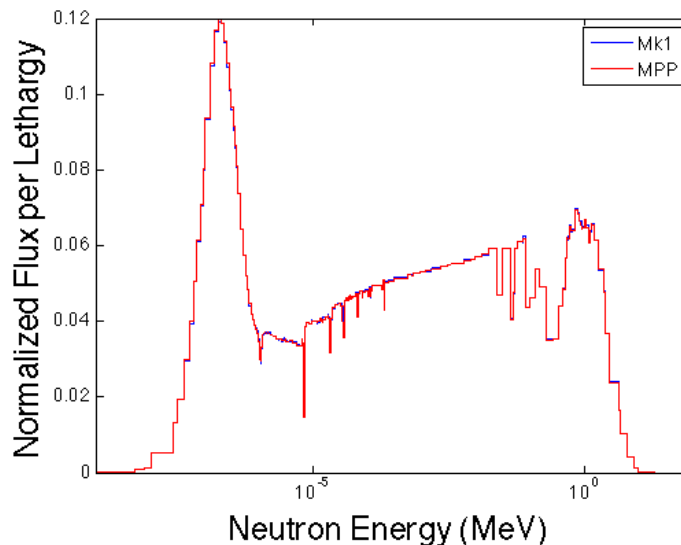


Figure 5-5. Flux distribution in the PB-FHR averaged over all the fuel kernels: MPP (red); Mark 1 (blue)

These neutron flux spectra are very similar.

The flux distribution decomposed into coarse axial segments is presented in Table 5-6.

Table 5-6. Axial flux distribution in Mk1 PB-FHR reactor core full-core models.

	Mark 1 Design	MPP PB-FHR
Average neutron flux entrance (n/cm ² s)	2.27*10 ¹⁴	2.01*10 ¹⁴
Average neutron flux expansion (n/cm ² s)	2.92*10 ¹⁴	3.90*10 ¹⁴
Average neutron flux active (n/cm ² s)	4.73*10 ¹⁴	4.69*10 ¹⁴
Average neutron flux converging (n/cm ² s)	1.92*10 ¹⁴	2.05*10 ¹⁴
Average neutron flux defueling chute (n/cm ² s)	4.48*10 ¹⁴	1.62*10 ¹⁴
Average Flux (n/cm ² s)	3.95*10 ¹⁴	3.86*10 ¹⁴

The flux is significantly lower in the MPP PB-FHR’s defueling chute design because this region is closer to the control rods nosing into the core – the power density in this region can be mitigated by lowering the shutdown blades during normal operation. However, operating with control elements partially inserted results the potential for inserting reactivity in control element removal accidents and in lower burnup.

This power density distribution decomposed into coarse axial segments is presented in Table 5-7.

Table 5-7 Axial power density distribution in PB-FHR full-core models

	Mark 1 Design	MPP PB-FHR
Power density entrance (MWt/m ³)	13.3	11.1
Power density expansion (MWt/m ³)	15.1	19.4
Power density active (MWt/m ³)	24.1	24.1
Power density converging (MWt/m ³)	9.4	10.0
Power density defueling chute (MWt/m ³)	3.4	1.2
Power Total (MWt)	20.2	19.6

Like the power density distribution, the power density is significantly lower in the MPP PB-FHR’s defueling chute design.

5.3 Results

5.3.1 Burnup

The maximum attainable burnup in the PB-FHR is a key performance parameter for economics, fuel utilization, and severe accident management (the burnup determines the amount of Cs-137 in the core: due to its 30-year half life and strong gamma emission, Cs-137 is the isotope that was primary cause of long-term ground contamination following the Chernobyl and Fukushima accidents). Key performance parameters related to burnup are presented in Table 5-8.

Table 5-8. Burnup performance metrics.

Average burnup (MWd/MT)	177948
FIMA (%)	17.4
Cesium 137 Inventory (moles/core)	22.4
Average cycle length (EFPD)	549
Pebble recirculation rate (pebble/day)	7185

The burnup evolution for each radial branch is presented in Figure 5-6. The pebbles accumulate burnup pretty evenly throughout the core for the first 4 passes, after which the pebbles need to be in the high flux inner region – radial branch 1 – to accumulate additional burnup. The saw-tooth burnup evolution is a result of the approximation of mixing the composition of the radial branches between each pass through the core as shown in Figure 5-4. Actual pebbles will accumulate unique burnup histories during each pass through the core, but by passing randomly through different paths on each pass will accumulate approximately similar burnup histories.

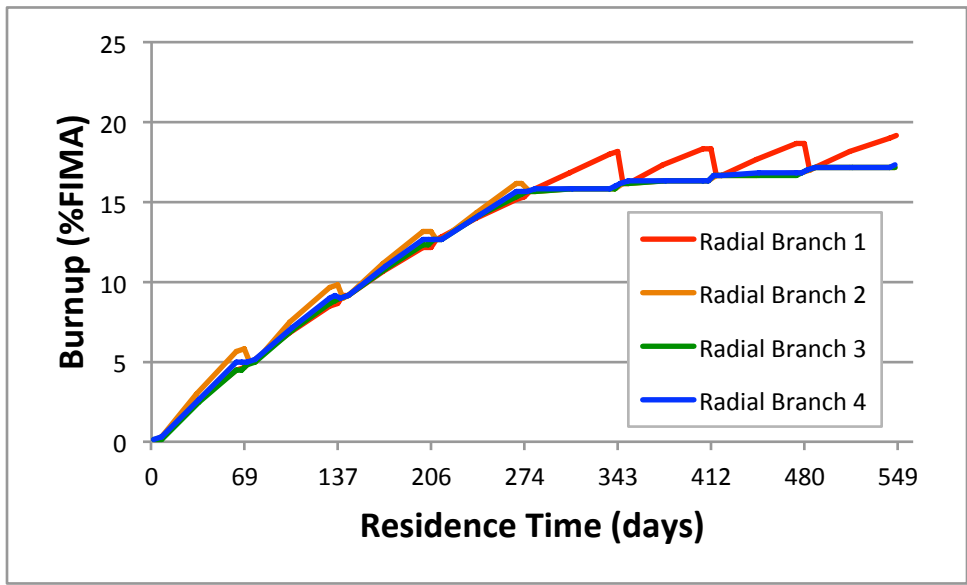


Figure 5-6. Evolution of Burnup in radial channels

5.3.2 Reactivity Coefficients

The temperature in the equilibrium PB-FHR neutronics model is adjusted in three ways. The material cards are changed to use different temperature dependent cross section libraries – $k_{\text{effective}}$ is only calculated at temperatures (600, 900 and 1200 K) with dedicated cross section libraries. For graphite-based materials, this is complicated because they required temperature dependent scattering kernels also – so for the graphite materials, $k_{\text{effective}}$ is only calculated at temperatures with dedicated cross section libraries and dedicated scattering kernel libraries. The temperature card is updated in the cells of the components with temperatures being adjusted. Finally, the density is updated for cells with flibe – including cells composed of homogenized

materials including flibe – according to the thermal expansion of the coolant (thermal expansion is much smaller for the graphite and fuel kernels); the material composition of the homogenized materials must be updated because of differential expansion rates of flibe and other components.

The temperature effects in the PB-FHR are segregated into 5 groups: fuel, coolant, graphite moderator, inner reflector, and outer reflector. The temperature of all the cells in these groups are set to a single temperature. The fuel group includes only the fuel kernels. The coolant group includes the flibe coolant in the active part of the pebble bed – the coolant in the pebble reflector is included in the outer reflector group. The moderator group includes all the carbonous materials in the fuel pebbles: TRISO layers, the TRISO graphite matrix, the pebble core and the pebble shell. The inner graphite reflector group includes the cells representing the solid portion of the central graphite reflector as well as the cells composed of homogenized flibe and graphite representing the coolant inlet into the pebble bed. The outer graphite reflector group includes the inter graphite pebbles in the the pebble reflector, the coolant in this pebble reflector, the solid outer graphite reflector and the cells composed of homogenized flibe and graphite represented the coolant channels carrying the flibe out of the core.

The reactivity coefficients for each of these groups are presented in Table 5-9. All the elements of the active pebble bed have negative temperature reactivity feedback, while the two graphite reflectors have positive temperature reactivity feedback.

Though the fuel and coolant have strongly negative temperature reactivity feedback, the positive sign of the inner- and outer graphite reflector suggest more complicated physics than assumed in the simplified ATWS model. Future work should include coupled thermal hydraulics and neutronics transient analysis to understand ATWS in the PB-FHR, though that is outside the scope of this work.

Table 5-9. Global reactivity coefficients.

Component	Temperature Reactivity Coefficient (pcm/K)
Fuel	-3.8
Coolant	-1.8
Graphite moderator	-0.7
Inner graphite reflector	+0.9
Outer graphite reflector	+0.9

These reactivity coefficients are used in the simplified ATWS model to find the asymptotic hot shutdown temperature. The hot shutdown coolant outlet temperature is dependent not only on this hot shutdown temperature but the temperature rise across the core required to remove decay heat with the DRACS in natural circulation mode of operation. The hot shutdown coolant outlet temperature is plotted as a function of this temperature rise as the design of the DRACS system has not yet been finalized.

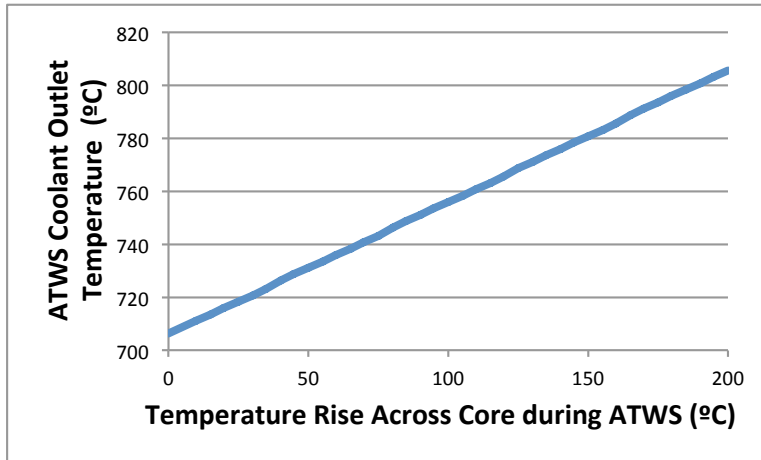


Figure 5-7. Equilibrium ATWS temperature in the PB-FHR as a function of temperature rise in passive DRACS cooling operation mode

5.3.3 Power Distribution

5.3.3.1 Local Fission Power Distribution

The local average fission power distribution is displayed in Figure 5-8. This is the expected power distribution and should be used in applications like system codes or the optimization of thermal hydraulics. This average power density is calculated similarly to the first pass power distribution in section 3.3.3.3.8. The 238-group flux is tallied on a mesh that encompasses the entire core. Corresponding 238-group macroscopic fission cross sections are calculated for each burnup state of the PB-FHR. The power in each mesh point is calculated by first determining if this point is in the boundaries one of the pebble bed's 20 (R,Z) nodes. If so, the multi-group flux is crossed with the corresponding multi-group macroscopic cross sections and multiplied by an assumed fission energy of 200MeV for each of the burnup states in this (R,Z) node. The average of these burnup states' power density is the reported average local fission power density. The discontinuities are interfaces between the control volumes of depletion zone – in reality these would be continuous.

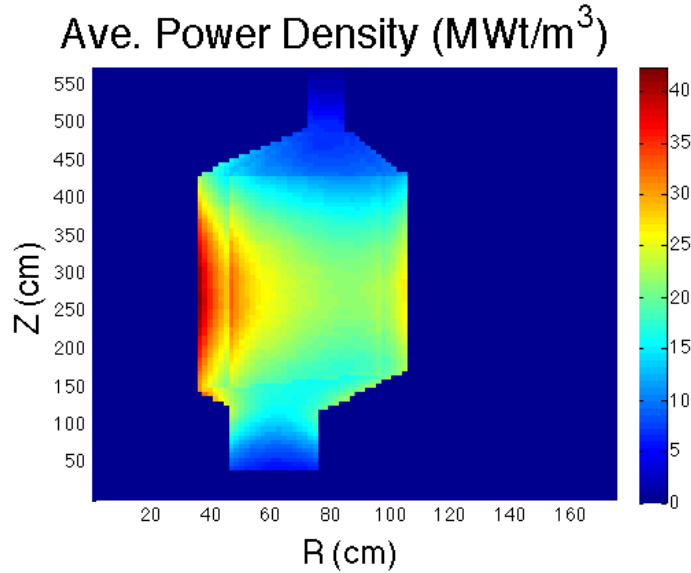


Figure 5-8. Full-core fission power density distribution.

5.3.3.2 First Pass Local Fission Power Distribution

The first pass local fission power distribution is displayed in Figure 5-9. This power distribution is used to identify the limiting case in terms of peak fuel temperature. This peak power density is calculated similarly to the average fission power distribution. The 238-group flux distribution and 238-group macroscopic fission cross sections used for the burnup averaged power distribution are reused in this calculation. The power in each mesh point is calculated by first determining if this point is in the boundaries one of the pebble bed's 20 (R,Z) nodes. If so, the multi-group flux is crossed with the multi-group macroscopic cross sections for the first pass pebble fuel in this (R,Z) node and multiplied by an assumed fission energy of 200MeV to account for the burnup peaking.

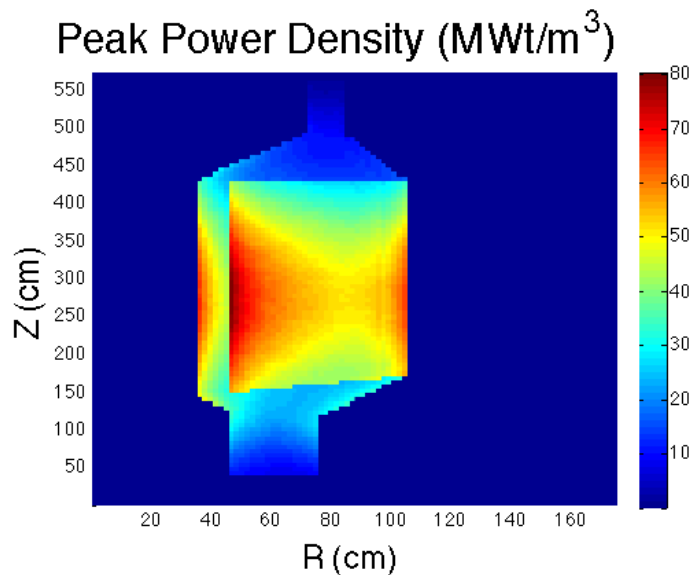


Figure 5-9. Full-core first pass (i.e. burnup peaking) fission power density distribution.

5.3.3.3 Power Peaking Factors Summary

The peak power densities from these power distributions are summarized in Table 5-10 along with estimates of the peak fuel temperature in a pebble unit cell with that power density. The peak to average power density in the PB-FHR pebble bed is approximately 4.

Table 5-10. Power density metrics

	(MWt/m ³)	(°C)
Fission power density averaged over pebble bed	20.2	726
Fission power density averaged over expansion-, active- and converging regions	23.0	733*
Peak average power density in PB-FHR	42.2	809
Peak local first pass (i.e. burnup peaking) pebble power density in PB-FHR	80.2	983

*Assumed temperature of neutronics model

5.3.4 Radiation Damage

Radiation damage limits the service lifetime of structural components of the PB-FHR. The radiation damage rate and gas production (in the core barrel) have been explicitly calculated for the inner graphite reflector, outer graphite reflector and core barrel are components as these structural integrity of these elements has been identified as vital to safety. The dose- and gas production rates and service lifetimes for these three components have calculated explicitly in sections 5.3.4.1-5.3.4.4.

5.3.4.1 Inner Graphite Reflector

The radiation damage rate and service lifetime for the inner graphite reflector are displayed in Figure 5-10. The peak radiation rate and the limiting service lifetime are presented in Table 5-11. This analysis shows that the inner graphite reflector assembly can survive longer than 10 EFPY assuming a 22.1 DPA limit. Therefore, the inner graphite reflector can be replaced while the core is shutdown for a 10-year reactor vessel service.

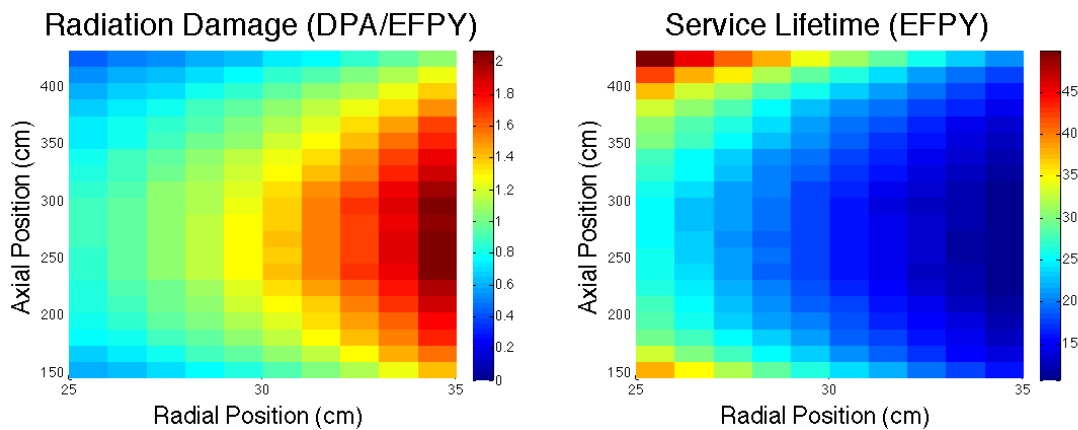


Figure 5-10. Radiation damage in PB-FHR Mark 1 inner reflector: (left) DPA rate (DPA/EFPY); (right) Time to reach DPA limit (EFPY).

Table 5-11. Inner reflector radiation damage parameters

Peak radiation damage rate (DPA/EFPY)	2.07
Service lifetime of inner graphite reflector assembly (EFPY)	10.7

5.3.4.2 Outer Graphite Reflector

The radiation damage rate and service lifetime for the outer graphite reflector assuming a 19.1 DPA limit are displayed in Figure 5-11. The peak radiation rate and the limiting service lifetime are presented in Table 5-12. Figure 5-11 and Table 5-12 show that the outer solid graphite reflector is very far from the outer graphite dose limit.

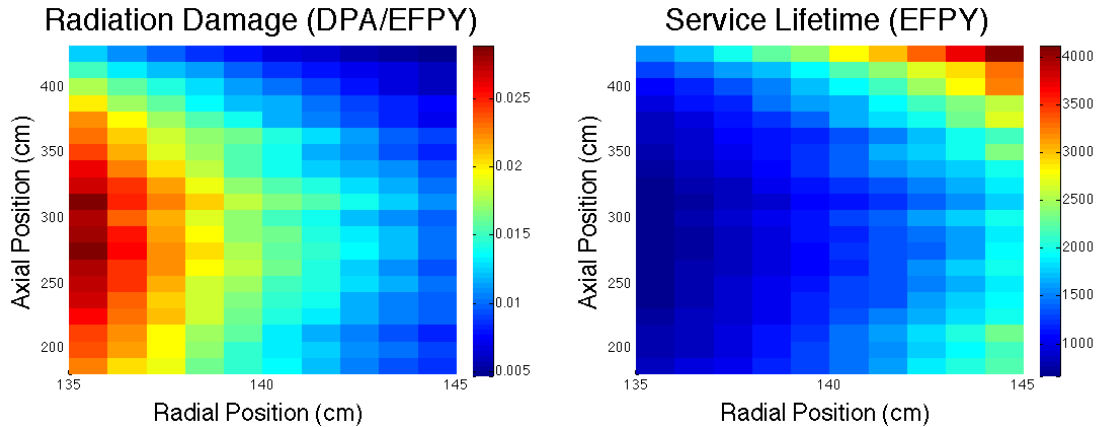


Figure 5-11. Radiation damage in PB-FHR Mark 1 outer reflector: (left) DPA rate (DPA/EFPY); (right) Time to reach DPA limit (EFPY).

Table 5-12. Outer reflector radiation damage parameters

Peak radiation damage rate (DPA/EFPY)	$2.88 \cdot 10^{-2}$
Service lifetime of outer graphite reflector assembly (EFPY)	662

5.3.4.3 Core Barrel

The radiation damage rate, gas production rate and corresponding service lifetimes for the outer graphite reflector are displayed in Figure 5-12 and Figure 5-13. The peak radiation rate, gas production rate and the limiting service lifetime are presented in Table 5-13. This gas production rate is the sum of the linear instantaneous production of gas particles a nonlinear production of helium from the activation of ^{58}Ni to ^{59}Ni as shown in Equation 5-1. This gas production limit is more limiting than the dose to the outer graphite reflector. The use of a neutron shield in the periphery of the solid radial reflector just inside the core barrel extends the operating life of the core barrel – without a shield the core barrel lifetime is less than 10 EFPY.

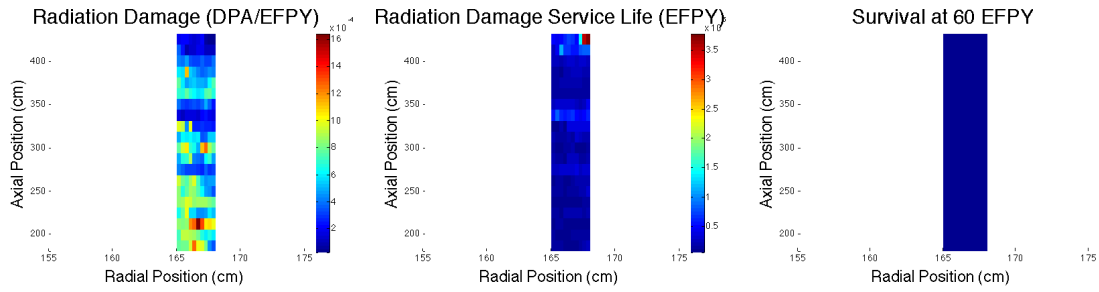


Figure 5-12. Radiation damage in PB-FHR Mark 1 core barrel: (left) DPA rate (DPA/EFPY); (right) gas production rate (ppm/EFPY).

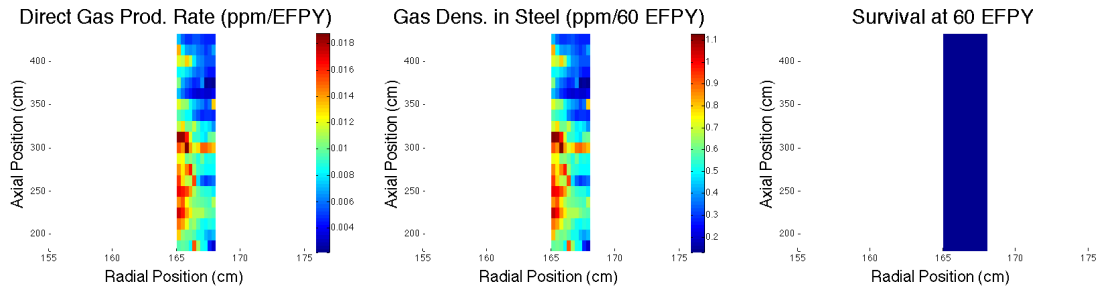


Figure 5-13. Service lifetime as a function of radiation damage limit: (left) service lifetime with respect to DPA; (right) service lifetime with respect to gas concentration.

Table 5-13. Outer reflector radiation damage parameters

Peak radiation damage rate (DPA/EFPY)	$1.20 \cdot 10^{-3}$
Peak gas production density at 60 EFY (ppm)	1.13
Nominal service lifetime of core barrel (EFY)	510

$$N_{\alpha}^{Ni-59}(T) = \Phi \sigma_{\alpha} \frac{\Phi \sigma_{\gamma} N_{Ni-58}}{(\Phi \sigma_{abs} + \lambda_{Ni-59})} \left(t + \frac{1 - e^{-(\Phi \sigma_{abs} + \lambda_{Ni-59})t}}{(\Phi \sigma_{abs} + \lambda_{Ni-59})} \right)$$

Equation 5-1

5.3.4.4 Full Core Radiation Damage

Future designs of the Mk1 PB-FHR might want to consider putting graphite- or steel elements as either test coupons or functional devices at an undetermined location in the core. So the dose rate for graphite and SS-316 were estimated everywhere in core and 10- and 60 EFY survival maps are produced to guide future design work. These survival maps are maps that identify whether a component of a given material would meet (blue) or exceed (red) the radiation damage constant proposed in Chapter 4. These material design constraints are reproduced in Table 5-14.

Table 5-14. Radiation damage limits for full core radial damage studies.

Material	10 EFPY limit	60 EFPY limit
Graphite	22.1 DPA*	19.1 DPA*
SS-316	10 DPA	10 DPA
SS-316	10 ppm	10 ppm

*Different radiation damage limits were used for the 10- and 60 EFPY cases because these correspond to inner reflector and outer reflector, which have different temperature dependent turnaround point dose limits.

The graphite dose rate distribution and survival maps are presented in Figure 5-14. The dose rate is very high within the active core, but the inner graphite reflector can survive for 10 years and the outer reflector should last over 60 years – the baseline life of plant.

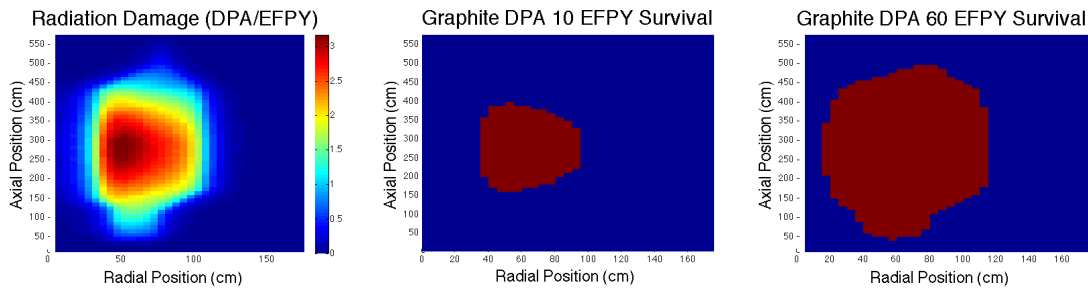


Figure 5-14. Full-core Graphite DPA parameters: (left) graphite DPA rate; (middle) 10 EFPY graphite radiation damage survival; (right) 60 EFPY graphite damage survival.

The SS-316 dose rate distribution and survival maps are presented in Figure 5-15.

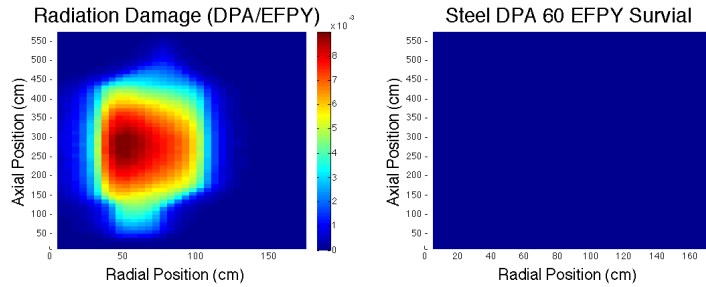


Figure 5-15. Full-core SS-316 DPA parameters: (left) graphite DPA rate; (right) 60 EFPY SS-316 damage survival.

The SS-316 gas production rate distribution and survival maps are presented in Figure 5-16. This gas production rate is the sum of the direct production of gas particles and the indirect production of helium from the activation of ^{58}Ni to ^{59}Ni as shown in Equation 5-1. The gas production rate severely limits the locations where structural steel can be implemented in the PB-FHR. In the current design the core barrel is the only structure explicitly shielded from neutron flux to mitigate gas production; future iterations of the PB-FHR must shield the lower portion of the reactor barrel to limit the accumulation of voids.

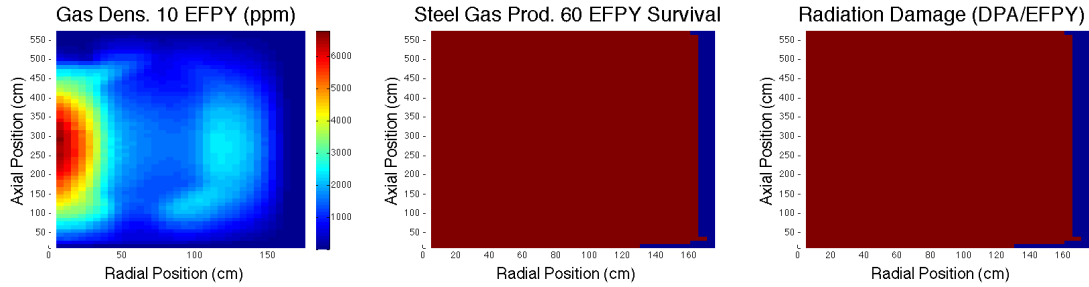


Figure 5-16. Full-core SS-316 gas production: (left) integrated gas accumulation at 10 EFPY (including N-59 mechanism); (middle) 10 EFPY gas production survival; (right) 60 EFPY gas production survival.

5.3.5 Control Element Worth

The Mk1 PB-FHR must have two diverse reactivity control mechanisms that must be able to independently maintain the core in a subcritical state even if the core is operated with xenon decayed, in a cold state producing no power – cold zero power (CZP). To model this state every material uses cross sections corresponding to 460 °C – the freezing point of flibe and the temperature cards are set to that temperature. The density of flibe is conservatively extrapolated up to 2.06 g/cm³ using the temperature dependent density correlation for flibe. Because the reactor must maintain subcriticality even after the ¹³⁵Xe has decayed away, for CZP all the ¹³⁵Xe is removed from these neutronics model.

5.3.5.1 Control Rods

The long-term CZP multiplication factor was estimate for the Mk1 PB-FHR for several control rod insertion patterns as shown in Figure 5-17. The dimensions of these control rods are presented in Table 5-3. The control rods are inserted in symmetric patterns – these insertion patterns are presented in Table 5-15. Only three control rods are required to maintain long-term subcriticality in the Mk1 PB-FHR. Therefore, the PB-FHR could maintain subcriticality assuming a single failure of a bank of rods, even if the control rods are engaged in banks of 2 or 4 control rods.

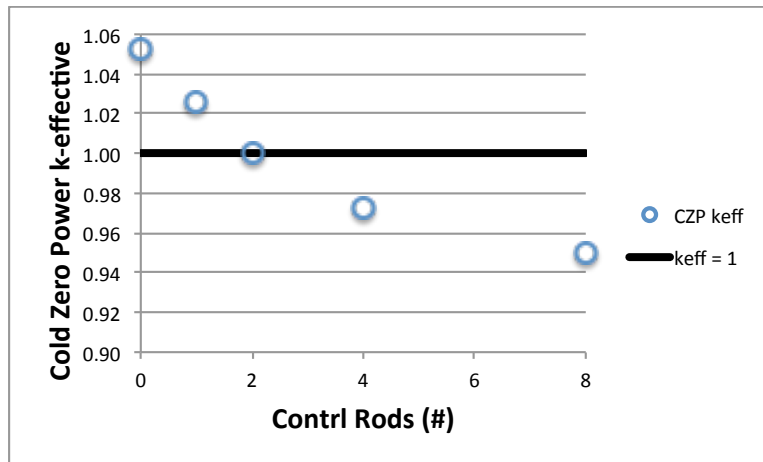


Figure 5-17. longterm cold zero power k-effective as a function of number of control rods inserted.

Table 5-15. Control rod worth study case definitions: (Y) yes, the control rod is engaged, (N) the control rod is not engaged.

Control Rods	Azimuthal Position of Control Rod							
	0	45	90	135	180	225	270	315
0	N	N	N	N	N	N	N	N
1	Y	N	N	N	N	N	N	N
2	Y	N	N	N	Y	N	N	N
4	Y	N	Y	N	Y	N	Y	N
8	Y	Y	Y	Y	Y	Y	Y	Y

5.3.5.2 Shutdown Blades

The long-term CZP multiplication factor was estimate for the Mk1 PB-FHR for several shutdown blade insertion patterns as shown in Figure 5-18. The dimensions of these shutdown blades are presented in Table 5-3. The shutdown blades are inserted in symmetric patterns – these insertion patterns are presented in Table 5-16. Only four shutdown blades are required to maintain long-term subcriticality in the Mk1 PB-FHR. Therefore, the PB-FHR could maintain subcriticality even assuming a single failure.

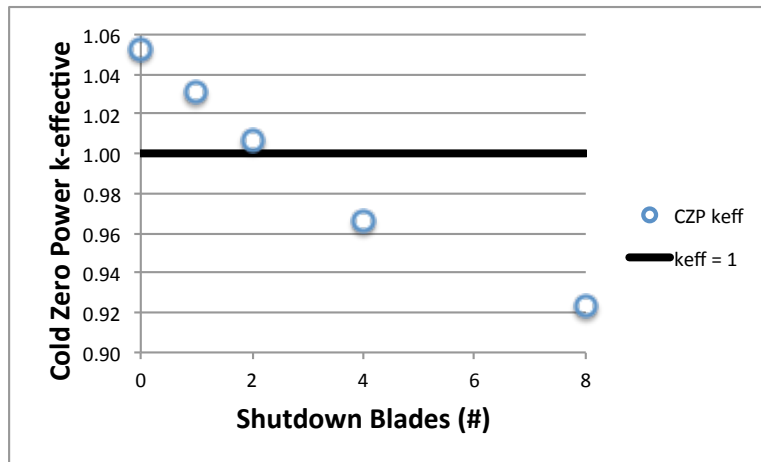


Figure 5-18. longterm cold zero power k-effective as a function of number of shutdown blades inserted.

Table 5-16. Control blade worth study case definitions: (Y) yes, the control blade is engaged, (N) the control blade is not engaged.

Shutdown Blades	Azimuth Position of Shutdown Blade							
	22.5	67.5	112.5	157.5	202.5	247.5	292.5	337.5
0	N	N	N	N	N	N	N	N
1	Y	N	N	N	N	N	N	N
2	Y	N	N	N	Y	N	N	N
4	Y	N	Y	N	Y	N	Y	N

8	Y	Y	Y	Y	Y	Y	Y	Y
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5.3.6 Decay Heat Curve

The decay heat curve for the Mk1 PB-FHR was calculated and presented in Figure 5-19. The decay heat curve was calculated by consolidating the equilibrium state material vectors of each of the burnup states into a single isotopic inventory for the entire core. ORIGEN is used to perform a depletion analysis on this core wide equilibrium isotopic inventory keeping track of decay heat produced by the three classes of isotopes in ORIGEN – actinides, fission products and activation products – at each timestep. This heat generation curve does not include subcritical fission multiplication preceding the insertion of control rods, but this additional heating occurs for only a short time.

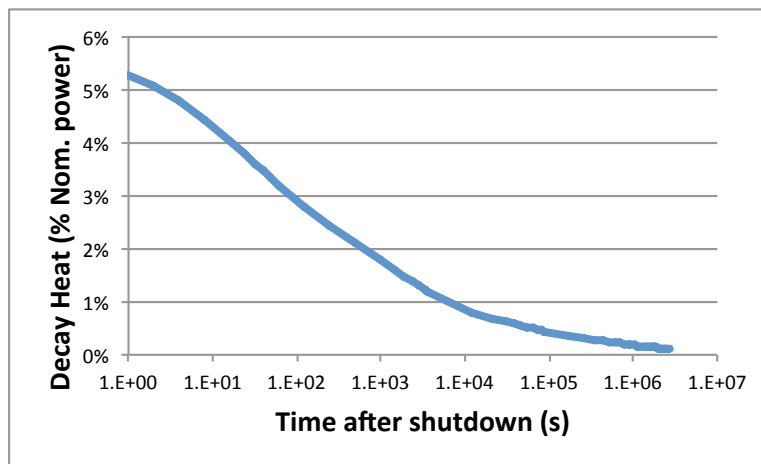


Figure 5-19. Decay heat curve for PB-FHR Mark 1 from equilibrium state

5.3.7 Tritium Production

The equilibrium tritium production rate, average flux and cross section are presented in Table 5-17. The flux and reaction rates are presented in Table 5-17 so that evolution of the tritium production rate can be estimated starting from an initial state with an arbitrary concentration of ${}^6\text{Li}$ according to Equation 5-2. The time to establish the equilibrium state is increased by the volume of coolant circulated outside the PB-FHR pebble bed. The additionally coolant inventory is taken account for using the loop to core coolant volume ratio in Equation 5-2.

Figure 5-20. Tritium production evolution from an initial state of 100 atomic ppm ${}^6\text{Li}$ presents the evolution of tritium production rate assuming an initial concentration of 100 atomic ppm (relative to ${}^7\text{Li}$) ${}^6\text{Li}$ and a loop to core coolant volume ratio of 4. At equilibrium 64.4% of the tritium is produced from reactions on the ${}^6\text{Li}$.

Table 5-17. Tritium production rate and nuclear data

Equilibrium Tritium Production (mole T/day)	$2.27 \cdot 10^{-2}$
Equilibrium Tritium Production (Cu T/day)	285
Flux in Coolant (n/cm ² s)	$3.41 \cdot 10^{14}$
Li-6 (n,T) (bn)	148.026
Li-6 (n,abs) (bn)	148.032
Be-9 (n,alpha) (bn)	$3.63 \cdot 10^{-3}$
Li-7 (n,T) (bn)	$1.00 \cdot 10^{-3}$

$$\frac{dN_T}{dt} = \Phi \sigma_{Li-7}^T N_{Li-7} + \Phi \sigma_{Li-6}^T \left(N_{Li-6}^o e^{-\frac{\Psi_{coolant}^{core}}{\Psi_{coolant}^{loop}} \Phi \sigma_{Li-6}^{abs} t} + \frac{\Phi \sigma_{Be-9}^{\alpha} N_{Be-9}}{\Phi \sigma_{Li-6}^{abs}} \left(1 - e^{-\frac{\Psi_{coolant}^{core}}{\Psi_{coolant}^{loop}} \Phi \sigma_{Li-6}^{abs} t} \right) \right)$$

Equation 5-2

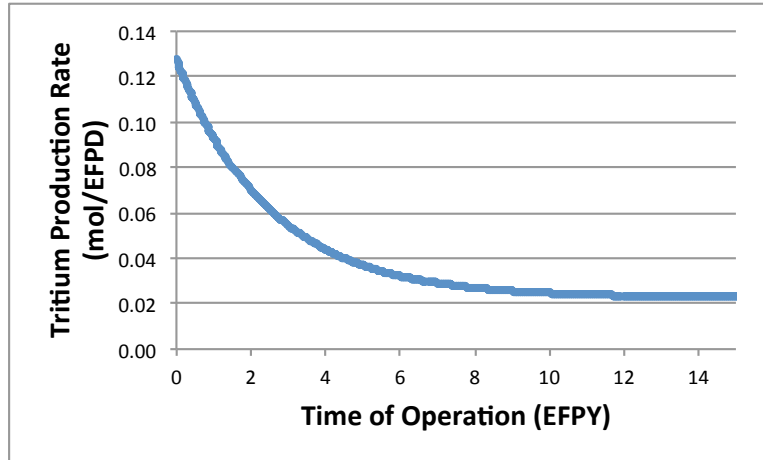


Figure 5-20. Tritium production evolution from an initial state of 100 atomic ppm ⁶Li

5.4 Conclusions

A detailed description of the Mk1 PB-FHR pebble bed core and its neutronics and depletion models is presented in the first section of this Chapter.

A set of high level reactor characteristics for the Mk 1 PB-FHR was compared the MPP PB-FHR for the same set of high level pebble bed core design parameters (power density, active thickness, pebble reflector thickness and active height). These reactor designs are similar enough so that the parametric trends from Chapter 3 and the Economics implications from Chapter 4 apply to the Mk1 PB-FHR.

The detailed neutronics characteristics for the Mk1 are presented for several safety and economic performance parameters including: burnup, temperature reactivity coefficients, the power distribution, radiation damage distribution, control element worth, decay heat evolution and tritium production. The Mk1 PB-FHR satisfies the PB-

FHR safety criteria outlined in chapter 4. The fuel, moderator (pebble core, pebble shell, graphite matrix, TRISO layers) and coolant have bulk negative temperature reactivity coefficients. The fuel temperatures are well within their temperature limits. However, shielding should be added to maintain the metallic structures at the bottom of the reactor vessel with their gas concentration limit.

6 Conclusions and Future Work

This project performed design and analysis that optimized the core and reflector design of the Pebble Bed Fluoride Salt Cooled High Temperature Reactor (PB-FHR) – a small and modular reactor (SMR) annular pebble bed fluoride salt cooled high temperature reactor (FHR). Effective passive decay heat removal by the direct reactor auxiliary cooling system (DRACS), replaceable central graphite reflector assembly and the inert pebble reflector assembly enable the PB-FHR to operate with high power densities (20-30 MWt/m³), thereby minimizing the capital costs of the reactor core.

A new computational tool– Burnup Equilibrium Analysis Utility (BEAU) – was developed to couple continuous energy Monte Carlo neutron transport and point depletion analysis to effectively identify the equilibrium state of the PB-FHR core. BEAU can be applied to any continuously refueled nuclear reactor system core analysis. BEAU uses a novel equilibrium depletion algorithm, the Multiple Burnup State Method (MBSM), which is an evolution of the full-core method proposed by Dr. Fratoni. The MBSM calculates burnup dependent cross sections in each depletion control volume accurately modeling depletion analysis in systems the composition of which changes significantly between the beginning- and end of cycle. Furthermore, BEAU can model advanced pebble bed fuel cycles with multiple fuel types each with independent cycle lengths. The results from BEAU was satisfactorily benchmarked against published results from deterministic equilibrium depletion codes VSOP and PEBBED for a high temperature gas cooled reactor pebble bed reactor.

Several parametric studies were performed to guide the optimization of the PB-FHR. These parametric studies covered a broad design space, assessed the implications of design changes on reactor safety and economics performance parameters (temperature reactivity coefficients, burnup, etc.) and the underlying reactor physics that are responsible for these effects.

A pebble fuel parametric study determined that pebble fuel should be designed with a carbon to heavy metal ratio (C/HM) less than 400 to maintain negative coolant temperature reactivity coefficients.

Seed and thorium blanket-, seed and inert pebble reflector- and seed only core configurations were investigated for annular FHR PBRs – the C/HM of the blanket pebbles and discharge burnup of the thorium blanket pebbles were additional design variable for core configurations with thorium blankets. Either a thorium blanket or graphite pebble reflector is required to shield the outer graphite reflector enough to extend its service lifetime to 60 EFPY. The fuel fabrication costs and long cycle lengths of the thorium blanket fuel limit the potential economic advantages of using a thorium blanket. Therefore, the seed and pebble reflector core configuration was adopted as the baseline core configuration.

The design space of an SMR annular FHR PBR was explored, developing correlations of reactor performance parameters (burnup, reactivity coefficients, radiation damage rates,

peak power, etc.) as a function of the design parameters (power density, active core thickness, pebble reflector thickness and active core height) within the explored design space. Safety design criteria were derived and these physical constraints were applied along in a design tool, Nuclear Application Value Estimator, that evaluated a simplified cash flow economics model based on estimates of reactor performance parameters calculated using correlations based on the results of parametric design studies for a specific PB-FHR design and a set of economic assumptions about the electricity market to evaluate the economic implications of design decisions.

The sensitivity of the economic performance of the PB-FHR was quantified to several high uncertainty cost drivers to guide future research. As the cost of natural gas electricity (due to scarcity or carbon taxes) increases the PB-FHR becomes more attractive. The flexibility of the PB-FHR's power conversion system should increase the revenue of the PB-FHR by providing ancillary services or using the process heat market to put a floor on the price of heat from the PB-FHR. The fuel fabrication costs of TRISO fuel strongly affect the levelized unit electricity cost and research should be performed to leverage the strong safety basis of FHRs into lower fuel costs for FHRs.

The economic performance in LEUC of the optimized design point was similar to that of the Mk1 PB-FHR and given the high uncertainties in the economics assumptions the Mk1 PB-FHR pebble bed core geometry was adopted as the baseline geometry.

A detailed summary is presented of the design and neutronics performance characteristics of results for the Mark 1 PB-FHR (Mk1 PB-FHR) are presented for several safety and economic performance parameters including: burnup, temperature reactivity coefficients, the power density distribution, radiation damage distribution, control element worth, decay heat evolution and tritium production. The Mk1 PB-FHR satisfies the PB-FHR safety criteria. The fuel, moderator (pebble core, pebble shell, graphite matrix, TRISO layers) and coolant have global negative temperature reactivity coefficients and the fuel temperatures are well within their limits. However, shielding should be added to maintain the bottom of the reactor vessel with its gas concentration limit.

There is still a lot of work to be done with respect to modeling of FHR PBRs. BEAU was designed to allow the user to select their own balance between precision and speed. Implications of this trade off should be quantified and documented by exploring the sensitivity of neutronics results to increasingly detailed levels of heterogeneity in the pebble bed. At the same time, BEAU can be further accelerated by parallelizing the depletion module (however, this requires switching to python 3).

Very simplified design criteria were used for complicated failure mechanisms like fuel performance, ATWS response, radiation damage and radiation induced geometry distortion (particularly with respect to control element alignment). Detailed multi-physics analysis should be performed to confirm these simplified models or develop new design criteria for future iterations of the PB-FHR (or other FHR PBRs).

NAVE is a powerful tool for assessing the impacts of design decisions and economic market conditions. As the design of the Mk1 PB-FHR develops NAVE should be integrated with better, PB-FHR specific cost estimate models and be used to continue to inform design. The initial operation mode optimization analysis high lighted the economic impacts of these operating modes but did not go into detail about the implications on safety or maintenance – these issues merit further study. Additionally, most of the economics analysis was performed assuming time-averaged electricity prices from the California Independent Service Operator historical data for 2012-2013 additional electricity markets should be studied to understand how the evolution of electricity markets (the transition to more intermittent electricity generating sources, price of natural gas, changes in incentive structures, etc.) impact the economics of PB-FHRs.

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Appendix A: mocup.py

A.1 Introduction

Often in the analysis of advanced or novel nuclear systems, no existing cross sections for neutron transport or depletion analysis will be available for the system of interest. Monte Carlo methods can be utilized to simultaneously perform neutron transport with continuous energy neutron cross sections – so the physics are modeled without approximations – and the results from this calculation can be used to generate a set of one group cross sections for point depletion analysis. This methodology has been performed before in many Monte Carlo neutron transport to point depletion couplings: MOCUP, MONTE-BURNS, TRITON, VESTA, SERPENT, etc.

The UC-Berkeley Neutronics Group has traditionally used MOCUP for depletion analysis (Moore, Schnitzler, Wemple, Babcock, & Wessel, 1995). MOCUP couples MCNP for neutron transport with ORIGEN2 for point depletion analysis. MOCUP offers a high level of flexibility because its functional modules are organized into several shell scripts that can be modified for analysis of advanced fuel cycles. However, managing this set of scripts was challenging, especially with respect to teaching new users how to set up their own depletion analyses. Additionally, MOCUP limited the number of depletion zones to 50 because of the way it organizes tallies.

The UC-Berkeley Neutronics Group developed mocup.py to have all the flexibility of MOCUP while making setting up depletion analyses much simpler set up.

Table A-1 Qualities of mocup.py

Simplicity	Simple user interface enables intuitive problem definition and minimum effort to produce useful results.
Neutron transport flexibility	mocup.py is set up to be compatible with MCNP5 and SERPENT to generate single group cross sections and flux distributions
Modular framework	Modular framework enables easy to use depletion modules to model novel fuel cycles.
Open source code	Since the source code is open, it can be altered for specific applications.
Materials definition tools	mocup.py material objects are powerful tools that can be utilized for analysis or development of Monte Carlo input decks.
User friendly output files	mocup.py depletion results are summarized in formats compatible with python, matlab or excel for easy post processing and analysis.

The next sections of this report will describe the calculational algorithm (section A.2), the implementation of a depletion calculation (section A.3) and some benchmarking results (section A.4).

A.2 Calculational Algorithm

mocup.py uses a calculational algorithm similar to other depletion coupling utilities. First it processes the neutron transport output file to extract and/or calculate parameters required for depletion analysis. Then mocup.py generates input decks for ORIGEN2.2 and performs a point depletion analysis for every depletion zone in the depletion calculation. Then mocup.py processes the output from the point depletion calculations. Finally, mocup.py generates a new neutron transport cycle input deck based on the updated material composition vectors calculated from ORIGEN2.2. This algorithm is presented in Figure A- 1. The details of each step are presented in the subsequent sub-sections.

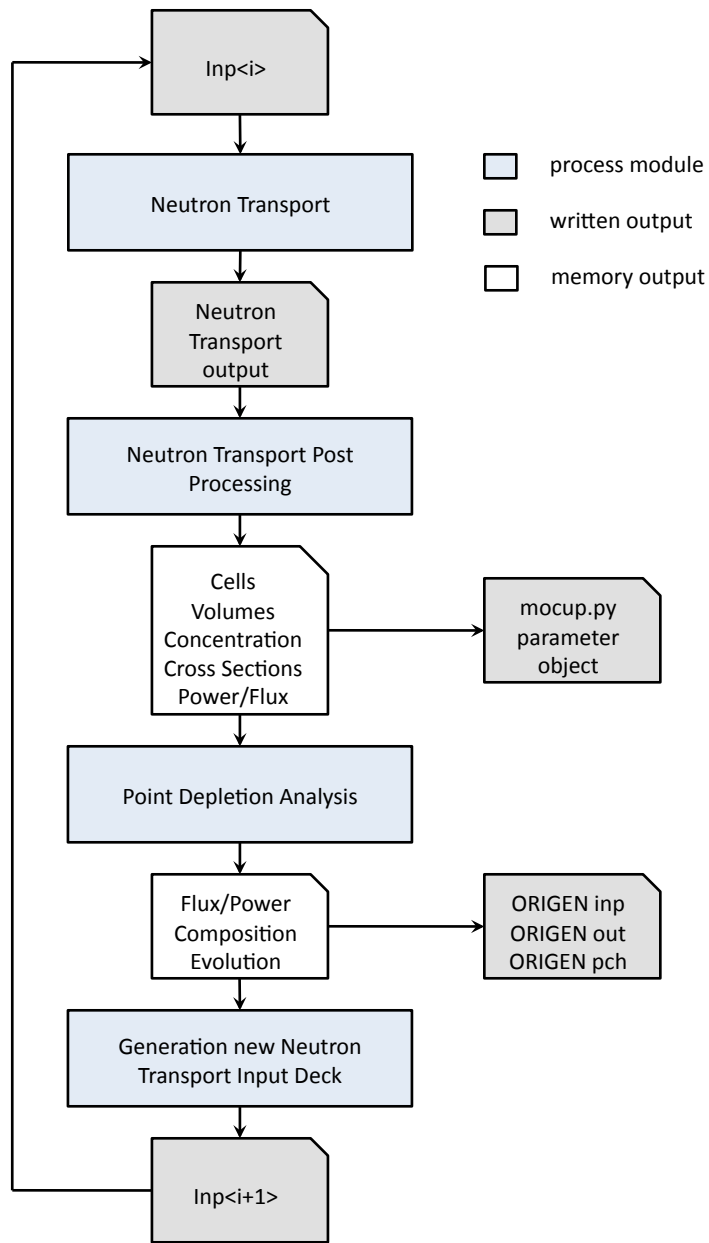


Figure A- 1 mocup.py calculational algorithm

A.2.1 Neutron Transport

mocup.py is written to use either MCNP5 or SERPENT to perform neutron transport with the *transportBRO* module. A tally must be written in MCNP5 find the flux in each depletion zone and to fold the flux in each depletion zone by the energy dependent cross section for each isotope. Serpent is setup to automatically calculate these fluxes and the single group cross sections in its externally coupled burnup mode. The equation for the single group cross sections for reaction, R, for isotope, i, is presented in equation 1.

$$\bar{\sigma}_R^j = \frac{\int_0^\infty \int_{4\pi} \int_{\text{volume}} \sigma_R^j(E) \psi(\vec{r}, E, \Omega) d\vec{r} dE d\Omega}{\int_0^\infty \int_{4\pi} \int_{\text{volume}} \psi(\vec{r}, E, \Omega) d\vec{r} dE d\Omega}$$

Eq 1

Neutron transport output files are labeled as `outp.*iteration*` (MCNP5) or `inp.*iteration*_res.m` and `inp.*iteration*_dep.m` (SERPENT) – the $k_{\text{effective}}$ and fission source term are written in the `res.m` file and all over results are written in the `dep.m` file.

A.2.2 Neutron Transport Post Processing

The neutron transport post processing is performed by a `mocup.py` post processing object (`mpo`). This object holds all the parameters required for setting up the point depletion analysis for a single burnup step.

Table A- 2 Attributes of `mpo` objects

attribute	description
<code>timestep</code>	This is the timestep of iteration
<code>cell</code>	This is a list of the cell IDs in the depletion problem
<code>volume</code>	This is a dictionary of the volumes of the cells indexed by cell ID
<code>power</code>	This is a dictionary of the normalized power of the cells indexed by the cell ID
<code>power_density</code>	This is a dictionary of the power density normalized by source neutron in (MeV/fission source neutron-cm ³) indexed by the cell ID
<code>concentration</code>	This is a dictionary of the material composition vector indexed by cell ID and nuclide in ZAM format
<code>flux_tally</code>	This is a dictionary of flux normalized per fission source neutron indexed by cell ID
<code>mcnp_nuclides</code>	This is a list of cross section library identifiers indexed by MCNP material number in MCNP or indexed by cross section library identifiers in SERPENT (so it's a pair of the same values)
<code>N2N</code>	This is a list of the (n,2n) cross sections indexed by cell ID and nuclide in ZAM format
<code>N3N</code>	This is a list of the (n,3n) cross sections indexed by cell ID and nuclide in ZAM format
<code>NF</code>	This is a list of the (n,γ) cross sections indexed by cell ID and nuclide in ZAM format
<code>NG</code>	This is a list of the (n,α) cross sections indexed by cell ID and nuclide in ZAM format

The post processing is performed by the `populate` methods of the `mpo` object. There are separate post processing methods for depletion analyses using MCNP5 or SERPENT – `mpo.populate(*iteration*)` or `mpo.populate_serpent(*iteration*)`. These post processing

results are written in the same mpo format internally and printed to text output files by default.

A.2.3 Point Depletion Analysis

The depletion analysis is performed by ORIGEN2.2 (Croff, 1983). The information in the mpo file is used to generate a ORIGEN2.2 input decks for each depletion zone in the depletion calculation by updating a skeleton file. This input deck passes the problem-, depletion zone- and burnup- specific set of single group cross sections and communicates the flux rate either in terms of neutron flux or derived based on power. mocup.py can be run in either mode – flux or power. Please note that whichever value is defined (power or flux) is assumed to be constant over the duration of the depletion step. For reactions whose cross sections are not calculated in the neutron transport, ORIGEN will defer to a set of pre-calculated cross sections from a standard ORIGEN cross section library.

mocup.py can perform depletion analysis assuming the conditions at the beginning of the time step ('beginning of step' mode) or at the middle of the timestep ('predictor corrector' mode). In the beginning of step mode, the flux or power distribution and problem-, depletion zone- and burnup-specific cross sections calculated at the beginning of the time step are used over the duration of the depletion step. In a standard time step run in 'predictor corrector' mode, two depletion calculations are performed with the flux or power distribution and problem-, depletion zone- and burnup-specific cross sections calculated at the middle of the time step: one to the middle of the next timestep – the predictor step, and another to end of the current timestep – the corrector timestep. The next neutron transport calculation will be performed assuming the predicted middle of timestep composition and corrector step will calculate the best estimate of the composition evolution. Figure A- 2 Relationship of timesteps, depletion steps and burnup at which neutron transport is performed shows the relationship between the user defined timesteps, depletion steps and the burnup at which neutron transport calculations are performed.

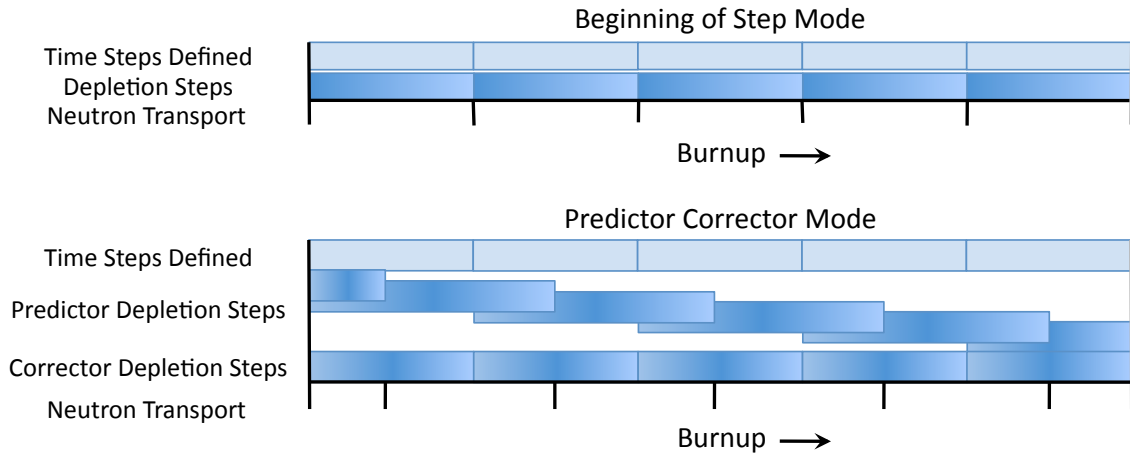


Figure A- 2 Relationship of timesteps, depletion steps and burnup at which neutron transport is performed

The generation of ORIGEN2.2 input decks, moving cross section libraries into the working directory, moving ORIGEN2.2 beginning of step composition files into the working directory, moving the results into the moi_files directory and cleaning up the working directory are all performed by the *origenBRO* module.

The user defines the set of isotopes whose concentrations are updated for subsequent neutron transport and whose transmutation cross sections are calculated in the neutron transport calculation. Only isotopes classified as fission products or actinides in ORIGEN's cross section library are passed between ORIGEN2.2 and neutron transport. Isotopes in the depletion material definition not classified as a fission product or actinides remain at their initial concentration throughout the depletion calculation. Currently, mocup.py only updates cross sections for (n,gamma), (n,fission), (n,2n) and (n,3n) transmutation reactions and the metastable branching ratios in the ORIGEN cross section library are preserved (i.e. the total cross section calculated in neutron transport is split based on the ratio of cross section to ground state and to the metastable state in the ORIGEN cross section library).

A.2.4 Generation of New Neutron Transport Input Deck

The *compBRO* module updates the material composition definition in the neutron transport input decks when generating a new neutron transport input deck. *compBRO* reads the ORIGEN2.2 composition files corresponding to each depletion zone and generates corresponding material objects. These material objects are normalized from molar masses to atom concentrations and then material cards are generated using the mcf or scf method of the mocup material object to generate material cards for MCNP5 or SERPENT input decks. These are new material cards replace the old material cards and a new input deck is generated for the next depletion step.

A.3 Implementation

This section describes how to set up a depletion calculation with the mocup.py framework. This set up includes changes to a standard MCNP5 or SERPENT input deck,

definition of the depletion calculation, and calculational options, and execution of mocup.py on UC Berkeley's Berkeley cluster. Furthermore, there is a section about the depletion summary output files generated by mocup.py. Finally, there is a discussion about how to implement this depletion framework on another computational platform.

A.3.1 MCNP5 input deck

This section presents how to convert a standard MCNP5 input deck into a mocup.py formatted input deck.

Recall that the purpose of the neutron transport step is to generate a flux or power distribution in the nuclear system and generate a set of system-, depletion zone- and burnup- specific single group cross sections. These parameters are calculated with flux multiplication tallies in MCNP5. To expand the number of depletion zones past 50 – as is the case in MOCUP – a single tally is used for all the flux multiplication tallies required calculate the full set of single group cross sections, rather than a flux multiplication and flux tally for each depletion zone. This tally also provides a means to identify which cells are depletion zones. This tally much be identified by a commented out token:

c time dependent reaction rates

mocup.py assumes the following f4 tally will be flux multiplication tally used to calculate the flux- and power- distribution and set of system-, depletion zone- and burnup-specific single group cross sections.

There must be a flux tally and a set of flux multiplication tallies for each isotope mocup.py follows. The flux tally is simply the flux multiplied by 1. The flux multiplication tallies are the cross sections for each isotope-reaction pair mocup.py will update; note that for any isotope-reaction pair that a cross section not generated in neutron transport, mocup.py will defer to what every cross section is in the reference ORIGEN cross section library. The recommended format is that a new line is used for each isotope's flux multiplication tallies with the follow format:

*(1 *mat_id* (16) (17) (-6) (102)) \$ example actinide*

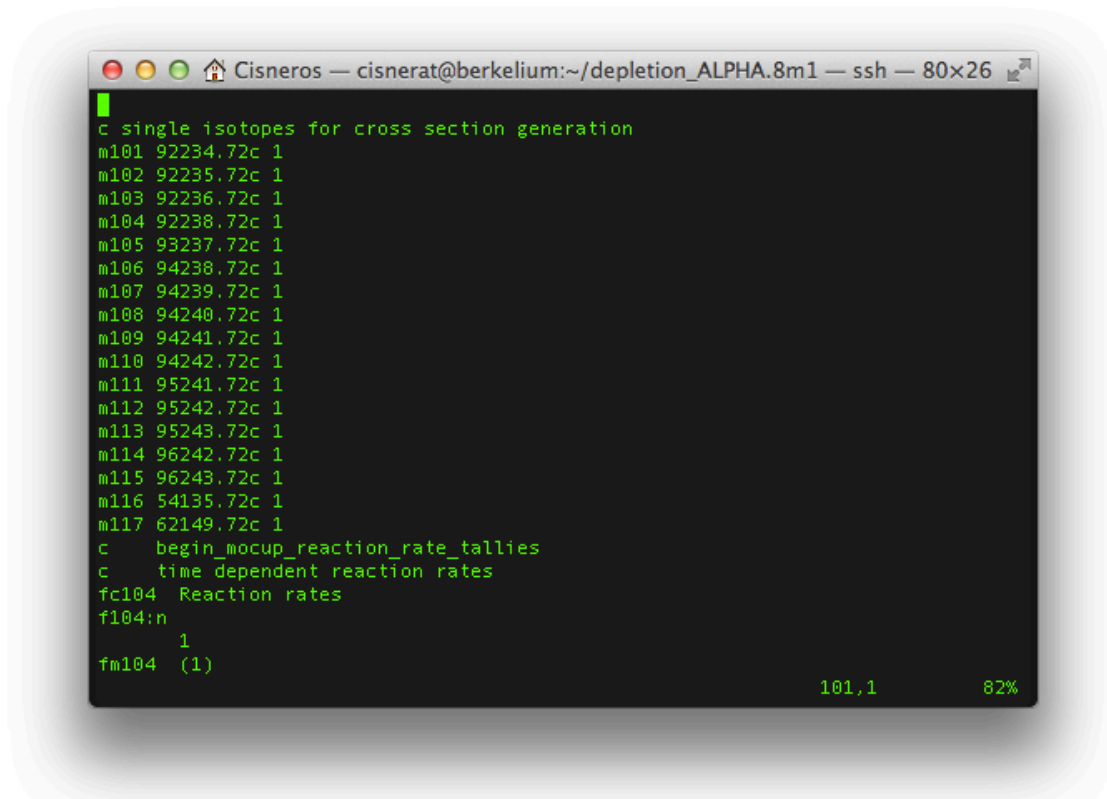
*(1 *mat_id* (16) (17) (102)) \$ example fission product*

Here *mat_id* is the material number of a material composed only of the single isotope in the set of isotopes tracked by mocup.py. Figure A- 3 shows an example of a simple mocup.py formatted flux multiplier tally.

```
Cisneros — cisnerat@berkelium:~/depletion_ALPHA.8m1 — ssh — 80x26
c begin_mocup_reaction_rate_tallies
c time dependent reaction rates
fc104 Reaction rates
f104:n
1
fm104 (1)
(1 101 (16) (17) (-6) (102) )
(1 102 (16) (17) (-6) (102) )
(1 103 (16) (17) (-6) (102) )
(1 104 (16) (17) (-6) (102) )
(1 105 (16) (17) (-6) (102) )
(1 106 (16) (17) (-6) (102) )
(1 107 (16) (17) (-6) (102) )
(1 108 (16) (17) (-6) (102) )
(1 109 (16) (17) (-6) (102) )
(1 110 (16) (17) (-6) (102) )
(1 111 (16) (17) (-6) (102) )
(1 112 (16) (17) (-6) (102) )
(1 113 (16) (17) (-6) (102) )
(1 114 (16) (17) (-6) (102) )
(1 115 (16) (17) (-6) (102) )
(1 116 (16) (17) (102) )
(1 117 (16) (17) (102) )
print 4
kcode 1000 1 10 110
143.8 98%
```

Figure A- 3 mocup.py formatted flux multiplier tallies

The single isotopes for single group cross section generation must be defined. mocup.py was set up assuming that all the isotopes tracked are defined in a single cross section library (i.e. they have the same cross section library suffix) – this played into how new material cards are written. Figure A- 4 shows an example of a simple mocup.py formatted set of single isotope material cards.

A terminal window titled "Cisneros — cisnerat@berkelium:~/depletion_ALPHA.8m1 — ssh — 80x26" displays the output of a Python script. The output consists of a list of material cards, each starting with 'm' followed by a material ID, a numerical value, and a unit. The cards are numbered from m101 to m117. Below the list, there are commands to begin reaction rate tallies and to print time-dependent reaction rates. The output shows 'fc104 Reaction rates' and 'f104:n' with a value of 1. At the bottom right, the text '101,1 82%' is visible.

```
c single isotopes for cross section generation
m101 92234.72c 1
m102 92235.72c 1
m103 92236.72c 1
m104 92238.72c 1
m105 93237.72c 1
m106 94238.72c 1
m107 94239.72c 1
m108 94240.72c 1
m109 94241.72c 1
m110 94242.72c 1
m111 95241.72c 1
m112 95242.72c 1
m113 95243.72c 1
m114 96242.72c 1
m115 96243.72c 1
m116 54135.72c 1
m117 62149.72c 1
c begin_mocup_reaction_rate_tallies
c time dependent reaction rates
fc104 Reaction rates
f104:n
1
fm104 (1)
101,1 82%
```

Figure A- 4 mocup.py formatted set of single isotope material cards

The material cards for each depletion zone must include all the actinides and fission products tracked by mocup.py and any structural materials that are not depleted (i.e. carbon or oxygen in TRISO fuel). At the beginning of cycle isotopes can be defined with an arbitrarily low composition (1e-30 at/bn-cm) – these concentrations must be greater than zero otherwise MCNP5 will crash. The material composition vectors are read from the MCNP5 output file in table 40. Therefore, the formatting in the initial card is not important. However, it is important that there is a card that reads:

print 40

and that a lower case m (rather than an upper case M) is used in the material identification card and it is followed by a space.

m*mat_id* \$

The string “m*mat_id* “ will be used as a token to indicate the beginning of the material card. Subsequent material cards will be generated automatically by mocup.py in the *compBRO* module with the *mcf* method of the *material* object. Examples of user generated- and mocup.py generated– material cards are presented in Figure 5.

```

Cisneros — cisnerat@berkelium:~/depletion_ALPHA.8m1 — ssh — 80x26
m1      6000.72c 1.18234E-02 $ TRISO particle fuel (900K)
        8016.72c 3.54703E-02
        92234.72c 1.00000E-30
        92235.72c 4.75372E-03
        92236.72c 1.00000E-30
        92238.72c 1.88931E-02
        93237.72c 1.00000E-30
        94238.72c 1.00000E-30
        94239.72c 1.00000E-30
        94240.72c 1.00000E-30
        94241.72c 1.00000E-30
        94242.72c 1.00000E-30
        95241.72c 1.00000E-30
        95242.72c 1.00000E-30
        95243.72c 1.00000E-30
        96242.72c 1.00000E-30
        96243.72c 1.00000E-30
        54135.72c 1.00000E-30
        62149.72c 1.00000E-30
m2      6000.72c 1 $ Graphite (900K)
mt2     grph.63t
m3      6000.72c 0.5 $ Silicon Carbide (900K)
        14028.72c 0.5
m4      6000.71c 1 $ Graphite (900K)
mt4     grph.63t
-- INSERT --                                     77,28      58%

```

```

Cisneros — cisnerat@berkelium:~/depletion_ALPHA.8m1 — ssh — 80x26
-----
C -----
m1 $
        6000.72c 3.32022E-02
        8016.72c 9.96046E-02
        92234.72c 2.54582E-07
        92235.72c 2.60862E-04
        92236.72c 6.68359E-04
        92238.72c 1.57869E-02
        93237.72c 1.03707E-04
        94238.72c 6.36338E-05
        94239.72c 1.69856E-04
        94240.72c 1.08417E-04
        94241.72c 8.77579E-05
        94242.72c 1.45882E-04
        95241.72c 7.76229E-07
        95242.72c 1.10377E-08
        95243.72c 5.29935E-05
        96242.72c 1.97753E-06
        96243.72c 6.90620E-08
        54135.72c 1.55481E-08
        62149.72c 1.97321E-07
m2      6000.72c 1 $ Graphite (1200K)
mt2     grph.63t
m3      6000.72c 0.5 $ Silicon Carbide (1200K)
        14028.72c 0.5
                                           94,1      57%

```

Figure A- 5 Examples of user generated (top) and mocupy generated (bottom) material cards

Volumes are important in depletion analysis for the normalization of flux tallies and the conversion of isotopic concentrations to molar masses. MCNP5 can calculate volume for simple geometries like spheres and prisms. However, MCNP5 cannot calculate the volume of irregular shapes and volumes calculated by MCNP5 of repeated structures are not accurate. Therefore it is **strongly** recommended that the user define volumes explicitly in the cell cards of the depletion zones. Volume can be defined in the cell cards with a volume card:

```
vol=*volume*
```

Here, *volume* is the volume of the depletion zone in cm³.

The source distribution between each depletion step is likely very similar. Therefore, it is a good practice to recycle source tape files from previous depletion steps. mocup.py expects a source tape (srctp) file in the working directory named 'source'. Alternatively, a source definition in the input deck can be used but this will take longer to converge and not take advantage of the similarity of the source distributions between each depletion step.

A.3.2 SERPENT input deck

SERPENT can be used as a neutron transport module using its externally coupled burnup capability. This enables depletion controllers for iterative depletion analysis for multi-batch- and continuously refueling- fuel cycles – otherwise, it is better to use SERPENT's internal coupling. This sub-section presents how to set up the SERPENT input deck for external depletion coupling to utilize mocup.py.

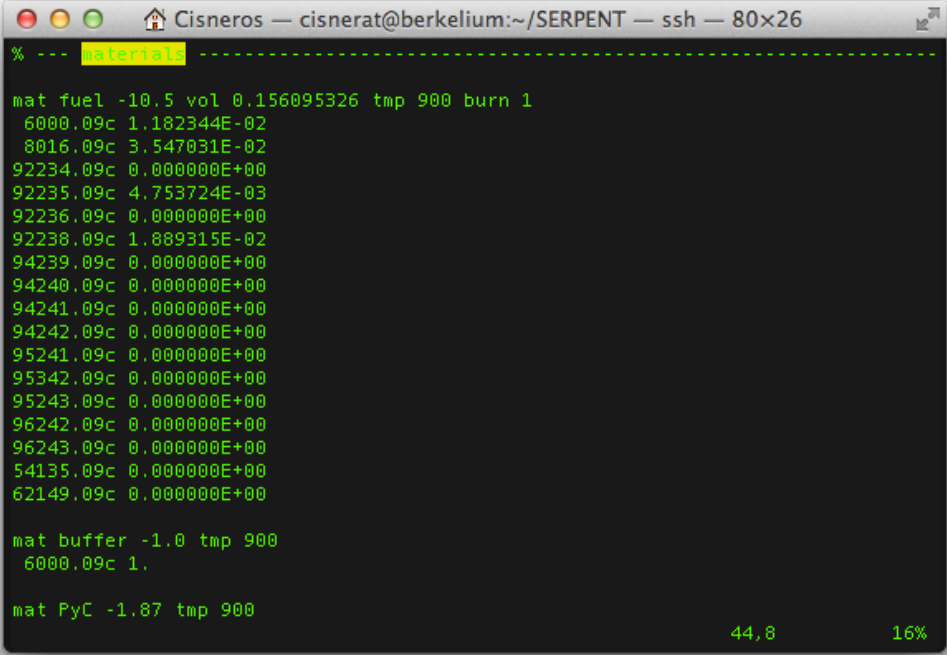
The burnup functionality in SERPENT is engaged if any material has a burn flag. So the first step is to use a burn flag (burn 1) in the material options for all the materials that are to be depleted with mocup.py. All the isotopes tracked by mocup.py must be present in the material definition because system-, depletion zone- and burnup- specific single group cross sections will only be generated for isotopes identified in the material cards. Furthermore, mocup.py only uses a single list of isotopes when generating new material cards so the isotope lists should be consistent between each depletion zone's material definitions.

The user should define the volume of the depletion zone in the material card options line because this is important for the coupling between isotopic concentrations for the neutron transport input deck and the molar mass composition vectors for the point depletion input deck. Like MCNP5, SERPENT has some basic geometric volume calculating capabilities, but these break down for irregular geometries and repeated structures. It is **strongly** recommended that the user manually define the volume in the material options:

```
vol *volume*
```

Here *volume* is the volume of the depletion zone in cm³.

Figure A- 6 presents a simple mocup.py formatted material.



```
% --- material -----  
  
mat fuel -10.5 vol 0.156095326 tmp 900 burn 1  
6000.09c 1.182344E-02  
8016.09c 3.547031E-02  
92234.09c 0.000000E+00  
92235.09c 4.753724E-03  
92236.09c 0.000000E+00  
92238.09c 1.889315E-02  
94239.09c 0.000000E+00  
94240.09c 0.000000E+00  
94241.09c 0.000000E+00  
94242.09c 0.000000E+00  
95241.09c 0.000000E+00  
95342.09c 0.000000E+00  
95243.09c 0.000000E+00  
96242.09c 0.000000E+00  
96243.09c 0.000000E+00  
54135.09c 0.000000E+00  
62149.09c 0.000000E+00  
  
mat buffer -1.0 tmp 900  
6000.09c 1.  
  
mat PyC -1.87 tmp 900  
  
44,8 16%
```

Figure A- 6 mocup.py formatted SERPENT material definition

Do not define an irradiation history otherwise serpent will execute in the internally coupled depletion mode and print cross sections in a different format, which mocup.py will not be able to read.

A.3.3 Depletion Calculation Definition (gofiss.py)

To perform a mocup.py depletion calculation one must define a depletion object and execute the gofiss method of the depletion object. This procedure can be implemented in a simple python script. In this script, mocup.py must be imported. Then a Depletion object must be created. The power vector and timestep vector must be defined. The power vector is a list of powers at which the nuclear system operates in MWt. The timestep vector is a list of timestep lengths for which the nuclear system is depleted at the corresponding power with the system-, depletion zone- and burnup- specific set of cross sections. Finally, the gofiss method of the depletion object sets up the depletion calculation, iteratively executes the four main modules (neutron transport, neutron transport post processing, point depletion analysis, and generation of the new neutron transport input deck) and creates depletion summary output files. Figure A-7 presents an example of a simple script to define and execute the depletion calculation.

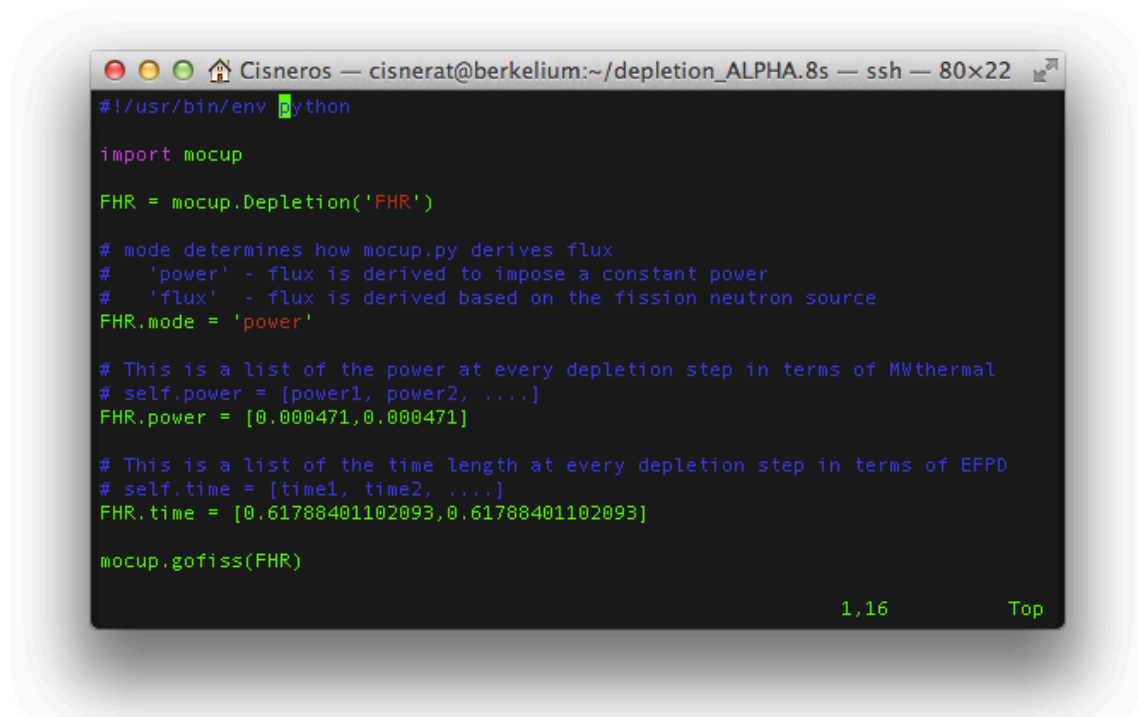
A terminal window titled "Cisneros — cisnerat@berkelium:~/depletion_ALPHA.8s — ssh — 80x22". The terminal shows a Python script for defining a depletion calculation. The script starts with a shebang line and imports the 'mocup' module. It then creates a 'FHR' object using 'mocup.Depletion('FHR')'. Comments explain that the 'mode' parameter determines how flux is derived: 'power' for constant power and 'flux' for flux derived from the fission neutron source. The script sets 'FHR.mode = 'power''. It then defines a list of power values in MWthermal: 'FHR.power = [0.000471, 0.000471]'. Next, it defines a list of time lengths in EFPD: 'FHR.time = [0.61788401102093, 0.61788401102093]'. Finally, it calls 'mocup.gofiss(FHR)'. The terminal output shows '1,16' and 'Top' at the bottom right.

Figure A-7 Simple depletion calculation definition script

A.3.4 mocup.py Output files

By default mocup.py creates summaries of the depletion analysis in three formats: a matlab m-file, a python executable, and in comma separated values, csv, format that can be opened by excel. These output files contain the evolutions of system $k_{\text{effective}}$, flux in each depletion zone, power in each depletion zone and the isotopic concentration vector. Additional information can be obtained from the ORIGEN2.2 input, material punch and output files organized by depletion zone and depletion step in the moi_files director within the working directory. A summary of the information used for each depletion analysis step for each cell is located in the mpo files organized by depletion zone and depletion step in the working directory.

A.3.5 Options

mocup.py has a few options to give the user flexibility in setting up the depletion calculation. These options are presented in Table 3.

Table A-3 Options for depletion calculation

Option	flag	Description
mode	power	Flux in depletion is derived based on maintaining constant power throughout the depletion step.
	flux	Flux in depletion is derived based on a fission neutron source. Note that if flux mode is engaged, a vector, <i>depletion.flux</i> , defining the fission neutron source in neutrons per section for each timestep, similar to the <i>depletion.power</i> vector, must be defined in the depletion calculation definition.
transport module	mcnp	MCNP5 is used as the neutron transport module
	serpent	SERPENT is used as the neutron transport module
method	predictor	Predictor corrector algorithm is used for depletion analysis; see Figure A- 2 Relationship of timesteps, depletion steps and burnup at which neutron transport is performed.
	beginning	Beginning of step algorithm is used for depletion analysis; see Figure A- 2 Relationship of timesteps, depletion steps and burnup at which neutron transport is performed.
runTRANS	yes	Neutron transport is performed to determine the problem-, depletion zone- and burnup specific single group cross sections.
	no	Previously generated neutron transport outputs are recycled for depletion analysis – this option is helpful for trouble shooting.
library	pwru50	This determines the ORIGEN cross section library used for all the isotopes whose cross sections are not generated by neutron transport.
format	all	This option prints summary output files in all three formats
	python	Summary output is printed as a python file
	matlab	Summary output is printed as a matlab, m-file
	csv	Summary output is printed in an excel formatted csv file

* bold options are the default options

Figure 8 presents the simplified depletion calculation description from section 3.3 updated with the explicit definitions of all mocup.py's options.

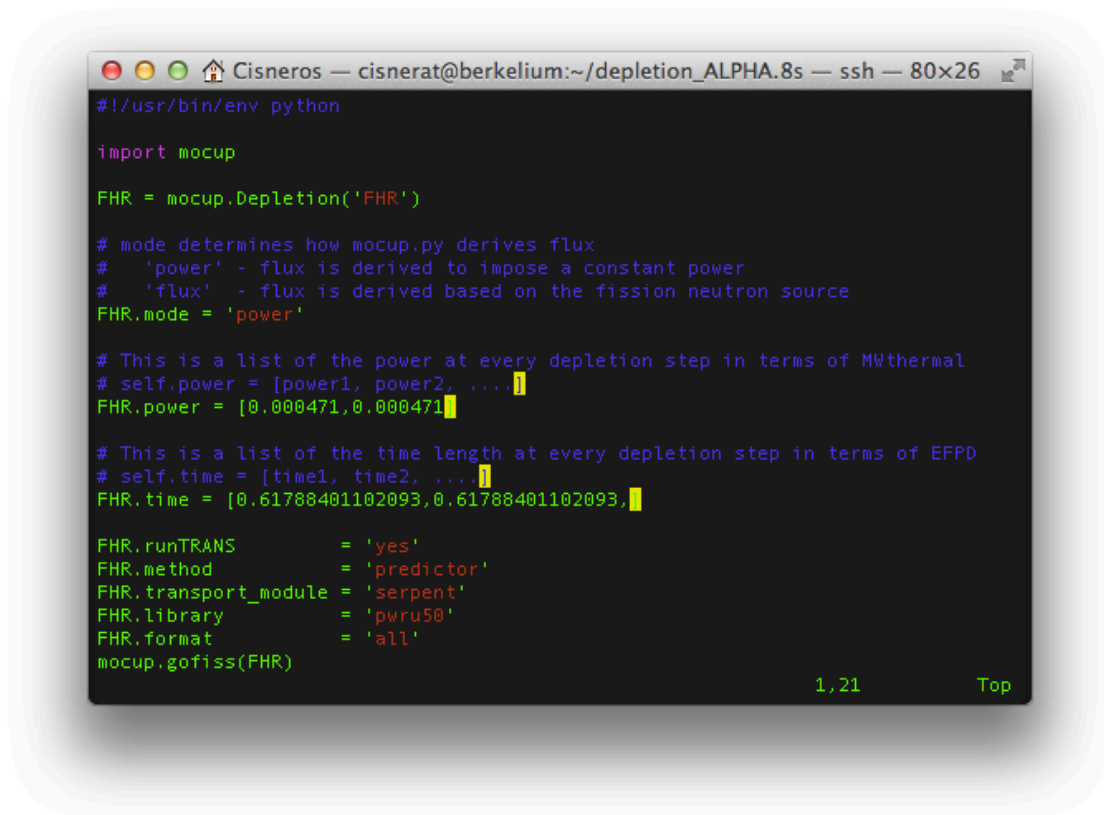
A terminal window titled 'Cisneros — cisnerat@berkelium:~/depletion_ALPHA.8s — ssh — 80x26'. The terminal shows a Python script for mocup. The script starts with '#!/usr/bin/env python' and 'import mocup'. It then creates an object 'FHR = mocup.Depletion('FHR')'. Comments explain that 'mode' determines how flux is derived: 'power' (constant power) or 'flux' (based on fission neutron source). The script sets 'FHR.mode = 'power''. It then defines a list of power values: 'FHR.power = [0.000471, 0.000471]'. Next, it defines a list of time lengths: 'FHR.time = [0.61788401102093, 0.61788401102093]'. Finally, it sets several options: 'FHR.runTRANS = 'yes'', 'FHR.method = 'predictor'', 'FHR.transport_module = 'serpent'', 'FHR.library = 'pwr50'', and 'FHR.format = 'all''. The script ends with 'mocup.gofiss(FHR)'. The terminal shows the script is being executed, with a cursor at the end of the last line. The bottom right corner of the terminal shows '1,21' and 'Top'.

Figure A-8 Simple depletion calculation definition script with options explicitly defined

A.3.5 Execution on Berkelium Cluster

The UC-Berkeley Neutronics Group’s cluster, Berkelium, uses a q-submission process to allocate calculation time. Therefore, a submission file must be generated to submit the depletion calculation definition and execution script described in section 3.3. The depletion calculation definition and execution file included in the example depletion directory is named `gofiss.py` and the corresponding submission file is named `run_mocup.sh`. Figure A-9 presents the example q submission file.

```
#!/bin/sh
#
## -N mocup
## -cwd
## -pe ompi 8
## -S /bin/bash
## -q x.q
## -V

python gofiss.py
```

Figure A-9 Example q-submission file for mocup.py

The user's settings must be set up to run MCNP5, SERPENT and ORIGEN2.2 and include permissions to execute these programs. Furthermore, for parallelization mocup.py 8.1 uses mvapich so the mvapich2-1.8a2 module must be loading into the user's .bashrc file in their home directory. As of June 2012, Ryan Bergman manages the Berkelium cluster and is the best reference to set up one's user account and permissions.

A.3.6 Executing mocup.py on other platforms

mocup.py was written to run on The UC-Berkeley Neutronics Group's cluster, Berkelium. However, this depletion coupling framework might be attractive to other groups. The berkelium-specific directory locations, commands and nuclide correspondence tables must be updated. The location of the ORIGEN cross-section libraries is defined in the *dir* attribute of the depletion object (this location can even be changed in the depletion calculation definition); this framework also, assumes that the decay library (decay.lib) and gamma cross sections (gxuo2brm.lib) are in the same directory as the depletion cross section libraries, so if this assumption does not hold these file locations must be updated as well in the *origenB* and *origenPC* modules of mocup.py. The commands to execute ORIGEN2.2, *o2_fast* and *o2_therm*, must be updated if different commands are used to execute ORIGEN2.2. Furthermore, there is some logic based on the names of the cross section libraries that come standard with ORIGEN2.2 whether *o2_fast* or *o2_therm* is to be utilized. If the names of these ORIGEN2.2 cross section libraries change, this logic in *origenBRO* and *origenPC* must be updated. MCNP5 is executed as a neutron transport module with the mpiexec command:

```
mpiexec mcnp5.mvapich i=inp.*depletion step* o=outp.*depletion step*
mc=mcctal.*depletion step* runtpe=runtp.*depletion step*
```

If MCNP5 is not executed with this command it must be updated in the *mocup* iterative depletion step module and the *gofiss* iterative depletion step controller module of mocup.py. SERPENT is executed as a neutron transport module with the mpiexec command:

mpiexec sss117 inp.*depletion step*

If SERPENT is not executed with this command it must be updated in the *mocup* iterative depletion step module and the *gofiss* iterative depletion step controller module of *mocup.py*. Finally, the nuclide correspondence table, *depletion.nct*, in the depletion object initialization method was developed based on the nuclear data that comes standard with SERPENT and MCNP5. This table must be looked over and updated when porting *mocup.py* to a new cluster or updating the cross sections used by MCNP5 or SERPENT.

A.4 Verification

To have confidence in *mocup.py*'s results it is important to at a minimum benchmark *mocup.py* against depletion frameworks with similar functionality. *mocup.py* was developed as the depletion framework to analyze pebble bed fluoride salt cooled high temperature reactors (FHR), so it was benchmarked for a simple unit cell of a single pebble. SERPENT (SERPENT for neutron transport and an internal point depletion solver) and TRITON (KENO6 for neutron transport and ORIGEN-S as a point depletion solver) were used as reference codes.

A.4.1 Test case definition

An infinitely reflected FHR annular pebble unit cell was used as the test case. This pebble is similar to those utilized in high temperature gas-cooled reactors, but it has a smaller outer diameter and an inner low-density graphite core to control buoyancy in the salt coolant; as shown in Figure A-10.

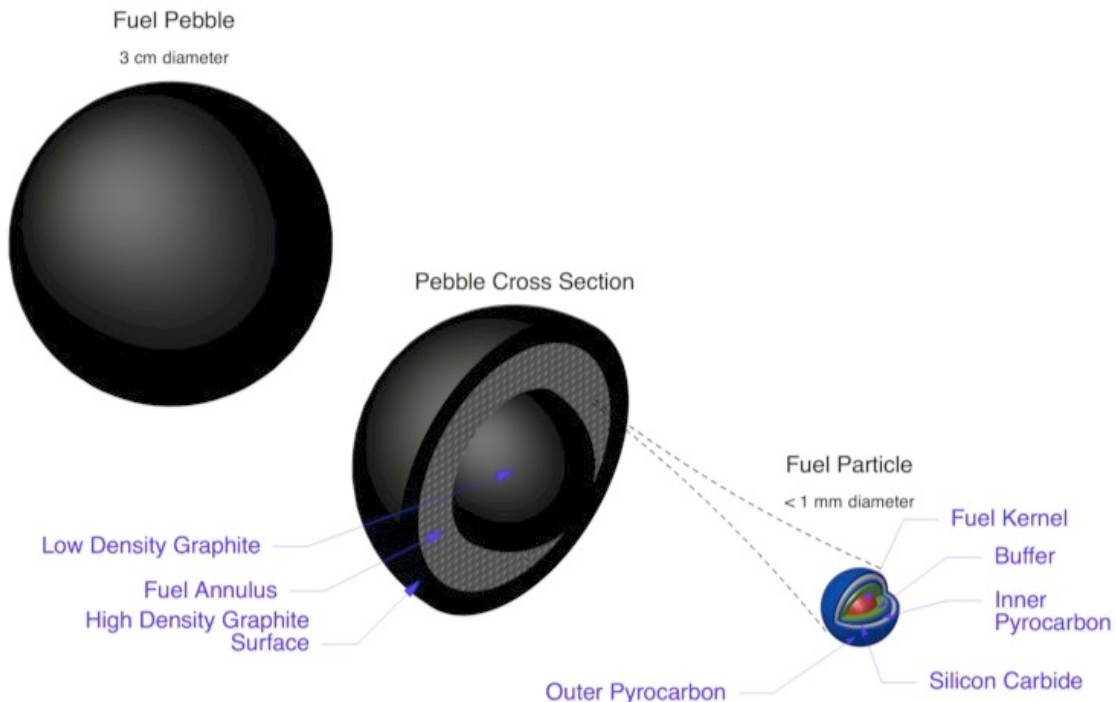


Figure A-10 Fluoride salt cooled high temperature reactor fuel pebble and TRISO particle.

The dimensions for the TRISO particles are presented in Table A-4.

Table A-4 Dimensions for the TRISO particles of the test case

Parameter	Dimension	Material	Density (g/cc)
fuel kernel diameter (μm)	500	$\text{UC}_{0.5}\text{O}_{1.5}$	10.5
buffer thickness (μm)	100	Carbon	1.0
inner pyrolytic carbon layer thickness (μm)	35	Carbon	1.87
silicon carbide Layer Thickness (μm)	35	Silicon Carbide	3.2
outer pyrolytic carbon layer thickness (μm)	35	Carbon	1.87
TRISO packing fraction (%)	11.171	Carbon	1.6

The dimensions for the annular fuel pebble are presented in Table A-5.

Table A-5 Dimensions for the FHR fuel pebble of the test case

Parameter	Dimension	Material	Density (g/cc)
inner pebble core diameter (cm)	1.1415	low density graphite	0.5
active region outer diameter (cm)	2.6	TRISO + matrix	-
pebble shell outer diameter (cm)	3.0	graphite	1.74
pebble volume fraction (salt) (%)	60	flibe	1.96

The active region is modeled as a simple cubic lattice of the TRISO particles suspended in a graphite matrix. The unit cell is a hexagonal prism with reflective boundary conditions.

To make each of the depletion calculations as consistent as possible, these benchmarking depletion analyses were performed with all the materials at 900K so that the standard endfb-vii cross sections distributed with MCNP5 and SERPENT can be utilized. Unfortunately, thermal scattering libraries for graphite do not exist at these temperatures so the thermal scattering libraries at 800K will be utilized for MCNP5 and SERPENT. However, KENO6 uses thermal scattering laws to for scattering at energies below 5eV.

Each neutron transport codes is allowed to use 100 active cycle with 10000 neutrons per cycle to calculate keffective, the flux distribution and the set of problem-, depletion zone- and burnup- specific single group cross sections.

Only a single depletion zone was utilized for this benchmark, though the multi-depletion functionality was confirmed in the initial verification.

SERPENT and MCNP5 use continuous energy Monte Carlo neutron transport. The KENO6 Module uses a 238 multi-group structure, where the cross sections for the fuel are

based on imposing a dancoff factor tuned such that the multi-group model k_{∞} matches a KENO6 continuous energy reference as described by Kelly and Ilas.

A.4.2 Results Comparisons

The three metrics utilized to compare depletion results between SERPENT, TRITON and mocup.py using SERPENT and MCNP5 as neutron transport drivers: the k_{∞} evolution, the isotopic concentration evolution and the cross section evolution are presented in sections 4.2.1-4.2.3.

A.4.2.1 k_{∞} evolution

Figure A-11 presents the k_{∞} evolution for the test case.

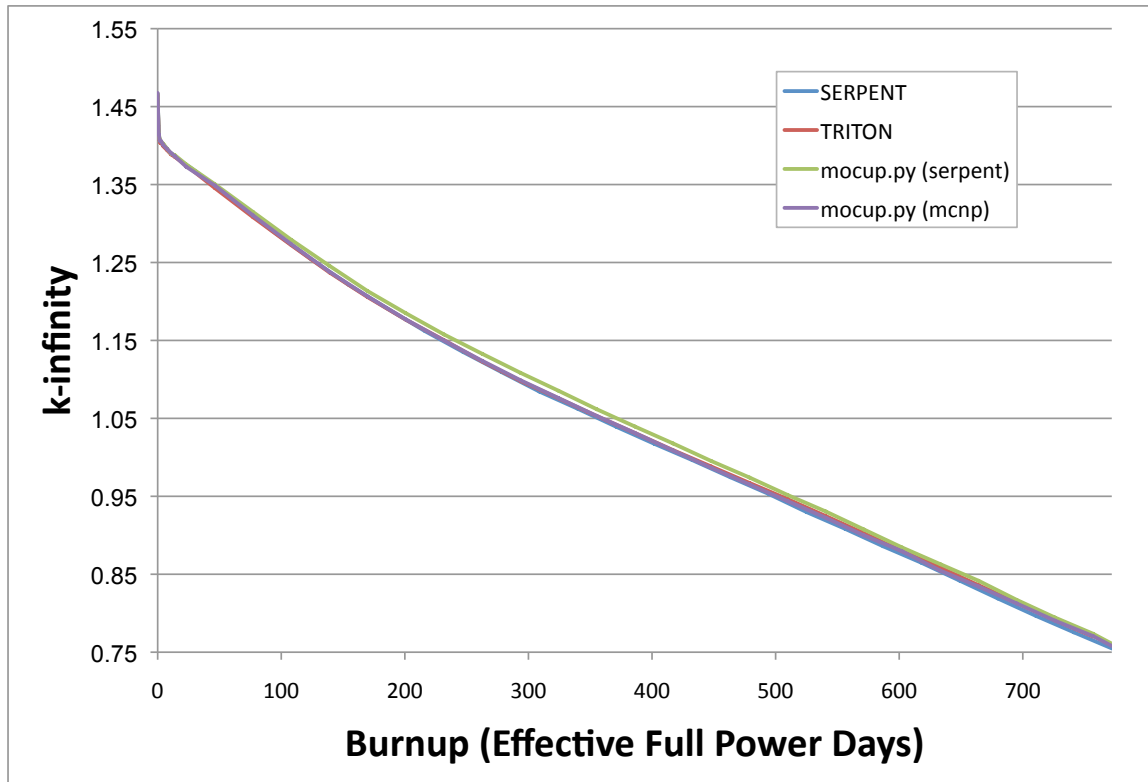


Figure A-11 k_{∞} evolution

A.4.2.2 Isotopic composition evolution

Figure A-12 - Figure A-18 present the concentration evolutions for ^{235}U , ^{238}U , ^{239}Pu , ^{240}Pu , ^{241}Pu , ^{135}Xe , and ^{149}Sm .

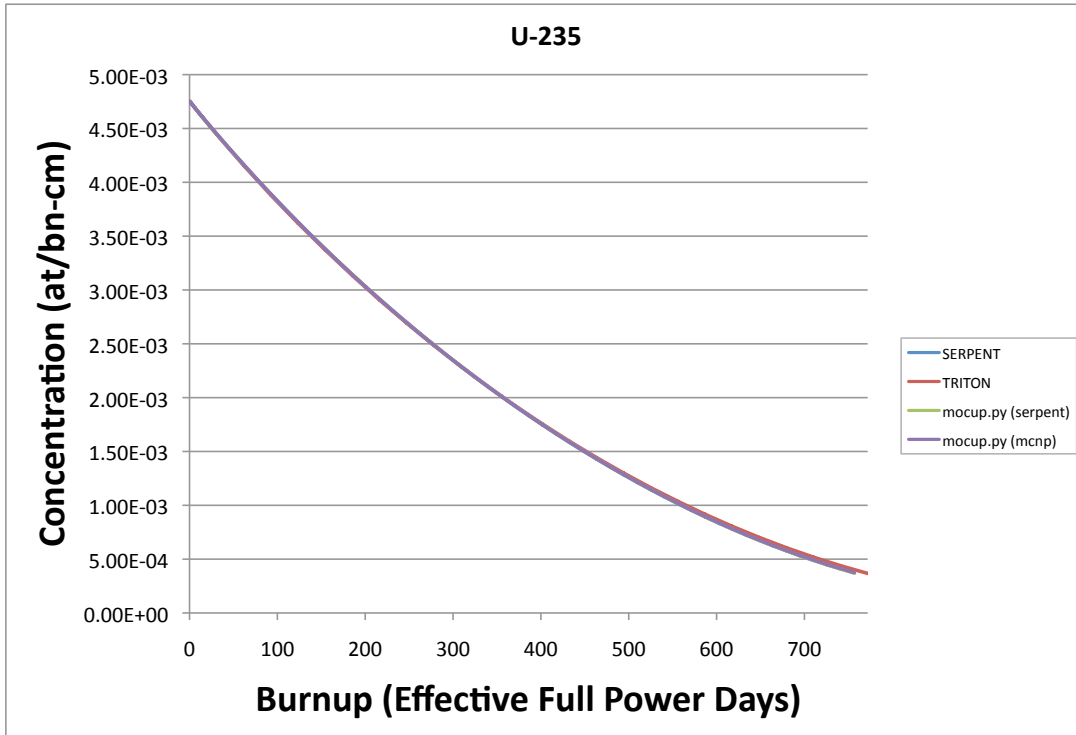


Figure A-12 U-235 concentration evolution

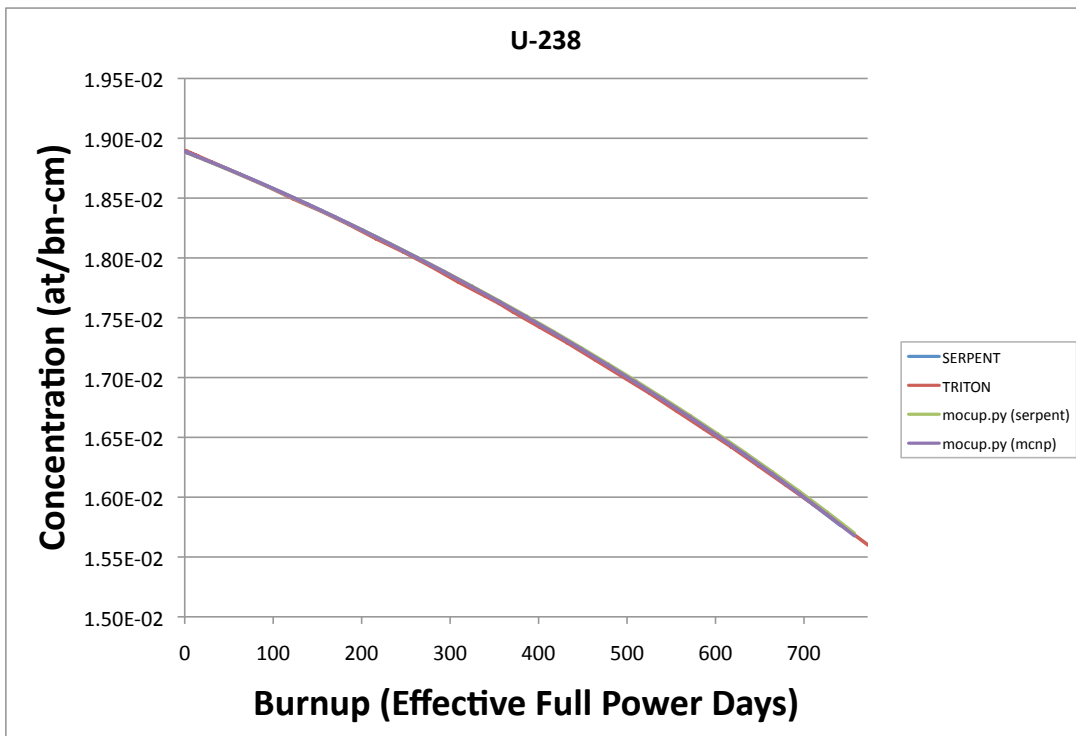


Figure A-13 U-238 concentration evolution

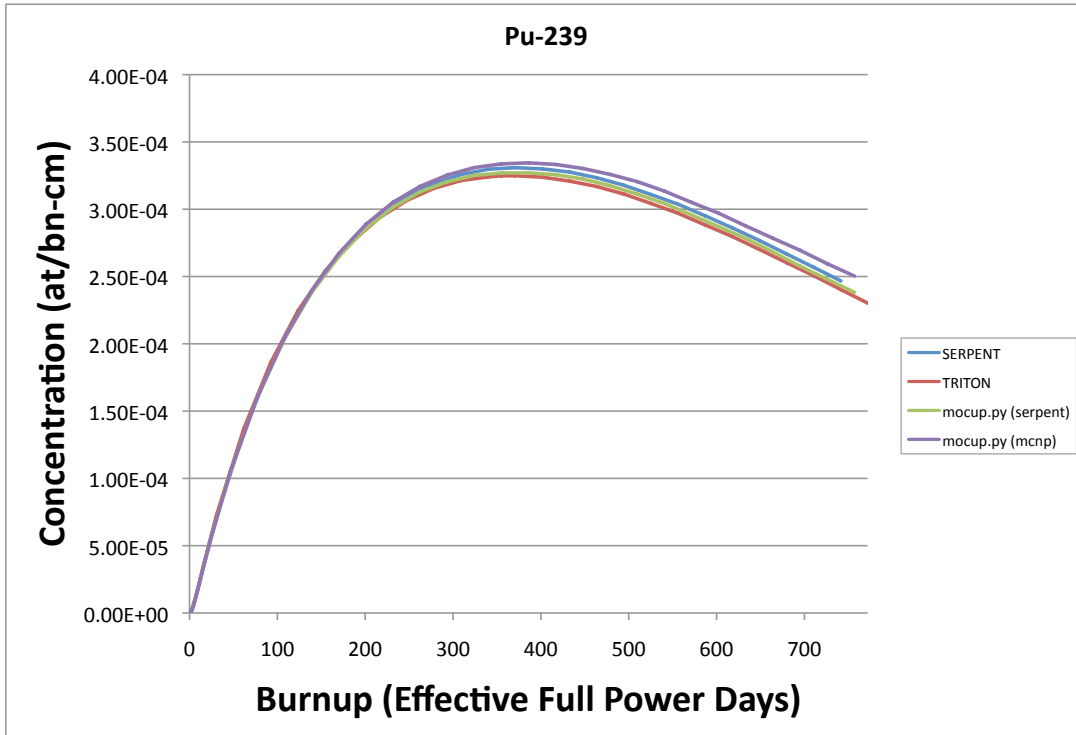


Figure A- 14 Pu-239 concentration evolution

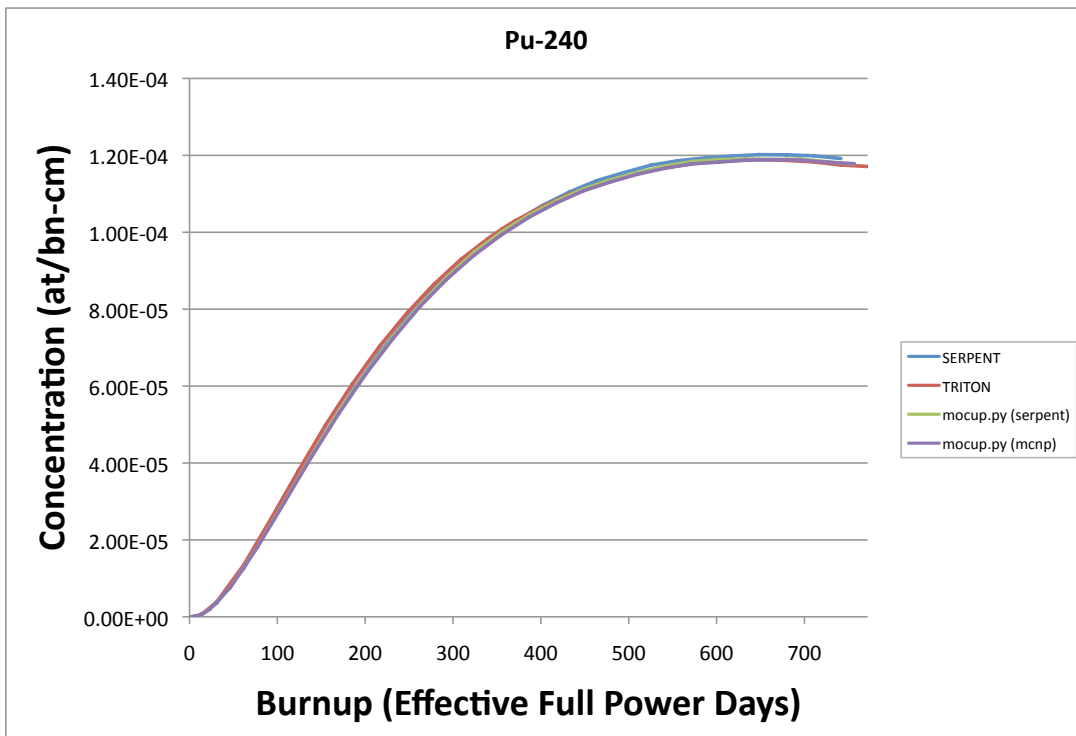


Figure A- 15 Pu-240 concentration evolution

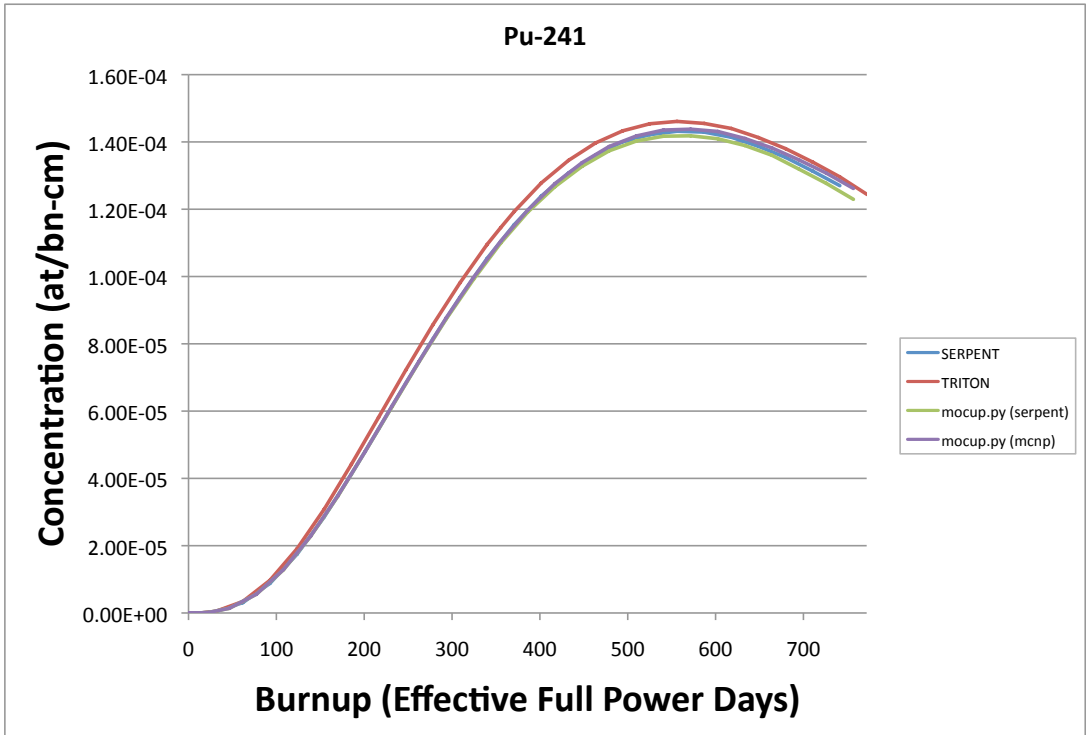


Figure A- 16 Pu-241 concentration evolution

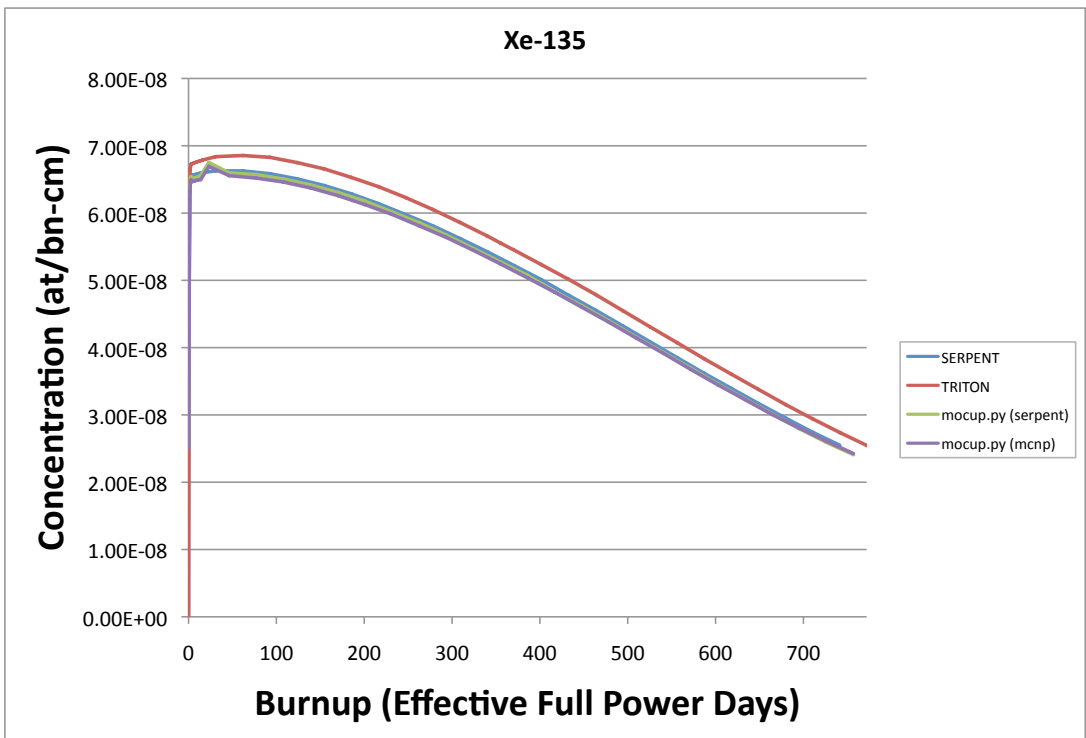


Figure A-17 Xe-135 concentration evolution

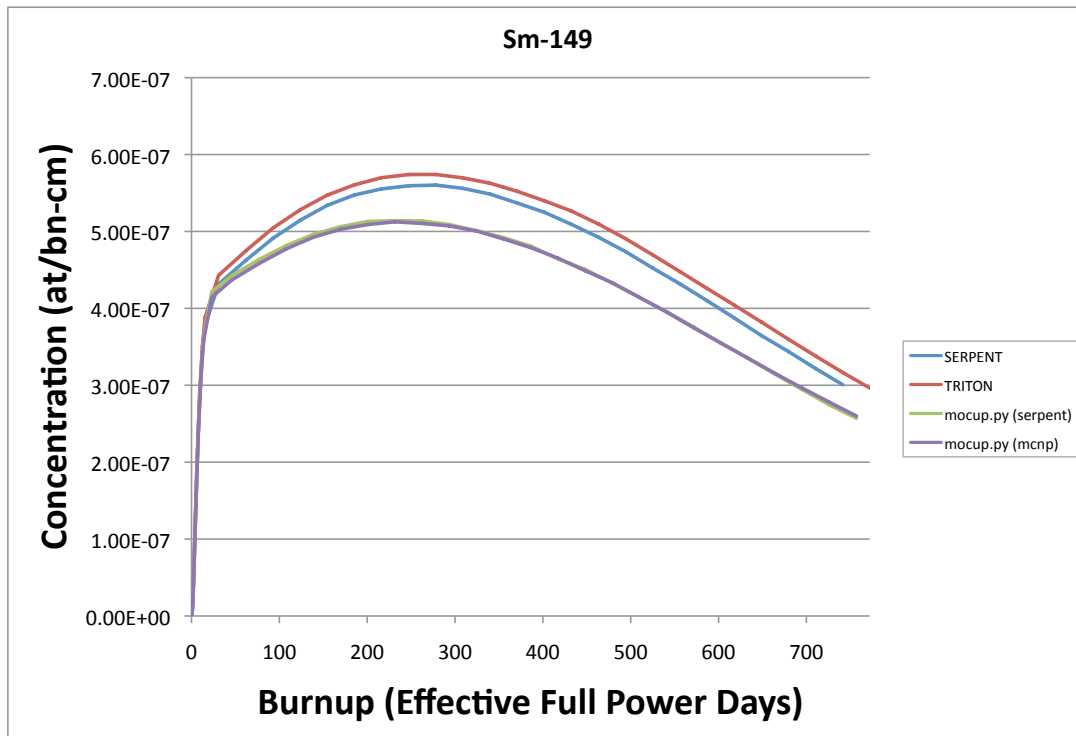


Figure A-18 Sm-149 concentration evolution

The concentration evolutions match well for the actinides. There is a spike in the ^{135}Xe concentration at 32 effective full power days where the burnup steps transition from lengths of 1 GWd/MT to 5 GWd/MT. This discrepancy could be due to differences in fission yields libraries between the more recent nuclear data in SERPENT and ORIGEN-S verses the older data used in the ORIGEN2 data library.

A.4.2.3 Single group cross section evolution

Figure A- 19 - Figure A-25 present the single group (n,gamma) cross section evolutions for ^{235}U , ^{238}U , ^{239}Pu , ^{240}Pu , ^{241}Pu , ^{135}Xe , and ^{149}Sm . Figure A-26 - Figure A-30 present the single group (n,fission) cross section evolution for ^{235}U , ^{238}U , ^{239}Pu , ^{240}Pu , and ^{241}Pu .

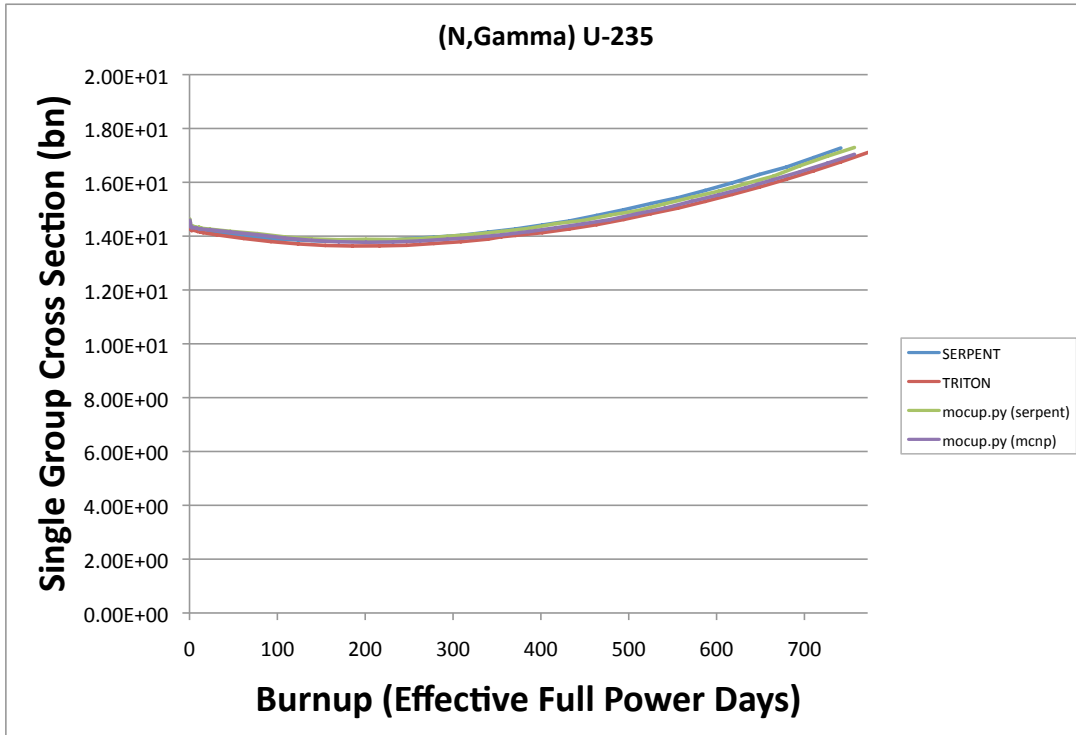


Figure A- 19 (n,gamma) U-235 single-group cross section evolution

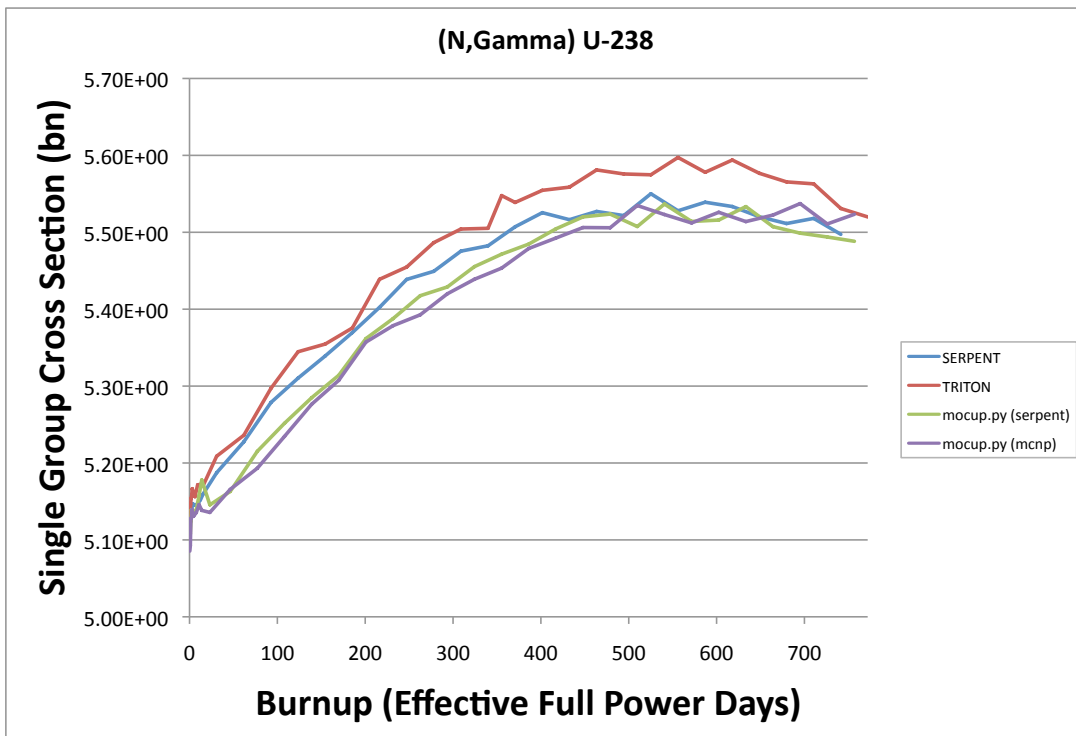


Figure A-20 (n,gamma) U-238 single-group cross section evolution

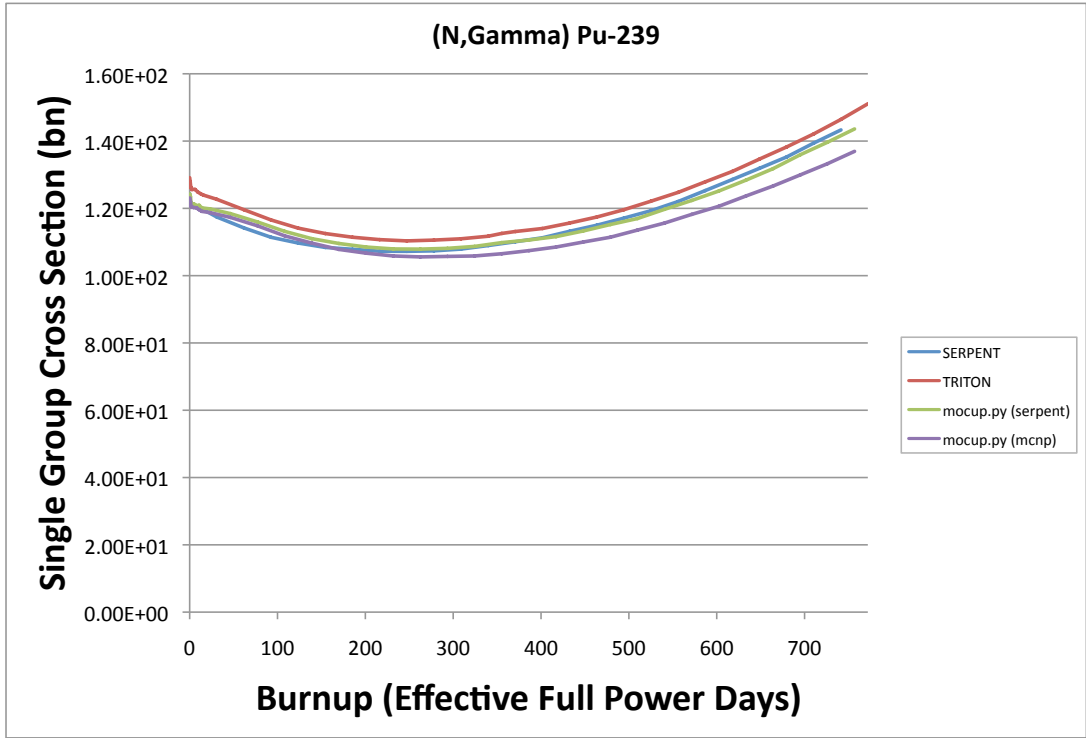


Figure A-21 (n,gamma) Pu-239 single-group cross section evolution

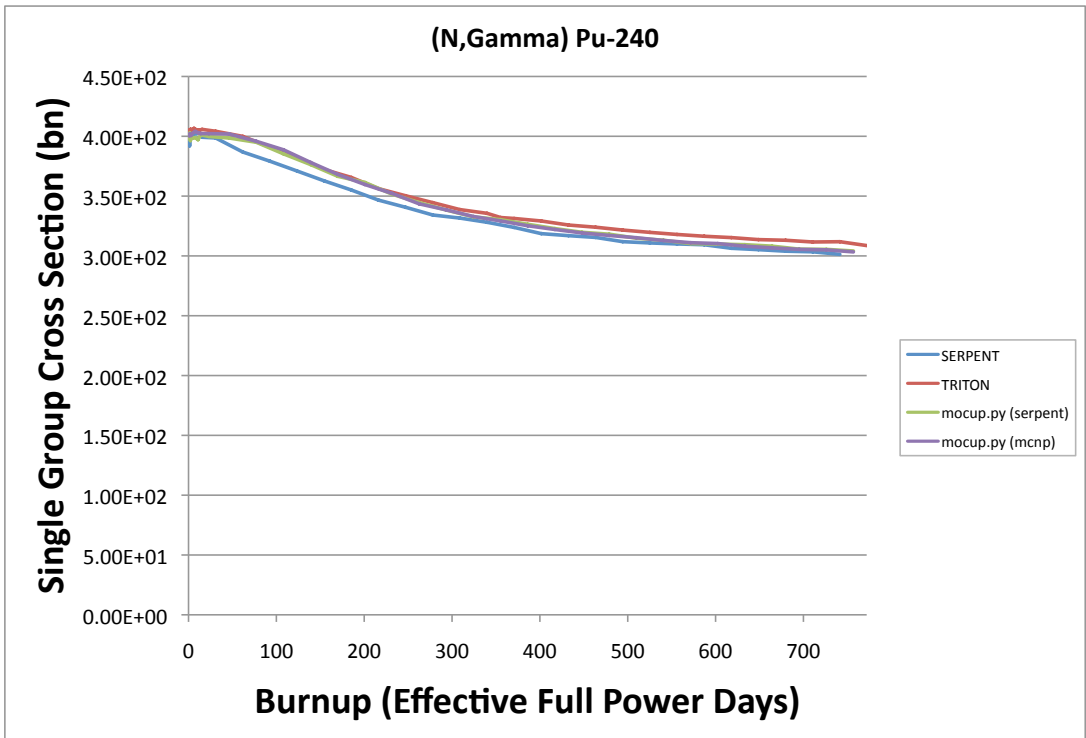


Figure A-22 (n,gamma) Pu-240 single-group cross section evolution

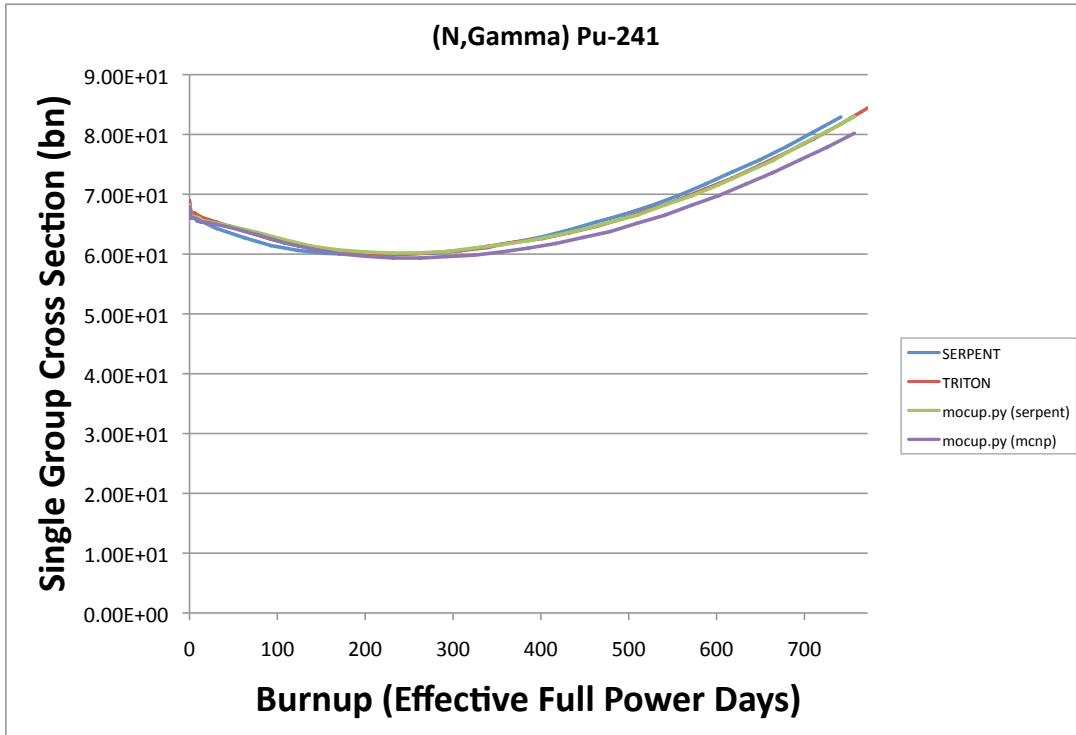


Figure A-23 (n,gamma) Pu-241 single-group cross section evolution

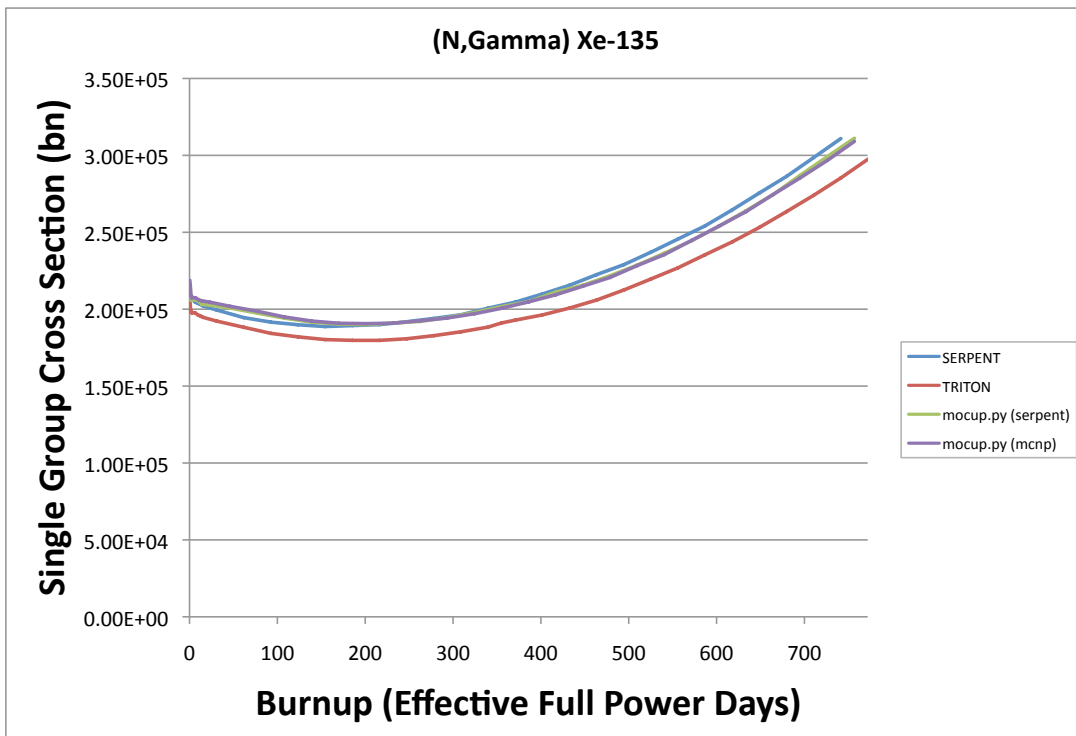


Figure A-24 (n,gamma) Xe-135 single-group cross section evolution

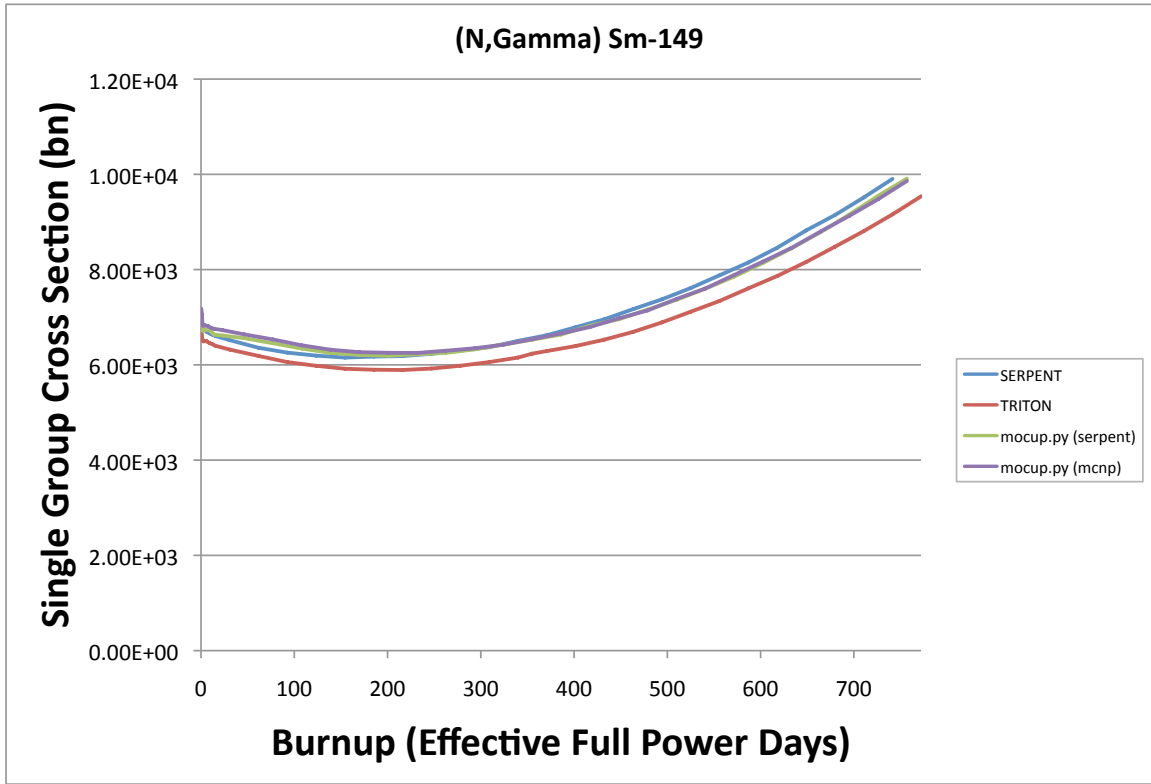


Figure A-25 (n,gamma) Sm-149 single-group cross section evolution

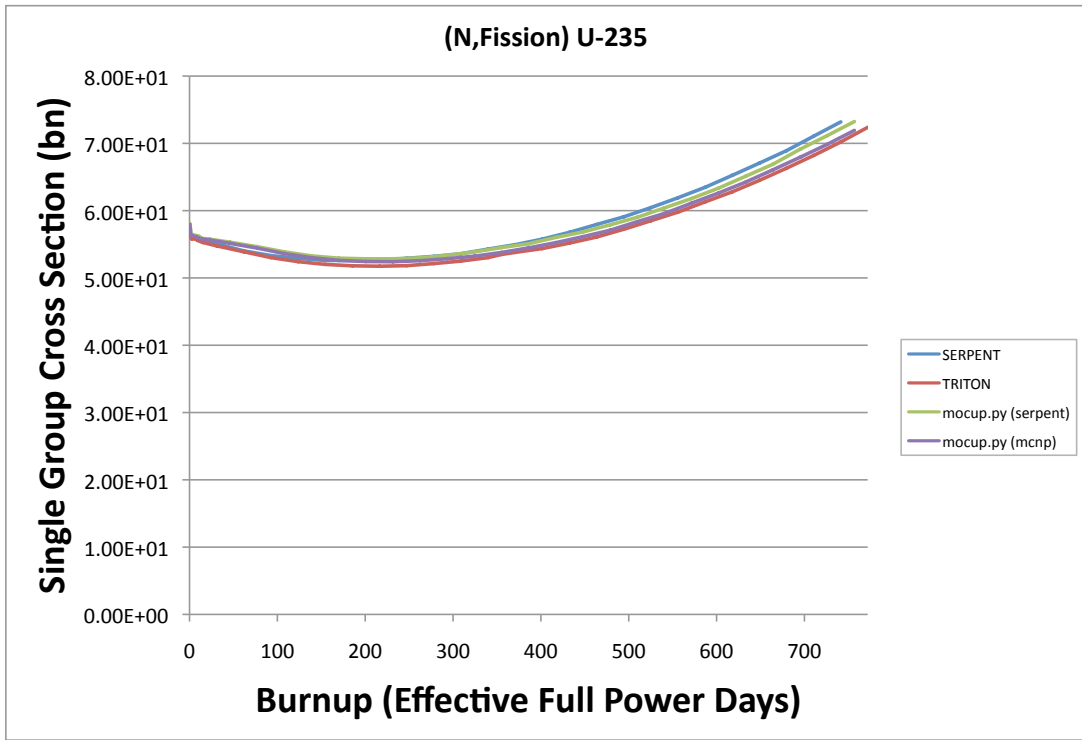


Figure A-26 (n,fission) U-235 single-group cross section evolution

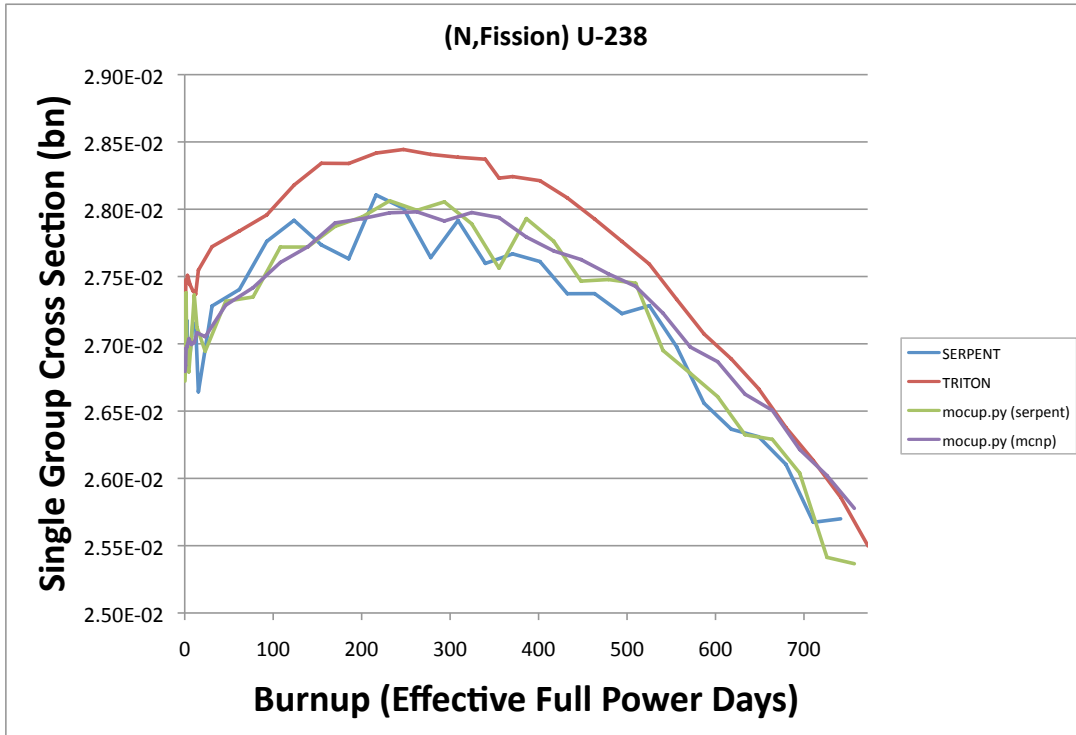


Figure A-27 (n,fission) U-238 single-group cross section evolution

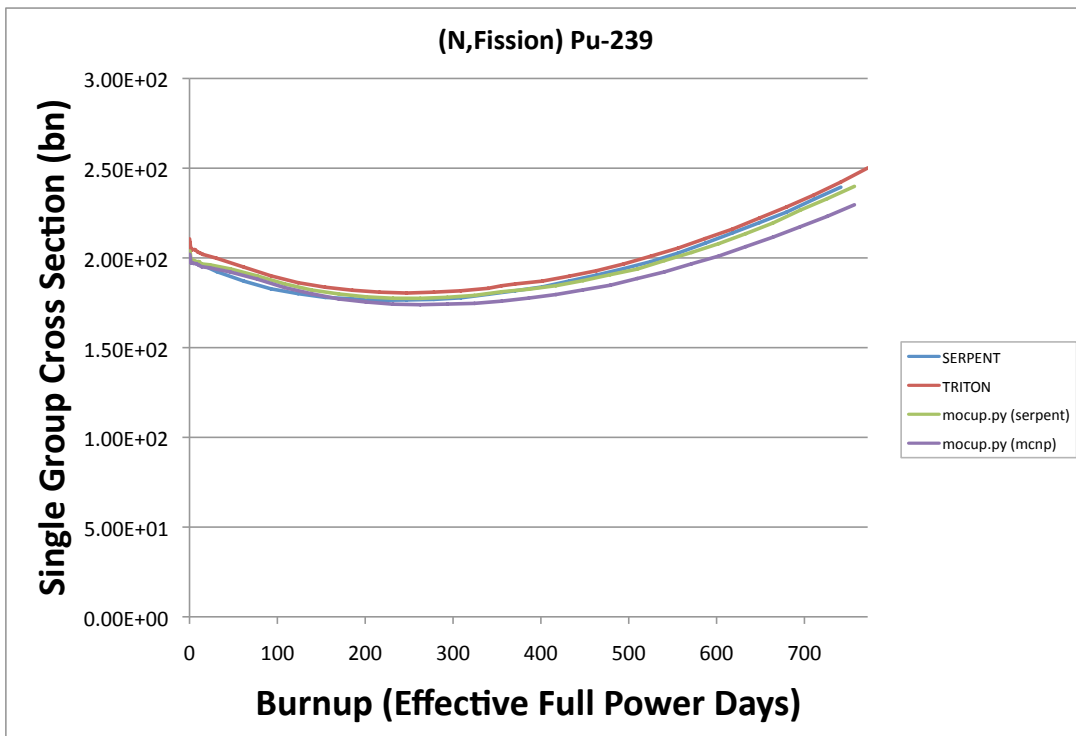


Figure A-28 (n,fission) Pu-239 single-group cross section evolution

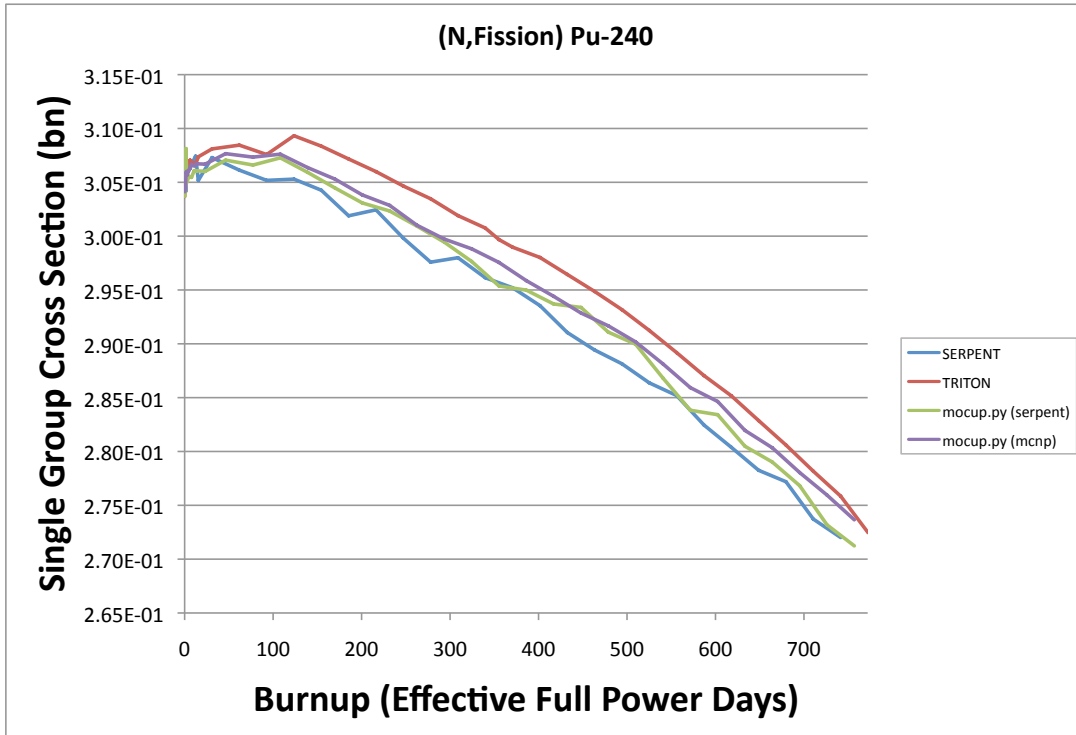


Figure A-29 (n,fission) Pu-240 single-group cross section evolution

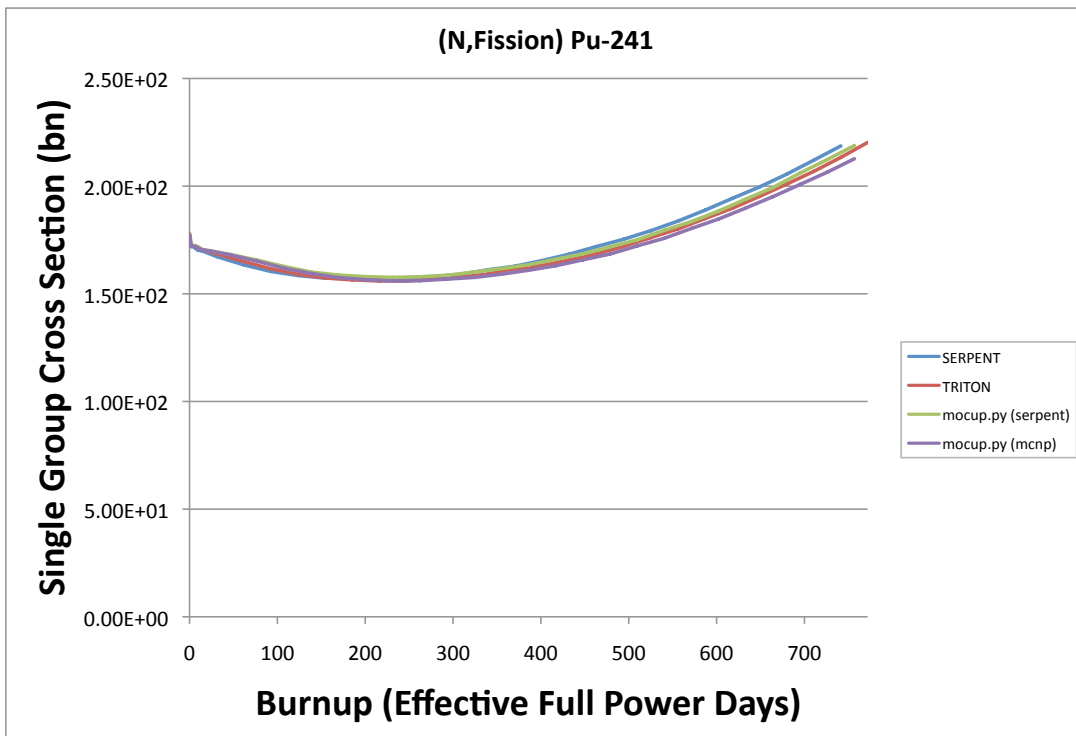


Figure A-30 (n,fission) Pu-241 single-group cross section evolution

A.4.3 Conclusion

mocup.py depletion analysis reasonably predicts similar k_{∞} evolutions, isotopic concentration evolutions- and cross section evolutions- for important isotopes compared to SERPENT and TRITON. Therefore, one can use mocup.py with reasonable confidence.

A.5 References

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A.6 Acknowledgements

Special thanks to Ryan Bergman for maintaining the Berkelium cluster and the mocup.py user base (in alphabetical order): Ryan Bergmann, Alejandra Jolodosky, Madicken Monk, Staffan Qvist, Suso Soto, Chris Varela, and Guanheng "George" Zhang.

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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 Office of Nuclear Energy.

Appendix B: Face Centered Cubic Pebble Unit Cell MCNP5 Input Deck

Infinite Pebble Bed - Advanced High Temperature Reactor

```
c -----
c Cells
c -----
c Depletion Zone 1
10 10 7.137462E-02 -1 u=101 tmp=8.93144E-08 imp:n=1 $ Kernel 1
    vol=7.92486E-02
11 12 -1.70386E+00 1 u=101 tmp=8.56711E-08 imp:n=1 $ Coatings and Matrix
12 0 -11 12 -13 14 -15 16 lat=1      imp:n=1 fill=101 u=91 $ TRISO lattice
c Depletion Zone 2
20 20 7.221424E-02 -1 u=102 tmp=8.93144E-08 imp:n=1 $ Kernel 2
    vol=7.92486E-02
21 12 -1.70386E+00 1 u=102 tmp=8.56711E-08 imp:n=1 $ Coatings and Matrix
22 0 -11 12 -13 14 -15 16 lat=1      imp:n=1 fill=102 u=92 $ TRISO lattice
c Depletion Zone 3
30 30 7.298886E-02 -1 u=103 tmp=8.93144E-08 imp:n=1 $ Kernel 3
    vol=7.92486E-02
31 12 -1.70386E+00 1 u=103 tmp=8.56711E-08 imp:n=1 $ Coatings and Matrix
32 0 -11 12 -13 14 -15 16 lat=1      imp:n=1 fill=103 u=93 $ TRISO lattice
c Depletion Zone 4
40 40 7.369491E-02 -1 u=104 tmp=8.93144E-08 imp:n=1 $ Kernel 4
    vol=7.92486E-02
41 12 -1.70386E+00 1 u=104 tmp=8.56711E-08 imp:n=1 $ Coatings and Matrix
42 0 -11 12 -13 14 -15 16 lat=1      imp:n=1 fill=104 u=94 $ TRISO lattice
c Depletion Zone 5
50 50 7.432124E-02 -1 u=105 tmp=8.93144E-08 imp:n=1 $ Kernel 5
    vol=7.92486E-02
51 12 -1.70386E+00 1 u=105 tmp=8.56711E-08 imp:n=1 $ Coatings and Matrix
52 0 -11 12 -13 14 -15 16 lat=1      imp:n=1 fill=105 u=95 $ TRISO lattice
c Depletion Zone 6
60 60 7.487843E-02 -1 u=106 tmp=8.93144E-08 imp:n=1 $ Kernel 6
    vol=7.92486E-02
61 12 -1.70386E+00 1 u=106 tmp=8.56711E-08 imp:n=1 $ Coatings and Matrix
62 0 -11 12 -13 14 -15 16 lat=1      imp:n=1 fill=106 u=96 $ TRISO lattice
c Depletion Zone 7
70 70 7.538065E-02 -1 u=107 tmp=8.93144E-08 imp:n=1 $ Kernel 7
    vol=7.92486E-02
71 12 -1.70386E+00 1 u=107 tmp=8.56711E-08 imp:n=1 $ Coatings and Matrix
72 0 -11 12 -13 14 -15 16 lat=1      imp:n=1 fill=107 u=97 $ TRISO lattice
c Depletion Zone 8
80 80 7.583687E-02 -1 u=108 tmp=8.93144E-08 imp:n=1 $ Kernel 8
```

```

vol=7.92486E-02
81 12 -1.70386E+00 1 u=108 tmp=8.56711E-08 imp:n=1 $ Coatings and Matrix
82 0 -11 12 -13 14 -15 16 lat=1      imp:n=1 fill=108 u=98 $ TRISO lattice
c -----
c Pebbles
110 21 7.99780E-02  -110      u=1 imp:n=1 $ Depletion Region 1
111 0      -111 110 fill=91 u=1 imp:n=1
112 22 8.72414E-02  -112 111      u=1 imp:n=1
113 21 7.99780E-02  -113      u=1 imp:n=1 $ Depletion Region 1
114 0      -114 113 fill=91 u=1 imp:n=1
115 22 8.72414E-02  -115 114      u=1 imp:n=1
116 21 7.99780E-02  -116      u=1 imp:n=1 $ Depletion Region 1
117 0      -117 116 fill=91 u=1 imp:n=1
118 22 8.72414E-02  -118 117      u=1 imp:n=1
119 21 7.99780E-02  -119      u=1 imp:n=1 $ Depletion Region 1
120 0      -120 119 fill=91 u=1 imp:n=1
121 22 8.72414E-02  -121 120      u=1 imp:n=1
c
123 21 7.99780E-02  -130      u=1 imp:n=1 $ Depletion Region 2
124 0      -131 130 fill=92 u=1 imp:n=1
125 22 8.72414E-02  -132 131      u=1 imp:n=1
126 21 7.99780E-02  -133      u=1 imp:n=1 $ Depletion Region 2
127 0      -134 133 fill=92 u=1 imp:n=1
128 22 8.72414E-02  -135 134      u=1 imp:n=1
129 21 7.99780E-02  -136      u=1 imp:n=1 $ Depletion Region 2
130 0      -137 136 fill=92 u=1 imp:n=1
131 22 8.72414E-02  -138 137      u=1 imp:n=1
132 21 7.99780E-02  -139      u=1 imp:n=1 $ Depletion Region 2
133 0      -140 139 fill=92 u=1 imp:n=1
134 22 8.72414E-02  -141 140      u=1 imp:n=1
c
135 21 7.99780E-02  -150      u=1 imp:n=1 $ Depletion Region 3
136 0      -151 150 fill=93 u=1 imp:n=1
137 22 8.72414E-02  -152 151      u=1 imp:n=1
c
138 21 7.99780E-02  -153      u=1 imp:n=1 $ Depletion Region 4
139 0      -154 153 fill=94 u=1 imp:n=1
140 22 8.72414E-02  -155 154      u=1 imp:n=1
c
141 21 7.99780E-02  -160      u=1 imp:n=1 $ Depletion Region 5
142 0      -161 160 fill=95 u=1 imp:n=1
143 22 8.72414E-02  -162 161      u=1 imp:n=1
c
144 21 7.99780E-02  -163      u=1 imp:n=1 $ Depletion Region 6

```

145 0 -164 163 fill=96 u=1 imp:n=1
 146 22 8.72414E-02 -165 164 u=1 imp:n=1
 c
 147 21 7.99780E-02 -170 u=1 imp:n=1 \$ Depletion Region 7
 148 0 -171 170 fill=97 u=1 imp:n=1
 149 22 8.72414E-02 -172 171 u=1 imp:n=1
 c
 150 21 7.99780E-02 -173 u=1 imp:n=1 \$ Depletion Region 8
 151 0 -174 173 fill=98 u=1 imp:n=1
 152 22 8.72414E-02 -175 174 u=1 imp:n=1
 c
 153 24 8.34284E-02 112 115 118 121
 132 135 138 141
 152 155 162 165 172 175 u=1 imp:n=1 \$ Coolant
 154 0 -900 901 -902 903 -904 905 fill=1 imp:n=1 \$ FCC Unit Cell
 999 0 900:-901:902:-903:904:-905 imp:n=0 \$ Vacuum Boundary Conditions

c -----

c Surface

c -----

c TRISO surfaces

1 so 2.00000E-02 \$ Kernel

c Fuel Particles Lattice

11 px 4.43031E-02

12 px -4.43031E-02

13 py 4.43031E-02

14 py -4.43031E-02

15 pz 4.43031E-02

16 pz -4.43031E-02

c Pebble Surfaces

110 s 2.27541E+00 2.27541E+00 2.27541E+00 1.25114E+00

111 s 2.27541E+00 2.27541E+00 2.27541E+00 1.40000E+00

112 s 2.27541E+00 2.27541E+00 2.27541E+00 1.50000E+00

c

113 s -2.27541E+00 2.27541E+00 2.27541E+00 1.25114E+00

114 s -2.27541E+00 2.27541E+00 2.27541E+00 1.40000E+00

115 s -2.27541E+00 2.27541E+00 2.27541E+00 1.50000E+00

c

116 s -2.27541E+00 -2.27541E+00 2.27541E+00 1.25114E+00

117 s -2.27541E+00 -2.27541E+00 2.27541E+00 1.40000E+00

118 s -2.27541E+00 -2.27541E+00 2.27541E+00 1.50000E+00

c

119 s 2.27541E+00 -2.27541E+00 2.27541E+00 1.25114E+00

120 s 2.27541E+00 -2.27541E+00 2.27541E+00 1.40000E+00

121 s 2.27541E+00 -2.27541E+00 2.27541E+00 1.50000E+00
 c
 130 s 2.27541E+00 2.27541E+00 -2.27541E+00 1.25114E+00
 131 s 2.27541E+00 2.27541E+00 -2.27541E+00 1.40000E+00
 132 s 2.27541E+00 2.27541E+00 -2.27541E+00 1.50000E+00
 c
 133 s -2.27541E+00 2.27541E+00 -2.27541E+00 1.25114E+00
 134 s -2.27541E+00 2.27541E+00 -2.27541E+00 1.40000E+00
 135 s -2.27541E+00 2.27541E+00 -2.27541E+00 1.50000E+00
 c
 136 s -2.27541E+00 -2.27541E+00 -2.27541E+00 1.25114E+00
 137 s -2.27541E+00 -2.27541E+00 -2.27541E+00 1.40000E+00
 138 s -2.27541E+00 -2.27541E+00 -2.27541E+00 1.50000E+00
 c
 139 s 2.27541E+00 -2.27541E+00 -2.27541E+00 1.25114E+00
 140 s 2.27541E+00 -2.27541E+00 -2.27541E+00 1.40000E+00
 141 s 2.27541E+00 -2.27541E+00 -2.27541E+00 1.50000E+00
 c
 150 s 2.27541E+00 0 0 1.25114E+00
 151 s 2.27541E+00 0 0 1.40000E+00
 152 s 2.27541E+00 0 0 1.50000E+00
 c
 153 s -2.27541E+00 0 0 1.25114E+00
 154 s -2.27541E+00 0 0 1.40000E+00
 155 s -2.27541E+00 0 0 1.50000E+00
 c
 160 s 0 2.27541E+00 0 1.25114E+00
 161 s 0 2.27541E+00 0 1.40000E+00
 162 s 0 2.27541E+00 0 1.50000E+00
 c
 163 s 0 -2.27541E+00 0 1.25114E+00
 164 s 0 -2.27541E+00 0 1.40000E+00
 165 s 0 -2.27541E+00 0 1.50000E+00
 c
 170 s 0 0 2.27541E+00 1.25114E+00
 171 s 0 0 2.27541E+00 1.40000E+00
 172 s 0 0 2.27541E+00 1.50000E+00
 c
 173 s 0 0 -2.27541E+00 1.25114E+00
 174 s 0 0 -2.27541E+00 1.40000E+00
 175 s 0 0 -2.27541E+00 1.50000E+00

c Unit Cell Surfaces
 *900 px 2.27541E+00

*901 px -2.27541E+00
*902 py 2.27541E+00
*903 pz -2.27541E+00
*904 px 2.27541E+00
*905 pz -2.27541E+00

c -----
c Data

c -----
c Fuel Mix (800 C - Scattering Kernel at 727 C)

m10 6000.73c 1.182344e-02
8016.73c 3.547031e-02
90232.73c 2.242860E-13
90233.73c 1.547395E-18
91233.73c 1.698920E-14
92233.73c 6.268153E-12
92234.73c 6.549850E-09
92235.73c 4.147410E-03
92236.73c 1.217444E-04
92237.73c 7.925811E-07
92238.73c 1.866327E-02
92239.73c 1.630300E-07
93236.73c 6.438980E-14
93237.73c 1.584326E-06
93238.73c 1.355063E-08
93239.73c 2.194682E-05
94236.73c 9.825573E-14
94237.73c 9.072507E-14
94238.73c 6.464513E-08
94239.73c 1.481207E-04
94240.73c 1.615710E-05
94241.73c 5.340841E-06
94242.73c 2.740218E-07
94243.73c 2.659136E-10
94244.73c 3.796942E-12
95241.73c 8.986638E-09
95642.73c 2.048780E-10
95242.73c 7.020080E-11
95243.73c 1.005810E-08
95244.73c 1.505068E-12
96242.73c 6.850849E-10
96243.73c 2.794931E-12
96244.73c 5.198131E-10
96245.73c 8.429628E-12

96246.73c 9.239687E-14
96247.73c 2.139133E-16
96248.73c 2.431772E-18
97249.73c 8.985118E-21
98249.73c 9.661434E-23
98250.73c 1.249816E-21
35081.73c 1.087044E-06
36083.73c 2.608906E-06
36084.73c 5.251552E-06
36086.73c 1.000111E-05
37085.73c 4.976695E-06
37087.73c 1.276261E-05
38088.73c 1.824380E-05
38089.73c 1.781294E-05
38090.73c 2.836802E-05
39089.73c 5.925588E-06
39091.73c 2.261934E-05
40091.73c 6.366257E-06
40092.73c 2.974725E-05
40093.73c 3.201100E-05
40094.73c 3.177923E-05
40095.73c 2.601155E-05
40096.73c 3.224809E-05
42095.73c 1.767083E-06
42097.73c 3.013860E-05
42098.73c 3.042736E-05
42099.73c 3.477326E-06
42100.73c 3.295253E-05
43099.73c 2.683605E-05
44101.73c 2.653513E-05
44102.73c 2.299701E-05
44103.73c 1.251336E-05
44104.73c 1.141833E-05
44105.73c 6.171494E-08
44106.73c 3.505594E-06
45103.73c 4.932392E-06
45105.73c 3.840029E-07
46104.73c 3.082099E-07
46105.73c 5.220320E-06
46106.73c 1.515023E-06
46107.73c 2.131458E-06
46108.73c 1.287203E-06
46110.73c 4.024534E-07
47109.73c 7.042269E-07

48110.73c 5.176094E-08
48111.73c 1.692841E-07
48113.73c 1.296398E-08
48114.73c 2.231537E-07
49115.73c 6.807230E-08
52120.73c 1.000000E-30
53127.73c 6.976993E-07
53129.73c 3.727183E-06
54131.73c 1.018349E-05
54132.73c 2.053948E-05
54134.73c 4.001053E-05
54135.73c 6.855180E-08
54136.73c 6.432293E-05
55133.73c 2.599635E-05
55134.73c 6.449771E-07
55135.73c 4.230772E-06
55137.73c 3.187878E-05
56138.73c 3.526264E-05
56140.73c 1.285532E-05
57139.73c 3.314554E-05
58141.73c 1.966331E-05
58142.73c 3.044028E-05
58143.73c 1.735091E-06
59141.73c 1.034079E-05
59143.73c 1.194875E-05
60143.73c 1.563581E-05
60144.73c 2.286099E-06
60145.73c 1.939050E-05
60146.73c 1.597549E-05
60147.73c 4.115494E-06
60148.73c 9.203971E-06
60150.73c 3.511142E-06
61147.73c 6.362534E-06
61148.73c 1.464869E-07
61548.73c 4.804956E-08
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62147.73c 8.776144E-08
62149.73c 3.807049E-07
62150.73c 4.945767E-06
62151.73c 1.039398E-06
62152.73c 2.380175E-06
62153.73c 1.322159E-07
62154.73c 5.257100E-07
63153.73c 1.121544E-06

63154.73c 8.980559E-08
63155.73c 9.619639E-08
63156.73c 1.384699E-07
64155.73c 2.243164E-10
64156.73c 1.384319E-07
64157.73c 5.297755E-09
64158.73c 1.129903E-07

c

m20 6000.73c 1.182344e-02

8016.73c 3.547031e-02
90232.73c 1.394426E-12
90233.73c 9.650035E-18
91233.73c 2.006150E-13
92233.73c 1.649374E-11
92234.73c 1.690941E-08
92235.73c 3.133849E-03
92236.73c 3.164777E-04
92237.73c 2.408443E-06
92238.73c 1.820809E-02
92239.73c 1.631744E-07
93236.73c 8.033717E-13
93237.73c 1.125723E-05
93238.73c 1.043198E-07
93239.73c 2.331769E-05
94236.73c 1.528245E-12
94237.73c 7.123351E-13
94238.73c 1.093807E-06
94239.73c 3.239932E-04
94240.73c 6.543771E-05
94241.73c 4.224541E-05
94242.73c 4.849183E-06
94243.73c 4.740744E-09
94244.73c 1.563733E-10
95241.73c 1.451647E-07
95642.73c 6.125139E-09
95242.73c 1.166910E-09
95243.73c 3.735162E-07
95244.73c 5.757193E-11
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96244.73c 4.047559E-08
96245.73c 1.273525E-09
96246.73c 2.872669E-11
96247.73c 1.297310E-13

96248.73c 2.949192E-15
97249.73c 1.903106E-17
98249.73c 3.794738E-19
98250.73c 4.554263E-18
35081.73c 3.079820E-06
36083.73c 7.034594E-06
36084.73c 1.496937E-05
36086.73c 2.767954E-05
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37087.73c 3.533027E-05
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38089.73c 3.450957E-05
38090.73c 7.818664E-05
39089.73c 3.098361E-05
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40094.73c 9.037552E-05
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42097.73c 8.904568E-05
42098.73c 8.858214E-05
42099.73c 3.504379E-06
42100.73c 9.646996E-05
43099.73c 8.273847E-05
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44103.73c 2.677070E-05
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46105.73c 1.955540E-05
46106.73c 6.585566E-06
46107.73c 9.805816E-06
46108.73c 6.367549E-06
46110.73c 1.922788E-06
47109.73c 3.503315E-06
48110.73c 5.517671E-07
48111.73c 8.858214E-07

48113.73c 1.904322E-08
48114.73c 8.015479E-07
49115.73c 1.935706E-07
52120.73c 1.000000E-30
53127.73c 2.597052E-06
53129.73c 1.202474E-05
54131.73c 3.592300E-05
54132.73c 6.980489E-05
54134.73c 1.164935E-04
54135.73c 6.491262E-08
54136.73c 1.894216E-04
55133.73c 8.722191E-05
55134.73c 5.050102E-06
55135.73c 1.261139E-05
55137.73c 9.308838E-05
56138.73c 1.013182E-04
56140.73c 1.654086E-05
57139.73c 9.550488E-05
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60145.73c 5.443124E-05
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60148.73c 2.713165E-05
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61147.73c 2.041637E-05
61148.73c 5.435373E-07
61548.73c 1.681442E-07
61149.73c 6.525533E-07
62147.73c 6.757228E-07
62149.73c 4.847055E-07
62150.73c 1.712446E-05
62151.73c 1.852801E-06
62152.73c 7.602851E-06
62153.73c 2.771146E-07
62154.73c 1.870279E-06
63153.73c 4.940219E-06
63154.73c 6.680934E-07
63155.73c 2.789384E-07

63156.73c 4.339970E-07
64155.73c 7.356034E-10
64156.73c 9.724506E-07
64157.73c 1.158323E-08
64158.73c 5.509312E-07

c

m30 6000.73c 1.182344e-02

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92233.73c 2.112612E-11
92234.73c 2.631324E-08
92235.73c 2.353426E-03
92236.73c 4.537090E-04
92237.73c 3.690023E-06
92238.73c 1.775062E-02
92239.73c 1.676655E-07
93236.73c 2.678970E-12
93237.73c 2.750553E-05
93238.73c 2.717421E-07
93239.73c 2.390966E-05
94236.73c 6.141249E-12
94237.73c 2.204257E-12
94238.73c 4.566802E-06
94239.73c 3.810544E-04
94240.73c 1.005506E-04
94241.73c 9.564166E-05
94242.73c 1.924308E-05
94243.73c 1.956832E-08
94244.73c 1.165618E-09
95241.73c 5.013094E-07
95642.73c 3.092510E-08
95242.73c 4.246350E-09
95243.73c 2.409811E-06
95244.73c 3.912372E-10
96242.73c 1.452255E-07
96243.73c 1.878713E-09
96244.73c 4.314057E-07
96245.73c 2.023323E-08
96246.73c 7.434988E-10
96247.73c 5.177082E-12
96248.73c 1.865263E-13
97249.73c 1.643599E-15

98249.73c 4.548488E-17
98250.73c 5.311205E-16
35081.73c 4.803284E-06
36083.73c 1.036131E-05
36084.73c 2.367636E-05
36086.73c 4.245058E-05
37085.73c 2.144072E-05
37087.73c 5.411816E-05
38088.73c 7.733555E-05
38089.73c 3.673761E-05
38090.73c 1.193355E-04
39089.73c 6.333429E-05
39091.73c 5.195015E-05
40091.73c 7.340912E-05
40092.73c 1.299058E-04
40093.73c 1.421251E-04
40094.73c 1.420795E-04
40095.73c 6.600156E-05
40096.73c 1.462817E-04
42095.73c 4.918866E-05
42097.73c 1.424898E-04
42098.73c 1.422998E-04
42099.73c 3.273671E-06
42100.73c 1.556058E-04
43099.73c 1.308633E-04
44101.73c 1.243585E-04
44102.73c 1.168962E-04
44103.73c 3.210979E-05
44104.73c 6.843402E-05
44105.73c 9.877247E-08
44106.73c 2.768866E-05
45103.73c 5.074039E-05
45105.73c 6.265494E-07
46104.73c 1.252324E-05
46105.73c 3.684020E-05
46106.73c 1.455142E-05
46107.73c 2.106989E-05
46108.73c 1.416919E-05
46110.73c 4.257748E-06
47109.73c 7.493652E-06
48110.73c 1.962607E-06
48111.73c 1.994979E-06
48113.73c 2.233589E-08
48114.73c 1.509552E-06

49115.73c 2.798275E-07
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53127.73c 4.677596E-06
53129.73c 2.056911E-05
54131.73c 5.609848E-05
54132.73c 1.216000E-04
54134.73c 1.869747E-04
54135.73c 5.901499E-08
54136.73c 3.066141E-04
55133.73c 1.399289E-04
55134.73c 1.355139E-05
55135.73c 2.064282E-05
55137.73c 1.498913E-04
56138.73c 1.612747E-04
56140.73c 1.528169E-05
57139.73c 1.519278E-04
58141.73c 3.610309E-05
58142.73c 1.409472E-04
58143.73c 1.452179E-06
59141.73c 1.006266E-04
59143.73c 1.454762E-05
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60145.73c 8.363516E-05
60146.73c 7.735834E-05
60147.73c 4.634965E-06
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61147.73c 2.866362E-05
61148.73c 8.165941E-07
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61149.73c 7.076389E-07
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62151.73c 2.183967E-06
62152.73c 1.166606E-05
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63156.73c 8.673557E-07
64155.73c 1.550206E-09

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64157.73c 1.849077E-08
64158.73c 1.288799E-06

c

m40 6000.73c 1.182344e-02

8016.73c 3.547031e-02
90232.73c 5.951425E-12
90233.73c 4.348329E-17
91233.73c 1.198522E-12
92233.73c 2.209880E-11
92234.73c 3.905760E-08
92235.73c 1.755077E-03
92236.73c 5.467973E-04
92237.73c 4.461479E-06
92238.73c 1.729164E-02
92239.73c 1.639875E-07
93236.73c 5.136199E-12
93237.73c 4.652671E-05
93238.73c 4.661334E-07
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94238.73c 1.114477E-05
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94241.73c 1.372997E-04
94242.73c 4.188749E-05
94243.73c 4.287689E-08
94244.73c 4.153869E-09
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95642.73c 7.392585E-08
95242.73c 8.114267E-09
95243.73c 7.481190E-06
95244.73c 1.212733E-09
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96244.73c 1.938746E-06
96245.73c 1.207717E-07
96246.73c 6.438600E-09
96247.73c 6.152192E-11
96248.73c 3.101477E-12
97249.73c 3.354373E-14
98249.73c 1.194115E-15
98250.73c 1.318283E-14

35081.73c 6.267622E-06
36083.73c 1.275349E-05
36084.73c 3.147147E-05
36086.73c 5.475572E-05
37085.73c 2.775402E-05
37087.73c 6.971446E-05
38088.73c 9.960077E-05
38089.73c 3.366836E-05
38090.73c 1.531589E-04
39089.73c 9.495015E-05
39091.73c 4.935280E-05
40091.73c 1.126179E-04
40092.73c 1.690409E-04
40093.73c 1.856752E-04
40094.73c 1.873166E-04
40095.73c 6.620597E-05
40096.73c 1.939430E-04
42095.73c 8.964601E-05
42097.73c 1.902271E-04
42098.73c 1.911693E-04
42099.73c 2.928142E-06
42100.73c 2.096274E-04
43099.73c 1.711230E-04
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44102.73c 1.627793E-04
44103.73c 3.338035E-05
44104.73c 9.998072E-05
44105.73c 1.030735E-07
44106.73c 4.221045E-05
45103.73c 7.363329E-05
45105.73c 6.273625E-07
46104.73c 2.735203E-05
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61548.73c 2.411559E-07
61149.73c 5.574664E-07
62147.73c 5.552171E-06
62149.73c 3.989882E-07
62150.73c 7.595176E-05
62151.73c 3.054439E-06
62152.73c 1.973930E-05
62153.73c 6.000895E-07
62154.73c 1.286595E-05
63153.73c 3.104212E-05
63154.73c 7.760911E-06
63155.73c 2.203269E-06
63156.73c 3.428844E-06
64155.73c 7.605890E-09
64156.73c 3.432871E-05
64157.73c 6.177573E-08
64158.73c 1.002695E-05

c

c Single isotopes for depletion

m101 90232.73c 1.

m102 90233.73c 1.

m103 91233.73c 1.

m104 92233.73c 1.

m105 92234.73c 1.

m106 92235.73c 1.
m107 92236.73c 1.
m108 92237.73c 1.
m109 92238.73c 1.
m110 92239.73c 1.
m111 93236.73c 1.
m112 93237.73c 1.
m113 93238.73c 1.
m114 93239.73c 1.
m115 94236.73c 1.
m116 94237.73c 1.
m117 94238.73c 1.
m118 94239.73c 1.
m119 94240.73c 1.
m120 94241.73c 1.
m121 94242.73c 1.
m122 94243.73c 1.
m123 94244.73c 1.
m124 95241.73c 1.
m125 95642.73c 1.
m126 95242.73c 1.
m127 95243.73c 1.
m128 95244.73c 1.
m129 96242.73c 1.
m130 96243.73c 1.
m131 96244.73c 1.
m132 96245.73c 1.
m133 96246.73c 1.
m134 96247.73c 1.
m135 96248.73c 1.
m136 97249.73c 1.
m137 98249.73c 1.
m138 98250.73c 1.
m139 35081.73c 1.
m140 36083.73c 1.
m141 36084.73c 1.
m142 36086.73c 1.
m143 37085.73c 1.
m144 37087.73c 1.
m145 38088.73c 1.
m146 38089.73c 1.
m147 38090.73c 1.
m148 39089.73c 1.
m149 39091.73c 1.

m150 40091.73c 1.
m151 40092.73c 1.
m152 40093.73c 1.
m153 40094.73c 1.
m154 40095.73c 1.
m155 40096.73c 1.
m156 42095.73c 1.
m157 42097.73c 1.
m158 42098.73c 1.
m159 42099.73c 1.
m160 42100.73c 1.
m161 43099.73c 1.
m162 44101.73c 1.
m163 44102.73c 1.
m164 44103.73c 1.
m165 44104.73c 1.
m166 44105.73c 1.
m167 44106.73c 1.
m168 45103.73c 1.
m169 45105.73c 1.
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m171 46105.73c 1.
m172 46106.73c 1.
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m174 46108.73c 1.
m175 46110.73c 1.
m176 47109.73c 1.
m177 48110.73c 1.
m178 48111.73c 1.
m179 48113.73c 1.
m180 48114.73c 1.
m181 49115.73c 1.
m182 52120.73c 1.
m183 53127.73c 1.
m184 53129.73c 1.
m185 54131.73c 1.
m186 54132.73c 1.
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m188 54135.73c 1.
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m202 60143.73c 1.
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m204 60145.73c 1.
m205 60146.73c 1.
m206 60147.73c 1.
m207 60148.73c 1.
m208 60150.73c 1.
m209 61147.73c 1.
m210 61148.73c 1.
m211 61548.73c 1.
m212 61149.73c 1.
m213 62147.73c 1.
m214 62149.73c 1.
m215 62150.73c 1.
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m218 62153.73c 1.
m219 62154.73c 1.
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m221 63154.73c 1.
m222 63155.73c 1.
m223 63156.73c 1.
m224 64155.73c 1.
m225 64156.73c 1.
m226 64157.73c 1.
m227 64158.73c 1.

c

c Carbon

m2 6000.73c 1.

mt2 grph.64t

c

c Pyretic Carbon

m3 6000.73c 1.

mt3 grph.64t

c

c SiC

```

m4 6000.73c 1.
    14000.16c 1.
c
c Graphite in the Matrix
m6 6000.71c 1.
mt6 grph.64t
c
c Coatings and matrix mix
m12 6000.73c 7.62188E-02
    14028.73c 3.98085E-03
mt12 grph.64t
c
c Graphite in the pebble central core (750 C - Scattering Kernel at 727 C)
m21 6000.73c 1.
mt21 grph.64t
c
c Graphite in the pebble shell (690 C - Scattering Kernel at 727 C)
m22 6000.73c 1.
mt22 grph.64t
c
c Salt in the Core - 2LiF-BeF2 - (655 C)
m24 3006.73c 3.28E-07
    3007.73c 2.38358E-02
    4009.73c 1.19185E-02
    9019.73c 4.76740E-02
m536 3006.73c 1
m549 4009.73c 1
f64:n 153
fm64 (1 536 -2)
sd64 1
f74:n 153
fm74 (1 549 107)
sd74 1
c begin_mocup_reaction_rate_tallies
c time dependent reaction rates
fc104 Reaction rates
f104:n
    10 20 30 40 50 60 70 80
fm104 (1)
    (1 101 (16) (17) (-6) (102) )
    (1 102 (16) (17) (-6) (102) )
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c

f314:n 10
fm314 (-1 10 (-6))
sd314 1
f324:n 20
fm324 (-1 20 (-6))
sd324 1
f334:n 30
fm334 (-1 30 (-6))
sd334 1

f344:n 40
 fm344 (-1 40 (-6))
 sd344 1
 f354:n 50
 fm354 (-1 50 (-6))
 sd354 1
 f364:n 60
 fm364 (-1 60 (-6))
 sd364 1
 f374:n 70
 fm374 (-1 70 (-6))
 sd374 1
 f384:n 80
 fm384 (-1 80 (-6))
 sd384 1
 fc404 Neutron Spectrum Tally
 f404:n 10 20 30 40 50 60 70 80
 e404 1.0000E-11 1.0000E-10 5.0000E-10 7.5000E-10 1.0000E-09 1.2000E-09
 1.5000E-09 2.0000E-09 2.5000E-09 3.0000E-09 4.0000E-09 5.0000E-09
 7.5000E-09 1.0000E-08 2.5300E-08 3.0000E-08 4.0000E-08 5.0000E-08
 6.0000E-08 7.0000E-08 8.0000E-08 9.0000E-08 1.0000E-07 1.2500E-07
 1.5000E-07 1.7500E-07 2.0000E-07 2.2500E-07 2.5000E-07 2.7500E-07
 3.0000E-07 3.2500E-07 3.5000E-07 3.7500E-07 4.0000E-07 4.5000E-07
 5.0000E-07 5.5000E-07 6.0000E-07 6.2500E-07 6.5000E-07 7.0000E-07
 7.5000E-07 8.0000E-07 8.5000E-07 9.0000E-07 9.2500E-07 9.5000E-07
 9.7500E-07 1.0000E-06 1.0100E-06 1.0200E-06 1.0300E-06 1.0400E-06
 1.0500E-06 1.0600E-06 1.0700E-06 1.0800E-06 1.0900E-06 1.1000E-06
 1.1100E-06 1.1200E-06 1.1300E-06 1.1400E-06 1.1500E-06 1.1750E-06
 1.2000E-06 1.2250E-06 1.2500E-06 1.3000E-06 1.3500E-06 1.4000E-06
 1.4500E-06 1.5000E-06 1.5900E-06 1.6800E-06 1.7700E-06 1.8600E-06
 1.9400E-06 2.0000E-06 2.1200E-06 2.2100E-06 2.3000E-06 2.3800E-06
 2.4700E-06 2.5700E-06 2.6700E-06 2.7700E-06 2.8700E-06 2.9700E-06
 3.0000E-06 3.0500E-06 3.1500E-06 3.5000E-06 3.7300E-06 4.0000E-06
 4.7500E-06 5.0000E-06 5.4000E-06 6.0000E-06 6.2500E-06 6.5000E-06
 6.7500E-06 7.0000E-06 7.1500E-06 8.1000E-06 9.1000E-06 1.0000E-05
 1.1500E-05 1.1900E-05 1.2900E-05 1.3750E-05 1.4400E-05 1.5100E-05
 1.6000E-05 1.7000E-05 1.8500E-05 1.9000E-05 2.0000E-05 2.1000E-05
 2.2500E-05 2.5000E-05 2.7500E-05 3.0000E-05 3.1250E-05 3.1750E-05
 3.3250E-05 3.3750E-05 3.4600E-05 3.5500E-05 3.7000E-05 3.8000E-05
 3.9100E-05 3.9600E-05 4.1000E-05 4.2400E-05 4.4000E-05 4.5200E-05
 4.7000E-05 4.8300E-05 4.9200E-05 5.0600E-05 5.2000E-05 5.3400E-05
 5.9000E-05 6.1000E-05 6.5000E-05 6.7500E-05 7.2000E-05 7.6000E-05
 8.0000E-05 8.2000E-05 9.0000E-05 1.0000E-04 1.0800E-04 1.1500E-04
 1.1900E-04 1.2200E-04 1.8600E-04 1.9250E-04 2.0750E-04 2.1000E-04

2.4000E-04 2.8500E-04 3.0500E-04 5.5000E-04 6.7000E-04 6.8300E-04
9.5000E-04 1.1500E-03 1.5000E-03 1.5500E-03 1.8000E-03 2.2000E-03
2.2900E-03 2.5800E-03 3.0000E-03 3.7400E-03 3.9000E-03 6.0000E-03
8.0300E-03 9.5000E-03 1.3000E-02 1.7000E-02 2.5000E-02 3.0000E-02
4.5000E-02 5.0000E-02 5.2000E-02 6.0000E-02 7.3000E-02 7.5000E-02
8.2000E-02 8.5000E-02 1.0000E-01 1.2830E-01 1.5000E-01 2.0000E-01
2.7000E-01 3.3000E-01 4.0000E-01 4.2000E-01 4.4000E-01 4.7000E-01
4.9952E-01 5.5000E-01 5.7300E-01 6.0000E-01 6.7000E-01 6.7900E-01
7.5000E-01 8.2000E-01 8.6110E-01 8.7500E-01 9.0000E-01 9.2000E-01
1.0100E+00 1.1000E+00 1.2000E+00 1.2500E+00 1.3170E+00 1.3560E+00
1.4000E+00 1.5000E+00 1.8500E+00 2.3540E+00 2.4790E+00 3.0000E+00
4.3040E+00 4.8000E+00 6.4340E+00 8.1873E+00 1.0000E+01 1.2840E+01
1.3840E+01 1.4550E+01 1.5683E+01 1.7333E+01 2.0000E+01

sd404 1 7r

kcode 10000 1.0 10 110

prdmp 10000 10000 10000

print

mode n

Appendix C: Mark 1 PB-FHR full-core MCNP5 input deck

c Mark I PB-FHR

c

c Power (MW): 236.0

c Power Density (MWt/m3): 23.0

c Fuel Fraction (%): 68.1

c Graphite Reflector Thickness (cm): 40.0

c Fuel Design

c CHM: 300.8

c Kernel Diameter (um): 400.0

c Ave Density (g/cc): 1.745

c

c -----

c Cells

c

c

c Depletion Zones 11

c

11100 11100 7.09608E-02 -1 u=1111 tmp=8.66798E-08 imp:n=1 \$ Kernel 11 pass:1
vol=6.71867E+01

11101 11 -1.70386E+00 1 u=1111 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix

11102 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=1111 u=1110

c

11200 11200 7.19257E-02 -1 u=1121 tmp=8.66798E-08 imp:n=1 \$ Kernel 11 pass:2
vol=6.71867E+01

11201 11 -1.70386E+00 1 u=1121 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix

11202 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=1121 u=1120

c

11300 11300 7.27633E-02 -1 u=1131 tmp=8.66798E-08 imp:n=1 \$ Kernel 11 pass:3
vol=6.71867E+01

11301 11 -1.70386E+00 1 u=1131 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix

11302 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=1131 u=1130

c

11400 11400 7.34815E-02 -1 u=1141 tmp=8.66798E-08 imp:n=1 \$ Kernel 11 pass:4
vol=6.71867E+01

11401 11 -1.70386E+00 1 u=1141 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix

11402 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=1141 u=1140

c

11500 11500 7.41002E-02 -1 u=1151 tmp=8.66798E-08 imp:n=1 \$ Kernel 11 pass:5
vol=6.71867E+01

11501 11 -1.70386E+00 1 u=1151 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix

11502 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=1151 u=1150

c

11600 11600 7.42213E-02 -1 u=1161 tmp=8.66798E-08 imp:n=1 \$ Kernel 11 pass:6
 vol=6.71867E+01
 11601 11 -1.70386E+00 1 u=1161 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 11602 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=1161 u=1160
 c
 11700 11700 7.43340E-02 -1 u=1171 tmp=8.66798E-08 imp:n=1 \$ Kernel 11 pass:7
 vol=6.71867E+01
 11701 11 -1.70386E+00 1 u=1171 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 11702 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=1171 u=1170
 c
 11800 11800 7.44396E-02 -1 u=1181 tmp=8.66798E-08 imp:n=1 \$ Kernel 11 pass:8
 vol=6.71867E+01
 11801 11 -1.70386E+00 1 u=1181 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 11802 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=1181 u=1180
 c
 c Depletion Zones 21
 c
 21100 21100 7.09578E-02 -1 u=2111 tmp=8.66798E-08 imp:n=1 \$ Kernel 21 pass:1
 vol=9.99403E+01
 21101 11 -1.70386E+00 1 u=2111 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 21102 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=2111 u=2110
 c
 21200 21200 7.19226E-02 -1 u=2121 tmp=8.66798E-08 imp:n=1 \$ Kernel 21 pass:2
 vol=9.99403E+01
 21201 11 -1.70386E+00 1 u=2121 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 21202 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=2121 u=2120
 c
 21300 21300 7.27608E-02 -1 u=2131 tmp=8.66798E-08 imp:n=1 \$ Kernel 21 pass:3
 vol=9.99403E+01
 21301 11 -1.70386E+00 1 u=2131 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 21302 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=2131 u=2130
 c
 21400 21400 7.34787E-02 -1 u=2141 tmp=8.66798E-08 imp:n=1 \$ Kernel 21 pass:4
 vol=9.99403E+01
 21401 11 -1.70386E+00 1 u=2141 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 21402 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=2141 u=2140
 c
 21500 21500 7.40984E-02 -1 u=2151 tmp=8.66798E-08 imp:n=1 \$ Kernel 21 pass:5
 vol=9.99403E+01
 21501 11 -1.70386E+00 1 u=2151 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 21502 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=2151 u=2150
 c
 21600 21600 7.42191E-02 -1 u=2161 tmp=8.66798E-08 imp:n=1 \$ Kernel 21 pass:6
 vol=9.99403E+01

21601 11 -1.70386E+00 1 u=2161 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 21602 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=2161 u=2160
 c
 21700 21700 7.43324E-02 -1 u=2171 tmp=8.66798E-08 imp:n=1 \$ Kernel 21 pass:7
 vol=9.99403E+01
 21701 11 -1.70386E+00 1 u=2171 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 21702 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=2171 u=2170
 c
 21800 21800 7.44374E-02 -1 u=2181 tmp=8.66798E-08 imp:n=1 \$ Kernel 21 pass:8
 vol=9.99403E+01
 21801 11 -1.70386E+00 1 u=2181 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 21802 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=2181 u=2180
 c
 c Depletion Zones 31
 c
 31100 31100 7.09546E-02 -1 u=3111 tmp=8.66798E-08 imp:n=1 \$ Kernel 31 pass:1
 vol=4.19117E+02
 31101 11 -1.70386E+00 1 u=3111 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 31102 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=3111 u=3110
 c
 31200 31200 7.19202E-02 -1 u=3121 tmp=8.66798E-08 imp:n=1 \$ Kernel 31 pass:2
 vol=4.19117E+02
 31201 11 -1.70386E+00 1 u=3121 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 31202 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=3121 u=3120
 c
 31300 31300 7.27588E-02 -1 u=3131 tmp=8.66798E-08 imp:n=1 \$ Kernel 31 pass:3
 vol=4.19117E+02
 31301 11 -1.70386E+00 1 u=3131 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 31302 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=3131 u=3130
 c
 31400 31400 7.34772E-02 -1 u=3141 tmp=8.66798E-08 imp:n=1 \$ Kernel 31 pass:4
 vol=4.19117E+02
 31401 11 -1.70386E+00 1 u=3141 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 31402 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=3141 u=3140
 c
 31500 31500 7.40976E-02 -1 u=3151 tmp=8.66798E-08 imp:n=1 \$ Kernel 31 pass:5
 vol=4.19117E+02
 31501 11 -1.70386E+00 1 u=3151 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 31502 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=3151 u=3150
 c
 31600 31600 7.42171E-02 -1 u=3161 tmp=8.66798E-08 imp:n=1 \$ Kernel 31 pass:6
 vol=4.19117E+02
 31601 11 -1.70386E+00 1 u=3161 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 31602 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=3161 u=3160

c
31700 31700 7.43313E-02 -1 u=3171 tmp=8.66798E-08 imp:n=1 \$ Kernel 31 pass:7
vol=4.19117E+02
31701 11 -1.70386E+00 1 u=3171 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
31702 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=3171 u=3170

c
31800 31800 7.44358E-02 -1 u=3181 tmp=8.66798E-08 imp:n=1 \$ Kernel 31 pass:8
vol=4.19117E+02
31801 11 -1.70386E+00 1 u=3181 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
31802 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=3181 u=3180

c
c Depletion Zones 41

c
41100 41100 7.09563E-02 -1 u=4111 tmp=8.66798E-08 imp:n=1 \$ Kernel 41 pass:1
vol=1.27498E+02
41101 11 -1.70386E+00 1 u=4111 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
41102 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=4111 u=4110

c
41200 41200 7.19215E-02 -1 u=4121 tmp=8.66798E-08 imp:n=1 \$ Kernel 41 pass:2
vol=1.27498E+02
41201 11 -1.70386E+00 1 u=4121 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
41202 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=4121 u=4120

c
41300 41300 7.27594E-02 -1 u=4131 tmp=8.66798E-08 imp:n=1 \$ Kernel 41 pass:3
vol=1.27498E+02
41301 11 -1.70386E+00 1 u=4131 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
41302 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=4131 u=4130

c
41400 41400 7.34775E-02 -1 u=4141 tmp=8.66798E-08 imp:n=1 \$ Kernel 41 pass:4
vol=1.27498E+02
41401 11 -1.70386E+00 1 u=4141 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
41402 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=4141 u=4140

c
41500 41500 7.40973E-02 -1 u=4151 tmp=8.66798E-08 imp:n=1 \$ Kernel 41 pass:5
vol=1.27498E+02
41501 11 -1.70386E+00 1 u=4151 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
41502 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=4151 u=4150

c
41600 41600 7.42184E-02 -1 u=4161 tmp=8.66798E-08 imp:n=1 \$ Kernel 41 pass:6
vol=1.27498E+02
41601 11 -1.70386E+00 1 u=4161 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
41602 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=4161 u=4160

c
41700 41700 7.43317E-02 -1 u=4171 tmp=8.66798E-08 imp:n=1 \$ Kernel 41 pass:7

vol=1.27498E+02
 41701 11 -1.70386E+00 1 u=4171 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 41702 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=4171 u=4170
 c
 41800 41800 7.44367E-02 -1 u=4181 tmp=8.66798E-08 imp:n=1 \$ Kernel 41 pass:8
 vol=1.27498E+02
 41801 11 -1.70386E+00 1 u=4181 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 41802 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=4181 u=4180
 c
 c
 c Depletion Zones 12
 c
 12100 12100 7.09974E-02 -1 u=1211 tmp=8.66798E-08 imp:n=1 \$ Kernel 12 pass:1
 vol=4.11301E+01
 12101 11 -1.70386E+00 1 u=1211 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 12102 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=1211 u=1210
 c
 12200 12200 7.19594E-02 -1 u=1221 tmp=8.66798E-08 imp:n=1 \$ Kernel 12 pass:2
 vol=4.11301E+01
 12201 11 -1.70386E+00 1 u=1221 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 12202 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=1221 u=1220
 c
 12300 12300 7.27930E-02 -1 u=1231 tmp=8.66798E-08 imp:n=1 \$ Kernel 12 pass:3
 vol=4.11301E+01
 12301 11 -1.70386E+00 1 u=1231 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 12302 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=1231 u=1230
 c
 12400 12400 7.35075E-02 -1 u=1241 tmp=8.66798E-08 imp:n=1 \$ Kernel 12 pass:4
 vol=4.11301E+01
 12401 11 -1.70386E+00 1 u=1241 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 12402 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=1241 u=1240
 c
 12500 12500 7.41216E-02 -1 u=1251 tmp=8.66798E-08 imp:n=1 \$ Kernel 12 pass:5
 vol=4.11301E+01
 12501 11 -1.70386E+00 1 u=1251 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 12502 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=1251 u=1250
 c
 12600 12600 7.42419E-02 -1 u=1261 tmp=8.66798E-08 imp:n=1 \$ Kernel 12 pass:6
 vol=4.11301E+01
 12601 11 -1.70386E+00 1 u=1261 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 12602 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=1261 u=1260
 c
 12700 12700 7.43533E-02 -1 u=1271 tmp=8.66798E-08 imp:n=1 \$ Kernel 12 pass:7
 vol=4.11301E+01

12701 11 -1.70386E+00 1 u=1271 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 12702 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=1271 u=1270
 c
 12800 12800 7.44581E-02 -1 u=1281 tmp=8.66798E-08 imp:n=1 \$ Kernel 12 pass:8
 vol=4.11301E+01
 12801 11 -1.70386E+00 1 u=1281 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 12802 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=1281 u=1280
 c
 c Depletion Zones 22
 c
 22100 22100 7.09901E-02 -1 u=2211 tmp=8.66798E-08 imp:n=1 \$ Kernel 22 pass:1
 vol=6.77434E+01
 22101 11 -1.70386E+00 1 u=2211 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 22102 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=2211 u=2210
 c
 22200 22200 7.19502E-02 -1 u=2221 tmp=8.66798E-08 imp:n=1 \$ Kernel 22 pass:2
 vol=6.77434E+01
 22201 11 -1.70386E+00 1 u=2221 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 22202 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=2221 u=2220
 c
 22300 22300 7.27848E-02 -1 u=2231 tmp=8.66798E-08 imp:n=1 \$ Kernel 22 pass:3
 vol=6.77434E+01
 22301 11 -1.70386E+00 1 u=2231 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 22302 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=2231 u=2230
 c
 22400 22400 7.35011E-02 -1 u=2241 tmp=8.66798E-08 imp:n=1 \$ Kernel 22 pass:4
 vol=6.77434E+01
 22401 11 -1.70386E+00 1 u=2241 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 22402 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=2241 u=2240
 c
 22500 22500 7.41173E-02 -1 u=2251 tmp=8.66798E-08 imp:n=1 \$ Kernel 22 pass:5
 vol=6.77434E+01
 22501 11 -1.70386E+00 1 u=2251 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 22502 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=2251 u=2250
 c
 22600 22600 7.42370E-02 -1 u=2261 tmp=8.66798E-08 imp:n=1 \$ Kernel 22 pass:6
 vol=6.77434E+01
 22601 11 -1.70386E+00 1 u=2261 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 22602 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=2261 u=2260
 c
 22700 22700 7.43499E-02 -1 u=2271 tmp=8.66798E-08 imp:n=1 \$ Kernel 22 pass:7
 vol=6.77434E+01
 22701 11 -1.70386E+00 1 u=2271 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 22702 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=2271 u=2270

c
22800 22800 7.44535E-02 -1 u=2281 tmp=8.66798E-08 imp:n=1 \$ Kernel 22 pass:8
vol=6.77434E+01
22801 11 -1.70386E+00 1 u=2281 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
22802 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=2281 u=2280
c
c Depletion Zones 32
c
32100 32100 7.09882E-02 -1 u=3211 tmp=8.66798E-08 imp:n=1 \$ Kernel 32 pass:1
vol=5.28651E+02
32101 11 -1.70386E+00 1 u=3211 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
32102 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=3211 u=3210
c
32200 32200 7.19498E-02 -1 u=3221 tmp=8.66798E-08 imp:n=1 \$ Kernel 32 pass:2
vol=5.28651E+02
32201 11 -1.70386E+00 1 u=3221 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
32202 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=3221 u=3220
c
32300 32300 7.27844E-02 -1 u=3231 tmp=8.66798E-08 imp:n=1 \$ Kernel 32 pass:3
vol=5.28651E+02
32301 11 -1.70386E+00 1 u=3231 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
32302 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=3231 u=3230
c
32400 32400 7.34998E-02 -1 u=3241 tmp=8.66798E-08 imp:n=1 \$ Kernel 32 pass:4
vol=5.28651E+02
32401 11 -1.70386E+00 1 u=3241 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
32402 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=3241 u=3240
c
32500 32500 7.41187E-02 -1 u=3251 tmp=8.66798E-08 imp:n=1 \$ Kernel 32 pass:5
vol=5.28651E+02
32501 11 -1.70386E+00 1 u=3251 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
32502 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=3251 u=3250
c
32600 32600 7.42370E-02 -1 u=3261 tmp=8.66798E-08 imp:n=1 \$ Kernel 32 pass:6
vol=5.28651E+02
32601 11 -1.70386E+00 1 u=3261 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
32602 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=3261 u=3260
c
32700 32700 7.43496E-02 -1 u=3271 tmp=8.66798E-08 imp:n=1 \$ Kernel 32 pass:7
vol=5.28651E+02
32701 11 -1.70386E+00 1 u=3271 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
32702 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=3271 u=3270
c
32800 32800 7.44522E-02 -1 u=3281 tmp=8.66798E-08 imp:n=1 \$ Kernel 32 pass:8

vol=5.28651E+02
 32801 11 -1.70386E+00 1 u=3281 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 32802 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=3281 u=3280
 c
 c Depletion Zones 42
 c
 42100 42100 7.10002E-02 -1 u=4211 tmp=8.66798E-08 imp:n=1 \$ Kernel 42 pass:1
 vol=1.58102E+02
 42101 11 -1.70386E+00 1 u=4211 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 42102 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=4211 u=4210
 c
 42200 42200 7.19604E-02 -1 u=4221 tmp=8.66798E-08 imp:n=1 \$ Kernel 42 pass:2
 vol=1.58102E+02
 42201 11 -1.70386E+00 1 u=4221 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 42202 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=4221 u=4220
 c
 42300 42300 7.27935E-02 -1 u=4231 tmp=8.66798E-08 imp:n=1 \$ Kernel 42 pass:3
 vol=1.58102E+02
 42301 11 -1.70386E+00 1 u=4231 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 42302 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=4231 u=4230
 c
 42400 42400 7.35074E-02 -1 u=4241 tmp=8.66798E-08 imp:n=1 \$ Kernel 42 pass:4
 vol=1.58102E+02
 42401 11 -1.70386E+00 1 u=4241 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 42402 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=4241 u=4240
 c
 42500 42500 7.41249E-02 -1 u=4251 tmp=8.66798E-08 imp:n=1 \$ Kernel 42 pass:5
 vol=1.58102E+02
 42501 11 -1.70386E+00 1 u=4251 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 42502 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=4251 u=4250
 c
 42600 42600 7.42436E-02 -1 u=4261 tmp=8.66798E-08 imp:n=1 \$ Kernel 42 pass:6
 vol=1.58102E+02
 42601 11 -1.70386E+00 1 u=4261 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 42602 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=4261 u=4260
 c
 42700 42700 7.43554E-02 -1 u=4271 tmp=8.66798E-08 imp:n=1 \$ Kernel 42 pass:7
 vol=1.58102E+02
 42701 11 -1.70386E+00 1 u=4271 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 42702 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=4271 u=4270
 c
 42800 42800 7.44591E-02 -1 u=4281 tmp=8.66798E-08 imp:n=1 \$ Kernel 42 pass:8
 vol=1.58102E+02
 42801 11 -1.70386E+00 1 u=4281 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix

42802 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=4281 u=4280
c
c Depletion Zones 13
c
13100 13100 7.14302E-02 -1 u=1311 tmp=8.66798E-08 imp:n=1 \$ Kernel 13 pass:1
vol=6.35465E+02
13101 11 -1.70386E+00 1 u=1311 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
13102 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=1311 u=1310
c
13200 13200 7.23154E-02 -1 u=1321 tmp=8.66798E-08 imp:n=1 \$ Kernel 13 pass:2
vol=6.35465E+02
13201 11 -1.70386E+00 1 u=1321 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
13202 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=1321 u=1320
c
13300 13300 7.30898E-02 -1 u=1331 tmp=8.66798E-08 imp:n=1 \$ Kernel 13 pass:3
vol=6.35465E+02
13301 11 -1.70386E+00 1 u=1331 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
13302 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=1331 u=1330
c
13400 13400 7.37563E-02 -1 u=1341 tmp=8.66798E-08 imp:n=1 \$ Kernel 13 pass:4
vol=6.35465E+02
13401 11 -1.70386E+00 1 u=1341 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
13402 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=1341 u=1340
c
13500 13500 7.43507E-02 -1 u=1351 tmp=8.66798E-08 imp:n=1 \$ Kernel 13 pass:5
vol=6.35465E+02
13501 11 -1.70386E+00 1 u=1351 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
13502 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=1351 u=1350
c
13600 13600 7.44534E-02 -1 u=1361 tmp=8.66798E-08 imp:n=1 \$ Kernel 13 pass:6
vol=6.35465E+02
13601 11 -1.70386E+00 1 u=1361 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
13602 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=1361 u=1360
c
13700 13700 7.45541E-02 -1 u=1371 tmp=8.66798E-08 imp:n=1 \$ Kernel 13 pass:7
vol=6.35465E+02
13701 11 -1.70386E+00 1 u=1371 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
13702 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=1371 u=1370
c
13800 13800 7.46420E-02 -1 u=1381 tmp=8.66798E-08 imp:n=1 \$ Kernel 13 pass:8
vol=6.35465E+02
13801 11 -1.70386E+00 1 u=1381 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
13802 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=1381 u=1380
c

c Depletion Zones 23

c

23100 23100 7.15331E-02 -1 u=2311 tmp=8.66798E-08 imp:n=1 \$ Kernel 23 pass:1
vol=9.48491E+02

23101 11 -1.70386E+00 1 u=2311 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
23102 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=2311 u=2310

c

23200 23200 7.24173E-02 -1 u=2321 tmp=8.66798E-08 imp:n=1 \$ Kernel 23 pass:2
vol=9.48491E+02

23201 11 -1.70386E+00 1 u=2321 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
23202 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=2321 u=2320

c

23300 23300 7.31845E-02 -1 u=2331 tmp=8.66798E-08 imp:n=1 \$ Kernel 23 pass:3
vol=9.48491E+02

23301 11 -1.70386E+00 1 u=2331 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
23302 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=2331 u=2330

c

23400 23400 7.38431E-02 -1 u=2341 tmp=8.66798E-08 imp:n=1 \$ Kernel 23 pass:4
vol=9.48491E+02

23401 11 -1.70386E+00 1 u=2341 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
23402 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=2341 u=2340

c

23500 23500 7.41371E-02 -1 u=2351 tmp=8.66798E-08 imp:n=1 \$ Kernel 23 pass:5
vol=9.48491E+02

23501 11 -1.70386E+00 1 u=2351 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
23502 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=2351 u=2350

c

23600 23600 7.42576E-02 -1 u=2361 tmp=8.66798E-08 imp:n=1 \$ Kernel 23 pass:6
vol=9.48491E+02

23601 11 -1.70386E+00 1 u=2361 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
23602 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=2361 u=2360

c

23700 23700 7.43673E-02 -1 u=2371 tmp=8.66798E-08 imp:n=1 \$ Kernel 23 pass:7
vol=9.48491E+02

23701 11 -1.70386E+00 1 u=2371 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
23702 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=2371 u=2370

c

23800 23800 7.44683E-02 -1 u=2381 tmp=8.66798E-08 imp:n=1 \$ Kernel 23 pass:8
vol=9.48491E+02

23801 11 -1.70386E+00 1 u=2381 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
23802 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=2381 u=2380

c

c Depletion Zones 33

c

33100 33100 7.14126E-02 -1 u=3311 tmp=8.66798E-08 imp:n=1 \$ Kernel 33 pass:1
 vol=4.08274E+03
 33101 11 -1.70386E+00 1 u=3311 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 33102 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=3311 u=3310
 c
 33200 33200 7.23211E-02 -1 u=3321 tmp=8.66798E-08 imp:n=1 \$ Kernel 33 pass:2
 vol=4.08274E+03
 33201 11 -1.70386E+00 1 u=3321 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 33202 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=3321 u=3320
 c
 33300 33300 7.31051E-02 -1 u=3331 tmp=8.66798E-08 imp:n=1 \$ Kernel 33 pass:3
 vol=4.08274E+03
 33301 11 -1.70386E+00 1 u=3331 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 33302 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=3331 u=3330
 c
 33400 33400 7.37775E-02 -1 u=3341 tmp=8.66798E-08 imp:n=1 \$ Kernel 33 pass:4
 vol=4.08274E+03
 33401 11 -1.70386E+00 1 u=3341 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 33402 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=3341 u=3340
 c
 33500 33500 7.41395E-02 -1 u=3351 tmp=8.66798E-08 imp:n=1 \$ Kernel 33 pass:5
 vol=4.08274E+03
 33501 11 -1.70386E+00 1 u=3351 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 33502 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=3351 u=3350
 c
 33600 33600 7.42587E-02 -1 u=3361 tmp=8.66798E-08 imp:n=1 \$ Kernel 33 pass:6
 vol=4.08274E+03
 33601 11 -1.70386E+00 1 u=3361 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 33602 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=3361 u=3360
 c
 33700 33700 7.43695E-02 -1 u=3371 tmp=8.66798E-08 imp:n=1 \$ Kernel 33 pass:7
 vol=4.08274E+03
 33701 11 -1.70386E+00 1 u=3371 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 33702 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=3371 u=3370
 c
 33800 33800 7.44690E-02 -1 u=3381 tmp=8.66798E-08 imp:n=1 \$ Kernel 33 pass:8
 vol=4.08274E+03
 33801 11 -1.70386E+00 1 u=3381 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 33802 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=3381 u=3380
 c
 c Depletion Zones 43
 c
 43100 43100 7.14720E-02 -1 u=4311 tmp=8.66798E-08 imp:n=1 \$ Kernel 43 pass:1
 vol=1.25316E+03

43101 11 -1.70386E+00 1 u=4311 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 43102 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=4311 u=4310
 c
 43200 43200 7.23783E-02 -1 u=4321 tmp=8.66798E-08 imp:n=1 \$ Kernel 43 pass:2
 vol=1.25316E+03
 43201 11 -1.70386E+00 1 u=4321 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 43202 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=4321 u=4320
 c
 43300 43300 7.31460E-02 -1 u=4331 tmp=8.66798E-08 imp:n=1 \$ Kernel 43 pass:3
 vol=1.25316E+03
 43301 11 -1.70386E+00 1 u=4331 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 43302 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=4331 u=4330
 c
 43400 43400 7.38072E-02 -1 u=4341 tmp=8.66798E-08 imp:n=1 \$ Kernel 43 pass:4
 vol=1.25316E+03
 43401 11 -1.70386E+00 1 u=4341 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 43402 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=4341 u=4340
 c
 43500 43500 7.41545E-02 -1 u=4351 tmp=8.66798E-08 imp:n=1 \$ Kernel 43 pass:5
 vol=1.25316E+03
 43501 11 -1.70386E+00 1 u=4351 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 43502 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=4351 u=4350
 c
 43600 43600 7.42720E-02 -1 u=4361 tmp=8.66798E-08 imp:n=1 \$ Kernel 43 pass:6
 vol=1.25316E+03
 43601 11 -1.70386E+00 1 u=4361 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 43602 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=4361 u=4360
 c
 43700 43700 7.43809E-02 -1 u=4371 tmp=8.66798E-08 imp:n=1 \$ Kernel 43 pass:7
 vol=1.25316E+03
 43701 11 -1.70386E+00 1 u=4371 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 43702 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=4371 u=4370
 c
 43800 43800 7.44823E-02 -1 u=4381 tmp=8.66798E-08 imp:n=1 \$ Kernel 43 pass:8
 vol=1.25316E+03
 43801 11 -1.70386E+00 1 u=4381 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 43802 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=4381 u=4380
 c
 c Depletion Zones 14
 c
 14100 14100 7.18624E-02 -1 u=1411 tmp=8.66798E-08 imp:n=1 \$ Kernel 14 pass:1
 vol=9.98499E+01
 14101 11 -1.70386E+00 1 u=1411 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 14102 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=1411 u=1410

c
14200 14200 7.26687E-02 -1 u=1421 tmp=8.66798E-08 imp:n=1 \$ Kernel 14 pass:2
vol=9.98499E+01
14201 11 -1.70386E+00 1 u=1421 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
14202 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=1421 u=1420

c
14300 14300 7.33826E-02 -1 u=1431 tmp=8.66798E-08 imp:n=1 \$ Kernel 14 pass:3
vol=9.98499E+01
14301 11 -1.70386E+00 1 u=1431 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
14302 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=1431 u=1430

c
14400 14400 7.40010E-02 -1 u=1441 tmp=8.66798E-08 imp:n=1 \$ Kernel 14 pass:4
vol=9.98499E+01
14401 11 -1.70386E+00 1 u=1441 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
14402 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=1441 u=1440

c
14500 14500 7.45744E-02 -1 u=1451 tmp=8.66798E-08 imp:n=1 \$ Kernel 14 pass:5
vol=9.98499E+01
14501 11 -1.70386E+00 1 u=1451 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
14502 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=1451 u=1450

c
14600 14600 7.46612E-02 -1 u=1461 tmp=8.66798E-08 imp:n=1 \$ Kernel 14 pass:6
vol=9.98499E+01
14601 11 -1.70386E+00 1 u=1461 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
14602 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=1461 u=1460

c
14700 14700 7.47523E-02 -1 u=1471 tmp=8.66798E-08 imp:n=1 \$ Kernel 14 pass:7
vol=9.98499E+01
14701 11 -1.70386E+00 1 u=1471 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
14702 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=1471 u=1470

c
14800 14800 7.48247E-02 -1 u=1481 tmp=8.66798E-08 imp:n=1 \$ Kernel 14 pass:8
vol=9.98499E+01
14801 11 -1.70386E+00 1 u=1481 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
14802 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=1481 u=1480

c
c Depletion Zones 24

c
24100 24100 7.20742E-02 -1 u=2411 tmp=8.66798E-08 imp:n=1 \$ Kernel 24 pass:1
vol=1.31653E+02
24101 11 -1.70386E+00 1 u=2411 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
24102 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=2411 u=2410

c
24200 24200 7.28780E-02 -1 u=2421 tmp=8.66798E-08 imp:n=1 \$ Kernel 24 pass:2

vol=1.31653E+02
 24201 11 -1.70386E+00 1 u=2421 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 24202 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=2421 u=2420
 c
 24300 24300 7.35792E-02 -1 u=2431 tmp=8.66798E-08 imp:n=1 \$ Kernel 24 pass:3
 vol=1.31653E+02
 24301 11 -1.70386E+00 1 u=2431 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 24302 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=2431 u=2430
 c
 24400 24400 7.41819E-02 -1 u=2441 tmp=8.66798E-08 imp:n=1 \$ Kernel 24 pass:4
 vol=1.31653E+02
 24401 11 -1.70386E+00 1 u=2441 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 24402 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=2441 u=2440
 c
 24500 24500 7.41566E-02 -1 u=2451 tmp=8.66798E-08 imp:n=1 \$ Kernel 24 pass:5
 vol=1.31653E+02
 24501 11 -1.70386E+00 1 u=2451 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 24502 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=2451 u=2450
 c
 24600 24600 7.42759E-02 -1 u=2461 tmp=8.66798E-08 imp:n=1 \$ Kernel 24 pass:6
 vol=1.31653E+02
 24601 11 -1.70386E+00 1 u=2461 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 24602 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=2461 u=2460
 c
 24700 24700 7.43829E-02 -1 u=2471 tmp=8.66798E-08 imp:n=1 \$ Kernel 24 pass:7
 vol=1.31653E+02
 24701 11 -1.70386E+00 1 u=2471 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 24702 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=2471 u=2470
 c
 24800 24800 7.44816E-02 -1 u=2481 tmp=8.66798E-08 imp:n=1 \$ Kernel 24 pass:8
 vol=1.31653E+02
 24801 11 -1.70386E+00 1 u=2481 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 24802 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=2481 u=2480
 c
 c Depletion Zones 34
 c
 34100 34100 7.18346E-02 -1 u=3411 tmp=8.66798E-08 imp:n=1 \$ Kernel 34 pass:1
 vol=5.48216E+02
 34101 11 -1.70386E+00 1 u=3411 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 34102 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=3411 u=3410
 c
 34200 34200 7.26899E-02 -1 u=3421 tmp=8.66798E-08 imp:n=1 \$ Kernel 34 pass:2
 vol=5.48216E+02
 34201 11 -1.70386E+00 1 u=3421 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix

34202 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=3421 u=3420
 c
 34300 34300 7.34214E-02 -1 u=3431 tmp=8.66798E-08 imp:n=1 \$ Kernel 34 pass:3
 vol=5.48216E+02
 34301 11 -1.70386E+00 1 u=3431 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 34302 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=3431 u=3430
 c
 34400 34400 7.40540E-02 -1 u=3441 tmp=8.66798E-08 imp:n=1 \$ Kernel 34 pass:4
 vol=5.48216E+02
 34401 11 -1.70386E+00 1 u=3441 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 34402 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=3441 u=3440
 c
 34500 34500 7.41563E-02 -1 u=3451 tmp=8.66798E-08 imp:n=1 \$ Kernel 34 pass:5
 vol=5.48216E+02
 34501 11 -1.70386E+00 1 u=3451 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 34502 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=3451 u=3450
 c
 34600 34600 7.42748E-02 -1 u=3461 tmp=8.66798E-08 imp:n=1 \$ Kernel 34 pass:6
 vol=5.48216E+02
 34601 11 -1.70386E+00 1 u=3461 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 34602 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=3461 u=3460
 c
 34700 34700 7.43830E-02 -1 u=3471 tmp=8.66798E-08 imp:n=1 \$ Kernel 34 pass:7
 vol=5.48216E+02
 34701 11 -1.70386E+00 1 u=3471 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 34702 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=3471 u=3470
 c
 34800 34800 7.44797E-02 -1 u=3481 tmp=8.66798E-08 imp:n=1 \$ Kernel 34 pass:8
 vol=5.48216E+02
 34801 11 -1.70386E+00 1 u=3481 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 34802 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=3481 u=3480
 c
 c Depletion Zones 44
 c
 44100 44100 7.19175E-02 -1 u=4411 tmp=8.66798E-08 imp:n=1 \$ Kernel 44 pass:1
 vol=1.64005E+02
 44101 11 -1.70386E+00 1 u=4411 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 44102 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=4411 u=4410
 c
 44200 44200 7.27696E-02 -1 u=4421 tmp=8.66798E-08 imp:n=1 \$ Kernel 44 pass:2
 vol=1.64005E+02
 44201 11 -1.70386E+00 1 u=4421 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 44202 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=4421 u=4420
 c

44300 44300 7.34733E-02 -1 u=4431 tmp=8.66798E-08 imp:n=1 \$ Kernel 44 pass:3
 vol=1.64005E+02
 44301 11 -1.70386E+00 1 u=4431 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 44302 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=4431 u=4430
 c
 44400 44400 7.40833E-02 -1 u=4441 tmp=8.66798E-08 imp:n=1 \$ Kernel 44 pass:4
 vol=1.64005E+02
 44401 11 -1.70386E+00 1 u=4441 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 44402 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=4441 u=4440
 c
 44500 44500 7.41723E-02 -1 u=4451 tmp=8.66798E-08 imp:n=1 \$ Kernel 44 pass:5
 vol=1.64005E+02
 44501 11 -1.70386E+00 1 u=4451 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 44502 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=4451 u=4450
 c
 44600 44600 7.42885E-02 -1 u=4461 tmp=8.66798E-08 imp:n=1 \$ Kernel 44 pass:6
 vol=1.64005E+02
 44601 11 -1.70386E+00 1 u=4461 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 44602 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=4461 u=4460
 c
 44700 44700 7.43947E-02 -1 u=4471 tmp=8.66798E-08 imp:n=1 \$ Kernel 44 pass:7
 vol=1.64005E+02
 44701 11 -1.70386E+00 1 u=4471 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 44702 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=4471 u=4470
 c
 44800 44800 7.44943E-02 -1 u=4481 tmp=8.66798E-08 imp:n=1 \$ Kernel 44 pass:8
 vol=1.64005E+02
 44801 11 -1.70386E+00 1 u=4481 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 44802 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=4481 u=4480
 c
 c Depletion Zones 15
 c
 15100 15100 7.18943E-02 -1 u=1511 tmp=8.66798E-08 imp:n=1 \$ Kernel 15 pass:1
 vol=3.43321E+01
 15101 11 -1.70386E+00 1 u=1511 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 15102 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=1511 u=1510
 c
 15200 15200 7.26951E-02 -1 u=1521 tmp=8.66798E-08 imp:n=1 \$ Kernel 15 pass:2
 vol=3.43321E+01
 15201 11 -1.70386E+00 1 u=1521 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 15202 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=1521 u=1520
 c
 15300 15300 7.34065E-02 -1 u=1531 tmp=8.66798E-08 imp:n=1 \$ Kernel 15 pass:3
 vol=3.43321E+01

15301 11 -1.70386E+00 1 u=1531 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 15302 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=1531 u=1530
 c
 15400 15400 7.40208E-02 -1 u=1541 tmp=8.66798E-08 imp:n=1 \$ Kernel 15 pass:4
 vol=3.43321E+01
 15401 11 -1.70386E+00 1 u=1541 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 15402 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=1541 u=1540
 c
 15500 15500 7.45916E-02 -1 u=1551 tmp=8.66798E-08 imp:n=1 \$ Kernel 15 pass:5
 vol=3.43321E+01
 15501 11 -1.70386E+00 1 u=1551 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 15502 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=1551 u=1550
 c
 15600 15600 7.46775E-02 -1 u=1561 tmp=8.66798E-08 imp:n=1 \$ Kernel 15 pass:6
 vol=3.43321E+01
 15601 11 -1.70386E+00 1 u=1561 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 15602 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=1561 u=1560
 c
 15700 15700 7.47680E-02 -1 u=1571 tmp=8.66798E-08 imp:n=1 \$ Kernel 15 pass:7
 vol=3.43321E+01
 15701 11 -1.70386E+00 1 u=1571 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 15702 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=1571 u=1570
 c
 15800 15800 7.48390E-02 -1 u=1581 tmp=8.66798E-08 imp:n=1 \$ Kernel 15 pass:8
 vol=3.43321E+01
 15801 11 -1.70386E+00 1 u=1581 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 15802 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=1581 u=1580
 c
 c Depletion Zones 25
 c
 25100 25100 7.20992E-02 -1 u=2511 tmp=8.66798E-08 imp:n=1 \$ Kernel 25 pass:1
 vol=5.21411E+01
 25101 11 -1.70386E+00 1 u=2511 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 25102 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=2511 u=2510
 c
 25200 25200 7.28988E-02 -1 u=2521 tmp=8.66798E-08 imp:n=1 \$ Kernel 25 pass:2
 vol=5.21411E+01
 25201 11 -1.70386E+00 1 u=2521 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 25202 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=2521 u=2520
 c
 25300 25300 7.35970E-02 -1 u=2531 tmp=8.66798E-08 imp:n=1 \$ Kernel 25 pass:3
 vol=5.21411E+01
 25301 11 -1.70386E+00 1 u=2531 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 25302 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=2531 u=2530

c
25400 25400 7.41988E-02 -1 u=2541 tmp=8.66798E-08 imp:n=1 \$ Kernel 25 pass:4
vol=5.21411E+01
25401 11 -1.70386E+00 1 u=2541 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
25402 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=2541 u=2540

c
25500 25500 7.41709E-02 -1 u=2551 tmp=8.66798E-08 imp:n=1 \$ Kernel 25 pass:5
vol=5.21411E+01
25501 11 -1.70386E+00 1 u=2551 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
25502 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=2551 u=2550

c
25600 25600 7.42885E-02 -1 u=2561 tmp=8.66798E-08 imp:n=1 \$ Kernel 25 pass:6
vol=5.21411E+01
25601 11 -1.70386E+00 1 u=2561 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
25602 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=2561 u=2560

c
25700 25700 7.43951E-02 -1 u=2571 tmp=8.66798E-08 imp:n=1 \$ Kernel 25 pass:7
vol=5.21411E+01
25701 11 -1.70386E+00 1 u=2571 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
25702 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=2571 u=2570

c
25800 25800 7.44927E-02 -1 u=2581 tmp=8.66798E-08 imp:n=1 \$ Kernel 25 pass:8
vol=5.21411E+01
25801 11 -1.70386E+00 1 u=2581 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
25802 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=2581 u=2580

c
c Depletion Zones 35

c
35100 35100 7.18539E-02 -1 u=3511 tmp=8.66798E-08 imp:n=1 \$ Kernel 35 pass:1
vol=2.37262E+02
35101 11 -1.70386E+00 1 u=3511 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
35102 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=3511 u=3510

c
35200 35200 7.27060E-02 -1 u=3521 tmp=8.66798E-08 imp:n=1 \$ Kernel 35 pass:2
vol=2.37262E+02
35201 11 -1.70386E+00 1 u=3521 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
35202 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=3521 u=3520

c
35300 35300 7.34370E-02 -1 u=3531 tmp=8.66798E-08 imp:n=1 \$ Kernel 35 pass:3
vol=2.37262E+02
35301 11 -1.70386E+00 1 u=3531 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
35302 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=3531 u=3530

c
35400 35400 7.40675E-02 -1 u=3541 tmp=8.66798E-08 imp:n=1 \$ Kernel 35 pass:4

vol=2.37262E+02
 35401 11 -1.70386E+00 1 u=3541 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 35402 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=3541 u=3540
 c
 35500 35500 7.41670E-02 -1 u=3551 tmp=8.66798E-08 imp:n=1 \$ Kernel 35 pass:5
 vol=2.37262E+02
 35501 11 -1.70386E+00 1 u=3551 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 35502 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=3551 u=3550
 c
 35600 35600 7.42843E-02 -1 u=3561 tmp=8.66798E-08 imp:n=1 \$ Kernel 35 pass:6
 vol=2.37262E+02
 35601 11 -1.70386E+00 1 u=3561 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 35602 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=3561 u=3560
 c
 35700 35700 7.43923E-02 -1 u=3571 tmp=8.66798E-08 imp:n=1 \$ Kernel 35 pass:7
 vol=2.37262E+02
 35701 11 -1.70386E+00 1 u=3571 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 35702 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=3571 u=3570
 c
 35800 35800 7.44883E-02 -1 u=3581 tmp=8.66798E-08 imp:n=1 \$ Kernel 35 pass:8
 vol=2.37262E+02
 35801 11 -1.70386E+00 1 u=3581 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 35802 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=3581 u=3580
 c
 c Depletion Zones 45
 c
 45100 45100 7.19377E-02 -1 u=4511 tmp=8.66798E-08 imp:n=1 \$ Kernel 45 pass:1
 vol=7.38325E+01
 45101 11 -1.70386E+00 1 u=4511 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 45102 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=4511 u=4510
 c
 45200 45200 7.27858E-02 -1 u=4521 tmp=8.66798E-08 imp:n=1 \$ Kernel 45 pass:2
 vol=7.38325E+01
 45201 11 -1.70386E+00 1 u=4521 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 45202 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=4521 u=4520
 c
 45300 45300 7.34882E-02 -1 u=4531 tmp=8.66798E-08 imp:n=1 \$ Kernel 45 pass:3
 vol=7.38325E+01
 45301 11 -1.70386E+00 1 u=4531 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 45302 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=4531 u=4530
 c
 45400 45400 7.40966E-02 -1 u=4541 tmp=8.66798E-08 imp:n=1 \$ Kernel 45 pass:4
 vol=7.38325E+01
 45401 11 -1.70386E+00 1 u=4541 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix

45402 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=4541 u=4540
 c
 45500 45500 7.41839E-02 -1 u=4551 tmp=8.66798E-08 imp:n=1 \$ Kernel 45 pass:5
 vol=7.38325E+01
 45501 11 -1.70386E+00 1 u=4551 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 45502 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=4551 u=4550
 c
 45600 45600 7.42997E-02 -1 u=4561 tmp=8.66798E-08 imp:n=1 \$ Kernel 45 pass:6
 vol=7.38325E+01
 45601 11 -1.70386E+00 1 u=4561 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 45602 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=4561 u=4560
 c
 45700 45700 7.44052E-02 -1 u=4571 tmp=8.66798E-08 imp:n=1 \$ Kernel 45 pass:7
 vol=7.38325E+01
 45701 11 -1.70386E+00 1 u=4571 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 45702 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=4571 u=4570
 c
 45800 45800 7.45038E-02 -1 u=4581 tmp=8.66798E-08 imp:n=1 \$ Kernel 45 pass:8
 vol=7.38325E+01
 45801 11 -1.70386E+00 1 u=4581 tmp=8.40196E-08 imp:n=1 \$ Coatings and Matrix
 45802 0 -11 12 -13 14 -15 16 lat=1 imp:n=1 fill=4581 u=4580
 c
 c -----
 c Pebbles
 c -----
 c
 c Pebble: 11
 c
 1100 3 -1.59368E+00 -1000 u=110 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 1101 0 -1001 1000 fill=1110 u=110 imp:n=1
 1102 4 -1.73841E+00 -1002 1001 u=110 imp:n=1
 tmp=8.25186E-08
 1103 3 -1.59368E+00 -1003 u=110 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 1104 0 -1004 1003 fill=1110 u=110 imp:n=1
 1105 4 -1.73841E+00 -1005 1004 u=110 imp:n=1
 tmp=8.25186E-08
 1106 3 -1.59368E+00 -1006 u=110 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 1107 0 -1007 1006 fill=1110 u=110 imp:n=1
 1108 4 -1.73841E+00 -1008 1007 u=110 imp:n=1
 tmp=8.25186E-08
 1109 3 -1.59368E+00 -1009 u=110 imp:n=1 \$ Depletion Region 1

tmp=8.45293E-08
 1110 0 -1010 1009 fill=1110 u=110 imp:n=1
 1111 4 -1.73841E+00 -1011 1010 u=110 imp:n=1
 tmp=8.25186E-08
 1120 3 -1.59368E+00 -1020 u=110 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 1121 0 -1021 1020 fill=1150 u=110 imp:n=1
 1122 4 -1.73841E+00 -1022 1021 u=110 imp:n=1
 tmp=8.25186E-08
 1123 3 -1.59368E+00 -1023 u=110 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 1124 0 -1024 1023 fill=1150 u=110 imp:n=1
 1125 4 -1.73841E+00 -1025 1024 u=110 imp:n=1
 tmp=8.25186E-08
 1126 3 -1.59368E+00 -1026 u=110 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 1127 0 -1027 1026 fill=1150 u=110 imp:n=1
 1128 4 -1.73841E+00 -1028 1027 u=110 imp:n=1
 tmp=8.25186E-08
 1129 3 -1.59368E+00 -1029 u=110 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 1130 0 -1030 1029 fill=1150 u=110 imp:n=1
 1131 4 -1.73841E+00 -1031 1030 u=110 imp:n=1
 tmp=8.25186E-08
 c
 1140 3 -1.59368E+00 -1040 u=110 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 1141 0 -1041 1040 fill=1120 u=110 imp:n=1
 1142 4 -1.73841E+00 -1042 1041 u=110 imp:n=1
 tmp=8.25186E-08
 1143 3 -1.59368E+00 -1043 u=110 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 1144 0 -1044 1043 fill=1160 u=110 imp:n=1
 1145 4 -1.73841E+00 -1045 1044 u=110 imp:n=1
 tmp=8.25186E-08
 c
 1150 3 -1.59368E+00 -1050 u=110 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 1151 0 -1051 1050 fill=1130 u=110 imp:n=1
 1152 4 -1.73841E+00 -1052 1051 u=110 imp:n=1
 tmp=8.25186E-08
 1153 3 -1.59368E+00 -1053 u=110 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 1154 0 -1054 1053 fill=1170 u=110 imp:n=1

1155 4 -1.73841E+00 -1055 1054 u=110 imp:n=1
 tmp=8.25186E-08
 c
 1160 3 -1.59368E+00 -1060 u=110 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 1161 0 -1061 1060 fill=1140 u=110 imp:n=1
 1162 4 -1.73841E+00 -1062 1061 u=110 imp:n=1
 tmp=8.25186E-08
 1163 3 -1.59368E+00 -1063 u=110 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 1164 0 -1064 1063 fill=1180 u=110 imp:n=1
 1165 4 -1.73841E+00 -1065 1064 u=110 imp:n=1
 tmp=8.25186E-08
 1180 24 -1.96280E+00 1002 1005 1008 1011
 1022 1025 1028 1031
 1042 1045 1052 1055 1062 1065 u=110 imp:n=1 \$ Coolant
 tmp=7.95509E-08
 1181 0 -1080 1081 -1082 1083 -1084 1085 fill=110 imp:n=1 u=11 lat=1 \$ FCC Unit
 Cell
 c
 c Pebble: 21
 c
 2100 3 -1.59368E+00 -1000 u=210 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2101 0 -1001 1000 fill=2110 u=210 imp:n=1
 2102 4 -1.73841E+00 -1002 1001 u=210 imp:n=1
 tmp=8.25186E-08
 2103 3 -1.59368E+00 -1003 u=210 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2104 0 -1004 1003 fill=2110 u=210 imp:n=1
 2105 4 -1.73841E+00 -1005 1004 u=210 imp:n=1
 tmp=8.25186E-08
 2106 3 -1.59368E+00 -1006 u=210 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2107 0 -1007 1006 fill=2110 u=210 imp:n=1
 2108 4 -1.73841E+00 -1008 1007 u=210 imp:n=1
 tmp=8.25186E-08
 2109 3 -1.59368E+00 -1009 u=210 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2110 0 -1010 1009 fill=2110 u=210 imp:n=1
 2111 4 -1.73841E+00 -1011 1010 u=210 imp:n=1
 tmp=8.25186E-08
 2120 3 -1.59368E+00 -1020 u=210 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08

2121 0 -1021 1020 fill=2150 u=210 imp:n=1
 2122 4 -1.73841E+00 -1022 1021 u=210 imp:n=1
 tmp=8.25186E-08
 2123 3 -1.59368E+00 -1023 u=210 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2124 0 -1024 1023 fill=2150 u=210 imp:n=1
 2125 4 -1.73841E+00 -1025 1024 u=210 imp:n=1
 tmp=8.25186E-08
 2126 3 -1.59368E+00 -1026 u=210 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2127 0 -1027 1026 fill=2150 u=210 imp:n=1
 2128 4 -1.73841E+00 -1028 1027 u=210 imp:n=1
 tmp=8.25186E-08
 2129 3 -1.59368E+00 -1029 u=210 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2130 0 -1030 1029 fill=2150 u=210 imp:n=1
 2131 4 -1.73841E+00 -1031 1030 u=210 imp:n=1
 tmp=8.25186E-08
 c
 2140 3 -1.59368E+00 -1040 u=210 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2141 0 -1041 1040 fill=2120 u=210 imp:n=1
 2142 4 -1.73841E+00 -1042 1041 u=210 imp:n=1
 tmp=8.25186E-08
 2143 3 -1.59368E+00 -1043 u=210 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2144 0 -1044 1043 fill=2160 u=210 imp:n=1
 2145 4 -1.73841E+00 -1045 1044 u=210 imp:n=1
 tmp=8.25186E-08
 c
 2150 3 -1.59368E+00 -1050 u=210 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2151 0 -1051 1050 fill=2130 u=210 imp:n=1
 2152 4 -1.73841E+00 -1052 1051 u=210 imp:n=1
 tmp=8.25186E-08
 2153 3 -1.59368E+00 -1053 u=210 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2154 0 -1054 1053 fill=2170 u=210 imp:n=1
 2155 4 -1.73841E+00 -1055 1054 u=210 imp:n=1
 tmp=8.25186E-08
 c
 2160 3 -1.59368E+00 -1060 u=210 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2161 0 -1061 1060 fill=2140 u=210 imp:n=1

2162 4 -1.73841E+00 -1062 1061 u=210 imp:n=1
 tmp=8.25186E-08
 2163 3 -1.59368E+00 -1063 u=210 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2164 0 -1064 1063 fill=2180 u=210 imp:n=1
 2165 4 -1.73841E+00 -1065 1064 u=210 imp:n=1
 tmp=8.25186E-08
 2180 24 -1.96280E+00 1002 1005 1008 1011
 1022 1025 1028 1031
 1042 1045 1052 1055 1062 1065 u=210 imp:n=1 \$ Coolant
 tmp=7.95509E-08
 2181 0 -1080 1081 -1082 1083 -1084 1085 fill=210 imp:n=1 u=21 lat=1 \$ FCC Unit
 Cell
 c
 c Pebble: 31
 c
 3100 3 -1.59368E+00 -1000 u=310 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3101 0 -1001 1000 fill=3110 u=310 imp:n=1
 3102 4 -1.73841E+00 -1002 1001 u=310 imp:n=1
 tmp=8.25186E-08
 3103 3 -1.59368E+00 -1003 u=310 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3104 0 -1004 1003 fill=3110 u=310 imp:n=1
 3105 4 -1.73841E+00 -1005 1004 u=310 imp:n=1
 tmp=8.25186E-08
 3106 3 -1.59368E+00 -1006 u=310 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3107 0 -1007 1006 fill=3110 u=310 imp:n=1
 3108 4 -1.73841E+00 -1008 1007 u=310 imp:n=1
 tmp=8.25186E-08
 3109 3 -1.59368E+00 -1009 u=310 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3110 0 -1010 1009 fill=3110 u=310 imp:n=1
 3111 4 -1.73841E+00 -1011 1010 u=310 imp:n=1
 tmp=8.25186E-08
 3120 3 -1.59368E+00 -1020 u=310 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3121 0 -1021 1020 fill=3150 u=310 imp:n=1
 3122 4 -1.73841E+00 -1022 1021 u=310 imp:n=1
 tmp=8.25186E-08
 3123 3 -1.59368E+00 -1023 u=310 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3124 0 -1024 1023 fill=3150 u=310 imp:n=1

3125 4 -1.73841E+00 -1025 1024 u=310 imp:n=1
 tmp=8.25186E-08
 3126 3 -1.59368E+00 -1026 u=310 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3127 0 -1027 1026 fill=3150 u=310 imp:n=1
 3128 4 -1.73841E+00 -1028 1027 u=310 imp:n=1
 tmp=8.25186E-08
 3129 3 -1.59368E+00 -1029 u=310 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3130 0 -1030 1029 fill=3150 u=310 imp:n=1
 3131 4 -1.73841E+00 -1031 1030 u=310 imp:n=1
 tmp=8.25186E-08
 c
 3140 3 -1.59368E+00 -1040 u=310 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3141 0 -1041 1040 fill=3120 u=310 imp:n=1
 3142 4 -1.73841E+00 -1042 1041 u=310 imp:n=1
 tmp=8.25186E-08
 3143 3 -1.59368E+00 -1043 u=310 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3144 0 -1044 1043 fill=3160 u=310 imp:n=1
 3145 4 -1.73841E+00 -1045 1044 u=310 imp:n=1
 tmp=8.25186E-08
 c
 3150 3 -1.59368E+00 -1050 u=310 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3151 0 -1051 1050 fill=3130 u=310 imp:n=1
 3152 4 -1.73841E+00 -1052 1051 u=310 imp:n=1
 tmp=8.25186E-08
 3153 3 -1.59368E+00 -1053 u=310 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3154 0 -1054 1053 fill=3170 u=310 imp:n=1
 3155 4 -1.73841E+00 -1055 1054 u=310 imp:n=1
 tmp=8.25186E-08
 c
 3160 3 -1.59368E+00 -1060 u=310 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3161 0 -1061 1060 fill=3140 u=310 imp:n=1
 3162 4 -1.73841E+00 -1062 1061 u=310 imp:n=1
 tmp=8.25186E-08
 3163 3 -1.59368E+00 -1063 u=310 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3164 0 -1064 1063 fill=3180 u=310 imp:n=1
 3165 4 -1.73841E+00 -1065 1064 u=310 imp:n=1

tmp=8.25186E-08
 3180 24 -1.96280E+00 1002 1005 1008 1011
 1022 1025 1028 1031
 1042 1045 1052 1055 1062 1065 u=310 imp:n=1 \$ Coolant
 tmp=7.95509E-08
 3181 0 -1080 1081 -1082 1083 -1084 1085 fill=310 imp:n=1 u=31 lat=1 \$ FCC Unit
 Cell
 c
 c Pebble: 41
 c
 4100 3 -1.59368E+00 -1000 u=410 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4101 0 -1001 1000 fill=4110 u=410 imp:n=1
 4102 4 -1.73841E+00 -1002 1001 u=410 imp:n=1
 tmp=8.25186E-08
 4103 3 -1.59368E+00 -1003 u=410 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4104 0 -1004 1003 fill=4110 u=410 imp:n=1
 4105 4 -1.73841E+00 -1005 1004 u=410 imp:n=1
 tmp=8.25186E-08
 4106 3 -1.59368E+00 -1006 u=410 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4107 0 -1007 1006 fill=4110 u=410 imp:n=1
 4108 4 -1.73841E+00 -1008 1007 u=410 imp:n=1
 tmp=8.25186E-08
 4109 3 -1.59368E+00 -1009 u=410 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4110 0 -1010 1009 fill=4110 u=410 imp:n=1
 4111 4 -1.73841E+00 -1011 1010 u=410 imp:n=1
 tmp=8.25186E-08
 4120 3 -1.59368E+00 -1020 u=410 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4121 0 -1021 1020 fill=4150 u=410 imp:n=1
 4122 4 -1.73841E+00 -1022 1021 u=410 imp:n=1
 tmp=8.25186E-08
 4123 3 -1.59368E+00 -1023 u=410 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4124 0 -1024 1023 fill=4150 u=410 imp:n=1
 4125 4 -1.73841E+00 -1025 1024 u=410 imp:n=1
 tmp=8.25186E-08
 4126 3 -1.59368E+00 -1026 u=410 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4127 0 -1027 1026 fill=4150 u=410 imp:n=1
 4128 4 -1.73841E+00 -1028 1027 u=410 imp:n=1

tmp=8.25186E-08
 4129 3 -1.59368E+00 -1029 u=410 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4130 0 -1030 1029 fill=4150 u=410 imp:n=1
 4131 4 -1.73841E+00 -1031 1030 u=410 imp:n=1
 tmp=8.25186E-08
 c
 4140 3 -1.59368E+00 -1040 u=410 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4141 0 -1041 1040 fill=4120 u=410 imp:n=1
 4142 4 -1.73841E+00 -1042 1041 u=410 imp:n=1
 tmp=8.25186E-08
 4143 3 -1.59368E+00 -1043 u=410 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4144 0 -1044 1043 fill=4160 u=410 imp:n=1
 4145 4 -1.73841E+00 -1045 1044 u=410 imp:n=1
 tmp=8.25186E-08
 c
 4150 3 -1.59368E+00 -1050 u=410 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4151 0 -1051 1050 fill=4130 u=410 imp:n=1
 4152 4 -1.73841E+00 -1052 1051 u=410 imp:n=1
 tmp=8.25186E-08
 4153 3 -1.59368E+00 -1053 u=410 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4154 0 -1054 1053 fill=4170 u=410 imp:n=1
 4155 4 -1.73841E+00 -1055 1054 u=410 imp:n=1
 tmp=8.25186E-08
 c
 4160 3 -1.59368E+00 -1060 u=410 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4161 0 -1061 1060 fill=4180 u=410 imp:n=1
 4162 4 -1.73841E+00 -1062 1061 u=410 imp:n=1
 tmp=8.25186E-08
 4163 3 -1.59368E+00 -1063 u=410 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4164 0 -1064 1063 fill=4140 u=410 imp:n=1
 4165 4 -1.73841E+00 -1065 1064 u=410 imp:n=1
 tmp=8.25186E-08
 4180 24 -1.96280E+00 1002 1005 1008 1011
 1022 1025 1028 1031
 1042 1045 1052 1055 1062 1065 u=410 imp:n=1 \$ Coolant
 tmp=7.95509E-08

4181 0 -1080 1081 -1082 1083 -1084 1085 fill=410 imp:n=1 u=41 lat=1 \$ FCC Unit
 Cell
 c
 c Pebble: 12
 c
 1200 3 -1.59368E+00 -1000 u=120 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 1201 0 -1001 1000 fill=1210 u=120 imp:n=1
 1202 4 -1.73841E+00 -1002 1001 u=120 imp:n=1
 tmp=8.25186E-08
 1203 3 -1.59368E+00 -1003 u=120 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 1204 0 -1004 1003 fill=1210 u=120 imp:n=1
 1205 4 -1.73841E+00 -1005 1004 u=120 imp:n=1
 tmp=8.25186E-08
 1206 3 -1.59368E+00 -1006 u=120 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 1207 0 -1007 1006 fill=1210 u=120 imp:n=1
 1208 4 -1.73841E+00 -1008 1007 u=120 imp:n=1
 tmp=8.25186E-08
 1209 3 -1.59368E+00 -1009 u=120 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 1210 0 -1010 1009 fill=1210 u=120 imp:n=1
 1211 4 -1.73841E+00 -1011 1010 u=120 imp:n=1
 tmp=8.25186E-08
 1220 3 -1.59368E+00 -1020 u=120 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 1221 0 -1021 1020 fill=1250 u=120 imp:n=1
 1222 4 -1.73841E+00 -1022 1021 u=120 imp:n=1
 tmp=8.25186E-08
 1223 3 -1.59368E+00 -1023 u=120 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 1224 0 -1024 1023 fill=1250 u=120 imp:n=1
 1225 4 -1.73841E+00 -1025 1024 u=120 imp:n=1
 tmp=8.25186E-08
 1226 3 -1.59368E+00 -1026 u=120 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 1227 0 -1027 1026 fill=1250 u=120 imp:n=1
 1228 4 -1.73841E+00 -1028 1027 u=120 imp:n=1
 tmp=8.25186E-08
 1229 3 -1.59368E+00 -1029 u=120 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 1230 0 -1030 1029 fill=1250 u=120 imp:n=1
 1231 4 -1.73841E+00 -1031 1030 u=120 imp:n=1

tmp=8.25186E-08
 c
 1240 3 -1.59368E+00 -1040 u=120 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 1241 0 -1041 1040 fill=1220 u=120 imp:n=1
 1242 4 -1.73841E+00 -1042 1041 u=120 imp:n=1
 tmp=8.25186E-08
 1243 3 -1.59368E+00 -1043 u=120 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 1244 0 -1044 1043 fill=1260 u=120 imp:n=1
 1245 4 -1.73841E+00 -1045 1044 u=120 imp:n=1
 tmp=8.25186E-08
 c
 1250 3 -1.59368E+00 -1050 u=120 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 1251 0 -1051 1050 fill=1230 u=120 imp:n=1
 1252 4 -1.73841E+00 -1052 1051 u=120 imp:n=1
 tmp=8.25186E-08
 1253 3 -1.59368E+00 -1053 u=120 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 1254 0 -1054 1053 fill=1270 u=120 imp:n=1
 1255 4 -1.73841E+00 -1055 1054 u=120 imp:n=1
 tmp=8.25186E-08
 c
 1260 3 -1.59368E+00 -1060 u=120 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 1261 0 -1061 1060 fill=1240 u=120 imp:n=1
 1262 4 -1.73841E+00 -1062 1061 u=120 imp:n=1
 tmp=8.25186E-08
 1263 3 -1.59368E+00 -1063 u=120 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 1264 0 -1064 1063 fill=1280 u=120 imp:n=1
 1265 4 -1.73841E+00 -1065 1064 u=120 imp:n=1
 tmp=8.25186E-08
 1280 24 -1.96280E+00 1002 1005 1008 1011
 1022 1025 1028 1031
 1042 1045 1052 1055 1062 1065 u=120 imp:n=1 \$ Coolant
 tmp=7.95509E-08
 1281 0 -1080 1081 -1082 1083 -1084 1085 fill=120 imp:n=1 u=12 lat=1 \$ FCC Unit
 Cell
 c
 c Pebble: 22
 c
 2200 3 -1.59368E+00 -1000 u=220 imp:n=1 \$ Depletion Region 1

tmp=8.45293E-08
 2201 0 -1001 1000 fill=2210 u=220 imp:n=1
 2202 4 -1.73841E+00 -1002 1001 u=220 imp:n=1
 tmp=8.25186E-08
 2203 3 -1.59368E+00 -1003 u=220 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2204 0 -1004 1003 fill=2210 u=220 imp:n=1
 2205 4 -1.73841E+00 -1005 1004 u=220 imp:n=1
 tmp=8.25186E-08
 2206 3 -1.59368E+00 -1006 u=220 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2207 0 -1007 1006 fill=2210 u=220 imp:n=1
 2208 4 -1.73841E+00 -1008 1007 u=220 imp:n=1
 tmp=8.25186E-08
 2209 3 -1.59368E+00 -1009 u=220 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2210 0 -1010 1009 fill=2210 u=220 imp:n=1
 2211 4 -1.73841E+00 -1011 1010 u=220 imp:n=1
 tmp=8.25186E-08
 2220 3 -1.59368E+00 -1020 u=220 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2221 0 -1021 1020 fill=2250 u=220 imp:n=1
 2222 4 -1.73841E+00 -1022 1021 u=220 imp:n=1
 tmp=8.25186E-08
 2223 3 -1.59368E+00 -1023 u=220 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2224 0 -1024 1023 fill=2250 u=220 imp:n=1
 2225 4 -1.73841E+00 -1025 1024 u=220 imp:n=1
 tmp=8.25186E-08
 2226 3 -1.59368E+00 -1026 u=220 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2227 0 -1027 1026 fill=2250 u=220 imp:n=1
 2228 4 -1.73841E+00 -1028 1027 u=220 imp:n=1
 tmp=8.25186E-08
 2229 3 -1.59368E+00 -1029 u=220 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2230 0 -1030 1029 fill=2250 u=220 imp:n=1
 2231 4 -1.73841E+00 -1031 1030 u=220 imp:n=1
 tmp=8.25186E-08
 c
 2240 3 -1.59368E+00 -1040 u=220 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2241 0 -1041 1040 fill=2220 u=220 imp:n=1
 2242 4 -1.73841E+00 -1042 1041 u=220 imp:n=1

tmp=8.25186E-08
 2243 3 -1.59368E+00 -1043 u=220 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2244 0 -1044 1043 fill=2260 u=220 imp:n=1
 2245 4 -1.73841E+00 -1045 1044 u=220 imp:n=1
 tmp=8.25186E-08
 c
 2250 3 -1.59368E+00 -1050 u=220 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2251 0 -1051 1050 fill=2230 u=220 imp:n=1
 2252 4 -1.73841E+00 -1052 1051 u=220 imp:n=1
 tmp=8.25186E-08
 2253 3 -1.59368E+00 -1053 u=220 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2254 0 -1054 1053 fill=2270 u=220 imp:n=1
 2255 4 -1.73841E+00 -1055 1054 u=220 imp:n=1
 tmp=8.25186E-08
 c
 2260 3 -1.59368E+00 -1060 u=220 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2261 0 -1061 1060 fill=2240 u=220 imp:n=1
 2262 4 -1.73841E+00 -1062 1061 u=220 imp:n=1
 tmp=8.25186E-08
 2263 3 -1.59368E+00 -1063 u=220 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2264 0 -1064 1063 fill=2280 u=220 imp:n=1
 2265 4 -1.73841E+00 -1065 1064 u=220 imp:n=1
 tmp=8.25186E-08
 2280 24 -1.96280E+00 1002 1005 1008 1011
 1022 1025 1028 1031
 1042 1045 1052 1055 1062 1065 u=220 imp:n=1 \$ Coolant
 tmp=7.95509E-08
 2281 0 -1080 1081 -1082 1083 -1084 1085 fill=220 imp:n=1 u=22 lat=1 \$ FCC Unit
 Cell
 c
 c Pebble: 32
 c
 3200 3 -1.59368E+00 -1000 u=320 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3201 0 -1001 1000 fill=3210 u=320 imp:n=1
 3202 4 -1.73841E+00 -1002 1001 u=320 imp:n=1
 tmp=8.25186E-08
 3203 3 -1.59368E+00 -1003 u=320 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08

3204 0 -1004 1003 fill=3210 u=320 imp:n=1
 3205 4 -1.73841E+00 -1005 1004 u=320 imp:n=1
 tmp=8.25186E-08
 3206 3 -1.59368E+00 -1006 u=320 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3207 0 -1007 1006 fill=3210 u=320 imp:n=1
 3208 4 -1.73841E+00 -1008 1007 u=320 imp:n=1
 tmp=8.25186E-08
 3209 3 -1.59368E+00 -1009 u=320 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3210 0 -1010 1009 fill=3210 u=320 imp:n=1
 3211 4 -1.73841E+00 -1011 1010 u=320 imp:n=1
 tmp=8.25186E-08
 3220 3 -1.59368E+00 -1020 u=320 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3221 0 -1021 1020 fill=3250 u=320 imp:n=1
 3222 4 -1.73841E+00 -1022 1021 u=320 imp:n=1
 tmp=8.25186E-08
 3223 3 -1.59368E+00 -1023 u=320 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3224 0 -1024 1023 fill=3250 u=320 imp:n=1
 3225 4 -1.73841E+00 -1025 1024 u=320 imp:n=1
 tmp=8.25186E-08
 3226 3 -1.59368E+00 -1026 u=320 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3227 0 -1027 1026 fill=3250 u=320 imp:n=1
 3228 4 -1.73841E+00 -1028 1027 u=320 imp:n=1
 tmp=8.25186E-08
 3229 3 -1.59368E+00 -1029 u=320 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3230 0 -1030 1029 fill=3250 u=320 imp:n=1
 3231 4 -1.73841E+00 -1031 1030 u=320 imp:n=1
 tmp=8.25186E-08
 c
 3240 3 -1.59368E+00 -1040 u=320 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3241 0 -1041 1040 fill=3220 u=320 imp:n=1
 3242 4 -1.73841E+00 -1042 1041 u=320 imp:n=1
 tmp=8.25186E-08
 3243 3 -1.59368E+00 -1043 u=320 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3244 0 -1044 1043 fill=3260 u=320 imp:n=1
 3245 4 -1.73841E+00 -1045 1044 u=320 imp:n=1
 tmp=8.25186E-08

c

3250 3 -1.59368E+00 -1050 u=320 imp:n=1 \$ Depletion Region 1
tmp=8.45293E-08

3251 0 -1051 1050 fill=3230 u=320 imp:n=1

3252 4 -1.73841E+00 -1052 1051 u=320 imp:n=1
tmp=8.25186E-08

3253 3 -1.59368E+00 -1053 u=320 imp:n=1 \$ Depletion Region 1
tmp=8.45293E-08

3254 0 -1054 1053 fill=3270 u=320 imp:n=1

3255 4 -1.73841E+00 -1055 1054 u=320 imp:n=1
tmp=8.25186E-08

c

3260 3 -1.59368E+00 -1060 u=320 imp:n=1 \$ Depletion Region 1
tmp=8.45293E-08

3261 0 -1061 1060 fill=3240 u=320 imp:n=1

3262 4 -1.73841E+00 -1062 1061 u=320 imp:n=1
tmp=8.25186E-08

3263 3 -1.59368E+00 -1063 u=320 imp:n=1 \$ Depletion Region 1
tmp=8.45293E-08

3264 0 -1064 1063 fill=3280 u=320 imp:n=1

3265 4 -1.73841E+00 -1065 1064 u=320 imp:n=1
tmp=8.25186E-08

3280 24 -1.96280E+00 1002 1005 1008 1011
1022 1025 1028 1031

1042 1045 1052 1055 1062 1065 u=320 imp:n=1 \$ Coolant
tmp=7.95509E-08

3281 0 -1080 1081 -1082 1083 -1084 1085 fill=320 imp:n=1 u=32 lat=1 \$ FCC Unit
Cell

c

c Pebble: 42

c

4200 3 -1.59368E+00 -1000 u=420 imp:n=1 \$ Depletion Region 1
tmp=8.45293E-08

4201 0 -1001 1000 fill=4210 u=420 imp:n=1

4202 4 -1.73841E+00 -1002 1001 u=420 imp:n=1
tmp=8.25186E-08

4203 3 -1.59368E+00 -1003 u=420 imp:n=1 \$ Depletion Region 1
tmp=8.45293E-08

4204 0 -1004 1003 fill=4210 u=420 imp:n=1

4205 4 -1.73841E+00 -1005 1004 u=420 imp:n=1
tmp=8.25186E-08

4206 3 -1.59368E+00 -1006 u=420 imp:n=1 \$ Depletion Region 1
tmp=8.45293E-08

4207 0 -1007 1006 fill=4210 u=420 imp:n=1

4208 4 -1.73841E+00 -1008 1007 u=420 imp:n=1
 tmp=8.25186E-08
 4209 3 -1.59368E+00 -1009 u=420 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4210 0 -1010 1009 fill=4210 u=420 imp:n=1
 4211 4 -1.73841E+00 -1011 1010 u=420 imp:n=1
 tmp=8.25186E-08
 4220 3 -1.59368E+00 -1020 u=420 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4221 0 -1021 1020 fill=4250 u=420 imp:n=1
 4222 4 -1.73841E+00 -1022 1021 u=420 imp:n=1
 tmp=8.25186E-08
 4223 3 -1.59368E+00 -1023 u=420 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4224 0 -1024 1023 fill=4250 u=420 imp:n=1
 4225 4 -1.73841E+00 -1025 1024 u=420 imp:n=1
 tmp=8.25186E-08
 4226 3 -1.59368E+00 -1026 u=420 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4227 0 -1027 1026 fill=4250 u=420 imp:n=1
 4228 4 -1.73841E+00 -1028 1027 u=420 imp:n=1
 tmp=8.25186E-08
 4229 3 -1.59368E+00 -1029 u=420 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4230 0 -1030 1029 fill=4250 u=420 imp:n=1
 4231 4 -1.73841E+00 -1031 1030 u=420 imp:n=1
 tmp=8.25186E-08
 c
 4240 3 -1.59368E+00 -1040 u=420 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4241 0 -1041 1040 fill=4220 u=420 imp:n=1
 4242 4 -1.73841E+00 -1042 1041 u=420 imp:n=1
 tmp=8.25186E-08
 4243 3 -1.59368E+00 -1043 u=420 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4244 0 -1044 1043 fill=4260 u=420 imp:n=1
 4245 4 -1.73841E+00 -1045 1044 u=420 imp:n=1
 tmp=8.25186E-08
 c
 4250 3 -1.59368E+00 -1050 u=420 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4251 0 -1051 1050 fill=4230 u=420 imp:n=1
 4252 4 -1.73841E+00 -1052 1051 u=420 imp:n=1
 tmp=8.25186E-08

4253 3 -1.59368E+00 -1053 u=420 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4254 0 -1054 1053 fill=4270 u=420 imp:n=1
 4255 4 -1.73841E+00 -1055 1054 u=420 imp:n=1
 tmp=8.25186E-08
 c
 4260 3 -1.59368E+00 -1060 u=420 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4261 0 -1061 1060 fill=4240 u=420 imp:n=1
 4262 4 -1.73841E+00 -1062 1061 u=420 imp:n=1
 tmp=8.25186E-08
 4263 3 -1.59368E+00 -1063 u=420 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4264 0 -1064 1063 fill=4280 u=420 imp:n=1
 4265 4 -1.73841E+00 -1065 1064 u=420 imp:n=1
 tmp=8.25186E-08
 4280 24 -1.96280E+00 1002 1005 1008 1011
 1022 1025 1028 1031
 1042 1045 1052 1055 1062 1065 u=420 imp:n=1 \$ Coolant
 tmp=7.95509E-08
 4281 0 -1080 1081 -1082 1083 -1084 1085 fill=420 imp:n=1 u=42 lat=1 \$ FCC Unit
 Cell
 c
 c Pebble: 13
 c
 1300 3 -1.59368E+00 -1000 u=130 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 1301 0 -1001 1000 fill=1310 u=130 imp:n=1
 1302 4 -1.73841E+00 -1002 1001 u=130 imp:n=1
 tmp=8.25186E-08
 1303 3 -1.59368E+00 -1003 u=130 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 1304 0 -1004 1003 fill=1310 u=130 imp:n=1
 1305 4 -1.73841E+00 -1005 1004 u=130 imp:n=1
 tmp=8.25186E-08
 1306 3 -1.59368E+00 -1006 u=130 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 1307 0 -1007 1006 fill=1310 u=130 imp:n=1
 1308 4 -1.73841E+00 -1008 1007 u=130 imp:n=1
 tmp=8.25186E-08
 1309 3 -1.59368E+00 -1009 u=130 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 1310 0 -1010 1009 fill=1310 u=130 imp:n=1
 1311 4 -1.73841E+00 -1011 1010 u=130 imp:n=1

tmp=8.25186E-08
 1320 3 -1.59368E+00 -1020 u=130 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 1321 0 -1021 1020 fill=1350 u=130 imp:n=1
 1322 4 -1.73841E+00 -1022 1021 u=130 imp:n=1
 tmp=8.25186E-08
 1323 3 -1.59368E+00 -1023 u=130 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 1324 0 -1024 1023 fill=1350 u=130 imp:n=1
 1325 4 -1.73841E+00 -1025 1024 u=130 imp:n=1
 tmp=8.25186E-08
 1326 3 -1.59368E+00 -1026 u=130 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 1327 0 -1027 1026 fill=1350 u=130 imp:n=1
 1328 4 -1.73841E+00 -1028 1027 u=130 imp:n=1
 tmp=8.25186E-08
 1329 3 -1.59368E+00 -1029 u=130 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 1330 0 -1030 1029 fill=1350 u=130 imp:n=1
 1331 4 -1.73841E+00 -1031 1030 u=130 imp:n=1
 tmp=8.25186E-08
 c
 1340 3 -1.59368E+00 -1040 u=130 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 1341 0 -1041 1040 fill=1320 u=130 imp:n=1
 1342 4 -1.73841E+00 -1042 1041 u=130 imp:n=1
 tmp=8.25186E-08
 1343 3 -1.59368E+00 -1043 u=130 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 1344 0 -1044 1043 fill=1360 u=130 imp:n=1
 1345 4 -1.73841E+00 -1045 1044 u=130 imp:n=1
 tmp=8.25186E-08
 c
 1350 3 -1.59368E+00 -1050 u=130 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 1351 0 -1051 1050 fill=1330 u=130 imp:n=1
 1352 4 -1.73841E+00 -1052 1051 u=130 imp:n=1
 tmp=8.25186E-08
 1353 3 -1.59368E+00 -1053 u=130 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 1354 0 -1054 1053 fill=1370 u=130 imp:n=1
 1355 4 -1.73841E+00 -1055 1054 u=130 imp:n=1
 tmp=8.25186E-08

c

1360 3 -1.59368E+00 -1060 u=130 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 1361 0 -1061 1060 fill=1340 u=130 imp:n=1
 1362 4 -1.73841E+00 -1062 1061 u=130 imp:n=1
 tmp=8.25186E-08
 1363 3 -1.59368E+00 -1063 u=130 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 1364 0 -1064 1063 fill=1380 u=130 imp:n=1
 1365 4 -1.73841E+00 -1065 1064 u=130 imp:n=1
 tmp=8.25186E-08
 1380 24 -1.96280E+00 1002 1005 1008 1011
 1022 1025 1028 1031
 1042 1045 1052 1055 1062 1065 u=130 imp:n=1 \$ Coolant
 tmp=7.95509E-08
 1381 0 -1080 1081 -1082 1083 -1084 1085 fill=130 imp:n=1 u=13 lat=1 \$ FCC Unit
 Cell
 c
 c Pebble: 23
 c
 2300 3 -1.59368E+00 -1100 u=230 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2301 0 -1101 1100 fill=2310 u=230 imp:n=1
 2302 4 -1.73841E+00 -1102 1101 u=230 imp:n=1
 tmp=8.25186E-08
 2303 3 -1.59368E+00 -1103 u=230 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2304 0 -1104 1103 fill=2310 u=230 imp:n=1
 2305 4 -1.73841E+00 -1105 1104 u=230 imp:n=1
 tmp=8.25186E-08
 2306 3 -1.59368E+00 -1106 u=230 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2307 0 -1107 1106 fill=2310 u=230 imp:n=1
 2308 4 -1.73841E+00 -1108 1107 u=230 imp:n=1
 tmp=8.25186E-08
 2309 3 -1.59368E+00 -1109 u=230 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2310 0 -1110 1109 fill=2310 u=230 imp:n=1
 2311 4 -1.73841E+00 -1111 1110 u=230 imp:n=1
 tmp=8.25186E-08
 2320 3 -1.59368E+00 -1120 u=230 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2321 0 -1121 1120 fill=2310 u=230 imp:n=1
 2322 4 -1.73841E+00 -1122 1121 u=230 imp:n=1
 tmp=8.25186E-08

2323 3 -1.59368E+00 -1123 u=230 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2324 0 -1124 1123 fill=2310 u=230 imp:n=1
 2325 4 -1.73841E+00 -1125 1124 u=230 imp:n=1
 tmp=8.25186E-08
 2326 3 -1.59368E+00 -1126 u=230 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2327 0 -1127 1126 fill=2310 u=230 imp:n=1
 2328 4 -1.73841E+00 -1128 1127 u=230 imp:n=1
 tmp=8.25186E-08
 2329 3 -1.59368E+00 -1129 u=230 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2330 0 -1130 1129 fill=2310 u=230 imp:n=1
 2331 4 -1.73841E+00 -1131 1130 u=230 imp:n=1
 tmp=8.25186E-08
 c
 2340 3 -1.59368E+00 -1140 u=230 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2341 0 -1141 1140 fill=2320 u=230 imp:n=1
 2342 4 -1.73841E+00 -1142 1141 u=230 imp:n=1
 tmp=8.25186E-08
 2343 3 -1.59368E+00 -1143 u=230 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2344 0 -1144 1143 fill=2320 u=230 imp:n=1
 2345 4 -1.73841E+00 -1145 1144 u=230 imp:n=1
 tmp=8.25186E-08
 c
 2350 3 -1.59368E+00 -1150 u=230 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2351 0 -1151 1150 fill=2330 u=230 imp:n=1
 2352 4 -1.73841E+00 -1152 1151 u=230 imp:n=1
 tmp=8.25186E-08
 2353 3 -1.59368E+00 -1153 u=230 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2354 0 -1154 1153 fill=2330 u=230 imp:n=1
 2355 4 -1.73841E+00 -1155 1154 u=230 imp:n=1
 tmp=8.25186E-08
 c
 2360 3 -1.59368E+00 -1160 u=230 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2361 0 -1161 1160 fill=2340 u=230 imp:n=1
 2362 4 -1.73841E+00 -1162 1161 u=230 imp:n=1
 tmp=8.25186E-08
 2363 3 -1.59368E+00 -1163 u=230 imp:n=1 \$ Depletion Region 1

tmp=8.45293E-08
 2364 0 -1164 1163 fill=2340 u=230 imp:n=1
 2365 4 -1.73841E+00 -1165 1164 u=230 imp:n=1
 tmp=8.25186E-08
 2380 24 -1.96280E+00 1102 1105 1108 1111
 1122 1125 1128 1131
 1142 1145 1152 1155 1162 1165 u=230 imp:n=1 \$ Coolant
 tmp=7.95509E-08
 2381 0 -1180 1181 -1182 1183 -1184 1185 fill=230 imp:n=1 u=23 lat=1 \$ FCC Unit
 Cell
 c
 c Pebble: 33
 c
 3300 3 -1.59368E+00 -1100 u=330 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3301 0 -1101 1100 fill=3310 u=330 imp:n=1
 3302 4 -1.73841E+00 -1102 1101 u=330 imp:n=1
 tmp=8.25186E-08
 3303 3 -1.59368E+00 -1103 u=330 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3304 0 -1104 1103 fill=3310 u=330 imp:n=1
 3305 4 -1.73841E+00 -1105 1104 u=330 imp:n=1
 tmp=8.25186E-08
 3306 3 -1.59368E+00 -1106 u=330 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3307 0 -1107 1106 fill=3310 u=330 imp:n=1
 3308 4 -1.73841E+00 -1108 1107 u=330 imp:n=1
 tmp=8.25186E-08
 3309 3 -1.59368E+00 -1109 u=330 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3310 0 -1110 1109 fill=3310 u=330 imp:n=1
 3311 4 -1.73841E+00 -1111 1110 u=330 imp:n=1
 tmp=8.25186E-08
 3320 3 -1.59368E+00 -1120 u=330 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3321 0 -1121 1120 fill=3310 u=330 imp:n=1
 3322 4 -1.73841E+00 -1122 1121 u=330 imp:n=1
 tmp=8.25186E-08
 3323 3 -1.59368E+00 -1123 u=330 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3324 0 -1124 1123 fill=3310 u=330 imp:n=1
 3325 4 -1.73841E+00 -1125 1124 u=330 imp:n=1
 tmp=8.25186E-08
 3326 3 -1.59368E+00 -1126 u=330 imp:n=1 \$ Depletion Region 1

tmp=8.45293E-08
 3327 0 -1127 1126 fill=3310 u=330 imp:n=1
 3328 4 -1.73841E+00 -1128 1127 u=330 imp:n=1
 tmp=8.25186E-08
 3329 3 -1.59368E+00 -1129 u=330 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3330 0 -1130 1129 fill=3310 u=330 imp:n=1
 3331 4 -1.73841E+00 -1131 1130 u=330 imp:n=1
 tmp=8.25186E-08
 c
 3340 3 -1.59368E+00 -1140 u=330 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3341 0 -1141 1140 fill=3320 u=330 imp:n=1
 3342 4 -1.73841E+00 -1142 1141 u=330 imp:n=1
 tmp=8.25186E-08
 3343 3 -1.59368E+00 -1143 u=330 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3344 0 -1144 1143 fill=3320 u=330 imp:n=1
 3345 4 -1.73841E+00 -1145 1144 u=330 imp:n=1
 tmp=8.25186E-08
 c
 3350 3 -1.59368E+00 -1150 u=330 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3351 0 -1151 1150 fill=3330 u=330 imp:n=1
 3352 4 -1.73841E+00 -1152 1151 u=330 imp:n=1
 tmp=8.25186E-08
 3353 3 -1.59368E+00 -1153 u=330 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3354 0 -1154 1153 fill=3330 u=330 imp:n=1
 3355 4 -1.73841E+00 -1155 1154 u=330 imp:n=1
 tmp=8.25186E-08
 c
 3360 3 -1.59368E+00 -1160 u=330 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3361 0 -1161 1160 fill=3340 u=330 imp:n=1
 3362 4 -1.73841E+00 -1162 1161 u=330 imp:n=1
 tmp=8.25186E-08
 3363 3 -1.59368E+00 -1163 u=330 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3364 0 -1164 1163 fill=3340 u=330 imp:n=1
 3365 4 -1.73841E+00 -1165 1164 u=330 imp:n=1
 tmp=8.25186E-08
 3380 24 -1.96280E+00 1102 1105 1108 1111
 1122 1125 1128 1131

1142 1145 1152 1155 1162 1165 u=330 imp:n=1 \$ Coolant
 tmp=7.95509E-08
 3381 0 -1180 1181 -1182 1183 -1184 1185 fill=330 imp:n=1 u=33 lat=1 \$ FCC Unit
 Cell
 c
 c Pebble: 43
 c
 4300 3 -1.59368E+00 -1100 u=430 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4301 0 -1101 1100 fill=4310 u=430 imp:n=1
 4302 4 -1.73841E+00 -1102 1101 u=430 imp:n=1
 tmp=8.25186E-08
 4303 3 -1.59368E+00 -1103 u=430 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4304 0 -1104 1103 fill=4310 u=430 imp:n=1
 4305 4 -1.73841E+00 -1105 1104 u=430 imp:n=1
 tmp=8.25186E-08
 4306 3 -1.59368E+00 -1106 u=430 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4307 0 -1107 1106 fill=4310 u=430 imp:n=1
 4308 4 -1.73841E+00 -1108 1107 u=430 imp:n=1
 tmp=8.25186E-08
 4309 3 -1.59368E+00 -1109 u=430 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4310 0 -1110 1109 fill=4310 u=430 imp:n=1
 4311 4 -1.73841E+00 -1111 1110 u=430 imp:n=1
 tmp=8.25186E-08
 4320 3 -1.59368E+00 -1120 u=430 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4321 0 -1121 1120 fill=4310 u=430 imp:n=1
 4322 4 -1.73841E+00 -1122 1121 u=430 imp:n=1
 tmp=8.25186E-08
 4323 3 -1.59368E+00 -1123 u=430 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4324 0 -1124 1123 fill=4310 u=430 imp:n=1
 4325 4 -1.73841E+00 -1125 1124 u=430 imp:n=1
 tmp=8.25186E-08
 4326 3 -1.59368E+00 -1126 u=430 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4327 0 -1127 1126 fill=4310 u=430 imp:n=1
 4328 4 -1.73841E+00 -1128 1127 u=430 imp:n=1
 tmp=8.25186E-08
 4329 3 -1.59368E+00 -1129 u=430 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08

4330 0 -1130 1129 fill=4310 u=430 imp:n=1
 4331 4 -1.73841E+00 -1131 1130 u=430 imp:n=1
 tmp=8.25186E-08
 c
 4340 3 -1.59368E+00 -1140 u=430 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4341 0 -1141 1140 fill=4320 u=430 imp:n=1
 4342 4 -1.73841E+00 -1142 1141 u=430 imp:n=1
 tmp=8.25186E-08
 4343 3 -1.59368E+00 -1143 u=430 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4344 0 -1144 1143 fill=4320 u=430 imp:n=1
 4345 4 -1.73841E+00 -1145 1144 u=430 imp:n=1
 tmp=8.25186E-08
 c
 4350 3 -1.59368E+00 -1150 u=430 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4351 0 -1151 1150 fill=4330 u=430 imp:n=1
 4352 4 -1.73841E+00 -1152 1151 u=430 imp:n=1
 tmp=8.25186E-08
 4353 3 -1.59368E+00 -1153 u=430 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4354 0 -1154 1153 fill=4330 u=430 imp:n=1
 4355 4 -1.73841E+00 -1155 1154 u=430 imp:n=1
 tmp=8.25186E-08
 c
 4360 3 -1.59368E+00 -1160 u=430 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4361 0 -1161 1160 fill=4340 u=430 imp:n=1
 4362 4 -1.73841E+00 -1162 1161 u=430 imp:n=1
 tmp=8.25186E-08
 4363 3 -1.59368E+00 -1163 u=430 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4364 0 -1164 1163 fill=4340 u=430 imp:n=1
 4365 4 -1.73841E+00 -1165 1164 u=430 imp:n=1
 tmp=8.25186E-08
 4380 24 -1.96280E+00 1102 1105 1108 1111
 1122 1125 1128 1131
 1142 1145 1152 1155 1162 1165 u=430 imp:n=1 \$ Coolant
 tmp=7.95509E-08
 4381 0 -1180 1181 -1182 1183 -1184 1185 fill=430 imp:n=1 u=43 lat=1 \$ FCC Unit
 Cell
 c
 c Pebble: 14

c

1400 3 -1.59368E+00 -1000 u=140 imp:n=1 \$ Depletion Region 1
tmp=8.45293E-08
1401 0 -1001 1000 fill=1410 u=140 imp:n=1
1402 4 -1.73841E+00 -1002 1001 u=140 imp:n=1
tmp=8.25186E-08
1403 3 -1.59368E+00 -1003 u=140 imp:n=1 \$ Depletion Region 1
tmp=8.45293E-08
1404 0 -1004 1003 fill=1410 u=140 imp:n=1
1405 4 -1.73841E+00 -1005 1004 u=140 imp:n=1
tmp=8.25186E-08
1406 3 -1.59368E+00 -1006 u=140 imp:n=1 \$ Depletion Region 1
tmp=8.45293E-08
1407 0 -1007 1006 fill=1410 u=140 imp:n=1
1408 4 -1.73841E+00 -1008 1007 u=140 imp:n=1
tmp=8.25186E-08
1409 3 -1.59368E+00 -1009 u=140 imp:n=1 \$ Depletion Region 1
tmp=8.45293E-08
1410 0 -1010 1009 fill=1410 u=140 imp:n=1
1411 4 -1.73841E+00 -1011 1010 u=140 imp:n=1
tmp=8.25186E-08
1420 3 -1.59368E+00 -1020 u=140 imp:n=1 \$ Depletion Region 1
tmp=8.45293E-08
1421 0 -1021 1020 fill=1450 u=140 imp:n=1
1422 4 -1.73841E+00 -1022 1021 u=140 imp:n=1
tmp=8.25186E-08
1423 3 -1.59368E+00 -1023 u=140 imp:n=1 \$ Depletion Region 1
tmp=8.45293E-08
1424 0 -1024 1023 fill=1450 u=140 imp:n=1
1425 4 -1.73841E+00 -1025 1024 u=140 imp:n=1
tmp=8.25186E-08
1426 3 -1.59368E+00 -1026 u=140 imp:n=1 \$ Depletion Region 1
tmp=8.45293E-08
1427 0 -1027 1026 fill=1450 u=140 imp:n=1
1428 4 -1.73841E+00 -1028 1027 u=140 imp:n=1
tmp=8.25186E-08
1429 3 -1.59368E+00 -1029 u=140 imp:n=1 \$ Depletion Region 1
tmp=8.45293E-08
1430 0 -1030 1029 fill=1450 u=140 imp:n=1
1431 4 -1.73841E+00 -1031 1030 u=140 imp:n=1
tmp=8.25186E-08

c

1440 3 -1.59368E+00 -1040 u=140 imp:n=1 \$ Depletion Region 1
tmp=8.45293E-08

1441 0 -1041 1040 fill=1420 u=140 imp:n=1
1442 4 -1.73841E+00 -1042 1041 u=140 imp:n=1
tmp=8.25186E-08
1443 3 -1.59368E+00 -1043 u=140 imp:n=1 \$ Depletion Region 1
tmp=8.45293E-08
1444 0 -1044 1043 fill=1460 u=140 imp:n=1
1445 4 -1.73841E+00 -1045 1044 u=140 imp:n=1
tmp=8.25186E-08
c
1450 3 -1.59368E+00 -1050 u=140 imp:n=1 \$ Depletion Region 1
tmp=8.45293E-08
1451 0 -1051 1050 fill=1430 u=140 imp:n=1
1452 4 -1.73841E+00 -1052 1051 u=140 imp:n=1
tmp=8.25186E-08
1453 3 -1.59368E+00 -1053 u=140 imp:n=1 \$ Depletion Region 1
tmp=8.45293E-08
1454 0 -1054 1053 fill=1470 u=140 imp:n=1
1455 4 -1.73841E+00 -1055 1054 u=140 imp:n=1
tmp=8.25186E-08
c
1460 3 -1.59368E+00 -1060 u=140 imp:n=1 \$ Depletion Region 1
tmp=8.45293E-08
1461 0 -1061 1060 fill=1440 u=140 imp:n=1
1462 4 -1.73841E+00 -1062 1061 u=140 imp:n=1
tmp=8.25186E-08
1463 3 -1.59368E+00 -1063 u=140 imp:n=1 \$ Depletion Region 1
tmp=8.45293E-08
1464 0 -1064 1063 fill=1480 u=140 imp:n=1
1465 4 -1.73841E+00 -1065 1064 u=140 imp:n=1
tmp=8.25186E-08
1480 24 -1.96280E+00 1002 1005 1008 1011
1022 1025 1028 1031
1042 1045 1052 1055 1062 1065 u=140 imp:n=1 \$ Coolant
tmp=7.95509E-08
1481 0 -1080 1081 -1082 1083 -1084 1085 fill=140 imp:n=1 u=14 lat=1 \$ FCC Unit
Cell
c
c Pebble: 24
c
2400 3 -1.59368E+00 -1000 u=240 imp:n=1 \$ Depletion Region 1
tmp=8.45293E-08
2401 0 -1001 1000 fill=2410 u=240 imp:n=1
2402 4 -1.73841E+00 -1002 1001 u=240 imp:n=1
tmp=8.25186E-08

2403 3 -1.59368E+00 -1003 u=240 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2404 0 -1004 1003 fill=2410 u=240 imp:n=1
 2405 4 -1.73841E+00 -1005 1004 u=240 imp:n=1
 tmp=8.25186E-08
 2406 3 -1.59368E+00 -1006 u=240 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2407 0 -1007 1006 fill=2410 u=240 imp:n=1
 2408 4 -1.73841E+00 -1008 1007 u=240 imp:n=1
 tmp=8.25186E-08
 2409 3 -1.59368E+00 -1009 u=240 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2410 0 -1010 1009 fill=2410 u=240 imp:n=1
 2411 4 -1.73841E+00 -1011 1010 u=240 imp:n=1
 tmp=8.25186E-08
 2420 3 -1.59368E+00 -1020 u=240 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2421 0 -1021 1020 fill=2450 u=240 imp:n=1
 2422 4 -1.73841E+00 -1022 1021 u=240 imp:n=1
 tmp=8.25186E-08
 2423 3 -1.59368E+00 -1023 u=240 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2424 0 -1024 1023 fill=2450 u=240 imp:n=1
 2425 4 -1.73841E+00 -1025 1024 u=240 imp:n=1
 tmp=8.25186E-08
 2426 3 -1.59368E+00 -1026 u=240 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2427 0 -1027 1026 fill=2450 u=240 imp:n=1
 2428 4 -1.73841E+00 -1028 1027 u=240 imp:n=1
 tmp=8.25186E-08
 2429 3 -1.59368E+00 -1029 u=240 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2430 0 -1030 1029 fill=2450 u=240 imp:n=1
 2431 4 -1.73841E+00 -1031 1030 u=240 imp:n=1
 tmp=8.25186E-08
 c
 2440 3 -1.59368E+00 -1040 u=240 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2441 0 -1041 1040 fill=2420 u=240 imp:n=1
 2442 4 -1.73841E+00 -1042 1041 u=240 imp:n=1
 tmp=8.25186E-08
 2443 3 -1.59368E+00 -1043 u=240 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2444 0 -1044 1043 fill=2460 u=240 imp:n=1

2445 4 -1.73841E+00 -1045 1044 u=240 imp:n=1
 tmp=8.25186E-08
 c
 2450 3 -1.59368E+00 -1050 u=240 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2451 0 -1051 1050 fill=2430 u=240 imp:n=1
 2452 4 -1.73841E+00 -1052 1051 u=240 imp:n=1
 tmp=8.25186E-08
 2453 3 -1.59368E+00 -1053 u=240 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2454 0 -1054 1053 fill=2470 u=240 imp:n=1
 2455 4 -1.73841E+00 -1055 1054 u=240 imp:n=1
 tmp=8.25186E-08
 c
 2460 3 -1.59368E+00 -1060 u=240 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2461 0 -1061 1060 fill=2440 u=240 imp:n=1
 2462 4 -1.73841E+00 -1062 1061 u=240 imp:n=1
 tmp=8.25186E-08
 2463 3 -1.59368E+00 -1063 u=240 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2464 0 -1064 1063 fill=2480 u=240 imp:n=1
 2465 4 -1.73841E+00 -1065 1064 u=240 imp:n=1
 tmp=8.25186E-08
 2480 24 -1.96280E+00 1002 1005 1008 1011
 1022 1025 1028 1031
 1042 1045 1052 1055 1062 1065 u=240 imp:n=1 \$ Coolant
 tmp=7.95509E-08
 2481 0 -1080 1081 -1082 1083 -1084 1085 fill=240 imp:n=1 u=24 lat=1 \$ FCC Unit
 Cell
 c
 c Pebble: 34
 c
 3400 3 -1.59368E+00 -1000 u=340 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3401 0 -1001 1000 fill=3410 u=340 imp:n=1
 3402 4 -1.73841E+00 -1002 1001 u=340 imp:n=1
 tmp=8.25186E-08
 3403 3 -1.59368E+00 -1003 u=340 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3404 0 -1004 1003 fill=3410 u=340 imp:n=1
 3405 4 -1.73841E+00 -1005 1004 u=340 imp:n=1
 tmp=8.25186E-08
 3406 3 -1.59368E+00 -1006 u=340 imp:n=1 \$ Depletion Region 1

tmp=8.45293E-08
 3407 0 -1007 1006 fill=3410 u=340 imp:n=1
 3408 4 -1.73841E+00 -1008 1007 u=340 imp:n=1
 tmp=8.25186E-08
 3409 3 -1.59368E+00 -1009 u=340 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3410 0 -1010 1009 fill=3410 u=340 imp:n=1
 3411 4 -1.73841E+00 -1011 1010 u=340 imp:n=1
 tmp=8.25186E-08
 3420 3 -1.59368E+00 -1020 u=340 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3421 0 -1021 1020 fill=3450 u=340 imp:n=1
 3422 4 -1.73841E+00 -1022 1021 u=340 imp:n=1
 tmp=8.25186E-08
 3423 3 -1.59368E+00 -1023 u=340 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3424 0 -1024 1023 fill=3450 u=340 imp:n=1
 3425 4 -1.73841E+00 -1025 1024 u=340 imp:n=1
 tmp=8.25186E-08
 3426 3 -1.59368E+00 -1026 u=340 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3427 0 -1027 1026 fill=3450 u=340 imp:n=1
 3428 4 -1.73841E+00 -1028 1027 u=340 imp:n=1
 tmp=8.25186E-08
 3429 3 -1.59368E+00 -1029 u=340 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3430 0 -1030 1029 fill=3450 u=340 imp:n=1
 3431 4 -1.73841E+00 -1031 1030 u=340 imp:n=1
 tmp=8.25186E-08
 c
 3440 3 -1.59368E+00 -1040 u=340 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3441 0 -1041 1040 fill=3420 u=340 imp:n=1
 3442 4 -1.73841E+00 -1042 1041 u=340 imp:n=1
 tmp=8.25186E-08
 3443 3 -1.59368E+00 -1043 u=340 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3444 0 -1044 1043 fill=3460 u=340 imp:n=1
 3445 4 -1.73841E+00 -1045 1044 u=340 imp:n=1
 tmp=8.25186E-08
 c
 3450 3 -1.59368E+00 -1050 u=340 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3451 0 -1051 1050 fill=3430 u=340 imp:n=1

3452 4 -1.73841E+00 -1052 1051 u=340 imp:n=1
 tmp=8.25186E-08
 3453 3 -1.59368E+00 -1053 u=340 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3454 0 -1054 1053 fill=3470 u=340 imp:n=1
 3455 4 -1.73841E+00 -1055 1054 u=340 imp:n=1
 tmp=8.25186E-08
 c
 3460 3 -1.59368E+00 -1060 u=340 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3461 0 -1061 1060 fill=3440 u=340 imp:n=1
 3462 4 -1.73841E+00 -1062 1061 u=340 imp:n=1
 tmp=8.25186E-08
 3463 3 -1.59368E+00 -1063 u=340 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3464 0 -1064 1063 fill=3480 u=340 imp:n=1
 3465 4 -1.73841E+00 -1065 1064 u=340 imp:n=1
 tmp=8.25186E-08
 3480 24 -1.96280E+00 1002 1005 1008 1011
 1022 1025 1028 1031
 1042 1045 1052 1055 1062 1065 u=340 imp:n=1 \$ Coolant
 tmp=7.95509E-08
 3481 0 -1080 1081 -1082 1083 -1084 1085 fill=340 imp:n=1 u=34 lat=1 \$ FCC Unit
 Cell
 c
 c Pebble: 44
 c
 4400 3 -1.59368E+00 -1000 u=440 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4401 0 -1001 1000 fill=4410 u=440 imp:n=1
 4402 4 -1.73841E+00 -1002 1001 u=440 imp:n=1
 tmp=8.25186E-08
 4403 3 -1.59368E+00 -1003 u=440 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4404 0 -1004 1003 fill=4410 u=440 imp:n=1
 4405 4 -1.73841E+00 -1005 1004 u=440 imp:n=1
 tmp=8.25186E-08
 4406 3 -1.59368E+00 -1006 u=440 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4407 0 -1007 1006 fill=4410 u=440 imp:n=1
 4408 4 -1.73841E+00 -1008 1007 u=440 imp:n=1
 tmp=8.25186E-08
 4409 3 -1.59368E+00 -1009 u=440 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08

4410 0 -1010 1009 fill=4410 u=440 imp:n=1
 4411 4 -1.73841E+00 -1011 1010 u=440 imp:n=1
 tmp=8.25186E-08
 4420 3 -1.59368E+00 -1020 u=440 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4421 0 -1021 1020 fill=4450 u=440 imp:n=1
 4422 4 -1.73841E+00 -1022 1021 u=440 imp:n=1
 tmp=8.25186E-08
 4423 3 -1.59368E+00 -1023 u=440 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4424 0 -1024 1023 fill=4450 u=440 imp:n=1
 4425 4 -1.73841E+00 -1025 1024 u=440 imp:n=1
 tmp=8.25186E-08
 4426 3 -1.59368E+00 -1026 u=440 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4427 0 -1027 1026 fill=4450 u=440 imp:n=1
 4428 4 -1.73841E+00 -1028 1027 u=440 imp:n=1
 tmp=8.25186E-08
 4429 3 -1.59368E+00 -1029 u=440 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4430 0 -1030 1029 fill=4450 u=440 imp:n=1
 4431 4 -1.73841E+00 -1031 1030 u=440 imp:n=1
 tmp=8.25186E-08
 c
 4440 3 -1.59368E+00 -1040 u=440 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4441 0 -1041 1040 fill=4420 u=440 imp:n=1
 4442 4 -1.73841E+00 -1042 1041 u=440 imp:n=1
 tmp=8.25186E-08
 4443 3 -1.59368E+00 -1043 u=440 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4444 0 -1044 1043 fill=4460 u=440 imp:n=1
 4445 4 -1.73841E+00 -1045 1044 u=440 imp:n=1
 tmp=8.25186E-08
 c
 4450 3 -1.59368E+00 -1050 u=440 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4451 0 -1051 1050 fill=4430 u=440 imp:n=1
 4452 4 -1.73841E+00 -1052 1051 u=440 imp:n=1
 tmp=8.25186E-08
 4453 3 -1.59368E+00 -1053 u=440 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4454 0 -1054 1053 fill=4470 u=440 imp:n=1
 4455 4 -1.73841E+00 -1055 1054 u=440 imp:n=1

tmp=8.25186E-08
 c
 4460 3 -1.59368E+00 -1060 u=440 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4461 0 -1061 1060 fill=4440 u=440 imp:n=1
 4462 4 -1.73841E+00 -1062 1061 u=440 imp:n=1
 tmp=8.25186E-08
 4463 3 -1.59368E+00 -1063 u=440 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4464 0 -1064 1063 fill=4480 u=440 imp:n=1
 4465 4 -1.73841E+00 -1065 1064 u=440 imp:n=1
 tmp=8.25186E-08
 4480 24 -1.96280E+00 1002 1005 1008 1011
 1022 1025 1028 1031
 1042 1045 1052 1055 1062 1065 u=440 imp:n=1 \$ Coolant
 tmp=7.95509E-08
 4481 0 -1080 1081 -1082 1083 -1084 1085 fill=440 imp:n=1 u=44 lat=1 \$ FCC Unit
 Cell
 c
 c Pebble: 15
 c
 1500 3 -1.59368E+00 -1000 u=150 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 1501 0 -1001 1000 fill=1510 u=150 imp:n=1
 1502 4 -1.73841E+00 -1002 1001 u=150 imp:n=1
 tmp=8.25186E-08
 1503 3 -1.59368E+00 -1003 u=150 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 1504 0 -1004 1003 fill=1510 u=150 imp:n=1
 1505 4 -1.73841E+00 -1005 1004 u=150 imp:n=1
 tmp=8.25186E-08
 1506 3 -1.59368E+00 -1006 u=150 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 1507 0 -1007 1006 fill=1510 u=150 imp:n=1
 1508 4 -1.73841E+00 -1008 1007 u=150 imp:n=1
 tmp=8.25186E-08
 1509 3 -1.59368E+00 -1009 u=150 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 1510 0 -1010 1009 fill=1510 u=150 imp:n=1
 1511 4 -1.73841E+00 -1011 1010 u=150 imp:n=1
 tmp=8.25186E-08
 1520 3 -1.59368E+00 -1020 u=150 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 1521 0 -1021 1020 fill=1550 u=150 imp:n=1

1522 4 -1.73841E+00 -1022 1021 u=150 imp:n=1
 tmp=8.25186E-08
 1523 3 -1.59368E+00 -1023 u=150 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 1524 0 -1024 1023 fill=1550 u=150 imp:n=1
 1525 4 -1.73841E+00 -1025 1024 u=150 imp:n=1
 tmp=8.25186E-08
 1526 3 -1.59368E+00 -1026 u=150 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 1527 0 -1027 1026 fill=1550 u=150 imp:n=1
 1528 4 -1.73841E+00 -1028 1027 u=150 imp:n=1
 tmp=8.25186E-08
 1529 3 -1.59368E+00 -1029 u=150 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 1530 0 -1030 1029 fill=1550 u=150 imp:n=1
 1531 4 -1.73841E+00 -1031 1030 u=150 imp:n=1
 tmp=8.25186E-08
 c
 1540 3 -1.59368E+00 -1040 u=150 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 1541 0 -1041 1040 fill=1520 u=150 imp:n=1
 1542 4 -1.73841E+00 -1042 1041 u=150 imp:n=1
 tmp=8.25186E-08
 1543 3 -1.59368E+00 -1043 u=150 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 1544 0 -1044 1043 fill=1560 u=150 imp:n=1
 1545 4 -1.73841E+00 -1045 1044 u=150 imp:n=1
 tmp=8.25186E-08
 c
 1550 3 -1.59368E+00 -1050 u=150 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 1551 0 -1051 1050 fill=1530 u=150 imp:n=1
 1552 4 -1.73841E+00 -1052 1051 u=150 imp:n=1
 tmp=8.25186E-08
 1553 3 -1.59368E+00 -1053 u=150 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 1554 0 -1054 1053 fill=1570 u=150 imp:n=1
 1555 4 -1.73841E+00 -1055 1054 u=150 imp:n=1
 tmp=8.25186E-08
 c
 1560 3 -1.59368E+00 -1060 u=150 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 1561 0 -1061 1060 fill=1540 u=150 imp:n=1
 1562 4 -1.73841E+00 -1062 1061 u=150 imp:n=1

tmp=8.25186E-08
 1563 3 -1.59368E+00 -1063 u=150 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 1564 0 -1064 1063 fill=1580 u=150 imp:n=1
 1565 4 -1.73841E+00 -1065 1064 u=150 imp:n=1
 tmp=8.25186E-08
 1580 24 -1.96280E+00 1002 1005 1008 1011
 1022 1025 1028 1031
 1042 1045 1052 1055 1062 1065 u=150 imp:n=1 \$ Coolant
 tmp=7.95509E-08
 1581 0 -1080 1081 -1082 1083 -1084 1085 fill=150 imp:n=1 u=15 lat=1 \$ FCC Unit
 Cell
 c
 c Pebble: 25
 c
 2500 3 -1.59368E+00 -1000 u=250 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2501 0 -1001 1000 fill=2510 u=250 imp:n=1
 2502 4 -1.73841E+00 -1002 1001 u=250 imp:n=1
 tmp=8.25186E-08
 2503 3 -1.59368E+00 -1003 u=250 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2504 0 -1004 1003 fill=2510 u=250 imp:n=1
 2505 4 -1.73841E+00 -1005 1004 u=250 imp:n=1
 tmp=8.25186E-08
 2506 3 -1.59368E+00 -1006 u=250 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2507 0 -1007 1006 fill=2510 u=250 imp:n=1
 2508 4 -1.73841E+00 -1008 1007 u=250 imp:n=1
 tmp=8.25186E-08
 2509 3 -1.59368E+00 -1009 u=250 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2510 0 -1010 1009 fill=2510 u=250 imp:n=1
 2511 4 -1.73841E+00 -1011 1010 u=250 imp:n=1
 tmp=8.25186E-08
 2520 3 -1.59368E+00 -1020 u=250 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2521 0 -1021 1020 fill=2550 u=250 imp:n=1
 2522 4 -1.73841E+00 -1022 1021 u=250 imp:n=1
 tmp=8.25186E-08
 2523 3 -1.59368E+00 -1023 u=250 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2524 0 -1024 1023 fill=2550 u=250 imp:n=1
 2525 4 -1.73841E+00 -1025 1024 u=250 imp:n=1

tmp=8.25186E-08
 2526 3 -1.59368E+00 -1026 u=250 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2527 0 -1027 1026 fill=2550 u=250 imp:n=1
 2528 4 -1.73841E+00 -1028 1027 u=250 imp:n=1
 tmp=8.25186E-08
 2529 3 -1.59368E+00 -1029 u=250 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2530 0 -1030 1029 fill=2550 u=250 imp:n=1
 2531 4 -1.73841E+00 -1031 1030 u=250 imp:n=1
 tmp=8.25186E-08
 c
 2540 3 -1.59368E+00 -1040 u=250 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2541 0 -1041 1040 fill=2520 u=250 imp:n=1
 2542 4 -1.73841E+00 -1042 1041 u=250 imp:n=1
 tmp=8.25186E-08
 2543 3 -1.59368E+00 -1043 u=250 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2544 0 -1044 1043 fill=2560 u=250 imp:n=1
 2545 4 -1.73841E+00 -1045 1044 u=250 imp:n=1
 tmp=8.25186E-08
 c
 2550 3 -1.59368E+00 -1050 u=250 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2551 0 -1051 1050 fill=2530 u=250 imp:n=1
 2552 4 -1.73841E+00 -1052 1051 u=250 imp:n=1
 tmp=8.25186E-08
 2553 3 -1.59368E+00 -1053 u=250 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2554 0 -1054 1053 fill=2570 u=250 imp:n=1
 2555 4 -1.73841E+00 -1055 1054 u=250 imp:n=1
 tmp=8.25186E-08
 c
 2560 3 -1.59368E+00 -1060 u=250 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2561 0 -1061 1060 fill=2540 u=250 imp:n=1
 2562 4 -1.73841E+00 -1062 1061 u=250 imp:n=1
 tmp=8.25186E-08
 2563 3 -1.59368E+00 -1063 u=250 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 2564 0 -1064 1063 fill=2580 u=250 imp:n=1
 2565 4 -1.73841E+00 -1065 1064 u=250 imp:n=1
 tmp=8.25186E-08

2580 24 -1.96280E+00 1002 1005 1008 1011
 1022 1025 1028 1031
 1042 1045 1052 1055 1062 1065 u=250 imp:n=1 \$ Coolant
 tmp=7.95509E-08
 2581 0 -1080 1081 -1082 1083 -1084 1085 fill=250 imp:n=1 u=25 lat=1 \$ FCC Unit
 Cell
 c
 c Pebble: 35
 c
 3500 3 -1.59368E+00 -1000 u=350 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3501 0 -1001 1000 fill=3510 u=350 imp:n=1
 3502 4 -1.73841E+00 -1002 1001 u=350 imp:n=1
 tmp=8.25186E-08
 3503 3 -1.59368E+00 -1003 u=350 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3504 0 -1004 1003 fill=3510 u=350 imp:n=1
 3505 4 -1.73841E+00 -1005 1004 u=350 imp:n=1
 tmp=8.25186E-08
 3506 3 -1.59368E+00 -1006 u=350 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3507 0 -1007 1006 fill=3510 u=350 imp:n=1
 3508 4 -1.73841E+00 -1008 1007 u=350 imp:n=1
 tmp=8.25186E-08
 3509 3 -1.59368E+00 -1009 u=350 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3510 0 -1010 1009 fill=3510 u=350 imp:n=1
 3511 4 -1.73841E+00 -1011 1010 u=350 imp:n=1
 tmp=8.25186E-08
 3520 3 -1.59368E+00 -1020 u=350 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3521 0 -1021 1020 fill=3550 u=350 imp:n=1
 3522 4 -1.73841E+00 -1022 1021 u=350 imp:n=1
 tmp=8.25186E-08
 3523 3 -1.59368E+00 -1023 u=350 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3524 0 -1024 1023 fill=3550 u=350 imp:n=1
 3525 4 -1.73841E+00 -1025 1024 u=350 imp:n=1
 tmp=8.25186E-08
 3526 3 -1.59368E+00 -1026 u=350 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3527 0 -1027 1026 fill=3550 u=350 imp:n=1
 3528 4 -1.73841E+00 -1028 1027 u=350 imp:n=1
 tmp=8.25186E-08

3529 3 -1.59368E+00 -1029 u=350 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3530 0 -1030 1029 fill=3550 u=350 imp:n=1
 3531 4 -1.73841E+00 -1031 1030 u=350 imp:n=1
 tmp=8.25186E-08
 c
 3540 3 -1.59368E+00 -1040 u=350 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3541 0 -1041 1040 fill=3520 u=350 imp:n=1
 3542 4 -1.73841E+00 -1042 1041 u=350 imp:n=1
 tmp=8.25186E-08
 3543 3 -1.59368E+00 -1043 u=350 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3544 0 -1044 1043 fill=3560 u=350 imp:n=1
 3545 4 -1.73841E+00 -1045 1044 u=350 imp:n=1
 tmp=8.25186E-08
 c
 3550 3 -1.59368E+00 -1050 u=350 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3551 0 -1051 1050 fill=3530 u=350 imp:n=1
 3552 4 -1.73841E+00 -1052 1051 u=350 imp:n=1
 tmp=8.25186E-08
 3553 3 -1.59368E+00 -1053 u=350 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3554 0 -1054 1053 fill=3570 u=350 imp:n=1
 3555 4 -1.73841E+00 -1055 1054 u=350 imp:n=1
 tmp=8.25186E-08
 c
 3560 3 -1.59368E+00 -1060 u=350 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3561 0 -1061 1060 fill=3540 u=350 imp:n=1
 3562 4 -1.73841E+00 -1062 1061 u=350 imp:n=1
 tmp=8.25186E-08
 3563 3 -1.59368E+00 -1063 u=350 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 3564 0 -1064 1063 fill=3580 u=350 imp:n=1
 3565 4 -1.73841E+00 -1065 1064 u=350 imp:n=1
 tmp=8.25186E-08
 3580 24 -1.96280E+00 1002 1005 1008 1011
 1022 1025 1028 1031
 1042 1045 1052 1055 1062 1065 u=350 imp:n=1 \$ Coolant
 tmp=7.95509E-08
 3581 0 -1080 1081 -1082 1083 -1084 1085 fill=350 imp:n=1 u=35 lat=1 \$ FCC Unit
 Cell

c

c Pebble: 45

c

4500 3 -1.59368E+00 -1000 u=450 imp:n=1 \$ Depletion Region 1
tmp=8.45293E-08
4501 0 -1001 1000 fill=4510 u=450 imp:n=1
4502 4 -1.73841E+00 -1002 1001 u=450 imp:n=1
tmp=8.25186E-08
4503 3 -1.59368E+00 -1003 u=450 imp:n=1 \$ Depletion Region 1
tmp=8.45293E-08
4504 0 -1004 1003 fill=4510 u=450 imp:n=1
4505 4 -1.73841E+00 -1005 1004 u=450 imp:n=1
tmp=8.25186E-08
4506 3 -1.59368E+00 -1006 u=450 imp:n=1 \$ Depletion Region 1
tmp=8.45293E-08
4507 0 -1007 1006 fill=4510 u=450 imp:n=1
4508 4 -1.73841E+00 -1008 1007 u=450 imp:n=1
tmp=8.25186E-08
4509 3 -1.59368E+00 -1009 u=450 imp:n=1 \$ Depletion Region 1
tmp=8.45293E-08
4510 0 -1010 1009 fill=4510 u=450 imp:n=1
4511 4 -1.73841E+00 -1011 1010 u=450 imp:n=1
tmp=8.25186E-08
4520 3 -1.59368E+00 -1020 u=450 imp:n=1 \$ Depletion Region 1
tmp=8.45293E-08
4521 0 -1021 1020 fill=4550 u=450 imp:n=1
4522 4 -1.73841E+00 -1022 1021 u=450 imp:n=1
tmp=8.25186E-08
4523 3 -1.59368E+00 -1023 u=450 imp:n=1 \$ Depletion Region 1
tmp=8.45293E-08
4524 0 -1024 1023 fill=4550 u=450 imp:n=1
4525 4 -1.73841E+00 -1025 1024 u=450 imp:n=1
tmp=8.25186E-08
4526 3 -1.59368E+00 -1026 u=450 imp:n=1 \$ Depletion Region 1
tmp=8.45293E-08
4527 0 -1027 1026 fill=4550 u=450 imp:n=1
4528 4 -1.73841E+00 -1028 1027 u=450 imp:n=1
tmp=8.25186E-08
4529 3 -1.59368E+00 -1029 u=450 imp:n=1 \$ Depletion Region 1
tmp=8.45293E-08
4530 0 -1030 1029 fill=4550 u=450 imp:n=1
4531 4 -1.73841E+00 -1031 1030 u=450 imp:n=1
tmp=8.25186E-08

c

4540 3 -1.59368E+00 -1040 u=450 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4541 0 -1041 1040 fill=4520 u=450 imp:n=1
 4542 4 -1.73841E+00 -1042 1041 u=450 imp:n=1
 tmp=8.25186E-08
 4543 3 -1.59368E+00 -1043 u=450 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4544 0 -1044 1043 fill=4560 u=450 imp:n=1
 4545 4 -1.73841E+00 -1045 1044 u=450 imp:n=1
 tmp=8.25186E-08
 c
 4550 3 -1.59368E+00 -1050 u=450 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4551 0 -1051 1050 fill=4530 u=450 imp:n=1
 4552 4 -1.73841E+00 -1052 1051 u=450 imp:n=1
 tmp=8.25186E-08
 4553 3 -1.59368E+00 -1053 u=450 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4554 0 -1054 1053 fill=4570 u=450 imp:n=1
 4555 4 -1.73841E+00 -1055 1054 u=450 imp:n=1
 tmp=8.25186E-08
 c
 4560 3 -1.59368E+00 -1060 u=450 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4561 0 -1061 1060 fill=4540 u=450 imp:n=1
 4562 4 -1.73841E+00 -1062 1061 u=450 imp:n=1
 tmp=8.25186E-08
 4563 3 -1.59368E+00 -1063 u=450 imp:n=1 \$ Depletion Region 1
 tmp=8.45293E-08
 4564 0 -1064 1063 fill=4580 u=450 imp:n=1
 4565 4 -1.73841E+00 -1065 1064 u=450 imp:n=1
 tmp=8.25186E-08
 4580 24 -1.96280E+00 1002 1005 1008 1011
 1022 1025 1028 1031
 1042 1045 1052 1055 1062 1065 u=450 imp:n=1 \$ Coolant
 tmp=7.95509E-08
 4581 0 -1080 1081 -1082 1083 -1084 1085 fill=450 imp:n=1 u=45 lat=1 \$ FCC Unit
 Cell
 c
 c Pebble: 50
 c
 5002 2 -1.73841E+00 -1002 u=500 imp:n=1
 tmp=8.38596E-08
 5005 2 -1.73841E+00 -1005 u=500 imp:n=1

tmp=8.38596E-08
 5008 2 -1.73841E+00 -1008 u=500 imp:n=1
 tmp=8.38596E-08
 5011 2 -1.73841E+00 -1011 u=500 imp:n=1
 tmp=8.38596E-08
 5022 2 -1.73841E+00 -1022 u=500 imp:n=1
 tmp=8.38596E-08
 5025 2 -1.73841E+00 -1025 u=500 imp:n=1
 tmp=8.38596E-08
 5028 2 -1.73841E+00 -1028 u=500 imp:n=1
 tmp=8.38596E-08
 5031 2 -1.73841E+00 -1031 u=500 imp:n=1
 tmp=8.38596E-08
 c
 5042 2 -1.73841E+00 -1042 u=500 imp:n=1
 tmp=8.38596E-08
 5045 2 -1.73841E+00 -1045 u=500 imp:n=1
 tmp=8.38596E-08
 c
 5052 2 -1.73841E+00 -1052 u=500 imp:n=1
 tmp=8.38596E-08
 5055 2 -1.73841E+00 -1055 u=500 imp:n=1
 tmp=8.38596E-08
 c
 5062 2 -1.73841E+00 -1062 u=500 imp:n=1
 tmp=8.38596E-08
 5065 2 -1.73841E+00 -1065 u=500 imp:n=1
 tmp=8.38596E-08
 5080 25 -1.93840E+00 1002 1005 1008 1011
 1022 1025 1028 1031
 1042 1045 1052 1055 1062 1065 u=500 imp:n=1 \$ Coolant
 tmp=8.38596E-08
 5081 0 -1080 1081 -1082 1083 -1084 1085 fill=500 imp:n=1 u=50 lat=1 \$ FCC Unit
 Cell
 c
 c Pebble: 51
 c
 5102 2 -1.73841E+00 -1102 u=510 imp:n=1
 tmp=8.38596E-08
 5105 2 -1.73841E+00 -1105 u=510 imp:n=1
 tmp=8.38596E-08
 5108 2 -1.73841E+00 -1108 u=510 imp:n=1
 tmp=8.38596E-08
 5111 2 -1.73841E+00 -1111 u=510 imp:n=1

```

    tmp=8.38596E-08
5122 2 -1.73841E+00 -1122      u=510 imp:n=1
    tmp=8.38596E-08
5125 2 -1.73841E+00 -1125      u=510 imp:n=1
    tmp=8.38596E-08
5128 2 -1.73841E+00 -1128      u=510 imp:n=1
    tmp=8.38596E-08
5131 2 -1.73841E+00 -1131      u=510 imp:n=1
    tmp=8.38596E-08
c
5142 2 -1.73841E+00 -1142      u=510 imp:n=1
    tmp=8.38596E-08
5145 2 -1.73841E+00 -1145      u=510 imp:n=1
    tmp=8.38596E-08
c
5152 2 -1.73841E+00 -1152      u=510 imp:n=1
    tmp=8.38596E-08
5155 2 -1.73841E+00 -1155      u=510 imp:n=1
    tmp=8.38596E-08
c
5162 2 -1.73841E+00 -1162      u=510 imp:n=1
    tmp=8.38596E-08
5165 2 -1.73841E+00 -1165      u=510 imp:n=1
    tmp=8.38596E-08
5180 25 -1.93840E+00 1102 1105 1108 1111
    1122 1125 1128 1131
    1142 1145 1152 1155 1162 1165 u=510 imp:n=1 $ Coolant
    tmp=8.38596E-08
5181 0      -1180 1181 -1182 1183 -1184 1185 fill=510 imp:n=1 u=51 lat=1 $ FCC Unit
Cell
c
c Shield Unit Cell
c
6000 60 -2.52000E+00 -6000      imp:n=1 u=61 tmp=7.52422E-08 $ shield absorber
6001 61 -8.00000E+00 -6001 6000 imp:n=1 u=61 tmp=7.52422E-08 $ shield clad
6002 1 -1.74000E+00      6001 imp:n=1 u=61 tmp=7.52422E-08 $ shield block
c
6003 0      -6010 6011 -6012 6013 -6014 6015 imp:n=1 lat=2 fill=61 u=60 $ infinite
array of shield absorber pins
c
c Full Core
c
c control rod cells
c

```

8000 23 -1.98720E+00 -8000 -600 302 imp:n=1 tmp=7.52422E-08
8001 50 -2.40000E+00
(-8001 -8002 8003 8004 -602 600):(-8005 -8006 8007 8008 -602 600)
imp:n=1 tmp=7.52422E-08
8002 23 -1.98720E+00 -8000 #8001 -602 600 imp:n=1 tmp=7.52422E-08
8010 23 -1.98720E+00 -8010 -600 302 imp:n=1 tmp=7.52422E-08
8011 50 -2.40000E+00
(-8011 -8012 8013 8014 -602 600):(-8015 -8016 8017 8018 -602 600)
imp:n=1 tmp=7.52422E-08
8012 23 -1.98720E+00 -8010 #8011 -602 600 imp:n=1 tmp=7.52422E-08
8020 23 -1.98720E+00 -8020 -600 302 imp:n=1 tmp=7.52422E-08
8021 50 -2.40000E+00
(-8021 -8022 8023 8024 -602 600):(-8025 -8026 8027 8028 -602 600)
imp:n=1 tmp=7.52422E-08
8022 23 -1.98720E+00 -8020 #8021 -602 600 imp:n=1 tmp=7.52422E-08
8030 23 -1.98720E+00 -8030 -600 302 imp:n=1 tmp=7.52422E-08
8031 50 -2.40000E+00
(-8031 -8032 8033 8034 -602 600):(-8035 -8036 8037 8038 -602 600)
imp:n=1 tmp=7.52422E-08
8032 23 -1.98720E+00 -8030 #8031 -602 600 imp:n=1 tmp=7.52422E-08
8040 23 -1.98720E+00 -8040 -600 302 imp:n=1 tmp=7.52422E-08
8041 50 -2.40000E+00
(-8041 -8042 8043 8044 -602 600):(-8045 -8046 8047 8048 -602 600)
imp:n=1 tmp=7.52422E-08
8042 23 -1.98720E+00 -8040 #8041 -602 600 imp:n=1 tmp=7.52422E-08
8050 23 -1.98720E+00 -8050 -600 302 imp:n=1 tmp=7.52422E-08
8051 50 -2.40000E+00
(-8051 -8052 8053 8054 -602 600):(-8055 -8056 8057 8058 -602 600)
imp:n=1 tmp=7.52422E-08
8052 23 -1.98720E+00 -8050 #8051 -602 600 imp:n=1 tmp=7.52422E-08
8060 23 -1.98720E+00 -8060 -600 302 imp:n=1 tmp=7.52422E-08
8061 50 -2.40000E+00
(-8061 -8062 8063 8064 -602 600):(-8065 -8066 8067 8068 -602 600)
imp:n=1 tmp=7.52422E-08
8062 23 -1.98720E+00 -8060 #8061 -602 600 imp:n=1 tmp=7.52422E-08
8070 23 -1.98720E+00 -8070 -600 302 imp:n=1 tmp=7.52422E-08
8071 50 -2.40000E+00
(-8071 -8072 8073 8074 -602 600):(-8075 -8076 8077 8078 -602 600)
imp:n=1 tmp=7.52422E-08
8072 23 -1.98720E+00 -8070 #8071 -602 600 imp:n=1 tmp=7.52422E-08
c
c control blade cells
c
9000 25 -1.93840E+00

(9001 9002 -9003 -9004 -602 -501):(9005 9006 -9007 -9008 -602 -501)
 imp:n=1 tmp=8.38596E-08
 9010 25 -1.93840E+00
 (9011 9012 -9013 -9014 -602 -501):(9015 9016 -9017 -9018 -602 -501)
 imp:n=1 tmp=8.38596E-08
 9020 25 -1.93840E+00
 (-9021 9022 9023 -9024 -602 -501):(-9025 9026 9027 -9028 -602 -501)
 imp:n=1 tmp=8.38596E-08
 9030 25 -1.93840E+00
 (-9031 9032 9033 -9034 -602 -501):(-9035 9036 9037 -9038 -602 -501)
 imp:n=1 tmp=8.38596E-08
 9040 25 -1.93840E+00
 (-9041 -9042 9043 9044 -602 -501):(-9045 -9046 9047 9048 -602 -501)
 imp:n=1 tmp=8.38596E-08
 9050 25 -1.93840E+00
 (-9051 -9052 9053 9054 -602 -501):(-9055 -9056 9057 9058 -602 -501)
 imp:n=1 tmp=8.38596E-08
 9060 25 -1.93840E+00
 (9061 -9062 -9063 9064 -602 -501):(9065 -9066 -9067 9068 -602 -501)
 imp:n=1 tmp=8.38596E-08
 9070 25 -1.93840E+00
 (9071 -9072 -9073 9074 -602 -501):(9075 -9076 -9077 9078 -602 -501)
 imp:n=1 tmp=8.38596E-08
 903 6 -1.81936E+00 -905 413 -602 404 imp:n=1 \$ outer porous reflector
 tmp=8.38596E-08
 904 2 -1.74000E+00 (-905 413 -404 204) imp:n=1 \$ outer solid reflector
 tmp=8.38596E-08
 905 1 -1.74000E+00 (-907 905 -602 204) imp:n=1 \$ outer solid reflector
 tmp=7.52422E-08
 906 0 -602 204 907 -909 imp:n=1 fill=60 \$ shield
 907 31 -8 -602 204 909 -911 imp:n=1 \$ core barrel
 tmp=7.52422E-08
 908 23 -1.98720E+00 -602 204 911 -913 imp:n=1 \$ down comer
 tmp=7.52422E-08
 909 31 -8 -602 204 913 -915 imp:n=1 \$ reactor vessel
 tmp=7.52422E-08
 c -----
 c lower region
 c
 100 31 -8 -204 -100 102 imp:n=1 \$ plenum reactor vessel
 tmp=7.52422E-08
 101 23 -1.98720E+00 -204 -102 104 imp:n=1 \$ down comer in plenum
 tmp=7.52422E-08
 102 31 -8 -204 -104 106 imp:n=1 \$ core barrel

tmp=7.52422E-08
 103 1 -1.74000E+00 -204 -106 211 imp:n=1 \$ outer solid reflector
 tmp=7.52422E-08
 104 23 -1.98720E+00 -204 -106 -211 201 imp:n=1 \$ lower coolant plenum
 tmp=7.52422E-08
 105 1 -1.74000E+00 -204 -106 -901 imp:n=1 \$ solid graphite
 tmp=7.52422E-08
 106 5 -1.81936E+00 -204 -106 -201 901 imp:n=1 \$ porous graphite
 tmp=7.52422E-08

c -----

c pebble entrance

c

200 1 -1.74000E+00 -300 204 -901 400 302
 8000 8010 8020 8030 8040 8050 8060 8070
 tmp=7.52422E-08
 imp:n=1 \$ solid graphite
 201 5 -1.81936E+00 -300 204 -201 901 400 302
 8000 8010 8020 8030 8040 8050 8060 8070
 tmp=7.52422E-08
 imp:n=1 \$ porous graphite
 202 1 -1.74000E+00 -300 204 -901 400 -302
 tmp=7.52422E-08
 imp:n=1 \$ solid graphite
 203 5 -1.81936E+00 -300 204 -201 901 400 -302
 tmp=7.52422E-08
 imp:n=1 \$ porous graphite
 204 11100 -10.5 -300 204 -203 201 imp:n=1 fill=11 \$ depletion zone 11
 205 21100 -10.5 -300 204 -205 203 imp:n=1 fill=21 \$ depletion zone 21
 206 31100 -10.5 -300 204 -207 205 imp:n=1 fill=31 \$ depletion zone 31
 207 41100 -10.5 -300 204 -209 207 imp:n=1 fill=41 \$ depletion zone 41
 208 5 -1.74 -300 204 -211 209 imp:n=1 fill=50 \$ pebble reflector
 209 2 -1.74000E+00 -300 204 -413 211 imp:n=1 \$ solid graphite
 tmp=8.38596E-08

c -----

c expansion region

c

300 1 -1.74000E+00 400 300 -901
 8000 8010 8020 8030 8040 8050 8060 8070
 tmp=7.52422E-08
 imp:n=1 \$ solid graphite
 c
 301 5 -1.81936E+00 400 300 -301 901
 8000 8010 8020 8030 8040 8050 8060 8070
 tmp=7.52422E-08

imp:n=1 \$ porous graphite
 302 12100 -10.5 400 300 -303 301 imp:n=1 fill=12 \$ depletion zone 12
 303 22100 -10.5 400 300 -305 303 imp:n=1 fill=22 \$ depletion zone 22
 304 32100 -10.5 400 300 -307 305 imp:n=1 fill=32 \$ depletion zone 32
 305 42100 -10.5 400 300 -309 307 imp:n=1 fill=42 \$ depletion zone 42
 306 5 -1.74 400 300 -311 309 imp:n=1 fill=50 \$ pebble reflector
 307 2 -1.74000E+00 400 300 -413 311 imp:n=1 \$ solid graphite reflector
 tmp=8.38596E-08

c -----

c active region

c

400 1 -1.74000E+00 -500 -400 -901
 8000 8010 8020 8030 8040 8050 8060 8070
 tmp=7.52422E-08
 imp:n=1 \$ solid graphite

c

401 5 -1.81936E+00 -500 -400 -401 901
 8000 8010 8020 8030 8040 8050 8060 8070
 tmp=7.52422E-08
 imp:n=1 \$ porous graphite

402 13100 -10.5 -500 -400 -403 401

c CB engaged #9000
 c CB engaged #9010
 c CB engaged #9020
 c CB engaged #9030
 c CB engaged #9040
 c CB engaged #9050
 c CB engaged #9060
 c CB engaged #9070

imp:n=1 fill=13 \$ depletion zone 13

403 23100 -10.5 -500 -400 -405 403

c CB engaged #9000
 c CB engaged #9010
 c CB engaged #9020
 c CB engaged #9030
 c CB engaged #9040
 c CB engaged #9050
 c CB engaged #9060
 c CB engaged #9070

imp:n=1 fill=23 \$ depletion zone 23

404 33100 -10.5 -500 -400 -407 405

c CB engaged #9000
 c CB engaged #9010
 c CB engaged #9020

```

c CB engaged      #9030
c CB engaged      #9040
c CB engaged      #9050
c CB engaged      #9060
c CB engaged      #9070
                    imp:n=1 fill=33 $ depletion zone 33
405 43100 -10.5 -500 -400 -409 407
c CB engaged      #9000
c CB engaged      #9010
c CB engaged      #9020
c CB engaged      #9030
c CB engaged      #9040
c CB engaged      #9050
c CB engaged      #9060
c CB engaged      #9070
                    imp:n=1 fill=43 $ depletion zone 43
406  6 -1.74 -500 -400 -411 409
c CB engaged      #9000
c CB engaged      #9010
c CB engaged      #9020
c CB engaged      #9030
c CB engaged      #9040
c CB engaged      #9050
c CB engaged      #9060
c CB engaged      #9070
                    imp:n=1 fill=51 $ pebble reflector (bulk)
407  5 -1.74 -500 -400 -413 411
c CB engaged      #9000
c CB engaged      #9010
c CB engaged      #9020
c CB engaged      #9030
c CB engaged      #9040
c CB engaged      #9050
c CB engaged      #9060
c CB engaged      #9070
                    imp:n=1 fill=50 $ pebble reflector (wall)
c -----
c converging region
c
500 1 -1.74000E+00 -600 500 -501
    8000 8010 8020 8030 8040 8050 8060 8070
    #9000
    #9010
    #9020

```


#9030
 #9040
 #9050
 #9060
 #9070
 tmp=7.52422E-08
 imp:n=1 \$ solid graphite
 502 14100 -10.5 -600 500 -503 501
 c CB engaged #9000
 c CB engaged #9010
 c CB engaged #9020
 c CB engaged #9030
 c CB engaged #9040
 c CB engaged #9050
 c CB engaged #9060
 c CB engaged #9070
 imp:n=1 fill=14 \$ depletion zone 14
 503 24100 -10.5 -600 500 -505 503
 c CB engaged #9000
 c CB engaged #9010
 c CB engaged #9020
 c CB engaged #9030
 c CB engaged #9040
 c CB engaged #9050
 c CB engaged #9060
 c CB engaged #9070
 imp:n=1 fill=24 \$ depletion zone 24
 504 34100 -10.5 -600 500 -507 505
 c CB engaged #9000
 c CB engaged #9010
 c CB engaged #9020
 c CB engaged #9030
 c CB engaged #9040
 c CB engaged #9050
 c CB engaged #9060
 c CB engaged #9070
 imp:n=1 fill=34 \$ depletion zone 34
 505 44100 -10.5 -600 500 -509 507
 c CB engaged #9000
 c CB engaged #9010
 c CB engaged #9020
 c CB engaged #9030
 c CB engaged #9040
 c CB engaged #9050

```

c CB engaged      #9060
c CB engaged      #9070
                    imp:n=1 fill=44 $ depletion zone 44
506  5 -1.74 -600 500 -511 509
c CB engaged      #9000
c CB engaged      #9010
c CB engaged      #9020
c CB engaged      #9030
c CB engaged      #9040
c CB engaged      #9050
c CB engaged      #9060
c CB engaged      #9070
                    imp:n=1 fill=50 $ pebble reflector
510 6 -1.81936E+00 -413 511 -600 500
                    imp:n=1 tmp=8.38596E-08
c -----
c defueling chute
c
600 1 -1.74000E+00 -602 600 -601
    8000 8010 8020 8030 8040 8050 8060 8070
    #9000
    #9010
    #9020
    #9030
    #9040
    #9050
    #9060
    #9070
    imp:n=1 $ solid graphite
602 15100 -10.5 -602 600 -603 601 imp:n=1 fill=15 $ depletion zone 15
603 25100 -10.5 -602 600 -605 603 imp:n=1 fill=25 $ depletion zone 25
604 35100 -10.5 -602 600 -607 605 imp:n=1 fill=35 $ depletion zone 35
605 45100 -10.5 -602 600 -609 607 imp:n=1 fill=45 $ depletion zone 45
606  5 -1.74 -602 600 -611 609 imp:n=1 fill=50 $ pebble reflector
610 6 -1.81936E+00 -413 611 -602 600
                    imp:n=1 tmp=8.38596E-08
c -----
c boundary conditions
c
9997 0 204 -602 915 imp:n=0
9998 0 -204 100 imp:n=0
9999 0 602 imp:n=0
c 9999 0 (602):(204 915 -602):(-204 100) imp:n=0 $ vacuum boundary conditions

```

```

c -----
c Surfaces
c -----
c
c TRISO surfaces
c
1 so 2.00000E-02 $ Kernel
c
c Seed Fuel Particles Lattice (PF: 40.0)
c
11 px 4.43031E-02
12 px -4.43031E-02
13 py 4.43031E-02
14 py -4.43031E-02
15 pz 4.43031E-02
16 pz -4.43031E-02
c
c Pebble surfaces (PF: 57.0)
c
1000 s 2.31465E+00 2.31465E+00 2.31465E+00 1.25114E+00
1001 s 2.31465E+00 2.31465E+00 2.31465E+00 1.40000E+00
1002 s 2.31465E+00 2.31465E+00 2.31465E+00 1.50000E+00
c
1003 s -2.31465E+00 2.31465E+00 2.31465E+00 1.25114E+00
1004 s -2.31465E+00 2.31465E+00 2.31465E+00 1.40000E+00
1005 s -2.31465E+00 2.31465E+00 2.31465E+00 1.50000E+00
c
1006 s -2.31465E+00 -2.31465E+00 2.31465E+00 1.25114E+00
1007 s -2.31465E+00 -2.31465E+00 2.31465E+00 1.40000E+00
1008 s -2.31465E+00 -2.31465E+00 2.31465E+00 1.50000E+00
c
1009 s 2.31465E+00 -2.31465E+00 2.31465E+00 1.25114E+00
1010 s 2.31465E+00 -2.31465E+00 2.31465E+00 1.40000E+00
1011 s 2.31465E+00 -2.31465E+00 2.31465E+00 1.50000E+00
c
1020 s 2.31465E+00 2.31465E+00 -2.31465E+00 1.25114E+00
1021 s 2.31465E+00 2.31465E+00 -2.31465E+00 1.40000E+00
1022 s 2.31465E+00 2.31465E+00 -2.31465E+00 1.50000E+00
c
1023 s -2.31465E+00 2.31465E+00 -2.31465E+00 1.25114E+00
1024 s -2.31465E+00 2.31465E+00 -2.31465E+00 1.40000E+00
1025 s -2.31465E+00 2.31465E+00 -2.31465E+00 1.50000E+00
c
1026 s -2.31465E+00 -2.31465E+00 -2.31465E+00 1.25114E+00

```

1027 s -2.31465E+00 -2.31465E+00 -2.31465E+00 1.40000E+00
 1028 s -2.31465E+00 -2.31465E+00 -2.31465E+00 1.50000E+00
 c
 1029 s 2.31465E+00 -2.31465E+00 -2.31465E+00 1.25114E+00
 1030 s 2.31465E+00 -2.31465E+00 -2.31465E+00 1.40000E+00
 1031 s 2.31465E+00 -2.31465E+00 -2.31465E+00 1.50000E+00
 c
 1040 s 2.31465E+00 0 0 1.25114E+00
 1041 s 2.31465E+00 0 0 1.40000E+00
 1042 s 2.31465E+00 0 0 1.50000E+00
 c
 1043 s -2.31465E+00 0 0 1.25114E+00
 1044 s -2.31465E+00 0 0 1.40000E+00
 1045 s -2.31465E+00 0 0 1.50000E+00
 c
 1050 s 0 2.31465E+00 0 1.25114E+00
 1051 s 0 2.31465E+00 0 1.40000E+00
 1052 s 0 2.31465E+00 0 1.50000E+00
 c
 1053 s 0 -2.31465E+00 0 1.25114E+00
 1054 s 0 -2.31465E+00 0 1.40000E+00
 1055 s 0 -2.31465E+00 0 1.50000E+00
 c
 1060 s 0 0 2.31465E+00 1.25114E+00
 1061 s 0 0 2.31465E+00 1.40000E+00
 1062 s 0 0 2.31465E+00 1.50000E+00
 c
 1063 s 0 0 -2.31465E+00 1.25114E+00
 1064 s 0 0 -2.31465E+00 1.40000E+00
 1065 s 0 0 -2.31465E+00 1.50000E+00
 c
 c Pebble Unit Cell Surfaces (PF: 57.0)
 c
 1080 px 2.31465E+00
 1081 px -2.31465E+00
 1082 py 2.31465E+00
 1083 py -2.31465E+00
 1084 pz 2.31465E+00
 1085 pz -2.31465E+00
 c
 c Pebble surfaces (PF: 61.0)
 c
 1100 s 2.26291E+00 2.26291E+00 2.26291E+00 1.25114E+00
 1101 s 2.26291E+00 2.26291E+00 2.26291E+00 1.40000E+00

1102 s 2.26291E+00 2.26291E+00 2.26291E+00 1.50000E+00
 c
 1103 s -2.26291E+00 2.26291E+00 2.26291E+00 1.25114E+00
 1104 s -2.26291E+00 2.26291E+00 2.26291E+00 1.40000E+00
 1105 s -2.26291E+00 2.26291E+00 2.26291E+00 1.50000E+00
 c
 1106 s -2.26291E+00 -2.26291E+00 2.26291E+00 1.25114E+00
 1107 s -2.26291E+00 -2.26291E+00 2.26291E+00 1.40000E+00
 1108 s -2.26291E+00 -2.26291E+00 2.26291E+00 1.50000E+00
 c
 1109 s 2.26291E+00 -2.26291E+00 2.26291E+00 1.25114E+00
 1110 s 2.26291E+00 -2.26291E+00 2.26291E+00 1.40000E+00
 1111 s 2.26291E+00 -2.26291E+00 2.26291E+00 1.50000E+00
 c
 1120 s 2.26291E+00 2.26291E+00 -2.26291E+00 1.25114E+00
 1121 s 2.26291E+00 2.26291E+00 -2.26291E+00 1.40000E+00
 1122 s 2.26291E+00 2.26291E+00 -2.26291E+00 1.50000E+00
 c
 1123 s -2.26291E+00 2.26291E+00 -2.26291E+00 1.25114E+00
 1124 s -2.26291E+00 2.26291E+00 -2.26291E+00 1.40000E+00
 1125 s -2.26291E+00 2.26291E+00 -2.26291E+00 1.50000E+00
 c
 1126 s -2.26291E+00 -2.26291E+00 -2.26291E+00 1.25114E+00
 1127 s -2.26291E+00 -2.26291E+00 -2.26291E+00 1.40000E+00
 1128 s -2.26291E+00 -2.26291E+00 -2.26291E+00 1.50000E+00
 c
 1129 s 2.26291E+00 -2.26291E+00 -2.26291E+00 1.25114E+00
 1130 s 2.26291E+00 -2.26291E+00 -2.26291E+00 1.40000E+00
 1131 s 2.26291E+00 -2.26291E+00 -2.26291E+00 1.50000E+00
 c
 1140 s 2.26291E+00 0 0 1.25114E+00
 1141 s 2.26291E+00 0 0 1.40000E+00
 1142 s 2.26291E+00 0 0 1.50000E+00
 c
 1143 s -2.26291E+00 0 0 1.25114E+00
 1144 s -2.26291E+00 0 0 1.40000E+00
 1145 s -2.26291E+00 0 0 1.50000E+00
 c
 1150 s 0 2.26291E+00 0 1.25114E+00
 1151 s 0 2.26291E+00 0 1.40000E+00
 1152 s 0 2.26291E+00 0 1.50000E+00
 c
 1153 s 0 -2.26291E+00 0 1.25114E+00
 1154 s 0 -2.26291E+00 0 1.40000E+00

```

1155 s 0 -2.26291E+00 0 1.50000E+00
c
1160 s 0 0 2.26291E+00 1.25114E+00
1161 s 0 0 2.26291E+00 1.40000E+00
1162 s 0 0 2.26291E+00 1.50000E+00
c
1163 s 0 0 -2.26291E+00 1.25114E+00
1164 s 0 0 -2.26291E+00 1.40000E+00
1165 s 0 0 -2.26291E+00 1.50000E+00
c
c Pebble Unit Cell Surfaces (PF: 61.0)
c
1180 px 2.26291E+00
1181 px -2.26291E+00
1182 py 2.26291E+00
1183 py -2.26291E+00
1184 pz 2.26291E+00
1185 pz -2.26291E+00
c
c Shield Unit Cell Surfaces
c
6000 cz 3.16100E-01
6001 cz 3.72000E-01
c
6010 px 5.00000E-01
6011 px -5.00000E-01
6012 p 1. 1.732050808 0 1.00000E+00
6013 p 1. 1.732050808 0 -1.00000E+00
6014 p -1. 1.732050808 0 1.00000E+00
6015 p -1. 1.732050808 0 -1.00000E+00
c
c -----
c
c Full Core Surfaces
c
901 cz 2.50000E+01 $ between porosity region and solid region
903 cz 3.50000E+01
905 cz 1.35000E+02 $ between porosity region and solid region
907 cz 1.62000E+02 $ outer radius of sold graphite
909 cz 1.64700E+02 $ outer radius of shield
911 cz 1.67200E+02 $ outer radius of core barrel
913 cz 1.70000E+02 $ outer radius of down comer
915 cz 1.75000E+02 $ outer radius of reactor vessel
c

```

c plenum region

c

100 sq 3.26531E-05 3.26531E-05 0.000577848
0 0 0 -1 0 0 41.6 \$ Lower Surface of Reactor Pressure Vessel
102 sq 3.46021E-05 3.46021E-05 0.000746514
0 0 0 -1 0 0 41.6 \$ Lower Surface of Coolant Plenum
104 sq 3.57707E-05 3.57707E-05 0.001321565
0 0 0 -1 0 0 41.6 \$ Lower Surface of core barrel
106 sq 3.68649E-05 3.68649E-05 0.001599003
0 0 0 -1 0 0 41.6 \$ Lower Surface of graphite structure

c

c entrance region

c

c axial surfaces

c

200 sq 3.54308E-05 3.54308E-05 1.04817E-04
0 0 0 -1 0 0 8.40000E+01
202 pz 0
204 pz 41.6
206 pz 8.40000E+01

c

c radial surfaces

c

201 cz 4.50000E+01 \$ outer inner solid reflector
203 cz 4.83865E+01 \$ outer radial zone 2
205 cz 5.31185E+01 \$ outer radial zone 3
207 cz 7.07613E+01 \$ outer radial zone 4
209 cz 7.54140E+01 \$ outer radial zone 4
211 cz 8.57400E+01 \$ outer pebble reflector

c

c expansion region

c

c axial surfaces

c

300 kz 144.068483063328 7.376656 -1
302 pz 112.5

c

c radial surfaces

c

301 kz 205.44 0.333352890036215 -1
303 kz 612.281107254241 0.00991119998716127 -1 \$ outer radial zone 1
305 kz -176.586233100836 0.0311228045689724 1 \$ outer radial zone 2
307 kz -24.9390791589181 0.245017934327766 1 \$ outer radial zone 3
309 kz -27.1272572624689 0.276458256720492 1 \$ outer radial zone 4

311 kz -36.0053489556801 0.333336418685121 1 \$
 c
 c active region
 c
 400 kz 130.944444444444 6.36261034808663 1
 402 pz 4.86599E+02
 404 pz 1.80500E+02
 406 pz 1.64848E+02
 408 pz 4.14600E+02
 c
 c radial surfaces
 c
 401 cz 35
 403 cz 4.61000E+01 \$ outer radial zone 1
 405 cz 5.83333E+01 \$ outer radial zone 2
 407 cz 9.60000E+01 \$ outer radial zone 3
 409 cz 1.05000E+02 \$ outer radial zone 4
 411 cz 1.13900E+02
 413 cz 1.25000E+02
 c
 c converging region
 c
 500 pz 4.30500E+02
 c
 c radial surfaces
 c
 501 kz 369.881944444444 0.333374276290362 1
 503 kz 320.34772599041 0.175151925600586 1 \$ outer radial zone 1
 505 kz 197.916514496768 0.0629035770305654 1 \$ outer radial zone 2
 507 kz 845.26913113413 0.0535709826006711 -1 \$ outer radial zone 3
 509 kz 739.203292887614 0.115690067384004 -1 \$ outer radial zone 4
 511 kz 646.993055555556 0.333374276290362 -1
 c
 c defueling chute
 c
 600 pz 4.92850E+02
 602 pz 5.72850E+02
 c
 c radial surfaces
 c
 601 cz 7.10000E+01
 603 cz 7.21942E+01 \$ outer radial zone 1
 605 cz 7.39710E+01 \$ outer radial zone 2
 607 cz 8.15688E+01 \$ outer radial zone 3

609 cz 8.37927E+01 \$ outer radial zone 4
611 cz 8.90000E+01
c
c heat exchanger
c
700 pz 5.26600E+02
c
c radial surfaces
c
701 cz 7.60000E+01
c
c
c control rod surfaces
c
8000 c/z 2.50000E+01 0.00000E+00 5.00000E+00
8001 py 4.00000E+00
8002 px 2.60000E+01
8003 py -4.00000E+00
8004 px 2.40000E+01
8005 py 1.00000E+00
8006 px 2.90000E+01
8007 py -1.00000E+00
8008 px 2.10000E+01
8010 c/z 1.76777E+01 1.76777E+01 5.00000E+00
8011 py 2.16777E+01
8012 px 1.86777E+01
8013 py 1.36777E+01
8014 px 1.66777E+01
8015 py 1.86777E+01
8016 px 2.16777E+01
8017 py 1.66777E+01
8018 px 1.36777E+01
8020 c/z 1.53081E-15 2.50000E+01 5.00000E+00
8021 py 2.90000E+01
8022 px 1.00000E+00
8023 py 2.10000E+01
8024 px -1.00000E+00
8025 py 2.60000E+01
8026 px 4.00000E+00
8027 py 2.40000E+01
8028 px -4.00000E+00
8030 c/z -1.76777E+01 1.76777E+01 5.00000E+00
8031 py 2.16777E+01
8032 px -1.66777E+01

8033 py 1.36777E+01
8034 px -1.86777E+01
8035 py 1.86777E+01
8036 px -1.36777E+01
8037 py 1.66777E+01
8038 px -2.16777E+01
8040 c/z -2.50000E+01 3.06162E-15 5.00000E+00
8041 py 4.00000E+00
8042 px -2.40000E+01
8043 py -4.00000E+00
8044 px -2.60000E+01
8045 py 1.00000E+00
8046 px -2.10000E+01
8047 py -1.00000E+00
8048 px -2.90000E+01
8050 c/z -1.76777E+01 -1.76777E+01 5.00000E+00
8051 py -1.36777E+01
8052 px -1.66777E+01
8053 py -2.16777E+01
8054 px -1.86777E+01
8055 py -1.66777E+01
8056 px -1.36777E+01
8057 py -1.86777E+01
8058 px -2.16777E+01
8060 c/z -4.59243E-15 -2.50000E+01 5.00000E+00
8061 py -2.10000E+01
8062 px 1.00000E+00
8063 py -2.90000E+01
8064 px -1.00000E+00
8065 py -2.40000E+01
8066 px 4.00000E+00
8067 py -2.60000E+01
8068 px -4.00000E+00
8070 c/z 1.76777E+01 -1.76777E+01 5.00000E+00
8071 py -1.36777E+01
8072 px 1.86777E+01
8073 py -2.16777E+01
8074 px 1.66777E+01
8075 py -1.66777E+01
8076 px 2.16777E+01
8077 py -1.86777E+01
8078 px 1.36777E+01

c

c control blade surfaces

c

9001 p 4.14214E-01 -1 0 -2.43538E+00
9002 p -2.41421E+00 -1 0 -1.43069E+02
9003 p 4.14214E-01 -1 0 2.43538E+00
9004 p -2.41421E+00 -1 0 -1.39149E+02
9005 p 4.14214E-01 -1 0 -8.11794E-01
9006 p -2.41421E+00 -1 0 -1.64627E+02
9007 p 4.14214E-01 -1 0 8.11794E-01
9008 p -2.41421E+00 -1 0 -1.17591E+02
9011 p 2.41421E+00 -1 0 -5.87953E+00
9012 p -4.14214E-01 -1 0 -5.92610E+01
9013 p 2.41421E+00 -1 0 5.87953E+00
9014 p -4.14214E-01 -1 0 -5.76374E+01
9015 p 2.41421E+00 -1 0 -1.95984E+00
9016 p -4.14214E-01 -1 0 -6.81907E+01
9017 p 2.41421E+00 -1 0 1.95984E+00
9018 p -4.14214E-01 -1 0 -4.87076E+01
9021 p -2.41421E+00 -1 0 5.87953E+00
9022 p 4.14214E-01 -1 0 -5.92610E+01
9023 p -2.41421E+00 -1 0 -5.87953E+00
9024 p 4.14214E-01 -1 0 -5.76374E+01
9025 p -2.41421E+00 -1 0 1.95984E+00
9026 p 4.14214E-01 -1 0 -6.81907E+01
9027 p -2.41421E+00 -1 0 -1.95984E+00
9028 p 4.14214E-01 -1 0 -4.87076E+01
9031 p -4.14214E-01 -1 0 2.43538E+00
9032 p 2.41421E+00 -1 0 -1.43069E+02
9033 p -4.14214E-01 -1 0 -2.43538E+00
9034 p 2.41421E+00 -1 0 -1.39149E+02
9035 p -4.14214E-01 -1 0 8.11794E-01
9036 p 2.41421E+00 -1 0 -1.64627E+02
9037 p -4.14214E-01 -1 0 -8.11794E-01
9038 p 2.41421E+00 -1 0 -1.17591E+02
9041 p 4.14214E-01 -1 0 2.43538E+00
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9043 p 4.14214E-01 -1 0 -2.43538E+00
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9045 p 4.14214E-01 -1 0 8.11794E-01
9046 p -2.41421E+00 -1 0 1.64627E+02
9047 p 4.14214E-01 -1 0 -8.11794E-01
9048 p -2.41421E+00 -1 0 1.17591E+02
9051 p 2.41421E+00 -1 0 5.87953E+00
9052 p -4.14214E-01 -1 0 5.92610E+01
9053 p 2.41421E+00 -1 0 -5.87953E+00

9054 p -4.14214E-01 -1 0 5.76374E+01
9055 p 2.41421E+00 -1 0 1.95984E+00
9056 p -4.14214E-01 -1 0 6.81907E+01
9057 p 2.41421E+00 -1 0 -1.95984E+00
9058 p -4.14214E-01 -1 0 4.87076E+01
9061 p -2.41421E+00 -1 0 -5.87953E+00
9062 p 4.14214E-01 -1 0 5.92610E+01
9063 p -2.41421E+00 -1 0 5.87953E+00
9064 p 4.14214E-01 -1 0 5.76374E+01
9065 p -2.41421E+00 -1 0 -1.95984E+00
9066 p 4.14214E-01 -1 0 6.81907E+01
9067 p -2.41421E+00 -1 0 1.95984E+00
9068 p 4.14214E-01 -1 0 4.87076E+01
9071 p -4.14214E-01 -1 0 -2.43538E+00
9072 p 2.41421E+00 -1 0 1.43069E+02
9073 p -4.14214E-01 -1 0 2.43538E+00
9074 p 2.41421E+00 -1 0 1.39149E+02
9075 p -4.14214E-01 -1 0 -8.11794E-01
9076 p 2.41421E+00 -1 0 1.64627E+02
9077 p -4.14214E-01 -1 0 8.11794E-01
9078 p 2.41421E+00 -1 0 1.17591E+02

c -----

c Data

c -----

c

m11100 \$ tmp=1.00588E+03K

6000.72c 7.43138E-03 6000.73c 4.39234E-03
8016.72c 2.22941E-02 8016.73c 1.31770E-02
90232.72c 4.99649E-16 90232.73c 2.95319E-16
90233.72c 1.11420E-21 90233.73c 6.58552E-22
91233.72c 3.79636E-19 91233.73c 2.24385E-19
92233.72c 5.82397E-19 92233.73c 3.44227E-19
92234.72c 1.15116E-10 92234.73c 6.80398E-11
92235.72c 2.96832E-03 92235.73c 1.75443E-03
92236.72c 3.44904E-06 92236.73c 2.03856E-06
92237.72c 1.83824E-09 92237.73c 1.08650E-09
92238.72c 1.18718E-02 92238.73c 7.01684E-03
92239.72c 2.59535E-08 92239.73c 1.53399E-08
93236.72c 3.31068E-19 93236.73c 1.95679E-19
93237.72c 2.83173E-10 93237.73c 1.67370E-10
93238.72c 3.56543E-13 93238.73c 2.10735E-13
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94237.72c 7.51334E-22 94237.73c 4.44078E-22
94238.72c 1.28402E-13 94238.73c 7.58923E-14
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94240.72c 9.24419E-09 94240.73c 5.46380E-09
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94243.72c 4.55481E-17 94243.73c 2.69213E-17
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95642.72c 2.39394E-17 95642.73c 1.41495E-17
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46108.72c 1.55472E-08 46108.73c 9.18919E-09
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47109.72c 5.75011E-09 47109.73c 3.39862E-09
48110.72c 1.23637E-11 48110.73c 7.30757E-12
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48114.72c 3.79936E-09 48114.73c 2.24562E-09
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63156.72c 2.21638E-09 63156.73c 1.31000E-09
64155.72c 3.37919E-12 64155.73c 1.99728E-12
64156.72c 2.03930E-10 64156.73c 1.20533E-10
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64158.72c 1.01799E-09 64158.73c 6.01688E-10
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90232.72c 4.80555E-13 90232.73c 2.84033E-13
90233.72c 9.14034E-19 90233.73c 5.40242E-19
91233.72c 5.19514E-14 91233.73c 3.07060E-14
92233.72c 3.49062E-12 92233.73c 2.06314E-12
92234.72c 8.39626E-09 92234.73c 4.96263E-09
92235.72c 2.17307E-03 92235.73c 1.28440E-03
92236.72c 1.54866E-04 92236.73c 9.15337E-05
92237.72c 5.23364E-07 92237.73c 3.09336E-07
92238.72c 1.16271E-02 92238.73c 6.87223E-03
92239.72c 2.21895E-08 92239.73c 1.31152E-08
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93237.72c 3.63346E-06 93237.73c 2.14756E-06
93238.72c 1.11680E-08 93238.73c 6.60086E-09
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94242.72c 9.85896E-07 94242.73c 5.82716E-07
94243.72c 2.18982E-10 94243.73c 1.29430E-10
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95242.72c 4.84266E-10 95242.73c 2.86226E-10
95243.72c 3.98918E-08 95243.73c 2.35782E-08
95244.72c 1.64796E-12 95244.73c 9.74030E-13
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m129 96242.72c 1.0
m130 96243.72c 1.0
m131 96244.72c 1.0
m132 96245.72c 1.0
m133 96246.72c 1.0
m134 96247.72c 1.0
m135 96248.72c 1.0
m136 97249.72c 1.0
m137 98249.72c 1.0
m138 98250.72c 1.0
m139 35081.72c 1.0
m140 36083.72c 1.0
m141 36084.72c 1.0
m142 36086.72c 1.0
m143 37085.72c 1.0
m144 37087.72c 1.0
m145 38088.72c 1.0
m146 38089.72c 1.0
m147 38090.72c 1.0
m148 39089.72c 1.0
m149 39091.72c 1.0
m150 40091.72c 1.0
m151 40092.72c 1.0
m152 40093.72c 1.0
m153 40094.72c 1.0
m154 40095.72c 1.0
m155 40096.72c 1.0
m156 42095.72c 1.0
m157 42097.72c 1.0
m158 42098.72c 1.0

m159 42099.72c 1.0
m160 42100.72c 1.0
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m169 45105.72c 1.0
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m171 46105.72c 1.0
m172 46106.72c 1.0
m173 46107.72c 1.0
m174 46108.72c 1.0
m175 46110.72c 1.0
m176 47109.72c 1.0
m177 48110.72c 1.0
m178 48111.72c 1.0
m179 48113.72c 1.0
m180 48114.72c 1.0
m181 49115.72c 1.0
m182 53127.72c 1.0
m183 53129.72c 1.0
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m188 54136.72c 1.0
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m190 55134.72c 1.0
m191 55135.72c 1.0
m192 55137.72c 1.0
m193 56138.72c 1.0
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m197 58142.72c 1.0
m198 58143.72c 1.0
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m201 60143.72c 1.0
m202 60144.72c 1.0

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m204 60146.72c 1.0
m205 60147.72c 1.0
m206 60148.72c 1.0
m207 60150.72c 1.0
m208 61147.72c 1.0
m209 61148.72c 1.0
m210 61548.72c 1.0
m211 61149.72c 1.0
m212 62147.72c 1.0
m213 62149.72c 1.0
m214 62150.72c 1.0
m215 62151.72c 1.0
m216 62152.72c 1.0
m217 62153.72c 1.0
m218 62154.72c 1.0
m219 63153.72c 1.0
m220 63154.72c 1.0
m221 63155.72c 1.0
m222 63156.72c 1.0
m223 64155.72c 1.0
m224 64156.72c 1.0
m225 64157.72c 1.0
m226 64158.72c 1.0
m1 \$ tmp=8.73150E+02K
 6000.71c 1.16653E-02 6000.72c 1.33335E-01
mt1 grph.15t
m2 \$ tmp=9.73150E+02K
 6000.72c 1.07392E-01 6000.73c 3.76077E-02
mt2 grph.16t
m3 \$ tmp=9.80922E+02K
 6000.72c 1.03507E-01 6000.73c 4.14926E-02
mt3 grph.16t
m4 \$ tmp=9.57589E+02K
 6000.72c 1.15218E-01 6000.73c 2.97822E-02
mt4 grph.16t
m5 \$ tmp=8.73150E+02K
 6000.71c 6.99919E-03 6000.72c 8.00008E-02
 9019.71c 2.58378E-03 9019.72c 2.95326E-02
 3006.71c 1.58803E-08 3006.72c 1.81512E-07
 3007.71c 1.29187E-03 3007.72c 1.47661E-02
 4009.71c 6.45945E-04 4009.72c 7.38315E-03
mt5 grph.15t
m6 \$ tmp=9.73150E+02K

6000.72c 6.44354E-02 6000.73c 2.25646E-02
9019.72c 2.32024E-02 9019.73c 8.12526E-03
3006.72c 1.42606E-07 3006.73c 4.99391E-08
3007.72c 1.16011E-02 3007.73c 4.06259E-03
4009.72c 5.80060E-03 4009.73c 2.03132E-03
mt6 grph.16t
m11 \$ tmp=9.75007E+02K
14028.72c 4.85348E-03 14028.73c 1.75687E-03
6000.72c 9.29267E-02 6000.73c 3.36377E-02
mt11 grph.16t
m23 \$ tmp=8.73150E+02K
9019.71c 3.21802E-01 9019.72c 3.67820E+00
3006.71c 1.97785E-06 3006.72c 2.26068E-05
3007.71c 1.60899E-01 3007.72c 1.83908E+00
4009.71c 8.04505E-02 4009.72c 9.19549E-01
m24 \$ tmp=9.23150E+02K
9019.72c 3.66250E+00 9019.73c 3.37496E-01
3006.72c 2.25103E-05 3006.73c 2.07430E-06
3007.72c 1.83123E+00 3007.73c 1.68746E-01
4009.72c 9.15626E-01 4009.73c 8.43740E-02
m25 \$ tmp=9.73150E+02K
9019.72c 2.96255E+00 9019.73c 1.03745E+00
3006.72c 1.82083E-05 3006.73c 6.37636E-06
3007.72c 1.48126E+00 3007.73c 5.18723E-01
4009.72c 7.40636E-01 4009.73c 2.59364E-01
m700 \$ tmp=9.23150E+02K
11023.72c 3.05958E-04 11023.73c 2.81937E-05
14028.72c 1.15130E-03 14028.73c 1.06091E-04
19039.72c 1.04210E-03 19039.73c 9.60285E-05
19040.72c 1.34090E-07 19040.73c 1.23563E-08
19041.72c 7.52019E-05 19041.73c 6.92978E-06
24050.72c 8.90473E-06 24050.73c 8.20562E-07
24052.72c 1.65114E-04 24052.73c 1.52151E-05
24053.72c 1.83694E-05 24053.73c 1.69272E-06
24054.72c 4.48784E-06 24054.73c 4.13550E-07
26054.72c 7.86154E-06 26054.73c 7.24433E-07
26056.72c 1.19881E-04 26056.73c 1.10469E-05
26057.72c 2.82503E-06 26057.73c 2.60324E-07
26058.72c 3.53349E-07 26058.73c 3.25608E-08
28058.72c 1.20274E-03 28058.73c 1.10831E-04
28060.72c 4.47850E-04 28060.73c 4.12689E-05
28061.72c 1.91503E-05 28061.73c 1.76468E-06
28062.72c 6.00612E-05 28062.73c 5.53458E-06
28064.72c 1.48263E-05 28064.73c 1.36623E-06

42092.72c 4.25032E-05 42092.73c 3.91663E-06
42094.72c 2.59292E-05 42094.73c 2.38935E-06
6000.72c 1.06755E-01 6000.73c 9.83737E-03
8016.72c 5.19175E-04 8016.73c 4.78415E-05
9019.72c 3.93524E-02 9019.73c 3.62629E-03
92235.72c 6.95798E-05 92235.73c 6.41171E-06
92238.72c 2.76537E-04 92238.73c 2.54826E-05
3006.72c 2.44789E-07 3006.73c 2.25571E-08
3007.72c 2.10581E-02 3007.73c 1.94048E-03
4009.72c 9.95699E-03 4009.73c 9.17527E-04
42095.72c 4.41565E-05 42095.73c 4.06898E-06
42096.72c 4.57826E-05 42096.73c 4.21882E-06
42097.72c 2.59422E-05 42097.73c 2.39055E-06
42098.72c 6.48794E-05 42098.73c 5.97857E-06
42100.72c 2.53747E-05 42100.73c 2.33825E-06
54135.72c 1.83125E-30 54135.73c 1.68748E-31
m50 \$ tmp=8.73150E+02K
5010.71c 6.53444E-04 5010.72c 7.46887E-03
5011.71c 2.63019E-03 5011.72c 3.00631E-02
6000.71c 1.31346E-02 6000.72c 1.50128E-01
c
m51 \$ tmp=9.23150E+02K
5010.72c 7.43700E-03 5010.73c 6.85312E-04
5011.72c 2.99349E-02 5011.73c 2.75847E-03
6000.72c 1.49487E-01 6000.73c 1.37751E-02
c
m60 \$ tmp=8.73150E+02K
5010.71c 7.88017E-03 5010.72c 9.00704E-02
5011.71c 7.16379E-03 5011.72c 8.18821E-02
6000.71c 3.76099E-03 6000.72c 4.29881E-02
c
m61 \$ tmp=8.73150E+02K
24050.71c 3.91504E-05 24050.72c 4.47490E-04
24052.71c 7.25940E-04 24052.72c 8.29749E-03
24053.71c 8.07626E-05 24053.72c 9.23117E-04
24054.71c 1.97312E-05 24054.72c 2.25528E-04
26054.71c 3.45640E-05 26054.72c 3.95066E-04
26056.71c 5.27066E-04 26056.72c 6.02436E-03
26057.71c 1.24205E-05 26057.72c 1.41966E-04
26058.71c 1.55353E-06 26058.72c 1.77569E-05
28058.71c 5.28797E-03 28058.72c 6.04415E-02
28060.71c 1.96901E-03 28060.72c 2.25058E-02
28061.71c 8.41960E-05 28061.72c 9.62361E-04
28062.71c 2.64065E-04 28062.72c 3.01826E-03

28064.71c 6.51850E-05 28064.72c 7.45065E-04
42092.71c 1.86869E-04 42092.72c 2.13591E-03
42094.71c 1.14000E-04 42094.72c 1.30302E-03
42095.71c 1.94138E-04 42095.72c 2.21900E-03
42096.71c 2.01287E-04 42096.72c 2.30071E-03
42097.71c 1.14057E-04 42097.72c 1.30367E-03
42098.71c 2.85248E-04 42098.72c 3.26038E-03
42100.71c 1.11562E-04 42100.72c 1.27516E-03

c

m536 3006.72c 1

m549 4009.72c 1

c Steel for Reactor Pressure Vessel and Core Barrel

m31 \$ SS-316

6000.72c 0.00032117333333333333
25055.72c 0.00175185454545455
15031.72c 6.99329032258065E-05
16032.72c 4.26911656381377E-05
16033.72c 3.36964578284606E-07
16034.72c 1.89149449943759E-06
16036.72c 8.9857220875895E-07
14028.72c 0.00129042857142857
24050.72c 0.000683602384459175
24052.72c 0.013182591528527
24053.72c 0.00149480005862983
24054.72c 0.000372087373819551
28058.72c 0.00719820016768188
28060.72c 0.00277271917089651
28061.72c 0.000120539215758
28062.72c 0.000384245184267167
28064.72c 9.79116787648313E-05
42092.72c 0.000186130725692655
42094.72c 0.000116018141014626
42095.72c 0.00019967662756247
42096.72c 0.000209208928878267
42097.72c 0.00011978089153402
42098.72c 0.000302650566776533
42100.72c 0.000120784291672524
7014.72c 0.000342765218807083
7015.72c 1.25912911338893E-06
26054.72c 0.003273142835476
26056.72c 0.0517608035982515
26057.72c 0.00124153693759435
26058.72c 0.000158013792057462

c begin_mocup_reaction_rate_tallies

c time dependent reaction rates

fc4 Reaction rates

f4:n

11100 11200 11300 11400 11500 11600 11700 11800
21100 21200 21300 21400 21500 21600 21700 21800
31100 31200 31300 31400 31500 31600 31700 31800
41100 41200 41300 41400 41500 41600 41700 41800
12100 12200 12300 12400 12500 12600 12700 12800
22100 22200 22300 22400 22500 22600 22700 22800
32100 32200 32300 32400 32500 32600 32700 32800
42100 42200 42300 42400 42500 42600 42700 42800
13100 13200 13300 13400 13500 13600 13700 13800
23100 23200 23300 23400 23500 23600 23700 23800
33100 33200 33300 33400 33500 33600 33700 33800
43100 43200 43300 43400 43500 43600 43700 43800
14100 14200 14300 14400 14500 14600 14700 14800
24100 24200 24300 24400 24500 24600 24700 24800
34100 34200 34300 34400 34500 34600 34700 34800
44100 44200 44300 44400 44500 44600 44700 44800
15100 15200 15300 15400 15500 15600 15700 15800
25100 25200 25300 25400 25500 25600 25700 25800
35100 35200 35300 35400 35500 35600 35700 35800
45100 45200 45300 45400 45500 45600 45700 45800

fm4 (1)

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(1.0 102 (16) (17) (-6) (102))
(1.0 103 (16) (17) (-6) (102))
(1.0 104 (16) (17) (-6) (102))
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(1.0 224 (16) (17) (102))
(1.0 225 (16) (17) (102))
(1.0 226 (16) (17) (102))

f64:n (1180

2180
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2280
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4280
1380
2380
3380
4380
1480
2480
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4480
1580
2580
3580
4580)

fm64 (1 536 -2)

sd64 1

f74:n (1180

2180
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4180

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3280

4280

1380

2380

3380

4380

1480

2480

3480

4480

1580

2580

3580

4580)

fm74 (1 549 107)

sd74 1

kcode 10000 1.0 10 110

prdmp 10000 10000 10000

print 30 40 60 72

mode n