OPTIMUM UTILIZATION OF FISSION POWER WITH GAS CORE REACTORS

By

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Dedicated to my parents for encouraging me to do my best.

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A gas or vapor core reactor (GCR) with power conversion cycles including a cascaded combination of Brayton, superheated Rankin, and magnetohydrodynamic (MHD) generator forms the basis for a Generation IV concept that is expected to set the upper performance limits in sustainability, proliferation resistance, and power conversion efficiency among all existing and proposed fission-powered systems. The GCR system described here is not constrained by solid fuel-cladding temperature limitations, and is only constrained by the less restrictive vessel limits. Because the GCR is a circulating fuel reactor, it can potentially reach the maximum theoretical burnup, while current reactors only obtain approximately 4% of that limit. Generation IV reactors only propose to attain 15 to 20% percent of theoretical burnup, without use of reprocessing. The gaseous fuel is also inherently proliferation resistant as it exists in the core in small quantities compared to a solid fuel reactor, and the mix of actinides in the fuel is shown to be very poor for weapons construction.

This research studies the actinide production and depletion features of three GCR designs. The research is done using Monteburns, which integrates MCNP and Origen computer codes to simulate a time-dependent monte-carlo method depletion analysis. Several designs were analyzed during the research and three specific designs were chosen to evaluate in depth. These designs were then compared to a LWR design to evaluate how the GCR functions compared to current power reactors. The research simulates an operation time of 10 years, studying the effects of parametric changes in design specifications on actinide inventory.

Core depletion analysis has shown that using a GCR for power production limits both the actinide inventory produced, as well as the unused spent fuel. The GCR is up to 15 times more efficient than a typical LWR in terms of fuel utilization (percent of fuel fissioned). The GCR produces as little as one-tenth the actinides of a conventional LWR, while limiting the overall waste by two orders of magnitude, and is inherently resistant to weapons proliferation. The GCR shows promise as being a viable means of power production while limiting nuclear waste.

CHAPTER 1
INTRODUCTION

Introduction

The Gaseous Core Reactor (GCR) concept has the potential to surpass all existing and current plans for advanced reactors in the areas of sustainability, proliferation resistance, and power conversion efficiency. The fuel in a GCR (as its name implies) is in a gaseous state throughout the lifecycle of the core. Because of its gaseous state, the fuel can be pumped into the system allowing continuous online fueling, giving operators greater control over the nuclear environment in the core. The unique ability to rapidly change the neutron environment through online fueling opens up a range of abilities to manage fission products, manage actinide production, and limit proliferation dangerous materials.

The GCR design is being studied as a viable alternative to solid-fuel power reactors, because of its promise in the areas of waste products and high burnup capabilities. Currently there are no regulations differentiating solid or gas fuel reactors for core limitations, therefore the GCR must abide by the current regulations placed on existing reactors. The GCR design is being studied under the assumption that these current regulations and restrictions do not change. The GCR design would prove even more effective if limitations on the reactor were less severe than on solid-fuel reactors.

Features of the Gas Core Reactor

The GCR design can be characterized as an externally moderated, circulating fuel reactor with a thermal neutron spectrum. The fuel will typically be enriched uranium

tetra-fluoride (UF_4) and will remain with current licensing limitations for the purpose of this study.

The GCR features a unique list of traits that promise to make it an exceptionally safe and environmentally suitable reactor. The GCR design features a very low fuel inventory, requiring nearly 2 orders of magnitude lower inventory than a Light Water Reactor (LWR). Because the fuel is constantly being fed into the core excess fuel will be required on site, so that the core can remain critical, however this can be kept secure while not in use. The system's criticality is a direct function of the core pressure, therefore any pressure leak will immediately cause a loss in reactivity, thus protecting the workers and the public from criticality accidents. The reactor will be kept isolated from the workers, and operated remotely, limiting the radiation dose to plant workers compared to current operating reactors. The GCR design also shows promise of being able to achieve extremely high burnups, thus limiting the amount of nuclear waste that must be stored in containment.

Neutronically and electrically the GCR design promises to set the upper limit in performance for power reactors. Plant operators can directly change the neutronics in the core through a variety of methods. Rotating BeO columns, mass flow rate, and pressure can be adjusted to control core reactivity and neutron energy spectrum, allowing the users to achieve optimum neutronics. The system is designed to use a minimum of a combined cycle (Brayton and Rankine), as well as possibly including a Magneto-Hydro-Dynamic (MDH) third tier system. The electrical efficiencies are expected to achieve upwards of 60% for the combined cycle design and 70% when including the MHD topping cycle [1].

Mechanically the system is very simple compared to current nuclear power plants. There will be no control or fuel rods, or any other mechanical structures in the core: simply a cylinder containing the gaseous fuel, and an outside cylinder containing the reflector material. This drastically reduces the risk of accidents caused from mechanical limitations. There will be no cladding as the fuel is gaseous, further eliminating any mechanical or material concerns. The MHD system will present a mechanical challenge, however the system with just the combined cycle is no different than current combined-cycle systems used throughout the world, and therefore imposes no additional danger to mechanical limits.

The backend of the cycle will function like existing power plants, there will be filters to remove unwanted products (in this case fission products will most likely be chemically separated) and the fuel will be pressurized to preset specifications. The backend of the cycle will also include a feeding valve where new fuel is mixed with the existing fuel, thus removing the current shutdown period required by current nuclear power plants to reload the fuel.

Thesis Objectives

The goal of this research is to study the design of a GCR and analyze its ability to reduce the production of nuclear waste while operating as a power reactor. The work will focus on reasons why waste is produced, which waste products are most important, methods in which to reduce their production, and how well the GCR design functions in reducing waste compared to a solid-fuel reactor. The main objectives of this research can be classified as follows:

- Evaluate reflector size changes and their effects on criticality, spectrum, and actinide inventory.

- Evaluate effects on spectrum and actinide inventory for varying core pressures.
- Evaluate effects on spectrum and actinide inventory for varying fuel. Enrichments.

- Evaluate effects on spectrum and actinide inventory for varying core sizes.
- Evaluate different fuel-feeding schemes and their affects on spectrum and actinide inventory.

- Determine important and dangerous actinides and evaluate the GCRs ability to limit these actinides.

- Compare GCR designs to current operating LWR designs.

It is recognized that assumptions made in the analysis of the core will affect the results, however reasons will be given as to why they may or may not be important. Additionally it should be noted that there is acoustic phenomena related with GCR designs with the potential to significantly affect the results. This research does not include acoustic effects, as the necessary tools to evaluate their effects are not available.

Thesis Organization

A brief description of current work that may reduce nuclear waste is given in Chapter 2. The chapter includes methods to reduce waste (such as transmutation), as well as other reactor designs that will limit waste production.

Current design methodology for the GCR is described in Chapter 3. This includes a description of the GCR, and its subsystems. The chapter also includes current licensing, material and mechanical limitations, as well as any limitations the GCR possesses in the areas of meeting those licensing requirements.

A section describing how the materials and geometries were selected for this research is detailed in Chapter 4. It includes detailed reasons why every choice in the GCR design was made for this research, as well as the effects of that design choice. This

chapter includes the step by step methodology performed in designing the systems to be evaluated.

A complete analysis of three different GCR designs, as well as a reference LWR design is included in Chapter 5. This chapter analyzes reasons why the GCRs performed as they did, as well as describing any limitations found in the research. The neutronic analysis was performed using MCNP4C2, in conjunction with Origin2.1. The code package Monteburns assimilated both packages to work in conjunction with each other, to produce time-dependent results.

A brief summary of results and recommendations for future work is included in Chapter 6. This includes any problems found during the course of the research, as well as the overall performance of the GCR compared to LWR designs.

The appendix section lists sample inputs used in the code packages and any insights that were thought relevant. Inputs were taken from actual code inputs used in the research, and can therefore be used to recreate some of the results obtained in this research.

CHAPTER 2
STATE OF THE ART STUDY

Designs discussed in this chapter include vapor core reactors (VCRs), neutron accelerators, liquid metal fast breeder reactors (LMFBRs), molten salt cooled reactors, and gas cooled reactors. Each design has some benefit in the area of lowering waste production, or depleting current waste, and should be discussed in comparison to the GCR design that is being studied in this research.

Vapor Core Reactors

Vapor core reactors (VCR) design emerged as an alternative to the GCR, with functions tailored to utilizing a faster neutron spectrum. Analyses of the VCR cycle and neutronics have been performed by Kahook [2]. Vapor core reactors function similarly to GCRs, except the fuel changes phase from gaseous to liquid during its cycle through the core. Typically VCRs are associated with very high operating temperatures and efficiencies, simple geometry, and a variety of power control mechanisms. The secondary side for a VCR is similar to the GCR discussed in this research and would most likely include MHD as an additional means of power conversion. Therefore an additional working fluid must be used to increase conductivity of the fuel. The fissioning fuel would most likely be UF_4, the same as in a GCR design, however the operating temperature for the fuel will be around 3000 K. The main geometry difference between the VCR and GCR design is the inclusion of boiler columns in a VCR, where the fuel vaporizes and flows through the core, as opposed to the open cylinder geometry in a GCR design. The main use for VCR appears to be space transportation, because in a terrestrial

design, building and maintaining a GCR is easier while still providing most of the benefits from using a gaseous fuel. The only benefits GCRs do not also possess is the smaller size that can be used for the VCR because of its liquid stage, as well as its slightly higher efficiency because of the temperature increase. The high operating temperature is of course tailored for space power, as the only method of heat rejection is thermal radiation. The VCR should be just as proficient at reducing actinide inventories, however is more costly and less practical than a GCR design for terrestrial concerns.

Actinide Transmutation Through Neutron Acceleration

Accelerator driven transmutation system using particle accelerators to eliminate waste is an alternative (or can be used in conjunction) to minimizing waste production. Ideally almost all waste can be transmuted into nonradioactive material. Of course all of the produced actinides will be radioactive, but they can be transmuted into actinides with short half-lives, and then be allowed to decay naturally.

The process works by accelerating protons through a linac or cyclotron, and impinging them onto a system comprised of a subcritical nuclear assembly and target. The protons enter the target (typically W or Pb/Bi) and cause spallations, thus sending neutrons into the fuel, and causing power to be generated. The idea is that the spallations will cause a high enough neutron flux (with the correct energy) to transmute any transuranics. The power generated will then be used to power the accelerator with some energy left over for commercial use. Thermal efficiencies are believed to be in the 10% to 20% range [3], with the accelerator taking up approximately 10% of the power. Fission products would be separated out chemically in-between cycles, which would take place approximately three times a year. The system provides a unique way of attempting to eliminate waste, however one major problem exists in this design. The accelerators

have to remain on at all times, or the system will shut down, as it is sub critical by itself. For commercial use this presents a large problem as any shutdown time will cost a company a large amount of money lost in profits.

The actual transmutation of waste as opposed to a GCR should be comparable, as the main factor in determining how much waste is transmuted is neutron energy. It should be noted that while the GCR can vary the neutron energy drastically through user control, the average spallation neutron energy may be past what the GCR is capable of achieving, therefore transmuting a greater portion of actinides. An increased flux does not correlate to an increased transmutation, as more actinides will also be produced as well (the ratio of actinides removed to produced is dependent only on cross-sections), the only way to transmute the actinides is to use a very specific neutron energy. The Accelerator Driven System (ADS) also heavily relies on reprocessing fuel, which is another expensive process. Commercially the accelerator driven systems do not possess the same possibility for economical power generation as the GCR system does. As far as waste transmutation is concerned, the ADS design may prove to be effective yet costly and not for commercial use.

Reprocessing

Although fuel reprocessing is not a method of power generation, it is a method that enables power reactors to approach theoretical maximum burnup. Spent fuel is taken from reactors and deposited into the reprocessing plant, where chemical processes will separate the unused fuel from the nuclear waste. Reprocessing potentially allows maximum burnup in fuel, as any fuel that is not burned, can be placed back into another core.

Currently the only type of reprocessing used commercially is PUREX, which uses nitric acid to remove useful fuel from waste products. The downside to this method of reprocessing is the expense that it entails. Lamarsh [4] gives a simple explanation of the reprocessing method. Reprocessing is currently not very economical not only from an operations standpoint but from a capital cost as well. The plants require very strict regulations because they are dealing with large amounts of hazardous materials (radioactive and proliferation dangerous), thus raising the capital costs considerably. As fuel becomes more expensive in the future the operation viability may increase however. From an engineering standpoint, the main concern is limiting the potential for flammability due to the organic solvent.

Liquid Metal Fast Breeder Reactors

The Liquid Metal Fast Breeder Reactor (LMFBR) is a concept that increases the electric efficiency (40%) and fuel utilization compared to a LWR. The fuel is typically enriched uranium or plutonium (15-30%) and is surrounded by liquid sodium as the coolant. The fuel is in solid form, in fuel pins, much like a LWR, except the cladding is usually stainless steel.

Sodium has a much higher thermal conductance than water and therefore makes a better heat transfer medium. Using sodium as a coolant allows higher maximum temperatures because of its high thermal conductance (141 w/mK compared to 0.6 w/mK for water), allowing for faster removal of heat. There is no moderator in this reactor design as the purpose is to utilize high-energy neutrons to breed fissile material [4]. Sodium is heavier than water, which will limit the ability of the system to thermalize neutrons compared to a LWR, allowing for a higher neutron energy spectrum. Plutonium will then be bread from absorptions in U238, which can then be used as fissile fuel to

extend the life of the core. Maximum burnup is expected between 100 and 150 GWD/MTU, which is three to four times greater than burnups for LWRs.

The most significant problem with the system is the sodium is highly reactive with air and water. Therefore extreme precautions must be taken to insure the system is sealed off from the outside environment. The secondary side in this system (the turbine side) requires a working fluid to go through the turbine. The working fluid is typically water (steam), which creates the potential for sodium-water interaction. Therefore the secondary and primary side must be very carefully sealed off from each other. If any leaks were to occur and the sodium came into contact with water, the system would undergo a violent explosion depending on the break in the system. This is a very large detriment to this system, as any sodium leaks will lead to a loss of the system. Sodium also has a very high melting point (98 ºC), therefore requiring the system to remain above room temperature at all times. This causes problems in the case of emergency if the reactor needed to be shut down, the sodium would solidify, thus ruining the core. At then end of core life, the sodium is also very radioactive and must be handled with care, further increasing safety problems and costs associated with LMFBRs. Finally since the fuel used in the LMFBR is enriched well above current limits, licensing problems are certain to arise. If the licensing restrictions can be changed, and the system can be sealed perfectly, this reactor design shows promise for reducing nuclear waste by not only using more of the initial uranium, but by breeding plutonium from unfissioned uranium to then be used as fuel.

CHAPTER 3
DESIGN OF THE URANIUM TETRA-FLUORIDE GAS CORE REACTOR

Introduction

In the design of a GCR the choice of geometry, pressure, enrichment and fuel feeding must be based on the application of the plant. The purpose of this research is to study the GCR's ability to limit actinide inventory, either by burning actinides as they are produced, or by limiting the production of those actinides. The biggest factor in determining actinide production and depletion is the neutron energy spectrum, which is affected by the physical properties of the core. Therefore once a general design is established the purpose of the research is to parametrically evaluate the properties of the core, and their effects on actinide production and depletion. When establishing the general design it must be known how strict current regulations will be followed as well as the limiting effect of current material and mechanical properties. The reactor must also be able to maintain criticality throughout the life of the core. Although the primary focus of this research is to evaluate actinide production, the system must be feasible from both a neutronics and materials standpoint.

Preliminary Design Considerations

This particular GCR design is being developed for terrestrial power production, and as such must attempt to adhere as closely as possible to current restrictions placed on existing designs. Therefore the following issues should be addressed:

- The pressure of the core should not reach an unattainable state. Current reactor pressure vessels for Pressurized Water Reactors (PWR) operate at approximately 2000 psi or 138 bar. As long as the pressure of the GCR does not overly exceed

the nominal PWR pressure, there is no cause for concern. Pressure vessels can be built to operate at higher pressures currently, however in the scope of the research the limiting pressure will be 175 bar.

- The dimensions of the reactor must be feasible. Reactor dimensions will vary between three and four meters for a right circular cylinder.

- Material considerations for wall liners, coolant flow, and reflector liners and coolant are not considered in the scope of this research. They will be discussed later as to how they will affect the reactor, but are not included in any calculations.

- The lifetime for the core is 10 years, where radiation damage is considered the limiting factor on core length.

- Fueling will done on-line, meaning fuel will be added to the core during operation, either at state points, or continuously. During operation, fission product removal must be done online, also at either state points or continuously.

The above issues are considered to be the most important from a design standpoint for the scope of this research.

Material Selection

The GCR design being study in this research is a BeO externally moderated core with UF_4 circulating as the fissioning fuel. The choice for BeO as the moderator/reflector and UF_4 as the fuel is based on considerations outlined in the following sections.

Moderator-Reflector Material

Conventional thermal reactors use low mass number isotopes with small absorption cross-sections relative to their scattering cross sections. The low mass number allows rapid loss of energy upon scattering, and the lower the ratio of absorption to total cross section allows for more neutrons to reach thermal energies without being absorbed in the moderator-reflector. Traditional materials include normal water (H_2O), deuterium (D_2O), beryllium (Be) and beryllium oxide (BeO), and pyrolitic graphite, which have all been studied in [5-7]. The decision of material must be based on the specific design goals for

the material. Since the purpose of this research is to evaluate actinide inventory, the spectrum desired for the reactor is faster than the conventional LWR. As will be shown later, faster spectrums have higher fission to capture cross section ratios, which lead to higher removal rates. The choice of BeO will allow the reactor to operate at a faster spectrum than traditional thermal reactors, while also allowing spectrum shifts if desired. The BeO can be processed into columns that can be externally adjusted to change the spectrum inside the core. By making the external moderator BeO instead of a liquid, there is no pumping required, which lessens the number of movable parts. As the GCR is a high temperature reactor Be may be an unsafe material to implement as the reflector. Operating temperatures for Be must remain in the region between 1600 and 2000K, while this reactor design will reach temperatures of up to 2400 K. The BeO reflector can operate safely at temperatures up to 2800K. As far as thermodynamic safety is concerned graphite is superior as it can operate safely at temperatures of up to 4000K, exceeding the system requirements. Neutronically it was shown by S.D. Kahook [2] that graphite is inferior to the BeO reflector as it provides a lower reactivity gain for the same size reflector. In terrestrial applications size is not an explicit demand for feasibility, however as the reactor may be moderated by rotating reflector columns, graphite becomes cumbersome to maneuver and may pose problems for moderation control systems.

From a chemical standpoint BeO and Be prove to be superior to H_2O or D_2O as well. Chemically H_2O and D_2O are not compatible with either choice of fuel, UF_4 or UF_6. If there were to be any leak in the system the chance for the fuel and reflector coming into contact is possible, and would cause a violent reaction if the reflector were D_2O or H_2O.

For the purposes of code calculations the reflector is assumed to be a cylindrical shell around the fuel, in reality it will be many columns containing graphite on one side, and an absorber material on the other. Criticality control will be accomplished by rotating the graphite columns to achieve an optimum mix of absorption and moderation/reflection. Reflector size in a GCR has a direct effect on the core's spectrum and ability to limit actinide production, and various sizes will be studied and discussed in chapter 5.

Fissioning Fuel Material

The possible choices for fuel for this reactor consist of only UF_4 and UF_6. The University of Florida has done research on both the above fuel types examining their viability as fuel choices [2,5-7]. The choice of fuel for this reactor is based on chemical stability and thermo-physical properties.

For the purpose of this research, it can be shown in Figure 3-1 that UF4 is the ideal choice. Pressure for the core may exceed 100 bar to harden the spectrum, therefore at those pressures and the 2400 ºK operating temperature of the core, UF6 would be in a liquid phase. It should be noted that UF6 is also not as stable as UF4 and will lead to material and decomposition problems at the operating pressure and temperatures used in the reactor. Uranium metal is not being looked at as a feasible choice as it would remain solid at the operating temperature and pressure as shown in Figure 3-2. Fuel enrichment, as well as feed fuel enrichment must also be considered when designing a core. The lower the enrichment, in principal, the higher the conversion ratio (it will be shown in Chapter 5 that this does not represent the best limitation of actinides as was originally thought). Another important parameter affecting fuel enrichment is the legality of the

fuel. Currently 5%-enriched fuel is the upper limit for fuel used in power production. This research is focused on designed a viable means of power production and therefore must try and comply with current standards (although the limit on enrichment has steadily risen over recent years and is being proposed to be raised again). The enrichment of the feed fuel can also be varied, however it was chosen to be the same enrichment as was being burned (if the core burns 35% U238 and 65% U235, then the feed fuel enrichment will be the same mix). Although this will cost more than conventional fuel for the increase in seperative work units (SWU) needed, this method was chosen to try and keep the U238 constant, and to limit pressure increases from excess fuel. It is possible to operate the core with low feed fuel enrichment (although this will decrease the effectiveness of the reactor), however for this research the feed fuel enrichment will not be limited to conventional fuel enrichment limitations.

Power Cycle Description

Ideally the power cycle for the GCR would include MHD power conversion, which leads to a combined brayton-rankine cycle. The schematic design for this system is shown in Figure 3-3. The design shown operates at 70% efficiency, however for this research it is assumed that the design will not encompass the use of MHD, thus only being 60% efficient. For the purpose of this research it is assumed that the fission products can be completely removed chemically during operation. This chemical cleanup would be done in the backend of the cycle before the fuel is sent back to the core. The assumption that all fission products can be removed may shown not be viable in future studies, however for the scope of this study it is assumed acceptable

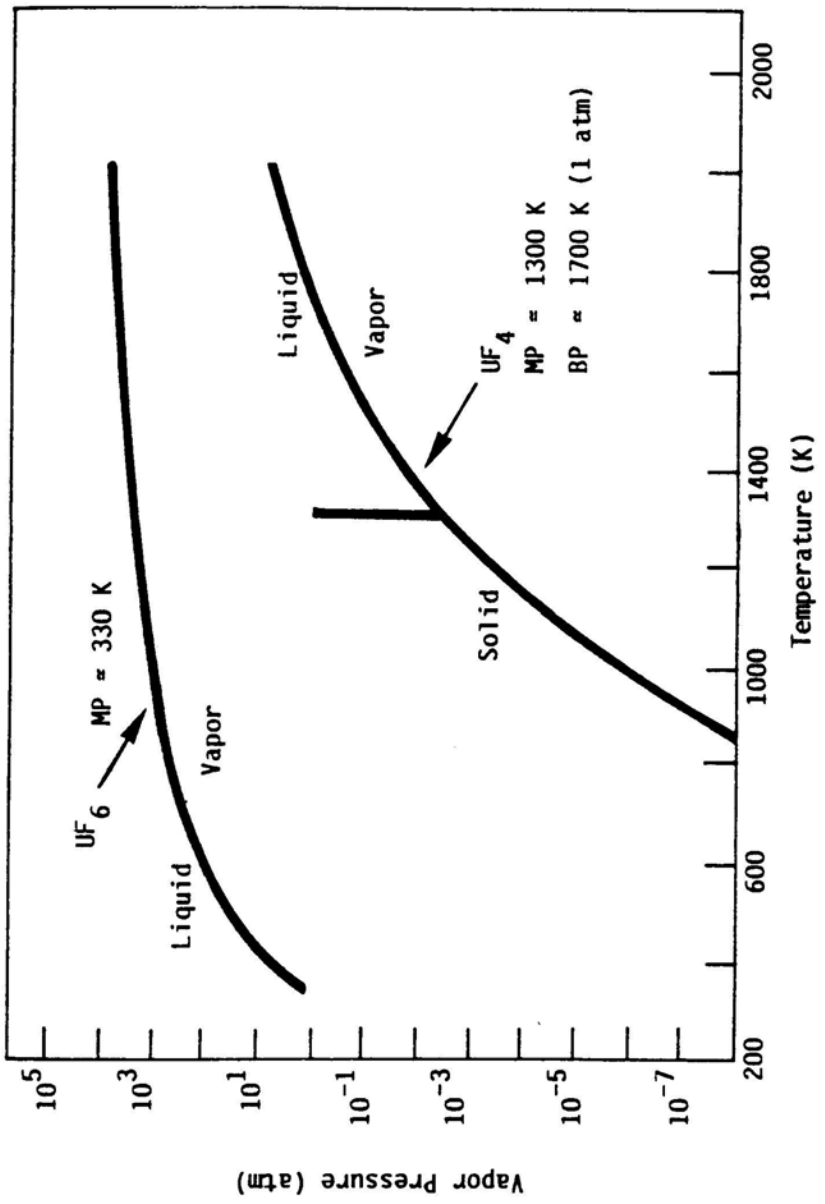

Figure 3-1. Saturation curves for UF4 and UF6.

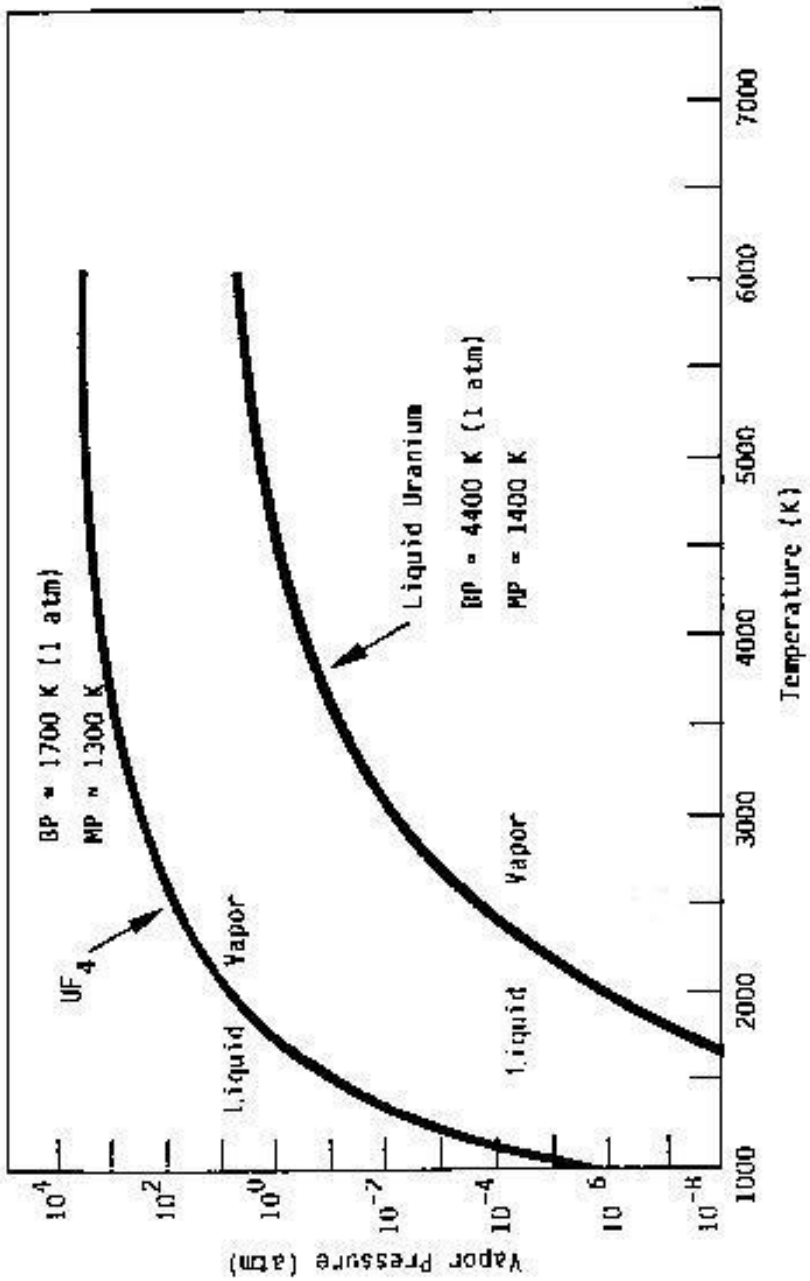

Figure 3-2. Saturation curves for UF4 and uranium metal.

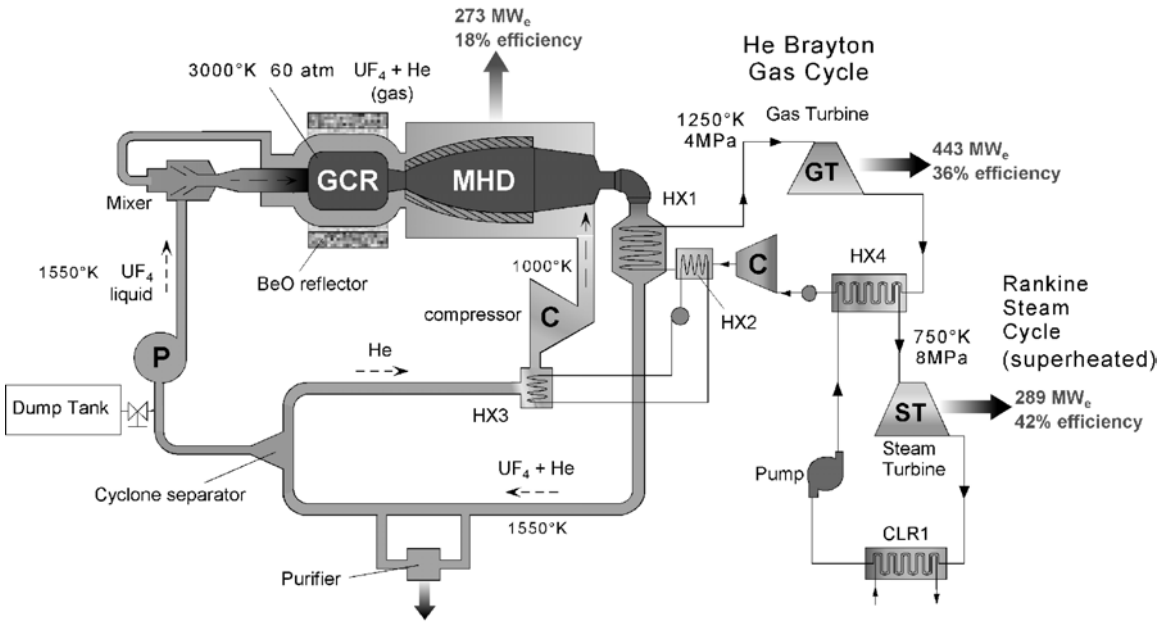

Figure 3-3. Example of a combined cycle schematic for a GCR. A 1500MWth, 1000MWe G/VCR power plant with a combined cycle efficiency of 69%.

Neutronic Analysis for the Gas Core Reactor

The code package used in the research is Monteburns, which integrates MCNP4C2's ability to calculate fluxes and criticality, and Origen2.2's ability to simulate core depletion. Monteburns requires a standard MCNP input card, and a specialized input card that the code uses to execute Origen with user defined parameters, such as specification of which actinides to monitor. The code initially executes a standard MCNP run, calculating flux plots, cross sections, and criticality, and then passes the flux values and cross sections to Origen, to produce one-group cross sections, which are then used to burn the core. Origen will burn the core for a user specified time step, tally the user defined important isotopes, and then generate a new MCNP input card. The code

will then run the new MCNP input card, and repeat the process for a specified number of iterations, simulating time dependent depletion. Monteburns allows the input of fuel feeding, which allows the user to determine how much fuel should be added per cycle, as Origen will be depleting the current inventory. The feed fuel can be input daily, or at specified state points throughout the life of the core. When evaluating longer cycle lengths (10 years), it is assumed acceptable to input the fuel continuously throughout the core life. In practice the fuel will most likely be fed into the core every few days or weeks, however to do this explicitly in Monteburns would require a tremendous amount of CPU time, as the code has to execute an MCNP run every state point. Feeding the fuel continuously should not significantly change the neutronics in the core, as the extra fuel would only be present for a few days or weeks at most. Adding the fuel continuously or at state points will not affect the reactivity of the core, as the external BeO reflector cylinders can be rotated to maintain criticality. It should be noted that Monteburns does not calculate burnup correctly (when including online fueling), it will only calculate burnup as a function of Metric Ton Heavy Metal (MTHM) of initial inventory, the user must tally the total MTHM put into the core through the feed file, and recalculate the correct burnup. It is also assumed that the removal of fission products can also be done continuously online, and this is simulated in Monteburns as well. The lack of liners and wall cooling materials will lead to an overestimation of K_{eff}, as will the assumption that the fuel remains in a static position. Fuel in the GCR is circulating which means that when the fuel exits the core, delayed neutrons will still be produced, however the code packages used do not account for this effect which will cause a phantom increase in reactivity that would not be experienced in practice. This means that in practice delayed

neutrons will be released while in the backend of the cycle, which will not be contributing to the criticality of the core, whereas in the research those neutrons are emitted while still in the core, causing the phantom increase in reactivity. It was shown by S.D. Kahook [2] that these assumptions do not severely affect the reactivity worth in the core, and values obtained under these assumptions are reasonable.

CHAPTER 4
CONFIGURATION AND OPTIMIZATION OF DESIGN PARAMETERS

The core geometry consists of only two concentric right circular cylinders, containing the fuel and the reflector as shown in Figure 4-1. The inner cylinder consists of the fissioning fuel (UF4), and the outer consists of the reflector (BeO). In practice there will be additional lining materials and coolants, however they are beyond the scope of this study, as their effect on core neutronics is small.

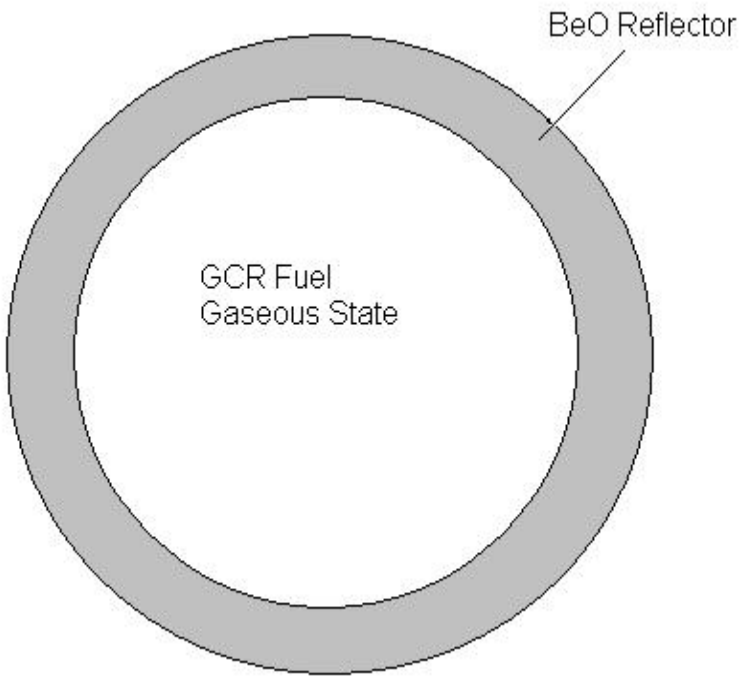

Figure 4-1. Sample core geometry

Reflector and Fuel Density Optimization

Reflector size and fuel density are optimized simultaneously as varying one requires a change in the other to compensate for reactivity change. For instance

increasing the reflector size introduces a positive reactivity insertion and must be offset with a decrease in fuel density. Conversely shrinking the reflector size requires an increase in fuel density to maintain criticality. It should be noted that instead of lowering the fuel density the enrichment could be adjusted, however evaluating the system with three variables simultaneously significantly increases the difficulty. Therefore it was decided to evaluate reflector and fuel density parameters for chosen enrichments. Pressure limitations were not placed on the core at this stage, as it is just scoping calculations to determine viable reflector size and pressure combinations (viable in the sense that the reactor would start its cycle critical).

In this stage of optimization varying reflector sizes were matched up with fuel densities to achieve several critical reactor design possibilities. This stage of the research is not concerned with actinide inventory, but is only concerned with producing several critical reactor designs that can be studied in depletion analysis. The initial enrichment used in this optimization phase was 10% U235 by atom. This value will be changed during depletion analysis, but in this stage of the research it was used as a benchmark value. For a given enrichment, there is a minimum reflector thickness where no amount of pressure increase will cause the reactor to reach criticality. It was also initially assumed that the fastest neutron spectrum would produce the best actinide composition, so reflector size was minimized in many cases. A smaller reflector will provide a faster neutron spectrum because a neutron scattered into the reflector will either escape the reflector and disappear from the system or be reintroduced into the fuel region with a lower average number of scatters than when a larger reflector is present. The lower number of scatters is because in a larger reflector the neutron may backscatter towards

the core from a point in space where if a smaller reflector were present the neutron would have already escaped the geometry. The faster neutron spectrum also produces higher fission to capture ratios for the actinides, resulting in more fissions per interaction, thus depleting the actinides. However it will be shown in Chapter 5 that the assumption that the fastest neutron spectrum results in the best actinide inventory is false and therefore other designs were researched.

One benefit to a smaller reflector is an increase in plutonium generation, which increases the conversion ratio of the core (fissile material produced compared to fissile material depleted). A faster spectrum causes more U238 to undergo capture thus eventually creating Pu239, which has high relative neutron cross-sections within the energy regime that the GCR operates. A higher conversion ratio in a LWR leads to a higher achievable burnup. It was also initially believed that this would increase the GCRs effectiveness, and designs were based on achieving the highest plutonium production.

It should also be noted that rotating the BeO columns to increase or decrease neutron thermalization can control the spectrum of the core. As was previously mentioned the reflector may consist of many BeO columns as shown in Figure 4-2 (The figure shows the BeO and absorber material with a random configuration, this is just a sample made to show the general setup of the columns). However, the columns were simulated by one concentric cylinder enveloping the core, which they will approximate in an actual design. Criticality of the core can also be managed by rotating these BeO reflector columns; however, an easier method is to adjust the flow rate of the fuel. As there is no way to model dynamically rotating BeO columns in MCNP, the geometry

used was a concentric cylinder around the fuel core. Because of code limitations the only way to manage criticality is through the fuel feed rate in Monteburns (as opposed to the rotating column method) and enrichment, however in practice this will not be the case.

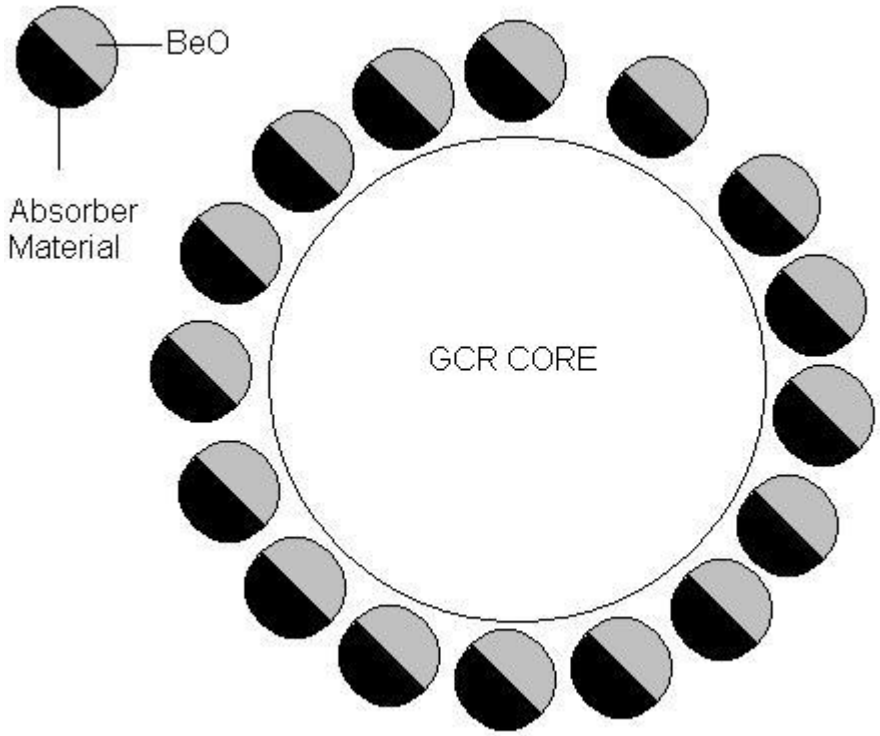

Figure 4-2. Sample BeO column configuration

The initial reflector material used in this research was Be, however because of temperature limitations BeO will be used instead. The change in reflector thickness required to achieve the same criticality when going from Be to BeO is approximately 2.5 cm, not a significant change in size. Changing from Be to BeO does not significantly affect the neutron spectrum as most of the scatters are from interactions with Be, and the few scatters that occur from an interaction with oxygen will result in almost the same energy loss as from Be (although the maximum energy loss possible from an interaction with Be compared to O is different, when taking into account the probability of

interaction with O the average energy loss per scatter is nearly the same for Be and BeO). The use of BeO became necessary as a shielding material when considering a highly pressurized core, since UF4 would be in a liquid phase at the temperatures needed to keep Be from melting. Using UF4 requires a higher temperature therefore BeO is needed because its melting point is 2800 K, well above the operational temperature of 2400 K used in this research.

Tables 4-1 through 4-5 show the varying pressure and reflector size combinations studied to achieve criticality. The minimum shield thickness to reach criticality in a 10% enriched core is approximately 30 cm. This reflector thickness is the base design used in the next phase of optimization – actinide inventory optimization. No amount of pressure increase using a 20 cm reflector thickness and 10% U235 enrichment will cause criticality to be achieved (pressure surpasses the previous statement of the 175 bar limitation to show that pressure is not the limiting agent in reaching criticality in that design). When using a 50 cm reflector size, which also gives the softest neutron spectrum, enrichment can be decreased under the initial 10% as shown in the inventory optimization stage.

Core Size Optimization

The core varied between a three and four meter right circular cylinder. Reasons for varying the size include effects on neutron spectrum and leakage. A larger core (for a given gas pressure and enrichment) features lower leakage, as well as a faster neutron spectrum. The lower leakage is an obvious result of increased core size as an increased core size increases the number of mean free paths in the core, causing more neutrons to interact before escaping into the reflector. Most of the interactions take place from neutrons that have backscattered into the core from the reflector, and those neutrons will

see more mean free paths before being able to pass through the core into the reflector again, thus causing more interactions and a lower leakage. The faster spectrum is a result of more interactions taking place in the core without having seen the reflector region (without having seen the reflector region as often as would occur with a smaller core). Neutrons must pass through more mean free paths to reach the reflector and thermalize, increasing the chance that a fast reaction will occur. A larger core will raise k_{eff}, however the hardened neutron spectrum causes more actinide buildup in the core. Mechanical properties also favor a smaller core, as it is easier to manufacture, and to maintain pressure in compared to a larger core. Since active LWRs today are around the same size as a three-meter right circular cylinder now, production of such a core should pose no problem.

Table 4-1. Keff values for 10 cm reflected core

Fuel *	Enrichment	U *	U235 *	U238 *	F *	P (bar)	K eff	%Rel Error	MFP (cm)
1.42E-04	10.00%	2.84E-05	2.84E-06	2.56E-05	1.14E-04	9.42	0.2133	1.41E-03	1210.5
1.42E-03	10.00%	2.84E-04	2.84E-05	2.56E-04	1.14E-03	94.16	0.4558	2.06E-03	131.2
1.42E-02	10.00%	2.84E-03	2.84E-04	2.56E-03	1.14E-02	941.6	0.7974	1.74E-03	12.899
3.40E-02	10.00%	6.80E-03	6.80E-04	6.12E-03	2.72E-02	2253.9	0.9048	1.21E-03	5.351
4.40E-02	10.00%	8.80E-03	8.80E-04	7.92E-03	3.52E-02	2916.81	0.9227	1.55E-03	4.129
8.40E-02	10.00%	1.68E-02	1.68E-03	1.51E-02	6.72E-02	5568.45	0.9553	1.25E-03	2.159
1.40E-01	10.00%	2.80E-02	2.80E-03	2.52E-02	1.12E-01	9280.74	0.9647	1.03E-03	1.295
2.80E-01	10.00%	5.60E-02	5.60E-03	5.04E-02	2.24E-01	1.86E+04	0.9668	9.90E-04	0.647
5.60E-01	10.00%	1.12E-01	1.12E-02	1.01E-01	4.48E-01	3.71E+04	0.9668	9.90E-04	0.324
1.20E+00	10.00%	2.40E-01	2.40E-02	2.16E-01	9.60E-01	7.95E+04	0.9668	9.90E-04	0.151
1.20E+01	10.00%	2.40E+00	2.40E-01	2.16E+00	9.60E+00	7.95E+05	0.9668	9.90E-04	0.015
1.20E+04	10.00%	2.40E+03	2.40E+02	2.16E+03	9.60E+03	7.95E+08	0.9668	9.90E-04	1.51E-05
1.42E-02	15.00%	2.84E-03	4.26E-04	2.41E-03	1.14E-02	941.6	0.9436	1.20E-03	12.923
1.93E-02	15.00%	3.86E-03	5.79E-04	3.28E-03	1.54E-02	1279.42	0.9996	1.26E-03	9.48
2.00E-02	15.00%	4.00E-03	6.00E-04	3.40E-03	1.60E-02	1325.82	1.0084	1.23E-03	9.145
2.50E-02	15.00%	5.00E-03	7.50E-04	4.25E-03	2.00E-02	1657.28	1.0422	1.36E-03	7.304
3.40E-02	15.00%	6.80E-03	1.02E-03	5.78E-03	2.72E-02	2253.9	1.084	1.48E-03	5.36

* Units of (atoms/barn*cm)

Volume (m^3) 21.21

BeO Thickness(cm) 10

Table 4-2. Keff values for 20 cm reflected core

Fuel *	Enrichment	U *	U235 *	U238 *	F *	P (bar)	K eff	%Rel Error	MFP (cm)
1.42E-04	10%	2.84E-05	2.84E-06	2.56E-05	1.14E-04	9.42	0.61938	2.64E-03	1093.3
1.42E-03	10%	2.84E-04	2.84E-05	2.56E-04	1.14E-03	94.16	0.93954	2.64E-03	29.04
2.80E-03	10%	5.60E-04	5.60E-05	5.04E-04	2.24E-03	185.61	0.95242	2.24E-03	65.74
5.60E-03	10%	1.12E-03	1.12E-04	1.01E-03	4.48E-03	371.23	0.96322	2.47E-03	32.9
1.20E-02	10%	2.40E-03	2.40E-04	2.16E-03	9.60E-03	795.49	0.96661	2.11E-03	15.25
2.40E-02	10%	4.80E-03	4.80E-04	4.32E-03	1.92E-02	1590.98	0.96661	2.11E-03	7.59
4.80E-02	10%	9.60E-03	9.60E-04	8.64E-03	3.84E-02	3181.97	0.96661	2.11E-03	3.79
8.40E-02	10%	1.68E-02	1.68E-03	1.51E-02	6.72E-02	5568.45	0.96661	2.11E-03	2.16
8.40E-01	10%	1.68E-01	1.68E-02	1.51E-01	6.72E-01	5.57E+04	0.96661	2.11E-03	0.22
8.40E+00	10%	1.68E+00	1.68E-01	1.51E+00	6.72E+00	5.57E+05	0.96661	2.11E-03	0.02

* Units of (atoms/barn*cm)

Volume (m^3) 21.21

BeO Thickness(cm) 20

Table 4-3. Keff values for 30 cm reflected core

Fuel *	Enrichment	Core Radius (m)	U *	U235 *	U238 *	F *	P (bar)	K eff	%Rel Error	MFP (cm)
1.42E-04	10.00%	3	2.84E-05	2.84E-06	2.56E-05	1.14E-04	9.42	0.80734	2.15E-03	1041
1.42E-03	10.00%	3	2.84E-04	2.84E-05	2.56E-04	1.14E-03	94.16	1.13973	2.17E-03	126.17
5.00E-04	10.00%	3	1.00E-04	1.00E-05	9.00E-05	4.00E-04	33.15	1.07378	2.41E-03	333.45
2.50E-04	10.00%	3	5.00E-05	5.00E-06	4.50E-05	2.00E-04	16.57	0.96226	2.27E-03	623.16
4.00E-04	10.00%	3	8.00E-05	8.00E-06	7.20E-05	3.20E-04	26.52	1.03446	2.36E-03	432.77
3.25E-04	10.00%	3	6.50E-05	6.50E-06	5.85E-05	2.60E-04	21.54	1.01311	2.52E-03	492.21
3.13E-04	10.00%	3	6.25E-05	6.25E-06	5.63E-05	2.50E-04	20.72	1.0078	2.32E-03	509.8
3.00E-04	10.00%	3	6.00E-05	6.00E-06	5.40E-05	2.40E-04	19.89	1.00045	2.19E-03	529.18
1.80E-03	7.50%	3	3.60E-04	2.70E-05	3.33E-04	1.44E-03	119.32	1.14515	3.16E-03	101.54
2.00E-03	7.50%	4	4.00E-04	3.00E-05	3.70E-04	1.60E-03	132.58	1.11223	2.60E-03	91
2.17E-03	7.50%	4	4.34E-04	3.26E-05	4.01E-04	1.74E-03	143.85	1.07764	4.72E-03	100.64
2.17E-03	6.00%	4	4.34E-04	2.60E-05	4.08E-04	1.74E-03	143.85	1.07764	4.72E-03	69.22
1.80E-03	5.00%	4	3.60E-04	1.80E-05	3.42E-04	1.44E-03	119.32	1.04945	3.63E-03	101.72
2.40E-03	5.00%	4	4.80E-04	2.40E-05	4.56E-04	1.92E-03	159.1	1.04255	2.99E-03	76.64
2.66E-03	5.00%	4	5.32E-04	2.66E-05	5.05E-04	2.13E-03	176.33	1.03093	3.51E-03	69.22

* Units of (atoms/barn*cm)

Table 4-4. Keff values for 40 cm reflected core

Fuel *	Enrichment	U *	U235 *	U238 *	F *	P (bar)	K eff	%Rel Error	MFP (cm)
1.42E-04	10.00%	2.84E-05	2.84E-06	2.56E-05	1.14E-04	9.42	0.93312	2.02E-03	1018.2
2.80E-04	10.00%	5.60E-05	5.60E-06	5.04E-05	2.24E-04	18.56	1.09133	2.04E-03	556.01
2.00E-04	10.00%	4.00E-05	4.00E-06	3.60E-05	1.60E-04	13.26	1.02232	2.45E-03	749.61
1.80E-04	10.00%	3.60E-05	3.60E-06	3.24E-05	1.44E-04	11.93	0.99529	1.88E-03	824.42
1.90E-04	10.00%	3.80E-05	3.80E-06	3.42E-05	1.52E-04	12.6	1.00431	2.36E-03	786.08

* Units of (atoms/barn*cm)

Volume (m^3) 21.21

BeO Thickness(cm) 40

Table 4-5. Keff values for 50 cm reflected core.

Fuel *	Enrichment	U *	U235 *	U238 *	F *	P (bar)	K eff	%Rel Error	MFP (cm)
1.42E-04	10.00%	2.84E-05	2.84E-06	2.56E-05	1.14E-04	9.42	0.986	2.12E-03	1010.8
1.45E-04	10.00%	2.90E-05	2.90E-06	2.61E-05	1.16E-04	9.61	0.9899	2.57E-03	992.09
1.50E-04	10.00%	3.00E-05	3.00E-06	2.70E-05	1.20E-04	9.94	0.9986	2.06E-03	962.65
1.52E-04	10.00%	3.04E-05	3.04E-06	2.74E-05	1.22E-04	10.08	1.0003	2.39E-03	908.47
1.55E-04	10.00%	3.10E-05	3.10E-06	2.79E-05	1.24E-04	10.28	1.0116	2.50E-03	934.71
1.60E-04	10.00%	3.20E-05	3.20E-06	2.88E-05	1.28E-04	10.61	1.018	1.96E-03	951.1
3.60E-04	5.00%	7.20E-05	3.60E-06	6.84E-05	2.88E-04	23.86	1.0207	2.06E-03	460.45

* Units of (atoms/barn*cm)

Volume (m^3) 21.21

BeO Thickness(cm) 50

Feed Fuel and Enrichment Optimization

The defining aspect of the GCR is that it allows online fuel feeding, enabling the reactor operator to very accurately control criticality and actinide production. The feed fuel can vary by enrichment and flow. Adjusting the enrichment affects not only the spectrum and reactivity of the core, but also affects the overall enrichment of the core, which could pose licensing problems. The flow at which feed is introduced into the core affects both the criticality and the spectrum of the core. A higher flow rate will induce a positive reactivity insertion, as fissile fuel will be replaced faster after fissions. The

spectrum is dependent on actinide inventory in the core, and a lower flow rate will affect the ratio of U235 to other actinides in the core at a given time.

The enrichment of the feed in this research was chosen to match the enrichment at which the fuel was depleted. This does not have to be the case; the enrichment can be of any amount as long as the core remains critical. Monteburns outputs how much U235 and U238 are depleted at each step point in the depletion analysis. Using those values to derive an average "depletion enrichment" generates the enrichment used for the feed fuel. One reason for using this "depletion enrichment" method is simplicity; it allows a single enrichment to be used for fuel feeding. In practice, the feed fuel enrichment would vary every time fuel was injected into the core, be it at state points, or continuously. However this research is used mainly for scoping calculations and it is assumed acceptable to use an average enrichment for the feed fuel. Changing the enrichment would allow a tighter control on spectrum and actinide production, although would be much more computationally difficult, and require new methodology to be developed in the area of fuel management. It is possible to use highly enriched fuel, which then requires less fuel to be injected into the core, however the concern then arises with transporting highly enriched fuel, and raising the overall core enrichment to dangerous levels (with respect to licensing), as well as cost increases to generate the fuel. Using low enriched fuel is the safest from a proliferation aspect, and cheapest to produce; however, it may produce a problem in pressure control. Because more U238 is introduced into the core than depleted, the pressure of the core will gradually rise, and if the enrichment of the feed fuel is lowered the rise of pressure becomes more dramatic, and eventually will strain the mechanical limits of the core. Highly (>20% U235 by atom) enriched fuel will also

allow a higher burnup to be achieved however commercially highly enriched fuel is not viable at this time (commercial reactors can not operate with greater than 20% U235 by atom enriched fuel). Therefore using approximately the depletion enrichment limits pressure increase as well as simplifies the fuel management process.

The initial core enrichment is varied between 5 to 10% throughout the research, however this does not drastically affect the fuel cycle as a whole. The reason for this is that no matter the initial enrichment, the feed fuel will vary the overall enrichment. Although current limitations restrict commercial reactors from operating with high-enriched fuel (>20%) it was felt that these reactors should operate with as low enrichment as possible (lower enrichments lead to less proliferation concerns and cost of fuel production). The only reactivity restriction on the fuel is based on the reflector size, for a given reflector size there is a minimum enrichment below which the reactor will not be critical. Therefore ideally the fuel enrichment should be as low as manageable to not only keep costs down, but to be proliferation resistant, and to reside within the bounds of current legislation.

Iterative Optimization of Feed Fuel Enrichment

Unfortunately the optimization of the feed fuel is an iterative process and can only be done by evaluating the change in criticality and actinide inventory of the previous execution. This is to say that an initial guess must be made of the feed fuel amount per step, and then after evaluating the results of that step another input must be determined using the output from that step. This is the drawback in using the Monteburns code package, because each step may take a large amount of CPU time, and if the change in feed fuel does not reach the desired results, the problem must be executed again, requiring more CPU time, until acceptable results are achieved.

When optimizing the feed fuel, a few restrictions must be in place on the design specifications and the output, these are:

- Reactor must remain critical at all step points (it should actually be supercritical at the start of every step to account for burnup between steps). K_{eff} for the core should remain between 1.0 and 1.05, with an error less than 1%.
- The U235 enrichment should not significantly change throughout the life of the core, and should preferably remain as low as possible.
- The pressure of the core should not rise above current mechanical limits (these are assumed to be around 175 bar, ideally however, the pressure will remain below 140 bar).
- All fission products are assumed to be removable throughout the life of the core.

Using these initial guidelines for optimization the iterative process can then begin with the goal of limiting actinide inventory. It should be noted however that Monteburns allows the fuel to be input in either a specified amount of grams per day, or a specified amount in discrete step points. Because this reactor is being operated for 10 years the method chosen was in grams per day, as to do it in step points every two weeks would require too much CPU time. Also of note is that the MCNP mockup of the core does not include the backend cycle, and assumes all fuel in the reactor is residing in the core at all times, this is obviously not the case, and in practice the reactor would require slightly more fuel in it during operation. The fact that in practice the core would require slightly more fuel does not affect the results of this research as the burnup would not significantly change, as the reactor would not require a large amount of fuel to be added to compensate for the fuel in the backend of the cycle. The backend of the cycle itself only requires a small mass inventory and thus is considered negligible for the purposes of scoping calculations.

Important Actinide Inventory

Once a reactor design has been simulated for 10 years while maintaining criticality and operating within all the limits placed upon it by this research the actinide inventory is then evaluated and compared to other GCR designs and a reference LWR design. However a decision must be made as to which actinide s are considered “important”. It is important to limit overall actinide inventory however certain actinides pose a higher threat to the environment than others. The actinide inventories were evaluated according in four different categories, total actinide inventory, and actinides with over 100, 1000, and 10000 year half-lives. The short-lived actinides can be contained while they decay naturally without posing a threat to the environment. It is the actinides with long half lives that pose a threat to long term containment, which is a problem the nuclear industry is currently facing (storing waste for up to 10,000 years). If the long-lived actinides can be reduced, then much less storage space is required for nuclear waste, as the short half live actinides can be stored on-site or in a remote location, but do not require the long-term storage of a facility like Yucca Mountain. Another criteria on which actinides are evaluated is their usefulness, certain actinides can be reused for other purposes, research or commercially, while others may pose a proliferation threat. Based on those criteria each cores final actinide inventory will be evaluated and discussed.

This basis for this research is to determine a general idea of how efficient a GCR can be in the area of actinide inventory, therefore the main concerns of the research is limiting overall actinides. The main criteria in evaluating the actinide inventory is therefore overall mass, while the secondary evaluations are based on mass for specific half-life intervals. A tertiary concern, and one on which the research does not adjust to correct, is the usefulness and safety concerns of the actinides. The next chapter will

discuss the three GCR designs chosen and their relative performance compared to a reference LWR design.

Reference LWR Design

A reference LWR design is used to evaluate the performance of the GCR designs, as the LWR is the only currently operating power reactor. The LWR design features 10.2 g/cc fuel density comprised of UO_2 with 100 MTU operating at 1000ºF. The cladding is zircaloy, and the moderator is light water, with an operational temperature of 588ºF. The radii for the fuel, gap, clad, and moderator are 0.4096 in, 0.4178 in, 0.4750 in, and 0.7094 in respectively. The code used to simulate the fuel cycle was CASMO3. The core was simulated for a burnup of 35 MWD, producing the same total thermal power as the GCR designs studied in this research. Because the goal of the research is to compare how equal power designs (GCR and LWR) compare, electrical efficiency must be taken into account. The electrical efficiency assumed for the LWR is 30%, while the GCR design uses 60% for its efficiency. Because the LWR operates at half the electrical efficiency it must produce twice the total thermal power and therefore the LWR design would require two operating reactors to produce an equivalent power. The actinide inventory of the LWR is therefore doubled to take into account the electrical efficiency.

CHAPTER 5
CORE COMPARISONS

The actinide inventories for the different cores will be evaluated in this chapter according to the guidelines specified in chapter 4. The list of cores evaluated in this chapter is shown in Table 5-1. As shown in Table 5-2, the optimal design is the lower pressure core, which features both the lowest total actinide inventory, as well as lowest inventory of isotopes with long half-lives. The reference LWR core output file does not show inventories for some of the listed isotopes, therefore making it appear to have a lower inventory then an actual LWR core design. Both cores produce approximately the same thermal power (3600 GWD) during operation. The table takes into account electrical efficiencies of 30% for the LWR and 60% for the GCR.

The important actinides are considered to be those with half-lives of greater than 100 years, because these represent the largest concern for the issue of waste storage. The shorter-lived actinides will decay naturally, and do not present problems for long-term storage. The low-pressure core has significantly lower amounts of important actinides at the end of core life when compared to the other gas core designs and the reference LWR. Two factors are important when calculating actinide production and depletion, overall cross section and fission to capture ratio. These two parameters vary as a direct function of neutron energy [8]. When comparing the gas core designs, the spectrum is the most important criteria for determining actinide inventory. The lower pressure core is the most thermal, as the reflector is largest in this design, forcing more neutrons to thermalize and return to the core from the reflector.

Table 5-1. The three analyzed GCR design specifications.

Design	Core 1	Core 2	Core 3	LWR
Height (m)	4	4	3	3.657
Radius (m)	4	4	3	0.014*
Pressure (bar)	143.85	119.32	26.5	138
Enrichment (%U235)	6%	6%	5%	3.20%
Reflector Thickness (cm)	32	32	50	0.145 *
Reflector Material	BeO	BeO	BeO	H2O **
Cycle Length (yrs)	10	10	10	1
Total Electric Power (GWD)	2160	2160	2160	2160
Fuel Density (g/cc)	0.226	0.188	0.042	10.2
Power Density (w/cc)	20.02	19.85	47.56	100
MFP Core (cm)	84.53	101.61	460.17	***
MFP Reflector (cm)	1.46	1.47	1.42	***
Burnup (GWD/MTU)	260	290.1	556	36

* The thickness is only for 1 pin, and not for the core as a whole
** The reflector in the LWR is the water surrounding the pin
*** Reactor geometry is single pin for the LWR, so MFP can not be evaluated for the core/reflector

The lower pressure in the core results in less mean free paths experienced by the neutrons, thus more neutrons reach the reflector, where they will thermalize and return to the core. The smaller size of the lower pressure core causes a similar effect, essentially lessening the number of mean free paths that a neutron needs to travel to go from one side of the core to the other, where it will enter the reflector. The downside to the low-pressure core is that in general the more thermal a spectrum is the lower the fission to capture ratio is for the actinides. The fission to capture ratio for U235 for the low-pressure core, is 20% greater than for the other two GCR designs, and as most of the interactions occur in U235, fewer actinides are produced initially. This initial decrease in actinide inventory causes an avalanche effect resulting in a decrease of total actinide inventory despite the more thermal spectrum.

Specific isotopes may be present in a greater concentration in the low-pressure core as the thermal spectrum is not as conducive for the removal of those isotopes as it is in other core designs. Such an example is U237, much less U237 is produced from capture

in the low-pressure core however the cross-section ratios (fission to capture) are higher in the harder spectrum cores, and thus remove much more than the low pressure core does. The net effect is to see a higher concentration of U237 in the low-pressure core. A similar effect occurs in the low-pressure core as well for Cm246, Cm247, and Cm248. For those three isotopes, the fission to capture ratio for the high-pressure cores is much higher compared to the more thermal low-pressure core, this increase in fission to capture ratio causes more of those actinides to be fissioned in the high-pressure core, whereas in the low-pressure the dominant effect is the transmutation of those isotopes into heavier actinides.

During the initial design of the GCR for this research it was assumed that the optimum design would also contain the highest conversion ratio (fissile material produced compared to fissile material depleted). The logic behind this is that if the core is producing and then burning more plutonium, the same effect should be seen on the higher actinides. Moreover at higher energies the fission to capture for most actinides favors fission, however, it was shown that this is not the determining factor for actinide inventory. In this study the low-pressure core produces 50% of the total actinides compared to the high-pressure core. The cores were then modified to try and obtain a very thermal spectrum, as the high-pressure core's spectrum was optimal for minimizing actinide inventory.

As shown the reference LWR design fares poorly when comparing it to the GCR designs. Although some of the isotopes are not listed in the output (certain isotopes are not explicitly listed in the output however their aggregate mass is included), the core still possesses a higher actinide concentration at the end of life. The LWR design has an

increase in total actinide inventory of 342% over the worst of the GCR designs. When comparing to the optimum GCR design the actinide inventory of the LWR is 703% of the GCR's (This does not include unspent fuel, the inclusion of the unspent uranium increases the factor from 704% to almost 9,800%).

Table 5-2. Core isotopic abundance in Kg.

Isotope	Core 1 (Kg)	Core 2 (Kg)	Core 3 (Kg)	LWR (Kg)
U234	1.4	1.32	0.25	28.43
U236	518	510	367	818.24
U237	0.51	0.56	0.94	*
Np237	39.5	36.1	14.7	102.2
Pu238	43.9	43.2	19.8	38.95
Pu239	130	95.9	9.57	1339.59
Pu240	120	95	13.3	514.91
Pu241	46.2	37.5	5.08	336.78
Pu242	41.8	39.7	19.2	108.7
Am241	1.67	1.13	0.03	9.11
Am242	1.81E-01	1.02E-01	6.62E-04	0.14
Am243	3.18	3.32	1.65	24.8
Cm242	0.68	0.58	0.09	3.96
Cm243	4.22E-02	3.63E-02	4.39E-03	*
Cm244	21.6	24.2	17.8	7.47
Cm245	3.13	3.08	1.15	*
Cm246	1.11	1.45	3.07	*
Cm247	0.08	0.11	0.19	*
Cm248	0.02	0.04	0.32	*
$T_{1/2}$ > 100a	860.07	787.25	430.43	2946.12
$T_{1/2}$ >1000a	858.4	786.12	430.41	2937.01
$T_{1/2}$ >1E4a	730.8	683.17	411.24	2397.16
Total	973	893.32	474.14	3333.29

*Not explicitly given in Casmo output.
The $T_{1/2}$ designator represents total actinides with half life greater than that listed.

The cause for the increase in actinide inventory is because the fuel in the LWR does not experience the same burnup that the GCR fuel does. The burnup required for the same total power output in the LWR is 35 GWD/MTU, compared to a burnup of 560 GWD/MTU for the gas core reactor (design 3), over an order of magnitude lower.

Essentially this means that the fuel in the LWR does not remain in the core long enough to burn the actinides that are being produced. Typical LWR fuel will remain in the core until a burnup of around 40 MWD/KgU, which means conventional LWR's are getting an order of magnitude less power from the same amount of fuel used in a GCR. This translates to excess waste being produced by LWR's from unspent fuel. It should be noted that the maximum possible power for uranium (assuming 200 MeV per fission) is about 950 MWD/kgU, the GCR design is operating at 59% of that upper limit, whereas the LWR is operating typically at around 4.2% of that limit.

Actinide Inventory Comparison

The GCR's ability to preferentially produce or burn, actinides is also shown in Table 5-2. The Pu238 inventory for the high-pressure (harder spectrum) design is 2.22 times that of the low-pressure more thermal spectrum core. The Pu240 inventory for the high-pressure core is 9.02 times that of the low-pressure core and Am242 is increased by a factor of 273.4 when going from low to high pressure. This gives GCRs the ability to preferentially produce particular isotopes that could then be potentially used for other research or space power as is the case for Pu238. The ability also allows the GCR to preferentially burn dangerous actinides, thus making it proliferation resistant. The ratios of Pu238, Pu240, and Pu242, to overall plutonium inventory determine how proliferation resistant a core will be, as those isotopes are difficult to separate from the useful isotopes of plutonium. Those three isotopes have very low fission cross-sections, and when present in high concentrations make weapon production unfeasible. The ratios for those plutonium isotopes to the overall plutonium inventories are 53.9%, 57.1%, and 78.1% for the high, intermediate, and low-pressure cores respectively (meaning the best GCR design possesses 21.9% usable plutonium for weapons purposes). The LWR design has a

ratio of only 28.3%, which makes it much more dangerous from a proliferation safety standpoint. The other inherent proliferation resistant feature the GCR possesses is the fuel is gaseous, and the mass inventory is very low. A typical LWR will contain up to 100 MTU in the core at any given time, whereas the GCR contains between 600 and 4000 kilograms at any given time, depending on which core design as well as how far into core life the inventory is measured. Through the 10 year cycle studied the low-pressure core only contained a maximum of 6.51 MTU (time integrated). The mass of plutonium in a typical LWR is around 2300 Kg, 71.7% of which is useful in weapons productions, whereas the low-pressure GCR design only contains 67.0 kg, only 21.9% of which is useful in weapons productions. It should also be noted that other weapons grade materials such as Am242 can be preferentially depleted by slightly changing the spectrum. As shown in Table 5-2 the high-pressure core has over a two order of magnitude increase in the amount of Am242 at the end of core life. A probable reason for this phenomenon is a large decrease in the absorption cross-section in the epithermal range, so when the spectrum is hardened in the high-pressure core, the cross section decreases by approximately two orders of magnitude.

Evaluation of Integrated Cycle Performance

To demonstrate the effects of the GCR's continuous fuel feeding ability, two designs were analyzed to compare actinide buildup. The first case is the standard core used in this research, where the fuel is allowed to burn continuously to the end of core life (feeding fuel online as the reactor operates), and the second is burning the fuel for one year, and then replacing it with fresh fuel (similar to a LWR).

Table 5-3. Integrated cycle performance comparison, representing the benefits of switching to online fuel feeding, as opposed to regular fuel reload dates.

Isotope	Continuous Burn (Kg)	Multiple Restart (Kg)
U234	0.25	0.07
U236	367	624
U237	0.94	4.72
Np237	14.7	21.5
Pu238	19.8	18
Pu239	9.57	18.2
Pu240	13.3	26
Pu241	5.08	10.2
Pu242	19.2	36.5
Am241	0.03	0.01
Am242	6.62E-04	1.21E-04
Am243	1.65	7.01
Cm242	0.09	0.13
Cm243	4.39E-03	3.91E-03
Cm244	17.8	23.8
Cm245	1.15	1.26
Cm246	3.07	1.22
Cm247	0.19	0.03
Cm248	0.32	0.01
$T_{1/2}$ > 100a	430.43	735.82
$T_{1/2}$ >1000a	430.41	735.8
$T_{1/2}$ >1E4a	411.24	700.31
Total	474.14	2964.59

A comparison between the integrated fuel cycle designs is shown in Table 5-3. There is a dramatic increase in actinides produced when the fuel is replaced every year. The reason for this increase is that the actinides produced in the initial year are removed from the core and are not allowed to approach equilibrium concentration. For the majority of actinides there is an equilibrium state, where the actinides are being burned as fast as they are produced. The GCR design retains the actinides during the life of the core and allows them to approach that equilibrium state. As the actinides approach the equilibrium point, the net gain of each particular actinide is decreased over the same

change in time, as the ratio of actinide production to depletion approaches unity. Heavier actinides are produced in greater quantities in the continuous burn design, as opposed to the multiple burns, because in a multiple burn scenario actinides are removed before they are able to transmute in significant quantities. Although the continuous burn design produces more curium than a multiple burn scenario, the overall actinide production (total as well as those with long half-lives) is lower for a continuous burn design, meaning it is more effective at limiting overall actinide (as well as dangerous actinide) production.

The effects of the integrated fuel cycle on actinide production can also be seen in figures 5-1 through 5-6, which show the actinide buildup as a function of time for both the major and minor actinides. In the high and intermediate-pressure cores Pu239, Pu240, and Pu241 approach the equilibrium concentration, whereas in the low-pressure core Pu241 and Am243 are approaching their equilibrium point. The reason for the difference in actinides approaching their equilibrium point is the changes in spectrum between the core designs. The harder spectrum core will burn plutonium compared to uranium faster than the more thermal design, thus the trend for those isotopes to "peak out" in the intermediate and high-pressure cores. In the low-pressure core the higher actinides (americium and curium) are shown to approach equilibrium, except for Cm246 and Cm248, because of the nature of the cross-sections in the epithermal regime. The higher-pressure cores see an increase in actinides such as Am242 because there is a large drop in the interaction cross section as the spectrum becomes hardens for those actinides therefore actinides like Am242 are produced at a high rate, while maintaining a very low depletion rate.

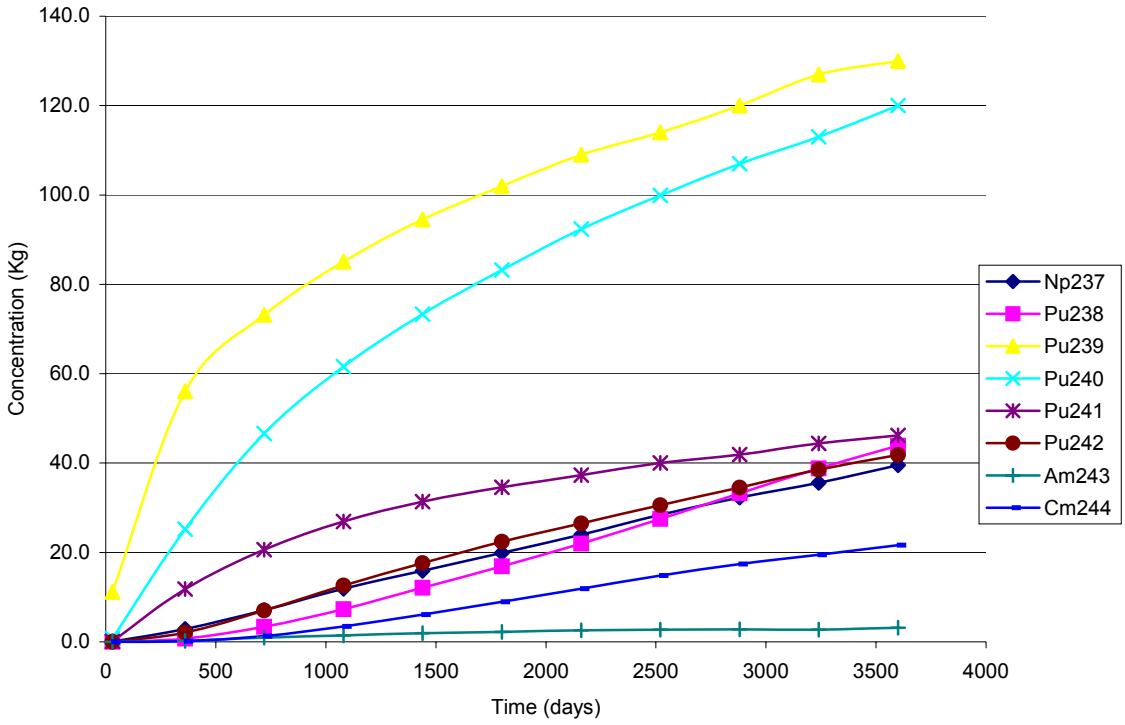

Figure 5-1. Mass of the major actinides present in the high-pressure core (Core 1) as a function of time.

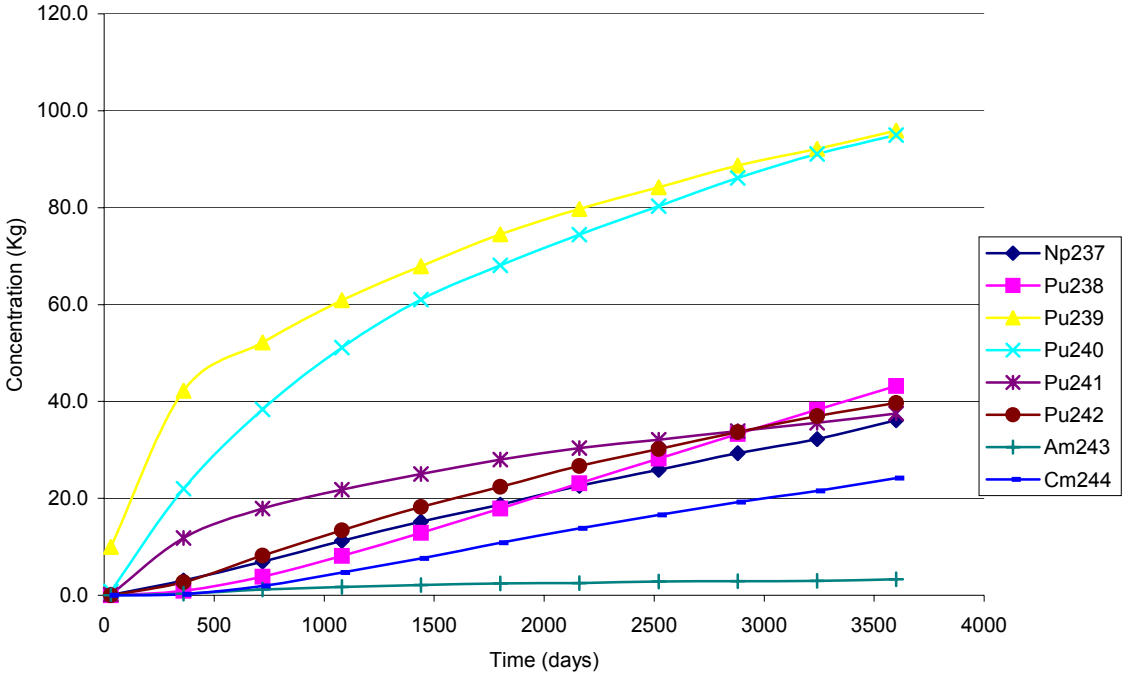

Figure 5-2. Mass of the major actinides present in the intermediate-pressure core (Core 2) as a function of time.

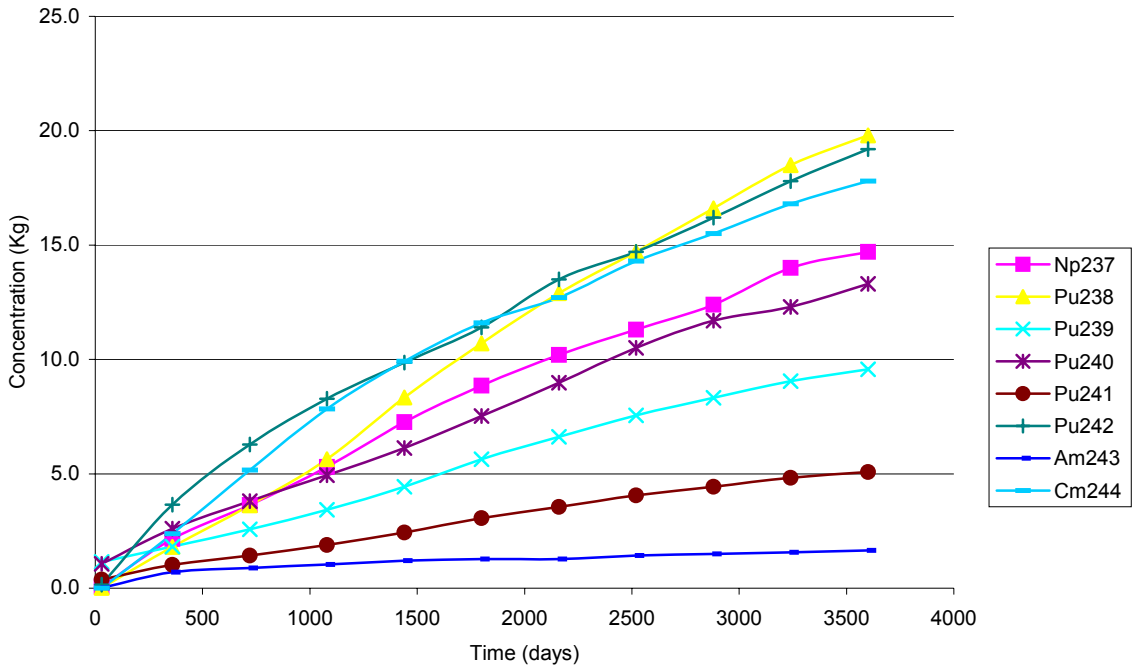

Figure 5-3. Mass of the major actinides present in the low-pressure core (Core 3) as a function of time.

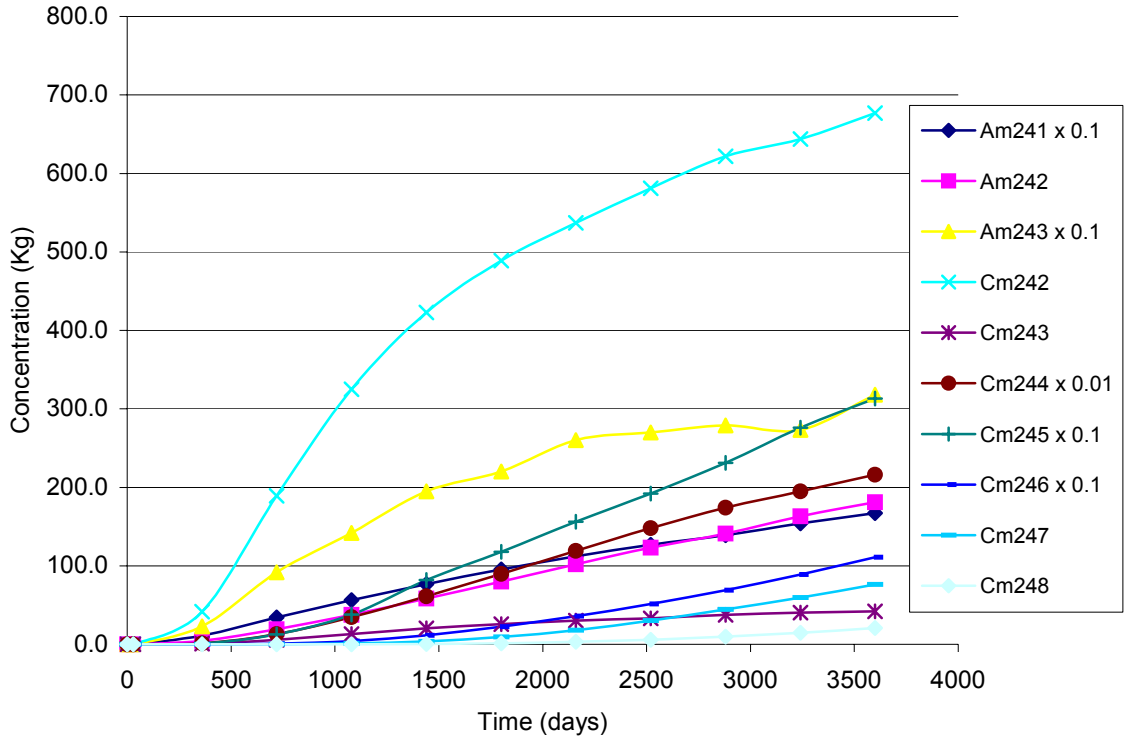

Figure 5-4. Mass abundance of minor actinides present in the high-pressure (Core 1) core as a function of time.

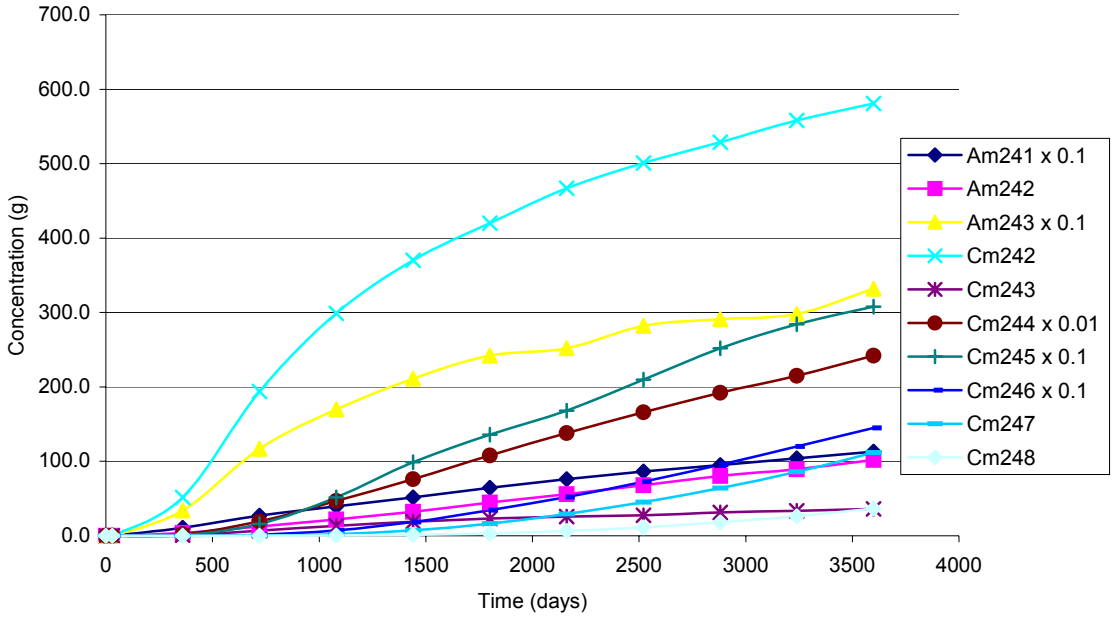

Figure 5-5. Mass abundance of the minor actinides in the intermediate-pressure core (Core 2) as a function of time.

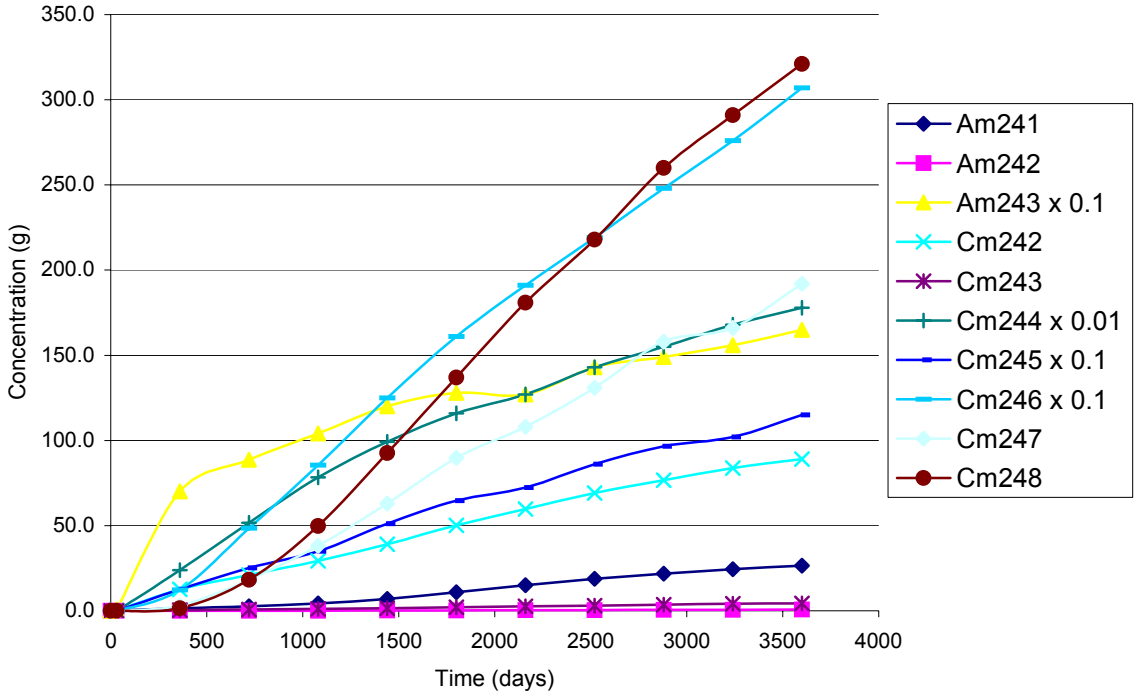

Figure 5-6. Mass abundance of the minor actinides in the low-pressure core (Core 3) as a function of time.

Flux Calculations

The second part of the core evaluation is demonstrating a difference in the spectrum for the GCRs, and to evaluate this effect on the overall actinide inventory. The limitation for this part of the research is that the spectrum can only be analyzed at specific state-points, and therefore only represents accurately the spectrum at a given time and material composition. However this is assumed to be acceptable because each core is compared at the same time (in terms of days as well as energy). The comparisons are also only made in between GCRs, so using state-points to describe the flux is acceptable as the relative values at state-points can give insight as to the effects on actinide production through changes in average neutron energy.

The code package MCNP was also used to determine flux values for the core and reflector for the design cases. The flux was separated into 648 groups, covering the total energy spectrum. The group structure was chosen by evaluating the ACE files containing the MCNP library cross section values, and determining an appropriate number of points per energy decade. The points per decade were scaled according to the ACE file format for the Be library, however the lower and upper energy bins were kept unchanged (i.e. for energies less than 1E-08 MeV, the bin structure was kept constant as there were few bins in the original ACE file). The flux in the thermal region is fairly smooth and does not require a large number of points to describe the shape, while the flux in the resonance region is rapidly changing and requires more grid points to accurately show the flux. The reason for shrinking the number of energy grid points is to decrease CPU time required to obtain results with sufficient accuracy and precision. Flux calculations were done for the core (the core was separated into five cells to evaluate the spatial flux) and the reflector using a cell flux tally in MCNP (f4:n tally) to obtain a cell averaged flux. Fluxes were

then used to calculate average energies for the cores. Average neutron energy was calculated by weighting the flux with the energy for energies under 4 eV. The cutoff of 4ev was chosen based on thermal scattering for the operational temperature. The average neutron energy for the high-pressure core is 0.49 eV, while the average neutron energy for the low-pressure core is 0.293 eV. The disparity in energies shows the ease of which the GCR can manipulate the neutron spectrum.

As shown in Figure 5-7 the low-pressure core exhibits a more thermal spectrum, which of course has a lower fission to capture ratio for the actinides. However, U235 had a higher fission to capture ratio for this spectrum compared to the harder neutron spectrum core, thus causing less actinides to be produced from uranium. Once actinides are produced, a harder spectrum is favorable to deplete them, however as shown in Table 5-2, the lower pressure core has a lower actinide inventory. The reason is the initial lack of production in actinides from the uranium. Although the high-pressure core is better at depleting actinides, the low-pressure core produces less, which appears to be the dominant factor shown by this research.

The flux plots for the reflector region are similar, however the low-pressure core again exhibits a more thermal spectrum as shown in Figure 5-8. The more thermal spectrum results in less leakage from the reflector, thus lowering radiation damage to any containment vessels and the surrounding environment. As the reactor will be remotely operated this decrease in leakage will not drastically affect worker shielding (they will be sealed off from the reactor so that almost no dose will reach them, regardless of leakage).

The spatial fluxes in the high and low-pressure cores are shown in Figure 5-9. and Figure 5-10. respectively. . The core is broken up into 4 cells each with a different

thickness. The first cell (Cell 1) represents the volume encompassed by the outer 2cm of thickness (radial and axial, the volumes are all concentric cylinders). The second represents the next 2 cm of thickness, the third is the next 26 cm, and the fourth is the next 60 cm. The high-pressure core features varying neutron energy plots for the different cells, while the low-pressure core's cells all exhibit almost the same properties. Because the average number of mean-free-paths in the low-pressure core is greater than the high-pressure core by a factor of 6 more of the thermal neutrons will travel into the central regions of the core and have their interactions there. Thermal neutrons entering the high-pressure core must travel through more MFPs to reach the central region without undergoing fission, and therefore the thermal flux decreases towards the center of the core.

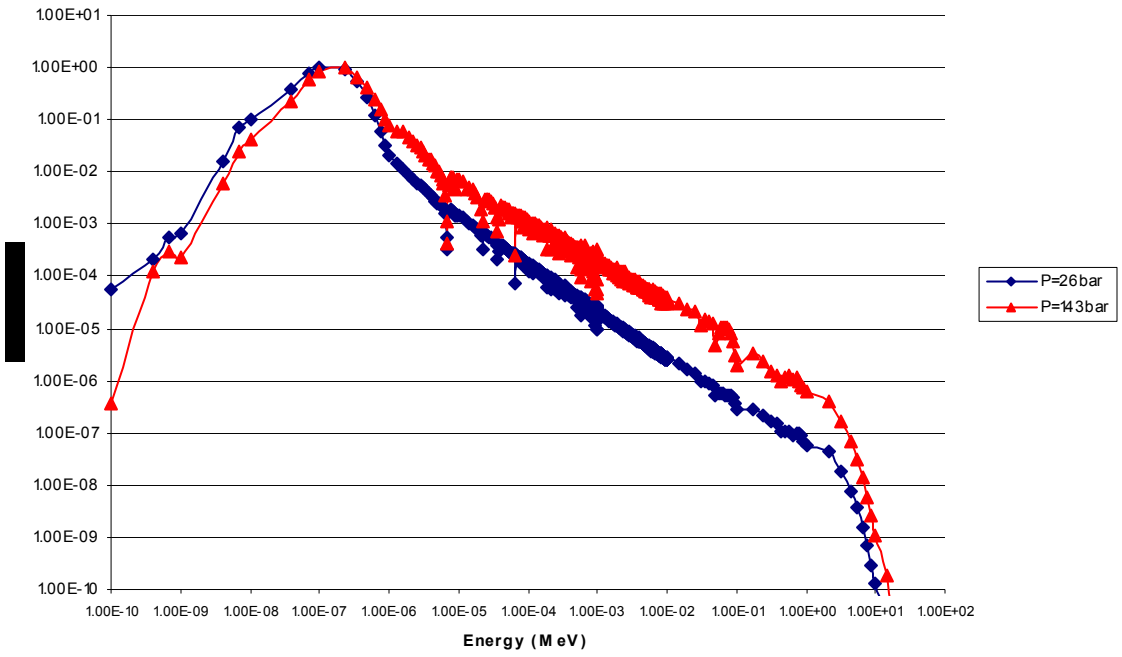

Figure 5-7. Relative flux plots for the low and high-pressure cores. Fluxes were normalized by maintaining the maximum value of 1.0.

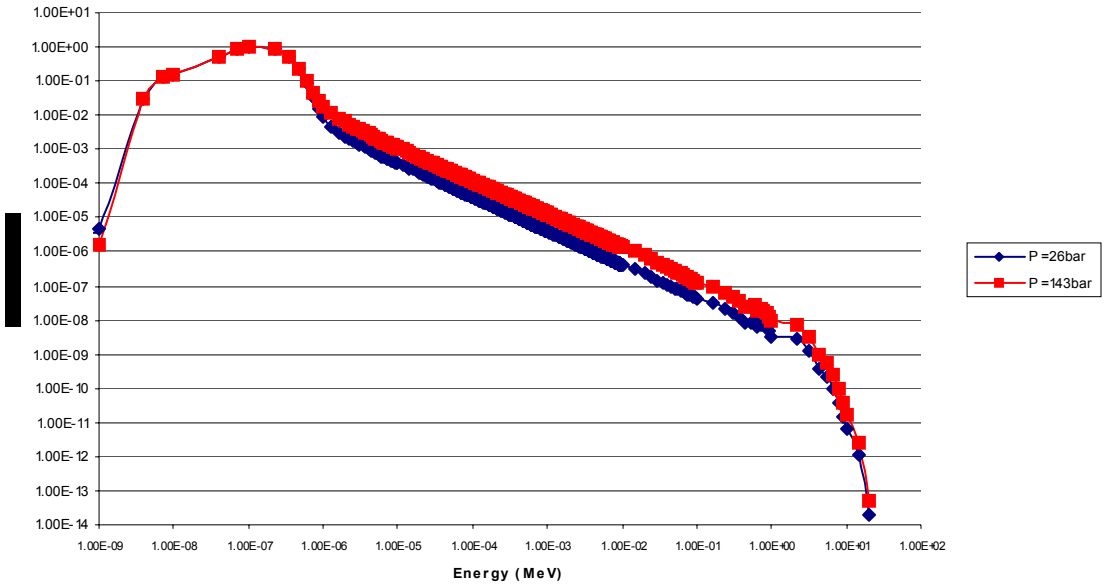

Figure 5-8. Relative flux plots for the low and high-pressure core reflector regions. Fluxes were normalized by maintaining the maximum value of 1.0.

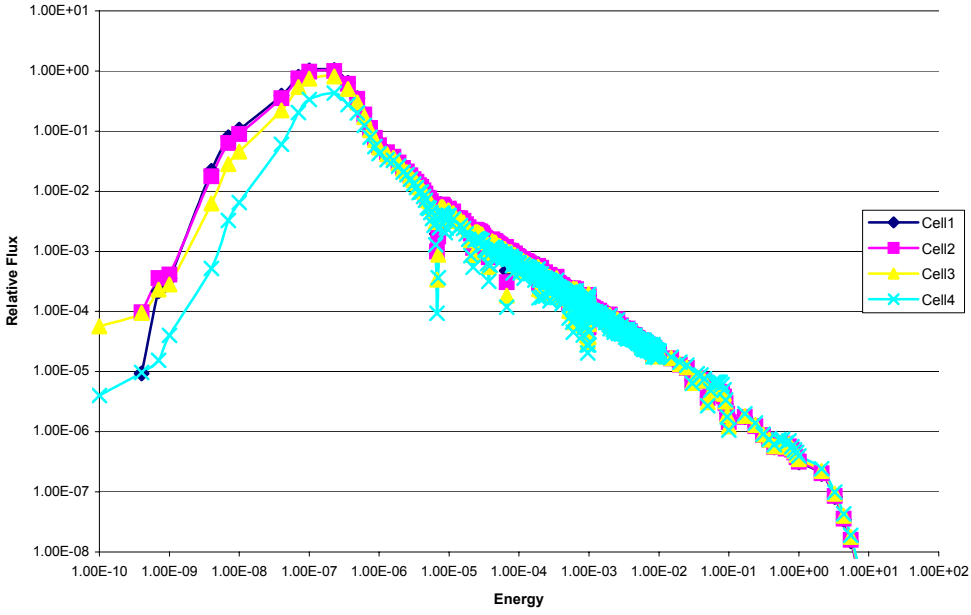

Figure 5-9. Spatial flux plots for the high-pressure core, split into different volumes.

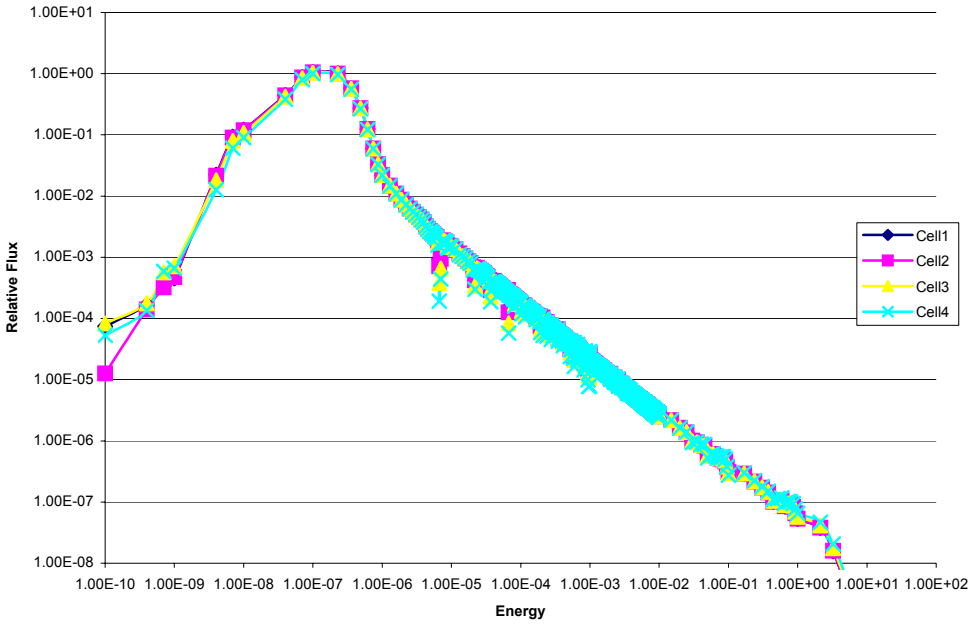

Figure 5-10. Spatial flux plots for the low-pressure core, split into different volumes.

CHAPTER 6
RESULTS AND RECOMMENDATIONS FOR FUTURE RESEARCH

Introduction

This chapter will consist of a concise summary of the results obtained in chapter 5 on comparing the GCR design to a typical LWR design, for the purposes of reducing overall actinide inventory. The three different GCR design comparisons will also be made in this chapter, as well as recommendations for future work in the area of gas core design.

Summary of Results

Results obtained from the Monteburns code package indicate that unlike the initial assumption, a higher energy neutron spectrum will not lead to the most efficient way of reducing actinide inventory. In terms of overall actinide production, the fastest neutron spectrum produces slightly over twice as much waste. Results on core comparison can be summarized as follows:

- K_{eff} was maintained within 5% of criticality at all times (with under 1% relative error). A varying K_{eff} does not drastically affect actinide inventory, as it is not directly related to K_{eff}.
- Reflector size can not be decreased to less than approximately 30cm when using enrichments of less than 10% U235. However the optimum reflector size for this design is around 50cm, past that size no additional reactivity is gained.
- Enrichment can be lowered to current operational limits (5% U235) and maintain criticality.
- Pressure can be maintained well within current mechanical limits ($p<175$bar), and in fact is beneficial to actinide depletion to maintain a low pressure.

- Core size can be kept within current mechanical limitations, and in fact is beneficial to use a smaller core.

- The optimum spectrum for limiting actinide inventory is a more thermal spectrum, resulting in half the waste produced compared to a faster neutron spectrum core.

- Overall fuel inventory (inclusion of all actinides and fuel) is approximately 98 times greater in an LWR than a similar total power GCR (total electrical power produced). Accordingly almost 2 orders of magnitude less waste is produced by the GCR.

- Compared to LWRs the GCR design limits actinide production by a factor of 7.03.
- Weapons grade plutonium in the GCR design is only 28.3% by atom of total plutonium inventory, compared to 78.1% by atom of an LWR design.

- Fuel efficiency is approximately 15 times greater in a GCR than in an LWR based on burnup attainable. Burnup in a LWR is approximately 40GWD/MTU, whereas the GCR can surpass 600GWD/MTU (4% and 62% of theoretical limits respectively).

The most noticeable result should be the decrease in overall waste produced. Decreasing the total waste by almost two orders of magnitude is a significant change in waste material. The lower waste material produced results in significantly less storage required, which is a problem current power reactors are facing. Spent fuel casks for waste products are currently very expensive and not economically efficient. Even if Yucca Mountain does go into operation it is predicted that it will only delay the spent fuel problem for a few years. The GCR design effectively decreases the danger from waste products as most of the waste products can be allowed to decay naturally. The half-lives of U235 and U238 are in the orders of billions of years, and per reactor approximately 90MTU is unused at the end of a life cycle (life cycle meaning if the core is a three batch cycle, then after the three batches are finished). The GCR effectively depletes almost all of its U235 and U238 content while in operation. The GCR waste product could then be solidified chemically and placed into storage the same as current LWR fuel is, except

requiring one percent of the space used by an LWR for the same electrical power. Also many of the waste products could easily be separated and put to use in other areas of research, such as Pu238 for space thermoelectric propulsion systems.

This research has shown that the GCR has the potential to set the upper limits in performance in the areas of burnup and waste production. The GCR concept can be considered a viable alternative to LWRs when dealing with fuel management and waste products, as well as electrical efficiency and proliferation resistance.

Future work should include design parameters that were left out in this research, as well as other code packages to benchmark results. Parameters that were left out include material designs (liners and extra coolant materials), secondary side considerations (fission product removal, flow rates, and online fuel feeding limitations), as well as radiation damage to the core and mechanical parts. Currently MCNP is the only package that uses a continuous energy spectrum to calculate interactions and fluxes, however it should be acceptable to benchmark the code with a transport or diffusion package.

Recommendations for future ideas in this area of research include studying continuous online fuel feeding designs, in this research it was limited to set intervals. Continuous online fueling may turn out to not be a viable or workable method, however it allows for higher burnups since the core can be more tightly controlled. More advanced spectrum analysis would also be useful in determining the effectiveness of the GCR's ability to limit actinide inventories. The neutron energy spectrum is the fundamental property affecting all waste production and depletion, and should therefore be studied in depth. Cost estimate analysis should also be done on the GCR, to evaluate its viability in the nuclear power production field.

Controlling the fuel mass flow rate and its subsequent effects on spectrum and criticality should be studied. The code packages used in this research only allow for steady state analysis as far as fuel location is concerned (the fuel can not be simulated as flowing). As the mass flow rate can affect critically it may be shown that rotating BeO columns are not necessary and the geometry used in this research (concentric cylinders) could be used. This research did not study at all the effects of circulating fuel, and that facet of GCR design needs to be more intricately examined, as it has a direct affect on the viability of the system.

The last parameter area of future work should be done in the area of optimizing enrichments. For the purpose of simplicity certain enrichments were not varied (the fuel feed enrichment was kept the same as the average fuel depletion enrichment), and for the purpose of legality the limits on enrichment were kept to within current standards. This severely limits the effectiveness of the GCR, as increasing enrichment in a GCR does not result in the same increase of proliferation danger as in a solid fuel reactor. Because the fuel is gaseous and in exists in such low quantities compared to a solid fuel reactor, there is no inherent proliferation danger from increasing the enrichment past five percent. Increasing fuel enrichment is believed to decrease the plutonium production from U238 captures, therefore increasing the fission to capture ratio of the fuel. Increasing the ratio of fissions to captures inevitably decreases the amount of waste products produced, thus lowering the total actinide inventory.

APPENDIX
SAMPLE INPUTS

A sample input for each file used in Monteburns will be presented along with a sample MCNP input used for tallying the flux. A brief paragraph will accompany the input explaining its purpose.

Sample MCNP Input

The following input is for the MCNP flux calculation. The input is also used in Monteburns, however the F14, F24, and E0 card would be removed for the Monteburns execution. The number of histories would also be much smaller for a Monteburns execution. The number of histories is large in this input because the small energy bins require a large number of histories to obtain reasonable statistics.

```
Gas Core Design SAMPLE MCNP INPUT – FLUX TALLY INCLUSION
1   1 3.60E-04 -1            $Core
2   2 -3.01 1 -2           $BeO Reflector
3   0 2                  $Rest of world

1 RCC 0 0 50 0 0 300 150     $Core
2 RCC 0 0 0 0  0 400 200     $BeO reflector

mode    n
imp:n   1 1 0
kcode   100000 1 10 100    $Large histories for flux tally
ksrc    0 0 150
f14:n 1
f24:n 2
E0       1e-10       4e-10       7e-10       1e-09       4e-09
         7e-09       1e-08       4e-08       7e-08       1e-07
         2.2857e-07  3.5714e-07  4.8571e-07  6.1429e-07  7.4286e-07
         8.7143e-07  1e-06       1.3e-06     1.6e-06     1.9e-06
         2.2e-06     2.5e-06     2.8e-06     3.1e-06     3.4e-06
         3.7e-06     4e-06       4.3e-06     4.6e-06     4.9e-06
         5.2e-06     5.5e-06     5.8e-06     6.1e-06     6.4e-06
         6.7e-06     7e-06       7.3e-06     7.6e-06     7.9e-06
         8.2e-06     8.5e-06     8.8e-06     9.1e-06     9.4e-06
         9.7e-06     1e-05       1.15e-05    1.3e-05     1.45e-05
         1.6e-05     1.75e-05    1.9e-05     2.05e-05    2.2e-05
```

2.35e-05 2.5e-05 2.65e-05 2.8e-05 2.95e-05
3.1e-05 3.25e-05 3.4e-05 3.55e-05 3.7e-05
3.85e-05 4e-05 4.15e-05 4.3e-05 4.45e-05
4.6e-05 4.75e-05 4.9e-05 5.05e-05 5.2e-05
5.35e-05 5.5e-05 5.65e-05 5.8e-05 5.95e-05
6.1e-05 6.25e-05 6.4e-05 6.55e-05 6.7e-05
6.85e-05 7e-05 7.15e-05 7.3e-05 7.45e-05
7.6e-05 7.75e-05 7.9e-05 8.05e-05 8.2e-05
8.35e-05 8.5e-05 8.65e-05 8.8e-05 8.95e-05
9.1e-05 9.25e-05 9.4e-05 9.55e-05 9.7e-05
9.85e-05 0.0001 0.0001036 0.0001072 0.0001108
0.0001144 0.000118 0.0001216 0.0001252 0.0001288
0.0001324 0.000136 0.0001396 0.0001432 0.0001468
0.0001504 0.000154 0.0001576 0.0001612 0.0001648
0.0001684 0.000172 0.0001756 0.0001792 0.0001828
0.0001864 0.00019 0.0001936 0.0001972 0.0002008
0.0002044 0.000208 0.0002116 0.0002152 0.0002188
0.0002224 0.000226 0.0002296 0.0002332 0.0002368
0.0002404 0.000244 0.0002476 0.0002512 0.0002548
0.0002584 0.000262 0.0002656 0.0002692 0.0002728
0.0002764 0.00028 0.0002836 0.0002872 0.0002908
0.0002944 0.000298 0.0003016 0.0003052 0.0003088
0.0003124 0.000316 0.0003196 0.0003232 0.0003268
0.0003304 0.000334 0.0003376 0.0003412 0.0003448
0.0003484 0.000352 0.0003556 0.0003592 0.0003628
0.0003664 0.00037 0.0003736 0.0003772 0.0003808
0.0003844 0.000388 0.0003916 0.0003952 0.0003988
0.0004024 0.000406 0.0004096 0.0004132 0.0004168
0.0004204 0.000424 0.0004276 0.0004312 0.0004348
0.0004384 0.000442 0.0004456 0.0004492 0.0004528
0.0004564 0.00046 0.0004636 0.0004672 0.0004708
0.0004744 0.000478 0.0004816 0.0004852 0.0004888
0.0004924 0.000496 0.0004996 0.0005032 0.0005068
0.0005104 0.000514 0.0005176 0.0005212 0.0005248
0.0005284 0.000532 0.0005356 0.0005392 0.0005428
0.0005464 0.00055 0.0005536 0.0005572 0.0005608
0.0005644 0.000568 0.0005716 0.0005752 0.0005788
0.0005824 0.000586 0.0005896 0.0005932 0.0005968
0.0006004 0.000604 0.0006076 0.0006112 0.0006148
0.0006184 0.000622 0.0006256 0.0006292 0.0006328
0.0006364 0.00064 0.0006436 0.0006472 0.0006508
0.0006544 0.000658 0.0006616 0.0006652 0.0006688
0.0006724 0.000676 0.0006796 0.0006832 0.0006868
0.0006904 0.000694 0.0006976 0.0007012 0.0007048
0.0007084 0.000712 0.0007156 0.0007192 0.0007228
0.0007264 0.00073 0.0007336 0.0007372 0.0007408
0.0007444 0.000748 0.0007516 0.0007552 0.0007588
0.0007624 0.000766 0.0007696 0.0007732 0.0007768
0.0007804 0.000784 0.0007876 0.0007912 0.0007948
0.0007984 0.000802 0.0008056 0.0008092 0.0008128
0.0008164 0.00082 0.0008236 0.0008272 0.0008308
0.0008344 0.000838 0.0008416 0.0008452 0.0008488
0.0008524 0.000856 0.0008596 0.0008632 0.0008668
0.0008704 0.000874 0.0008776 0.0008812 0.0008848
0.0008884 0.000892 0.0008956 0.0008992 0.0009028
0.0009064 0.00091 0.0009136 0.0009172 0.0009208

0.0009244 0.000928 0.0009316 0.0009352 0.0009388
0.0009424 0.000946 0.0009496 0.0009532 0.0009568
0.0009604 0.000964 0.0009676 0.0009712 0.0009748
0.0009784 0.000982 0.0009856 0.0009892 0.0009928
0.0009964 0.001 0.001036 0.001072 0.001108
0.001144 0.00118 0.001216 0.001252 0.001288
0.001324 0.00136 0.001396 0.001432 0.001468
0.001504 0.00154 0.001576 0.001612 0.001648
0.001684 0.00172 0.001756 0.001792 0.001828
0.001864 0.0019 0.001936 0.001972 0.002008
0.002044 0.00208 0.002116 0.002152 0.002188
0.002224 0.00226 0.002296 0.002332 0.002368
0.002404 0.00244 0.002476 0.002512 0.002548
0.002584 0.00262 0.002656 0.002692 0.002728
0.002764 0.0028 0.002836 0.002872 0.002908
0.002944 0.00298 0.003016 0.003052 0.003088
0.003124 0.00316 0.003196 0.003232 0.003268
0.003304 0.00334 0.003376 0.003412 0.003448
0.003484 0.00352 0.003556 0.003592 0.003628
0.003664 0.0037 0.003736 0.003772 0.003808
0.003844 0.00388 0.003916 0.003952 0.003988
0.004024 0.00406 0.004096 0.004132 0.004168
0.004204 0.00424 0.004276 0.004312 0.004348
0.004384 0.00442 0.004456 0.004492 0.004528
0.004564 0.0046 0.004636 0.004672 0.004708
0.004744 0.00478 0.004816 0.004852 0.004888
0.004924 0.00496 0.004996 0.005032 0.005068
0.005104 0.00514 0.005176 0.005212 0.005248
0.005284 0.00532 0.005356 0.005392 0.005428
0.005464 0.0055 0.005536 0.005572 0.005608
0.005644 0.00568 0.005716 0.005752 0.005788
0.005824 0.00586 0.005896 0.005932 0.005968
0.006004 0.00604 0.006076 0.006112 0.006148
0.006184 0.00622 0.006256 0.006292 0.006328
0.006364 0.0064 0.006436 0.006472 0.006508
0.006544 0.00658 0.006616 0.006652 0.006688
0.006724 0.00676 0.006796 0.006832 0.006868
0.006904 0.00694 0.006976 0.007012 0.007048
0.007084 0.00712 0.007156 0.007192 0.007228
0.007264 0.0073 0.007336 0.007372 0.007408
0.007444 0.00748 0.007516 0.007552 0.007588
0.007624 0.00766 0.007696 0.007732 0.007768
0.007804 0.00784 0.007876 0.007912 0.007948
0.007984 0.00802 0.008056 0.008092 0.008128
0.008164 0.0082 0.008236 0.008272 0.008308
0.008344 0.00838 0.008416 0.008452 0.008488
0.008524 0.00856 0.008596 0.008632 0.008668
0.008704 0.00874 0.008776 0.008812 0.008848
0.008884 0.00892 0.008956 0.008992 0.009028
0.009064 0.0091 0.009136 0.009172 0.009208
0.009244 0.00928 0.009316 0.009352 0.009388
0.009424 0.00946 0.009496 0.009532 0.009568
0.009604 0.00964 0.009676 0.009712 0.009748
0.009784 0.00982 0.009856 0.009892 0.009928
0.009964 0.01 0.015 0.02 0.025
0.03 0.035 0.04 0.045 0.05

```
        0.055       0.06        0.065       0.07        0.075
        0.08        0.085       0.09        0.095       0.1
        0.16923     0.23846     0.30769     0.37692     0.44615
        0.51538     0.58462     0.65385     0.72308     0.79231
        0.86154     0.93077     1           2.125       3.25
        4.375       5.5         6.625       7.75        8.875
        10          15          20
m1    9019.60c .8 92235.17c  .010 92238.17c  .19  GAS=1 $fuel gaseous
m2    4009.60c .5 8016.54c .5                        $BeO shield
mt2   beo.06t
totnu
print
```

Monteburns Input

The following is a sample input for the Monteburns code package. This particular input requires a feed file in conjunction with the MCNP input cards. The ! symbol denotes a comment marker, much like the $ used in MCNP. Of note is that the user is required to select a starting flux estimate from the Origen library. This flux plot is used to calculate initial averaged cross-sections and after the first step is replaced by MCNPs determination of the flux. The fractional importance in this input is listed as 0.1, meaning only isotopes with 10% of the fuels amount (in atoms) is recorded, problems will arise if this number is set too low, as MCNP will require to many libraries to be loaded and will exceed memory limitations.

```
MONTEBURNS SAMPLE INPUT – REQUIRES FEED FILE
PC        !Operating System
1         !Number of Materials **
1         !Material Number (will burn all cells with this mat)
0.0       !Volume (0.0 uses MCNP volume)
1000      !Power MWt
-200.0    !Recov. energy per Fission (-200.0 includes actinides)
0         !Total number of days burned (Non 0 if No feed file is used)
11        !Number of outer burn steps
100       !Number of internal burn steps
1         !Number of predictor steps
0         !Step to restart after (0 = beginning)
PWRU      !Number of default origen library (next line is location)
D:\Origen2\Libs !Origen Library Location
0.1       !Fractional importance (track isos with abs,fis,atom,mass)
1         !Intermediate Keff calc. 0=no, 1=yes (0 for no feed file)
26        !Number of automatic tally isotopes, followed by list
55137.60c
55135.50c
```

```
54135.50c
53135.50c
62149.50c
92234.35c
92235.17c
92236.60c
92237.35c
92238.17c
93237.60c
94238.35c
94239.60c
94240.50c
94241.60c
94242.50c
95241.50c
95242.50c
95243.50c
96242.60c
96243.60c
96244.50c
96245.60c
96246.60c
96247.60c
96248.60c
```

Monteburns Feed File

The feed file is used to inject or "feed" isotopes into the geometry, as well as remove unwanted isotopes (in this case fuel is fed into the core, while fission products are removed). A feed file is only required if the user wishes to inject or remove isotopes into or from the core respectively during operation.

FEED FILE – No title card used in feed file, input starts with "Time".

```
Time   Days   Power MBMat Feed Begin&EndRates Remov. Fraction F.P.Removed
Step Burned  Fract. #  #      grams/day    Group#
1    30.00   1.000   1   1     2000.0  2000.0   1     1.000 !
2    330.00  1.000   1   1      -1.0    1900.0   1     1.000 !
3    360.00  1.000   1   1      -1.0    1900.0   1     1.000 !
4    360.00  1.000   1   1      -1.0    1800.0   1     1.000 !
5    360.00  1.000   1   1      -1.0    1850.0   1     1.000 !
6    360.00  1.000   1   1      -1.0    1700.0   1     1.000 !
7    360.00  1.000   1   1      -1.0    1500.0   1     1.000 !
8    360.00  1.000   1   1      -1.0    1450.0   1     1.000 !
9    360.00  1.000   1   1      -1.0    1350.0   1     1.000 !
10   360.00  1.000   1   1      -1.0    1300.0   1     1.000 !
11   360.00  1.000   1   1      -1.0    1300.0   1     1.000 !
1                              ! # of feed specs
2                              ! # of isos in Feed Group 1
92235 .650000
92238 .350000
```

```
1                   ! # of removal groups
1                   ! # of ranges
28 68                 ! range of atomic numbers
```

LIST OF REFERENCES

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BIOGRAPHICAL SKETCH

Robert Edward Norring was born on January 7, 1980, in New Jersey. At the age of 10 he moved to Sarasota Florida and attended Gulf Gate Elementary School. One year later he enrolled in Pine View School for the Gifted, where he stayed until his high school graduation in 1998.

Robert Norring enrolled at the University of Florida in August of 1998, and received his undergraduate degree in fall of 2002. During his undergraduate studies he also enrolled as a graduate student at the University of Florida and began his thesis research. During his work he attained the position of vice president of Alpha Nu Sigma for the year of 2002.

Currently he is enrolled in the Nuclear and Radiological Engineering graduate program at the University of Florida and is working on his doctorate degree.