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Management Tool for Assessment of Alternative Fuel Cycles

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To the Graduate Council:

I am submitting herewith a dissertation written by Jeffrey R. Preston entitled "Management Tool for Assessment of Alternative Fuel Cycles." I have examined the final electronic copy of this dissertation for form and content and recommend that it be accepted in partial fulfillment of the requirements for the degree of Doctor of Philosophy, with a major in Nuclear Engineering.

Laurence F. Miller, Major Professor

We have read this dissertation and recommend its acceptance:

Wes Hines, Ron Pevey, Dayakar Penumadu

Accepted for the Council:
Dixie L. Thompson

Vice Provost and Dean of the Graduate School

(Original signatures are on file with official student records.)

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Management Tool for Assessment of Alternative Fuel Cycles

A Dissertation Presented for
the Doctorate of Philosophy
Degree
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Jeffrey Robert Preston
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ABSTRACT

A new approach to fuel cycle uncertainty analysis and optimization is presented that combines reactor physics information, spent fuel management, and economic forecasting, which may be used to investigate effects of decisions in the design of advanced nuclear fuel cycles. The Matlab-based simulation includes isotopic mass and integral decay heat data produced by reactor physics codes in the SCALE package (SAS2, ORIGEN-ARP, and ORIGEN-S). Reactor physics data for Light Water Reactor (LWR), and metal- and oxide-fueled Liquid Metal-cooled Fast Burner Reactor (LMFBR) designs are stored in databases that the code uses as needed. Detailed models of the once through and hybrid LWR-LMFBR fuel cycles have been developed for repository decay heat analysis, determination of levelized unit electric cost (LUEC), and reprocessing of spent fuel into fast reactor fuel or targets as a means of isotopic inventory minimization. The models may be run for single estimates based on best estimates of model parameters as either a Monte Carlo uncertainty analysis or as an optimization using Genetic Algorithms (GA).

Results from the LUEC calculations show the once through cycle has a bus bar cost of about $19.0mills/kWh (excluding repository and interim storage costs), and the hybrid cycle has a bus bar cost of about $26.5mills/kWh. Implementation of the hybrid cycle compared to the closed once through cycle yields an effective repository mass capacity increase by a percentage of about 30% to 60% through full reprocessing of LWR spent fuel compared to original mass definitions of the Yucca Mountain repository. The GA optimization routine allows the user to define any one of the variables present in the output structure as the fitness parameter; thus, optimization of any calculated value is possible, including economic cost, isotopic inventory, or required repository capacity. Optimization of the once through cycle with respect to LUEC gives a result of $19.2 mills/kWh when burn up approaches the upper limit of 60 GWd/t and delay time spent fuel cools after discharge approaches 200 years (including repository and interim storage costs).

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AFCI	Advanced Fuel Cycle Initiative
BWR	Boiling Water Reactor
CPU	Central Processing Unit
DOE	United States Department of Energy
DU	Depleted Uranium
EMWG	GIF Economic Modeling Working Group
FCRD	Fuel Cycle Research and Development Program
FP	Fission Product
FR	Fast reactor
GA	Genetic Algorithm
Gen IV	Generation IV Nuclear Energy Systems Initiative
GIF	Generation IV International Forum
GPU	Graphics Processing Unit
IDH	Integral Decay Heat
LUEC	Levelized Unit of Electric Cost
LMFBR	Liquid Metal Fast Burner Reactor
LWR	Light Water Reactor
MC	Monte Carlo
MOX	Mixed Oxide Fuel
MTHM	Metric Tons Heavy Metal
NEI	Nuclear Energy Institute
NWPA	Nuclear Waste Policy Act of 1982
OCRWM	DOE Office of Civilian Radioactive Waste Management
PWR	Pressurized Water Reactor
PUREX	Plutonium and Uranium Extraction
SF	Spent Fuel
SWU	Separative Work Unit
TRU	Transuranic Isotope
UOX	Uranium Oxide
UREX	Uranium Extraction
USEC	United States Enrichment Corporation
YMP	Yucca Mountain Project

CHAPTER 1
Introduction and General Design of research

Current Status of Nuclear Fuel Cycle

New construction of nuclear power in the United States has been stagnant over the past 30 years due to economic, technological, and political constraints. Rising oil and natural gas prices along with fears of anthropogenic global warming have renewed interest in nuclear power as a sustainable energy platform that is both economically viable and environmentally sound. While the nuclear industry is on the cusp of entering a renaissance, new power plant construction still faces many of the same economic, technological, and political barriers that depend heavily on the solution for long term suitable handling of spent fuel and financing of reactors and supporting facilities.

Revitalizing the nuclear power industry in preparation for new plant construction and maintenance of the existing reactor fleet is currently hindered because of two key elements. Decline in demand for new reactors after a period of rapid growth in the 1970's resulted in severely diminished capacity to build new plants because of the re-tasking or closure of steel foundries used to forge reactor components [1]. Also, the industry is currently sustaining a loss of engineering knowledge and experience as the workforce of the 1970's and 1980's enters retirement [2]. Mitigating these circumstances requires a significant capital investment in facilities and sustained investment in training of a new workforce; however, these tasks could possibly require about 5 to 10 years before beginning large scale construction of new plants, leaving the issues of waste storage and financing as the major issues.

The enduring problem of waste storage poses a similar hindrance to the construction of new plants as it remains a political obstacle. Legislation currently exists in California and Illinois that prohibits the opening of new power plants at least until this issue is resolved [3]. Congress addressed this issue in the Nuclear Waste Policy Act (NWPA) of 1982 that opted for a secured, centralized geologic repository and later selected Yucca Mountain as the site [4,5]. The Obama administration defunded the Yucca Mountain Project (YMP) in March 2009, effectively reopening the question of how to manage the existing and future spent fuel inventory [6].

The NWPA states that the federal government assumes custody of all spent fuel discharged from commercial reactors; however, Congress has repeatedly delayed the YMP for a variety of reasons. Justifications for the delays include concern the environmental impact and protests over the concept of using the mountain as a nuclear waste dump [7]. Delays in opening are not without some benefit as they have allowed time for discussions as to whether the facility is large enough to store all of the existing and future waste from the current reactor fleet and of new reactors that may be added [8]. YMP is legally restricted to store 63,000 metric tons of civilian spent fuel, with the remaining 7,000 metric tons reserved for national defense waste. At the time of writing, nearly 62,000 metric tons are in storage at various sites around the country [9,10]. Thus YMP will reach capacity in 2010 at the current spent fuel production rates of

approximately 2,000 metric tons generated per year from the 104 reactors in operation [10,11]. Assuming that additional nuclear power plants will be built and the existing plants pursue 20 year license extensions, spent fuel production rates will increase, exacerbating the problem of having more waste than repository capacity. This renews the question of where the federal government will store the excess spent fuel once YMP reaches capacity, with the prospect of a second repository as a consideration [7,9,10,12,13,14,15].

Current research aims to increase the capacity of YMP by relying on the passive heat removal rate limit rather than the statutory mass limit in the NWPA [7,10,12]. The current mass limit assumes a specific fuel burn up, corresponding to a specific amount of decay heat that would be within the limits of passive heat removal; however, the 70,000 metric ton limit was enacted by the legislation before much of the relevant research was performed. The Department of Energy (DOE) estimates the capacity to be greater than 130,000 metric tons, and the Secretary of Energy urged Congress to remove the current limit in 2008 [10]. Because the current mass limit infers a comparable decay heat limit, replacing the mass limit with a decay heat limit adds the possibility for a greater total mass of spent fuel that could possibly be stored in addition to the planned amounts. Decreases in decay heat can be achieved from a combination of higher fuel burn ups, longer cooling periods after discharge, and a coupled reprocessing and recycling of isotopes that significantly contribute to the decay heat [12,16]. Industry currently uses higher burn up fuels, currently 40GWd/t compared to 30GWd/t 20 years ago, as a means to lower refueling costs and maximize capacity factor. Longer cooling periods in interim storage are currently the norm as there is no facility accepting spent fuel for permanent storage. Commercial reprocessing and recycling of spent fuel is currently not practiced.

Advanced reprocessing could partition the spent fuel into constituent products of fission products and actinides that would then be sent to long term storage or fuel fabrication for recycling. Fission products could be stored in an alternate facility to minimize repository heat load as fission products comprise nearly 50% of the total decay heat 25 years after discharge. Actinides could be converted into new reactor fuel or targets to burn the minor actinides in Mixed-Oxide (MOX) Light Water Reactors (LWR) or Fast Reactors (FR). Transuranic isotopes generally have longer half lives than the fission products and comprise nearly 94% of the total mass of spent fuel and contribute about 80% of the total heat load over 1500 years [7,17,18]. Some estimate that transuranic removal could reduce the repository heat load by a theoretical factor of 50 or greater, but these estimates do not necessarily account for secondary waste streams and associated costs with multiple recycle stages [9,16]. A reduction factor between 4 and 15 could be realistically achieved depending on the recycling method and extent to which recycling is applied.

Since much of the actinide mass is either fissionable or fertile, reduction of the repository heat load requirement is possible through removal and usage as MOX fuel or actinide targets. France currently uses reprocessing technology for MOX-fueled LWR's, while the United States has only used MOX fuel in a few trials (excluding university and experimental reactors) [19,20]. Implementing any strategy that utilizes reprocessing will be a shift away from the currently used once through fuel cycle design.

The once through fuel cycle, consisting of light water reactors burning uranium oxide fuel without recycling, has existed from the beginnings of the large-scale commercialization of nuclear power as it was the economically optimal option at the time; however, some fuel cycle designers preferred fast reactors with fuel recycling in order to have a seemingly infinite fuel supply. The once through cycle had the definite economic advantage because of abundant and low cost uranium resources when compared to fossil fuels; however, recent reports of diminishing uranium supplies have increased the spot price, reducing the economic advantage [21,22,23]. As prices rise and the waste issue becomes increasingly pronounced, a fundamental break from the open once through fuel cycle is necessary as a solution to some problems with waste storage and the longevity of electrical production [7,24].

Advanced fuel cycle design focuses on closing the fuel cycle by depositing spent fuel into the geologic repository. DOE had previously set a tentative schedule for completion of YMP with the opening in 2017, but has recently reversed that decision citing a variety of reasons [1,6,25,26]. Previous plans stated that upon final approval, the first fuel shipment was expected within 10 years as efforts would be made to at least partially complete the facility during that time. Despite this recent setback, waste must still be stored in a secured facility, leaving the geologic repository as the most viable option for a long term solution. Delays in opening have thus far been beneficial in allowing time for additional studies that investigate the benefits of advanced fuel cycles for both the industry and repository [8,10].

Advanced fuel cycle designs are expected to be thoroughly modeled and evaluated before any major steps are taken economically and politically. Systems analysis is a common approach to modeling fuel cycle scenarios with some uncertainty information. Existing models, namely DANESS and VISION, apply sensitivity analysis methods in determining results of initial parameter selection or growth models [27,28,29,30]. These nominal value systems analysis codes reduce the complexity of mass flow in the fuel cycle to a solution of a system of equations, where the results include a single number for each parameter. DANESS is limited to 8 sensitivity variables for the entire analysis, including reactor operation parameters, fuel production, waste management, and economic cost estimation [29]. Estimates of uncertainty are generated using a range of inputs for a single variable, providing insight into model sensitivity to changes in that value. Only limited insight into variable effects is obtained when performing a sensitivity analysis on a small number of variables, whereas applying variations to many variables provides better insight into entire system operations under uncertainty.

Contributions and Research Overview

Presented in this research is a systems analysis approach to fuel cycle modeling that incorporates reactor physics data and economic assessment with Monte Carlo sampling for detailed uncertainty analysis and optimization. New and significant contributions to the field of fuel cycle research and development include:

- A new methodology for simulation and uncertainty analysis of fuel cycle design that integrates economic and physical modeling
- An optimization procedure for fuel cycle design using Genetic Algorithm optimization [31,32]
- A highly configurable management and analysis code with supporting tools that include an interface for importing physical reactor data, plotting, and batch file processing
- A procedure for generation of reactor isotopic decay heat and mass data for a range of operating conditions for LWR and fast reactor designs
- A parallelized code capable of running on distributed CPU or GPU configurations for large analyses [33]

This dissertation begins in Chapter 2 with a brief overview of the current state of the art of fuel cycle design and economic analysis methods. Descriptions of the once through and hybrid fuel cycle designs are provided in Chapter 3. Chapter 4 presents the analysis methodology used, and results of the Matlab analysis tool are given in Chapter 5. Conclusions and future work are given in Chapter 6. The Appendices include flow charts of the Matlab implementation, code usage descriptions, and data creation guidelines.

CHAPTER II
Literature Review

Nuclear power will likely play a larger role in energy production over the coming decades, requiring both a sustainable fuel supply and secured long term waste storage, while remaining economically competitive to alternative energy sources. Many approaches have been proposed for analysis of advanced fuel cycles for a range of scenarios with varied capabilities for detailed multivariate uncertainty analysis. DOE programs supporting the development of advanced fuel cycles include the Advanced Fuel Cycle Initiative (AFCI), recently renamed as the Fuel Cycle Research and Development program (FCRD), and the Generation IV Nuclear Energy Systems Initiative (Gen IV). Both programs support development of computational tools that analyze various components of advanced fuel cycles. Components consist of fuel cycle aspects such as the numbers of reactors needed for a specified electrical generation capacity or the estimated reduction of the plutonium stockpile over time from the addition of new reactor technology.

Advanced Fuel Cycle Initiative

The need for sustainable, clean energy emerged as both an economic necessity and paramount national security issue in the 1970's; however, few efforts have pushed the need for nuclear power as the cornerstone for future electrical production as has the Bush administration's AFCI and Gen IV programs [34,35,36,37]. Nuclear power is a candidate for long-term base load electrical production as an alternative to coal and natural gas. Issues concerning waste storage, proliferation risk, and operational safety cause reluctance for expanding nuclear power usage; however, the AFCI and Gen IV programs have advanced technological development with a transitional period leading towards long term energy stability that will assuage these concerns.

The AFCI began as a long term research and development program, funded through the DOE, and was renamed the FCRD in 2009 with similarly stated goals [38]. Goals of the program include reducing environmental impact of nuclear power, minimizing non-proliferation risk, ensuring energy security through the reprocessing of spent fuel to remove usable components, and to improve fuel cycle management in terms of economic costs and operational safety [34,35]. Meeting these goals requires research in repository analysis, new reactor technology, and cost estimation of all necessary components to operate various fuel cycle scenarios, all of which operate as closed cycles with a central repository for long term spent fuel storage.

The NWPA set the groundwork for storing spent nuclear fuel in a geologic repository in 1982, with the location selection dependent on various environmental and safety criteria [4]. Congress selected Yucca Mountain in Nevada as the geologic repository site, later named the YMP. Congress defined the repository as 63,000 metric tons for civilian usage and 7,000 metric tons for defense waste. The U.S. commercial

power industry has accrued nearly 62,000 metric tons of spent fuel as of December of 2009. These 104 reactors generate about 2,000 metric tons of spent fuel each year of operation [10]. YMP was tentatively scheduled to open in 2017 until Congress cut funding in 2009, and spent fuel inventory currently stored at reactor sites will exceed the licensed 63,000 metric tons in 2010 [6,25,34,35]. The 104 power reactors will continue operations after this threshold is passed, continuing to produce electricity and spent fuel, resulting in an excess that the repository cannot hold based on current statutory limit in the NWPA [8]. Securing spent fuel for long term requires a choice of either increasing YMP effective capacity, opening a second repository, or both [10].

Increasing capacity is quite feasible since the current mass limit is based on estimates derived from incomplete data, according to the DOE, with a realistic capacity of greater than 130,000 metric tons of civilian spent fuel [10,13,14,16]. The FCRD is continuing the AFCI program's investigation into new reactor technologies that reduce capacity required by removing long lived actinides from spent fuel inventories through reprocessing and transmutation in either fast reactors or mixed oxide thermal reactors, while minimizing proliferation risk. Collocating reprocessing and fast reactor facilities minimizes the proliferation or diversion risk from fuel transportation and minimize the stockpile of reclaimable plutonium from spent LWR waste currently stored around the country [19,22,34,36,39]. The Gen IV program researches advanced reactor designs capable of this task; however, these reactors are beyond the scope of the transitional period as they are still largely in early design phases [37].

While the advanced Gen IV reactors are still conceptual, the choice of a transitional reactor design that bridges the gap between legacy reactors and advanced reactors remains open. The FCRD researches several reactor types, including gas cooled reactors, mixed oxide fueled LWR's, and fast reactors. Each of these reactor types has some degree of operational experience [24,40,41]. The AFCI did not select a preferred reactor design at the outset of the program, nor does the FCRD. Currently, however, liquid metal cooled fast reactor appears to be preferred over the alternatives cited above. The liquid metal cooled fast burner reactor (LMFBR) design that has been developed is based on the SuperPRISM design that was a liquid metal cooled breeder reactor (further references to the "LMFBR" design refers to the burner variant in this research) [42,43,44,45,46,47]. Regardless of the choice of reactor, the AFCI initially set a timeline to begin implementation of the best available technology in 2010 that meets the previously listed goals as well as possible; although, this has been delayed due to the cancellation of funding for YMP as this was an integral part of all of the AFCI fuel cycles [6,34].

The original AFCI program and its successor, FCRD, administer parallel research projects into separations research, fuel design, reactor design, temporary storage methods, transportation, and repository storage methods, all with the goal of an optimal fuel cycle design that provides for longevity and security of energy production [12,34]. Conservation of fuel resources is a concern for long term energy security, even when implementing reprocessing and burner reactors that add to the longevity to the fuel cycle when compared to the approach currently used in the once through fuel cycle. In the once through cycle, nuclear power generation will cease once the uranium supply is

exhausted, where as in a scenario with reprocessing and advanced burner reactors, the usage of the plutonium stockpile will delay this cessation according to the fuel consumption rates of the reactors used. Current estimates predict that the inventory of recoverable plutonium from LWR legacy spent fuel and future production in the current LWR fleet will begin to tail off around 2050-2070 assuming burner fast reactors are deployed with a single reprocessing step between the LWR and FR stages [17,29,48,37]. Another fuel cycle option combines the existing reactor fleet with fast breeder reactor technology, where U-238 is transmuted into plutonium and recycled for a seemingly infinite fuel supply given the abundance of the isotope [7,12].

Modern Fuel Cycle Analysis

One method of fuel cycle analysis involves the use of mass flow equations. Modern approaches have applied these sets of equations to computer-based equation solvers that include varying degrees of detail as a simulation method. Mass flow of fuel throughout the cycle is expressed with rate or mass balance equations that use nominal values to estimate masses at various steps of the fuel cycle, described at length later in the theory section [24,49,50]. Modern computerized methods for large-scale fuel cycle analysis add complexity to these equations to include far more detail with isotopic data, decision making capabilities, and ability for sensitivity analysis.

The DOE funded the creation of two recent fuel cycle analysis codes, DANESS and VISION, that are developed by Argonne National Laboratory and Idaho National Laboratory, respectively [28,29,30,48,51,52,53]. Both packages are designed to obtain a time-dependent variation of the standard analytical equations used in traditional fuel cycle analysis. These codes are built inside systems analysis software packages, such as iThink/Stella or PowerSim, which were not intended for computationally intense models with massive databases and large numbers of uncertainty variables [54,55]. Thus, they have limited uncertainty analysis capability [28,29,56]. This limits resulting models to perform sensitivity analyses rather than uncertainty analyses as they typically use less than 10 random variables. For uncertainty analysis packages, parameters are sampled from distributions, through Monte Carlo, Latin Hypercube, or empirical methods, in order to perform the variation on parameters; thus, the extent to which existing codes can apply sampling techniques is limited by software environment since these code systems often use interpreted code for distribution sampling rather than compiled code [28,29,56,57,58,59].

Sensitivity and uncertainty analyses for fuel cycle analysis are similar in the initial approach, as both use the same mass flow equations with differing numbers of random variables assigned through distribution sampling methods [59,60,61,62,63]. A sensitivity analysis may be performed by perturbation of some parameter, possibly fuel burn up or enrichment, and noting the variation of results. Uncertainty analysis performs a similar task by perturbing multiple parameters and obtaining distributions of results as a means to quantify variance in the calculated values. Sensitivity analyses provide useful information as to which parameters constitute major components of variation. Once the

ideal range for the parameter is found, a larger uncertainty analysis may be performed by sampling many parameters and noting the resulting distributions of results.

Economic Analysis

Economic studies were included in the AFCI program as a goal to create a technologically feasible, affordable, and sustainable fuel cycle. This requires intensive study of the costs associated with various design choices and operational parameters. The Generation IV International Forum (GIF) created the Economic Modeling Working Group (EMWG) in 2003 with an objective to create cost estimation methodologies for various fuel cycle scenarios, all common with those that are studied in the AFCI program [35,37]. Each aspect of the fuel cycle has associated costs for mass and processes, as is described further in Chapter 4. Because the AFCI assumes usage of large scale reprocessing facilities and operation of new reactors, cost estimation must be accomplished for new facilities of a type that has not been previously constructed in the US. This process is inherently uncertain since incomplete designs and estimates for material cost, labor contracts, facility operational costs, transportation costs, and maintenance cannot be accurately determined for these facilities. Estimating these costs is an extensive effort that requires a bottom up approach, similar to cost engineering techniques, or a top-down method based on scaling the cost of similar facilities to the specifications of the needed facility [64,65,66].

A top-down approach for a conceptual facility begins with comparison to a similar system. In the case of a large scale reprocessing facility, a plant as large as the AFCI intends has not been built in the US; however, France operates a facility that is smaller (two 800 metric tons per year) than the estimated size of the US design (about 2000 tons per year) [21]. The comparison of the French plant is the basis for extrapolation, but it is noted that the plant has most likely received subsidies from the French government. Scaling the cost initially based on the facility size allows a reasonable estimate for the cost, whereas a second tier extrapolation for large equipment and construction costs leads to a better estimate; still, a bottom up approach is typically a more trusted method [67,68,69].

Construction of a facility begins from the bottom-up, hence the name of the method and the general approach taken. Building a nuclear facility begins with a land purchase, then a series of surveying, soil work, foundation pouring, and structural construction. Each step has an associated cost as does each brick and hour of labor spent on construction. This method may be favored as it is easily explained, but it is difficult to estimate the cost of specialized, one-of-a-kind pieces of equipment that require special machining and maintenance; thus, the top-down approach has some advantages. Both methods have their own respective advantages over the other, and a mixture of the two is used to produce a final estimate that minimizes uncertainty of the project's cost.

The EMWG has developed software, G4ECONS, which is capable of sensitivity and uncertainty analysis of fuel cycle economics [70]. An uncertainty analysis may be

performed on the various cost parameters to determine the largest contributors of variance. Determination of model sensitivity identifies parameters of interest that may lead to an optimal design by either minimization of the value or variance outside of the code.

Levelized unit of electric cost (LUEC) is used as a basis for comparison of fuel cycle design, given in terms of cost per kilowatt-hour generated, where each fuel cycle component adds a portion to the overall cost of electricity. The cost includes facilities, operations, fuel, and decommissioning [66]. Components of the LUEC are annualized for the equilibrium fuel cycle design, and are dependent on reactor lifetime, interest rates, and growth models.

Once the major facility costs are included, operating and fuel cost may also be included in the unit cost of electricity. Operational costs refer to maintenance, labor, property taxes, insurance, furnishings, and other overhead costs external to the capital cost of the facility. Maintenance costs depend on equipment and plant reliability, and generally are included in the cost estimate of the facility on the basis of the mean failure rate of similar equipment, similar to the top-down approach of cost estimation.

Refueling costs are dependent on the fuel mass, enrichment, and fuel type required by the reactor, operational costs to the fuel producer, and frequency of refueling events. For a BWR, the cost of an 18 month refueling is around $10 million (in 2007 dollars) [24,49,64,65,70]. This leads reactor owners to seek longer burn up fuels to maximize revenue per refueling, which has been the case as applications have been filed for burn up increases approaching 60 GWd/t [37,64,67]. Costs associated with fuel production are dependent on the spot price of uranium ore, facility operations, and the energy required for separation. The spot price of uranium is around $90/kg in 2010 [71]. Vendor costs include raw ore, transportation, assembly and cladding material, burnable poison applications, and core engineering services. Except for uranium, these costs are rather stable and are forecast using inflationary growth models. While uranium ore and enrichment are large components of LWR fuel cost and have a history of reasonable forecasts, estimation of fast reactor fuel costs is much different as it requires a different set of support facilities for fuel production.

The size of a reprocessing facility required for 100 LWR's is estimated between 1500 and 3000 MTHM per year with a cost estimated around $6,000/kg processed [64]. Reprocessing plants require a large initial capital investment, estimated at more than $20 billion, where costs are incurred annually as the cost is amortized over the life of the facility [66,67,72]. While capital costs of the enrichment facility are treated similarly to those of a reactor, the operating cost treatment varies.

The choice of separation methodology has a large portion of the reprocessing costs. In advanced methods, fission products are almost entirely removed from the product stream for disposal, while the actinides may be separated into uranium and minor actinides or into constituent element streams for full partitioning. Separations processes for minor actinide partitioning methods could be used in a cycle that uses fast reactors or LWR targets [12,19,48,67,68]. Uranium removed from the process may be used to fill the remainder of the fast reactor fuel mass. Reusing uranium from the LWR enrichment

stream in fast reactor fuel is a more efficient use of the resource than using mined natural uranium [22].

Fuel scarcity is a concern as the developed nations continue a steady growth in annual electricity demand. Lower emissions standards will deter construction of new coal plants, with nuclear power as a major factor in new electrical generation. Thus the cost of fuel may increase if suppliers throttle production to maintain profits from ore sales and sustain production as resources are depleted. Estimates on the amount of remaining ore vary greatly, with some estimating that the supply is already very low while others report an overabundance of ore deposits [21,22,23]. The cost of uranium was much higher in 2008 than in 1980; thus there was an incentive to minimize fuel consumption in order to maintain profitability [24,67].

Summary

The current state of nuclear fuel cycle design lacks a direction for long term spent fuel storage, which is a major criterion for the future of nuclear power. The U.S. DOE has sponsored large research and development programs to investigate design choices of the next generation nuclear fuel cycle as it pertains to spent fuel management and resource conservation. Economic analysis is becoming an increasingly important aspect in the design of the next generation nuclear fuel cycle.

CHAPTER III
Advanced Fuel Cycle designs

Each of the designs being investigated in the FCRD, previously AFCI, represents a transition away from the open once through LWR fuel cycle, shown in Figure 1, and eventually towards a closed fuel cycle with actinide recycling in LWR's or fast reactors. Of the technologically feasible fuel cycles that use either a currently available or mature advanced reactor design, two options for actinide recycling include the following: a combination of uranium oxide-fueled (UOX) and mixed oxide-fueled (MOX) LWR's, and a hybrid reactor fleet of UOX-fueled LWR's and MOX-fueled fast reactors (also referred to as a one-tier cycle), shown in Figure 2. MOX fuel consists of a mixture of uranium and higher actinides present in spent LWR fuel, including plutonium, americium, and curium. Because these are transition-focused fuel cycles, each has stages common to the once through fuel cycle; thus, the case of a once through LWR cycle is presented as a common foundation that will also be used for a comparison.

Once Through LWR

The once through fuel cycle is a good reference for comparison with advanced fuel cycle scenarios because it has been used in the United States since the beginning of large scale commercial nuclear power generation. Components of the fuel cycle, namely uranium enrichment plants, fuel fabrication facilities, and legacy reactors, are significant portions of any transition to an advanced fuel cycle as they represent the largest capital investments of the once through cycle and critical fuel production pathways to any transition type Gen IV reactor. The cycle is described in this section from the perspectives of mass flow, reactor physics, and economic cost, while later sections will refer to this in comparison.

A once through cycle follows the path of uranium mass through the fuel cycle, shown previously in Figure 1. Uranium enters the fuel cycle as ore that is separated and purified in the milling stage as U_3O_8, referred to as yellow cake [24,49]. As a general rule, the spot price of uranium ore varies in the international open market according to supply and demand, but has deviated from around $140/kg in 2008 to $90/kg in 2010 [64,67,71]. For an equilibrium cycle, no stockpiles of uranium are kept and the amount of uranium required per year to continue the fuel cycle is dependent on the fuel consumption of the reactors used. Costs of milled uranium ore per year are usually stable as they are set by multi-year contracts to minimize variability in forecasting, but are subject to sharp turns in the event of unforeseen production discontinuities. A nominal value, as shown in

Table 1 along with other associated fuel cycle costs, may be used with some estimated growth factor to account for increases over time [64,67]. Mass dependent costs also indicate the composition of the mass as "kgU" refers to "kilograms of uranium" and "kgHM" refers to "kilograms of heavy metal" where this implies fuel mass that includes uranium and other actinides.

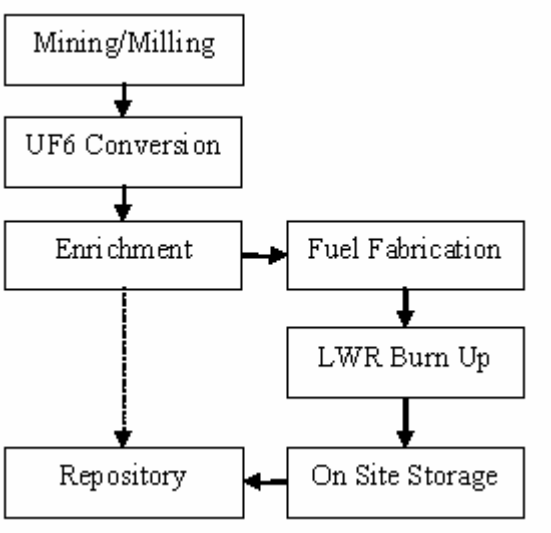

Figure 1: Once through fuel cycle mass flow diagram where the dashed line refers to the enrichment tails sent to long term storage or repository

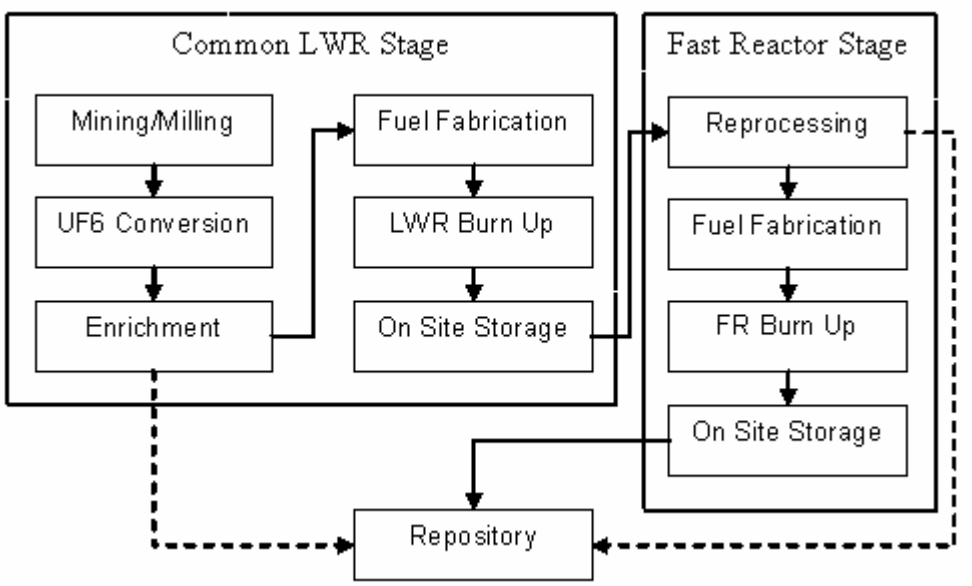

Figure 2: Hybrid fuel cycle mass flow diagram where dashed lines refer to enrichment and reprocessing waste sent to long term storage or repository

Table 1: Estimated Nominal Costs Associated with the Once Through Fuel Cycle with Referenced Page Numbers Provided [64]

Stage	Uranium Status	Fissile Enrichment	Cost
Mining and Milling (A-10)	U_3O_8	0.711% U-235	$100/kgU
UF_6 Conversion (B-2)	UF_6	0.711% U-235	$10/kgU
Enrichment (C1-3)	UF_6	3-5% U-235	$115/SWU (~7.2 SWU/kg for 5% enriched fuel)
DU Disposal (K2-1)	DUF_6	~.2-.3% U-235	$8/kgU
Fuel Fabrication (D1-4)	UO_2	3-5% U-235	$220/kg (PWR) $270/kg (BWR)
Light Water Reactor (R1-8)	UO_2	3-5% U-235	$1,800/kWe (installed capacity)
Spent Fuel Storage (Dry) (E2-5)	UO_2 + Fission Products	~1% U-235+Pu	$120/kgHM
Spent Fuel Storage (Wet) (E1-4)	UO_2 + Fission Products	~1% U-235+Pu	$300/kgHM
Repository (L-9)	UO_2 + Fission Products	~1% U-235+Pu	$900/kgHM (for current capacity)

Once the ore is purified, it is then converted into uranium hexafluoride (UF_6) that can be used in the enrichment process. The process has a relatively stable cost of around $10/kg, which is rather low compared to the rest of the fuel cycle costs as it is a basic chemical conversion, while the enrichment process contributes the majority of the fuel cost [24,64,65,67,69].

Enrichment of natural uranium to a specific U-235 percentage is costly both in terms capital and operations and is one of the most expensive stages during the fuel cycle. Because of reactor operation and design, the cost of fuel enrichment may vary slightly for LWR's that produce the same energy over a comparable time period. Reaching a specific enrichment requires a certain number of Separative Work Units (SWU) to perform enrichment to the amount specified by the utility. This term may also be used as a coefficient for the amount of natural uranium required for higher enrichments of some amount of output product. A mass balance, given in Equation (1), shows the relation between the enriched product, $M_{Enriched}$, feed of natural uranium, $M_{Nat\ U}$, and the depleted uranium tails, M_{DU}, with units of mass per year.

$$M_{Enriched} = M_{Nat\ U} - M_{DU} \tag{1}$$

Each of the mass terms has an associated weight percent of U-235 content, given as X in the U-235 mass balance in Equation (2).

$$x_{Enriched} M_{Enriched} = \left(x_{Nat\ U} M_{Nat\ U}\right) - \left(x_{DU} M_{DU}\right) \tag{2}$$

From this, the number of SWU's required by the enrichment facility during a time period, T, is defined according to Equation (3), with the units of kg-SWU [24].

$$SWU = \left[M_{Enriched} \cdot V\left(x_{Product}\right) + M_{DU} \cdot V\left(x_{Tails}\right) - M_{Nat\ U} \cdot V\left(x_{Feed}\right)\right] \cdot T \tag{3}$$

Each of the x terms are mass fractions of U-235 in the enriched product, depleted uranium tails, and natural uranium feed, respectively [24,49,20]. V is a dimensionless value function shown in Equation (4).

$$V(x) = (1-2x)\ln\left(\frac{1-x}{x}\right) \tag{4}$$

A major goal of fuel vendors is to reduce energy consumption of the fuel production process. Thus gas centrifuges that are currently under construction in the US may be evaluated instead of diffusion as this method requires significantly less power per SWU than diffusion. The separation factor of the centrifuge method is around 1.2 as opposed to 1.004289 for diffusion, requiring fewer stages to reach a specific enrichment and less natural uranium in the feed [24]. For example, a plant producing 25MT of 5% enriched fuel with a tail fraction of 0.3% requires 286MT of natural uranium and produces 261MT of depleted uranium.

The number of SWU's per kg of product is termed the SWU factor, SF, shown in Equation (5) below (note that the units of SWU are actually kg-SWU, which is truncated to units of "SWU") [24].

$$SF = \frac{SWU}{\left(\mathrm{M}_{\text{Enriched}} \cdot T\right)} = V\left(x_p\right) + \left(\frac{\mathrm{M}_{\text{Nat U}}}{\mathrm{M}_{\text{Enriched}}} - 1\right) \cdot V\left(x_w\right) - \frac{\mathrm{M}_{\text{Nat U}}}{\mathrm{M}_{\text{Enriched}}} \cdot V\left(x_f\right) \tag{5}$$

The hypothetical enrichment plant previously described has a SF of about 7.2 SWU/kg. The total number of SWUs per year required is shown in Equation (6), where T is defined as 1 year since the mass balances are in mass per year.

$$N_{\text{SWU per year}} = SF \cdot \mathrm{M}_{\text{Enriched}} \tag{6}$$

The total number of SWU's per year for the hypothetical case would thus be about 180,000. At $115 per SWU, as specified in

Table 1, the total cost of enrichment would be nearly $21 million.

For commercial reactors, uranium may be enriched to any percentage below 5% as ordered by a plant due to licensing restrictions, and plants may order multiple enrichment percentages for a single refueling as a result of reactor operation. Once the enriched product is ready, it is transported to a conversion facility where the UF_6 is converted into uranium oxide, UO_2. After conversion to the oxide form, the enriched product is transported to a fuel fabrication facility where it is pressed into fuel pellets before being loaded into the fuel rods. Cost of PWR fuel fabrication is about $220/kg of UO_2, which includes the conversion from UF_6 to UO_2 and the production of the fuel pellets and assemblies. Because this cost is based on mass, the cost of refueling varies according to reactor requirements. For the 25MT production example, the cost of fuel fabrication would be about $5 million. All fuel costs are combined into the total fuel cost shown in Equation (7), where each term is defined in a later section.

$$\mathrm{C}_{\text{Total Fuel Cost}} = \mathrm{C}_{\text{Mining/Milling}} + \mathrm{C}_{\text{UF}_6\text{ Conversion}} + \mathrm{C}_{\text{Enrichment}} + \mathrm{C}_{\text{DU Disposal}} + \mathrm{C}_{\text{Fuel Fabrication}} \tag{7}$$

While the fuel is burned within the reactor core, the amount of electricity generated per year, in kWh/yr, is proportional to the thermal power, thermal efficiency, and capacity factor as shown in Equation (8).

$$\mathrm{E}_{\text{Electricity Generated}} = \left(\mathrm{P}_{\text{Reactor Thermal Power}} \cdot \eta_{\text{Efficiency}}\right) \cdot \left(f_{\text{Capacity Factor}}\right) \tag{8}$$

Thermal power is limited by the reactor's NRC operating license and only changes through a power uprating if the reactor can handle the extra load. Thermal efficiency of a light water reactor is typically around 34% as a result of thermodynamics of the steam cycle. Capacity factor of the current nuclear fleet is around 0.9 due to longer burn up fuels and shorter and less frequent outages. Using a thermal power of 3,000 MWth, electricity generated is nearly 1,000 MWe. The total electrical production for the

year would be about 7.9E9 kWh. Nuclear waste fund contributions are determined as a flat rate of \$1mill/kWh electricity produced. The cost to the NWPA fee is given in Equation (9).

$$C_{\text{NWPA Fee}} = E_{\text{Electricity Generated}} \cdot c_{\$1\text{mill/kWh}} \tag{9}$$

Total cost to the NWPA fee would be around \$7.9 million for a yearly electrical production of 7.9E9 kWh for a single reactor.

Revenue from electrical sales is dependent the state or regional utility commission since the utility cannot set the price for electricity for the entire US fleet. Assuming that taxes, distribution losses, and the NWPA fee are excluded from the model, the total income from electrical sales is given in Equation (10).

$$P_{\text{Electricity Sales}} = E_{\text{Electricity Generated}} \cdot c_{\text{Consumer price in \$/kWh}} \tag{10}$$

Thus, a utility may increase profits in this scenario by increasing the time between refueling and outages as this pertains to optimal usage of the fuel while it is in the core. This is typically done by shaping the neutron flux distribution in core by varying enrichments in assemblies, using burnable poisons, and shuffling the fuel to maximize thermal energy production [4,5]. Higher enrichment fuels enable longer fuel burn ups, but increased enrichment costs are detrimental to the profit. Also, the upper limit for burn up is about 60 GWd/t due to the lifetime of the cladding materials.

A clear economic goal exists for the reactor owners to increase burn up when profitable, but as this is sought after, more fission products and transuranic elements are created, increasing the difficulties of long term waste storage, although utilities are not liable for storage after the initial cool down period [11]. Fuel discharged from the reactor is initially stored in a spent fuel pool to remove heat from the assemblies, during which time the decay of very short lived isotopes is the primary contributor to heat and radiation output. Depending on the reactor license and storage capacity, the assembly may be removed from the spent fuel pool and placed into dry storage in large casks for a interim storage that reduces the dose rate outside of the cask to safe levels for facility personnel to monitor the integrity of the cask. Casks for interim storage of the spent fuel assemblies are around \$120/kg of spent fuel stored, including assembly mass.

Currently, this is the end of the once through fuel cycle as the repository is not operational and is thus an open cycle where spent fuel is stored at many reactor sites around the country with the government funding the storage cost. Were the repository operational, spent fuel would be transported to YMP at an estimated one-time disposal cost of about \$900/kg of heavy metal including transportation and cask costs, under the assumption that the repository mass capacity is permanent [64]. Repository cost is shown in Equation (11), where the cost of storage term is discussed in Chapter 4.

$$\mathrm{C}_{\text{Repository}} = \mathrm{C}_{\text{Cost of Storage}} \cdot \mathrm{M}_{\text{Spent Fuel}} \tag{11}$$

If the funding for the YMP is reinstated and the facility begins accepting spent fuel, management of the spent fuel sent to the repository is rather important as this affects the capacity of the storage facility. Proposed loading schemes for the repository include variations on the loading concept, where higher heat producing casks may be placed in different tunnels to average heat load and maintain the passive heat removal requirement. In this research, approaches for loading of the repository are not addressed; however, methods for maximization of total repository heat load capacity are investigated through additions of delay times, reprocessing applications, and burner reactor usage.

Heat load of the repository is defined as the integral of the decay heat from the time of initial deposition through 1500 years, which allows adequate decay time for fission products and results in a decay heat term dominated by transuranic isotopes [17,73,74,75,76]. This integral decay heat term can be determined for individual assemblies or constituent isotopes, allowing for analyses to show key contributors over the duration of decay time as well as the effect of removing isotopes through reprocessing has on this integral.

While the effect of reprocessing is left to sections concerning advanced fuel cycles, the effect of a delay time between reactor discharge and deposition in the repository provides a significant decrease in repository heat load for the closed once through cycle scenario. Spent fuel is typically stored for about 15 years in wet storage after discharge to remove heat from the decay of short lived isotopes. The cost function is shown in Equation (12).

$$C_{\text{Wet Storage}} = C_{\text{Wet Storage Costs}} M_{\text{Mass Spent Fuel}} T_{\text{Wet Storage Duration}} \tag{12}$$

After this initial cool down period, spent fuel may then be transferred to dry storage if space in the pool is at a premium. Because fission products supply the majority of the integral decay heat for the first 200 years after discharge, one approach to repository heat load minimization is to not send the fuel to the repository until the fission products have decayed such that the long lived isotopes dominate the heat load. This is plausible because of industry's experience with storing spent fuel on site for over nearly 30 years already; thus, storing the fuel onsite at least until decommissioning is a historically supported option.

The DOE estimated the government's liability from YMP-related lawsuits to be nearly $7 billion in 2007, and would continue to increase every year after 2017 that the repository is not accepting spent fuel; therefore, this option would require more effort to pursue as utilities are not required to pay for dry cask storage because of the settlements [11,78]. Regardless of where the spent fuel is stored, a longer duration of storage will have an economic penalty in terms of regulatory and operational fees associated with managing increased storage capacity. Cost of interim storage is shown in Equation (13).

$$C_{\text{Interim Storage}} = C_{\text{Storage Material Costs}} M_{\text{Mass Spent Fuel}} + C_{\text{Operating Costs for Storage}} T_{\text{Post LWR Delay Time}} \tag{13}$$

The added cost of the delay time parameter is a function of the stored fuel mass, length of storage, and the cost per year per mass stored. The cost of storage includes

materials, regulatory recovery fees, and added operational costs. Costs that are likely to be incurred only by the government are collected in a long term storage term, shown in Equation (14), that may include a repository cost.

$$C_{\text{Long Term Storage}} = C_{\text{Interim Storage}} + C_{\text{Repository}} \tag{14}$$

Facility capital cost of a new reactor is introduced in Equation (15).

$$C_{\text{Facility}} = C_{\text{Initial Facility Cost}} \frac{\left[i \cdot (1+i)^{T_{\text{Repayment}}} \right]}{\left[(1+i)^{T_{\text{Repayment}}} - 1 \right]} \tag{15}$$

The amortized payment amount is determined using a mortgage payment function with an estimated interest rate (in annual percentage yield), i, over the lifetime of the loan, $T_{\text{Repayment}}$. The lifetime of the loan is assumed to be 40 years. Total operating costs are combined in Equation (16).

$$C_{\text{Total Cost of Operation}} = C_{\text{Facility}} + C_{\text{Operations}} + C_{\text{NWPA Fee}} + C_{\text{Wet Storage}} \tag{16}$$

The total cost of the fuel cycle, shown in Equation (17), is the sum of costs for operations, fuel, decommissioning, and long term storage.

$$C_{\text{Total}} = C_{\text{Total Cost of Operation}} + C_{\text{Total Fuel Cost}} + C_{\text{Decomissioning}} + C_{\text{Long Term Storage}} \tag{17}$$

Each term is given in units of dollars per year, with the terms being variable according to underlying distributions. Depending on the perspective, the total cost may be from the standpoint of government, utilities, or tax payers. The difference is important as some costs are not similar, specifically the repository cost of $1 mills/kWh is a cost for utilities, but may be considered revenue for the government. Due to recent litigation, the government must reimburse utilities for interim storage at reactor sites as a result of delays in opening the repository. Currently, utilities are not obligated to pay for on-site interim storage after the initial cool down period [4,11]. The total cost is also increased by including terms for facility and operating costs. Costs for facility and operations are subject to market forces that are difficult to predict for long term planning. Each term is also subject to individual interest formulas.

Levelized Unit Electric Cost (LUEC) is a term describing the cost of producing 1kWh of electricity using any form of power generation. LUEC is defined as an entity's total cost for the total electricity produced, given in Equation (18), with units of $mills/kWh. Nuclear power is less costly than other forms when compared on the basis of production costs, as shown in Table 2 [74].

$$\text{LUEC} = \frac{C_{\text{Total}}}{E_{\text{Electricity Generated}}} \tag{18}$$

Table 2: LUEC Comparison of Electrical Generation Methods in $mills/kWh [74]

Year	Coal	Gas	Nuclear	Petroleum
2008	27.5	80.9	18.7	172.6
2007	25.7	66.9	18.5	108.3
2006	25.3	69.5	18.9	102.8
2005	24.3	80.0	18.7	89.6
2004	22.4	64.2	19.5	65.4
2002	21.9	47.0	20.3	57.6
2000	21.5	72.8	21.7	65.1
1998	22.9	40.8	24.6	37.6
1996	24.2	45.7	25.3	59.5

Hybrid LWR and FR Fuel Cycle

Transitions from the open once through fuel cycle will most likely continue to use existing light water reactors as additional reactor types, particularly liquid metal cooled fast reactors, are introduced to the fuel cycle, creating a hybrid or one-tier fuel cycle. This option recycles plutonium from LWR spent fuel into a plutonium-rich fuel mixture containing transuranics and uranium, either in a metallic or Mixed Oxide (MOX) form. Fast reactors with low conversion ratios (amount of fissile material at discharge per amount fissile material at input) will reduce the plutonium stockpile and the repository heat load from long lived transuranics [34,35,64,65]. Some motivations for using this option include:

- Utilization of the existing fuel production facilities for LWR portion of fuel cycle
- Efficient use of enrichment tails
- Reduction of plutonium inventory for nonproliferation purposes
- Increasing repository capacity through removal of long lived actinides from spent LWR fuel to reduce the integral decay heat

Any deviation from the current once through cycle poses a significant capital investment to fund R&D, new reprocessing and fuel fabrication facilities, and plant construction. Comparing the mass flow diagram of the once through cycle in Figure 1 with the hybrid cycle shown in Figure 2, the cycles are nearly identical until LWR discharge where the fuel is now sent through the FR stages after a period of storage at the LWR site. In the reprocessing stage, minor actinides are fully partitioned to allow for control of FR fuel composition. For nonproliferation purposes, some suggest that the reprocessing plant and fuel fabrication facility be collocated in central or regional sites to avoid concerns of plutonium transportation [42,43,44,77]. Once fuel is delivered to the reactor, which may also be collocated with the reprocessing and fabrication facilities, the

fuel is burned for some duration that is dependent on physical aspects of the reactor core and fuel types [42,43,77].

After irradiation, the fuel is stored in a spent fuel pool, as is the case with a LWR, except the pool is a liquid metal rather than water in the case of the LMFBR [45]. A period of dry storage may be used once the short lived fission products have been given adequate time to decay, until final deposition in the repository. Some estimate that the added FR stages to the once through cycle create large repository capacity savings factors of nearly an 80 to 100 fold, while others estimate more moderate increases, nearly 4 to 15 fold due to methods for handling secondary waste streams [7,12,17,18,50,51]. Costs associated with the wet and dry storage of FR fuel are identical to those given in Equations (12) and (13), respectively.

Proliferation concerns are heightened from the lack of a repository because of the many locations storing spent LWR fuel, with each having significant amounts of plutonium in storage. PWR spent fuel contains, by weight percent, about 0.8% of plutonium, with .1% minor actinides that include americium and curium. A single PWR assembly contains about .5 metric tons of heavy metal (MTHM) at discharge, with nearly 10kg of that mass being fissile material. While about 60 assemblies are discharged each refueling in a PWR, the resulting spent fuel becomes a large stockpile of plutonium, which some believe could be either diverted for weapons programs or recycled into fuel, assuming adequate reprocessing technology is available.

Major contributors to the long term heat load in the repository is due to plutonium, americium, and curium, while shorter lived actinides and fission products are primary contributors in the first 200 years; thus, the repository heat load may be reduced by chemically separating spent LWR fuel into components of uranium, transuranics, and fission products, and then storing the fission products and reusing the uranium and transuranics as fuel for fast reactors. A mixture of uranium and plutonium fueled LWR's are used in France and were tested in the United States [20,77]. For a fast reactor, the target conversion ratio is dependent on reactor core design and reactor physics, which primarily determines the effectiveness in burning the plutonium. A low conversion ratio FR would have less U-238 than would a higher conversion ratio FR due to the fast neutron capture of U-238 that permits for some plutonium breeding in order to reach higher burn ups. Low conversion ratio reactors, also known as fast burner reactors, are intended to reduce both the plutonium stockpile and repository heat load. While the hybrid cycle has thus far appeared as a panacea for many of the problems facing future nuclear power, a major barrier to usage of the hybrid cycle is the economic cost associated with research, construction, and operating costs of reprocessing and fast reactor fuel fabrication facilities.

Economic cost is a large factor in the seemingly advantageous hybrid fuel cycle as it requires many new facilities and technologies to function, not all of which have been proven on a large scale; however, as the costs of uranium ore and spent fuel disposal increase with diminishing supplies and storage space, these large capital costs may not pose an insurmountable barrier to usage as expected [64,69]. Some costs are mitigated by using legacy LWR's for the duration of their licenses as newer Generation III+ and Generation-IV reactors are built, creating an overlap in technologies that does not require

a drastic shift in electrical generation capacity, but rather a controlled transition, with costs similar to those described in

Table 1. Because loans on older light water reactors are repaid during the initial 40 year licenses, reactor owners have less overhead cost associated with operating past the original life expectancy; thus, the NRC expects that nearly every reactor will opt for the 20 year license extension and possibly further extensions in the future with proper research, design, and maintenance [12,67]. With the 20 year extension, legacy LWR's will begin to go offline in the mid 2020's and all will be offline by 2040, barring further license extensions. Currently new reactors cannot be built quickly and in sufficient quantity enough to replace the current fleet let alone expansion given the current state of large commercial steel foundries; thus, a significant investment into nuclear power is required to entice foundries to reopen or re-task to produce reactor components.

Facility cost estimates are required to provide a realistic expectation of cost associated with R&D, construction, and operation. Estimates for the cost of a combined reprocessing and fuel fabrication facility are with undetermined uncertainty since as no similar facility has been recently built; thus, the costs for existing reprocessing plants are scaled by capacity and combined with cost estimates for the fuel fabrication facility, and are calculated similarly to the reactor facility cost in Equation (15). A reprocessing facility capable of handling the United States for AFCI goals would have a capacity between 1500 MTHM to 3000 MTHM per year. This includes cladding and fuel assembly structural materials, which correspond to handling all the spent fuel waste for a range of 60 to 120 PWR's, respectively.

Costs of the hybrid cycle are also combined into subsets of fuel, operations, D&D, and repository. Total fuel costs for the fast reactor fleet are shown in Equation (19).

$$C_{\text{Total FR Fuel Cost}} = C_{\text{Cost of Reprocessing}} \cdot M_{\text{Reprocessed LWR SF Mass}} + C_{\text{FR Fuel Fabrication}} \cdot M_{\text{FR Fuel}} \tag{19}$$

Cost of long term storage is calculated using Equation (20).

$$C_{\text{FR Long Term Storage}} = C_{\text{Interim Storage}} + C_{\text{Repository}} + C_{\text{Reprocessing Waste}} \tag{20}$$

Total operational cost is calculated using Equation (16). Total cost of the FR portion of the hybrid cycle is calculated similarly to the LWR tier in Equation (17). The entire combined one tier (hybrid) fuel cycle cost is shown in Equation (21).

$$C_{\text{Total}} = C_{\text{Total LWR}} + C_{\text{Total FR}} \tag{21}$$

LUEC of the hybrid cycle is calculated using Equation (18), where the electrical generation is combined according to Equation (22).

$$E_{\text{Total Electricity Generated}} = E_{\text{LWR Electricity Generated}} + E_{\text{FR Electricity Generated}} \tag{22}$$

Estimated costs are provided in Table 3 [64,65,69,70]. The cost of the metal fast reactor is higher than that of the oxide reactor because of a higher power density for the

same energy production; thus, these values are similar when normalized to energy production.

Summary

The once through and hybrid fuel cycles have been described in terms of detailed mass flow equations and economic costs. Equations presented in this chapter form the basis of the analytical methods described in chapter 4.

Table 3: Estimated Amortized Nominal Costs Associated with the Hybrid Fuel Cycle after LWR Stages with Referenced Page Numbers Provided [64]

Stage	Capacity	Cost
UREX Reprocessing (F1-4)	3,000 MTHM/yr	$502/kgHM
Uranium Disposal (K2-1)	47 MTU/yr	$5/kgU
Fuel Fabrication (D1-1)	4.5 MTHM/yr	$5,000/kgHM
Fast Reactor (D1-12)	Dependent on Burn Up	$5,150/kgHM (Metal) $1,650/kgHM (Oxide)
Pyro Reprocessing (F2-2)	500 MTHM/yr	$2,700/kgHM
Repository (L-9)	63,000 MTHM	$528/kgHM

CHAPTER IV
Analysis Methodology

Overview

Development of an advanced fuel cycle management tool begins with a detailed mass flow model that is expanded to include reactor physics, policy considerations, and economic forecasting. Each fuel cycle modeled has a specific set of mass flow equations that are evaluated within the Matlab toolbox environment, which provides a function toolbox of distribution sampling, data interpolation, economic, and optimization routines written for this task. User input is read into the sampling routine, which parses distribution information to sample fuel cycle parameters. These parameters subsequently drive the mass flow models and call the interpolation function to provide data for isotopic and heat load analysis. Economic analysis assigns costs to various aspects of the fuel cycle. An analysis code specific to a fuel cycle design may be used in either the uncertainty analysis or genetic algorithm optimization routines, as parameter sampling is a common point of entry to the analysis. In the optimization routine, the target is defined by the user as any one of the variables present in the output structure; thus, optimization of any parameter is possible, including economic cost, isotopic inventory, or required repository capacity. With this methodology, a tool for policy and decision makers in designing future fuel cycles is created that examines the case of an equilibrium once through and one-tier fuel cycle.

Fuel Cycle Analysis Toolbox Design

The analysis tool was created in the Matlab development environment where the user provides input to a driver function that manages the execution of the requested routines, as shown in Figure 3. Matlab was chosen for speed of execution, multiprocessing capability, ability to handle very large data sets, and it is an extensible framework that can be upgraded in functionality with subsequent releases without significant code rewrites. The bulk of the analysis toolbox consists of helper functions that direct data flow and pass a common user-defined input structure.

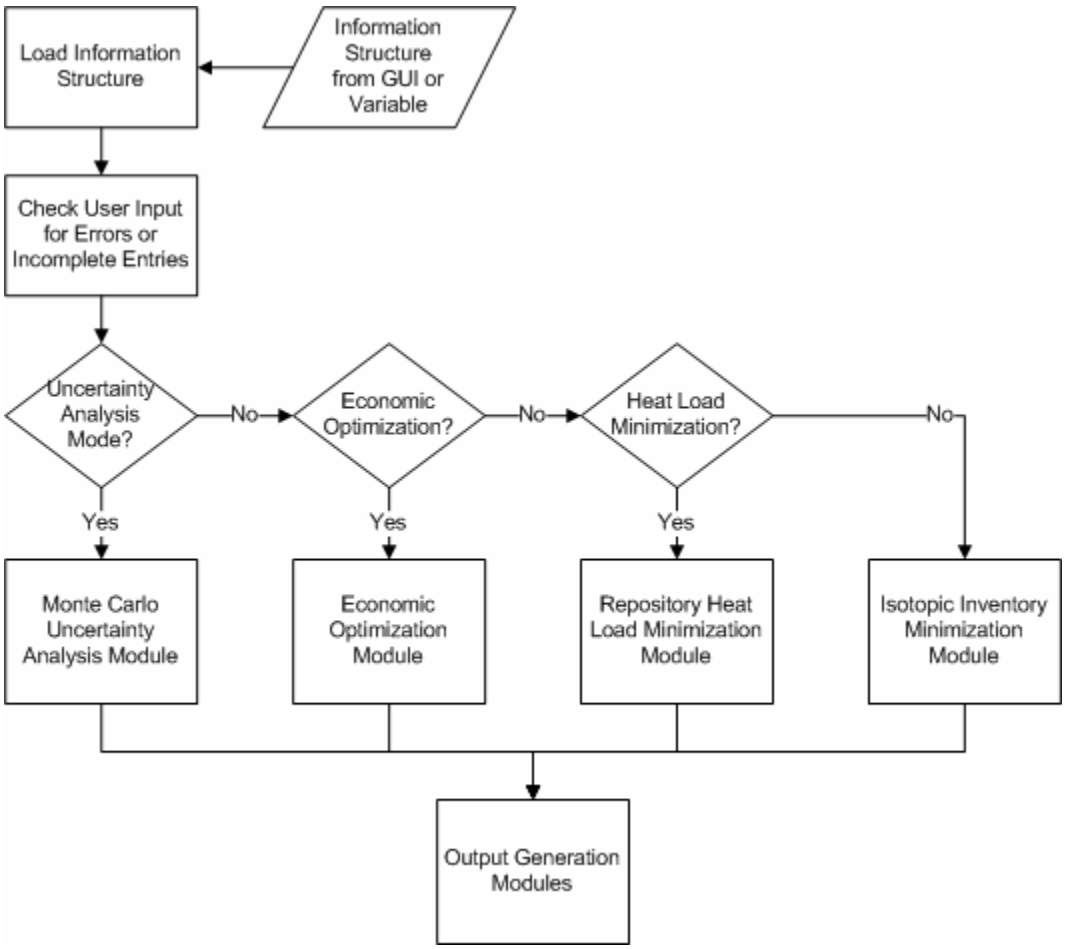

Figure 3: Analysis Tool Flow Chart of Driver Program

User input directs nearly every aspect of code operation and consists of a data structure that contains information as to which modules the driver program should select, fuel cycle model to evaluate, parameters or distributions are to be used in the model, etc. Input structures are broken into sections or fields that contain similar information such as attributes pertinent to analysis or requested outputs. Examples of input structures are given in Appendix E for uncertainty analysis and optimization modes.

A naming convention has been designed for consistency between fuel cycle models. Matlab has a variable type or class called a structure, similar to those in C++, which is comprised of fieldnames that represent other variables or cascaded structures. In the data structure, information on the analysis method is in the fieldname, ***analysis***, reactor data are stored in the fieldnames ***LWR*** or ***FR***, and economic information is stored in the fieldname ***econ***. For the once through example, the data structure ***FuelCycleParameters*** contains the fieldname ***LWR***, which contains parameter and distribution types relevant to LWR operations. The fieldname, representing a second structure, contains the fieldnames ***budist*** and ***burange***. The former is a string value that gives the name of the sampling function to use when selecting values during runtime,

while the latter gives bounds for the sampling function. Sampling functions available include ***single***, ***uniform***, ***triangular***, or any of the intrinsic Matlab distributions. Distributions created are assumed to have boundaries; thus, care should be taken when evaluating with an unconstrained sampling function, such as the normal distribution, as values may be well outside of the realistic range. The ***single*** distribution function is the default and takes a scalar value unless otherwise directed by the user.

The ***analysis*** fieldname is also a structure that contains the fieldname ***mode***, which directs the driver program to run either the optimization or uncertainty analysis modules. The uncertainty analysis module requires an additional fieldname for the number of trials, ***numtrials***, and it straightforward as to the execution. In the optimization module, fieldnames specific to Genetic Algorithms are required for runtime. Variables for population size, number of generations, iteration termination limits, and plotting functions are expected; however, some of these have default values built into the analysis. The Genetic Algorithm in the code has been written specifically for this analysis, but may also be replaced by the Matlab toolbox version if available. For the optimization case to run properly, the fieldname ***optimization.fitness*** must be set as a string value of the output structure's fieldnames. Results are given in the structure, ***results***, which contain identical fieldnames to the input structure. An economic optimization of the LUEC would point the input for fitness function to the result structure fieldname, ***results.econ.luec***. The Genetic Algorithm would then look to this value when sorting the population according to best values.

Uncertainty Analysis Modules

The uncertainty analysis module, shown in Figure 4, begins by evaluating a common base of parameters for the once through fuel cycle since LWR's provide the foundation for both the once through and hybrid fuel cycles.

In block A, which is evaluated for both cycles, the first function performs the Monte Carlo sampling of the LWR distributions for initial enrichment, burn up, and delay time as defined in Table 4. All of the trials are sampled simultaneously before the analysis code is executed to "vectorize" the code for a reduction in computation time and to provide capability for plotting results against sampled values to determine trends. An isotopic database is sampled using either a generalized regression neural network, described in Appendix D, or a multiple linear interpolation, which can be used depending on the user input and data constraints. Dimensions of the isotopic database are enrichment, burn up, decay time, mass, time dependent decay heat, and integral decay heat.

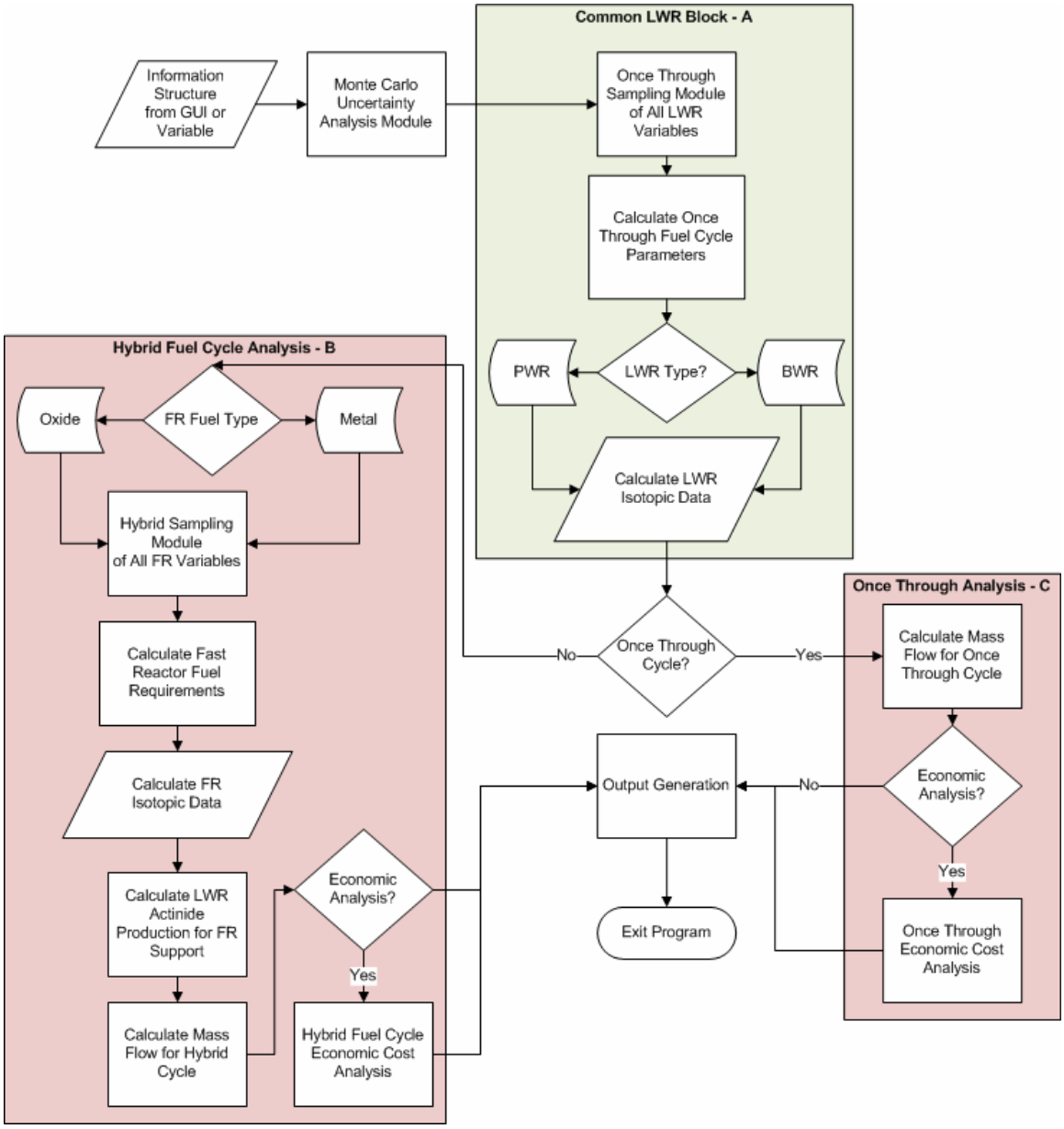

Figure 4: Uncertainty Analysis Module

Table 4: Light water reactor parameter distributions

Parameter	Distribution	Low Value	High Value	Peak
Enrichment	Triangular	3.0%	5.0%	4.5%
Burn up	Triangular	30 GWd/t	60 GWd/t	45 GWd/t
Delay Time (After LWR Discharge)	Triangular	5 years	30 years	10 years

After the common LWR database is calculated, the user definition for fuel cycle type, either the once through or hybrid cycle is used to direct the flow of subsequent analysis to either Block C for the once through fuel cycle or Block B for the hybrid fuel cycle. In both cases, the mass flow model begins with determination of parameters that are common to any reactor type.

Reactor Independent Mass Flow Derivation

Analysis of the nuclear fuel begins with definitions of the demand driven mass flow throughout the cycle. A demand driven model is where the reactor mass input requirements back-propagate the mass requirements for fuel production. In the case of the nuclear fuel cycle, the mass flow of the cycle begins with the amount of fuel required by a reactor, assuming that the individual requirements of the reactor do not exceed the fuel production infrastructure capacity.

The steady state mass flow balance of a nuclear reactor is the basis for analysis, given by Equation (23).

$$\dot{M}_{in} = \dot{M}_{out} \tag{23}$$

In Equation (23), there is technically a loss of input mass because of the mass to energy conversion, but this is several orders of magnitude smaller than the masses and is therefore neglected in calculations. Rate terms are annualized as this analysis is concerned with the equilibrium case rather than a time dependent fuel cycle that has dependencies on refueling outage durations or other intermittent states of operation.

Depending on the reactor type specified by the user, the thermal energy production or the efficiency should be specified. Assuming that the thermal power and steam cycle efficiencies are known for the reactor, the electrical generation may be determined by Equation (8), which is equivalent to Equation (24).

$$\mathrm{E}_{\text{Electricity Generated}} = S_{\text{Reactor}} \cdot M_{\text{Core}} \cdot \eta_{\text{Efficiency}} \cdot \left(f_{\text{Capacity Factor}} \right) \cdot \left(g_{\text{Unit Conversion}} \right) \tag{24}$$

Where M_{Core} is the mass of the reactor core, $g_{UnitConversion}$ is a conversion factor, and $S_{Reactor}$ is the specific power that is be determined by Equation (25) in units of GW per metric ton of fuel.

$$S_{\text{Reactor}} = \frac{\text{P}_{\text{Reactor Thermal Power}}}{M_{\text{Core}}} \tag{25}$$

Evaluations are performed as though the fuel cycle were operating in steady state; thus, a common time interval is used to convert all rates to a per year basis. This equalization of terms begins with determination of the length of time between refueling, $T_{Refueling}$, shown in Equation (26), in units of the number of core refuelings per year.

$$\text{T}_{\text{Refueling}} = \left(\frac{1\text{yr}}{365\text{days}} \right) \cdot \left(\frac{\text{BU}}{\text{S}_{\text{Reactor}}} \right) \cdot \left(\frac{1}{\text{N}_{\text{Batches}}} \right) \tag{26}$$

Where $N_{Batches}$ is the number of batches in the fuel and BU is the fuel burn up, shown in Equation (27), with units of GWd per metric ton of fuel. In-core fuel is divided into batches that correspond to the length of time spent in the core, where multiple batches are used to achieve higher burn ups in order to keep the flux profile of the reactor constant over time.

$$\text{BU} = \frac{\text{P}_{\text{Reactor Thermal Power}} \cdot T_{\text{Operation}}}{M_{\text{Core}}} \tag{27}$$

Where $T_{Operation}$ is the length of time the fuel is burned in the core. The number of fuel batches typically ranges from 3 to 5 for a Gen III+ LWR, where each batch has resided in the core for a different amount of time.

The fuel burnup and energy demand specification determine the fuel mass flow through the reactor. The peak fuel burn up is a regulatory limit placed on the reactor design and operating license; thus, the maximum burn up is a parameter that can be used in calculating other terms as is discussed later. Included reactor data is restricted to the average burn up ranges from 30 GWd/t to 60 GWd/t for LWR's and between 80 GWd/t to 180 GWd/t for some fast reactors, as described in Appendix C.

With the length between refueling defined, the next step towards determining the equilibrium cycle's average yearly mass requirement is defining the total number of refuelings required over the lifetime of the reactor. Equation (28) uses the relation of the reactor lifetime, $T_{Lifetime}$, to the time between refueling to determine the number of refueling, $N_{Refuelings}$.

$$N_{\text{Refuelings}} = \frac{T_{\text{Lifetime}}}{T_{\text{Refueling}}} \tag{28}$$

The sum of the number of $N_{Refuelings}$ and the initial core mass, M_{Core}, determine the total mass used in the reactor, shown in Equation (29).

$$M_{\text{Total}} = M_{\text{Core}} \cdot \left(\frac{N_{\text{Refuelings}}}{N_{\text{Batches}}} + 1 \right) \tag{29}$$

A reactor begins operation with a single core loading, represented by the 1 in the parenthesis. Each refueling replaces a fraction of the core, which can be multiplied by the total number of refuelings to determine the total mass throughput over the lifetime of the reactor. Because this calculation is independent of the fuel composition, it is performed in both once through and hybrid cycle analysis.

Once the total mass is determined, the amount can then be averaged into a per year amount that is used as a common denominator for analysis, shown in Equation (30).

$$\dot{M}_{in} = \frac{M_{\text{total}}}{T_{\text{lifetime}}} \tag{30}$$

The mass input rate is the same as that defined in Equation (23), although this is scaled according to the number of reactors in the system. Equation (30) is utilized in both the once through and hybrid cycle analyses.

In the fuel production pathway leading to the reactor stage, the type of reactor and fuel defines the origin of the fuel used in that stage. In the case of LWR's in a once through cycle, fresh fuel is produced from uranium enrichment, while a fast reactor in a hybrid cycle would have fuel produced from reprocessed LWR fuel. In either case, the basic mass flow equations presented thus far are applicable to any reactor in an equilibrium fuel cycle.

Integral Decay Heat Calculation

Repository heat load analysis is performed by evaluating the decay heat and the integral decay heat. Isotopic decay heat data is obtained directly from ORIGEN-S calculations in SCALE, as with the isotopic mass data [79,81]. Decay is stored in the databases according to burn up, enrichment, and time after discharge for LWR's. Databases for FR's are stored according to burn up and target conversion ratio in FR's. Interpolation of these data points will provide values for the instantaneous decay heat at some point in time after discharge up to 1500 years for all isotopes listed in Appendix B.

Integrations of the decay heat data are stored in a database of integral decay heats. Integrations are performed using the trapezoidal rule, given in Equation (31), where t_a and t_b correspond to times after discharge at which ORIGEN-S decay heat and mass data was created, and $H(t_a)$ and $H(t_b)$ correspond to the decay heat data at their respective time steps. An array, d, is created for each isotope, *i*, for each set of reactor operating parameters, *parms*, such as fuel burn up. The array is indexed according to the low decay time, t_a.

$$d_{i,parms}\left(t_a\right)=\left(t_b-t_a\right)\left[\frac{H\left(t_a\right)-H\left(t_b\right)}{2}\right] \tag{31}$$

Equation (32) uses the integral array to produce the integral decay heat database, D, where *cumsum* refers to a cumulative summation formula, $T_{Initial}$ corresponds to the first time after discharge for which data exists, and T_{End} is the last time in the decay (1500 years).

$$D_{i,parms}(t)=\sum_{t=T_{Initial}}^{T_{End}-1} cumsum\left(d_{i,parms}\left(t\right),d_{i,parms}\left(T_{End}-1\right)\right) \tag{32}$$

Because of the cumulative summation, the integral decay heat always refers to the integral decay heat from some time after discharge to 1500 years.

Once Through Fuel Cycle

Using the basic mass flow from Equation (23) as a foundation for analysis, the fuel cycle mass flow is completed by determining the underlying mass equations for input and output terms as they relate to the preceding stages of fuel production and the succeeding stages for spent fuel storage. Continuing from the generic reactor model described thus far, the once through design in Figure 5 continues with the fuel production pathway calculations after yearly mass input requirements in Block B.

Mass input into the LWR consists of uranium oxide fuel, which has a certain U-235 enrichment that requires inclusion of the enrichment process in the analysis. Beginning with the amount of enriched fuel required, the amount of feed material can be determined. Equation (2) uses the required core mass to determine the amount of natural uranium that must be mined, milled, converted to UF_6, and then delivered to the enrichment facility.

Mass output from the mass balance formula in Equation (23), specifically isotopic mass output, is a function of the reactor's operating characteristics, namely burnup. Isotopic masses are determined using the SCALE software package to simulate the reactor (described further in Appendix C) [79]. The analysis package models the reactor's fuel pin geometry, operating characteristics, charge masses, and decay times, all to produce a time-dependent isotopic composition database. After discharge, the fuel may reside in a spent fuel pool for some time before being sent to reprocessing or the repository, which is referred to as the delay time. Depending on the reactor type and fuel cycle characteristics, the destination of the fuel output is fuel cycle specific and will be described according to the specific fuel cycle later.

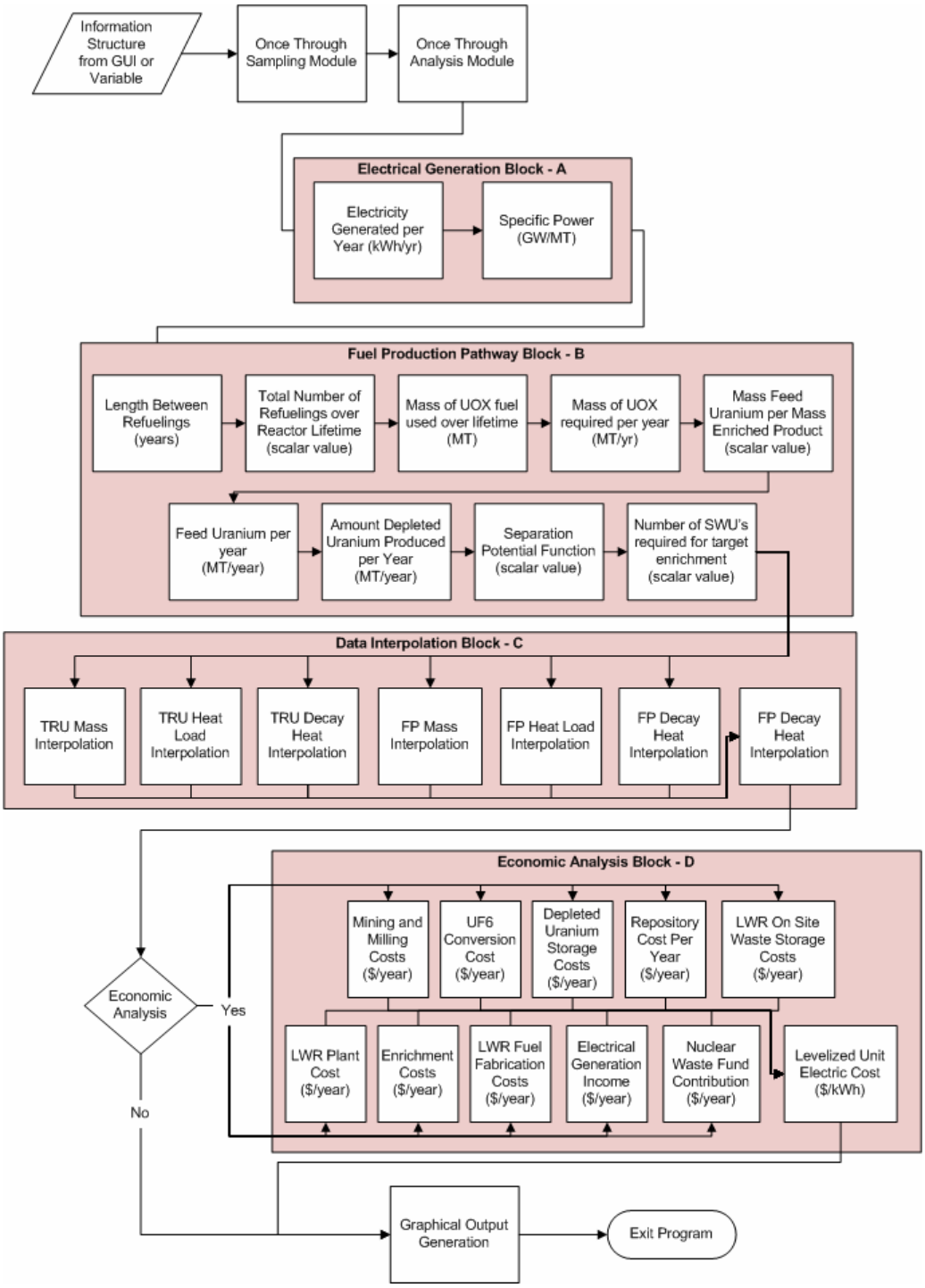

Figure 5: Once Through Fuel Cycle Flow Chart

In this research, emphasis is on creating an effective analysis tool; thus, reactor simulation is performed a single time and results are stored in a multidimensional database that can be interpolated by the analysis program to obtain data without the need for repeating reactor physics calculations when parameters change. Output from the SCALE package includes isotopic mass and decay heat for 66 isotopes. Totals of all fission products and actinides are tabulated separately [79,80]. Each isotope has a time history from discharge until 1500 years after discharge, allowing for delay time-dependent analyses to be performed. Decay heat data is integrated by the trapezoidal method in order to produce a term for integral decay heat from some time after discharge until 1500 years after discharge. Further explanation is available in Appendix C. A data preparation tool is provided in the code to convert from ORIGEN-S output to the specified format because data processing only needs to be performed a single time [81].

The data interpolation step replaces reactor simulation by using the pre-calculated data files and provides isotopic information for a range of reactor operation parameters. Information stored in the database may be interpolated and included in the analysis results; thus, interpolation steps are included for mass, decay heat, and integral decay heat and are performed for actinides and fission products separately. Interpolation calls a multiple linear interpolation function or neural network (Appendix D). Individual isotopes must be selected numerically from the list in Appendix B. Isotopes of interest, such as Pu-239 or Cs-137, may be selected for analysis rather than performing interpolations for 68 separate terms. In Figure 5, block C illustrates how the interpolations are performed and how the interpolated data are organized into a results structure that may be passed to the user at the end of analysis.

Using the fission product and actinide mass totals from the interpolation stage, the annualized mass output from the system is determined by Equation (33).

$$\dot{M}_{\text{Out}} = \dot{M}_{\text{Actinides}} + \dot{M}_{\text{Fission Products}} = \sum_{\text{Actinides}} \dot{M}_{\text{i}} + \sum_{\text{Fission Products}} \dot{M}_{\text{j}} \tag{33}$$

The mass output term is the spent fuel mass, and is the summation of all isotopes present in the analysis. Each isotopic mass term is separately stored for further analysis. Because of the changes in isotopic composition as a result of nuclear reactions, the total mass output of the system is the summation of all isotopes present in the spent fuel. While the annualized mass output can be viewed as a single term, there is a time dependence due to the continual, and unequal decay rates of different isotopes. This results in the time dependence of the mass flow term that enables viewing the mass composition out of the reactor at any given time after discharge.

For the once through fuel cycle, the program may perform an economic analysis, Block D of Figure 5, that uses mass flow quantities previously described and assigns a cost to each term. Economic costs are all specified in terms of dollars per year for a single reactor, which may be scaled to apply to the total number of reactors in the system. Four cost areas are associated with the overall fuel cycle cost: fuel, operations, decommissioning, and long term storage. Stages within these groupings all have some associated annual costs that are dependent costs defined in

Table 1.

Beginning with the fuel cost grouping, the total fuel group cost is given in Equation (7). The mining and milling stage cost is calculated according to Equation (34), where the cost of uranium ore is giving in dollars per kg.

$$\mathrm{C}_{Mining/Milling} = \mathrm{M}_{\text{Nat U}} \cdot c_{\text{Uranium Ore}} \tag{34}$$

The U_3O_8 to UF_6 conversion cost is given in Equation (35), where the conversion cost is in dollars per kg.

$$\mathrm{C}_{\mathrm{UF_6\ Conversion}} = \mathrm{M}_{\text{Nat U}} \cdot c_{\text{Conversion Cost}} \tag{35}$$

Enrichment costs are more dependent on the energy required and operational costs to produce the product than the cost of the feed uranium; thus, the cost of enrichment a function of the number of SWU's, Equation (6), required to reach a specific enrichment, as shown in Equation (36).

$$\mathrm{C}_{\text{Enrichment}} = N_{\text{SWU per year}} \cdot c_{\text{Cost per SWU}} \tag{36}$$

Depleted uranium from the enrichment tails does not need to be stored in the repository when a less costly facility may be used because of the low radioactivity levels of the material. Costs for this storage are also on a mass basis, shown in Equation (37).

$$\mathrm{C}_{\text{DU Disposal}} = \mathrm{M}_{\text{DU}} \cdot c_{\text{DU Storage cost per mass}} \tag{37}$$

Fuel fabrication costs include transportation and facility operations, and are a function of the amount of fuel to be fabricated [66,70]. The resulting formula is shown in Equation (38).

$$C_{\text{LWR Fuel Fab}} = \dot{M}_{\text{In}} \cdot c_{\text{Fuel Fab Cost}} \tag{38}$$

Operational costs include facility costs, operations, NWPA fees, and on-site storage (not including interim storage). Facility costs are calculated according to Equation (15), where the resulting amortized cost is dependent on the financed terms and the initial estimated cost. Operational costs are sampled from the distributions in

Table 1when an uncertainty analysis is performed. NWPA fees are calculated according to Equation (9). Wet storage costs are defined in Equation (12). The total cost of operation is given in Equation (16).

Long term storage costs consist of the interim storage and repository costs. Due to the government covering both of these costs, they are left out of the total costs from the utility perspective. Long term storage is calculated according to Equation (14).

Repository cost is based on either mass capacity or decay heat capacity. The current license of YMP has the mass capacity limited to 63,000 metric tons of spent fuel,

where each unit of mass has a fixed cost, shown in Equation (39).

$$C_{\text{Repository}} = \dot{M}_{\text{Out}} \cdot C_{\text{Cost per unit mass deposited}} \tag{39}$$

The capacity for spent fuel storage can be converted into an equivalent unit of integral decay heat capacity by using a fuel burn up of 50 GWd/t as a reference decay heat. The conversion between integral decay heat capacity and mass capacity is shown in Equation (40), where $M_{SFMassCapacity}$ is the repository mass limit of 63,000 metric tons.

$$\mathrm{D}_{\text{Integral Decay Heat Capacity}} = \mathrm{M}_{\text{SF Mass Capacity}} \cdot \mathrm{B}_{\text{Heat Basis}} \tag{40}$$

The total cost of the repository can be used to define the cost per unit mass deposited in Equation (39).

$$C_{\text{Cost per unit mass deposited}} = \frac{\mathrm{C}_{\text{Total Repository Cost}}}{\mathrm{M}_{\text{SF Mass Capacity}}} \tag{41}$$

Or equivalently to define the cost per unit integral decay heat in Equation (42).

$$C_{\text{Cost per unit IDH deposited}} = \frac{\mathrm{C}_{\text{Total Repository Cost}}}{\mathrm{D}_{\text{Integral Decay Heat Capacity}}} \tag{42}$$

Equation (39) can also be converted using the same factor to produce the repository cost as shown in Equation (43).

$$C_{\text{Repository}} = D\left(T_{\text{Delay Time}}\right) \cdot C_{\text{Cost per unit IDH deposited}} \tag{43}$$

D is the IDH of the spent fuel at a given time after discharge, $T_{\text{Delay Time}}$. Thus, the method for repository cost calculation is dependent on selection of either a mass or IDH basis. Each of the aforementioned economic functions may be used in the estimation of the LUEC, given in Equation (18).

Upon exiting the program, control of the once through module is returned to the calling function, either the uncertainty analysis or optimization modules. Output data consists of all variables and calculated values from the module. If a user wants to reduce the size of the output, then this is performed in the calling function according to the user-defined input structure.

Hybrid Fuel Cycle Module

A hybrid or one-tier fuel cycle consists of two reactor technologies existing in the same fleet, where one is dependent on the other for fuel. In the case of the hybrid LWR-FR cycle, LWR spent fuel is used in production of fuel for fast reactors. The mass flow is evaluated according to the flow chart in Figure 6.

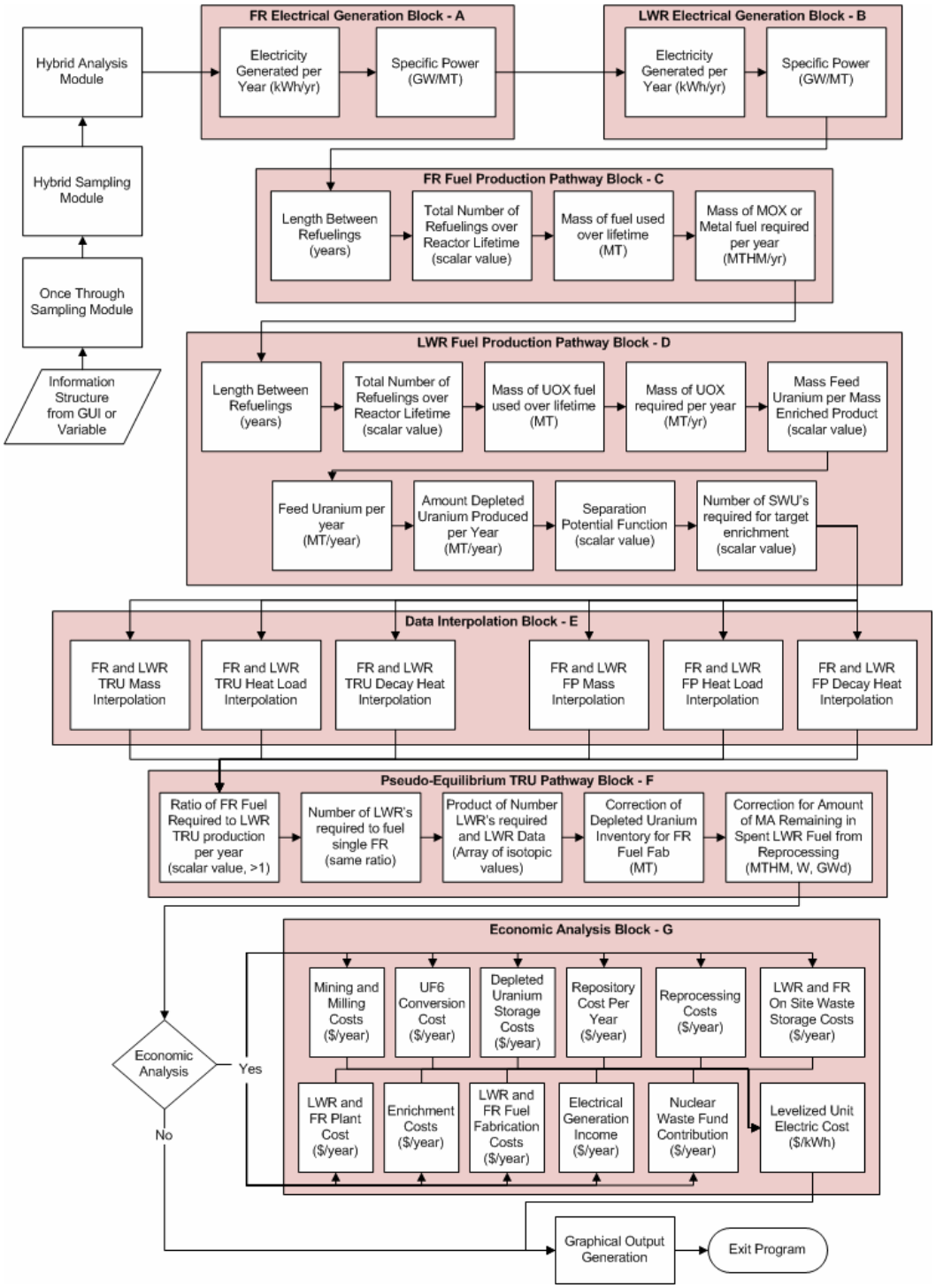

Figure 6: Hybrid Fuel Cycle Fuel Cycle Flow Chart

Using many of the same functions as in the once through case, the hybrid cycle appends reprocessing and fast reactor stages before deposition in the repository. The hybrid cycle begins with the fuel production pathway terms before data interpolations. The electrical generation terms, shown in blocks A and B of Figure 6, determine electricity generated per year, in kWh/yr, and specific power, in GW/MT, for both reactor technologies. After the generation blocks, the reactor types are treated differently. Block D contains the same code as in the once through module, shown in block B of Figure 5, which is the LWR fuel production pathway as described in the previous section. Mass flow through the production pathway in the hybrid cycle differs from the once through only after the LWR reactor stage.

Block C of Figure 6 illustrates the mass flow through FR fuel production. Calculations for time between refueling, number of refuelings, and total mass consumed over the lifetime, Equations (26), (28), and (29), respectively, are similarly applied to a FR as to a LWR.

The fast reactor mass input requirements differ from the LWR in that the fuel is not uranium enriched to some percentage, but rather a combination of many minor actinides that make up the fuel. Charge mass is the weight percent of the fuel's isotopic composition used in reactor physics calculations. The yearly mass flow term from Equation (30) is multiplied by the each member of the array of isotopic compositions, given in weight fraction, to create an array of isotopic inputs. Equation (44) provides the yearly mass of each charge mass isotope, denoted *i*.

$$\dot{M}^{FR}_{\text{Charge mass in}}(i) = \dot{M}^{FR}_{in} \cdot w(i) \tag{44}$$

Where $w(i)$ is the weight fraction of each isotope in the fuel. Data interpolation steps are identical for both reactor types, shown in block E of Figure 6, unless using the forced removal option as discussed later in this section. Interpolation functions selected by the user calculate terms for isotopic mass and decay heat based on sampled fuel cycle parameters used in optimization and uncertainty analyses. Available functions include generalized regression neural networks, cubic splines, and multiple linear interpolations; however, the default method uses multiple linear interpolations because the data spacing is sufficiently small. Data are scaled to proper units where data stored is on the basis of either one metric ton or some fraction thereof due to data handling in SCALE.

Actinide recycling is handled by three possible modes of calculation:

1. Set number of LWR's and variable number of FR's
2. Set number of FR's and variable number of LWR's
3. Set number of FR's and set number of LWR's

In the mode where a set number of LWR's exist, the goal is to determine the maximum number of FR's that may be fueled by that fleet. The isotopic masses from the LWR spent fuel as some time after discharge determine the number of LWR's required to fuel a single fast reactor, shown in Equation (45).

$$P_{\text{Ratio LWR to FR}} = \begin{cases} \max\left(\dfrac{\dot{M}^{FR}_{\text{Charge mass in}}(i)}{\eta_{\text{Relative}}(i) \cdot \dot{M}^{LWR}_{Out}\left(T^{\text{LWR}}_{\text{Delay Time}}, i\right)} \right) & \dot{M}^{LWR}_{Out} \le \dot{M}_{\text{Reprocessing Capacity}} \\ \max\left(\dfrac{\dot{M}^{FR}_{\text{Charge mass in}}(i)}{\eta_{\text{Relative}}(i) \cdot \dot{M}_{\text{Reprocessing Capacity}}\left(T^{\text{LWR}}_{\text{Delay Time}}, i\right)} \right) & \dot{M}^{LWR}_{Out} > \dot{M}_{\text{Reprocessing Capacity}} \end{cases} \quad (45)$$

Where $\eta_{\text{Relative}}(i)$ is the separation efficiency of reprocessing for each isotope processed. The ratio of LWR's to FR's is then scaled according to the number of LWR's present in the fleet, shown in Equation (46), where floor refers to the intrinsic function to round a decimal down to the nearest integer.

$$N_{FR} = floor\left(\frac{N_{LWR}}{P_{\text{Ratio LWR to FR}}} \right) \quad (46)$$

In the case for a set number of FR's with a variable number of LWR's, Equation (45) is also used as this scales according to the number of FR's present to determine the number of LWRs, shown in Equation (47). The intrinsic function ceiling is used to round the decimal up to the nearest integer as there must be at least this number of LWR's to support the specified FR fleet.

$$N_{LWR} = ceiling\left(N_{FR} \cdot P_{\text{Ratio LWR to FR}} \right) \quad (47)$$

The final case where the entire reactor fleet consists of a set number of FR's and LWR's again uses Equation (45) to verify the feasibility of the specified quantities. In each case, isotopic inventories are subject to the mass balance in Equation (48).

$$\dot{M}_{Excess}(i) = N_{LWR} \cdot \dot{M}^{LWR}_{Out}\left(T^{\text{LWR}}_{\text{Delay Time}}, i\right) - N_{FR} \cdot \dot{M}^{FR}_{\text{Charge mass in}}(i) \quad (48)$$

The isotopic mass excess refers to the isotopic mass that is not used in the creation of fast reactor fuel and will be deposited in the repository or specified for use as a target using the forced reduction model. For repository deposition, the total isotopic mass addition to the repository is given in Equation (49).

$$\dot{M}(i) = N_{FR} \cdot \dot{M}^{FR}_{\text{Out}}\left(T^{\text{FR}}_{\text{Delay Time}}, i\right) + \dot{M}_{Excess}(i) \quad (49)$$

Repository IDH for the hybrid cycle is given by Equation (50), where D_{Total} refers to the total IDH added to the repository yearly.

$$D_{Total} = N_{FR} \cdot D_{FR}\left(T^{FR}_{\text{Delay Time}}\right) + N_{LWR} \cdot \sum_i \left(D_{LWR}\left(T^{LWR}_{\text{Delay Time}}, i\right) \cdot \left(1 - \frac{N_{FR} \cdot \dot{M}^{FR}_{\text{Charge mass in}}(i)}{N_{LWR} \cdot \dot{M}^{LWR}_{\text{Out}}\left(T^{LWR}_{\text{Delay Time}}, i\right)} \right) \right) \quad (50)$$

Fast reactor isotopic data has a large uncertainty associated due to many factors described in Appendix C; thus, functionality is included that allows the user to specify mass removal fractions and force a reactor simulation rather than rely on the provided data files. One significant issue is that cross section libraries for fast reactors were not available in SCALE when these calculations were performed. This removes the fast reactor interpolation step and introduces new steps for decay heat scaling and mass determination, shown in Block F of Figure 7.

Forced removal assumes an annualized mass rate into the fast reactor direct from reprocessing, with isotopic composition defined along with fractions for production or destruction. For example, to simulate a conversion ratio of 0.5, half of the fissile mass into the reactor is destroyed after burn up, therefore the fraction for each fissile isotope could be 0.5 or some weighted sum equaling 0.5. This mode also allows for non-charge mass isotopes to be included as targets where the reprocessed mass is evenly distributed between all of the fast reactors. In Equation (51), the fractional removal term, f, is multiplied by the charge composition, w, and core mass, M, to produce the mass output, which replaces the term produced from the interpolation block. Updates for decay chain information and decay heat terms must be updated accordingly.

$$\dot{M}^{Forced}_{Out}(i) = \dot{M}^{Forced}_{In}(i) \cdot f(i)_{\text{i}} \quad (51)$$

Without data files for a simulated reactor, the decay heat and integral decay heat terms must be estimated. A specific decay heat term can be estimated on a per gram basis for isotopes listed in Appendix B; however, since the code lacks capability for point depletion, estimation of the composition of fission products from the fast reactor is assumed to be similar to that of LWR's. Only the summation term is used for fission products and is used in scaling the value on the basis of fuel burn up, shown in Equation (52).

$$D^{FPs}_{FR} = \frac{D^{FPs}_{LWR}}{BU_{LWR}} \cdot BU_{FR} \quad (52)$$

The summation term from the LWR spent fuel is normalized to the burn up of the LWR, and is then multiplied by the burn up of the fast reactor fuel, resulting in an estimate of the fast reactor fission product integral decay heat or mass. Once the scaling is completed, the code follows the same path of correcting masses as in the interpolated data module.

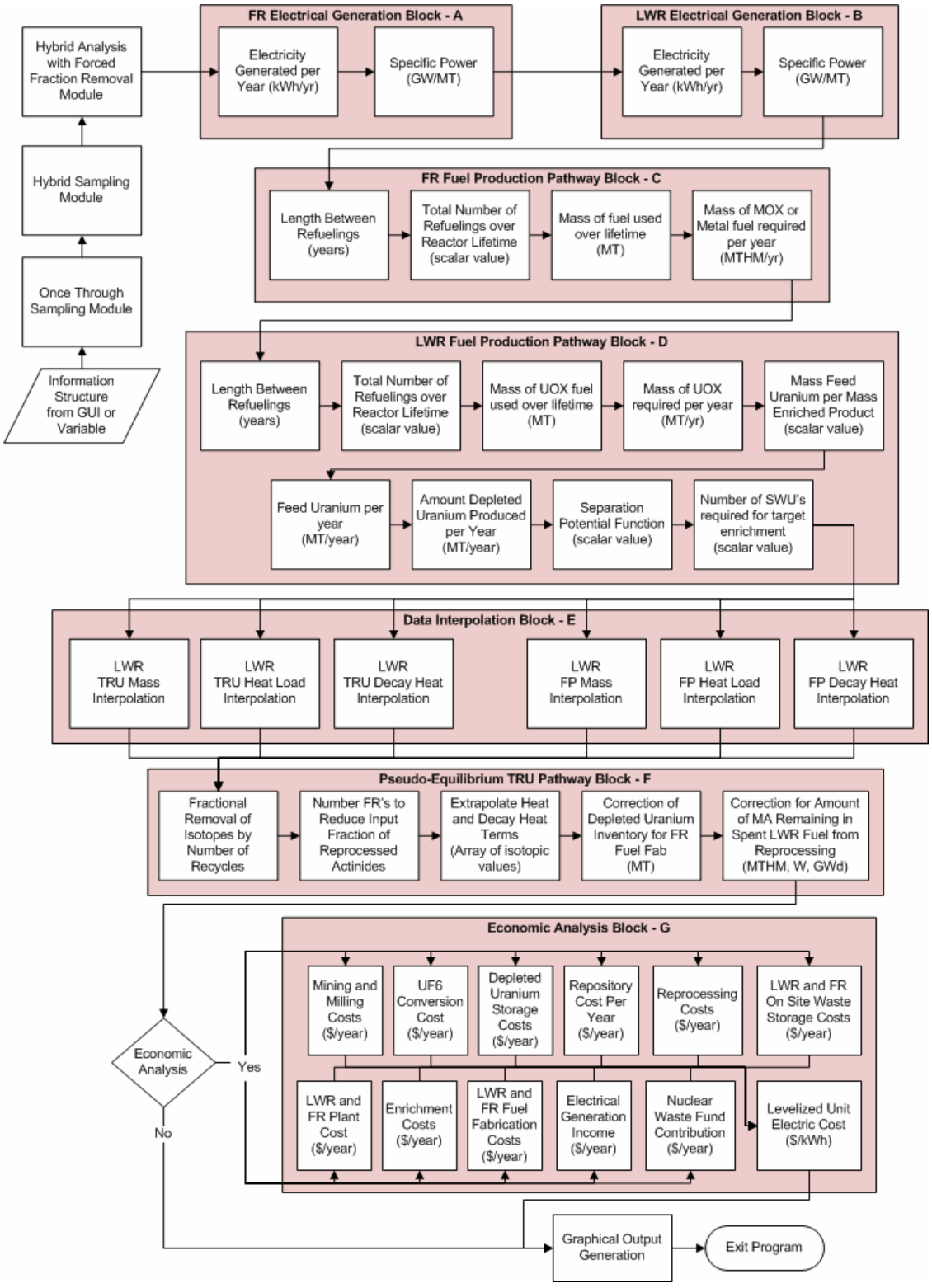

Figure 7: Forced Fractional Removal Hybrid Fuel Cycle Fuel Cycle Flow Chart

Fast reactor fuel production uses depleted uranium from the LWR enrichment process rather than reprocessing product, thus the amount of depleted uranium may be modified according to Equation (53). This term is calculated on a mass per year basis as done with the once through cycle.

$$\dot{M}_{\mathrm{DU}} = \dot{M}_{DU}^{\text{Enrichement Tails}} - \dot{M}_{DU}^{\text{FR Fuel Fabrication}} \tag{53}$$

Economic analysis of the hybrid cycle includes the terms for the once through cycle and additional terms for reprocessing, FR operation, and FR fuel fabrication. Each term is given in Equations (19), (20), and (21). LUEC of the hybrid cycle is calculated similarly to Equation (18) where the total cost of is determined in Equation (21) and total electrical production is determined in Equation (22).

Optimization

Optimization is performed using a single goal Genetic Algorithm (GA). Should the user choose an optimization function, the driver program directs the flow of the program to the optimization module. An outline of the optimization routine is given in Figure 8.

Because the procedure is meant to be flexible, user input defines the variable or variables to optimize rather than providing hard coded options. Optimization parameters are selected by the user in the input structure and will direct the genetic algorithm input in the number of parameters to optimize, limits on the magnitudes of the values chosen, and methods of choosing best individuals.

At the beginning of the routine, parameters independent of any optimization are defined and stored in a data set that is passed on to the optimization routine in Genetic Algorithm Block. Constant values may include any value as defined by the user in the input structure. Regardless of the optimization type chosen, an initial population is selected that includes all parameters required for the once through or hybrid fuel cycle modules. Using the output from these modules, which includes all calculated values for each individual in the population, a fitness or objective function is evaluated for the specific goal. The optimization procedure is same regardless of the choice in fitness value or function. The fitness parameter may be selected according to the specified goals such as isotopic minimization, repository heat load minimization, or economic cost minimization.

An isotopic minimization requires the user to select an isotope from the list in Appendix B and the algorithm will aim to minimize its inventory by varying other parameters. At the end of the once through or hybrid fuel cycle modules, the waste stream mass output is parsed for the isotope of interest and presented to the genetic algorithm for evaluation of stopping criteria or new population selection. Any variable in the output structure may be chosen as the fitness parameter, leaving the possibility for optimization of any parameter or variable in the computational model.

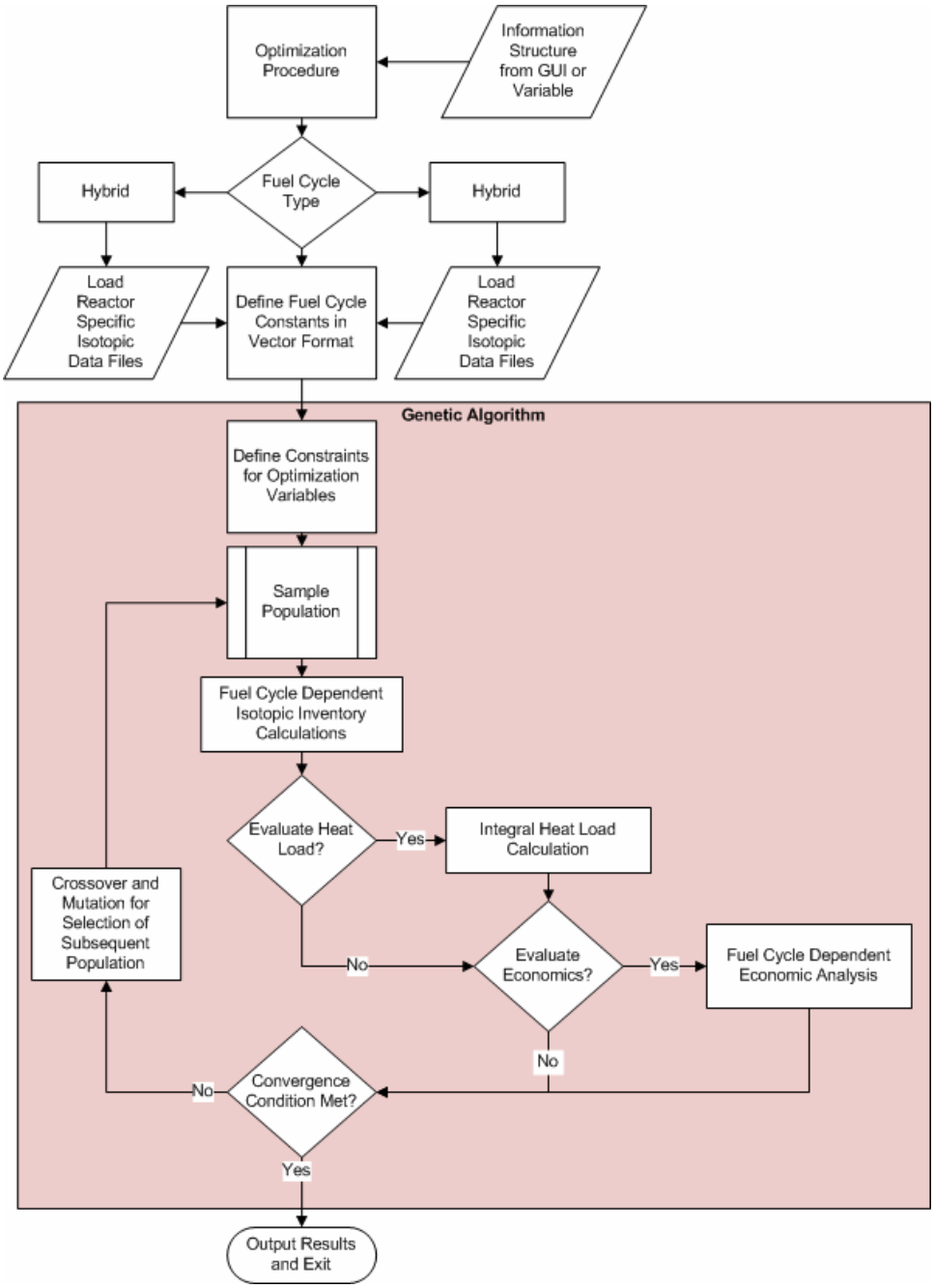

Figure 8: Optimization Procedure

For example, a repository decay heat minimization using the hybrid fuel cycle module is evaluated to return integral decay heat. Here, the number of LWR's and FR's is defined in the output with the length of the delay times. Using these parameters with the integral decay heats for the after reprocessing LWR data and FR spent fuel data, Equation (54) is used to define the contribution to integral decay heat on a per year basis.

$$D_{\text{Total IDH}} = N_{\text{FR}} \cdot D_{\text{FR IDH per year}} \cdot T_{\text{Reactor Lifetime}} + N_{\text{LWR}} \cdot D_{\text{LWR IDH per year}} \cdot T_{\text{Reactor Lifetime}} \quad (54)$$

Economic cost minimization is similar to the isotopic minimization case as the LUEC is used as the fitness value, which is an output of the once through and hybrid fuel cycle modules.

Once the fitness function is specified, the optimization routine manages the optimization process. First, the convergence parameter values are compared to stopping criteria for convergence or computing time for each iteration. Should the value not converge, a new population is created using the free optimization parameter values of the top 5 individuals, dropping the bottom 30% of individuals, and crossing over the top 70% of the population values to fill the population back to the set number. An adaptive mutation function is included that reduces the mutation as the number of generations increases, reducing the time for convergence.

For an arbitrary variable optimization, the user selects one of the inputs as the variable to optimize, which is entered as a string value equivalent to one of the result structure's fieldnames. Variables remaining that are not the target of optimization may either have bounds for constrained or blank fields for an unconstrained optimization. After the algorithm finishes, output returned to the user consists of the optimized parameters and a single can of the corresponding uncertainty analysis code with those parameters. A single Monte Carlo evaluation of 100,000 trials takes about 20 seconds on a dual core system, while an optimization case runs in less than 3 minutes, depending on the interpolation method selected.

Summary

The presented analysis methodology is rooted in basic mass flow equations. Models of the once through and hybrid fuel cycles have been produced that follow the mass flow throughout the fuel cycle and economic costs associated with different stages in the fuel cycle. The same models are used for uncertainty analysis or optimization purposes, and no not require modification for other reactor isotopic data unless the once through or hybrid cycle is not used.

CHAPTER V
Results and Discussion

Simulations of the once through and hybrid cycles are presented and compared in this chapter. Results are provided that demonstrate the functionality of the analysis code for uncertainty analysis, optimization, and single parameter variations. Comparisons are performed with regard to actinide inventory in repository waste, repository decay heat, and sustained economic cost of the fuel cycle. Optimization results include Pu-239 inventory reduction, repository heat load minimization, and LUEC minimization.

Once Through Fuel Cycle

Example Description

The developed theory is evaluated using best estimates for the once through cycle because of its long usage and available data for comparison. Isotopic and economic results are compared to other analyses using nominal values from the resulting distributions. A once through fuel cycle is modeled using the distributions given in Table 5. In order to keep consistent results with the current fleet of 104 reactors in operation, the number and type of reactors is set to 69 PWR's. Distributions for reactor operation parameters cover the range of those in the current PWR fleet. Internal variables, including separation efficiency are not varied but rather use nominal values reported by the United States Enrichment Corporation (USEC) and Nuclear Energy Institute (NEI) [37,67].

Economic costs are defined for each stage of the fuel cycle from plant capital costs to long term waste storage. New plant costs are estimated as n^{th} of-a-kind designs with a 5% interest rate on capital costs. Literature suggests that the cost of a new AP1000 was about 3 billion dollars as of 2006, and is currently estimated at about 7 billion dollars in 2010 [65,82]. Contributions to the nuclear waste fund are included at the constant rate of 1 mill per kWh generated, and this value has not changed since the initial passage of the NWPA in 1982 [4,5]. Repository costs are also unknown as the facility has yet to near completion let alone provide exact storage and handling costs per ton of spent fuel; thus, an estimate from the EMWG is used [70]. The EMWG estimated the repository cost based on differing repository capacities, where the \$528/kg and \$381/kg estimates assume theoretical capacities of 129,000 Metric Tons Heavy Metal (MTHM) and 270,000 MTHM, respectively. The current repository capacity is represented as the \$900/kg cost.

Table 5: Once Through Fuel Cycle Parameter Distributions Used in Analyses of PWRs

Parameter	Class	Distribution	Low Value	Peak	High Value
Number Reactors [37]	Fuel Cycle Specific	None		69 reactors	
Enrichment	Reactor Operation	Triangular	3%	4.5%	5%
Burn up	Reactor Operation	Triangular	30 GWd/t	45 GWd/t	60 GWd/t
Delay Time	Reactor Operation	Triangular	5 years	20 years	200 years
Reactor Lifetime	Reactor Operation	None		60 years	
Plant Cost [70]	Economic	Triangular	$1.5E9/reactor	$2E9/reactor	$3E9/reactor
Mining and Milling [70]	Economic	Triangular	50 $/kg	100 $/kg	150 $/kg
UF6 Conversion [70]	Economic	Triangular	5 $/kg	10 $/kg	15 $/kg
Enrichment [70]	Economic	Triangular	$100/SWU	$115/SWU	$130/SWU
Depleted Uranium Storage (per year stored) [70]	Economic	Triangular	$4/kgU	$8/kgU	$30/kgU
Fuel Fabrication [70]	Economic	Triangular	$210/kg	$220/kg	$264/kg
Onsite SNF Storage cost (per year stored) [70]	Economic	Triangular	$100/kg	$120/kg	$300/kg
Operating Costs [70]	Economic	Triangular	$50E6 /year/reactor	$75E6 /year/reactor	$100E6 /year/reactor
Repository Cost [70]	Economic	Triangular	$381/kg	$528/kg	$900/kg

Isotopic Inventory and Repository Heat Load

Simulations of the fuel cycle using the aforementioned distributions provide estimates on the annualized production of fission products and transuranic isotopes that will contribute to repository heat load. In Figure 9, the PWR contribution to integral decay heat (IDH) in repository is shown for various burn ups as a function of delay time before deposition. This and subsequent IDH related figures are based on the assumption that IDH is a discrete quantity that is summed for a total repository heat load rather than repeat integration steps after adding additional time-dependent masses to the repository isotopic mass tallies. The calculation theory described in chapter 4 makes use of this approximation.

Figure 9 shows that burn up has little effect on the integral decay heat when normalized to electrical energy produced. Fission product decay heat dissipates by a factor of 10 within a 150 year period, giving ample time for short lived isotopes to decay into stable isotopes. Short lived fission products are those with half lives less than 30 years, while long lived have half lives greater than 30 years. The fission product terms all appear to overlap due to the normalization to energy produced; however, the transuranic contributions are much higher and appear to spread out as a function of burn up. In the span of 200 years after discharge, the fission products contribute very little compared to the transuranic isotopes; thus, any reduction of the long term integral decay heat depends on the removal of the highest contributing transuranic isotopes.

Longer fuel burn ups result in less plutonium than shorter burn ups due to the plutonium being used in power production as the concentration of fissile uranium and plutonium is depleted. Longer burn ups also generate more higher actinides that produce additional decay heat as plutonium is transmuted to americium, curium, and other actinides. In Figure 10, the total integral decay heat is shown as a function of burn up, and presented by delay time between discharge and repository deposition. For the longer delay times and higher burn ups, the curve has a greater negative slope than for long burn ups and short delay times. As shown in Figure 9, this is due primarily to the contribution from fission products, where the remainder comprises long lived fission products and transuranic isotopes. It is important to note that for a 200 year delay time with fuel burned to 60 GWd/t, the IDH is nearly half of what the 10 year delay time would add to the repository. A 200 year delay may seem like an unreasonably long time; however, many existing reactors are reaching their original operation lifetimes of about 40 years, while safely storing spent fuel for that entire duration. This operating history demonstrates that long term storage before deposition is feasible.

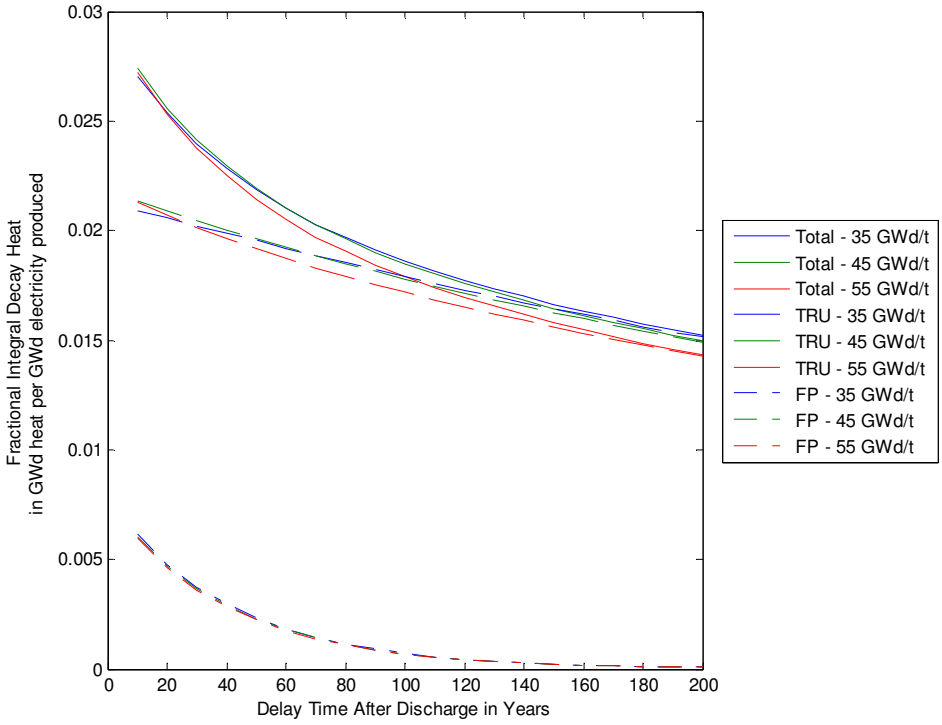

Figure 9: Fractional integral decay heat as function of delay time by burn up

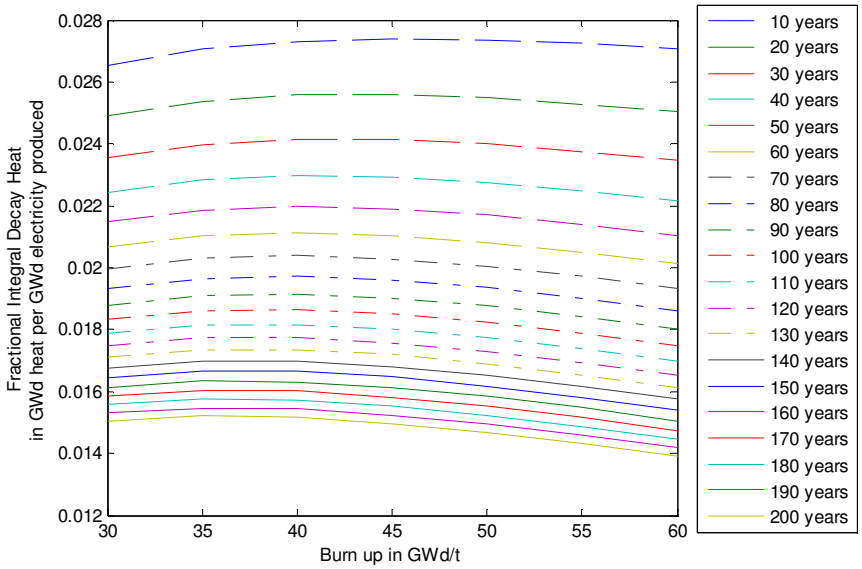

Figure 10: PWR Burn up vs. repository integral decay heat by delay time in once through fuel cycle for total spent fuel mass with fission products included

Main contributors to the integral decay heat are not necessarily the most abundant isotopes in the spent fuel. Results shown in Table 6 and Table 7 provide isotopic mass and fractional integral decay heat data, respectively, for selected actinides, all actinides, all fission products, and the total of all isotopes. Table 8 show the fractional integral decay heat of each component in Table 7 as a percentage of the total integral decay heat at each delay time listed. Am-241 has a fractional mass of about 0.1% to the total spent fuel mass, but comprises nearly 54% of the integral decay heat over the period of 30 to 1500 years after discharge. Fission products have a fractional mass of nearly 4.7% after 30 years of decay, but comprise around 15% of the integral decay heat over the period of 30 to 1500 years after discharge. This is because of the half lives and decay energies of the initial isotopes, and the amount and rate of energy released over the entire decay chain after the initial decay. Data listed in Table 6 are restricted to isotopes that are both significant in terms of inventory and decay heat.

Isotopes of interest for decay heat analysis are broken into three categories, short, medium, and long lived isotopes. Short lived isotopes include those with half lives around 30 years or less, which include many of the fission products, such as Cs-137, and the high-Z transuranics, including Cm-244 and Pu-241. Medium lived isotopes have half lives ranging between 30 and 1000 years, which include Pu-238 and Am-241. Long live isotopes include those with half lives greater than 1000 years, including Pu-239 and Np-237.

Table 6: Once Through Isotopic Composition of Spent Fuel for a Single PWR of 45 GWd/t Burnup, 4.5% Initial Enrichment, for Delay Times of 30, 50, 100, and 200 Years

Isotope	Mass (MT/yr)			
	30 Year Delay	50 Year Delay	100 Year Delay	200 Year Delay
Am-241	3.28E-02	3.79E-02	3.82E-02	3.29E-02
Cm-244	4.72E-04	2.20E-04	3.24E-05	7.03E-07
Np-237	1.85E-02	1.97E-02	2.27E-02	2.83E-02
Pu-238	5.59E-03	4.78E-03	3.22E-03	1.47E-03
Pu-239	1.62E-01	1.62E-01	1.62E-01	1.61E-01
Pu-240	6.81E-02	6.82E-02	6.80E-02	6.73E-02
Pu-241	9.98E-03	3.80E-03	3.39E-04	2.82E-06
U-235	3.24E-08	2.66E-08	1.62E-08	6.00E-09
U-238	2.59E-07	3.87E-07	7.42E-07	1.60E-06
Selected Actinides	2.98E-01	2.97E-01	2.94E-01	2.91E-01
All Actinides	2.48E+01	2.48E+01	2.48E+01	2.48E+01
All Fission Products	1.22E+00	1.22E+00	1.22E+00	1.22E+00
Total	2.60E+01	2.60E+01	2.60E+01	2.60E+01

Table 7: Once Through Fractional Isotopic Integral Decay Heat of Spent Fuel for a Single PWR of 45 GWd/t Burnup, 4.5% Initial Enrichment, for Delay Times of 30, 50, 100, and 200 Years

Isotope	Fractional Integral Decay Heat (GWd Heat Deposited/GWd Electricity Produced)			
	30 Year Delay	50 Year Delay	100 Year Delay	200 Year Delay
Am-241	1.29E-02	1.26E-02	1.15E-02	9.62E-03
Cm-244	1.76E-04	8.18E-05	1.21E-05	2.73E-07
Np-237	6.12E-06	6.08E-06	5.98E-06	5.74E-06
Pu-238	1.91E-03	1.63E-03	1.10E-03	5.03E-04
Pu-239	2.11E-03	2.08E-03	2.00E-03	1.86E-03
Pu-240	3.08E-03	3.04E-03	2.92E-03	2.70E-03
Pu-241	3.49E-06	1.33E-06	1.21E-07	3.14E-09
U-235	1.10E-08	9.04E-09	5.50E-09	2.04E-09
U-238	1.79E-08	1.79E-08	1.79E-08	1.77E-08
Selected Actinides	2.02E-02	1.94E-02	1.76E-02	1.47E-02
All Actinides	2.05E-02	1.96E-02	1.78E-02	1.49E-02
All Fission Products	3.68E-03	2.28E-03	6.95E-04	6.89E-05
Total	2.41E-02	2.19E-02	1.85E-02	1.50E-02

Table 8: Percentage of Total Fractional Isotopic Integral Decay Heat of Spent Fuel for a Single PWR of 45 GWd/t Burnup, 4.5% Initial Enrichment, for Delay Times of 30, 50, 100, and 200 Years

Isotope	Percentage of Total Fractional Integral Decay Heat (GWd Heat Deposited/GWd Electricity Produced)			
	30 Year Delay	50 Year Delay	100 Year Delay	200 Year Delay
Am-241	53.59548%	57.33246%	62.33253%	64.30860%
Cm-244	0.72900%	0.37344%	0.06524%	0.00183%
Np-237	0.02534%	0.02776%	0.03235%	0.03838%
Pu-238	7.92069%	7.45642%	5.96070%	3.35968%
Pu-239	8.72620%	9.48173%	10.83756%	12.42254%
Pu-240	12.76141%	13.85598%	15.80585%	18.04385%
Pu-241	0.01447%	0.00607%	0.00065%	0.00002%
U-235	0.00005%	0.00004%	0.00003%	0.00001%
U-238	0.00007%	0.00008%	0.00010%	0.00012%
Selected Actinides	83.77270%	88.53398%	95.03500%	98.17503%
All Actinides	84.76044%	89.59987%	96.23999%	99.53946%
All Fission Products	15.23956%	10.40013%	3.76001%	0.46054%
Total	100.00000%	100.00000%	100.00000%	100.00000%

Repository heat load capacity is increased because of the rapid decay of short lived fission products and actinides over the delay time. Data in Table 7 show that the decay heat from fission products decrease by nearly 97 percent over a time period of 170 years. This decrease is illustrated in Figure 11, where the fractional decay heat is shown as a function of delay time after discharge. Long lived fission products, including Tc-99 and I-129, have a flat relation as the half life is far greater than the 200 year window shown. Cs-137 has a 30 year half life and undergoes multiple half lives as is indicated by the negative slope in the Figure. After a 100 year decay time, the heat load contribution from fission products is a factor of about 8 less active than when initially discharged, while the contribution from actinides is about a factor of 1.2 less.

Medium- and long-lived actinides contribute more heavily to the integral heat load than the short lived isotopes. At 30 years after discharge, the contribution from actinides is about 85 percent, while after 200 years this figure increases to about 99 percent. A comparison of the fractional decay heat of actinides of interest as a function of delay time is shown in Figure 12. Long lived products have a flat appearance, medium lived have a negative slope, and short lived have a sharply negative slope. With a 432 year half life, Am-241 is a medium lived isotope, but has a long lived daughter product, Np-237. Contributions from Am-241 are about 53% of the total fraction decay heat after a 30 year cooling period, while this number increases to nearly 64% after 200 years. It is important to note that these percentages are tabulated using a fixed upper limit on the integration time of 1500 years; thus, if an assessment were to be performed for 10,000 years, this percentage would decrease.

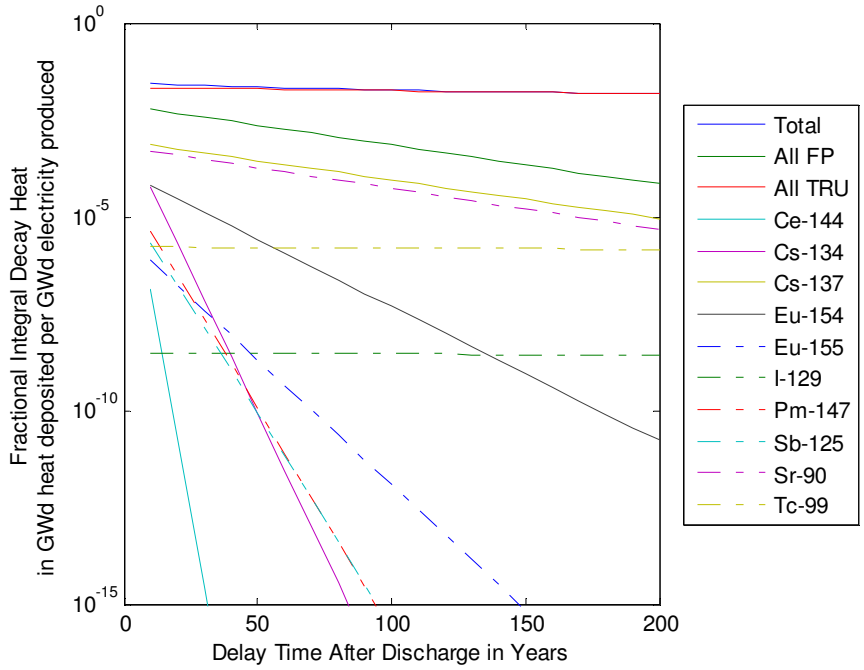

Figure 11: Isotopic components of actinide contribution to repository integral decay heat as function of delay time in once through cycle

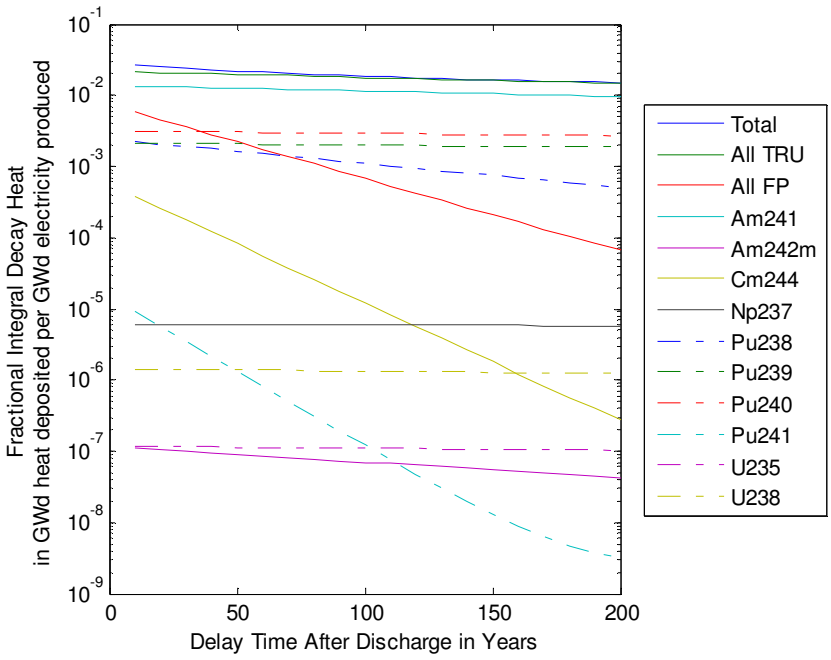

Figure 12: Isotopic components of actinide contribution to repository integral decay heat as function of delay time in once through cycle

Decreasing the heat load may also be achieved through reprocessing of spent fuel. Although this is not currently used in the once through fuel cycle, it is introduced here to show the potential gains to the repository heat load capacity. Selective removal of components of the spent fuel is possible through chemical separation; thus, possible options for selective removal of isotopes are shown in Table 9 with the assumption of perfect separations. Percent reductions are all based on a 30 year cooling period. The first row gives the fractional integral decay heat for delay times of 30, 50, 100, and 200 years, and the percent reduction as a result of storing fuel longer without any reprocessing. A 50 year delay gives a savings of about 9% over the 30 year delay. Results for spent fuel component removal are shown in the remaining rows.

Five separation options are evaluated to reduce the fractional decay heat by the theoretical value of nearly 99.9%. Fission products are separated first as in the UREX separations process, reducing the heat load by a percentage that is heavily dependent on the delay time. Storing the fuel for 200 years yields nearly the same percent reduction as reprocessing and removing the fission products. Americium is removed next as this is the largest contributor over the 1500 year integral. The combination of removing the fission products and americium reduces the initial heat load by nearly 70% with a cooling time of 30 years. Removal of plutonium reduces the heat load by a theoretical value of about 99%. Curium and neptunium separation removes only a slight amount compared to the initial. From Table 9 it is shown that a combination of longer cooling times and reprocessing can make a significant difference in repository heat load capacity.

Table 9: Single Isotope Removal Scenario in the Once Through Cycle for Minimization of Fractional Integral Decay Heat as a Function of Delay Time

Isotopes Removed	Fractional Integral Decay Heat (GWd Heat Deposited/GWd Electricity Produced)				Percent Reduction of Fractional Integral Decay Heat			
	30 Year Delay	50 Year Delay	100 Year Delay	200 Year Delay	30 Year Delay	50 Year Delay	100 Year Delay	200 Year Delay
None	0.0241	0.0219	0.0185	0.0150	0.00%	9.24%	23.38%	38.01%
Fission Products	0.0205	0.0196	0.0178	0.0149	15.24%	18.68%	26.26%	38.29%
Fission Products Americium	0.0073	0.0069	0.0061	0.0051	69.57%	71.44%	74.72%	78.81%
Fission Products Americium Plutonium	2.3E-4	1.3E-4	5.8E-5	4.1E-5	99.06%	99.45%	99.76%	99.83%
Fission Products Americium Plutonium Curium	5.0E-5	4.7E-5	4.4E-5	3.9E-5	99.79%	99.80%	99.82%	99.84%
Fission Products Americium Plutonium Curium Neptunium	4.3E-5	4.1E-5	3.8E-5	3.3E-5	99.82%	99.83%	99.84%	99.86%

Economic Costs

The overall cost of operating the once through fuel cycle is evaluated in terms of the LUEC. From Equations (17) and (18), LUEC is given as a function of fuel cost, capital investment, and continuing operations. With the costs defined in Table 5, LUEC is evaluated from the standpoint of the utility. This neglects interim storage, transportation, and repository costs as those are either reimbursed by the government or paid in the nuclear waste fee.

LUEC varies significantly due to refueling costs, while the capital and continuing operations costs are more stable. Fluctuations in uranium prices, and by extension refueling costs, have caused the LUEC to vary, as shown in Table 2. In 2008, the NEI assessed the probability mass function for LUEC of the fuel cycle modeled is shown in Figure 13 [74]. The distribution has a mean value of nearly \$19.0mills/kWh, standard deviation of \$1.7mills/kWh, minimum value of \$13.2mills/kWh, and maximum value of \$26.4mills/kWh.

Similar analyses for the once through fuel cycle are shown in Table 10. Comparison of the results of this analysis with those found in literature shows the author's result to be in between the other estimates. Differences between the EMWG and NEI estimates are reflections of differences in the year of calculation as fuel costs have changed as have the costs for interim storage [65,74].

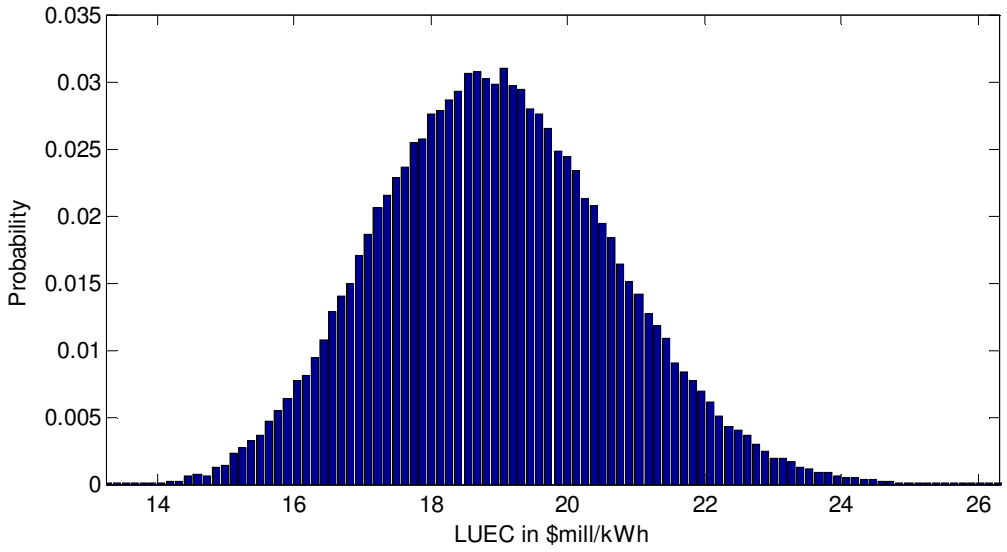

Figure 13: Once Through Fuel Cycle LUEC

Table 10: Comparison of Once Through Fuel Cycle LUEC to Existing Estimates

Source	LUEC ($mills/kWh)
Preston - Utility Perspective	19.0
Preston - Total Cost (Open Cycle)	19.1
Preston - Total Cost (Closed Cycle)	20.8
EMWG (2004) [65]	17.5
NEI (Utility Averaged 2008) [74]	18.7

A similar probability mass function is shown in Figure 14, where the previous distribution is separated into burnup ranges. Here the LUEC estimates of the selected burnup ranges are $18.8mills/kWh for burnups between 30 GWd/t and 40 GWd/t, $15.36mills/kWh for burnups between 40 GWd/t and 50 GWd/t, and $15.02mills/kWh for burnups between 50 GWd/t and 60 GWd/t. The trend towards lower refueling costs by increasing burnup is evident in the lower LUEC. When evaluating the LUEC as a function of delay time for seven distinct burn ups, shown in Figure 15, the same trend of lower LUEC for higher burnup fuel is present. The stratifications of the burn up lines show the LUEC is somewhat independent of the delay time after discharge.

Longer delay times increase the cost of interim storage, but this not reflected in the LUEC evaluations of Figure 15 due to the analysis being conducted from a utility's perspective rather than the total cost of electrical production. The recent lawsuits that forced the government to pay utilities for interim storage costs enable the utilities to not take a loss for dry storage of spent fuel as they previously had [11]. Were this evaluation to be performed from the vantage of the government and tax payers, this would not be the case. From the perspective of government, extended interim storage increases LUEC because of recent legal decisions ordering the government pay the costs of interim storage, as shown in Table 11. The LUEC is compared for the once through cycle from the perspective of a utility and the consumer. LUEC shows an increase to nearly $19.1mills/kWh after including interim storage costs for the open cycle. This number increases further to about $21.0mills/kWh for the closed fuel cycle. The difference between the open and closed cycle is nearly twice the NWPA fee of $1mills/kWh. This shows that the fee was likely an overestimate when determined in 1984 dollars; however, after discounting at 5% through 2009, the original fee is now probably too low to cover the cost of the repository in 2010 dollars. The $1 mills/kWh fee results in a repository cost of about $25.7 billion assuming an average burn up of 50GWd/t with a thermal efficiency of 34%.

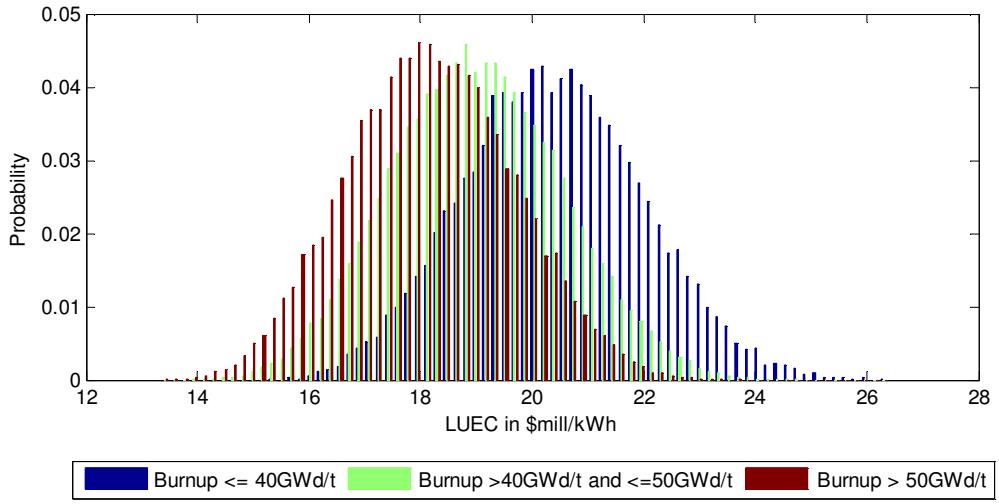

Figure 14: Once Through Fuel Cycle LUEC Sorted by Burnup

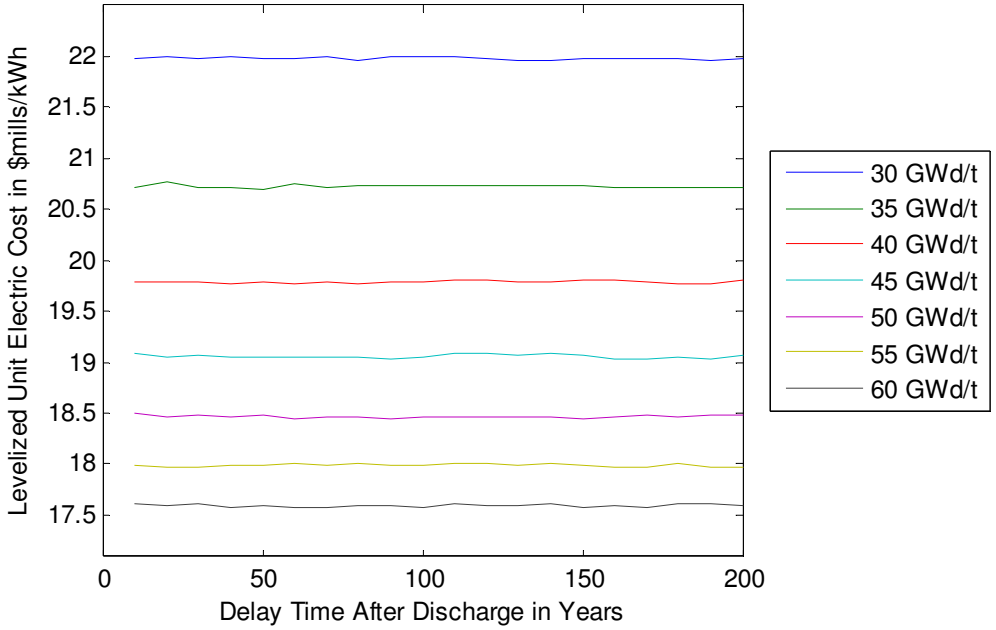

Figure 15: LUEC as function of delay time by burn up for once through cycle from the perspective of the utilities not responsible for interim storage costs

Table 11: Comparison of LUEC and Fuel Cycle Annual Costs for Utility (Neglecting Interim Storage and Repository Costs) and Total (Including All Costs) Methods for

	Open Cycle		Closed Cycle	
	Utility	**Total**	**Total (Mass Basis)**	**Total (IDH basis)**
LUEC ($mills/kWh)	$18.96	$19.08	$21.00	$21.01
Fuel Costs (millions of dollars per reactor per year)	$54.90	$54.84	$54.85	$54.86
Operating Costs (millions of dollars per reactor per year)	$114.49	$114.54	$114.41	$114.51
Backend Costs (millions of dollars per reactor per year)	N/A	$1.11	$18.34	$18.34
Total Costs (millions of dollars per reactor per year)	$169.39	$170.49	$187.61	$187.71

As discussed earlier, converting the repository mass limit into a heat load limit opens the possibility for storing more mass, reducing the cost per unit mass for deposition. In Table 11, the columns for closed cycle are labeled according to the heat load cost basis for a delay time of 30 years. The mass basis shows a similar backend cost to the IDH basis, because the burnup is set to 50 GWd/t in both examples, showing the equivalence of the two models. These values should be similar because the repository cost term is taken from the median value of the repository cost in Table 5, corresponding to a total repository cost of about $75 billion. The intersection of the mass basis and heat load basis at this total repository cost is illustrated in Figure 16.

The comparison of the mass and decay heat capacity model shows a decreasing repository disposal cost with the delay time increases and a significant negative slope for the mass based model with increasing delay time and burn up. A decrease in the disposal cost is one result of the longer delay times that allow for short lived fission products and actinides to decay. The decrease with respect to burn up is a result of the reactors requiring fewer refueling over the same time period as a comparable reactor with a lower burnup. This causes the mass through put to decrease proportionally to the burnup, resulting in the negative slope. The degree of the slope is dependent on the combination of delay time and burn up. Longer burn ups do require higher costs from interim storage, but this cost is significantly less than the cost to deposit the spent fuel into the repository, as shown in

Table 1. Thus, the dominant economic term in the back end of the fuel cycle is the cost per unit mass to deposit in the repository.

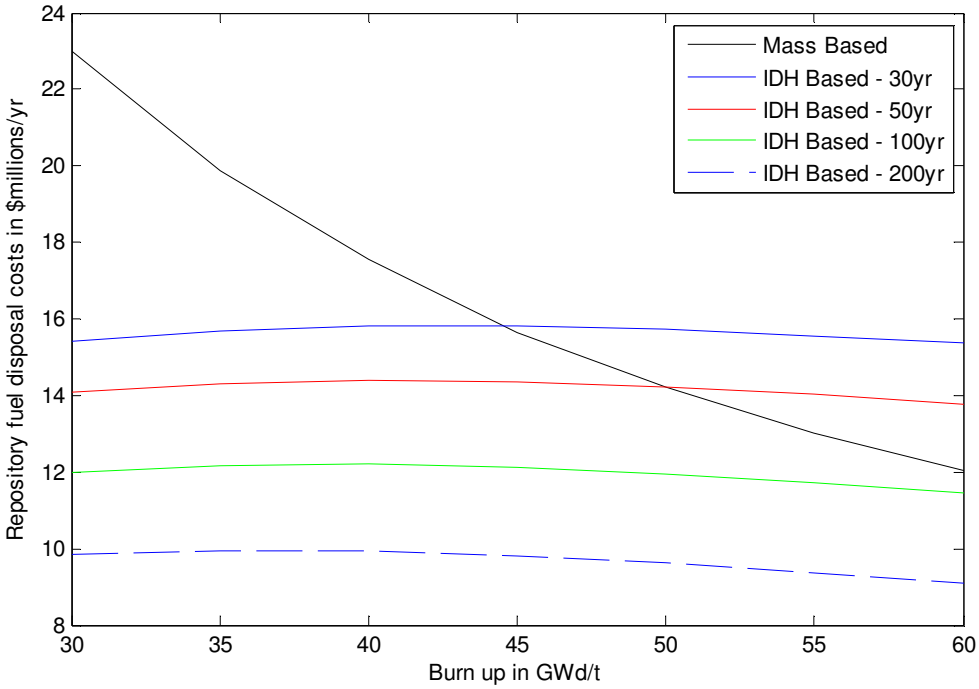

Figure 16: Comparison of Repository Fuel Disposal Costs for a Single PWR for Mass and Heat Load Bases for Delay Times of 30, 50, 100, and 200 Years

From evaluation of the once through cycle many properties of the currently used cycle have been presented in terms of fuel burn up, delay time, repository heat capacity, and economic costs. Optimization has been presented from the point of view of the utility and of the government that includes repository and interim storage costs. With the current cycle presented as a baseline for comparison use, results from the hybrid LWR-FR advanced fuel cycle are now introduced.

Optimization Example

LUEC minimization using the government point of view model results is an example of how strictly dominated solutions appear within the fuel cycle when there is a lack of outside influences that bias the results. Strictly dominated refers to the game theory concept of an obvious strategy or trivial solution. In the case of LUEC minimization, the major cost added is from the repository, where a smaller cost is added from interim storage. The obvious solution is to delay the fuel in interim storage as long as possible because the repository deposition costs are much higher than for extending interim storage, as shown in Table 12.

Table 12: LUEC Optimization Example Using Once Through Fuel Cycle

Parameter	Optimized Values	Closest Boundary
Fuel Burn Up (GWd/t)	59.987	60
Enrichment (wt-% U-235)	4.998	5.0
Delay Time (years)	199.998	200
LUEC $mills/kWh	19.21	N/A

Without taking into account external factors, namely politics or security concerns, this optimization is correct. When accounting for these concerns, the result may be quite different. Manipulation of the boundary conditions cannot perform the task of these externalities; thus, the code may be included in models suited for this task.

Hybrid Fuel Cycle

Example Description

Analysis of the hybrid fuel cycle begins by defining parameter distributions and loading into the module for evaluation. Several parameters use nominal values while others use triangular or uniform distributions where distributions are defined. Both the interpolated data and forced isotope reduction methods are investigated using similar distributions where possible. In each isotopic inventory model, the reactor park configuration consists of a fixed size fleet of 69 PWR's with as many fast reactors that can be fueled by the LWR fleet. The number of fast reactors depends on the number required to consume the plutonium produced in the LWRs, neglecting fractional reactors. Results for each case show differences in the calculation methodology and in the specific difficulties in tabulating fast reactor data using existing reactor physics codes.

Isotopic data libraries for mass, decay heat, and integral decay heat are available for several reactor types, including a LWR type, PWR of BWR, and a fuel selection for the fast reactor, metal or oxide. Libraries for the metal-fueled fast burner reactor have a target conversion ratio of 0.75. Conversion ratio is defined here as the ratio of all fissile isotopes present at the end of fuel life to all fissile isotopes present at beginning of fuel life. Fast reactor specific distributions based on the usage of the 0.75 conversion ratio library is shown in Table 13, while parameters relevant to LWR's are similar to the once through cycle, shown in Table 5. A delay time exists after fuel is discharged from a reactor, with the first delay being after the LWR stage, and the second stage being after the fast reactors and before the repository. The delay after the LWR stage is limited to 50 years to allow for cooling of spent LWR fuel before reprocessing.

Fast reactor costs are more uncertain than LWR's. No FR's have been built in the US for over 40 years and the supporting facility cost estimates are scaled from similar, but smaller existing facilities. Reprocessing facility cost is estimated at $543 million, in 2002 dollars, for a plant with a capacity between 2,000 and 3,000 tons per year, which is extrapolated from smaller designs [65].

Table 13: Hybrid Fuel Cycle Parameter Distributions Used in Analyses

Parameter	Class	Distribution	Low Value	Peak	High Value
Number LWR's [37]	Fuel Cycle Specific	None		69 reactors	
Number FR	Fuel Cycle Specific	None		Maximum Supported by LWR Fleet	
Delay Time 1 (after LWR)	LWR Operation	Triangular	15 years	30 years	50 years
Reactor Lifetime	LWR and FR Operation	None		60 years	
Delay Time 2 (after FR)	FR Operation	Triangular	5 years	20 years	200 years
FR Burn Up	FR Operation	Triangular	90 GWd/t	150 GWd/t	180 GWd/t
FR Plant Cost [70]	Economic	Triangular	$2.5E9 per reactor	$5E9 per reactor	$6E9 per reactor
Reprocessing Facility Cost [70]	Economic	Triangular	$15E9	$20E9	$25E9
Aqueous Reprocessing Costs [70]	Economic	Triangular	$460/kg	$502/kg	$829/kg
FR Fuel Fabrication [70]	Economic	Triangular	$2200/kg	$5000/kg	$6000/kg
FR Operating Costs [70]	Economic	Triangular	$50E6 per year per reactor	$75E6 per year per reactor	$100E6 per year per reactor
Repository Cost [70]	Economic	Triangular	$381/kg	$528/kg	$900/kg

Isotopic Inventory and Repository Heat Load

Simulations of the fuel cycle using the aforementioned distributions provide estimates on the annualized production of fission products and transuranic isotopes that contribute to the repository heat load. In Figure 17, the total fast reactor contribution to integral decay heat is shown for various burn ups as a function of delay time before deposition in the repository. As in the once through cycle, fission products provide the greatest fraction within the first 100 years as short lived products to decay quickly into stable isotopes.

A similar diagram for the entire fuel cycle including the LWR component is shown in Figure 18. Compared to Figure 9, which shows a similar view for the once through cycle, the fission products are a smaller component of the total decay heat with fast reactors. The transuranic contributions still maintain a relatively flat term that is stratified by burn up because there is far less production of higher actinides than in a thermal reactor. This is also related to conversion ratio in that a higher charge mass of U-238 acts as a production pathway to higher actinides, where usage of an inert matrix fuel would severely inhibit that pathway and allow much lower conversion ratio designs.

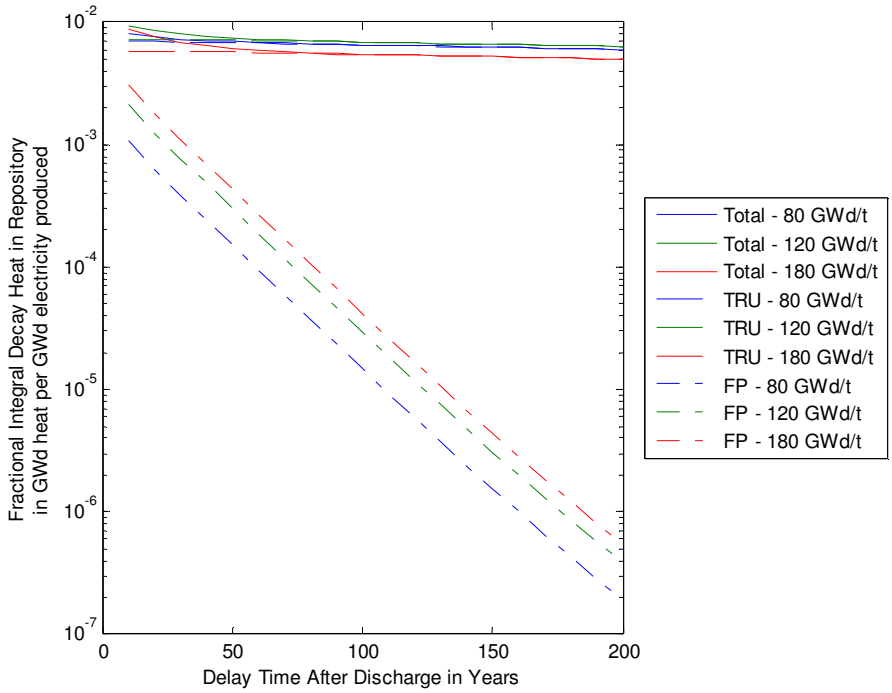

Figure 17: Components of FR contribution to repository integral decay heat as function of delay time by burn up

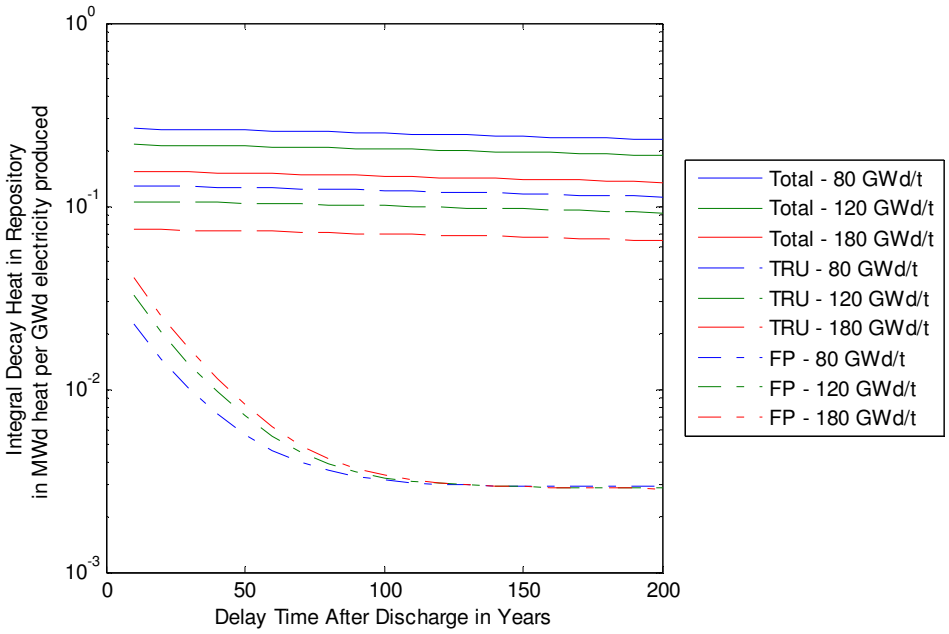

Figure 18: Entire hybrid fleet components of repository integral decay heat as function of delay time by burn up

LMFBR's can use higher burn up fuels than LWR's because of the longer lifetime of the fuel cladding material. Longer fuel lifetime enables increased electrical production and significant reductions in plutonium and minor actinide inventory. For the case of 69 LWRs and 13 fast reactors, the hybrid cycle produces 12% more total electrical generation than the once through with 69 LWR's, where the reactor power for the FR's are 1000MWth with a thermal efficiency of about 60% and the LWR's have ~3000MWth with a thermal efficiency of 34%. As fuel burn up increases, the transuranics are effectively removed from the long term decay heat and replaced with fission products that are likely to be short lived, where the relation is shown in Figure 19 by delay time. There is an asymptotic behavior for the fission product IDH due to the reduction in inventory of short lived fission products for longer interim storage durations. The long term decay of actinides is the lower limit of the integral decay heat.

Data shown in Figure 19 illustrates the benefit of LMFBR usage for repository capacity maximization. For the 10 year delay time, the fractional IDH is decreased by a factor of about 1.25 when fast reactor fuel burn up is increased to 180 GWd/t. Employing fast reactors can gain capacity by a factor of nearly 1.5 when compared to the data in Figure 10, which shows a very different trend over reactor burn up with respect to IDH.

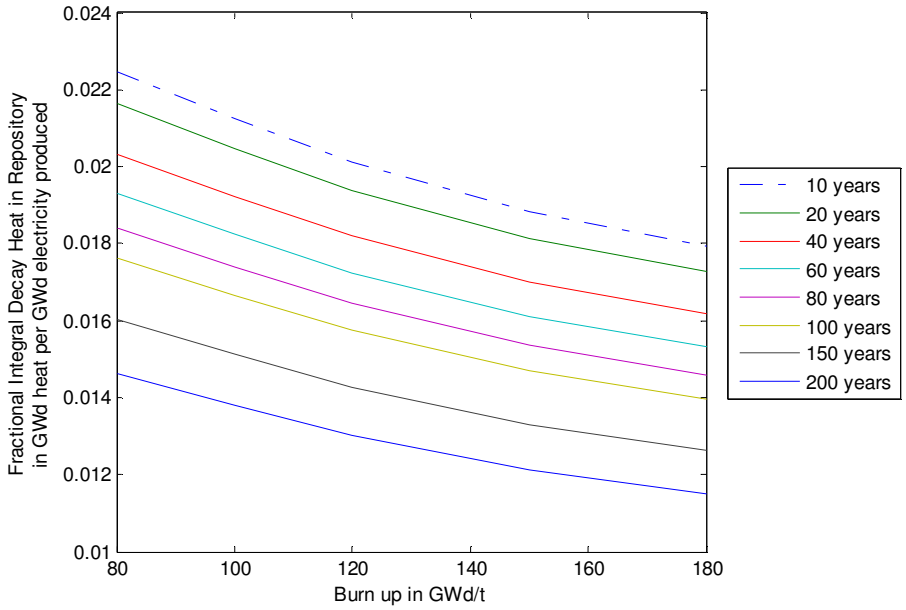

Figure 19: FR Burn up vs. repository integral decay heat by delay time

Simulations of the hybrid cycle provide estimates of fission products and transuranic isotopes that will contribute to repository heat load. In Table 14, the mean values for charge isotopes are shown for the fast reactor fleet, where the units are annualized in terms of mass composition of spent fuel and integral decay heat. In this fast reactor design, the mass input composition of fuel is 25% Pu-239, 8% Pu-240, and 67% U-238. The fraction to total mass is about 9% Pu-239, 7.5% Pu-240, and 79% U-238, where other actinides make up the remaining mass: a significant reduction in plutonium inventory using fast reactors. The actinide decay heat term is dominated by the remaining plutonium, while the fission products account for nearly 20% of the IDH.

Isotopic masses for the hybrid cycle to be deposited in the repository are somewhat different than for the once through example. Spent fuel consists primarily of U-238, which is the primary component of both LWR fuel and FR fuel; however, the fast reactor fuel may not use the U-238 from the reprocessing stream, but instead use depleted uranium from the enrichment tails. Usage of U-238 from enrichment tails in a net increase in waste mass when compared to the once through cycle, but adds little to the decay heat [43]. This effectively shifts the amount of depleted uranium from low level storage into the repository mass required, which is the more expensive disposal method. While this is discussed in literature, it is evident that it would be more beneficial to the repository if the reprocessed uranium were used instead; however, this would increase the cost of fuel fabrication from the contamination of the reclaimed material [42,43]. Isotopes in LWR spent fuel not present in the charge mass of the fast reactor fuel are deposited along with LWR high level waste from the reprocessing stream. These actinides are included in the “All Actinides” term in Table 14. The isotopes are chosen for compatibility with results from the forced removal method as will be discussed later along with the use of isotope targets.

Table 14: Hybrid Cycle Isotopic Composition and Integral Decay Heat of Spent Fuel for FR Fleet of 9 Reactors with 120 GWd/t Burnup and 30 Year Delay Time

Isotope	Mass (MT/yr)	Fraction of Total Mass	Integral Decay Heat (GWd/yr)	Fraction of Total Integral Decay Heat
Pu-239	1.96	0.089889	122740.457	0.243466
Pu-240	1.63	0.074988	299560.638	0.594205
U-238	17.17	0.788114	42.402	0.000084
Selected Actinides	20.77	0.952990	422343.497	0.837755
All Actinides	21.40	0.982078	422343.497	0.837755
All Fission Products	0.39	0.017922	81793.557	0.162245
Total	21.79	1.000000	504137.054	1.000000

Reduction of plutonium inventory is of particular interest in the hybrid cycle when using burner reactors. The PWR component (69 reactors) of the once through cycle creates 13.0 MT/yr of Pu-239 and Pu-240 that could be available for fast reactors or sent to the repository. Reducing this inventory serves to increase the repository mass capacity by decreasing the heat load added. Over 95 percent of spent fuel mass from LWR's is U-238; thus, removing the plutonium and minor actinides has little effect on the mass. In the hybrid cycle, the amount of plutonium sent to the repository is about 6.1 MT/yr for 9 fast reactors present in the fleet with a target burn up of 120 GWd/t, a target conversion ratio of 0.75, and without reprocessing fast reactor spent fuel. This represents a two-fold decrease in plutonium mass sent to the repository; however, a back of the envelope calculation does not agree with this result because this would correspond to a conversion ratio of ~0.5. The cause of this is the lack of fast reactor cross section data available in ORIGEN-S that is reducing the plutonium inventory more than it should for the geometry; thus, the forced reduction model is used later to assume a conversion ratio and estimate the corresponding decay heat.

Figure 20 shows the factor decrease in the amount of plutonium mass deposited in the repository for fast reactors of increasing fuel burn up for the case of 69 PWRs in a hybrid cycle. Despite having a target conversion ratio of 0.75, this would be the conversion ratio if defined as the ratio of fissile material output to fissile material at time of input. Reactor physics permitting, a conversion ratio of 0.3 would be obtained by burning the fuel to 180 GWd/t [42]. In reality this is unrealistic given the power density of the reactor as this corresponds to an in-core fuel lifetime of nearly 13 years per batch, while a more realistic 80 GWd/t corresponds to a lifetime of nearly 5.75 years [43].

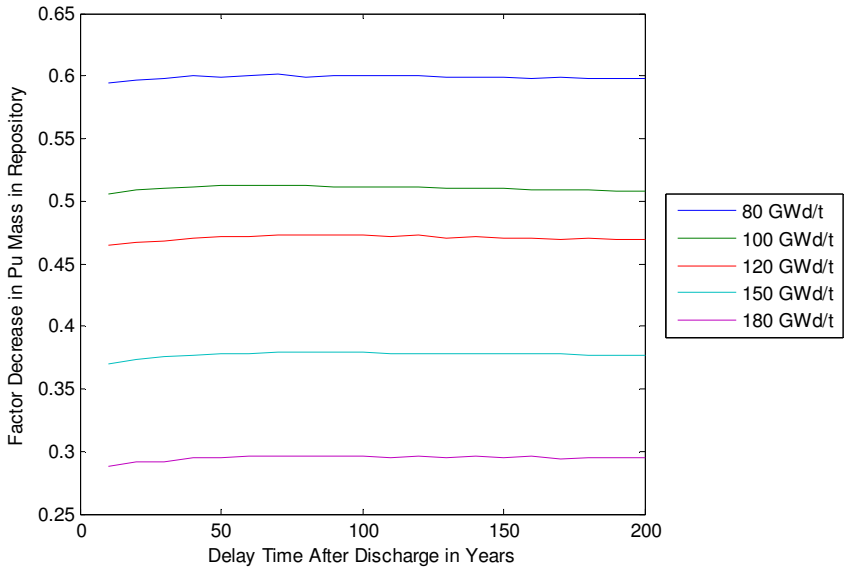

Figure 20: Factor decrease of Pu mass using one-tier fuel cycle by burn up for 9 FR's

There is a factor gain for repository heat load capacity when using the hybrid cycle. Figure 21 shows this gain as a function of delay time, where this is the ratio of the total integral decay heats for the once through cycle relative to the hybrid cycle. The values assume a 30 year delay time for both LWR and fast reactor spent fuel. A trend shows a decrease in the factor for increased burn up. This follows the logic that as more plutonium is being burned, the integral decay heat of the spent fuel inventory will decrease, causing an increase in the available thermal capacity of the repository. It shows that the 63,000 MT repository at Yucca mountain would be able to hold nearly 82,000 MT of fuel for a 80 GWd/t fast reactor fuel burn up and up to nearly 103,000 MT for a 180 GWd/t burn up, where mass is defined as having the equivalent decay heat regardless of burn up. This yields a percentage capacity gain of between 30% and 63%. Some estimate that a hybrid cycle would have a single or double digit factor increase in capacity, but results are likely more modest when secondary waste streams are considered.

The charge masses for a fast burner reactor design contain some questionable aspects as to the feasibility of the design with regard to minor actinide components [43]. Using the referenced fuel specification, masses of Cm-244 and Am-241 would be limiting factors in the size number of supported fast reactors rather than Pu-239 when adhering strictly to the charge mass definitions. Plutonium should be the limiting factor as this is the major fissile component of the fast reactor fuel, which is the approach of the previous section [43]. Figure 22 shows the average number of fast reactors as a function of burn up for a single reprocessing step (only LWR fuel is reprocessed). Because the data are for a single conversion ratio reactor, increasing the burn up decreases the amount of fuel needed when operating with the same power density and core size; thus, more fast reactors can operate as there is a greater amount of available fuel.

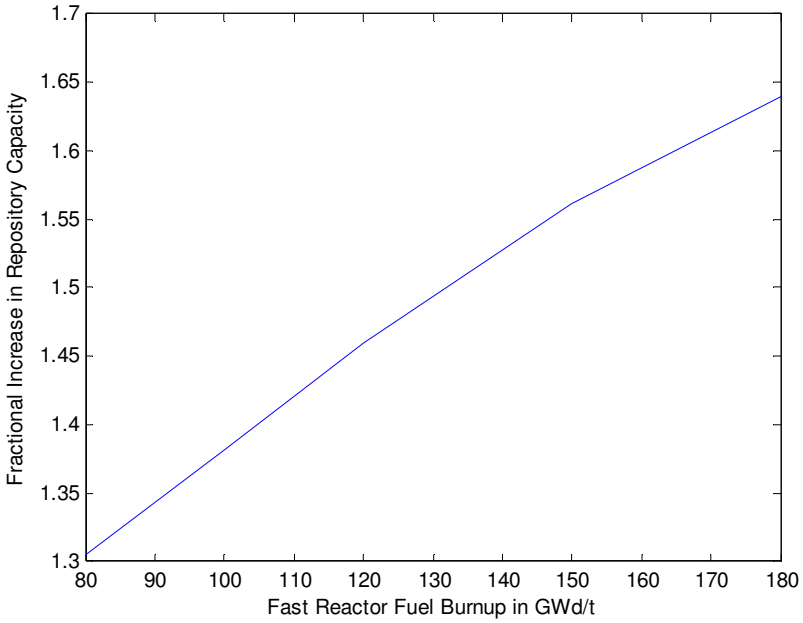

Figure 21: Fractional increase in repository mass capacity from using hybrid cycle with 9 fast reactors operating at 400MW

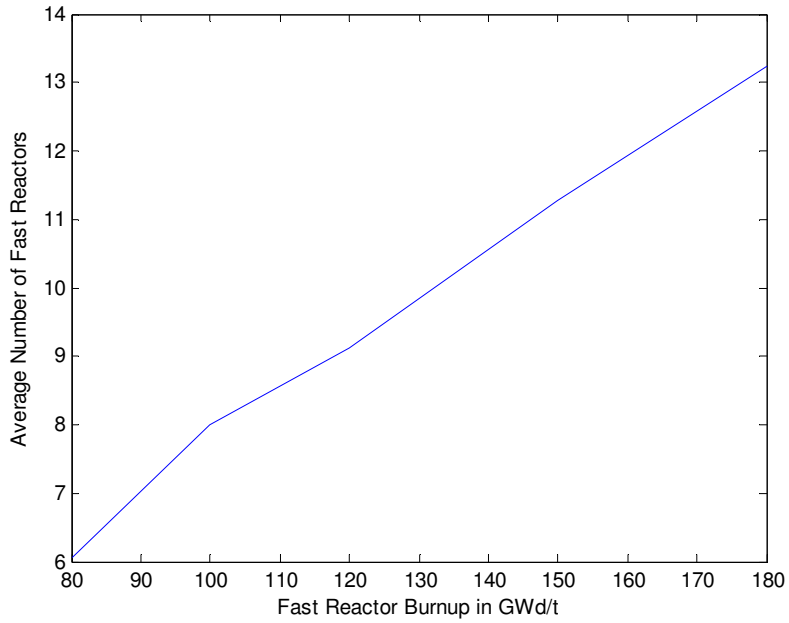

Figure 22: Average number of fast reactors for each burn up

The ratio of LWR's to FR's ranges between about 5 and 12 as a function of fast reactor burn up, while literature suggests this ratios should be around 2-4; however, this depends on the assumptions of whether fast reactor spent fuel is also reprocessed [65,70]. Reprocessing fast reactor spent fuel would result in a lower ratio because the recoverable plutonium would be a function of the conversion ratio of the fast reactors. Reducing the LWR to FR ratio in the equilibrium case would burn more of the plutonium rather than stockpile remaining amounts. This analysis assumes the excess material is sent to the repository rather than stored. In a non-equilibrium model, reducing this ratio indicates higher conversion ratio designs that require many more reactors for the same inventory reduction as obtained with this reactor model.

Using the same reactor input definitions, the forced reduction model is used to evaluate the possibility of multiple reprocessing steps and isotopic reduction without relying on the provided data libraries. This model allows the user to define the percentage to which each input isotope is transmuted or fissioned in the reactor. Forcing the reduction of the charge mass allows the user to specify reactor designs for which there are not data sets available, but at the cost of infeasible designs.

Results using the forced reduction method show a ratio of LWRs to FRs between 6.9 and 7.7, when having the initial mass as defined according to Table 15. Here the charge mass only relies on masses of Pu-239, Pu-240, and U-238, and all of the reprocessed Am-241, Cm-244, and Np-237 are used as targets with a set fraction remaining. The fraction remaining after burn up is equivalent to the conversion ratio of 0.6.

The forced reduction model interpolates integral decay heat based on comparable isotopes from the LWR results that are scaled by burn up. Figure 23 shows the integral decay normalized to energy produced produced for the fleet of all reactors. The values here are less than both the once through cycle and the interpolated hybrid cycle model, by a factor of about 6 and 5, respectively. This shows that scaling from LWR results are likely missing components of the burn up that would be included in data produced from ORIGEN-S, resulting in a lower IDH.

Table 15: Hybrid Cycle Forced Reduction Charge Mass and Remaining Isotopic Fractions for Input Fuel and Targets

Isotope	Initial Charge Fraction	Fraction Remaining After Burnup
Am-241	* All Reprocessed	0.80
Cm-244	* All Reprocessed	0.80
Np-237	* All Reprocessed	0.80
Pu-239	0.25	0.60
Pu-240	0.08	0.98
U-238	0.67	0.99

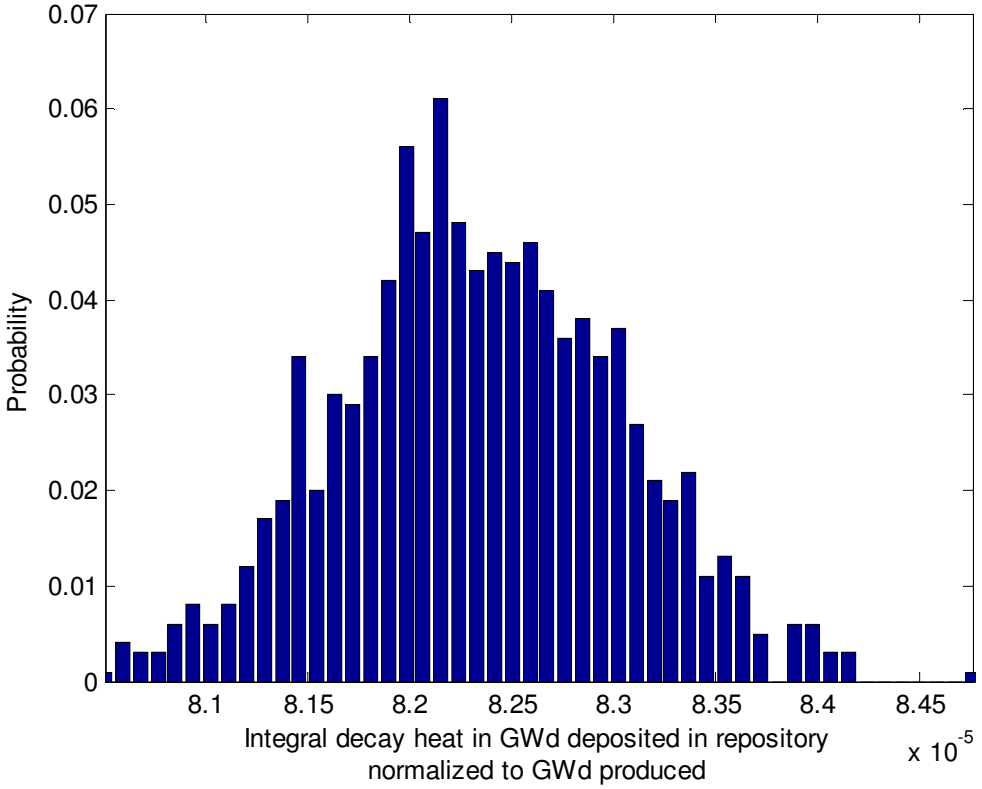

Figure 23: Integral decay heat normalized to electricity produced for forced reduction model of hybrid cycle

If the IDH results shown in Figure 23 are indeed accurate, this results in a nearly 6 fold gain in repository heat load. Since about 7 LWRs support each fast reactor in this model, the reduction in plutonium inventory is more significant than in the interpolated model where about 5 to 12 LWRs supported a single fast reactor. From this standpoint, the mass reduction appears to be more realistic than for the interpolated model. Figure 24 shows the probability mass of the total plutonium inventory sent to the repository after discharge from the fast reactor, normalized to electrical energy produced.

The forced reduction model can be applied to using isotope targets in the fuel, including Am-241. Figure 25 shows the results when using a fractional destruction of 0.8.

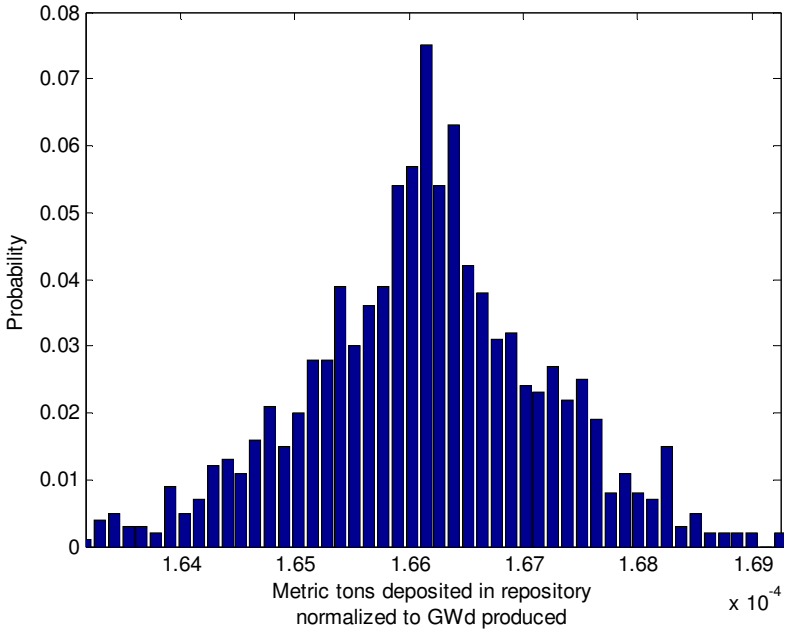

Figure 24: Plutonium inventory in repository calculated using forced reduction model

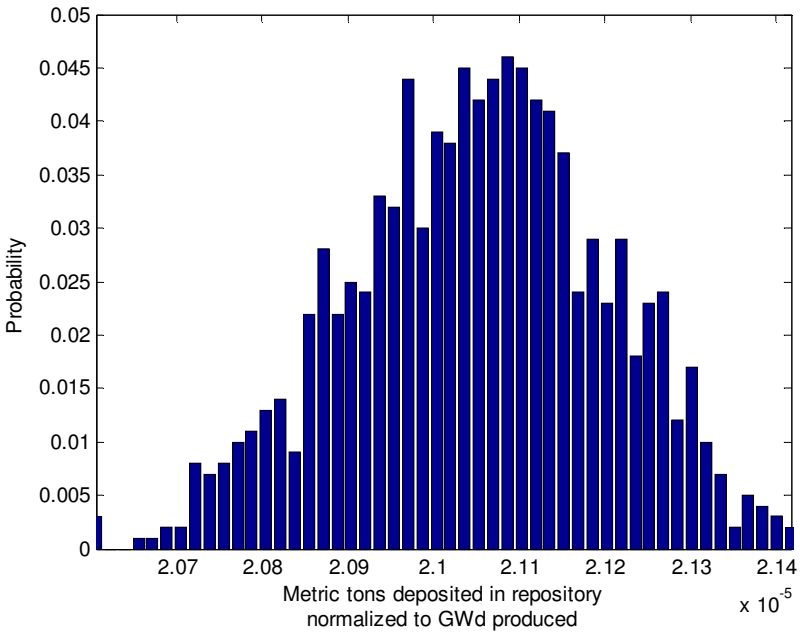

Figure 25: Mass of Am-241 using targets in forced reduction model

Economic Costs

Estimates of the hybrid cycle's LUEC are higher than comparable analyses, as shown in Table 16. LUEC is higher than for a once through cycle because of the added reactors, reprocessing facility, operating costs, and decreased electrical output for those costs. Here the mean value is estimated at nearly 26.5 $mills/kWh, for the interpolated model using a single reprocessing step, while the estimate from literature is significantly lower. Investigating the method used by the comparable analysis finds that reprocessing facility costs are not included in the LUEC calculation directly, but are included into the estimated costs for reprocessing [70]. Not including the facility cost assumes that the reprocessing costs per unit mass include some fraction of the facility cost that is not defined in the reference. Based on an amortization of the facility cost, the fast reactor site is collocated with the reprocessing and fuel fabrication facilities, and that the Department of Energy will operate both the reactor and reprocessing stages, it is the author's intent to keep the facility costs separate from the specific mass-dependent costs. Estimating that a 3000MTHM/yr reprocessing facility would cost nearly $20 billion significantly adds to the capital costs of only 12 fast reactors in the fleet, taken as the median of the distribution. Growth in the fast reactor fleet would further distribute these costs among more facilities, decreasing the LUEC contribution.

LUEC is a function of total cost and electricity produced; thus, adding additional reactors affects the LUEC. In Figure 26, the LUEC follows this trend, but it is important to also note that the number of reactors is dependent on the fast reactor fuel burn up.

A stated goal of optimization of the fuel cycle is maximization of the repository heat load capacity, while maintaining minimal economic costs. In Figure 27, the largest decrease in plutonium inventory in the repository has the highest fast reactor fuel burn up, and costs the most in terms of LUEC. This also agrees with the trend noted in Figure 26.

Table 16: LUEC Comparisons of Hybrid Fuel Cycle for the Interpolated Data Model and the Forced Reduction Model to an Estimate from Literature

Source	LUEC ($mill/kWh)
Interpolated Model	26.47
Forced Reduction Model	26.37
EMWG (LWR-CFR) [65]	8.19

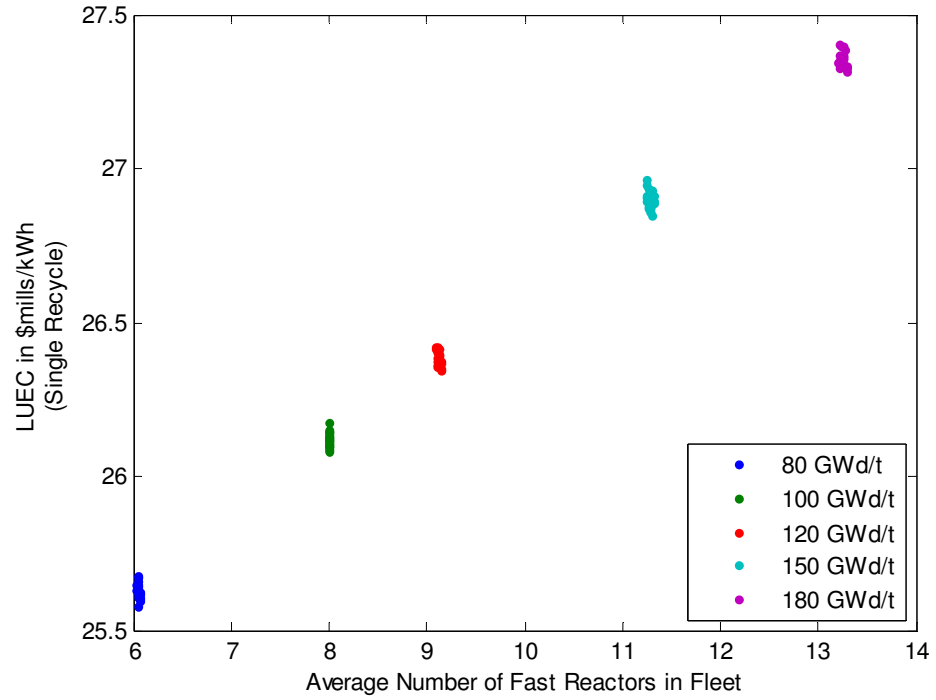

Figure 26: LUEC of hybrid cycle as function of number of fast reactors

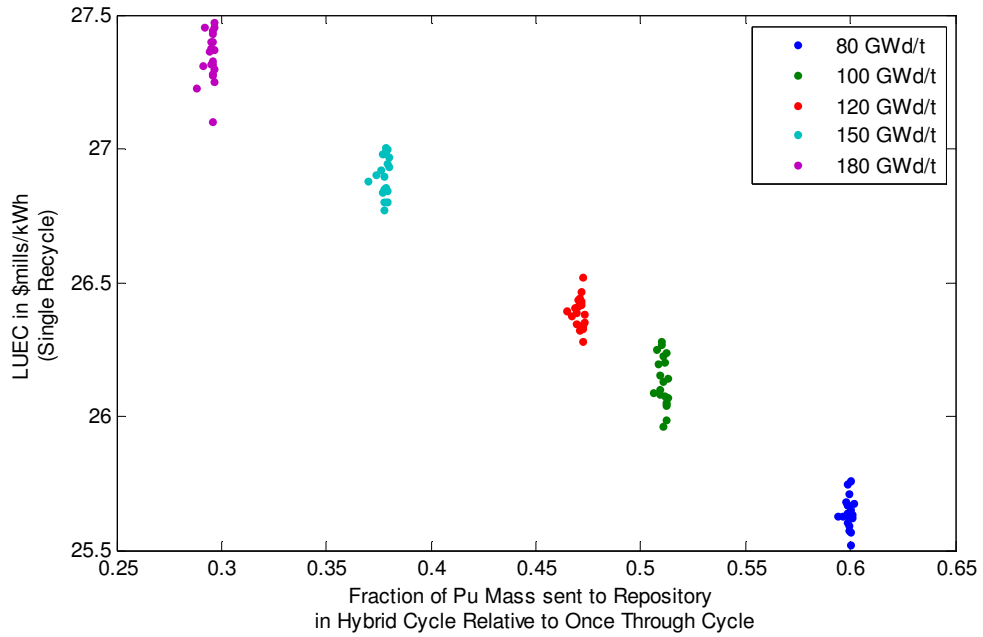

Figure 27: Hybrid cycle LUEC as function of the fractional amount of plutonium inventory sent to repository in the hybrid cycle relative to the once through cycle

From the previous figures shown, results of LUEC optimization tend toward the lowest fast reactor fuel burn up since it costs the least in terms of facilities and reprocessing; however, a single optimization goal is not a suitable solution as there are many factors as to why a hybrid cycle may be used. Minimization of plutonium inventory, maximization repository capacity, and maximization of electrical output are all key goals that must carry some weight in the optimization.

Comparison of Once Through and Hybrid Cycles

A primary focus in this research has been creating a tool that may aid in the analysis of the next generation fuel cycle. Two of the FCRD fuel cycles have been modeled using the developed methodology. Results for actinide inventory in spent fuel, decay heat, and economic costs were obtained for the once through and hybrid cycles. A technical comparison of the two models is given based on repository heat load reduction, isotopic inventory, and economic costs. An interpretation of these comparisons will describe a potential direction for the future of the nuclear fuel cycle using these results.

One goal of using the hybrid cycle rather than the once through cycle is repository capacity maximization through inventory reduction of higher actinides that are major contributors to the long term decay heat. Results from the analysis show that this is an attainable goal when implementing the hybrid fuel cycle design. Assuming that utilities operate the LWR fleet at the maximum possible burn up to maximize revenues, only the duration spent fuel resides in interim storage will reduce the repository heat load, barring reprocessing and transmutation. This delay time only method for heat load reduction is limited by the long lived actinides, i.e. Pu-239, where the possible delay time will be much less than the half lives of the isotopes. As shown previously in Figure 9, once the fission products and short lived actinides decay, the decay heat levels off as long lived fission products and actinides are the remaining factor; therefore only reprocessing and transmutation of the LWR spent fuel can reduce the contribution from long lived actinides.

For the hybrid cycle without reprocessing fast reactor spent fuel, the inventory of the long lived actinides can be effectively reduced by some percentage that is dependent on the fast reactor conversion ratio. A conversion ratio of 0.75 would theoretically reduce the long live actinide inventory about 25%. After discharge, the delay time only method for reducing repository heat load would apply to fast reactor spent fuel if it is not also reprocessed.

Economic cost is a strong factor in choosing portions of an advanced fuel cycle. The industry will not be profitable if the LUEC is too high, which may reduce the usage of nuclear power for base load power generation if other options are cheaper and stable. Table 17 gives estimates of the total costs for the models evaluated, which include repository costs. The once through cycle is less costly to operate than the hybrid cycle; however, lower cost does not necessarily imply an optimal choice.

Table 17: Comparison of LUEC for Once Through and Hybrid Fuel Cycles

Once Through	Hybrid Cycle	
	Interpolated Model	**Forced Reduction Model**
$20.8 mill/kWh	$26.47 mill/kWh	$23.37 mill/kWh

If a decision function were to weight on the issues of cost, repository usage, and longevity of electrical production, then the choice of fuel cycle becomes more complex. Cost favors the once through, but results show that the plutonium inventory is controlled in the hybrid cycle, which is beneficial to repository capacity. Fast reactor usage also adds to the longevity of fuel supply because plutonium from spent LWR fuel may be reused in fast reactors as new fuel, which reduces the number of LWR's necessary in a reactor fleet to reach a target electrical production compared to a LWR-only fleet in the once through cycle.

Choosing a direction for the future of nuclear power is a difficult task that requires methodical planning. Of the two choices evaluated in this research, the decision favors the hybrid cycle for controlling repository usage and offering more efficient uranium usage. The economic cost may be higher, but this cost increase is preferable compared to opening multiple repositories to handle more waste.

Summary

Uncertainty analyses for the once through and hybrid fuel cycles have been performed using the developed method. Results from the fuel cycle models generally agree with the estimates from EMWG and NEI reports. The optimization procedure returns the optimal technical parameters for the fuel cycle models, but neglects political factors that cannot be accurately modeled or forecasted, which heavily influence design choices.

CHAPTER VI
CONCLUSIONS AND RECOMMENDATIONS

The goal of producing an extensible foundation for fuel cycle evaluation and characterization has been demonstrated. While the usage and applications of this code may change, the framework presented for uncertainty analysis and optimization is dynamic with respect to changes in reactor type or fuel cycle design. Current results and future usefulness of the work are largely dependent on the quality of the data sets, distributions selected for parameters, and fuel cycle types to be evaluated. Conclusions for the selected fuel cycle cases and optimizations are discussed with possible areas of interest for future research.

Both the once through and hybrid fuel cycles were evaluated for economic cost, decay heat, and repository utilization. Evaluations assumed usage of a repository for long-term spent fuel storage, contrary to the recent decision by the DOE and Congress. Major conclusions of the research include:

- Short term decay heat can be diminished by increasing a delay time spent fuel resides in interim storage to allow decay of short lived fission products and actinides.
- Long term decay heat within the repository may be effectively reduced by reducing the inventory of plutonium through usage of fast reactors.
- Long term decay heat is dependent on the amount of Am-241 present; thus, burning Am-241 in fast reactors or targets would effectively reduce the repository decay heat.
- Cost of repository deposition is much greater than the cost of long term interim storage; thus, long term interim storage will both reduce costs and decay heat within the repository.
- Costs of interim storage may be reduced by avoiding lawsuits if DOE takes possession of spent fuel once it has sufficiently cooled, and transports it to a centrally located interim storage facility.
- The once through cycle currently has an LUEC of about $19.0 mills/kWh, while the hybrid cycle is expected to have an LUEC of around $26.4 mills/kWh.
- The nearly 50% increase in LUEC from implementing the hybrid cycle adds only 12% more electrical generation compared to the once through cycle. This shows that there is a premium to be paid for using the hybrid cycle to reduce heat load and plutonium inventories associated with the once through cycle.
- Optimization of the fuel cycle must take many parameters into account rather than a single goal to avoid rapid approaches to the boundaries.
- Economic cost and political feasibility control the development of new nuclear power plants.

Contributions to Fuel Cycle Analysis

This work combines various techniques commonly used in and outside of fuel cycle research and development that build upon prior modeling efforts with regard to uncertainty analysis and optimization. The author believes that this approach is a step towards more intricate designs that will bridge the gaps between physics, economics, and resource management. Methods for uncertainty analysis, reactor physics, economic forecasting, and advanced computing are combined into a multidisciplinary modeling code to evaluate two of the possible AFCI scenarios.

The two major codes for fuel cycle analysis, VISION and DANESS, provide detailed models using nominal value analysis with limited uncertainty analysis for a time-dependent fuel cycle [28,29]. This research presents and demonstrates methods to extend these models for optimization under uncertainty, data simulation, and uncertainty analysis. Aspects of this research have been aimed at reducing computational time, while allowing rapid evaluation of changes to a fuel cycle. Preprocessing the reactor physics data provides a means to calculate results for a wide range of systems without large amounts of computing time. Simplifying data storage and interpolation with neural networks and other methods provides a quick data retrieval system for use in both the Monte Carlo uncertainty analysis and Genetic Algorithm optimization functions. GA optimization methods are applied to optimize the entire fuel cycle, where the choice of fitness function is not restricted to a default variable or figure of merit, but is instead customizable to any parameter or figure of merit chosen by the user. The result of this research is a new method and tool that allows for model, data, and method customization for the evaluation and characterization of advanced nuclear fuel cycles.

Recommendations for Future Research

The code was intended to be modified and expanded upon for the purposes of different reactors, fuel cycle models, and economic studies. Functions used for data handling and parameter sampling can be applied to thorium fuel cycle analysis, other Gen IV fuel cycle models with new reactors, and other analyses without significant modification to the basic structure. A particularly interesting application would be to expand the code to perform time-dependent analyses that may be used in planning phases with various decision criteria and growth models.

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APPENDIX

A. Code Information

Matlab Environment Requirements

- Matlab R2009b or later (64-bit recommended for large Monte Carlo sets)
- Optimization Toolbox
- Genetic Algorithm and Direct Search Toolbox
- Neural Network Toolbox **(*Recommended*)**
- Statistics Toolbox **(*Recommended*)**
- Parallel Computing Toolbox **(*Recommended*)**
- Matlab Common Runtime (Required if using compiled version only on machine without Matlab R2009b)

Table A1: Function Names and Descriptions

Name	Group	Called By	Description
SimFuelCycle	Driver	Matlab command prompt	Main program that directs user input to appropriate functions
dist_single.m	Random Number Generator	helper_sampler	Replicates single value to number of trials in analysis
dist_triangular.m	Random Number Generator	helper_sampler	Samples variable data according to triangular distribution
dist_uniform.m	Random Number Generator	helper_sampler	Samples variable data according to uniform distribution
fc_hybrid	Fuel Cycle Model	SimFuelCycle	Hybrid fuel cycle with fast reactor data interpolation
fc_hybrid_forced	Fuel Cycle Model	SimFuelCycle	Hybrid fuel cycle with forced reduction of fast reactor mass
fc_oncethrough	Fuel Cycle Model	SimFuelCycle	Once through LWR fuel cycle
helper_dataloader	Helper Function	Fuel Cycle Module	Loads data libraries into analysis code for interpolation
helper_interp	Helper Function	Fuel Cycle Module	Coordinates data interpolation
helper_optimizer	Helper Function	SimFuelCycle	Optimization control function
helper_sampler	Helper Function	Fuel Cycle Module	Parses user input for variable definitions and calls appropriate sampling function
plot_pdf	Plotting	Matlab command prompt	Produces probability density functions of uncertainty analysis data
plot_SaveFigure	Plotting	Matlab command prompt	Exports multiple figures to specific format
plot_dsxy2figxy	Plotting	Matlab command prompt	Transforms point or position from data space coordinates to figure coordinates

B. Tracked Isotope Lists

Table B1: Fission Product List

Number	Isotope
1	Ce-144
2	Cs-134
3	Cs-137
4	Eu-154
5	Eu-155
6	I-129
7	Pm-147
8	Sb-125
9	Sr-90
10	Tc-99
11	Totals for all fission products

Table B2: TRU List

Number	Isotope		Number	Isotope		Number	Isotope
1	Ac-225		20	Cm-250		39	Ra-224
2	Ac-227		21	Es-253		40	Ra-225
3	Am-241		22	Es-254		41	Ra-226
4	Am-242m		23	Es-255		42	Ra-228
5	Am-243		24	Np-235		43	Rn-222
6	Bk-249		25	Np-236		44	Th-227
7	Cf-249		26	Np-237		45	Th-228
8	Cf-250		27	Pa-231		46	Th-229
9	Cf-251		28	Pa-233		47	Th-230
10	Cf-252		29	Pu-236		48	Th-232
11	Cf-253		30	Pu-237		49	Th-234
12	Cf-254		31	Pu-238		50	U-232
13	Cm-242		32	Pu-239		51	U-233
14	Cm-243		33	Pu-240		52	U-234
15	Cm-244		34	Pu-241		53	U-235
16	Cm-245		35	Pu-242		54	U-236
17	Cm-246		36	Pu-244		55	U-237
18	Cm-247		37	Pu-246		56	U-238
19	Cm-248		38	Ra-223		57	Total for all Actinides

C. SCALE Data Creation

The Standardized Computer Analyses for Licensing Evaluation (SCALE) package produced by ORNL is a combination of many codes useful in the evaluation of criticality safety and LWR core analysis. A few of the codes in the package are used in this research, thus a brief overview of their functionality is included [79].

SAS2

The Shielding Analysis Sequence number 2 (SAS2), as used in this research, is a one-dimensional code used to determine isotopic compositions, decay heat, and burn up dependent cross sections for a given reactor type and core composition [80]. A 1-D calculation begins with material homogenization in a fuel pin lattice and then creates a representation of the fuel assembly. After homogenization, assembly averaged fluxes are collapsed into 3-group fluxes, used in ORIGEN-S, to provide irradiation and decay compositions. The homogenization assumption for FR's is more valid than for LWR's as modern LWR's have multiple fuel batches, burnable poison rods, control blades, multiple coolant channels where a 1-D approach loses much of the particle transport information that affects core analysis; thus, a 2-D analysis yields a more accurate approximation for a LWR, but suitable for a FR.

Cross section libraries that are included with SCALE releases 5 and 5.1lack fast fission information for a number of higher actinides, particularly those that were in the driver fuel for the Super PRISM, which necessitates the usage of a separate library to correct the missing data. With fast reactor calculations depending on the fast fission yields, fast fission cross sections, and neutron production, omitting this information yields incorrect flux magnitudes as the source terms are incorrectly calculated; however, the shape of the flux remains the same, which shows that the transport module is working correctly, but that the source terms are incorrect. Using a modified library, the source terms are calculated with full fast fission data that results in a flux on the order of 10^{15} n/cm^2 which is on the order of the design specifications for the Super PRISM [42,43,44].

ORIGEN-S

The Oak Ridge Isotope Generation code (ORIGEN-S) provides time-dependent isotopic composition and depletion within the SCALE package [81]. Cross section libraries and flux information are output from one of the physics codes, SAS2 or TRITON, and imported into ORIGEN. In this module, the source terms and reaction rates are calculated for user defined time steps to deplete fuel by producing fission products, higher actinides through neutron absorption, and spontaneous decays to allow for isotopic depletion analysis over an arbitrary period of time.

ORIGEN-ARP/ARP

Automatic Rapid processing (ARP) is a SCALE module that interpolates pre-calculated burn up dependent cross section libraries from a SAS2 or TRITON model.

Interpolation of cross sections without recalculating the physics data allows for significant reduction in time required for analysis [81,83,84]. For light water reactors, SCALE includes TRITON libraries for various PWR and BWR reactors of differing characteristics such as moderator density, assembly type, enrichment, etc. ORIGEN-ARP is a graphical front end to the ORIGEN-S and ARP modules that allow a user to select a reactor type from a list and enter various parameters specify irradiation and decay times for ORIGEN-S to calculate decay heat, mass, or other units of interest. As a time saving feature, this is rather useful when reactor parameters may change by small amounts.

OPUS

OPUS is a SCALE module that parses ORIGEN-S output and provides information for a variety of plots [85]. Plots of isotopic composition versus time are not necessarily what are needed for this research; however, because the output of ORIGEN-S is rather lengthy and has a tendency to change location between releases of SCALE, OPUS is used solely for the plot data files that ORIGEN-S creates. Within these files, data are stored in tabular format where each column is a separate isotope and rows are the time-dependent mass or decay heat, depending on the user designated output type. This formatting allows for a much more simplified and human readable output of ORIGEN-S data than ORIGEN-S produces itself.

D. Neural Network Design

Generalized Regression Neural Networks

While neural networks are usually thought of as regression or classification tools, certain types, such as the generalized regression neural network (GRNN), can perform difficult, nonlinear interpolations easier than would be possible with other types of algebraic interpolations [86,87,88,89,90]. In the case of the radioactive decay chain for a given isotope, an equation may be written using multiple exponentials for a simple case, but a more detailed case would require multiple branches of exponentials for a single initial isotope, which is performed using matrix multiplication in ORIGEN-S; however, for the case of several thousand Monte Carlo trials, individual calculations require a large amount of CPU time, which is why the neural network interpolation method is implemented. A useful feature of the GRNN is that it can simulate the intricate math associated with the radioactive decay chain without using as much CPU time.

This type of network architecture, shown from in Figure 28 below, is built on the storing of training values inside of radial basis function (RBF) hidden nodes [86,88]. Each of the training inputs, if recalled exactly in a test data set, is automatically given a weight of 1 when determining the next layer as expected; however, a RBF does not automatically give the training output for the trained input, as it uses a spread constant to define a region over which weights are assigned to nearest neighbor values before calculating the next layer.

A Gaussian normal distribution may be used to define these weights as the distance from the training point with regard to the spread constant as an analog to the standard deviation. If the test input is too distant from area defined by the spread constant, a low weight, approaching zero, is assigned, while the converse is true for a point within the spread constant's reach receiving a higher weight. Shown in Equation 1, the interpolated value, $\hat{Y}$ as a function of x, is determined by the summation of all training values, Y_i, according to the spread constant, σ, and distance from the training point, D_i.

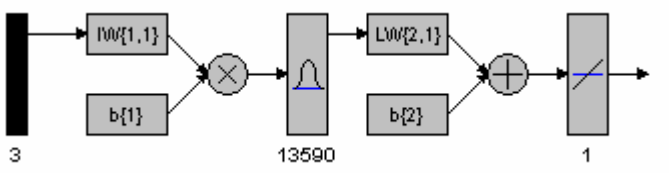

Figure 28: Generalized Regression Neural Network Architecture

$$\hat{Y}(x) = \frac{\sum_{i=1}^{n} Y_i \exp\left(\frac{-D_i^2}{2\sigma^2}\right)}{\sum_{i=1}^{n} \exp\left(\frac{-D_i^2}{2\sigma^2}\right)}$$

Choice of spread constant is extremely important as a small spread constant will not generalize or interpolate well, but too large a spread constant can interpolate incorrectly. A spread constant should be chosen such that at least 2 points are included, assuming sufficient training data is applied, to allow a decent guess as to the interpolated value, where only a single point may give a decent approximation in some cases, it begins to fail as the test point moves further away from the training point. In the event that a test point is too distant from any of the training points, the radial basis function returns very low weights that force the output to approach zero.

Training of the GRNN can use scored or unscored data, depending on the variability. If the data have too large of variance, the function may not operate as intended, thus validation data should be used during the training procedure to assure that the function is operating correctly. A benefit of this network design is that the output layer can use either natural or scaled units regardless of the number or variation of data. In testing, standardized scores are created according to Equation 2, where X is the data set, μ is the population mean, and σ is the population standard deviation.

$$Z = \frac{X - \mu}{\sigma}$$

Because the network is designed to store training data locations, the network is largely subject to the curse of dimensionality and requires a large number of points to train correctly for a large range of training data. Training is of feed-forward style that is performed quickly, but simulates at a decreasing speed as the amount of data trained into the network increases, thus networks should be kept as simple as possible if speed of simulation is a source of concern.

VITA

Jeff Preston was originally born in Michigan, but raised in Georgia. In 2000 he decided to begin his undergraduate degree at North Carolina State University after attending a summer program sponsored by the NCSU Nuclear Engineering Department. In fall of 2002 he transferred to Georgia Tech where he graduated with a degree in Nuclear and Radiological Engineering. After graduation, he began graduate school at the University of Tennessee in Knoxville. He graduated with a Masters in Nuclear Engineering in 2006 and chose to continue towards a PhD in Nuclear Engineering.

In spring of 2008, he accepted a job at ICx Technologies in Oak Ridge, TN as a physicist. At ICx he develops new radiation detection techniques. He is a member of IEEE, American Nuclear Society, Institute of Nuclear Material Management, and the American Physical Society.

In summer of 2008, his wife, Shannon, had their first child, Grace. They enjoy Dr. Seuss and Legos.