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I am submitting herewith a thesis written by Gary Alan Sweder entitled "Methodology for Uncertainty Analysis of Advanced Fuel Cycles and Preliminary Results." I have examined the final electronic copy of this thesis for form and content and recommend that it be accepted in partial fulfillment of the requirements for the degree of Master of Science, with a major in Nuclear Engineering.

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Methodology for Uncertainty Analysis of Advanced Fuel Cycles and Preliminary Results

A Thesis

Presented for the

Master of Science Degree

The University of Tennessee, Knoxville

Gary Alan Sweder
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Abstract

This report assesses the sensitivity and uncertainty associated with certain advanced nuclear fuel cycles due to the variance of chosen parameters and how these results relate to the deep geological nuclear waste repository. High burn up uranium oxide, mixed oxide, and fast spectrum nuclear fuels are the advanced fuel cycles considered. The parameters that are varied in these cases are: the time of advanced fuel implementation, energy growth rate, fuel burn up, and reprocessing introduction and capacity. The results analyzed are the amount of spent fuel and the amount of Pu in spent fuel in the year 2099. The advanced fuel cycle scenarios are modeled using the DANESS code developed by Argonne National Laboratory. All the fuel cycles modeled in this report are highly sensitive to the above-mentioned varied parameters. In a 0% energy growth rate case the plutonium fast burner reactor significantly reduces the amount of waste destined to the repository. Compared to current once-through fuel cycle practices, the fast reactor reduces waste by 50-52 percent. As energy demand grows, the high burn up case of 100 (GWd per ton heavy metal) fuel, as modeled in this thesis, reduces the mass destined for the repository greatest. In the 1.5% energy growth rate, spent fuel mass is reduced 32-44 percent, and in the 3.0% energy growth rate those numbers are 43-49 percent.

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CHAPTER 1: Introduction and Background to Thesis

1.1 Introduction

The current U.S. energy market is in a transitional period in its history. Record energy demand has encouraged the nuclear market to seriously consider the construction of new nuclear energy plants. However, the licensing and construction process is time consuming and requires enormous investment capital. The initial investment in a new nuclear plant is greater than all fossil-based energy sources. Therefore it is pertinent that there be clear evidence of what the future holds for the nuclear industry.

On August 8th, 2005 President Bush signed the Energy Policy Act of 2005, the country's first comprehensive energy bill in thirteen years, into law at Sandia National Laboratories in Albuquerque, New Mexico. The Energy Policy Act of 2005 includes a wide range of measures designed to support the current nuclear fleet as well as bolster the construction of new nuclear energy generating facilities.

The nuclear energy related provisions in this Act include:

- Stand-by support to offset the financial impact of delays beyond industry's control that might occur during construction and beginning of operations for as many as six new reactors.
- Extension of the Price-Anderson Act, which provides for industry self-funded liability insurance for 20 years.
- Authorization of a production tax credit of 1.8 cents per kilowatt-hour for the first 6,000 mega-watt hours from new nuclear power plants for the first eight years of operation, subject to a \$125 million annual limit.
- Authorization of \$1.25 billion to fund a Next Generation Nuclear Plant project at Idaho National Laboratory that would generate both electricity and hydrogen.
- Additional nuclear power plant security measures.
- Updates of tax treatment of decommissioning funds.

- A provision directing the Department of Energy to report to Congress within one year with a long-term plan for the disposal of high-level nuclear waste.
- Authorization of funding for the Advanced Fuel Cycle Initiative, which would provide for research in developing advanced nuclear plants, better proliferation resistant fuel cycles, and improved management of spent nuclear fuel.

(wikipedia.org/wiki/Energy_Policy_Act_2005)

The last two bulleted items are significant to the work contained in this thesis. First, a major item in this thesis is to examine the repository benefits from the implementation of advanced fuel cycles. In terms of repository benefits, spent fuel mass totals and plutonium in spent fuel mass are presented in this thesis. This will be discussed in greater detail later in the report. Secondly, this research project by the University of Tennessee receives its funding from the Department of Energy through the Advanced Fuel Cycle Initiative.

During the first year of this project, work consisted primarily of three objectives:

1. Design and analyze advanced fuel cycles for LWRs, including BWRs
2. Examine the effects of nuclear material data and industry infrastructure uncertainties on fuel cycles and analyze corresponding feedback
3. Identify and assess the repository benefits of the advanced fuel cycles

In order to accomplish these objectives several tasks were identified for focus in the first year.

1. Review current advance fuel cycle studies
2. Develop fuel cycles strategies that have the potential to improve sustainability, reduce waste, and decrease proliferation of nuclear power production
3. Conduct conferences between the University of Tennessee and Argonne National Laboratory in order to come to consensus on advanced fuel cycle designs and strategies
4. Identify pertinent fuel cycle parameters that may have an effect on results
5. Perform a sensitivity analysis on those agreed upon nuclear parameters in the DANESS code

6. Evaluate repository benefits of the modeled advanced fuel cycles against current once-through cycle

Significant progress has been made on tasks 1-5 in the first year. Progress has also been made on task 6, however not to the extent of tasks 1-5. This is because we have not been able to fully analyze the heat loads corresponding to the modeled advanced fuel cycles. Currently, graduate students Jeff Clark and Jeff Preston from the University of Tennessee are performing calculations with the SCALE and MATLAB codes to obtain the nuclear data needed to accomplish this task. The information obtained by these students will eventually be incorporated into a customized version of DANESS, so that the desired heat load calculations may be obtained in the code.

1.2 Advanced Fuel Cycle Initiative (AFCI)

The Department of Energy created the Nuclear Energy Research Initiative (NERI) in order to address the technical and scientific concerns with sustaining the current nuclear fleet as well as expanding nuclear power in the United States. NERI provides funding for research through university, national laboratory, and industry partnerships. The goal is to maintain the U.S nuclear infrastructure and technology as well as develop new research and design projects to ensure the United States remains a global force in the nuclear industry.

In 2004, the NERI research moved to exclusively fund research and development supporting three of DOE's nuclear research and design initiatives: the Generation IV Nuclear Energy Systems Initiative, the Advanced Fuel cycle Initiative, and the Nuclear Hydrogen Initiative. Also only U.S. universities are permitted to serve as the chief investigators for these new projects. National laboratories and private companies are still allowed to participate as collaborating partners (www.ne.doe.gov/infosheets/afci.pdf). Argonne National Laboratory has been working with the University of Tennessee on the research project presented in this thesis. This new direction by the DOE aims to preserve

and advance the U.S. nuclear infrastructure as well as equip the next generation of engineers with the skills necessary to develop these programs in the future.

The mission of the AFCI is to develop proliferation resistant spent fuel management and transmutation technologies in order to facilitate a transition from the current once-through nuclear fuel cycle to a sustainable, proliferation-resistant closed fuel cycle. The primary goals of the AFCI program are to:

- Reduce the long-term radiological impact of nuclear waste;
- Development of a simpler, more economic repository;
- Reduce nuclear proliferation risks;
- Improve long-term outlook for nuclear power.

Success in these goals creates the opportunity to significantly delay and even possibly eliminate the need for a second geological repository, reduce the amount of civilian plutonium, and recover the viable energy source in spent nuclear fuel.

The AFCI aims at addressing both intermediate and long-term issues. The intermediate issues deal with the spent nuclear fuel, chiefly the reduction of material and the associated heat generation of material requiring disposal. Advanced separation technologies and proliferation-resistant recycled fuels to be used in advanced light water reactors will deal with these issues. The long-term effort of this program will develop fuel cycle technologies that will eliminate minor actinides in a fast neutron spectrum system, dramatically reducing the long-term heat load and radiotoxicity of the waste being sent to a geological repository; this will be addressed by the development of a transmutation fuel cycle through the Generation IV fast reactor program.

In order to accomplish these goals, researchers are focusing their efforts in several areas. The major efforts are in:

- System Studies
- Separation technologies
- Fuel development

- Materials research
- Physics Research

With all that is involved in the AFCI, there is much interesting and beneficial work in progress. However there is still much to do in order for a revitalization of the U.S. nuclear industry to take place. The AFCI aims to advance the nuclear infrastructure, technologies, and manpower in order to see this goal is met.

1.3 Motivation for Research

Advanced fuel cycles offer the opportunity to significantly reduce both the mass of waste and the accompanying heat load destined for geological storage. However, the implementation of advanced fuel cycles require the construction of expensive and technically challenging infrastructure, such as advanced reprocessing technologies and hard-spectrum nuclear reactors. With such a great amount of time and money at risk, it is important that the uncertainties associated with the deployment of these fuels be finely understood and minimized.

Virtually all important business decisions are made under uncertainty. Decision making ability requires the processing of information, construction of knowledge, and the assessment of associated uncertainties and risks. Uncertainty modeling and analysis improves the ability to make the correct decision. Modeling and analyzing uncertainty stems from the fact that the abundance of data does not necessarily denote certainty, and in some cases may lead to errors in decision-making due to over-confidence or overwhelming data.

As the nuclear industry advances in scientific understanding and develops new technologies, human knowledge is expanded due to this excess information. It is critical that the uncertainties associated with this information are assessed and the level of knowledge/ignorance is quantified.

Decision-making involves risk control and management. Decision makers in the nuclear industry generally have an idea of the risks involved in implementing new technologies; the reduction of risk needs to be pursued by government, industry, and universities. Policy makers are required to subjectively weigh the benefits against the risks and associate the uncertainty inherent in such decisions. Along with considering the economic and infrastructure demands on the nuclear industry, public opinion and government regulations add to the overwhelming amount of information a decision maker must deal with. Uncertainty modeling and analysis are imperative to such risk-based decisions.

1.4 Scope of Thesis

The purpose of the work presented in this thesis is to perform an initial sensitivity analysis as well as to collaborate with the Argonne National Laboratory members on development of a customized DANESS code. The research members from the University of Tennessee performed no coding of DANESS; the primary function of the graduate students was to familiarize themselves with DANESS and facilitate the exchange of ideas about the software between the University of Tennessee and ANL. The goal of this ongoing dialogue is the development of a customized version of DANESS that meets the goals set out in the research proposal.

The work in this thesis focuses on the dynamic analyses of several advanced fuel cycles. Dynamic simulation models were constructed using the DANESS code, which is discussed in more detail in a subsequent chapter. For this thesis the following advanced fuel cycles were modeled:

1. Once-through cycle utilizing high burn up fuels (UOX 60 and UOX 100 GWd/thm)
2. Single MOX recycle
3. Fast spectrum reactor (Plutonium burner)

This study focuses on the time of implementation of the above fuel cycles, as well as a general sensitivity and uncertainty analysis of several parameters.

The analysis performed in this thesis is primarily a sensitivity analysis; however it may also be considered a small-scale incremental uncertainty analysis. The work presented represents a manual, incremental sampling of a specific input parameter and the resultant distributions of output parameters of interest. The incremental input distribution was placed on the nuclear reactor shutdown profile.

Also several strictly sensitivity analyses are performed in conjunction with the work of this thesis. Sensitivity analysis is performed regarding the input parameters: energy demand and fuel type usage, in conjunction with the incremental input distribution placed on the reactor shutdown profile.

CHAPTER 2: DANESS (Dynamic Analysis of Nuclear Energy System Strategies)

2.1 Introduction

Developed by the Nuclear Division of Argonne National Laboratories, DANESS (Dynamic Analysis of Nuclear Energy System Strategies) is an integrated dynamic nuclear process model for analysis of today's and future nuclear energy markets on the fuel batch, reactor, country, and even worldwide scale. Beginning from the current nuclear reactor park and fuel cycle scenario, DANESS will analyze the energy-demand motivated nuclear energy systems over time and accommodates changes in the reactor parks and fuel cycle options as defined by the user. New reactors are introduced based on the energy demand and the economic and technological availability to do so. DANESS is not intended to predict the future of the nuclear market, but instead is intended for projecting and analyzing different nuclear energy paths in a robust and consistent manner. In order to accomplish this task, DANESS encompasses the major aspects of nuclear energy process models, such as:

- Technical Aspects: different reactor types, fuels, and fuel cycle technologies interact in a symbiotic manner
- Economic Aspects: in a competitive energy market, nuclear energy and nuclear technologies must be economically competitive
- Socio-political Aspects: safety, waste management, and non-proliferation are all vital socio-political concerns of nuclear energy and thus are a major aspect of nuclear development

This report focuses heavily on the socio-political aspects of nuclear energy development, especially the concerns of waste management. However all the above aspects receive significant consideration in developing results and conclusions (Van Den Durpel, 2004).

The utility of DANESS is found in nuclear scenario analysis of different possible developmental paths for nuclear energy systems from a governmental, utility, or research

and development perspective. Students, professionals, researchers, and policy makers can use the program, or anyone interested in situation-based analysis of today's and future nuclear energy systems. The program's intended use is as follows:

- Analysis of exploratory paths for nuclear energy: the impact of new developments in the nuclear reactor park and fuel cycle facilities may be analyzed in an incorporated manner. The impact on the cost, major aspects of the fuel cycle, and the sustainability of a desired industry path may be analyzed.
- Integrated process model: research and design of new nuclear technologies and facilities is a costly endeavor and includes significant amounts of data generated from programs such as DANESS. An integrated process model that includes accurate mathematical modeling of the physics and cost-scaling relationships facilitates the optimization of parameters for the industrial system and the economic feasibility of the system in question.
- Parameter investigation model for new designs: DANESS can assist in examining the influence of key parameters in the complete nuclear system, guide research and development efforts in identifying the major drivers for new technologies, and help analyze the trade-offs between parameters.
- Economic analysis of nuclear energy systems: utilities are continually striving to lower the costs of generating electricity. DANESS can be used to calculate today's as well as future energy cost based on the technical aspects of the plant, the fuel, the fuel cycle, as well as government influence.
- Governmental role: DANESS may be used as a policy advisor for interested governments. Several government policy options such as tax rates and price premiums may be examined in order to analyze the role of government policy in the sustainability of nuclear energy.
- Educational Use: the ability to simulate and develop numerous nuclear energy scenarios may facilitate the understanding of nuclear energy systems for students as well as the general public and policy makers (Van Den Durpel, 2004).

This list, of course, may be extended. However it does indicate the major intended uses of the program. This thesis and research work uses the DANESS program primarily for the first three uses bulleted above. Economics and governmental roles were not scrutinized heavily for this thesis.

The DANESS program utilizes the iThink/StellaTM software environment, which is developed by iseeTM systemsTM. The iThink/StellaTM software is a graphical user interface software that facilitates complex system modeling. It allows the simulation of a system over time and permits the building of sub-models to support hierarchical model structures. The software consists of basic stock and flow diagrams, which support the basic language of systems thinking. Enhanced stock types that allow for discrete or continuous process, action arrows, and decision diamonds allow for complex models such as DANESS to be built. The software easily communicates with the user through input knobs, sliders, switches, and buttons and output graphs and tables. Figure 1 shows the graphical interface for the DANESS software.

2.2 Architecture of DANESS

The model allows the simulation of up to 10 reactor types and 10 fuels simultaneously, and is composed of interconnected sub-models each of which is intended to perform a specific part of the simulation. Figure 2 shows a schematic of the basic design of DANESS. These sub-models in Figure 2 will be briefly described in a following section. The sub-models depicted in Figure 2 have a symbiotic relationship, where the initial parameters defined in the user interface are utilized throughout the course of the model.

The major sub-models are as follows:

- Energy-demand scenario model: The DANESS program is an energy driven model, where the user defines the energy demand scenario. In this case, "energy driven model" implies that based on the specific energy demand scenario the model will attempt to equate the defined energy demand with the energy produced

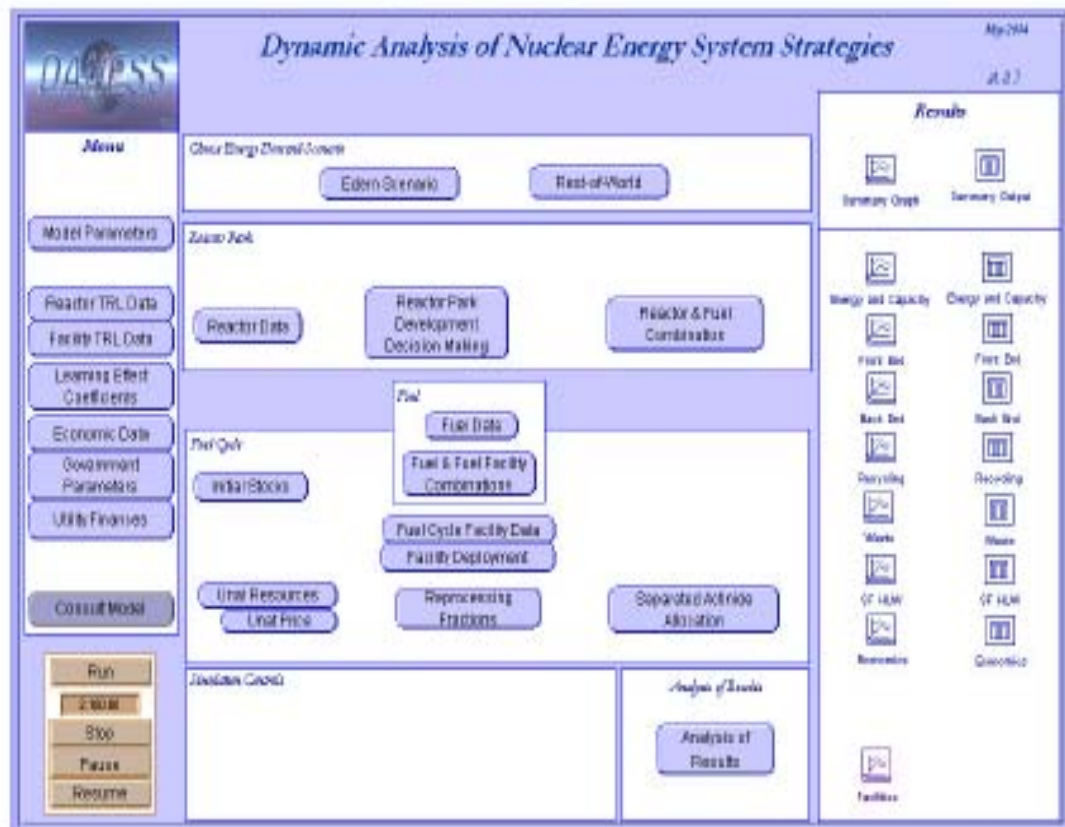
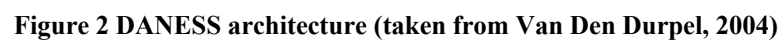


Figure 1 DANESS graphical user interface (taken from Van Den Durpel , 2004)



by the reactor park. There are several pre-defined energy scenarios taken from IIAS/WEC or other such studies, exponential energy growths, or graphically user defined scenarios. This thesis utilized the exponential energy growth scenario.

- Rest of World model: The Rest of the World model is related to the "U-Price" model. The user may decide to simulate the rest of the world energy demand, but primarily this model simulates the depletion of natural uranium and the effect of this depletion on the natural uranium price. This model was not used in this research.
- Energy utility strategic model: The DANESS model simulates the general behavior of the nuclear utility sector as a whole over an extended period of time. This sub-model encompasses the core of the financial assessment by utilities and incorporates the essential decision drivers for utilities to invest in new nuclear generating capacity.
- Reactor and Fuel Cycle Facility Technology development model: Each new reactor or fuel facility type must result from research and development and will progress in technical readiness. This sub-model traces the technological readiness for each reactor and fuel cycle facility. However this sub-model does not find prominent use in this research being that the user defines the new reactor capacity being built.
- New Reactor Capacity Decision model: As mentioned earlier, DANESS is an energy driven model. Thus the amount of new generating capacity that is built is driven by the shortage of energy production and by the shutdown of existing reactors. The ordering of new reactors depends on three decisions:
 - The expected shortage in energy production;
 - The economic performance of fuel and reactor combinations;
 - The availability of fissile material.

Being that economic decision making was not used in this research, the user sets the distribution of new reactor and fuel combinations. Figure 3 shows a flow chart for the new reactor decision process

- Reactor and Fuel Cycle Facility History model: These two models track the advancement of the reactor and fuel cycle facilities and provide feedback to the available, projected, and replacement capacity decision models
- Fuel Cycle mass-flow model: The fuel-cycle mass-flow model is the core of DANESS and accounts for all aspects of the fuel cycle for a given fuel type. Up to ten types of fuel may be followed.
- Fuel Cycle costing model: This model calculates the levelized fuel cycle costs for the different reactors and fuels.
- U Price model: This models the evolution of the uranium price and takes into account the different timing of operations for each fuel batch. The price is ultimately used as input for the fuel cycle costing model.
- Energy costing model: This model calculates the actual and net-present value of the energy produced by each reactor type. The energy cost is divided into a capital cost term, an operation and maintenance cost term, and a fuel cycle cost term. Each of these cost terms is calculated and used in the new capacity decision-making model for investment appraisals of new reactors.
- Government role model: The government may have an important role in the development of new nuclear capacity. Government may influence tax rates on investment, risk premium reduction for new technologies, and much more. This model allows the user to simulate the long-term policy impacts of varying these means of government action (Van Den Durpel, 2004).

Many of these sub-models, such as the costing, pricing, and government role, were not used for the work presented in this report. The sub-models of importance are also the sub-models central to the DANESS program. Primarily those models are the energy demand, new reactor capacity decision, and fuel-cycle mass flow models.

2.3 *Uncertainty Analysis in DANESS*

Computer based modeling and analysis of physical processes has increased dramatically over the past several years. Mathematical models are built from independent variables, dependent variables, and a system of relationships between the two. Even if the process is modeled mathematically accurately, there is still error from the numerical methods employed to solve the equations. Mathematical models also involve the input of parameters that have been found experimentally or through other mathematical models; these values are not known precisely but instead have a range of uncertainty associated with them. It is important that parameter uncertainties and variations in results be assessed. This is accomplished by sensitivity/uncertainty analysis.

Uncertainty analysis and sensitivity analysis are closely related, but are uniquely different. Uncertainty analysis measures the variability and probability in model outputs that result from the uncertainty in model inputs. Input variables are given a specific distribution of values that results in a corresponding distribution of the output variables in question. Uncertainty analysis executes a more complex analysis of the relationship between input variability and output variability than does a sensitivity analysis. A sensitivity analysis measures the rate of change of the output with respect to the variation in input. Using this definition, the sensitivities are the partial derivatives of the output with respect to each input. The drawback of considering only the sensitivity is that it ignores the degree of uncertainty in each input. An input with a small sensitivity may have a large uncertainty or vice versa. Therefore both sensitivity and uncertainty must be considered in the analysis.

This thesis primarily involves a small-scale uncertainty analysis associated with alternative fuel cycle scenarios. The varied input parameter is "the shutdown of existing reactors", as it is named in the code. Thus, varying this input effectively changes the time in which advanced fuel cycles are implemented into the power producing reactor park. This type of analysis is a simple and effective means of analyzing the uncertainty, which isolates the change in when these fuel cycles are introduced on several output variables.

There are several advantages to this type of analysis. First, the process provides more information upon which to base a decision. Second, because the process requires examination of the factors most likely to influence pertinent output, the analyst is better informed as to what the results represent.

Disadvantages to this method are also prevalent. First, this approach fails to account for factor interaction. When there are dependencies between parameter and output, single value sensitivity analyses does not sufficiently analyze uncertainty. Joint sensitivity analysis can more accurately measure output sensitivity. However, quantifying the interdependencies in variations in input parameters can be difficult, being that DANESS is a non-linear model due to time delay and decision making events.

The different methods of performing uncertainty or sensitivity analysis in DANESS will be discussed with no distinction between the two. The methods described may be used for either type analysis. Figure 4 shows a schematic of the sensitivity/uncertainty capabilities in DANESS.

The uncertainty/sensitivity analysis capabilities in DANESS come via the STELLA interface. There are three methods by which to perform uncertainty/sensitivity analysis:

- 1. Manually set values** - This is a basic approach to an uncertainty/sensitivity analysis, and is the mode used in this thesis. Various input values for a parameter are manually entered into the DANESS interface, and the influence on the results is examined. *"Years of life extension beyond predefined Lifetime of Existing Reactors"* is the manipulated parameter, as labeled in DANESS, for this thesis work. This parameter was varied from 0 to 50 years in 10-year increments.
- 2. Distribution of values** – DANESS allows the same sampling as above in an automated manner instead of performing a manual setup of parameters. Four variation types are available under "Sensi-Specs" in the 'Run' menu of STELLA.

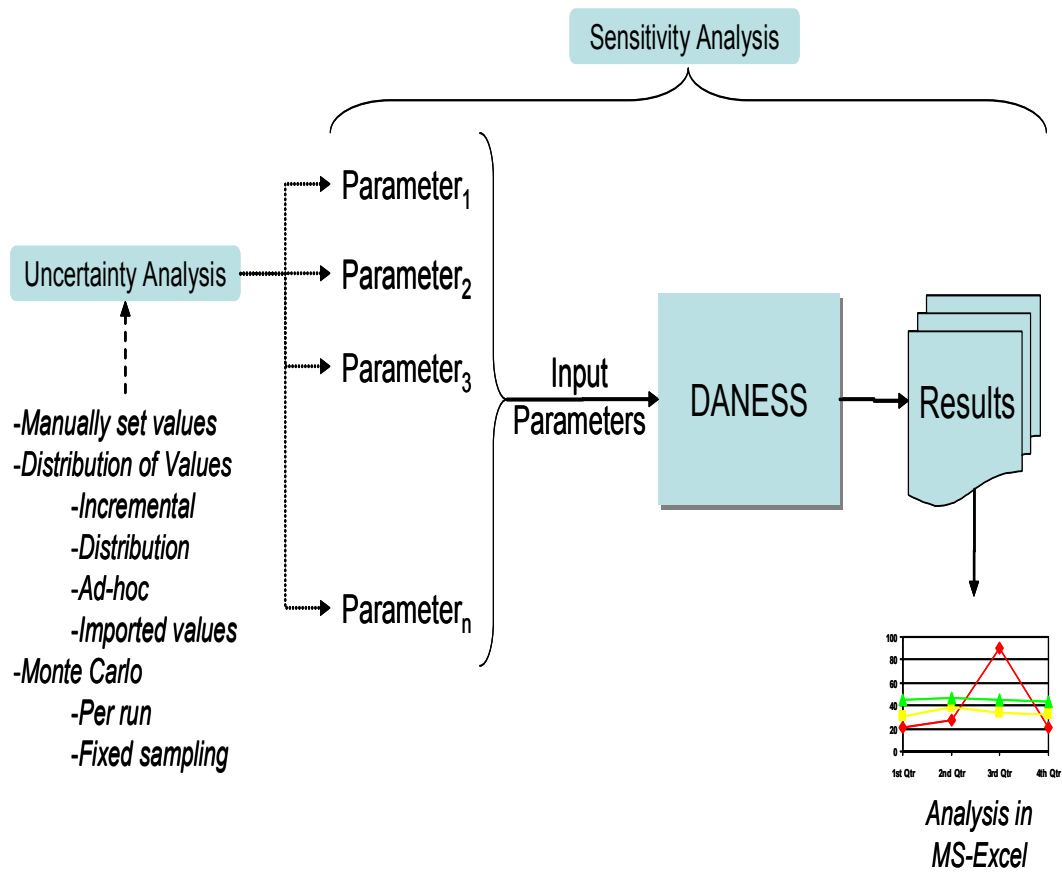


Figure 4 Sensitivity and Uncertainty capabilities in DANESS (Taken from Van Den Durpel 2004)

- a. **Incremental:** By defining a "Start" and an "End" value along with the number of runs, the selected parameter will be sampled linearly between these values, and DANESS will execute the corresponding parameter values for the number of runs.
 - b. **Distribution:** The selected parameter may also be sampled from a normal or rectangular distribution. Given the mean value, the standard deviation and the seed value for the Monte Carlo sampling of the distribution, the parameter will be sampled the number of times the user defines.
 - c. **Ad-hoc:** This option allows to user to specify an ad-hoc list of n values for the parameter of interest, upon which DANESS will be run n times assuming these n values per run. This option is similar to "manually set value", however the ad-hoc method is more automated and the distribution can be saved for later use.
 - d. **Imported values:** Similar to the previous option, a list of values for the parameter can be imported into DANESS. DANESS will run as many times as the number of values contained in the imported list.
3. **Pure Monte Carlo:** This approach consists of assuming uncertainty distribution for the parameters of interest and allowing these parameters to be sampled within each run. There are two options for this approach.
- a. **Per Run:** The sample assumes a random seed, which is selected by DANESS, and each DANESS run results in a different parameter sample per time step and per run.
 - b. **Fixed Sampling:** With a specified seed for the sampling, the sequence of sampled values will be the same for each run.

Figure 5 shows the uncertainty/sensitivity user interface in DANESS. This figure displays how the above analysis methods are implemented in the DANESS environment.

Select desired variable(s) from the allowable list (the list contains all model variables which have a constant value). Click the ">>" button to load variable(s) into the selected list. Reverse the procedure to remove variables from the selected list.

Define how many runs you want (up to 1000) in the sensitivity analysis session. Note that this option is unavailable if "Import Data" has been checked.

Buttons take you directly to new pages of Graph or Table Pads.

This option is available in the Windows version only. Selecting this radio button allows for the import of data, formatted as numbers, from the Windows clipboard into the scrollable Run#/Value list to the right.

Select a variable from within the selected list. Then, define how its values will vary during sensitivity analysis. Values will appear in the scrollable Run#/Value box.

Check boxes for turning sensitivity analysis on, and for printing a record of setups with comparative graphs & tables.

Allowable

- ☐ Stock_1
- ☒ flow_1
- ☒ flow_2
- ☒ Noname_1
- ☒ converter_1

Selected (Value)

- converter_1 (8.00)

Variation Type:

- ☒ Incremental
- ☐ Distribution
- ☐ Ad hoc
- ☐ Import data

of Runs: 3

Define: Graph Table

Start: 4.00 **End:** 12.0

Run# Value

Run#	Value
1	4.00
2	8.00
3	12.0

☒ Sensitivity On
☐ Print Setups

Cancel OK

Variation Type:

- ☒ Incremental
- ☐ Distribution
- ☐ Ad hoc
- ☐ Import data

Start:

End:

For Incremental variation, type in values for Start and End. Then click the "Set" button. The software will assign the appropriate intermediate values.

Variation Type:

- ☐ Incremental
- ☐ Distribution
- ☒ Ad hoc
- ☐ Import data

Ad hoc Values:

For ad hoc variation, type in each ad hoc value and hit the "Enter" key.

Variation Type:

- ☐ Incremental
- ☒ Distribution
- ☐ Ad hoc
- ☐ Import data

Mean:

S.D.:

Seed:

To specify a normal distribution for sensitivity analysis, type in the mean, the standard deviation, and a randomizing seed. Then click the "set" button. Click on the bell curve button to toggle to a uniform distribution.

Variation Type:

- ☐ Incremental
- ☒ Distribution
- ☐ Ad hoc
- ☐ Import data

Min:

Max:

Seed:

To specify a uniform distribution for sensitivity analysis, type in the minimum, the maximum, and a randomizing seed. Then click the "set" button. Click on the distribution button to toggle to a normal distribution.

For distributions, the seed must be a non-negative integer. A seed of 0 will cause the software to choose a random seed, which means your sequence of random numbers will not be repeatable if you ever change the setup for the variable. When you use a positive integer for the seed, you ensure the ability to replicate a particular random number sequence in subsequent sensitivity runs.

Variation Type:

- ☐ Incremental
- ☐ Distribution
- ☐ Ad hoc
- ☒ Import data

Run# Value

Run#	Value
1	2.00
2	4.00
3	6.00
4	8.00

Set Import

Import Data (Windows only):
To select Import Data, you must have copied a numeric data set to the clipboard. Click the Import Data radio button and a new Import button will appear to the left of the scrollable Run # / Value list. Clicking the Import button pastes the data into that list. The number of runs will be automatically set to match the number of values pasted in.

Figure 5 Uncertainty/Sensitivity Analysis options in DANESS (Taken from Van Den Durpel 2004)

CHAPTER 3: Methodology, Set up, and Results of Simulated Scenarios

3.1 Introduction

In this report, variations on specific cases are considered. Those cases are the implementation of higher burn-up, MOX, and fast reactor fuels. Currently nuclear power plants are utilizing uranium oxide fuels with specific burn ups of 40-50 GWd/thm. These cases consider the time of implementing higher burn up, MOX, or fast reactor fuel into reactor parks. Two different fuel burn ups are considered in the higher burn up case: UOX60 and UOX100 fuel. In the MOX scenario, the fuel being introduced is MOX50. This report looks at the time these advanced fuel cycles are implemented into reactors and the results are compared. The fuels are defined as UOX (uranium oxide) or MOX (mixed oxide) where the number that follows is the fuel burn up in units of GWd/thm.

The time of implementation schedule is as follows: The present day reactor shutdown profile is considered first, then additional ten year increments are added to the present planned shutdown for up to fifty years. Simply stated this analysis examines the effects of delaying the integration of different fuels and fuel cycles into the current U.S. nuclear reactor park. Figure 6 graphically shows how this is accomplished.

As can be seen in Figure 6, delaying the planned shutdown of the current reactor park effectively delays the implementation of advanced fuels cycles, thus manufacturing a range of uncertainty. The planned reactor shutdown profile was taken from the Energy Information Administration website (Energy Information Administration, 2005). Renewal of reactor operating licenses after 2002 has not been considered. This is performed for three energy scenarios: 0% growth, 1.5% exponential growth, and 3% exponential growth in demand for nuclear energy. The energy demand growth begins in the year 2010.

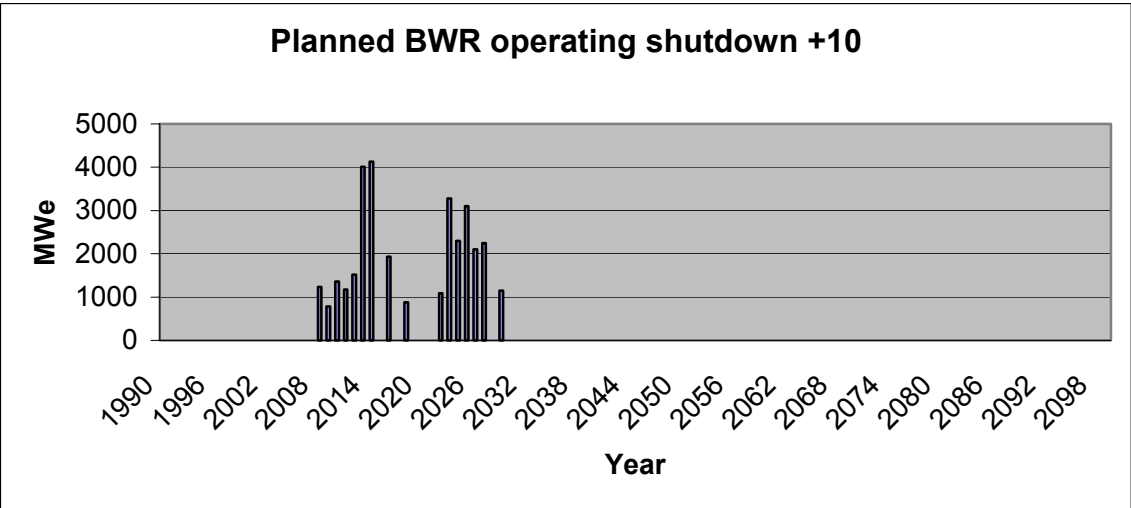
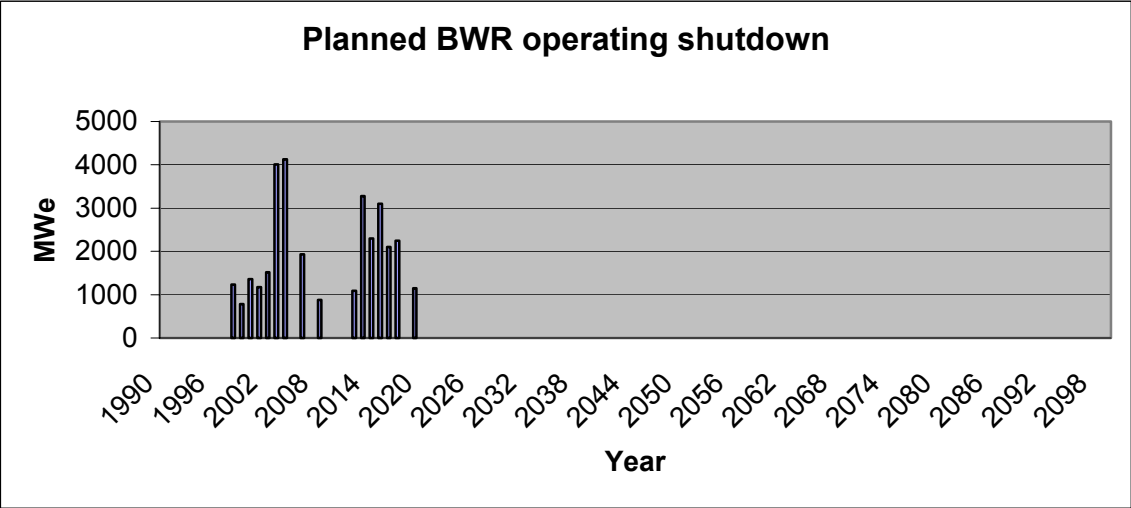


Figure 6 Time implementation of reactor shutdown and subsequent advanced fuel usage

Total electricity sales are projected to increase on an average annual rate of 1.9% in the AEO2005 reference case. Therefore the spread of energy growths used in this analysis provides a range of uncertainty for energy growth. Energy demand is an important factor in these simulations being that the construction of new reactors is based on meeting the desired energy demand. New nuclear construction is designed to meet the entirety of new energy demand.

Spent fuel accumulated and plutonium in the spent fuel are the results examined in this analysis. Spent fuel consists of spent fuel at-reactor and spent fuel interim. This report discusses trends related to the varying of certain parameters within the model. The key parameters that are varied are as follows:

- Time of new fuel implementation in reactors
- Energy growth scenario
- Type of new fuel utilized (High burn up, MOX, or fast spectrum)

This sensitivity/uncertainty analysis is done through the use of DANESS.

3.2 Same Trend Fuel Use Scenario

This scenario models the nuclear reactor park with no change in its fuel usage. "Same Trend" in this context implies the current once-through fuel cycle utilizing UOX40 in BWRs and UOX50 in PWRs throughout the remainder of this century. This case was modeled to gain a nominal spent fuel value for comparison with the more advance fuel cycles discussed in this report.

The initial BWR capacity is 31,789 MWe and the initial PWR capacity is 62,997 MWe corresponding to the current capacity in the United States (eia.doe.gov/nuclear, 2006). The model is set up so that with future energy growth the nuclear reactor park will be composed of 30% BWR and 70% PWR in order to meet energy demand. The BWR reactors utilize UOX40 while the PWR reactors use UOX50. This fuel/reactor combination remains the same throughout the simulation. The fuel cycle facilities are set to unlimited. This means that all stages of the fuel cycle, from uranium mining to spent

fuel disposal, have unlimited processing capacity. This option is chosen in order to reduce the number of input variables that may constrain the simulation to run as intended. Initial energy demand was set at 770 TWe/yr as taken from the Energy Information Administration website. Figure 7 displays the spent fuel results, while Figure 8 shows the amount Pu in the spent fuel.

Figure 7 and Figure 8 show that the results are highly sensitive to the energy growth. Being that energy demand is the only varied parameter in this case, the output's sensitivity to this parameter is clear. Also due to the exponential growth rate of energy, the discrepancies between values in the three graphs of Figure 7 and Figure 8 become much more pronounced with time.

3.3 Methodology, Set up, and Results for High Burn up, MOX, and Fast Spectrum Cases

There are several input parameters common to the high burn up, MOX, and fast spectrum fuel implementation scenarios being modeled for this analysis. In terms of initial energy demand, both cases consider 770 TWe/yr the initial energy demand. Also as stated earlier, three energy demands are considered: 0%, 1.5%, and 3.0% exponential growth rates beginning in the year 2010. The initial reactor park is modeled in the same manner as well. The initial BWR capacity is 31,789 MWe and the initial PWR capacity is 62,997 MWe. The reactor data also remained consistent in all cases. The input data for all the simulations presented in this thesis can be found in the appendices.

In order to look at how the time of implementation affected the results, six separate runs in each case were done. The standard shutdown profile was taken from the Energy Information Administration website, which can be found in the reference section. The manner of time implementation was consistent in all cases. Additional 10-year increments were added to the shutdown profile, thereby effectively delaying the change in current nuclear industry trends.

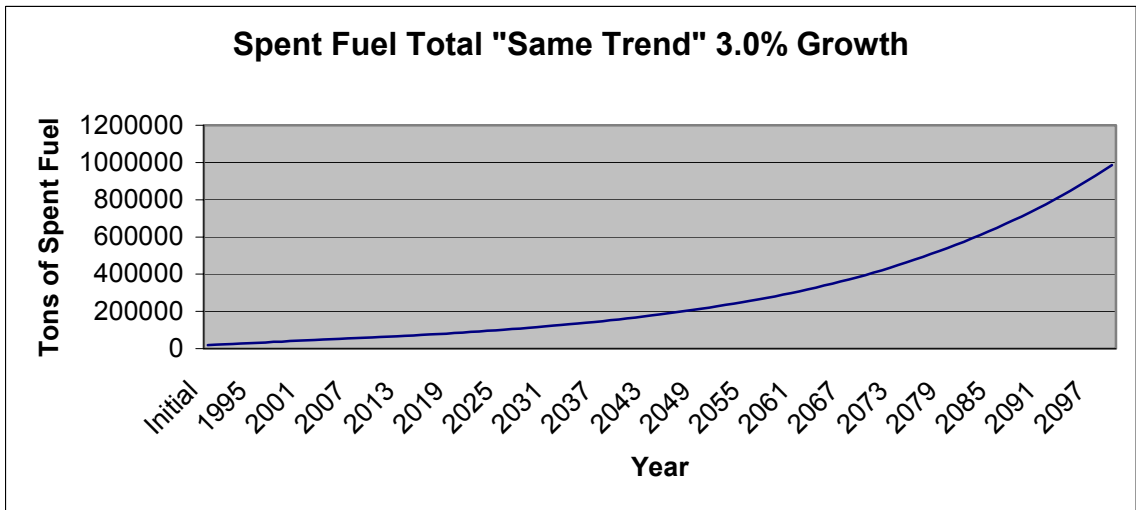
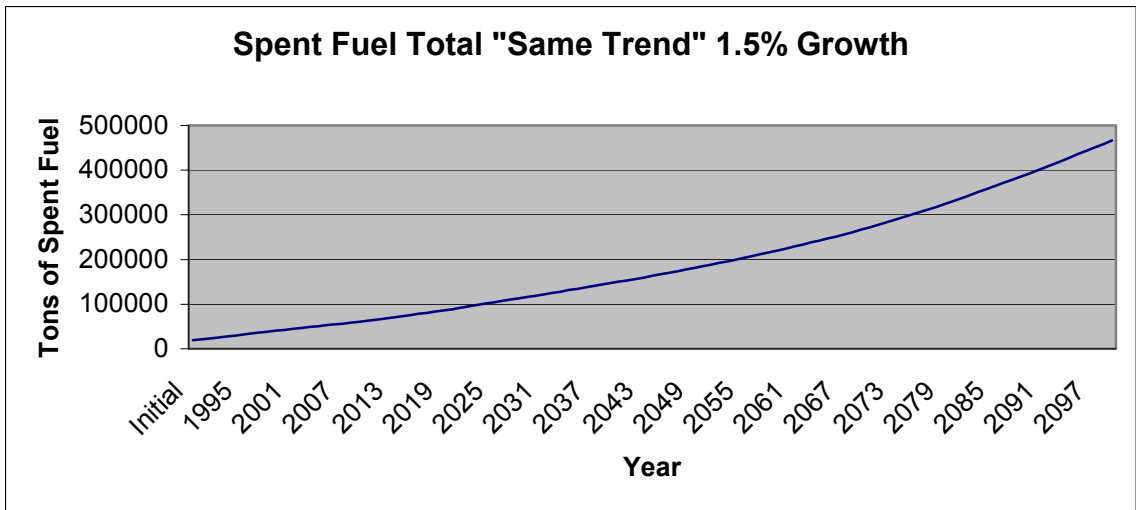
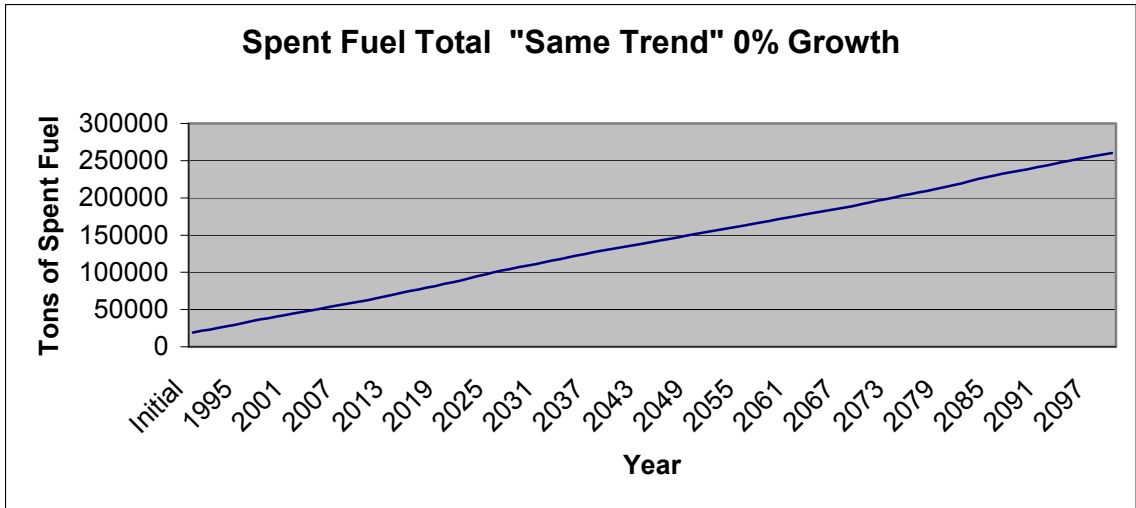


Figure 7 Spent fuel totals for "Same Trend scenario" for 0%, 1.5%, and 3.0% energy growth

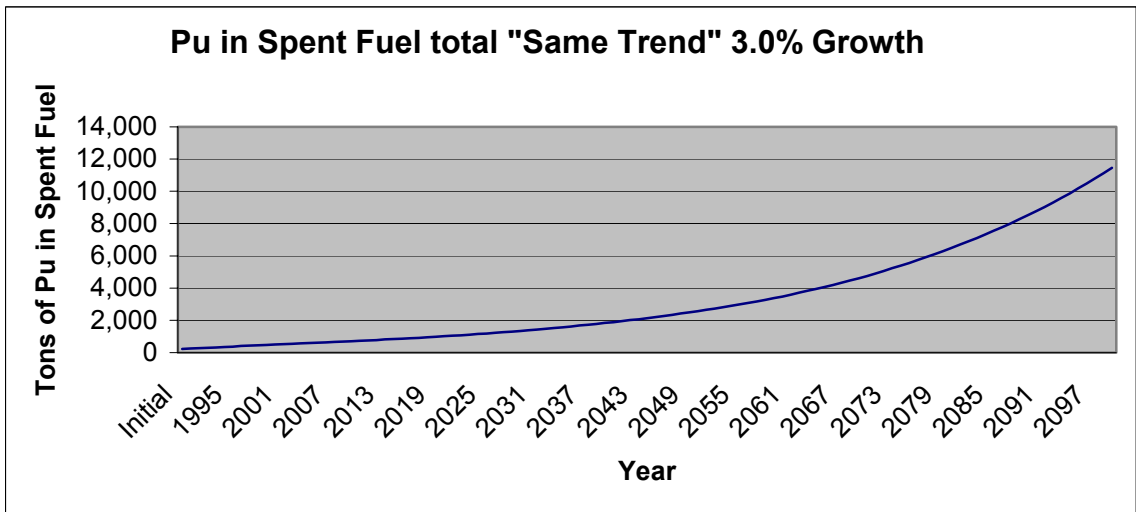
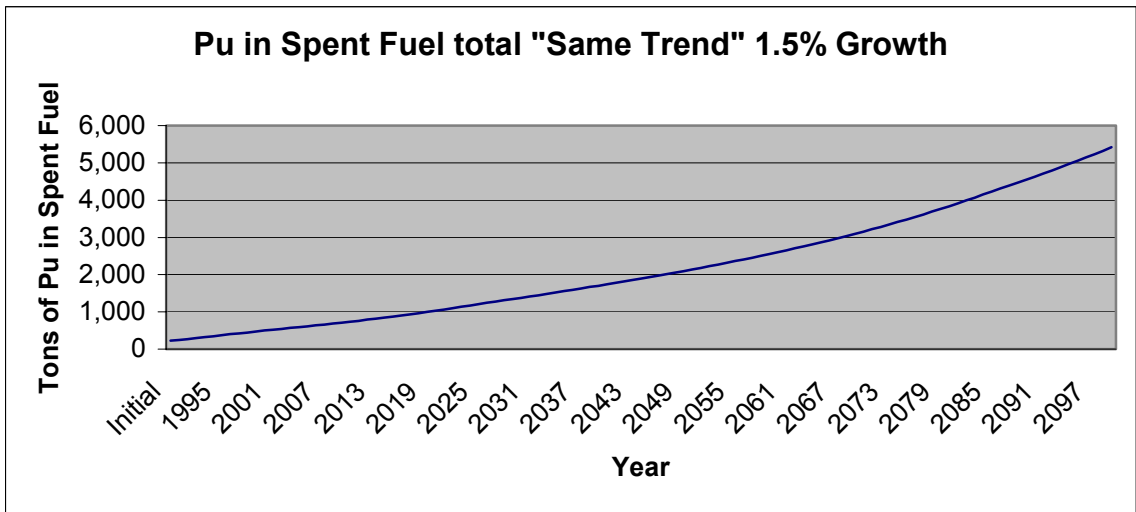
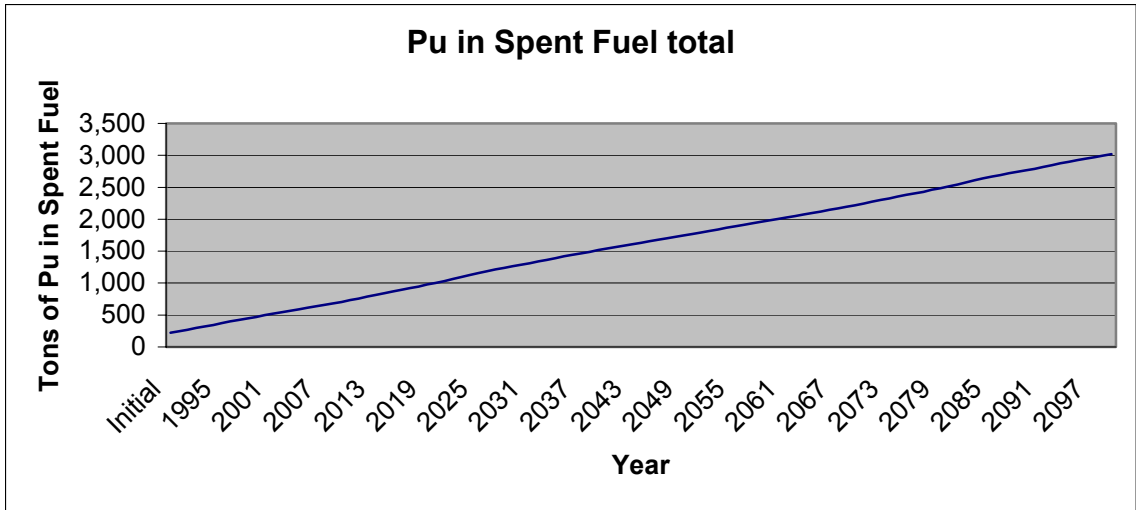


Figure 8 Pu in spent Fuel "Same Trend scenario" for 0%, 1.5%, and 3.0% energy growth

The first run was accomplished using the standard shut down profile; the second and each of the next five simulations were done by extending the shutdown profile by 10 years. The data collected was for the standard shut down, standard shut down +10, standard shut down +20, standard shut down +30, standard shut down +40, and the standard shut down +50. This extension of the standard shut down profile can be seen as an extension of the licensing of the reactors for longer life.

As the old reactor capacity is shutdown, new reactors are built in order to make up for the loss in energy produced. When this report discusses "construction percentages of new reactor", it is describing the share of reactor capacity that is supplied in order to meet energy demand.

Using the method mentioned above, the three cases were then simulated three more times with varying energy demands applied to them. The following sections discuss the details specific to the high burn up, MOX, and fast spectrum cases.

3.3.1 High Burn up Scenario Description and Results

Using the DANESS code, the effects of implementing higher burn-up fuel over time was analyzed. This was accomplished by using a standard shut down profile taken from the D.O.E. website to phase out the current reactor fleet, and implementing new reactors that were fueled with higher burn-up fuels. Two cases were looked at, one in which all the old reactors that were using UOX40 in BWRs and UOX50 in PWRs were replaced with new reactors using UOX60 and another where they all use UOX100. The DANESS setup and data for the UOX60 and UOX 100 cases are found in Appendices A and B respectively.

The construction of new reactors was set to 30% BWR's and 70% PWR's, roughly the same as the current reactor fleet in the U.S. The lifetime of all new reactors was set to 100 years so that the effects of shutting down these new reactors would not be seen. As stated above all the new reactors either used UOX60 or UOX100, depending on the case.

To remove a limiting factor in the construction of the new reactors, all the fuel cycle facilities have unlimited processing capacity.

3.3.1.1 UOX60

The total amounts of spent fuel and Pu in spent fuel were plotted for each shutdown scenario according to the energy growth scenario, as described in previous sections. The total spent fuel includes both the amount of fuel at the reactor site and the fuel sent to interim storage. The transition from at-reactor to interim storage is 5 years. Figure 9 displays the curves of spent fuel mass and Figure 10 displays the Pu in the spent fuel for each energy scenario.

The spent fuel amounts in Figure 9 follow a similar trend regardless of the energy growth rate. For every additional 10-year increment of reactor shutdown, there is roughly a 4,000 ton difference in spent fuel totals. For all three energy scenarios where UOX60 is implemented, there is a maximum spread of spent fuel totals in the year 2099 around 22,000 tons heavy metal. Figure 10 displays similar results, except there is roughly a 40 ton increase in tons of Pu per additional 10-year increment, and for all three energy scenarios where UOX60 is implemented, there is a maximum spread of Pu in spent fuel in the year 2099 around 200 tons heavy metal

3.3.1.2 UOX100

The implementation of UOX 100 fuel is the next case. Again this simulation is modeled in the exact manner as the UOX60 case, except UOX100 burn up fuel is transitioned into the new reactor fleet. Figure 11 and Figure 12 displays the mass of spent fuel and Pu in spent fuel results respectively.

Figure 11 and Figure 12 results follow the trends of the UOX60 scenario, except the results show larger discrepancies between the 10-year increments. In Figure 11, for every additional 10-year increment of reactor shutdown, there is roughly a 10,000 ton difference in spent fuel totals. For all three energy scenarios where UOX1000 is implemented, there is a maximum spread of spent fuel totals in the year 2099 around

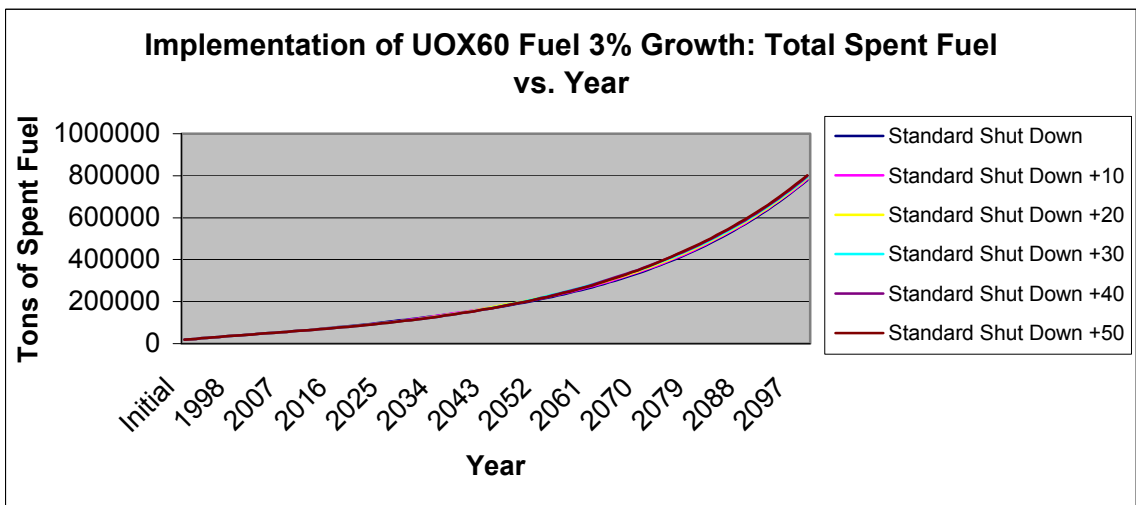
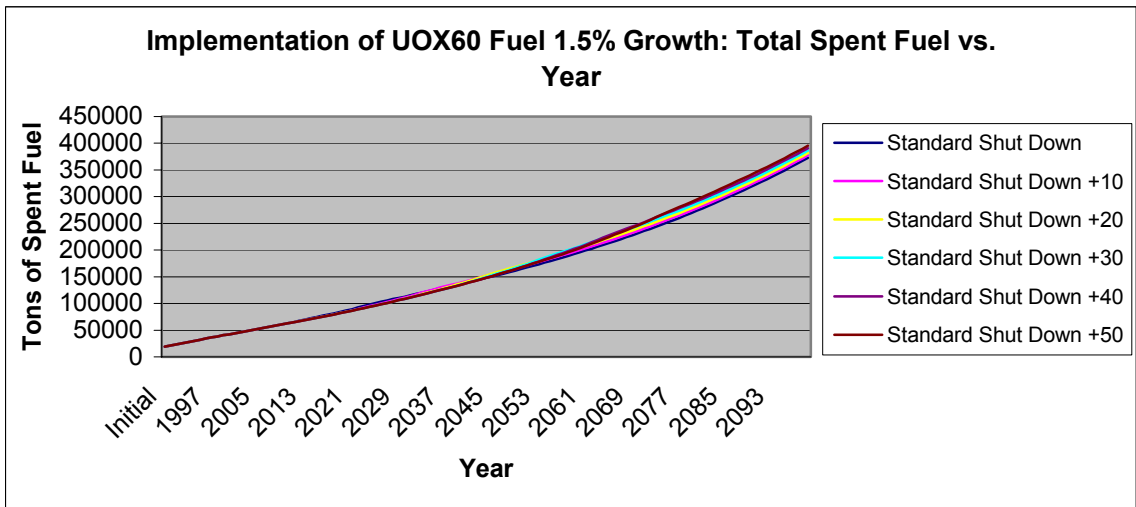
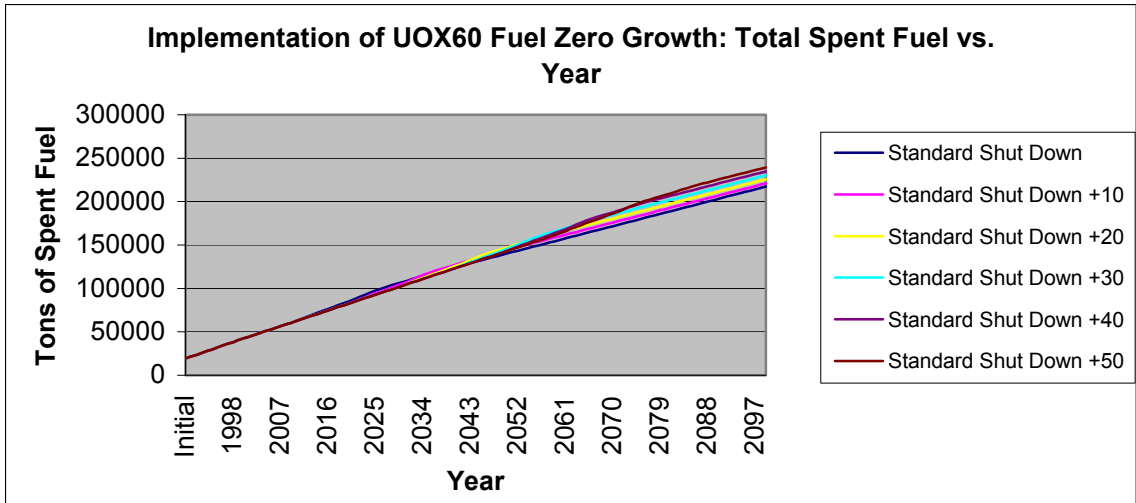


Figure 9 Spent fuel totals for time of implementation utilizing UOX60 for 0%, 1.5% and 3.0% energy growth

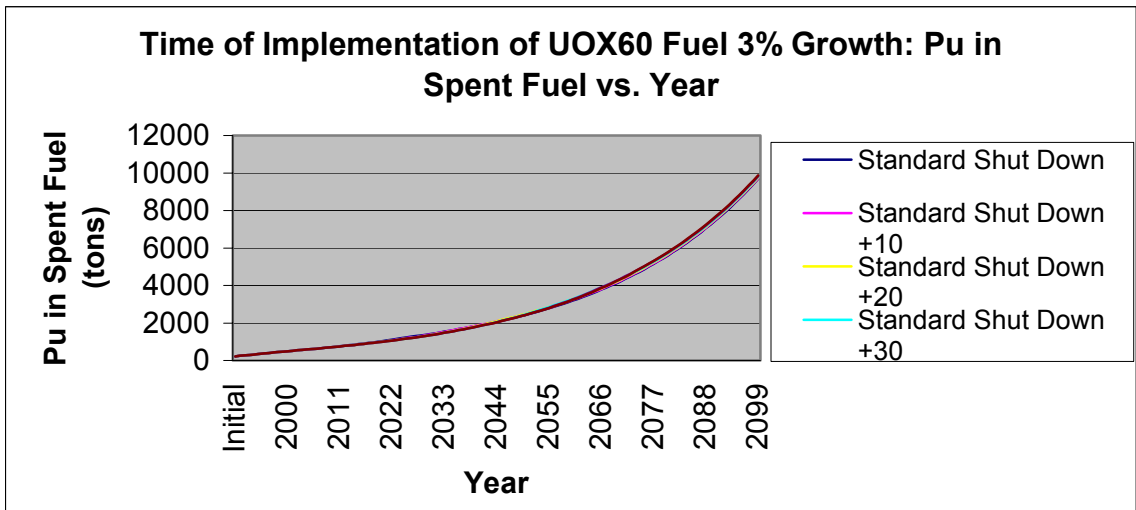
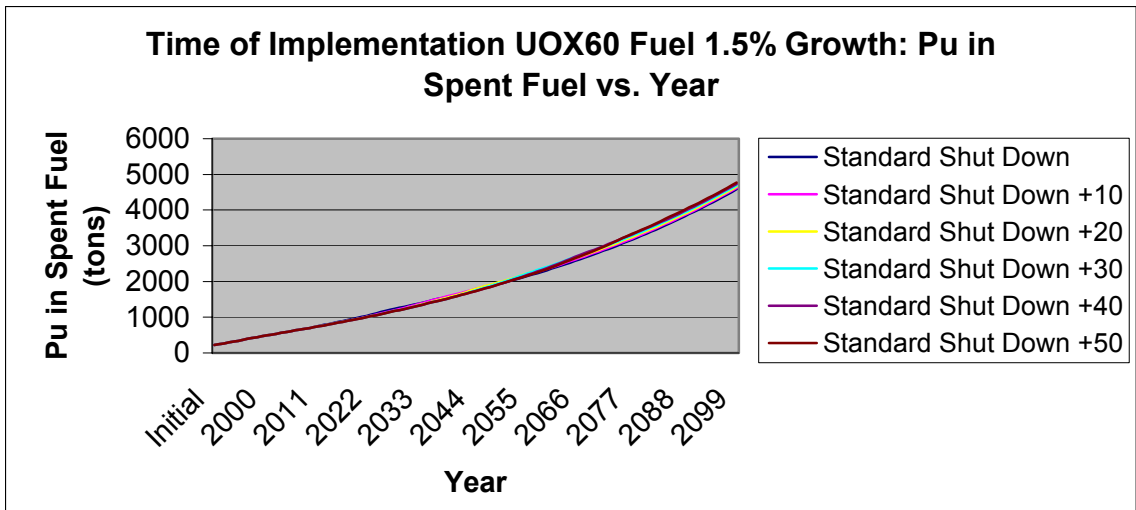
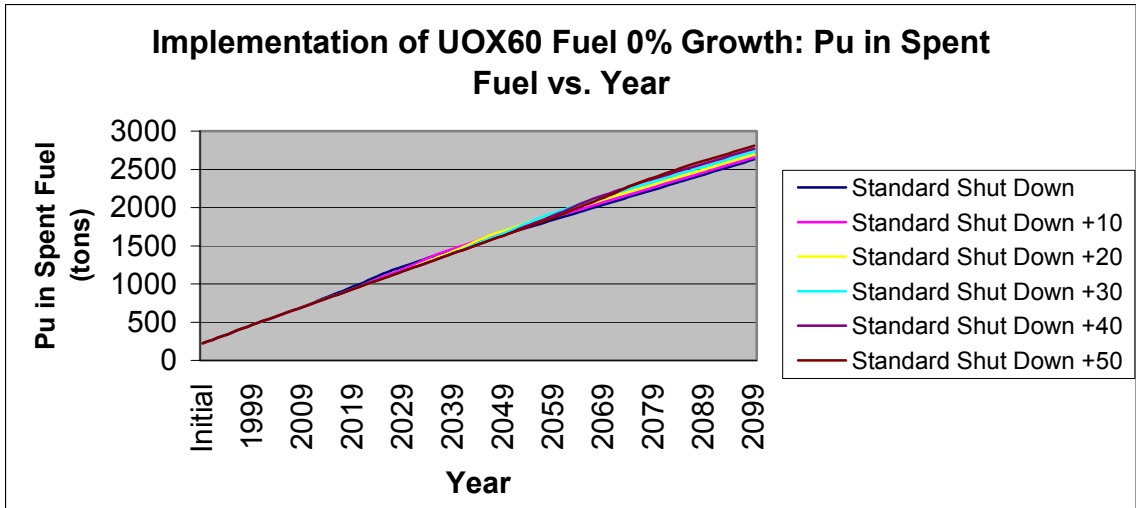


Figure 10 Pu in spent fuel totals for time of implementation utilizing UOX60 for 0%, 1.5%, and 3.0% energy growth

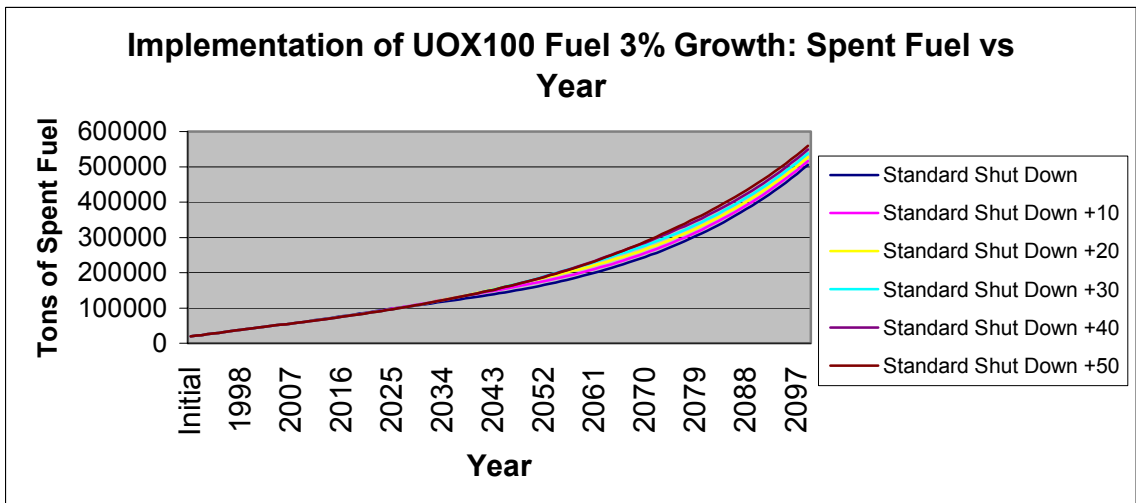
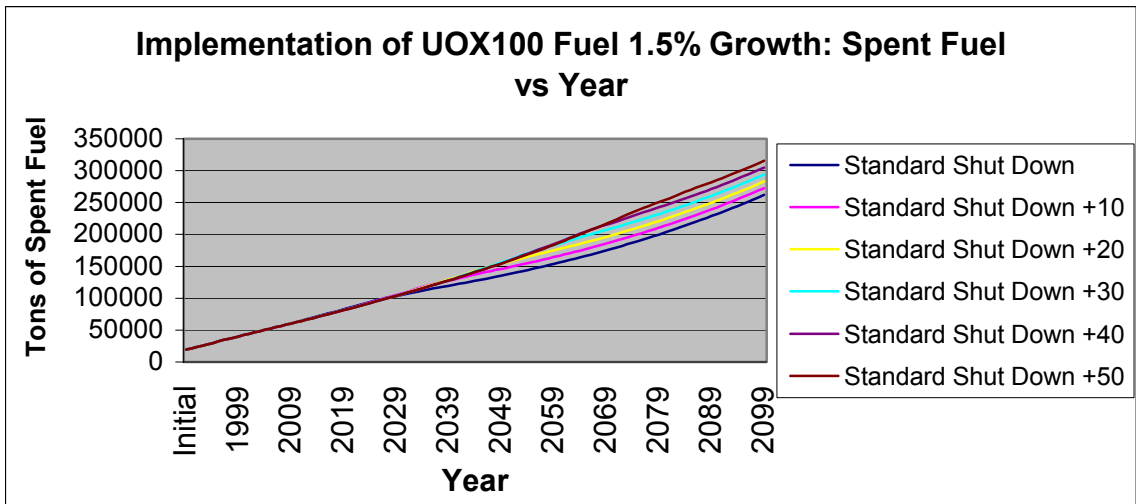
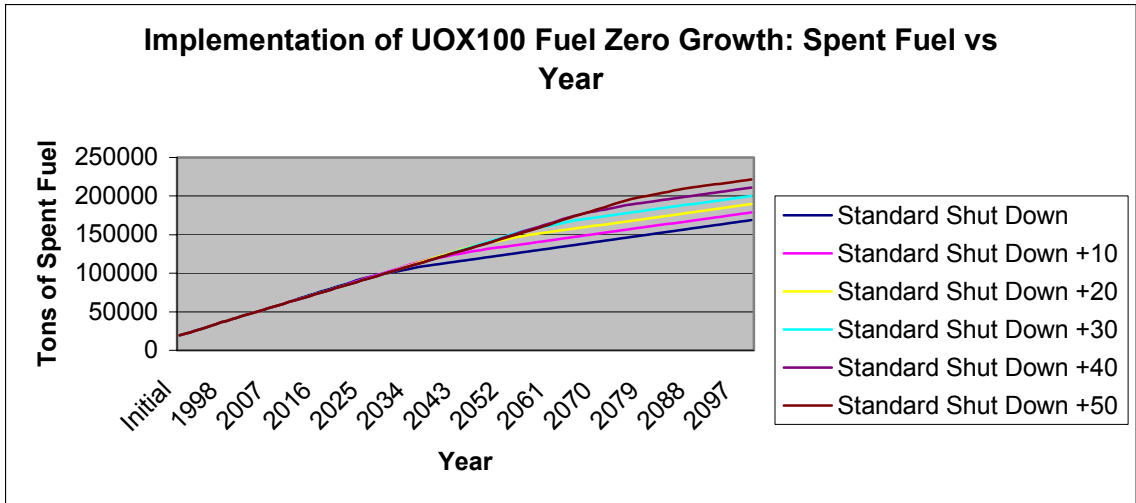


Figure 11 Spent fuel totals for time of implementation utilizing UOX100 for 0%, 1.5%, and 3.0% energy growth

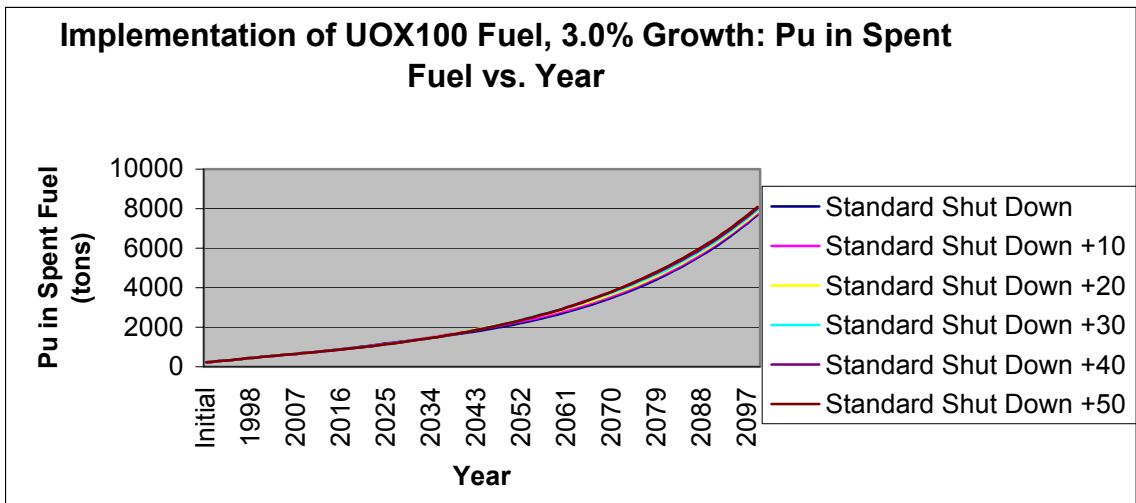
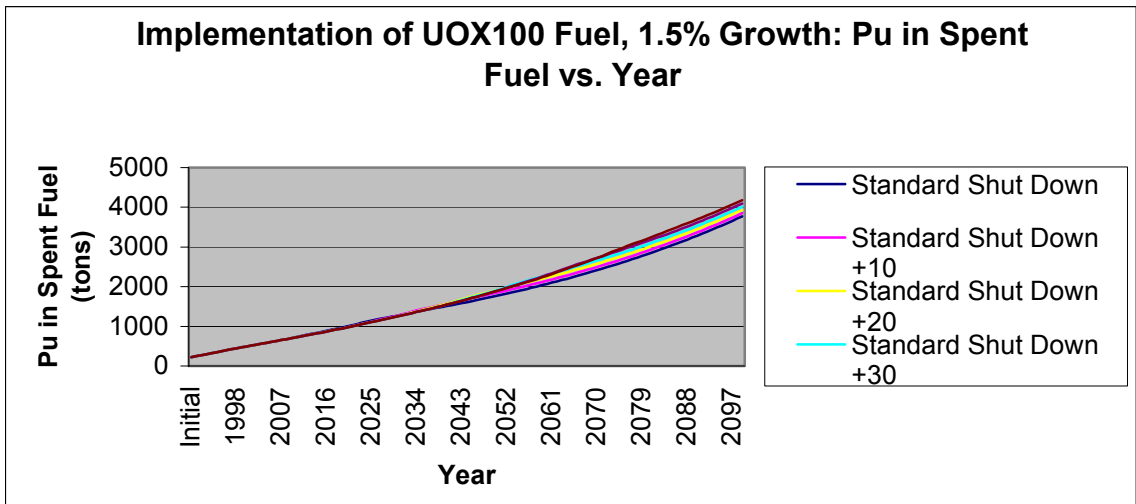
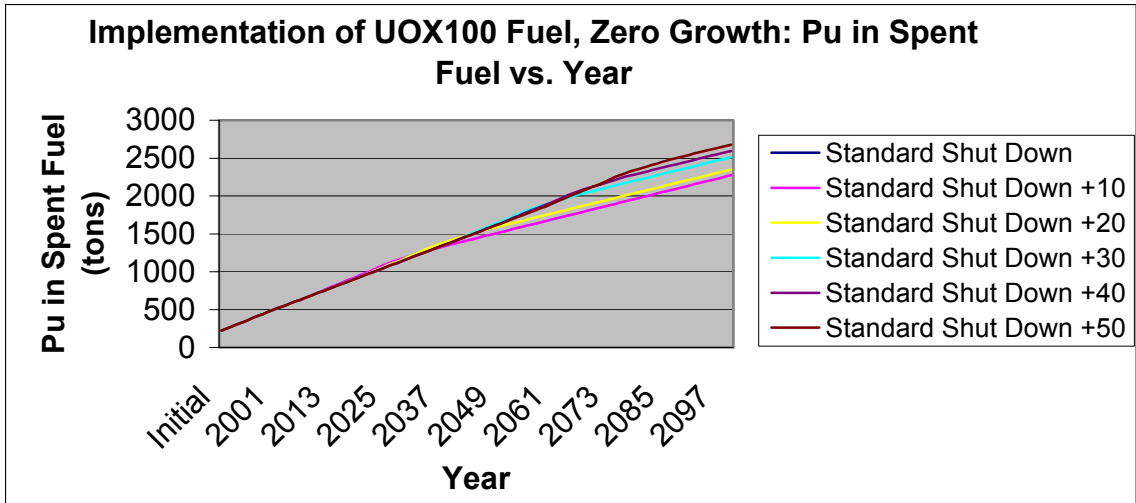


Figure 12 Pu in spent fuel totals for time of implementation utilizing UOX100 for 0%, 1.5% and 3.0% energy growth

53,000 tons heavy metal. In Figure 12 there is roughly an 80 ton increase in Pu per additional 10-year increment. For all three energy scenarios where UOX100 is implemented, there is a maximum spread of Pu in spent fuel in the year 2099 of around 400 tons heavy metal.

The spread of results in the UOX100 case are approximately double those found in the UOX60 scenario. This can be attributed to the difference in fuel burn up. The burn up of the uranium oxide fuel greatly influences the results examined in this report. It is interesting to note that the spread in results at the end of the simulation are roughly the same regardless of the energy growth rate.

3.3.2 MOX50 Scenario Description and Results

The MOX50 case was modeled in a manner similar to the higher burn up scenarios, although there were several different methods and scenarios ran for the MOX50 implementation case. In all cases, the time of MOX fuel implementation was performed as discussed in previous sections. A total of five different MOX fuel implementation scenarios were run. They are as follows:

- 10% Reactor Park PWR-MOX, loaded with 10% MOX core, "User Defined"
- 10% Reactor Park PWR-MOX, loaded with 30% MOX core, "User Defined"
- 26% Reactor Park PWR-MOX, loaded with 30% MOX core, "User Defined"
- 10% Reactor Park PWR-MOX, loaded with 30% MOX core, "Automatic Reprocessing Deployment"
- 26% Reactor Park PWR-MOX, loaded with 30% MOX core, "Automatic Reprocessing Deployment"

"User Defined" and "Automatic Reprocessing Deployment" defined the manner in which DANESS deploys annual reprocessing capacity. DANESS allows the user to choose among three approaches in defining the fuel cycle facility capacity. Two of those three were used for the MOX case. "User Defined" fuel cycle facility deployment scenario allows the user to input the desired fuel cycle facility capacity by graph or table format.

"Automatic Deployment" is a DANESS sub-routine, which will deploy fuel cycle facility capacity according to projected needs.

For the "User Defined" cases a 1500 thm/yr reprocessing capacity began in the year 2000, and stayed constant throughout the duration of the simulation. For the "Automatic Deployment" this reprocessing capacity changes according to projected fuel cycle needs. Values for the "Automatic Deployment" reprocessing capacity will be discussed in their subsequent section.

The results that are presented in this thesis are for one "automatic deployment" and one "user defined". These cases were chosen to display how the different fuel cycle facility functions affect the output. The "automatic deployment" scenarios create more realistic models. The "user defined" cases are limited by the fixed reprocessing capacities. The results from the "user defined" simulations reflect the "hard-wired" nature of the reprocessing deployment, and in turn results in output totals that are not as trustworthy as the "automatic deployment" cases. Another reason the "automatic" model is presented in this thesis is for reasons of comparison. The fast reactor models were also built using "automatic reprocessing deployment", making a comparison of results more effective.

3.3.2.1 10% Reactor Park PWR-MOX, loaded with 30% MOX core, "User Defined"

For this case the new reactor park was set to:

- 30% BWR_n
- 60% PWR_n
- 10% PWR_nM

The BWR_n and PWR_n reactor park retains the same fuel use: 100%, UOX40 and UOX50 respectively. The PWR_nM reactor park will utilize 30% MOX50 and 70%UOX50. The letter "n" that follows the reactor type stands for "new"; thus these are the newly constructed reactors that meet energy demands as existing reactors are retired.

As was discussed in the previous section, the capacity of aqueous reprocessing was set at 1500thm/yr, beginning in the year 2000. This reprocessing capacity remains at this value through out the simulation. The remainder of the fuel cycle facilities, such as fuel fabrication, conversion, and spent fuel storage were set to unlimited in order to not constrain the model. The setup data for this model can be found in Appendix D.

Figure 13 displays the spent fuel totals and Figure 14 displays the Pu in the spent fuel for this case at all energy growths. Figure 13 and Figure 14 display significantly different behaviors than those from the higher burn up cases. In Figure 13 the spent fuel totals converge to roughly the same value by the end of the simulation in the year 2099. This is due to the fact that the mixed oxide fuel has a burn up of 50 GWd/thm; therefore, there is no transition to a higher fuel burn in the new reactors. In Figure 13, the advanced fuel cycle's sensitivity to the initiation of utilizing MOX50 is evident. Roughly between the years 2015 and 2075, the lines diverge from one another. This is attributed to the uncertainty in the introduction of MOX50. This will be discussed in a later section. Figure 14 displays similar behavior to that of Figure 13. The figures show that spent fuel totals and Pu in spent fuel totals are sensitive to the time of implementation of mixed oxide fuel. However, this influence is stronger during the times of transition than in the long-term result.

3.3.2.2 10% Reactor Park PWR-MOX, loaded with 30% MOX core, "Automatic Reprocessing Deployment"

For this case the new reactor park was set to:

- 30% BWRn
- 60% PWRn
- 10% PWRnM

The BWRn and PWRn reactor park retains the same fuel use: 100%, UOX40 and UOX50 respectively. The PWRnM reactor park will utilize 30% MOX50 and 70%UOX50. The difference between this case and the MOX case in the previous section is the use of

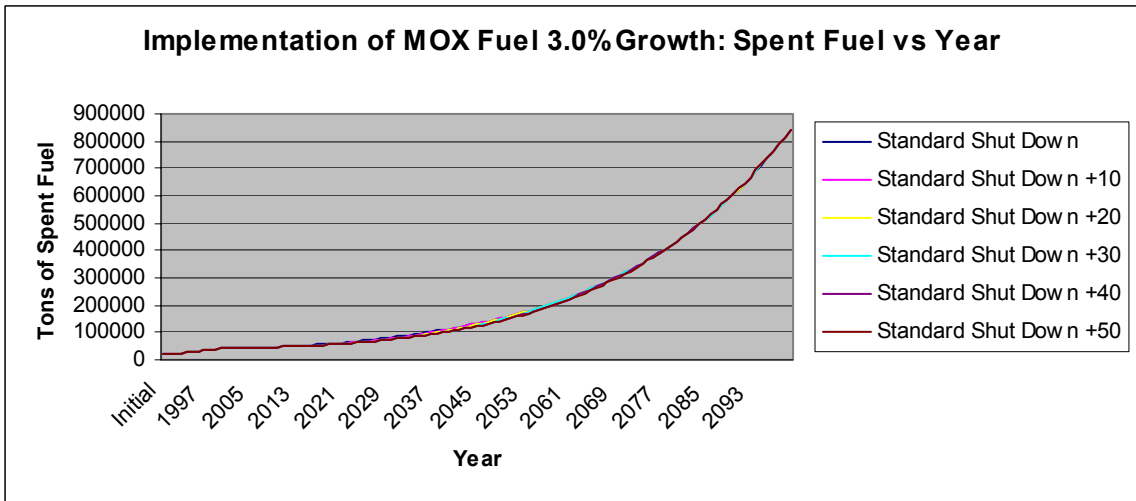
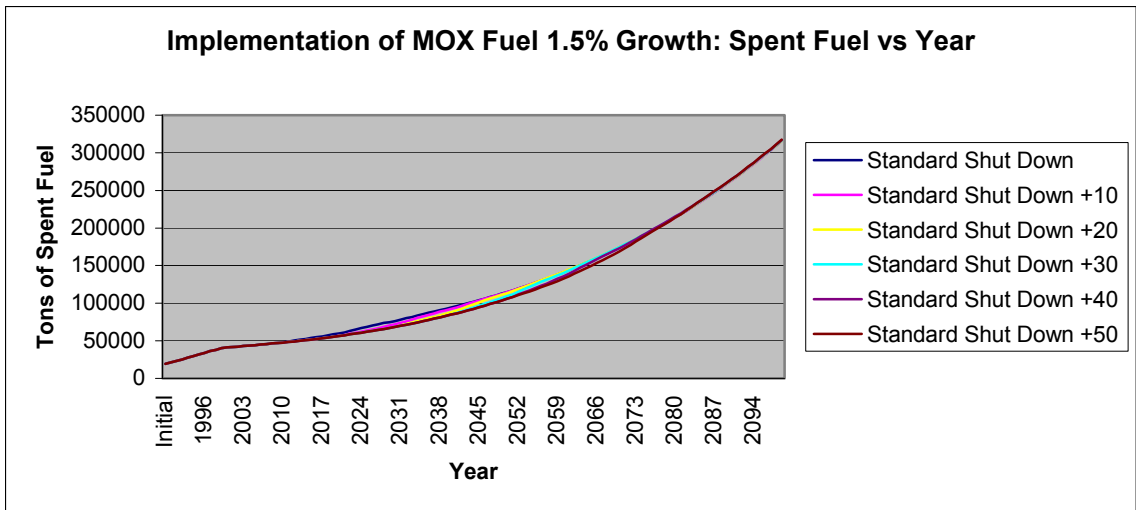
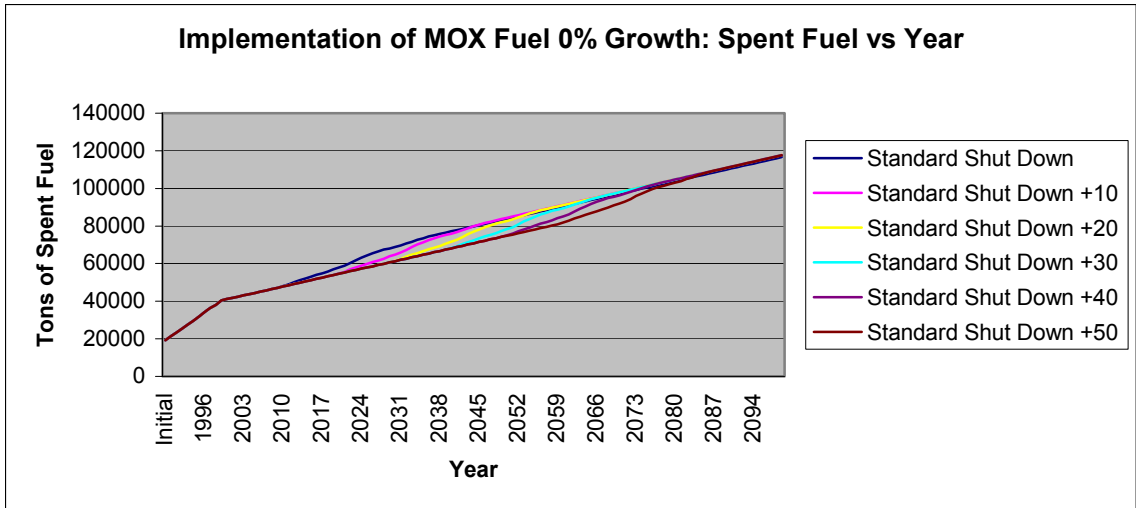


Figure 13 Spent fuel totals for time of implementation utilizing MOX50 for 0%, 1.5% and 3.0% energy growth "User Defined"

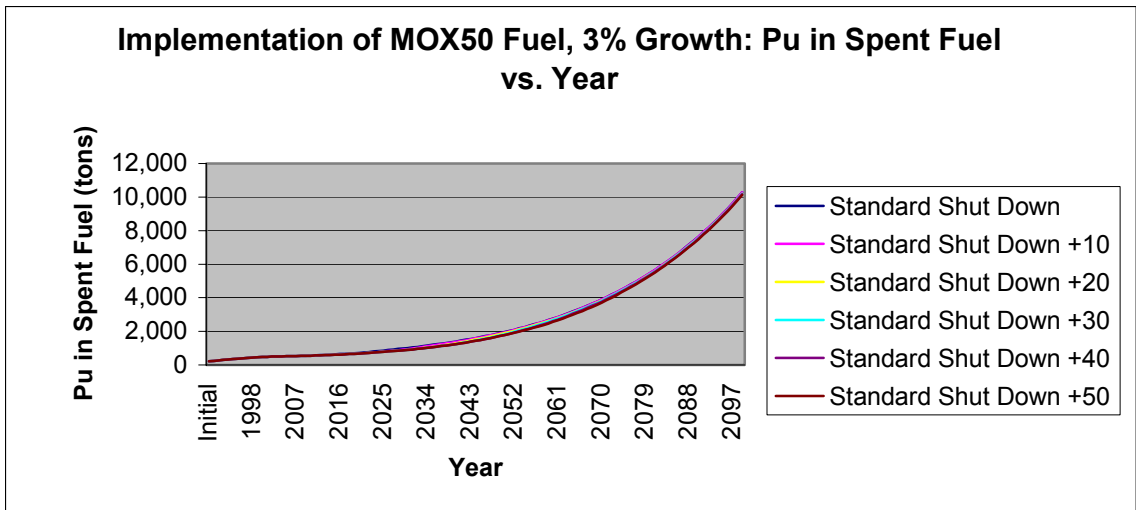
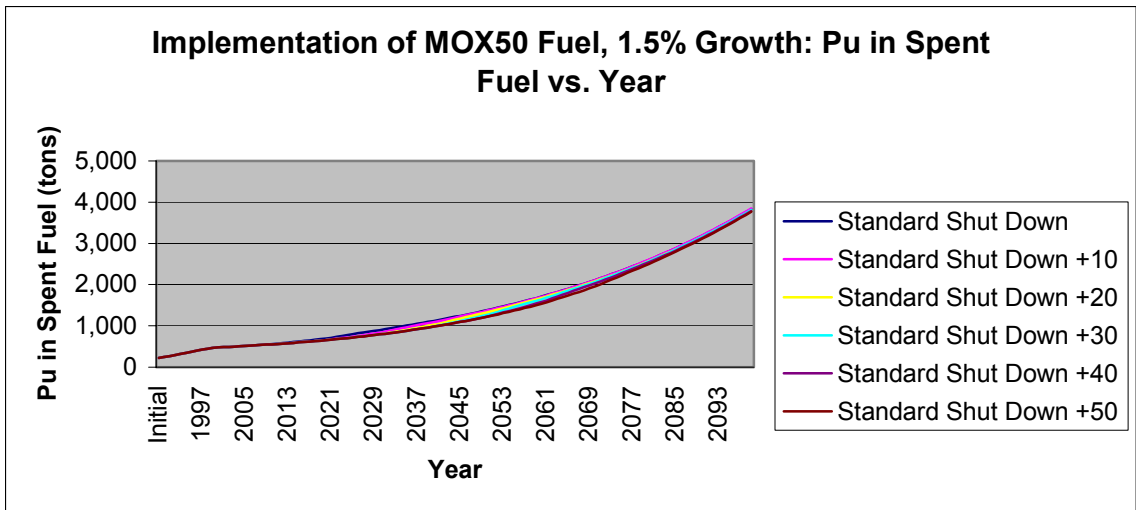
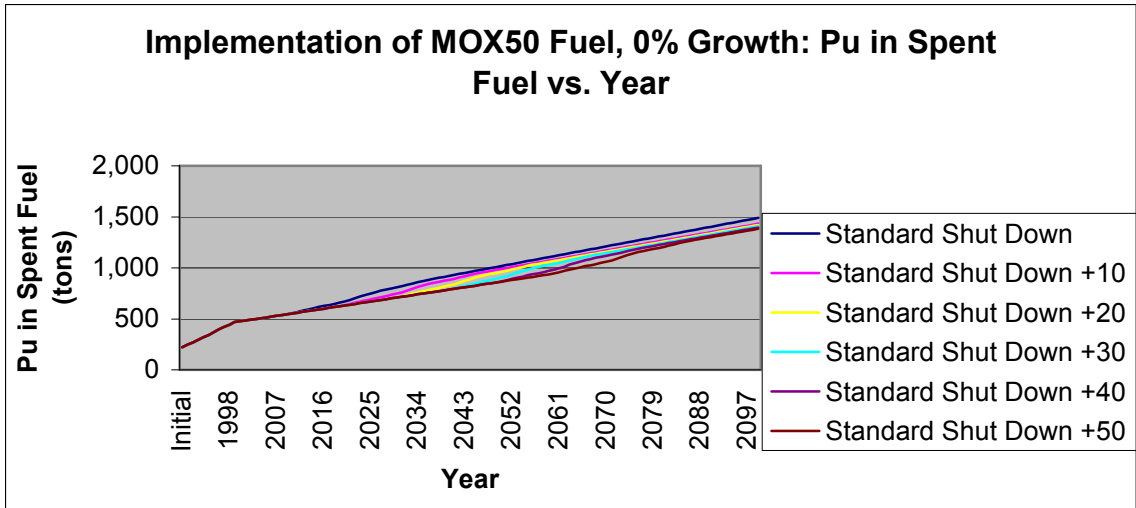


Figure 14 Pu in spent fuel totals for time of implementation utilizing MOX50 for 0%, 1.5%, and 3.0% energy growth "User Defined"

"Automatic Facility Deployment" instead of "User Defined", specifically the deployment of aqueous reprocessing capacity. For the "Automatic" cases there is no set reprocessing capacity, as the 1500 thm/yr, in the "User" scenario. A forecasting sub-model in DANESS determines the reprocessing capacity. The remainder of the fuel cycle facilities (those facilities other than reprocessing) is set to unlimited capacity.

For the "Automatic" MOX case the maximum deployed reprocessing capacity varies. The 0% energy growth has a maximum capacity deployment of around 1400 thm/yr, the 1.5% energy growth has a maximum of around 2,600 thm/yr, and the 3.0% growth case has a maximum capacity of around 5,650 thm/yr. The difference in reprocessing capacity between "User" and "Automatic" is evident in the spent fuel totals. This will be discussed further in a following section of this report. Figure 15 and Figure 16 represent the total spent fuel mass and the amount of Pu in the spent fuel. The setup data for this model can be found in Appendix C.

The results shown in Figures 15 and 16 mimic those of the previous mixed oxide fuel scenario. Again the results converge to roughly the same totals at the end of the simulation, and the largest fluctuation of result values occurs during the time of MOX50 fuel transition periods. The results in Figures 15 and 16 display much smoother lines than the previous mixed oxide fuel case. This is due to the use of the "Automatic Reprocessing" option in DANESS. This option creates a more "realistic" fuel cycle scenario where the reprocessing capacity is based on the demands of the model.

3.3.3 Fast Reactor

The setup and methodology for the fast reactor runs is the same as it was for the high-burn up and MOX fuel cases, as stated before. The fast reactor modeled is a Pu fast burner. The data for this simulation is based on data supplied by Argonne National Laboratory for a plutonium burner reactor with a conversion ratio of 0.5.

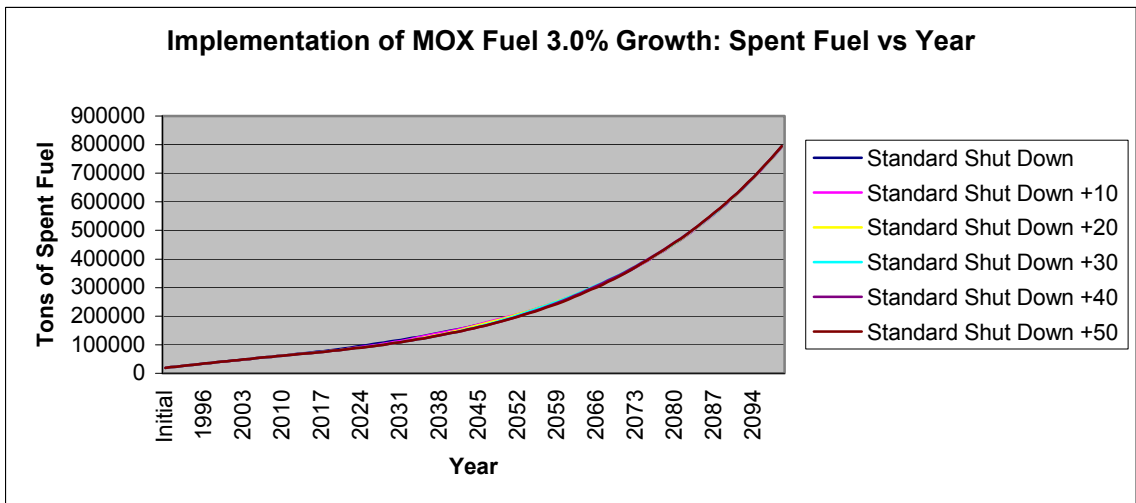
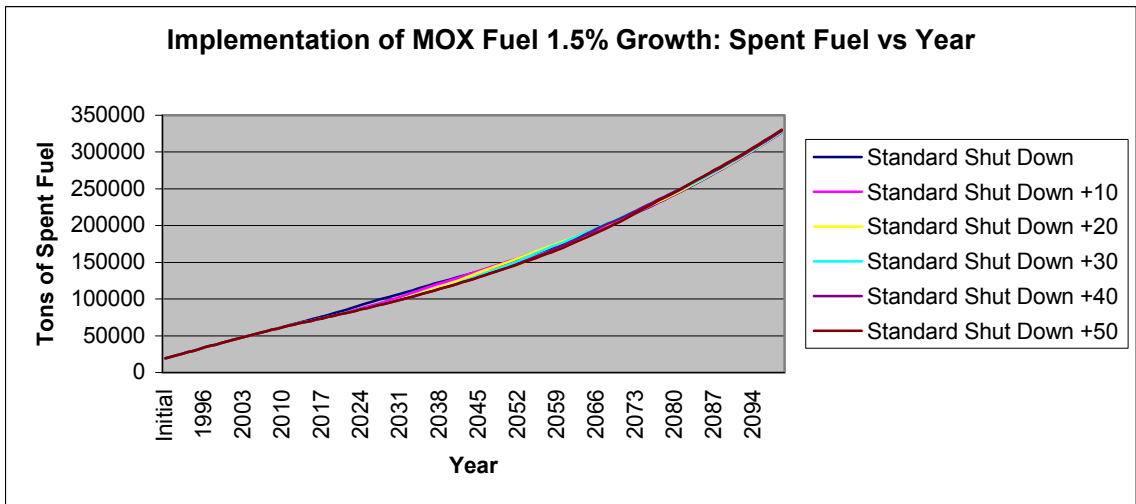
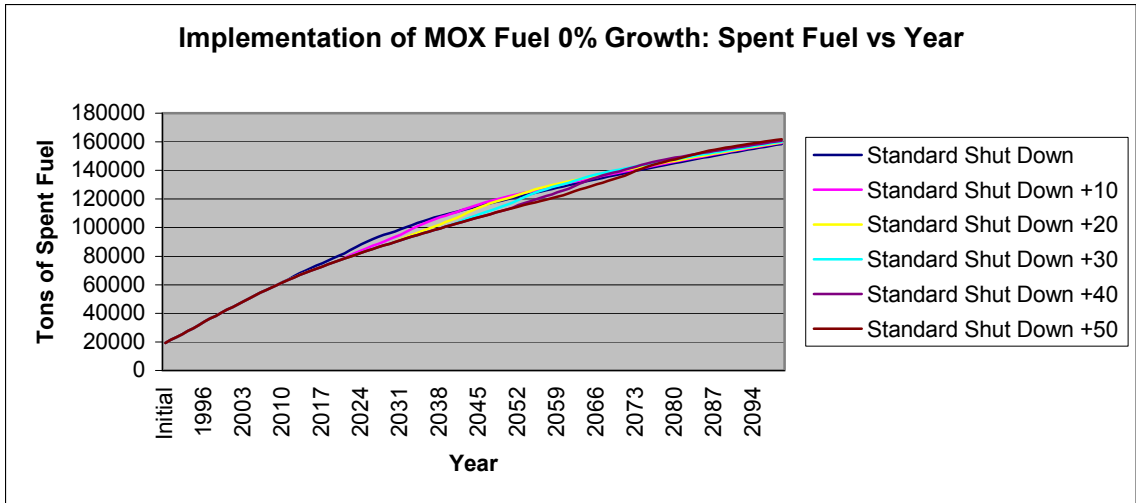


Figure 15 Spent fuel totals for time of implementation utilizing MOX50 for 0%, 1.5% and 3.0% energy growth "Automatic"

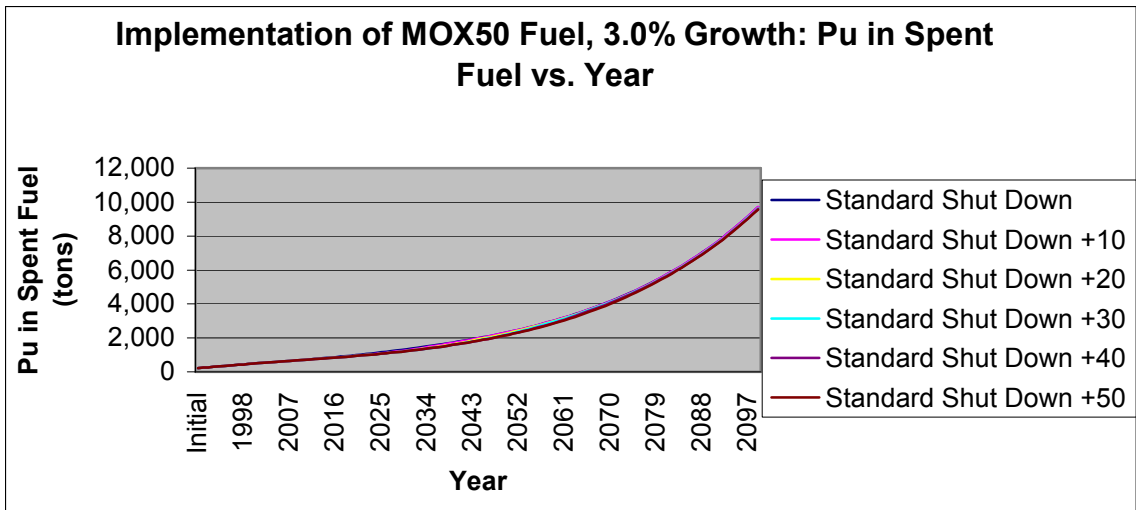
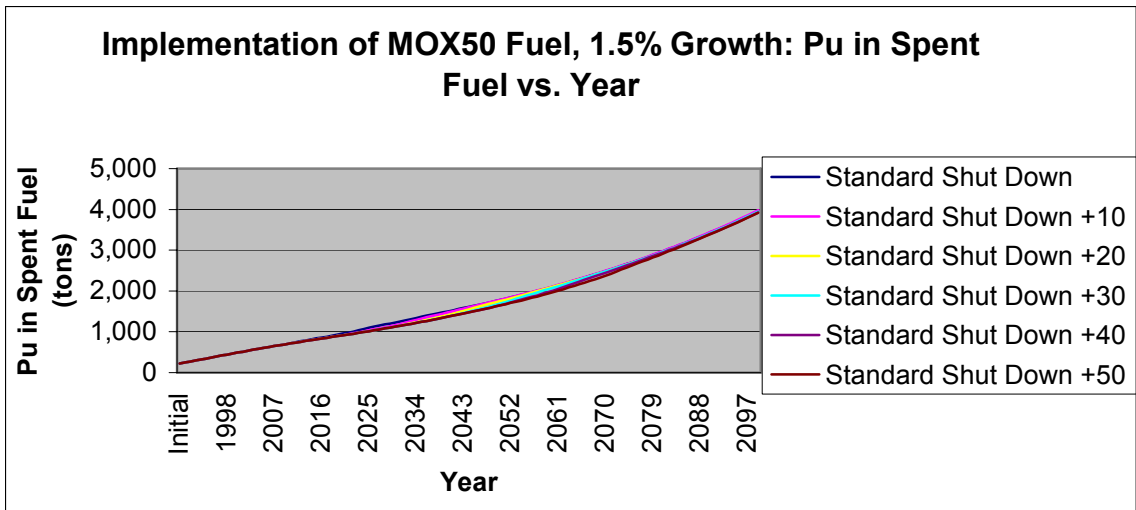
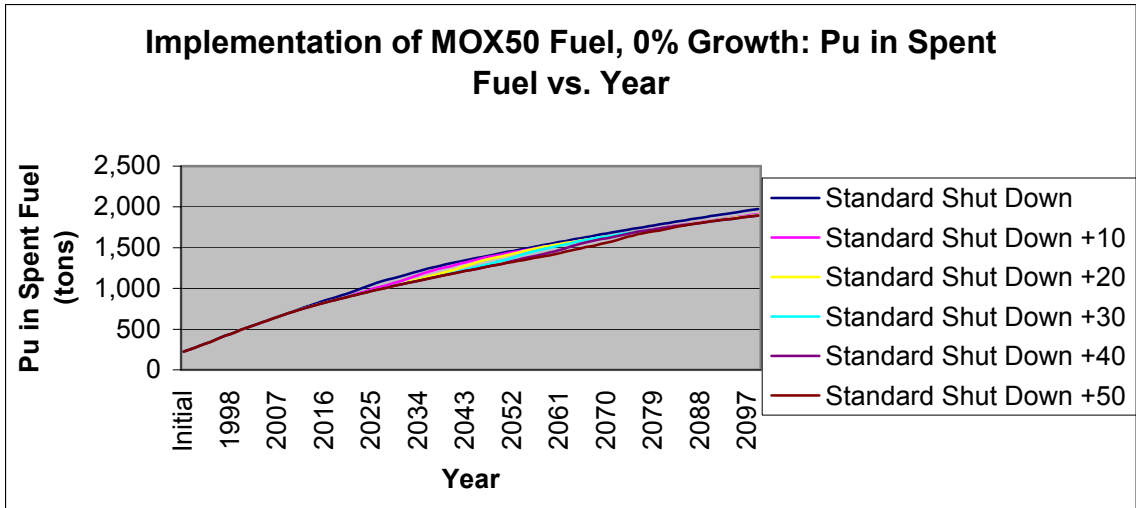


Figure 16 Pu in spent fuel totals for time of implementation utilizing MOX50 for 0%, 1.5% and 3.0% energy growth "Automatic"

For this case the new reactor park was set to:

- 30% BWRn
- 60% PWRn
- 10% Frbu (Fast Reactor burner)

The BWRn and PWRn reactor park retains the same fuel use: 100%, UOX40 and UOX50 respectively. The fast reactor park will utilize 100% fast driver fuel. The data for this reactor and fuel can be found in Appendix E. This scenario utilizes the "automatic deployment" functionality for the aqueous reprocessing facilities, and all other fuel cycle facilities are set to unlimited capacity. For the "Automatic" fast reactor case the maximum deployed reprocessing capacity varies. The 0% energy growth has a maximum capacity deployment of around 1,890 thm/yr, the 1.5% energy growth has a maximum of around 3,500 thm/yr, and the 3.0% growth case has a maximum capacity of around 8,100 thm/yr. These reprocessing capacity values are greater than those of the MOX50 "Automatic" case discussed earlier.

Figure 17 shows that, as in the MOX case, the spent fuel totals have the greatest fluctuation during the period of advanced fuel transition then tend to equal out by the end of the simulation. Regardless of the energy growth rate in Figure 17, the spread in spent fuel totals in the year 2099 is about 2,000 tons. Figure 18 again shows the period of fuel transition to be the time of greatest result fluctuation. Also by the end of the simulation there is no significant difference in the results, regardless of when the fast reactor fuel is implemented.

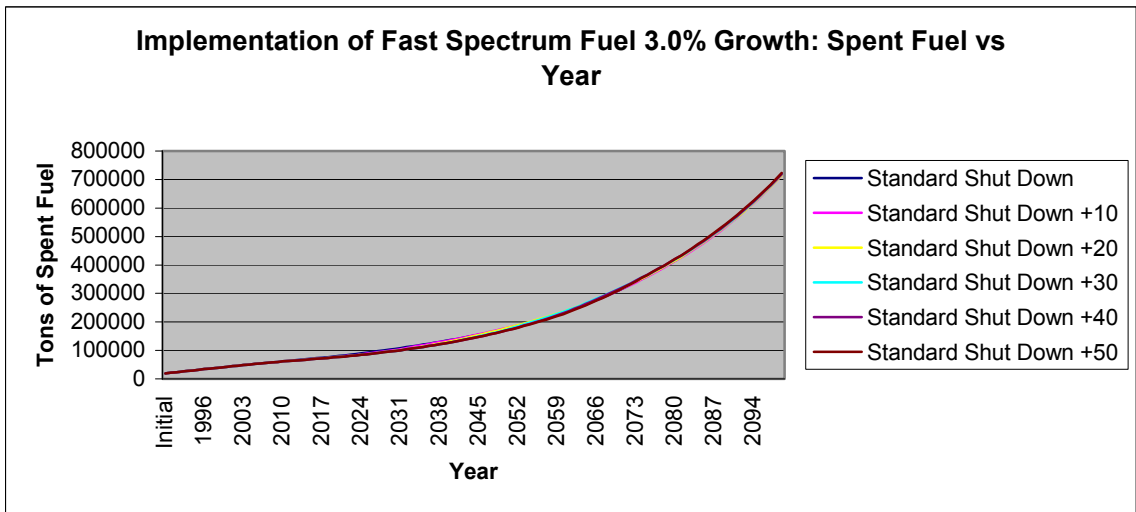
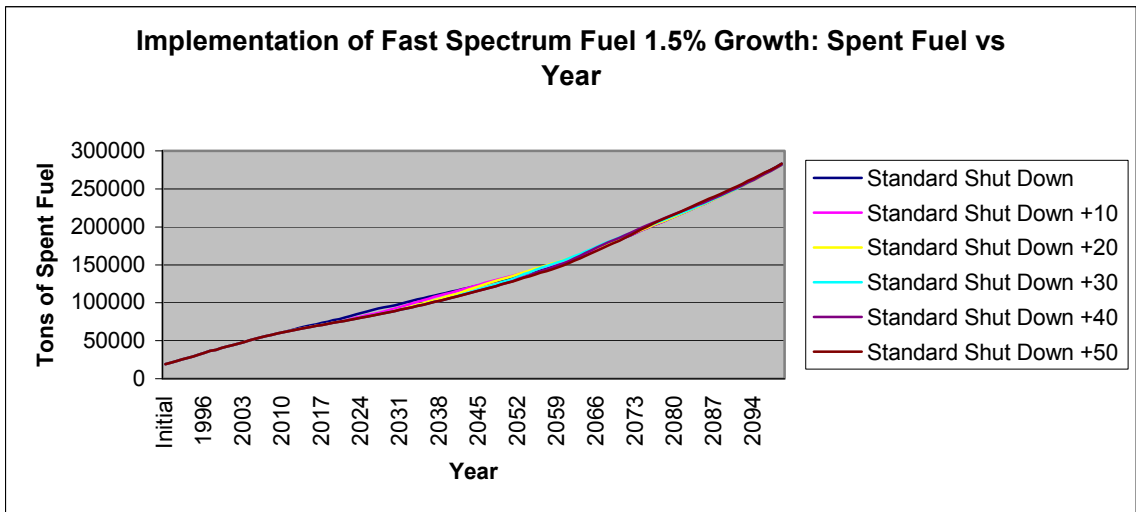
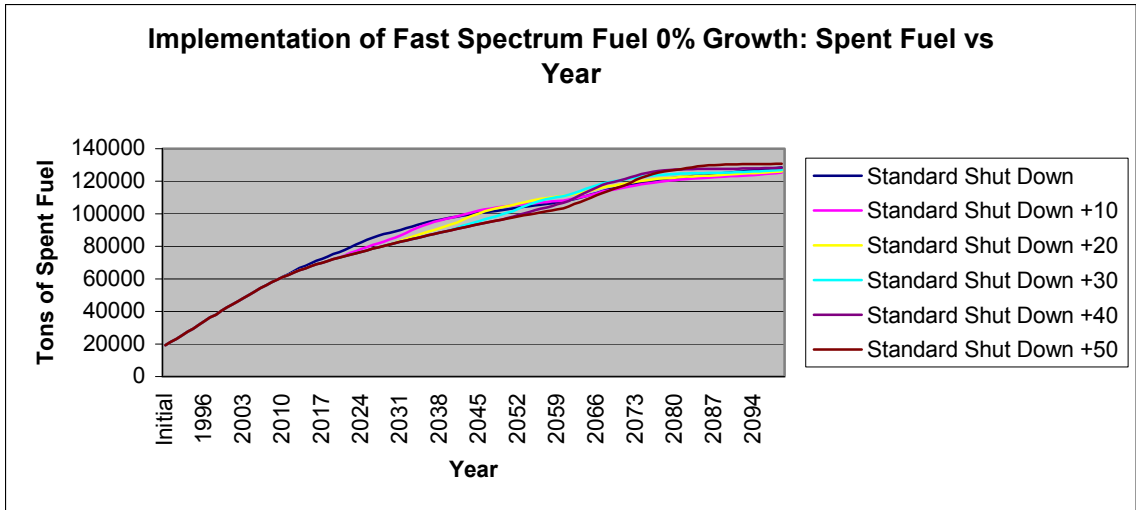


Figure 17 Spent fuel totals for time of implementation utilizing fast burner fuel 0%, 1.5% and 3.0% energy growth

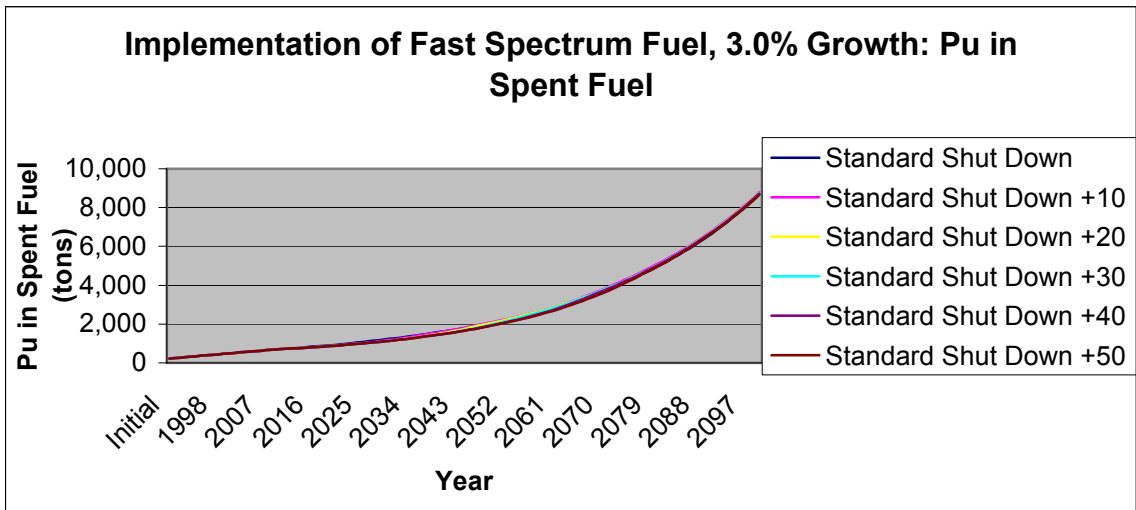
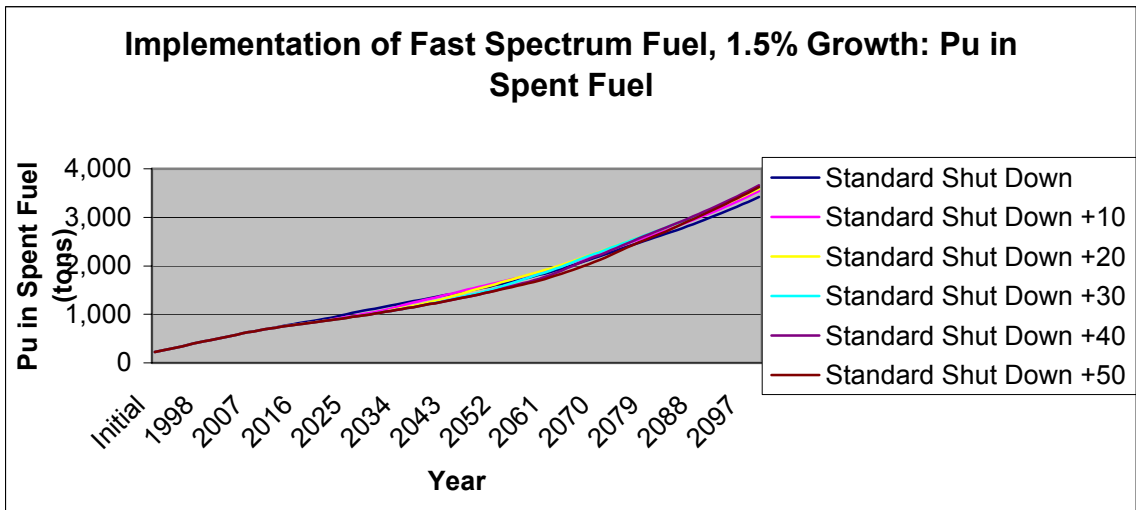
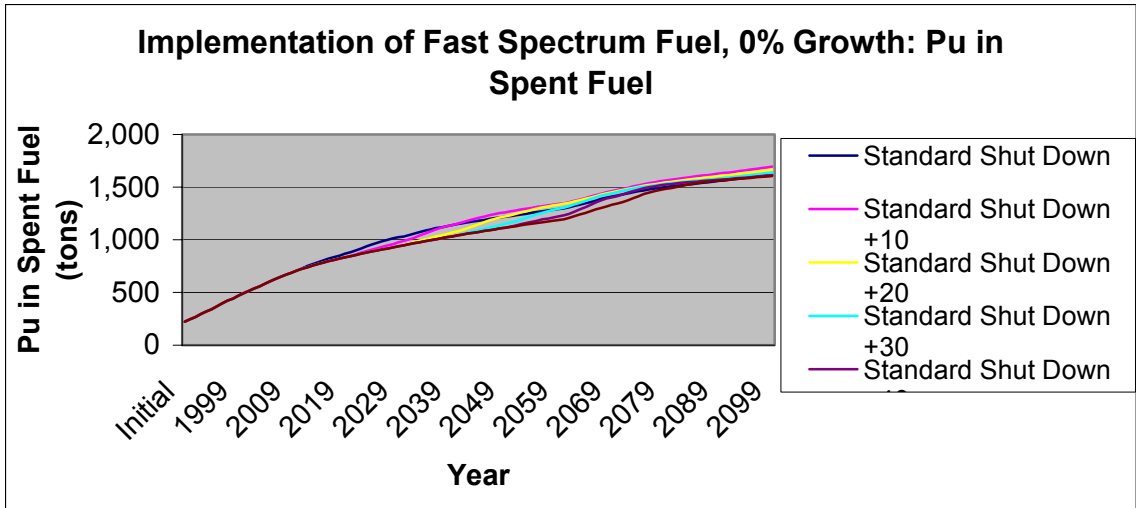


Figure 18 Pu in spent fuel totals for time of implementation utilizing fast burner fuel for 0%, 1.5%, and 3.0% energy growth

CHAPTER 4: Analysis of Results

4.1 Comparison of Results

Every fuel cycle scenario modeled in this thesis reduces the total mass of spent fuel and the amount of Pu in the spent fuel when compared to the once through cycle. Table 1 displays the spent fuel total ranges from the simulation models presented in this report. Table 2 lists the amount of Pu in the spent fuel. The values in Tables 1 and 2 have been normalized by the total amount of energy produced by the reactor park in the simulation, i.e. the total electricity produced over the 110-year simulation. Table 1 is in units of tons of heavy metal per terawatt-electric. Table 2 is in units of tons of Pu in spent fuel per terawatt-electric. The value ranges in the tables are the high and low values at the end of the 100-year simulations. The variance in values is a result of varying the time of implementing the advanced fuels and associated fuel cycle facilities.

In a zero energy growth scenario, the lowest spent fuel totals result in the reprocessing cases. The "User Defined" MOX scenario has the lowest spent fuel total of the 0% growth cases. Here the spent fuel total is dominated by the fact that there is a reprocessing capacity and some fuel mass is being diverted to the reprocessing plants. However the high level waste from reprocessing should be considered in order to gain a full understanding of the total amount of waste that would need to go to geological storage.

In the zero energy growth case the advanced fuel cycles utilizing reprocessing show the lowest spent fuel mass values. As energy demand grows there is a change in the trend of spent fuel mass arising. The mass saving of spent fuel is greater in the higher burn up cases than in the MOX fuel cases. This is especially true with UOX100 fuel. From Table 1 it can be seen that the UOX100 stands out, especially in scenarios of high-energy demand. The UOX60 and MOX cases are comparable in mass of spent fuel generated, and the Pu fast burner scenario lies somewhere between the two.

Table 1 Summary Table of Spent Fuel ranges for different fuel cycles in (thm/TWe)

<i>Energy Growth</i>	Same Trend	UOX60	UOX100	30% MOX Core, 10% Reac Park, "Automatic"	30% MOX Core, 10% Reac Park "User Defined"	Pu Fast Burner (CR=0.5) 10% Reac Park
<i>0%</i>	3.07	2.56-2.82	1.98-2.62	1.87-1.91	1.37-1.38	1.48-1.54
<i>1.5%</i>	2.87	2.29-2.44	1.61-1.94	2.02-2.04	1.94-1.96	1.73-1.75
<i>3.0%</i>	2.68	2.12-2.19	1.38-1.53	2.16-2.17	2.29-2.30	1.95-1.97

Table 2 Summary Table of Pu in spent Fuel ranges for different fuel cycles in (tons Pu /TWe)

<i>Energy Growth</i>	Same Trend	UOX60	UOX100	30% MOX Core, 10% Reac Park, "Automatic"	30% MOX Core, 10% Reac Park "User Defined"	Pu Fast Burner (CR=0.5) 10% Reac Park
<i>0%</i>	0.036	0.031-0.033	0.027-0.032	0.022-0.023	0.016-0.018	0.019-0.020
<i>1.5%</i>	0.033	0.028-0.029	0.023-0.026	0.024-0.025	0.022-0.023	0.022-0.023
<i>3.0%</i>	0.031	0.026-0.027	0.021-0.022	0.026-0.0265	0.027-0.028	0.023-0.024

The amount of Pu in the spent fuel, which can be found in Table 2, follow similar trends as those seen in Table 1. Again in the zero energy growth cases, the lowest values are found in the reprocessing scenarios, with the Pu fast burner being the lowest. As energy demand increases the values in Tables 1 and 2 for the non-reprocessing cases (UOX60 and UOX100) decrease, while the opposite trend is event in those scenarios that utilize fuel reprocessing.

The values listed in Tables 1 and 2 provide useful information concerning the amount of heavy metal generated outside the reprocessing loop by the advanced fuel cycles simulated in this thesis. This information is useful when considering the mass of heavy metal being sent to the repository. However, as stated before when comparing scenarios that utilize reprocessing against those that do not, it is important to consider mass flows that are diverted in the reprocessing cycle. Accounting for these considerations yields Tables 3 and 4. Tables 3 and 4 have been normalized by the total amount of energy produced by the reactor park in the simulation.

Table 3 accounts for the mass of separated uranium found in the reprocessing loop of the fuel cycle. Thus while Table 1 accounts only for the heavy metal mass accruing from used fuel destined for the repository, Table 3 incorporates the mass of uranium associated with the reprocessing cycle. This yields significantly different values from Table 1.

The same trend, UOX60 and UOX100 values from Table 1 to Table 3 remain the same due to the fact there is no reprocessing in the cycle. The interesting changes occur in those scenarios that incorporate a reprocessing capacity. The two MOX cases and the fast burner reactor values all increase significantly when the entire amount of heavy metal circulating in the fuel cycle is accounted for; the values between reprocessing and non-reprocessing scenarios in table 3 are not as comparable as those in Table 1. In fact, in the 3.0% energy growth case the values of the three reprocessing scenarios are almost double that of the UOX100 value. This can attributed to the importance of high fuel burn up has on spent heavy metal mass.

Table 3 Summary of Spent Fuel and Separated Uranium ranges for different fuel cycles in (thm/TWe)

<i>Energy Growth</i>	Same Trend	UOX60	UOX100	30% MOX Core, 10% Reac Park, "Automatic"	30% MOX Core, 10% Reac Park "User Defined"	Pu Fast Burner (CR=0.5) 10% Reac Park
<i>0%</i>	3.07	2.56-2.82	1.98-2.62	2.88-2.90	2.85-2.86	2.80-2.84
<i>1.5%</i>	2.87	2.29-2.44	1.61-1.94	2.71-2.72	2.72-2.73	2.67-2.69
<i>3.0%</i>	2.68	2.12-2.19	1.38-1.53	2.61-2.62	2.63-2.64	2.58-2.60

Table 4 Summary Table of Pu Out-of-Pile ranges for different fuel cycles in (tons Pu/TWe)

<i>Energy Growth</i>	Same Trend	UOX60	UOX100	30% MOX Core, 10% Reac Park, "Automatic"	30% MOX Core, 10% Reac Park "User Defined"	Pu Fast Burner (CR=0.5) 10% Reac Park
<i>0%</i>	0.036	0.031-0.033	0.027-0.032	0.032-0.034	0.032-0.033	0.030-0.032
<i>1.5%</i>	0.033	0.028-0.029	0.023-0.026	0.030-0.031	0.031-0.032	0.028-0.031
<i>3.0%</i>	0.031	0.026-0.027	0.021-0.022	0.028-0.029	0.028-0.029	0.028-0.029

Comparison of results in Tables 2 and 4 show similar trends. Table 4 accounts for all the Pu out-of-pile, which consists of the Pu in the spent fuel plus the Pu diverted into the reprocessing loop. Table 4 now shows that the UOX100 case contains the lowest amount of Pu in the fuel cycle for all energy growths. Also the three reprocessing case values are now comparable to the UOX60 case, whereas in Table 2 they are lower.

It is significant to note that all the advanced fuel scenario values in Tables 1-4 are less than that of the Same Trend. Thus the mass-flow of heavy metal, which is primarily uranium, and the plutonium specific mass-flow of the advanced fuel cycles is less than that of the projected current industry trends of the once through cycle.

4.2 Uncertainty of Spent Fuel Totals

The uncertainties generated in this report are solely for the spent fuel totals. The analysis is not performed for the Pu in spent fuel totals. The uncertainty in spent fuel totals is generated in the following manner. The absolute difference between the standard shut down simulation and the standard shut down +50 simulation values is calculated for each year of the simulation. That is to say, the difference in spent fuel values between the model of earliest advanced fuel implementation and the most delayed advanced fuel implementation model is calculated. The average and standard deviation of these values is then calculated. The values used in these calculations are taken beginning in the year 2010 and end 2099. The year 2010 is significant because it is the year energy demand begins; hence it is the first year a discrepancy between simulation values arises.

Table 5 displays the uncertainty values and standard deviations calculated for each scenario presented in this thesis. The values in Table 5 represent the uncertainty associated with the spent fuel total for any given year of the simulation, beginning in the year 2010. The high burn up cases have the greatest uncertainty and standard deviations. The UOX100 case has an uncertainty over double that of the UOX60 case and roughly five times the uncertainties generated from the MOX50 and fast reactor cases.

Table 5 Uncertainties and Standard Deviations (thm)

Energy Growth	UOX60		UOX100		30% MOX Core, 10% Reac Park, "Automatic"		30% MOX Core, 10% Reac Park "User Defined"		Pu Fast Burner (CR=0.5)10% Reac Park	
		Uncertainty	Stan. Dev	Uncertainty	Stan. Dev	Uncertainty	Stan. Dev	Uncertainty	Stan. Dev	Uncertainty
0%	9,566	8,239	26,147	20,853	4,847	2,507	5,024	3,467	4,602	2,227
1.50%	9,902	8,486	26,244	20,976	5,007	2,580	5,175	3,542	4,161	2,626
3.00%	10,082	8,672	26,390	21,067	4,819	2,647	5,114	3,509	4,789	2,343

The fast reactor case has the lowest uncertainty associated with it; however this value is only slightly less than the MOX50 scenarios. The burn up of the implemented advanced fuel has the highest correlation with the model uncertainty.

CHAPTER 5: Conclusions and Future Work

5.1 Conclusions

Conclusions from the work performed in this thesis may be divided into two primary categories. The first category consists of those conclusions drawn from the sensitivity analyses. Second, conclusions are drawn based on the results supplied by the advanced fuel cycles modeled in this report. An important aspect of this research project is the ability to draw conclusive decisions from the data provided by this research. The "User Defined" MOX50 case is not considered in the conclusions because it unrealistically models the reprocessing cycle by beginning reprocessing in the year 2000 and by not adapting the capacity to meet reactor needs.

There were four varied parameters that constitute the scenario analysis of this report. Those parameters are:

1. Time of advanced fuel implementation: this was accomplished by varying the lifetime of the existing reactors.
2. Energy growth
3. Fuel Burn up
4. Reprocessing introduction and capacity

The effect of when the advanced fuels are introduced into the reactor park is an important parameter when analyzing these advanced fuel cycles. The time of implementing the advanced fuel has the greatest impact in the 0% energy growth cases and its impact declines as energy growth increases. The greatest discrepancy in output values is in a 0% energy growth rate case. The sensitivity to time of advanced fuel implementation is greatest in scenarios with no reprocessing. In those cases in which reprocessing is utilized, the results are highly sensitive during the fuel transition periods, but converge to similar values as the simulation runs.

The output is also highly sensitive to the energy growth of the industry. In cases with no reprocessing the normalized values of heavy metal or Pu mass decrease with energy growth, i.e. less heavy metal is produced per terawatt as energy demand increases. Those scenarios with reprocessing capacity follow a similar trend to once-through cycles, however their sensitivity to energy growth is less.

The preliminary results of the simulations show that the output is most sensitive to the burn up of the fuel. Scenarios that move to high burn up fuels (UOX60 and UOX100) have a larger spread of values.

In a zero energy growth case, the Pu fast burner reactor saves the most in terms of spent fuel destined for the repository as well as the amount of Pu in the fuel, when compared to the current once through cycle. Those savings are 50% to 52% in terms of spent fuel mass and 44% to 47% in mass of Pu in the spent fuel.

As the energy demand grows the UOX100 case begins to dominate in savings when compared to the current once through cycle. For 1.5% energy growth, the UOX100 scenario saves 32% to 44% in terms of spent fuel mass and 21% to 30% of the Pu in spent fuel. In the 3.0% energy growth case, the UOX100 scenario saves between 43% and 49% in spent fuel mass and between 29% and 32% in Pu in the spent fuel.

Reprocessing capacity and introduction date are also highly influential parameters in terms of spent heavy metal masses. Fuel cycles become much more complex with the addition of reprocessing capacity. The analyses of these fuel cycles require the comparison of several additional variables. Given an equal share in the new reactor order fraction (both 10%), the fast reactor case requires greater reprocessing capacity than the MOX case. The fast reactor scenario requires an extra capacity of about 35% in the 0% energy growth case, 35% in the 1.5% case, and 43% in the 3.0% energy growth rate case.

Of the advanced fuel cycles model in this thesis, the two that reduced spent fuel and Pu most significantly are the UOX100 high burn up scenario and the Pu fast burner case. The UOX100 stands out in terms of savings in mass of spent fuel as well as the amount of Pu contained in that fuel. However concerns with enrichment percent, pellet-cladding interaction, and a plethora of other technical challenges beg the question of whether such a high fuel burn is feasible. The Pu fast burner reactor also presents a viable option in terms of mass savings destined for the repository, however the introduction of a reprocessing cycle and infrastructure needs pose a downside to this advanced fuel cycle as well.

5.2 *Future Work*

The work presented in this thesis is primarily the preliminary work of this research project. The pith of this research project will be accomplished in the next two years. The advanced fuel cycles have been chosen, and initial runs have been accomplished as seen in this report. Also key sensitivity parameters relevant to these advanced nuclear fuel cycles have been identified; the output variables of interest have been chosen via dialogue between the University of Tennessee and Argonne National Laboratory.

The research work is now moving into the next phase of the project. This next step is performing in depth uncertainty analysis of the identified variables. Additional literature research is required to provide data for reasonable input uncertainty distributions. Also the development of combinations of advanced cycles such as doubled tiered reprocessing or mixed oxide reactors in combination with fast reactors.

A key output variable of interest that has not been included in this research is the cumulative heat load destined for the repository. This key parameter has not been included in this report, due to the continuing development of DANESS and the ongoing ORIGEN work being performed by another graduate student working on the project. Heat load calculations have always been in DANESS, however they were not based on

extensive isotopic compositions specific to fuel burn up. Also the project researchers at the University of Tennessee desire a heat calculation normalized by GWd. Therefore, the program developers from ANL have been collaborating with the university staff to customize the code as desired. Other graduate students working with ORIGEN, as part of this research, have been calculating burn-up dependent isotopic compositions of spent fuel. These calculations will be incorporated into DANESS, creating a burn-up specific library that tracks some 60 isotopes. This will provide much more detailed heat load values than what is previously available.

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Appendices

APPENDIX A: UOX60

Edem Scenario

Rest of World

Edem Scenario	Edem Exponential
Initial Edem (Twe/yr)	770
Start Year of Edem Growth	2010
Edem annual growth rate (%yr)	0, 1.5 or 3.0

Not Used

Reactor Data

Reactorsdata[PWRe,Pe] MWe	950
Reactorsdata[PWRe, Effth] %	34
Reactorsdata[PWRe fLoad] %	90
Reactorsdata[PWRe, Fuel Use Flag] 0/1	0
Reactorsdata[PWRe, Reactor Group] 1...5	1
Reactorsdata[PWRe, Licensing Time] year	2
Reactorsdata[PWRe, Construction Time] year	4
Reactorsdata[PWRe, Lifetime] years	60
Reactorsdata[PWRe, Construction Cost] B\$/unit	1
Reactorsdata[PWRe, Other Capital cost] B\$/unit	0
Reactorsdata[PWRe, DecomCost] B\$/unit	0
Reactorsdata[PWRe, Contingencies] B\$/unit	0
Reactorsdata[PWRe, OMCost] B\$/unit	0
Reactorsdata[PWRe, InitialTRL] 1...9	9

Reactorsdata[BWRe,Pe] MWe	900
Reactorsdata[BWRe, Effth] %	34
Reactorsdata[BWRe fLoad] %	90
Reactorsdata[BWRe, Fuel Use Flag] 0/1	0
Reactorsdata[BWRe, Reactor Group] 1...5	1
Reactorsdata[BWRe, Licensing Time] year	2
Reactorsdata[BWRe, Construction Time] year	4
Reactorsdata[BWRe, Lifetime] years	60
Reactorsdata[BWRe, Construction Cost] B\$/unit	1
Reactorsdata[BWRe, Other Capital cost] B\$/unit	0
Reactorsdata[BWRe, DecomCost] B\$/unit	0
Reactorsdata[BWRe, Contingencies] B\$/unit	0
Reactorsdata[BWRe, OMCost] B\$/unit	0
Reactorsdata[BWRe, InitialTRL] 1...9	9
Reactorsdata[PWReM,Pe] MWe	950
Reactorsdata[PWReM, Effth] %	34
Reactorsdata[PWReM fLoad] %	90
Reactorsdata[PWReM, Fuel Use Flag] 0/1	0
Reactorsdata[PWReM, Reactor Group] 1...5	1
Reactorsdata[PWReM, Licensing Time] year	2
Reactorsdata[PWReM, Construction Time] year	4
Reactorsdata[PWReM, Lifetime] years	60
Reactorsdata[PWReM, Construction Cost] B\$/unit	1
Reactorsdata[PWReM, Other Capital cost] B\$/unit	0
Reactorsdata[PWReM, DecomCost] B\$/unit	0
Reactorsdata[PWReM, Contingencies] B\$/unit	0
Reactorsdata[PWReM, OMCost] B\$/unit	15
Reactorsdata[PWReM, InitialTRL] 1...9	9
Reactorsdata[BWRn,Pe] MWe	910
Reactorsdata[BWRn, Effth] %	34
Reactorsdata[BWRn fLoad] %	90
Reactorsdata[BWRn, Fuel Use Flag] 0/1	0
Reactorsdata[BWRn, Reactor Group] 1...5	3
Reactorsdata[BWRn, Licensing Time] year	2
Reactorsdata[BWRn, Construction Time] year	3
Reactorsdata[BWRn, Lifetime] years	100
Reactorsdata[BWRn, Construction Cost] B\$/unit	1
Reactorsdata[BWRn, Other Capital cost] B\$/unit	0
Reactorsdata[BWRn, DecomCost] B\$/unit	0
Reactorsdata[BWRn, Contingencies] B\$/unit	0
Reactorsdata[BWRn, OMCost] B\$/unit	15
Reactorsdata[BWRn, InitialTRL] 1...9	9

Reactorsdata[PWRn,Pe] MWe	950
Reactorsdata[PWRn, Effth] %	34
Reactorsdata[PWRn fLoad] %	90
Reactorsdata[PWRn, Fuel Use Flag] 0/1	0
Reactorsdata[PWRn, Reactor Group] 1...5	3
Reactorsdata[PWRn, Licensing Time] year	2
Reactorsdata[PWRn, Construction Time] year	4
Reactorsdata[PWRn, Lifetime] years	60
Reactorsdata[PWRn, Construction Cost] B\$/unit	1
Reactorsdata[PWRn, Other Capital cost] B\$/unit	0
Reactorsdata[PWRn, DecomCost] B\$/unit	0
Reactorsdata[PWRn, Contingencies] B\$/unit	0
Reactorsdata[PWRn, OMCost] B\$/unit	0
Reactorsdata[PWRn, InitialTRL] 1...9	9

Fuel Data

Fueldata[UOX40, BU] GWd/tHM	40
Fueldata[UOX40,CycleLength] months	12
Fueldata[UOX40,NumberBatches] 2..50	4
Fueldata[UOX40,Initial U] t/tHM	1
Fueldata[UOX40,Initial REPU] t/tHM	0
Fueldata[UOX40,Initial DU] t/tHM	0
Fueldata[UOX40,Initial Enr] %	3.7
Fueldata[UOX40,Initial Pu] t/tHM	0
Fueldata[UOX40,Initial MA] t/tHM	0
Fueldata[UOX40,Initial Np] t/tHM	0
Fueldata[UOX40,Initial Am] t/tHM	0
Fueldata[UOX40,Initial Cm] t/tHM	0
Fueldata[UOX40,Initial SLFP] t/tHM	0
Fueldata[UOX40,Initial LLFP] t/tHM	0
Fueldata[UOX40,Spent U] t/tHM	0.945
Fueldata[UOX40,Spent Enr] t/tHM	0.81
Fueldata[UOX40,Spent Pu] t/tHM	0.01085
Fueldata[UOX40,Spent MA] t/tHM	0.00115
Fueldata[UOX40,Spent Np] t/tHM	0
Fueldata[UOX40,Spent Am] t/tHM	0
Fueldata[UOX40,Spent Cm] t/tHM	0
Fueldata[UOX40,Spent FP] t/tHM	0.04225
Fueldata[UOX40,Spent SLFP] t/tHM	0
Fueldata[UOX40,Spent LLFP] t/tHM	0
Fueldata[UOX40, Spent Activity] kBq/tHM	0
Fueldata[UOX40,Spent Heat KW/tHM]	0
Fueldata[UOX40,Fuel Group] 1...5	1

Fueldata[UOX50, BU] GWd/tHM	50
Fueldata[UOX50,CycleLength] months	12
Fueldata[UOX50,NumberBatches] 2..50	5
Fueldata[UOX50,Initial U] t/tHM	1
Fueldata[UOX50,Initial REPU] t/tHM	0
Fueldata[UOX50,Initial DU] t/tHM	0
Fueldata[UOX50,Initial Enr] %	4.7
Fueldata[UOX50,Initial Pu] t/tHM	0
Fueldata[UOX50,Initial MA] t/tHM	0
Fueldata[UOX50,Initial Np] t/tHM	0
Fueldata[UOX50,Initial Am] t/tHM	0
Fueldata[UOX50,Initial Cm] t/tHM	0
Fueldata[UOX50,Initial SLFP] t/tHM	0
Fueldata[UOX50,Initial LLFP] t/tHM	0
Fueldata[UOX50,Spent U] t/tHM	0.93545
Fueldata[UOX50,Spent Enr] t/tHM	0.82
Fueldata[UOX50,Spent Pu] t/tHM	0.012
Fueldata[UOX50,Spent MA] t/tHM	0.00184
Fueldata[UOX50,Spent Np] t/tHM	0
Fueldata[UOX50,Spent Am] t/tHM	0
Fueldata[UOX50,Spent Cm] t/tHM	0
Fueldata[UOX50,Spent FP] t/tHM	0.0513
Fueldata[UOX50,Spent SLFP] t/tHM	0
Fueldata[UOX50,Spent LLFP] t/tHM	0
Fueldata[UOX50, Spent Activity] kBq/tHM	0
Fueldata[UOX50,Spent Heat KW/tHM]	0
Fueldata[UOX50,Fuel Group] 1...5	1

Fueldata[UOX60, BU] GWd/tHM	60
Fueldata[UOX60,CycleLength] months	12
Fueldata[UOX60,NumberBatches] 2..50	5
Fueldata[UOX60,Initial U] t/tHM	1
Fueldata[UOX60,Initial REPU] t/tHM	0
Fueldata[UOX60,Initial DU] t/tHM	0
Fueldata[UOX60,Initial Enr] %	4.9
Fueldata[UOX60,Initial Pu] t/tHM	0
Fueldata[UOX60,Initial MA] t/tHM	0
Fueldata[UOX60,Initial Np] t/tHM	0
Fueldata[UOX60,Initial Am] t/tHM	0
Fueldata[UOX60,Initial Cm] t/tHM	0
Fueldata[UOX60,Initial SLFP] t/tHM	0
Fueldata[UOX60,Initial LLFP] t/tHM	0
Fueldata[UOX60,Spent U] t/tHM	0.92396
Fueldata[UOX60,Spent Enr] t/tHM	0.25
Fueldata[UOX60,Spent Pu] t/tHM	0.01256
Fueldata[UOX60,Spent MA] t/tHM	0.00171
Fueldata[UOX60,Spent Np] t/tHM	0
Fueldata[UOX60,Spent Am] t/tHM	0
Fueldata[UOX60,Spent Cm] t/tHM	0
Fueldata[UOX60,Spent FP] t/tHM	0.06177
Fueldata[UOX60,Spent SLFP] t/tHM	0
Fueldata[UOX60,Spent LLFP] t/tHM	0
Fueldata[UOX60, Spent Activity] kBq/tHM	0
Fueldata[UOX60,Spent Heat KW/tHM]	0
Fueldata[UOX60,Fuel Group] 1...5	2

Reactor Park Development Decision Making

Allow new Reactors? On

Modify Capacity Factors? Off

Use Economics Decision Making? Off

Use Existing reactors? On

Years of life extension beyond predefined Lifetime of Existing Reactor-Vary from 0 to 50 in steps of 10 years

User-Defined New Reactor Order Fractions (%)

30 BWRn

70 PWRn

Initial Reactor Capacity

BWRe-31,789

PWRe-36838

PWReM-26159

Planned Shutdown

PWRe		BWRe		PWReM	
1990	0	1990	0	1990	0
1991	1939	1991	0	1991	0
1992	2836	1992	0	1992	0
1993	5838	1993	0	1993	0
1994	2743	1994	0	1994	0
1995	2637	1995	0	1995	0
1996	2515	1996	0	1996	1106
1997	1813	1997	0	1997	1060
1998	0	1998	1229	1998	0
1999	1765	1999	784	1999	0
2000	1935	2000	1362	2000	2270
2001	0	2001	1171	2001	1136
2002	1909	2002	1524	2002	0
2003	1975	2003	4009	2003	2272
2004	1075	2004	4132	2004	4472
2005	2540	2005	0	2005	3366
2006	1733	2006	1938	2006	2278
2007	1270	2007	0	2007	3491
2008	0	2008	878	2008	2408
2009	1161	2009	0	2009	1150
2010	0	2010	0	2010	0
2011	0	2011	0	2011	0
2012	0	2012	1090	2012	1150
2013	0	2013	3281	2013	0
2014	0	2014	2298	2014	0
2015	1154	2015	3101	2015	0
2016	0	2016	2099	2016	0
2017	0	2017	2243	2017	0
2018	0	2018	0	2018	0
2019	0	2019	1150	2019	0
2020	0	2020	0	2020	0

Reactor and Fuel Combination

Initial Reactor Fuel Combination

BWRe-UOX40 (1), All other Rows (0)

PWRe-UOX50 (1), All other Rows (0)

PWReM-UOX50 (1), All other Rows (0)

BWRn-UOX60 (1), All other Rows (0)

PWRn-UOX60 (1), All other Rows (0)

Intended Reactor Fuel Combination

Same as "Initial"

Possible Reactor Fuel Combination

Same as "Initial"

Fuel and Fuel Facility Combinations

The following are On for UOX40, UOX50, and UOX60. All others fuels and facilities are not in use.

- Unat
- Conv
- Gas Enr
- UOX Fab
- Aq Rep
- SF Stor
- SF Int Stor
- HLW Stor
- SF Cond
- HLW Cond
- SF Geo

Fuel Facility Data

Set on "Unlimited Amount of Fuel Cycle"

Facility Deployment

Not in Use

Existing Facility Shutdown

Not in Use

Reprocessing Fractions

Not in Use

Separated Actinide Allocation

Not in Use

Priorities for Conditioning and Reprocessing

Not in Use

Initial Stocks

UOX40- 7494 (SF Interim)

UOX50-11,820 (SF Interim)

UOX60-200 (Fresh Fuel)

Months of FF Stock -2

Unat Resources

Natural Uranium Resources-KCR+RAR+EAR-I

Average Age Legacy SF-25

Waste Fee-0

Unat Price

Use U-Price Escalation Model? On

Initial U-Price-30

U-Price Start Year-2010

U-Price Escalation Rate-0

MENU Buttons-Not Applicable

APPENDIX B: UOX100

Set up the same as Appendix A, except replace UOX60 with UOX 100.

Fuel Data

Fueldata[UOX100,BU] GWd/tHM	100
Fueldata[UOX100,CycleLength] months	12
Fueldata[UOX100,NumberBatches] 2..50	5
Fueldata[UOX100,Initial U] t/tHM	1
Fueldata[UOX100,Initial REPU] t/tHM	0
Fueldata[UOX100,Initial DU] t/tHM	0
Fueldata[UOX100,Initial Enr] %	8
Fueldata[UOX100,Initial Pu] t/tHM	0
Fueldata[UOX100,Initial MA] t/tHM	0
Fueldata[UOX100,Initial Np] t/tHM	0
Fueldata[UOX100,Initial Am] t/tHM	0
Fueldata[UOX100,Initial Cm] t/tHM	0
Fueldata[UOX100,Initial SLFP] t/tHM	0
Fueldata[UOX100,Initial LLFP] t/tHM	0
Fueldata[UOX100,Spent U] t/tHM	0.88084
Fueldata[UOX100,Spent Enr] t/tHM	0.25
Fueldata[UOX100,Spent Pu] t/tHM	0.01604
Fueldata[UOX100,Spent MA] t/tHM	0.003069
Fueldata[UOX100,Spent Np] t/tHM	0
Fueldata[UOX100,Spent Am] t/tHM	0
Fueldata[UOX100,Spent Cm] t/tHM	0
Fueldata[UOX100,Spent FP] t/tHM	0.1005
Fueldata[UOX100,Spent SLFP] t/tHM	0
Fueldata[UOX100,Spent LLFP] t/tHM	0
Fueldata[UOX100, Spent Activity] kBq/tHM	0
Fueldata[UOX100,Spent Heat KW/tHM]	0
Fueldata[UOX100,Fuel Group] 1...5	2

APPENDIX C 10% Reactor Park PWR-MOX, loaded with 30% MOX core, "Automatic Reprocessing Deployment

Edem Scenario

Same

Reactor Data

Reactorsdata[PWRnM,Pe] MWe	1000
Reactorsdata[PWRnM, Effth] %	34
Reactorsdata[PWRnM fLoad] %	90
Reactorsdata[PWRnM, Fuel Use Flag] 0/1	0
Reactorsdata[PWRnM, Reactor Group] 1...5	4
Reactorsdata[PWRnM, Licensing Time] year	2
Reactorsdata[PWRnM, Construction Time] year	3
Reactorsdata[PWRnM, Lifetime] years	60
Reactorsdata[PWRnM, Construction Cost] B\$/unit	1
Reactorsdata[PWRnM, Other Capital cost] B\$/unit	0
Reactorsdata[PWRnM, DecomCost] B\$/unit	0
Reactorsdata[PWRnM, Contingencies] B\$/unit	0
Reactorsdata[PWRnM, OMCost] B\$/unit	0
Reactorsdata[PWRnM, InitialTRL] 1...9	9

Fuel Data

Fueldata[MOX50, BU] GWd/tHM	50
Fueldata[MOX50,CycleLength] months	12
Fueldata[MOX50,NumberBatches] 2...50	5
Fueldata[MOX50,Initial U] t/tHM	1
Fueldata[MOX50,Initial REPU] t/tHM	0
Fueldata[MOX50,Initial DU] t/tHM	0.91903
Fueldata[MOX50,Initial Enr] %	0.25
Fueldata[MOX50,Initial Pu] t/tHM	0.08097
Fueldata[MOX50,Initial MA] t/tHM	0
Fueldata[MOX50,Initial Np] t/tHM	0
Fueldata[MOX50,Initial Am] t/tHM	0
Fueldata[MOX50,Initial Cm] t/tHM	0
Fueldata[MOX50,Initial SLFP] t/tHM	0
Fueldata[MOX50,Initial LLFP] t/tHM	0
Fueldata[MOX50,Spent U] t/tHM	0.88753
Fueldata[MOX50,Spent Enr] t/tHM	0.25
Fueldata[MOX50,Spent Pu] t/tHM	0.0512
Fueldata[MOX50,Spent MA] t/tHM	0.0074
Fueldata[MOX50,Spent Np] t/tHM	0
Fueldata[MOX50,Spent Am] t/tHM	0
Fueldata[MOX50,Spent Cm] t/tHM	0
Fueldata[MOX50,Spent FP] t/tHM	0.04996
Fueldata[MOX50,Spent SLFP] t/tHM	0
Fueldata[MOX50,Spent LLFP] t/tHM	0
Fueldata[MOX50, Spent Activity] kBq/tHM	0
Fueldata[MOX50,Spent Heat KW/tHM]	0
Fueldata[MOX50,Fuel Group] 1...5	3

Reactor Park Development Decision Making

Allow new Reactors? On

Modify Capacity Factors? Off

Use Economics Decision Making? Off

Use Existing reactors? On

Years of life extension beyond predefined Lifetime of Existing Reactor-Vary from 0 to 50 in steps of 10 years

User-Defined New Reactor Order Fractions (%)

30 BWRn

60 PWRn

10 PWRnM

Initial Reactor Capacity

BWRe-31,789

PWRe-36838

PWReM-26159

Reactor and Fuel Combination

Initial Reactor Fuel Combination

BWRe-UOX40 (1), All other Rows (0)

PWRe-UOX50 (1), All other Rows (0)

PWRM-UOX50 (1), All other Rows (0)

BWRn-UOX40 (1), All other Rows (0)

PWRn-UOX50 (1), All other Rows (0)

PWRnM-MOX50 (0.3), UOX50 (0.7), All other Rows (0)

Intended Reactor Fuel Combination

Same as "Initial"

Possible Reactor Fuel Combination

Same as "Initial"

Fuel and Fuel Facility Combinations

The following are On for UOX40, UOX50, and MOX50. All others fuels and facilities are not in use.

- Unat
- Conv
- Gas Enr
- UOX Fab
- MOX fab (MOX 50 only)
- Aq Rep
- SF Stor
- SF Int Stor
- HLW Stor
- SF Cond
- HLW Cond
- SF Geo

Fuel Facility Data

Set "InitCap" to 1e+009 (i.e. unlimited) for all except "Aqueous Reprocessing" here
"InitCap"=0

Facility Deployment

Automatic Fuel Cycle Facility deployment

Existing Facility Shutdown

Not in Use

Reprocessing Fractions

UOX40 and UOX50=1

Separated Actinide Allocation

REPU, Pu and MA from UOX40 and UOX50 to MOX50 are (1); rest of the rows (0)

Priorities for Conditioning and Reprocessing

Equal Priority

Initial Stocks

UOX40- 7494 (SF Interim)

UOX50-11,820 (SF Interim)

UOX60-200 (Fresh Fuel)

Months of FF Stock -2

Unat Resources

Natural Uranium Resources-KCR+RAR+EAR-I

Average Age Legacy SF-25

Waste Fee-0

Unat Price

Use U-Price Escalation Model? On

Initial U-Price-30

U-Price Start Year-2010

U-Price Escalation Rate-0

MENU Buttons-Not Applicable

APPENDIX D: 10% Reactor Park PWR-MOX, loaded with 30% MOX core, "User Defined"

Set up the same as Appendix C except,

Fuel Facility Data

Set "InitCap" to 1e+009 (i.e. unlimited) for all except "Aqueous Reprocessing" here
"InitCap"=1500

Facility Deployment

User Defined Fuel Cycle Facility deployment

APPENDIX E: 10% Reactor Park Fast Reactor "Pu-burner", CR=0.5, Automatic Reprocessing Deployment

Edem Scenario

Edem Scenario	Edem Exponential
Initial Edem (Twe/yr)	770
Start Year of Edem Growth	2010
Edem annual growth rate (%yr)	0, 1.5 or 3.0

Reactor Data

Reactorsdata[FRbu,Pe] MWe	1000
Reactorsdata[FRbu, Effth] %	34
Reactorsdata[FRbu, fLoad] %	90
Reactorsdata[FRbu, Fuel Use Flag] 0/1	0
Reactorsdata[FRbu, Reactor Group] 1...5	4
Reactorsdata[FRbu, Licensing Time] year	2
Reactorsdata[FRbu, Construction Time] year	3
Reactorsdata[FRbu, Lifetime] years	60
Reactorsdata[FRbu, Construction Cost] B\$/unit	1
Reactorsdata[FRbu, Other Capital cost] B\$/unit	0
Reactorsdata[FRbu, DecomCost] B\$/unit	0
Reactorsdata[FRbu, Contingencies] B\$/unit	0
Reactorsdata[FRbu, OMCost] B\$/unit	0
Reactorsdata[FRbu, InitialTRL] 1...9	9

Fuel Data

Fueldata[FRd,BU] GWd/tHM	185
Fueldata[FRd,CycleLength] months	51
Fueldata[FRd,NumberBatches] 2..50	6
Fueldata[FRd,Initial U] t/tHM	0
Fueldata[FRd,Initial REPU] t/tHM	0.46589
Fueldata[FRd,Initial DU] t/tHM	0.0.8977
Fueldata[FRd,Initial Enr] %	0.44
Fueldata[FRd,Initial Pu] t/tHM	0.44434
Fueldata[FRd,Initial MA] t/tHM	0
Fueldata[FRd,Initial Np] t/tHM	0
Fueldata[FRd,Initial Am] t/tHM	0
Fueldata[FRd,Initial Cm] t/tHM	0
Fueldata[FRd,Initial SLFP] t/tHM	0
Fueldata[FRd,Initial LLFP] t/tHM	0
Fueldata[FRd,Spent U] t/tHM	0.465889
Fueldata[FRd,Spent Enr] t/tHM	0.32262
Fueldata[FRd,Spent Pu] t/tHM	0.322262
Fueldata[FRd,Spent MA] t/tHM	0.0074
Fueldata[FRd,Spent Np] t/tHM	0
Fueldata[FRd,Spent Am] t/tHM	0
Fueldata[FRd,Spent Cm] t/tHM	0
Fueldata[FRd,Spent FP] t/tHM	0.184623
Fueldata[FRd,Spent SLFP] t/tHM	0
Fueldata[FRd,Spent LLFP] t/tHM	0
Fueldata[FRd, Spent Activity] kBq/tHM	0
Fueldata[FRd,Spent Heat KW/tHM]	0
Fueldata[FRd,Fuel Group] 1...5	4

Reactor Park Development Decision Making

Allow new Reactors? On

Modify Capacity Factors? Off

Use Economics Decision Making? Off

Use Existing reactors? On

Years of life extension beyond predefined Lifetime of Existing Reactor-Vary from 0 to 50 in steps of 10 years

User-Defined New Reactor Order Fractions (%)

30 BWRn

60 PWRn

10 FRbu

Initial Reactor Capacity

BWRe-31,789

PWRe-36838

PWReM-26159

Reactor and Fuel Combination

Initial Reactor Fuel Combination

BWRe-UOX40 (1), All other Rows (0)

PWRe-UOX50 (1), All other Rows (0)

PWReM-UOX50 (1), All other Rows (0)

BWRn-UOX40 (1), All other Rows (0)

PWRn-UOX50 (1), All other Rows (0)

FRbu-Frd (1), All other Rows (0)

Intended Reactor Fuel Combination

Same as "Initial"

Possible Reactor Fuel Combination

Same as "Initial"

Fuel and Fuel Facility Combinations

The following are On for UOX40, UOX50, and MOX50. All others fuels and facilities are not in use.

- Unat
- Conv
- Gas Enr
- UOX Fab
- MOX fab (FRd only)
- Aq Rep
- SF Stor
- SF Int Stor
- HLW Stor
- SF Cond
- HLW Cond
- SF Geo

Fuel Facility Data

Set "InitCap" to 1e+009 (i.e. unlimited) for all except "Aqueous Reprocessing" here
"InitCap"=0

Facility Deployment

Automatic Fuel Cycle Facility deployment

Existing Facility Shutdown

Not in Use

Reprocessing Fractions

UOX40 and UOX50=1

Separated Actinide Allocation

REPU, Pu and MA from UOX40 and UOX50 to FRd are (1); rest of the rows (0)

Priorities for Conditioning and Reprocessing

Equal Priority

Initial Stocks

UOX40- 7494 (SF Interim)

UOX50-11,820 (SF Interim)

UOX60-200 (Fresh Fuel)

Months of FF Stock -2

Unat Resources

Natural Uranium Resources-KCR+RAR+EAR-I

Average Age Legacy SF-25

Waste Fee-0

Unat Price

Use U-Price Escalation Model? On

Initial U-Price-30

U-Price Start Year-2010

U-Price Escalation Rate-0

MENU Buttons-Not Applicable

APPENDIX F: Helpful Hints on using DANESS

The order of the headings in Appendices A-E follow the same basic layout of the DANESS code graphical user interface. That is when using DANESS, the user should begin from the top of the screen and work their way down through the interface model buttons. All headings in the Appendices are as listed in the DANESS program. Therefore if the users begins with the first heading in the Appendices (Edem Scenario) and follows the headings accordingly, the input of data into DANESS is relatively simple. The input of data into DANESS may occur by several different means. For input decks such as "Fuel Data" the information exists in table form. Other headings such as "Edem Scenario" and "Fuel & Fuel Facility Combinations" use input means such as sliders and matrices respectively.

Once the prescribed data is inputted into DANESS, the next task is to run the simulation. The most common error that will occur is that a variable cannot be evaluated because of "division by zero or value has become too large to represent." There is no definite way to track an error. The best option is to go to where the input for the variable in question is located and by trial and error find the value that is causing the error. Another useful heading in DANESS that assists in ensuring proper input is the "Analysis of Results" button located at the bottom of the interface. This button contains several matrices that indicate the fuel, reactor, and/or facility combinations the user has entered. It is a means by double-checking your input.

Once the program has been sufficiently modeled and runs without error, the next step is to analyze the output. The output is in both table and graph form. It is best to initially scroll through the graphs and look for any obvious problems. The first item to check is the output variable "Ecov" under the "Summary Graph". If "Ecov" drops below 94-95% the model should be re-worked. Other informative graphs are: "Ready Reactors", "In-Fabrication Fuels", "Fresh Fuel Stocks", and "Available Pu Fuels" (for MOX or fast fuel cases).

These are the basics to running DANESS. However, these methods are from personal experience. Different users may find other methods of data input simpler and different output graphs more informative. The design of the program facilitates usage, and just like with any other computer program the user will become more adept with time and practice.

VITA

Gary Alan Sweder was born in Morgantown, West Virginia on September 5, 1980 to John and Cheryl Sweder. He attended Albert Gallatin public high school in Uniontown, Pennsylvania. Upon graduation from high school in 1999, he enrolled at the University of Maryland located in College Park, Maryland. There he majored in Nuclear Engineering and received a Bachelor of Science in that field in December 2003. The subsequent year he worked as a Bank of America customer service representative. Wanting to pursue the field of his undergraduate study, he enrolled in the Master of Science program in Nuclear Engineering at the University of Tennessee, Knoxville in January 2005. He is presently a Graduate Research Assistant in the program and plans to receive a Master of Science degree in May of 2006.