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Uncertainty Analyses of Advanced Fuel Cycles

Final Report for April 1, 2005 through March 31, 2008

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1. Introduction

The Department of Energy is developing technology, experimental protocols, computational methods, systems analysis software, and many other capabilities in order to advance the nuclear power infrastructure through the Advanced Fuel Cycle Initiative (AFCI). Our project, "Uncertainty Analyses of Advanced Fuel Cycles," is intended to facilitate will-informed decision making for the selection of fuel cycle options and facilities for development. In order to achieve this goal the following objectives were proposed and are accomplished by this research:

1) design and analyze advanced fuel cycles for LWRs, including BWRs,
2) identify and assess the repository benefits of advanced fuel cycles,
3) determine the effect of uncertainties on repository benefit assessments,
4) conduct dynamic fuel cycle scenario studies to develop an understanding of the issues in the transition from thermal reactor to a mixed thermal/fast reactor fleet, or a fleet in which the majority of reactors are fast and/or accelerator-driven,
5) optimize the use of key resources, e.g., repository capacity and uranium ore, in the long term for advanced fuel cycles, and
6) evaluate the optimal use of fast reactors.

These objectives are accomplished by a team effort between Argonne National Laboratory (ANL) and The University of Tennessee Department of Nuclear Engineering (UTNE) with ANL and UTNE receiving 25 and 75 % of the funding, respectively. The primary contributions of ANL are the enhancement of the DANESS (Dynamic Analysis of Nuclear Energy System Strategies) software and of reactor analysis computational support. The objectives cited above are accomplished, in part, through the following activities:

1) utilization of the DANESS code as a computational tool for scenario analyses,
2) generation end-of-cycle Light Water Reactor (LWR) and Fast Reactor (FR) isotopic compositions for DANESS and for decay heat analyses,
3) incorporation of uncertainty information into the analyses for optimization of equilibrium fuel cycles under uncertainty,
4) identification of uncertainties associated with important fuel cycle parameters,
5) analyses of dynamic and equilibrium fuel cycle scenarios, and
6) development of software for uncertainty analysis of equilibrium fuel cycles.

Light Water Reactors generate all of the nuclear power in the United States and will most likely continue to do so for at least several decades. Thus, and evaluation of various scenarios for their use offers an opportunity for modestly enhancing the sustainability of power production by nuclear energy. Some issues considered in this study include the following:

1) Sustainability,
2) Alternative fuel cycles,

3) Impact of Time-of-implementation of alternative fuel cycles.
4) Use high burnup fuel (i.e. 70 GWth-d/tonne),
5) Reprocessing of LWR fuel,
6) Use Mixed Oxide (MOX) fuel,
7) Introduction of FRs in to the LWR fleet,
8) Inventories of Plutonium (Pu) and other actinides in and out of reactors,
9) Economic impact of alternative fuel cycles, and
10) Uncertainties of parameters and outcomes.

Salient issues associated with these topics are addressed in the main body of this report and details are presented in the appendices.

2. Sustainability and Fuel Cycle Options

The ability to use nuclear energy for hundreds of years depends on the raw materials and on maintenance of organizational infrastructures for management, training, and education. As for fuel supply for the once-through fuel cycle, it may be effectively infinite if suitably technology is developed to extract uranium from seawater. If may, however, be more cost effective to generate fissile fuel with breeder reactors than to restrict the use of nuclear power to the once-through option when life-cycle costs are properly taken into account. This is discussed in more detail in section Appendix B.

Nuclear energy currently utilizes the once-through fuel cycle, and it produces about twenty percent of the electrical power generated in the US. However, expansion of power production with the current once-through fuel cycle is limited to about 1,000 reactors for 100 years, based on an estimate of about 15×10^8 metric tons of natural uranium with production cost of less than $130/kg and on a burn up of about 50 GWd/ton. Thus, the current technology for utilization of nuclear energy with relatively inexpensive fuel is not considered to be sustainable with relatively inexpensive uranium.

Recycle of fissile material provides about a thirty percent gain in resource utilization, which is much less than the uncertainty in the availability of economically priced natural uranium. Thus, there is little economic or technical incentive for implementation of a fuel cycle that only recycles fissile material. This option does offer the opportunity for reducing the requirements for disposal of spent fuel and for reducing the short-term heat load, if fission products are handled separately, since the actinides contribute nearly all of the 1500-year integral decay heat to the repository if fission produces are handled separately. Requirements for repository space are reduced by about a factor of two if this option is implemented.

Recycle of fissile and fertile material and most (about 95%) of the actinides, based on previously demonstrated reprocessing technology, would result in about a factor of three reduction in repository space. However, this approach, coupled with breeder reactors, could provide an essentially infinite supply of electrical energy. If essentially all of the actinides (99.9%) are recycled into advanced reactors or accelerator driven systems, a reduction factor for the required repository space of about 60 can be realized.

The MIT Report on the Future of Nuclear Energy[2] claims that the availability of uranium is not the limiting constraint for expanding the use of nuclear energy. Instead, economics, waste management, proliferation, and public acceptance are more important than uranium resources. Wineberg[1] claims that the use of breeder reactors is necessary for nuclear power to have a long-term impact on human energy consumption; where "long-term" is several thousand years. The MIT article appears to be focused of electrical energy usage during the relatively near future, such as the next hundred years. Thus, there is not a real contradiction, but only a difference in perspective.

The Department of Energy (DOE) recognizes the need for the development of advanced reactors and advanced fuel cycles to assure that the use of nuclear fission remains a viable option for production of electrical power. In particular, it supports the Generation IV reactor development program, and it supports the *Advanced Fuel Cycle Initiative* of alternative fuel cycles with a focus on Generation IV systems[3].

Sustainability is an important issue relative to choices for fuel cycles and for reactor systems. In particular, the following conditions should be satisfied for sustainability:

1) The fuel supply should last for several thousand years,
2) The radioactive materials produced from operation should be effectively managed and not exceed the original activity of ore after about one thousand years,
3) Materials required to construct and operate reactors should be in adequate supply, and
4) Byproducts of operation should not create a significant risk, which includes proliferation of nuclear weapons.

Sustainability issues considered to be important are as follows:

1) Resources,
2) Environmental Effects,
3) Economics,
4) Societal Impacts,
5) Proliferation,
6) Infrastructure Commitments,
7) Waste Management, and
8) Vulnerability for Disruptions.

These topics are discussed in Appendix B.

Fuel Cycles of Interest and Repository Issues

The transuranics and minor actinides (MA) in spent fuel from reactors use the Pu/U fuel cycle may be disposed of directly with the spent fuel, or they can be separated for subsequent disposition or use in reactor facilities. Implementation of practices that

manage these transuranics require investments in support facilities and infrastructure that competes with the much more straightforward practice of direct disposal of spent fuel. Some of the generic fuel cycle concepts under consideration that may be optimal for sustainability or life cycle costs include the following:

1) once-through LWRs that use enriched natural uranium,
2) recycle of Pu extracted from spent fuel into LWR fuel,
3) recycle of Pu and MAs extracted from spent fuel into LWR fuel,
4) use of Pu and MAs in FRs in conjunction with LWRs, and
5) use of the Th-232/U-233 fuel cycle.

Numerous variations on these generic fuel cycle concepts are documented by very qualified researchers. Apparently, the optimal fuel cycle is difficult to determine since there is no general consensus after of over 50 years of research on nuclear energy. One objective of this study is to better quantify the uncertainties associated with alternative fuel cycles and to develop a computational tool that will assist in this assessment.

Figures 2-1 through 2-4 illustrate four fuel cycle concepts that are candidates for implementation. Variations of these are described in Appendix C, and are analyzed in this study.

Once-Through Fuel Cycle

The once-through fuel cycle is well established and does not require additional technology development. Thus, there is little commercial incentive to develop other fuel cycles given that disposal is handled through a tax of 1 mil/kW-h. A significant disadvantage of this fuel cycle is that notably more repository space is required (based on long-term heat load) than if an advanced fuel cycle with reprocessing is implemented. Legacy waste, with the addition of spent fuel being produced by operating reactors, will require all of the Yucca mountain physical capacity within about ten years. In order to extend the Yucca mountain capacity, it is necessary to separate several actinides and fission products from spent fuel.

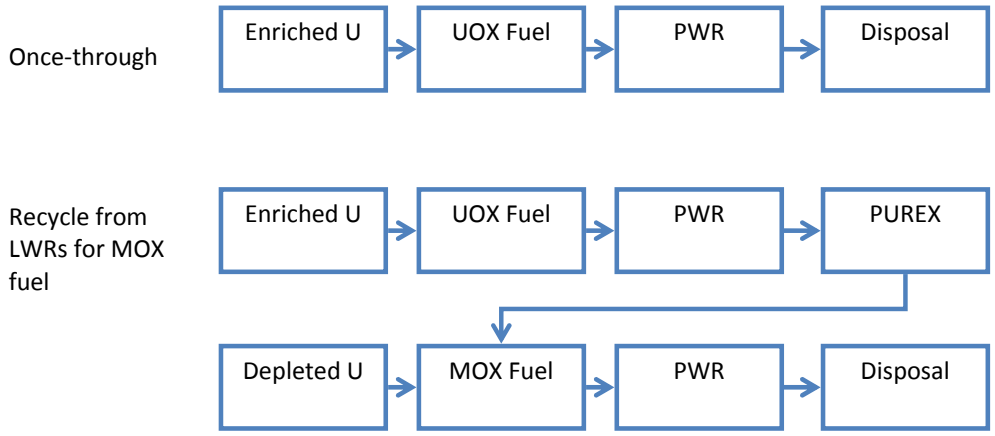

Figure 2-1: Illustrations of the once-through and of recycle to LWRs fuel cycle concepts with depleted uranium feed.
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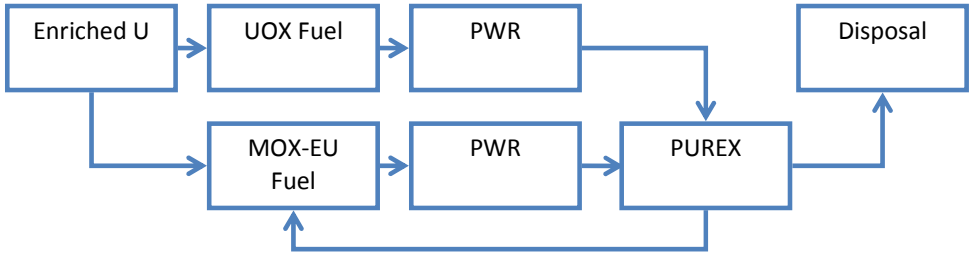

Figure 2-2: Illustrations of the once-through and of recycle to LWRs fuel cycle concepts with enriched uranium feed

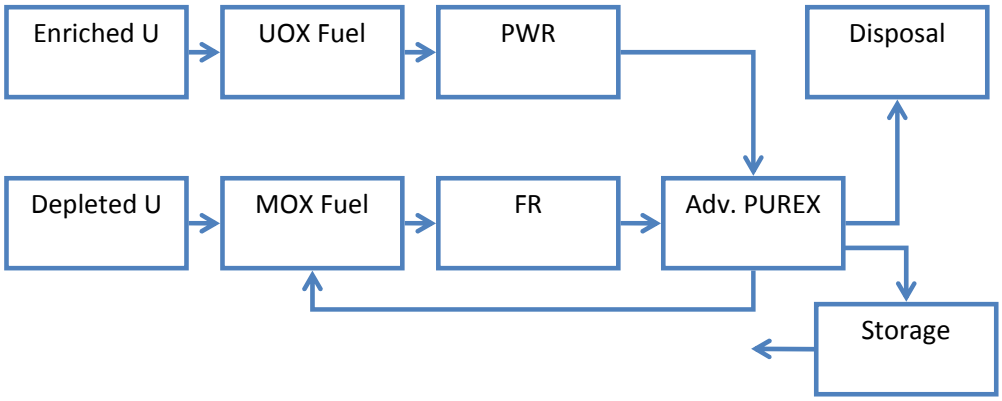

Figure 2-3: A two-tier concept with recycle from FRs.

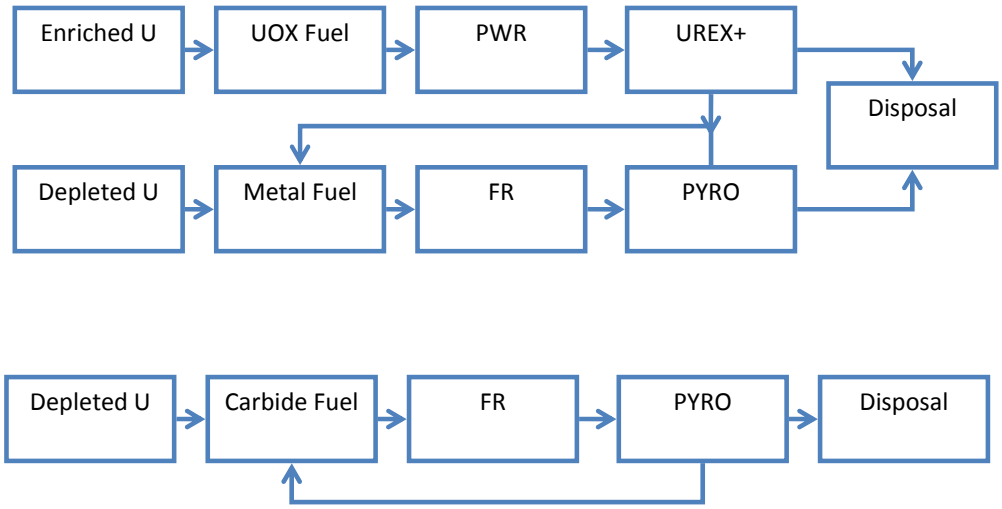

Figure 2-4: A two-tier fuel concept with actinide recycle from PWRs and FRs

MOX Recycle

The MOX recycle option could be handled in several ways. One is to recycle into existing reactors, another is to recycle into new reactors, and a third would be to recycle into fast reactors.

The most likely approach is that existing reactors that are capable of accepting MOX fuel will be fueled with one-third MOX fuel. There are twenty reactors in the US fleet that could use MOX fuel. If full-core reloads could be utilized, only about 12 percent of the fleet needs to be MOX-fueled in order utilize all of the Pu generated by the fleet. Utilization of this approach could keep the Pu stocks from increasing.

Recycle into new reactors has some advantage of being able to use a full-core reload, which is the case with the EPR. Currently, for example, the MOX recycle approach in French reactors utilizes one-third core reloads.

There are some disadvantages with recycling MOX fuel into thermal reactors. One is that the fraction of minor actinides is increased, which creates a larger heat load for the repository. Hence, this may be counterproductive from a waste management perspective. Another possible disadvantage is the more Pu-240 is produced, which results in the requirement that more Pu is needed for achievement of criticality in a fast reactor.

If fast reactors are used for controlling the Pu inventory, minor actinides can most probably be utilized as fuel and their inventory will most likely not increase during the MOX fuel residence in the fast reactor core. If fast reactors are used as burners, rather than breeders, or near breeders, then it is expected that about twelve percent of the reactor fleet would need to be fast reactors of this type. If fast reactors are operated with conversion ratios near unity, Pu inventories will continue to increase if U-238 is included in the fuel. Alternatively, Pu could be utilized in a metal matrix fuel that does not produce additional Pu.

Thermal and Fast Reactors

If it is an objective to increase the fast reactor fleet, it may be preferred to utilize fast reactors with conversion ratios near or greater than unity. In the case with conversion ratios near unity, the reactivity swing between refuelings is minimal, and technology for these reactors is proven. However, reactivity control for burner fast reactors may present a technological challenge. An alternative approach for utilization of fast reactors is to use thorium-plutonium based fuels to avoid reactivity swings and to control the total Pu inventory in the fuel cycle.

Mixed Reactor Fleet

A mix of LWRs, FRs and HTBRs could be deployed for effective production of electricity and hydrogen while controlling the fleet-wide or system-wide inventory of Pu

and other actinides. A study of a mixed reactor fleet should be considered, first for fast reactors and second for HTBRs. The problem for fast reactors would be reactivity control if they are to be used for plutonium consumption. There may be advantages to using Th as a burnable poison and to produce U-233, which could be used for fueling HTBRs, which could most likely then use Th for fuel. A mix of LWRs, FRs, and HTBRs could most likely be sustainable without increasing the PU inventory. U-233 would be present in relatively large quantities, but is has never been considered as a concern for proliferation.

Pebble Bed Reactors

One advantage of pebble bed reactors is that reactivity control is accomplished on line. This would permit these reactors to be used for Pu-burner reactors, while at the same time permit production of hydrogen with high temperature reactors. The overall balance among LWRs, FRs, and HTBRs would be determined from the desired growth rate of a particular type. If pebble bed reactors are operated as Pu burners, and they are expanded, then it may be necessary to operate fast reactors with a conversion ratio greater than unity in order to produce an adequate supply of fissile fuel.

Minimization of Pu in the Reactor Fleet

This objective will most likely require the deployment of fast reactors that can burn actinides, since minor actinides are also important contributors to the long term heat load.

Reduction of Repository Space Required

The repository space required is determined by the long-term heat load generated by the spent fuel or high level waste, and one of the primary contributors to the heat load is Am, which can be separated. Since the half life is only about 20 years, it could be allowed to decay to Pu and the Pu could subsequently be used for fuel. Separation of minor actinides for recycle into LWRs or FRs for destruction or for fuel has the potential for reducing repository space required by a factor of 50 if secondary wastes can be effectively minimized.

3. Dynamic Analyses of Nuclear Energy System Strategies (DANESS)

The Dynamic Analysis of Nuclear Energy System Strategies (DANESS) code was developed by the Argonne National Laboratory (ANL) with much of the recent development work accomplished by Luc Van Den Durpel. This software is used extensively for evaluating scenarios for implementation of various fuel cycle concepts. In particular, DANESS is used for determination of composition and mass of spent fuel in LWR and FR fleets for the following scenarios:

1) growth rates of LWRs,

2) times-of-implementation of high burnup fuel (i.e. 40 to 100 GWd/t)
3) times-of-incorporation of MOX fuel into a selected fraction of the LWR fleet,
4) times-of-incorporation of FRs into a LWR fleet under several LWR and FR growth rates, and
5) uncertainty analyses.

The relatively detailed description of the capabilities of the DANESS code if presented in Appendix A.

4. Determination of Spent Fuel Composition

DANESS uses libraries of composition of fuel that at tabulated for specific burnups, fuel type, and reactor type. At the time of this research, additional compositions for the DANESS libraries were needed in order to run meaningful studies. Examples of descriptions for generation of LWR and FR cross sections (for the same nuclides used by DANESS) are shown in Appendix D. Results for the decay heat from actinides are also shown.

5. Effects of Time-of-Implementation of Parameters on Fuel Cycle Characteristics

Implementation of advanced fuel cycle options that include recycling require the development of extensive support infrastructure for fuel fabrication, reprocessing, safeguards, security and non-proliferation. For example, some reprocessing is required to supply LWRs or FRs with MOX fuel. Most likely remote fuel fabrication is needed to fabricate MOX fuel, and reprocessing is required to extract Pu and MAs from spent fuel. For the case of high burnup, only improved fuel and in-core fuel management in needed.

High Burnup Fuel: The use of high burnup fuel extends fuel supply in a nearly linear relationship and lengthens the time between refuelings. Thus increasing burnup from 40 to 70 GWth/tonne could increase by about 75 %, and improve sustainability. However, decay heat increases with burnup, and this makes direct disposal and reprocessing more difficult. It is the opinion that use of high burnup fuel has only a models impact (about 25 %) on any parameter of interest.

Reprocessing: Estimates for the cost of reprocessing plants range from about 20 to 30 billion dollars for a facility with the capacity to reprocess about 2,000 MTHM per year. Some[11] expect that reprocessing will cost about 1,000 $/kg with a break even cost of uranium of $420/kg.

MOX Fuel: The use of MOX fuel can very modestly extend fuel supplies, about 25 %, and it can reduce system Pu inventories by about the same amount. Our view is that use of MOX fuel in LWRs is very counterproductive. It increases decay heat in

spent fuel, complicates fuel fabrication, diminishes reactor margins, requires reprocessing, and is the most expensive fuel cycle option of all studied in this research.

Introduction of Fast Reactors into a LWR Fleet: Fast reactors in a mixed LWR/FR fleet could facilitate an expansion of the fuel supply for LWRs and for FRs if operated in a breeder mode. However, the current philosophy for FRs is to use them as burners of Pu and minor actinides, not breeders. Since the current fleet of about 100 LWRs generate about 2,000 tons of spent fuel (SF) per year and the Pu concentration in SF is slightly less than one percent, there is a maximum of about 15 to 20 tons of Pu in SF generated each year. If this Pu is fabricated into an inert matrix fuel, about 15 to 20 FRs of equal power to the LWRs could burn all of the Pu generated by 100 LWRs. If U-238 is in the fuel and the conversion ratio is as high as 0.5, then about 40 FRs would be required, or about one FR for two LWRs. The number of FRs required to burn minor actinides requires sophisticated neutronics calculations where the target (or fuel) design is very important.

Inventories of Pu and Other Actinides: Inventories of Pu and other actinides in a particular fuel cycle can be estimated by back-of-the-envelope calculations to obtain qualitative assessments of inventories. Given a fleet of 100 LWR that produce 15 to 20 tons of Pu annually, the total Pu in the spent fuel would be from 600 to 800 tons after 40 years of operation and 1,500 to 2,000 after 100 years of operation. The Pu generated after 40 years of operation could be used to start about 30 to 70 FRs, depending on conversion ratios of the FRs. Introduction of about 50 FR with conversion ratios could easily keep the Pu inventory below a few hundred tons. Results from calculations using the DANESS code to model specific scenarios are given in Sections 5.1 and 5.2.

5.1 Fuel Cycle Simulations with Thermal Reactors

The time of implementation incorporated the standard shutdown profile was taken from eia.doe.gov, and the manner of time implementation was consistent in all cases. Additional 10-year increments were added to the current shutdown profile, thereby effectively delaying the change in current nuclear industry trends. The first run was done using the standard shut down profile; the second and each of the next five were done by extending the shutdown profile by 10 years. So the data collected were 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. In order to build reprocessing facilities in correspondence to the building of the fast reactors, a 1500 tHM/yr, tons of heavy metal per year, reprocessing plant was implemented 10 years after the shut down profile started to take effect. This means that for the base case the reprocessing plant was constructed in 2010 and then for the 50 year delayed case it was constructed in 2060. The reprocessing fractions for the UOX50 and UOX40 were set to 1 so as to make all of the spent UOX fuel available to be reprocessed.

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 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 two cases were then done three more times with varying energy demands applied to them.

5.2 Fuel Cycle Simulations with Fast Reactors

The results from DANESS v1.3.1r US were exported into Microsoft Excel sheets. The total amount of spent fuel was 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. The resulting fast reactor data are shown below in Figures 5-1 – 5-3.

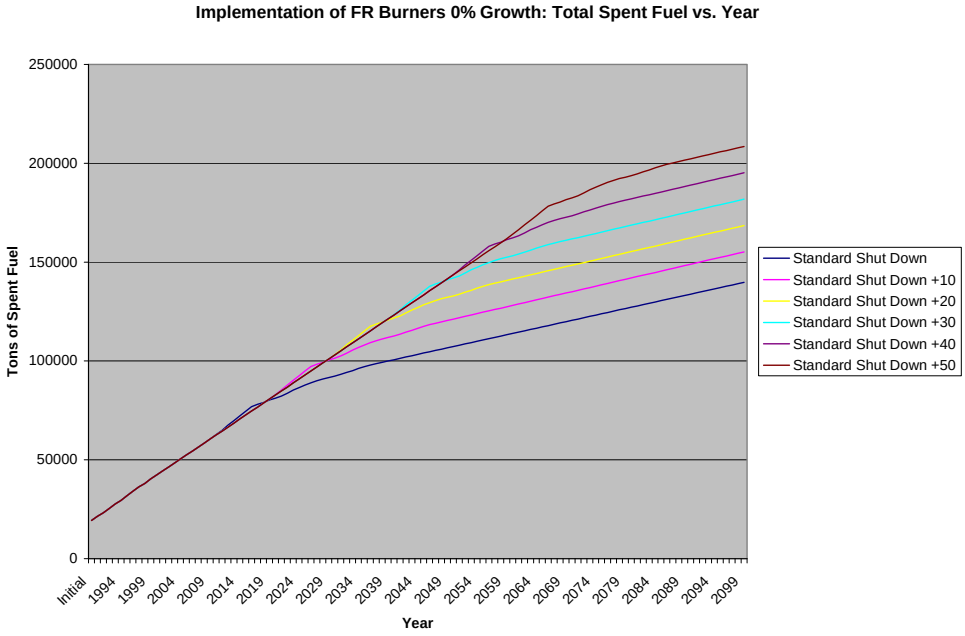

Figure 5-1: Spent Fuel Totals for Time of Implementation of Fast Burner Reactors 0%Growth

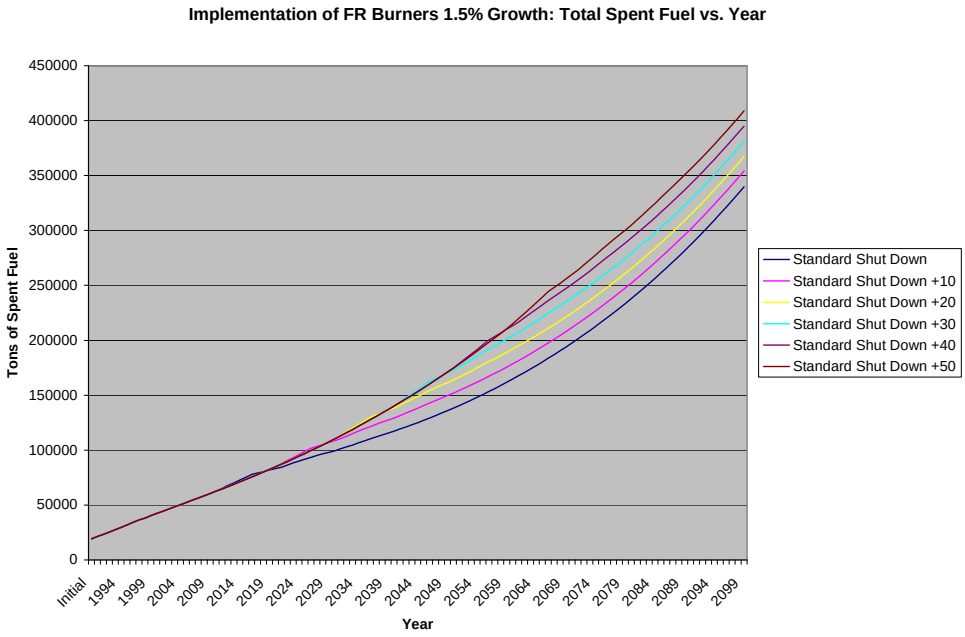

Figure 5-2: Spent Fuel Totals for Time of Implementation of Fast Burner Reactors 1.5%Growth

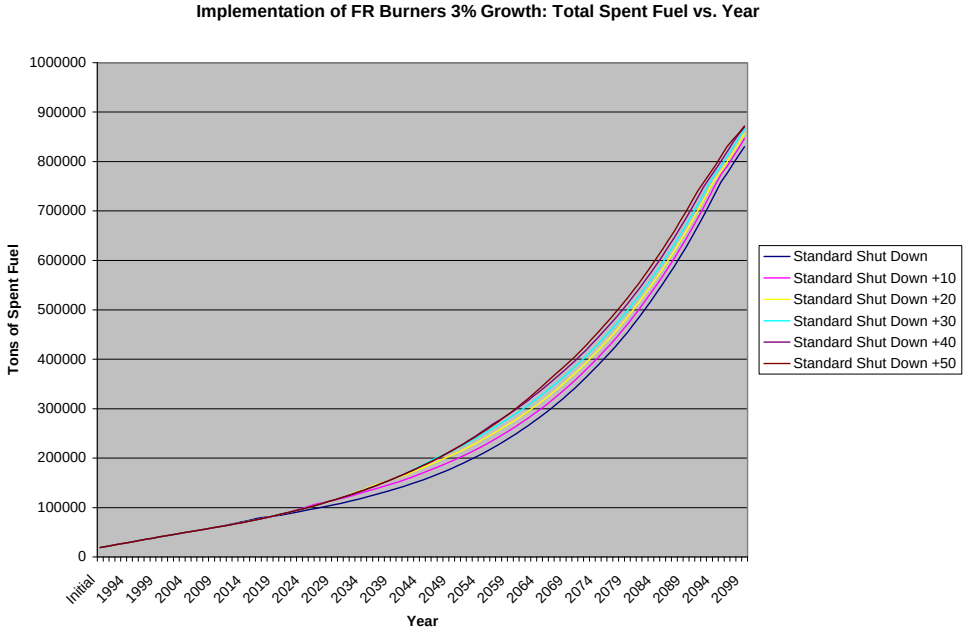

Figure 5-3: Spent Fuel Totals for Time of Implementation of Fast Burner Reactors 3%Growth

Integration of fast reactors (FRs) into the US fleet of reactors offers some significant opportunities to diminish requirements for radioactive waste repository space and to extend uranium ore resources. If fast reactors are operated as burners they can control inventories of problematic radioactive isotopes, such as Plutonium (Pu) and various minor actinides (MA). If they are operated with conversion ratios near or above unity, the fuel supply for fission reactors becomes essentially unlimited. Results reported for this quarter focus on a mixed fleet of LWR and FR fleet of reactors and an assessment of actinide inventories. Comments of uncertainties are also included.

Simulations reported in the sixth-quarter report focused on time-dependent profiles of Pu and MA inventories relative to time-of-implementation of mixed reactor fleets. Scenarios evaluated for this quarter include inventories of Pu and minor actinides (MA) at year 2100, and some additional time-dependent calculations are reported. Results from DANESS calculations that include fast reactors (FRs) are somewhat constrained by specifications that to date cannot be changed by the user and by availability of data through the seventh quarter.

Fuel cycle scenarios evaluated this quarter include the following: unlimited reprocessing capacity, constant LWR fleet of 105 reactors, a growing fleet of FRs with several times-of-implementation and with growth rates of 1, 3, or 5 Rx/yr, and growing fleets of LWRs and FRs. These simulations are analyzed with three initial values of spent fuel, which include: the actual initial legacy spent fuel (55,000 tHM), no initial spent fuel, and an unlimited amount of spent fuel. The specifications for LWRs and FRs employed in the scenarios cited above are listed in Tables 5-1 and 5-2.

Results with no growth in LWRs

Results listed in Table 5-3 illustrate that increasing the nuclear reactor capacity by adding one FR per year significantly decreases the total and out-of-pile amount of Pu associated with the rector fleet. In particular, note that the out-of-pile Pu at year 2100 with about 160 reactors is about one-half of that for a reference fleet of 105 LWRs and that the total Pu at 2100 is about 30 % less with 105 LWRs and 60 FRs than with 105 LWRs. The Pu inventory could be much further reduced with inert matrix fuel; however, the choice to minimize Pu inventory diminishes the feasibility for extensive expansion of production of nuclear power. You may note that inventories of MA are not reduced. This is because there is insufficient fuel composition data incorporated into the current version DANESS for explicitly recycling these materials into LWRs, FRs, or accelerators; however, results from recently completed neutronic calculations performed at ANL should largely resolve this problem. Inventories relative to the reference case are listed in Table 4.

Results listed in Tables 5 and 6 are comparable to those in Tables 5-3 and 5-4 except that the input specification for DANESS is to increase the FR part of the fleet by three reactors per year, rather than one reactor per year. Likewise results listed in Tables 7 and 8 are based on specifications for DANESS to increase the FR portion of the fleet by

five reactors. Regardless of the requested rate of increase of FRs only about 35 to 70 FRs are built due to constraints on the availability of Pu inherent to the DANESS simulation software, even with unlimited reprocessing capacity. If one assumes that results from these simulation runs represents reality, only 35 to 70 FRs should be expected to be in operation by 2100 if these reactors have conversion ratios of 0.5 and if the LWR fleet is limited to about 100. Correspondingly, the inventories of Pu and MA could vary by about a factor of two. However, fuel for many more FRs could be available with a growing LWR fleet and with FRs with conversion ratios of near unity. You may note from the tables that the choice to consider no initial availability of Pu results in about 20 fewer FRs than when Pu in the current inventory of spent fuel is utilized.

Results with growth in LWRs and FRs

The specifications for reactors and their corresponding fuels are the same as used for previous cases, listed in Tables 5-1 and 5-2. However, the reference case for these runs implements 8 LWRs per year starting in the year 2025. The subsequent runs implement 6 LWRs and 2 FRs per year instead of the 8 LWRs. Reprocessing capacity is set to unlimited, and all of the LWR and FR fuel is set to recycle. The initial fleet is initialized to 105 LWRs, legacy spent fuel inventory is set at the current level of 55,000tHM, and times-of-implementation are 2020, 2030, and 2040. The results that illustrate the Pu and MA inventories as a function of time are illustrated in Figures 5-3 and 5-4, and the corresponding number of reactors is illustrated in Figure 5. The basic fuel cycle scenario for the DANESS runs with FRs illustrated in Figure 5-6. Absolute and relative inventories at year 2100 are listed in Tables 5-9 and 5-10.

It is apparent that the introduction of the FRs, as opposed to a total LWR reactor fleet, results in a reduction of Pu inventory of about a factor of two, but the impact on MA inventory is rather modest. Note that implementation of the 2 FRs per year instead of an additional 2 LWRs yields a 40% reduction in total Pu inventory and that the majority of that inventory is in-pile rather than out-of-pile. Inventories of U, Pu, MA, Np, Am, and FP (fission products) are listed in Tables 5-11, 5-12 and 5-13. These values are based on the total spent fuel (SF) reported in the corresponding tables.

Comments on Uncertainties

The number of fast reactors that are likely to be built during the next 100 years will significantly influenced by policy decisions relative to nonproliferation, the capacity and features of reprocessing plants, conversion ratios of reactors, and international agreements. Other decisions, such as how spent fuel is to be stored and policies for assurance of fuel supply are expected to be less important. For example, DANESS requires that sufficient fuel must be available to meet reactor needs for 15 years before it will build a new reactor. If the LWR fleet is held constant at about 100, then relatively minor variations in construction rates could influence fuel supply availability by a factor of two. If the LWR fleet is expanded rather rapidly, then Pu would be available for many more FRs. Likewise, if CRs of near one are selected many more reactors could be

constructed. Variations on parameters that relate to policy will be thoroughly investigated next quarter.

Table 5-1: Specifications for Light Water Reactors used in the DANESS fuel cycle simulations.

Power (Electric)	900 MWe
Thermal Efficiency	34%
Load Factor	90%
Reactor Lifetime	100 Years
Fuel Burn-Up	50 GWd/tHM
Cycle Length	12 Months
Number of Batches	5
Initial Uranium	1 t/tHM
Initial Enrichment	4.7%
Spent Uranium	.93545 t/tHM
Spent Enrichment	.82%
Spent Pu	.012 t/tHM
Spent MA	.00184 t/tHM
Spent Fission Products	.0513 t/tHM

Table 5-2: Specifications for Fast Reactor (FR) used in the DANESS fuel cycle simulations. All FRs have conversion ratios of 0.5.

Power (Electric)	1500 MWe
Thermal Efficiency	42.425%
Load Factor	76%
Reactor Lifetime	100 Years
Fuel Burn-Up	136 GWd/tHM
Cycle Length	14.6979 Months
Number of Batches	5
Initial Depleted Uranium	.7452 t/tHM
Initial Enrichment	.25%
Initial Pu	.25 t/tHM
Initial MA	.004794 t/tHM
Spent Uranium	.6374 t/tHM
Spent Enrichment	.09844%
Spent Pu	.2148 t/tHM
Spent MA	.009576 t/tHM
Spent Np	.0005577 t/tHM
Spent Am	.007231 t/tHM
Spent Cm	.001788 t/tHM
Spent Fission Products	.1381 t/tHM

Table 5-3: Absolute Inventories of Pu and MA at 2100 with the specification that DANESS construct one FR per year. Due to constraints on fuel inventories specified for DANESS, the number of FRs actually built is limited as shown in parentheses with the year of initial construction.

	Pu Inventories			MA Inventories		
FR Growth Act Initial Pu	In-Pile	Out-Of-Pile	Total	In-Pile	Out-Of-Pile	Total
Reference(No Growth)	55	2865	2920	8	438	446
2020 (60)	680	1309	1988	29	532	561
2030 (65)	733	1379	2113	31	511	542
2040 (60)	687	1592	2279	29	491	520
FR Growth No Initial Pu						
Reference(No Growth)	55	2203	2258	8	337	346
2020 (37)	436	1149	1585	20	414	434
2030 (45)	521	1072	1593	23	407	430
2040 (53)	605	1055	1661	26	392	418

Table 5-4: Relative Inventories of Pu and MA at 2100 with the specification that DANESS construct one FR per year. Due to constraints on fuel inventories specified for DANESS, the number of FRs actually built is limited as shown in parentheses with the year of initial construction.

	Pu Inventories			MA Inventories		
FR Growth Act Initial Pu	In-Pile	Out-Of-Pile	Total	In-Pile	Out-Of-Pile	Total
Reference	1.00	1.00	1.00	1.00	1.00	1.00
2020 (60)	12.34	0.46	0.68	3.41	1.22	1.26
2030 (65)	13.31	0.48	0.72	3.64	1.17	1.21
2040 (60)	12.48	0.56	0.78	3.43	1.12	1.17
FR Growth No Initial Pu						
Reference(No Growth)	1.00	1.00	1.00	1.00	1.00	1.00
2020 (37)	7.91	0.52	0.70	2.41	1.23	1.26
2030 (45)	9.45	0.49	0.71	2.76	1.21	1.24
2040 (53)	10.99	0.48	0.74	3.10	1.16	1.21

Table 5-5: Absolute Inventories of Pu and MA at 2100 with the specification that DANESS construct three FRs per year. Due to constraints on fuel inventories specified for DANESS, the number of FRs actually built is limited as shown in parentheses with the year of initial construction.

	Pu Inventories			MA Inventories		
FR Growth Act Initial Pu	In-Pile	Out-Of-Pile	Total	In-Pile	Out-Of-Pile	Total
Reference(No Growth)	55	2865	2920	8	438	446
2020 (59)	662	1196	1858	27	585	613
2030 (71)	786	1104	1890	31	582	613
2040 (76)	839	1079	1917	33	564	597
FR Growth No Initial Pu						
Reference(No Growth)	55	2203	2258	8	337	346
2020 (35)	428	1327	1754	20	429	449
2030 (40)	475	1193	1668	21	430	451
2040 (51)	579	1023	1602	25	432	597

Table 5-6: Relative Inventories of Pu and MA at 2100 with the specification that DANESS construct three FRs per year. Due to constraints on fuel inventories specified for DANESS, the number of FRs actually built is limited as shown in parentheses with the year of initial construction.

	Pu Inventories			MA Inventories		
FR Growth Act Initial Pu	In-Pile	Out-Of-Pile	Total	In-Pile	Out-Of-Pile	Total
Reference	1.00	1.00	1.00	1.00	1.00	1.00
2020 (59)	12.01	0.42	0.64	3.23	1.34	1.37
2030 (71)	14.27	0.39	0.65	3.70	1.33	1.38
2040 (76)	15.22	0.38	0.66	3.91	1.29	1.34
FR Growth No Initial Pu						
Reference(No Growth)	1.00	1.00	1.00	1.00	1.00	1.00
2020 (35)	7.76	0.60	0.78	2.37	1.27	1.30
2030 (40)	8.63	0.54	0.74	2.54	1.27	1.30
2040 (51)	10.51	0.46	0.71	2.92	1.28	1.73

Table 5-7: Absolute Inventories of Pu and MA at 2100 with the specification that DANESS construct five FRs per year. Due to constraints on fuel inventories specified for DANESS, the number of FRs actually built is limited as shown in parentheses with the year of initial construction.

	Pu Inventories			MA Inventories		
FR Growth Act Initial Pu	In-Pile	Out-Of-Pile	Total	In-Pile	Out-Of-Pile	Total
Reference(No Growth)	55	2865	2920	8	438	446
2020 (64)	734	1215	1949	30	606	635
2030 (71)	802	1379	2181	32	599	631
2040 (79)	869	1592	2460	34	589	623
FR Growth No Initial Pu						
Reference(No Growth)	55	2203	2258	8	337	346
2020 (50)	581	1149	1729	25	461	486
2030 (58)	664	1072	1736	27	462	489
2040 (60)	691	1055	1746	28	449	478

Table 5-8: Relative Inventories of Pu and MA at 2100 with the specification that DANESS construct five FRs per year. Due to constraints on fuel inventories specified for DANESS, the number of FRs actually built is limited as shown in parentheses with the year of initial construction.

	Pu Inventories			MA Inventories		
FR Growth Act Initial Pu	In-Pile	Out-Of-Pile	Total	In-Pile	Out-Of-Pile	Total
Reference	1.00	1.00	1.00	1.00	1.00	1.00
2020 (64)	13.32	0.46	0.67	3.51	1.38	1.42
2030 (71)	14.55	0.48	0.75	3.76	1.37	1.42
2040 (79)	15.77	0.56	0.84	4.01	1.35	1.40
FR Growth No Initial Pu						
Reference(No Growth)	1.00	1.00	1.00	1.00	1.00	1.00
2020 (50)	10.54	0.52	0.77	2.92	1.37	1.40
2030 (58)	12.05	0.49	0.77	3.23	1.37	1.41
2040 (60)	12.53	0.48	0.77	3.34	1.33	1.38

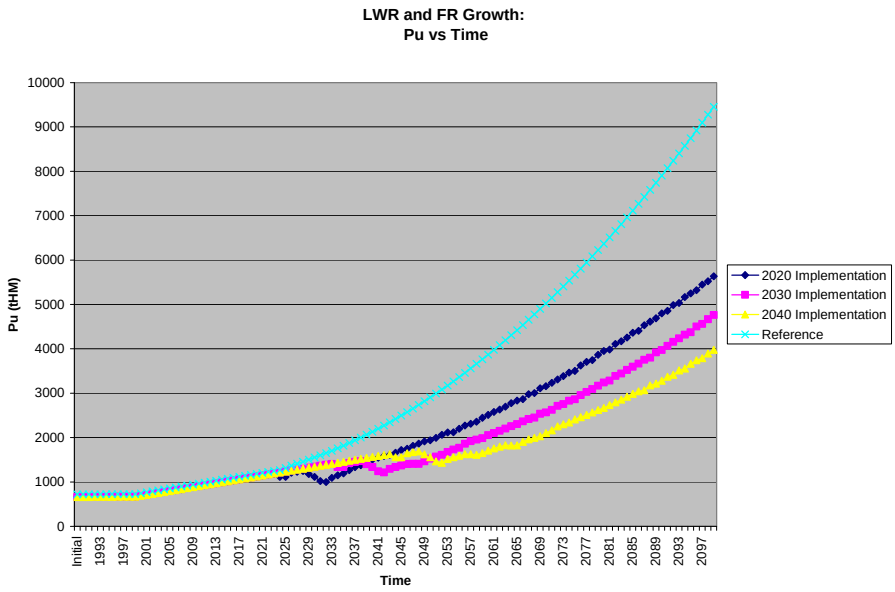

Figure 5-4: Total Amount of Pu vs. Time: LWR and FR Growth Case: 8Rx/Yr (The Reference run grows 8 LWRs per year and the other runs build 6 LWRs and 2 FRs per Year)

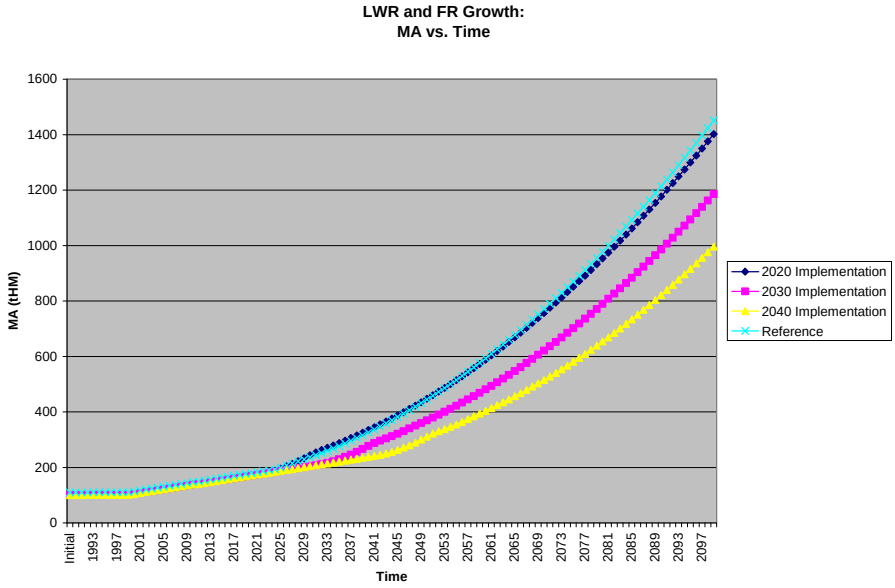

Figure 5-5: Total Amount of MA vs. Time: LWR and FR Growth Case: 8Rx/Yr (The Reference run grows 8 LWRs per year and the other runs build 6 LWRs and 2 FRs per Year)

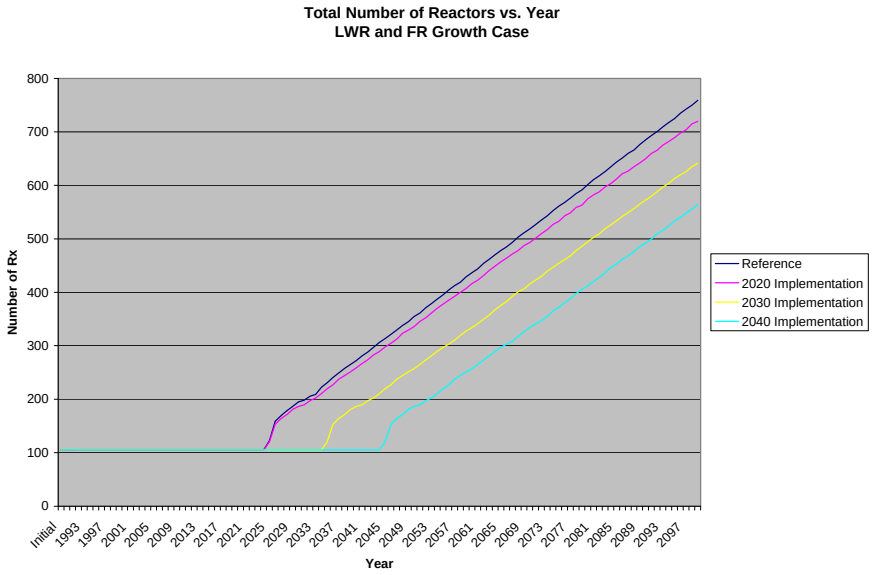

Figure 5-6: Total Number of Reactors vs. Year: LWR and FR Growth Case: 8Rx/Yr (The Reference run grows 8 LWRs per year and the other runs build 6 LWRs and 2 FRs per Year)

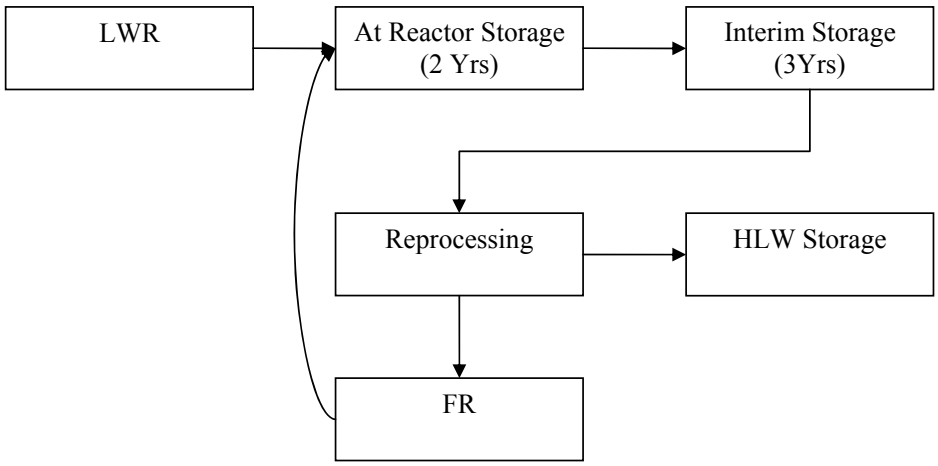

Figure 5-7: DANESS model for Fast reactor calculations

Table 5-9: Absolute Inventories of Pu and MA at 2100: 8Rx/Yr (The Reference run grows 8 LWRs per Year and the other runs build 6 LWRs and 2 FRs per Year)

	Pu Inventories			MA Inventories		
LWR & FR Growth Actual Initial Pu	In-Pile	Out-Of-Pile	Total	In-Pile	Out-Of-Pile	Total
Reference	442	9011	9453	67	1383	1451
2020 Implementation	1775	3860	5636	96	1305	1402
2030 Implementation	1545	3215	4761	85	1100	1185
2040 Implementation	1364	2612	3977	74	921	996

Table 5-10: Relative Inventories as Ratios: 8Rx/Yr (Ref 8 LWRs: others 6 LWRs and 2 FRs)

	Pu Inventories			MA Inventories		
LWR & FR Growth Actual Initial Pu	In-Pile	Out-Of-Pile	Total	In-Pile	Out-Of-Pile	Total
Reference	1.00	1.00	1.00	1.00	1.00	1.00
2020 Implementation	4.02	0.43	0.60	1.42	0.94	0.97
2030 Implementation	3.50	0.36	0.50	1.26	0.80	0.82
2040 Implementation	3.09	0.29	0.42	1.11	0.67	0.69

Table 5-11: Actinides In Spent Fuel at 2100: No Initial SF: Units tHM

1Rx/Yr : Units tHM	SF Tot	U	Pu	MA	Np	Am	Cm	FP
Reference(No Growth)	9393	8787	113	17	0	0	0	482
2020 (37)	10762	9659	412	30	1	10	2	671
2030 (45)	11058	9848	476	33	1	12	3	712
2040 (53)	11327	10019	535	36	1	14	3	749
3Rx/Yr								
2020 (35)	10688	9612	395	30	1	9	2	661
2030 (40)	10873	9730	436	31	1	11	3	686
2040 (51)	11280	9989	525	35	1	14	3	742
5Rx/Yr								
2020 (50)	11243	9966	517	35	1	13	3	737
2030 (58)	11539	10154	581	38	1	16	4	778
2040 (60)	11612	10202	597	39	1	16	4	788

Table 5-12: Actinides In Spent Fuel at 2100: Actual Initial SF: Units tHM

1Rx/Yr : Units tHM	SF Tot	U	Pu	MA	Np	Am	Cm	FP
Reference(No Growth)	9393	8787	113	17	0	0	0	482
2020 (60)	11612	10202	597	39	1	16	4	788
2030 (65)	11760	10296	630	40	1	17	4	809
2040 (60)	11492	10124	571	37	1	15	4	772
3Rx/Yr								
2020 (59)	11576	10178	589	38	1	16	4	783
2030 (71)	12019	10461	686	42	1	19	5	845
2040 (76)	12204	10579	727	44	2	20	5	870
5Rx/Yr								
2020 (64)	11760	10296	630	40	1	17	4	809
2030 (71)	12019	10461	686	42	1	19	5	845
2040 (79)	12315	10649	751	45	2	21	5	885

Table 5-13: Actinides In Spent Fuel at 2100: LWR and FR Growth Case: Units tHM

LWR & FR Growth Actual Initial Pu	SF Tot	U	Pu	MA	Np	Am	Cm	FP
Reference(8LWR/Yr)	72209.83	67548.68	866.52	132.87	0.00	0.00	0.00	3704.36
2020 Implementation	77085.56	70656.48	1931.38	179.56	2.72	35.26	8.72	4377.70
2030 Implementation	76410.81	70226.39	1784.01	173.09	2.34	30.38	7.51	4284.52
2040 Implementation	75989.54	69957.87	1692.01	169.06	2.11	27.33	6.76	4226.34

6. Determination of Decay Heat

Since the DANESS program utilizes a table lookup process to perform uncertainty analysis, it is imperative that the data tables be as accurate as possible. To aid in this effort, ORIGEN-ARP, when used with the specialized burn-up libraries, created data used in an integral decay heat calculation for the repository.

The libraries created for ORIGEN-ARP currently include both PWR and BWR type reactors, as fast reactors will be added in the future. From these libraries, over 50 data sets were completed that included each reactor type and the corresponding MOX configuration. The burn-ups ranged from 30GWd to 100GWd to allow for a good data table range from which to extrapolate the data to fit a desired burn-up for a DANESS run. Since the enrichment varies from 3-5%, each reactor type and burn-up included the 3 enrichments. From this, DANESS will use an extrapolation method between these 3 enrichment levels to reach the desired enrichment for the uncertainty model.

Accompanying the burn-up data in DANESS is a data table for the integral decay heat. ORIGEN-ARP was used to create output in units of watts per basis, where the basis used is 1 metric ton of uranium. A list of 56 actinides and 11 fission products were

supplied by ANL to use as a basis for these calculations. While not a perfect model, the majority of decay heat is represented by these individual isotopes with the rest being accounted for in a separate model discussed later. Usage of the PlotOPUS software in SCALE5 as an ARP utility allowed for pre-selection of the isotopes of interest. This enabled the data to be easily imported into a spreadsheet. With the data entered into the spreadsheet, the integration was performed using the midpoint formula. To achieve greater accuracy with the integration, an iterative method was used to find ideal time steps for the data. The final result yielded time steps at every 2 years from discharge until 200 years, then every 20 years from 200 to 400 years after discharge, every 25 years from 400 to 1100 years after discharge and every 50 years from 1100 to 1500 years after discharge. While time consuming, the variation of time steps allows for a more accurate data table to be entered into DANESS. Sample graphs of the 5 to 1500 year integral heat load in terms of MWd deposited in the repository per GWd burnup can be seen below for fission products, actinides, and a total.

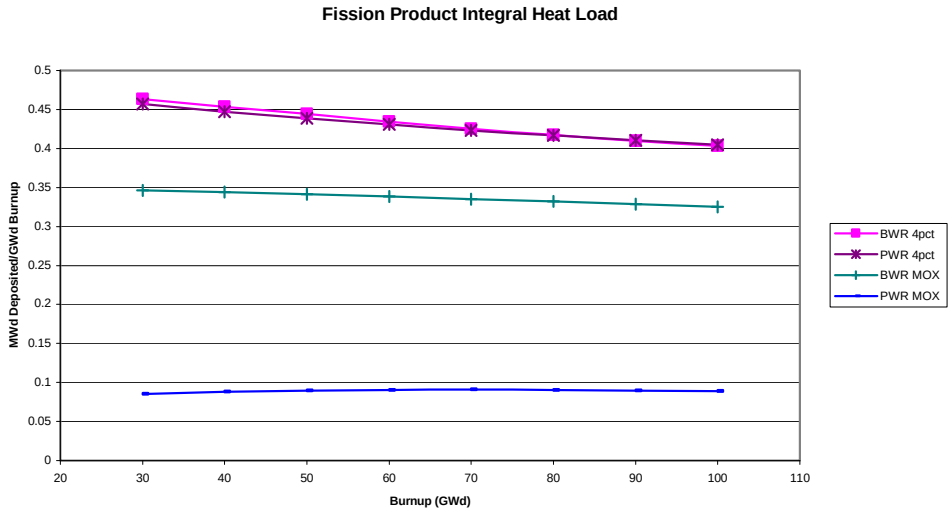

Figure 6-1: Decay heat integrated from 5 to 1500 years after discharge for fission products

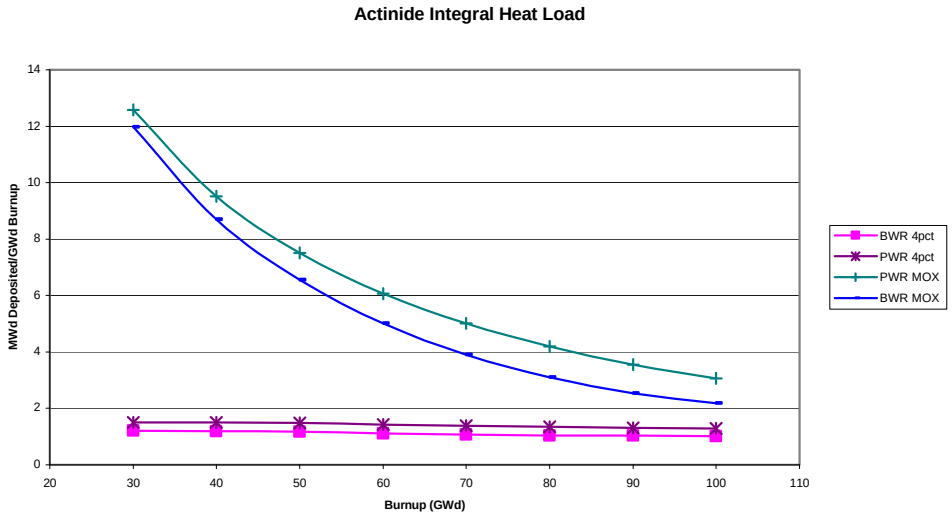

Figure 6-2: Decay heat integrated from 5 to 1500 years after discharge for Actinides

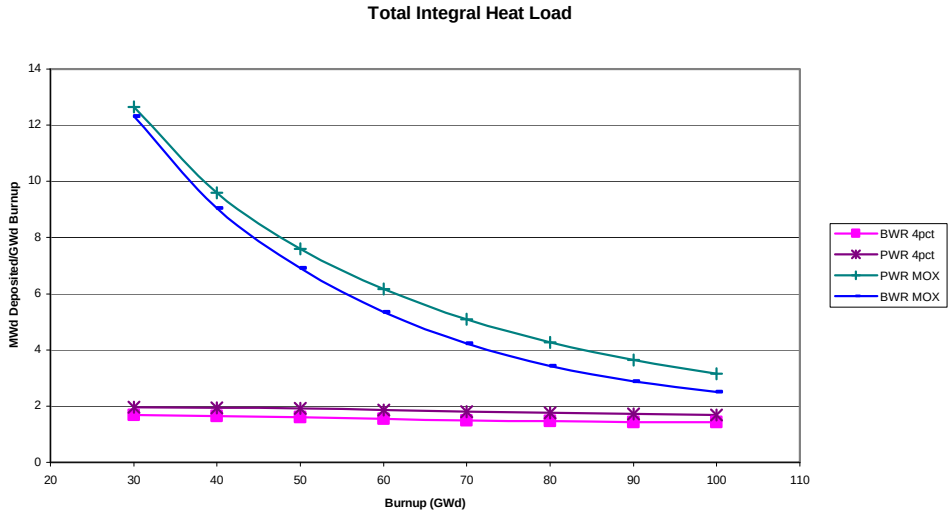

Figure 6-3: Total decay heat integrated from 5 to 1500 years after discharge

As mentioned previously, the list of isotopes supplied by ANL did not represent the total heat load on the repository since contributions from both short lived and long lived fission products are significant. To account for this discrepancy, a double exponential regression was created for each burn-up, enrichment, and reactor type. Data output from ORIGEN contains all isotopes created during the fission process, while ORIGEN-ARP restricts this list to those contributing more than a certain percentage to the total heat. For this case, that cutoff was .001%. From this list, the ANL supplied list

of fission products was removed to give a remainder heat contribution. The remaining isotopes had a time line of 5 to 1500 years after discharge. This list was then separated into short and long lived fission products and further restricted to isotopes contributing on the order of 10^{-6} W. While this low amount is likely excessive, the resulting model for heat decay from 5 to 1500 years was fit using Matlab to perform a double exponential curve fit. The resulting fits all had an R2 of 1 for the chosen isotopes. A sample figure for both SLFP and LLFP fits is shown below.

Fast reactors are likely to play a critical role in managing the inventory of transuranics as well as the heat load of spent fuel in repository. For this reason, a comparison of fast reactor designs is performed that will contribute isotopic data for uses in other portions of this research. A previous study on advanced burner reactors by Hoffman, E. A., et al at Argonne National Laboratory used the Super PRISM design to compare isotopic compositions in spent fuel with conversion ratios of 0, .25, .5, .75, and 1 for both metal and oxide fuels[1,2,3]. The results of the ANL study are compared with results from cores designed in SCALE 5.1.

SAS2, a 1-D point depletion code, is used as the driver for cross section generation and ORIGEN-S[4]. A different geometry is used for each fuel type and ideal conversion ratio, leading towards 10 individual reactor designs. Fuel assembly burn ups range from 80 to 200 GWd/t by steps of 10 GWd/t. The cross section libraries created by the SAS2 procedure are then used by the ARP module to reduce computing time for future analyses. In particular, the ARP module is used to calculate the masses and decay heats for 66 isotopes over a period of 1500 years, similar to the LWR portion of this research.

Parameters used in the design of the UT code are derived from the paper by Dubberly, et al[2]. For the entire core, the metal fuel has a mass of 26,181 kg and the oxide fuel has a mass of 34,914 kg. Power density, power per metric ton of fuel, for the two different types of fuel was calculated using a thermal power of 1000MWth divided by the mass of the fuel in metric tons. Table 1 shows the power densities for each reactor type.

Table 6-1: Power Density

Fuel Type	Metal	Oxide
Reactor power (MWth)	1000	1000
Core mass (kg)	26181	34914
Power Density	38.196	28.642

Differences between the ANL and UT data exist for what are likely a few reasons. Table 2 shows a comparison of the two results for a specific reactor type and burnup. The two codes used are rather different. ANL used DIF3D/REBUS-3 to calculate the isotopic compositions and perform fuel shuffling with the ENDF/B-V.2 cross section set underneath. The SAS2 code used by UT performs no fuel shuffling, thus isotopics are likely to be different. ANL also used multiple fuel batches with different enrichments for

blankets, whereas the 1-D code assumes a single assembly in an infinite medium without any blanket or reflector assemblies. Thus the differences should be at least subtle if not significant. Lastly, the ANL paper did not vary the burn ups as was done by UT, thus differences across the burnup of a single assembly will vary the isotopic composition of fission products and transmutation.

Table 6-2: Comparison of ANL and UT data for CR=1 at 109.9 GWd/t burnup using oxide fuel

	sas2 results	ANL results	Difference	Percent Difference %
Pu	0.121585897	0.167116	0.045530103	31.5412562
Np	0.00164968	0.000894	-0.00075568	-59.41625754
Am	0.005299192	0.003658	-0.001641192	-36.64523997
Cm	0.001763036	0.00174	-2.30362E-05	-1.315215235
U-235	0.000540962	0.000207	-0.000333962	-89.29912069
Pu-238	0.001773077	0.001924	0.000150923	8.164454247
Pu-239	0.073653846	0.103694	0.030040154	33.87710028
Pu-240	0.040467949	0.048982	0.008514051	19.03645872
Pu-241	0.005691026	0.007386	0.001694974	25.92293394
Am-241	0.004144872	0.002069	-0.002075872	-66.8141173
Cs-137	2.53141E-05	0.004799	0.004773686	197.9011232
U-235/U	0.00062518	0.0003	-0.00032518	-70.29548816

To illustrate the differences in isotopic compositions, Figure 2 shows the percent of U-235 with respect to the total amount of Uranium for a range of burnups. SAS2 calculates a higher percentage for the lower burnups while the difference between the UT and ANL results decrease as burnup increases. Figure 3 shows a similar plot for Pu-241.

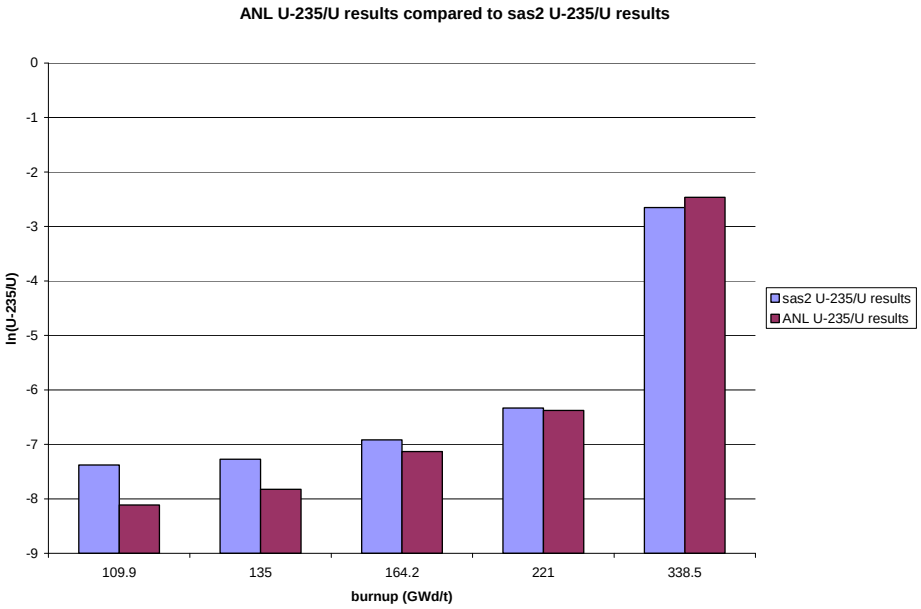

Figure 6-4: Comparison of percent U-235 in total Uranium for range of burnups

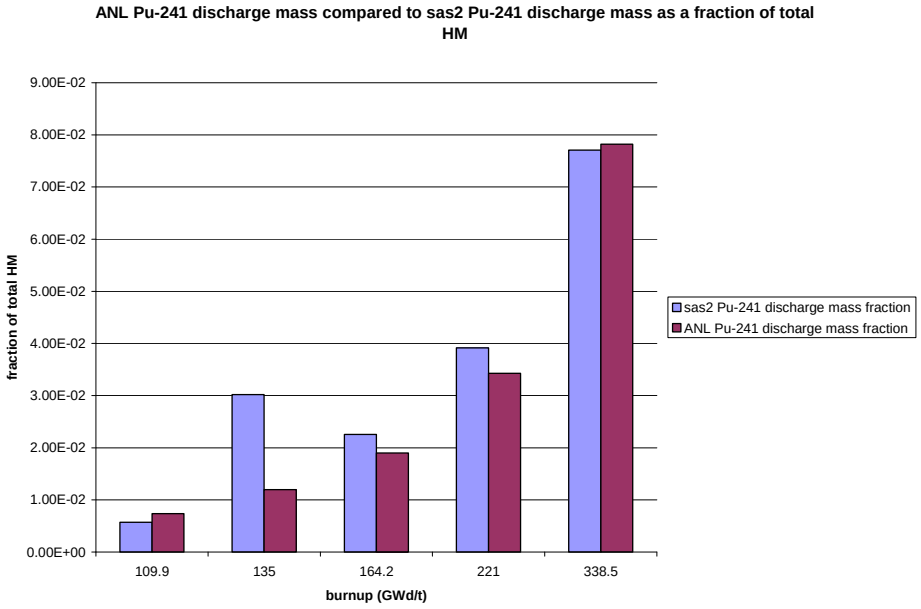

Figure 6-5: Comparison of Pu-241/MTHM for range of burnups

Plutonium concentrations are controlled by the various conversion ratios that are made possible by the different assembly dimensions. In this case, larger assemblies have more U-238, which gives a larger conversion ratio from self-shielding and resonance

effects. The smaller assemblies have lower conversion ratios from the smaller amount of U-238 present. Isotopic product is followed using the PlotOPUS module of SCALE. Figure 4 shows the concentrations of Pu-239 normalized to a burnup of 100 GWd/t for the 5 conversion ratios in the oxide fuel reactor group. In all cases, reactor 1 corresponds to an ideal conversion ratio of 0 with each subsequent reactor corresponding to an increase in conversion ratio of .25 with reactor 5 having an ideal conversion ratio of 1.

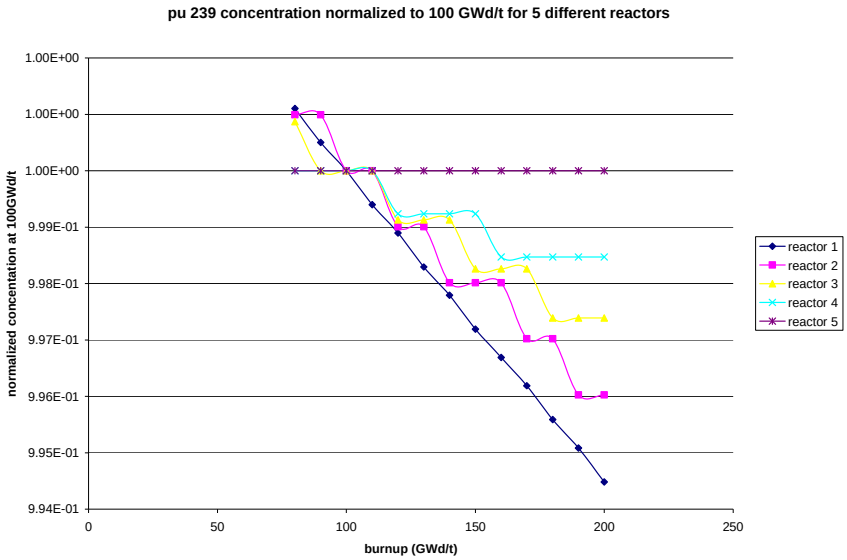

Figure 6-6: Normalized Pu-239 Concentration of oxide fueled reactor

Numerous studies have shown transuranics, specifically plutonium isotopes, are the largest contributors to the long term heat load of spent fuel. Figure 5 shows the decay heats in Watts for various isotopes from 1 MTHM of spent oxide fuel from a reactor with conversion ratio of .5.

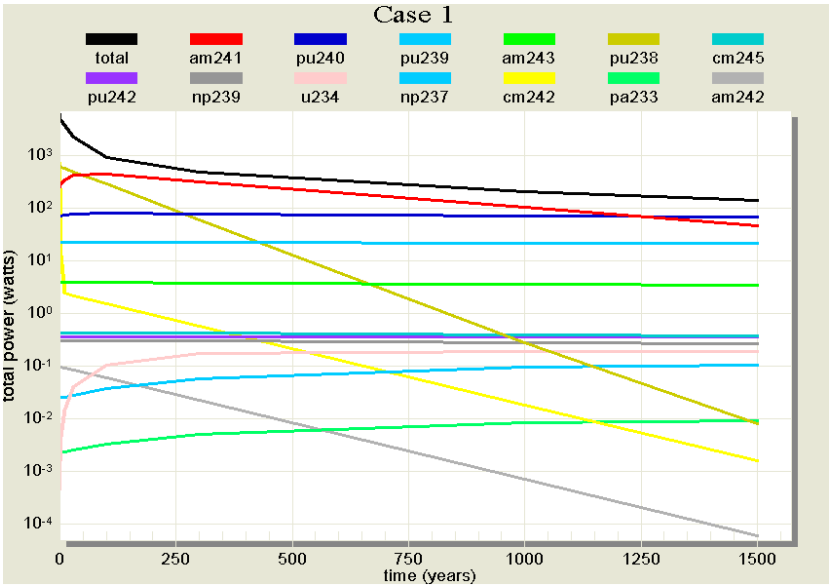

Figure 6-7: Decay heat of 1 MTHM oxide fuel from reactor with CR=.5

Preliminary work using the 1-D infinite lattice code shows some differences in the isotopic compositions; however, it is noted that the slopes of these concentrations nearly match that of the ANL data despite the use of the 1-D code without core shuffling.

7. Expert Elicitation

In an attempt to define distributions for various values in the fuel cycle for the dynamic uncertainty study and the economic one, literature studies were conducted and an expert elicitation survey was drawn up to survey experts in the nuclear field. The survey consisted of 27 multiple choice questions designed to aid in our research that ranged from expected times of implementation of advanced reactors to economics of those advanced fuel cycles. The survey was distributed to top experts in the nuclear field around the country and 17 such surveys were completed. The questions and there corresponding percentage of responses are included in Appendix A.

The respondents answers seem to correspond relatively well with our literature research and those that have been used thus far while conducting the work. The vast majority of respondents seem to agree on the time of implementation of new reactors needs to be soon, 94% say by the year 2010, and 76% say that this will happen. When asked about how many reactors will be constructed per year the range is spot on for the value used in this work; 44% say 1 Rx/yr, 38% say 3 Rx/yr, and 13% say 7 Rx/yr. This is almost exactly the distribution assigned to new LWR growth for the uncertainty analysis. As for the burn up of the LWR fuel 100% of respondents say that within the next 20yrs we will see burn ups of between 50-75 GWd/ton. Again this is spot on for the distribution of burn ups used. And 38% say we could see 90 GWd/ton burn ups. This is slightly lower

than the high burn up value used. Such consensus seems to break down when we get to the economic questions. This is most likely because the economics of advanced fuel cycles is not widely known or studied and thus may not be uniform throughout the nuclear community.

The questions and there corresponding percentage of responses are included in Appendix B.

Assessment of Survey Results

There is a strong consensus that new construction in the US should (94 %) begin by 2010 and a very positive attitude that it will (74 %) by 2010; whereas, internationally the corresponding numbers are 94 % and 100 %. As for the rate of construction in the US, the respondents expect one to three reactors to be built per year. For all other countries the expectation for "will" be built is relatively flat in the range of 5 to 20, and for "should" be built, the corresponding relatively-flat-range is 10 to 30.

The expectations for increasing burnup of LWR fuel is relatively consistent with experts with whom we have had personal communications, which is that the technology limit is about 75 GWd/ton. Survey results are that 50 % expect 60 GWd/ton will be achieved and 44 % expect that 75 GWd/ton will be achieved. None of the respondents expect that more than 75 GWd/ton will be readily obtained.

Most of the respondents (60 %) are of the opinion that LWR fuel will be cooled for 50 years (most probably assuming it will be reprocessed) before it is reprocessed, while none thought it would be cooled more than 100 years. Surprisingly, 56 % believe that it should only be cooled for 5 years, since based on results from our studies; we believe that it should be cooled about 100 years and that it will be cooled about 50 years. Respondents are generally of the opinion that reprocessing of LWR fuel will begin between 2030 (35 %) and 2040 (41 %), while 65% cited 2020 as the time it should begin. Our opinion is that it should and will occur within the 2030 to 2040 time frame.

As for when fast reactors should and will be built relative to initiation of reprocessing, there is a very weak preference for 10 to 20 years after it is implemented, which is consistent with our opinion. Straightforward calculations indicate that about 2,000 tons of spent fuel will be produced each year (depending on the assumption for burnup); thus, in order to provide some allowance most (56 %) believe that a 3,000 ton/yr reprocessing plant is required. There is good consistency regarding the size of a reprocessing plant that will support the operation of 100 LWRs.

There is a weak indication of preference for conversion ratios (CRs) for fast reactors (FRs), which is to operate with CR greater than one. However, there is a strong consensus (63 %) that it will be 0.75. This is consistent with the assumption that FRs will be operated primarily for the purpose of burning transuranics. It is our understanding that it may not be technologically feasible to operate FRs with CRs much less than 0.75 due primarily to reactivity control issues. It is also interesting to note that

the highest frequencies of responses for burnups corresponding to selected CRs is consistent with calculations reported in an Argonne National Laboratory Report.

Most of the respondents (75 %) indicated that mixed oxide fuel (MOX) should be used in LWRs. Our assessments imply that this is counterproductive since MOX fuel increases the decay heat load on the repository and since use of MOX fuel results in the most expensive fuel cycle considered. Fifty-six percent were of the opinion that the burnup achievable with MOX and UOX fuel is the same, which is consistent with our understanding of fuel performance. The expected fraction of MOX fuel in any particular is 0.3 (50 %) to 0.5 (36 %). It is our understanding that the French typically use about one-third MOX fuel in LWRs.

Forty-seven percent of the responds selected twenty billion dollars as the expected cost of a 2,000 tHM/yr PUREX reprocessing plant. Twenty percent selected ten billion and 13 % selected 40 and 50 billion each. The 20 billion dollar cost is often cited in presentations and seems to be a commonly accepted number even though we are of the opinion that no design has progressed to the point where a reliable cost estimate could be made. If additional processing steps are added to achieve a factor 30 reduction in heat load on the repository, 47 % expected the cost to increase by 30 %, 27 % expected it to be the same and 13 % expected it to cost 50 % more.

The final cost of disposing of spent fuel in Yucca Mountain is estimated to be more than a factor of two greater than revenue generated by the 1 mil/kW-hr waste management fee, while 69 % expect that dry cask storage for 100 years would be less or equal to 50 % of the waste management fee. About the same fraction of respondents expect that the cost interim storage for 100 years would be about one-half of the waste management fee.

8. Uncertainty Analysis of a Pu/U Equilibrium Fuel Cycle

Various parameters in DANESS have significantly different influence on results of calculations. In some cases, such as burnup, an input parameter may be input as continuously varying, but data used for obtaining results are pre-calculated based on discrete values. Thus, any value may be chosen as input, but the results are based on values associated with specific pre-calculated values. As a result, it is somewhat impractical to perform uncertainty assessments by well-established Monte Carlo, or Latin Hypercube, sampling methods with DANESS. Instead of using formal uncertainty analysis methods with DANESS, scenarios are run for feasible fuel cycles, and the range of resultant values is analyzed to obtain uncertainties of selected outcomes. Nevertheless, the current version (December, 2008) has been upgraded to perform useful uncertainty analyses.

Parameters associated with fuel cycles can be generally classified in the following categories: 1) physics, 2) economic, and 3) decision. If one limits consideration to an equilibrium fuel cycle model, it is straightforward to write material balances for various fuel cycle facilities and to solve algebraic equations for physics-related results of interest.

In order to conduct uncertainty analyses for equilibrium fuel cycle models it is necessary to specify probability density functions for parameters associated with the model and to solve equations for outcomes of interest. Some results of particular interest are as follows:

1) Energy deposition in the repository during a 1,500 year residence based on spent fuel and for cases with some isotopes removed,
2) Energy deposition in the repository during a 1,500 year residence for selected isotopes,
3) The amount of spent fuel generated,
4) The amount of natural uranium feed required,
5) Plutonium in spent fuel,
6) Short-lived fission products, and
7) Long-lived fission products.

Results for these outcomes can be obtained for specific models, such as the fuel cycle model shown in Figure 4, which considers the following facilities:

1) enrichment,
2) fuel fabrication,
3) reactor, and
4) reprocessing.

Material balances are written for total uranium, U-235 and Pu-239, and are based on one year. The model parameters are the following:

1) fraction of U-235 in the feed to the enrichment plant,
2) fraction of U-235 in the tails stream from the enrichment plant,
3) fraction of U-235 product stream from the enrichment plant,
4) fraction of Pu in spent fuel,
5) effectiveness of Pu in causing fission relative to U-235,
6) fraction of U-235 in makeup uranium for the fuel fabrication plant,
7) burnup of spent fuel, and
8) thermal energy produced per year.

Assumptions for the current model include the following:

1) no losses in any fuel cycle facility,
2) the enrichment of the product stream from the enrichment plant is equal to that required for new fuel,
3) natural uranium feed to the enrichment plant, and
4) depleted uranium for makeup feed to the fuel fabrication plant.

Derivation of Equilibrium Fuel Cycle Equations with Plutonium Recycle

Algebraic equations that characterize material flows in an equilibrium fuel cycle are obtained by writing material balances for each facility of interest in the fuel cycle. Equation (1) is a total material balance for the enrichment and Eq. (2) is an isotopic balance for U235.

$$F_{NU} = T + P_e \tag{1}$$

$$\alpha_1 F_{NU} = \alpha_2 T + \alpha_3 P_e \tag{2}$$

α_1 = fraction of U-235 in the feed stream to the enrichment plant
α_2 = fraction of U-235 in the tails stream from the enrichment plant
α_3 = fraction of U-235 in the product stream from the enrichment plant
P_e =quantity of material in the enrichment stream
T = quantity of material in the tails stream
F_{NU} = quantity of material in the feed stream

The quantity of spent fuel (energy/burnup)

$$F_{sf} = \left(\frac{E}{BU} \right) \tag{3}$$

E = thermal energy released from fission in one year (GWd/T)
BU=burnup of fuel (GWd/T)
F_{sf} = quantity of spent fuel

The quantity of Pu in spent fuel is estimated from a physics parameter that defines the fraction of Pu in spent fuel as follows,

$$F_{Pu} = \alpha_4 F_{sf} \tag{4}$$

α_4 = fraction of Pu in spent fuel
F_{Pu} = quantity of Pu in spent fuel

The following two equations are material balances for the fuel fabrication plant for the total material and for the fissile component.

$$P_e + F_{MU} + F_{Pu} = F_{sf} \tag{5}$$

$$\alpha_3 P_e + \alpha_6 F_{MU} + \alpha_5 F_{Pu} = \alpha_3 F_{sf} \tag{6}$$

α_5 = compensation for Pu being less effective in causing fission in LWRs than U-235
α_6 = fraction of U-235 in the makeup uranium stream to the fuel fabrication plant

The following equations are intermediate steps for solutions of Eqs. (1-6)

$$\alpha_1 F_{NU} = \alpha_2 \left(F_{NU} - P_e \right) + \alpha_3 P_e \tag{7}$$

$$F_{NU} \left(\alpha_1 - \alpha_2 \right) = P_e \left(\alpha_3 - \alpha_2 \right) \tag{8}$$

$$F_{NU} \frac{\left(\alpha_1 - \alpha_2 \right)}{\left(\alpha_3 - \alpha_2 \right)} + F_{MU} + \alpha_4 \frac{E}{BU} = \frac{E}{BU} \tag{9}$$

$$F_{MU} = \frac{E}{BU} \left(1 - \alpha_4 \right) + F_{NU} \left(\frac{\alpha_2 - \alpha_1}{\alpha_2 - \alpha_3} \right) \tag{10}$$

$$F_{NU} \alpha_3 \left(\frac{\alpha_1 - \alpha_2}{\alpha_3 - \alpha_2} \right) + \alpha_6 F_{MU} + \left(\alpha_5 \right) \alpha_4 \left(\frac{E}{BU} \right) = \alpha_3 \left(\frac{E}{BU} \right) \tag{11}$$

$$F_{NU} \alpha_3 \frac{\left(\alpha_1 - \alpha_2 \right)}{\left(\alpha_3 - \alpha_2 \right)} + \alpha_6 \left[\frac{E}{BU} \left(1 - \alpha_4 \right) + F_{NU} \left(\frac{\alpha_2 - \alpha_1}{\alpha_2 - \alpha_3} \right) \right] = \frac{E}{BU} \left(\alpha_3 - \alpha_4 \alpha_5 \right) \tag{12}$$

and

$$F_{NU} \left[\frac{\left(\alpha_3 - \alpha_6 \right) \left(\alpha_1 - \alpha_2 \right)}{\alpha_3 - \alpha_2} \right] = \frac{E}{BU} \left(\alpha_3 - \alpha_4 \alpha_5 - \alpha_6 + \alpha_4 \alpha_6 \right) \tag{13}$$

Table 8-1: Values and distribution of values used in the fuel cycle model.

Description of Parameter	Nominal Value	Expected Range	Distribution
Fraction of U235 in feed to enrichment plant	0.007	none	fixed value
Fraction of U235 in tails from enrichment plant	0.002	0.002-0.003	uniform
Fraction of U235 in product from enrichment plant	0.04	0.03-0.05	uniform
Fraction of Pu in spent fuel	0.007	0.007-0.008	uniform
Effectiveness of Pu in causing fission relative to U-235	0.8	none	fixed value
Fraction of U235 in makeup uranium for fuel fabrication	0.0025	0.002-0.003	uniform
Burnup of spent fuel (GWth-d)	50	40-70	triangular
Thermal energy produced per year (GWth-d)	1000	none	fixed value

Comments on Selection of Parameter Values

Burnup of spent fuel (GWth-d)

The energy yield from Light Water Reactor (LWR) fuel (Burnup) has generally increased during the past forty years. Initially (1960), the energy yield per metric ton of fuel (burnup) was about 30 GWd/T, whereas today (2006), some fuel assemblies achieve about twice this burnup. It is expected that essentially all fuel will achieve a burnup of at least 40 GWd/T and that very little will exceed 70 GWd/T. Thus a triangular distribution with a minimum of 40 and maximum of 70 GWd/T with a most probable of 50 GWd/T is chosen for results illustrated in this report. Given that fuel detailed information on the burnup is not currently compiled, a uniform distribution of burnups that range from about 40 to 70 GWd/T would also be a good choice.

Fraction of U235 in feed to enrichment plant

It is assumed that only natural uranium is used for feed to the enrichment plant. Thus, this parameter is set at 0.007.

Fraction of U235 in tails from enrichment plant

The optimal value for the enrichment of U235 in the tails stream from a gaseous diffusion plant is 0.002; however, increased throughput can be obtained if higher fractions of U235 remain in the waste stream. A uniform distribution with minimum and maximum values or 0.002 and 0.003 is used.

Fraction of U235 in product from enrichment plant

The maximum enrichment of U235 from an enrichment plant is 0.05, and it is unusual for new fuel assemblies to utilize uranium enriched to less than 0.03. Since no distributional information is obtained to date, a uniform distribution is used with minimum and maximum of 0.03 and 0.05.

Fraction of Pu in spent fuel

The fraction of Pu in spent fuel varies as a function of burnup, spectral characteristics of the reactor, enrichment, and general fuel management methods. References indicate that the fraction of Pu in spent fuel ranges from about 0.007 to 0.008 for typical LWRs. A uniform distribution over this range is used to represent this parameter.

Effectiveness of Pu in causing fission relative to U-235

The capture-to-fission ratio for Pu239 in thermal reactors is about 0.36 and for U235 it is about 0.18. Thus the likelihood of Pu239 nucleus undergoing fission relative to a U235 nucleus in a thermal reactor is about 0.8, which is the value chosen for this parameter.

Fraction of U235 in makeup uranium for fuel fabrication

It is assumed that depleted uranium is used and that the enrichment ranges from 0.002 to 0.003 as a uniform distribution.

Thermal energy produced per year (GWth-d)

A fixed value is chosen for a 1 GWe reactor that operates at 33 % thermal efficiency for 330 days per year.

Uncertainty Analyses for Equilibrium Fuel Cycles with Simulation Software

The uncertainty analysis described above was first performed with Crystal Ball, an overlaying program for Excel, designed for qualifying the effects of parameter variations on some output. The following results are obtained by using simulation software, Analytica by Lumina Decision Systems, Inc., developed to perform uncertainty analyses on algebraic equations. Analytica uses "Intelligent Arrays" to provide great flexibility in managing multiple dimensions (e.g., isotope). The user can easily add or subtract dimensions without the major surgery required by a spreadsheet. Changes to the dimensions of input arrays propagate through the model automatically without requiring any manual changes to downstream formulas. Each multidimensional table needs a single definition (formula), rather than one for each cell. Analytica estimates desired endpoints using probability distributions to represent uncertainties, and efficient Monte Carlo and Latin hypercube simulation to compute their implications. It also offers importance analysis to identify which uncertainties have the most impact on the results.

The diagram based modeling language starts with a standard flow chart model where the user selects variables and creates links between them using arrows and function boxes where equations or other relationships can be defined. The general Analytica model is shown in Figure 7, and selected submodels are illustrated in Figures 8 and 9. Some results are provided in Figures 10-12.

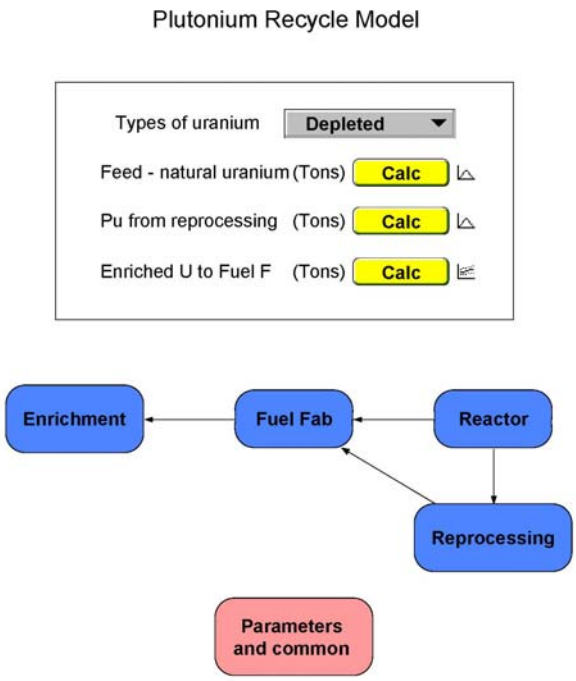

Figure 8-1: General block diagram of the Analytica model (the direction of arrows is not relevant to the calculations)

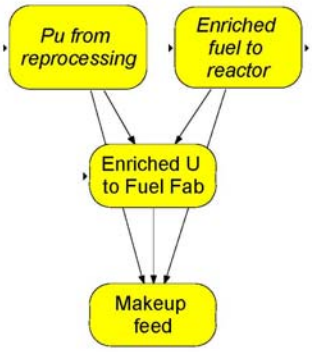

Figure 8-2: Block diagram of the fuel fabrication submodel

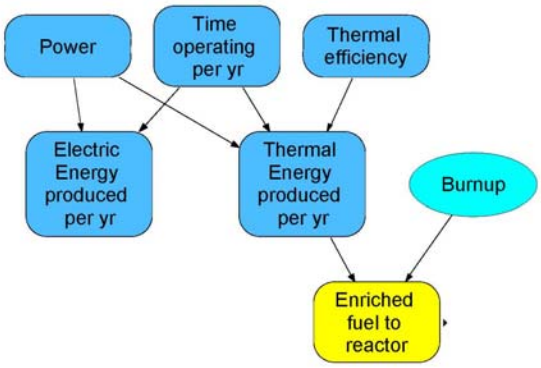

Figure 8-3: Block diagram of the reactor submodel.

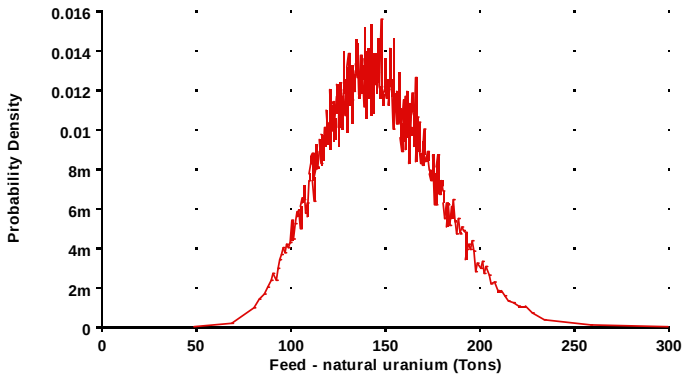

Figure 8-4: Natural Uranium Feed for a Single Reactor

Pu from Reprocessing

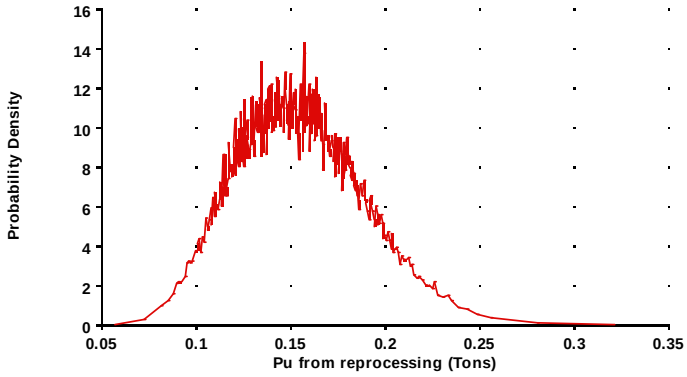

Figure 8-5: Expected mass of Plutonium from reprocessing fuel from a single reactor

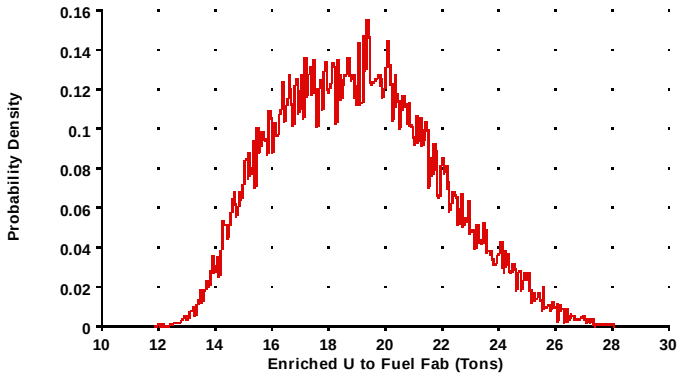

Figure 8-6: Requirement of Enriched Uranium Feed for Fuel Fabrication for a Single Reactor

9. Uncertainty Analyses for Dynamic Fuel Cycles

In addition to the static approach used a more realistic model is also needed. For this DANESS, Dynamic Analysis of Nuclear Energy Systems, a dynamic modeling code developed at Argonne National Labs, is utilized to model various fuel cycle options and obtain monte carlo uncertainties. In the previous quarter the implementation of FRs was analyzed and CDF plots were generated. This quarter's work has expanded on that work to include the implementation of high burn up LWRs, MOX reactors, and mixed fleets scenarios and the comparison of these various scenarios.

Set-Up and Methodology

DANESS offers a multitude of input parameter options. Through literature and expert solicitation a sample of eight of these parameters were selected to be monte carlo sampled. These eight parameters were chosen because of their importance to the overall fuel cycle scheme. The parameters selected were given triangular distributions according to the following tables below for each scenario analyzed.

Table 9-1: Input Variables, Ranges, and Nominal Values: FR Implementation

Input Variable	Range	Nominal Value
Growth rate of reactors LWRs	0 to 7	3
Growth rate of reactors FRs	0 to 3	2
SF At-Rx Cooling Time	1 to 10	3
SF Interim Cooling Time	1 to 10	3

Year of Implementation of New LWR Construction	2015 to 2050	2030
Year of Implementation Reprocessing	2015 to 2030	2020
Year of Implementation of New FR Construction	2030 to 2060	2040
LWR Burnup	40 to 70	50
FR Burnup	80 to 200	120

Table 9-2: Input Variables, Ranges, and Nominal Values: MOX Rx Implementation

Input Variable	**Range**	**Nominal Value**
Growth rate of reactors LWRs	0 to 7	3
Growth rate of reactors MOX	0 to 3	2
SF At-Rx Cooling Time	1 to 10	3
SF Interim Cooling Time	1 to 10	3
Year of Implementation of New LWR Construction	2015 to 2050	2030
Year of Implementation Reprocessing	2015 to 2030	2020
Year of Implementation of New MOX Construction	2030 to 2060	2040
LWR Burn up	40 to 70	50
MOX Burn up	40 to 70	50

Table 9-3: Input Variables, Ranges, and Nominal Values: High Burn up LWR Implementation

Input Variable	**Range**	**Nominal Value**
Growth rate of reactors LWRs	0 to 7	3
Growth rate of reactors High Burn up LWRs	0 to 3	2
SF At-Rx Cooling Time	1 to 10	3
SF Interim Cooling Time	1 to 10	3
Year of Implementation of New LWR Construction	2015 to 2050	2030
Year of Implementation of New High Burn up LWR Construction	2030 to 2060	2040
LWR Burn up	40 to 70	50
LWR High Burn up	75 to 100	87.5

The range values determined for the uncertainty input variables were not chosen randomly. Each variable was researched in order to give as realistic a distribution as possible. The growth rate for the LWRs is set from zero to seven with the nominal value at three reactors per year. This is based on historical data taken during the booming construction of the late sixties and all throughout the seventies it is not unthinkable to believe that this accelerated rate could be again obtained if the need is there.

The growth rate for the FRs is a little more subjective and required a more in depth search since historical data for these types of reactors does not exist. Taking into account the increased initial cost and the unfamiliarity with this type of reactor construction, a considerable reduction in the building capacity of FRs, zero to three, is seen as compared to that of the LWRs, zero to seven. However, the nominal value for the FRs is only one lower at two per year than that of the LWRs at three per year. This is due to the assumed need for an aggressive construction strategy not only to keep up with current power production levels but to expand power production to meet ever increasing needs. The rate for the implementation of the MOX reactors and high burn up reactors is set as the same value as the one used for the FRs. This is due to similar concerns as the FRs and also so that the results can be easily comparable.

The ranges for the two cooling time variables should be a simple matter of looking up how long spent fuel is kept at a reactor and cooled and how long it is kept in interim storage; however, since in the US there is currently no open repository or a operating reprocessing facility these become more problematic variables. Originally the idea was that spent fuel would be stored on-site at reactors in wet storage for a minimum of two years and a maximum of five years and then would be sent to a reprocessing facility or taken control of by the government and stored. This all changed when the US decided to not reprocess spent fuel instead opting for storage in a repository. This decision changed the amount of time that spent fuel would be stored at a reactor and introduced the need for interim storage. This provides for the reasoning behind the ranges of the cooling times to be between one and ten years. The nominal values were chosen as three years each for a total of six years of cooling as an attempt to stay true to the original thinking of five years of storage and adding an additional year due to the increased burnup of LWR fuel needing a little additional time to cool.

The variable Year of Implementation of New LWR Construction takes a very conservative look at new LWR reactor construction. The lower bound is set at the year 2015, which would have the construction of new reactors starting in the year 2010 assuming a five year construction time, and the upper bound set at the year 2050 with a very conservative nominal value set at the year 2030. The lower bound allows for the expansion of the LWR fleet in an attempt to keep up with current growths in energy demand. The upper bound would symbolize a reduction in the need for nuclear power and most likely would not even be able to account for current energy demand. And the nominal value represents an attempt to keep the fleet at current energy production and just replace old shutting down reactors with new ones.

The driving force of this research is to look into advanced fuel cycles; therefore, it becomes necessary to implement the use of a reprocessing facility in order to use LWR spent fuel again in the FRs or MOX reactors. This is the main influence on the range of Year of Implementation of Reprocessing variable. The lower bound is set to a very optimistic year of 2015, this representing the absolute earliest that a reprocessing plant could realistically be built in the US. The upper bound is perhaps to some another optimistic year of 2030, this year was chosen in order to allow for FRs and MOX reactors to start to be built in that same year. The nominal value is set at the year 2020, this value coincides with the thought that the US will need to have a better long term view for the control of spent fuel and before any utilities would likely build a FR or MOX reactor they would need to be certain that there will be fuel for them.

The construction of new advanced reactors: FR, MOX, and high burn up LWRs; distribution is set in a way as to assume that as some of the older LWRs begin to shut down they will be replaced with the advanced reactors. Thus the lower bound is set to the year 2030 accounting for the beginning of the decade of considerable LWRs end of life time frame. The upper bound is set to the year 2060 roughly after all of the old LWRs will be shut down. And the nominal value is set at the year 2040 to allow for some LWRs to be shut down and replaced with new LWRs before advanced reactor construction begins.

The burnup variables are given distributions given the best data available. The LWR burnup ranges coincide with relatively standard values in the 40-50 GWd/ton to more advanced burnups in the 70 GWd/ton range. Some research suggest that burnups going all the way up to 100 GWd/ton could perhaps be possible; however, most literature suggests that this high value is a long way off if so and thus it is not considered in the regular LWR distribution rather in the high burn up one. The FR burnup range in taken from the GE research into the Super Prism. Most of their research puts the burnup between the 80-200 GWd/ton range seen here with the most likely conversion ratio burner reactor having a burnup close to 120 GWd/ton. The distribution of the MOX reactor burn ups is set to the same as that of the LWR burn up. This is due to the fact that MOX fuel will most likely follow the same burn up trends that the UOX LWR fuel will follow. The distribution of the high burn up LWR fuel is set as an extremely high value burn up for an LWR ranging between 75 and 100 GWd/ton. This distribution is stretching the extreme limits of what is considered to be feasible for LWR burn ups; however, research in this range is ongoing, thus it is included in this work.

Triangular distributions were chosen for these distributions because of its simplicity and data correlation. The distributions were implemented by the use of a simple MatLab script. This method was used for each variable in the above tables and a list of one hundred input values for each variable were calculated for each scenario.

Results

The list of Monte Carlo sampled input variables was loaded in an Excel sheet and imported into DANESS via the MC Sampling function. The code was then run one hundred times utilizing the batch run capacity in DANESS. The output from these runs is extensive and it was decided to limit the output to areas of interest. Pu inventory was determined to be the most important output; and thus, for this paper Pu inventory will be the only output value discussed. The Pu inventory is reported for in-pile, actually physically in a reactor, and out-of-pile, in reprocessing, spent fuel storage, HLW, etc. Even with the reduction of the output down to only two parameters, a total of one hundred years for one hundred runs yields an enormous amount of data for each scenario. For the purpose of this paper a sample comparison CDF plot taken at the year 2100 for the Pu in-pile and out-of-pile will be representative of the results obtained. These plots are shown below in Figures 1 and 2.

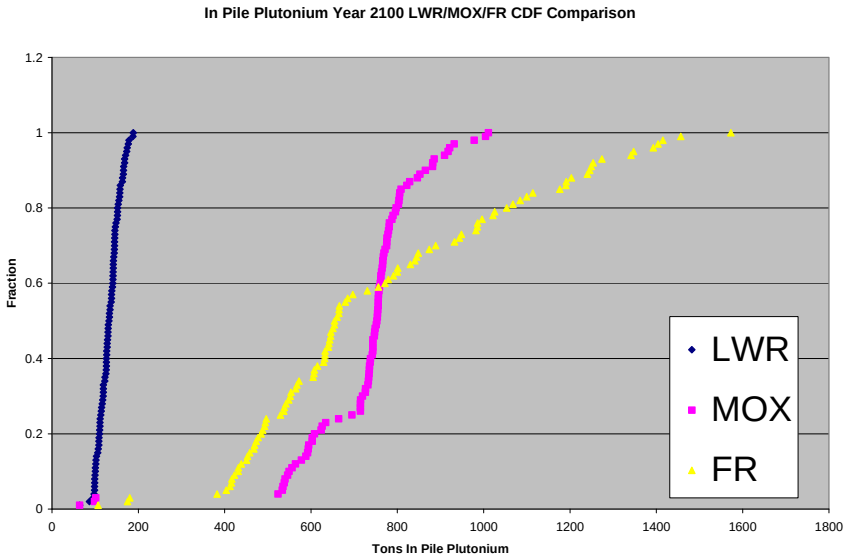

Figure 9-1: Sample Comparison CDF of Results. Pu In-Pile year 2100

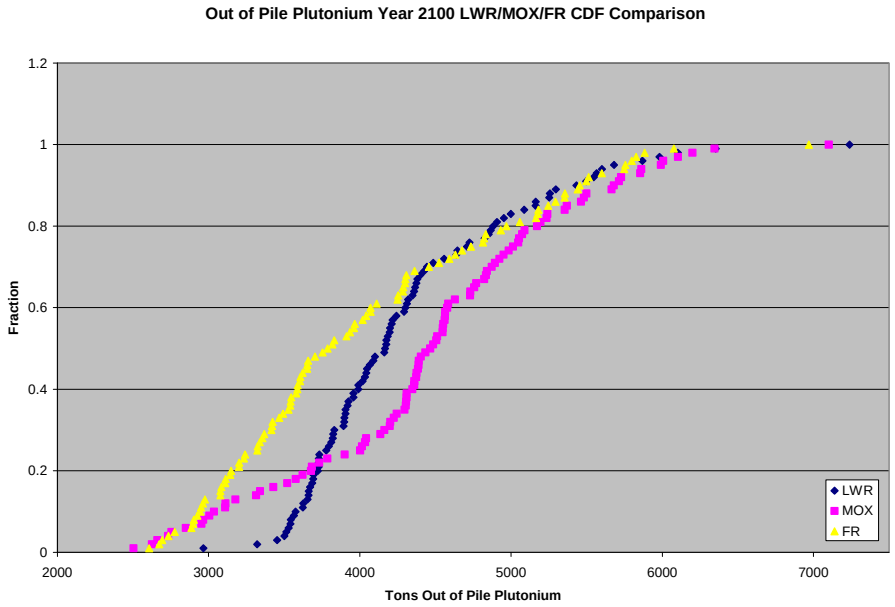

Figure 9-2: Sample Comparison CDF of Results. Pu Out-of-Pile year 2100

In Figure 1, it appears that the high burn up LWR scenario has the lowest amount of Pu never reaching above 200 tons of Pu opposed to the 1000 to 1500 tons that the MOX and FR cases reach. However, this is misleading due to the higher content of Pu in the MOX and FR fuel. Figure 2 appears to show very little savings going to the MOX and FR reactors over the high burn up LWRs scenario. At the 50% mark on the CDF the FR case is only saving about 200 tons of Pu over the high burn up case and about 600 tons of Pu over the MOX case which when added together with the in-pile results yields roughly identical values for Pu inventory. This plot appears to show that the MOX case is the worst in the Pu savings aspect and the FR is marginally the best. In order to better understand these results additional plots were made normalizing the data over energy produced and are shown below in Figures 3 and 4.

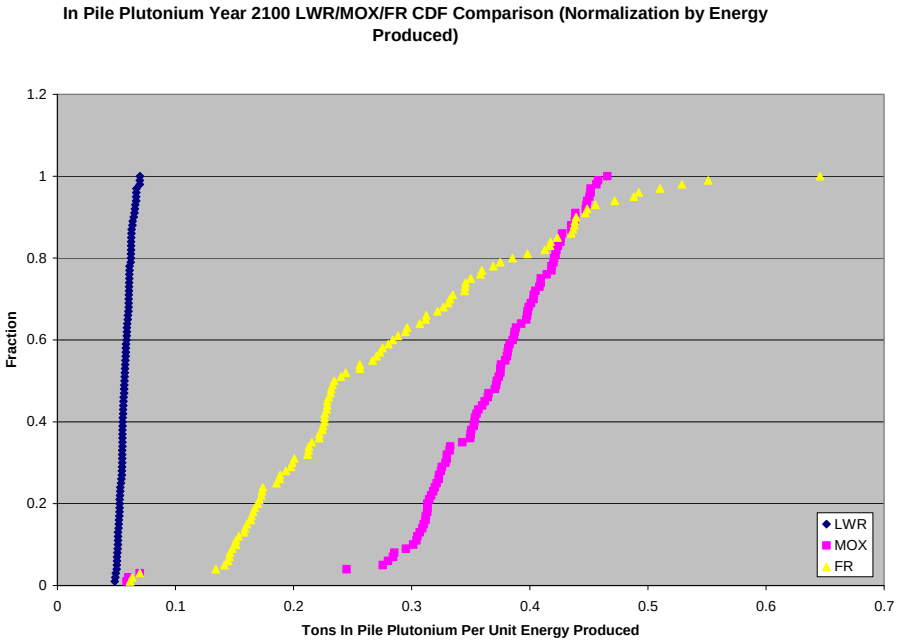

Figure 9-3: Sample Comparison CDF of Results Normalized Over Energy Produced. Pu In-Pile year 2100

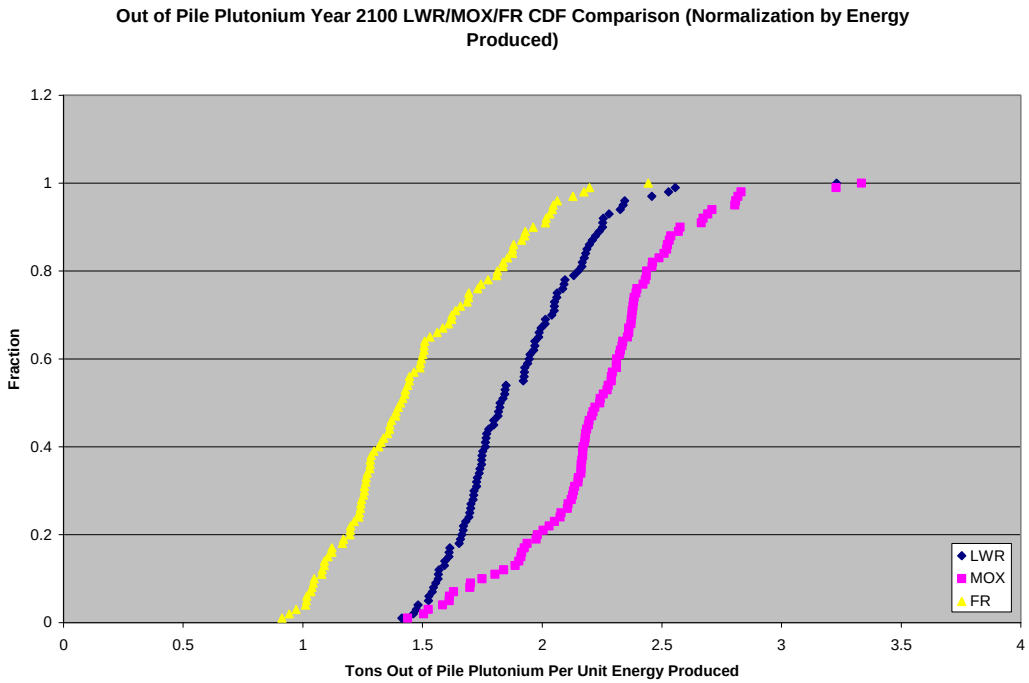

Figure 9-4: Sample Comparison CDF of Results Normalized Over Energy Produced. Pu Out-of-Pile year 2100

These energy normalized plot yields the same conclusions as the previous results in a much easier to view manner. The FR energy normalized Pu out-of-pile inventory for 50% of the runs is lower than all of the MOX and high Burn up runs and when adjusted for the in-pile inventories the FR scenario comes out slightly ahead of the high burn up scenario. This small Pu savings is further significant when coupled with the fact that a significantly more amount of Pu in the FR scenario is located in-pile and thus in a reactor than that of the high burn up scenario. Also as before the MOX case turns out to be the least favorable in terms of Pu inventory for in-pile and out-of-pile inventories.

10. Uncertainty Analyses of Fuel Cycle Economics

Estimates of life cycle costs associated with alternative fuel cycles are complicated by requirements for facilities that have not been built or operated. Nevertheless results obtained with the G4Econ code show less uncertainty and the less cost for the once-through fuel cycle than with fuel cycles that introduce support facilities for advanced fuel cycles.

In order to consider implementing any advanced fuel cycle, the cost of doing so will inevitably become a concern. Therefore, we have done a preliminary economic overview of the fuel cycles considered and have put together some uncertainty economic

data. The initial economic data is found using GNEP Excel code G4ECONS, which evaluates equilibrium scenarios from an economic point of view. This code is supplemented by the coupling of the @RISK code, which allows us to uncertainties from the G4ECONS code.

The @RISK software allows the use of the same monte carlo sampling done in DANESS by assigning distributions to cell values rather than constant numbers and then simulating the entire spreadsheet 100 times. This yields data that can be plotted in CDFs as before to compare uncertainties between different scenarios. An example flow diagram from the G4ECONS code for a totally closed fuel cycle is shown below in Figure 10-5.

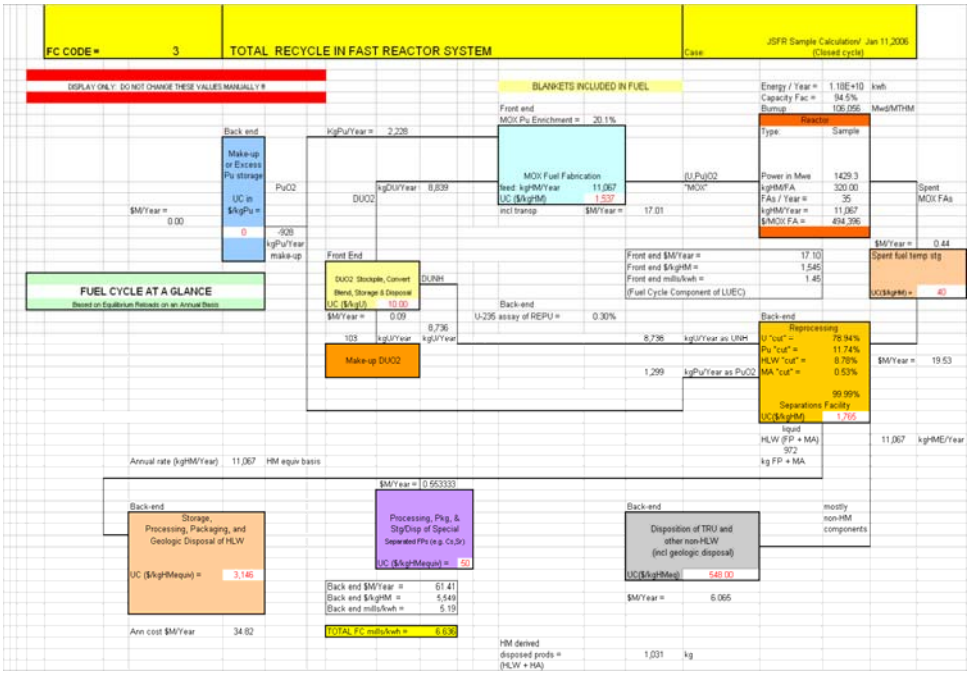

Figure 10-1: Example Flow Diagram from G4ECONS code for Totally Closed Fuel Cycle

As with DANESS, G4ECONS has numerous input parameters that can be Monte Carlo sampled. In this instance the input parameters of interest are more concrete and distributions are readily available in the "Advanced Fuel Cycle Cost Basis" report from Idaho National Labs, prepared for the Department of Energy. Using this document and the distribution functionality in @RISK, distributions are defined for several parameters inside the G4ECONS code. Table 10-4 shows an example of these parameters and there distributions for the open fuel cycle scenario.

Table 10-1: Parameter Distributions for LWR Open Cycle

Input	Minimum	Maximum	Mean
Reactor Average Capacity Factor over Life	0.74	0.97	0.87

Thermal Efficiency	0.31	0.34	0.33
Plant Economic and Operational Life	36.02	70.19	47.80
Years to Construct (up to 10 years allowed)	3.69	8.06	5.17
Real discount rate for Interest during Construction & Amortization	0.03	0.09	0.05
Estimated D&D cost for Reactor at end-of-life	274.96	939.33	533.14
Capital replacements as a % of direct capital	0.01	0.02	0.01
Contingency on non-fuel O&M cost	12.23	94.17	48.32
Required U-enrichment level for virgin EU reactor fuel (initial [first] core average)	0.03	0.03	0.03
Required U-enrichment level for virgin EU reactor fuel (reload average)	0.04	0.04	0.04
Uranium Ore (Mining and Milling U3O8)	12.49	75.89	42.66
Oxide to UF6 conversion (natural or virgin EU)	5.21	14.39	10.00

Results

The G4ECONS code run with @RISK coupled to it produced CDFs for the fuel cycle total cost for all three scenarios. These results are shown below in Figure 6.

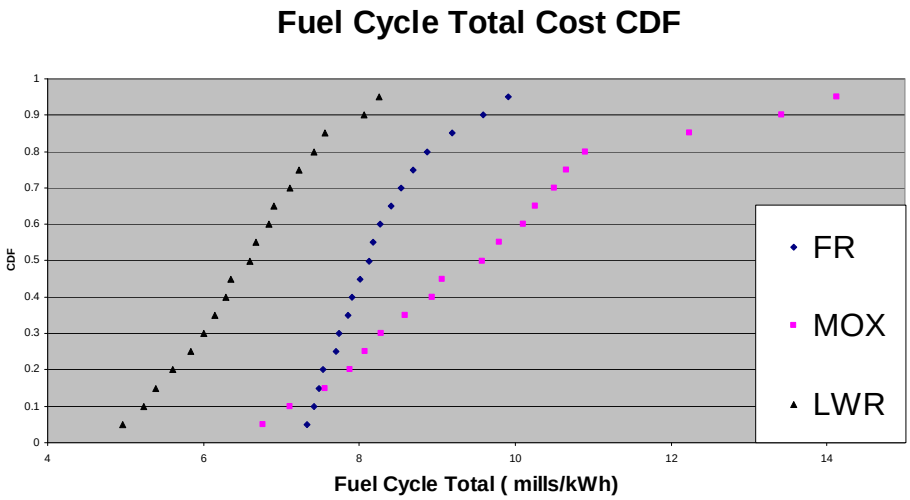

Figure 10-2: CDF of Fuel Cycle Total Cost

As can be seen in the CDF plot, the LWR open cycle is by far the cheapest with 80% of the runs coming in cheaper than both of the recycle cases. The FR full recycle comes in second and the MOX partial recycle cycle turns out to be the most expensive. This is a very limited uncertainty analysis and is only a preliminary study of the economics of the advanced fuel cycles. Future work will be done and will include additional parameters varied and a more in-depth break down of the cost structure; however, this initial work added to the Pu inventory uncertainty study would seem to

suggest that the MOX cycle is not only the most expensive but the worst at controlling the Pu supply.

11. Optimization of Advanced Fuel Cycles

In the past quarter, advances have been made in creating the advanced fuel cycle uncertainty analysis and optimization code in the Matlab environment. As an effort to create an easy, intuitive program, a graphical user interface (GUI) was designed such that design of a fuel cycle would require only a point and click interface. A screen capture of the GUI design is shown in Figure 1.

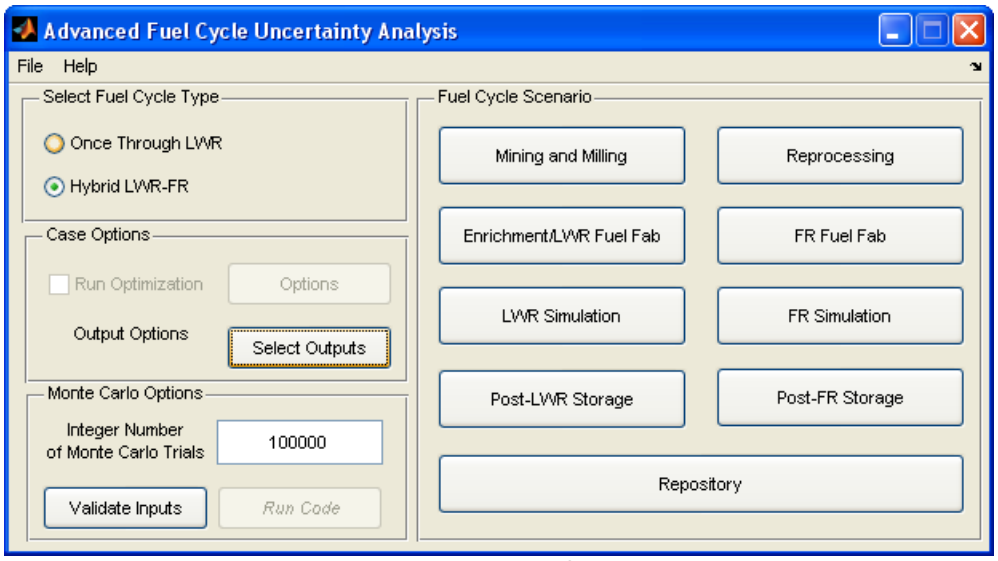

Figure 11-1: Graphical environment for uncertainty analysis

As a design, the user initially selects the fuel cycle type, either a once through or a hybrid cycle, then options are made available to the user to define parameters associated with the cycle. Specific options previously used in the script version of the code, such as burn up or enrichment, have had a GUI dialog added where the user can graphically select the distribution type and associated parameters, as shown in Figure 2. Other parameter types, such as natural uranium enrichment, are input as a single value.

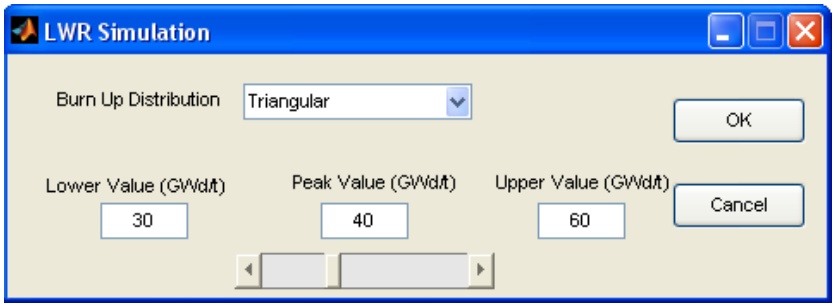

Figure 11-2: GUI for selecting LWR burn up

While parameter selection is arguably the most important part of the code, creating an output type that is useful for analysis is equally important. For some fuel cycle scenarios, isotopic compositions are rather important, thus a GUI is provided, Figure 3, where the user may select individual isotopes and corresponding output units for the table. Two major options are provided that select the main case of the code, heat load and isotopic masses. Both provide isotopic inventories, but the heat load analysis also provides units useful to repository heat load analysis for once through and hybrid cycles. Also, the user may select to save the outputs as a Matlab data file, where additional Monte Carlo trials may be added to the data to improve the statistics or to perform alternative analyses. As a similarity to ORIGEN-ARP, which allows a user to produce time-dependent isotopic plots or mass or decay heat, the user may choose to produce similar plots that give a graphical view of how the long term effects of the fuel cycle are manifested in the decay heat or isotopic inventories.

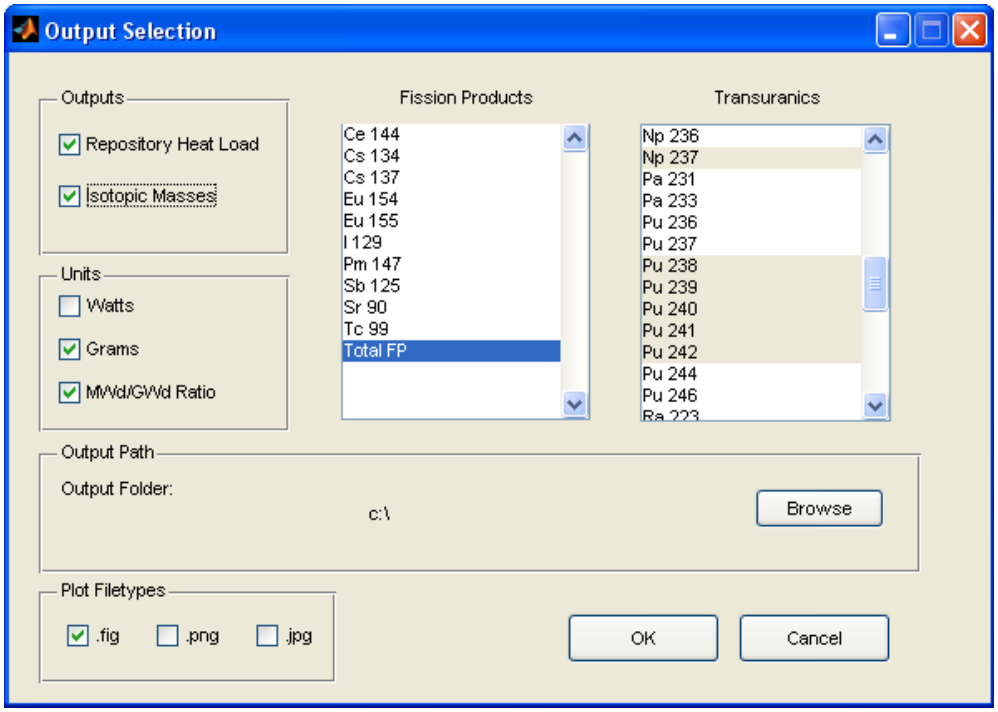

Figure 11-3: GUI for output selection

Underneath the GUI, a command line version of the code is provided that performs the calculations and simulations. Isotopic decay heat and mass data produced using SCALE 5.1 has been processed and trained into neural networks that perform the interpolations. As an effort to speed up calculations, each isotope has its work neural network for each unit of possible output, including mass, decay heat, and integral decay heat. LWR Mass and decay heat networks use the inputs of burn up and enrichment to produce an array of output data that includes the time at discharge to 1500 years. Networks for integral decay heat output a single value, but have 3 inputs of burn up, enrichment, and time after discharge, between 0 and 200 years. Because 68 isotopes are tracked in the program, there are 204 networks trained for each LWR type. Fast reactor networks are similar to the LWR networks, except the enrichment variable is replaced with the ideal conversion ratio of the fast reactor.

Because the code uses Monte Carlo sampling, neural networks had to be used to perform quick interpolations of intermediate values, but this has a double function in that genetic algorithms can use the same neural networks to optimize various fuel cycle parameters. Optimizations within the scope of this program include:

1. Minimization of specific isotopes, such as plutonium, americium and curium
2. Minimization of repository heat load
3. Minimization of fuel cycle cost

Various genetic algorithms have been designed that are powered by the output of a fitness function, an equation that uses a set of independent variables determined in a quasi-random fashion to provide a single output that determines how well that set of parameters performed. In the case of the isotopic minimization, a set of parameters for the hybrid cycle are selected, including burn ups, enrichments, conversion ratios, delay time after discharge, etc, such that for that set of parameters, the Monte Carlo code is run for a single trial using multiple sets of parameters to return numerical values for the mass of that isotope or isotopes. A user may decide to make a design decision for a parameter or set of parameters, such as setting the fast reactor conversion ratio to a constant. A similar optimization scheme for the second case is used such that instead of tracking the isotopic masses, the integral heat load becomes the parameter of interest.

Optimization of the fuel cycle cost is the more difficult of the tasks, as few sources for economic information of fast reactor and reprocessing facilities exist. Using an economic report from the Economic Modeling Working Group as the basis for the cost functions per kilogram of material, costs are applied to each step the material takes in a given fuel cycle. This optimization is supplied for both once through and fast reactors such that a comparison between the repository costs may be found. It is important to note that the code is written such that should distributions or parameters change, the code requires no changes except for the inputs supplied by the user.

In the next quarter, the code is on schedule for completion. An analysis of results for various optimization cases will be supplied along with sample uncertainty analyses for a once through and hybrid fuel cycle.

12. Management Tool for Assessment of Alternative Fuel Cycles

12.1 Overview

Development of an advanced fuel cycle management tool for uncertainty analyses and optimization requires a clear, detailed definition of the fuel cycle used with regard to mass flow, reactor physics, policy considerations, and economic forecasting. A set of mass flow equations is the foundation of the analysis code and provides a common set of equations necessary for all heat load and economic analyses. In the uncertainty analysis portion of the code, parameter distributions used are based on reported data for the current and possible future reactor fleets and supporting facilities. Economic costs for some stages, namely reprocessing and repository operation, are dependent on the isotopic compositions of spent fuel produced with the varying reactor physics parameters, requiring an individual assessment for each set of sampled parameters. Optimization using genetic algorithms is performed using the same code foundation as the uncertainty analysis portion, but rather selects parameter sets that maximize a specific output, such as economic cost, isotopic inventory, or repository capacity. While time dependent uncertainty analysis with stepwise long term optimization is the intended usage of this methodology as a tool for policy and decision makers in designing the future fuel cycle, the simplified case of the equilibrium cycle is evaluated here as a demonstration of the methodology.

12.2 Code Layout

Because the code includes both uncertainty analysis and optimization capabilities, many functions are shared between the divisions to reduce complexity and add uniformity to the evaluation methodologies. With the Genetic Algorithm optimization method, functions common to uncertainty analysis can be used with very few changes when defining the optimization cases.

From either the Matlab command line or the graphical user interface, input data is entered into a structure defined for case of uncertainty analysis, economic optimization, repository heat load optimization, or isotopic inventory minimization and subsequently loaded into a driver program that orchestrates the flow of the code, as shown in Figure J1 in the Appendix. In the flow chart of the driver program, the information structure contains all fuel cycle variables, parameters, distribution constraints, data locations, and output requirements. An error checking module verifies the input structure to ensure that a complete input is provided. Once the input is parsed, the driver selects the module to run based on the user request as either the uncertainty analysis or optimization modules.

12.2.1 Uncertainty Analysis Module

For the uncertainty analysis case, the flow chart of the module, shown in Figure J2 in the Appendix, begins with evaluating a common base of parameters for the once through fuel cycle, as LWR's provide the foundation for both the once through and hybrid fuel cycles. In Block A, which is evaluated for both cycles, the first function performs the Monte Carlo sampling of the LWR distributions established by the user, which are data limited here to initial enrichment, burn up, and delay time as defined in Table 1; however, the code is designed such that if the data allows for evaluation of additional parameters, no changes need to be made to the code itself, but rather the input definitions of parameters. In sampling, all of the trials are sampled at the same time to "vectorize" the code for a reduction in computation time and to provide output for plotting output results against sampled values to determine trends. A multidimensional isotopic database is calculated using either a generalized regression neural network, described in Appendix D, or a multiple linear interpolation, which can be used depending on the user input and data constraints.

Table 1: Light water reactor parameter distributions

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

After the common LWR database is calculated, the user definition for fuel cycle type, either the once through or hybrid cycle, directs the flow of subsequent analysis to either Block C for the once through fuel cycle or Block B for the hybrid fuel cycle.

12.2.2 Once Through Fuel Cycle Module

For the once through case, the mass flow is evaluated according to the flow chart in Figure J4 in the Appendix. Data calculated for the once through cycle in blocks are produced during the Monte Carlo sampling and common parameter evaluation and passed to the once through module. Block B is an example of the pseudo-equilibrium analysis, where reactor parameters are reduced to per-year units of fuel requirements as is customary of an equilibrium analysis. A pseudo-equilibrium analysis is performed for economic analysis where costs may increase over time, which requires non-equilibrium conditions. The first piece of the analysis prepares for the LUEC of the economic analysis by calculating the amount of electricity generated per year, shown in Equation (1).

$$\dot{E}_{Gen} = \dot{P}_{\text{Reactor}} \cdot \eta_{\text{Thermal}} \cdot \frac{1000\text{kW}}{1\text{MW}} \cdot \frac{365d}{1yr} \cdot \frac{24h}{1d} = \left[\frac{kWh}{yr} \right] \tag{1}$$

In the event that the specific power of the reactor is variable depending on the provided data, Equation (2) provides an estimate of the specific power, otherwise, the user may input a constant specific power.

$$SP = \frac{\dot{P}_{\text{Reactor}}}{M_{\text{Core}}} = \left[\frac{GW}{MT} \right] \tag{2}$$

For economic analysis that may dictate increasing fuel costs over time, a modification to the equilibrium fuel cycle case is made to calculate the total amount of fuel required over the lifetime of the reactor and is reduced to the mass required per year, shown in Equation (3).

$$\dot{M}_{\text{Reactor}} = \frac{M_{\text{Initial}} + \sum M_{\text{Refueling}}}{T_{\text{Reactor Lifetime}}} = \left[\frac{MT}{yr} \right] \tag{3}$$

Assuming an operating lifetime of 60 years, consisting of the 40 year initial licensing combined with a 20 year extension, the total core mass is the sum of the initial core loading and all refueling over the lifetime, shown in Equation (4).

$$M_{Total} = M_{\text{Initial}} \cdot \left(\frac{1}{N_{\text{FuelBatches}}} \cdot R_{\text{Number Refuelings}} + 1 \right) = \left[MT \right] \tag{4}$$

Over the entire reactor lifetime, the number of refuelings is a function of the fuel burn up and the number of batches in the fuel, shown in Equation (5), where the time between refueling is given in Equation (6).

$$R_{\text{Number Refuelings}} = \frac{T_{\text{Reactor Lifetime}}}{T_{\text{Time between refuelings}}} \tag{5}$$

$$T_{\text{Time between refuelings}} = \left(\frac{B_{\text{Reactor Burn Up [GWd/MT]}}}{SP_{\text{[GW/MT]}}} \right) \cdot \frac{1yr}{365d} \cdot \frac{1}{N_{\text{Fuel Batches}}} = \left[yr \right] \tag{6}$$

Assuming a gaseous diffusion enrichment method, the ratio of the amount of natural uranium to the mass of enriched uranium is calculated to estimate the amount of depleted uranium created and to provide estimates for costs. This feed to product ration is given in Equation (7) [1].

$$\frac{F}{P} = \frac{x_p - x_w}{x_f - x_w} \tag{7}$$

The amount of feed uranium required per year is given in Equation (8), with the amount of depleted uranium produced per year in Equation (9).

$$\dot{M}_{\text{Natural Uranium}} = \frac{F}{P} \cdot \dot{M}_{\text{Reactor}} = \left[\frac{MT}{yr} \right] \tag{8}$$

$$\dot{M}_{\text{Depleted Uranium}} = \dot{M}_{\text{Natural Uranium}} - \dot{M}_{\text{Reactor}} = \left[\frac{MT}{yr} \right] \tag{9}$$

In calculating the costs associated with enrichment, the number of separative work units, or SWU's, is required, shown in Equation (10).

$$V(x) = (2 \cdot x - 1) \cdot \ln\left(\frac{x}{1-x} \right) \tag{10}$$

The resulting number of SWU's required for a given amount of product is given in Equation (11) and the number of SWU's per year is given in Equation (12).

$$SWU_{\text{Per MT Product}} = V(x_p) + \left(\frac{F}{P} - 1 \right) \cdot V(x_w) - \frac{F}{P} \cdot V(x_f) \tag{11}$$

$$SWU = SWU_{\text{Per MT Product}} \cdot \dot{M}_{\text{Reactor}} = \left[\frac{SWU}{yr} \right] \tag{12}$$

Returning to Figure J4, the fuel production pathway for the LWR consists of the mass requirements, which may be scaled according to the number of LWR's in the fleet assuming standard reactor types. Block C performs the interpolation calculations to produce the reactor physics database. Mass data is automatically calculated for the totals of all actinides and fission products, but the user may select which isotopes to evaluate in addition, where the list is given in Appendix B. If the user wants to perform an analysis for decay heat, then interpolations are made for that data and the mass data. In the once through case, a reactor park is defined as a single LWR; thus, results may be scaled to a larger reactor park if applicable.

After the database is created, user input then may direct the code to perform an economic analysis, returning from the once through analysis block of Figure J2C to Figure J4D. Similar to the mass requirements in the pseudo-equilibrium case, the economic data can be reduced to the terms of a cost per year. Each stage of the fuel cycle has some associated annual cost that is

dependent on a mass flow, electrical generation, or facility costs as defined in **Error! Reference source not found.**. Beginning with the mining and milling stage, the cost per year is calculated according to Equation (13).

$$Cost_{Mining/Milling} = \dot{M}_{\text{Natural Uranium}} \cdot C_{\text{Uranium Ore}} = \left[\frac{\$}{yr}\right] \tag{13}$$

Conversion of uranium ore from U3O8 to UF6 is given in Equation (14).

$$Cost_{\text{UF6 Conversion}} = \dot{M}_{\text{Natural Uranium}} \cdot C_{\text{Cost per unit mass}} = \left[\frac{\$}{yr}\right] \tag{14}$$

Enrichment costs are more dependent on the energy required to produce the product rather than the cost of the feed uranium; thus, the cost of enrichment is dependent on the product enrichment and the number of SWU's required to reach that, as shown in Equation (15).

$$Cost_{Enrichment} = SWU_{\text{per year}} \cdot C_{\text{Cost per SWU}} = \left[\frac{\$}{yr}\right] \tag{15}$$

Depleted uranium does not need to be stored in the repository as a less costly facility may be used because of the low radioactivity of the material. Costs for this storage can be reduced to cost per year, shown in Equation (16), as with the rest of the cost terms.

$$C_{\text{DU Storage}} = \dot{M}_{\text{Depleted Uranium}} \cdot C_{\text{Storage cost per mass}} = \left[\frac{\$}{yr}\right] \tag{16}$$

Fuel fabrication costs are expanded to include transportation and facility operations [37,38]. The resulting formula is shown in Equation (17).

$$C_{\text{LWR Fuel Fab}} = \dot{M}_{\text{Reactor}} \cdot C_{\text{Cost per unit mass}} = \left[\frac{\$}{yr}\right] \tag{17}$$

Because a repository is expected to open at some point, there are costs associated with on-site temporary storage and long term disposal. On-site storage costs, neglecting any increased costs of diminishing capacity stemming from delays in repository opening, is given in Equation (18), while repository costs are given in Equation (19). Repository costs also neglect increases in storage price as space decreases and are based on mass deposition rather than heat load.

$$C_{\text{On-site Storage}} = \dot{M}_{\text{Reactor}} C_{\text{Storage cost per unit mass}} = \left[\frac{\$}{yr}\right] \tag{18}$$

$$C_{\text{Repository}} = \dot{M}_{\text{Reactor}} \cdot C_{\text{Cost per unit mass deposited}} = \left[\frac{\$}{yr}\right] \tag{19}$$

With the NWPA of 1982, the Nuclear Waste Fund was established to offset the government cost of building and operating a repository, as shown in Equation (20).

$$C_{\text{NWPA Fee}} = \dot{E}_{\text{Gen per year}} \cdot C_{\text{1 mill/kWh}} = \left[\frac{\$}{yr}\right] \tag{20}$$

While operating and maintenance costs are significant, the majority of the cost associated with nuclear power is the facility cost. Because no plant has been built for some time, the estimates vary as to the cost of a new facility. Using the uniform series approximation for pay off, the resulting facility cost is reduced to a per year cost in Equation (21), where i is the estimated interest rate.

$$C_{\text{Facility}} = P_{\text{Initial Facility Price}} \frac{\left[i \cdot (1+i)^{T_{\text{Reactor Lifetime}}}\right]}{\left[(1+i)^{T_{\text{Reactor Lifetime}}} - 1\right]} = \left[\frac{\$}{yr}\right] \tag{21}$$

Revenue is accounted for in terms of electricity sales, shown in Equation (22).

$$R_{\text{Electricity Sales}} = \dot{E}_{\text{Gen per year}} \cdot \left(R_{\text{Consumer Price}} - C_{\text{State and local taxes}}\right) = \left[\frac{\$}{yr}\right] \tag{22}$$

Each of the aforementioned costs includes a time-dependent amount that can be tied to inflation or scarcity costs according to either a flat rate, uniform series, or geometric gradient. In the simple case of a flat rate, the sum of all costs is the linear combination of all of the costs and revenue. Time-dependent increases require summation in each of the time steps separately, where a present-day fixed cost can be evaluated.

The levelized unit electric cost, LUEC, is the ratio of the summation of the costs and revenue to the electrical generation, given in Equation (23).

$$LUEC = \frac{R_{\text{Electricity Sales}} - \sum C}{\dot{E}_{Gen}} = \left[\frac{\$}{kWh}\right] \tag{23}$$

Upon exiting the program, control of the once through module is returned to the calling function, either the uncertainty analysis or optimization modules. Output data consists of all variables and calculated values from the module. If a

user wants to reduce the size of the output, then this is performed in the calling function according to the user input structure.

12.2.3 Hybrid Fuel Cycle Module

For the hybrid fuel cycle case, the mass flow is evaluated according to the flow chart in Figure J5 in Appendix K. Data calculated for the hybrid cycle are produced during the Monte Carlo sampling and common parameter evaluation and passed to the hybrid module, as is the case with the once through module. As with the once through module, a pseudo-equilibrium analysis is performed, where reactor parameters are reduced to per-year units of fuel requirements as is customary of an equilibrium analysis. A pseudo-equilibrium analysis is preferred over an actual equilibrium analysis where for economic costs may increase over time, which requires non-equilibrium calculations; thus, performing the calculations in this manner allows for operating in both cases.

Blocks A and B are rather similar in that the first piece of the analysis prepares for the LUEC of the economic analysis by calculating the amount of electricity generated per year, shown in Equation (1), for both the selected LWR and FR, and adding the values, shown in Equation (24).

$$\dot{E}_{\text{Gen total}} = \dot{E}_{\text{Gen FR}} + \dot{E}_{\text{Gen LWR}} \tag{24}$$

In the event that the specific power of the reactor is variable depending on the provided data, Equation (2) provides an estimate of the specific power, otherwise, the user may input a constant specific power; however, this must be done separately for LWR and FR terms. For economic analysis that may dictate increasing fuel costs over time, a modification to the equilibrium fuel cycle case is made to calculate the total amount of fuel required over the lifetime of the reactor and is reduced to the mass required per year, shown in Equation (3).

With the power terms finished, fuel requirements for the fast reactor are calculated in Block C and LWR requirements are calculated in Block D. Both blocks have similar calculations for the first few functions; however, the LWR block contains terms for the uranium production pathway of mining through enrichment. Block D is identical to the LWR fuel pathway described in the “Once Through Fuel Cycle Module” section; thus, only block C is described here.

Because a FR has a different input fuel consistency compared to the LWR, fuel requirements are done on a per isotope mass basis. Depending on the composition data provided to the program from the user, which is the same as that used in FR data creation, each isotope is evaluated. Because the reactor is assumed to have a single fuel type, a simplified case based on the Super PRISM, the number of refueling is the same regardless of the isotope, so this needs to be calculated only once. Assuming an operating lifetime of 60 years, consisting of the 40 year initial licensing combined with a 20 year extension, the total isotopic mass required by the core description is the sum of the initial core loading and all refueling over the lifetime, shown in Equation (25).

$$M_{\text{Isotope Total}} = M_{\text{Initial}} \cdot \left(\frac{1}{N_{\text{FuelBatches}}} \cdot R_{\text{Number Refuelings}} + 1 \right) = [MT] \quad (25)$$

Over the entire reactor lifetime, the number of refuelings is a function of the fuel burn up and the number of batches in the fuel; thus, the same formula as in the once through case may be applied, shown in Equation (5), where the time between refueling is given in Equation (6).

With the fast reactor fuel requirements and LWR fuel production pathway terms tabulated, the reactor database is used in the interpolation of values for the specified parameters. Block E performs these calculations to produce the problem specific reactor physics database. Mass data is automatically calculated for the totals of all actinides, fission products, and isotopes listed in the fast reactor initial composition, but the user may select additional isotopes listed in Appendix B. If the user wants to perform an analysis for decay heat, then interpolations are made for that data and the mass data. With limited fast reactor data, the core charge loading cannot be sampled as this pertains to the conversion ration and stability of the core, thus the conversion ratio is defined by the user and the data for burn up is then samples according to the distributions in Table 2.

Table 2: Fast reactor parameter distributions

Parameter	***CR***	***Distribution***	***Low Value***	***Peak***	***High Value***
Burn up (metal fueled)	0.00	Triangular	80 GWd/t	90 GWd/t	110 GWd/t
	0.25	Triangular	80 GWd/t	120 GWd/t	130 GWd/t
	0.50	Triangular	80 GWd/t	130 GWd/t	180 GWd/t
	0.75	Triangular	80 GWd/t	150 GWd/t	200 GWd/t
	1.00	Triangular	80 GWd/t	180 GWd/t	200 GWd/t
Burn up (oxide fueled)	0.00	Triangular	80 GWd/t	90 GWd/t	100 GWd/t
	0.25	Triangular	80 GWd/t	110 GWd/t	120 GWd/t
	0.50	Triangular	80 GWd/t	130 GWd/t	150 GWd/t
	0.75	Triangular	80 GWd/t	150 GWd/t	180 GWd/t
	1.00	Triangular	80 GWd/t	150 GWd/t	200 GWd/t
Delay Time	All	Triangular	5 years	30 years	10 years

After interpolation, Block F performs scaling calculations that depend on the view of the hybrid cycle. While the once through case uses a single reactor in the park, the hybrid cycle assumes a variable size reactor park. One of two reactor park models is chosen by the user. The first of which is a variable number of LWR's to support a single fast reactor. This results in a reactor cluster model that may be scaled up in integer numbers of clusters. The second model is a set number of LWR's and fast reactors.

In the cluster model, the number of LWR's to fast reactors is defined by the production of isotopes required for fast reactor fuel. The least produced isotope is used in determining the ratio, where any excess is left to the repository. The ratio, given in Equation (26) is performed on a per year production basis. Once the ratio is found, the number of LWR's is used as a scaling factor for all mass, decay heat, and cost terms where applicable.

$$P_{\text{Ratio LWR to FR}} = \frac{M_{\text{FR Isotopic Required}}}{M_{\text{LWR Isotopic Required}}} = N_{\text{Number of LWRs}} \tag{26}$$

In either the cluster or fixed park models, a calculation, representing the reprocessing stage, is performed to correct the magnitudes of LWR waste streams. In the once through case, the waste stream is the interpolated value, while in the hybrid cycle, the waste stream is the interpolated value with the amount diverted for fast reactor fuel production, shown in Equation (27), where the units are in mass or integral decay heat. Time dependent decay is not addressed as the interpolated value is taken at a certain time after LWR discharge, regarded as the delay time.

$$\dot{M}_{\text{Isotope Value}} = \left(N_{\text{Number of LWRs}} \cdot \dot{M}_{\text{Interpolated}}\left(T_{\text{Delay Time}}\right)\right) - \left(N_{\text{Number of FRs}} \cdot \dot{M}_{\text{Required for FR Fuel Fab}}\right) \tag{27}$$

Fast reactor fuel production uses depleted uranium from the LWR enrichment process rather than reprocessing product, thus the amount of depleted uranium is modified according to Equation (28). This term is calculated on a mass per year basis as done with the once through cycle.

$$\dot{M}_{\text{Depleted Uranium in Hybrid Cycle}} = \dot{M}_{\text{Depleted Uranium from LWR Enrichment}} - \dot{M}_{\text{Depleted Uranium Required in FR}} = \left[\frac{MT}{yr}\right] \tag{28}$$

Should an economic analysis be required, Block G performs an analysis very similar to that defined in the once through cycle. Beginning with the mining and milling stage, the cost per year is calculated according to Equation (13), where the amount of natural uranium required per year is scaled to the number of LWR's in the reactor park as given in Equation (29).

$$Cost_{Mining/Milling} = N_{\text{Number of LWRs}} \cdot \dot{M}_{\text{Natural Uranium}} \cdot C_{\text{Uranium Ore}} = \left[\frac{\$}{yr}\right] \quad (29)$$

Conversion of uranium ore from U3O8 to UF6 is given in Equation (14), but again the number of LWR's in the park changes the value as in Equation (30) ..

$$Cost_{\text{UF6 Conversion}} = N_{\text{Number of LWRs}} \cdot \dot{M}_{\text{Natural Uranium}} \cdot C_{\text{Cost per unit mass}} = \left[\frac{\$}{yr}\right] \quad (30)$$

Enrichment costs are more dependent on the energy required to produce the product rather than the cost of the feed uranium; thus, the cost of enrichment is dependent on the product enrichment and the number of SWU's required to reach that, as shown in Equation (15).

While depleted uranium does not need to be stored in the repository when a less costly facility may be used, the modified amount of depleted uranium is used in Equation (31) in the calculation of storage cost.

$$C_{\text{DU Storage}} = \dot{M}_{\text{Depleted Uranium in Hybrid Cycle}} \cdot C_{\text{Storage cost per mass}} = \left[\frac{\$}{yr}\right] \quad (31)$$

Fuel fabrication costs are expanded to include transportation and facility operations [37,38]. The resulting formula is shown in Equation (32) for LWR's and in Equation (33) for FR's. Reprocessing costs are included in the fast reactor cost per unit mass.

$$C_{\text{LWR Fuel Fab}} = N_{\text{Number of LWRs}} \cdot \dot{M}_{\text{Reactor}} \cdot C_{\text{Cost per unit mass}} = \left[\frac{\$}{yr}\right] \quad (32)$$

$$C_{\text{FR Fuel Fab}} = N_{\text{Number of FRs}} \cdot \dot{M}_{\text{Reactor}} \cdot C_{\text{Cost per unit mass}} = \left[\frac{\$}{yr}\right] \quad (33)$$

Because the hybrid cycle is assumed to be closed, there are costs associated with on-site temporary storage and long term disposal. On-site storage costs, neglecting any increased costs of diminishing capacity stemming from delays in repository opening, is given in Equation (34) that assumes a common cost for storage.

$$C_{\text{On-site Storage}} = \left(N_{\text{Number of LWRs}} \cdot \dot{M}_{\text{LWR}} \cdot T_{\text{Delay 1}} + N_{\text{Number of FRs}} \cdot \dot{M}_{\text{FR}} \cdot T_{\text{Delay 2}}\right) C_{\text{Storage cost}} = \left[\frac{\$}{yr}\right] \quad (34)$$

Repository costs are not well defined, thus the quality of result dependent on the quality of the estimate. Using the per unit mass cost as in the once

through cycle, which neglects reduced costs for different waste products, Equation gives an estimate to the cost per unit mass of the repository.

$$C_{\text{Repository}} = \left(N_{\text{Number of LWRs}} \cdot \dot{M}_{\text{LWR}} + N_{\text{Number of FRs}} \cdot \dot{M}_{\text{FR}} \right) \cdot C_{\text{Cost per unit mass deposited}} = \left[\frac{\$}{yr} \right] \quad (35)$$

Contribution to the Nuclear Waste Fund is taken into account in Equation (36) using the combined energy generation of LWR's and FR's.

$$C_{\text{NWPA Fee}} = \left(N_{\text{Number of LWRs}} \cdot \dot{E}_{\text{Gen per year LWR}} + N_{\text{Number of FRs}} \cdot \dot{E}_{\text{Gen per year FR}} \right) \cdot C_{\text{1 mill/kWh}} = \left[\frac{\$}{yr} \right] \quad (36)$$

While operating and maintenance costs are significant, the majority of the cost associated with nuclear power is the facility cost. Because no LWR plant has been built for some time, the estimates vary as to the cost of a new facility; however, since no Super PRISM has ever been built, the estimates on cost have a high amount of intrinsic variance. Included in the facility costs are the costs of the reprocessing facilities that use values scaled up from existing facilities in France. It is assumed that a reprocessing capacity is built of sufficient size such that only a single facility is required. Each facility cost is calculated separately as in Equation (21), where i is the estimated interest rate, and then scaled according to the number of facilities required, as in Equation (38).

$$C_{\text{Facility}} = P_{\text{Initial Facility Price}} \frac{\left[i \cdot (1+i)^{T_{\text{Reactor Lifetime}}} \right]}{\left[(1+i)^{T_{\text{Reactor Lifetime}}} - 1 \right]} = \left[\frac{\$}{yr} \right] \quad (37)$$

$$C_{\text{Facility Total}} = N_{\text{Number of LWRs}} \cdot C_{\text{LWR Facility}} + N_{\text{Number of FRs}} \cdot C_{\text{FR Facility}} + C_{\text{Reprocessing Facility}} \quad (38)$$

Revenue is accounted for in terms of electricity sales, shown in Equation (39).

$$R_{\text{Electricity Sales}} = \dot{E}_{\text{Gen per year Total}} \cdot \left(R_{\text{Consumer Price}} - C_{\text{State and local taxes}} \right) = \left[\frac{\$}{yr} \right] \quad (39)$$

As with the once through fuel cycle module, each of the aforementioned costs includes a time-dependent amount that can be tied to inflation or scarcity costs according to either a flat rate, uniform series, or geometric gradient. In the simple case of a flat rate, the sum of all costs is the linear combination of all of the costs and revenue. Time-dependent increases require summation in each of the time steps separately, where a present-day fixed cost can be evaluated.

The levelized unit electric cost, LUEC, is the ratio of the summation of the costs and revenue to the electrical generation, given in Equation (23).

$$LUEC = \frac{R_{\text{Electricity Sales}} - \sum C}{\dot{E}_{\text{Gen per year total}}} = \left[\frac{\$}{kWh}\right] \tag{40}$$

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

12.2.4 Optimization

Should the user choose an optimization function, the driver program directs the flow of the program to the optimization module. An outline of the optimization routine is given in Figure J3 in Appendix K. Because the procedure is meant to be flexible, user input is important to the flow of the program. Optimization parameters are selected by the user in the input and will direct the genetic algorithm input in the number of parameters to optimize, limits on the magnitudes of the values chosen, and methods of choosing best individuals.

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

In the case of isotopic minimization, an isotope from the list in Appendix B is chosen to minimize. At the end of the once through or hybrid fuel cycle modules, the waste stream mass output is parsed for the isotope of interest and presented to the genetic algorithm for evaluation of stopping criteria or new population selection.

For repository decay heat minimization, the hybrid fuel cycle module is evaluated to return integral decay heat. Here, the number of LWR's and FR's is defined in the output with the length of the delay times. Using these parameters with the integral decay heats for the after reprocessing LWR data and FR spent fuel data, Equation (41) is used to define the contribution to integral decay heat on a per year basis.

$$D_{IDH} = N_{\text{Number of FRs}} \cdot D_{\text{FR IDH per year}} \cdot T_{\text{Reactor Lifetime}} + N_{\text{Number of LWRs}} \cdot D_{\text{LWR IDH per year}} \cdot T_{\text{Reactor Lifetime}} \quad (41)$$

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

Once the fitness function is calculated, the genetic algorithm evaluates the output. First, the values are compared to stopping criteria of convergence or computing time, in the event of an infinite solution. Should the value not converge, a new population is created using the values for the top 5 individuals, dropping the bottom 30% of individuals, and crossing over the top 70% of the population values to fill the population back to the set number. A adaptive mutation function is included that reduces the mutation as the number of generations increases, reducing the time for convergence.

12.3 Mangement Tool Examples

12.3.1 Once Through Fuel Cycle Analysis

12.3.1.1 PWR Spent Fuel Composition

Composition of spent fuel is largely dependent on the burn up achieved. To illustrate this point, Table 12-3 shows the sensitivity of burn up on PWR spent fuel isotopic composition for a single enrichment and multiple burn ups spanning the range of the trained data. As expected, the mass of fission products increases almost linearly with burn up, while the creation of other isotopes is dependent on reactor characteristics rather than fission power.

Table 12-3: PWR spent fuel composition after 10 year decay

5% Enriched 60 GWd/t		5% Enriched 45 GWd/t		5% Enriched 30 GWd/t	
Isotope	**Mass (g/t)**	**Isotope**	**Mass (g/t)**	**Isotope**	**Mass (g/t)**
Total FP	1544.8	Total FP	1229.3	Total FP	892.095
Total TRU	939930	Total TRU	953370	Total TRU	966940
U 238	914960	U 238	926750	U 238	937580
Pu 239	6030	U 235	8660	U 235	14420
U 236	5720	Pu 239	6020	Pu 239	5780
U 235	4830	U 236	5420	U 236	4680
Pu 240	3190	Pu 240	2690	Pu 240	2000
Pu 242	1250	Pu 241	1010	Pu 241	740
Pu 241	1160	Pu 242	770	Am 241	490
Np 237	840	Am 241	670	Np 237	430
Am 241	770	Np 237	650	Pu 242	370
Pu 238	430	Pu 238	260	U 234	240
Am 243	400	Am 243	210	Pu 238	120
U 234	180	U 234	210	Am 243	70
Cm 244	150	Cm 244	60	Cm 244	10

Cm 245	10	Ac 225	0	Ac 225	0

The effect of burn up dependent conversion ratio is apparent by the reduction in uranium mass as burn up increases, showing the usage of plutonium as a fuel. Plutonium is a first step to the production of higher actinides, such as americium and curium, as is shown in the previous table.

While some isotopic inventories benefit from using longer burn ups, decay heat is a large concern. As shown in Table 12-4, the decay heat at 10 years after discharge increases greatly as burn up increases. Contributions from fission products remain on a seemingly linear path, while some actinides have values that appear to follow a power law.

Table 12-4: PWR isotopic decay heat 10 years after discharge

5% Enriched 60 GWd/t			**5% Enriched 45 GWd/t**			**5% Enriched 30 GWd/t**	
Isotope	**Heat (W/t)**		**Isotope**	**Heat (W/t)**		**Isotope**	**Heat (W/t)**
Total FP	1467.2		Total FP	1226.0		Total FP	975.147
Total TRU	699.1138		Total TRU	436.5884		Total TRU	238.3424
Cm 244	350.8067		Cm 244	175.3031		Pu 238	85.3749
Pu 238	220.7648		Pu 238	149.278		Am 241	61.1265
Am 241	85.8644		Am 241	75.3766		Cm 244	60.9714
Pu 240	21.768		Pu 240	18.9051		Pu 240	15.4338
Pu 239	11.6148		Pu 239	11.5424		Pu 239	11.2771
Pu 241	3.7059		Pu 241	3.2677		Pu 241	2.6706
Am 243	2.2706		Am 243	1.3811		Am 243	0.6572
Cm 243	1.4621		Cm 243	0.9009		Cm 243	0.4189
Cm 242	0.3651		Cm 242	0.3008		Cm 242	0.2141
Pu 242	0.1324		Pu 242	0.0919		Pu 242	0.0545
Cm 245	0.0556		U 234	0.0368		U 234	0.041
U 234	0.0334		Cm 245	0.0255		Np 237	0.0098
Cm 246	0.017		Np 237	0.013		U 236	0.0085
Np 237	0.0159		U 236	0.0094		U 238	0.008
U 236	0.0099		U 238	0.0079		Cm 245	0.0075
U 238	0.0078		Cm 246	0.0064		U 237	0.0039
U 237	0.0054		U 237	0.0048		Am 242m	0.0029
Am 242m	0.0049		Am 242m	0.004		Cm 246	0.0013
Pu 236	0.0028		Pu 236	0.0019		Pu 236	0.0011
Ra 224	0.0015		Pa 233	0.0012		Pa 233	0.0009
U 232	0.0015		Ra 224	0.001		U 235	0.0008
Pa 233	0.0014		U 232	0.001		Ra 224	0.0006
Th 228	0.0014		Th 228	0.0009		U 232	0.0006
U 235	0.0003		U 235	0.0005		Th 228	0.0005
Th 234	0.0001		Th 234	0.0001		Th 234	0.0001

While some of the higher activity actinides are short lived in comparison to the rest of those listed, they usually decay into long lived plutonium or uranium isotope, and thus integrating the decay heat over some duration gives a better estimate as to the effects

of these isotopes. By integrating over 1500 years, a footprint for each isotope in each assembly is calculated such that the long term effects may be explored.

Comparing the previous burn up dependent decay heat table with the burn up dependent plot in Figure 12-1, it is apparent that actinides produce that vast majority of the Integral Decay Heat (IDH). Fission products supply a good amount, but actinides comprise of at least 3 times that value for lower burn ups and nearly 4 times that for higher burn ups.

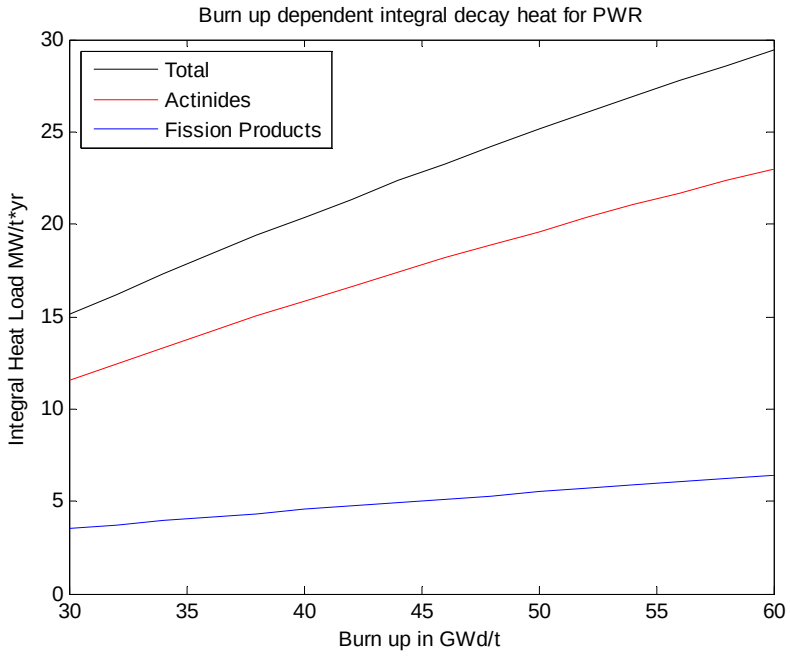

Figure 12-1: Burn up dependent IDH for PWR fuel

While actinides make up the majority of the IDH, contribution from individual isotopes shows which need to be minimized to reduce long term repository heat load. In Figure 12-2, contributions from selected long lived actinides are shown as a function of burn up. Americium-241 is a very large contributor, while the plutonium isotopes are rather small. Curium-244, when compared with the decay heat values in Table 12-4, is of far less significance than the table suggests.

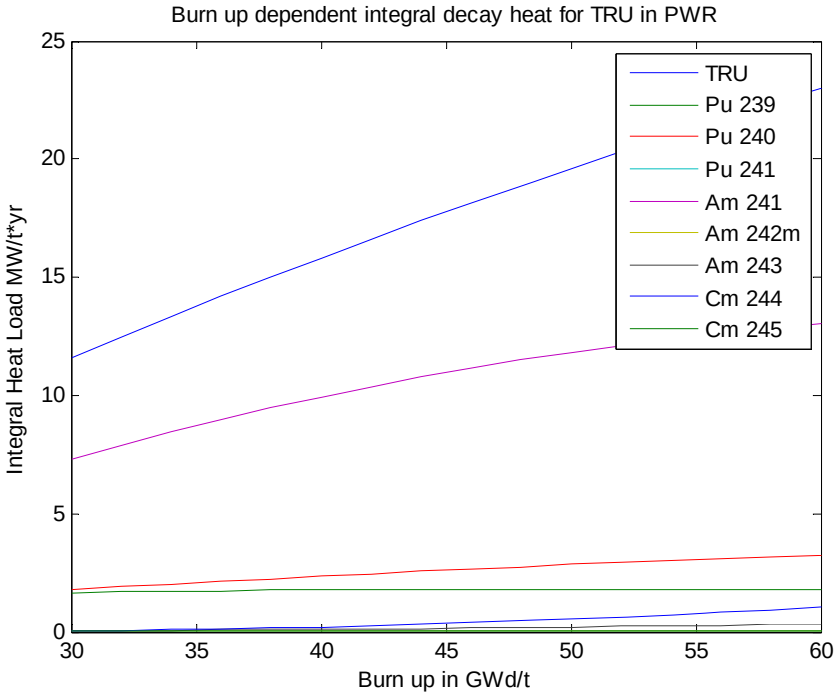

Figure 12-2: Composition of TRU burn up dependent IDH for PWR fuel

12.3.1.2 BWR Spent Fuel Composition

As with the PWR, similar relationships exist between the productions of isotopes; however, some changes are apparent in Table 12-5 as seemingly fewer actinides are produced in the BWR core. Despite using the same 1 metric ton basis for calculations, the BWR appears to be the better core in terms of creating fewer higher actinides while producing the same amount of power; however, when comparing the size of the cores, the PWR is nearly 50% smaller, about 100 metric tons, than the BWR core, about 150 metric tons.

Table 12-5: BWR spent fuel isotopic mass composition after 10 year decay

5% Enriched 60 GWd/t	
Isotope	**Mass (g/t)**
Total FP	1468.676
Total TRU	943030
U 238	922980
U 236	5750
Pu 239	4110
U 235	3490

5% Enriched 45 GWd/t	
Isotope	**Mass (g/t)**
Total FP	1230.888
Total TRU	953190
U 238	931250
U 235	6730
U 236	5470
Pu 239	4240

5% Enriched 30 GWd/t	
Isotope	**Mass (g/t)**
Total FP	981.367
Total TRU	963440
U 238	938850
U 235	11070
U 236	4970
Pu 239	4320

Pu 240	2810	Pu 240	2470	Pu 240	2030
Pu 242	1150	Pu 242	770	Pu 241	570
Pu 241	740	Pu 241	680	Pu 242	430
Np 237	620	Np 237	510	Np 237	380
Am 241	490	Am 241	450	Am 241	370
Am 243	300	U 234	200	U 234	230
Pu 238	270	Pu 238	190	Pu 238	110
U 234	180	Am 243	180	Am 243	80
Cm 244	100	Cm 244	50	Cm 244	10

Similar to the PWR table for heat contribution shown in the previous section, Table 12-6 shows heat contributions for a BWR core. It is important to note that the same bias from core size is apparent in this table as values are given as watts per initial fuel loading in metric tons, thus the BWR is actually a worse performer in regards to decay heat generation when values are scaled to core mass.

Table 12-6: BWR spent fuel isotopic heat contribution after 10 year decay

5% Enriched 60 GWd/t			**5% Enriched 45 GWd/t**			**5% Enriched 30 GWd/t**	
Isotope	**Heat (W/t)**		**Isotope**	**Heat (W/t)**		**Isotope**	**Heat (W/t)**
Total FP	1468.676		Total FP	1230.888		Total FP	981.367
Total TRU	518.3772		Total TRU	316.9635		Total TRU	171.468
Cm 244	274.715		Cm 244	129.156		Pu 238	62.2538
Pu 238	154.1503		Pu 238	106.8132		Am 241	42.8187
Am 241	55.8491		Am 241	50.9939		Cm 244	40.7502
Pu 240	19.8396		Pu 240	17.4501		Pu 240	14.3648
Pu 239	7.9366		Pu 239	8.1815		Pu 239	8.3408
Pu 241	2.4369		Pu 241	2.2244		Pu 241	1.874
Am 243	1.9486		Am 243	1.1338		Am 243	0.5037
Cm 243	0.9321		Cm 243	0.5885		Cm 243	0.2757
Cm 242	0.1578		Cm 242	0.1466		Cm 242	0.1172
Pu 242	0.1347		Pu 242	0.09		Pu 242	0.0505
U 234	0.0319		U 234	0.0367		U 234	0.0418
Cm 245	0.0259		Cm 245	0.0117		U 236	0.0087
Cm 246	0.0134		Np 237	0.0103		U 238	0.008
Np 237	0.0125		U 236	0.0096		Np 237	0.0077
U 236	0.0101		U 238	0.0079		Cm 245	0.0033
U 238	0.0079		Cm 246	0.0048		U 237	0.0027
U 237	0.0036		U 237	0.0032		Am 242m	0.0016
Am 242m	0.0021		Am 242m	0.002		Cm 246	0.0009
Pu 236	0.0017		Pu 236	0.0012		Pa 233	0.0007
Pa 233	0.0011		Pa 233	0.0009		Pu 236	0.0007
Ra 224	0.001		Ra 224	0.0007		U 235	0.0007
Th 228	0.001		Th 228	0.0007		Ra 224	0.0004
U 232	0.001		U 232	0.0007		Th 228	0.0004
U 235	0.0002		U 235	0.0004		U 232	0.0004
Th 234	0.0001		Th 234	0.0001		Th 234	0.0001

In the previous section for PWR analysis, the concept of integral decay heat was introduced with the assumption of a 10 year delay time before calculations began, but there is an obvious benefit of increasing that delay time to reduce short lived isotope contributions to IDH. In the previous decay heat tables, fission products comprise a large amount of the initial heat, thus increasing the delay time could effectively reduce IDH without requiring reprocessing.

A comparison the effect of delay time on fission product IDH is shown in Figure 12-3. As expected, longer delay times offer dramatic decreases in the IDH. As burn up increases, these decreases are greater, but the values are still higher than for lower burn ups.

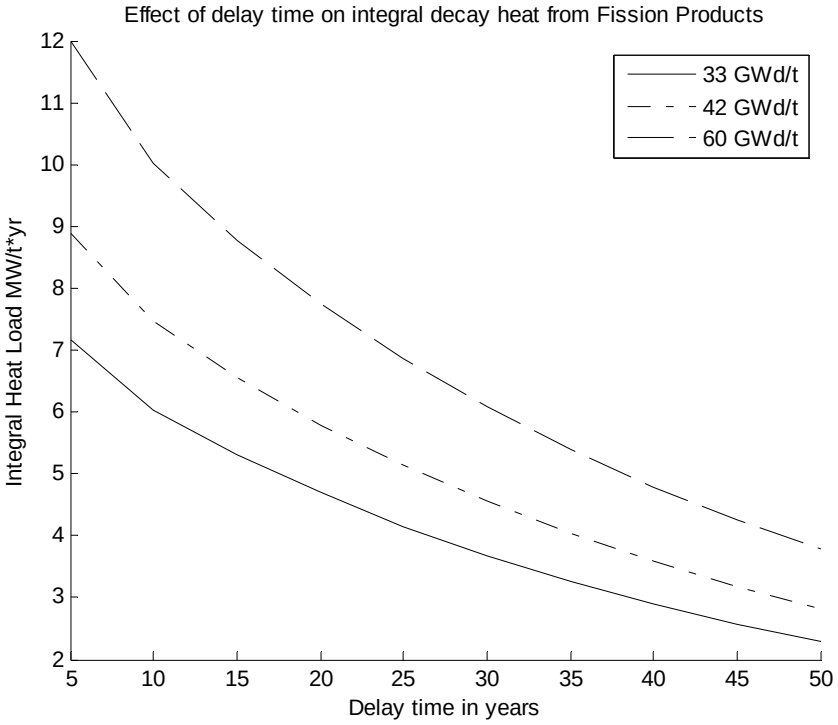

Figure 12-3: Composition of FP burn up dependent IDH for BWR fuel

While much of the initial heat load is from fission products, a greater amount still is supplied by actinides. An illustration showing the composition of the previously used burn ups with respect to fission products, actinides, and total IDH is shown in Figure 12-4. Even though the fission product contribution is greatly decreased with increasing delay time, the actinide IDH is relatively flat over the delay time range due to the long lived isotopes. This shows that repository heat capacity cannot be increased by merely adding a delay time to the spent LWR fuel, thus an advanced fuel cycle that uses

reprocessing and actinide burning fast reactors is needed to reduce this IDH in order to increase repository capacity.

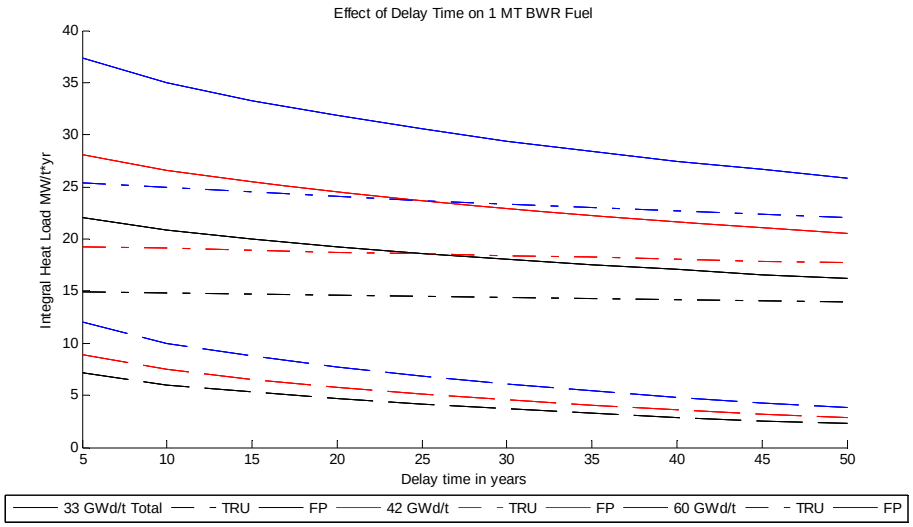

Figure 12-4: Effect of delay time on IDH of BWR fuel

12.3.2 Fast Reactor Heat Load

Using experimental fast reactor ORIGEN libraries, fast fission cross sections and fission product yields are added for the higher actinides that drive the reactions. Initially, SCALE calculates the flux to be nearly 10^{16} n/cm^2, which is rather high compared to a PWR with a flux on the order of 10^{15} n/cm^2. While the cross sections are based on older evaluations, ENDF-V, the magnitude of the flux illustrates the problem of reactivity control in the core. As the reactor burns up, more Pu-239 is created from the U-238 allowing for a longer possible burn up; however, because the 1-D model does not include control mechanisms, blanket fuel, or fuel shuffling, the flux is likely a bit higher than what would actually occur. The k-eigenvector begins at nearly 1.996 and decreases to .0045 as the reactor burns to 200 GWd/t, which is unlikely in a real system, but is believable in a mixed assembly environment with driver and blanket assemblies distributed.

As previously stated, U-238 converts to Pu-239 continuously; this translates to the target conversion ratio only being obtained near the end of the burn ups. The Super PRISM reactor has a target conversion ratio of 0.6 with a high power density and a relatively small core, but the conversion ratio is still burn up dependent as is the case with the LWR. Using a target conversion ratio of 0.75, as the 0.6 conversion ratio is not included in the data; an illustration of the conversion ratio is shown below in Figure 12-5.

At low burn ups, the core actually breeds fuel, but as the burn up increases, the reactor becomes an effective burner. A metal cooled Super PRISM is supposed to reach burn ups of 180 GWd/t with a conversion ratio around 0.6, but here the conversion ratio is about 0.4 for the same burn up. Given that the 1-D code does not account for fuel shuffling, assembly diversity, or control mechanisms, the slight discrepancy in conversion ratio is rather good. A noticeable difference between the burn up conversion ratios for LWR's and FR's is that the FR maintains a linear shape at the higher burn ups, whereas the LWR approaches an asymptote as fissile material decreases.

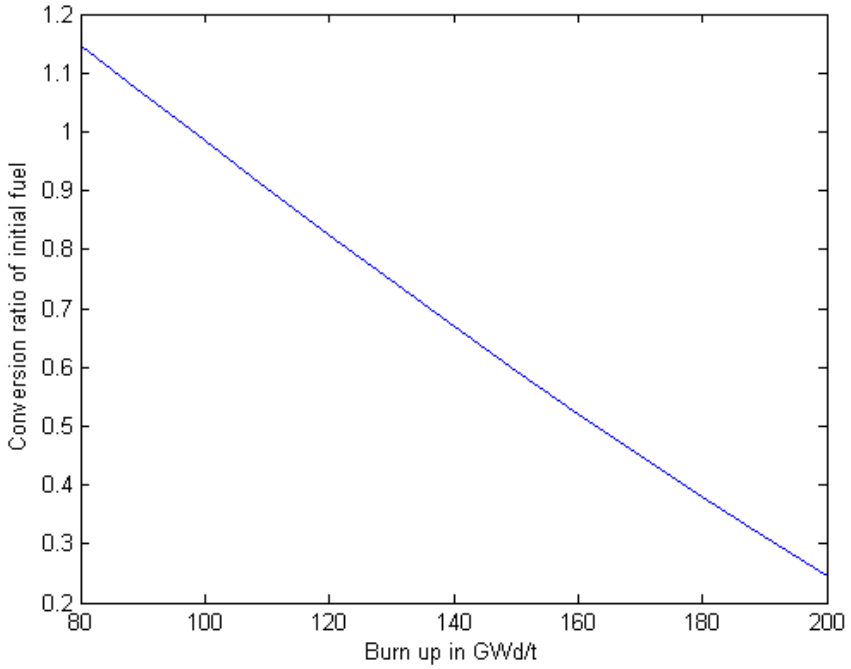

Figure 12-5: Burn up dependent conversion ratio of metal fueled Super PRISM

Unlike the LWR, as the burn up increases, the actinide contribution to integral decay heat decreases as the harder spectrum burns the higher actinides while producing much smaller amounts of higher actinides, as shown in Figure 12-6 below. Actinide contribution decreases almost linearly as the burn up increases, while the fission products show a slight curve as a saturation condition is approached. Total IDH decreases as the burn up does because of the decrease in actinide contribution.

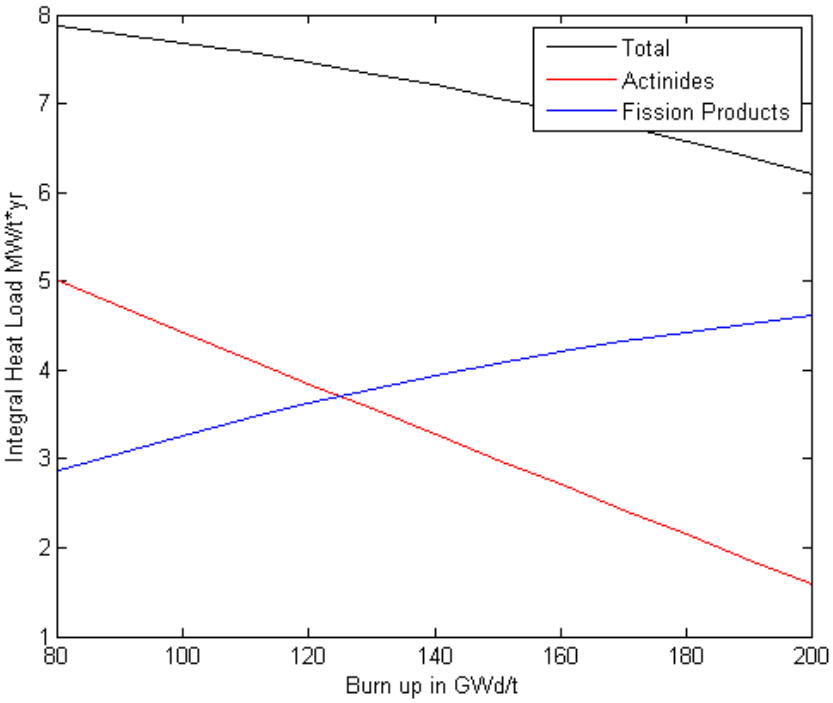

Figure 12-6: Effect of burn up on IDH in metal fueled Super PRISM

Because cross sections differ between the higher actinide isotopes, variations in the production and burn of these isotopes will differ. Individual isotope contribution to IDH is shown in Figure 12-7. Contribution from the plutonium and americium isotopes is still pronounced, but significantly decreased as the burn up increases.

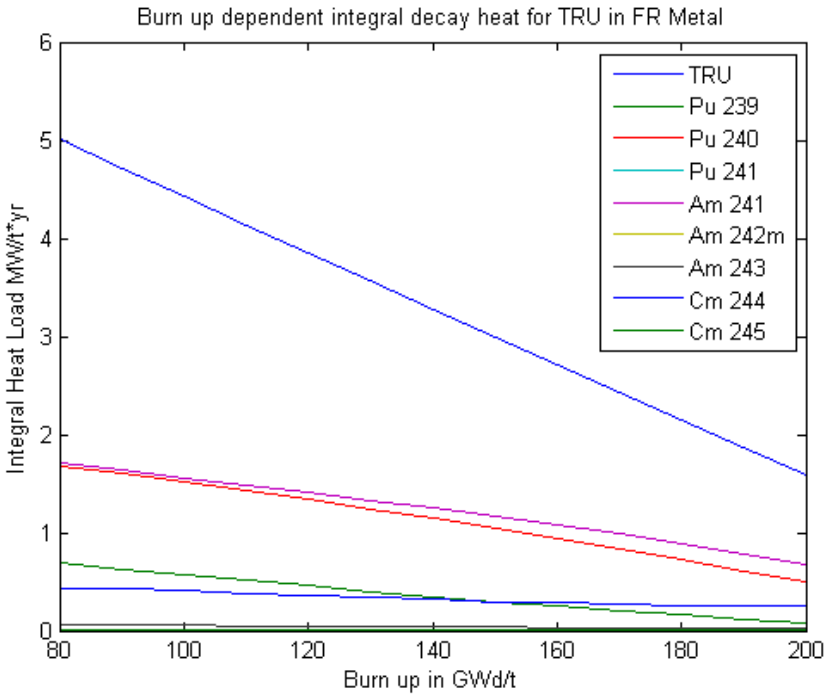

Figure 12-7: Burn up dependent actinide contribution to IDH for Super PRISM

Exploiting the decrease in total decay heat can be improved if a delay time is added after the burn. Using the same parameters as with the LWR delay time cases, the delay time dependent IDH is shown in Figure 12-8. As shown previously with the LWR's, delay time has little effect on the actinide contribution, but has a significant effect on the fission product contribution. In this instance of the 0.75 conversion ratio metal core, the fission product contribution is halved with a 20 year delay time. This is rather important as fast reactor spent fuel is not reprocessed in the hybrid fuel cycle, thus this is a significant savings to the repository heat load. The effect of burn up is also significant as the heat load is significantly less than for higher burn ups included with the effect of delay time.

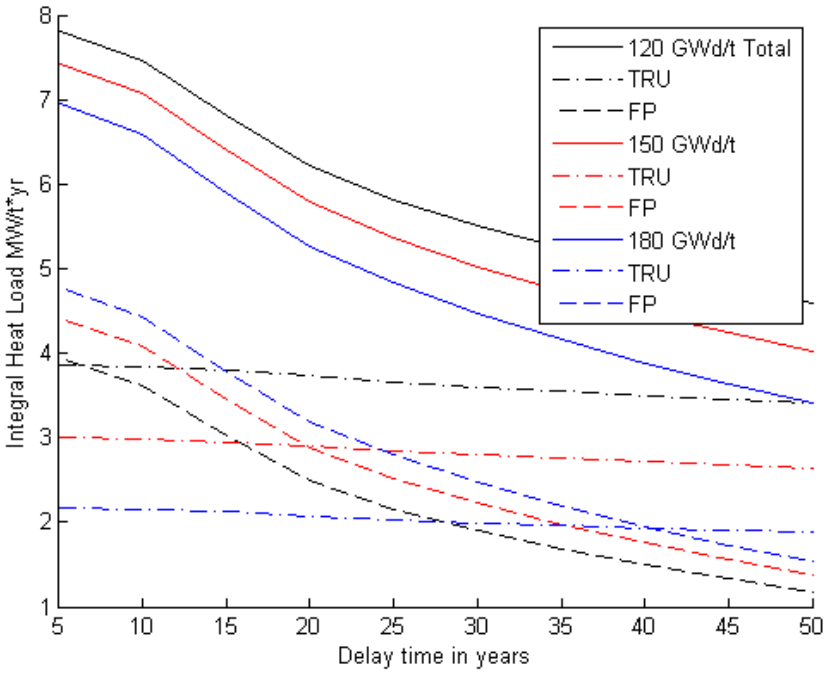

Figure 12-8: Effect of delay time on IDH in metal fueled Super PRISM

12.3.3 Repository Capacity Gain from Hybrid Cycle

A primary focus of fuel cycle analysis is the closing of the fuel by storing spent fuel indefinitely in a geologic repository. Currently, the Yucca Mountain Project is nearly at full capacity without having opened; thus, the hybrid fuel cycle aims to increase that capacity through reprocessing and fast burner reactors to reduce the amount of material that contributes to the heat load.

Increasing the capacity through removal of higher actinides and fission products requires the reprocessing stage to create enough fuel for the fast reactor. A PWR core has about 100 MT of UOX, while the Super PRISM has around 26 MT for the 0.75 conversion ratio metal fueled reactor. While a single PWR outputs a significant amount of plutonium that may be burned in the fast reactor, it cannot support a fast reactor by itself, but rather requires 11 PWR's of similar characteristics to support a single Super PRISM. The major reduction in heat load comes from the removal of higher actinides from spent fuel of multiple reactors by burning it in a low conversion ratio fast reactor to further decrease the higher actinide inventory.

In order to illustrate the factor increase in repository capacity, Table 12-7 provides a range of possible combinations with the respective factor increases. Using the same 11:1 ratio of PWR's to FR's as previously stated, the integral decay heat footprint of a single fast reactor is less than 11 PWR's. A single 0.75 conversion ratio metal fueled Super PRISM requires nearly 4200kg of transuranics, with remaining depleted uranium readily available and excess able to be stored at the WIPP site. For 11 PWR's, nearly 10 years worth of spent fuel is required for starting up a single fast reactor of this type. Removing the heat load footprint of 11 PWR's and replacing it with a single FR can reach a seemingly optimal factor increase of 4.6; however, this factor increase assumes a long burn up for the PWR and a short burn up for the fast reactor because of the large contribution of fission products to the heat load as fast reactor burn up increases. Looking at the individual components, the combination of both long burn ups yields an increase factor of 13 increase from actinide removal. This does raise the question of reprocessing the fast reactor fuel to gain an even larger increase, but given that the reprocessing facility cost for handling such high activity would be tremendous, it settles the idea that a second reprocessing step after fast burner reactors would be improbable.

Table 12-7: Factor increase in repository heat load capacity using hybrid fuel cycle

Burn Up		Component Factor		Combined Factor
LWR	**FR**	**FP**	**TRU**	**Total**
60	120	3.935528	7.212013	4.611128
60	160	3.237622	9.857225	4.13265
60	200	2.89714	13.0786	3.868326
45	120	3.285243	4.503818	3.536509
45	160	2.702655	6.155722	3.16954
45	200	2.418432	8.167431	2.966816
30	120	2.613046	2.458725	2.581226
30	160	2.149662	3.360532	2.313382

12.3.4 Economic Cost Analysis

Using the uncertainty analysis code to perform a Monte Carlo sampling of the parameters found in Table 1 and, the cost per kilowatt-hour is found on the basis of a hybrid fuel cycle. Estimates from an EMWG report are compared with estimates found using the Monte Carlo code with parameter estimates used by the author in Table 12-8 [35, 36]. A probability density function of the electrical generation costs for the metal fueled fast reactor hybrid cycle is shown below in Figure 12-9.

Table 12-8: Comparison of hybrid cycle fuel costs with EMWG

Parameter	Low cost in mills/kWh	High cost in mills/kWh
EMWG	4.4	6.5
Author	35.9817	77.9079

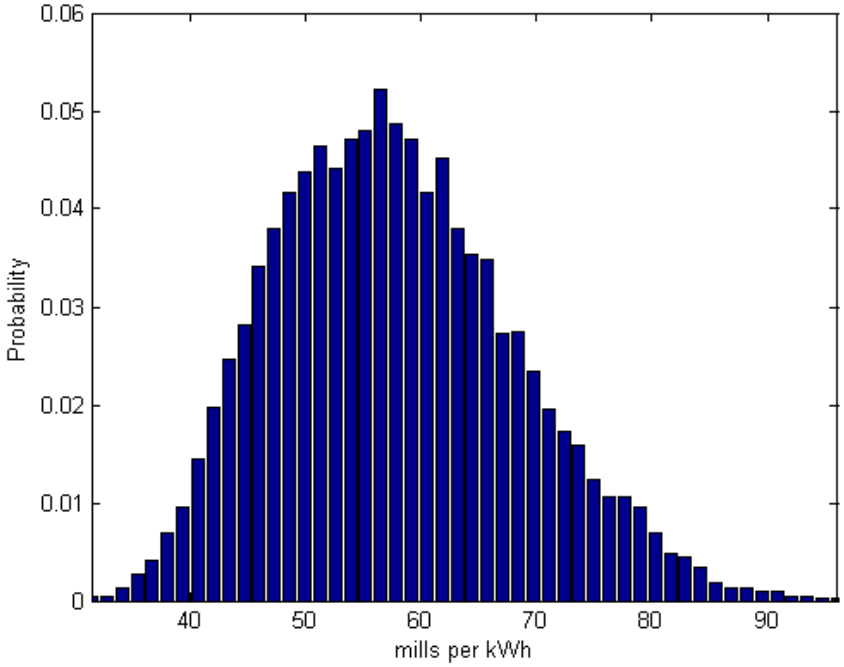

Figure 12-9: Estimate of cost of electricity generation of hybrid fuel cycle

Estimates are far higher than the EMWG reports for a number of reasons. Firstly, the estimates here are for an equilibrium fuel cycle, where all LWR fuel is reprocessed, requiring massive amounts of reprocessing capability that does not currently exist, thus the high estimates for reprocessing facility cost greatly affects the overall fuel cycle cost. Second, many unknowns exist for metal fuel fabrication and estimates for facility and R&D costs have been combined here in what is likely a significant overestimate on the cost. Combining costs into fewer, larger estimates is likely introducing error into the model along with unreliable cost estimates.

12.3.5 Isotopic Minimization

12.3.5.1 Optimization Results for LWR Plutonium Inventory Minimization

Using Pu-239 as a variable of interest, the *MinIso* function is used to drive the optimization. Results for both PWR and BWR reactors are shown in Table 12-9. As expected, results for both reactors are similar as they are reaching the constraints for burn up and delay time. Because there is no check against burning a reactor further than what the initial enrichment could reasonable attain, the optimal parameter for enrichment is insignificant.

Table 12-9: Optimal parameters for Pu-239 minimization

PWR			BWR	
Parameter	**Values**		**Parameter**	**Values**
Burn Up	59.7 GWd/t		Burn Up	59.9 GWd/t
Enrichment	4.01%		Enrichment	4.73%
Delay Time	29.56 years		Delay Time	29.91 years
Optimal Value	6072 g/t		Optimal Value	4123 g/t

As stated in the previous section on LWR spent fuel analysis, reducing the amount of higher actinides produced is a foregone conclusion as longer burn ups require less fuel than lower burn ups, while creating slightly more actinides for a single fuel batch. Because of this, any useful fuel cycle optimization really depends more on the type of reactor used as this relates back to the core mass differences between the LWR and PWR.

12.3.5.2 Optimization Results for FR Plutonium Inventory Minimization

Using Pu-239 as a variable of interest, the *MinIso* function is used to drive the optimization using a PWR driven hybrid fuel cycle. Results for both the oxide and metal fueled reactors are shown in Table 12-10. Because as FR burn up increases, the amount fissile material decreases, longer burn ups are the obvious result. Delay time plays a very small role in the reduction, but still manages to converge to the upper limit. Higher burn ups in LWR's lead to lower conversion ratios; thus, the higher burn up fuel reduces the Pu-239 inventory as it absorbs neutrons to form higher actinides while some fissions. LWR enrichment here is an independent variable from the burn up, while this is not necessarily true in practice as higher enrichments are needed to reach higher burn ups, thus the enrichment should follow the burn up trend more closely than it does.

Table 12-10: Optimal parameters for Pu-239 minimization using fast reactors

Metal CR=0.75			Oxide CR=0.75	
Parameter	**Values**		**Parameter**	**Values**
LWR Burn Up	59.4 GWd/t		LWR Burn Up	59.8 GWd/t
Enrichment	4.67%		Enrichment	4.78%
Delay Time 1	29.67 years		Delay Time 1	29.75 years

FR Burn Up	199.7 GWd/t		FR Burn Up	179.8 GWd/t
Delay Time 2	29.78 years		Delay Time 2	29.82 years
Optimal Value	11293.3g/t		Optimal Value	21192.7g/t

12.3.6 Heat Load Minimization

12.3.6.1 Optimization Results for LWR Heat Load Minimization

Using total heat load as a variable of interest, the *MinHeatLoad* function is used to drive the optimization. Results for both PWR and BWR reactors are shown in Table 12-11. As expected, results for both reactors are similar as they are reaching the constraints for burn up and delay time. As stated in the previous section, there is no check against burning a reactor further than what the initial enrichment could reasonable attain, thus optimal parameter for enrichment is insignificant.

Table 12-11: Optimal parameters for IDH minimization

PWR			BWR	
Parameter	**Values**		**Parameter**	**Values**
Burn Up	59.52 GWd/t		Burn Up	59.87 GWd/t
Enrichment	4.61%		Enrichment	4.23%
Delay Time	29.74 years		Delay Time	29.89 years
Optimal Value	38.67 MWd/t*yr		Optimal Value	31.34 MWd/t*yr

Again the issue of core size affects the seemingly optimal parameter as the heat load for 1 MT of PWR fuel is significantly higher than for 1 MT of BWR fuel; however, because there is 50% more BWR fuel, the advantage again goes to the PWR. As with the isotopic minimization cases, the optimizations quickly approach the upper bounds.

12.3.6.2 Optimization Results for FR Heat Load Minimization

Using total IDH as the variable of interest, the *MinHeatLoad* function is used to drive the optimization using a PWR driven hybrid fuel cycle. Results for both the oxide and metal fueled reactors are shown in Table 12-12. As with previous cases, the longer LWR fuel is burned, the larger the resulting heat load will be from higher actinide and fission product production, thus a short burn up will minimize the heat load. Again, the enrichment is selected at random and does not converge to any number as it should. There is a correlation between burn up and enrichment that is not accounted for in the SCALE data production. Delay times for both the LWR and FR spent fuel approaches the upper bound of 30 years. FR burn up follows a reverse trend than the LWR's. Because the initial fuel loading is such that high burn ups intend to reach a low conversion ratio, the longer the fuel is burned, the integral decay heat decreases as is the

opposite for the LWR case. In LWR's, both fission product and actinide contributions to IDH increase with burn up, but Figure 12-6 showed the inverse relation of actinide and fission product contribution to IDH for a fast reactor.

Table 12-12: Optimal parameters for IDH minimization using fast reactors

Metal CR=0.75			Oxide CR=0.75	
Parameter	**Values**		**Parameter**	**Values**
LWR Burn Up	30.2 GWd/t		LWR Burn Up	30.1 GWd/t
Enrichment	3.61%		Enrichment	4.29%
Delay Time 1	29.92 years		Delay Time 1	29.61 years
FR Burn Up	199.2 GWd/t		FR Burn Up	179.7 GWd/t
Delay Time 2	29.72 years		Delay Time 2	29.79 years
Optimal Value	55440/t		Optimal Value	76012/t

12.3.7 Economic Cost Minimization

12.3.7.1 Optimization Results

In this minimization problem, the total economic cost is the parameter of interest, and fuel cycle parameters are generated in the algorithm that are used in the calculations. Some parameters must be held constant if the algorithm is to work correctly, otherwise the number of generations is quickly exhausted.

As with the previous minimization cases, the problem is rather trivial as the optimal parameters all approach their respective upper limits, as shown in Table 12-13. Previously mentioned in the theory, high burn up reactors cannot run with low initial enrichments, thus a correction is needed to fix this discrepancy. The optimal parameter for LWR enrichment is shown with an alternate because the original algorithm did in fact minimize the cost by lowering the number of SWU's required to enrich the fuel to a high percentage, but this would have created an improbable situation. A simple linear relation was added to the fitness function that returns a minimum required enrichment for the sampled burn up. While a linear relation may not be completely appropriate, it does illustrate the need for a correction in the Monte Carlo sampling algorithm.

Table 12-13: Optimal parameters for economic cost minimization

Parameter	Optimal Parameter (Worst Case)	Optimal Parameter (Best Case)
LWR Enrichment	4.99% (alternatively 3.01%)	4.99%
LWR Burn Up	59.76 GWd/t	59.63 GWd/t

Delay Time (after LWR)	29.79 years	29.81 years
Burn up (metal fueled CR=0.75)	199.43 GWd/t	199.67 GWd/t
Burn up (oxide fueled CR=0.75)	179.72 GWd/t	179.91 GWd/t
Delay Time (after FR)	29.87 years	29.96 years
Cost per kWh of electricity	79.82 mills/kWh	36.97 mills/kWh

In the both the worst case, where all fixed costs are at the maximum, and the best case, where all fixed costs are at the minimum, the algorithm predicts a much higher cost of electricity than estimated by the EMWG [32,34,35,36,37,38]. Partially, this is due to the requirement of an equilibrium cycle, where all LWR fuel is reprocessed, which is not the case for the EMWG reports. Therefore some costs associated with multiple reprocessing facilities likely increase the estimates by this large factor. Also, here there is no constraint to keep electrical generation at a constant or a limited growth condition that would restrain the prices. Within the algorithm, all plants are assuming an Nth of a kind, where prices would reflect a long experience with construction and operation, which would decrease the estimates with respect to the EMWG studies [35].

A missing piece to this is a bottom up cost estimation. Because this model assumed a top-down estimate, where unknowns are scaled from similar designs, the estimates could be significantly inaccurate. In an ideal situation, as it is with the LWR cases in the EMWG reports, detailed information for the construction and operating costs, from initial design through decommissioning, allow a much better estimate to the actual cost of a new plant. Unfortunately, this does not exist at the current for the Super PRISM or large scale reprocessing facilities. While this is a guess as to the costs, it is significantly higher than some other energy sources because of the imperfect model.

Comment [FRM1]: Item 9 problems and delays and items 10 and 11 can be omitted if they do not apply.

13. Conclusions and Opinions

There are numerous options for potentially improving the sustainability and economic viability of nuclear power; however, in our opinion, there is no "best" option. One of the motivations for reprocessing and using breeder reactors during the 1950s and 1960s was the concern of the limited availability of uranium to support a once-through LWR fuel cycle. Recently, publications describe technology for extraction of uranium from seawater as an economic option. Thus, the limited availability of uranium may no longer be a valid reason for the development of reprocessing and advanced fuel cycles. Results reported herein show that the once through fuel cycle can produce electricity for less cost than other advanced fuel cycles considered. The overall life cycle cost for advanced fuel cycles is very uncertain since neither the cost of direct geologic disposal nor cost of reprocessing and destruction of actinides is well known (possibly no better than a factor of five or more).

Results obtained in this study show that modest (about 30 %) reductions in the required repository capacity can be obtained by increasing fuel burnup from about 40 to about 70 GWd/t. If Pu, Am, Cm and Np can be removed from spent fuel and recycled, then the 1500-year integrated heat load on the repository can be reduced by a factor of more than 50. Although this has the potential for very significantly reducing costs for geologic disposal, the time and cost required for development of reprocessing and remote fuel fabrication facilities to make this option possible most likely has an uncertainty of at least a factor of five. Since the 1 mil/kW-hr surcharge on nuclear power is utilities' charge for disposal of spent fuel, there is little commercial basis for developing advanced fuel cycles at this point in time, given current economic constraints and potential benefits.

Incorporation of fast burner reactors into the LWR fleet can limit ex-core Pu inventories, where with the once-through ex-core Pu and minor actinide inventories continue to grow linearly with time for a fixed number of reactors operating on the once-through fuel cycle. The number of FRs required to burn Pu and MAs from 100 LWRs could range from about 20 to 50, depending on the target designs, conversion ratios, etc. For example, if inert matrix fuel is used, the 20 equivalent-power FRs could burn all to the Pu from the LWRs annually. On the other hand the annual production of Pu and MAs from 100 LWRs could be recycled into 20 to 100 LWRs, depending on fuel design, fuel management, and reactor safety considerations.

Recycle of Pu and minor actinides into LWRs is an option currently being researched, and it may be economically advantages to accomplish destruction of actinides in LWRs and to not build fast reactors. Results from economic analyses indicate that the fuel cycle with the most uncertainty in cost is recycle of Pu into LWRs. This is a consequence of uncertainties assigned to infrastructure facilities relative to use of FRs, but these uncertainties are based on literature and results form an expert elicitation that may or may not be accurate. The cost for recycle of actinides into LWRs may be more or less economic than use of FRs to destroy actinides. Another that was evaluated, but not in depth, is to use the thorium/uranium fuel cycle. This fuel cycle produces about a factor

of 100 less problematic actinides, but it introduces numerous fuel handling and fabrication challenges.

Given the results and uncertainties of results obtained in this study the following conclusions are cited:

1) use of high burnup fuel (~70 GWd/t) could reduce the 1,500-year integral heat load on the repository by about 30 % relative to ~40 GWd/t burnup fuel,
2) twenty to 50 FRs are required to burn the production of Pu and MAs from 100 LWRs of equivalent power,
3) the once-through fuel cycle is the more economical than MOX recycle into LWRs or use in FRs,
4) there is less uncertainty associated with the cost of the once-through fuel cycle than the MOX recycle into LWRs or use in FRs,
5) there is less uncertainty in the cost of FRs than in MOX recycle, and
6) the cost of FRs to burn actinides may or may not be less than MOX recycle into LWRs because of the excessive uncertainty in the MOX recycle into LWRs fuel cycle.

Some opinions are as follows:

1) MOX fuel should not be recycled into LWRs,
2) FRs should be used to burn MAs and should operate with conversion ratios of near unity,
3) The delay time between withdrawal of spent fuel and reprocessing should be about 30 years,
4) The delay time between reprocessing and geologic disposal should be about 100 years,
5) A solvent extraction type reprocessing plant should built with a capacity of at least 1,000 metric tons per year,
6) Research on alternative reprocessing methods should be very actively pursued,
7) Interim storage sites with the expectation to be operated for several hundred years should be developed, and
8) Geologic storage of spent fuel should be delayed by about 100 years.

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Appendix A: Dynamic Analyses of Nuclear Energy System Strategies, Luc Van Den Durpel (ANL/NE)

DANESS, i.e. Dynamic Analysis of Nuclear Energy System Strategies, has been developed by ANL since the year 2000 and has been applied in a variety of projects (CA-Waste, Miller L.F. et al. 2007, OECD/NEA 2006, Roelofs F. 2008, Van Den Durpel et al. 2005) by universities, R&D-labs, industry and radioactive waste management organizations. DANESS is used within the NERI-project coordinated by Prof. Larry Miller (UTK) for the assessment of LWR and FR based nuclear energy systems in the USA.

DANESS is based on a system dynamics model, using the iThink-software (iseesystems), allowing to simulate the dynamic behavior of systems including multiple components and to simulate and investigate the dynamic interdependence of these components interacting between each other via feedback loops. System dynamics software also provides an easy way of communicating the set-up of models and the outcome of the simulations while also providing a good framework for quality assurance and control of the models.

Since early 2008, new functionalities have been developed within DANESS based on the latest version of the iThink-software and allowing hierarchical model architectures avoiding any model-size limitations for DANESS and allowing to perform multi-layered or hierarchical models. This hierarchical model architecture has been deployed at full from DANESS version 4.0 on which came available since late summer 2008 on. The architecture of DANESS v4.0 is shown in figure 1.

DANESS may start a case simulation from an existing nuclear energy system, composed of an existing NPP-park and fuel cycle facilities (including inventories of fuel and fissile material), and deploy time-varying nuclear energy systems where the deployment is defined according four possibilities:

- The user pre-defines to DANESS the NPP-park and nuclear fuel cycle capacity deployment, i.e. annual capacity extensions per type of NPP and fuel cycle facility, where the detailed mass-flow and economic and environmental analysis is then undertaken by DANESS;
- The deployment of the NPP-park is described as desired NPP-park fractions in the future and where DANESS will deploy the NPP-park and associated nuclear fuel cycle facilities accordingly to match this requested NPP-park in the future;
- The deployment of the NPP-park is driven by economic decision making where NPPs are ordered according their competitiveness with other NPP-types or with other non-nuclear energy generation technologies;
- Any combination of above options, for instance, NPP-park defined based on economic competitiveness with nuclear fuel cycle facility deployment exogenously defined by user.

As such, DANESS can be used in two essential modes, i.e. (nuclear) energy demand driven or exogenously defined NPP-capacity driven.

The life history of each of the ordered NPPs is followed until decommissioning of these NPPs and allowing to interact with other models, i.e. especially the 'Nuclear Fuel Cycle Option' model and the 'NPP Investment & Cash-Flow' model. These models may influence the decision making to order new NPPs due to, for instance, fissile material limitations and/or bad economics of the specific NPP.

The 'Nuclear Fuel Cycle Option' model analyses the various fuel cycle options (e.g. UOX or partial MOX for LWRs) available for the NPPs and then decides, based on preferences defined by the user or economic considerations, on the fuel to be used by the NPPs at each moment in time. This model interacts with the 'Nuclear Fuel Cycle Mass Flow' model which performs a detailed mass-flow analysis from cradle-to-grave, i.e. from U-mining till disposal of the ultimate waste. Obviously, inherent to system dynamic models, all operations in the nuclear fuel cycle facility are modeled based on flow-sheets of these facilities and thus also the time-delays in the nuclear fuel cycle. This 'Nuclear Fuel Cycle Mass Flow' model is very crucial within DANESS and interacts virtually with all other models in DANESS. Specifically, the 'Facility History' model specifies the available capacity for the various fuel cycle facilities which may be a limiting factor in the development of nuclear energy systems. This model also holds the capability to deploy nuclear fuel cycle facility capacity according the projected needs from the 'Nuclear Fuel Cycle Mass Flow' model. Once more, each nuclear fuel cycle facility follows a life-path from ordering till decommissioning while, as for the 'Reactor Technology Development' model, the 'Facility Technology' model may introduce delays in facility capacity deployment scenarios mimicking the transition towards an industrially available technology though impacting the fuel cycle mass flow and NPP ordering models by projecting the future availability of fissile materials and fuels for the various NPPs.

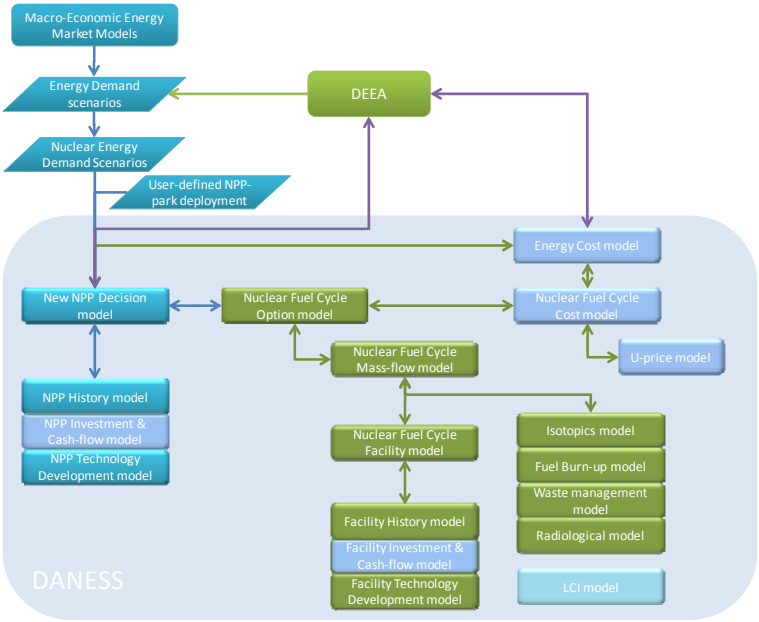

Figure A-1. Architecture of DANESS v4.0[1]

[1] Only the main interactions between the various sub-models are shown in this figure.

The ‘Fuel Cycle Mass Flow’ model makes use of four other models relating to:

- the isotopic follow-up of the fuels and fissile materials throughout the nuclear fuel cycle (the ‘Isotopics’ model). This isotopic model allows to trace 68 isotopes explicitly throughout the whole nuclear fuel cycle as well as two lumped representations of short- and long-lived fission and activation products that are not explicitly modeled. The list of isotopes (typical standard list shown in table 1) can be defined by the user depending on the specific scope of the assessment undertaken, e.g. one may be interested in operational dose aspects at fuel cycle facilities or in waste management aspects which involves different set of key isotopes to be important. Parameterization sheets for this ‘Isotopics’ model are provided to the user for these different scopes and multiple ‘Isotopics’ models may be run in parallel in case a user wants to cover both scopes in one simulation;
- the calculation of fresh and spent fuel compositions as function of burn-up of the fuels and/or of conversion ratio for fast reactors (the ‘Fuel Burn-up’ model). While DANESS is essentially using tabled fresh and spent fuel compositions for a variety of fuels and NPP combinations, this model allows to perform more detailed simulations with time-varying fuel burn-ups or core compositions. Part of the database has been developed under this NERI project in addition to the data related to developments undertaken by ANL itself;
- the ‘Radiological’ model calculates the activity, dose and decay heat for the various fissile material flows and inventories throughout the nuclear fuel cycle and in the disposal site;
- the ‘Waste Management’ model analysis the disposal size needed to dispose of the ultimate waste from the nuclear energy systems where a Yucca Mountain model and a Clay repository model has been developed so far. These models are essentially considering thermal load limitations for the disposed waste and feed-back to the other models any capacity limitations (being it restriction of mass or volume of waste being disposed of or economic cost changes per unit waste to be disposed of) which may trigger the ‘Fuel Cycle Option’ model to choose for other fuel cycle options for certain or all NPPs or even hint towards other NPPs, e.g. fast reactors, to be deployed to mitigate these waste disposal limitations.

For each of the nuclear energy system components, i.e. NPPs and fuel cycle facilities, a detailed investment and cash-flow analysis is undertaken in various models as shown in figure 1 where all these economic analyses leads to a nuclear fuel cycle cost per type of fuel and an energy generation cost per NPP or for the whole NPP-park. The DANESS model also tracks the cash-flow between NPPs and fuel cycle facilities allowing to use (user-specific) investment models for these NPPs or fuel cycle facilities.

U232	Np235	Th227	Cs134	Pd107
U233	Np236	Th228	Cs135	Ru106
U234	Np237	Th229	Cs137	Ag108m
U235	Am241	Th230	Sb125	Ag110m
U236	Am242m	Th232	Sn126	C14
U237	Am243	Th234	Pm147	Cl36
U238	Cm242	Pa231	Ce144	Ni59
Pu236	Cm243	Pa233	Eu154	Ni63
Pu237	Cm244	Ac227	Eu155	Ca41
Pu238	Cm245	Rn222	Sm151	Mn54
Pu239	Cm246		I129	Co60
Pu240	Cm247		Kr85	Sr90
Pu241	Cm248		Se79	SLFP
Pu242	Cm250		Zr93	LLFP
Pu244			Nb94	
Pu246			Tc99	

Table A- 1. Standard list of isotopes explicitly modeled in DANESS v4.0

The NPP-park evolution and the associated economics for each of the NPPs may also be coupled to a companion model called Dynamic Energy Economics Analysis (DEEA) which performs an energy market competitiveness analysis by simulating the cumulative supply curve versus energy demand curve and defining the market share for fossil-based, renewable and nuclear energy generation technologies (and also exogenously defined import).
The environmental life cycle inventory of all NPPs and nuclear fuel cycle facilities is analyzed in the 'LCI' model which is currently under development and verification and to be made available during 2009.

Each DANESS-model covering the previous capabilities may simulate in parallel 10 NPP-types and 10 fuel-types though the number of NPPs in a case simulation is not limited at all.

A unique feature of DANESS is the generic model architecture allowing to use the model for various applications without the need to reprogram or customize the model as such. All parameterization is provided via import of MS-Excel case specification sheets allowing to use the same DANESS-model to simulate one NPP or a world NPP-park covering multiple NPP-types.

New capabilities in DANESS v4.0

A diverse development program covering new functionalities and applications of DANESS has been set-up during the past few years including collaboration with various users of DANESS. The main new development relates to the hierarchical model architecture which came available since version 4.0 in 2008 and, as shown in figure 2, allows to expand the capabilities of DANESS in two main directions, i.e.:

- More detailed models composed of gradually more refine sub-models, i.e. layered structure of models with each part of a model consisting of a sub-model, are possible allowing a significant expansion of capabilities, e.g.:
  - More detailed flow-sheets for nuclear fuel cycle facilities may be used where these detailed flow-sheets may be hidden or locked for (non-authorized) users;
  - Existing models, such as the 'Isotopics' model, may be used multiple times in one DANESS model where each of these 'Isotopics' models are parameterized differently depending on the precise scope of the assessment by the user;
- Multiple DANESS-models may be run in parallel for multi-regional nuclear energy system assessment studies and allowing to interact between the various DANESS-models via an overarching model defining the mass-flow and cash-flow between the different DANESS-model or regions being modeled:
  - This also allows virtually unlimited detail in the assessment of nuclear energy systems, e.g. each of the parallel DANESS-models may be simulating a small NPP-park in all its detail with the combined set of DANESS-models simulating a country or world-region. Where previous DANESS-models were limited in case size due to software-limitations, this new version is not at all;
  - Coupling with other system dynamic (and later-on other software) models is made much easier and specifically the combinations of DANESS with DEEA (see figure 1).

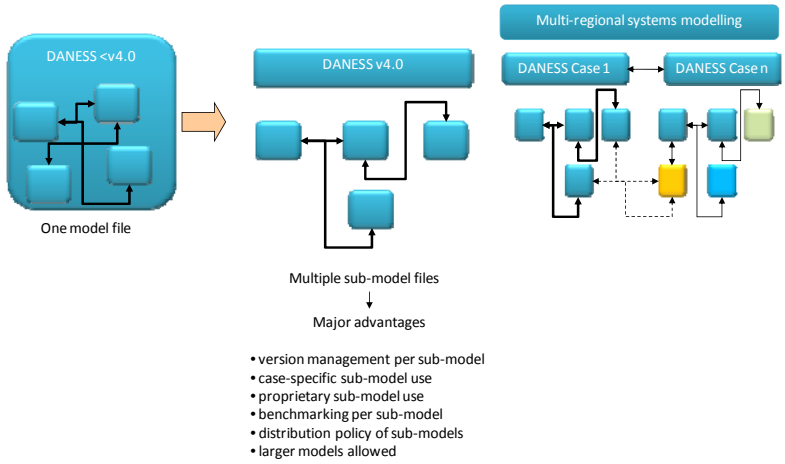

Fig A-2

Figure A- 2. New functionalities in DANESS v4.0 based on hierarchical model architecture.

This hierarchical model architecture also allows to have a different versioning-management and quality assurance program in place for the various models within what we may call from now on the DANESS-toolbox. Some of these have been indicated in figure 2 with two being very important, i.e. the possibility to use case or user-specific models for, for instance, detailed fuel cycle facility flow-sheets which are specific or commercially restricted for other use; and the possibility to use multiple DANESS's, each with sometimes specific models, in parallel for multi-regional nuclear energy system assessment applications.

Sustainability Dimension	Criterion	Indicator	Unit
Economic	Economic competitiveness	Levelised cost of energy generation	€/MWhe
		Levelised cost of nuclear fuel cycle	€/MWhe
		Levelised cost of waste management	€/MWhe
		Split-up per LLW, ILW, HLW	€/MWhe
	Financial investment	Total investment required for design and construction, operation and maintenance and decommissioning of necessary fue cycle and waste management facilities	[1..5]
		Annual operational cost waste management plants	M€/year
	Technological availability	Time required to reach desired end state in waste management	years
		Flexibility to switch to other nuclear fuel cycle scenario	[0..5]
		Period needed to dispose of all waste of a scenario ?	years
Environmental	Non-renewable resources use	Energy generated per ton U_{nat} mined	TWhe/tUnat
	Energy intensity	Ratio of energy used per energy generated	%
	Transport intensity	Amount of tonnes-kilometers needed per TWhe generated	t.km/TWhe
		Transport intensity	#/TWhe
		Number of LLW, ILW, HLW-transports	#/Twhe
	Land use	Size of geological disposal site	
		Footprint	km² / TWhe
		Disposal gallery length	m / TWhe

		Amount of land used for surface facilities	km²/TWhe
	Amount of Waste to be managed	Total volume of waste	m³/TWhe
		Volume of LLW, ILW, HLW waste	m³/TWhe
	Activity of disposed waste	Short-lived, i.e. T1/2 ≤ 30 yrs	TBq/TWhe
		α-emitters	
		β, γ -emitters	
		Long-lived, i.e. T1/2 > 30 yrs	TBq/TWhe
		α-emitters	
		B, γ -emitters	
	Radiotoxicity evolution of disposed waste	Radiotoxicity evolution of disposed waste	mSv/TWhe
		Radiotoxicity of disposed waste at 1000 years after emplacement	mSv/TWhe
	Decay heat of disposed waste	Decay heat evolution of disposed waste	kWth/TWhe
		Decay heat of disposed waste at 1000 years after waste emplacement operations	kWth/TWhe
	U_{nat}-equivalent radiotoxicity reduction period	Time needed to return to U_{nat}-mined natural radiotoxicity level	years
	Radiological Risk of Waste Disposal	Estimated maximum dose rate to an individual of the critical group	mSv/yr.TWhe
	Amount and Duration of storage	Maximum amount of SF to be stored	tHM
		Median of Duration of storage of SF	years
		Maximum amount of HLW to be stored	tHM
		Median of Duration of storage of HLW	years
Socio-Political	Siting of waste facilities	Number of waste management facilities needed	#
		Time needed before second unit repository is needed	years
	Proliferation risk	Amount of separated transuranics	tHM
		Pu	tHM
		MA	tHM
		Proliferation metric	[0..10]

Table A- 2. List of assessment criteria and indicators available in DANESS v4.0

Assessment criteria and indicators used by DANESS v4.0

As DANESS is used for the assessment of nuclear energy systems, a variety of technical-economic, environmental as well as socio-political criteria and indicators have been implemented allowing to perform comparative assessments of different nuclear energy systems using a consistent set of criteria and indicators. The list of criteria and indicators is listed in table 2 and covers virtually all indicators and criteria considered in many of the assessment studies referred to before. Some of the criteria and indicators are expressed on a numeric scale, e.g. [0..5], when they are a combination of other values calculated by DANESS or where the binning of the results is more appropriate.

Benchmarking of DANESS v4.0

A variety of benchmark or verification activities have been undertaken with DANESS, especially with the previous versions, where a renewed set of benchmark exercises is undertaken to confirm the correctness of modelling used in DANESS v4.0. Benchmarking is currently ongoing specifically re-establishing the correct application of the 'Isotopics' model throughout the whole nuclear fuel cycle and verification of the waste management models with published safety assessment studies. These benchmarks are undertaken in the context of EU-projects, IAEA-projects, OECD/NEA-projects and also one under the auspices of MIT.

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Appendix B: Questionnaire for Uncertainty Analyses of Advanced Fuel Cycle

This expert elicitation is distributed to facilitate uncertainty analyses for a NERI grant on "Uncertainty Analysis of Advanced Fuel Cycles." Results from this survey will be used to define distributions for parameters used to model fuel cycle scenarios with two codes. One is the DANESS code developed by Argonne National Laboratory and the other is a Matlab code written by a graduate student at The University of Tennessee. The uncertainty analyses will be performed by a Monte Carlo sampling method. Results will be ranked and will be evaluated using non-parametric statistical methods.

If you would like to receive a report on this study please provide your email address. If you feel that you are not qualified to answer a particular question you may leave it blank.

1. If the US production of nuclear power is to remain constant, in what decade will (should) construction of new reactors need to begin?

	2010	2020	2030	2040	2050
Will:	76%	18%	6%	0%	0%
Should:	94%	6%	0%	0%	0%

If the international production of nuclear power is to remain constant, in what decade will (should) construction of new reactors need to begin?

	2010	2020	2030	2040	2050
Will:	100%	0%	0%	0%	0%
Should:	94%	6%	0%	0%	0%

2. Given the current, and expected, infrastructure for construction of nuclear plants, how many LWRs (on average) will (should) be constructed per year during the next 20 years in the US?

	1	3	7	11	>11
Will:	44%	38%	13%	0%	6%
Should:	0%	44%	25%	19%	13%

	1	3	7	11	> 11
Should:	0%	44%	25%	19%	13%

3. Given the current, and expected, infrastructure for construction nuclear plants, how many LWRs (on average) will (should) be constructed per year during the next 20 years in all countries excluding the US?

	5	10	20	30	> 30
Will:	31%	25%	31%	6%	6%
Should:	0%	38%	31%	19%	13%

4. Given the trend for increased burn up fuels, how high of a burn up will (could) be achieved by PWRs (GWd/ton) during the next 20 years?

	50	60	75	90	>100
Will:	6%	50%	44%	0%	0%
Could:	6%	0%	56%	38%	0%

5. How long will (should) spent LWR fuel need to be cooled before being reprocessed? (years)

	1	5	10	50	100 or more
Will:	0%	7%	33%	60%	0%
Should:	0%	56%	25%	13%	6%

6. When do you expect a commercial-sized (~2000 t/yr) reprocessing plant in the US will (should) be completed?

	2020	2030	2040	2050	2060 or later
Will:	6%	35%	41%	6%	12%
Should:	65%	18%	6%	6%	6%

7. When will (should) a fast reactor be placed in operation relative to the operation of a commercial-sized reprocessing plant?

	10 years before	same year	10 years after	20 years after
Will:	13%	25%	25%	38%
Should:	24%	29%	29%	18%

8. How many fast reactors (FRs) will (should) be constructed per year on average during a 20 year period following implementation of reprocessing?

	1	2	3	4	>5
Will:	71%	29%	0%	0%	0%
Should:	38%	6%	19%	25%	13%

9. How large of reprocessing plant will need to be constructed to support the operation of 100 LWRs and 50 FRs? (tHM / yr)

1,500	2,000	3,000	5,000	>5,000
0%	25%	56%	19%	0%

What FR transuranic conversion ratio should, could or will be designed and operated?

	0.25	0.5	0.75	1.0	>1.0
Should:	24%	18%	12%	18%	29%
Could:	29%	12%	18%	6%	35%
Will:	0%	13%	63%	13%	13%

10. What is the expected burn up (GWd/tHM) of FRs with conversion ratios listed below for oxide fuel, including your insight into economic and technology issues?

	80	110	150	180	>180
CR<0.25	18%	9%	18%	27%	27%
CR~0.75	8%	33%	58%	0%	0%
CR~1.0	33%	42%	17%	8%	0%

11. What is the expected burn up (GWd/tHM) of FRs with conversion ratios listed below for metal fuel, including your insight into economic and technology issues?

	80	110	150	180	>180
CR<0.25	18%	9%	9%	36%	27%
CR~0.75	0%	33%	67%	0%	0%
CR~1.0	25%	33%	33%	8%	0%

12. Should MOX fuel be used in thermal reactors?

Yes	No
75%	25%

13. What burn up do you expect for MOX fuel relative to UOX fuel in LWRs?

The same	15 % higher	15 % lower	30 % higher	50 % higher
56%	19%	25%	0%	0%

14. If a reactor is licensed to burn MOX fuel, about what fraction of MOX fuel do you expect will be used relative to UOX fuel if the reactor is designed for a full load of MOX fuel?

10 %	30 %	50%	75%	100%
7%	50%	36%	0%	7%

15. What do you expect a 2000 tHM/year PUREX liquid extraction type reprocessing plant would cost (the reference case)?

$10x10^9$	$20x10^9$	$30x10^9$	$40x10^9$	$50x10^9$
20%	47%	7%	13%	13%

16. What do you expect a 2000 tHM/yr liquid extraction type reprocessing plant will cost relative to the base case if special effort is made to isolate actinides to achieve a factor of 30 reduction of decay heat in the product sent to the repository?

The same	30 % more	50 % more	100 % more	200 % more
27%	47%	13%	7%	7%

17. When do you expect Yucca Mountain to start accepting waste if it is tied to the current once-through fuel cycle?

2010	2020	2030	2040	>2040
0%	56%	38%	0%	6%

18. Would the adoption of recycling in the US delay or accelerate opening of Yucca Mountain?

Accelerate	Delay	No Impact
12%	35%	53%

19. The 1 mil/kW-hr fee for the waste management fund yields about $400/kg for the amount of spent fuel generated. What do you expect the final cost to be for disposing 70,000 tons of spent fuel (or spent fuel equivalent) in Yucca Mountain relative to funds provided by this fee (please ignore time value of money and cost to the waste management fund ($800/kg) issues)?

0.5	0.75	1.0	1.5	>=2
13%	6%	6%	13%	63%

20. How many tons of spent fuel (or spent fuel equivalent) do you believe will be disposed of in Yucca Mountain?

0	70,000	120,000	200,000	>200,000
0%	19%	19%	38%	25%

21. What do you expect the cost of dry cask storage of spent fuel for 100 years to be per kg of heavy metal relative to the 1 mil/kW-hr fee?

0.2	0.5	1.0	1.5	>2
38%	31%	15%	8%	8%

22. What do you expect the cost of dry cask storage of spent fuel for 200 years to be per kg of heavy metal relative to the 1 mil/kW-hr fee?

0.2	0.5	1.0	1.5	>2
15%	31%	31%	8%	15%

23. If a second U.S. geologic repository is built, how do you think will be the fractional cost relative to Yucca Mountain?

<=0.5	0.7	1.0	1.3	>1.3
19%	25%	13%	6%	38%

24. What would you expect the cost of interim storage of high level waste (or whatever you would like to call it) from a liquid extraction type reprocessing plant for 100 years per metric ton of heavy metal relative to the funds generated by the waste management fee?

<=0.1	0.3	0.5	1	>1
8%	38%	23%	8%	23%

25. What would you expect the cost of interim storage of high level waste (or whatever you would like to call it) from a liquid extraction type reprocessing plant for 200 years per metric ton of heavy metal relative to the funds generated by the waste management fee?

<=0.1	0.3	0.5	1	>1
8%	15%	23%	15%	38%

Appendix C: Sustainability Issues and Description of Fuel Cycles of Interest for Future Development

An estimate of distribution of uranium in the earth's crust is illustrated Figure 2-13. Note that the value shown for vein deposits, pegmatites, and unconformity deposits is about the same value listed in the MIT Report[2] of several million tons uranium. The World Energy Council (WEC) website[7,8] cites that the United States has about $1x10^5$ tonnes uranium below \$80/kg and about $3.5x10^5$ tonnes for costs up to \$130/kg. The World resources are estimated by the WEC, and others, to be about $10\text{-}15x10^6$ tonnes for costs up to \$130/kg. The feed for a 1 GWe (about 3 GWth) is about 20 tonnes of enriched uranium per year (based on a burn up of 50 GW/tonne), or about 150 tonnes of natural uranium. Thus, the world resources will fuel about $1x10^5$ Rx-years of 1 GWe reactors that operate on the once-through cycle, or about 1,000 reactors for a hundred years. If the estimates of uranium resources are correct, the once-through fuel cycle can fuel a growing nuclear power industry for about 100 years, which should not be considered as sustainable, but it may be used to argue, incorrectly in this authors opinion, that there is no need to initiate implementation of advanced fuel cycles at this point in time. Accord to the MIT report, it is believed that $15x10^6$ tonnes of uranium are available for modest prices to sustain a once-through fuel cycle. Thus, the authors of the MIT report conclude that there should be no particular concern with fuel supply during the next 50 years. Figure 2-14 illustrates the conclusions of this argument.

The once-through fuel cycle is currently deployed in the United States (US); although, some fuel assemblies that utilize mixed-oxides of plutonium and uranium (MOX) are scheduled for testing by Duke Power[6]. The motivation for use of MOX, however, is to burn plutonium and the fuel will be provided free of charge to Duke. Thus, the choice to test MOX fuel assemblies is apparently motivated by economics, but not free market economics.

Figure 2-1 is a very general illustration of resource utilization and waste production, and it identifies options for various fuel cycles. Figures 2-2 and 2-3 are representations of the four fuel cycles cited above as illustrated by two authors who are active in the DOE Advanced Fuel Cycle Initiative. These figures qualitatively illustrate the reduction of waste required for placement in repositories as more of the fuel, fission products, and transuranics are recycled.

Figure 2-4 illustrates fuel cycles currently deployed. Note that recycle of fissile material is implemented in Europe and in Japan. In France, for example, about one-third of the fuel utilized in PWRs is MOX fuel, whereas in the USA only the once-through fuel cycle is implemented. Figure 2-5 shows that the repository at Yucca Mountain will not be adequate to accommodate more fuel than will be produced by 2015 and that it would be filled to its theoretical capacity by 2050. Storage of spent fuel at nuclear power plants is also an option, which is also illustrated in Figure 2-5.

The importance of transmutation in accelerator driven systems or in hard-spectrum reactors is graphically represented in Figure 2-6. Note that with recycle of all actinides, spent fuel has less radio-toxicity than the original ore body in less than 1000 years. Also, it is apparent that the disposition of spent fuel requires much less repository capacity if actinides are transmuted and some fission products are handled separately, and it facilitates the implementation of a sustainable fuel cycle as illustrated in Figure 2-7.

For this example of a sustainable fuel cycle, the volume of repository space is reduced by 96% and the fuel supply is increased by a factor of 100. The economic benefit of implementing a sustainable fuel cycle is most likely dominated by reduction of costs for disposal of spent fuel since direct fuel costs currently constitute only 5% of the cost of production costs by nuclear energy. The overall goal of the Advanced Fuel Cycle Initiative (AFCI) is cited in Figure 2-8. In particular, the issues listed are essentially equivalent to the sustainability issues cited for this study.

Figure 2-9 illustrates that there is no technical necessity for a second repository if a sustainable fuel cycle is implemented, and fuel cycles of current practical interest are shown in Figure 2-10. It is shown in Figure 2-11 that the overall fuel cycle contributes about 20% of the cost of production; thus, it is unlikely that any of the fuel cycles cited in Figure 2-10 would have significantly different cost than any other comparable fuel cycle. Since disposal of spent fuel and high-level waste is a significant component of the overall fuel cycle costs, reduction of repository space is an important consideration for selection of an advanced fuel cycle. In order to minimize the volume required for disposal, the total energy released from decay heat of spent fuel must be minimized. During the first 100 years, fission products dominate decay heat production, while actinides dominate the long-term decay heat generation. Figure 2-12 qualitatively illustrates these issues.

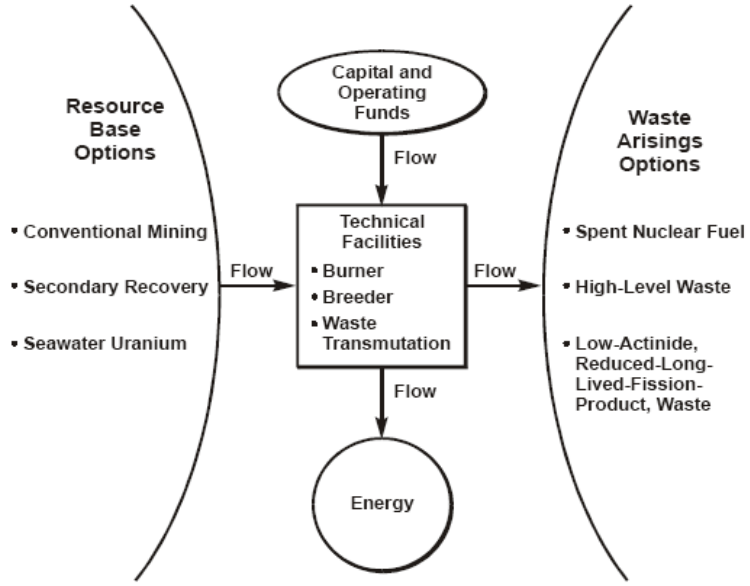

Figure C-1. General illustration of uranium resource utilization. (Charles Forsberg, "Generation IV Roadmap: Fuel Cycles," American Nuclear Society Winter Meeting, Reno NV, November 2001)[4]

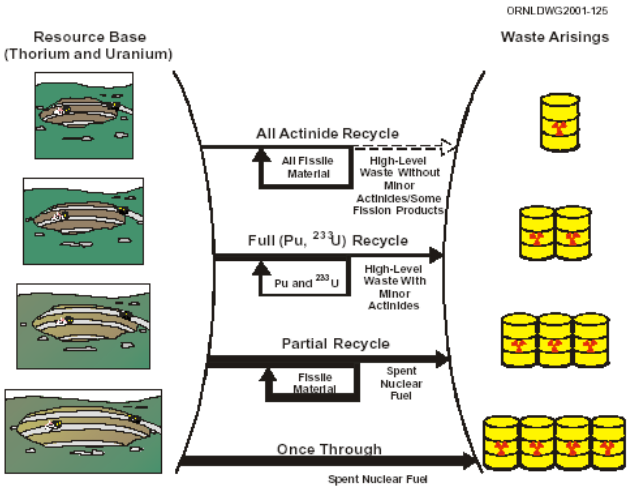

Figure C-2. General illustration of uranium resource utilization for four fuel cycles. (Charles Forsberg, "Generation IV Roadmap: Fuel Cycles," American Nuclear Society Winter Meeting, Reno NV, November 2001)[4]

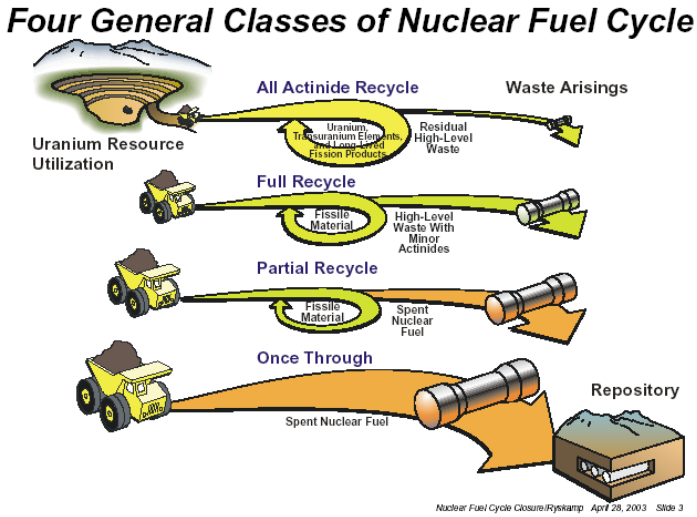

Figure C-3. Illustration of four general fuel cycles. (John M. Ryskamp[+], "Nuclear Fuel Cycle Closure," IEEE Power Engineering Society Meeting, April 28, 2003, +Idaho National Engineering and Environmental laboratory)[5]

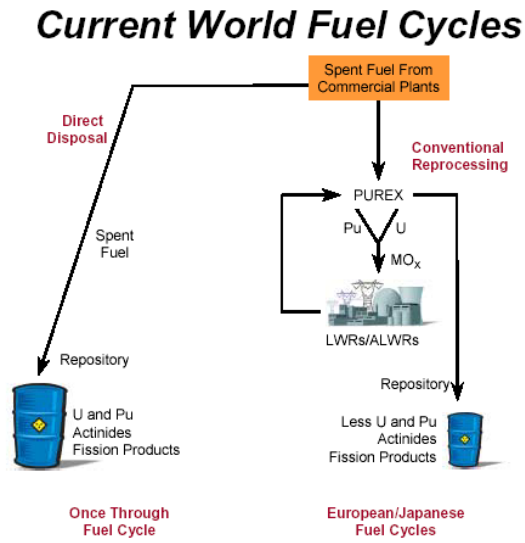

Figure C-4. Illustration of Current World Fuel Cycles. (John M. Ryskamp[+], "Nuclear Fuel Cycle Closure," IEEE Power Engineering Society Meeting, April 28, 2003, +Idaho National Engineering and Environmental Laboratory)[5]

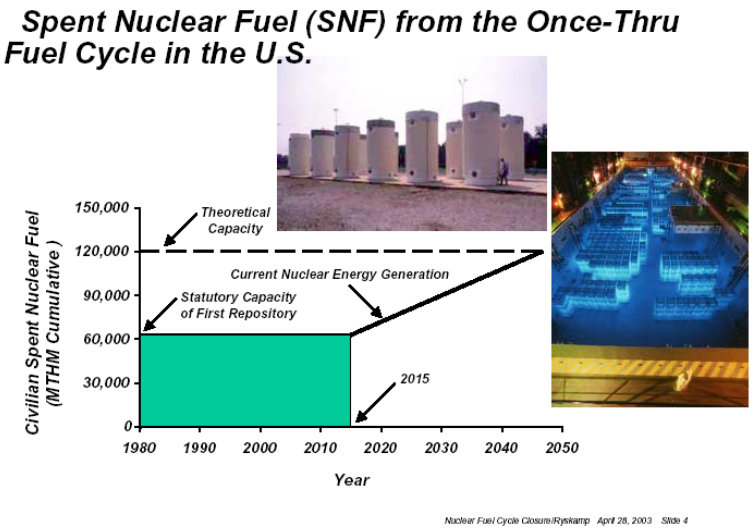

Figure C-5. Spent nuclear fuel from the once-through fuel cycle. (John M. Ryskamp[+], "Nuclear Fuel Cycle Closure," IEEE Power Engineering Society Meeting, April 28, 2003, +Idaho National Engineering and Environmental Laboratory)[5]

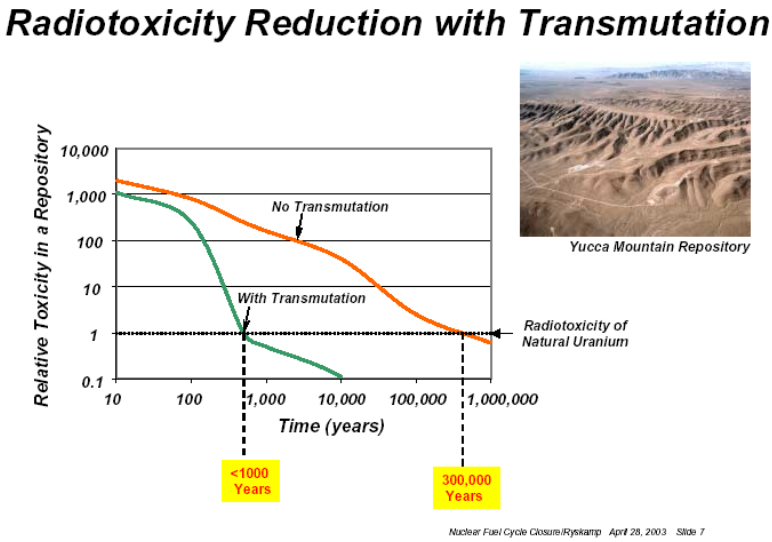

Figure C-6. Radiotoxicity with and without transmutation. (John M. Ryskamp[+], "Nuclear Fuel Cycle Closure," IEEE Power Engineering Society Meeting, April 28, 2003, +Idaho National Engineering and Environmental Laboratory)[5]

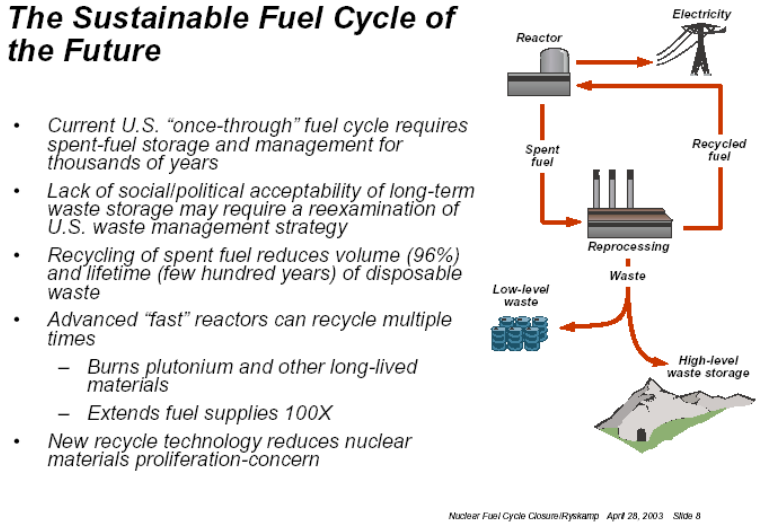

Figure C-7. Illustration of a sustainable fuel cycle. (John M. Ryskamp[+], "Nuclear Fuel Cycle Closure," IEEE Power Engineering Society Meeting, April 28, 2003, +Idaho National Engineering and Environmental laboratory)[5]

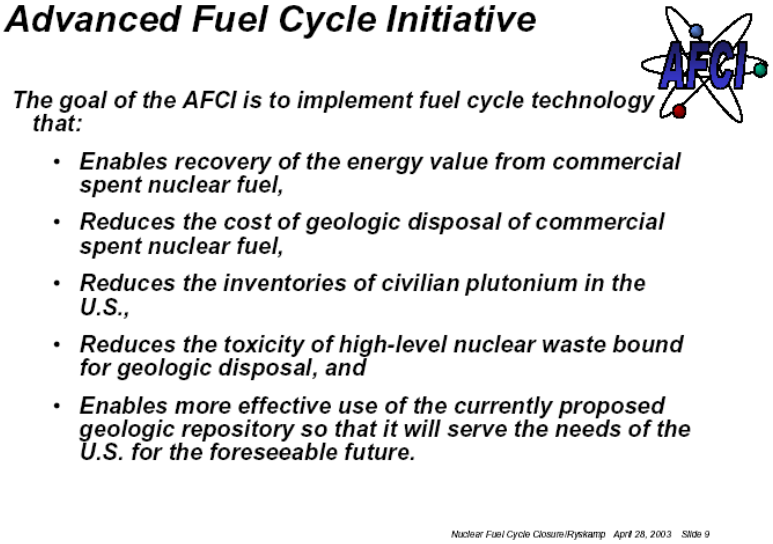

Figure C-8. Goal of the Advanced Fuel Cycle Initiative. (John M. Ryskamp[+], "Nuclear Fuel Cycle Closure," IEEE Power Engineering Society Meeting, April 28, 2003, +Idaho National Engineering and Environmental laboratory)[5]

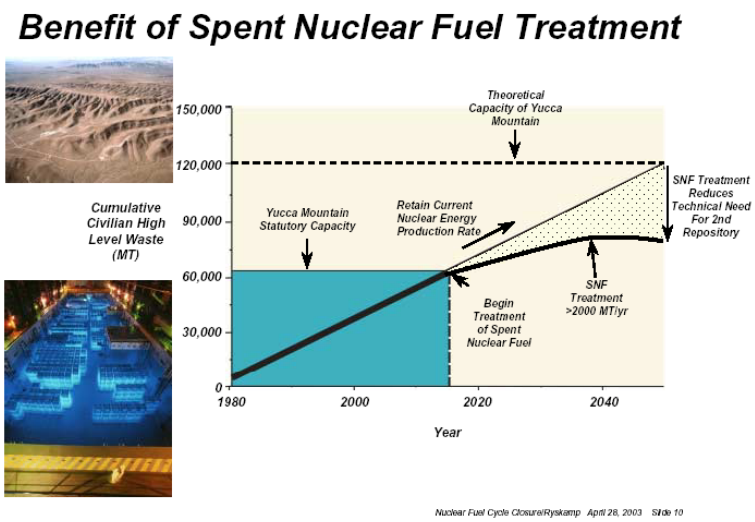

Figure C-9. Benefit of spent fuel treatment. (John M. Ryskamp[+], "Nuclear Fuel Cycle Closure," IEEE Power Engineering Society Meeting, April 28, 2003, +Idaho National Engineering and Environmental laboratory)[5]

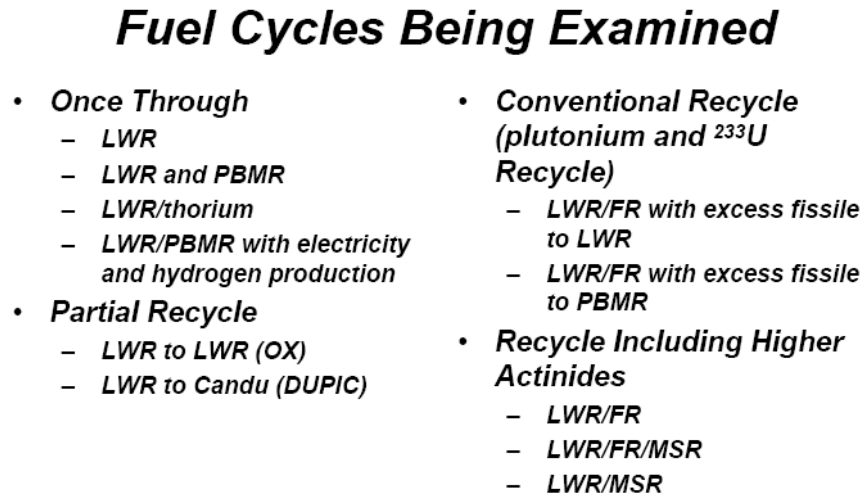

Figure C-10. Fuel cycles considered for Generation IV reactors. (Charles Forsberg, "Generation IV Roadmap: Fuel Cycles," American Nuclear Society Winter Meeting, Reno NV, November 2001)[4]

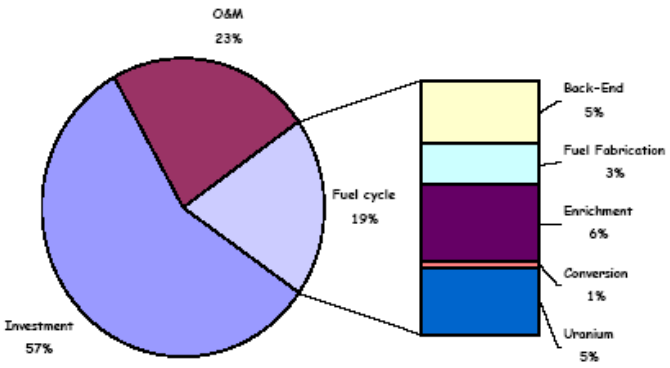

Figure C-11. Breakdown of cost in the production of nuclear power. (OECD/NEA. "Trends in the Nuclear Fuel Cycle," Paris, France, 2001, Luc Van Den Durpel)

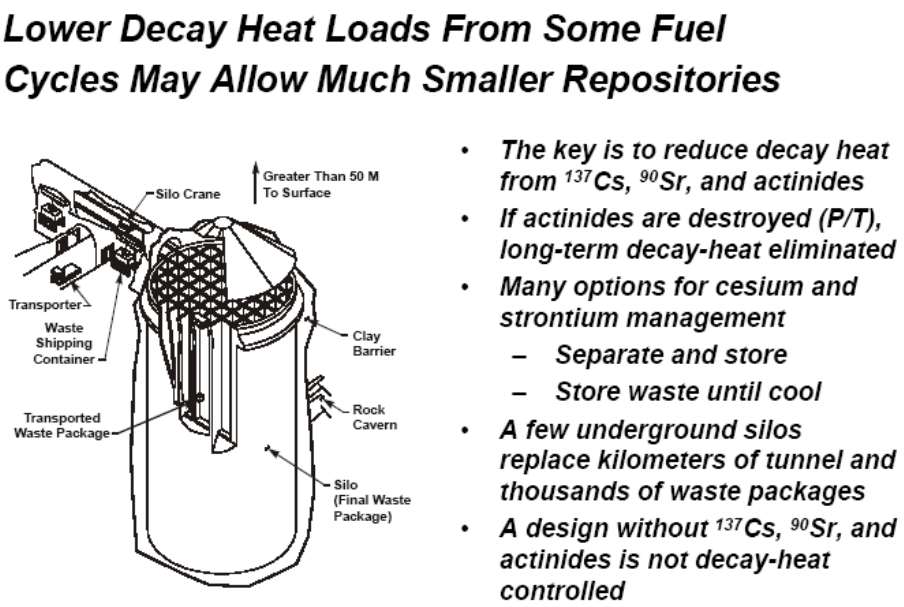

Figure C-12. Illustration of repository for consideration of decay heat removal. (Charles Forsberg, "Generation IV Roadmap: Fuel Cycles," American Nuclear Society Winter Meeting, Reno NV, November 2001)[4]

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Distribution of Uranium in the Earth's Crust

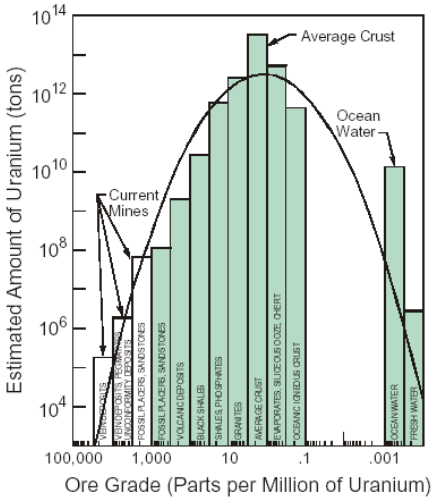

Figure C-13. Distribution of uranium in the earth. (Charles Forsberg, "Generation IV Roadmap: Fuel Cycles," American Nuclear Society Winter Meeting, Reno NV, November 2001)[4]

Sustainability I: Cost and Environmental Impacts, Not Resource Availability, Limit Uranium And Thorium Resources

- *Two Periods of Ore Exploration*
  - *1950's (Cold War Driven)*
  - *1970's (Oil Shock Driver)*
  - *Current Glut of Uranium – Negligible Prospecting Going On*
- *Three components in current estimates of ore*
  - *Redbook Known + Speculative Reserves: 4.5 + ~10 ~15 million tonnes U*
  - *Geologic estimates to crustal abundance (see Figure)*
  - *U in Seawater in parts per billion (Billions of tonnes)*
- *Harvesting Ore of 10 fold reduction in assay:*
  - *300 fold increase in reserves*
  - *10 fold increase in mining per kg of uranium*
  - *Cost and impacts determined by economics of scale and technological advances*
- *Sustainable ore availability is not the issue: Cost and ecological disruption are the issues and both will be impacted by:*
  - *Long-term competition between lower grade ores and recycle of discharged SNF*
  - *Differences in repositories with wastes from once-through and recycle fuel cycles*

Figure C-14. Constraints of sustainability of nuclear energy. (Charles Forsberg, "Generation IV Roadmap: Fuel Cycles," American Nuclear Society Winter Meeting, Reno NV, November 2001)[4]

In order for nuclear energy to make a substantial long-term contribution to overall energy supply, it is essential that advanced fuel cycles be implemented with a sufficient number of breeder reactors to fuel both breeder reactors and reactors that consume more fissile fuel than they produce. With the use of breeder reactors, the potential fuel supply is increased by about a factor of 100, and it would then become economical to mine uranium from the ocean. Thus, with breeder reactors the uranium and thorium fuel supplies would be adequate for thousands, perhaps millions, of years. Examples of fuel cycles considered in the MIT report[2] include the following:

1) once through with only thermal reactors,
2) partial recycle with only thermal reactors, and
3) combination of thermal and fast reactors.

These fuel cycles are illustrated in Figures C-15, C-16 and C-17, and mass flows are cited for 1500 GWe. It is apparent that substantial savings in resource demands can be gained with recycle of plutonium and uranium and that more significant benefit in resource utilization can be obtained when fast reactors are integrated into the fleet of reactors. However, for fission reactors to play a significant role in energy production for the world, the reactors deployed will need to have an overall conversion ratio near unity. These reactors, or accelerators, must also accomplish transmutation of actinides. Figure C-18 illustrates that without advanced fuel cycles, excess repository capacity at Yucca Mountain is eliminated within a few decades.

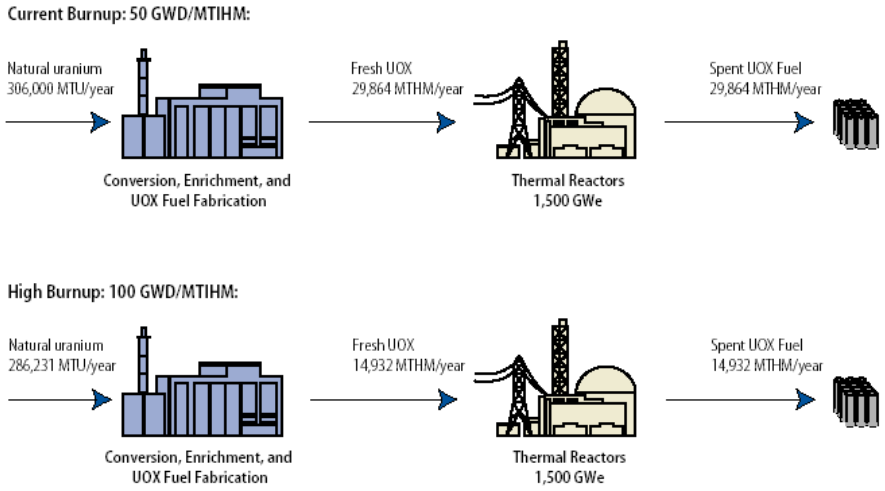

Figure C-15. Illustration of the once-through fuel cycle for 1500 GWe with mass flows cited for high and low burn up cases. (An Interdisciplinary MIT Study, "The Future of Nuclear Power,")[2]

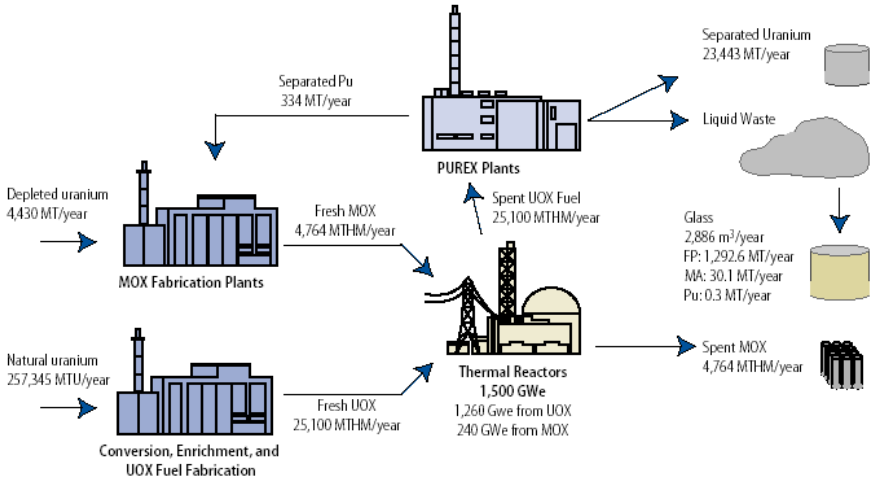

Figure C-16. Illustrations of fuel cycle with recycle of plutonium for production on MOX fuel into thermal reactors. (An Interdisciplinary MIT Study, "The Future of Nuclear Power,")[2]

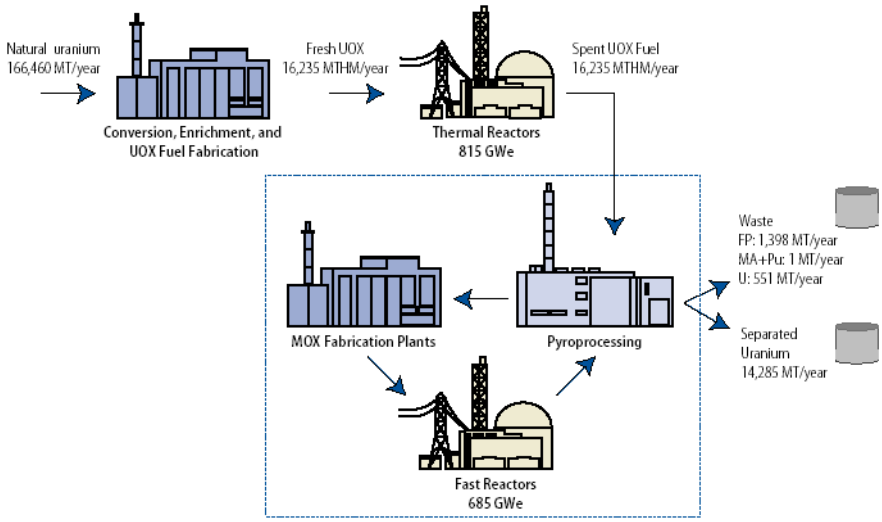

Figure C-17. Illustration of a fuel cycle with integration of fast and thermal reactors with pyroprocessing and MOX fabrication plants. (An Interdisciplinary MIT Study, "The Future of Nuclear Power,")[2]

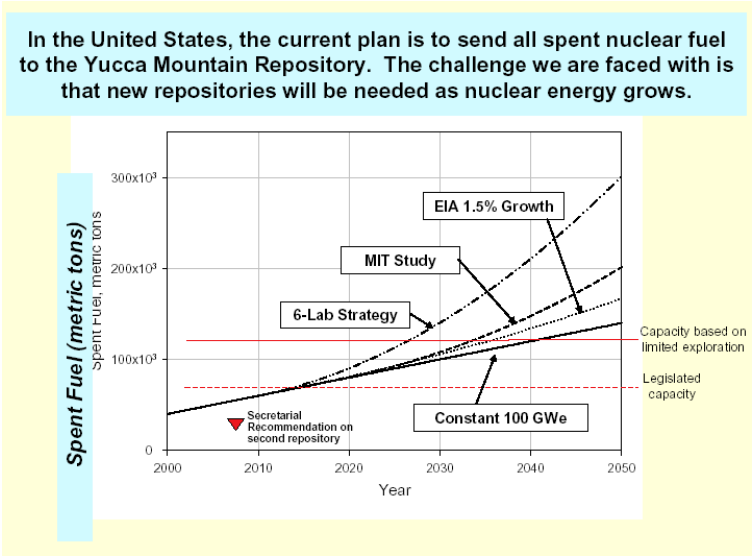

Figure C-18. Accumulation of spent nuclear fuel for several growth scenarios. (Mike Cappiello, "The Potential Role of Accelerator Driven Systems in the US," Presentation at ICRS-10/RPS 2004, Madeira Portugal)[9]

Environmental Effects

The nuclear fuel cycle encompasses many activities, and requires a number of processes, that have some potential for dispersion of radioactive materials. In the cases of mining, milling, conversion, enrichment, fuel fabrication, and most processing operations, there exists a possibility for dispersion of small amounts of radioactive materials. Since the potential contamination (at least for the once-through fuel cycle) has very low specific activity, any potential health effects are probably due to chronic exposure. However, care must be employed for managing these sources of contamination. For example, early mining activities in the US resulted in numerous piles of mill tailings that presented long term health issues comparable to disposal of spent fuel. Passage of the Uranium Mill Tailings Radiation Control Act has essentially eliminated this health risk, however. In the cases of power production, transmutation, and radionuclide disposition, the possibility for acute exposure with immediate health effects needs to be considered.

Economics

There is currently only one repository for commercial high-level waste under development in the United States, and it does have adequate capacity for even the lifetime production of spent fuel by the LWRs presently in operation in the US. It is apparent from Table 2-1 and Figure 2-19 that there is a potential for reducing repository space requirements by a factor of fifty if transmutation and isotopic separation options are optimized.

Table C-1. Accumulation of spent nuclear fuel for several growth scenarios. (Mike Cappiello, "The Potential Role of Accelerator Driven Systems in the US," Presentation at ICRS-10/RPS 2004, Madeira Portugal)[9]

Repository Packing Density Capacity can be Increased Dramatically if we Remove and Transmute Certain Elements

	TRU Recovery Efficiency		
Processing	*90%*	*99%*	*99.9%*
Pu, Am removal only	2.1X	3.2X	3.4X
Pu, Am, Cs, Sr removal	3.0X	20X	59X

Goal is to achieve this kind of capacity increase, and eliminate the need for additional repositories in the foreseeable future.

Note that the separation of Uranium alone is of minimal benefit without the processing and transmutation of actinides.

Societal Impacts

The current LWR fleet with a once-through fuel cycle is supported in the US by relatively few (about twenty) processing and handling facilities. Some of theses include the following:

conversion of U_3O_8 from mills to UF_6 for feed to enrichment plants, enrichment plants,

1) conversion of enriched UF_6 to UO_2 for feed to fuel fabrication plants,
2) fuel fabrication plants,
3) spent fuel storage facilities,
4) low level waste disposal facilities, and
5) spent fuel disposal facilities.

Note that in the case of advanced fuel cycles that include transmutation and isotopic separation, additional facilities must be built and operated, which involve the following operations:

spent fuel processing,

1) transmutation with accelerators or fast reactors,
2) conversion facilities, and
3) fuel fabrication plants.

Both PUREX and pyro-metallurgical methods for processing spent fuel are under consideration as part of the Advanced Fuel Cycle Initiative currently supported by the Department of Energy. These facilities may employ several hundred persons and would have a substantial impact on the local communities and would result in significant economic growth in the area of its location.

Accelerators or fast reactor transmutation facilities would probably employ about the same number of persons as a PUREX processing facility, but more than a facility based on pyro-technology.

The impact on community should be about the same as a commercial nuclear power plant presently in operation.

The development of advanced fuel cycles within the fifty years will enable the use of nuclear power to be sustainable by notably lessening the requirements for spent fuel or high-level waste disposal sites. Such an undertaking is apparently not economical based on fuel costs, but it provides the option for continuation of nuclear power and significantly less demand for foreign energy supplies.

Proliferation

The once-through fuel cycle continuously generates plutonium that could be used for production of weapons. Even if the spent fuel is disposed of in a facility comparable to Yucca Mountain, the reactor-produced plutonium could be mined for later use for weapons or for startup of fast reactors. In the case of advanced fuel cycles that separate isotopes and transmute actinides with accelerator driven systems or fast reactors, the plutonium inventory diminishes to an equilibrium inventory. Thus, implementation of advanced fuel cycles will lessen chances of reactor-produced plutonium being used for production of weapons. For the fuel cycles considered in this report, the amount of plutonium maintained in inventory probably determines the most to least proliferation resistance to be as follows:

1) full actinide recycle with isotopic separation and transmutation,
2) recycle of fissile actinides into accelerators or fast reactors,
3) recycle of uranium, and
4) once-through fuel cycle.

This point of view depends on where Pu is residing in the fuel cycle and for how long. Consequently, not everyone may agree with this perspective. In order to achieve a reduction of plutonium inventory, it would be useful to immediately implement plutonium recycle into MOX fuel utilized in present-day LWRs. This approach would permit technology for advanced fuel cycles to be progressively developed and implemented.

Infrastructure Commitments

Construction, operation and maintenance of advanced fuel cycles will require development of some new administrative, regulatory and security organizations because the required new facilities will have features that are not needed for the once-through fuel cycle. Some new educational programs may also be needed to operate PUREX processing facilities since radiochemistry academic programs are very few in number. Traditional engineering programs, including Nuclear Engineering, should be adequate to provide all the expertise required to support advanced fuel cycles. Communities that provide employees for operation, maintenance, and various other services may need to accommodate some restrictions relative to security services for these facilities.

Waste Management

Disposition of spent fuel and high level waste is of major concern to environmentalists, politicians, concerned citizens, governmental agencies, and scientific organizations. Thus,

implementation of advanced fuel cycles, even the initial phases, will significantly reduce the requirements for repository space, as shown in Table C-1 and Figures C-19 through C-23, and it will develop technology that is essential for nuclear energy to play a significant role in the world's supply of energy. The cost of spent fuel disposal is probably the primary motivation for development of advanced fuel cycles. The relative roles of accelerators and fast reactors are shown in Figure C-24.

The DOE is funding a program to dispose of 34 metric tons of weapons-grade plutonium in LWRs through the use of MOX fuel, and Russia will also dispose of 34 metric of weapons-grade plutonium. Duke Energy is currently implementing this program with Duke Cogema Stone & Webster (DCS), who will build a MOX fuel fabrication plant at Savannah River Plant and will use it at Catawba & McGuire. The fuel fabrication facility will be constructed at Savannah River Plant, and fuel assembly will be a Framatome design, similar to fuel recently used in McGuire and Catawba. The MOX and LEU assemblies will be dispersed throughout the core and will make up thirty to forty percent of the core.

A license amendment for the MOX lead assembly program was submitted to the NRC in February 2003 and MOX assemblies will be loaded in the Spring of 2005[6]. A license amendment for large-scale MOX fuel is expected to be submitted in August of 2005, and plant changes for use of MOX fuel will be implemented from 2004 through 2009. Large-scale use should begin in 2010 and should be completed by 2023.

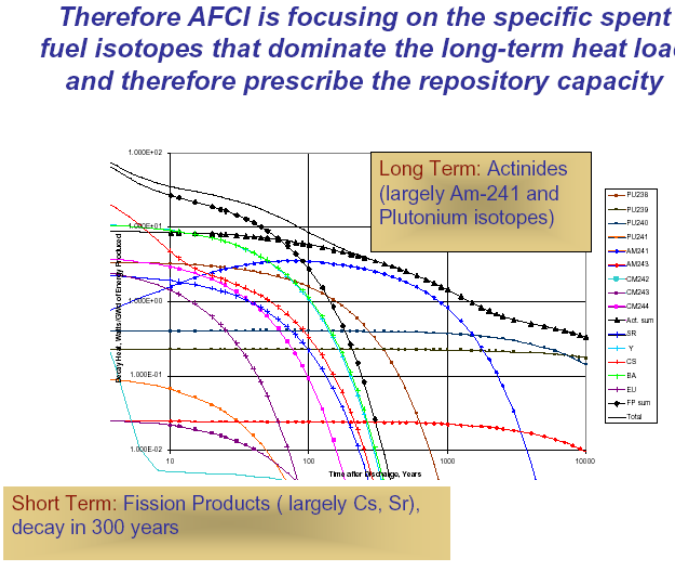

Figure C-19. Accumulation of spent nuclear fuel for several growth scenarios. (Mike Cappiello, "The Potential Role of Accelerator Driven Systems in the US," Presentation at ICRS-10/RPS 2004, Madeira Portugal)[9]

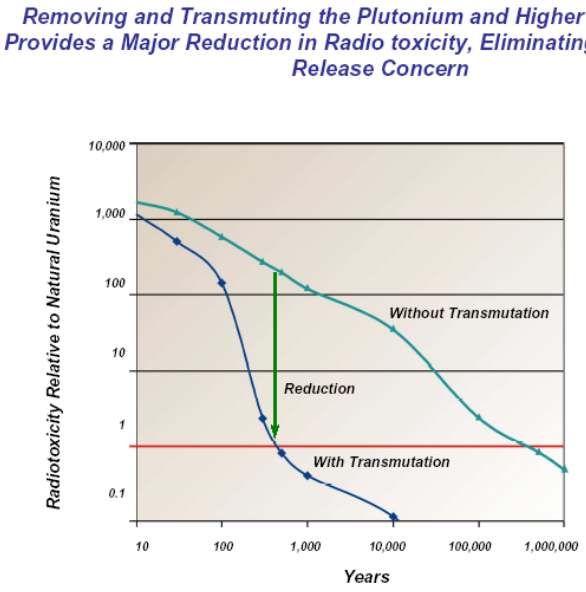

Figure C-20. Accumulation of spent nuclear fuel for several growth scenarios. (Mike Cappiello, "The Potential Role of Accelerator Driven Systems in the US," Presentation at ICRS-10/RPS 2004, Madeira Portugal)[9]

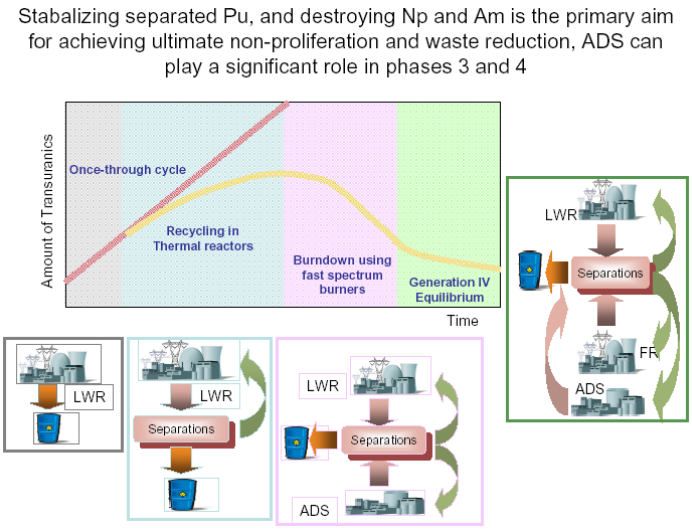

Figure C-21. Accumulation of spent nuclear fuel for several growth scenarios. (Mike Cappiello[+], "The Potential Role of Accelerator Driven Systems in the US," Presentation at ICRS-10/RPS 2004, Madeira Portugal, +Los Alamos National Laboratory)[9]

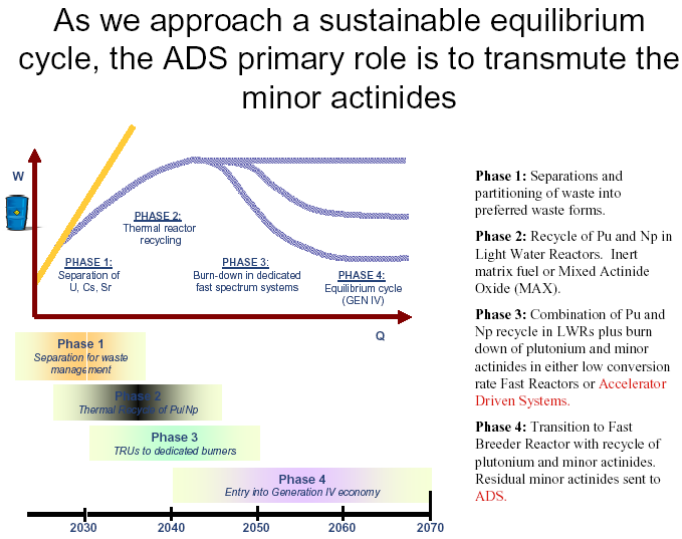

Figure C-22. Accumulation of spent nuclear fuel for several growth scenarios. (Mike Cappiello, "The Potential Role of Accelerator Driven Systems in the US," Presentation at ICRS-10/RPS 2004, Madeira Portugal)[9]

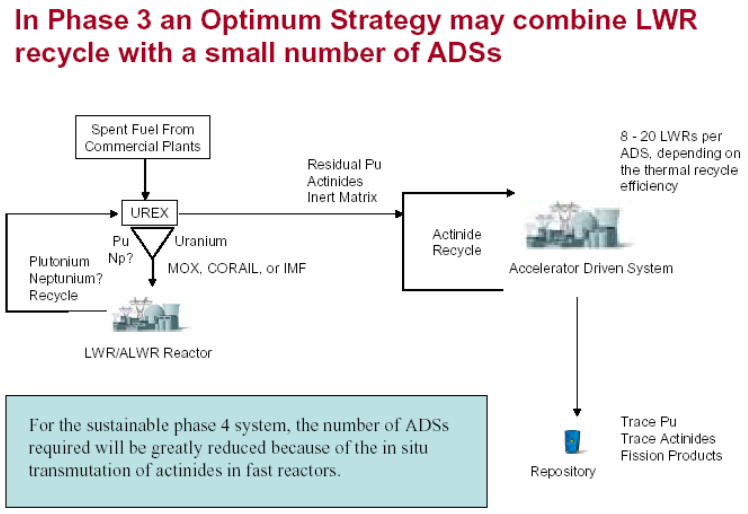

Figure C-23. Accumulation of spent nuclear fuel for several growth scenarios. (Mike Cappiello, "The Potential Role of Accelerator Driven Systems in the US," Presentation at ICRS-10/RPS 2004, Madeira Portugal)[9]

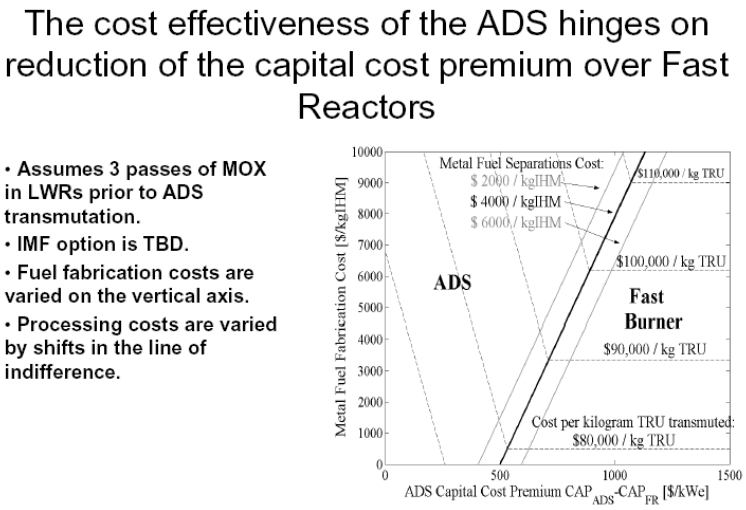

Figure C-24. Accumulation of spent nuclear fuel for several growth scenarios. (Mike Cappiello, "The Potential Role of Accelerator Driven Systems in the US," Presentation at ICRS-10/RPS 2004, Madeira Portugal)[9]

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Vulnerability for Disruptions

There are numerous options for implementing various fuel cycles for LWRs and for advanced reactors. The once-through option should be more robust to intentional or unintentional disruptions than advanced fuel cycles since there are fewer levels of processes in the once-through fuel cycle than in advanced fuel cycles. In addition there are typically several suppliers for each level of the once-through fuel cycle. Natural U_3O_8, natural and enriched UF_6, and enriched UO_2 are obtained from domestic and from international sources, and likewise fuel is fabricated in multiple locations. For the case of advance fuel cycles there would most likely be some processes, such as reprocessing, that take place on only one or two locations.

Conclusions and Recommendations

In order to meet energy needs of the US within the next several decades the use of nuclear power needs to be expanded. This follows from the need for security from disruptions of and from competition for international supplies of petroleum. Given this expansion, it is essential to implement advanced fuel cycles to eliminate the need for a second repository and for assurance of long-term fuel supplies.

Appendix D: Generation of Fuel Composition for DANESS

Generation of ORIGEN-ARP Cross-Section Libraries

SAS2 or SAS2H is used to calculate detailed nuclide concentrations, radiation source terms, and decay heat generation for spent nuclear fuel. The SAS2 code is part of the SCALE package offered by Radiation Safety Information Computational Center (RSICC) in Oak Ridge National Laboratory. SAS2 gives the user the ability to generate problem-dependent cross sections for burnup analysis based on specific assembly design characteristics, fuel type, enrichment, reactor operation, and irradiation and decay history. Cross-sections for the reactor fuel assembly is developed using the one-dimensional (1-D) neutron transport analysis code XSDRNPM. The cross sections developed for a specific assembly design, fuel type, and burnup can be saved and used in subsequent analysis calculations.

Having the ability to save developed cross sections is very useful when there are a wide range of scenarios to calculate. SAS2 cases can use up valuable computing time due to several reasons. For depletion analysis the SCALE manual recommends using the 44-group libraries. The standard energy group is 27. This increase from 27 to 44 will also increase the time needed for calculations. One of the goals of this project is to analyze the impact of high burnup reactors. High burnup is another factor that adds to computing time. A third factor that may increase computing time is the reactor design itself. In order to have flexibility with the reactor design and decrease the computing time, the ORIGEN-ARP program can use the saved cross sections calculated by SAS2.

ORIGEN-ARP takes less time to run and retains the accuracy of SAS2 calculations. Initially ORIGEN-ARP was programmed to calculate LWRs and MOX reactors with a burnup of up to 60 GWd/MTU. The initial fuel composition and enrichment can also be modified for all reactor types. The only set back in using ORIGEN-ARP is that this project is concerned with reactor burnup up to 100 GWd/MTU. In order to overcome this obstacle, new ORIGEN-ARP libraries must be created.

The first step in creating ORIGEN-ARP libraries is to run one SAS2 simulation for each reactor type under consideration, which also includes variations in initial fuel compositions. For this project that meant eight 17x17 PWR cases, twenty-four 8x8 BWR cases, nine 17x17 PWR MOX cases, and twenty-seven 9x9 BWR MOX cases. The reason there are more BWR cases than PWR is that moderator density is also varied for the BWRs. The fast reactor cases are addressed later in this section. For each LWR case the burnup is calculated up to 100 GWd/MTU with burnup steps of 3000 MWd/MTU. This is a total of 34 cross section libraries. The MOX cases were also calculated to 100 GWd/MTHM with burnup steps of 4800 MWd/MTHM for a total of 21 cross section libraries. All cross section libraries are saved in the SAS2 output.

The second step in creating ORIGEN-ARP libraries is to transfer the developed cross sections into ORIGEN-ARP. The PRISM module in SCALE 5 will pull the developed cross section libraries from the SAS2 output. The user can choose to take all the cross sections or just the ones that are needed. PRISM will produce a file that is read by ARPLIB. ARPLIB will format the cross section information into files that ORIGEN-ARP will be able to use. Once the ARPLIB files are created for every reactor type they are moved into the ORIGEN-ARP directory. The final step is to update the ORIGEN-ARP module.

Once the cross section libraries are configured into ORIGEN-ARP, a wide range of simulations can be performed in a very short period of time ~ 1-2 minutes for each simulation. The user interface for ORIGEN-ARP can be found in Figure 1. This interface is for a LWR. The interface for MOX is very similar. The fuel type is where you can choose the reactor type only BWR and PWR can simulate a burnup up to 100 GWd/MTU. The next line that needs to be changed is the enrichment line. For the libraries that were created for this project the enrichment of U-235 ranges from 1.5-8.0%. The burnup line will also need information; this information will range from 30-100 GWd/MTU. The average power will also need to be filled in, 40 MW/MTU was used for this project. The output from ORIGEN-ARP can be saved in variety of units (Ci, Watts, grams…). When a value is input within the above range, ORIGEN-ARP will use interpolation to determine the appropriate cross sections to use for all calculations.

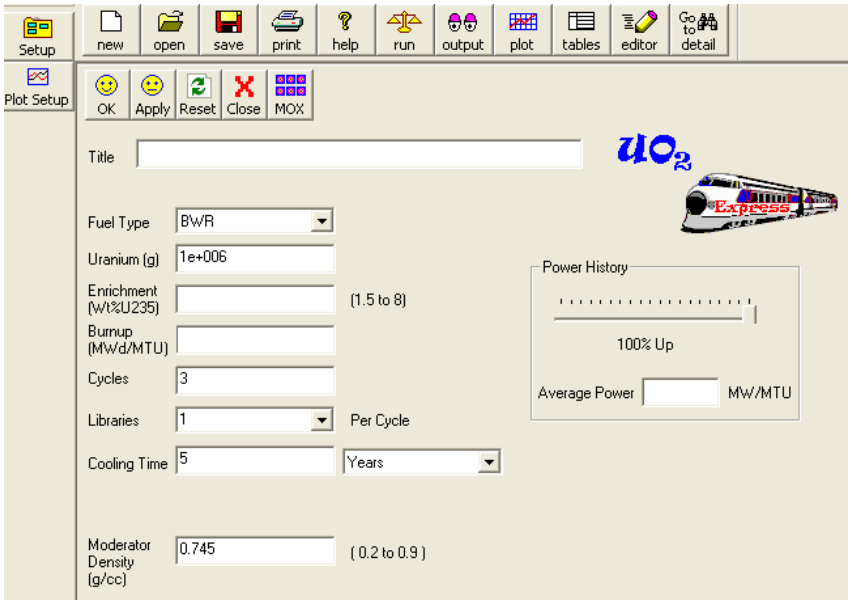


Figure D-1. ORIGEN-ARP user interface for LWR reactor simulation

Currently, the ORIGEN-ARP module for this project has been updated with eight 17x17 PWR cases, twenty-four 8x8 BWR cases, nine 17x17 PWR MOX cases, and twenty-seven 9x9 BWR MOX cases. The fast reactor model is still being designed. Table C-1. has a list of all ORIGEN-ARP libraries created. It is hopeful that SAS2 can be used to develop the fast reactor cross sections for oxide and metal reactors, which will then be transformed into ORIGEN-ARP libraries.

Table D-1. List of all ORIGEN-ARP libraries created for the Advanced Fuel Cycle Initiative project.

ORIGEN-ARP libraries for the Advanced Fuel Cycle Initiative project.

	ORIGEN-ARP Library Created	Reactor Type	wt% of U-235 Enrichment	Moderator Density	wt% of Plutonium	wt% of Pu-235 Enrichment	Burnup Range in Gwd/MTHM
1	**pwr1_17x17_2**	PWR	1.5	0.7295	0.0	0.0	0-100.
2	**pwr2_17x17_2**	PWR	2.0	0.7295	0.0	0.0	0-100.
3	**pwr3_17x17_2**	PWR	3.0	0.7295	0.0	0.0	0-100.
4	**pwr4_17x17_2**	PWR	4.0	0.7295	0.0	0.0	0-100.
5	**pwr5_17x17_2**	PWR	5.0	0.7295	0.0	0.0	0-100.
6	**pwr6_17x17_2**	PWR	6.0	0.7295	0.0	0.0	0-100.
7	**pwr7_17x17_2**	PWR	7.0	0.7295	0.0	0.0	0-100.
8	**pwr8_17x17_2**	PWR	8.0	0.7295	0.0	0.0	0-100.
9	**b1_bwr_2**	BWR	1.5	0.2	0.0	0.0	0-100.
10	**b1_bwr_5**	BWR	1.5	0.5	0.0	0.0	0-100.
11	**b1_bwr_9**	BWR	1.5	0.9	0.0	0.0	0-100.
12	**b2_bwr_2**	BWR	2.0	0.2	0.0	0.0	0-100.
13	**b2_bwr_5**	BWR	2.0	0.5	0.0	0.0	0-100.
14	**b2_bwr_9**	BWR	2.0	0.9	0.0	0.0	0-100.
15	**b3_bwr_2**	BWR	3.0	0.2	0.0	0.0	0-100.
16	**b3_bwr_5**	BWR	3.0	0.5	0.0	0.0	0-100.
17	**b3_bwr_9**	BWR	3.0	0.9	0.0	0.0	0-100.
18	**b4_bwr_2**	BWR	4.0	0.2	0.0	0.0	0-100.
19	**b4_bwr_5**	BWR	4.0	0.5	0.0	0.0	0-100.
20	**b4_bwr_9**	BWR	4.0	0.9	0.0	0.0	0-100.
21	**b5_bwr_2**	BWR	5.0	0.2	0.0	0.0	0-100.
22	**b5_bwr_5**	BWR	5.0	0.5	0.0	0.0	0-100.
23	**b5_bwr_9**	BWR	5.0	0.9	0.0	0.0	0-100.
24	**b6_bwr_2**	BWR	6.0	0.2	0.0	0.0	0-100.
25	**b6_bwr_5**	BWR	6.0	0.5	0.0	0.0	0-100.
26	**b6_bwr_9**	BWR	6.0	0.9	0.0	0.0	0-100.
27	**b7_bwr_2**	BWR	7.0	0.2	0.0	0.0	0-100.
28	**b7_bwr_5**	BWR	7.0	0.5	0.0	0.0	0-100.
29	**b7_bwr_9**	BWR	7.0	0.9	0.0	0.0	0-100.
30	**b8_bwr_2**	BWR	8.0	0.2	0.0	0.0	0-100.
31	**b8_bwr_5**	BWR	8.0	0.5	0.0	0.0	0-100.
32	**b8_bwr_9**	BWR	8.0	0.9	0.0	0.0	0-100.
33	**moxp_4_5**	MOX PWR	0.7	0.7295	4.0	50.0	0-100.
34	**moxp_4_6**	MOX PWR	0.7	0.7295	4.0	60.0	0-100.
35	**moxp_4_7**	MOX PWR	0.7	0.7295	4.0	70.0	0-100.
36	**moxp_6_5**	MOX PWR	0.7	0.7295	6.0	50.0	0-100.
37	**moxp_6_6**	MOX PWR	0.7	0.7295	6.0	60.0	0-100.
38	**moxp_6_7**	MOX PWR	0.7	0.7295	6.0	70.0	0-100.
39	**moxp_1_5**	MOX PWR	0.7	0.7295	10.0	50.0	0-100.
40	**moxp_1_6**	MOX PWR	0.7	0.7295	10.0	60.0	0-100.
41	**moxp_1_7**	MOX PWR	0.7	0.7295	10.0	70.0	0-100.
42	**moxb_4_5_2**	MOX BWR	0.7	0.2	4.0	50.0	0-100.
43	**moxb_4_5_5**	MOX BWR	0.7	0.5	4.0	50.0	0-100.
44	**moxb_4_5_8**	MOX BWR	0.7	0.8	4.0	50.0	0-100.
45	**moxb_4_6_2**	MOX BWR	0.7	0.2	4.0	60.0	0-100.
46	**moxb_4_6_5**	MOX BWR	0.7	0.5	4.0	60.0	0-100.
47	**moxb_4_6_8**	MOX BWR	0.7	0.8	4.0	60.0	0-100.
48	**moxb_4_7_2**	MOX BWR	0.7	0.2	4.0	70.0	0-100.
49	**moxb_4_7_5**	MOX BWR	0.7	0.5	4.0	70.0	0-100.
50	**moxb_4_7_8**	MOX BWR	0.7	0.8	4.0	70.0	0-100.
51	**moxb_6_5_2**	MOX BWR	0.7	0.2	6.0	50.0	0-100.
52	**moxb_6_5_5**	MOX BWR	0.7	0.5	6.0	50.0	0-100.
53	**moxb_6_5_8**	MOX BWR	0.7	0.8	6.0	50.0	0-100.
54	**moxb_6_6_2**	MOX BWR	0.7	0.2	6.0	60.0	0-100.
55	**moxb_6_6_5**	MOX BWR	0.7	0.5	6.0	60.0	0-100.
56	**moxb_6_6_8**	MOX BWR	0.7	0.8	6.0	60.0	0-100.
57	**moxb_6_7_2**	MOX BWR	0.7	0.2	6.0	70.0	0-100.
58	**moxb_6_7_5**	MOX BWR	0.7	0.5	6.0	70.0	0-100.
59	**moxb_6_7_8**	MOX BWR	0.7	0.8	6.0	70.0	0-100.

60	**moxb_1_5_2**	MOX BWR	0.7	0.2	10.0	50.0	0-100.
61	**moxb_1_5_5**	MOX BWR	0.7	0.5	10.0	50.0	0-100.
62	**moxb_1_5_8**	MOX BWR	0.7	0.8	10.0	50.0	0-100.
63	**moxb_1_6_2**	MOX BWR	0.7	0.2	10.0	60.0	0-100.
64	**moxb_1_6_5**	MOX BWR	0.7	0.5	10.0	60.0	0-100.
65	**moxb_1_6_8**	MOX BWR	0.7	0.8	10.0	60.0	0-100.
66	**moxb_1_7_2**	MOX BWR	0.7	0.2	10.0	70.0	0-100.
67	**moxb_1_7_5**	MOX BWR	0.7	0.5	10.0	70.0	0-100.
68	**moxb_1_7_8**	MOX BWR	0.7	0.8	10.0	70.0	0-100.

Generation of Spent Fuel Isotopic Data for Fast Reactors

Fast reactor data in DANESS currently includes models that utilize fresh fuel with conversion ratios of .5, 1.0, and 1.5, but only a few isotopes are included. This situation can now improved with new results recently received from Argonne National Laboratory (ANL) and from calculations that employed the SAS2H module in SCALE5. Both of these results are based on the Super PRISM model developed by General Electric and will be incorporated into the DANESS data base.

The calculations performed by ANL for isotopic compositions are for Super PRISM models with conversion ratios of 0.0, 0.25, 0.50, 0.75 and 1.0 for both oxide and metal fuel. These results include several hundred fission products and transmuted actinides in spent fuel that are based on burnup-independent cross sections and on constant flux for depletion. They further include both charge and discharge masses for inner and outer regions of the reactor as well as the total reactor.

The calculations performed at The University of Tennessee (UTK) utilize the SCALE/ORIGEN software developed by ORNL, and they obtain burnup-dependent results for the nuclides included in the DANESS data base for LWRs. Example plots of burnup-dependent results are shown for Pu-239, and U-238 in Figures 1 and 2.

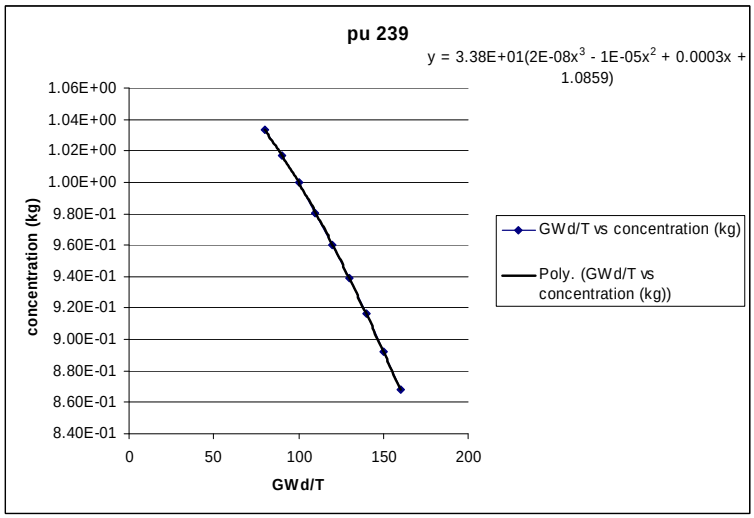


Figure D-2. Variation of Pu-239 concentration with burnup normalized at 100 GWd/T

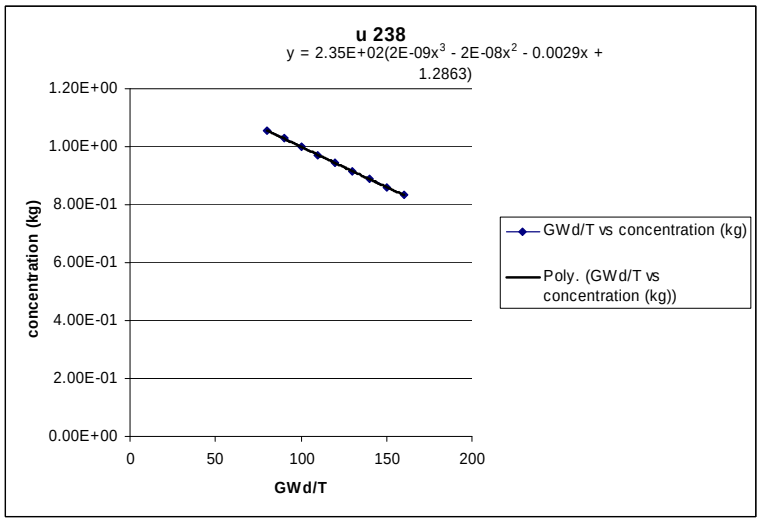


Figure D-3. Variation of U-238 concentration with burnup normalized at 100 GWd/T

Fast reactors are likely to play a critical role in managing the inventory of transuranics as well as the heat load of spent fuel in repository. For this reason, a comparison of fast reactor designs is performed that will contribute isotopic data for uses in other portions of this research. A previous study on advanced burner reactors by Hoffman, E. A., et al at Argonne National Laboratory used the Super

PRISM design to compare isotopic compositions in spent fuel with conversion ratios of 0, .25, .5, .75, and 1 for both metal and oxide fuels[1,2,3]. The results of the ANL study are compared with results from cores designed in SCALE 5.1.

SAS2, a 1-D point depletion code, is used as the driver for cross section generation and ORIGEN-S[4]. A different geometry is used for each fuel type and ideal conversion ratio, leading towards 10 individual reactor designs. Fuel assembly burn ups range from 80 to 200 GWd/t by steps of 10 GWd/t. The cross section libraries created by the SAS2 procedure are then used by the ARP module to reduce computing time for future analyses. In particular, the ARP module is used to calculate the masses and decay heats for 66 isotopes over a period of 1500 years, similar to the LWR portion of this research.

Parameters used in the design of the UT code are derived from the paper by Dubberly, et al[2]. For the entire core, the metal fuel has a mass of 26,181 kg and the oxide fuel has a mass of 34,914 kg. Power density, power per metric ton of fuel, for the two different types of fuel was calculated using a thermal power of 1000MWth divided by the mass of the fuel in metric tons. Table 1 shows the power densities for each reactor type.

Table D-2: Power Density

Fuel Type	Metal	Oxide
Reactor power (MWth)	1000	1000
Core mass (kg)	26181	34914
Power Density	38.196	28.642

Differences between the ANL and UT data exist for what are likely a few reasons. Table D-3 shows a comparison of the two results for a specific reactor type and burnup. The two codes used are rather different. ANL used DIF3D/REBUS-3 to calculate the isotopic compositions and perform fuel shuffling with the ENDF/B-V.2 cross section set underneath. The SAS2 code used by UT performs no fuel shuffling, thus isotopics are likely to be different. ANL also used multiple fuel batches with different enrichments for blankets, whereas the 1-D code assumes a single assembly in an infinite medium without any blanket or reflector assemblies. Thus the differences should be at least subtle if not significant. Lastly, the ANL paper did not vary the burn ups as was done by UT, thus differences across the burnup of a single assembly will vary the isotopic composition of fission products and transmutation.

Table D-3: Comparison of ANL and UT data for CR=1 at 109.9 GWd/t burnup using oxide fuel

	sas2 results	ANL results	Difference	Percent Difference %
Pu	0.121585897	0.167116	0.045530103	31.5412562
Np	0.00164968	0.000894	-0.00075568	-59.41625754
Am	0.005299192	0.003658	-0.001641192	-36.64523997
Cm	0.001763036	0.00174	-2.30362E-05	-1.315215235
U-235	0.000540962	0.000207	-0.000333962	-89.29912069
Pu-238	0.001773077	0.001924	0.000150923	8.164454247
Pu-239	0.073653846	0.103694	0.030040154	33.87710028
Pu-240	0.040467949	0.048982	0.008514051	19.03645872
Pu-241	0.005691026	0.007386	0.001694974	25.92293394

Am-241	0.004144872	0.002069	-0.002075872	-66.8141173
Cs-137	2.53141E-05	0.004799	0.004773686	197.9011232
U-235/U	0.00062518	0.0003	-0.00032518	-70.29548816

To illustrate the differences in isotopic compositions, Figure 2 shows the percent of U-235 with respect to the total amount of Uranium for a range of burnups. SAS2 calculates a higher percentage for the lower burnups while the difference between the UT and ANL results decrease as burnup increases. Figure D-4 shows a similar plot for Pu-241.

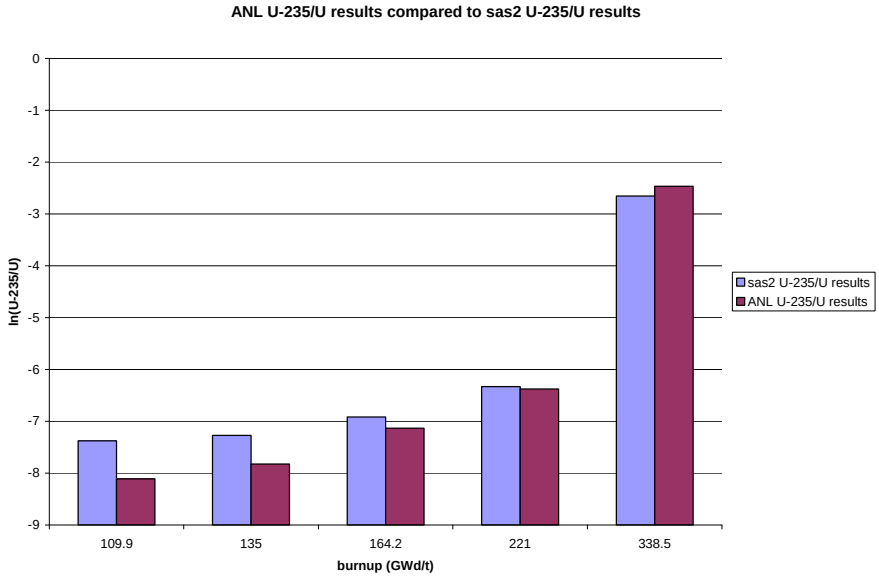


Figure D-4. Comparison of percent U-235 in total Uranium for range of burnups

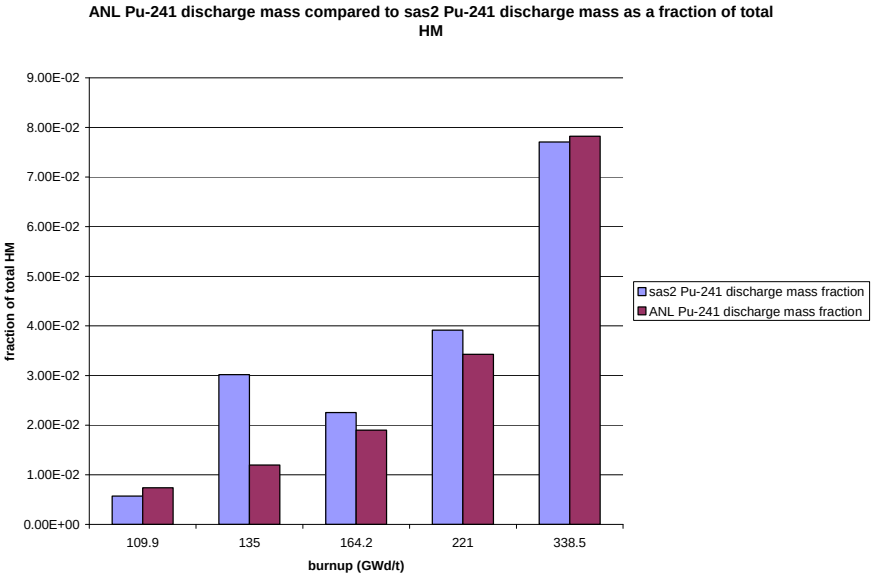


Figure D-5. Comparison of Pu-241/MTHM for range of burnups

Plutonium concentrations are controlled by the various conversion ratios that are made possible by the different assembly dimensions. In this case, larger assemblies have more U-238, which gives a larger conversion ratio from self-shielding and resonance effects. The smaller assemblies have lower conversion ratios from the smaller amount of U-238 present. Isotopic product is followed using the PlotOPUS module of SCALE. Figure 4 shows the concentrations of Pu-239 normalized to a burnup of 100 GWd/t for the 5 conversion ratios in the oxide fuel reactor group. In all cases, reactor 1 corresponds to an ideal conversion ratio of 0 with each subsequent reactor corresponding to an increase in conversion ratio of .25 with reactor 5 having an ideal conversion ratio of 1.

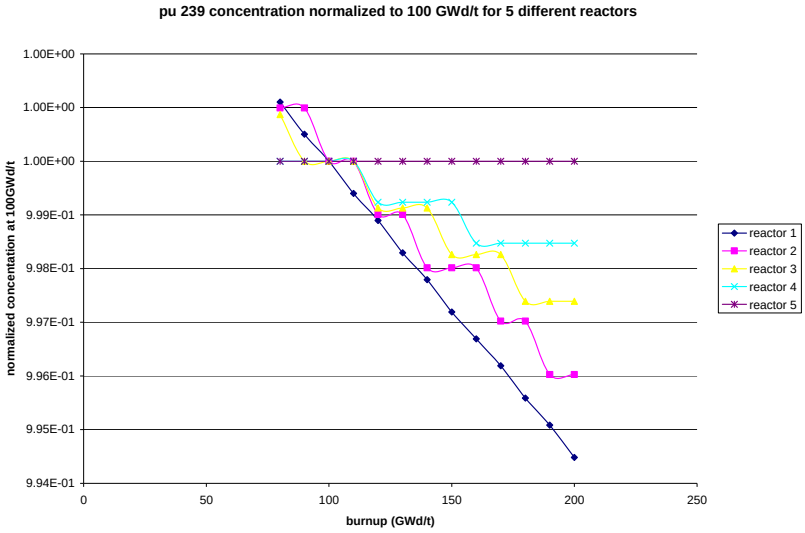


Figure D-6. Normalized Pu-239 Concentration of oxide fueled reactor

Numerous studies have shown transuranics, specifically plutonium isotopes, are the largest contributors to the long term heat load of spent fuel. Figure 5 shows the decay heats in Watts for various isotopes from 1 MTHM of spent oxide fuel from a reactor with conversion ratio of .5.

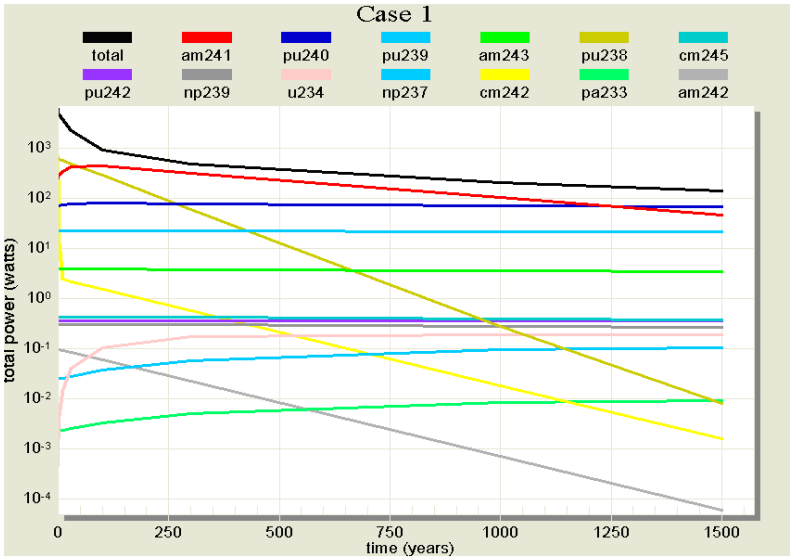


Figure D-7. Decay heat of 1 MTHM oxide fuel from reactor with CR=.5

Preliminary work using the 1-D infinite lattice code shows some differences in the isotopic compositions; however, it is noted that the slopes of these concentrations nearly match that of the ANL data despite the use of the 1-D code without core shuffling. Future work into the creation of isotopic data for fast reactors of varying conversion ratios will use the TRITON code, also present in the SCALE package and possibly fuel shuffling if time permits.

Appendix E: High Burnup and MOX Fuel Simulations

The U.S light-water reactor park is simulated beginning year 2000, where Table D-1 displays the initial fuel cycle conditions. The spent fuel accumulated at the year 2000 consists of what is called "legacy spent fuel", which is the amount of spent fuel stored in facilities away from the reactors, and the spent fuel still residing at reactor sites. This case takes these differences into account and transfers the spent fuel on-site to an interim storage located away from the reactors at a rate specified by DOE/EIA information on spent fuel generated in the US. Note that in Table E-1, "BWRe" and 'PWRe" represent the existing reactor parks. The new light-water park will be labeled "BWRn" and "PWRn". These new reactors have the same reactor attributes as the existing reactor parks, however the new reactor parks are loaded with a higher burn up fuel.

Table E-1. US Reactor Park and Fuel Cycle conditions in 2000 (Adapted from Durpel)

Parameters	BWR, i.e. 'BWRe'	PWR, i.e. 'PWRe'
Initial installed capacity (MWe)	32289	66160
Average Unit Capacity (MWe)	900	950
Average Load factor (%)	90	90
Average Thermal efficiency (%)	34	34
Expected technical lifetime (years)	60	60
Fuel used	UOX40	UOX50
Average Burnup (GWd/tHM)	40	50
Cycle Length (mo)	12	12
Number of batches	4	5
Fresh Fuel Composition (t/tHM)		
U_{nat}	1	1
DU	0	0
REPU	0	0
Initial U enrichment (%)	3.7	4.2
Pu	0	0
MA	0	0
Spent Fuel Composition		
U_{nat}	-	-
DU	-	-
REPU	0.945	0.935
Spent U enrichment (%)	0.81	0.81
Pu	0.01085	0.012
MA	0.00114	0.00125
FP	0.04225	0.05130
SF-amount in storage, i.e. legacy + At-Reactor storage (tHM)	16 707	30 292

Because this was a screening calculation, energy demand was set to zero growth. The energy demand stays at a constant 770 TWe/yr. Then as previously stated, the following two new fuels were added to the model:

1) Fuel 4 is 50 BU for the new BWR park.
2) Fuel 5 is 60 BU for the new PWR park.

The Fuel 4 information is given in Table E-1. The same input for 50 BU used for PWRe is used for BWRn. Original loadings for the existing LWR park and are given in Table E-1. Table E-2 lists the fuel data for Fuel 5, which is the 60 BU fuel used in the new PWR park.

Table E-2. Fuel data for 60 BU fuel used in PWRn

Average Burnup (GWd/tHM)	60
Cycle Length (mo)	12
Number of batches	5
Fresh Fuel Composition (t/tHM)	
U_{nat}	1
DU	0
REPU	0
Initial U enrichment (%)	4.8
Pu	0
MA	0
Spent Fuel Composition	
U_{nat}	-
DU	-
REPU	0.92553
Spent U enrichment (%)	0.85
Pu	0.01263
MA	0.00184
FP	0.06

Note that the burnup is increased by 10 GWd/thm for each park.

The front-end parameters of the fuel cycle are set extremely high so that there are virtually no limitations in front-end capacities in mining, converting, enriching, and fabrication of UOX fuels. The back-end of the fuel cycle parameters are set so that the spent fuel remains in interim storage; thus, there is subsequent geological disposal.

Utilization of MOX Fuel

The next step was to implement time of transition in fuel use. This was accomplished by inputting the shutdown of all existing reactor capacity at one year using the "Planned Shutdown" input graph in DANESS. Therefore the new reactor/fuel combination would be put into use immediately. For example in the 2010 implementation run, all 32,289 MWe and 66,160 MWe for BWRe and PWRe respectively, were planned for shutdown in the year 1990. In order to keep up with energy demand the new reactors were implemented immediately except for a 20-year lag from planned existing reactor shutdown to PWRn and BWRn implementation. The 20-year delay is apparently a feature of DANESS that requires time to build new capacity. The time of fuel implementation was varied in 10-year intervals, starting from 2010 and ending at 2090. The setup for this run is almost the exact information preset in DANESSv1.2.3r US model with PU-Mono recycle. There were, however, several key changes. First, energy demand was set to zero growth as before. The energy demand stays at a constant 770 TWe/yr. In the reactor park decision making, 30% of new reactors are set to reactor 5 (MOX reactor), and 70% are set to reactor 4 (PWRn reactor). There is an unlimited amount of fuel cycle facility capacity available for reprocessing. The reprocessing fractions for the UOX 40 and UOX 50 are set to one, meaning the entire amount of UOX spent fuel is available for reprocessing. Lastly under the reactor data, the lifetime of the reactor is set to 200 years so that it will continue to operate throughout the whole time period. Table E-3 below gives the fuel data for the MOX fuel

Table E-3. Fuel Data for MOX fuel (Adapted from Durpel).

Average Burnup (GWd/tHM)	50
Cycle Length (mo)	12
Number of batches	5
Fresh Fuel Composition (t/tIHM)	
U_{nat}	0
DU	0.91903
REPU	0
Initial U enrichment (%)	0.25
Pu	0.08097
MA	0
Spent Fuel Composition	
U_{nat}	0
DU	0
REPU	0.88753
Spent U enrichment (%)	0.18
Pu	0.05512
MA	0.00740
FP	0.04996

The next step was to implement time of transition in fuel use. This was accomplished by inputting the planned shutdown of all existing reactor capacity 20 years before actual shutdown would occur. With the shutdown of all existing reactors the new reactors would all be built in one year. The time of fuel implementation was varied in 10-year intervals, starting from 2010 and ending at 2090.

Results for Higher Burn up

Data were generated from the DANESS code and exported into Excel spreadsheets. The spreadsheets contain a summary tab, which lists the final accumulated data at year 2099. This summary data was used to generate the graphs of data used later in this section.

Again the results recorded are as follows:

1) Spent Fuel at Reactor (for each Reactor)
2) Spent Fuel at Interim (for each Reactor)
3) The sum of spent fuel (for each Reactor)
4) The sum of spent fuel total (all Reactors)

The results for the high-burn up case indicate the following trends:
The sooner the higher BU fuel is implemented the lower the total amount of spent fuel.
Figure E-1 shows the final spent fuel amounts for the different time implementations.

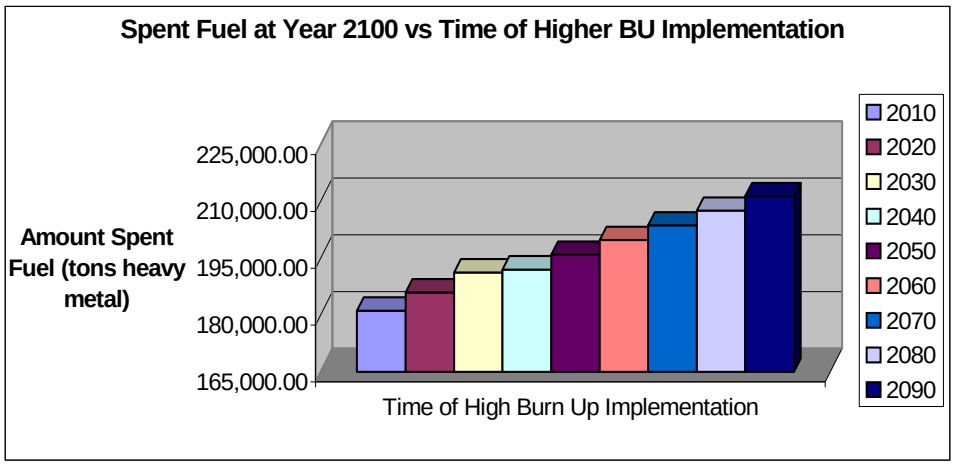
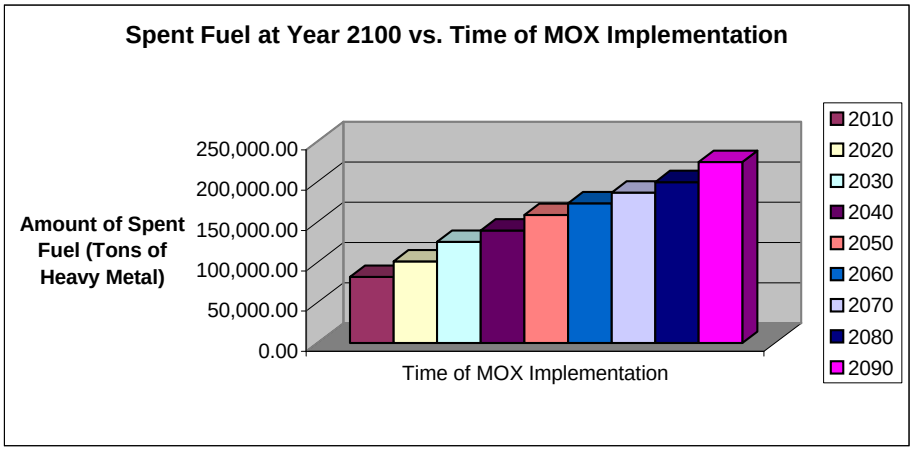


Figure E-1. Spent Fuel Totals at year 2100 vs. Time of New BU Fuel Implementation

Figure E-1 displays an expected steady increase in the amount of spent fuel on hand at 2100 as a function of time of implementation. However, the fractional savings is modest and there is only a 14% in savings in spent fuel for a 10 GWd incease in burnup. It is expected that increasing the burnup to 100 GWd would result in reducing the fuel inventory by about a factor of two, but the reduction in the required repository space is less since the higher burnup will produce more decay heat.

Results for MOX fuel



The results for the MOX case are illustrated by Figure E-2. The sooner that MOX reactors are implemented into the reactor fleet the less total spent fuel is accumulated in the year 2100. Figure E-2 shows the final spent fuel amounts for the different time implementations.

Figure E-2. Spent Fuel Totals at year 2100 vs. Time of New MOX Fuel Implementation

Figure E-2 shows that the sooner MOX fueled reactors are implemented into the reactor fleet the lower the total spent fuel build up will be. Note that the reduction in the amount of spent fuel on hand at 2100 could be reduced by over 60 % if MOX recycling were introduced in 2010 as compared to introduction in 2090.

Conclusions

There was a 14% in savings in spent fuel for a 10 GWd incease in burnup. It is expected, however, that increasing the burnup to 100 GWd would result in reducing the fuel inventory by about a factor of two and that the reduction in the required repository space would be notably less since the higher burnup fuel produces more decay heat. A realistic goal would be to utilize fuel with about 70 GWd within about 20 years. The implementation of MOX fueled reactors into the reactor fleet did show a considerable percentage decrease in spent fuel the sooner they were implemented. The difference was shown to be a 64% savings in total spent fuel by implementing in 2010 rather than 2090. Introduction of both high burnup fuel and MOX fuel could realistically reduce repository requirements by about a factor or two.

Time-of-Implementation of MOX Fuel

During the second quarter of this project, variations on specific cases are considered. Those cases are the implementation of higher burn-up and MOX fuels. Currently nuclear power plants are utilizing uranium oxide fuels with specific burnups of 40-50 GWd/thm. These cases consider the time of implementing higher burn up fuel into reactor parks in one scenario and MOX fuel in another. Two fuels are considered in the higher burn up case: UOX60 and UOX100 fuel. In the MOX scenario, the fuel being introduced is MOX50. This report will look at the time the higher burn-up and MOX fuel is implemented into reactors and the results will be compared.

The time 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. This was done for three energy scenarios: O% growth, 1.5% exponential growth, and 3% exponential growth in demand for nuclear energy. The energy demand growth begins in the year 2010. 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.

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

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

This DANESS scenario models the nuclear reactor park with no change in its fuel usage. This basic model determines the amount of spent fuel if the industry continued the once through 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.

Scenario Analysis

This basic model provides somewhat of a measuring stick against other fuel cycles in this report. The initial BWR capacity is 31,789 MWe and the initial PWR capacity is 62,997 MWe. The model is set up so that with future energy growth the nuclear reactor park will be made up of 30% BWR and 70% PWR in order to meet energy demand. The BWR reactors will utilize UOX40 while the PWR reactor uses UOX50. This fuel/reactor combination will stay consistent throughout the simulation. The fuel cycle facilities were set to unlimited. This was done in order to reduce variables that may constrain the simulation to run as intended. Figure E-3 shows the spent fuel total for this case with 0% growth.

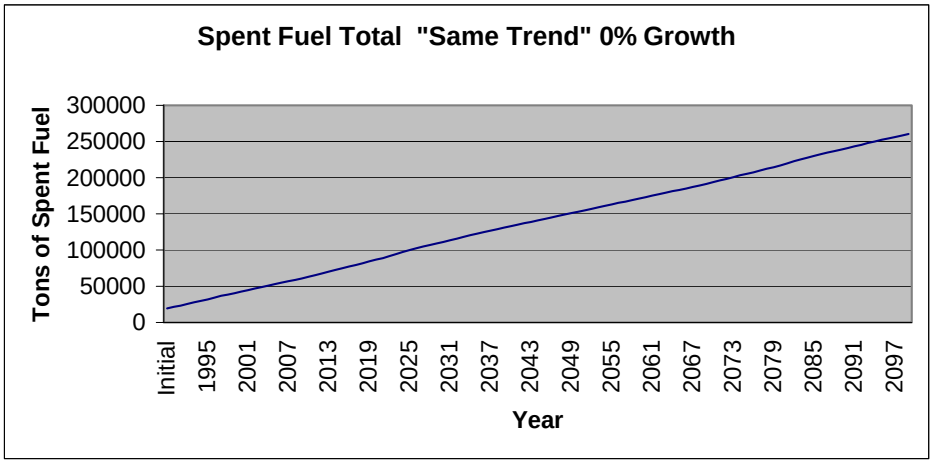


Figure E-3. Spent Fuel total for "Same Trend" case - 0% Growth

Figure E-4 shows the spent fuel mass total for this case with 1.5% energy growth.

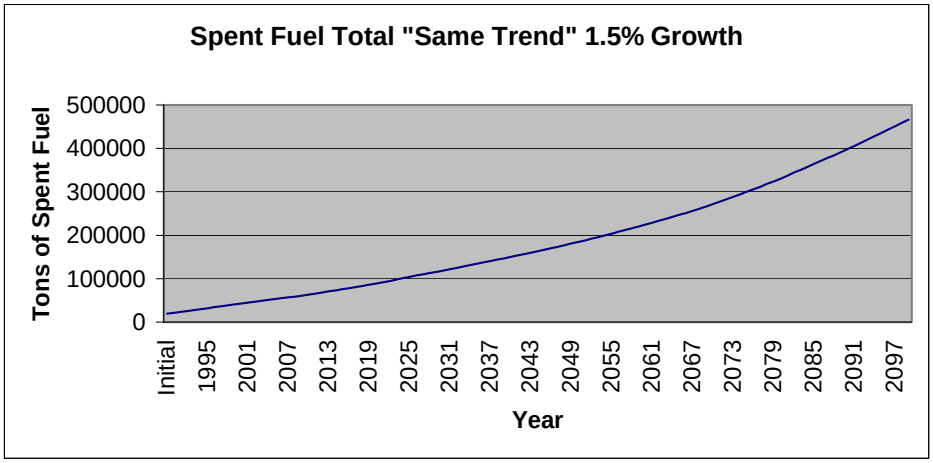


Figure E-4. Spent Fuel total "Same Trend" - 1.5% Growth

Finally Figure E-5 is the spent fuel total for 3 % energy growth.

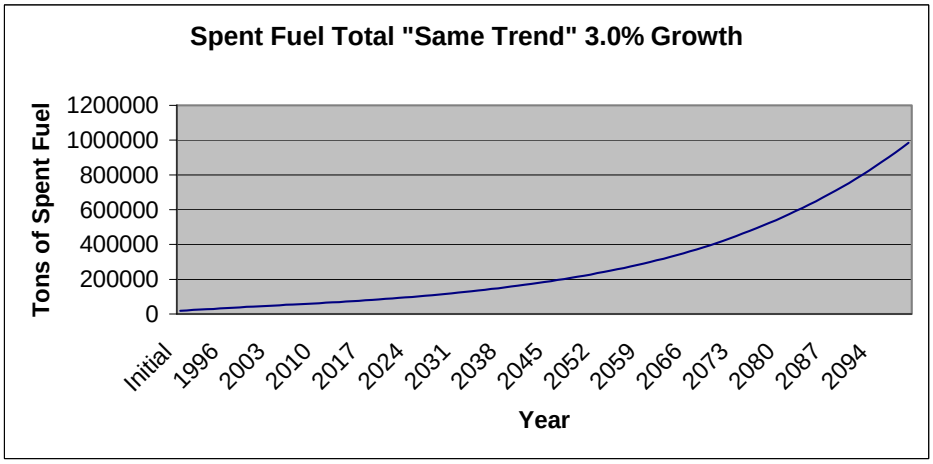


Figure E-5. Spent Fuel Totals "Same Trend" - 3.0% Growth

Methodology and Set up for High Burn up and MOX

There are several input parameters common to both the high burn up and MOX 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 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 both the high burn up and MOX cases. Table 1 displays this data.

	BWRe	PWRe	BWRn	PWRn	PWRnM
Power (MWe)	900	950	910	950	1000
Efficiency (%)	34	34	34	34	34
Fuel Load (%)	90	90	90	90	90
Licensing time (yrs.)	2	2	2	2	2
Construction time (yrs.)	4	4	3	3	3
Lifetime (yrs.)	60	60	100	100	100

Table E- 4. Reactor Data

Where the ending "e" in Table 1 signifies "existing" and "n" signifies "new" reactors. The "new" reactors will be the ones utilizing the high burn up fuels. The PWRnM reactor is only constructed in the MOX case. This will be the reactor that consumes MOX fuel.

Time Implementation for Higher Burn up and MOX

In order to look at how the time of implementation affected the data, six separate runs in each case were done. The standard shutdown profile was taken from eia.doe.gov, which is the Department of Energy website. The manner of time implementation was consistent in all cases. Additional 10-year increments were added to the current shutdown profile, thereby effectively delaying the change in current nuclear industry trends. The first run was done using the standard shut down profile; the second and each of the next five were done by extending the shutdown profile by 10 years. So 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 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 two cases were then done three more times with varying energy demands applied to them.

The following sections discuss the details specific to the high burn up and MOX case.

Higher Burn up Scenario

Using the DANESS code, the effects of implementing higher burn-up fuel over time was looked at. This was done 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 data for all of these fuels can be seen below in Table E-5.

Table E-5. Fuel Data

Fuel	UOX40	UOX50	UOX60	UOX100
Initial Enr (%)	3.70	4.70	4.90	8
Spent U (t/t int HM)	0.945	0.93454	0.92369	0.88084
Spent Enr (%)	0.81	0.82	0.25	0.25
Spent Pu (t/t int HM)	0.01085	0.012	0.01256	0.01604
Spent MA(t/t int HM)	0.00115	0.00184	0.00171	0.003069
Spent FP(t/t int HM)	0.04225	0.0513	0.06177	0.1005

A few parameters were put into place in order to get a good read on the data. 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.

High Burn Up Results: **UOX60**

The results from DANESS v1.3.1r US were exported in Microsoft Excel sheets. The total amount of spent fuel was 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 E-6 shows the first case. That is the implementation of UOX 60 fuel in a zero energy growth scenario.

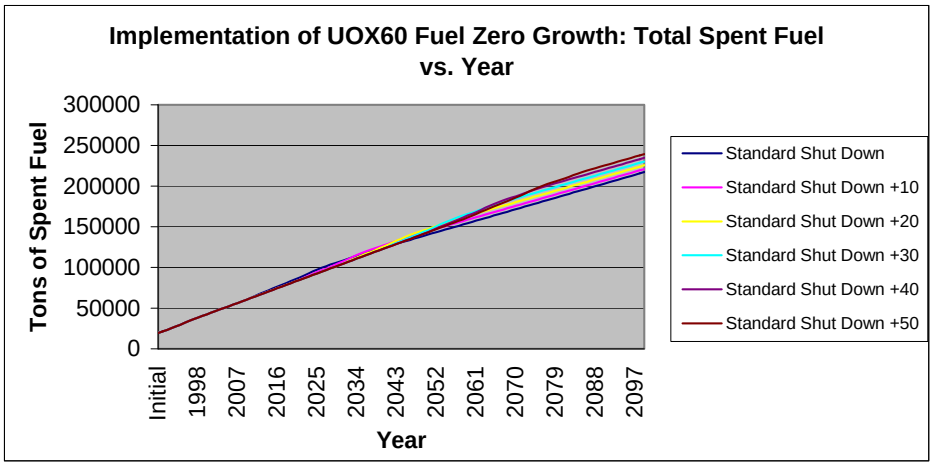


Figure E-6. Spent Fuel amounts for time of implementation with UOX 60—0%growth

The second case, shown in Fig. E-7, is for UOX 60 implementation with 1.5% exponential energy growth.

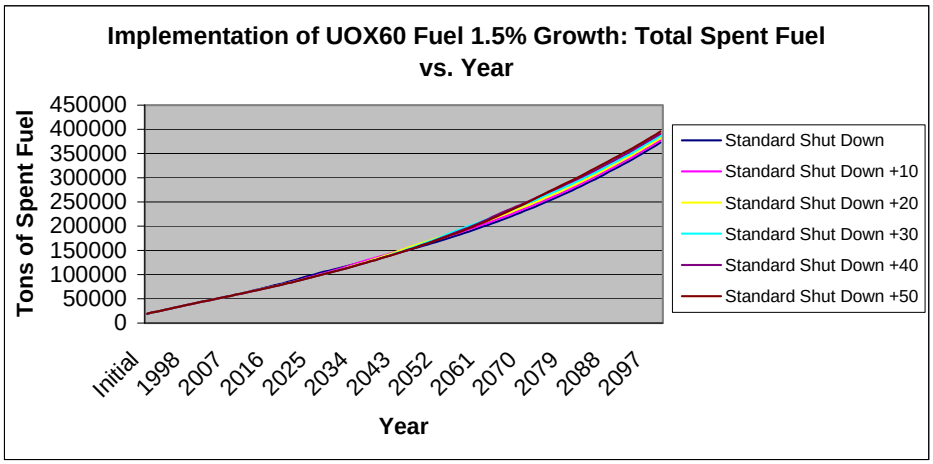


Figure E-7.
Spent Fuel amounts for time of implementation with UOX 60--1.5% growth

Figure E-8 shows the final case with UOX 60 implementation. That is the case with 3% exponential energy growth.

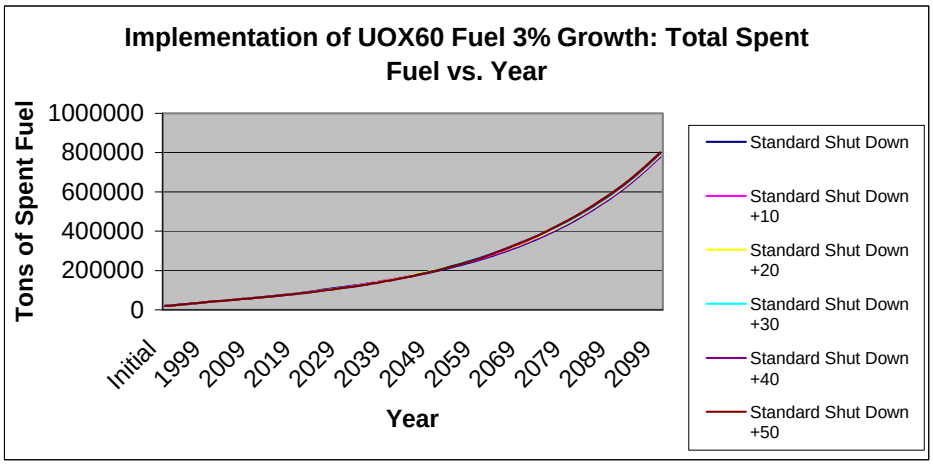


Figure E-8.
Spent Fuel amounts for time of implementation with UOX 60—3% growth

For all three energy scenarios where UOX60 is implemented, there is a maximum spread of spent fuels in the year 2099 around 22,000 tons heavy metal.

High Burn Up Results: **UOX100**

The implementation of UOX 100 fuel is the next case. Again this was done for three energy scenarios. The first scenario is 0% energy growth. The results are plotted in Figure E-9.

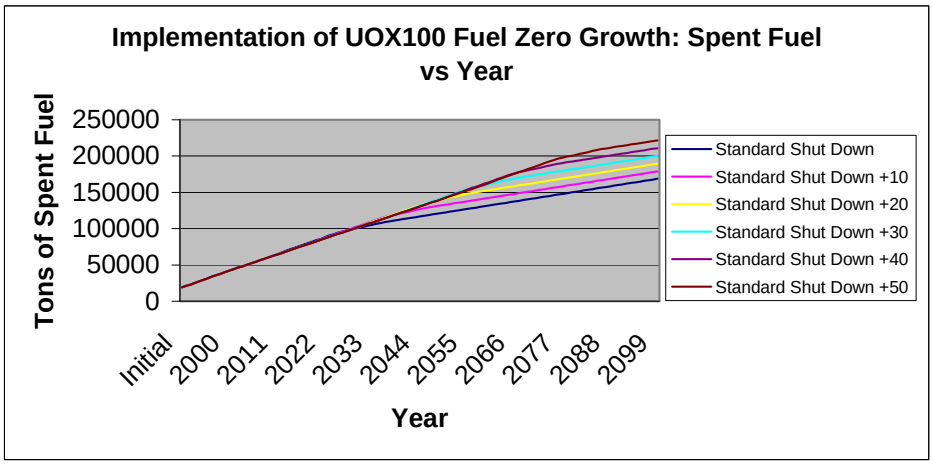


Figure E-9.
Spent Fuel with time implementation for UOX 100--0% growth

The case of 1.5% exponential energy growth for UOX 100 implementation is shown in Figure E-10.

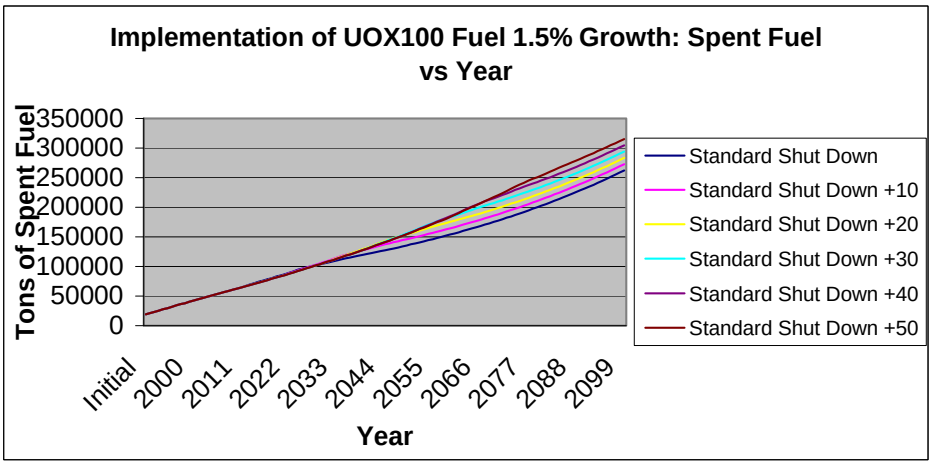


Figure E-10.
Spent Fuel for time of implementation for UOX 100--1.5% growth

The final plotted spent fuel result is that of the 3% energy growth case for UOX 100 implementation, which is plotted in Figure E-11.

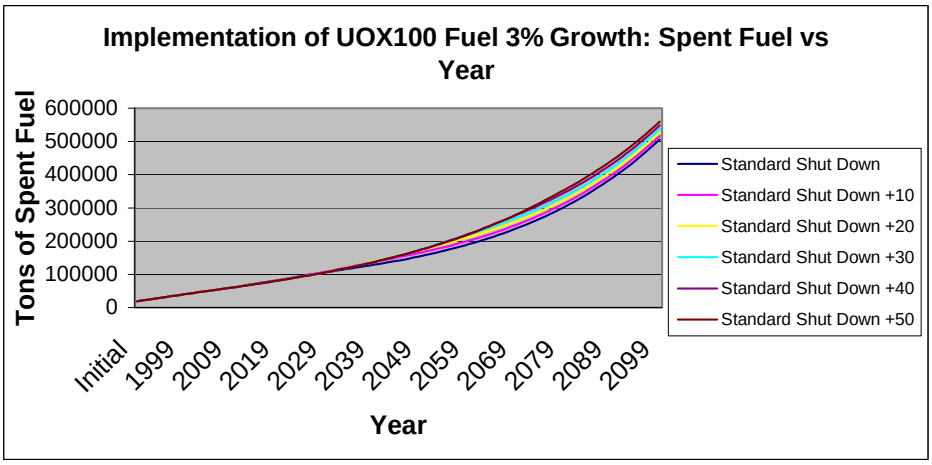


Figure E-11. Spent Fuel for time of implementation for UOX 100-3% growth

For all three energy scenarios where UOX100 is implemented, there is a maximum spread of spent fuels in the year 2099 around 53,000 tons heavy metal, which is over double what we saw in the UOX60 case.

Results: MOX Scenario

The MOX50 case was modeled primarily in the same manner as the higher burn up scenarios. Data for the MOX50 fuel is shown in Table E-6.

Table E-6. MOX50 Fuel Data as Implemented in DANESS

Fuel	MOX50
Initial DU (t/t int HM)	0.91903
Initial Enrichment (%)	0.25
Initial Pu (t/t int HM)	0.08097
Spent U (t/t int HM)	0.88753
Spent Enrichment (%)	0.25
Spent Pu (t/t int HM)	0.0512
Spent MA (t/t int HM)	0.0074
Spent FP (t/t int HM)	0.04996

There were several different methods and scenarios ran for the MOX50 implementation case. In all cases, the time of MOX fuel implementation was done 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"

What is meant by "User Defined" and "Automatic Reprocessing Deployment" simply lies in the manner in which DANESS deploys annual reprocessing capacity. DANESS allows the user to choose between 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 users to input by graph or table format the desire fuel cycle facility capacity. "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.

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

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 10% MOX50 and 90%UOX50. Figure E-12 displays the spent fuel totals for this case at zero energy growth. As was discussed in the previous section, the capacity of aqueous reprocessing was set at 1500thm/yr, beginning in the year 2000. The remainder of the fuel cycle facilities, such as fuel fabrication, conversion, and spent fuel storage were set to unlimited in order to constrain the model.

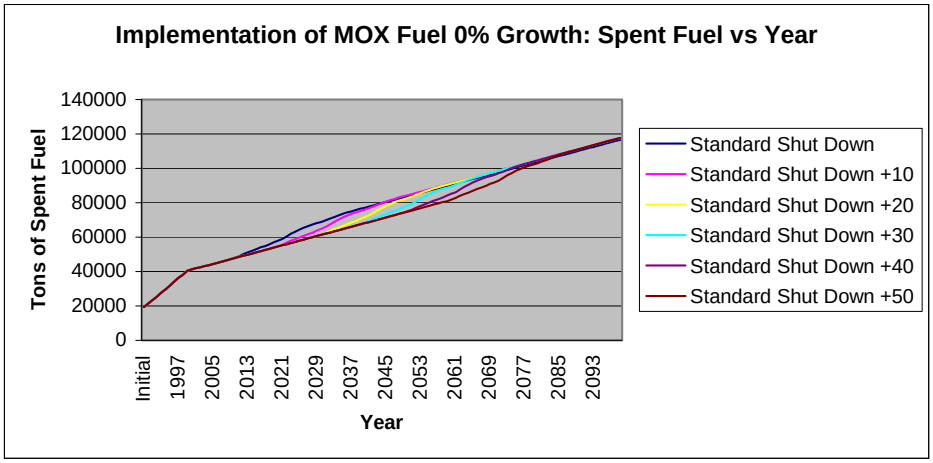


Figure E-12. Spent Fuel totals for time of fuel implementation for 10% Reac Park, 10% Core MOX "User Defined"-0% growth

Figure E-13 displays the spent fuel totals at 1.5% exponential energy growth.

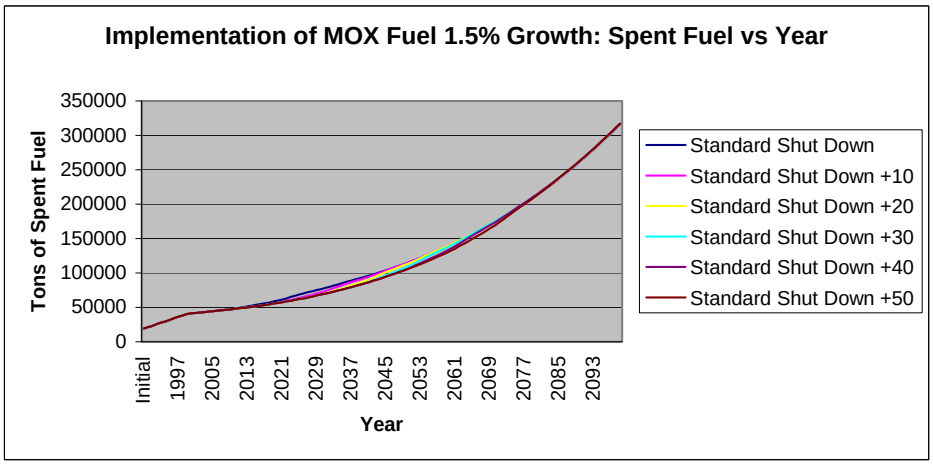


Figure E-13. Spent Fuel totals for time of implementation 10% Reac Park, 10% Core MOX "User Defined"-1.5% growth

Figure E-14 shows the spent fuel totals for this case with 3.0% growth.

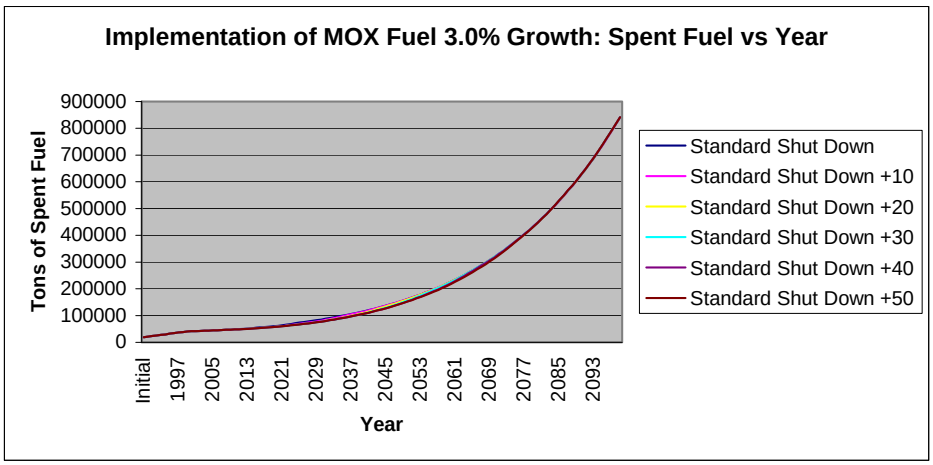


Figure E-14. Spent Fuel totals for time of implementation, 10% Reac Park, 10% Core MOX," User Defined"-3.0% growth

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

This case was set up in the same manner as the above section, with one exception. The MOX reactor core is fueled with 30% MOX50, and 70% UOX50. It was found that for this case the spent fuel totals do not change from cores fueled with 10% MOX50 and 90% UOX50. This is attributed to the fact that the overall burn-up of the remains 50GWd/thm, regardless of the amount of MOX fuel the core contains.

26% Reactor Park PWR-MOX, loaded with 30% MOX core, "User Defined"

For this case the new reactor park was set to:

- 30% BWRn
- 44% PWRn
- 26% PWRnM

The amount of nuclear capacity being supplied by MOX fueled cores is 26% of the total reactor park capacity. The BWRn and PWRn again continue use of the fuels UOX40 and UOX50 respectively. The PWR utilizing MOX fuel contains a core make-up of 70% UOX50 and 30%MOX50. The fuel cycle capacity setup and deployment is the same as the above cases with 1500thm/yr aqueous reprocessing starting in the year 2000 and continuing throughout the duration of the simulation. While the remaining fuel cycle facilities are set to unlimited. Figure E-15 show the spent fuel totals for the case of 0% energy growth.

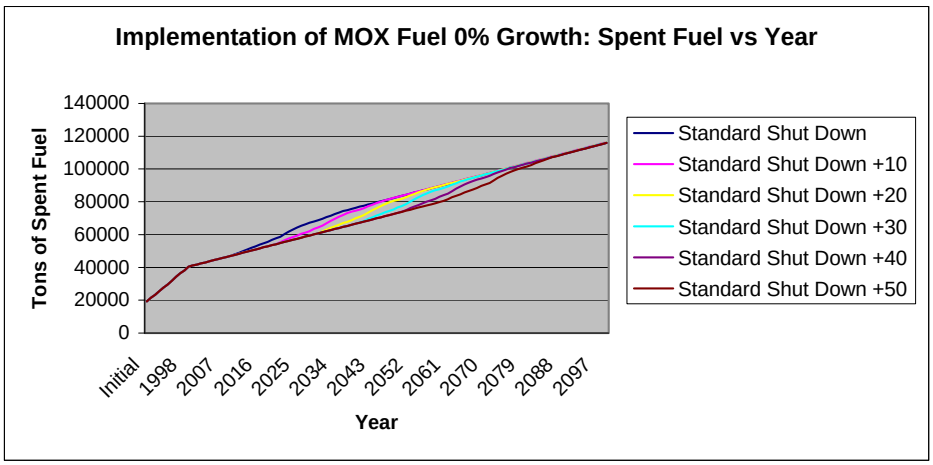


Figure E-15. Spent Fuel totals for time of implementation 26% Reac Park, 30% Core MOX, "User Defined" - 0% Growth

Figure E-16 shows the spent fuel totals for this case with 1.5% energy growth.

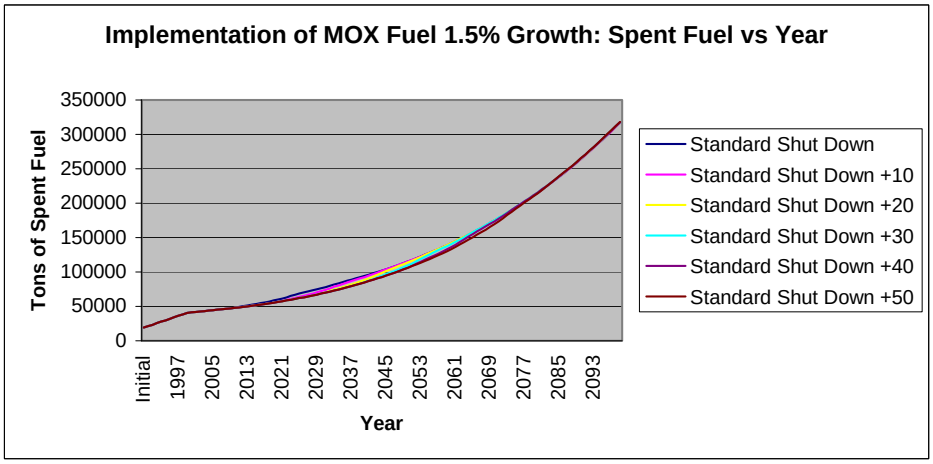


Figure E-16. Spent Fuel totals for time of implementation 26% Reac Park, 30% Core MOX, "User Defined" - 1.5%

Figure E-17 shows the spent fuel totals for this case with 3.0% energy growth.

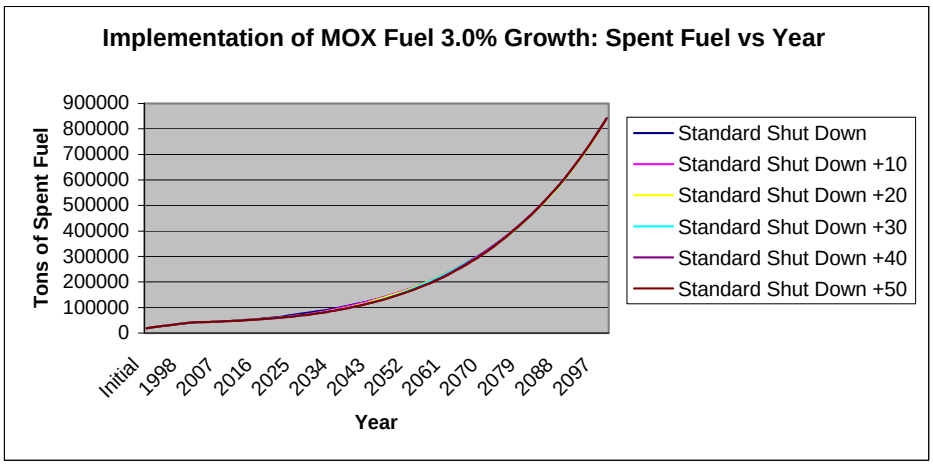


Figure E-17. Spent Fuel totals for time of implementation 26% Reac Park, 30% Core MOX, "User Defined" - 3.0% Growth

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 10% MOX50 and 90%UOX50. The difference between this case and the MOX cases in the previous sections is the use of "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" scenarios. The reprocessing capacity is determined by a forecasting sub-model in DANESS. The remainder of the fuel cycle facilities is set to unlimited as was done in the "User Defined" MOX cases.

For both "Automatic" MOX cases the maximum deployed reprocessing capacity varies. However for both this case (10% Reactor Park) and the next case (26% Reactor Park) the automatically deployed reprocessing capacity is about the same. 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 E-18 shows the spent fuel totals for this case with 0% growth.

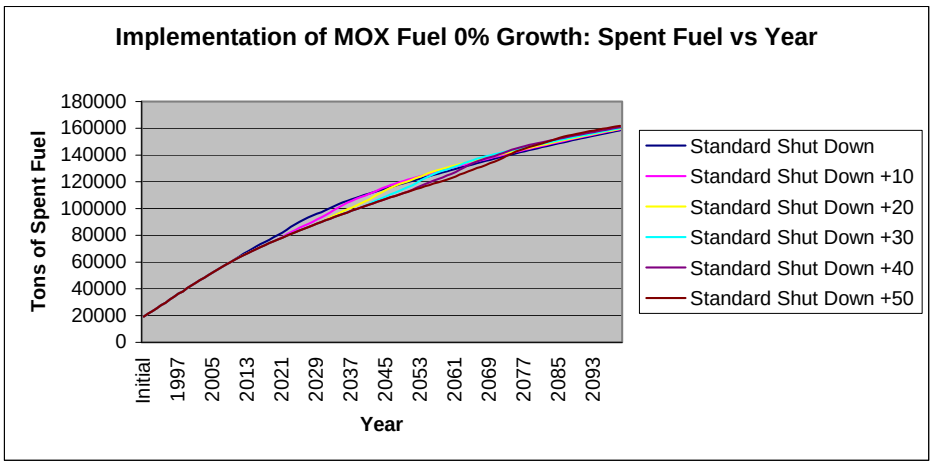


Figure E-18. Spent Fuel totals for time of implementation 10% Reac Park, 30% Core MOX, "Automatic" - 0% Growth

Figure E-19 shows the spent fuel totals from this case for 1.5% energy growth.

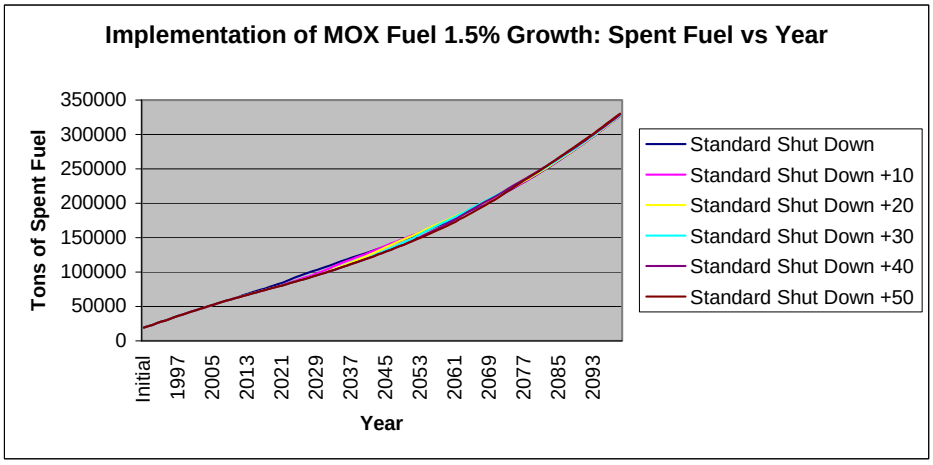


Figure E-19. Spent Fuel totals for time of implementation 10% Reac Park, 30% Core MOX, "Automatic" - 1.5% Growth

And finally for this scenario, Figure E-20 shows the spent fuel totals with 3.0% energy growth.

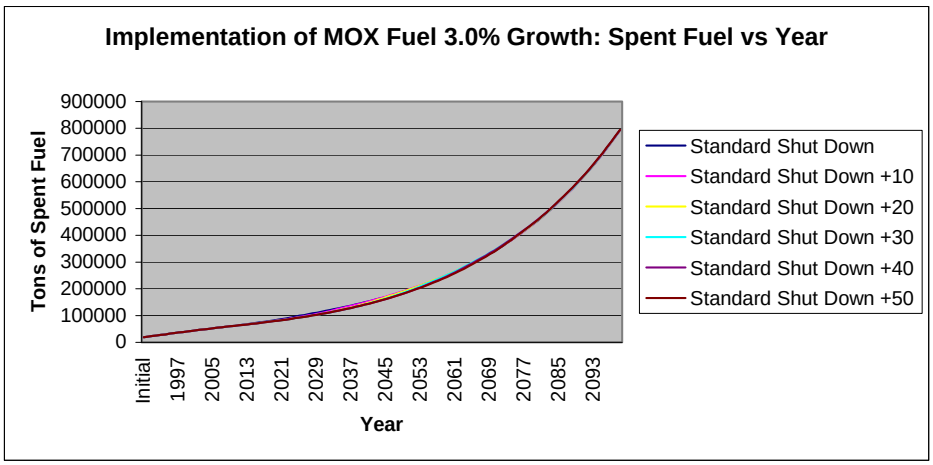


Figure E-20. Spent Fuel totals for time of implementation 10% Reac Park, 30% Core MOX, "Automatic" - 3.0% Growth

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

For this case the new reactor park was set to:

- 30% BWRn
- 44% PWRn
- 26% PWRnM

The amount of nuclear capacity being supplied by MOX fueled cores is 26% of the total reactor park capacity. The BWRn and PWRn again continue use of the fuels UOX40 and UOX50 respectively. The reprocessing capacity is described in the same manner as in the above section.

Figure E-21 shows the spent fuel totals in this case with 0% energy growth.

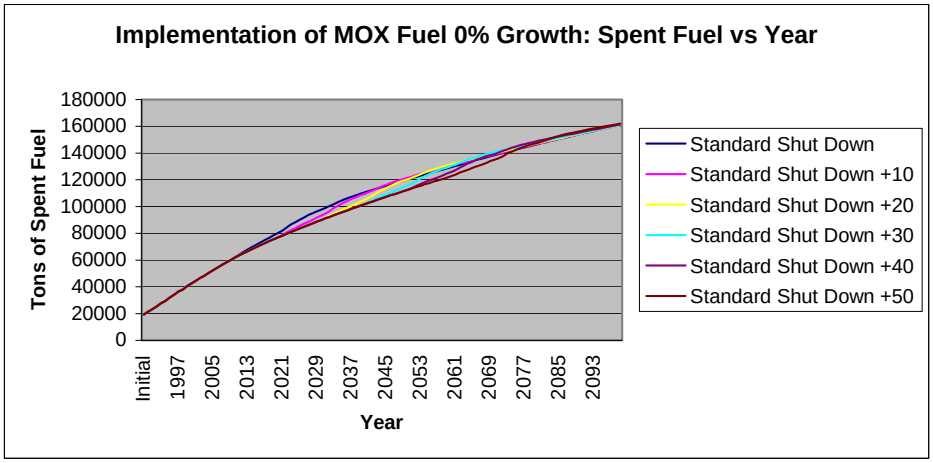


Figure E-21. Spent Fuel totals for time of implementation 26% Reac Park, 30% Core MOX, "Automatic" - 0% Growth

Figure E-22 shows the spent fuel totals for this case with 1.5% energy growth.

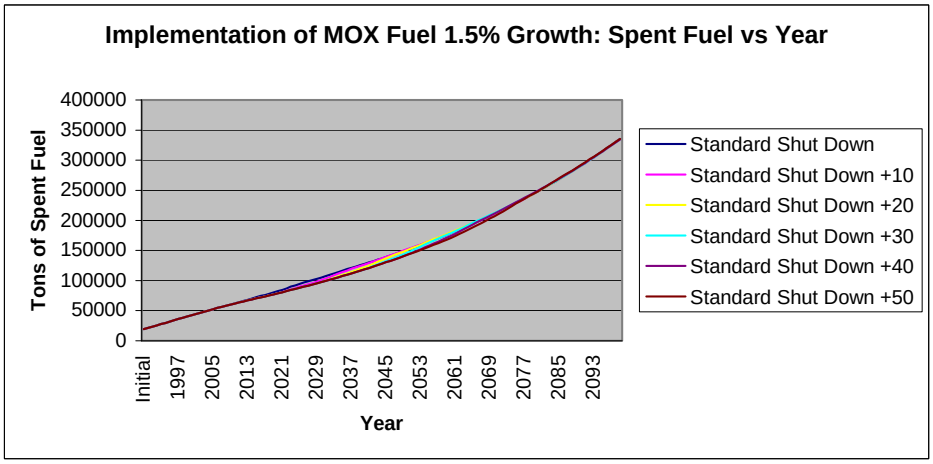


Figure E-22. Spent Fuel totals for time of implementation 26% Reac Park, 30% Core MOX, "Automatic" – 1.5% Growth

Figure E-23 shows the spent fuel totals for this case with 3.0% energy growth.

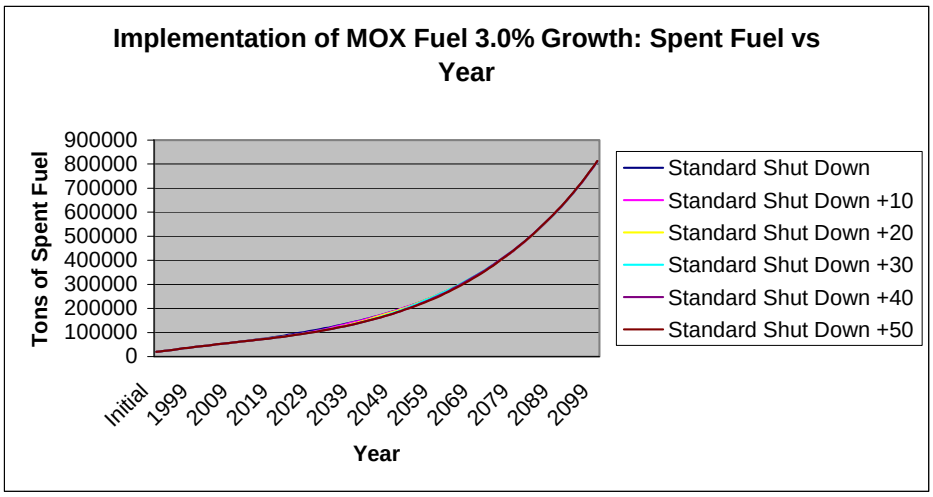


Figure E-23. Spent Fuel totals for time of implementation 26% Reac Park, 30% Core MOX, "Automatic" - 3.0% Growth

Comparison of Results

Table E-7 is a summary table of the above previous sections' spent fuel totals.

	Range of Spent Fuel Totals for individual fuel types (thm)						
Energy Growth	*Same Trend*	***12.3.7.2 UOX60***	***12.3.7.3 UOX100***	*30% MOX Core, 10% Reac Park, "User"*	*30% MOX Core, 26% Reac Park, "User"*	*30% MOX Core, 10% Reac Park, "Automatic"*	*30% MOX Core, 26% Reac Park, "Automatic"*
0%	260,000	217,000-239,000	168,000-222,000	116,000-118,000	115,900-116,300	158,600-161,900	160,900-162,000
1.5%	466,000	372,000-396,000	262,000-315,000	316,000-317,000	317,400-318,400	327,200-330,600	335,400-335,100
3.0%	985,000	779,000-802,200	505,000-560,000	841,000-842,000	838,500-842,000	792,700-795,000	811,000-813,000

Table E-7. Summary Table of Spent Fuel ranges for different fuel cycles

Every fuel cycle scenario reduces the total weight amount of spent fuel when comparing it to the same trend scenario. In a zero energy growth scenario, the lowest spent fuel totals result in the reprocessing 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 analyzed in order to gain a full understanding of the total amount of waste that would need to go to geological storage.

As energy demand growths 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 4 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.

Uncertainty of Spent Fuel Totals

The uncertainty analysis consists of examining the previous figures and the range of spent fuel totals exhibited for each year. Figure E-24 shows the uncertainty slopes for the individual cases.

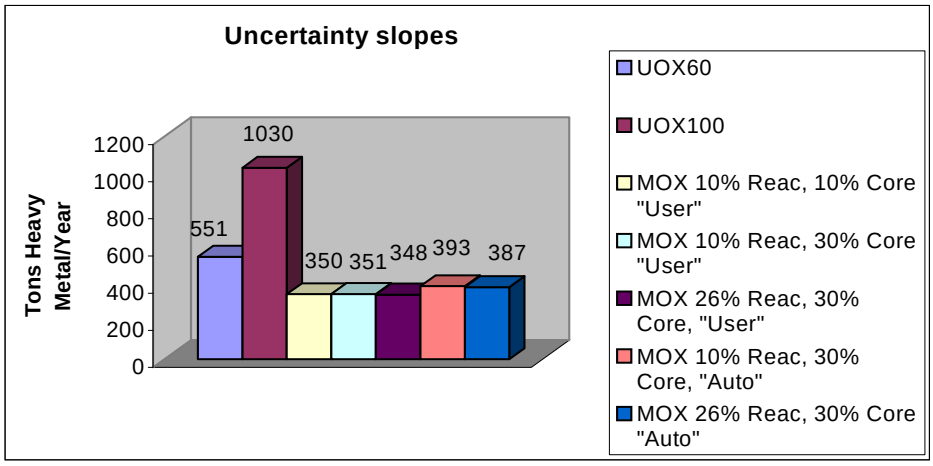


Figure E-24. Uncertainty Slopes

What is meant by uncertainty slopes is that beginning in the year 2010 the uncertainty in spent fuel mass follows a linear fit with the slopes in Figure E-24. The year 2010 is significant because it is the year energy demand begins.

The slopes are greatest for the burn up cases. This is due to the large discrepancy between the initial and implemented fuel burn-ups. The MOX uncertainty slopes stay relatively consistent, around 350 thm/yr. However, the "Automatic" MOX cases have a slightly higher uncertainty slope than "User Defined" due to the changing reprocessing capacity of the model.

Conclusions

All the advanced fuel cycles examined in this report can significantly reduce the mass of spent fuel when compared to the current industry standards. In higher energy demand scenarios high burn up UOX fuel significantly reduces spent fuel mass. Utilizing extremely high burn ups, coupled with a

median energy growth would require one less Yucca Mountain capacity. Reprocessing does not look like an immediate priority, unless the fuel cycle plans to encompass fast reactor or accelerated driven systems.

In order to better understand repository requirements, not just spent fuel mass, these spent fuel totals should be normalized according to their burn up. Less fuel mass may not result in less decay heat.

Appendix F: Time of Implementation of Fast Reactors

In order to look at how the time of implementation affected the data, six separate runs in each case were done. The standard shutdown profile was taken from eia.doe.gov, which is the Department of Energy website. The manner of time implementation was consistent in all cases. Additional 10-year increments were added to the current shutdown profile, thereby effectively delaying the change in current nuclear industry trends. The first run was done using the standard shut down profile; the second and each of the next five were done by extending the shutdown profile by 10 years. So the data collected were 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. In order to build reprocessing facilities in correspondence to the building of the fast reactors, a 1500 tHM/yr, tons of heavy metal per year, reprocessing plant was implemented 10 years after the shut down profile started to take effect. This means that for the base case the reprocessing plant was constructed in 2010 and then for the 50 year delayed case it was constructed in 2060. The reprocessing fractions for the UOX50 and UOX40 were set to 1 so as to make all of the spent UOX fuel available to be reprocessed.

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 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 two cases were then done three more times with varying energy demands applied to them.

Fast Reactors

The results from DANESS v1.3.1r US were exported into Microsoft Excel sheets. The total amount of spent fuel was 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. The resulting fast reactor data are shown below in Figures 7-9.

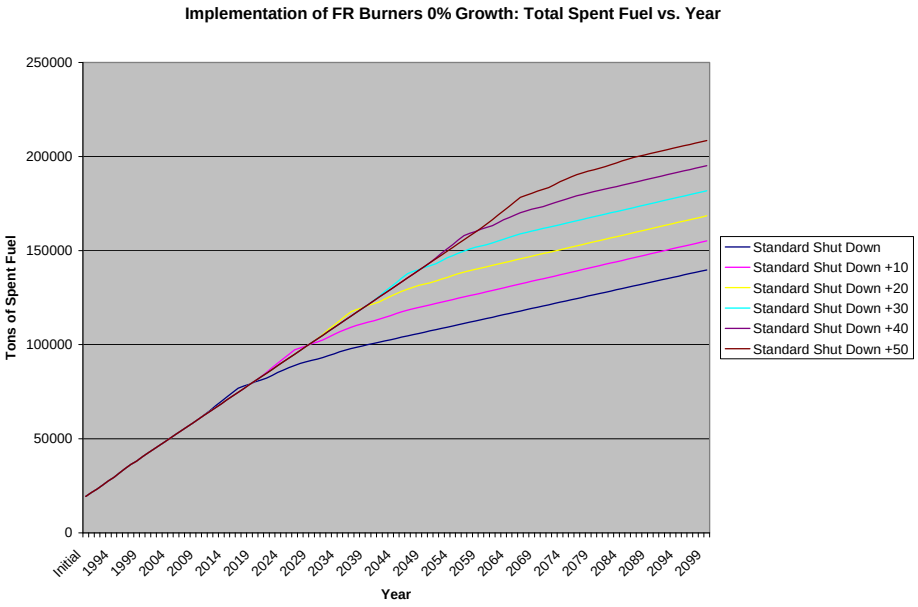


Figure F-1. Spent Fuel Totals for Time of Implementation of Fast Burner Reactors 0% Growth

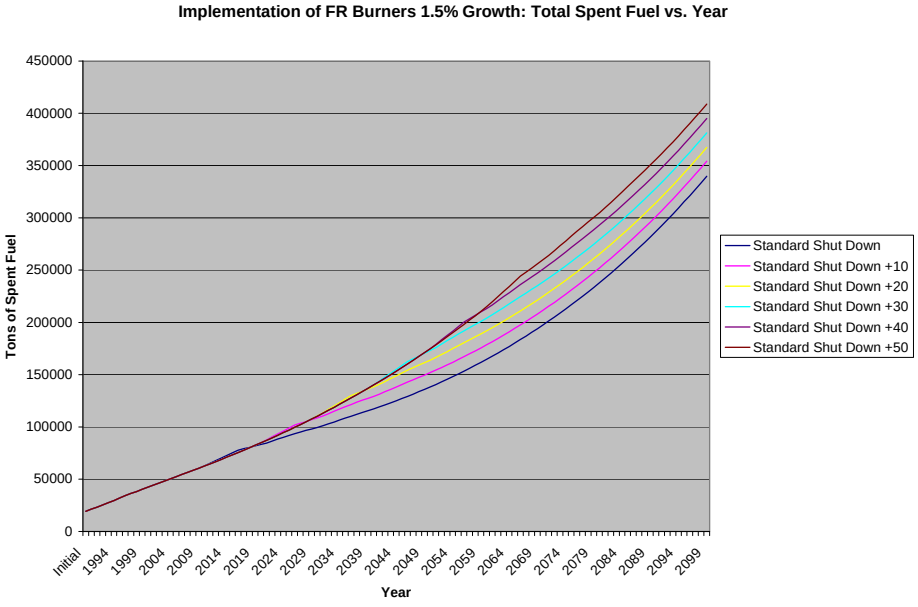


Figure F-2. Spent Fuel Totals for Time of Implementation of Fast Burner Reactors 1.5% Growth

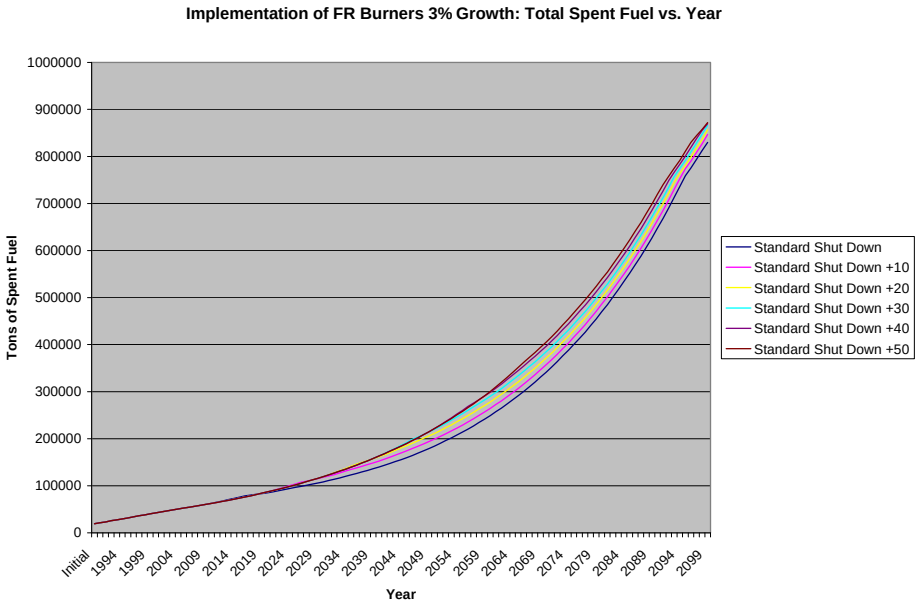


Figure F-3. Spent Fuel Totals for Time of Implementation of Fast Burner Reactors 3% Growth

Fast Reactors

Fast reactors have the capability to significantly impact the nuclear industry; however, in order for there use to be seriously considered the true effects of these reactors must be learned. Continued work into the discovery of these impacts was conducted this quarter. Using the DANESS code, multiple scenarios were looked at and studied. The following is a detailed description of the methodology, set-up, and results of the DANESS scenarios runs. As with all runs in DANESS, the first thing done is to determine what results are desired. The desired results for these cases are to attain a better understanding of the effects fast reactors will have on the reactor park. More specifically the Plutonium and minor actinide inventories and the heat load deposited to a repository from the spent fuel.

To accomplish these goals a non-realistic scenario was chosen to be run first. Taking the current US reactor fleet and extending its lifetime for the next century with no loss or addition of any other reactors is chosen as our reference case. Once the data for this reference case is collected, the Pu and MA inventories in the year 2100, a scenario taking the current US fleet of LWRs and putting a growing energy demand of 2% of original demand per year starting in the year 2010 and continuing till 2100 with no loss in the current reactor fleet and the addition of LWRs to cover the increased energy demand. This growing energy demand requires an average of two new reactors per year. Once this was conducted a third run was done exactly the same way as the second except instead of using additional LWRs to

compensate for the energy growth demand fast burner reactors are used. The reactor and fuel data used for these runs is listed in Tables F-1 and F-2.

Table F-1. LWR Reactor and Fuel Data

Power (Electric)	1500 MWe
Thermal Efficiency	34%
Load Factor	76%
Reactor Lifetime	150 Years
Fuel Burn-Up	60 GWd/tHM
Cycle Length	17.72 Months
Number of Batches	4
Initial Uranium	1 t/tHM
Initial Enrichment	4.95%
Spent Uranium	.9237 t/tHM
Spent Enrichment	.8235%
Spent Pu	.01295 t/tHM
Spent MA	.001557 t/tHM
Spent Np	.0009731 t/tHM
Spent Am	.0003912 t/tHM
Spent Cm	.0001926 t/tHM
Spent Fission Products	.06185 t/tHM

Table F-2. Fast Reactor and Fuel Data

Power (Electric)	1500 MWe
Thermal Efficiency	42.425%
Load Factor	76%
Reactor Lifetime	150 Years
Fuel Burn-Up	136 GWd/tHM
Cycle Length	14.6979 Months
Number of Batches	5
Initial Depleted Uranium	.7452 t/tHM
Initial Enrichment	.25%
Initial Pu	.25 t/tHM
Initial MA	.004794 t/tHM
Spent Uranium	.6374 t/tHM
Spent Enrichment	.09844%
Spent Pu	.2148 t/tHM
Spent MA	.009576 t/tHM
Spent Np	.0005577 t/tHM
Spent Am	.007231 t/tHM
Spent Cm	.001788 t/tHM
Spent Fission Products	.1381 t/tHM

The DANESS code was set-up using the above table data for the reactors and fuels. The code was set to have unlimited uranium resources, and unlimited amount of fuel cycle facility capacity such as milling, enriching, and reprocessing. This was done so that there would be no limitations on the runs from resources or fuel cycle facilities. The fuel for the fast reactor is just the driver fuel, no blankets were used, so that it would be a burner and not a breeder reactor. The first two runs were done using no reprocessing. The last run with the fast reactors all of the LWR and FR fuel was reprocessed.

Results from the DANESS code for three cases are shown in Figures F-4 and F-5. The reference case is for zero growth of a fleet of LWRs. The option of adding LWRs to increase capacity is denoted in Figures F-4 and F-5 as "LWR addition…" legend, and the option for increasing capacity by adding FRs is denoted by the "FR addition…" legend. The energy demand option for DANESS is utilized as two reactors per year. Note that DANESS responds to this input by adding reactor capacity as requested. In the case for addition of FRs, additional capacity is terminated after about 30 years. This is due to lack of the availability of Pu from LWRs to fuel FRs. You may note that Pu continues to increase even though no fuel is available for FRs. This is apparently due to Pu in fuel removed from FRs.

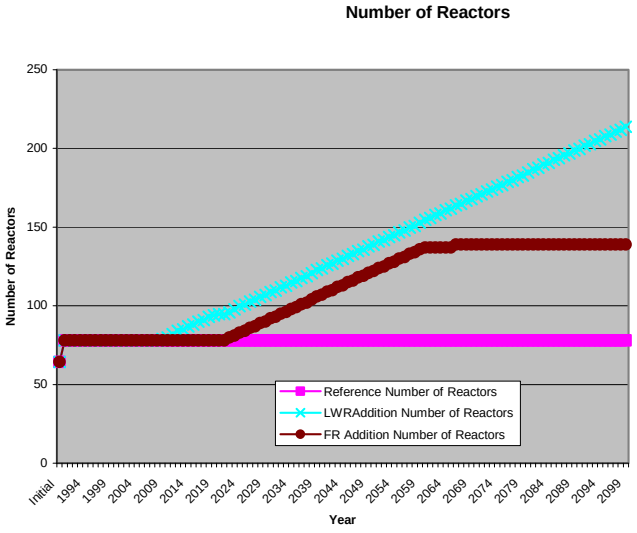


Figure F-4. Number of reactors for cases of zero growth, addition of light water reactors, and addition of fast reactors.

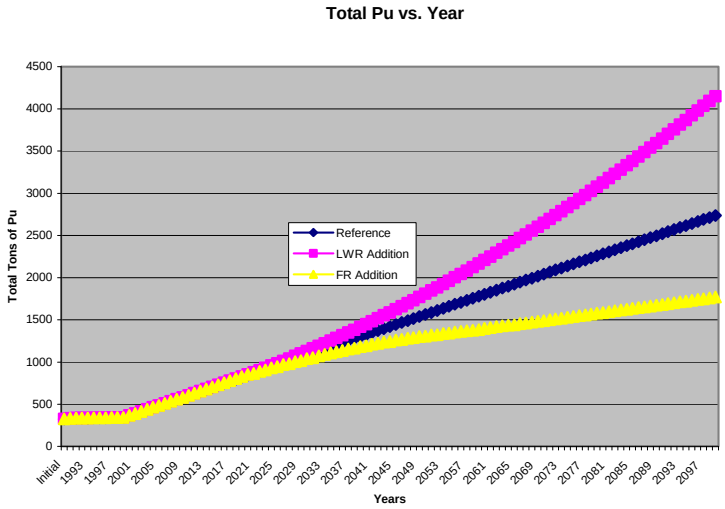


Figure F-5. Plutonium inventory for cases of zero growth, addition of light water reactors, and addition of fast reactors.

Appendix G: Shutdown Schedule for Existing Light Water Reactors

A vital part of this research is to look into the current reactor park and determine a scheme for replacing it. In order to do this effectively the license terms for all the U.S. reactors must be looked at, and then possible license extensions must be factored in to get a comprehensive overview of the U.S. reactor park lifetime.

Currently thirty-nine reactors have already completed the license renewal process and another twelve have applications under review by the NRC. In addition to these reactors, twenty-eight reactors have submitted letters of intent to apply for license renewal. This leaves just twenty-five current reactors that are not somewhere in the renewal process; however, the NRC believes that the remaining twenty-five will submit letters of intent to extend there licenses sometime in the near future.

Using the current expiration of licenses model a bar graph can be compiled showing the number of plants and which year they will need to be replaced. This is shown below in Figure F-6. Using the data from Figure F-6 a scheme for the replacement of these reactors can be produced. Taking a relatively optimistic look at reactor construction and the current political climate an assumption of a construction time of fours years per reactor is used and the first round of such construction will not be able to start for the next ten years. Using these assumptions the first reactor to be constructed will not start to be built until the year 2016 and will not be done until 2020. This means that by the time the first new reactor is under construction that twenty-two existing reactors will be shut down.

To develop a strategy for the replacement of these reactors five different linear replacement schemes are entertained. They consist of a constant construction tempo using the assumptions set forth above of four years per reactor and starting in the year 2016. The different schemes looked at are for the construction of two, three, four, five, and six reactors per year. Taking into account the shutdown of current reactors and the construction of new reactors using the above scheme a line graph is produced and is shown below in Figure F-7. Using the information in the above replacement chart it is easy to see that if maintaining the current level of reactors in the U.S. is a priority an aggressive construction plan for replacing the reactor fleet will be needed.

In addition to the current license expiration data, two more data sets were analyzed. The first one assumes that all reactors that are under review for extensions receive them, and the second one assumes that all currently active reactor receive license extensions. The first data set of under review reactors license expiration is shown below in Figure F-8. This data was used in the same manner as was done before. The use of five different linear reactor replacement schemes assuming the same four years for construction and the first ground breaking on a rector will not start until 2016. This chart is shown below in Figure F-9. As seen before in the current shut down replacement chart, an aggressive construction plan will need to be implemented in order to replenish the reactor fleet over the next thirty years.

The last data set analyzed assumed that all currently active reactors received license extensions. This perhaps is the most realistic of the three models. Using this model a bar graph is compiled in the same manner as before showing the number of reactors and in which year they shut down. This is shown below in Figure F-10. As before, the same five replacement schemes are used and coupled with the above data to produce a replacement chart. This replacement chart is shown below in Figure F-11. This chart gives a little different perspective than the previous charts. It shows that if construction is started in 2016, as assumed, it is possible to build up more than the U.S.'s current level of 104 reactors before the

current fleet starts to be shut down. In addition to this build up, if aggressive construction is undertaken reactor levels can be more than maintained they can be increased significantly.

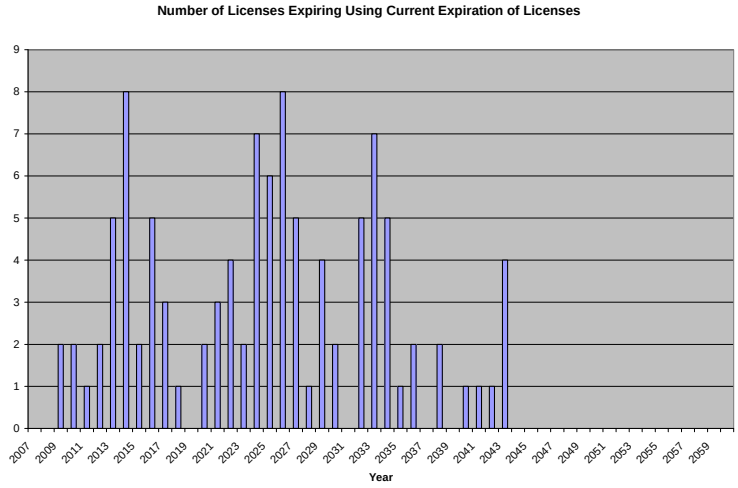


Figure F-6. Schedule of expiring reactor licenses.

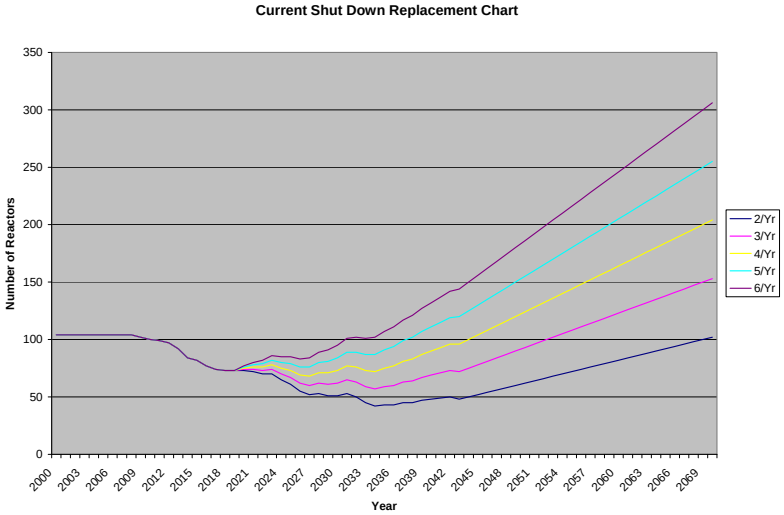


Figure F-7. Number of reactors in operation for replacement schedules of 2-6 years.

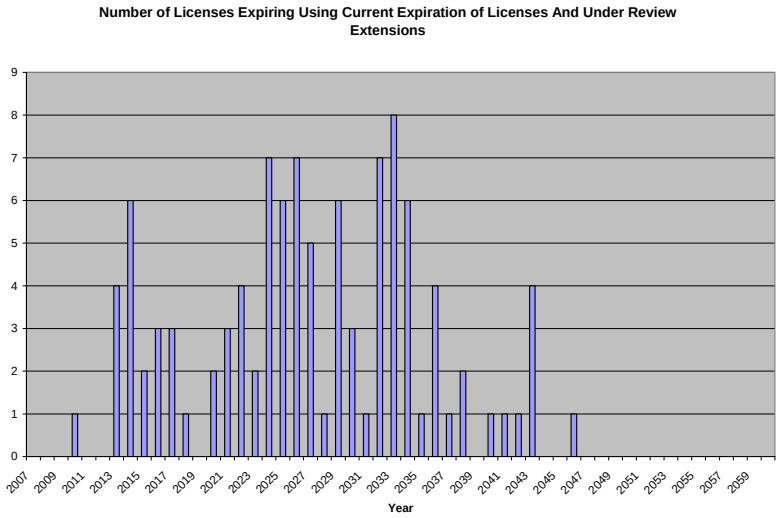


Figure F-8. Schedule of expiring reactor licenses with consideration of known extensions.

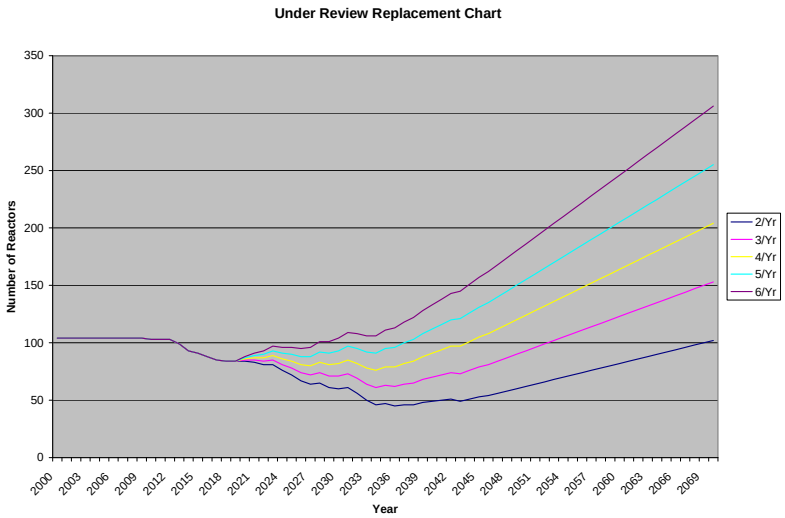


Figure F-9. Number of reactors in operation for replacement schemes with varying delay times and with licensing extensions.

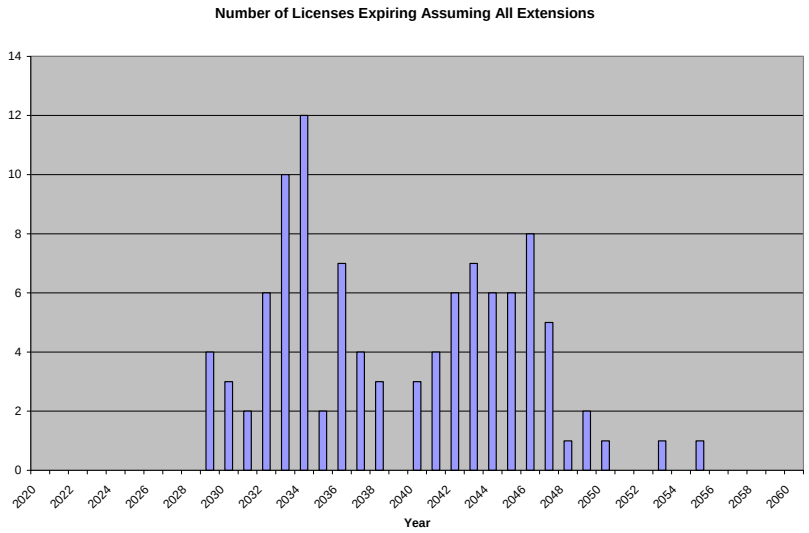


Figure F-10. Schedule of expiring reactor licenses with consideration of all reactors receiving extensions.

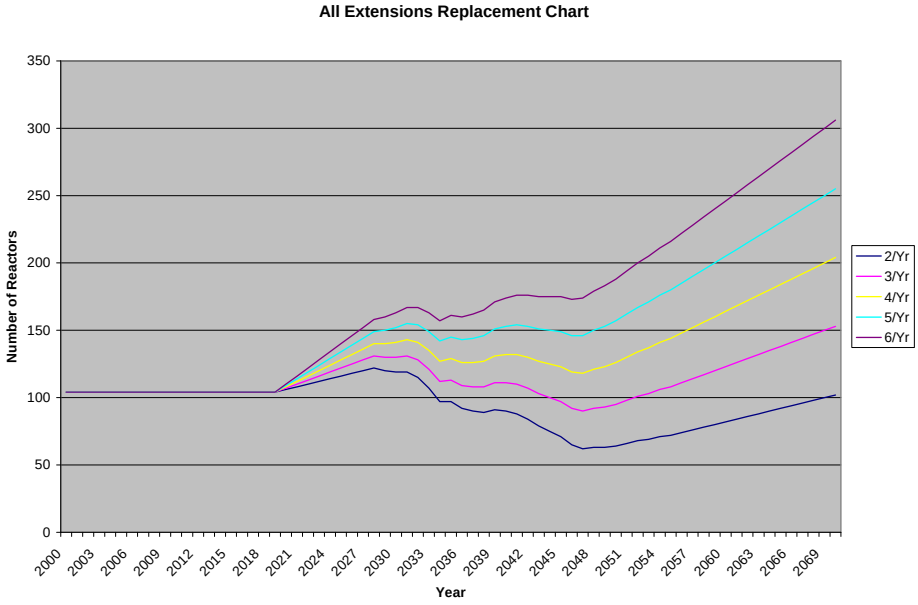


Figure F- 11. Extensions Replacement Chart

Appendix H: Gas Reactors

Purpose

The purpose of this analysis is to compare the actinide (americium, curium, and neptunium) production between light water reactors and gas-cooled reactors.

Method

In order to obtain spent fuel isotopics, the OrigenArp sequence of SCALE 5.1 was used. In OrigenArp are reactor specific libraries which were created by using the Origen-S depletion sequence to create parameterized cross section sets at an array of discrete enrichments, burn-ups, and moderator densities for each specific reactor type. These pre-loaded cross-section are utilized by OrigenArp (the suffix 'Arp' and acronym for Automatic Rapid Processing) as an interpolation mechanism, to create problem-dependent cross sections for the user specified depletion case. Because of this interpolation scheme, OrigenArp provides a mechanism of performing detailed depletion and decay analyses much more quickly than traditional transport calculations.

Using the OrigenArp sequence, depletion and decay cases were created for a typical LWR and a high temperature gas-cooled reactor. For the LWR scenario a Combustion Engineering 14 x 14 fuel lattice array, available in the OrigenArp library, was used. The input for each analysis was created using the express editor feature of OrigenArp. The input used for the LWR case is provided in Table 1:

Table H-1. OrigenArp Express Editor LWR Case Input Values

Enrichment (% U-235)	4
Burn-up (MWd/MTU)	55,000
Cycles	3
Libraries per Cycle	1
Cooling Time	5
Average Power (MW/MTU)	22.917
Moderator Density (g/cc)	0.7332

To make for an accurate comparison, the input values for the gas-reactor were set to mimic the input values of the LWR case. To accomplish this, the Advanced Gas-Cooled Reactor (AGR) OrigenArp library was used. However, the maximum burn-up which can be accepted by this library is 30,000 MWd/MTU. To calculate a higher burn-up the OrigenArp AGR source code was used. The source code used was a sas2h input file. The burn-up was changed to 55,000 MWd/MTU, and the gas-cooled reactor case was executed with sas2h. Although the inputs used for each reactor were utilizing different modules, the output accomplished the same desired end. The equivalent sas2h input for the AGR is as follows:

Table H-2. SAS2H Gas-Cooled Reactor Input Mimicing OrigenArp Input Fields

Enrichment (% U-235)	3.5
Burn-up (MWd/MTU)	55,000
Cycles	21

Libraries per Cycle	1
Average Power (MW/MTU)	22.917
Moderator Density (g/cc)	0.7332

For each reactor input case one metric ton of fuel was used. In OrigenArp this was explicitly set, and in sas2h the default for a unit cell is a normalization to one metric ton of fuel. The number of cycles differed between the two inputs, however, this field is arbitrary, and is just a means of expressing time steps.

In each case the irradiation time of the fuel was set for 2400 days. This fuel load was arbitrary in that the average power of each reactor could be set higher to produce a shorter fuel load time to achieve the desired burn-up of 55,000 MWd/MTU. The irradiated fuel was then modeled to decay for a period of 1500 years following removal from the core. The SCALE 5.1 module OPUS was used as the data storage and processing utility. Opus was called to collect mass (grams) and power (watts) data for the following nuclides of each reactor case: Am-241, Am-242m, Am-243, Cm-242, Cm-243, Cm-244, Cm-245, Cm-246, Cm-247, Cm-248, Cm-250, Np-235, Np-236, and Np-237. The selection of these nuclides was based on their long half-lives. The long half-lives of these nuclides accounts for a large amount of the long-term strain placed upon a repository.

Discussion of Results

In each case the output data was collected from opus and post-processed using Excel. Figure 1 shows the mass of actinides produced by each reactor scenario.

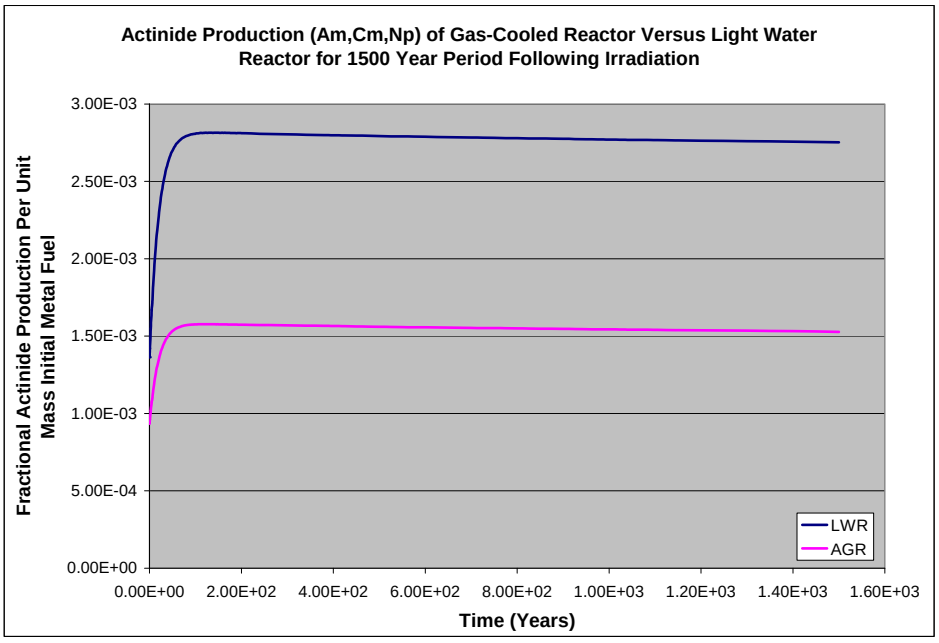


Figure H-1. Plot of reactor scenarios of CE 14 x 14 LWR and Advanced Gas-Cooled Reactor actinide mass produced per unit mass fuel as a function of decay time.

This plot exhibits an approximate factor of 1.8 increase in the actinide mass produced in the LWR over the gas-cooled reactor. As an example, on the lifetime scale of 100 years, the mass of actinides (for every 1000 kg of fuel) is: 2.81 kg for the LWR and 1.58 kg for the AGR. This is a decrease in actinide production of about 44%.

For the power analysis, integral decay heats were calculated for each of the reactor scenarios. This is a representation of the "capacity" displaced upon a repository by the storing of each type of reactor's spent fuel. Figure 2 shows a plot of this analysis.

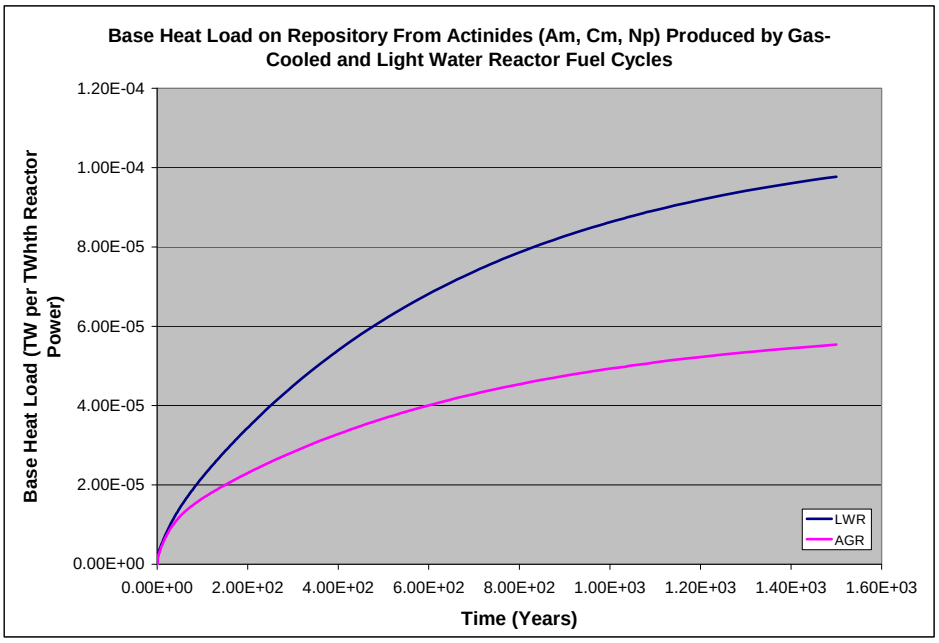


Figure H-2. Plot of the base heat load (represented as actinide power generated in terawatts per reactor power in terawatt-hours-thermal) as a function of spent fuel decay time.

As can be seen from Figure 2, the results of the repository heat load analysis show the same trend as the actinide mass analysis. Over the course of 1500 years of storage in a repository the LWR case exhibits a power production of 9.77×10^{-5} TW/TWhth, and the AGR case exhibits a power production of 5.53×10^{-5} TW/TWhth. This is a decrease in actinide power transferred upon a repository of 44%, as was the case with mass. The normalization of the actinide power produced is based upon a reactor power production of 1.32 TWth. Since both reactor scenarios were set to equal burn-ups, this analysis is independent of differences in reactor efficiencies. It is for this reason that reactor power is expressed in thermal units as opposed to electric units.

Future Work

Further work needs to be done to examine the impact of enrichment upon this analysis of the comparison of actinide production between LWRs and HTGRs. For this analysis the LWR was set with an enrichment of 4 % wt percent U-235, and the AGR with an enrichment of 3.5% U-235. In theory, if the enrichment of the AGR was increased to equal that of the LWR, the actinide production would decrease due to the decreasing in the amount of U-238 in the reactor (less U-238 neutron absorption would decrease the production of heavier nuclides). Also, further work can be done with gas reactors to incorporate them into the DANESS fuel cycle scenario models previously accomplished through this project. Through this work, gas-reactors could be compared with LWRs, FRs and MOX-fueled fuel cycle scenarios at a variety of levels.

Conclusions

The purpose of this analysis was to compare and contrast the actinide production between LWRs and HTGRs. Using the specific cases of a CE 14 x 14 PWR and the British design AGR the code package SCALE 5.1 was used to calculate the actinide production of each of these reactor categories. The results of this analysis indicate that gas-cooled reactors produce less actinides per unit mass of loaded fuel than do LWRs by an approximate factor of 0.55. Future work needs to be completed to examine the effect of differences of enrichment to determine it's impact on the results of actinide production.

References

1) SCALE Version 5.1. Code Package. Oak Ridge National Laboratory. November 2006.
2) SCALE Version 5.1. Manual. Oak Ridge National Laboratory. November 2006.

Appendix I: Thorium Fuel Cycle

Introduction

With the increase in worldwide awareness of and dependence upon nuclear power, which has been heightening for the last few decades, there has been an insurgence of research activity to create an optimum nuclear environment. This optimum nuclear environment includes, but is not limited to: extending fuel cycles, achieving higher burn-ups, creating less toxic fuel forms, decreasing proliferation concerns, and burning of weapons and civilian grade plutonium. Incorporation of thorium into the nuclear fuel cycle may be a way to address some of these issues. The following will discuss some of the potential advantages and disadvantages of the utilization of thorium fuel, ending with a discussion of a molten salt thorium system.

Potential Advantages [1]

1) Abundancy. Thorium is almost 4 times more abundant in nature than uranium. Most of the production of thorium is presently attributed to the mining of rare earth elements, namely uranium. Monzanite sand, usually found in beaches, contains a large fraction of thorium. This has been the source of thorium for India's nuclear power operations.

2) Proliferation Resistance. ^{232}U is formed in thorium based fuels. Some of the decay daughters of ^{232}U, such as ^{212}Bi and ^{208}Tl, emit very energetic gammas. This would help deter proliferation activities due to the radioactivity of the spent fuel. It should be noted though that excess amounts of ^{233}U would produce proliferation concerns in the same manner as ^{239}Pu.

3) Plutonium Destruction. The addition of plutonium to thorium based fuels is more attractive than the burning of plutonium with uranium based fuels. Uranium based fuels produce plutonium through the conversion of fertile ^{238}U to fissile ^{239}Pu. In a thorium system fissile ^{233}U would be produced from fertile ^{232}Th, and in this system almost no plutonium would be produced. This would lead to higher plutonium destruction rates in a thorium fueled system.

4) Nuclear Properties. $^{232}Th^{232}$ when compared to U^{238} as the primary fertile materials in their respective fuel cycles, has a larger capture cross section (7.4 barns to 2.7 barns, respectively). This would lead to a higher fissile conversion rate for ^{232}Th to ^{233}U than for ^{238}U to ^{239}Pu. Also, ^{233}U can achieve breeding in almost any neutron spectrum, including thermal due a thermal utilization factor greater than 2 at most neutron energies.

Potential Disadvantages [1]

1) Natural Composition. For the most part thorium is found in nature as ThO_2. Although this fuel form is favorable for fuel fabrication, thorium dioxide is somewhat chemically inert. This would lead to more difficult fuel fabrication activities.

2) Protactinium Production. For thorium based fuels, the production of protactinium is an issue. When ^{232}Th absorbs a neutron it is converted to ^{233}Th which decays with ~22 minute half life to ^{233}Pa. ^{233}Pa has a relatively long half-life of 27 days. This creates challenges due to: 1) neutron capture of ^{233}Pa, formation of ^{234}Pa, and subsequent decay of ^{234}U. ^{234}U is not fissile, but could capture to ^{235}U. Ultimately this sequence of capture would lead to fission, but would do so with detriment to neutron

economy 2) longer spent fuel cooling time than with an associated uranium system due to the decay of the intermediates. ^{239}Np decays to ^{239}Pu with a half-life of only 2.4 days, this is significantly quicker than ^{233}Pa decay.
3) ^{232}U. Even though the presence of ^{232}U in spent thorium based fuel is beneficial from a proliferation standpoint, it presents problems in other areas such as reprocessing. For reprocessing to be conducted it would have to take place in either a remote setting or a heavily shielded cell. In either case, the cost of such activities would be high.

4) Experience. One key disadvantage of the thorium fuel cycle and thorium based fuels is the extent with which they have been used to date. In the United States, very limited commercial activities have been conducted using thorium fuels. The experience using these types of fuels is miniscule when compared to that with uranium fuels. Because of this, thorium fuels would also have higher costs due to the necessity of implementing new types of fuel fabrication, and waste disposal activities.

Thorium Molten Fluoride System

One avenue for utilizing a thorium fueled system is a molten salt reactor with a fluoride based fuel salt. Fortunately, there has been experience with this type of reactor in the United States. In the 1960s the MSRE (Molten Salt Reactor Experiment) operated for over 4 years. It has been shown that this type of reactor will operate, and some key advantages of molten salt reactors were displayed through its operation. A molten salt reactor may be able to address some of the issues which are challenging to thorium based fuel system. A discussion of some of these topics follows.

One key advantage of molten salt reactors, theoretically, is their ability to manipulate it's core environment online, which cannot be done in the typical solid fueled reactors being used today for commercial purposes. This online processing would allow for: gaseous extraction inside the core to remove neutron robbing gaseous poisons for the core, namely ^{135}Xe, monitoring and processing of the fuel inside the core, and continual insertion of fresh fuel. Through these processes the fuel inside the core would be extracted, and processed to remove fission products, transuranics, and minor actinides. The fission products would be removed from the fuel, and the transuranics and minor actinides would be reinserted back into the core system for incineration. Online processing of the fuel would also allow for the removal of ^{233}Pa from the fuel, to allow for the subsequent decay to ^{233}U. This would be essential to maintain an adequate neutron economy inside the reactor core. The online processing capabilities of a molten salt system would alleviate the difficulties in fuel reprocessing and refabrication because a remote system would already be in place.

Since there exists no natural fissile isotope of thorium, a thorium based molten salt system would have to, initially, be "charged" with some fissile species. This is beneficial in the area of plutonium destruction, as the system could be loaded with a ThO_2-PuO_2 fuel.

As was mentioned previously, an advantage of a thorium fueled reactor is the decrease in transuranics and minor actinide production. A molten salt thorium system takes advantage of this concept the most be in situ processing and burning of these actinides. As the minor actinides are major contributors to the long term heat load for spent fuel storage, minimizing the production of these would greatly reduce the time necessary to reduce the activity of wastes to safe or background levels.

Molten Salt Fuel Cycle Comparisons

To analyze the amount of minor actinides produced in the various types of fuel cycles, a molten salt reactor was constructed using SCALE5.1, and loaded with four different fissile and fertile species. The fuel salt composition of this reactor is based on the MSRE fuel salt compositon of:

71.6 mol% ^{7}LiF
16.0 mol% BeF_2
12.0 mol% $^{232}ThF_4$
0.40 mol% $^{233}UF_4$

For the reactor power, 100 MWt was used, along with a burn length of 1000 days. The fuel salt was modeled as an 8 cm pin surrounded by 0.5 cm thick layer of graphite cladding. The associated secondary salt of the fuel was used as the coolant material, with a pin pitch of 15 cm. Four different cases were executed with the following fissile and fertile compositions: ^{235}U & ^{238}U, ^{239}Pu & ^{238}U, ^{239}Pu & ^{232}Th, and ^{233}U and ^{232}Th. In each case the reactor characteristics were held constant and the fertile and fissile molar fractions in the salt remained unchanged from above; in each case the fertile and fissile species were substituted into the above compositions. Upon SCALE modeling of each cycle, the isotopes of Americium, Curium, and Neptunium were isolated, and modeled to obtain heat load contributions for each cycle. The results are as follows:

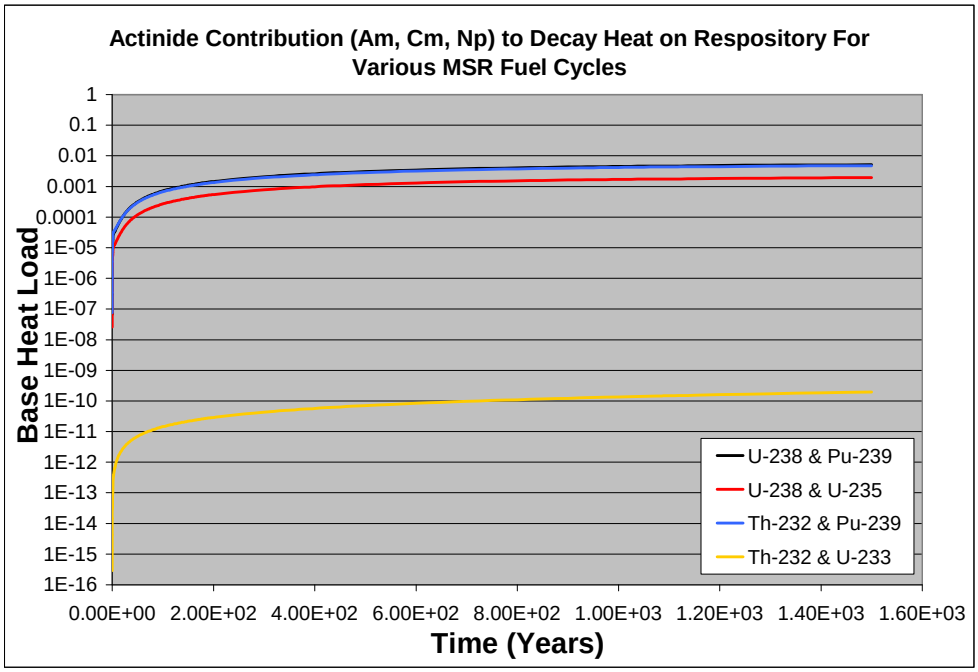


Figure I-1. Actinide Contribution (Am, Cm, Np) to Decay Heat on Repository For Various MSR Fuel Cycles. Base Heat Load given in units of: MWd energy produced by given nuclides per MWd energy produced by reactor.

As can be seen from the graph above the ^{232}Th & ^{233}U fuel cycle produces ~10,000,000 times less heat load (for a duration of 1500 years) than the other three studied. At 1500 years the integral heat load for each fuel cycle is as follows: 5.14 x 10^{-3} (^{238}U, ^{239}Pu), 4.76 x 10^{-3} (^{232}Th, ^{239}Pu), 1.94 x 10^{-3} (^{238}U, ^{235}U), and 1.96 x10^{-10} (^{232}Th, ^{233}U). The units for the previous given data are MWd actinide decay heat per MWd reactor power.

Some other nuclides are of concern to repository heat load for a thorium fuel cycle, such as: ^{231}Pa, ^{229}Th, and ^{230}U. The heat load contributions for these nuclides are given below.

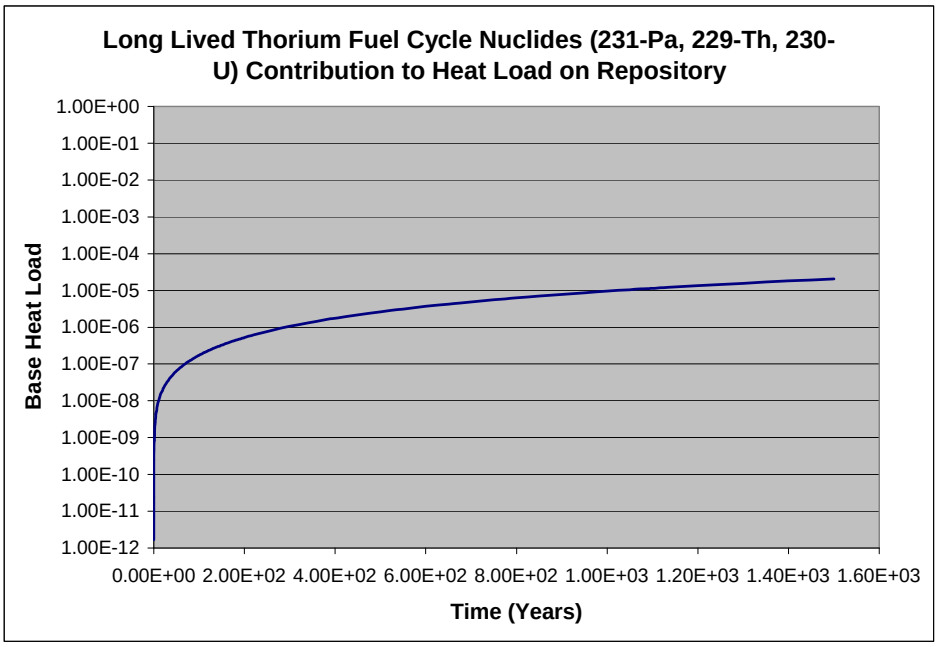


Figure I-2. ^{231}Pa, ^{229}Th, and ^{230}U Heat Load Contributions to Repository. These heat loads are for the ^{232}Th & ^{233}U fuel cycle. Base Heat Load given in units of: MWd energy produced by given nuclides per MWd energy produced by reactor.

At 1500 years, the heat load produced from these isotopes is 2.06 x 10^{-5}, which is substantially larger (~100,000 x) than the heat loads from the minor actinides (Am, Cm, Np) for the molten salt ^{232}Th and ^{233}U system. However, these heat load contributions are still ~90 times less than the minor actinide contributions of the ^{235}U, ^{238}U fuel cycle utilized in the current reactor park.

Future Work

Future studies need to be conducted to refine the SCALE molten salt model to define some of the necessary systems. Necessary changes in the model would account for gaseous Xe extraction and sequestering of ^{233}Pa. Also, future work needs to be conducted to compare the minor actinide

production of the above molten salt case with other reactor types (e.g. fast reactors, light water reactors, and possibly some Gen IV concepts).

References

1) International Atomic Energy Agency. "*Thorium fuel cycle—Potential benefits and challenges*". IAEA-TECDOC-1450. May 2005.
2) Kirk Sorensen, personal communications, December 2007.

Appendix J: Flow Charts for Advanced Fuel Cycle Management Tool

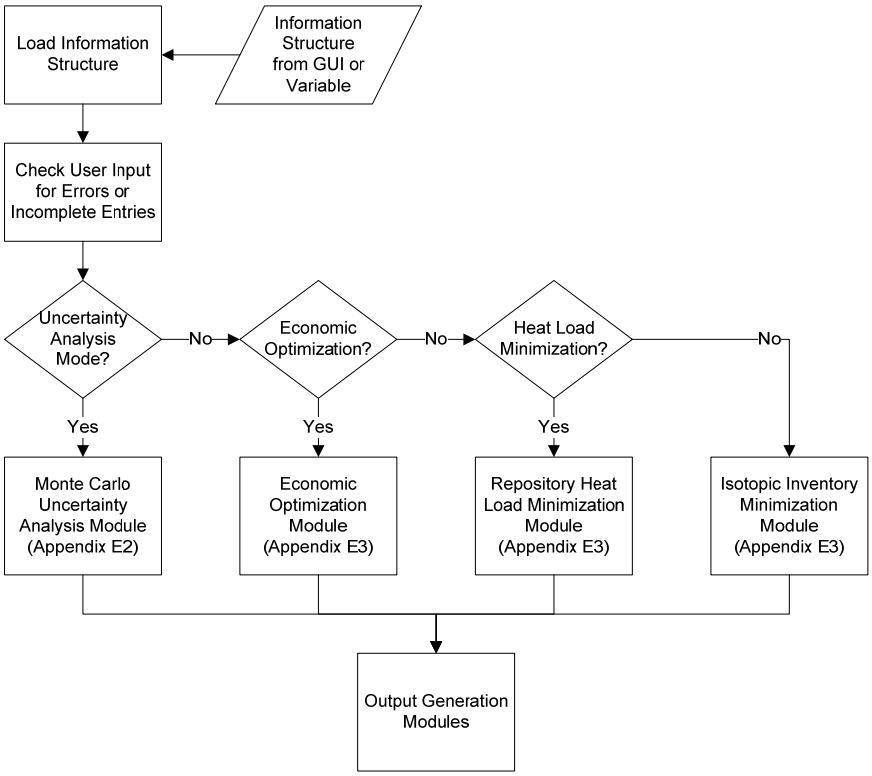


Figure J1: Overall Program Flow Chart

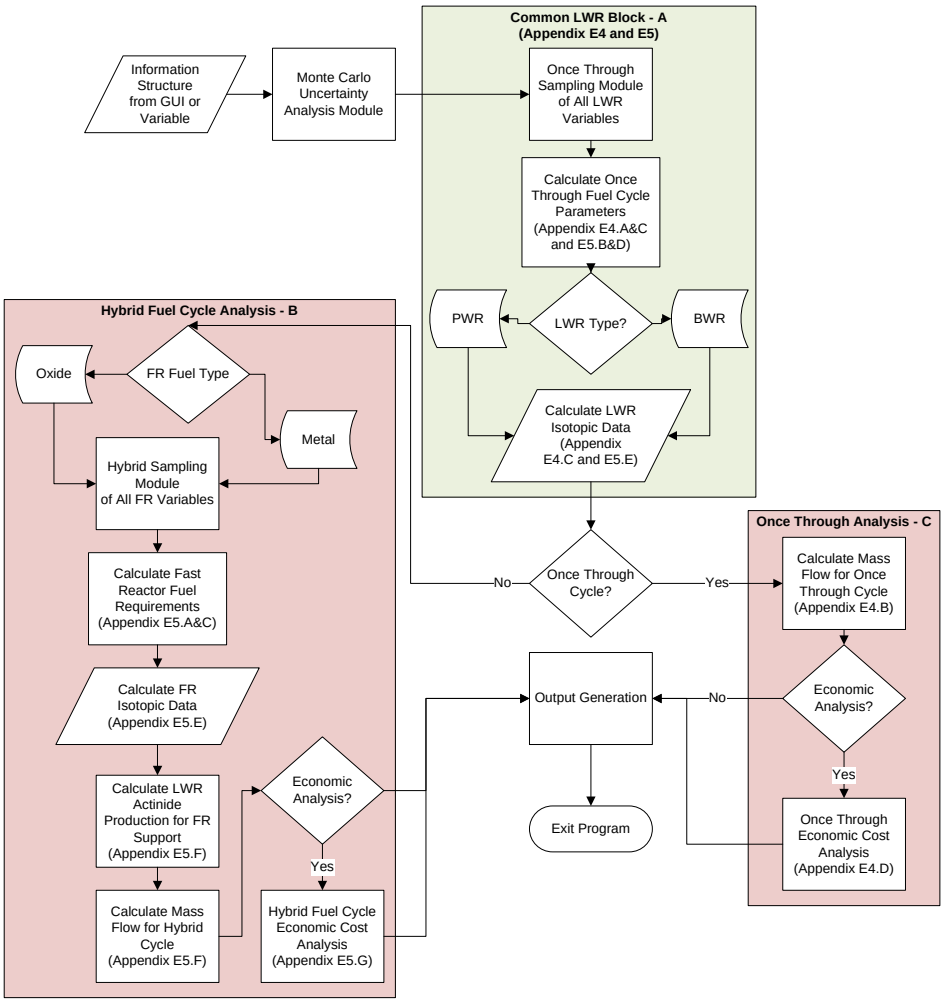


Figure J2: Uncertainty Analysis Module

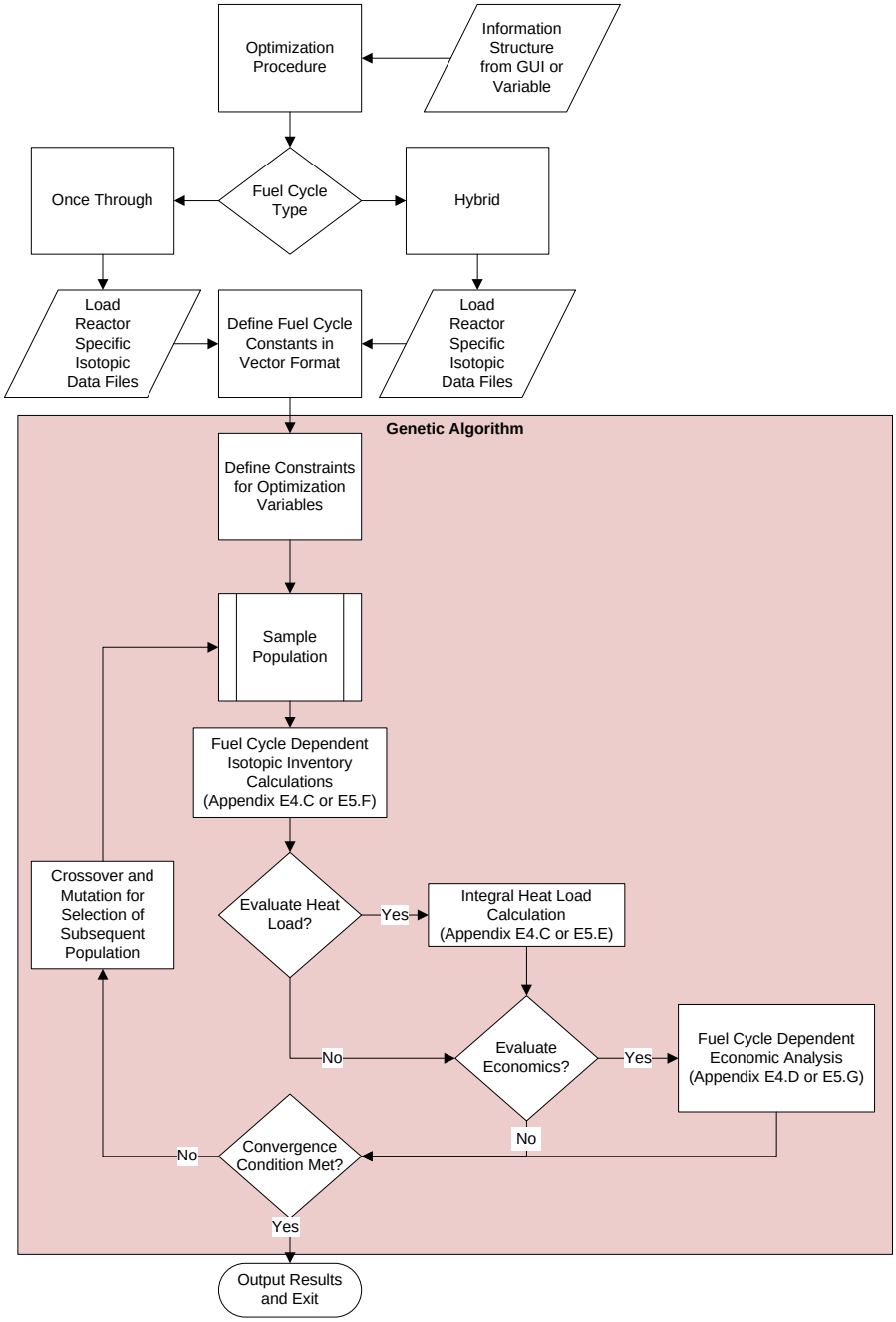


Figure J3: Optimization Procedure

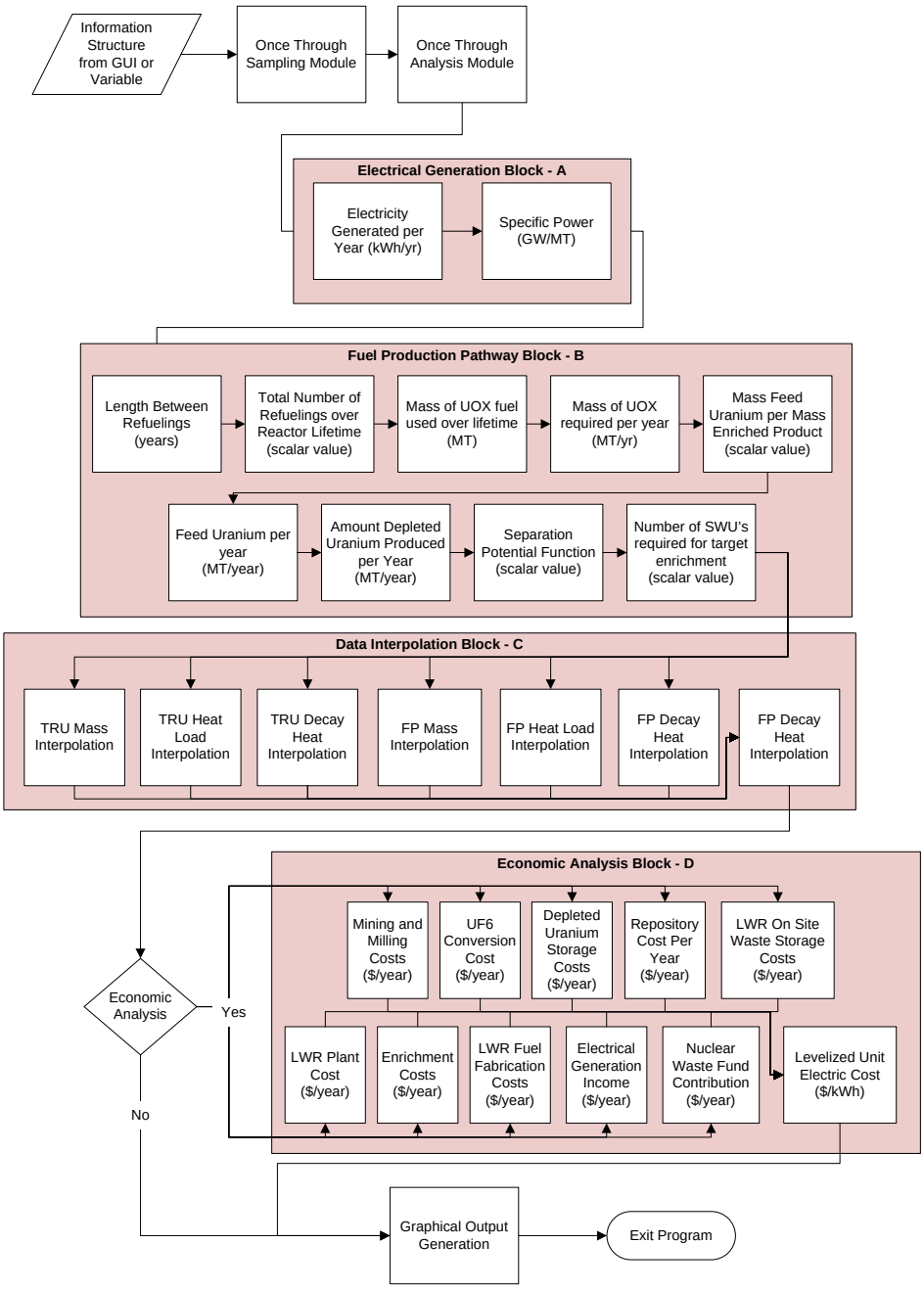


Figure J4: Once Through Fuel Cycle Flow Chart

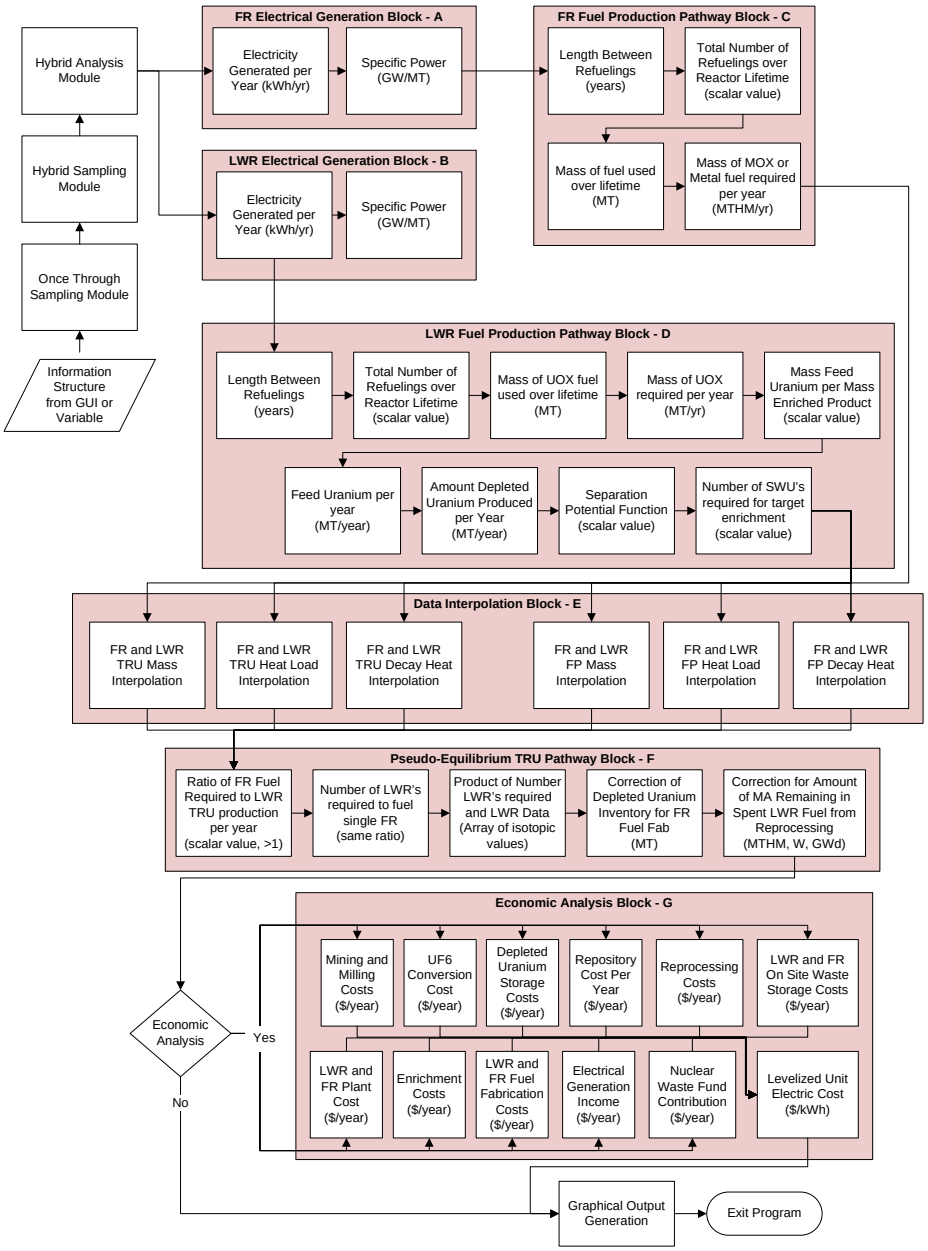


Figure J5: Hybrid Fuel Cycle Fuel Cycle Flow Chart

Appendix K: Presentations and Papers

Proceedings of ICAPP 2007
Nice, France, May 13-18, 2007
Paper 7288

Summary Presented at the Nov. 2005 Meeting of the American Nuclear Society

Methodology for Uncertainty Analysis of Advanced Fuel Cycles and Preliminary Results

L.F.Miller[1], Luc Van Den Durpel[2], A.M. Yacout[2], F.R. Mynatt[1], G. Sweder[1], T. Anderson[1], and Robert Hill[2]

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Advanced fuel cycles offer the opportunity of very significantly reducing the requirement for geologic repository space. In particular, if fission products, fissile material, actinides, and fertile material are separated, the need for repository space is about two percent of that required for the once-through cycle. Since implementation of advanced fuel cycles necessitates the construction of relatively expensive recycling facilities, and eventually of hard-spectrum reactors, it is important that uncertainties associated with deployment of these fuel cycles be understood and minimized. This paper describes the methodology for evaluating these uncertainties for advanced fuel cycles, and some preliminary results are presented.

The overall procedure for assessing uncertainties in system models with many unknown parameters involves the following general process:

1) identification of all parameters that may contribute to the uncertainties in results,
2) performance of sensitivity analyses on these parameters,
3) performance of scoping studies on all parameters that are identified to contribute to uncertainties, as determined by the sensitivity analyses,
4) conduct expert elicitations on parameters for which uncertainty distributions are not well defined,
5) assign uncertainty distributions to all parameters to be included in the uncertainty analyses,
6) conduct uncertainty analyses for all scenarios of interest,
7) refine estimates of uncertainties that contribute most to uncertainties in results, and
8) repeat steps 4 through 7 as required.

Studies for evaluating uncertainties in time for implementation of selected parameters in three fuel cycles are reported, and assessments of more detailed systems with additional parameters are in progress. Assessments that are reported include the following:

1) time to implement high burnup fuel in the once-through fuel cycle,
2) time to implement MOX recycle in LWRs, and
3) time to implement hard-spectrum reactors.

Output variables included in these assessments are the following:

1) mass of Plutonium in the repository at year 2100 per GW-d of energy produced,
2) integral heat produced during 1200 years after closure of the repository,

3) cost of disposing of spent fuel

Parameters selected for inclusion in an uncertainty analysis are based on discussions among developers of the DANESS code at Argonne National Laboratory and personnel at The University of Tennessee who are performing the uncertainty analyses with the DANESS code. Sensitivity analyses and screening calculations are performed on each of these parameters to determine if the parameter under consideration contributes to the uncertainty of variables included in the assessment. Where necessary, the uncertainty associated with variables selected for the assessment will be determined by structured expert judgment,[1,2,3] which has the following features:[4]

1) a group is selected,
2) experts are elicited individually within their domain of expertise,
3) experts also assess variables within their field whose true values are known,
4) experts are treated as statistical hypotheses and are scored,
5) scores are combined to form weights, and
6) the likelihood and information scores are used to obtain uncertainty distributions.

The DANESS (Dynamic Analysis of Nuclear Energy System Strategies)[5] code is developed by Argonne National Laboratory for the purpose of evaluating essentially all practically feasible fuel cycles, and it has the capability of performing sensitivity and uncertainty analyses. It is implemented in the IThink-software[6] and output is facilitated via MS-Excel templates. The fuel cycle mass-flow model incorporates 21 fuel cycle steps, and up to 10 different fuel types may be simulated in parallel. In addition each fuel type may have different characteristics and follow separate paths through the fuel cycle. Cross-flow of fissile material between these fuel types is possible and allocation of fuel to each fuel type may be a function of time. DANESS simulates energy systems ranging from a one-reactor level to a worldwide nuclear reactor park. Nuclear data are precalculated for specific levels of fuel utilization and fuel types, and they are utilized through data bases available to DANESS.

The objectives of this research are to evaluate a variety of issues that influence the sustainability of power generated by nuclear energy. Some of these are as follows:

1) design and analyze advanced fuel cycles for LWRs, including BWRs,
2) determine the effect of nuclear data uncertainties on fuel cycle evaluations and to determine the feedback of nuclear data needs,
3) identify and assess the repository benefits of advanced fuel cycles,
4) determine the effect of uncertainties on repository benefit assessments,
5) conduct dynamic fuel cycle scenario studies to develop an understanding of the issues in the transition from thermal reactor to a mixed thermal/fast reactor fleet, or a fleet in which the majority of reactors are fast and/or accelerator-driven,
6) optimize the use of key resources, e.g., repository capacity and uranium ore, in the long term for advanced fuel cycles, and
7) evaluate the optimal use of fast reactors and accelerator driven systems.

Results that include distributions for key parameters permit the quality of results to be evaluated and understood, and the lack of knowledge that contributes to the uncertainty of results can be identified.

This permits one to understand where improvements in knowledge are needed and where additional resources should be invested.

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1) Roger Cooke, *Experts in Uncertainty*, Oxford University press, New York (1991)
2) R. Budnitz, G. Apostolakis, D. Boore, L. Cluff, K. Coppersmith, C. Cornell and P. Morris, Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and Use of Experts, Technical Report NUREG/CR-6372; UCRL-ID-122160 Vol. 1. Lawrence Livermore National Laboratory (1997)
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6) IThink-software by high Performance Systems Inc., http://www.hps-inc.com

Proceedings of ICAPP 2007
Nice, France, May 13-18, 2007
Paper 7288

Summary Presented at the June 2006 Meeting of the American Nuclear Society

The Influence of Fuel Cycle and Spent Fuel Characteristics on Repository Heat Loads

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INTRODUCTION

The Department of Energy (DOE) currently supports the Advanced Fuel Cycle Initiative (AFCI) in order to facilitate a sustainable expansion of the use of nuclear power. Implementation of advanced fuel cycles has the potential of reducing requirements by as much as a factor fifty. Thus, it is imperative that optimal choices relative to type of reactors and to the manner in which they are operated enhance the feasibility of achieving the goal of sustainable nuclear power. A key parameter relative to this goal is the repository space required per unit of energy production. In order to investigate this issue, the decay heat generated by spent fuel, or recycled fuel, relative to the heat generated is calculated as a function of time between removal from the reactor and disposal, fuel burnup, and fuel composition. Results for this assessment are obtained from point depletion calculations and from simulations of alternative fuel cycles.

METHODOLOGY

Point depletions are obtained form several modules in SCALE and ORIGEN and fuel cycle simulations are accomplished using the DANESS code. Data processing is accomplished with a code written for MATLAB. This code reads isotope-specific, time-dependent, decay heat results from ORIGEN-ARP and integrates these results to obtain energy deposited in the repository relative to the fuel burnup. Selected isotopes are removed to evaluate the impact of reprocessing choices on repository requirements. The fuel cycles investigated for this assessment include: the current LWR, high burn-up fuels, MOX one tier recycle, and fast spectrum fuel one tier recycle. Since ORIGEN only uses a single group for performing depletion calculations, it is important that an accurate representative spectrum is obtained. This is accomplished using SCALE 5 with input parameters developed for several specific reactor types.

In order to accurately model the various fuel cycles of interest using the ORIGEN-ARP program, the current ORIGEN-ARP libraries are updated with new ARP cross-section libraries. These new ARP libraries are generated using SAS2/ORIGEN-S and are developed for an upper burn-up limit of 100GWd/MTU, whereas the previous upper limit was 60GWd/MTU. New fast reactor libraries are developed for ORIGEN-ARP so that analyses can be accomplished for advanced fuel cycles that include fast reactors.

RESULTS

The impact of removing actinides from spent fuel is illustrated in Figure 1. Results in this plot correspond to the energy deposited in the repository during 1200 years of residence as a function of cooling time before emplacement in the repository. These results show that disposal of actinides in fast reactors, rather than in repository, could provide a factor of over fifty savings in repository requirements

if cooling times of one hundred years are employed, and notably more if longer cooling times are considered. Note that only about a factor of five less repository space is required if interim storage is provided for about one hundred years, and actinides are not removed. Results from simulations indicate that only about a factor of two reduction of repository space requirement is obtained for high burnup fuels and that the benefit obtained from removing isotopes that contribute to the long-term heat load is much more effective for diminishing repository requirements than is utilization of high burnup or MOX fuels.

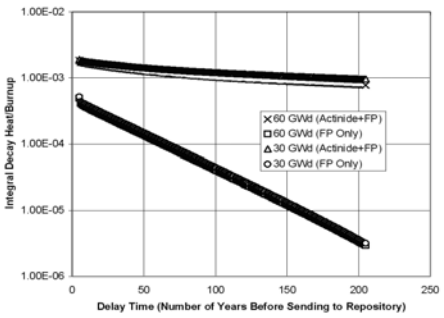

Fig. 1. Impact of removal of actinides from spent fuel for 30 GWd and 60 GWd burnup

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1. I.C. Gould, O.W. Hermann, R.M. Westfall, ORIGEN-S: SCALE System Module to Calculate Fuel Depletion, Actinide Transmutation, Fission Product Buildup and Decay, and Associated Radiation Source Terms, version 5, April 2005.

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Nice, France, May 13-18, 2007
Paper 7288

Summary Presented at the June 2007 Meeting of the American Nuclear Society

Uncertainty Analysis Methods for Equilibrium Fuel Cycles

L.F.Miller[1], J.Preston[1], Brian Thomas[2], J.McConn[1], J. Hou[1], T.Anderson,[1]and M.Humberstone[1]
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[2]*SENES Oak Ridge, Inc. Oak Ride, TN*

Good estimates for inventories of Plutonium and minor actinides associated with advanced fuel cycles are essential for determination of requirements for repository space and for selection of fuel cycles. Likewise it is important to determine the quality of these estimates. The overall procedure for assessing uncertainties in system models with many unknown parameters involves the following general process:

1) identification of all parameters that may contribute to the uncertainties in results,
2) performance of sensitivity analyses on these parameters,
3) performance of scoping studies on all parameters that are identified to contribute to uncertainties, as determined by the sensitivity analyses,
4) conduct expert elicitations on parameters for which uncertainty distributions are not well defined,
5) assign uncertainty distributions to all parameters to be included in the uncertainty analyses,
6) conduct uncertainty analyses for all scenarios of interest,
7) refine estimates of uncertainties that contribute most to uncertainties in results, and
8) repeat steps 4 through 7 as required.

Sensitivity analyses and screening calculations are performed to determine if the parameter under consideration contributes to the uncertainty of variables included in the assessment. Where necessary, the uncertainty associated with variables selected for the assessment is determined by structured expert judgment.[1,2,3,4]

Fuel cycles of varying complexities are modeled and uncertainties of selected outcomes are determined by random sampling of model parameters. Material balances are written for each module of the fuel cycle model for all isotopes included in the mode (currently 63) and for total material flows. Examples of parameters that are assigned either fixed, uniform or triangular uncertainty distributions:

1) fraction of U-235 in the feed to the enrichment plant,
2) fraction of U-235 in the tails stream from the enrichment plant,
3) fraction of U-235 in the product stream from the enrichment plant,
4) isotopic composition of fresh fuel,,
5) isotopic composition in spent fuel as a function of burnup and reactor type,
6) burnup of spent fuel, and
7) thermal energy produced per year.

Results are obtained from a commercial simulation code, Analytica[5], and from a Matlab code written by one of the authors. Both give the same results; thus, it is concluded that the uncertainty methodology in the code developed in-house is correct. The choice to use Matlab software was made since code for data handling of output form SCALE[6] was previously written for calculating heat loads on repositories for arbitrarily specified spent fuel compositions. Variations in composition are obtained from the ORIGEN[7] in combination with the SAS2 module of SCALE. The energy demand and the burnup are

used to determine fuel demand. Burnup is modeled as a random variable and energy demand is a user specified value.

Example outcomes form the uncertainty analyses include the following:

1) isotopic composition of spent fuel,
2) heat load on the repository,
3) natural uranium feed, and
4) makeup feed.

Table 1 lists some of the parameters used in a Pu recycle model with specifications of uncertainty distributions and Figure 1 illustrates an example of results.

Table 1. Specifications for parameters in the Pu recycle model

Parameter	Dist.	Specs.
Fraction of U-235 in feed	F	7 m
Fraction of U-235 in tails	U	2 m,3 m
Fraction of U-235 in fuel	T	0.03,0.045,0.05
Fraction of Pu-249 in spent fuel	T	4 m,7 m,0.01
Relative efficacy of Pu239 to U235	T	0.7,0.8,0.9
Fraction of U-235 in depleted stream	U	2 m,3 m
Fraction of U-235 in U in recycle	U	0.01,0.02

The unit m denotes milli, F denotes fixes distribution, U denotes uniform distribution, and T denotes a triangular distribution.

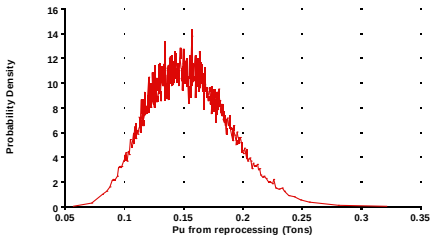

Figure 1. Probability density function of tons Pu generated per year per reactor.

Distributions of selected outcomes indicate that variations of a factor of two are typical for fuel cycles with parameter variations characteristics of current operational expectations for light water reactors and for fast reactors.

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2) R. Budnitz, G. Apostolakis, D. Boore, L. Cluff, K. Coppersmith, C. Cornell and P. Morris, Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and Use of Experts, Technical Report NUREG/CR-6372; UCRL-ID-122160 Vol. 1. Lawrence Livermore National Laboratory (1997)
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5) Lumina Decision Systems, www.lumina.com
6) SCALE
7) ORIGEN

Proceedings of ICAPP 2007
Nice, France, May 13-18, 2007
Paper 7288

Summary Presented at the Nov. 2006 Meeting of the American Nuclear Society

Utilization of Fast Reactors to Control Inventories of Plutonium and Minor Actinides in Advanced Fuel Cycles

L.F.Miller[1] J. Preston[1], T. Anderson[1], J. McComm[1], F.R. Mynatt[1], and Luc Van Den Durpel[2]
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INTRODUCTION

Inventories of Plutonium and minor actinides are important factors in determination of the risk associated with the use of nuclear energy. This includes the potential exceeding release limits from a repository and the potential for proliferation. The amount of these materials in any given fleet of reactors is determined in large part by the choice of fuel cycle and by the types of reactors selected for operation. Most of the US reactor fleet will need to be replaced within the next 30 years and additional reactors will need to be added if the contribution of power from nuclear energy is expanded. In order to minimize risk and to make judicious use of repository space, inventories all radionuclides will need to be effectively managed.

Modest reductions in requirements for repository space can be achieved through the use of high burnup fuels and of mixed oxides of Plutonium and uranium[1] (MOX). However, it is necessary to transmute actinides through the use of accelerators or fast reactors to achieve significant benefit from implementation of advanced fuel cycles. Use of hard-spectrum reactors to burn excess Plutonium and other actinides is technologically feasible and is most likely less costly than any other options for minimizing various risks.

RESULTS

Calculations for the inventories of several categories of radionuclides indicate that introduction of a modest fraction of fast reactors into the US reactor fleet is effective in stabilizing the growth of problematic radioisotopes. Results illustrated in Figure 1 are obtained from the DANESS (Dynamic Analysis of Nuclear Energy System Strategies)[2,3] code for a reactor park that consists of 30 % BWRs, 60 % PWR and 10 % fast burner reactors with a conversion ratio of 0.5 and with reactor attributes as listed in Table 1. Note that for this case the Plutonium inventory is stabilized in less than one hundred years. If a growth rate of 1.5 % is introduced and the reactor fleet is again comprised of 10 % fast reactors with a conversion ratio of 0.5, then the Plutonium inventory is not stabilized within one hundred years, as shown in Figure 2.

You may note from Figures 1 and 2 that the time of implementation of burner reactors over a thirty year period relative to the defined shutdown schedule has only a relatively small impact on Plutonium inventory. Results from calculations for the decay heat load on the repository per unit of energy generated[1] for the once-through and limited recycle for MOX fuel cycles will be compared with comparable results for fuel cycles that incorporate fast burner reactors with varying conversion ratios and fractions of hard-spectrum reactors in the fleet.

CONCLUSIONS

Inventories of Plutonium and minor actinides can be controlled through the use of recycle and hard-spectrum reactors, and significant reduction of repository space requirements can also be achieved.

Table 1. Fast Reactor Attributes

Reactor Attributes	
Pth [MWth]	3600
Pe [Mwe]	1450
Load Factor [%]	85
Coolant	Na
Fuel	FR-MOX
HM Inventory BOL [kg]	
(without blankets)	25690
Fuel Burn-up [GWd/tHM]	185
Power Split [%]	
Fuel Fractions [%]	
CR	
In-Pile time [d]	1553
No. of batches	6
TRU in top-up fuel [%]	57.6

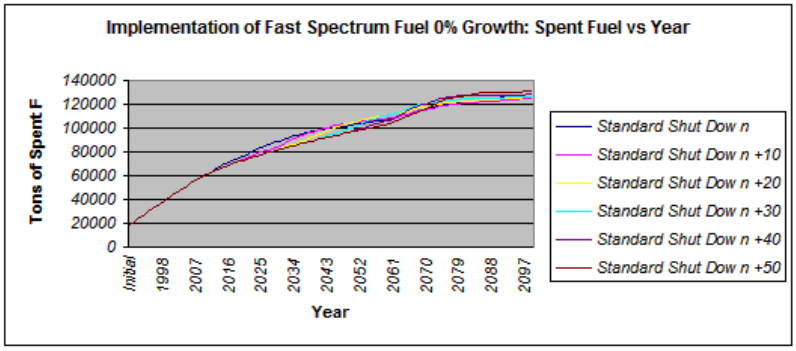

Figure 1. Spent fuel totals for time of implementation utilizing fast burner reactor with a conversion ration of 0.5 for the case of 0% growth rate.

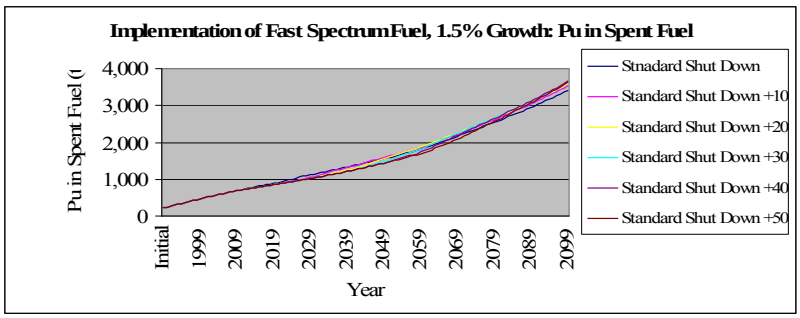

Figure 2. Spent fuel totals for time of implementation utilizing fast burner reactor with a conversion ration of 0.5 for the case of 1.5 % growth rate.

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Evaluation of Various Fuel Cycles to Control Inventories of Plutonium and Minor in Advanced Fuel Cycles

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Abstract – Inventories of Plutonium and minor actinides are important factors in determination of the risk associated with the use of nuclear energy. This includes the potential of exceeding release limits from a repository and the potential for proliferation. The amount of these materials in any given fleet of reactors is determined in large part by the choice of fuel cycle and by the types of reactors selected for operation. Most of the US reactor fleet will need to be replaced within the next 30 years and additional reactors will need to be added if the contribution of power from nuclear energy is expanded. In order to minimize risk and to make judicious use of repository space, inventories of all radionuclides will need to be effectively managed. Use of hard-spectrum reactors to burn excess Plutonium and other actinides is technologically feasible and is most likely less costly than any other options for minimizing various risks.

Calculations for the inventories of several categories of radionuclides indicate that introduction of a modest fraction of fast reactors into the US reactor fleet is effective in stabilizing the growth of problematic radioisotopes. Results are obtained from the DANESS (Dynamic Analysis of Nuclear Energy System Strategies)[1,2] Code and from the solution of algebraic equations that define steady state inventories. There are various different possible fuel cycle scenarios to utilize in the implementation of fast, thermal and intermediate spectrum reactors into the U.S. fleet. Results include various combinations of reactor types and fuel with varying times of implementations. Mass flows with uncertainties for equilibrium cycles will also be reported.

Time-dependent scenarios are modeled with the DANESS code, and algebraic equations for various fuel cycles are derived. Uncertainties are obtained using Monte Carlo simulations based on estimates of parameters in the models.

I. INTRODUCTION

Nuclear energy currently utilizes the once-through fuel cycle, and it produces about twenty percent of the electrical power produced in the US. However, expansion of power production with the current once-through fuel cycle is limited to about 1,000 reactors for 100 years. This is based on an estimate of about $15x10^8$ metric tons of natural uranium with production cost of less than $130/kg and on a burn up of about 50 GWd/ton. Thus, the current technology for utilization of nuclear energy with relatively inexpensive fuel is not considered to be sustainable. In order for nuclear energy to make a substantial contribution to the long-term energy supply for humans, sustainable nuclear fuel cycles must be implemented. It is also essential that advanced fuel recycling be implemented to eliminate excessive repository capacity.

Recycle of fissile material provides about a thirty percent gain in resource utilization, which is much less than the uncertainty in the availability of economically priced natural uranium. Thus, there is little economic or technical incentive for implementation of a fuel cycle that only recycles fissile material. This option does offer the opportunity for reducing the requirements for disposal of spent fuel and for reducing the short-term heat load, if fission products are handled separately, since the actinides contribute nearly the entire long-term heat load to the repository. Requirements for repository space are reduced by about a factor of two if this option is implemented.

Recycle of fissile and fertile material and most (about 95%) of the actinides, based on previously demonstrated reprocessing technology, would result in about a factor of three reduction in repository space. However, this approach, coupled with breeder reactors, could provide an essentially infinite supply of electrical energy. If essentially all of the actinides are recycled into advanced reactors or accelerator driven systems, a reduction factor for the required repository space of about 60 can be realized.

Integration of fast reactors (FRs) into the US fleet of offers some significant opportunities to diminish

requirements for radioactive waste repository space or to extend uranium ore resources. If fast reactors are operated as burners they can control inventories of problematic radioactive isotopes, such as Plutonium (Pu) and various minor actinides (MA). If they are operated with conversion ratios near or above unity, the fuel supply becomes essentially unlimited. Results reported herein include inventories of Pu and MA for several scenarios, uncertainties associated with an equilibrium fuel cycle, and heat load reductions with several reprocessing options.

II. METHODOLOGY

Calculations for the determination of Pu and MA inventories are accomplished by using the DANESS code that was recently developed by personnel from Argonne National Laboratory (ANL). The use of DANESS is focused on scenario-analysis of different development paths for nuclear energy systems from a governmental, utility or R&D perspective. It accomplishes this through analysis of development paths for nuclear energy. The impact of new developments in nuclear reactors and in fuel cycle operations may be analyzed from an integrated perspective. The impact on inventories in the fuel cycle, on costs of energy generation, waste production as well as the level of compliance with sustainability goals can also be analyzed.

III. DANESS NUCLEAR SYSTEMS MODEL

The DANESS code simulates nuclear fuel cycle systems form mining through final disposal. Some of its features include the following:

1) Timing of operations, i.e. including history of ordering, licensing, constructing, operating, and decommissioning of facilities,
2) Tracking primary mass flows but also secondary (waste) mass flows and Life-Cycle Inventory (LCI)-related flows and emissions,
3) Economics, including level cost cash-flow analysis, etc.,
4) Waste management impact analysis,
5) Intra-nuclear market penetration model included,
6) Nuclear energy policy decision-making,
7) 100 years 1 month time-step world simulation: <5 min on PC,
8) The simulation of nuclear energy demand driven development paths for a varying mix of reactors and fuel cycle options,
    i) up to ten nuclear reactor technologies can be modeled simultaneously
    ii) steady-state or transient conditions may be modeled,
    iii) simulation on reactor, utility, country, regional or even worldwide level are possible,
9) Detailed fuel cycle mass flow analysis from mining until final disposal of spent fuel or waste or up to ten different fuel types used in ten different reactors in varying degrees over time
    i) analysis on element level
    ii) cross-flows of materials between fuels and reactors,
10) Integrated process modeling of nuclear energy systems,
11) Analysis of governmental actions in guiding nuclear energy system development, and
12) Customization of the model for specific users or applications.

Figure 1 below shows the basic flow of the DANESS code.

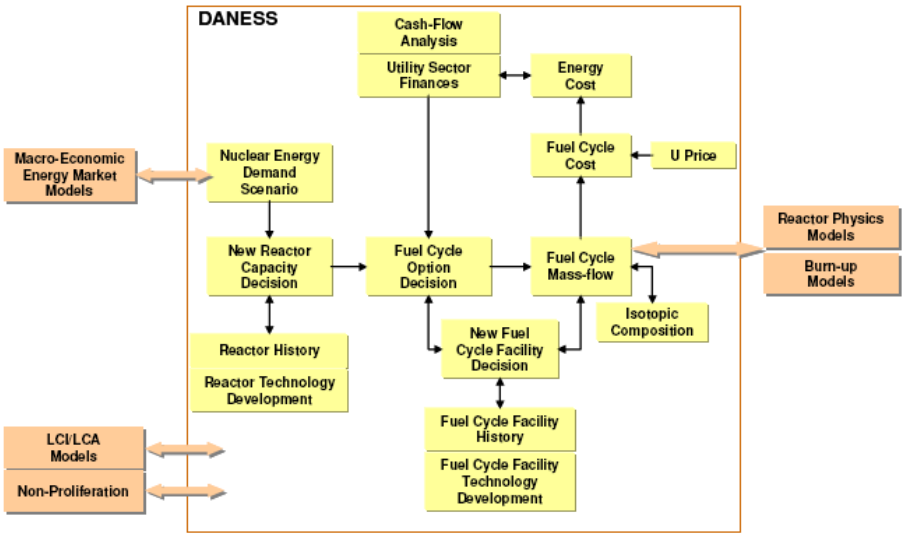

Figure 1: DANESS Flow Box Diagram

IV. EQUILIBRIUM FUEL CYCLE ANALYSES

Various parameters in DANESS have significantly different influence on results from simulations. In some cases, such as burnup, an input parameter may be input as continuously varying, but data used for obtaining results are pre-calculated based on discrete values. Thus, any value may be chosen as input, but the results are based on values associated with specific pre-calculated values. As a result, in the current version of DANESS, it is not feasible to perform uncertainty assessments by well-established Monte Carlo, or Latin Hypercube, sampling methods. Instead of using formal uncertainty analysis methods, scenarios are run for feasible fuel cycles, and the range of resultant values is analyzed to obtain uncertainties of selected outcomes.

Parameters associated with DANESS can be generally classified in the following categories: 1) physics, 2) economic, and 3) decision. If one limits consideration to an equilibrium fuel cycle model, it is straightforward to write material balances for various fuel cycle facilities and to solve algebraic equations for physics-related results of interest. In order to conduct uncertainty analyses for equilibrium fuel cycle models it is necessary to specify probability density functions for parameters associated with the model and to solve equations for outcomes of interest. Some results of particular interest are as follows:

1) Energy deposition in the repository during a 1,500 year residence based on spent fuel and for cases with some isotopes removed,
2) Energy deposition in the repository during a 1,500 year residence for selected isotopes,
3) The amount of spent fuel generated,
4) The amount of natural uranium feed required,
5) Plutonium in spent fuel,
6) Short-lived fission products, and
7) Long-lived fission products.

Results for these outcomes can be obtained for specific models, such as one which considers the following facilities:

1) enrichment,
2) fuel fabrication,
3) reactor, and
4) reprocessing.

Material balances are written for total uranium, U-235 and Pu-239, and are based on one year. The model parameters are the following:

1) fraction of U-235 in the feed to the enrichment plant,
2) fraction of U-235 in the tails stream from the enrichment plant,

3) fraction of U-235 product stream from the enrichment plant,
4) Pu isotopic composition in fresh fuel,
5) U isotopic composition in fresh fuel,
6) Pu isotopic composition in spent fuel,
7) U isotopic composition in spent fuel,
8) burnup of spent fuel, and
9) thermal energy produced per year.

Assumptions for the current model include the following:

1) no losses in any fuel cycle facility,
2) the enrichment of the product stream from the enrichment plant is equal to that required for new fuel,
3) natural uranium feed to the enrichment plant,
4) stockpiled plutonium for makeup feed to the fuel fabrication plant,
5) one third of irradiated fuel is replaced/recycled each year

Derivation of Equilibrium Fuel Cycle Equations with Plutonium Recycle

Algebraic equations that characterize material flows in an equilibrium fuel cycle are obtained by writing material balances for each facility of interest in the fuel cycle. Equation (1) is a total material balance for the enrichment and Eq. (2) is an isotopic balance for U235.

$$F_{NU} = T + P_e \tag{1}$$

$$\alpha_1 F_{NU} = \alpha_2 T + \alpha_3 P_e \tag{2}$$

α_1 = fraction of U-235 in the feed stream to the enrichment plant
α_2 = fraction of U-235 in the tails stream from the enrichment plant
α_3 = fraction of U-235 in the product stream from the enrichment plant
P_e =quantity of material in the enrichment stream
T = quantity of material in the tails stream
F_{NU} = quantity of material in the feed stream

These two can be manipulated to remove the mass of the tails stream to produce a relationship between the feed stream mass and the product stream mass:

$$F_{NU} = (\alpha_3 - \alpha_2)*P_e/(\alpha_1-\alpha_2) \tag{3}$$

The quantity of spent fuel (energy/burnup)

$$F_{sf} = \left(\frac{E}{BU}\right) \tag{4}$$

E = thermal energy released from fission in one year (GWd)
BU = burnup of fuel (GWd/T)
F_{sf} = quantity of spent fuel

The total quantity of fuel in reactor:

$$F_{nf} = 3* F_{sf} \tag{5}$$

Once the total mass of fuel in the reactor is known the mass of the ith isotope of Pu can be calculated

$$MPu_i = \Delta_i*F_{nf} - \Gamma_i*F_{sf} \tag{6}$$

where

Δ_i = fraction of isotope i in fresh fuel
Γ_i = fraction of isotope i in spent fuel

Similarly the necessary isotopic composition of the product stream is found from:

$$P_i = B_i*F_{nf} - K_i*F_{sf} \tag{7}$$

with the total mass as the sum of the isotopes. From here one can calculate the feed mass necessary to using Eq (3).

(2)

V. RESULTS

V.A. Time-Dependent Simulations with DANESS

Reactor fleets with time-dependent ratios of FRs and LWRs were simulated to determine inventories of plutonium and minor actinides in the overall system and in reactor cores. Results from these simulations show that inventories of plutonium can be significantly reduced through the use of fast reactors with conversion ratios (CRs) of about 0.5. However, development of fuel cycles to control of inventories of minor MA will require results from additional reactor physics calculations. Simulations that modeled mixed fleets of LWRs and FRs were completed for the following:

1) fixed and varying numbers of LWRs with three different growth rates of FRs,
2) different initial amounts of Pu, and
3) varying times of introduction of FRs.

Results are presented as time-dependent plots, and values for Pu and MA inventories at the year of 2100 are presented in Table 1, which include combined LWR and FR growth cases. The first and second sets of values assume no growth in LWRs and a growth rate of three FRs per year based on the Pu in spent fuel to date and on no initial Pu available, respectively. The third set of numbers is based on growth rate of 8 LWR Rx/Yr for the reference case, and 6 LWRs and 2 FRs for the other corresponding cases.

Table 1. Absolute Inventories of Pu and MA at 2100

3 Rx/Yr	Pu	tHM		MA	tHM	
FR Growth Act Initial Pu	In-Pile	Out-Of-Pile	Total	In-Pile	Out-Of-Pile	Total
Reference(No Growth)	55	2864	2920	8	438	446
2020 Implementation	662	1196	1858	27	585	613
2030 Implementation	786	1104	1890	31	580	611
2040 Implementation	839	1079	1917	33	564	597
FR Growth, No Initial Pu						
Reference(No Growth)	55	2203	2258	8	337	346
2020 Implementation	428	1327	1754	20	429	449
2030 Implementation	475	1193	1668	21	430	451
2040 Implementation	579	1023	1602	25	432	597
8Rx/Yr (Ref 8 LWRs: others 6 LWRs and 2 FRs)						
LWR & FR Growth Actual Initial Pu						
Reference	442	9012	9454	68	1384	1451
2020 Implementation	1776	3861	5636	97	1306	1402
2030 Implementation	1546	3216	4762	85	1101	1186
2040 Implementation	1365	2613	3977	75	921	996

It can be noted from the results in Table 1 that introduction of FRs with a CR=0.5 notably reduces Pu inventory relative to production of the same amount of energy with LWRs. The MA inventory is not decreased since applicable reactor physics data are not available in the current version of the DANESS data base.

Figure 2 illustrates the growth in Pu inventory if the once-through fuel cycle continues through 2100. If Pu generated from LWRs is recycled to startup FRs then the out-of-pile Pu could be reduced to inventories in fuel cycle facilities utilized to reprocess and fabricate fuel; however, the version of DANESS used to perform the calculations reported in this paper reserves fuel for 15 years for each reactor it adds to the fleet. Thus, out-of-pile inventories are reduced only by about a factor of two for the cases evaluated.

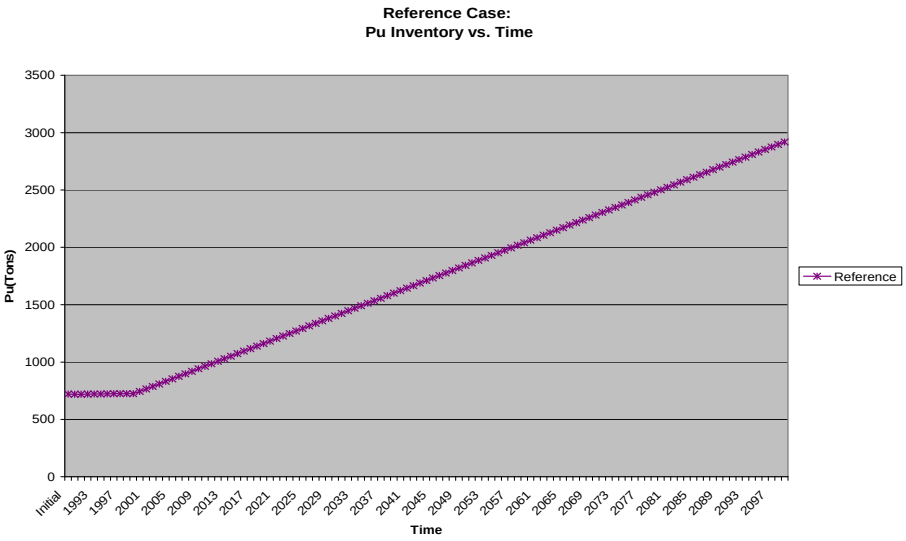

Figure 2. Growth of the Pu inventory based on 100 LWRs and the once-through fuel cycle.

An example of the number of reactors that DANESS includes in the simulated fleet with varying growth rates of FRs is shown in Figure 3, and the corresponding Pu inventory is shown in Figure 4. Note that the cited constraints limit the total number of reactors to about 70 and that about 900 tons of Pu would be required to start up these reactors, which is about one-half of the Pu generated over a 90 year period. If the fuel reserve constraint is removed, about 200 FRs could be fueled and could be continually operated if the CR=1.0. The inventory of minor actinides (MA) continues to increase in the simulation since they are not recycled in this simulation.

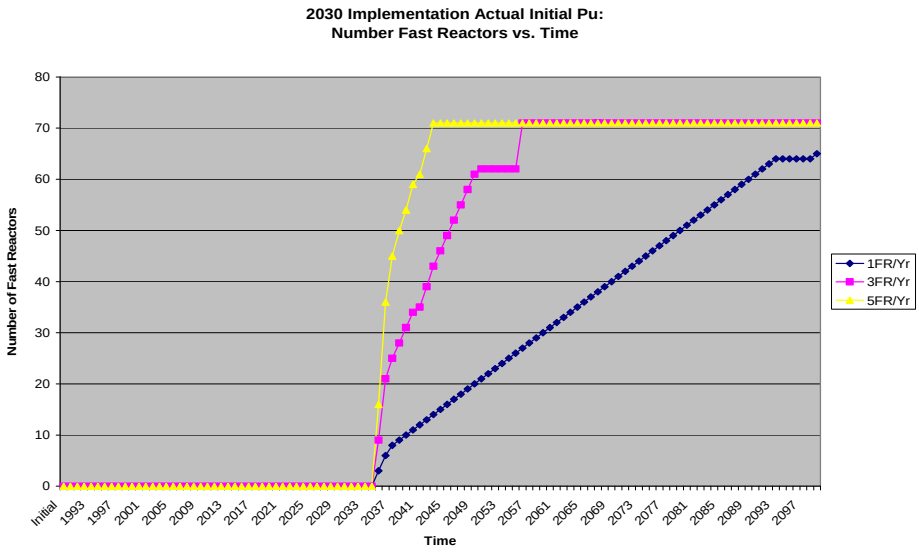

Figure 3. Number of FRs with CRs of 0.5 that DANESS includes in a simulated reactor fleet based on Pu from 100 LWR and a fuel reserve constraint of 15 years.

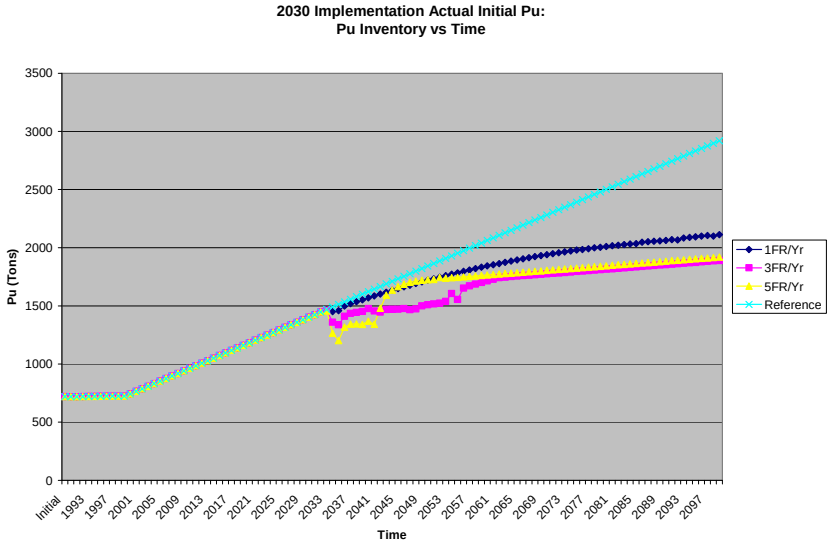

Figure 4. The Pu inventory that corresponds to the number of FRs with CRs of 0.5 that DANESS includes in a simulated reactor fleet based on Pu from 100 LWR and a fuel reserve constraint of 15 years.

V.B. Equilibrium Fuel Cycle Analyses

Equilibrium fuel cycles were analyzed using a Monte Carlo sampling method to propagate uncertainties through the model. Some of the results include the following:

burnup of spent fuel, composition of Pu and MA in spent fuel, requirements for natural uranium, mass flows for Pu and depleted uranium for blending, and the number of FRs to burn Pu and MA produced by LWRs. Parameters in the algebraic equations that model the equilibrium fuel cycles are specified by uniform and triangular distributions as listed in Table 2. An example result for the number of fast reactors required to utilize the Pu produced by 100 LWRs is shown in Figure 5. Note that if 100 LWRs generate about 20 tons Pu per year, then about 40 FRs with a CR of 0.5 would be required to consume this annual production of 20 tons per year. Plutonium in FRs is shown in Figure 6, and Pu generated in 100 LWRs by 2100 is shown in Figure 7.

Table 2. Values and distribution of values used in the fuel cycle model.

Description of Parameter	Nominal Value	Expected Range	Distribution
Fraction of U235 in feed to enrichment plant	0.007	none	fixed value
Fraction of U235 in tails from enrichment plant	0.002	0.002-0.003	uniform
Fraction of U235 in product from enrichment plant	0.04	0.03-0.05	uniform
Isotopic composition of Pu and U in fresh fuel	given	none	fixed value
Isotopic composition of Pu and U in spent fuel	varies	NA	function of burnup
Burnup of spent fuel (GWth-d)	150	130-180	triangular
Thermal energy produced per year (GWth)	1000	none	fixed value

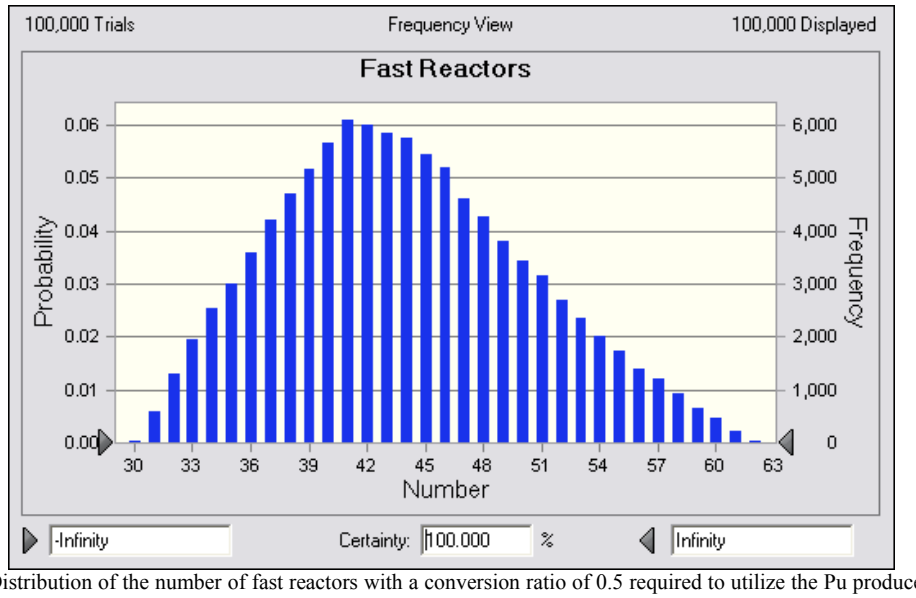

Figure 5. Distribution of the number of fast reactors with a conversion ratio of 0.5 required to utilize the Pu produced by 100 LWRs.

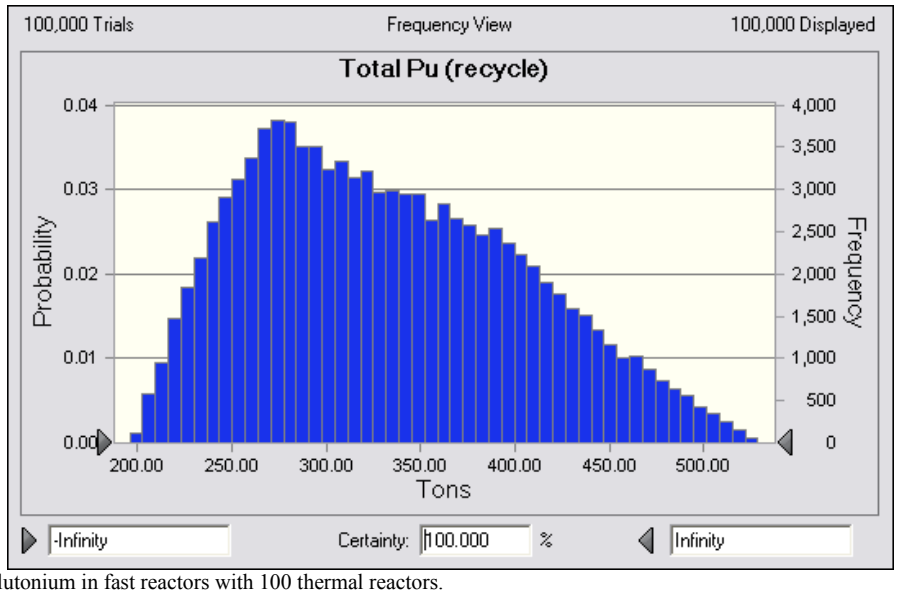

Figure 6. Plutonium in fast reactors with 100 thermal reactors.

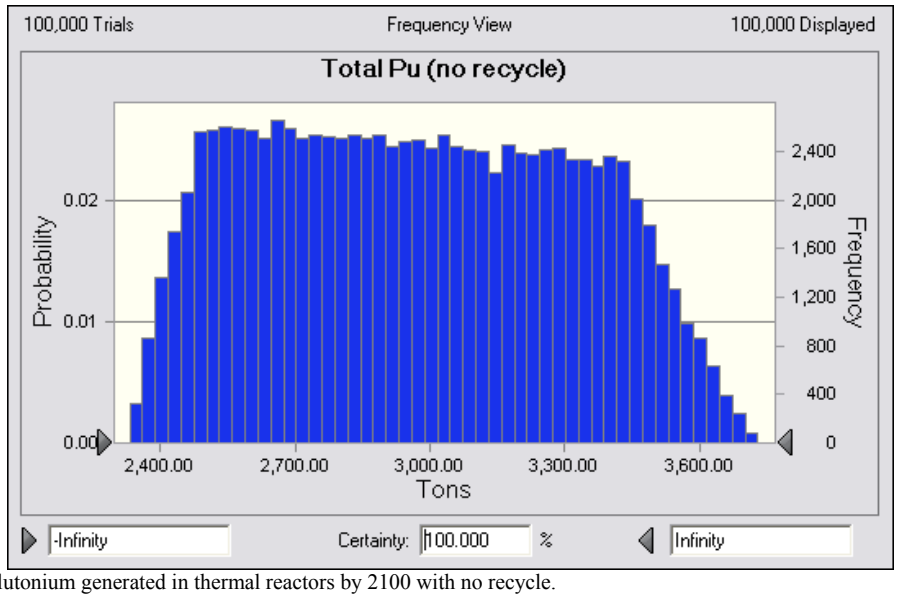

Figure 7. Plutonium generated in thermal reactors by 2100 with no recycle.

V.C. Relative Repository Heat Loads

Reactor physics calculations were preformed to extend ranges of burnup and enrichment data available in DANESS for running fuel cycle simulations and to obtain isotopic characteristics of spent fuel over a 1500 year time period. Results from these calculations were used to determine relative energy releases to a repository as a function of delay time for disposition, isotopic composition, reactor type, and burnup. These results permitted the determination of integrated heat loads as a function of choices in separation technology, and it is shown that repository heat load can be reduced by a factor of 100 with optimal management of transuranics and fission products. It is suggested that minor actinides (MA), such as Am and Cm, be considered for disposition in thermal reactors and that Pu and Np be utilized in fast reactors. Results that illustrate relative energy deposition, which includes fission products, as a function of delay time for various efficiencies of reprocessing are illustrated in Figure 8. Note that very significant reductions, greater than 50, in heat load may be accomplished by removal of actinides and by delay in closure of the repository.

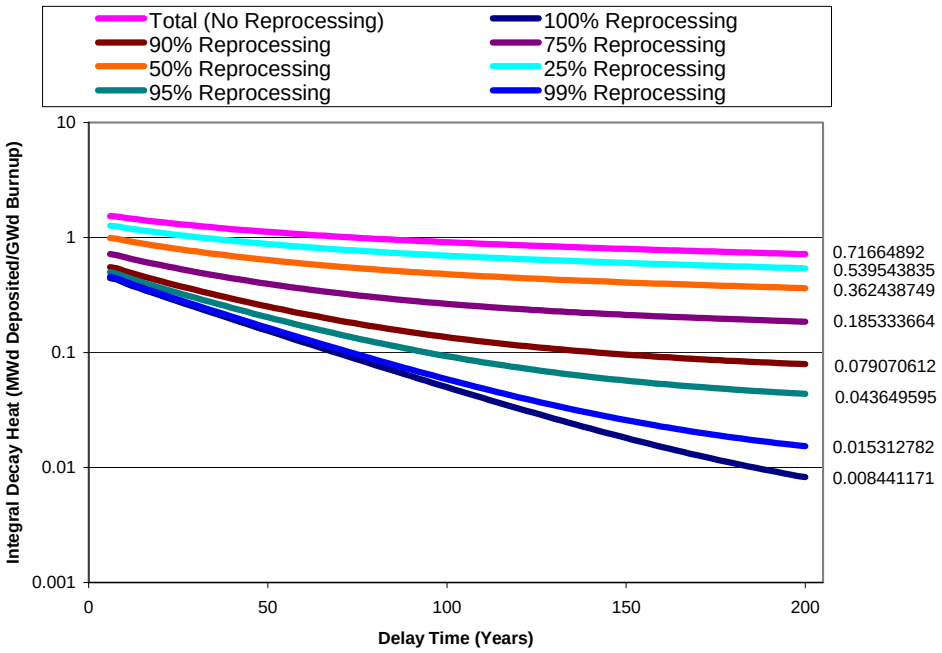

Figure 8. Ratio of energy yield to the repository relative to burnup versus delay time for various reprocessing efficiencies.

VI. CONCLUSION

Results from simulations of mixed reactor fleets show that Pu generated by LWRs can be controlled through the use of fast reactors with a CR of 0.5, and it is apparent that variations in design of both LWRs and FRs could significantly influence the quantities of in-pile and out-of-pile Pu and MA. Uncertainty analyses of equilibrium fuel cycles show that variations of a factor of fifty percent in inventories could be expected, depending of operational and deployment choices in fuel cycles.

Selective removal of specific actinides from spent fuel provides the opportunity of a very significant reduction of repository space requirements when time for out-of-pile decay time is utilized to diminish fission products.

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Repository Uncertainty Analysis for Advanced Fuel Cycles

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INTRODUCTION

Nearly 55,000 metric tons of spent fuel resides at reactor and storage sites around the United States with the fleeting chance of opening the central repository at Yucca Mountain in the near future. While this may not seem an optimal situation, it does open the possibility for revising the current strategy of the repository from one of storage of any and all once through commercial spent fuel to one of temporary storage of spent LWR fuel and long term storage of fast reactor fuel.

METHODOLOGY

Since uncertainty is ubiquitous in any fuel cycle, some assumptions must be made in order to simplify any advanced fuel cycle into something manageable. To this end, a Monte Carlo approach has been taken in building a code in Matlab to perform uncertainty analysis using data created in SCALE 5.1.

The foundation of the code is equations derived from mass flow rates through the fuel cycle, as seen in Figure 1.

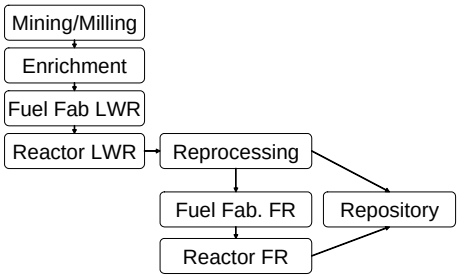

Figure 1: Simplified fuel cycle flow chart

Specific parameters of the LWR cycle are assigned empirically found distributions, Table 1, that are sampled to either initialize a fuel cycle to certain initial conditions or modify data calculated in the SAS2 sequence of SCALE[1]. Parameters for reprocessing and fast reactors are chosen as the mean values.

Table 1: Sample parameters used in analysis

Parameter	Nominal Value	Range	Distribution
Fuel Burnup (GWd)	40	30-60	Triangular
U-235 in enrichment tails (%)	.2	.2-.3	Uniform
U-235 Enrichment (%)	4.5	3-5	Triangular

LWR data used in this project was created using libraries from ORIGEN-ARP for both PWR and BWR reactor types that provided burnup data from 30 to 60 GWD/t for enrichments of 3%-5%. Fast reactor data was created with SAS2, a 1-D point depletion code, using the General Electric Super PRISM fast reactor design specifications as a model[2,3]. Both metal and oxide fuel types are modeled with burnups ranging from 80 to 200 GWd/t. A relevant ANL study was used in comparison to the data produced by the SAS2 code; however, this study uses an inner core fuel assembly without fuel shuffling whereas the ANL study used multiple fuel assembly types with shuffling[4]. Conversion ratios for this study are all less than 1, with 5 options for each fast reactor fuel type. Since these reactors are to be operated strictly as actinide burners, reprocessing is only performed on spent LWR fuel.

RESULTS

This study has shown that repository heat and mass capacity could be increased through various actions. Firstly, adding reprocessing capability on a large scale for spent LWR fuel removes the major heat and mass load contributors that would be otherwise present. This will recycle Plutonium, Americium, and Curium, while also reusing the massive amounts of usable Uranium. Secondly, lengthening the time spent FR fuel resides in a temporary facility, to around 30 years, allows short lived fission products at least one half life to decay before final deposition in the repository.

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Utilization of Fast Reactors to Control Inventories of Plutonium and Minor Actinides in Advanced Fuel Cycles

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INTRODUCTION

Inventories of Plutonium and minor actinides are important factors in determination of the risk associated with the use of nuclear energy. This includes the potential exceeding release limits from a repository and the potential for proliferation. The amount of these materials in any given fleet of reactors is determined in large part by the choice of fuel cycle and by the types of reactors selected for operation. Most of the US reactor fleet will need to be replaced within the next 30 years and additional reactors will need to be added if the contribution of power from nuclear energy is expanded. In order to minimize risk and to make judicious use of repository space, inventories all radionuclides will need to be effectively managed.

Modest reductions in requirements for repository space can be achieved through the use of high burnup fuels and of mixed oxides of Plutonium and uranium[1] (MOX). However, it is necessary to transmute actinides through the use of accelerators or fast reactors to achieve significant benefit from implementation of advanced fuel cycles. Use of hard-spectrum reactors to burn excess Plutonium and other actinides is technologically feasible and is most likely less costly than any other options for minimizing various risks.

RESULTS

Calculations for the inventories of several categories of radionuclides indicate that introduction of a modest fraction of fast reactors into the US reactor fleet is effective in stabilizing the growth of problematic radioisotopes. Results illustrated in Figure 1 are obtained from the DANESS (Dynamic Analysis of Nuclear Energy System Strategies)[2,3] code for a reactor park that consists of 30 % BWRs, 60 % PWR and 10 % fast burner reactors with a conversion ratio of 0.5 and with reactor attributes as listed in Table 1. Note that for this case the Plutonium inventory is stabilized in less than one hundred years. If a growth rate of 1.5 % is introduced and the reactor fleet is again comprised of 10 % fast reactors with a conversion ratio of 0.5, then the Plutonium inventory is not stabilized within one hundred years, as shown in Figure 2.

You may note from Figures 1 and 2 that the time of implementation of burner reactors over a thirty year period relative to the defined shutdown schedule has only a relatively small impact on Plutonium inventory. Results from calculations for the decay heat load on the repository per unit of energy generated[1] for the once-through and limited recycle for MOX fuel cycles will be compared with comparable results for fuel cycles that incorporate fast burner reactors with varying conversion ratios and fractions of hard-spectrum reactors in the fleet.

CONCLUSIONS

Inventories of Plutonium and minor actinides can be controlled through the use of recycle and hard-spectrum reactors, and significant reduction of repository space requirements can also be achieved.

Table 1. Fast Reactor Attributes

Reactor Attributes	
Pth [MWth]	3600
Pe [Mwe]	1450
Load Factor [%]	85
Coolant	Na
Fuel	FR-MOX
HM Inventory BOL [kg] (without blankets)	25690
Fuel Burn-up [GWd/tHM]	185
Power Split [%]	
Fuel Fractions [%]	
CR	
In-Pile time [d]	1553
No. of batches	6
TRU in top-up fuel [%]	57.6

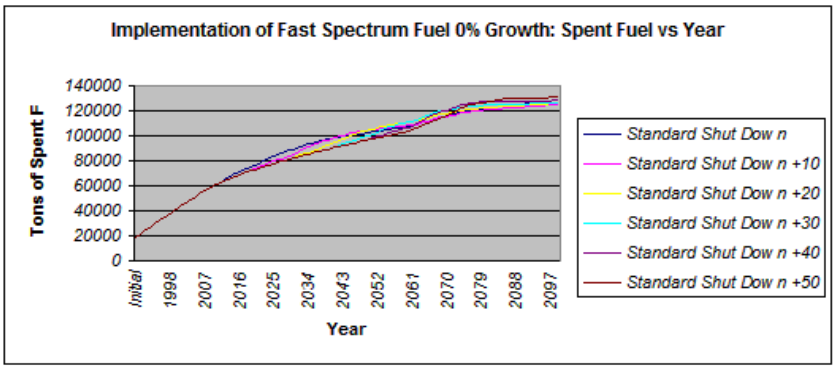

Figure 1. Spent fuel totals for time of implementation utilizing fast burner reactor with a conversion ration of 0.5 for the case of 0% growth rate.

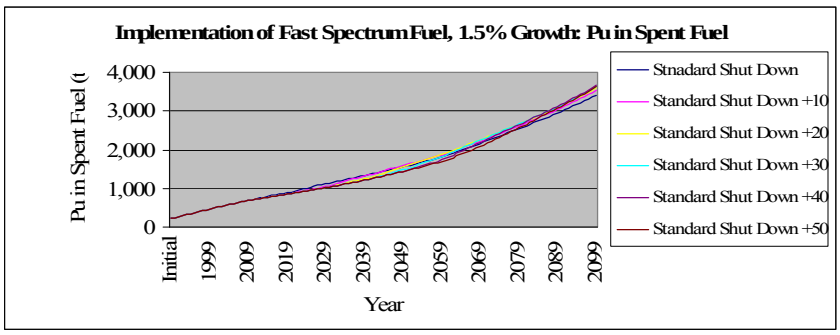

Figure 2. Spent fuel totals for time of implementation utilizing fast burner reactor with a conversion ration of 0.5 for the case of 1.5 % growth rate.

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Summary Presented at the Nov. 2007 Meeting of the American Nuclear Society

Repository Uncertainty Analysis for Advanced Fuel Cycles

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INTRODUCTION

Nearly 55,000 metric tons of spent fuel resides at reactor and storage sites around the United States with the fleeting chance of opening the central repository at Yucca Mountain in the near future. While this may not seem an optimal situation, it does open the possibility for revising the current strategy of the repository from one of storage of any and all once through commercial spent fuel to one of temporary storage of spent LWR fuel and long term storage of fast reactor fuel.

METHODOLOGY

Since uncertainty is ubiquitous in any fuel cycle, some assumptions must be made in order to simplify any advanced fuel cycle into something manageable. To this end, a Monte Carlo approach has been taken in building a code in Matlab to perform uncertainty analysis using data created in SCALE 5.1.

The foundation of the code is equations derived from mass flow rates through the fuel cycle, as seen in Figure 1.

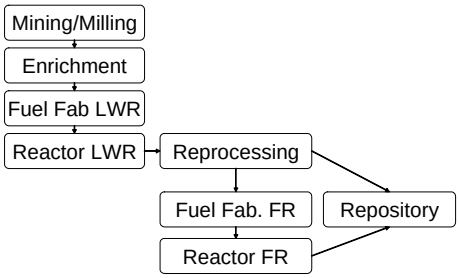

Figure 1: Simplified fuel cycle flow chart

Specific parameters of the LWR cycle are assigned empirically found distributions, Table 1, that are sampled to either initialize a fuel cycle to certain initial conditions or modify data calculated in the SAS2 sequence of SCALE[1]. Parameters for reprocessing and fast reactors are chosen as the mean values.

Table 1: Sample parameters used in analysis

Parameter	Nominal Value	Range	Distribution
Fuel Burnup (GWd)	40	30-60	Triangular
U-235 in enrichment tails (%)	.2	.2-.3	Uniform
U-235 Enrichment (%)	4.5	3-5	Triangular

LWR data used in this project was created using libraries from ORIGEN-ARP for both PWR and BWR reactor types that provided burnup data from 30 to 60 GWD/t for enrichments of 3%-5%. Fast reactor data was created with SAS2, a 1-D point depletion code, using the General Electric Super PRISM fast reactor design specifications as a model[2,3]. Both metal and oxide fuel types are modeled with burnups ranging from 80 to 200 GWd/t. A relevant ANL study was used in comparison to the data produced by the SAS2 code; however, this study uses an inner core fuel assembly without fuel shuffling whereas the ANL study used multiple fuel assembly types with shuffling[4]. Conversion ratios for this study are all less than 1, with 5 options for each fast reactor fuel type. Since these reactors are to be operated strictly as actinide burners, reprocessing is only performed on spent LWR fuel.

RESULTS

This study has shown that repository heat and mass capacity could be increased through various actions. Firstly, adding reprocessing capability on a large scale for spent LWR fuel removes the major heat and mass load contributors that would be otherwise present. This will recycle Plutonium, Americium, and Curium, while also reusing the massive amounts of usable Uranium. Secondly, lengthening the time spent FR fuel resides in a temporary facility, to around 30 years, allows short lived fission products at least one half life to decay before final deposition in the repository.

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Uncertainty Analysis of Advanced Fuel Cycles Using Dynamic System Modeling

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INTRODUCTION

Advanced fuel cycles offer the opportunity for significantly reducing the requirement for geologic repository space. In particular, if fission products, fissile material, actinides, and fertile material are separated, the need for repository space is about two percent of that required for the once-through cycle. Since implementation of advanced fuel cycles may require the construction of relatively expensive recycling facilities and of hard-spectrum reactors, it is important that uncertainties associated with deployment of these fuel cycles be understood and minimized. This paper describes the methodology for evaluating these uncertainties for advanced fuel cycles, and some preliminary results are presented.

METHODOLOGY

The DANESS (Dynamic Analysis of Nuclear Energy Systems[1,2]) code was developed by Argonne National Labs, and it has been recently upgraded to sequentially execute many scenarios and to save output for each of run at selected points in time. These new features permit one to obtain cumulative distribution functions of any output variable at any particular time.

There are well over 100 input parameters required by DANESS in order to perform a single fuel cycle scenario, and it is impractical to include all of them in an uncertainty analysis. Thus, the relative influences of these parameters on uncertainty estimates need to be determined through numerical experiments, expert elicitations[3], and engineering judgment. The DANESS model input parameters selected for inclusion in the assessment described in this paper are listed in Table 1. Triangular distributions were chosen since they permit one to establish realistic limits and since they represent the state of knowledge better than any other distributions considered.

The growth rate of LWRs will be primarily dependent on financial and on infrastructure issues. The current world-wide production capability of pressure vessels was stated to be six per year at a presentation at the ICAPP 2007 conference, and a plenary speaker at the June 2007 meeting of the ANS expressed considerable concern with the availability of sufficient personnel qualified for skilled craft work. Fast reactors

(FRs) are expected to be built at a slower rate than LWRs, and it is expected that construction will begin at least ten years following significant construction of LWRs. The values selected for storage of spent fuel (SF) at reactors reflects our perception of current policy, but it is most likely not optimal for optimization of overall fuel cycle costs. The same can be said for SF in an interim storage facility. Reprocessing should be implemented a few years prior to construction of FRs to assure an adequate fuel supply, unless non-spent fuel is used for initial loadings. The current burnup of LWR fuel is limited to less than 60 GWd/T, but upgrades of fuel fabrication facilities and licensing issues are under review.

Table 1. Input variables and specifications of parameters for triangular distributions.

Input Variable	Range	Nominal Value
Growth rate of reactors LWRs	0 to 7	3
Growth rate of reactors FRs	0 to 3	2
SF At-Rx Cooling Time	1 to 10	3
SF Interim Cooling Time	1 to 10	3
Year of Implementation of New LWR Construction	2015 to 2050	2030
Year of Implementation Reprocessing	2015 to 2030	2020
Year of Implementation of New FR Construction	2030 to 2060	2040
LWR Burnup	40 to 70	50
FR Burnup	80 to 200	120

RESULTS

A MatLab code was used to generate 100 input values for the parameters listed in Table 1, and these values were input into DANESS to calculate 100 outputs for the output variables selected for this study. Results from DANESS for each of the output variables were imported into an Excel spreadsheet and ranked to determine cumulative distribution functions (CDFs).

The parameters that are output from the current Monte Carlo version of DANESS include: Pu and minor actinides (in and out of pile), spent fuel and high level waste (at

several locations), energy produced, and number of reactors of each type. Others can be obtained if desired.

CDFs are obtained for each output variable cited above, and an example result is illustrated in Figure 1. Note that the in-pile Pu inventory could vary by over a factor of two in 2050 based on the uncertainties specified for this evaluation.

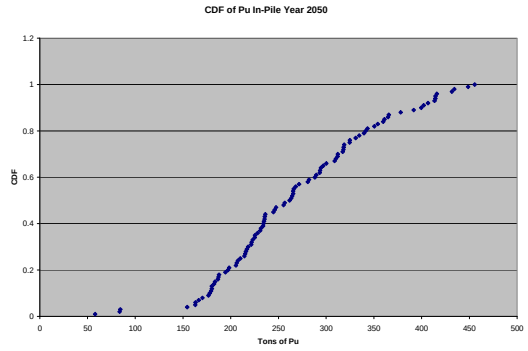

Figure 1: Sample CDF of Results. Pu In-Pile year 2050

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Isotopic Composition of Spent Fuel from Advanced Burner Reactors as a Function of Burnup

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INTRODUCTION

The isotopic composition of spent fuel from Advanced Burner Reactors (ABRs) is needed to determine their efficacy for improving spent fuel management. For this study, the result of primary interest is the heat load on a radioactive waste repository as function of burnup, which is required for performing uncertainty analyses.

A very useful study on ABRs is recently completed by Hoffman, E. A., Yang, W. S., and Hill, R. N. at Argonne National Laboratory[1] (ANL) for reactor designs that are based on reactor concepts similar to the Super Prism reactor designed by GE.[2] Both oxide and metal fuel designs are evaluated with conversion ratios of 0, 0.25, 0.5, 0.75 and 1.0, but their results are reported for a single burnup. Thus, they need to be extended to a range of burnups so they can be utilized for uncertainty analyses. This extension is accomplished by modeling these ten reactors with the sas2 module of SCALE 5.1 code[3] and by calculating spent fuel composition as a function of burnup. Scale5.1, a modular code system for performing standardized computer analyses for licensing evaluation.

DESCRIPTION OF WORK

The ten models cited above were used to study the isotopic production of each of the reactor designs for burnups of 80 through 200 GWd/t with steps of 10 GWd/t. ORIGEN-ARP was used for spent fuel depletion, decay, and source term analysis, to study the repository heat load due to the decay of the spent fuel for 1500 years. These results were fit to functions for interpolation as a function of burnup, and they were replicated by an artificial neural network.

Results obtained by Hoffman[1] utilize three-dimensional models of the reactors with multiple regions. The models for this study utilize a whole-core homogenization to determine initial composition for input to sas2. This approach is expected to obtain results relatively close to those reported by ANL since the neutron mean free path in fast reactors is significantly longer than fuel pin diameters. However, spectral effects associated with region-dependent fuel composition cannot be modeled with sas2.

RESULTS

Comparisons between SCALE 5.1 results and ANL reported values are obtained for the single burnup values cited in the ANL document[1] for over sixty isotopes. Many of

the results are very good, but some are quite poor. Figure 1 below is an example from five oxide-fueled reactor designs. The Pu 239 discharge mass, as a fraction of total heavy metal (HM), is compared to the ANL results at specified burnups, which correspond to reactors with conversion ratios of 1.0, 0.75, 0.5 and 0.0. Table 1 lists the heat load in watts for an oxide fueled reactor at a burnup of 140 GWd/t, which were calculated using ORIGEN-ARP.

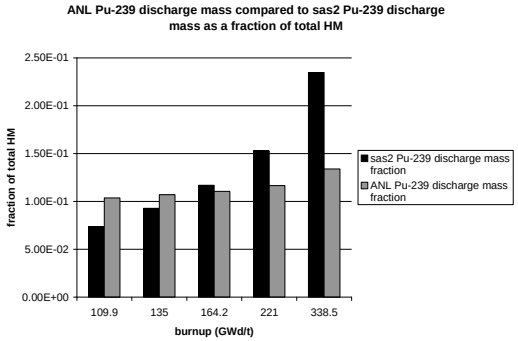

Figure 1: Pu 239 discharge mass comparison to ANL results at varying burnups.

Table 1. Heat load in watts per assembly for an oxide fueled reactor at a burnup of 140 GWd/t.

	Time (years)			
	5.0E+00	5.3E+02	1.0E+03	1.5E+03
am241	3.0E+02	2.2E+02	1.0E+02	4.7E+01
am243	3.9E+00	3.7E+00	3.6E+00	3.4E+00
am242m	3.3E-02	2.6E-03	2.5E-04	2.1E-05
cm244	3.2E+03	7.3E-06	9.2E-14	4.4E-22
pu239	2.2E+01	2.2E+01	2.2E+01	2.1E+01
pu240	7.2E+01	7.6E+01	7.2E+01	6.9E+01
pu241	5.7E+00	3.9E-04	3.8E-04	3.6E-04
total	4.3E+03	3.4E+02	2.0E+02	1.4E+02

FUTURE WORK

Work is currently ongoing based on use of three-dimensional modeling capabilities recently made available in SCALE 5.1

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