LA-UR-00-4526

RESULTS FROM A MULTIPLE-OBJECTIVE, NUCLEAR-FUEL-CYCLE OPTIMIZATION MODEL

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September 29, 2000

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ABSTRACT

A Nuclear Materials Management System (NMMS) analyzes the various material/process combinations in the nuclear cycle to determine how best to implement them. The fuel cycle (flow of materials between different materials and processes) can be optimized according to specific attributes, such as cost, proliferation risk, and environmental risk. This document describes an NMMS program, the Fuel Cycle Optimization, which optimizes the nuclear fuel cycle on the basis of economics. Although a more robust optimization program would also optimize simultaneously on other parameters, such as proliferation risk, environmental risk, and safety, the focus of this model is on the balance of the flow of materials and economic optimization. However, the model calculates estimates for proliferation and environmental risks associated with each fuel cycle under consideration.

This report details the second-year effort in a two-year Laboratory-Directed Research and Development initiative to produce a tool capable of supporting decisions regarding nuclear systems. Such a tool was indeed developed and has been shown to successfully optimize the processes desired in a nuclear fuel cycle based on given economic input parameters. The program allows decisions to be made based on input for different reactor types, pre- and post-reactor processes (reprocessing, transmutation of spent fuel, etc.), geologic repositories, and many other options.

1.0. INTRODUCTION

This report was written as part of a Laboratory-Directed Research and Development (LDRD) initiative at Los Alamos National Laboratory (LANL) to study Nuclear Materials Management Systems (NMMS) in the 21st century. The justification for this study is described in Sec. 1.1, and Sec. 1.2 contains a brief introduction to the model.

1.1. Background

The purpose of this project was to combine capabilities at LANL in nuclear system design, advanced simulation of complex systems, and nuclear materials management to initiate a new approach to the design and assessment of NMMS in the 21st century. The purpose also was to place LANL in a leadership role for current and future national and international nuclear energy and materials initiatives. Specifically, this LDRD initiative involves developing an analytic assessment tool to aid in the analysis of nuclear

systems by determining optimal mixes of nuclear reactors and fuel-cycle processes to meet the needs of future nuclear energy demand scenarios.[1,2] The primary customer(s) for this comprehensive nuclear-fuel-cycle assessment tool are the policy makers whose decisions are crucial for setting in motion the long-term research and development needed to bring a broadly optimal nuclear energy source to its highest potential.

The ultimate goal of this NMMS project was to assist future studies of such novel nuclear systems as (1) advanced nuclear power plants; (2) advanced, closed fuel cycles that minimize both proliferation and environmental risk; (3) coupled nuclear-power and proliferation-resistant systems that are economic and safe; and (4) detailed consideration of the physics, actinide, and fission-product chemistry and engineered-barrier systems for Monitored Retrievable Storage Systems (MRSS) and repositories. This goal was accomplished by coupling nuclear energy and fuel production/consumption concepts (reactors, subcritical systems, fuels, material processing, and safeguards) with nuclear material disposition concepts (interim retrievable storage, engineered barriers, and optimized disposal schemes) to determine overall performance levels as driven by economic factors (see Fig. 1-1). Achieving a balance in the approach was essential by including both the front- (i.e., fabrication and irradiation) and back-end (i.e., disposal) components of the nuclear fuel cycle connected via processing steps and transportation issues. The model also simultaneously integrated nuclear material legacies associated with the dismantling of nuclear weapon arsenals and calculated proliferation and environmental risk associated with each resulting fuel cycle.

The primary driver of the model was the global and long-term electricity demands for nuclear energy under a set of scenarios that reflect economic competition and environmental risks associated with alternative fossil and nonfossil energy sources.[3] These sources were defined in the E^3 study (energy/environment/economics), which is a global, long-term (to approximately the year 2100) Edmonds, Reilly, Barns (ERB) model.[4] Although this approach was sufficient for Fuel Cycle OPTimization (FCOPT) to present global and regional long-term demands for nuclear energy, more detailed processing/materials information may be needed in the future.[5,6] The study was initiated in fiscal year 1999 (FY99) as an LDRD project and has continued development as such. However, to carry out the ultimate goals, the project requires further progress.

1.2. Report Scope

The main starting point of this project was instantiation of the mass flow equations that govern the nuclear fuel cycle. These equations have been developed and refined over the last 2 years and represent a multireactor fuel cycle that is as accurate as possible. These equations then were optimized based solely on economic input factors.

This document describes final results obtained from the computational tool, FCOPT,[7] which was written to perform the above functions.

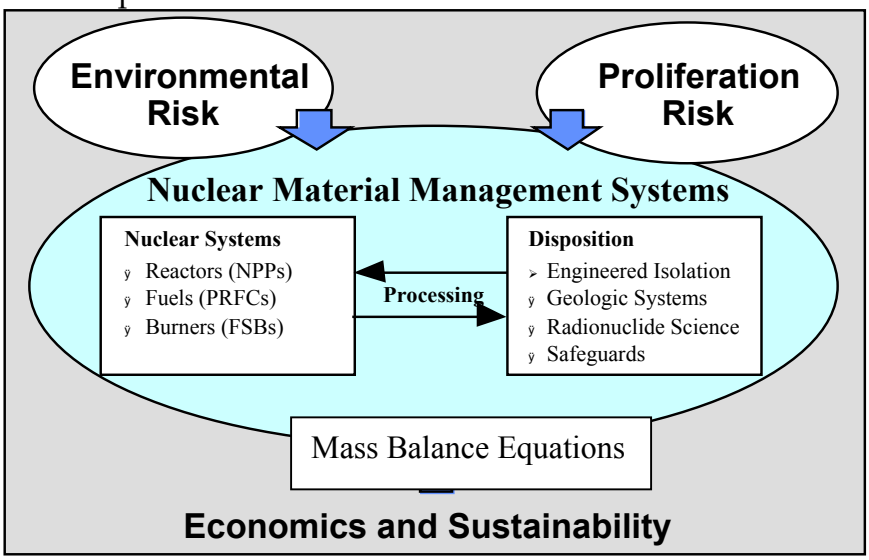

Fig. 1-1. Representation of major components of NMMS addressed in this LDRD effort and the linkages among them and performance drivers.

Results include both the costs of various fuel-cycle combinations and the proliferation and environmental risks associated with each fuel cycle once an optimization based on certain input parameters is performed for each year over a 100-year period. Descriptions of how these parameters were calculated are included, as is a basic description of how the model works (though more detailed information can be found in Ref. 1).

Although the project was initiated with the goal of finding the optimal nuclear fuel cycle based on three areas (economics, proliferation, and environment), the technique for optimizing only one of these (economics) has been implemented. Tying the others into the optimization will be the product of future work efforts, perhaps using multi-attribute and/or multi-criterion optimization techniques. In fact, development of this and other areas is crucial to meeting the ultimate goals of this project. Future work plans then can be made.

Thus, the main purpose of this document is to show the kind of results that can be obtained from FCOPT for future uses. For those unfamiliar with the code, Section 2.1 gives a brief description of the code itself (more information can be found in Ref. 7), as well as modifications made in the past year. Section 2 also describes the graphical user interface (GUI) developed for FCOPT, the cases chosen to represent certain results, and a short description of the proliferation and environmental risk metrics used in the analysis. Results from several sensitivity runs and a few of the aforementioned cases

are displayed in Section 3, and future work ideas (pending future funding) and conclusions appear in Section 4.

2.0. FCOPT MODEL

FCOPT is a Fortran 90 program (with some C functionality) that uses a linear programming (LP) technique (in the form of a code called CPLEX)[8] to optimize the total cost over 100 years. This is done using a set of constraints (material flow balances) between various materials/processes and input parameters specifying the cost per unit of each material. The program has over 100 constraints that relate the flow of material between one process in the nuclear fuel cycle and another (see Fig. 2-1). Some of these constraints are even time-dependent, and the balance for one timestep may depend on variables in the previous one (and if the required timestep is before the period of optimization, fixed data points are provided for past material flows). Additionally, the mix at one timestep does not have to be identical to that in the previous one (although it is often more economic that way). The constraints for a given timestep are given in Appendix A, and the equation listing the costs associated with each variable (called the objective function) is listed in Appendix B. More details about the model can be found in Ref. 7.

In the past year, the number of materials and processes analyzed by FCOPT has increased; Section 2.1 discusses more details about these modifications. The past year also has seen the further development of a GUI. The run time of FCOPT is a minimum

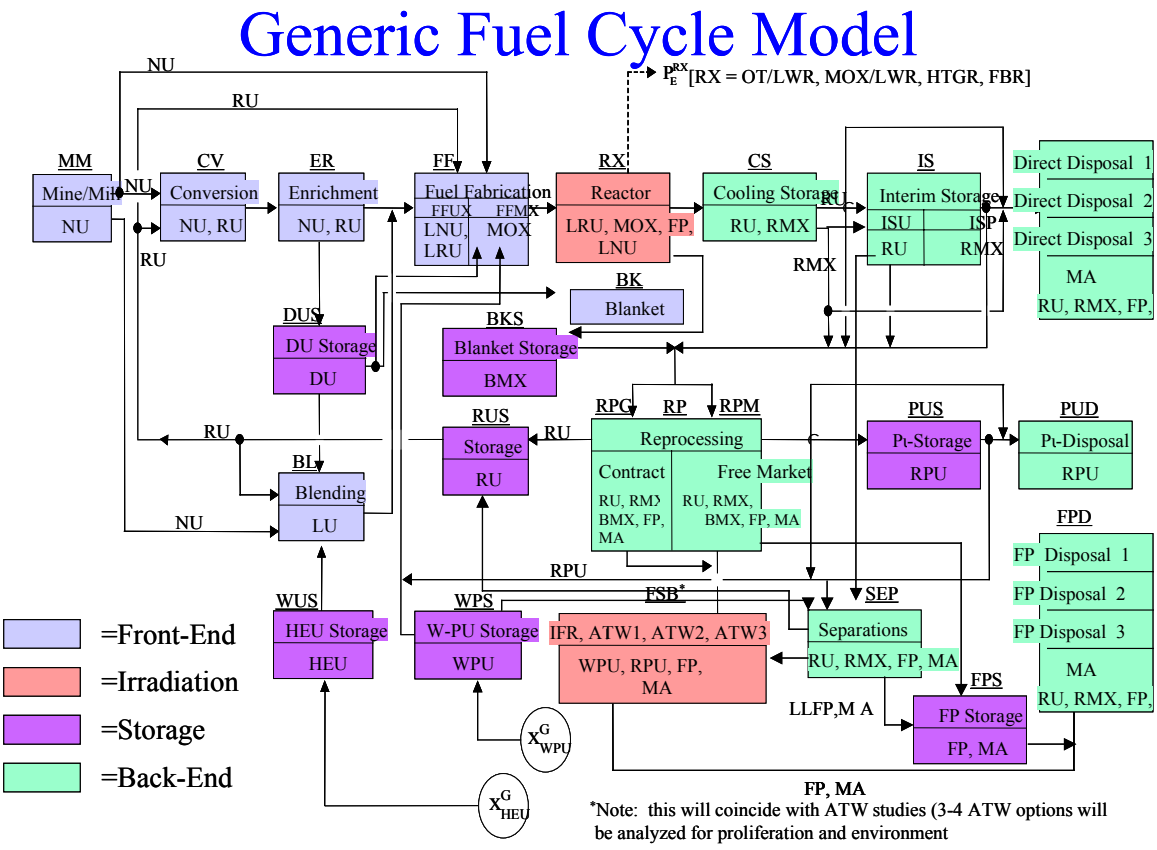

Fig. 2-1. Flow chart of nuclear fuel cycle for FCOPT.

of 20 minutes on a UNIX workstation, so it was not realistic for the GUI to actually run FCOPT on a laptop. Instead, a number of predefined cases were run on a UNIX system, and the results were transferred to a PC, where they were displayed using the GUI. These 90 cases are described in Section 2.2, the GUI in Section 2.3, and the proliferation and environmental calculations in Sections 2.4 and 2.5, respectively.

2.1. Modification of FCOPT Program

Modifications to FCOPT in the past year include the addition of new materials/processes (which are actually just refinements of previous materials/processes) and the addition of vintaging for economic purposes. The constraints also were modified to reflect the above two additions (see Appendix A for the updated version). These changes are described in more detail below.

2.1.1. Material and Processes Expansions

An updated list of materials examined by FCOPT appear in Table 2-1, and the fuel-cycle processes analyzed appear in Table 2-2. These materials encompass a wide range of nuclear-fuel-cycle options and allow a variety of nuclear materials to be examined.

TABLE 2-1
DESCRIPTION OF MATERIAL STREAMS CONSIDERED IN FCOPT MODEL

Material Streams	Acronym
Natural uranium	NU
Reactor-recycled uranium	RU
Depleted uranium	DU
High-enriched uranium (released from weapons)	HEU
Low-enriched uranium (LEU) derived from NU	LNU
LEU derived from RU	LRU
LEU derived from HEU blended with NU, RU, and DU	LWU
Weapons-released plutonium	WPU
Reactor-generated/released plutonium	RPU
Mixed (Pu/U) oxide fuel	MOX
MOX-reactor-released (fissioned) spent fuel	RMX
HTGR[a]-reactor-released (fissioned) spent fuel	RHTG
FBR[b]-reactor-released (fissioned) spent fuel	RFBR
FBR-blanket-generated/released MOX	BMX
Fission products	FPs
Long-lived fission products	LFPs
Minor actinides	MAs

[a]HTRG = high-temperature, gas-cooled reactor.

[b]FBR = fast breeder reactor.

The materials added to FCOPT included spent HTGR and spent FBR fuel. Previously, the category RMX covered all spent mixed oxide (MOX) fuel, including that from light-water reactors (LWRs), HTGRs, and FBRs. To separate the environmental risk associated with each, the three were separated.

TABLE 2-2
DESCRIPTION OF PROCESSES CONSIDERED BY FCOPT MODEL

Process	Acronym
Mining and milling	MM
UF_6 conversion	CV
Uranium enrichment	ER
Fuel fabrication of uranium oxide fuel (UO_2)	FFUX
Fuel fabrication of MOX from reprocessed plutonium	FFMXRP
Fuel fabrication of MOX from plutonium storage	FFMXPS
Fuel fabrication of MOX from weapons plutonium storage	FFMXWP
Blending of HEU with NU, DU, or RU	BL
Irradiation in once-through (OT) LWRs	RXOT
Irradiation in partial-core MOX recycle LWRs	RXMX
Irradiation in HTGRs	RXHG
Irradiation in FBRs	RXBR
Irradiation in IFRs[a]	IFR
Irradiation in ATW[b]1	ATW1
Irradiation in ATW2	ATW2
Irradiation in ATW3	ATW3
FBR plutonium-breeding blanket	BK
Cooling storage (reactor-side) for spent UO_2 or MOX fuel	CS
Interim storage for spent UO_2 fuel	ISU
Interim storage for spent MOX fuel	ISP
Depleted uranium storage	DUS
Recycled uranium storage	RUS
FBR BMX storage	BKS
Separated plutonium storage	PUS
Separated fission product (plus MA) storage from reprocessing	REFPS
Separated fission product storage from separations	SEFPS
Weapons plutonium storage	WPS
Weapons uranium storage	WUS
Contract reprocessing with feed from CS	RPCCS
Contract reprocessing with feed from IS	RPCIS
Contract reprocessing with feed from BK	RPCBKS
Market reprocessing with feed from CS	RPMCS
Market reprocessing with feed from IS	RPMIS

Market reprocessing with feed from BK	RPMBKS

TABLE 2-2 (cont)
DESCRIPTION OF PROCESSES CONSIDERED BY FCOPT MODEL

Amount of material going through separations	SEP
Direct disposal 1	DD1
Direct disposal 2	DD2
Direct disposal 3	DD3
Direct disposal from CS	DDCS
Direct disposal from IS	DDIS
Separated plutonium disposal	PUD
Separated plutonium disposal from RP	PUDRP
Separated plutonium disposal from PUS	PUDPUS
Separated fission product disposal 1	FPD1
Separated fission product disposal 2	FPD2
Separated fission product disposal 3	FPD3
Separated fission product disposal from REFPS	REFPD
Separated fission product disposal from SEFPS	SEFPD
Separated fission product disposal from IFR	FPDIFR
Separated fission product disposal from ATW1	FPDATW1
Separated fission product disposal from ATW2	FPDATW2
Separated fission product disposal from ATW3	FPDATW3

[a]IFR = integral fast reactor.

[b]ATW = accelerator transmutation of waste.

The process additions included separations and flows to and from separations ("separations" was differentiated from "reprocessing" because both minor actinides and fission products were part of the waste stream leaving reprocessing, whereas only fission products left separations) and the distinction of four separate fast spectrum burners (FSBs) and three separate direct and fission product geological repositories (i.e., disposal). The first type of FSB was the IFR, which irradiates either plutonium in storage (reactor grade or weapons grade) or spent fuel. The others are three different types of accelerator transmutation of waste (ATW) designs that only burn spent fuel. For the purpose of this project, only two were differentiated in terms of irradiation properties and resulting nuclide discharge (thus, repository performance): the molten salt system (ATW1) and the lead-bismuth system (ATW2). All four were associated with the same costs, and although ATW3 was given the same repository performance as ATW2 and IFRs, it was included for future use in ATW decision-making and general flexibility. The three different types of direct disposal and fission product disposal repositories were included so that detailed environmental studies could be performed on different geological media (for example, bedded salt vs volcanic tuff vs granite). This functionality has not yet been used yet; however, the capability exists.

2.1.2. Vintaging

Another modification to FCOPT is the addition of the economic concept of vintaging. Vintaging is the principle of making the cost of a reactor per year dependent on the time it took to construct the reactor and how long the reactor will be operating. Previously, one of the limitations of the program was the lack of a cost penalty incurred when one reactor type became slightly more costly and was then overtaken by one that was slightly cheaper. Vintaging attempts to prevent a "knife-edge" in- and out-swapping of systems by adding a cost penalty for a system that becomes cheaper by burdening it with the cost of future investment. In this way, a reactor cannot be easily turned on and off, but it is required to operate for a given lifetime, and it becomes more costly to implement a second type of reactor. The equations used to implement vintaging are described in the appendices.

2.2. Description of Cases

Although there are many parameters that can be changed in FCOPT and that can produce different results, the most significant are those associated with the reactor mix. The type of reactor(s) that operates influences both the front-end fuel requirements [what type of fuel is required (UO_2 vs MOX)] and the back-end processing (storage, FSB, and disposal) of that fuel. IFRs and ATWs are considered to be FSBs, not reactors in this model, so there are only four types (LWRs, MOXs, HTGRs, and FBRs), leading to eight different combinations. With each reactor combination, there is also the option of three different uranium prices and turning on/off reprocessing (except for a pure OTLWR case that does not use plutonium) and/or a fast spectrum burning (IFR or ATW). Thus, 90 "possible" combinations (not all of them equally likely) of the above variables were considered and run with FCOPT. Results from these 90 cases then were transferred to the GUI, from which they then could be viewed by a more widespread audience. Keep in mind that the cases without reprocessing can use plutonium from storage (either reactor-grade or weapons-grade plutonium) and that the FSB implemented in these cases was ATW2.

TABLE 2-3
KEY FOR 90 CASES

Reactor Combinations	Reprocessing	FSB	Uranium Cost
A. LWR	No	Yes/No	Hi/Med/Lo
B. LWR and MOX	Yes/No	Yes/No	Hi/Med/Lo
C. LWR and HTGR	Yes/No	Yes/No	Hi/Med/Lo
D. LWR, MOX, and HTGR	Yes/No	Yes/No	Hi/Med/Lo
E. LWR and FBR	Yes/No	Yes/No	Hi/Med/Lo
F. LWR, MOX, and FBR	Yes/No	Yes/No	Hi/Med/Lo
G. LWR, HTGR, and FBR	Yes/No	Yes/No	Hi/Med/Lo
H. LWR, MOX, HTGR, and FBR	Yes/No	Yes/No	Hi/Med/Lo

2.3. Description of Graphical User Interface

An important part of the FCOPT development is a GUI that can be used by decision makers to quickly view and understand the results of the optimization of possible future fuel-cycle scenarios. In this section, both the GUI itself and its interface to the optimization code are described.

A major design issue was whether the GUI should be an integral part of the optimization code or run separately, thus accessing a database calculated previously. The advantage of the first option is that the user has access to all possible combinations of input parameters, whereas with the second option, the number of combinations is limited to those included in the database. The second option was chosen primarily because of the constraint that the optimization code utilizes commercial software that was purchased originally for the Sun Solaris platform, whereas the GUI is intended to run on a Windows laptop. This option also has the advantage that the user has essentially immediate access to the results and does not have to wait for the completion of an optimization calculation. The database is created on the Sun platform, then written to a Windows-compatible CD, where it can be accessed by the GUI. Using the CD allows a large number of parametric cases to be included, giving the user a better feel for the sensitivity of the results to variations in input parameters.

A shadow database was created in the optimization code to facilitate the transfer of information to the GUI. A shadow database is a data structure that mimics, or shadows, the computational database but is maintained separately to allow maximum efficiency of the computation. The shadow database contains much more information (such as parameter names, units, and descriptions) than the computational database, and if the entire structure were accessed during low-level computational operations, the resulting code would run slower (this is a frequent cause of poor computational efficiency in object-oriented languages). The optimizing code was written in Fortran 90, which allowed clean usage of pointers to reference the computational variables from the shadow database.

Each run of FCOPT writes a binary file with the contents of the shadow database, including the numerical results of the optimization. The name of this file is given as the second line of the fc.dat input file. The results of a suite of cases are concatenated into a single file with the fcpuiDB tool, which also translates data byte ordering to the format used by the PC. This file, fcpuiDB.bin, is ported to the PC, where it is read during initialization of the GUI.

The GUI was designed to provide the user with a way to choose different cases quickly, view an intuitive overview of the results of each case, summarize graphs for each case, and have easy access to any of the detailed results included in the database. The intuitive overview is presented as a flowchart showing all possible processes and

material streams that were considered as part of the nuclear-fuel-cycle optimization problem, with the elements clearly highlighted that were chosen as active by the optimization solution. A dialog box allows the user to set values quickly for four parameters: (1) reactor types considered, (2) use of reprocessing, (3) use of ATW, and (4) fuel cost. The flow chart immediately shows the active components of the optimized system.

The user can point and click on the material streams of the process diagram to bring up time plots of material flow. If the depicted stream represents more than one material stream, a dialog box appears that allows the user to choose among them. Material streams, as well as all the other information in the database, also can be plotted by selecting Graph/Select Variable from the menu. These graphs show plots of the selected variable for the currently selected parametric case. If the "Dynamic Graphs of Applied Case" box on the case selection dialog box is checked, then subsequent selection of new cases will add plots for each case to the open graphs to a maximum of six plots per graph.

The individual plots on each graph are identified by "Plot 1," "Plot 2," …, "Plot 6," written in the same color as the associated plot in the right margin. The user can point and click on these identifiers to bring up another window with detailed information about the plot, including name, description, units, and a description of the parameter options that were selected to describe the case.

Plot packages can be viewed by selecting Graph/Select Plot Package from the menu. These packages are graphs with several plots of different variables for the same case, with the same units, packaged in a single graph for easy comparison. Graph packages can be viewed by selecting Graph/Select Graph Package from the menu. Graph packages are full-screen layouts of up to four plot packages. Only one graph package can be active simultaneously. The default graph package, iconized at GUI startup, is the case summary package. This shows time plots of electricity generation for the different reactor types, the cost of electricity (COE), a proliferation metric, and an environmental metric. The results depicted in the graph package, as well as those in any plot package, will change to show results for the current parametric case as the user selects it from the case selection dialog box.

The FCOPT GUI is written in Visual C++ and will run under any Windows (95, 98, or NT) on any machine that supports a screen resolution of at least 1024 x 768 pixels.

2.4. Description of Proliferation Risk Metric

Numerous studies have been performed regarding how best to calculate the proliferation risk of a certain fuel-cycle process. Members of the Paul Scherrer Institute (PSI) recently studied various ways to represent proliferation risks, and the results of

this study are forthcoming.[9] Part of their study included investigating a way to develop a simplified and transparent "recipe" that might be used by nuclear utilities to classify related source materials [SMs, e.g., fresh (F) and spent (S) nuclear fuels]; comparisons are made with spent and fresh fuels based on LEU, mixed (U, Pu) oxide fuel (MOX), and natural uranium [NU, as used in the Canadian deuterium-uranium reactors (CANDUs)]; highly enriched uranium (HEU) as used in some research reactors is also included as part of the comparison ensemble. Some results of this study are the measurement and comparison of proliferation risks of a range of commercial nuclear fuel cycles in FCOPT. This basic metric is known as the nonproliferation attractiveness level (AL) and is described more below.

The International Atomic Energy Agency (IAEA) has recommended a set of criteria for the level of protection needed to protect fissile materials ^{233}U, ^{235}U, and plutonium isotopes from illicit use through diversion or theft for the construction of nuclear explosive devices.[10] The United States Department of Energy (DOE) has elaborated and extended these recommendations in the form of "attractiveness levels" (AL = A - E) to offer a more graded and hopefully useful/usable set of criteria.[11] Both sets of protection categories are based on kind [^{233}U, ^{235}U, and Pu (all isotopes)], concentration (as in the case of ^{235}U), and mass of fissile material. A summary description of the DOE recommended ALs, along with comparable, but less-detailed (graded) groupings suggested by the IAEA,[10] are given in the following:

<u>DOE ALs or Grades:</u>[11]

- Grade A (weapons): assembled weapons and test devices;
- Grade B (pure products): pits, major components, button ingots, recastable metal, directly convertible materials;
- Grade C (high-grade materials): carbides, oxides, nitrate solutions (>25 g/l), fuel elements and assemblies, alloys and mixtures, UF_4 or UF_6 (>50% enriched), etc.;
- Grade D (low-grade materials): solutions (1–25 g/l), process residues requiring extensive reprocessing, moderately irradiated material, ^{238}Pu (except waste), UF_4 or UF_6 (>20% but <50% enriched); and
- Grade E (other materials): highly irradiated forms, solutions (<1 g/l), uranium containing < 20% ^{235}U (any form, any quantity).

<u>IAEA INFCIRC/225/Rev. 4 Designations:</u>[10]

- Refers to unirradiated material; irradiated material moves to the next category;
- Uranium subclasses:
  - (a): Uranium enriched to ≥20% ^{235}U;
  - (b): Uranium enriched to ≥10% ^{235}U, but <20% ^{235}U;

- (c): uranium enriched above natural levels (0.711%), but <10% ^{235}U.

Each of the AL values associated with the DOE[11] guidelines and the "subclasses" for the comparable IAEA[10] guidelines for a given mass of the respective weapons-usable material has associated with it a protection category, I, II, or III. The highest level of protection requires three barriers and corresponds to category I. The lowest level of protection, Level III, corresponds only to an unfenced, unbarred building. Although not duplicating the IAEA protection guidelines, the DOE regulations offer a greater category and material gradation. Figures 2-2 and 2-3 depict graphically the mass-AL-category "phase space" for the graded DOE regulations. Side-by-side comparisons with the IAEA guidelines for both ^{235}U and for ^{233}U/Pu (all isotopes) also are given.

Also shown on the figures are estimates of masses of weapons-usable materials for SM derived from fresh (F) and spent (S) fuels going to or taken from a CANDU reactor (NUF and NUS, with NU = natural uranium); an LWR reactor (LEUF and LEUS, with LEU = low-enriched uranium); an LWR operated on mixed (U, Pu) oxide fuel (MOXF and MOXS, with MOX = Mixed OXide); and a research reactor (10 MWt) operated on highly enriched uranium (HEUF and HEUS).[12] The CANDU and LWR reactors are of the 1000-MWe class. The length of the vertical line designating each entry is determined by (1) the mass (kilograms) of weapons-usable material per fuel element at

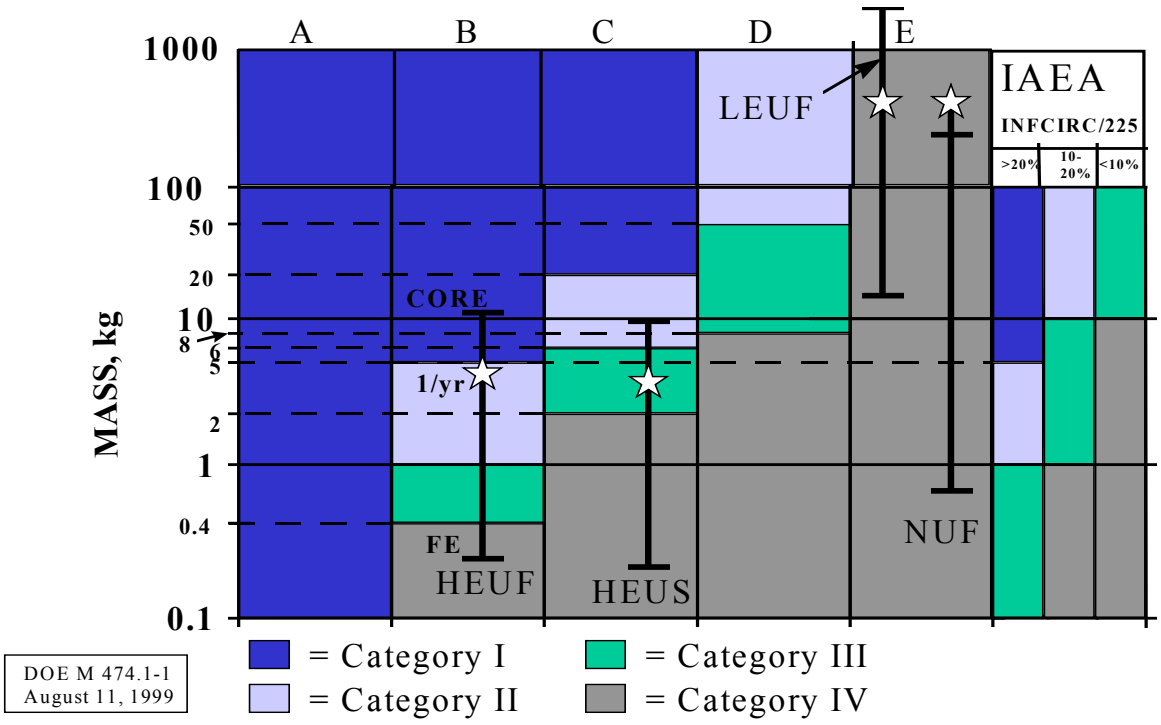

Fig. 2-2. Comparison of DOE[11] regulations and IAEA[10] guidelines for protection categories vs material mass and DOE-graded ALs and IAEA subclasses for ^{235}U.

the bottom; (2) the mass (kilograms) of weapons-usable material per core loading at the top; and (3) the mass of weapons-usable material transferred out of the respective cores each year (kilograms per year) and designated by the star-like shape.

If a numerical value is assigned to each of the DOE ALs described above and depicted in Fig. 2-2, the F (fresh) ⇒ S (spent) transition for each reactor type considered can be expressed graphically as a function of kilograms per reactor per year, kilograms per core, kilograms per fuel element, and nuclear weapon equivalent per reactor per year. The AL numerical assignments give AL = 10,000 for attractiveness level A and AL = 1 for attractiveness level E, as described below. This numerical reexpression of Figs. 2-2 and 2-3 is shown in Fig. 2-4. The MOX (LWR) and HEU (research reactor) fuel forms decrease in AL ranking as well as in magnitudes for all four rate or inventory measures. The NU (CANDU) and the LEU (LWR) increase in AL ranking, albeit from an initially lower level; with the exception of the NW/reactor/year metric, the magnitudes of all production or inventory measures also decrease.

The DOE ALs for each of the SM forms considered have been determined on the basis of judgments made by the authors. On this basis, only the MOXF fuel form requires Category-I physical protection for the commercial forms, with the same classification falling on the full core of an HEU-fueled, 10-MWt research reactor. For the case of ^{235}U-bearing SM, similar

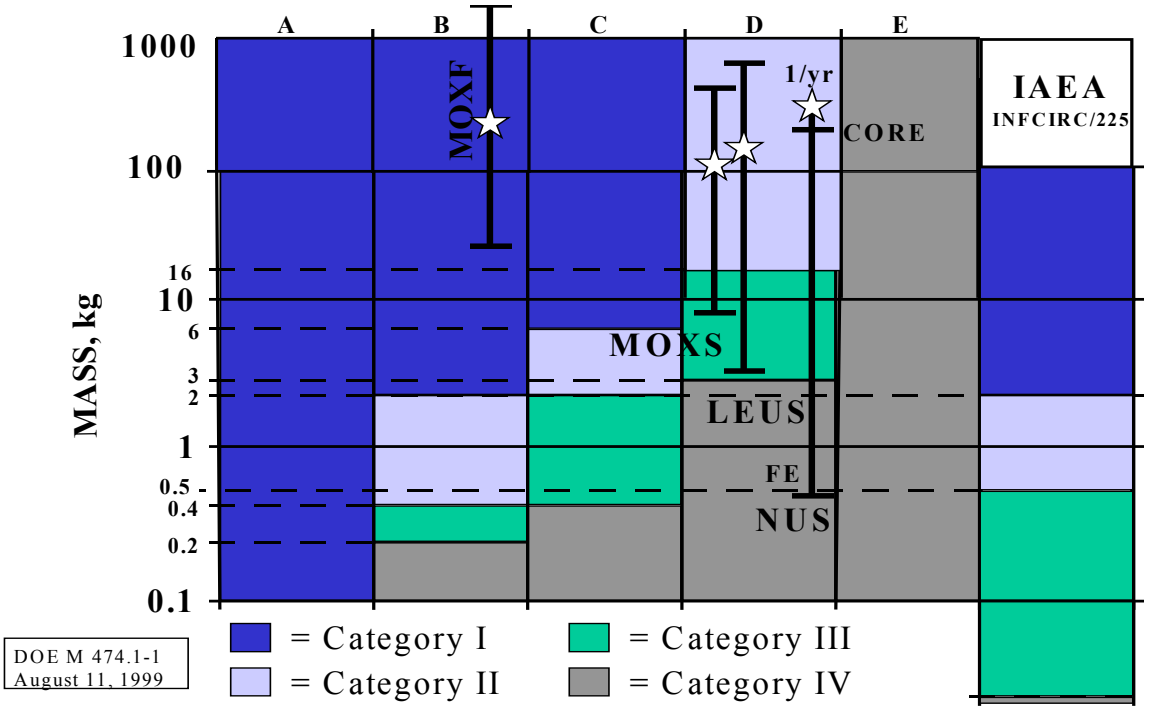

Fig. 2-3. Comparison of DOE[11] regulations and IAEA[10] guidelines for protection categories vs material mass and DOE-graded ALs (no IAEA subclasses) for $^{233}U/Pu$ (all isotopes).

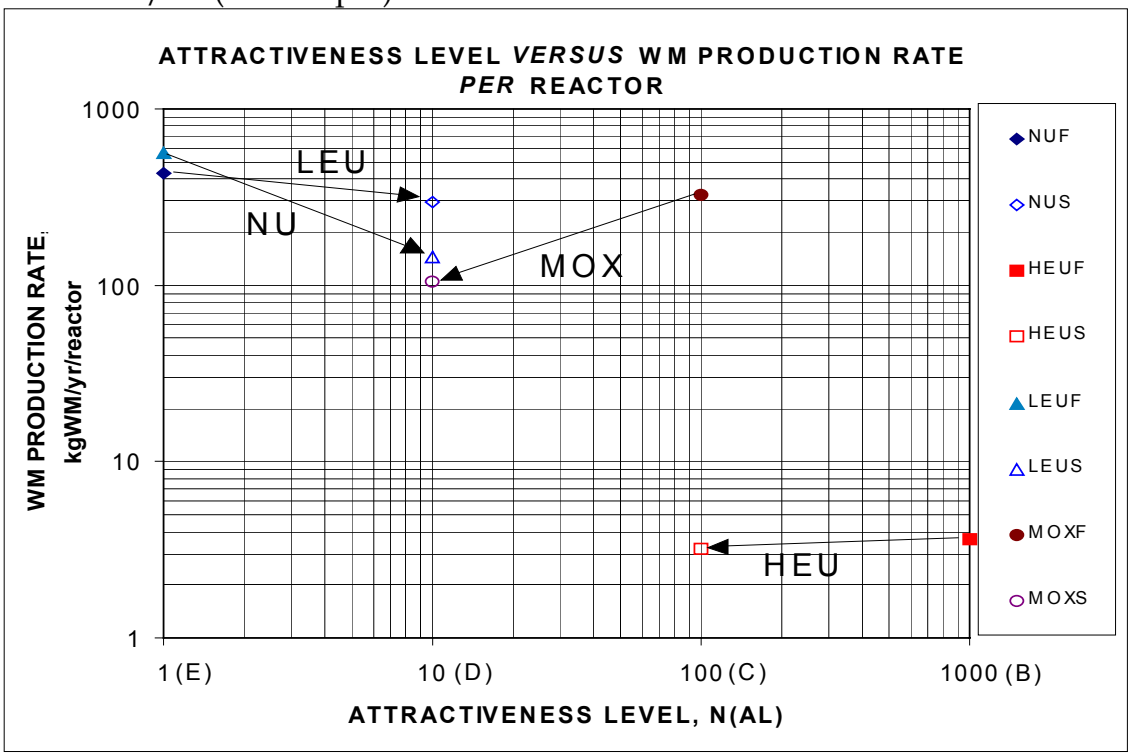

Fig. 2-4. Production of weapons-usable material per reactor year as a function of reactor type and DOE[11] AL.

conclusions would be reached using the IAEA guidelines (see Fig. 2-2). This would not be the case for plutonium-bearing SM, where all but the NUS (FE) would receive a Category-I rating if the IAEA guidelines were applied.

For FCOPT, the attractiveness of each material analyzed was input, and the final proliferation risk metric was the sum (over all materials) of this AL times the amount of material in a given process (i.e., reactor-grade plutonium in spent fuel is less attractive than plutonium in storage). Although this does not give a definitive quantity as a result, it is a metric that eventually can be used in linear optimization techniques for proliferation risk. Equation (2-1) used to calculate proliferation risk (below) uses the information obtained from Tables 2-4 to 2-6:

$$\sum_{m=1}^{m\max}\sum_{p=1}^{p\max} AL_{m,p}\, tres_{m,p}\, yPU_{m,p} \qquad (2\text{-}1)$$

where

AL = attractiveness level of the material in the given process,
tres = effective time constraints for plutonium-bearing material, and
yPU = effective plutonium concentration in material/process.

TABLE 2-4
ATTRACTIVENESS LEVEL FOR VARIOUS MATERIAL/PROCESS COMBINATIONS

	wpu	**rpu**	**ru**	**rmx**	**rfbr**	**rhtg**	**bmx**	**mox**	**heu**
pus	0	1000	0	0	0	0	0	0	0
pud	0	1000	0	0	0	0	0	0	0
bks	0	0	0	0	0	0	100	0	0
wps	1000	0	0	0	0	0	0	0	0
wus	0	0	0	0	0	0	0	0	1000
bl	0	0	0	0	0	0	0	0	100
ffmxrp	0	100	0	0	0	0	0	0	0
ffmxps	0	100	0	0	0	0	0	0	0
ffmxwp	100	0	0	0	0	0	0	0	0
cs	0	0	10	10	10	10	0	0	0
isu	0	0	10	0	0	0	0	0	0
isp	0	0	0	10	10	10	0	0	0
rpmcs	0	0	1000	1000	1000	1000	0	0	0
rpmis	0	0	1000	1000	1000	1000	0	0	0
rpmbks	0	0	0	0	0	0	1000	0	0

rpccs	0	0	1000	1000	1000	1000	0	0	0
rpcis	0	0	1000	1000	1000	1000	0	0	0
rpcbks	0	0	0	0	0	0	1000	0	0
sep	0	0	1000	1000	1000	1000	0	0	0
rxmx	0	0	0	0	0	0	0	100	0
rxhg	0	0	0	0	0	0	0	100	0
rxfb	0	0	0	0	0	0	0	100	0
rifr	100	100	100	100	100	100	0	0	0
atw1	0	0	100	100	100	100	0	0	0
atw2	0	0	100	100	100	100	0	0	0
atw3	0	0	100	100	100	100	0	0	0
dd1	0	0	10	10	0	0	0	0	0
dd2	0	0	10	10	0	0	0	0	0
dd3	0	0	10	10	0	0	0	0	0

TABLE 2-5
EFFECTIVE TIME CONSTRAINTS FOR PLUTONIUM-BEARING MATERIALS IN DIFFERENT PROCESSES

	wpu	rpu	ru	rmx	rfbr	rhtg	bmx	mox	heu
pus	1	1	1	1	1	1	1	1	1
pud	1	1	1	1	1	1	1	1	1
bks	1	1	1	1	1	1	1	1	1
wps	1	1	1	1	1	1	1	1	1
wus	1	1	1	1	1	1	1	1	1
bl	1	1	1	1	1	1	1	1	1
ffmxrp	1	1	1	1	1	1	1	1	1
ffmxps	1	1	1	1	1	1	1	1	1
ffmxwp	1	1	1	1	1	1	1	1	1
cs	1	1	4	4	4	4	1	1	1
isu	1	1	10	1	1	1	1	1	1
isp	1	1	1	10	10	10	1	1	1
rpmcs	1	1	3	3	3	3	1	1	1
rpmis	1	1	3	3	3	3	1	1	1
rpmbks	1	1	1	1	1	1	3	1	1
rpccs	1	1	3	3	3	3	1	1	1
rpcis	1	1	3	3	3	3	1	1	1
rpcbks	1	1	1	1	1	1	3	1	1
sep	1	1	3	3	3	3	1	1	1
rxmx	1	1	1	1	1	1	1	4	1
rxhg	1	1	1	1	1	1	1	4	1
rxfb	1	1	1	1	1	1	1	2	1
rifr	1	1	1	1	1	1	1	1	1
atw1	1	1	1	1	1	1	1	1	1
atw2	1	1	1	1	1	1	1	1	1
atw3	1	1	1	1	1	1	1	1	1
dd1	1	1	1	1	1	1	1	1	1
dd2	1	1	1	1	1	1	1	1	1
dd3	1	1	1	1	1	1	1	1	1

TABLE 2-6
EFFECTIVE PLUTONIUM CONCENTRATION FOR VARIOUS MATERIAL/PROCESS COMBINATIONS

	wpu	rpu	ru	rmx	rfbr	rhtg	bmx	mox	heu
pus	0	1	0	0	0	0	0	0	0
pud	0	1	0	0	0	0	0	0	0
bks	0	0	0	0	0		0.0300	0	0
wps	1	0	0	0	0	0	0	0	0
wus	0	0	0	0	0	0	0	0	1
bl	0	0	0	0	0	0	0	0	0.2
ffmxrp	0	1	0	0	0	0	0	0	0
ffmxps	0	1	0	0	0	0	0	0	0
ffmxwp	1	0	0	0	0	0	0	0	1
cs	0	0	0.0152	0.0376	0.0302	0.1320	0	0	0
isu	0	0	0.0146	0	0	0	0	0	0
isp	0	0	0	0.0349	0.0294	0.1310	0	0	0
rpmcs	0	0	0.0152	0.0376	0.0302	0.1310	0	0	0

2.5. Description of Environmental Risk Calculation

Several different methods are used to calculate the environmental risk associated with nuclear activities. Dose is a widely used metric, but it requires knowing where an individual is relative to the nuclear material and what type of transport medium exists between them. Toxicity is also another metric, but it is given in terms of the cubic meters needed to dilute a release from the material of interest so that it will be safe to inhale or ingest. The environmental risk associated with the front-end processes of the fuel cycle (mining, milling, conversion, etc.) is, for the most part, related to the risk of inhalation (due to air releases) of a certain isotope (or combination of isotopes). Thus, for the nonrepository (direct disposal) portions of the fuel cycle, the risk is measured in terms of inhalation toxicity (cubic meters of air needed to dilute the radioactivity to acceptable levels) per unit mass of a material times the amount of each material (summed over all materials). For this part of the fuel cycle, water releases are assumed to be negligible. Values of the results are on the order of 10^{15} to 10^{16}, which is expected for toxicity.

Conversely, the environmental risk for a geological repository lies more in the potential ingestion of radionuclides soluble in water that can infiltrate the groundwater in and around the repository. One of the advantages of geologic repositories is that there are several barriers that the radioactive waste must penetrate to reach a human, so the calculation of potential risk is complex and must be performed carefully. The techniques used (and discussed in Section 2.5.2) to determine environmental risk for the

repository involved calculating dose as a function of time over the next million years. The cumulative dose over this time frame then was calculated and normalized to that of LWR spent fuel leaving interim storage to produce a value between 0 and 5. This was multiplied by the flow of material from different processes to the repository to again produce a fairly large number (on the order of 10^8). It is assumed that air releases were negligible here. Thus, two different environmental metrics, both of which were effectively unitless, were produced. Because of funding limitations, only the environmental risk from air releases is shown in the GUI; however, repository risk hopefully will be included later.

2.5.1. Environmental Risk from Air Releases within the Fuel Cycle

Each portion of the nuclear fuel cycle generates radionuclides that are released into the environment. For simplicity, FCOPT embraced the assumption that a well-constructed nuclear cycle has air releases of nuclides from mining and milling through reprocessing but no significant water releases.[13] The high-level-waste repository produces no significant air releases, only potential releases into the ground water.

Total air releases of all of the important isotopes are counted for each process (representing a facility within the fuel cycle). Because each isotope represents a different degree of risk per curie emitted, the code scales the individual isotopes by their DOE-derived concentration guide (DCG) values from Order 5400.5. These DCG values are the number of cubic meters of air that each curie of a particular isotope must be diluted by to allow a standard person breathing 8400 m^3 of that air per year to receive <100 mrem of exposure. The DCGs are listed in column 3 of each of the following tables. The fourth column represents the total dilution in cubic meters for the total release of each particular radionuclide.

The information presented in the following tables then is scaled by the material flows and generated electricity calculated by the fuel-cycle optimization model. For example, if during a given year the fuel-cycle model calculates that five times the amount shown in the next table for uranium needs to be mined to sustain fuel production, the results of that table are linearly scaled (i.e., multiplied by five). The DCG factors used for the various fuel-cycle processes are listed by section below.[14]

2.5.1.1. Uranium Mining. An open pit mine is assumed to process 360,000 tons per year of uranium ore split equally between higher and lower grade ores containing 0.1% and 0.0015% U_3O_8, respectively (corresponding to 175,500 kg of uranium per year).[15] The important nuclides and their yearly release rate contained in dust from the ore and tailings are listed in Table 2-7.

2.5.1.2. Uranium Milling. The mill is assumed to process 2000 tons per day of uranium ore containing 0.2% U_3O_8 (850,000 kg of uranium per year).[13] The important nuclides and their yearly release rate contained in dust from the ore and tailings are listed in Table 2-8.

TABLE 2-7
RADIONUCLIDE AIR RELEASES FROM MINING OF URANIUM

Radionuclide	Release Rate (Ci/yr)	Toxicity Index (Ci/m^3)	Total Toxicity (m^3)
^{238}U	0.0271	1.0×10^{-13}	2.71×10^{11}
^{232}Th	0.00131	1.0×10^{-14}	1.31×10^{11}
^{222}Rn	500.	3.0×10^{-9}	1.67×10^{11}
		Sum of Toxicity (m^3/yr)=	5.69×10^{11}
		Sum of Toxicity (m^3/kg)=	3.24×10^{6}

TABLE 2-8
RADIONUCLIDE AIR RELEASES FROM MILLING OF URANIUM

Radionuclide	Release Rate (Ci/yr)	Toxicity Index (Ci/m^3)	Total Toxicity (m^3)
^{234}U	0.090	9.0×10^{-14}	1.0×10^{12}
^{235}U	0.0022	1.0×10^{-13}	2.2×10^{10}
^{238}U	0.090	1.0×10^{-13}	9.0×10^{11}
^{226}Ra	0.010	1.0×10^{-12}	1.0×10^{10}
^{230}Th	0.0087	5.0×10^{-14}	1.74×10^{11}
^{234}Th	0.0048	4.0×10^{-10}	1.2×10^{7}
^{210}Pb	0.0087	9.0×10^{-13}	9.67×10^{9}
^{210}Po	0.0087	1.0×10^{-12}	8.7×10^{9}
^{210}Bi	0.0087	6.0×10^{-11}	1.45×10^{7}
^{222}Rn	5800.	3.0×10^{-9}	1.93×10^{12}
		Sum of Toxicity (m^3/yr)=	4.05×10^{12}
		Sum of Toxicity (m^3/kg)=	4.78×10^{6}

2.5.1.3. Uranium-Fluorination Plant. This plant converts yellowcake to uranium hexafluoride with a capacity of 10,000 metric tons per year (8,500,000 kg of uranium per year).[13] The important nuclides and their yearly release rate from the fluorination facility are listed in Table 2-9.

2.5.1.4. Uranium-Enrichment Plant. This gaseous diffusion plant enriches the uranium stream in the fissile isotope ^{235}U. A capacity of 8.75×10^6 separative work units (SWUs) per year is assumed (16,800,000 kg of uranium per year).[13] A typical pressurized water reactor (PWR) with 4% ^{235}U in the fuel requires approximately 2600 SWU per assembly and uses approximately 43 assemblies per year. The important nuclides and their yearly release rate from the enrichment facility are listed in Table 2-10.

TABLE 2-9
RADIONUCLIDE AIR RELEASES FROM THE URANIUM-FLUORINATION PLANT

Radionuclide	Release Rate (Ci/yr)	Toxicity Index (Ci/m^3)	Total Toxicity (m^3)
^{234}U	0.11	9.0×10^{-14}	1.22×10^{12}
^{235}U	0.0026	1.0×10^{-13}	2.6×10^{10}
^{238}U	0.11	1.0×10^{-13}	1.1×10^{12}
^{226}Ra	0.0013	1.0×10^{-12}	1.3×10^{9}
^{230}Th	0.012	5.0×10^{-14}	2.4×10^{11}
^{234}Th	0.21	4.0×10^{-10}	5.25×10^{8}
^{234}Pa	0.21	2.0×10^{-8}	1.05×10^{7}
^{222}Rn	73.	3.0×10^{-9}	2.43×10^{10}
		Sum of Toxicity (m^3/yr)=	2.61×10^{12}
		Sum of Toxicity (m^3/kg)=	3.08×10^{5}

TABLE 2-10
RADIONUCLIDE AIR RELEASES FROM THE URANIUM-ENRICHMENT PLANT

Radionuclide	Release Rate (Ci/yr)	Toxicity Index (Ci/m^3)	Total Toxicity (m^3)
^{234}U	0.2	9.0×10^{-14}	2.2×10^{12}
^{235}U	0.0075	1.0×10^{-13}	7.5×10^{10}
^{236}U	0.0055	1.0×10^{-13}	5.5×10^{10}
^{238}U	0.033	1.0×10^{-13}	3.3×10^{11}
^{99}Tc	0.54	2.0×10^{-9}	2.7×10^{8}
^{106}Ru	0.0072	3.0×10^{-11}	2.4×10^{8}
$Zn/^{95}Nb$	0.0015	9.0×10^{-10}	1.7×10^{6}
		Sum of Toxicity (m^3/yr)=	2.66×10^{12}
		Sum of Toxicity (m^3/kg)=	1.58×10^{5}

2.5.1.5. Fuel Fabrication Plant. This plant makes reactor fuel with a throughput capacity of 1500 metric tons of uranium per year.[13] The important nuclides and their yearly release rate from the facility are listed in Table 2-11.

2.5.1.6. LWRs. Both PWRs and boiling water reactors (BWRs) having a capacity of 3500 MWt with 80% availability are considered.[13] The principal releases are from roof vents. The important nuclides and their yearly release rate from each model facility are listed in Table 2-12. These are an average of typical PWR and BWR releases.

TABLE 2-11
RADIONUCLIDE AIR RELEASES FROM THE FUEL FABRICATION PLANT

Radionuclide	Release Rate (Ci/yr)	Toxicity Index (Ci/m^3)	Total Toxicity (m^3)
^{234}U	0.15	9.0×10^{-14}	1.67×10^{12}
^{235}U	0.0051	1.0×10^{-13}	5.1×10^{10}
^{236}U	0.0080	1.0×10^{-13}	8.0×10^{10}
^{238}U	0.019	1.0×10^{-13}	1.9×10^{11}
^{231}Th	0.0051	2.0×10^{-8}	2.55×10^{5}
^{234}Th	0.019	4.0×10^{-10}	4.75×10^{7}
^{234}Pa	0.019	2.0×10^{-8}	9.5×10^{5}
		Sum of Toxicity (m^3/yr)=	1.99×10^{12}
		Sum of Toxicity (m^3/kg)=	1.33×10^{6}

TABLE 2-12
RADIONUCLIDE AIR RELEASES FROM LWRs

Radionuclide	Release Rate (Ci/yr)	Toxicity Index (Ci/m^3)	Total Toxicity (m^3)
^{14}C	25	5.0×10^{-7}	5.0×10^{7}
^{85m}Kr	19.5	1.0×10^{-7}	1.95×10^{8}
^{85}Kr	870	3.0×10^{-6}	2.9×10^{8}
^{88}Kr	2461	9.0×10^{-9}	2.73×10^{11}
^{131m}Xe	187	2.0×10^{-6}	9.35×10^{7}
^{133m}Xe	98.5	6.0×10^{-7}	1.64×10^{8}
^{133}Xe	26,350	5.0×10^{-7}	5.27×10^{10}
^{135}Xe	204	8.0×10^{-8}	2.55×10^{9}
^{131}I	0.0068	4.0×10^{-10}	1.7×10^{7}
^{133}I	0.0248	2.0×10^{-9}	1.24×10^{7}
		Sum of Toxicity (m^3/yr)=	3.29×10^{11}
		Sum of Toxicity (m^3/kg)=	3.56×10^{8}

2.5.1.7. Fuel Reprocessing Plant. This plant reprocesses spent reactor fuel with a capacity of 1500 metric tons per year.[13] Gases are released during the shear and subsequent dissolution steps and are assumed to be vented up a stack. The important nuclides and their yearly release rate from the facility are listed in Table 2-13.

2.5.2. Environmental Risk from Repositories

The repository model takes the high-level waste generated by the fuel cycle and, by inspecting the isotopic composition, uses a simulation code based on Yucca Mountain characteristics to estimate the total exposure (in millirems) to an inhabitant of the surrounding valley from ground water intrusion over 1 million years.[16] This is referenced as repository risk in this report.

In particular, to evaluate different waste stream inventories, the Simplified Total System Performance Assessment (STSPA) model based on GoldSim, Version 6.04.000 was used.[17] GoldSim is an object-oriented computer model developed by Golder Associates. The STSPA model is a simplified version of the performance assessment model (TSPA-VA) being developed for the proposed nuclear waste repository at Yucca Mountain, Nevada. As such, it should not be considered an official and accurate evaluation of the risk associated with disposal of nuclear waste at Yucca Mountain. Rather, the software is more a tool to improve the transparency and understanding of the performance assessment model itself.

TABLE 2-13
RADIONUCLIDE AIR RELEASES FROM THE FUEL REPROCESSING PLANT

Radionuclide	Release Rate (Ci/yr)	Toxicity Index (Ci/m^3)	Total Toxicity (m^3)
^{3}H	7.7×10^{5}	2.0×10^{-2}	3.85×10^{7}
^{14}C	700	5.0×10^{-7}	1.4×10^{9}
^{85}Kr	1.4×10^{7}	3.0×10^{-6}	4.67×10^{12}
^{129}I	2.5	7.0×10^{-11}	3.6×10^{10}
^{131}I	35	4.0×10^{-10}	8.75×10^{10}
^{106}Ru	5.7	3.0×10^{-11}	1.9×10^{11}
^{90}Sr	0.23	5.0×10^{-11}	4.6×10^{9}
^{134}Cs	0.66	2.0×10^{-10}	3.3×10^{9}
^{137}Cs	0.32	4.0×10^{-10}	8.0×10^{8}
^{154}Eu	0.035	5.0×10^{-11}	7.0×10^{8}
^{238}Pu	0.021	4.0×10^{-14}	5.25×10^{11}
^{240}Pu	0.0036	4.0×10^{-14}	9.0×10^{10}
^{241}Pu	0.77	2.0×10^{-12}	3.85×10^{11}
		Sum of Toxicity (m^3/yr)=	5.99×10^{12}
		Sum of Toxicity (m^3/kg)=	4.53×10^{6}

The STSPA model consists of numerous distinct modules: the infiltration model, the climate model, the biosphere model, repository model, the unsaturated zone flow and transport models, and the saturated zone flow and transport models. Although there are over 200 radionuclides in the Yucca Mountain waste inventory, the TSPA-VA has significantly reduced the number of radionuclides that must be considered for post-closure performance. The TSPA-VA has reduced the number of radionuclides to nine by taking into consideration the decay of radionuclides with short half-lives, sorption characteristics, and low biosphere dose conversion factors. The nine radionuclides are listed in Table 2-14, along with their half-lives.

Monte Carlo simulations were performed for several hundred system parameters. In the calculations presented below, 100 realizations were evaluated over a time span of 1 million years. Thirteen different sample waste inventories were evaluated. These values were obtained from sample systems irradiated using the codes ORIGEN2[18] and Monteburns[19] and may not be representative of every type of system in each category. The inventory types used in the simulations are listed in Table 2-15. The results are shown in Fig. 2-5, which shows a plot of mean dose for each material assuming 8.1 metric tons of heavy metal per package over 1 million years. The inventory identified as Min ACT was not simulated because it contains radionuclides not included in the inventory list in the current version of the code. It is important to note that because the radionuclide inventory is restricted to the species listed in Table 2-15, it is not possible to account for other radionuclides that are present in spent fuel or other waste streams and could yield a significant contribution to the dose rate. Furthermore, the same heat loading is used for all waste inventories based on that of spent fuel. Differences in heat production caused by radioactive decay resultingrom the different inventories are not accounted for in the simulations.

Results for the mean dose rate for each fuel inventory (in units of millirems per year) up to 1 million years are shown in Fig. 2-5 using the preliminary STSPA model. Peaks in the profiles are caused by a stochastic description of canister failure in the model. The

TABLE 2-14
RADIONUCLIDES AND HALF-LIVES USED IN THE STSPA MODEL

Radionuclides	Half-Life [Years]
^{14}C	5.73e3
^{129}I	1.57e7
^{237}Np	2.14e6
^{231}Pa	3.25e4
^{239}Pu	2.41e4
^{242}Pu	3.75e5
^{79}Se	6.50e4

^{99}Tc	2.13e5
^{234}U	2.46e5

TABLE 2-15
EXPLANATION OF WASTE INVENTORIES

CS	Cooling storage (assumed 4 years)
IS	Interim storage (assumed 10 years beyond cooling storage)
OTLWR	OT LWR, with full core of UO_2 fuel (after both CS and IS)
MOXLWR	MOX-fueled LWR, with partial UO_2 fuel core and partial-core MOX fuel loading (after both CS and IS)
HTGR	High-temperature, gas-cooled reactor (after both CS and IS)
FBR	Fast breeder reactor (after both CS and IS)
ATW	Accelerator transmutation of waste, lead bismuth
MS	Molten salt ATW
Min ACT	Minor actinides – not a large-enough effect to show a result here
Rep. FPs	Waste products from reprocessing (includes both minor actinides and fission products)
Sep. FPs	Waste products from separations (includes only specific fission products—excludes minor actinides and ^{99}Tc and ^{129}I)

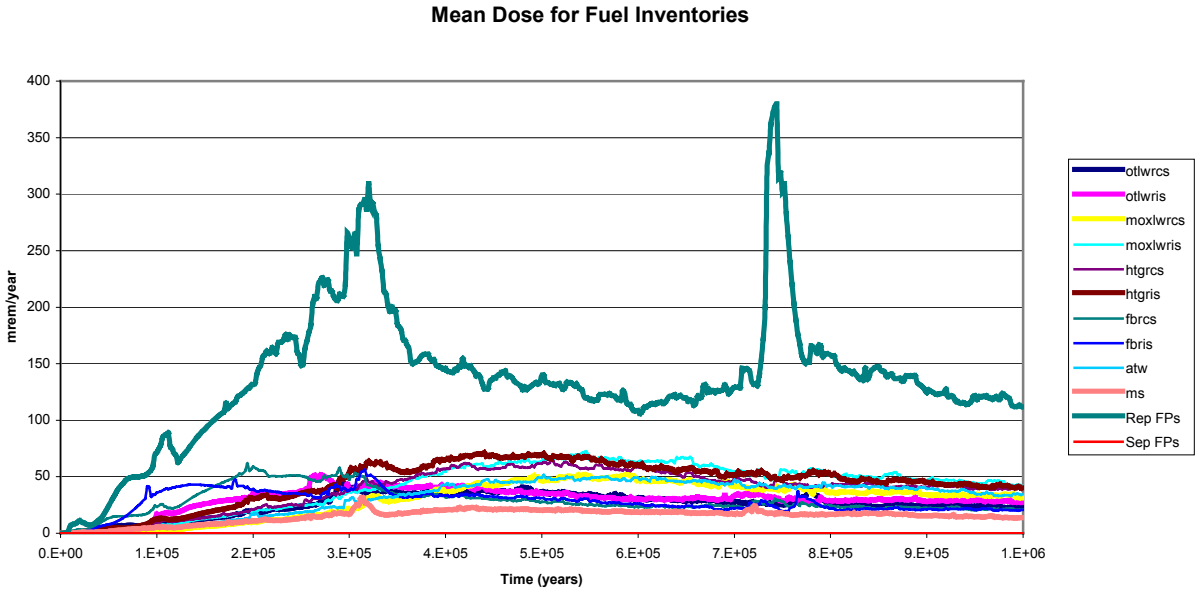

Fig. 2-5. Mean dose from inventories listed in Table 2-16 for 1 million years.

mean dose rate from the Rep FPs material stream is much higher than the other material streams, ranging from 100 to 400 mrem/yr over the time span of the simulations, because of the large amount of neptunium initially present compared to the other inventories (see Table 2-16). Mean dose rates of the remaining 10 inventories show similar behavior, ranging from 20 to 70 mrem/yr after approximately 300,000 years (Fig. 2-6). Separated fission products do not yield as high of a dose rate because

many of the isotopes being examined (^{99}Tc, ^{129}I, and ^{237}Np) are separated along with the actinides and are transmuted in an FSB instead of going to the disposal. Although the actual amount (kilograms) of those isotopes going to a repository may in reality be the same in the form of spent fuel or reprocessed fission products, the dose from a kilogram of reprocessed fission products is greater than that from a kilogram of spent fuel because the weight fraction of isotopes examined in the material stream is greater.

In Fig. 2-7, the contribution from individual radionuclides to the dose rates is shown for a representative inventory, OTLWRIS. Np-237 dominates the dose rate for this and all inventories. Pu-239 is the second most important radionuclide up to approximately 200,000 years, whereas ^{242}Pu becomes the second most important after 200,000 years. I-129, Tc-99, and U-234 rank third, fourth, and fifth, respectively. Se-79 ranks sixth and appears only during the first 400,000 years. C-14 does not contribute to the mean dose rate of this inventory. All other inventories display similar relationships as depicted for OTLWRIS.

To derive an overall risk factor (R) for each radionuclide for a particular inventory, the dose rate (D) for the *i*th radionuclide resulting from the *k*th inventory is integrated over time as (t) in Eq. (2-2):

$$R_{ik} = \int_0^{t_{max}} D_{ik} dt \quad . \tag{2-2}$$

TABLE 2-16
INVENTORIES USED IN THE STSPA MODEL (CI/PACKAGE)[a]

	^{14}C	^{129}I	^{237}Np	^{231}Pa	^{239}Pu	^{242}Pu	^{79}Se	^{99}Tc	^{234}U
OTLWRCS	2.93E-03	3.64E-01	4.06E+00	0	5.03E+03	3.16E+01	4.42E+00	1.40E+02	1.64E-01
OTLWRIS	2.92E-03	3.64E-01	4.13E+00	0	5.03E+03	3.16E+01	4.42E+00	1.40E+02	4.43E-01
MOXLWRCS	1.98E-03	4.44E-01	1.49E+00	0	7.69E+03	9.59E+01	3.66E+00	1.42E+02	7.30E-02
MOXLWRIS	1.97E-03	4.44E-01	1.75E+00	0	7.69E+03	9.59E+01	3.66E+00	1.42E+02	2.13E-01
HTGRCS	8.35E-03	8.66E-01	2.74E+00	0	7.68E+03	9.80E+01	8.10E+00	2.43E+02	2.89E-01
HTGRIS	8.34E-03	8.66E-01	2.89E+00	0	7.68E+03	9.80E+01	8.09E+00	2.43E+02	9.55E-01
FBRCS	5.85E-03	1.07E+00	3.51E+00	0	4.67E+04	1.43E+01	9.50E+00	3.26E+02	2.71E-01
FBRIS	5.84E-03	1.07E+00	3.64E+00	0	4.67E+04	1.43E+01	9.50E+00	3.26E+02	7.51E-01
ATW	1.91E-03	4.46E-01	2.86E-01	0	1.17E+03	8.89E+01	1.13E+00	3.86E+02	1.59E-01
MS	1.07E+00	0	1.15E+00	2.34E-06	1.76E+02	2.76E+01	0	2.10E+02	1.98E-01
Min. Act.	0	0	0	0	0	0	0	0	0
Rep. FPs	6.45E-02	9.29E+00	4.09E+01	0	0	0	8.68E+01	2.88E+03	0
Sep. FPs	6.68E-02	0	0	0	0	0	8.99E+01	0	0

[a]Based on 8.1 metric tons of heavy metal per package and input weight fractions of each isotope in each material.

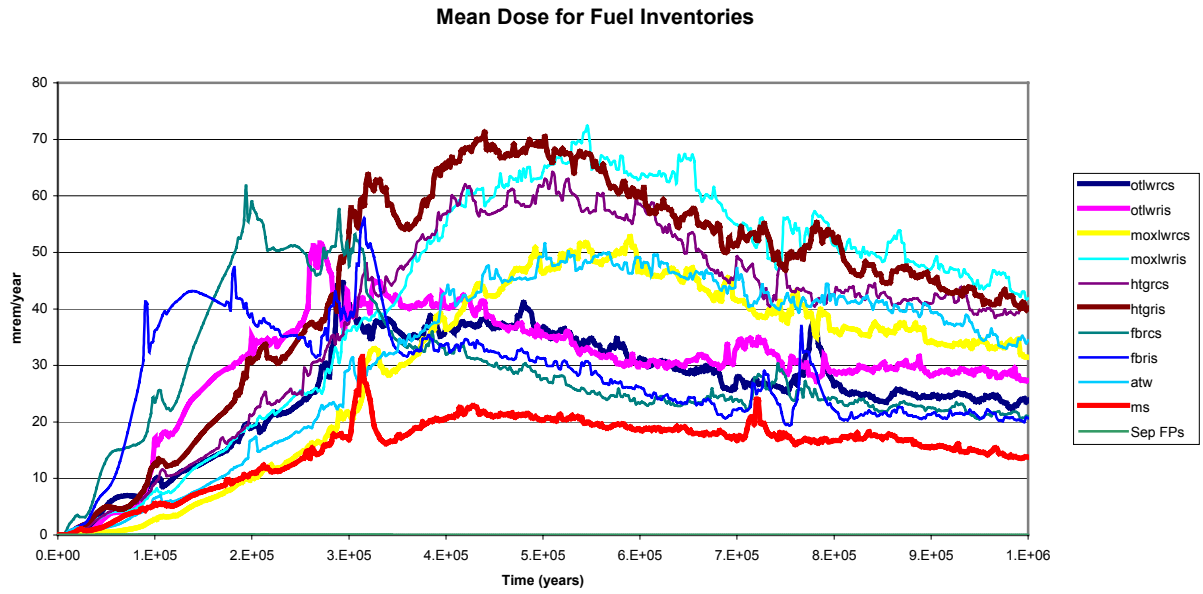

Fig. 2-6. Mean dose from select (important) radionuclides from Fig. 2-5.

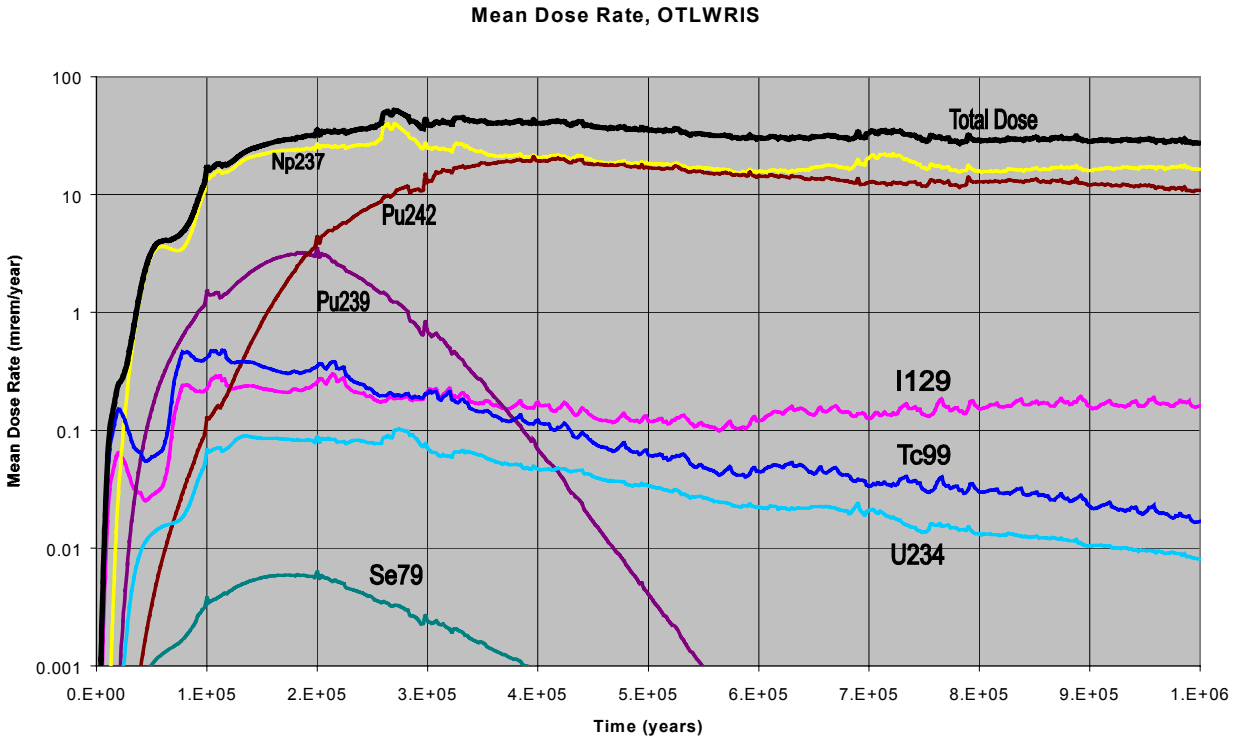

Fig. 2-7. Mean dose rates from each of the nine radionuclides from OTLWRIS.

The risk factor can be used in a decision-making algorithm to evaluate the relative risk for a given waste stream within the nuclear fuel cycle. The risk factor is subject to the same considerations as the dose rate regarding the limited set of radionuclides and constant heat load from which it is derived. Specific values are expected to change as additional radionuclides are added to the Goldsim model and the appropriate heat load is used. Nevertheless, this formulation provides a methodology for comparative risk analysis of different waste streams from the nuclear fuel cycle.

The risk factors normalized to the inventory OTLWRIS are listed in Table 2-17. The inventory Rep FP represents a relative risk 4.5 times greater than that for OTLWRIS. Both high–temperature, gas-cooled reactor inventories present somewhat greater risk than OTLWRIS. The ATW molten salt inventory represents substantially less risk than OTLWRIS. Again, these results should be taken only as a guide because of the approximations in the models, as discussed previously.

TABLE 2-17
RISK FACTORS CALCULATED BY NORMALIZING MEAN DOSE RATES, INTEGRATED OVER 1 MILLION YEARS FOR EACH INVENTORY, TO THE INTEGRATED MEAN DOSE RATE OF OTLWRIS

Inventory	Normalized Risk Factor
OTLWRCS	0.860
OTLWRIS	1.000
MOXLWRCS	0.998
MOXLWRIS	1.410
HTGRCS	1.296
HTGRIS	1.492
FBRCS	0.957
FBRIS	0.934
ATW	1.071
MS	0.508
Sep FPs	0.001
Rep FPs	4.507

3.0. RESULTS FROM OPTIMIZATIONS

The Fuel Cycle Optimization Model (FCOPT) is designed to output a variety of information about the optimal (most cost-efficient) fuel cycle for a given set of input costs. The most straightforward result output is the generation mix: the power produced by each reactor type implemented as a function of time. This generation requirement is driven by an increasing nuclear power demand, as given by the ERB model.[4] Other important results include the COE [both in terms of today's dollars and as present value (PV) COE], proliferation risk, and environmental risk from both air and water releases (each is plotted separately).

To show what type of output is feasible from FCOPT, results are displayed here for a case involving solely OT/LWRs. This case does not involve reprocessing, any type of plutonium reactor, or an FSB. The simple reactor mix in this case appears in Fig. 3-1, the COE in Fig. 3-2, proliferation risk in Fig. 3-3, environmental risk in Fig. 3-4, and repository risk in Fig. 3-5. It is desireable to analyze other reactor types, but this is left to future work.

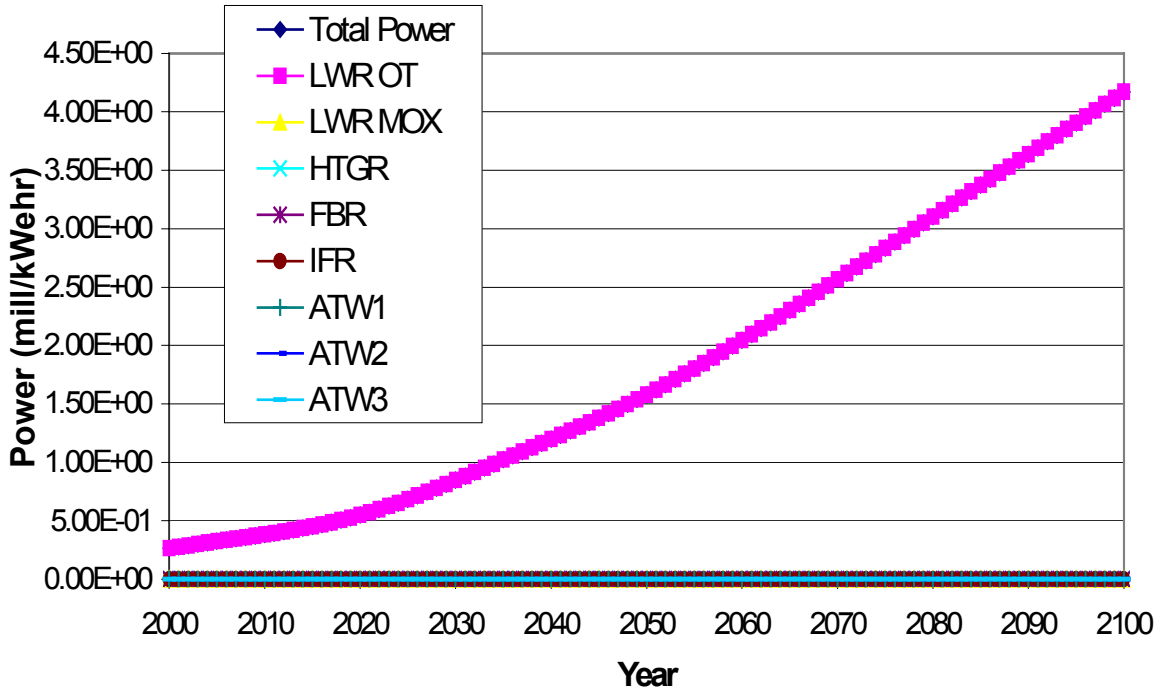

Fig. 3-1. Power generation mix.

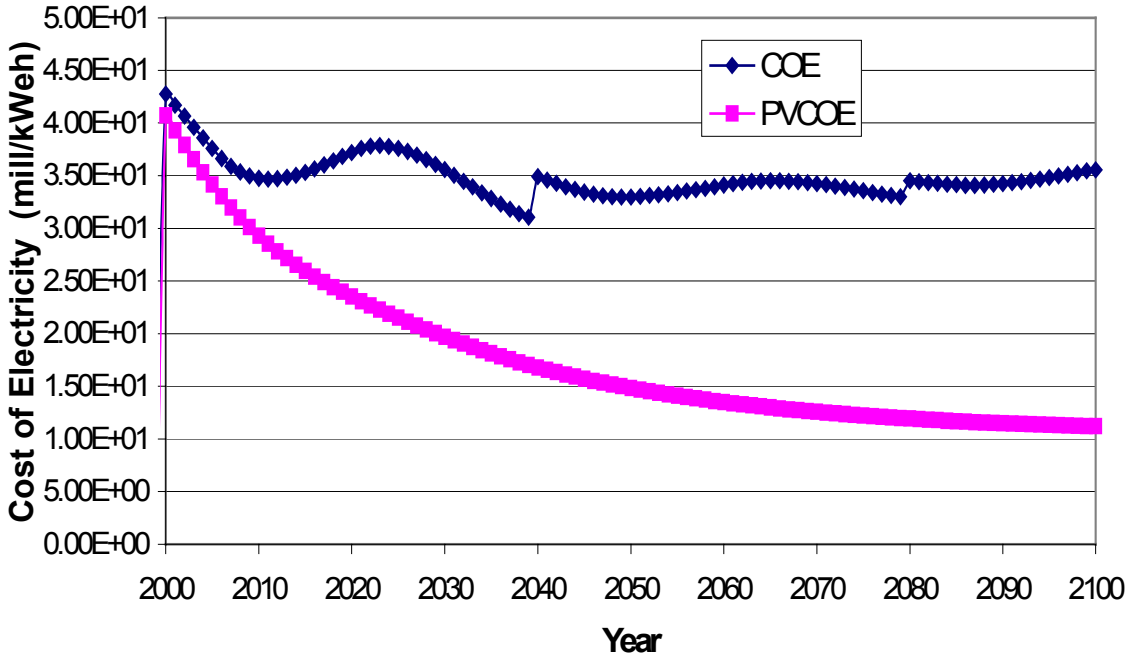

Fig. 3-2. Cost of electricity.

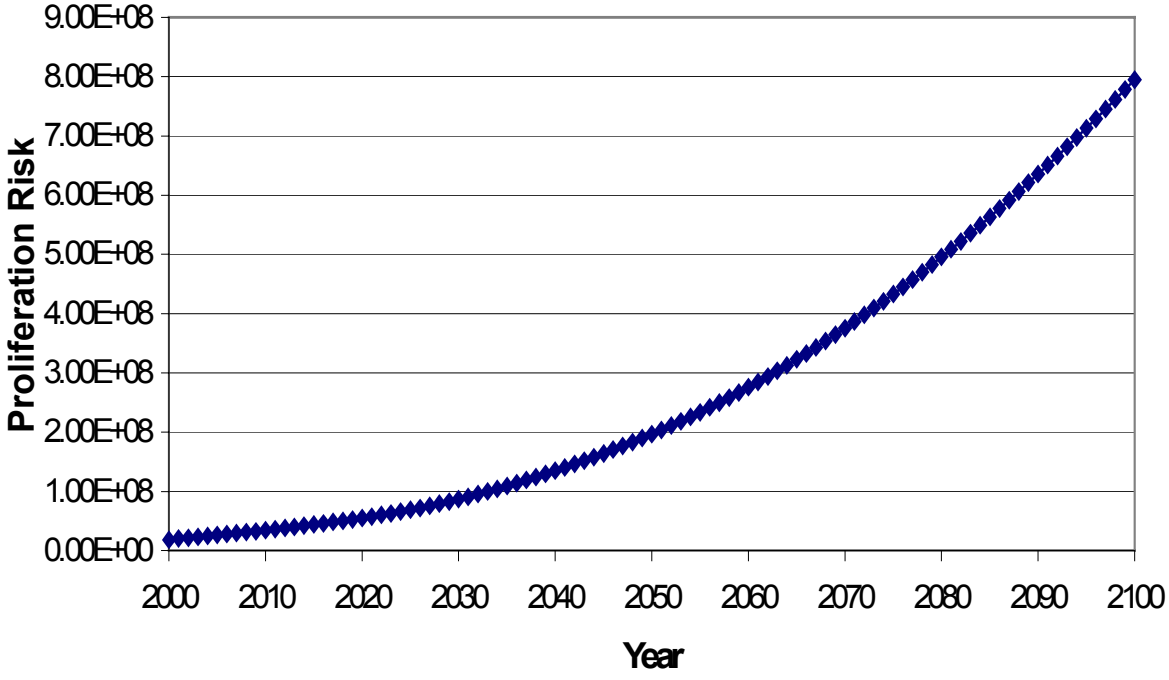

Fig. 3-3. Proliferation risk.

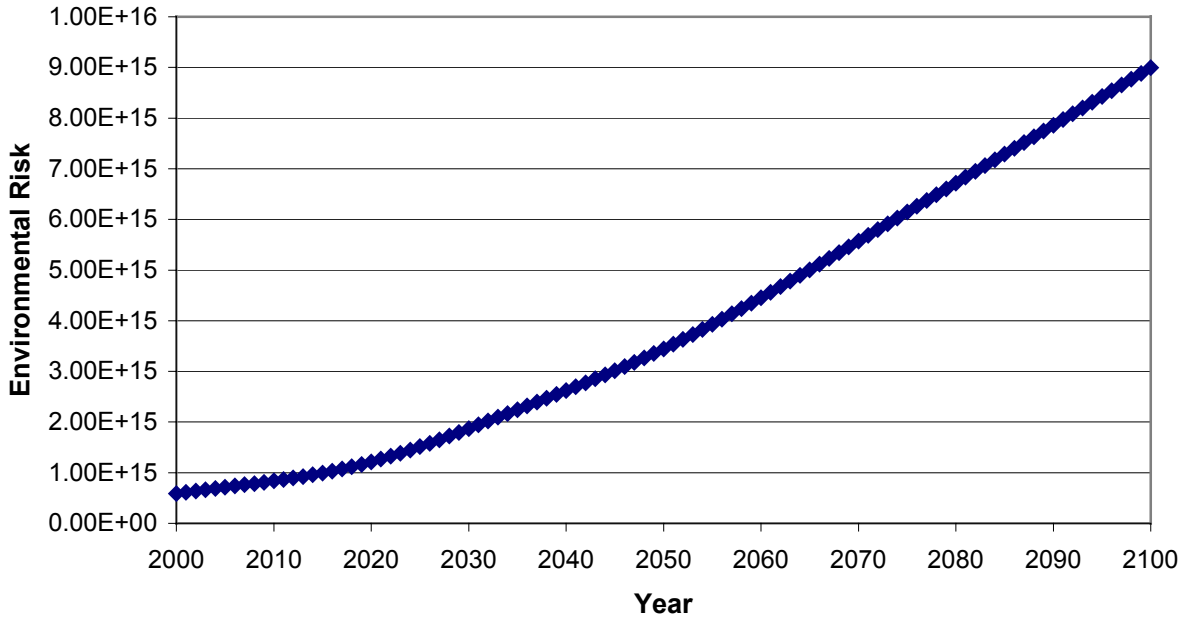

Fig. 3-4. Environmental risk.

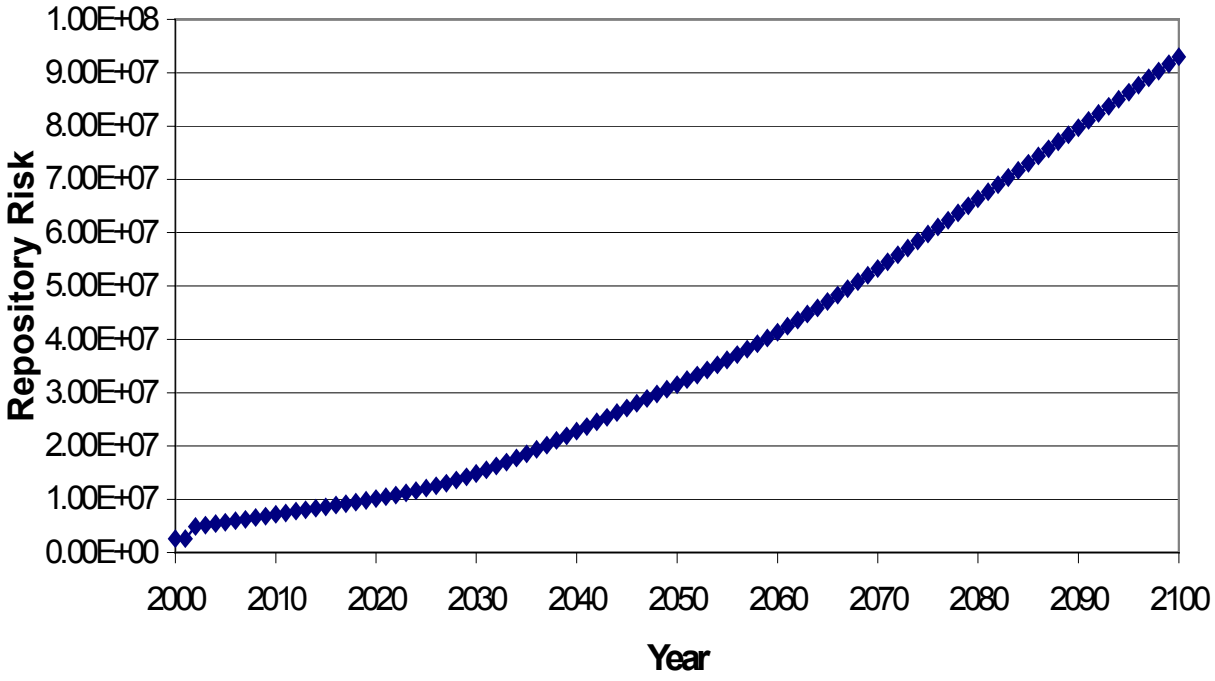

Fig. 3-5. Repository risk.

The COE fluctuates as a function of time; this warrants more analysis to determine why. The PVCOE is smoother, but it decreases as a function of time using an input discounted rate. The proliferation, environmental, and repository risks increase with the total power requirements because more material is present and is processed within the system. As more plutonium from spent fuel is produced, the proliferation risk increases. As more uranium is mined and milled, the environmental risk increases. As more spent fuel is sent to the repository, the repository risk increases. These kinds of analyses can be made for various reactor combinations in future analyses.

The GUI is also capable of showing the above information. On the GUI, the user selects the case of interest (see Fig. 3-6) and then views the resulting fuel cycle (see Fig. 3-7) where the processes that have been implemented are in orange. The costs and proliferation, environmental, and repository risks are also displayed as in Fig. 3-8.

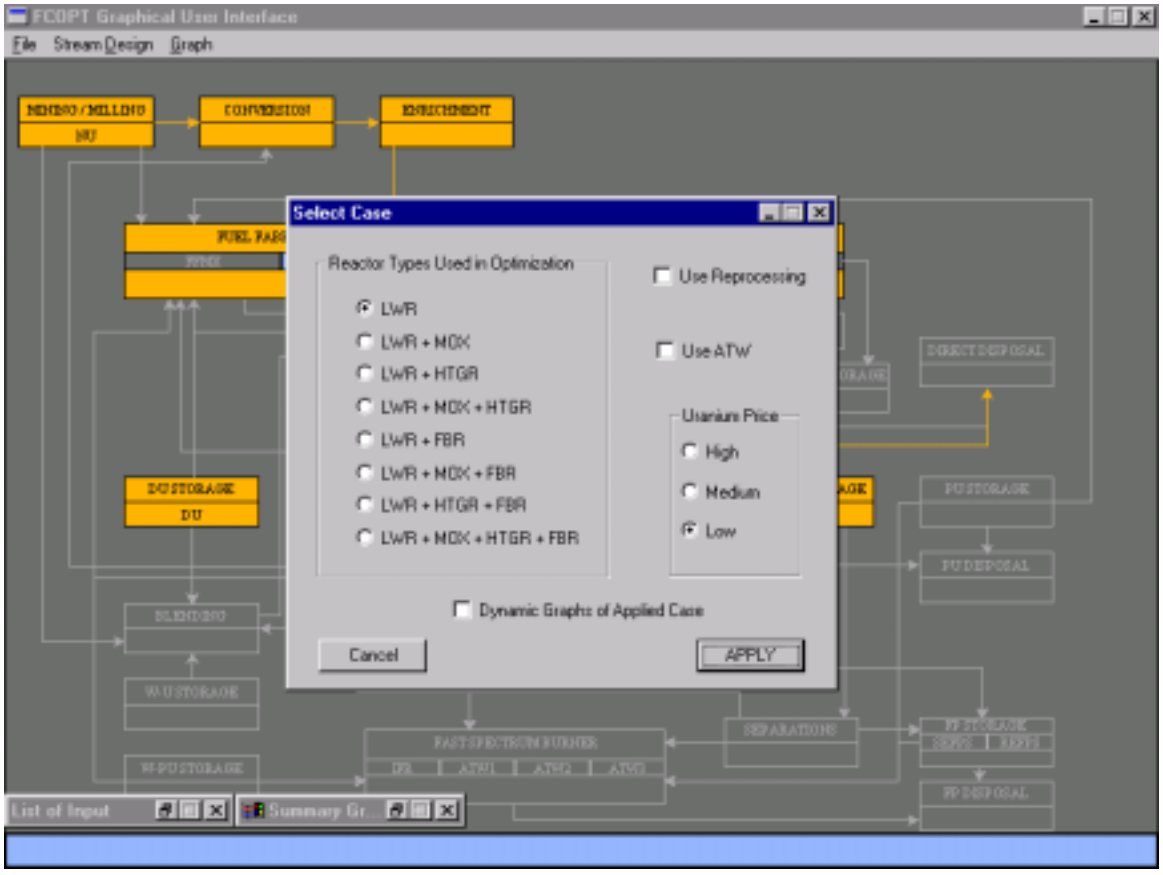

Fig. 3-6. Screen to choose case in GUI.

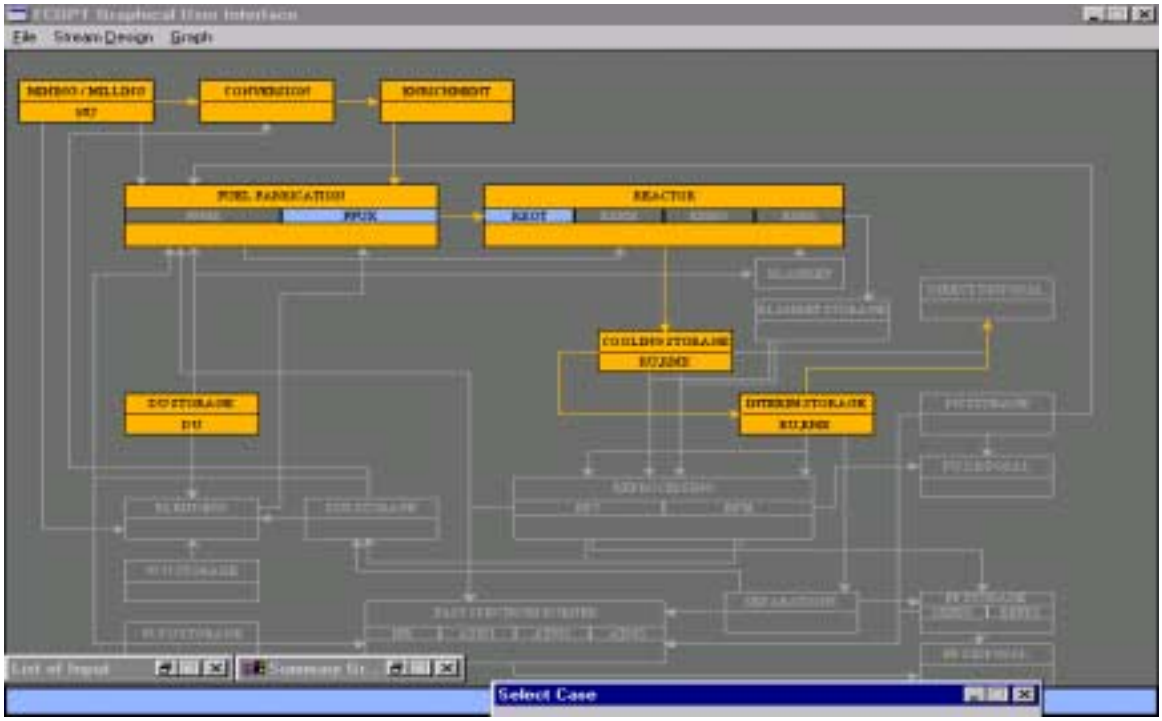

Fig 3-7. Resulting OT/LWR, as shown in the GUI.

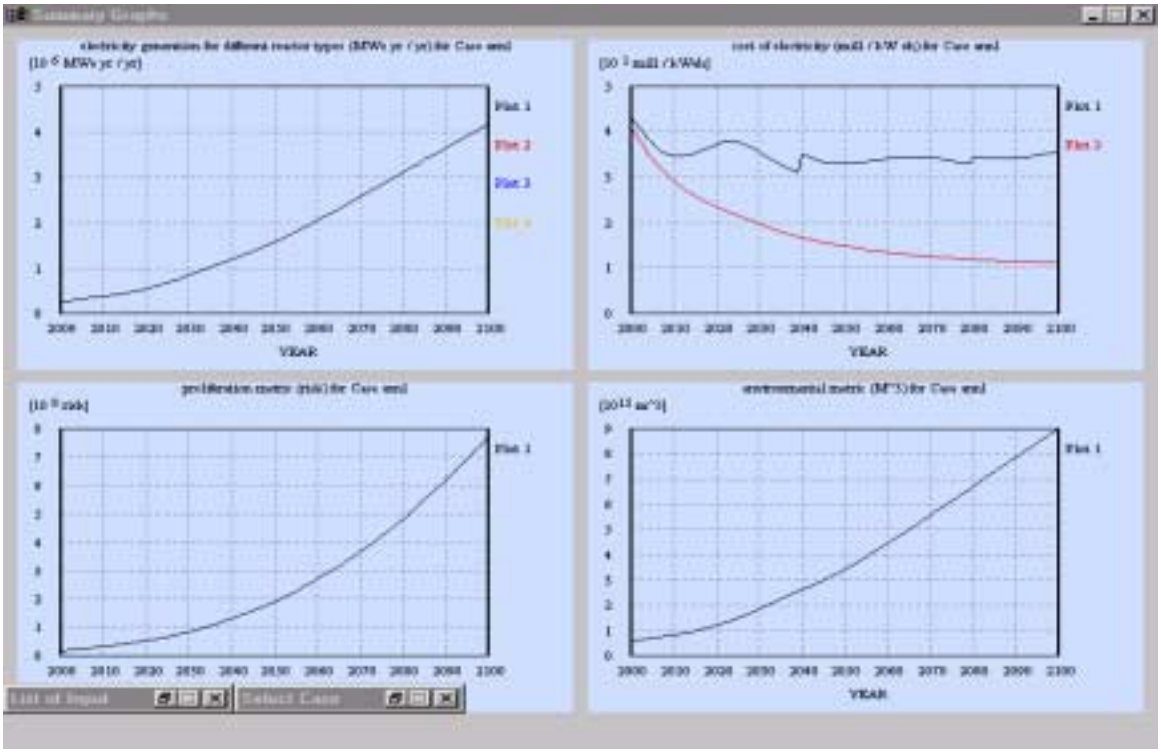

Fig. 3-8. Display of results in the GUI.

Other types of results that can be obtained include the cumulative uranium usage, the associated uranium cost, the material in storage (recycled uranium, depleted uranium, and reprocessed plutonium), and the amount of material (spent fuel from each reactor source, in this case just from OT/LWRs) in cooling storage and direct disposal (assuming that everything from cooling storage goes directly to direct disposal without an interim storage facility). These quantities are shown in Figs. 3-9 to 3-13, respectively. As the uranium requirements increase, so does the cumulative amount of uranium used and the associated cost of uranium. Similarly, as the amount of material going through conversion and enrichment increases, the amount of uranium tailings increases, as does the inventory of material in depleted uranium storage (other storages are unaffected in this case). The LWR spent fuel inventory in both cooling storage and direct disposal also increases as the power requirements do because an incremental amount of spent-fuel is produced each year, adding to the inventory.

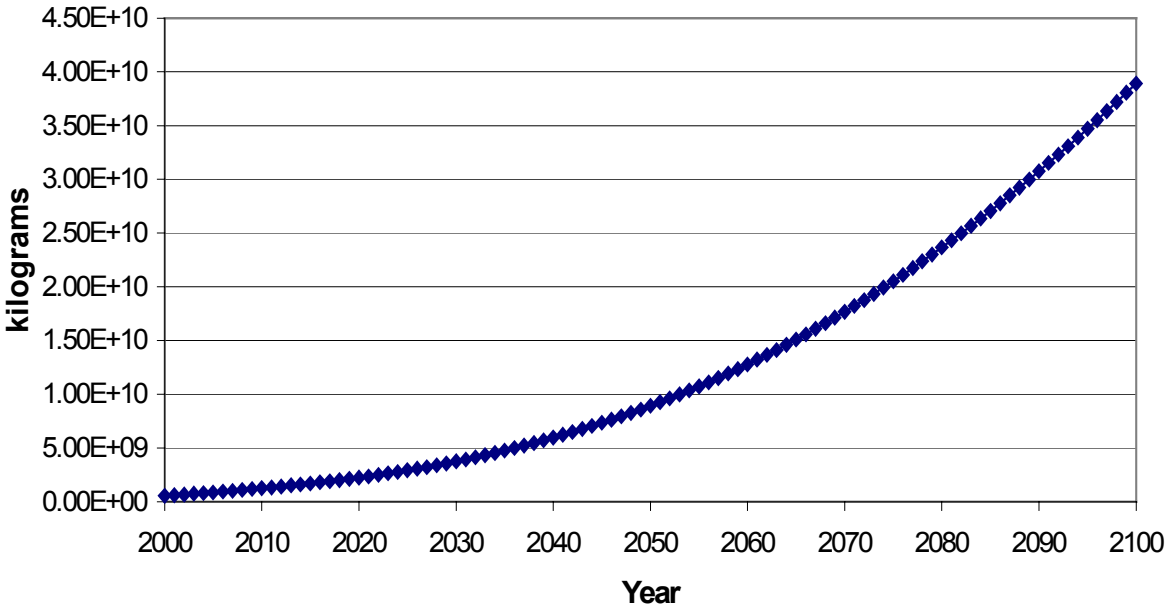

Fig. 3-9. Cumulative uranium usage.

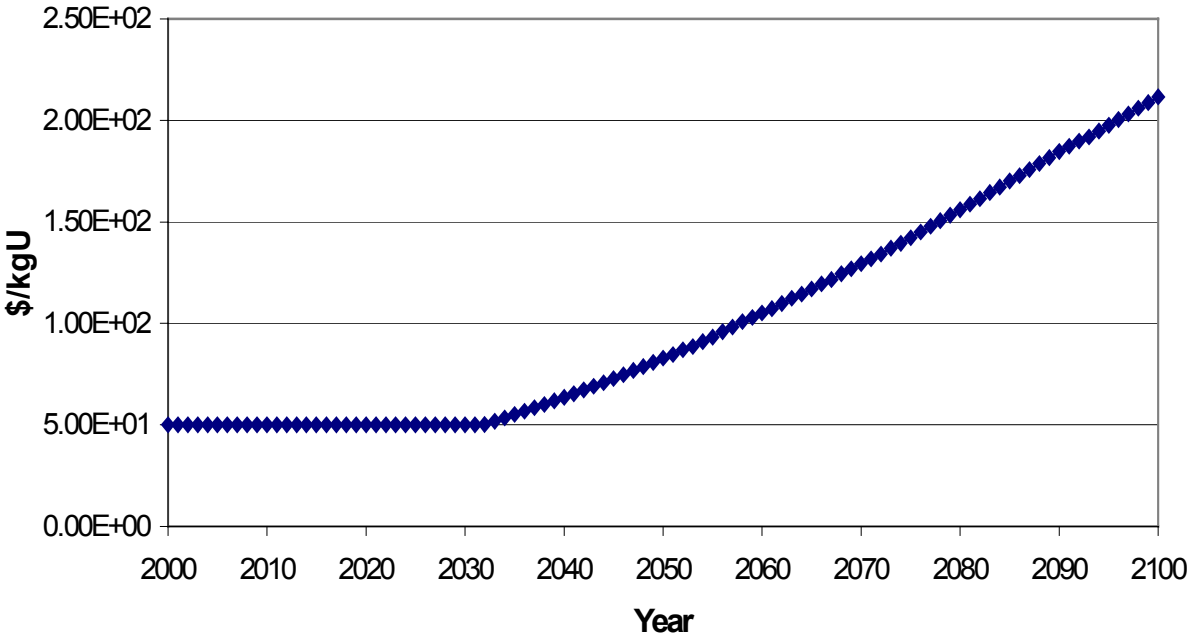

Fig. 3-10. Associated uranium cost.

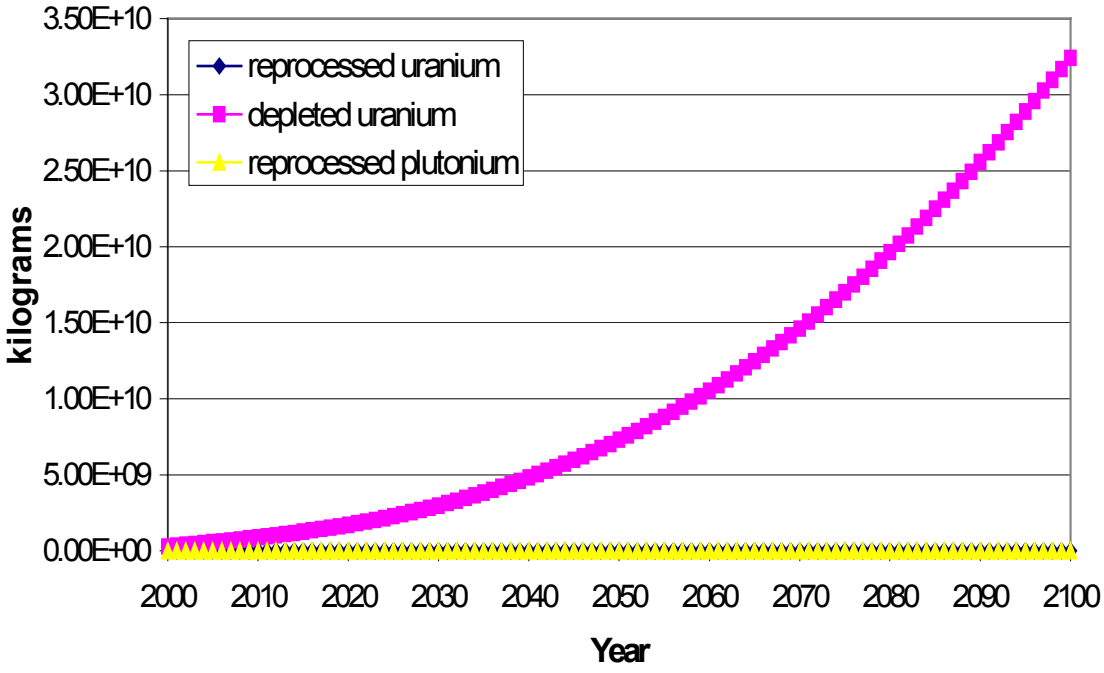

Fig. 3-11. Materials in storage.

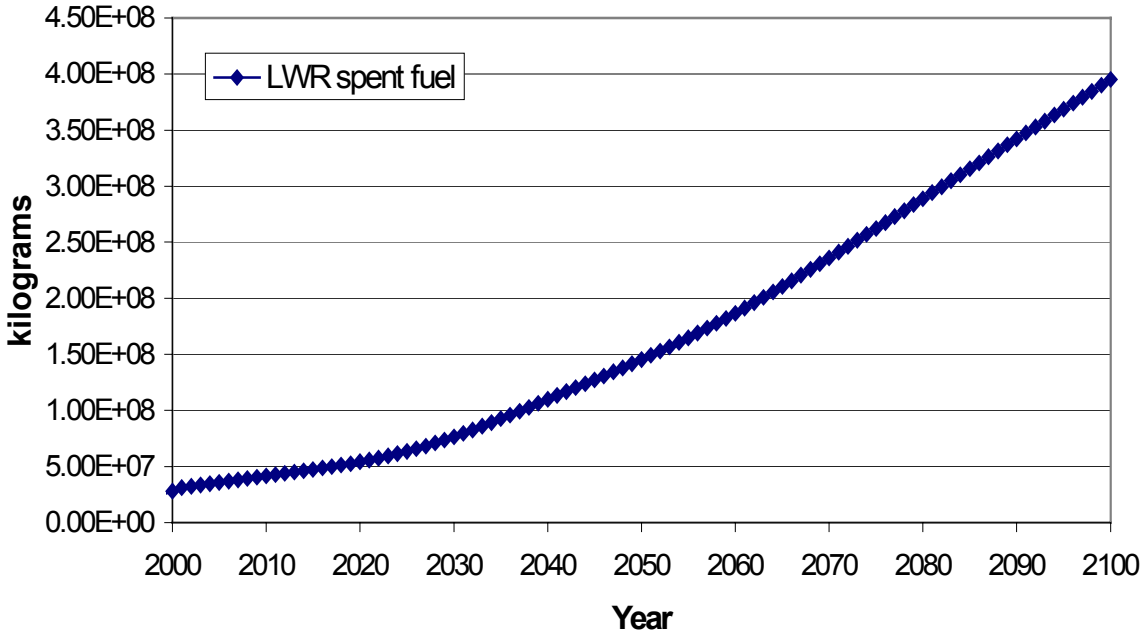

Fig. 3-12. Spent-fuel cooling storage inventory.

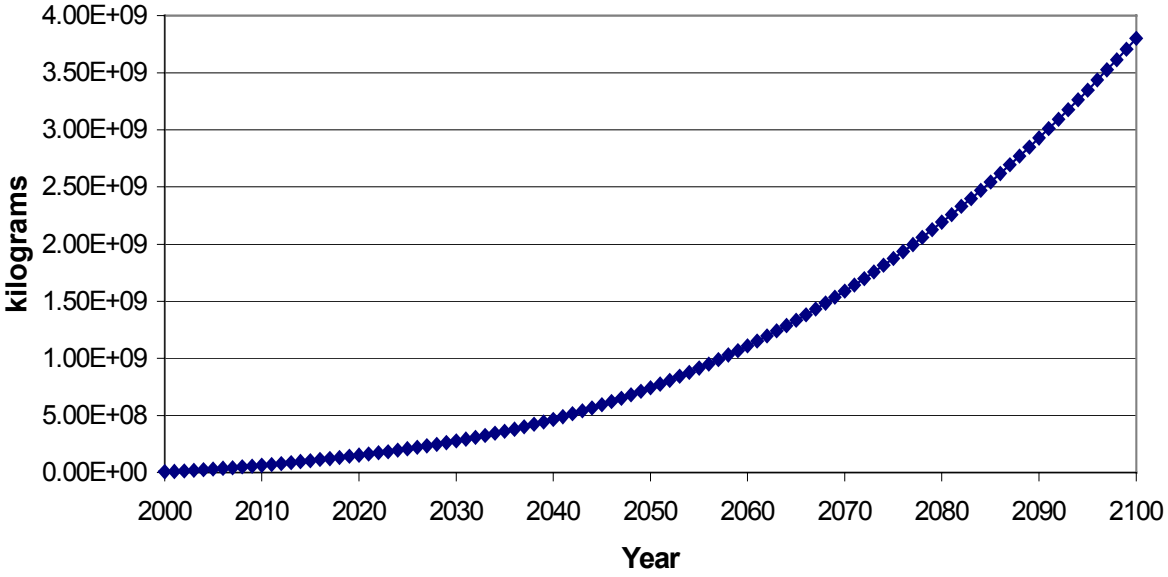

Fig. 3-13. LWR spent fuel direct disposal inventory.

4.0. CONCLUSIONS AND FUTURE WORK

The LDRD-funded NMMS project integrated an array of new analytical methods and novel approaches to nuclear energy and waste disposal into a set of global nuclear energy and materials management system options. Results from this project provide the ability to identify optimal nuclear energy architectures for global nuclear materials management and to assess overall performance in terms of national and international interests, requirements, and resources. The final product should be able to

- identify near-term critical (nuclear) technologies;
- provide information to support near- and long-term deployment of nuclear technologies and nuclear systems in a way that minimizes global proliferation, environmental, and economic risks;
- optimize repository performance; and
- improve the sustainability of future domestic and global energy supplies in a framework wherein nuclear energy competes with all major alternative sources of primary energy.

After 2 years of preliminary development of this project, many conclusions can be reached. First, it was extremely difficult to develop a technique to measure the three different objectives of the project: economics, proliferation, and environment. Each of these three are measured using specific units, and combining the units together was not practical at this point in the project. Thus, the optimization that was supposed to cover all three areas but ended up occurring only for economics. It was difficult to calculate the proliferation metrics; thus, two different metrics were calculated and the old-style multi-attribute utility analysis technique was discarded for a more realistic calculation. However, the results from this calculation do not seem to be intuitive; as plutonium from storage is processed and irradiated, the proliferation risk increases instead of decreases. This is a result of the fact that the material is now in the fuel cycle and goes through several processes and resides in numerous facilities other than just plutonium storage. The overall effect should probably reduce proliferation risk, but under the current implementation, this is not the case. Two different environmental metrics were developed, one for front-end (plus reprocessing) processes and one for the repository. These produced reasonable and expected results and can be used in future expansions of the code. Nonetheless, an integrated method for optimizing all three areas still needs to be developed to strengthen the impact of the results.

There is still much future work to be performed for FCOPT. First, a technique of combining the three objectives (economics, proliferation, environment) must be developed and implemented. Next, even though much work was put into developing the economic optimization of the fuel cycle, the results are still not perfectly smooth,

nor are they easy to explain fully (especially for proliferation risk). More research and studies must be performed to level out the results. Third, the capability to analyze different geological repositories and/or ATW systems must be utilized more fully. These activities can assist efforts other than just fuel-cycle analyses (transmutation of waste concepts and geological repository studies). Finally, the addition of new fuel types and fuel cycles that may decrease environment and proliferation risks should occur to represent fully the spectrum of activities capable of FCOPT and encourage the development of new technologies.

APPENDIX A. CONSTRAINTS

The purpose of the constraints in the FCOPT program is to represent the material flow between various steps in the fuel cycle. The rules for making constraints appear in Ref. 1, but they are fairly fixed from a user's standpoint and appear below.

Mining and Milling

The first constraint deals with the cumulated inventory of natural uranium that has been used, and in this sense differs from all other inventories considered in this model. At any time t, it is equal to the previous inventory (t-1) plus the material that goes to conversion, blending, and fabrication of MOX/LWR fuel (from any of the three plutonium sources).

Constraint 1

x(nu,mm,t) = x(nu,mm,t-1) + x(nu,cv,t) + x(nu,bl,t) + x(nu,ffmxrp,t) + x(nu,ffmxps,t) + x(nu,ffmxwp,t)

Conversion

The conversion constraints deal with the balance of how much natural and recycled uranium (respectively) from the conversion process goes to enrichment (i.e., U_3O_8 to UF_6). The loss of material from conversion is noted here by the parameter "1 - ycv," where ycv is the efficiency of conversion. However, the placement of this "lost" material is not yet accounted for in this model.

Constraint 2

x(nu,er,t) = x(nu,cv,t)*ycv

Constraint 3

x(ru,er,t) = x(ru,cv,t)*ycv

Enrichment

The enrichment Constraints 4 and 5 relate the amount of natural and recycled uranium powder, respectively, going to fuel fabrication to the amount leaving enrichment. The term "yernu" represents the efficiency for nu enrichment (how many tries it takes to enrich the nu to the desired lnu enrichment), "yerru" is the efficiency for ru enrichment, and "1 - yer" is the fraction of material lost during the enrichment process. "Ter" is the time (in years) required for enrichment (the residence time of a material in the facility).

Constraint 4

x(nu,er,t-ter) = x(lnu,ffux,t)/yer/yernu

Constraint 5

x(ru,er,t-ter) = x(lru,ffux,t)/yer/yerru

Fuel Fabrication

The first fuel fabrication constraint involves the uranium dioxide (UO_2) fuel going into OT and the non-MOX/LWR fuel portion of MOX/LWR-recycle LWRs. Note: All of the fuel fabrication equations are expressed in terms of t-tffux or t-tffmx, of which "tffux" and "tffmx" are the lengths of time (in years) it takes to fabricate UO_2 and MOX/LWR fuel, respectively. The parameter "yffux" is the efficiency of UO_2 fabrication. It is assumed that a fraction of MOX/LWR-recycle LWR cores will be UO_2 and the other part will be MOX/LWR fuel. Enriched uranium is needed for this portion, as well as in the full core of an OT LWR. The amount of enriched natural, recycled, and weapons-blended uranium all must be accounted for separately to make the amount of each fabricated correlate appropriately.

Constraint 6

x(lnu,ffux,t-tffux) = (x(lnu,rxot,t) + x(lnu,rxmx,t))/yffux

Constraint 7

x(lru,ffux,t-tffux) = (x(lru,rxot,t) + x(lru,rxmx,t))/yffux

Constraint 8

x(lwu,ffux,t-tffux) = (x(lwu,rxot,t) + x(lwu,rxmx,t))/yffux

The next fuel fabrication constraint deals with the uranium within the MOX/LWR fuel in the MOX/LWR-recycle LWRs. The parameter "mox" is the fraction of fuel that is plutonium in the MOX/LWR fuel itself; the rest (1 - ymox) is UO_2. "yffmx" is the efficiency of MOX/LWR fabrication.

Constraint 9

x(nu,ffmxrp,t-tffmx) + x(nu,ffmxps,t-tffmx) + x(nu,ffmxwp,t-tffmx)
+ x(du,ffmxrp,t-tffmx) + x(du,ffmxps,t-tffmx) + x(du,ffmxwp,t-tffmx)
+ x(ru,ffmxrp,t-tffmx) + x(ru,ffmxps,t-tffmx) + x(ru,ffmxwp,t-tffmx)
= (x(mox(rxbr,t)*(1 – ymox(inbr)))/yffmx
+ (x(mox,rxhg,t)*(1 – ymox(inhg)))/yffmx
+ (x(mox,rxmx,t)*(1 – ymox(inmx)))/yffmx

The following fuel fabrication constraint involves the plutonium within the MOX/LWR fuel in the MOX/LWR recycle LWRs.

Constraint 10

x(rpu,ffmxrp,t-tffmx) + x(rpu,ffmxps,t-ttfmx) + x(wpu,ffmxwp,t-tffmx)
= x(mox,rxmx,t)*ymox(inmx)/yffmx + x(mox,rxbr,t)*ymox(inbr)/yffmx
+ x(mox,rxhg,t)*ymox(inhg)/yffmx

These constraints are generally balanced on the amount of uranium vs plutonium in MOX/LWR fuel made from different plutonium sources [reprocessed plutonium (Pu) from both reprocessing and storage (RP and PS, respectively), and WPU] that keep the plutonium sources separate. The term "fmox" represents the fraction of the fuel that is MOX/LWR and not UO_2.

Constraint 11

x(nu,ffmxrp,t)*fmox(t) + x(du,ffmxrp,t)*fmox(t) + x(ru,ffmxrp,t)*fmox(t)
> x(rpu,ffmxrp,t)*(1 – fmox(t))

Constraint 12

x(nu,ffmxps,t)*fmox(t) + x(du,ffmxps,t)*fmox(t) + x(ru,ffmxps,t)*fmox(t)
> x(rpu,ffmxps,t)*(1 – fmox(t))

Constraint 13

x(nu,ffmxwp,t)*fmox(t) + x(du,ffmxwp,t)*fmox(t) + x(ru,ffmxwp,t)*fmox(t)
> x(wpu,ffmwp,t)*(1 – fmox(t))

Blending

The first blending constraint defines how much HEU must be blended with NU, DU, and RU to get a final blended stream. The term "yuox" is the concentration (weight fraction) of the feed material in parentheses as defined in "fc.dat". The amount coming out is ratioed to that coming in from natural, recycled, depleted, and highly enriched uranium.

Constraint 14

x(heu,bl,t) = x(nu,bl,t)*(yuox(lnu) - yuox(nu))/(yuox(heu) - yuox(nu))
+ x(ru,bl,t)*(yuox(lru) - yuox(ru))/(yuox(heu) - yuox(ru))
+ x(du,bl,t)*(yuox(ldu) - yuox(du))/(yuox(heu) - yuox(du))

The next blending constraint defines how much material comes out of the blending step based on the fraction and masses of material that went in.

Constraint 15

x(lwu,ffux,t)*yuox(lnu) = x(heu,bl,t-tbl)*yuox(heu) +
x(ru,bl,t-tbl)*yuox(ru) + x(du,bl,t-tbl)*yuox(du)

Depleted Uranium Storage

The dus constraint is a balance of the inventory of du that accumulates in storage. It is equal to the inventory at the previous timestep (t-1) plus that which comes in as losses

from the enrichment process minus what goes to MOX/LWR fuel fabrication (from all three plutonium sources), the blanket, and blending.

Constraint 16

x(du,dus,t) = x(du,dus,t-1) + x(nu,er,t-ter)*(1 - yernu)*yer
+ x(ru,er,t-ter)*(1 - yerru)*yer - x(du,ffmxrp,t) - x(du,ffmxps,t)
- x(du,ffmxwp,t) - x(du,bk,t) - x(du,bl,t)

Weapons Uranium Storage

This constraint describes a balance on the inventory of material in weapons uranium storage. The current inventory is equal to the previous inventory plus what is released from the weapons programs into the commercial fuel cycle (this amount is the exogenous release rate, gheu) minus what is sent to blending. The amount of weapons-grade uranium (heu) that can be released in any given year is regulated by policy issues outside the scope of this model. The inventory of heu tracked here is only that which has entered the commercial reactor stream and does not include heu stored at weapons facilities. However, the expression used to describe the rate "gheu" parametrically is constrained by estimates of the total amount of heu available for release to commercial use.

Constraint 17

x(heu,wus,t) = x(heu,wus,t-1) + gheu - x(heu,bl,t)

Weapons Plutonium Storage

As with heu storage, there is an exogenous release rate, "gwpu," for wps so that only a limited quantity can be introduced into MOX/LWR fuel fabrication at once. Similar to heu, this corresponds to policy issues that are outside the scope of this model. The current inventory of weapons plutonium in storage is equal to what was there previously plus gwpu minus what goes to the FSB and fuel fabrication. Plutonium going to the FSB technically goes through the separations box, although there is no cost associated there.

Constraint 18

x(wpu,wps,t) = gwpu + x(wpu,wps,t-1) - x(wpu,sep,t) - x(wpu,ffmxwp,t)

Separations

Separations technology involves the separation of spent fuel into uranium (which goes to uranium storage),other actinides [for the FSB(s)], and fission products (long-lived ones may go to the FSB; others go to storage and/or disposal). This varies from "reprocessing" because minor actinides are used as "fuel," not sent to disposal. ATWs take this spent fuel only for fuel; IFRs take RPU and WPU from storage as well and process it into a fuel form for IFRS. The

following constraints show the balances on the plutonium and spent fuel, where "ysep" is the separations efficiency.

Constraint 19
x(wpu,sep,t)*ysep = x(wpu,ifr,t)

Constraint 20
x(rpu,sep,t)*ysep = x(rpu,ifr,t)

Constraint 21
x(ru,sep,t)*othsep(inot)*ysep = x(ru,ifr,t) + x(ru,atw1,t) + x(ru,atw2,t) + x(ru,atw3,t)

Constraint 22
x(rmx,sep,t)*othsep(inmx)*ysep = x(rmx,ifr,t) + x(rmx,atw1,t) + x(rmx,atw2,t)
+ x(rmx,atw3,t)

Constraint 23
x(rhtg,sep,t)*othsep(inhg)*ysep = x(rhtg,ifr,t) + x(rhtg,atw1,t) + x(rhtg,atw2,t)
+ x(rhtg,atw3,t)

Constraint 24
x(rfbr,sep,t)*othsep(inbr)*ysep = x(rfbr,ifr,t) + x(rfbr,atw1,t) + x(rfbr,atw2,t)
+ x(rfbr,atw3,t)

Irradiation

The total amount of power generated by all reactor types must equal the total amount of electrical energy that must be generated each year by all reactor types. This power balance is represented in the following constraint, where gent(t) is the total, exogenously specified demand for nuclear energy, as determined for a range of global energy scenarios,[3] and "gen(reactor)" is the annual generation rate (megawatt electric year per year) for each reactor.

Constraint 25
gent(t) = gen(rxot,t) + gen(rxmx,t) + gen(rxhg,t) + gen(rxbr,t) + gen(rifr,t) +
gen(atw1,t) + gen(atw2,t) + gen(atw3,t)

The amount of fuel needed to power an OT reactor (the "load") is equal to the amount of power generated by that reactor in a given timestep divided by the term "alf", where:

$$alf(rx,t) = \eta_{th}(1-\varepsilon) * Bu/(days/year) \qquad \text{(A-1)}$$

where

η_{th} = thermal conversion efficiency,
ε = recirculation power fraction—the fraction of power needed to operate the reactor), and
Bu = burnup ($MW_t d/kg$ HM).

These terms consider that fuel is kept in the reactor for a certain period of time (the irradiation period); only the amount of fuel to replenish the reactor is needed. It is then assumed that this replenishing amount will move to cooling storage after irradiation.

Constraint 26
gen(rxot,t)/alf(inot,t) = x(lwu,rxot,t) + x(lnu,rxot,t) + x(lru,rxot,t)

The same is true for the other reactor types with their respective fuels, with the exception of the fast burner, which will be discussed later.

Constraint 27
gen(rxmx,t)/alf(inmx,t) = x(lwu,rxmx,t) + x(lnu,rxmx,t) + x(lru,rxmx,t) + x(mox,rxmx,t)

Constraint 28
gen(rxhg,t)/alf(inhg,t) = x(mox,rxhg,t)

Constraint 29
gen(rxbr,t)/alf(inbr,t) = x(mox,rxbr,t)

Constraint 30
x(bmx,bks,t) + x(bmx,rpcbks,t) + x(bmx,rpmbks,t) = x(bmx,bks,t-1) + gen(rxbr,t-trx(inbr))*br/alf(inbr,t-trx(inbr))

Discharge from FSB

This shows the amount of minor actinides, short-lived and long-lived fission products, respectively, that leave the FSB with the amount of material going to disposal from the burner (this assumes all products from the FSB will go directly to disposal). For the FSBs,

$$alf(rx,t) = \eta_{th}(1-\varepsilon) * N_A * Q * e / A / (\text{sec}onds / year) \quad \text{(A-2)}$$

where

η_{th} = thermal conversion efficiency,
ε = recirculation power fraction - the fraction of power needed to operate,
N_A = Avogadro's number ($6.022*10^{23}$ atoms/mole),
Q = energy per fission (200 MeV),
e = electric charge ($1.602*10^{-19}$ J/eV), and

A = average mass of actinides in burner (assume 239 g).

Constraint 31

x(ru,ifr,t)*yfsb + x(rhtg,ifr,t)*yfsb + x(rfbr,ifr,t)*yfsb + x(rmx,ifr,t)*yfsb + x(rpu,ifr,t) + x(wpu,ifr,t) = gen(rifr,t)/alf(inif,t)

Constraint 32

x(ru,atw1,t)*yfsb + x(rhtg,atw1,t)*yfsb + x(rfbr,atw1,t)*yfsb + x(rmx,atw1,t)*yfsb = gen(atw1,t)/alf(ina1,t)

Constraint 33

x(ru,atw2,t)*yfsb + x(rhtg,atw2,t)*yfsb + x(rfbr,atw2,t)*yfsb + x(rmx,atw2,t)*yfsb = gen(atw2,t)/alf(ina2,t)

Constraint 34

x(ru,atw3,t)*yfsb + x(rhtg,atw3,t)*yfsb + x(rfbr,atw3,t)*yfsb + x(rmx(atw3,t)*yfsb = gen(atw3,t)/alf(ina3,t)

Vintaging for Reactor Costs

Constraints 35-42

The main addition of vintaging included defining a new variable that reflects a reactor's yearly addition to capital stock and then limiting the rate at which the annual deployment of that reactor can occur each year. This annual addition of generating capacity or inventory is represented in the constraints as gen(inrx,t) for the inventory associated with a certain generation technique and has the units of capacity (**MWe**). The capacity at a given time step gen(rx,t) is equal to the sum of the annual addition of inventories for the past t-tlife years (where tlife is the lifetime of the reactor) divided by the availability of that reactor type. Also, to make the problem easier to "start," a residual inventory is set for the 40 years before the problem and diminishes to zero over the first 10 to 15 years (unless that reactor was nonexistent previously and the residual inventories remain zero for all time steps). Under these conditions, the capacity of rx at time t is given by

$$\mathrm{gen}(\mathrm{rx},\mathrm{t}) = \mathrm{resid}(\mathrm{rx},\mathrm{t}) + \sum_{\mathrm{i}=\mathrm{t}-\mathrm{tlife}+1}^{\mathrm{t}} \mathrm{gen}(\mathrm{inrx},\mathrm{i}) * \mathit{avail}(\mathit{inrx}) \quad .$$

Discharge from IFR

These constraints account for the minor actinides, short-lived and long-lived fission products that are being produced by IFRs powered on RU, RMX, RHTG, and RFBR (byproducts of using RPU and WPU are not included here). These fission products are sent directly to FPD. Similar equations follow for the three types of ATW.

Constraint 43

x(ma,fpdif,t) = fpdma*x(ru,ifr,t-trx(inif)) + fpdma*x(rmx,ifr,t-trx(inif))
+ fpdma*x(rhtg,ifr,t-trx(inif)) + fpdma*x(rfbr,ifr,t-trx(inif))

Constraint 44

x(fp,fpdif,t) = fpdfp*x(ru,ifr,t-trx(inif)) + fpdfp*x(rmx,ifr,t-trx(inif))
+ fpdfp*x(rhtg,ifr,t-trx(inif)) + fpdfp*x(rfbr,ifr,t-trx(inif))

Constraint 45

x(lfp,fpdif,t) = fpdlfp*x(ru,ifr,t-trx(inif)) + fpdlfp*x(rmx,ifr,t-trx(inif))
+ fpdlfp*x(rhtg,ifr,t-trx(inif)) + fpdlfp*x(rfbr,ifr,t-trx(inif))

Discharge from ATW1

Constraint 46

x(ma,fpda1,t) = fpdma*x(ru,atw1,t-trx(ina1))
+ fpdma*x(rmx,atw1,t-trx(ina1)) + fpdma*x(rhtg,atw1,t-trx(ina1))
+ fpdma*x(rfbr,atw1,t-trx(ina1))

Constraint 47

x(fp,fpda1,t) = fpdfp*x(ru,atw1,t-trx(ina1)) + fpdfp*x(rmx,atw1,t-trx(ina1))
+ fpdfp*x(rhtg,atw1,t-trx(ina1)) + fpdfp*x(rfbr,atw1,t-trx(ina1))

Constraint 48

x(lfp,fpda1,t) = fpdlfp*x(ru,atw1,t-trx(ina1)) + fpdlfp*x(rmx,atw1,t-trx(ina1))
+ fpdlfp*x(rhtg,atw1,t-trx(ina1)) + fpdlfp*x(rfbr,atw1,t-trx(ina1))

Discharge from ATW2

Constraint 49

x(ma,fpda2,t) = fpdma*x(ru,atw2,t-trx(ina2)) + fpdma*x(rmx,atw2,t-trx(ina2))
+ fpdma*x(rhtg,atw2,t-trx(ina2)) + fpdma*x(rfbr,atw2,t-trx(ina2))

Constraint 50

x(fp,fpda2,t) = fpdfp*x(ru,atw2,t-trx(ina2)) + fpdfp*x(rmx,atw2,t-trx(ina2))
+ fpdfp*x(rhtg,atw2,t-trx(ina2)) + fpdfp*x(rfbr,atw2,t-trx(ina2))

Constraint 51

x(lfp,fpda2,t) = fpdlfp*x(ru,atw2,t-trx(ina2)) + fpdlfp*x(rmx,atw2,t-trx(ina2))
+ fpdlfp*x(rhtg,atw2,t-trx(ina2)) + fpdlfp*x(rfbr,atw2,t-trx(ina2))

Discharge from ATW3

Constraint 52

x(ma,fpda3,t) = fpdma*x(ru,atw3,t-trx(ina3)) + fpdma*x(rmx,atw3,t-trx(ina3))

+ fpdma*x(rhtg,atw3,t-trx(ina3)) + fpdma*x(rfbr,atw3,t-trx(ina3))

Constraint 53

x(fp,fpda3,t) = fpdfp*x(ru,atw3,t-trx(ina3)) + fpdfp*x(rmx,atw3,t-trx(ina3))
+ fpdfp*x(rhtg,atw3,t-trx(ina3)) + fpdfp*x(rfbr,atw3,t-trx(ina3))

Constraint 54

x(lfp,fpda3,t) = fpdlfp*x(ru,atw3,t-trx(ina3)) + fpdlfp*x(rmx,atw3,t-trx(ina3))
+ fpdlfp*x(rhtg,atw3,t-trx(ina3)) + fpdlfp*x(rfbr,atw3,t-trx(ina3))

MOX/LWR Fractions

This constraint restricts the number of LWRs available to operate on partial-MOX/LWR cores to a certain fraction (fmx). Gen/alf represents the amount of material (load) that goes into OT and MOX-recycle LWRs, respectively.

Constraint 55

gen(rxot,t)/alf(inot,t) > gen(rxmx,t)/alf(inmx,t)*(1-fmx)/fmx

Even in the MOX/LWRs, only a fraction of the core is loaded with MOX/LWR fuel; the rest is UO_2 fuel. The two equations below set the maximum and minimum fraction of MOX/LWR fuel that can exist in the core of one of these reactors, respectively (this prevents a MOX/LWR reactor from being all UO_2 or all MOX/LWR fuel). The parameter "fmox" is the fraction of the core that is MOX/LWR fuel (vs UO_2).

Constraint 56

gen(rxmx,t)/alf(inmx,t)*fmox(t) > x(mox,rxmx,t)

Constraint 57

gen(rxmx,t)/alf(inmx,t)*fmoxmin < x(mox,rxmx,t)

Limit on FSB

The amount of power produced by the FSB (i.e., ATW) is restricted for the purposes of this model to be less than a certain fraction of the overall power produced. In later versions of this model, economic and/or material constraints should set this limit.

Constraint 58

gen(rifr,t) + gen(atw1,t) + gen(atw2,t) + gen(atw3,t) < ffsb*gent(t)

Limits on Growth and Decline of Each Reactor

The amount of energy produced by each type of reactor at a given time must not be greater or less than the amount generated previously times a fraction of the total power for that timestep [i.e., there cannot be all OT/LWR reactors, one timestep, and then all

MOX/LWR-recycle reactors because it takes time (i.e., more than 1 year) to conduct a safety analyses and implement a new fuel type in a reactor]. This somewhat artificial constraint is a surrogate for a constraint that must be imposed by a more realistic capital vintaging constraint that remains to be developed and implemented. Each type of reactor has this constraint. The parameter "fgrow" is equal to "fgrowrx"(which is input in the file *fc.dat* for each reactor type) *fgrowmn(t). The value "fgrowmn" is a minimum growth constraint for a reactor and equals 2*epsrx(t)/(2+epsrx(t)/(nrxts), where "epsrx" = 2*(gent(t+1)-gent(t-1))/(gent(t+1)+gent(t-1)) and "nrxts" is the number of reactors and FSBs (currently eight combined).

Constraint 59
gen(rxot,t) < fseed(inot,t)*gent(t) + gen(rxot,t-1)*njump*(1+fgrow(inot,t))

Constraint 60
gen(rxot,t) > gen(rxot,t-1)*(1-fgrow(inot,t))

Constraint 61
gen(rxmx,t) < fseed(inmx,t)*gent(t) + gen(rxmx,t-1)*njump*(1+fgrow(inmx,t))

Constraint 62
gen(rxmx,t) > gen(rxmx,t-1)*(1-fgrow(inmx,t))

Constraint 63
gen(rxbr,t) < fseed(inbr,t)*gent(t) + gen(rxbr,t-1)*njump*(1+fgrow(inbr,t))

Constraint 64
gen(rxbr,t) > gen(rxbr,t-1)*(1-fgrow(inbr,t))

Constraint 65
gen(rxhg,t) < fseed(inhg,t)*gent(t) + gen(rxhg,t-1)*njump*(1+fgrow(inhg,t))

Constraint 66
gen(rxhg,t) > gen(rxhg,t-1)*(1-fgrow(inhg,t))

Constraint 67
gen(rifr,t) < fseed(inif,t)*gent(t) + gen(rifr,t-1)*njump*(1+fgrow(inif,t))

Constraint 68
gen(rifr,t) > gen(rifr,t-1)*(1-fgrow(inif,t))

Constraint 69
gen(atw1,t) < fseed(ina1,t)*gent(t) + gen(atw1,t-1)*njump*(1+fgrow(ina1,t))

Constraint 70
gen(atw1,t) > gen(atw1,t-1)*(1-fgrow(ina1,t))

Constrant 71
gen(atw2,t) < fseed(ina2,t)*gent(t) + gen(atw2,t-1)*njump*(1+fgrow(ina2,t))

Constraint 72
gen(atw2,t) > gen(atw2,t-1)*(1-fgrow(ina2,t))

Constraint 73
gen(atw3,t) < fseed(ina3,t)*gent+ gen(atw3,t-1)*njump*(1+fgrow(ina3,t))

Constraint 74
gen(atw3,t) > gen(atw3,t-1)*(1-fgrow(ina3,t))

Rate of Blanket Replenishment

As soon as the blanket for an FBR completes its yet-to-be-specified breeding cycle, it must be replenished with more du to initiate a new blanket. Thus, the amount of du needed is equal to the load currently being used in a breeder reactor times the fraction of the reactor that is the blanket.

Constraint 75
x(du,bk,t) = gen(rxbr,t)*br/alf(rxbr,t)

Cooling Storage

This section contains balances for cooling storage of both spent UO_2 and MOX/LWR fuel (RU, RMX, RHTG, and RFBR). For the purpose of this model, it is assumed that material going into cooling storage (from irradiation in each reactor being considered) must be cooled for a set amount of time (tcs) before being transferred to interim storage (IS), or reprocessing. From IS, it then can go to direct disposal. The amount of material that actually is transferred is equal to what came into cooling storage at t-tcs, or the material that came into the reactor at t-trx-tcs (t-trxcs). Thus, the term "bnucsis" is

x(lnu,rxot,t-trxcs(inmx)) + x(lru,rxot,t-trxcs(inmx))
+ x(lwu,rxot,t-trxcs(inmx)) + x(lnu,rxmx,t-trxcs(inmx))
+ x(lru,rxmx,t-trxcs(inmx)) + x(lwu,rxmx,t-trxcs(inmx)).

The new inventory of material in cooling storage is equal to what was previously there, minus what was removed (bnucsis), plus what was added (removed from irradiation).

Constraint 76
x(ru,cs,t) = x(ru,cs,t-1) - bnucsis

+ x(lru,rxmx,t-trx(inmx)) + x(lnu,rxmx,t-trx(inmx))
+ x(lwu,rxmx,t-trx(inmx)) + x(lru,rxot,t-trx(inhg))
+ x(lnu,rxot,t-trx(inhg)) + x(lwu,rxot,t-trx(inhg))

The same condition applies for storage of spent MOX/LWR fuel from plutonium-based reactors, except that bnpcsis is used instead of bnucsis, and is

Constraint 77
x(rmx,cs,t) = x(rmx,cs,t-1) – x(mox,rxmx,t-trxcs(inmx)) + x(mox,rxmx,t-trx(inmx))

Constraint 78
x(rhtg,cs,t) = x(rhtg,cs,t-1) – x(mox,rxhg,t-trxcs(inhg)) + x(mox,rxhg,t-trx(inhg))

Constraint 79
x(rfbr,cs,t) = x(rfbr,cs,t-1) - x(mox,rxbr,t-trxcs(inbr)) + x(mox,rxbr,t-trx(inbr))

Interim Storage

The balance on interim storage (one for ru, one for rmx) specifies that the inventory of material in interim storage is equal to the amount that was previously there, plus what leaves cooling storage (bnucsis and bnpcsis respectively), minus what goes out (to direct disposal).

Constraint 80
x(ru,isu,t) + x(ru,rpccs,t) + x(ru,rpmcs,t) + x(ru,ddis,t) + x(ru,rpcis,t)
+ x(ru,rpmis,t) + x(ru,sep,t) = x(ru,isu,t-1) + x(lnu,rxot,t-trxcs(inmx))
+ x(lru,rxot,t-trxcs(inmx)) + x(lwu,rxot,t-trxcs(inmx)) + x(lnu,rxmx,t-trxcs(inmx))
+ x(lru,rxmx,t-trxcs(inmx)) + x(lwu,rxmx,t-trxcs(inmx))

Constraint 81
x(rmx,isp,t) + x(rmx,rpccs,t) + x(rmx,rpmcs,t) + x(rmx,ddis,t) +
x(rmx,rpcis,t) + x(rmx,rpmis,t) + x(rmx,sep,t) = x(rmx,isp,t-1) +
x(mox,rxmx,t-trxcs(inmx))

Constraint 82
x(rhtg,isp,t) + x(rhtg,rpccs,t) + x(rhtg,rpmcs,t)+ x(rhtg,ddis,t)
+ x(rhtg,rpcis,t) + x(rhtg,rpmis,t) + x(rhtg,sep,t) = x(rhtg,isp,t-1)
+ x(mox,rxhg,t-trxcs(inhg))

Constraint 83
x(rfbr,isp,t) + x(rfbr,rpccs,t) + x(rfbr,rpmcs,t) + x(rfbr,ddis,t)
+ x(rfbr,rpcis,t) + x(rfbr,rpmis,t) + x(rfbr,sep,t) = x(rfbr,isp,t-1)
+ x(mox,rxbr,t-trxcs(inbr))

Reprocessing

The amount of material that can go to reprocessing currently is restricted to less than a certain fraction of the total power generated from various reactors. This constraint is imposed because it is assumed that it takes time to develop the technology to reprocess material initially or to increase reprocessing capabilities significantly; it cannot increase by large increments at once. It is assumed that the reprocessing capacity (kilogram heavy metal/yr) = pnxyyy*"load(rx)"(t - trx), where

x = c,m; yyy = ru,mx,bmx; rx = ot,mx,fbr,fsb.

The values currently used for these variables are as follows, where the parameter "pnxyyy" is the factor by which the reprocessing capacity exceeds the reactor load [gen(rx,t)/alf(rx)].

uox under contracted market, pncru = 6
uox under free market, pnmru = 3
mox under contracted market, pncmx = 6
mox under free market, pnmmx = 3
bmx under contracted market, pncbmx = 6, and
bmx under free market, pnmbmx = 3.

Constraint 84

x(ru,rpccs,t) + x(ru,rpcis,t) + x(rmx,rpccs,t) + x(rmx,rpcis,t) + x(rhtg,rpccs,t)
+ x(rhtg,rpcis,t) + x(rfbr,rpccs,t) + x(rfbr,rpcis,t) + x(bmx,rpcbks,t)
< pncru*gen(rxot,t-1)/alf(inot,t-1) + pnc*gen(rxmx,t-1)/alf(inmx,t-1)
+ pncmx*gen(rxhg,t-1)/alf(inhg,t-1) + pncbmx*gen(rxbr,t-1)/alf(inbr,t-1)

Constraint 85

x(ru,rpmcs,t) + x(ru,rpmis,t) + x(rmx,rpmcs,t) + x(rmx,rpmis,t) + x(rhtg,rpmcs,t)
+ x(rhtg,rpmis,t) + x(rfbr,rpmcs,t) + x(rfbr,rpmis,t) + x(bmx,rpmbks,t)
< pnmru*gen(rxot,t-1)/alf(inot,t-1) + pnm*gen(rxmx,t-1)/alf(inmx,t-1)
+ pnmmx*gen(rxhg,t-1)/alf(inhg,t-1) + pnmbmx*gen(rxbr,t-1)/alf(inbr,t-1)

MOX/LWR Recycle Limit

The amount of MOX/LWR fuel that can be recycled must be less than the amount of MOX/LWR fuel currently in a reactor times a fraction representing the total number of times the fuel can be recycled (ncyc). This limits the amount of fuel that can go to reprocessing and thus represents how many times recycled fuel can be reprocessed.

Constraint 86

x(rmx,rpccs,t) + x(rmx,rpmcs,t) + x(rmx,rpcis,t) + x(rmx,rpmis,t) <
x(mox,rxmx,t-trx(inmx))*(1 – 1/ncyc)

Constraint 87

x(rhtg,rpccs,t) + x(rhtg,rpmcs,t) + x(rhtg,rpcis,t) + x(rhtg,rpmis,t) <
x(mox,rxhg,t-trx(inhg))*(1 - incyc)

Constraint 88

x(rfbr,rpccs,t) + x(rfbr,rpmcs,t) + x(rfbr,rpcis,t) + x(rfbr,rpmis,t) <
x(mox,rxbr,t-trx(inbr))*(1 - incyc)

Recycled Uranium Storage

The inventory of material in cooling storage is equal to what was there previously (inventory at t-1), plus what comes in from reprocessing, minus what goes to blending, conversion, and fuel fabrication.

Constraint 89

x(ru,rus,t) + x(ru,bl,t) + x(ru,cv,t) + x(ru,ffmxps,t) + x(ru,ffmxrp,t) + x(ru,ffmxwp,t)
= x(ru,rus,t-1) + x(bmx,rpcbks,t-trp)*(1 - ybpu)*yrp
+ x(bmx,rpmbks,t-trp)*(1 - ybpu)*yrp + x(rmx,rpccs,t-trp)*(1 - ympu)*yrp
+ x(rmx,rpcis,t-trp)*(1 - ympu)*yrp + x(rmx,rpmcs,t-trp)*(1 - ympu)*yrp
+ x(rmx,rpmis,t-trp)*(1 - ympu)*yrp + x(rhtg,rpccs,t-trp)*(1 - ympu)*yrp
+ x(rhtg,rpcis,t-trp)*(1 - ympu)*yrp + x(rhtg,rpmcs,t-trp)*(1 - ympu)*yrp
+ x(rhtg,rpmis,t-trp)*(1 - ympu)*yrp + x(rfbr,rpccs,t-trp)*(1 - ympu)*yrp
+ x(rfbr,rpcis,t-trp)*(1 - ympu)*yrp + x(rfbr,rpmcs,t-trp)*(1 - ympu)*yrp
+ x(rfbr,rpmis,t-trp)*(1 - ympu)*yrp + x(ru,rpccs,t-trp)*(1 - yupu)*yrp
+ x(ru,rpcis,t-trp)*(1 - yupu)*yrp + x(ru,rpmcs,t-trp)*(1 - yupu)*yrp
+ x(ru,rpmis,t-trp)*(1 - yupu)*yrp + x(ru,sep,t-tsp)*usep
+ x(rmx,sep,t-tsp)*usep + x(rhtg,sep,t-tsp)*usep + x(rfbr,sep,t-tsp)*usep

Plutonium Storage

The amount of material that goes to (separated) plutonium storage is equal to that which was in plutonium storage previously; minus what goes to fuel fabrication (ffmxrp and ffmxps), disposal (pudpus and pudrp), and the fast burner (rxfb); plus what is added through reprocessing at the previous time t-trp. The parameter "yupu" is the fraction of plutonium in ru (spent UO_2 fuel from rxot), "ympu" is the fraction of plutonium in rmx (spent MOX/LWR fuel from rxmx), "ybpu" is the fraction of plutonium in fbr blanket discharged, and "yrp" is the efficiency of reprocessing. The time"trp" is how long it takes to reprocess the fuel (in years).

Constraint 90

x(rpu,pus,t) = x(rpu,pus,t-1) – x(rpu,ffmxps,t) – x(rpu,pudpus,t) – x(rpu,ffmxrp,t)
– x(rpu,pudrp,t) – x(rpu,sep,t) + x(ru,rpcis,t-trp)*yupu*yrp
+ x(ru,rpccs,t-trp)*yupu*yrp + x(ru,rpmis,t-trp)*yupu*yrp

+ x(ru,rpmcs,t-trp)*yupu*yrp + x(rmx,rpcis,t-trp)*ympu*yrp
+ x(rmx,rpccs,t-trp)*ympu*yrp + x(rmx,rpmis,t-trp)*ympu*yrp
+ x(rmx,rpmcs,t-trp)*ympu*yrp + x(rhtg,rpcis,t-trp)*ympu*yrp
+ x(rhtg,rpccs,t-trp)*ympu*yrp + x(rhtg,rpmis,t-trp)*ympu*yrp
+ x(rhtg,rpmcs,t-trp)*ympu*yrp + x(rfbr,rpcis,t-trp)*ympu*yrp
+ x(rfbr,rpccs,t-trp)*ympu*yrp + x(rfbr,rpmis,t-trp)*ympu*yrp
+ x(rfbr,rpmcs,t-trp)*ympu*yrp + x(bmx,rpcbks,t-trp)*ybpu
+ x(bmx,rpmbks,t-trp)*ybpu

Minor Actinides and Fission Product Storage

The current inventory of minor actinides is equal to the previous inventory, minus what goes to disposal and the fast burner, plus a fraction of the total material that is reprocessed. The parameter "fruma" is the fraction of minor actinides in spent UO_2 fuel, "frmxma" is the fraction of minor actinides in spent MOX/LWR fuel, and "fbmxma" is the fraction of actinides in an irradiated breeder blanket.

Constraint 91

x(ma,refps,t) + x(ma,refpd,t) = x(ma,refps,t-1) + x(ru,rpccs,t-trp)*fruma
+ x(ru,rpmcs,t-trp)*fruma + x(ru,rpcis,t-trp)*fruma + x(ru,rpmis,t-trp)*fruma
+ x(rmx,rpccs,t-trp)*frmxma + x(rmx,rpmcs,t-trp)*frmxma
+ x(rmx,rpcis,t-trp)*frmxma + x(rmx,rpmis,t-trp)*frmxma
+ x(rhtg,rpccs,t-trp)*frmxma + x(rhtg,rpcis,t-trp)*frmxma
+ x(rhtg,rpmcs,t-trp)*frmxma + x(rhtg,rpmis,t-trp)*frmxma
+ x(rfbr,rpccs,t-trp)*frmxma + x(rfbr,rpcis,t-trp)*frmxma
+ x(rfbr,rpmcs,t-trp)*frmxma + x(rfbr,rpmis,t-trp)*frmxma
+ x(bmx,rpcbks,t-trp)*fbmxma + x(bmx,rpmbks,t-trp)*fbmxma

The balances on long-lived and regular fission product storage are almost identical, except that they use different variables.

Constraint 92

x(lfp,refps,t) + x(lfp,refpd,t) = x(lfp,refps,t-1) + x(ru,rpccs,t-trp)*frulfp
+ x(ru,rpmcs,t-trp)*frulfp + x(ru,rpmcs,t-trp)*frulfp + x(ru,rpmis,t-trp)*frulfp
+ x(rmx,rpccs,t-trp)*frmxlfp + x(rmx,rpmcs,t-trp)*frmxlfp
+ x(rmx,rpmcs,t-trp)*frmxlfp + x(rmx,rpmis,t-trp)*frmxlfp
+ x(rhtg,rpccs,t-trp)*frmxlfp + x(rhtg,rpcis,t-trp)*frmxlfp
+ x(rhtg,rpmcs,t-trp)*frmxlfp + x(rhtg,rpmis,t-trp)*frmxlfp
+ x(rfbr,rpccs,t-trp)*frmxlfp + x(rfbr,rpcis,t-trp)*frmxlfp
+ x(rfbr,rpmcs,t-trp)*frmxlfp + x(rfbr,rpmis,t-trp)*frmxlfp
+ x(bmx,rpcbks,t-trp)*fbmxlfp + x(bmx,rpmbks,t-trp)*fbmxlfp

Constraint 93

x(lfp,sefps,t) + x(lfp,sefpd,t) = x(lfp,sefps,t-1) + x(ru,sep,t-trp)*fseprulfp

+ x(rmx,sep,t-trp)*fseprmxlfp + x(rhtg,sep,t-trp)*fseprmxlfp
+ x(rfbr,sep,t-trp)*fseprmxlfp

Fission Product Storage Mass Balance

This constraint computes the inventory of material (short- and long-lived fission products and minor actinides) in fission product disposal at any given time based on what was there previously, plus what comes in from fission product storage (of both separated and reprocessed fission products-the difference being that reprocessed fission products include minor actinides) and the FSB.

Constraint 94

x(fp,refps,t) + x(fp,refpd,t) = x(fp,refps,t-1) + x(ru,rpccs,t-trp)*frufp
+ x(ru,rpmcs,t-trp)*frufp + x(ru,rpmcs,t-trp)*frufp + x(ru,rpmis,t-trp)*frufp
+ x(rmx,rpccs,t-trp)*frmxfp + x(rmx,rpmcs,t-trp)*frmxfp
+ x(rmx,rpmcs,t-trp)*frmxfp + x(rmx,rpmis,t-trp)*frmxfp
+ x(rhtg,rpccs,t-trp)*frmxfp + x(rhtg,rpcis,t-trp)*frmxfp
+ x(rhtg,rpmcs,t-trp)*frmxfp + x(rhtg,rpmis,t-trp)*frmxfp
+ x(rfbr,rpccs,t-trp)*frmxfp + x(rfbr,rpcis,t-trp)*frmxfp
+ x(rfbr,rpmcs,t-trp)*frmxfp + x(rfbr,rpmis,t-trp)*frmxfp
+ x(bmx,rpcbks,t-trp)*fbmxfp + x(bmx,rpmbks,t-trp)*fbmxfp

Constraint 95

x(fp,sefps,t) + x(fp,sefpd,t) = x(fp,sefps,t-1) + x(ru,sep,t-trp)*fseprufp
+ x(rmx,sep,t-trp)*fseprmxfp + x(rhtg,sep,t-trp)*fseprmxfp
+ x(rfbr,sep,t-trp)*fseprmxfp

Fission Product Disposal

The inventory of material in fission product disposal is balanced in the equations that follow. It basically equals what was there previously, plus what is added from separations, reprocessing, and the FSBs.

Constraint 96

x(fp,fpd1,t) + x(fp,fpd2,t) + x(fp,fpd3,t) = x(fp,refpd,t) + x(fp,sefpd,t) + x(fp,fpdif,t)
+ x(fp,fpda1,t) + x(fp,fpda2,t) + x(fp,fpda3,t) + x(fp,fpd1,t-1) + x(fp,fpd2,t-1)
+ x(fp,fpd3,t-1)

Constraint 97

x(lfp,fpd1,t) + x(lfp,fpd2,t) + x(lfp,fpd3,t) = x(lfp,refpd,t) + x(lfp,sefpd,t)
+ x(lfp,fpdif,t) + x(lfp,fpda1,t) + x(lfp,fpda2,t) + x(lfp,fpda3,t) + x(lfp,fpd1,t-1)
+ x(lfp,fpd2,t-1) + x(lfp,fpd3,t-1)

Constraint 98

x(ma,fpd1,t) + x(ma,fpd2,t) + x(ma,fpd3,t) = x(ma,refpd,t) + x(ma,fpdif,t)
+ x(ma,fpda1,t) + x(ma,fpda2,t) + x(ma,fpda3,t) + x(ma,fpd1,t-1)
+ x(ma,fpd2,t-1) + x(ma,fpd3,t-1)

Constraint 99
x(fp,fpd1,t) + x(lfp,fpd1,t) + x(ma,fpd1,t) > x(fp,fpd1,t-1) + x(lfp,fpd1,t-1)
+ x(ma,fpd1,t-1)

Constraint 100
x(fp,fpd2,t) + x(lfp,fpd2,t) + x(ma,fpd2,t) > x(fp,fpd2,t-1) + x(lfp,fpd2,t-1)
+ x(ma,fpd2,t-1)

Constraint 101
x(fp,fpd3,t) + x(lfp,fpd3,t) + x(ma,fpd3,t) > x(fp,fpd3,t-1) + x(lfp,fpd3,t-1)
+ x(ma,fpd3,t-1)

Plutonium Disposal

The inventory of fuel in plutonium disposal is equal to what was there previously, plus that coming in from reprocessing and plutonium storage.

Constraint 102
x(rpu,pud,t) = x(rpu,pud,t-1) + x(rpu,pudrp,t) + x(rpu,pudpus,t)

The inventory of material in plutonium disposal is required to be less than a certain value that represents the amount of room available at the site.

Constraint 103
x(rpu,pud,t) < pudcap(t)

Direct Disposal

This constraint addresses the balance of material inventory in direct disposal. Spent fuel residing in dd at the current time t is equal to what was there at t-1, plus what comes in from interim storage (for RU, RMX, RHTG, and RFBR).

Constraint 104
x(ru,dd1,t) + x(ru,dd2,t) + x(ru,dd3,t) = x(ru,ddis,t) + x(ru,dd1,t-1)
+ x(ru,dd2,t-1) + x(ru,dd3,t-1)

Constraint 105
x(rmx,dd1,t) + x(rmx,dd2,t) + x(rmx,dd3,t) = x(rmx,ddis,t)
+ x(rmx,dd1,t-1) + x(rmx,dd2,t-1) + x(rmx,dd3,t-1)

Constraint 106

x(rhtg,dd1,t) + x(rhtg,dd2,t) + x(rhtg,dd3,t) = x(rhtg,ddis,t)
+ x(rhtg,dd1,t-1) + x(rhtg,dd2,t-1) + x(rhtg,dd3,t-1)

Constraint 107
x(rfbr,dd1,t) + x(rfbr,dd2,t) + x(rfbr,dd3,t) = x(rfbr,ddis,t)
+ x(rfbr,dd1,t-1) + x(rfbr,dd2,t-1) + x(rfbr,dd3,t-1)

Direct Disposal Inventory

The following equations indicate that the inventory in direct disposal in one year must be greater than what was there the year before (at least a little spent fuel must go to direct disposal each year).

Constraint 108
x(ru,dd1,t) + x(rmx,dd1,t) + x(rhtg,dd1,t) + x(rfbr,dd1,t)
> x(ru,dd1,t-1) + x(rmx,dd1,t-1) + x(rhtg,dd1,t-1) + x(rfbr,dd1,t-1)

Constraint 109
x(ru,dd2,t) + x(rmx,dd2,t) + x(rhtg,dd2,t) + x(rfbr,dd2,t)
> x(ru,dd2,t-1) + x(rmx,dd2,t-1) + x(rhtg,dd2,t-1) + x(rfbr,dd2,t-1)

Constraint 110
x(ru,dd3,t) + x(rmx,dd3,t) + x(rhtg,dd3,t) + x(rfbr,dd3,t)
> x(ru,dd3,t-1) + x(rmx,dd3,t-1) + x(rhtg,dd3,t-1) + x(rfbr,dd3,t-1)

Spent Fuel Capacity

The inventory of material in direct disposal (ru and rmx combined) must be less than a certain value, representing how much room will be available in chosen repositories on a given year. However, currently this value is rather large, so it will not pose too much of a restriction.

Constraint 111
x(ru,dd1,t) + x(ru,dd2,t) + x(ru,dd3,t) + x(rmx,dd1,t) + x(rmx,dd2,t)
+ x(rmx,dd3,t) + x(rhtg,dd1,t) + x(rhtg,dd2,t) + x(rhtg,dd3,t)
+ x(rfbr,dd1,t) + x(rfbr,dd2,t) + x(rfbr,dd3,t) < ddcap(t)

APPENDIX B. OBJECTIVE FUNCTION

The purpose of the objective function is to represent variables in the fuel-cycle flow model that pose a cost, with the cost per unit of that material. By multiplying the two, an overall cost for that material/process in the fuel cycle can be obtained. The optimizer CPLEX tries to optimize this cost. Most unit costs are input in the input file "cost.dat", but instead of using a fixed charge rate for reactor inventory costs (as was before done), the economic principle of vintaging was recently implemented.

The piece of the objective function that deals with a fixed charge rate for reactors is represented as Eq. (B-1) instead of the simple product described in the above paragraph.

$$\sum_{rx=1}^{nrx}\sum_{t=1}^{nt}\frac{icap(rx,t)}{(1+dr)^{t-1}} \quad , \tag{B-1}$$

where

nrx	=	the number of reactors,
nt	=	the number of timesteps over which optimization occurs,
icap(rx,t)	=	the capital expended for reactor rx in time t, and
dr	=	the economic discount rate.

One way of computing icap(rx,t) is first to determine the cost icost(rx,t) [represented by Eq. (B-2)] of bringing on the new capacity, icap(rx,t), having a construction time, tconst(rx,t), and a lifetime, tlife(rx,t), as

$$icost(rx,t) = \text{gen(inrx,t)*utc(rx)*idc(rx)*crf(rx)} \quad , \tag{B-2}$$

where

crf(rx) = the capital recovery factor [see Eq. (B-3)]:

$$crf(rx) = \frac{dr*(1+dr)^{tlife(rx)}}{(1+dr)^{tlife(rx)}-1} \quad , \tag{B-3}$$

where

idc(rx) = the interest incurred during construction [see Eq. (B-4)].

$$idc(rx) = \sum_{i=1}^{tconstr(rx)}\frac{(1+dr)^{i}}{tconstr(rx)} \quad , \tag{B-4}$$

where

utc(rx) = unit cost of reactor ($/We).

The cost icap(rx,t) needed to compute the capital component of the cost-based objective function given above is derived from icost(rx,t) by discounting all charges back to the time of interest, t, as in Eq. (B-5):

$$icap(rx,t) = \sum_{i=t}^{\max(t+tlife,nt)} \frac{i\cos t(rx,t)}{(1+dr)^{i-t}} \quad . \tag{B-5}$$

However, for the format of the objective function needed in this program, the variable gen(inrx,t) had to be factored out of the above sum. Thus, the variable ucrx is multiplied by gen(inrx,t) to get the actual costs for the objective function (the denominator in Eq. (B-1) is already implicit to the objective function). Ucrx is basically equal to icap for each reactor and time divided by the inventory [see Eq. (B-6)]:

$$ucrx(rx,t) = \sum_{i=t}^{\max(t+tlife,nt)} \frac{\mathrm{utc(rx)} * \mathrm{idc(rx)} * \mathrm{crf(rx)}}{(1+dr)^{i-t}} \quad . \tag{B-6}$$

A list of all terms appearing in the objective function are shown in Table B-1. These are all multiplied by the term pdr, which is the present value discount rate [given by Eq. (B-7)]:

$$\sum_{rx=1}^{nrx}\sum_{t=1}^{nt} \frac{icap(rx,t)}{(1+dr)^{t-1}} \quad . \tag{B-7}$$

TABLE B-1
TERMS IN OBJECT FUNCTION

Coefficient	**Variable**
Ucmm	x(nu,bl,t)
Ucmm	x(nu,cv,t)
Ucmm	x(nu,ffmxrp,t)
Ucmm	x(nu,ffmxps,t)
Ucmm	x(nu,ffmxwp,t)
Uccv	x(nu,cv,t)
Uccv	x(ru,cv,t)
Ucsw *sofnu	x(nu,er,t)
Ucsw* sofru	x(ru,er,t)
Ucdus	x(du,dus,t)
Ucrus	x(ru,rus,t)
Ucwus	x(heu,wus,t)

Ucwps	x(wpu,wps,t)
Ucbl	x(ru,bl,t)

TABLE B-1 (cont)
TERMS IN OBJECT FUNCTION

Coefficient	**Variable**
Ucbl	x(du,bl,t)
Ucbl	x(nu,bl,t)
ucbl	x(heu,bl,t)
ucffux	x(lnu,ffux,t)
ucffux	x(lru,ffux,t)
ucffux	x(lwu,ffux,t)
ucffmx	x(nu,ffmxrp,t)
ucffmx	x(ru,ffmxrp,t)
ucffmx	x(du,ffmxrp,t)
ucffmx	x(nu,ffmxps,t)
ucffmx	x(ru,ffmxps,t)
ucffmx	x(du,ffmxps,t)
ucffmx	x(nu,ffmxwp,t)
ucffmx	x(ru,ffmxwp,t)
ucffmx	x(du,ffmxwp,t)
ucffmx	x(rpu,ffmxrp,t)
ucffmx	x(rpu,ffmxps,t)
ucffmx	x(wpu,ffmxwp,t)
ucrx(inot)	gen(inot,t)
ucrx(inmx)	gen(inmx,t)
ucrx(inhg)	gen(inhg,t)
ucrx(infb)	gen(infb,t)
ucrx(inif)	gen(inif,t)
ucrx(ina1)	gen(ina1,t)
ucrx(ina2)	gen(ina2,t)
ucrx(ina3)	gen(ina3,t)
fom(rxot)*utc(rxot)* 1000000/ avail(rxot)/(1-eps(rxot))	gen(rxot,t)
fom(rxmx)*utc(rxmx)* 1000000/ avail(rxmx)/(1-eps(rxmx))	gen(rxmx,t)
fom(rxhg)*utc(rxhg)* 1000000/ avail(rxhg)/(1-eps(rxhg))	gen(rxhg,t)
fom(rxbr)*utc(rxbr)* 1000000/	gen(rxbr,t)

avail(rxbr)/(1-eps(rxbr))	
fom(rxfb)*utc(rxfb)* 1000000/ avail(rxfb)/(1-eps(rxfb))	gen(rxfb,t)

TABLE B-1 (cont)
TERMS IN OBJECT FUNCTION

Coefficient	**Variable**
fom(rxhg)*utc(rxhg)* 1000000/ avail(rxhg)/(1-eps(rxhg))	gen(rxhg,t)
fom(rxbr)*utc(rxbr)* 1000000/ avail(rxbr)/(1-eps(rxbr))	gen(rxbr,t)
fom(rxfb)*utc(rxfb)* 1000000/ avail(rxfb)/(1-eps(rxfb))	gen(rxfb,t)
uccs	x(ru,cs,t)
uccs	x(rmx,cs,t)
uccs	x(rhtg,cs,t)
uccs	x(rfbr,cs,t)
ucisu	x(ru,isu,t)
ucisp	x(rmx,isp,t)
ucisp	x(rhtg,isp,t)
ucisp	x(rfbr,isp,t)
ucbks	x(bmx,bks,t)
ucpus	x(rpu,pus,t)
ucfps	x(fp,refps,t)
ucfps	x(fp,sefps,t)
ucrpcux	x(ru,rpccs,t)
ucrpcux	x(ru,rpcis,t)
ucrpcmx	x(rmx,rpccs,t)
ucrpcmx	x(rmx,rpcis,t)
ucrpmux	x(ru,rpmcs,t)
ucrpmux	x(ru,rpmis,t)
ucrpmmx	x(rmx,rpmcs,t)
ucrpmmx	x(rmx,rpmis,t)
ucrpcux	x(rhtg,rpccs,t)
ucrpcux	x(rhtg,rpcis,t)
ucrpcmx	x(rfbr,rpccs,t)
ucrpcmx	x(rfbr,rpcis,t)
ucrpmux	x(rhtg,rpmcs,t)
ucrpmux	x(rhtg,rpmis,t)

ucrpmmx	x(rfbr,rpmcs,t)
ucrpmmx	x(rfbr,rpmis,t)
ucrpcbmx	x(bmx,rpcbks,t)
ucrpmbmx	x(bmx,rpmbks,t)

TABLE B-1 (cont)
TERMS IN OBJECT FUNCTION

Coefficient	**Variable**
ucsep	x(ru,sep,t)
ucsep	x(rmx,sep,t)
ucsep	x(rhtg,sep,t)
ucsep	x(rfbr,sep,t)
ucpurp	x(rpu,ffmxrp,t)
ucpurp	x(rpu,ffmxps,t)
ucpurp	x(wpu,ffmxwp,t)
ucpurp	x(rpu,sep,t)
ucpurp	x(wpu,sep,t)
ucdd1	x(ru,dd1,t)
ucdd1	x(rmx,dd1,t)
ucdd1	x(rhtg,dd1,t)
ucdd1	x(rfbr,dd1,t)
ucdd2	x(ru,dd2,t)
ucdd2	x(rmx,dd2,t)
ucdd2	x(rhtg,dd2,t)
ucdd2	x(rfbr,dd2,t)
ucdd3	x(ru,dd3,t)
ucdd3	x(rmx,dd3,t)
ucdd3	x(rhtg,dd3,t)
ucdd3	x(rfbr,dd3,t)
ucpud	x(rpu,pud,t)
ucfpd1	x(fp,fpd1,t)
ucfpd1	x(lfp,fpd1,t)
ucfpd1	x(ma,fpd1,t)
ucfpd2	x(fp,fpd2,t)
ucfpd2	x(lfp,fpd2,t)
ucfpd2	x(ma,fpd2,t)
ucfpd3	x(fp,fpd3,t)
ucfpd3	x(lfp,fpd3,t)
ucfpd3	x(ma,fpd3,t)

APPENDIX C. VALID MATERIAL/PROCESS COMBINATIONS

The following matrix shows the material and process combinations addressed in this program and if they are valid, whether they are analyzed as a flow rate or an inventory.

M/P	MM(1)	CV(2)	ER(3)	FF				BL(8)
				FFUX(4)	FFMX			
					FFRP(5)	FFPS(6)	FFWP(7)	
NU(1)	I	F	F		F	F	F	F
RU(2)		F	F		F	F	F	F
DU(3)					F	F	F	F
LNU(4)				F				
LRU(5)				F				
LWU(6)				F				
WPU(7)							F	
HEU(8)								F
RPU(9)					F	F		
MOX(10)								
RMX(11)								
RHTG(12)								
RFBR(13)								
BMX(14)								
FP(15)								
LFP(16)								
MA(17)								

M/P	RX							
	RXOT(9)	RXMX(10)	RXHG(11)	RXBR(12)	IFR(13)	ATW1(14)	ATW2(15)	ATW3(16)
NU(1)								
RU(2)					F	F	F	F
DU(3)								
LNU(4)	F	F						
LRU(5)	F	F						
LWU(6)	F	F						
WPU(7)								
HEU(8)								
RPU(9)								
MOX(10)		F	F	F				
RMX(11)					F	F	F	F
RHTG(12					F	F	F	F
RFBR(13)					F	F	F	F
BMX(14)								
FP(15)								
LFP(16)								
MA(17)								

M/P	BK(17)	CS(18)	IS		DUS(21)	RUS(22)	BKS(23)	PUS(24)
			ISU(19)	ISP(20)				
NU(1)								
RU(2)		I	I			I		
DU(3)	F				I			
LNU(4)								
LRU(5)								
LWU(6)								
WPU(7)								
HEU(8)								
RPU(9)								I
MOX(10)								
RMX(11)		I		I				
RHTG(12)		I		I				
RFBR(13)		I		I				
BMX(14)							I	
FP(15)								
LFP(16)								
MA(17)								

M/P	REFPS(25)	SEFPS(26)	WPS(27)	WUS(28)	RP					
					CONTRACT(C)			MARKET(M)		
					CS(29)	IS(30)	BKS(31)	CS(32)	IS(33)	BKS(34)
NU(1)										
RU(2)					F	F		F	F	
DU(3)										
LNU(4)										
LRU(5)										
LWU(6)										
WPU(7)			I							
HEU(8)				I						
RPU(9)										
MOX(10)										
RMX(11)					F	F		F	F	
RHTG(12)					F	F		F	F	
RFBR(13)					F	F		F	F	
BMX(14)							F			F
FP(15)	I	I								
LFP(16)	I	I								
MA(17)	I									

M/P	SEP(35)	DD1(36),DD2(37),DD3(38)		PUD(41)		FPD1(44),FPD2(45),FPD3(46)	
		CS(39)	IS(40)	RP(42)	PUS(43)	REFPS(47)	SEFPS(48)
NU(1)							
RU(2)	F	F, I	F, I				
DU(3)							
LNU(4)							
LRU(5)							
LWU(6)							
WPU(7)							
HEU(8)							
RPU(9)				F, I	F,I		
MOX(10)							
RMX(11)	F	F, I	F, I				
RHTG(12)	F	F, I	F, I				
RFBR(13)	F	F, I	F, I				
BMX(14)							
FP(15)						F,I	F,I
LFP(16)						F,I	F,I
MA(17)						F,I	

M/P	FPD1(44),FPD2(45),FPD3(46)			
	FPDIFR(49)	FPDATW1(50)	FPDATW2(51)	FPDATW3(52)
NU(1)				
RU(2)				
DU(3)				
LNU(4)				
LRU(5)				
LWU(6)				
WPU(7)				
HEU(8)				
RPU(9)				
MOX(10)				
RMX(11)				
RHTG(12)				
RFBR(13)				
BMX(14)				
FP(15)	F, I	F, I	F, I	F, I
LFP(16)	F, I	F, I	F, I	F, I
MA(17)	F, I	F, I	F, I	F, I

APPENDIX D. SAMPLE INPUT FILES

The following listings are the two most important input files for FCOPT, fc.dat and cost.dat. They should be self-explanatory. Other required files were described in Ref. 1 or somewhere in Section 2, including

- Constraints.dat (contains constraints described in Appendix A);
- Initial.dat (contains initial quantity of materials in various processes for times before the start time—the year 2000);
- Meandose.dat (contains the cumulative normalized dose parameters described in Section 2.5.2);
- Pne.baubo.dat (contains power generation estimates from the E^3 model);
- Pne.edbo.dat (contains power generation estimates from the E^3 model);
- Prolifdat_pu.dat (contains the proliferation risk parameters described in Section 2.4); and
- Uinv.dat (contains estimate of uranium cost, which must be verified by the user as being close to that produced by FCOPT periodically).

"fc.dat"

```
DATA INPUT FOR LP OPTIMIZATION OF NUCLEAR FUEL CYCLES (fc.dat)
 bynl
 Generation parameters              k =  OT        MX       HTGR      FBR       FSB       ATW1      ATW2      ATW3
  o thermal conv. efficiency, etath(k)  0.33      0.33      0.39      0.40      0.32      0.32      0.32      0.32
  o plant availability, avail(k)        0.90      0.90      0.90      0.90      0.90      0.90      0.90      0.90
  o recirc. power fraction, epsilon(k)  0.01      0.01      0.01      0.01      0.01      0.01      0.01      0.01
  o fuel burnup, bu(k, MWtd/kgHM)       45.       45.       100.      100.      100.      100.      100.      100.
  o fuel residence times, trx(k, yr)    4         4         4         2         1         1         1         1
  o reactor construct times, tconstr    8         8         8         8         8         8         8         8
  o reactor construct times, tlife      40        40        40        40        40        40        40        40
```

```
 o pu fissile plutonium fraction, fpuf 0.60      0.50      0.6       0.70      0.70      0.70      0.70      0.70
 o fraction of gent each grows, fseed  0.01      0.01      0.01      0.01      0.001     0.001     0.001     0.001
Initial time frame information
 o history year, yroo (est. history)   1990.
 o base year, yro (unopt. growth)      1995.
 o start year, yri (start opt.)        2000.
Power gen.opt.(igent) 1=edbo,2=baubo   1
OT/(MX + OT) constraint, fmx           0.95
MOX core fraction constraints, fmox(t) = (fmoxf - fmoxo)(1 - exp(-t/taumox)) + fmoxo
 o minimum, fmoxmin                    0.10
 o initial value, fmoxo                0.33
 o final value, fmoxf                  0.67
 o MOX time constant, taumox(yr)       1000.
Max fract. total gen prov by fsb, ffsb 0.200
Process residence times (integer values, if .lt. 1.0, = 0.0)
 o enrichment, ter(yr)                 0
 o UOX fuel fabrication, tffux(yr)     1
 o MOX fuel fabrication, tffmx(yr)     1
 o weapons heu blending time, tbl(yr)  4
 o cooling storage time, tcs(yr)       4
 o interim storage time, tis(yr)      10
 o reprocessing, trp(yr)               3
 o separations, tsp(yr)                3
Number of MOX recycles plus one, ncyc  1
CS storage, ncs(yr)                    10
IS storage, nis(yr)                    20
Reprocessing capacity, repxyyy(kgHM/yr) = pnxyyy*loadrx(t - trx); where,
                        x = c,m; yyy = ru,mx,bmx; rx = ot,mx,fbr,fsb
 o uox under contracted market, pncru  6.
 o uox onder free market, pnmru        3.
 o mox under contracted market, pncmx  6.
 o mox onder free market, pnmmx        3.
 o bmx under contracted market, pncbmx 6.
 o bmx under free market, pnmbmx       3.
Process efficiencies, yxx; xx = mm, cv, er, ff, rp
 o ore => u3o8, ymm                    0.96
 o u3o8 => uf6 conversion, ycv         0.98
 o enrichment uranium losses, yer      0.98
 o fuel fab. losses for uox, yffux     0.98
 o fuel fab. losses for mox, yffmx     0.98
 o reprocessing losses, yrp            0.99
 o sep efficiency, ysep                0.99
 o usep = fr. uranium to RUS from SEP  0.956
 o fseprufp = fr.FP in RU (SEP to DIS) 0.03
 o fseprulfp = fr.LFP in RU(SEPtoDIS)  0.003
```

```
 o fseprmxfp=fr.FP in RMX(SEP to DIS)  0.03
 o fseprmxlfp=fr.FP in RMX(SEP to DIS) 0.003
 o fsb consumption efficiency, yfsb    0.99
    Note: (1 - yfsb) must = fpdma + fpdfp + fpdlfp
 o fpdma = fr. fsb gen of ma 4 disp.   0.005
 o fpdfp = fr. fsb gen of fp 4 disp.   0.003
 o fpdlfp = fr. fsb gen of lfp 4 disp. 0.002
Enrichment paramters (default values if ierop = 0)
 o enrichment tailings option, ieropt  0
 o nu feed concentration, xfnu         0.00711
 o ru feed concentration, xfru         0.01000
 o nu product concentration, xpnu      0.03500
 o ru product concentration, xpru      0.04000
 o nu tailings concentration, xtnu     0.00300
 o ru tailings concentration, xtru     0.00300
 o heu concentration,xheu              0.9500
Uranium and plutonium concentrations in feed and discharge fuels
 o u25 in uox from nu, yunu            0.00711
 o u25 in uox from ru, yulnu           0.03500
 o u25 in uox from nu, yuru            0.01000
 o u25 in uox from ru, yulru           0.04000
 o u25 in uox from ru, yudu            0.00000
 o u25 in uox from nu, yuldu           0.03500
 o u25 in uox from nu, yuheu           0.95000
 o pu in uox to lwr,  ymox(1)          0.00000
 o pu in mox to mx/lwr, ymox(2)        0.05000
 o pu in mox to htgr, ymox(3)          0.04000
 o pu in mox to fbr, ymox(4)           0.10000
 o pu in discharged uox, yupu          0.01520
 o pu in discharged mox, ympu          0.03760
 o pu in discharged htgr, yhpu         0.03020
 o pu in discharged fbr, yfpu          0.13200
 o pu in fbr blanket discharged, ybpu  0.03000
MA and LLFP fractions (of actinides and fp) from discharged fuels, fxxyy:
                                        xx = ma,fp; yy = ru, rmx
 o ma in ru fraction of act, fruma     0.01
 o ma in rmx fraction of act, frmxma   0.01
 o ma in bmx fraction of act, fbmxma   0.001
 o llfp in ru fraction of fp, frulfp   0.004
 o llfp in rmx fraction of fp, frmxlfp 0.004
 o llfp in rbx fraction of fp, frbxlfp 0.004
 o fp fraction in ru, frufp            0.04
 o fp fraction in rmx, frmxfp          0.04
 o fp fraction in rbx, frbxfp          0.04
Weap. matl. dispos., gxx(kgHM/yr) = mxx*(1-exp(-t))/(tfxx+tauxx*exp(-tfxx/tauxx);
```

```
                                                                  xx = heu, wpu
  o maximum heu disposal, mheu(kg)      5.00e+00
  o maximum heu dispos. time, tfheu(yr) 50.
  o heu disposal constant, tauheu(yr)   10.
  o maximum wpu disposal, mwpu(kg)      1.00e+05
  o maximum wpu dispos. time, tfwpu(yr) 50.
  o wpu disposal constant, tauwpu(yr)   10.
 fbr breeding ratio, br                  1.2
 dd cap. & rate, ddcap, epsdd(1/yr)     8.0e+10   0.001
 pud cap. & rate, pudcap, epspud(1/yr)  8.0e+10   0.001
```

"cost.dat"

```
COSTING DATA (cost.dat)
 MM cost fits, ucmm($/kgU) = u1*[umm(kgU)]**pnu:   1(cr)     2(kr)    3(tr)
  o mm cost coefficent, u1                        1.35e-22  3.13e-11  6.54e-06
  o mm cost exponent, pnu                         2.48      1.26      0.71
  o mm miniumum cost, ucmmmin                      50.       50.       50.
 Uranium mm cost scaling, iucost                  3
 Unit cost of conversion, uccv($/kgU)             5.
 Unit cost of separative work, ucsw($/kgU/SWU)   100.
 Unit cost of fabrication
  o uox fabrication, ucffux($/kgU)                200.
  o mox fabrication, ucffmx($/kgU)                400.
 Unit cost of sf transport, uctr($/kgHM)          10.
 Unit cost of plutonium
  o pu from weapons storage, ucpuwp($/kgPu/yr)    10000.0
  o pu from reprocessing, ucpurp($/kgPu/yr)       10000.0
  o pu from plutonium storage, ucpups($/kgPu/yr)  10000.0
 Unit costs of storage:
  o du storage, ucdus($/kgU/yr)                   5.
  o bks storage, ucbks($/kgU/yr)                  3.
  o cs  storage, uccs($/kgHM/yr)                  20.
  o is  storage, ucisu($/kgHM/yr)                 15.
  o is  storage, ucisp($/kgHM/yr)                 15.
  o ru storage, ucrus($/kgHM/yr)                  5.
  o rpu storage, ucpus($/kgHM/yr)                 1.
  o fp storage, ucfps($/kgHM/yr)                  20.
  o heu storage, ucwus($/kgHM/yr)                 100.
  o wpu storage, ucwps($/kgHM/yr)                 10.
 Unit costs of disposal:
  o direct disposal (dd1), ucdd1($/kgHM/yr)       10.
  o direct disposal (dd2), ucdd2($/kgHM/yr)       15.
  o direct disposal (dd3), ucdd3($/kgHM/yr)       20.
  o separated plutonium (pud), ucpud($/kgHM/yr)   50.
```

o fission product (fpd1), ucfpd1($/kgHM/yr)	10.							
o fission product (fpd2), ucfpd2($/kgHM/yr)	15.							
o fission product (fpd3), ucfpd3($/kgHM/yr)	20.							
Unit cost of heu blending, ucbl($/kgU)	40.							
Unit costs of reprocessing:								
o contract (c)								
- once-through uox, ucrpcux($/kgU)	1000.							
- recycled mox, ucrpcmx($/kgU)	1500.							
- fbr blanket, ucrpcbmx($/kgU)	1500.							
o free market (m)								
- once-through uox, ucrpmux($/kgU)	1200.							
- recycled mox, ucrpmmx($/kgU)	1800.							
- fbr blanket, ucrpmbmx($/kgU)	1800.							
Unit costs of separations:								
- spent fuel separations, ucsep($/kgHM)	1000.							
Generation cost paramters k =	OT	MX	HTGR	FBR	IFR	ATW1	ATW2	ATW3
o unit total cost, utc(k,$/We)	2.50	2.00	5.00	5.00	5.00	5.00	5.00	5.00
o fixed charge rate, fcr(k, 1/yr)	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00
o O&M, fom(k, 1/yr)	0.03	0.03	0.03	0.04	0.05	0.05	0.05	0.05
Discount rate, dr(1/yr)	0.05							
Discount rate for proliferation, drp (1/yr)	0.05							

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