Analysis of Alternative Light Water Reactor (LWR) Fuel Cycles

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ANALYSIS OF ALTERNATIVE LIGHT WATER
REACTOR (LWR) FUEL CYCLES

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ABSTRACT

Nine alternative LWR fuel cycles are analyzed in terms of the isotopic content of the fuel material, the relative amounts of primary and recycled material, the uranium and thorium requirements, the fuel cycle costs and the fraction of energy which must be generated at secured sites. The fuel materials include low-enriched uranium (LEU), plutonium-uranium (MOX), highly-enriched uranium-thorium (HEU-Th), denatured uranium-thorium (DU-Th) and plutonium-thorium (Pu-Th).

The analysis is based on tracing the material requirements of a generic pressurized water reactor (PWR) for a 30-year period at constant annual energy output. During this time period all the created fissile material is recycled unless its reactivity worth is less than 0.2% uranium enrichment plant tails.

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Among the authors, R. L. Aaberg performed the analysis of Cycles A and B as well as programming the routine which computes the separative work, and migration of the minor isotopes in the enrichment cascade. A. J. Boegel performed the analysis of Cycle C, and wrote the fuel cycles description section of the report. He also contributed to the original plan of analysis. C. M. Heeb guided the revisions to the LEOPARD code, and was responsible for the overall analysis effort. U. P. Jenquin studied the effect of resonance interference and evaluated the ^{232}Th resonance integral measurements and calculations. D. A. Kottwitz created the new ^{240}Pu resonance treatment in LEOPARD. M. A. Lewallen selected the unit cost assumptions and wrote the section on uranium and thorium requirements. E. T. Merrill wrote the original flow diagrams which form the basis of the analysis, did the cash flow analysis for the power costs, and wrote the economics sections of this report. A. M. Nolan performed the analysis of Cycles D, E, G and H, wrote the fissile loading search subroutines in LEOPARD, wrote the introduction, and the section on non-proliferation and diversion resistance.

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1.0 INTRODUCTION

1.0 INTRODUCTION

Within the last 2 years the Federal government has made major policy changes in regard to light water reactor (LWR) spent fuel reprocessing and plutonium recycle. The objective of current U.S. policy is to control the proliferation of nuclear weapons by limiting the spread of sensitive technologies and facilities and by limiting access to materials that could be used in weapons. The Nonproliferation Alternative Systems Assessment Program (NASAP) and the International Fuel Cycle Evaluation (INFCE) studies were initiated to aid in selecting future nuclear fuel cycle policies.

Currently, 60 different fuel cycles are being studied under NASAP and INFCE programs. As part of the Fuels Refabrication and Development Program, PNL 2779, "Description of Alternative Steady-State Fuel Cycles for LWRs," presented the results of an analysis of several diverse fuel cycles in a steady-state mode. The present study presents the results of an analysis of a 30-year transition to the steady-state for each of the fuel cycles.

This 30-year analysis deals with pressurized water reactors (PWRs) only. Boiling water reactors should yield similar overall fuel cycle mass flows. The concept of a closed cycle is used, wherein all fissile material recovered from spent fuel is either recycled into fresh fuel or retired to waste when its net reactivity worth is equal to or less than enrichment plant tails. Thus, fissile material is introduced into the cycle only through the enrichment cascade. The analysis uses five generic fuel types in nine different fuel cycles. The method of analysis includes all major effects which relate to fuel cycle comparisons.

The factors considered in this study that relate to the better definition of alternative nuclear fuel cycles are compatibility with present PWRs, resource utilization, economics, and proliferation and diversion resistance. The aim of this study is to provide information on these factors that will define the compositions of fuel materials involved in these cycles and to establish economic, raw material, and proliferation resistance trade-offs between the proposed fuel cycles.

2.0 TECHNICAL APPROACH

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A generic PWR design⁽²⁾ was chosen to generate a basis for LWR fuel cycle comparisons. The fuel requirements for a 30-year period were determined for each closed fuel cycle. Fissile material generated during irradiation was recycled or discarded as required by the fuel cycle scenario. The subsequent irradiation of the refabricated fissile material was followed through as many reprocessing and refabrication steps as were required until the end of the 30-year period was reached or the fissile content was reduced to the reactivity equivalent of tails. Fissile material was not stored or exchanged with an outside source to avoid the problem of assigning a dollar value to the fissile content for the economic evaluation. The only source of fissile material was therefore natural uranium. Enrichment plant tails were considered to have no economic value. Each closed cycle system accepted natural uranium feed material and reduced it to a zero value commodity by the energy extraction process. Comparisons were then made between various fuel cycle schemes on the basis of relative fuel material requirements, the cost of producing energy and the amount of energy that could be generated outside a secured center.

The generic reactor was assumed to be fueled with UO_2 on an equilibrium cycle. A fuel reload of the alternative cycle material was introduced each year for 5 years. Recycled material from the first load was introduced on the sixth year. Annual refueling was assumed. The residence time of the fuel in the reactor was 3 years (one-third core replacement each year). Two years were allowed for cooling, reprocessing and refabrication for a total of 5 years delay before charging back into the reactor. The properties of the generic reactor are shown in Table 2.1.

Fuel compositions were determined using a cell burnup code which is an extensive revision of LEOPARD.⁽³⁾ Initial fissile loadings were determined by matching the core end-of-life (two-thirds of the goal exposure) k -infinity values with the standard cycle low-enriched UO_2 value. Neutron cross sections were ENDFB-IV⁽³⁰⁾ values except for ^{238}U and ^{232}Th . Neutron absorption in these important fertile isotopes was based on experimental determinations of the resonance integrals.⁽⁴⁾ Accuracy of the discharged fuel isotopic content was addressed by comparisons between predicted and measured isotopic content of UO_2 , $\text{PuO}_2\text{-UO}_2$, and $\text{UO}_2\text{-ThO}_2$ discharged fuel.^(5,6,7)

TABLE 2.1. Generic Reactor Description

	<u>Datum</u>	<u>Units</u>
Thermal Power	3400	MW
Fuel Assemblies	193	Assemblies
Rod Array	17 x 17	Rods
Assembly Pitch	8.4853	Inches
Rods Per Assembly	264	Rods
Rod Pitch	0.4960	Inches
Rod O.D.	0.3740	Inches
Clad I.D.	0.3290	Inches
Pellet O.D.	0.3225	Inches
Pellet Density	95.0	% Theoretical
Core Loading	89.25	MTHM
Plant Capacity Factor	70.5	%
Discharge Exposure	30,000	MWd/MTHM
Net Thermal Efficiency	32.5	%

The fissile content of denatured uranium was selected by requiring that a minimum critical mass of 200 kg would be considered unusable as a weapons material. This leads to approximately 20% pure ^{235}U through intermediate values of ^{235}U - ^{233}U combinations shown to 12% for pure ^{233}U .

The changes in concentration of the minor isotopes ^{234}U and ^{236}U during the enrichment process were calculated using the equations developed by A. de La Garza.⁽⁸⁾

The economic analysis was based on comparisons between that portion of the total cost of electric power due to nuclear fuel expenditures. Fuel cycle costs in mills per kilowatt hour were calculated using an analysis of cash flows over the assumed 35-year period from initial purchase to final disposition. Two interest rate assumptions were made. In the typical-risk situation up to 50% of the financing could be from bonds. This results in an effective rate of 9.3%. For the high-risk condition 100% equity funding was assumed for an effective interest rate of 15%.

Several means of artificially increasing the radioactivity of fuel material to discourage diversion have been proposed. To assess the economic impact of these spiking proposals, a set of costs⁽¹⁴⁾ for fully remote fabrication were included along with a more usual set of fabrication costs where the degree of remote fabrication depends on the radioactivity of the unspiked fuel material.

3.0 SUMMARY OF RESULTS

3.0 SUMMARY OF RESULTS

Nine LWR fuel cycles were analyzed including the plutonium recycle in uranium. These were compared to the present once-through low enriched uranium fuel cycle. The items used to make the comparison were:

- 1) Uranium requirement,
- 2) Separative work requirements,
- 3) Fuel cycle costs,
- 4) The fraction of the energy generation from material which could be used in nuclear weapons and would therefore be subject to a higher level of security.

The use of denatured uranium of less than 20% fissile content with thorium as the primary fertile material (DU-Th) and without plutonium recycle can lower the uranium requirement by 28% compared to the present once-through low enriched (LEU) uranium fuel cycle. However, secured center reprocessing would be required to recover the bred ^{233}U for recycle. The cost penalty of DU-Th is in the 3 to 3.8 mill/kWhr ranges compared to the once-through LEU cycle.

With LEU plutonium recycle, no significant reduction in uranium requirements can be realized when thorium is used in place of ^{238}U as the fertile material for recycled plutonium. The cost penalty for using thorium in this mode is of the order of three mills per kilowatt hour. The uranium savings associated with any plutonium recycle are in the 31 to 34% range. However, recycle of bred ^{233}U from Pu-Th can lower the secured center energy generation required by 10% when compared to Pu-U recycle.

Recycle of plutonium from the DU-Th cycle as either Pu-Th or Pu-U increases the uranium savings from 28% to roughly 40%. Selecting Pu-Th as the DU-Th recycle mode required only 9% energy generation in a secured center which is the lowest of any plutonium recycle mode. Recycling DU-Th plutonium reduces the DU-Th cost penalty by three-tenths of a mill per kilowatt hour.

The maximum uranium saving of 41% comes from using highly-enriched uranium (93% ^{235}U) in thorium. However, this option also has the largest cost penalty (1 to 4 mills/kWhr). Separative work requirements are actually about

3% less for this cycle than for LEU. Of course, all of the HEU-Th energy generation would be in the secured center category.

Plutonium recycle in the context of the LEU cycle saves roughly 20% separative work. DU-Th has an 8% separative work penalty. Plutonium recycle in DU-Th can reduce this penalty to a savings of roughly 10%, but none of the DU-Th plutonium recycle options reaches the 20% separative work saving of LEU plutonium recycle.

4.0 ALTERNATIVE FUEL CYCLE DESCRIPTIONS

4.0 ALTERNATIVE FUEL CYCLE DESCRIPTIONS

In all of the fuel cycles the fissile isotope, ^{235}U , was used in either uranium or thorium or both to initiate each fuel cycle. Diagrams illustrating the flow of fissile and fertile material within each fuel cycle can be seen in Figures 4.1 through 4.3. Each diagram may be interpreted as a single PWR, producing 1105 MWe annually, with its core composed of the fractional amounts of each fuel type as indicated. It should be noted here that each reprocessing step indicated in the detailed fuel cycle diagrams is also the fuel fabrication step. Detailed mass flow diagrams are shown in Appendix A for each of these cycles.

LEU, Pu-U RECYCLE, CYCLE A

Cycle A in Figure 4.1 depicts the low-enriched uranium (LEU) fuel cycle with plutonium recycle. Initially, natural uranium is enriched to approximately 3% ^{235}U . After an in-core residence time of 3 years, the LEU fuel is discharged and subsequently reprocessed. The separated plutonium is blended with a uranium stream from the enrichment cascade to the desired enrichment. An option available to the nuclear industry is to blend this recycled plutonium with uranium containing 0.4-0.5% ^{235}U instead of depleted uranium or tails containing 0.2% ^{235}U . This has three primary advantages: 1) more energy can be derived from the same amount of plutonium, 2) at goal exposure the uranium can be discarded because it has the same ^{235}U content as that of tails, and 3) the plutonium is concentrated in less fuel which reduces its fabrication penalty. This option was investigated in Cycle A and Cycle H only.

The separated uranium from the initial LEU fuel is sent back to the enrichment cascade for re-enriching. The increased concentrations of ^{234}U and ^{236}U in the product stream as a result of the re-enrichment process were calculated using the methodology described by A. de la Garza.⁽⁸⁾

One basic assumption of each fuel cycle was a constant annual energy production. In Cycle A, the energy derived from the bred plutonium plus the recycled uranium is insufficient to meet the total energy generation assumed

FUEL CYCLE A
LEU, Pu IN PARTIALLY DEPLETED U

FUEL CYCLE B
LEU, PuTh, ^{233}U IN ^{238}U

FUEL CYCLE C
LEU, PuTh, DENATURED ^{233}U IN Th

FIGURE 4.1. Alternative Fuel Cycles A Through C

constant for a given increment of time. For this reason enough natural uranium is enriched, assuming 0.2% ^{235}U tails, at each step and blended with the re-enriched uranium to provide the necessary amount of fuel to make up this deficit.

The plutonium bred in the LEU fuel plus the plutonium generated in the mixed oxide fuel are blended to the desired fissile content with 0.5% ^{235}U tails at each succeeding step of the 30-year cycle. This fuel type must be reprocessed, fabricated, and irradiated in a secured center.

LEU, Pu-Th, ^{233}U IN ^{238}U , CYCLE B

Figure 4.1, Fuel Cycle B, illustrates a means of starting a thorium cycle. The initial fuel type is low-enriched uranium (LEU) from which the bred plutonium is recycled and blended to the desired fissile loading with fresh thorium. The uranium from the LEU fuel is recycled back to the enrichment cascade for re-enriching. Enough natural uranium is enriched and blended with the enriched uranium to the desired fissile content to maintain a constant energy production.

The bred ^{233}U from the $\text{PuO}_2\text{-ThO}_2$ fuel is blended with depleted uranium as a denaturant from the enrichment cascade (0.2% ^{235}U) to the desired fissile content. The irradiated thorium can be stored until its radioactivity is at or below maximum working levels, at which time it can be recycled, however thorium recycle was not assumed for the purposes of this study. For the remainder of this fuel cycle, the bred plutonium from this fuel plus the plutonium from the $\text{PuO}_2\text{-ThO}_2$ and that bred in the LEU are blended with thorium to the desired fissile content.

LEU, Pu-Th, DU-Th, CYCLE C

Figure 4.1, Fuel Cycle C, depicts another method for initiating a thorium cycle. Essentially, this cycle is similar to Cycle B with the exception that in this instance, the bred ^{233}U from $\text{PuO}_2\text{-ThO}_2$ is combined with the denaturant of depleted uranium and blended with thorium to form denatured $\text{UO}_2\text{-ThO}_2$. Similarly, the plutonium bred from the three fuel types illustrated (LEU, Pu-Th, DU-Th) is blended with thorium to the desired fissile content to form $\text{PuO}_2\text{-ThO}_2$ fuel. The enrichment cascade is operated at 0.2% ^{235}U tails throughout the lifetime of this cycle.

LEU, Pu-Th, HEU-Th, CYCLE D

Fuel Cycle D, illustrated in Figure 4.2, is a variation of the thorium cycle depicted by Cycle C. The difference here is that instead of blending with depleted uranium the bred ^{233}U and ^{235}U are combined directly with thorium. The result is a high-enriched uranium-thorium (HEU-ThO_2) fuel; the ^{232}U in the ^{233}U may make the fuel radioactive enough to permit its use outside a secured center, but this was not assumed.

HEU-Th, CYCLE E

Figure 4.2, Fuel Cycle E, illustrates another type of thorium cycle. Fully-enriched uranium (93% ^{235}U) is blended with thorium to the desired fuel enrichment. For the first 5 years each reload consists entirely of this fuel. However, once recycle has been initiated, this fuel type contributes roughly 30% to the total reload. The other 70% comes from the recycled bred ^{233}U and ^{235}U blended with fresh thorium. The irradiated thorium is stored until such time as its radioactivity has decayed to maximum workable levels. Since both fuel types comprising this cycle contain high-enriched uranium, the entire cycle must be deployed within a secured center unless ^{232}U spiking is considered sufficient.

DU-Th, Pu DISCARD, CYCLE F

Figure 4.2, Fuel Cycle F, depicts another thorium cycle but with plutonium throwaway. Initially, 20% ^{235}U enriched uranium is blended with thorium to the required fuel enrichment. Once recycle starts, the bred ^{233}U and ^{235}U are blended with thorium to the desired fuel enrichment. At each time increment, the fresh 20% ^{235}U -Th and the recycled U-Th are kept separate from one another. Each recycled fuel type is also segregated from the other, as is illustrated in Appendix A. Associated with each fuel type, whether fresh or recycled, is a separate reprocessing block. This does not imply separate reprocessing plants, but rather segregated or campaign reprocessing. The last fuel type in the recycle chain is the 4.6% fissile uranium in thorium. This material is not reprocessed and recycled after irradiation because of the high concentrations of neutron absorbers (^{234}U and ^{236}U) within the uranium.

FIGURE 4.2. Alternative Fuel Cycles D Through F

DU-Th, Pu-Th, CYCLE G

Cycle G, illustrated in Figure 4.3, is basically the same cycle as F, except that the bred plutonium is recycled and blended with thorium. The cycle starts with the irradiation of 20% ^{235}U enriched uranium blended with thorium to the required fuel enrichment. Two types of fissile fuel are bred from this initial fuel type: ^{233}U and $^{239,241}\text{Pu}$. Each type of fissile material is recycled and blended with thorium to the desired enrichment to form $\text{UO}_2\text{-ThO}_2$ and $\text{PuO}_2\text{-ThO}_2$. At each time increment, enough fresh 20% ^{235}U enriched uranium-thorium fuel is fabricated so that the annual energy production remains constant.

As in Cycle F, the fresh 20% ^{235}U enriched fuel plus each individual recycled fuel type are segregated from one another. In the last 5 years of this particular fuel cycle, a portion of the bred plutonium is blended with some recycle uranium to form $\text{PuO}_2\text{-UO}_2$. This recycle uranium contains enough fissile material to warrant its use in a fuel but not enough to reach goal exposure. For this reason, plutonium is blended with the uranium to the required fissile content to reach goal exposure. The uranium is then retired from the system since its effective fissile content is approximately that of tails (0.2% ^{235}U). The remainder of the bred plutonium is blended with thorium to the desired enrichment.

DU-Th, DU-Th, Pu-U, CYCLE H

Figure 4.3, Cycle H, depicts still another thorium cycle. Initially, 20% ^{235}U enriched uranium is blended with thorium to the required fissile content. After reprocessing, the bred ^{233}U and ^{235}U are blended with thorium to form the first recycled $\text{UO}_2\text{-ThO}_2$ fuel. The bred plutonium from the initial fuel type is blended with enough uranium taken from the enrichment cascade at about 0.4% ^{235}U to form $\text{PuO}_2\text{-UO}_2$ fuel at the desired enrichment. The plutonium is recycled and blended with plutonium bred from the uranium-thorium fuels to make up the next recycled plutonium-uranium fuel.

All fuel types within this cycle are segregated from one another. The remainder of the natural uranium is enriched to 20% ^{235}U at 0.2% tails. After four recycles, this uranium is retired from the system since its effective fissile content has been reduced to that of uranium tails (0.2% ^{235}U).

FUEL CYCLE H
DENATURED URANIA-THORIA $\text{PuO}_2\text{-UO}_2$ RECYCLE

FUEL CYCLE I
ENRICHED URANIA-THORIA, URANIA WITH Pu DISCARD

FIGURE 4.3. Alternative Fuel Cycles G Through I

HEU, LEU, LEU, CYCLE I

Cycle I shown in Figure 4.3 offers some nonproliferation advantages in that all bred plutonium is discarded while recycle occurs only for bred ^{233}U material. Initially, two fuel types are fabricated: 1) highly enriched uranium (93% ^{235}U) blended with thorium to the desired enrichment, and 2) low-enriched uranium-oxide (3% ^{235}U). At discharge the bred plutonium is discarded. The bred ^{233}U and ^{235}U from the HEU-Th fuel is blended to the desired enrichment with the uranium discharged from the LEU fuel as the denaturant. Fresh LEU fuel and HEU-Th fuel provide the remainder of the energy generation for the 6-10 year time block. The 3% ^{235}U enriched uranium oxide is employed in this time frame so that the desired fissile content of 3% $^{233,235}\text{U}$ will be attained in the recycled uranium oxide for the 11 to 15 year time frame.

The discharged uranium from the low-enriched UO_2 plus the bred ^{233}U and ^{235}U from the HEU-Th fuel are reprocessed and blended to form low-enriched $^{233,235}\text{U}$ fuel. Beyond 16 years enough additional fissile material in the form of fresh 93% ^{235}U uranium oxide is blended to this fuel material to increase its fissile content to 3%. From this point on, this fuel cycle is composed of about 55% LEU and 45% HEU-Th fuel.

The HEU-Th fuel would have to be produced, fabricated and irradiated in a secured center. The recycled low-enriched uranium oxide, however, is completely denatured and available for export. The plutonium bred in this fuel type would be sufficiently contaminated at discharge with fission products and uranium isotopics as to make its use as weapons material largely prohibitive.

RECYCLE ENERGY CONTRIBUTIONS

The fraction of the total energy produced from the primary fuel and subsequent recycles is shown in Figures 4.4 to 4.12 for each of the fuel cycles. The relative fraction of energy is shown for the six 5-year time periods. Steady-state is reached when the relative proportions of primary and recycle material energy fractions do not change significantly. Figures 4.4 through 4.8 show that the LEU cycles with recycle are essentially at steady-state after five cycles. Cycle E, the highly enriched uranium cycle, is also at steady-state at the end of the 30-year period. The dilute uranium-thorium cycles F, G and H are not at steady-state even after six irradiation steps.

FIGURE 4.4. Fraction of Energy Contribution by Fuel Type for Cycle A

FIGURE 4.5. Fraction of Energy Contribution by Fuel Type for Cycle B

FIGURE 4.6. Fraction of Energy Contribution by Fuel Type for Cycle C

FIGURE 4.7. Fraction of Energy Contribution by Fuel Type for Cycle D

FIGURE 4.8. Fraction of Energy Contribution by Fuel Type for Cycle E

FIGURE 4.9. Fraction of Energy Contribution by Fuel Type for Cycle F

FIGURE 4.10. Fraction of Energy Contribution by Fuel Type for Cycle G

FIGURE 4.11. Fraction of Energy Contribution by Fuel Type for Cycle H

FIGURE 4.12. Fraction of Energy Contribution by Fuel Type for Cycle I

5.0 RELATIVE URANIUM-THORIUM REQUIREMENTS

5.0 RELATIVE URANIUM-THORIUM REQUIREMENTS

In comparing alternative fuel cycles, one pertinent parameter is the relative level of resource utilization. For the fuel cycles discussed in this document the relative utilization of resources is summarized in Tables 5.1 and 5.2. Table 5.1 displays a constant energy (23.4 GWYe) comparison of uranium, thorium, and total heavy metal utilization. The once-through LEU cycle uses 28% more uranium than the next most resource-intensive cycle, the DUTh with plutonium discard. The least use of uranium is made by the HEU-Th system. However, this system is the most intensive user of thorium and its total heavy metal usage is within the range of the other cycles. Total resource usage of heavy metals is bounded by the maximum LEU once-through and the minimum LEU-Pu recycle system. Note that for the purposes of analysis, thorium was not recycled. If thorium recycle was included, the HEU-Th cycle would be the least intensive resource user.

For the alternative cycles analyzed, a maximum of 4.1 million separative work units was used in the HEUTh-LEU cycle. The minimum value of separative work (2.9 million separative work units) was required in the LEU-Pu recycle case.

The last two rows of Table 5.1 show relative resource utilization based on electrical production (GWYe) per unit of resource. Resource units include both heavy metals and uranium. For the heavy metal basis, the most efficient cycle is the LEU-Pu recycle system. This system has a utilization level of 7.6 GWYe per 1000 MT heavy metals. The minimum utilization is achieved on the LEU-once-through cycle (4.9 GWYe/1000 MTHM).

When the resource basis is uranium, the most efficient use of uranium is achieved in the HEU-Th cycle. This system achieves an energy conversion of 8.5 GWYe per 1000 MT uranium. As on the heavy metal basis, the least utilization is achieved on the LEU-once-through cycle. This cycle achieves a utilization level of 4.9 GWYe per 1000 MT uranium.

Table 5.2 shows relative energy production assuming a limited uranium reserve of 3.7 million short tons of U_3O_8 . Theoretically achievable energy values are shown as well as total usage of heavy metals. Material required to start-up and reach steady-state on the present LEU once-through cycle is not included.

TABLE 5.1. Resource Requirements for Generic Reactor Lifetime (23.4 GWYe)

Resource	Once Through LEU	LEU-Pu Recycle	LEU PuTh-LEU	LEU-PuTh -DUTH	LEU-PuTh -HEUTH	HEU-Th	DUTH-Pu to Waste	DUTH-PuTh	DUTH-PuU	HEUTH-LEU
	A'	A	B	C	D	E	F	G	H	I
Natural Uranium (MT)	4807	3309	3205	3171	3172	2846	3435	2903	2873	3326
Thorium (MT)(a)	--	--	95	135	148	771	536	541	457	346
Total Resource, U+Th (MT)	4807	3309	3300	3306	3321	3617	3976	3444	3330	3672
Natural Uranium Usage Relative to LEU Throwaway	1.000	0.688	0.667	0.660	0.660	0.592	0.715	0.604	0.598	0.692
Total Resource Usage Relative to LEU Throwaway	1.000	0.688	0.686	0.688	0.691	0.752	0.827	0.716	0.693	0.764
SWU's at 0.2% Tails	3.77+6	3.08+6	3.11+6	3.09+6	3.14+6	3.66+6	4.06+6	3.42+6	3.34+6	4.08+6
Resource Utilization (GWYe/1000 MTHM)	4.87	7.06	7.08	7.07	7.04	6.46	5.88	6.79	7.02	6.36
Resource Utilization (GWYe/1000 MTU)	4.87	7.06	7.29	7.37	7.37	8.21	6.80	8.05	8.13	7.03

(a) Thorium is not recycled in the analysis.

TABLE 5.2. Theoretical Energy Production for 3.7×10^6 ST U_3O_8 (2.85×10^6 MTU)

Resource	Once Through LEU	LEU-Pu Recycle	LEU PuTh-LEU	LEU-PuTh -DUTh	LEU-PuTh -HEUTh	HEU-Th	DUTh-Pu to Waste	DUTh-PuTh	DUTh-PuU	HEUTh-LEU
	A'	A	B	C	D	E	F	G	H	I
Energy Production (GWYe)	14,000	21,700	21,500	21,800	21,800	24,300	19,500	23,000	23,500	20,400
Required Thorium (MT)	--	--	8.69+4	1.25+5	1.37+5	7.97+5	4.43+5	5.29+5	4.56+5	2.99+5
Heavy Metals Used (MT)	2.85+6	2.85+6	2.94+6	2.98+6	2.99+6	3.65+6	3.29+6	3.38+6	3.31+6	3.15+6

6.0 RELATIVE PROLIFERATION AND DIVERSION RESISTANCE

6.0 PROLIFERATION AND DIVERSION RESISTANCE

The aim of proliferation and diversion resistance is to keep fissile material from being diverted and used for making nuclear weapons by nations or subnational groups with no present nuclear weapons capability. While it is not possible to ensure completely against diversion of fissile materials, diversion may be made so difficult and dangerous and so politically and economically unattractive that some other source of fissile material (e.g., uranium deposits within the national borders) would be preferred by a nation or subnational group intent on obtaining nuclear weapons. Proliferation and diversion resistance can be enhanced either by physical security methods or by intrinsic characteristics of the nuclear fuel such as high radiation levels or lowered fissile isotope content (denaturing), which make the material unattractive to diversion efforts. These fuel characteristics will be discussed here.

6.1 METHODS FOR ENHANCING PROLIFERATION AND DIVERSION RESISTANCE

There are basically two approaches to making fissile material less attractive to diversion or weapons production. The first of these is spiking, whereby a high-level radiation source is mixed with the nuclear fuel. This method increases resistance to the diversion of fissile material, since heavy shielding and remote handling of the fissile material or fuel assemblies would be required to protect personnel involved in any diversion attempts. The goal in spiking is to attain a dose rate of at least 100 R/hr/kg of fissile material at a distance of 1 meter. This criterion was used in a February 1978 SRL report⁽⁹⁾ and is a reasonable figure of merit.

If the material could be diverted, however, a small, clandestine reprocessing facility could separate the radiation source from the heavy metals, leaving the fissile material included with the heavy metals available for use in constructing nuclear weapons. Thus, spiking can offer some subnational diversion resistance, but the obstacles to proliferation by national states are not as great as with isotopically denatured fuel.

The other approach to proliferation resistance is that of denaturing. This involves increasing the weight fraction of nonfissile isotopes of uranium

or plutonium so that weapons-grade concentrations of fissile materials cannot be achieved without an isotopic separations process that requires advanced technology. The denaturing can enhance proliferation resistance by limiting the reactivity of the material or by causing the material to heat and melt when the elemental density reaches some specific level. A discussion of the various methods of spiking and denaturing and their applicability follows.

6.2 SPIKING

Spiking involves the introduction of a high-level radiation source into fresh fuel in order to discourage diversion of the fissile materials. This can be done in three ways: a) by retaining selected fission products during reprocessing (or by adding selected fission products back to the heavy metals just after reprocessing); b) addition of an activated radiation source such as ^{60}Co after reprocessing; or c) irradiation of fresh fuel onsite after fabrication. The first of these methods is similar to the CIVEX⁽¹⁰⁾ process, in which uranium and plutonium are co-processed and certain fission products are retained with the heavy metals. These methods and their relative merits are discussed in detail in Reference 9.

There are a variety of highly radioactive fission products present in spent fuel prior to reprocessing and they are in sufficient quantity to be used as spikants in refabricated fuel. However, many of these have rather short half-lives and decay to unsuitably low levels during typical spent fuel storage times (on the order of a year). Some of the fission products with longer half-lives, such as ruthenium and cesium, are volatile and may not be compatible with fuel fabrication processes such as sintering.

A study of dose rates from certain refabricated fuels spiked with selected fission products is presented in Reference 10 and discussed in Reference 9. Assuming a CIVEX-like spent fuel reprocessing scheme that retains 15% of the ruthenium and 100% of the zirconium and niobium, rather high radiation levels can be attained, easily exceeding the NASAP criterion of 100 R/hr/kg of fissile material at a distance of 1 meter. However, the radiation levels of all the fuels analyzed fell below the criterion within 2 years of discharge from the reactor, or about 1 year after reprocessing. The only fission product

with a half-life long enough to provide high radiation levels for long periods of time is ^{137}Cs which may be incompatible with present technology fuel fabrication processes.

Pre-irradiation is another spiking option that was studied in Reference 10. Fresh fuel would be fabricated into fuel elements which would then undergo exposure in an irradiation facility co-located with the fuel fabrication plant. The fresh fuel would attain some small percent (e.g., about 1%) of goal exposure. This process would cause fission products to form in the fresh fuel in quantities sufficient to provide high radiation levels. Radiation levels from all of the fuels analyzed in Reference 10 exceeded the NASAP criterion for 300 days after irradiation to 1% of goal exposure. The chief disadvantage of the pre-irradiation method is that the fissile material would be vulnerable to diversion during the entire fuel fabrication process. Thus, the reprocessing, fabrication, and pre-irradiation facilities would probably need to be co-located within a secured fuel center.

The third spiking concept is the addition of an activated radiation source to the reprocessed heavy metals. Of such sources, Reference 9 concludes that ^{60}Co is probably the best candidate. This choice is based on ^{60}Co 's small reactivity penalty, long half-life (5.3 years) and high radioactivity per unit mass. Volatility during the fuel pellet sintering process is not a problem with ^{60}Co . Indeed, when all three spiking methods discussed above are compared, spiking with ^{60}Co is probably the most effective method.

6.3 DENATURING

The denaturing of fissile material approaches proliferation resistance differently. Instead of relying on high radiation levels for deterrence, denaturing requires that large quantities of material would need to be diverted before the critical mass needed for a weapon could be obtained. High technology isotopic separation processes would be necessary to separate the fissile materials in order to produce a smaller weapon. In this study, a fuel is considered to be denatured if the mass of heavy metal needed to produce a nuclear weapon exceeds 200 kg.

Denaturing of uranium fuels is accomplished by the inclusion of a sufficient quantity of the nonfissile isotopes ^{234}U , ^{236}U , and ^{238}U . For ^{235}U fuels, an enrichment of 20% or less ^{235}U in ^{238}U will produce a denatured fuel. Enrichment in ^{233}U fuels must be held to 12% or less for denaturing, owing to the higher reactivity of ^{233}U compared to ^{235}U . For fuels containing both ^{233}U and ^{235}U , the maximum total fissile content in uranium for denaturing will be between 12% and 20%, depending on the relative quantities of the two fissile isotopes.

Uranium-235 can be separated from ^{238}U using current gas centrifuge or diffusion cascade technology. Uranium-233, however, presents a more difficult separation problem because of the ^{232}U that occurs with ^{233}U at a concentration up to 1300 ppm in uranium. The ^{232}U decay chain includes the high-energy gamma emitter ^{208}Tl . Heavy shielding and remote operations and maintenance procedures would be required for a facility which isotopically separates ^{233}U .

The thermal denaturing of plutonium fuels can be accomplished by including 5-10% ^{238}Pu . The heat produced by ^{238}Pu alpha decay may be sufficient to cause melting of weapons-grade plutonium components, making practical weapons difficult to construct. The technology for isotopic separation of useful quantities of plutonium is not available; separating ^{238}Pu from ^{239}Pu would, in any case, be an order of magnitude more difficult using gaseous diffusion or the centrifuge process than uranium enrichment since the mass numbers differ by only one mass unit.

Plutonium produced in recycled DU-Th fuels is about 7.5% ^{238}Pu , probably enough for denaturing. Plutonium-238 is produced mostly by neutron capture in ^{237}Np , which in turn is produced by neutron capture in ^{236}U . The ^{237}Np produced in recycled DU-Th fuels can be extracted in reprocessing and added to nonrecycle DU-Th or LEU fuels to enhance ^{238}Pu production. Without addition of ^{237}Np , the plutonium produced in LEU fuels is about 1% ^{238}Pu , while that produced in nonrecycle DU-Th fuels is about 3% ^{238}Pu .

Denaturing of uranium and plutonium fuels as described above can be an effective barrier to proliferation. Denaturing can discourage diversion without requiring much additional shielding or remoteness of operation and maintenance at fuel fabrication facilities.

6.4 COMPARISON OF SPIKING AND DENATURING

As discussed above, the two basic approaches to making nuclear fuel more proliferation resistant are spiking and denaturing. The applicability of these approaches, their relative costs, and their effectiveness can be compared.

The objective of spiking is to discourage the actual diversion of fissile material by the presence of high radiation levels. Heavy shielding would be needed to prevent exposure of personnel to a lethal dose of radiation--the material could not be smuggled out in a briefcase, for instance. Spiking with fission products or ^{60}Co , however, also requires that fuel fabrication operations and plant maintenance be done remotely due to the presence of the high radiation levels. The cost of remote fabrication and maintenance is much higher than that of contact operations and can be a major contributor to the costs of a particular fuel cycle. The fabricated fuel elements would also need a great deal of shielding, raising transportation charges and making the fuel, in general, more difficult to handle. It should be noted that fuels containing ^{233}U will require remote operations and plant maintenance in any case, since these fuels have higher intrinsic radiation sources than ^{235}U or plutonium fuels; the addition of a spikant will only increase the shielding requirements.

Pre-irradiation of fresh fuel has a further disadvantage in that throughout the fabrication process, the fuel includes no high-level radiation sources and thus is vulnerable to diversion until irradiation occurs. This requires that the reprocessing, fabrication, and irradiation take place within a secured facility. The design, construction, and operation of such a facility are more complex and more costly than for nonsecured facilities.

Denaturing, on the other hand, does not directly discourage the diversion of fissile materials. However, even if the material is diverted, it cannot readily be used for nuclear weapons without undergoing isotopic separation processes. Isotopic separation is an expensive, high-technology process, even for the easiest case of enrichment of ^{235}U . A great deal of shielding and more complex operations and maintenance procedures are necessary for ^{233}U enrichment. The technology for plutonium enrichment in practical quantities is not available.

Since no high-level radiation sources are present in denatured fuels, fabrication operations and plant maintenance would be less expensive than those for spiked fuels for all but the fuels containing ^{233}U . Less shielding than that required for spiked fuels would be needed for the denatured fuels.

A further advantage of denaturing is that the isotopes used to denature the fissile isotopes have very long half-lives, much longer than the 5.3 year half-life of ^{60}Co . Uranium-238 has a 4.5 billion year half-life, while ^{238}Pu has a half-life of 87.7 years. Thus, while radiation levels from ^{60}Co may decrease noticeably over typical storage times, the denaturing abilities of ^{238}U and ^{238}Pu will remain essentially unchanged over any reasonable storage times.

6.5 APPLICABILITY OF DENATURING TO THE NINE FUEL CYCLES

The present study concerns itself with nine closed fuel cycles using five generic oxide fuel types: low-enriched uranium (LEU), high-enriched uranium in thorium (HEU-Th), denatured uranium in thorium (DU-Th), plutonium in uranium (Pu-U), and plutonium in thorium (Pu-Th). Denaturing has varying degrees of applicability to these five fuels and nine cycles.

All LEU and Du-Th fuels are denatured, since the fissile uranium weight fractions are so low as to require more than 200 kg of uranium to attain critical mass. HEU-Th fuels are not denatured and must be protected in a secured fuel center unless spiked. Recycle of the plutonium formed during burnup of the uranium fuels provides the material for Pu-U and Pu-Th fuels. Unburned plutonium from these latter fuels is also recycled. The plutonium produced in LEU fuels is not sufficiently high in ^{238}Pu content to be denatured. DU-Th fuels produced more ^{238}Pu , but only the recycle DU-Th fuels (which start with some ^{236}U content) produce enough ^{238}Pu to denature plutonium.

Fuel materials as discharged from the reactor contain extremely high radiation sources and thus such materials can be considered to be safe from diversion. When the fission products are separated from the heavy elements during reprocessing, however, the fissile material becomes vulnerable to diversion until high radiation sources are re-introduced by spiking. Denatured

uranium or plutonium, however, would not need spiking since such materials are essentially useless in weapons production and thus are not an attractive diversion target.

Reprocessing plants that handle vulnerable fissile materials will need to be located within secured fuel centers in order to provide physical security from diversion efforts. In addition, if the chosen spiking method is pre-irradiation of fresh fuel, the fuel fabrication plant and irradiation facility will also need physical protection. Ideally, the reprocessing, fabrication and irradiation facilities would be co-located to reduce security and transportation costs, and to increase diversion resistance. Only spiked or denatured fuels would be allowed to leave the secured fuel centers.

It is difficult to estimate the costs of proliferation resistance; however, it is plain that secured facilities would be more costly to build and operate than nonsecured facilities. In addition, the increased radiation levels generated by spikants would demand more personnel protection in the form of shielding and remote operations and maintenance in the fuel fabrication facility. These, too, add to the cost of energy production. So, fuel cycles that can make the least use of secured fuel centers and spiking may be more economical.

Table 6.1 compares the percentage of energy that would require a secured generation site versus nonsecured site for each of the nine fuel cycles. Fuels that require spiking for diversion deterrence would need a secured center for reprocessing and perhaps fuel fabrication as well; denatured fuels would require neither secured centers nor spiking.

It is well to remember that recycle DU-Th and HEU-Th fuels would contain significant radiation sources that, while not as intense as those of spikants, would still require extra shielding and remote processes that are not needed for fuels with lower radiation levels. Thus, it should be noted that the need for secured versus nonsecured fuel centers is only one factor in determining fuel cycle costs.

TABLE 6.1. Percent of Energy Generation Inside Secured Center

Year	Cycle ^(a)									
	A ¹	A	B	C	D	E	F	G	H	I
1-5	0.0	0.0	0.0	0.0	0.0	100	0.0	0.0	0.0	46.1
6-10	0.0	19.4	13.5	13.6	13.6	100	0.0	6.0	7.7	42.6
11-15	0.0	24.7	15.7	15.8	21.4	100	0.0	8.0	11.8	43.9
16-20	0.0	28.6	18.1	16.4	25.5	100	0.0	9.5	14.8	45.3
21-25	0.0	28.7	17.3	16.5	28.8	100	0.0	9.5	18.3	46.7
26-30	0.0	30.0	16.4	16.4	31.0	100	0.0	19.0	32.0	48.0
AV	0.0	20.2	13.5	13.1	20.1	100	0.0	8.7	14.1	45.4
% Fuel Fabrication Inside	0.0	21.9	12.7	12.3	18.8	100.0	0.0	8.6	14.0	43.3

(a) A¹: LEU Once Through
 A: LEU, Pu Recycle
 B: LEU, Pu-Th, LEU
 C: LEU, Pu-Th, DU-Th
 D: LEU, Pu-Th, HEU-Th

E: HEU-Th
 F: DU-Th, Pu Discard
 G: DU-Th, Pu-Th
 H: DU-Th, Pu-U
 I: HEU-Th, LEU

7.0 RELATIVE FUEL CYCLE COSTS

7.0 RELATIVE FUEL CYCLE COSTS

7.1 FUEL CYCLE UNIT COSTS

The unit costs used in performing the economic analysis of the alternative LWR cycles are described in Appendix C and summarized here. Table 7.1 displays the unit costs used for raw materials (U_3O_8 and ThO_2), conversion, enrichment waste disposal, and transportation.

TABLE 7.1. Unit Costs for U_3O_8 , ThO_2 , Conversion, Enrichment, Waste Disposal, and Transportation (1978 Dollars)

<u>Item</u>	<u>Unit</u>	<u>Cost (\$)</u>
U_3O_8	lb	30 to 90
ThO_2	lb	15
Conversion	kg U	5
Enrichment	SWU	90
Waste Disposal	kg HM	66 to 72
Transportation	kg HM	40 to 71
Spent Fuel Storage	kg HM	32

Table 7.2 summarizes unit costs for reprocessing while Table 7.3 shows costs for the fabrication operation. All costs are reported in 1978 dollars and are assumed to be constant (in real terms) over the period analyzed (1980-2015). An exception to this is the U_3O_8 cost which rises during the period owing to the consumption of the easily accessible and higher grade ores. The basis and rationale for these unit costs are discussed in Appendix C.

TABLE 7.2. Reprocessing Unit Costs (1978 Dollars)⁽¹³⁾

<u>Fuel Type</u>	<u>High Costs \$/kg-HM</u>	<u>Low Costs \$/kg-HM</u>
(Pu, U) O_2	350	200
(U, Th) O_2	420	300

TABLE 7.3. Fabrication Unit Costs (1978 Dollars) NASAP

<u>Fuel Type</u>	<u>Facility Operation</u>	<u>High Costs \$/kg-HM</u>
LEU	Contact Operation Contact Maintenance	140
(²³³ U, Th)O ₂	Remote Operations Remote Maintenance	620
(Pu, U)O ₂	Remote Operations Contact Maintenance	400
(Pu, U)O ₂	Remote Operations Remote Maintenance	620

As shown in Table 7.2, unit costs are provided for two types of reprocessing operations, U-Pu fuels and U-Th fuels. The "high cost" and "low cost" sets of unit costs were excerpted from the NASAP data base.⁽¹³⁾ The high and low values correspond to the unit cost range given in Reference 13 which was based on estimates from AGNS, Exxon and Dupont.

Table 7.3 displays unit costs for the fabrication of various fuel types also taken from the NASAP data base. Fuel types include LEU, ²³⁵U and ²³³UO₂ in ThO₂, and PuO₂ in UO₂. Several combinations of facility design are included. Both remote and contact operations with either contact or remote maintenance are included.

7.2 FUEL CYCLE COST COMPARISON

Fuel cycle costs were computed using the PACTOLUS⁽¹²⁾ program. The input data required in addition to the unit costs given above are listed in Table 7.4. The price of the isotopes in recycle fuel was the exchange price for the various fuel types in Fuel Cycle C as described in Appendix C. The calculated fuel cycle costs are given in Table 7.5 for the once-through LEU or throwaway cycle and the nine fuel cycles under consideration. The only fuel cycle indicated to be cheaper (about 1%) than the once-through cycle is the LEU-Pu recycle system (Cycle A) using unspiked plutonium and the low reprocessing cost. For spiked plutonium, Cycle A costs increase about 2%, and if the high reprocessing cost is used the Cycle A costs incur about 7%.

TABLE 7.4. PACTOLUS Input Data

Price of Isotopes in Recycle Fuel, \$/g		
^{233}U	59 to 65	
^{235}U	40 to 44	
^{236}U	-15	
^{239}Pu	25 to 26	
^{241}Pu	29 to 31	
^{242}Pu	-8 to -9	
Load factor		70.5%
Fabrication losses		0.5%
Reprocessing losses		0.5%
Pre-irradiation lead time		1 year
Post-irradiation log time		1 year
Fraction of cost financed by banks		55%
Interest rate on bonds		9%
Earning rate on equity (after taxes)		12%
Federal income tax rate		48%
State income tax rate		4%
Cascade tails composition		0.2% ^{235}U

TABLE 7.5. Fuel Cycle Cost Comparison

	<u>Low Cost Estimates</u>		<u>High Cost Estimates</u>	
	<u>Partitioned</u>	<u>Spiked</u>	<u>Partitioned</u>	<u>Spiked</u>
A': Once-through LEU				7.05
A: LEU-Pu Recycle	6.98	7.15	7.44	7.60
B: LEU-Pu-Th-LEU	7.21	7.31	7.66	7.76
C: LEU-Pu-Th-DU-Th	7.18	7.28	7.64	7.75
D: LEU-Pu-Th-HEU-Th	7.17	7.27	7.63	7.74
E: HEU-Th	9.60	11.40	9.97	11.78
F: DU-Th, Pu-Th to Waste	10.37		10.75	
G: DU-Th-Pu-Th	10.07	10.12	10.45	10.50
H: DU-Th-Pu-U	10.00	10.07	10.38	10.45
I: HEU-Th-LEU	8.82	10.07	9.27	10.52

The three cycles (B, C, and D) using thorium as the fertile material for plutonium recycle are only 2 or 3% more expensive than Cycle A which uses uranium. In view of the uncertainties in the unit costs, all four of these cycles may be considered essentially equivalent in cost and any one of them could be competitive with the once-through cycle. The five cycles (E, F, G, H, and I) which use thorium in the primary fuel have costs which range from 25% to 50% higher than the once-through cycle. This difference is believed to be significant even though some of the unit costs are uncertain.

If fissile material must be spiked or is highly radioactive the only cycles drastically affected are E (HEU-Th) and I (HEU-Th+LEU), which required spiking of the primary fuel. The increases for Cycles G and H are less than 1%.

The same information presented in Table 7.5 is given in Figure 7.1. For each fuel cycle two bars are given. The left-hand bar represents the low cost estimate and the right-hand bar represents the high cost estimate. The width of the bar represents the variation in cost from a low when fabricating purified fissile material to a high when fabricating highly radioactive fissile material.

THE LOWER END OF THE COST RANGE SHOWN APPLIES TO UNSPIKED FUEL, WHILE THE UPPER LIMIT OF THE COST RANGE APPLIES TO SPIKED FUEL. LINES REPRESENT NO SPIKING EFFECT.

FIGURE 7.1. Fuel Cycle Cost Comparison

8.0 FABRICATED FUEL COMPOSITIONS

8.0 FABRICATED FUEL COMPOSITIONS

The nine fuel cycles considered in this study include several types of fuel which vary in elemental composition and fissile material enrichment. The design of a fuel fabrication plant requires a specification of average feed and fuel compositions which may be encountered over the life of the plant. Extremes in compositions are also necessary data.

Table 8.1 presents, for each fuel cycle, compositions of materials with which a fuel fabrication plant would have to deal. The data include input and output for the fabrication plant for each fuel type within the cycle. For most of the fuel types, compositions change somewhat over the 30 years considered in this study. For this reason, compositions are given for both the first appearance of the particular fuel type in the cycle and for its final appearance in cases where the change with time occurs.

The data is presented in the form of elemental weight fractions in the first three columns followed by isotopic weight fractions within each element for uranium and plutonium. All thorium considered in this study is ^{232}Th .

Most of the fuels considered are fabricated from two or more feed streams input to the fabrication plant. Explicit feed compositions are given only in cases that have two or more feed streams of the same element; in these cases, the feed compositions are listed before the final fuel composition. When there is only one feed stream of a particular element, e.g., uranium, the isotopic breakdown of that element in the fabricated fuel is the same as that in the feed stream and thus a separate feed composition listing would be redundant and is not included. Thorium feed streams are also not included, since as stated above, all the thorium is assumed to be ^{232}Th . The presence of thorium can be inferred from the fabricated fuel compositions.

Appendix D of this report supplies further information on fuel compositions. Charge and discharge compositions are presented for every fuel considered in the nine fuel cycles.

TABLE 8.1. Refabrication Plant Feed and Product Compositions

Material ^(a)	Description	Th	U	Pu	Uranium					Plutonium				
					233	234	235	236	238	238	239	240	241	242
A 1.1	LEU	--	1.000	--	--	3.-4 ^(b)	0.030	--	0.970	--	--	--	--	--
A 6.1	LEU	--	1.000	--	--	5.-4	0.032	0.006	0.962	--	--	--	--	--
A 2.2	MOX	--	0.953	0.047	--	3.-5	0.005	--	0.995	0.013	0.594	0.232	0.119	0.042
A 6.2	Feed U	--	1.000	--	--	3.-5	0.005	--	0.995	--	--	--	--	--
	Feed Pu	--	--	1.000	--	--	--	--	--	0.024	0.591	0.220	0.127	0.038
	Feed Pu	--	--	1.000	--	--	--	--	--	0.045	0.373	0.291	0.176	0.115
	Product	--	0.926	0.074	--	3.-5	0.005	--	0.995	0.040	0.443	0.275	0.148	0.094
B 1.1	LEU	--	1.000	--	--	3.-4	0.030	--	0.970	--	--	--	--	--
B 6.1	LEU	--	1.000	--	--	5.-4	0.033	0.004	0.963	--	--	--	--	--
B 2.2	Pu-Th	0.926	--	0.074	--	--	--	--	--	0.013	0.594	0.232	0.119	0.042
B 6.2	Feed Pu	--	--	1.000	--	--	--	--	--	0.025	0.592	0.220	0.126	0.037
	Feed Pu	--	--	1.000	--	--	--	--	--	0.045	0.273	0.360	0.198	0.124
	Feed Pu	--	--	1.000	--	--	--	--	--	4.-4	0.586	0.241	0.130	0.043
	Product	0.869	--	0.131	--	--	--	--	--	0.037	0.400	0.313	0.157	0.093
B 3.3	²³³ U-LEU	--	1.000	--	0.023	0.001	0.002	8.-6	0.974	--	--	--	--	--
B 6.3	²³³ U-LEU	--	1.000	--	0.023	0.005	0.002	7.-4	0.970	--	--	--	--	--
C 1.1	LEU	--	1.000	--	--	3.-4	0.030	--	0.970	--	--	--	--	--
C 6.1	LEU	--	1.000	--	--	0.001	0.033	0.007	0.959	--	--	--	--	--
C 2.2	Pu-Th	0.926	--	0.074	--	--	--	--	--	0.013	0.593	0.232	0.120	0.042
C 6.2	Feed Pu	--	--	1.000	--	--	--	--	--	0.028	0.593	0.218	0.125	0.036
	Feed Pu	--	--	1.000	--	--	--	--	--	0.048	0.270	0.356	0.197	0.129
	Feed Pu	--	--	1.000	--	--	--	--	--	0.004	0.641	0.184	0.139	0.032
	Product	0.877	--	0.123	--	--	--	--	--	0.042	0.395	0.309	0.158	0.096
C 3.3	DU-Th	0.722	0.278	--	0.118	0.008	0.003	4.-5	0.871	--	--	--	--	--
C 6.3	DU-Th	0.707	0.293	--	0.114	0.021	0.007	0.002	0.856	--	--	--	--	--

(a) See flow sheets in Appendix A.

(b) 3.-4 means 3.0×10^{-4} etc.

TABLE 8.1. (contd)

Material	Description	Th	U	Pu	Uranium					Plutonium				
					233	234	235	236	238	238	239	240	241	242
D 1.1	LEU	--	1.000	--	--	3.-4	0.030	--	0.970	--	--	--	--	--
D 6.1	LEU	--	1.000	--	--	0.001	0.033	0.007	0.959	--	--	--	--	--
D 2.2	Pu-Th	0.926	--	0.074	--	--	--	--	--	0.013	0.593	0.232	0.120	0.042
D 6.2	Feed Pu	--	--	1.000	--	--	--	--	--	0.028	0.593	0.218	0.125	0.036
	Feed Pu	--	--	1.000	--	--	--	--	--	0.050	0.272	0.356	0.194	0.128
	Feed Pu	--	--	1.000	--	--	--	--	--	1.000	--	--	--	--
	Product	0.874	--	0.126	--	--	--	--	--	0.043	0.394	0.312	0.155	0.096
D 3.3	HEU-Th	0.966	0.034	--	0.932	0.060	0.008	3.-4	--	--	--	--	--	--
D 6.3	HEU-Th	0.960	0.040	--	0.785	0.156	0.046	0.013	--	--	--	--	--	--
E 1.1	HEU-Th	0.957	0.043	--	--	0.009	0.931	--	0.060	--	--	--	--	--
E 2.2	HEU-Th	0.950	0.050	--	0.403	0.048	0.337	0.138	0.074	--	--	--	--	--
E 6.2	HEU-Th	0.930	0.070	--	0.391	0.127	0.154	0.211	0.117	--	--	--	--	--
F 1.1	DU-Th	0.768	0.232	--	--	--	0.200	--	0.800	--	--	--	--	--
F 2.2	DU-Th	0.688	0.312	--	0.057	0.005	0.084	0.023	0.831	--	--	--	--	--
F 3.3	DU-Th	0.555	0.445	--	0.060	0.010	0.039	0.031	0.860	--	--	--	--	--
F 4.4	DU-Th	0.353	0.647	--	0.047	0.013	0.021	0.032	0.887	--	--	--	--	--
F 5.5	DU-Th	0.103	0.897	--	0.033	0.013	0.013	0.031	0.910	--	--	--	--	--
G 1.1	DU-Th	0.768	0.232	--	--	--	0.200	--	0.800	--	--	--	--	--

TABLE 8.1. (contd)

Material	Description	Th	U	Pu	Uranium					Plutonium				
					233	234	235	236	238	238	239	240	241	242
G 2.2	Pu-Th	0.941	--	0.059	--	--	--	--	--	0.027	0.646	0.168	0.127	0.032
G 6.2	Feed Pu	--	--	1.000	--	--	--	--	--	0.027	0.631	0.168	0.142	0.032
	Feed Pu	--	--	1.000	--	--	--	--	--	0.102	0.242	0.313	0.208	0.135
	Feed Pu	--	--	1.000	--	--	--	--	--	0.066	0.603	0.169	0.132	0.030
	Feed Pu	--	--	1.000	--	--	--	--	--	0.061	0.636	0.165	0.116	0.022
	Feed Pu	--	--	1.000	--	--	--	--	--	0.057	0.664	0.161	0.102	0.016
	Feed Pu	--	--	1.000	--	--	--	--	--	0.052	0.669	0.168	0.096	0.015
	Product	0.910	--	0.090	--	--	--	--	--	0.079	0.443	0.247	0.149	0.082
G 2.3	DU-Th	0.688	0.312	--	0.057	0.005	0.084	0.023	0.831	--	--	--	--	--
G 3.4	DU-Th	0.555	0.445	--	0.060	0.010	0.039	0.031	0.860	--	--	--	--	--
G 4.5	DU-Th	0.353	0.647	--	0.048	0.013	0.021	0.032	0.886	--	--	--	--	--
G 5.6	DU-Th	0.103	0.897	--	0.033	0.013	0.013	0.031	0.910	--	--	--	--	--
G 6.7	DU-Th	--	0.960	0.040	0.018	0.012	0.009	0.030	0.931	0.079	0.443	0.247	0.149	0.082
H 1.1	DU-Th	0.768	0.232	--	--	--	0.200	--	0.800	--	--	--	--	--
H 2.2	MOX	--	0.958	0.042	--	--	0.004	--	0.996	0.038	0.633	0.169	0.128	0.032
H 6.2	Feed Pu	--	--	1.000	--	--	--	--	--	0.027	0.631	0.168	0.142	0.032
	Feed Pu	--	--	1.000	--	--	--	--	--	0.063	0.396	0.270	0.173	0.098
	Feed Pu	--	--	1.000	--	--	--	--	--	0.064	0.622	0.164	0.125	0.025
	Feed Pu	--	--	1.000	--	--	--	--	--	0.061	0.643	0.163	0.112	0.021
	Feed Pu	--	--	1.000	--	--	--	--	--	0.056	0.667	0.161	0.100	0.016
	Feed Pu	--	--	1.000	--	--	--	--	--	0.052	0.669	0.168	0.096	0.015
	Product	--	0.936	0.064	--	--	0.003	--	0.997	0.060	0.492	0.238	0.139	0.071
H 2.3	DU-Th	0.688	0.312	--	0.057	0.005	0.084	0.023	0.831	--	--	--	--	--
H 3.4	DU-Th	0.555	0.445	--	0.060	0.010	0.039	0.031	0.860	--	--	--	--	--

TABLE 8.1. (contd)

Material	Description	Th	U	Pu	Uranium					Plutonium				
					233	234	235	236	238	238	239	240	241	242
H 4.5	DU-Th	0.353	0.647	--	0.048	0.013	0.021	0.032	0.886	--	--	--	--	--
H 5.6	DU-Th	0.103	0.897	--	0.033	0.013	0.013	0.031	0.910	--	--	--	--	--
H 6.7	Pu-Th	--	0.965	0.035	0.018	0.013	0.009	0.030	0.930	0.060	0.492	0.238	0.139	0.071
I 1.1	HE-Th	0.957	0.043	--	--	0.009	0.931	--	0.060	--	--	--	--	--
I 1.2	LEU	--	1.000	--	--	3.-4	0.030	--	0.970	--	--	--	--	--
I 2.3	LEU	--	1.000	--	0.011	0.002	0.017	0.008	0.962	--	--	--	--	--
I 6.3	Feed U	--	1.000	--	--	3.-4	0.030	--	0.970	--	--	--	--	--
	Feed U	--	1.000	--	0.403	0.048	0.337	0.138	0.074	--	--	--	--	--
	Feed U	--	1.000	--	0.004	0.004	0.007	0.019	0.966	--	--	--	--	--
	Product	--	1.000	--	0.015	0.005	0.016	0.022	0.942	--	--	--	--	--

9.0 CONCLUSIONS

9.0 CONCLUSIONS

The closed cycle method of analysis gives an integrated view of the resource requirements and the comparative power costs that would result from changing the present LWR UO_2 once-through fuel cycle to other cycles involving thorium. The amounts of power that could be generated in diversion and proliferation-resistant fuel outside secured centers have also been established. These, and the savings in separative work, are summarized in Table 9.1.

Cycles A through D involve plutonium recycle with variations in the form of the recycled plutonium. All result in savings in uranium of about 31 to 34%. Cycle C would require the least amount of energy to be generated inside a secured center. The estimates of the cost differentials vary widely depending on the interest rate assumptions and the type of refabrication plant operation assumed. On the average there is a small cost penalty for these cycles.

Cycle E, the highly enriched uranium-thorium cycle, offers no apparent advantage except that it makes maximum use of thorium which results in the greatest uranium saving.

Cycles F through H, the dilute uranium-thorium cycles, will produce more expensive power, but can be used to some advantage to obtain some degree of diversion and proliferation resistance. Cycle F, the DU-Th cycle with plutonium discard saves 28% of the uranium consumed by the LEU once-through cycle and also would require no secured center energy generation. However, it requires approximately 8% more separative work. Cycles G and H, by recycling the bred plutonium, can produce lower power costs, savings in separative work, and some reduction in power costs when compared to the Cycle F, DU-Th with plutonium discard.

Cycle I, the highly enriched uranium-thorium with bred ^{233}U recycled in LEU and plutonium discarded has few distinctive advantages. It saves less uranium than the other recycle mode; uses more separative work; almost half of its energy must be generated in a secured center; and it has the same cost penalties that the DU-Th cycle has.

TABLE 9.1. Summary of Alternative Fuel Cycle Characteristics in Relation to LEU

Cycle	Uranium Savings (%)	Separative Work Savings (%)	Energy Generated in Secured Center (%)	Fuel Cycle Cost Increment Relative to LEU Once-Through ^(a) (mills/kWhr)		
				Maximum	Minimum	Average
A: LEU, Pu Recycle	31.2	18.3	21.9	+0.55	-0.07	+0.24
B: LEU, Pu-Th-LEU	33.3	17.5	12.7	+0.71	+0.16	+0.44
C: LEU, Pu-Th, DU-Th	33.6	18.0	12.3	+0.70	+0.13	+0.42
D: LEU, Pu-Th, HEU-Th	34.0	16.7	18.8	+0.69	+0.12	+0.41
E: HEU-Th	40.8	2.9	100.0	+4.73	+2.55	+3.64
F: DU-Th, Pu Discard	28.5	-7.7	0.0	+3.70	+3.32	+3.51
G: DU-Th, Pu-Th	39.6	9.3	8.6	+3.45	+3.02	+3.24
H: DU-Th, Pu-U	40.2	11.4	14.0	+3.40	+2.95	+3.18
I: HEU-Th, LEU	30.8	-8.2	43.3	+3.47	+1.77	+2.62

(a) Positive increments with respect to LEU once-through indicate additional costs. Negative increments are cost incentives favoring the fuel cycle. Maximum increments originate from the higher interest rate cash flow analysis, while minimum increments stem from the lower interest rate cases.

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APPENDIX A
FUEL CYCLE SCHEMATICS

APPENDIX A

FUEL CYCLE SCHEMATICS

The detailed mass flow charts for nine fuel cycles are presented in this appendix. On the flow diagram for each fuel cycle, the time is divided into 5-year intervals during which identical material is charged. Enrichment, reprocessing and fabrication-irradiation are each represented by a characteristic block shape on the diagram. The largest rectangles represent the fabrication and irradiation steps. The percentages shown are weight percents of the fissile isotopes. The material amounts shown are determined by the amount of material available for recycle from previous steps plus the amount of new material required to generate a constant amount of electrical energy for that year.

Each material is identified by a unique material designation. For example, Material A4.2 means Cycle A, fourth 5-year block, Material Type 0.2 which happens to be $\text{PuO}_2\text{-UO}_2$ for that particular fuel cycle. This system of designation can be used to find the detailed isotopic composition from the fuel material composition tables in Appendix D.

FUEL CYCLE A
LEU, PU IN PARTIALLY DEPLETED U

YEARS
1-5

6-10

11-15

16-20

21-25

26-30

YEARS
1-5

FUEL CYCLE C
LEU, PuTh, DENATURED U233 IN Th

YEARS
1-5

6-10

11-15

16-20

21-25

26-30

FUEL CYCLE D
LEU, PuTh, U233 IN THORIUM

YEARS
1-5

6-10

11-15

16-20

21-25

26-30

FUEL CYCLE E
ENRICHED URANIUM - THORIUM

FUEL CYCLE F
DENATURED URANIA-THORIA Pu DISCARD

YEARS

1-5

6-10

11-15

16-20

21-25

26-30

YEARS
1-5

YEARS
1-5

FUEL CYCLE I
ENRICHED URANIA - THORIA, URANIA WITH Pu DISCARD

APPENDIX B

NEUTRONICS

APPENDIX B

NEUTRONICS

A key problem in the analysis of nuclear reactor fuel cycles is the calculation of fissile loading requirements and the accurate prediction of spent fuel isotopics. The standard cell depletion code LEOPARD⁽³⁾ was selected as a starting point for the neutronic analysis of alternative LWR cycles. Extensive modifications were made to the original version in order to make it suitable for ^{233}U , ^{232}Th and higher plutonium loadings. Each of the major modifications will be discussed below.

B.1 PLUTONIUM-240 RESONANCE ABSORPTION

The calculation of neutron absorption in the ^{240}Pu isotope in the standard version of LEOPARD makes no provision for the change in the effective resonance integral with ^{240}Pu density nor for Doppler broadening. The calculation of the slowing down spectrum in the MUFT portion of LEOPARD divides the neutron energy range into 54 fine groups. The large 1.053 eV ^{240}Pu resonance integral is allocated to the single group containing the peak of the resonance, but the resonance is broad enough to have appreciable absorption in at least five of the neighboring energy groups.

The existing ^{240}Pu treatment was removed from the code and replaced by a new algorithm which is based on results obtained from the EGGNIT⁽²⁵⁾ code. The nuclear data on the LEOPARD library and on the EGGNIT library are from the ENDF/B-IV⁽²⁹⁾ data file. In EGGNIT the region of the 1.053 eV resonance is covered by seven MUFT energy groups. For a specific case the resonance is Doppler-broadened and a fine mesh integration of the reaction rate is carried out by the Nordheim integral⁽²⁶⁾ method. An effective flux-averaged cross section is then calculated for each of the seven large energy groups around the resonance. Heterogeneous effects are introduced by first order collision probability methods.

To apply the higher-order EGGNIT method to LEOPARD, a set of 80 cases was constructed for various rod sizes and fuel temperatures. From the value of

the resonance integral for each of these cases the values of the constants were determined for the following expression of the resonance integral:

$$RI = B_0 + B_1 X + B_2 X^2 + (B_3 + B_4 X + B_5 X^2) \sqrt{T},$$

where

RI = the resonance integral

B_i = constants evaluated from the EGGNIT cases

X = a parameter related to the surface-to-mass ratio of the fuel rod

T = the absolute temperature of the absorber.

This equation yielded an excellent fit to the EGGNIT results.

The allocation to the energy group of the resonance integral calculated by the previous equation is accomplished with the following equation for allocation factors.

$$\begin{aligned} AF_g = & A_{0g} + A_{1g} (RI)^{-2} + A_{2g} (RI)^{-1} + A_{3g} (RI)^2 \\ & + \left[A_{5g} + A_{6g} (RI) + A_{7g} (RI)^2 \right] T \\ & + \left[A_{8g} + A_{9g} (RI) + A_{10g} (RI)^2 \right] T^2, \end{aligned}$$

where

AF_g = the allocation factor for the gth MUFT energy group

A_{ig} = the ith term constant for gth group allocation factor.

The allocation factors are normalized:

$$\sum_{g=1}^7 AF_g = 1.0.$$

The 70 A_{ig} constants were evaluated from the same 80 EGGNIT cases that were used to evaluate the resonance integral. Here too, the fits were excellent using this particular form of equation.

The use of the two previous expressions to replace the existing ^{240}Pu resonance treatment supplies a much more satisfactory treatment at a very modest increase in computing time.

B.2 THORIUM-232 RESONANCE ABSORPTION

The treatment of ^{232}Th resonance absorption in the original version of LEOPARD is completely analogous to the ^{238}U treatment. An experimentally-determined function of the rod surface-to-mass ratio and the absorber temperature is used to provide a case-dependent value of the ^{232}Th resonance integral.

Several cases were run with EGGNIT to compare the ENDF/B-IV resonance data with measured resonance integral values.⁽²⁷⁾ The results are shown in Figure B.1. The EGGNIT ENDF/B-IV calculated points are lower than the measured points by amounts ranging from 1.2 to 2.5 barns, the amount of the discrepancy being dependent both on temperature and rod surface-to-mass ratio. Possible reasons for the discrepancy are many, but the integral measurements are to be given greatest weight, hence an adjustment to force better agreement with the experimental values was made.

Several EGGNIT cases were run at various temperatures and rod sizes. These provided a basis for evaluating the constants in the LEOPARD resonance integral equation:

$$\text{RI} = A_0 + A_1 X + (A_2 + A_3 X) \sqrt{T},$$

where

$$X = \left[\frac{\Sigma M_0}{N_0} P_0 + \frac{D}{N_0} \right]^{1/2}, \text{ and}$$

ΣM_0 = the macroscopic cross section for admixed scatterers

P_0 = single rod escape probability, a function of ΣM_0

$\bar{\ell}$ = rod mean chord length

N_0 = ^{232}Th concentration in the rod

D = the Dancoff factor.

A correction to the EGGNIT-ENDF/B-IV values of the constants was then made by increasing the value of A_0 by 1.8 barns. The resulting equation is shown plotted at 1200°K and at 300°K in Figure B.1. The adjustment brings the calculated and measured values to within 0.5 barns over the design range of PWR rods loaded with 20% uranium-thorium fuel. The final form of the equation used is:

$$\text{RI} = -4.48 + 2.21X + (0.0660X - 0.239) \sqrt{T}.$$

FIGURE B.1. Measured Versus Calculated ^{232}Th Resonance Integral

B.3 RESONANCE INTERFERENCE

With a mixture of several isotopes, each with some resonances, there exists the possibility of resonance interference. Resonance interference occurs when the $1/E$ asymptotic neutron spectrum is perturbed by a resonance that does not recover its $1/E$ dependence by the next resonance. The second resonance is then interfered with by the first resonance. The effect is largest for resonances close together, large resonances, and large concentrations of interfering resonances. The presence of ^{232}Th and ^{233}U introduces the possibility of significant interference effects which are not present in UO_2 or $\text{UO}_2\text{-PuO}_2$ fuel.

To evaluate potential interference effects from alternate LWR cycle material, the EGGNIT code was used to evaluate this effect for all resonance integrals larger than 10 barns from 1 eV to 100 eV. Compositions for eight different fuel types were used. The results in the form of a correction to the calculated resonance integrals are shown in Table B.1.

The most significant effects are for ^{232}Th and ^{238}U for the relative concentration involved in the LWR alternate fuels. These fertile nuclides are present in sufficient concentrations to alter the effective resonance integrals by a few tenths of a barn. The other nuclide effects were judged not to be as significant. The equations for ^{232}Th and ^{238}U were incorporated into LEOPARD as corrections to the existing expression for the resonance integrals of these two isotopes.

B.4 FISSION PRODUCT ABSORPTION

The neutron absorption in fission product isotopes is a strong determiner of the reactivity loss rate with burnup. The original version of LEOPARD provides a pair of polynomials for lumped fission product absorption cross sections as a function of burnup. This model applies to all fuel types. Fission product yields from the greater variety of fissile nuclide concentrations in the LWR alternate fuel types make this single fission product unacceptable.

TABLE B.1. Corrections for Interference

$$RI^{232} = RI^{calc} \frac{-11.7N^{233} - 5.1N^{239} - 1.05N^{238}}{N^{232}}$$

$$RI^{238} = RI^{calc} \frac{-6.1N^{233} - 1.53N^{235} - 4.5N^{239} - 0.193N^{232}}{N^{238}}$$

$$RI^{234} = RI^{calc} - 6.7 \times 10^4 N^{236}$$

$$RI^{236} = RI^{calc} - 5.5 \times 10^4 N^{234}$$

$$RI^{233} = RI^{calc} - 440N^{232} - 1560N^{238}$$

$$RI^{235} = RI^{calc} - 280N^{232} - 10/N^{238} + 5 N^{238}$$

$$RI^{239} = RI^{calc} - 150N^{232} - 220N^{238}$$

$$RI^{241} = RI^{calc} - 1600N^{238}$$

where: N^i is the nuclear density of the i th isotope

RI^{232} is the corrected resonance integral for ^{232}Th

RI^{238} is the corrected resonance integral for ^{238}U , etc.

RI^{calc} is the resonance integral without interference.

Five different fission product absorption models were derived from the CINDER⁽²⁸⁾ code using an ENDF/B-IV nuclear property library. The five fuel types covered are:

- . UO_2 (3.0% ^{235}U)
- . $\text{PuO}_2\text{-UO}_2$
- . $\text{PuO}_2\text{-ThO}_2$
- . $\text{UO}_2\text{-ThO}_2$ (4.0% ^{235}U)
- . $\text{UO}_2\text{-ThO}_2$ (3.8% ^{233}U)

The original fission-product polynomial was removed and replaced by tables of cross-section values at various exposures for each of the five fuel types. An interpolation routine was written to provide values at each step during the burnup calculation. At the beginning of the burnup calculation a choice of one of the five available sets of values which is most appropriate to the problem can be made.

B.5 OTHER CODING CHANGES

The experimentally-based resonance integral expression is used in place of the MUFT library values for either UO_2 or ThO_2 , but not for both of the materials. This presents a problem because some of the alternate LWR fuels have appreciable amounts of both UO_2 and ThO_2 present, hence, both resonance integrals are heavily shielded. LEOPARD was changed to permit the resonance integral calculation for both ^{232}Th and ^{238}U whenever both isotopes are present in appreciable amounts.

To determine the fissile loading in a single submittal, two subroutines were written which can vary the fissile loading and compare the value of k-infinity to a preselected value at a preselected exposure. When the desired value is reached to within a specified tolerance, the search is terminated and a normal burnup with the converged fissile loading is performed. The search option works on binary mixtures of U-Th, Pu-Th, Pu-U and on a mixture of two different uranium compositions.

B.6 COMPARISONS WITH MEASURED FUEL ISOTOPICS

Burnup isotopics from three LEU-MOX reactors were used to verify the ability of the modified code to predict spent fuel isotopics. Data comparison for the HB Robinson 2, Yankee-Rowe, and San Onofre reactors are shown in Tables B.2 to B.5. The first two reactors are fueled with UO_2 while San Onofre uses PuO_2 - UO_2 material and Elk River used $^{235}\text{UO}_2$ - ThO_2 material. In general, the overall agreement is quite good. However, there is a systematic underprediction of ^{235}U in all of the reactors.

Table B.5 compares the uranium isotopic composition of three samples from the $^{235}\text{UO}_2$ - ThO_2 core of the Elk River BWR. Even in this data the ^{235}U concentration is low, although the discrepancy is not as large. The amount of ^{233}U is overpredicted from 5% to 14%, but the experimental data is from dissolver samples of seven elements with batch-averaged exposures. Experimental errors can be quite large, so that there is little basis to conclude that adjustments to the nuclear data are warranted.

TABLE B.2. HB Robinson Isotopics, (30) UO₂

Sample		Uranium Isotopics, %				Plutonium Isotopics, %				Pu/U
		²³⁴ U	²³⁵ U	²³⁶ U	²³⁸ U	²³⁹ Pu	²⁴⁰ Pu	²⁴¹ Pu	²⁴² Pu	
HBR-A	meas.	0.0157	0.806	0.323	98.85	60.1	23.6	12.0	4.3	0.00892
	calc.	0.01	0.78	0.31	98.90	60.4	23.6	12.3	3.7	0.0085
HBR-B	meas.	0.017	0.765	0.343	98.9	58.8	23.4	13.0	4.8	0.00941
	calc.	0.01	0.66	0.33	99.6	57.7	24.9	12.9	4.6	0.00910
HBR-C	meas.	0.01	0.60	0.35	99.0	55.1	25.5	13.3	6.1	0.00952
	calc.	0.01	0.54	0.34	99.1	55.1	26.0	13.4	5.5	0.00950

SAMPLE LOCATIONS AND IDENTIFICATION

A: 12 in. from the core bottom P-811, 24,570 (MWd/MT)
 B: center near grid spacer P-879, 27,620 (MWd/MT)
 C: center P-868, 30,920 (MWd/MT)

TABLE B.3. Yankee-Rowe Isotopics,⁽⁵⁾ UO₂

	Sample(a) G4-C-f6 (13,629 MWd/MTHM)		Sample(a) F5-C-a1 (23,770 MWd/MTHM)	
	<u>Measured</u>	<u>Calculated</u>	<u>Measured</u>	<u>Calculated</u>
Uranium Isotopics, %				
234 _U	0.0161	0.016	0.144	0.14
235 _U	2.3	2.1	1.664	1.48
236 _U	0.254	0.275	0.369	0.39
238 _U	97.42	97.61	97.82	97.99
Pu/U	0.00674-0.00705	0.68	0.01	0.01
Plutonium Isotopics, %				
239 _{Pu}	79.32	78.7	68.0	68.9
240 _{Pu}	12.79	13.4	18.6	17.7
241 _{Pu}	7.20	7.2	11.3	11.3
242 _{Pu}	0.693	0.70	2.1	2.1

(a) Both samples were taken from interior regions of the reactor where the neutron spectrum is more nearly that of an infinite lattice.

TABLE B.4. San Onofre Isotopics,⁽⁶⁾ PuO₂-UO₂

	Sample A(a) (8,501 MWd/MTHM)		Sample B(a) (21,103 MWd/MTHM)	
	<u>Measured</u>	<u>Calculated</u>	<u>Measured</u>	<u>Calculated</u>
Uranium Isotopics, %				
234U	0.006	0.005	0.005	0.004
235U	0.633	0.621	0.464	0.437
236U	0.018	0.022	0.052	0.063
238U	99.343	99.35	99.479	99.50
Pu/U	3.03	3.15	2.97	2.86
Plutonium Isotopics, %				
239Pu	73.5	72.7	57.9	57.9
240Pu	18.9	18.0	26.7	25.2
241Pu	6.5	8.1	12.7	13.9
242Pu	1.1	1.2	3.1	3.0

(a) Both samples were taken from interior regions of the reactor where the neutron spectrum is more nearly that of an infinite lattice.

TABLE B.5. Elk River Isotopics, (7) $^{235}\text{UO}_2\text{-ThO}_2$

Sample	Uranium Isotopics, %					Burnup (MWd/MTM)
	^{233}U	^{234}U	^{235}U	^{236}U	^{237}U	
R-20	meas.	5.31	0.92	85.26	2.56	4730
	calc.	5.57	1.12	84.77	2.42	
R-11	meas.	6.11	0.99	83.83	2.91	5660
	calc.	6.65	1.14	83.13	2.90	
R-14	meas.	6.06	0.98	84.04	2.90	5860
	calc.	6.88	1.15	82.77	3.00	

APPENDIX C
FUEL CYCLE ECONOMICS

APPENDIX C

FUEL CYCLE ECONOMICS

The cost for the comparisons in Section 6 was computed using the computer code PACTOLUS.⁽¹²⁾ This code was chosen since the methodology is widely accepted and since it accepts fuel charge and discharge data for the life of a plant. The estimates of reactor costs associated with reactor construction, decommissioning, insurance, plant operation and maintenance, and land or land rights were not required since these costs are essentially unchanged regardless of the fuel cycle chosen. Therefore, the data for these costs were input as zero. The fuel of interest was assumed to begin with the first annual reload and to end with the 30th annual reload, so the first core masses were input as zero and three annual reloads of zero mass were added to the end of life data to displace the fuel of interest. Thus, a cash flow table for a given fuel type for a 35-year period is generated and appropriately discounted to the beginning year. The 35-year period arises since the uranium for the first reload is purchased more than 1 year before the first reload is charged to the reactor, and the credits and/or charges for waste treatment, reprocessing and recovered fuel values for the 30th reload are assessed in the year following its discharge from the reactor. The costs for the various fuel types for each fuel cycle can then be added to obtain the total cost as given in Table C.1. The code has provision for combining the calculations for primary and recycle fuel if the reprocessing costs are the same, which helps to reduce the total number of calculations.

The input data required by PACTOLUS that is the same for all calculations is given in Table C.2. The costs of the isotopes in recycle fuel should have a rather small impact on the overall costs since each cycle was designed to use all of the recovered fuel until reactor shutdown. The sale of recovered fuel from one fuel type would be compensated by a purchase in another type in the same year except for the losses and the last five discharges. Discounting the sale of the recovered fuel for 30 to 35 years decreases the impact of any error in these values. This decrease in direct fuel cycle cost is offset by an increase in carrying charges as the value of recycled fuel increases.

TABLE C.1. Fuel Cycle Costs

		Fuel Cycle Costs in Mills Per kWh			
		Low Estimate		High Estimate	
	Fuel Type	PF ^(a)	Spike	PF	Spike
A'	Once-through LEU				7.05
A	LEU-Pu Recycle	6.98	7.15	7.44	7.60
B	LEU Primary and Recycle	6.49	6.49	6.90	6.90
	Pu-Th	0.72	0.82	0.76	0.86
	Total	7.21	7.31	7.66	7.76
C	LEU	6.21	6.21	6.60	6.60
	Pu-Th	0.71	0.82	0.76	0.86
	DU-Th	0.25	0.25	0.28	0.28
	Total	7.18	7.28	7.64	7.75
D	LEU	6.21	6.21	6.60	6.60
	Pu-Th	0.70	0.80	0.75	0.85
	HEU-Th	0.27	0.27	0.29	0.29
	Total	7.17	7.27	7.63	7.74
E	HEU-Th Primary and Recycle	9.60	11.40	9.97	11.78
F	DU-Th Primary and Recycle	10.25		10.63	
	DU-Th Discard Spent Fuel	0.12		0.12	
	Total	10.37		10.75	
G	DU-Th Primary and Recycle	9.72	9.72	10.07	10.07
	Pu-Th	0.32	0.37	0.34	0.39
	Pu-U Recover Pu Discard U	0.04	0.04	0.04	0.04
	Total	10.07	10.12	10.45	10.50
H	DU-Th Primary and Recycle	9.58	9.58	9.93	9.93
	Pu-U Recover Pu Discard U	0.42	0.49	0.44	0.52
	Total	10.00	10.07	10.38	10.45
I	HEU-Th	3.85	5.11	3.84	5.09
	LEU Primary and Recycle	4.96	4.96	5.42	5.42
	Total	8.82	10.07	9.27	10.52

(a) PF: Purified Fissile Material

TABLE C.2. PACTOLUS Input Data

Price of Isotopes in Recycle Fuel, \$/g

233U	59 to 65
235U	40 to 44
236U	-15
239Pu	25 to 26
241Pu	29 to 31
242Pu	-8 to -9
Load factor	70.5%
Fabrication losses	0.5%
Reprocessing losses	0.5%
Pre-irradiation lead time	1 year
Post-irradiation log time	1 year
Fraction of cost financed by bonds	55%
Interest rate on bonds	9%
Earning rate on equity (after taxes)	12%
Federal income tax rate	48%
State income tax rate	4%
Cost of U ₃ O ₈ in first year in 36th year	\$30/lb (approximately 3.2% per \$90/lb year increase)
Cost of Th	\$37.60/kg
Conversion cost U ₃ O ₈ to UF ₆	\$5/kg
Cascade tails composition	0.2% 235U
Cost of separative work	\$90/kg
Cost of terminal waste disposal (paid 10 years after discharge)	\$66/kg for once-through cycle \$68/kg for uranium fuel \$72/kg for thorium fuel

TABLE C.2. (contd)

Spent fuel storage	\$32/kg
Pre-irradiation shipping	\$9/kg for spiked fuel \$6/kg for ^{233}U fuel \$1/kg for other fuel
Reactor waste shipping	\$17/kg based on fuel discharged
Post-irradiation shipping (based on fuel discharged)	\$22/kg for spent fuel \$6/kg reprocessing wastes \$1/kg PuO_2 \$10/kg ^{233}U or spiked PuO_2 \$7/kg for HLW
Fabrication cost - contact O&M	\$140/kg
Remote operation, contact maintenance	\$400/kg
Remote operation, remote maintenance	\$620/kg
Reprocessing cost - uranium fuels	\$200 to 350/kg
thorium fuels	\$300 to 420/kg

If the cost of money is low (no inflation assumed) the direct costs are more important, while if the cost of money is high the indirect costs are more important. For the intermediate situation assumed in Table C.2, the direct and indirect effects tend to be roughly equal. Thus it was not deemed necessary to determine the value of the recycled fuel for each fuel cycle. Several calculations were made for Fuel Cycle C (LEU-Pu-Th-DU-Th) using various isotopic costs with the remaining input as given in Table C.2 or as described in the following sections. It was found that a price increase of \$30 per gram for fissile plutonium increased the fuel cycle cost 0.10 mills/kWh, while a \$30 per gram price increase for fissile uranium caused a fuel cycle cost increase of 0.05 mills/kWh. Thus, in this case setting the prices chosen to zero would decrease the fuel cycle cost less than 3%. The prices chosen were derived to cause unit costs for each fuel type (costs for the type divided by the fraction of energy generated by that type) to be equal.

The financial parameters were selected to be consistent with the financial parameters used to derive the unit fabrication and reprocessing charges. In

the derivation of the unit costs bond interest of 8.3% and a 14% return on equity was used. In GEIS⁽³¹⁾ it was determined utilities paid about 0.8% more for bond issues than large industrial concerns, and earned about 2% less on their equity. We therefore selected 9% interest for bonds and a 12% return on equity.

C.1 U₃O₈ UNIT COST

The future price of yellowcake is a function of both its demand and relative availability (supply). The demand, both domestic and foreign, is directly related to the installed nuclear generating supply. The supply of U₃O₈ is dependent upon the availability of ore reserves at a recoverable price low enough to attract and/or sustain material producers. Recently, several independent predictions of future U₃O₈ price were completed. The results of these analyses are shown on the middle four columns of Table C.3. A large disparity of opinion is seen and a consensus on U₃O₈ price is not apparent.

TABLE C.3. U₃O₈ Market Price Predictions (1978 Dollars)

Year	U ₃ O ₈ Price \$/lb				
	<u>Geller⁽¹⁵⁾</u>	<u>EPRI⁽¹⁴⁾</u>	<u>GESMO⁽¹⁶⁾</u>	<u>AHMED⁽¹⁷⁾</u>	<u>This Report</u>
1980	31	--	22	24	30
1985	33	47	30	35	35
1990	37	52	34	44	41
1995	42	58	39	51	48
2000	48	75	45	55	56
2005	54	--	--	--	66
2010	--	--	--	--	77
2015	--	--	--	--	90

The last column on Table C.3 represents the range of U_3O_8 unit cost used in the economic analysis for this document. It is intended to be representative and not a rigid econometric result. This schedule allows the various comparisons between alternative fuel cycles summarized in Section 6.0 and this appendix to be made.

C.2 ThO₂ UNIT COSTS

Relatively little work has been done on estimating thorium reserves and the cost of recovering those reserves. Figure C.1 summarizes ongoing work at PNL. Although preliminary in nature this figure shows approximate unit costs for ThO₂ at various levels of cumulative demand. Using this data in conjunction with theoretical demands summarized in Table 4.2 will allow an estimate of possible thorium prices. For the comparisons and analysis of the fuel cycles in this document a thorium cost of \$15/lb-ThO₂ was used. This value is representative and feasible demands for thorium could be met at this price.

C.3 UF₆ CONVERSION UNIT COSTS

The conversion step in the nuclear fuel cycle is an established chemical process operating in a competitive market. The real price level should change little in the future unless technological breakthroughs occur. The analysis for this report has used a conversion cost of \$5.00/kg uranium as the constant value for the conversion step. Table C.4 shows how this value compares with other estimates.

C.4 ENRICHMENT UNIT COST

The isotopic enrichment operation is assumed to be a government operation subject to commercial pricing. Future technological advances should help to stabilize enrichment costs and for the scenario used in the analysis for this document a constant \$90/SWU was used. Table C.5 shows this value compared with other estimates.

FIGURE C.1. Estimation of Thorium Supply⁽¹⁸⁾

TABLE C.4. UF₆ Conversion Unit Cost Comparisons (1978 Dollars)

<u>Source</u>	<u>\$/kg U</u>
GESMO(16)	4.20
Geller(14)	5.27
South Australian Government(7,19)	5.19
TRW(20)	3.82
This Report	5.00

TABLE C.5. Enrichment Unit Costs (1978 Dollars)

<u>Source</u>	<u>\$/SWU</u>
Stoller(21)	84
GESMO(16)	90
URENCO(22)	124
EURODIF(23)	118
This Report	90

C.5 WASTE DISPOSAL UNIT COST

The terminal waste disposal of the HLW or of spent fuel is taken from the NASAP data base⁽¹³⁾ and is somewhat less than potential fees for this service as described in Reference 24 since R&D costs are not included and over capacity is not assumed. The spent fuel disposal cost is based upon 10 year old fuel so we have assumed that reactor basins can be expanded for a unit cost of \$32 per kg. If stand-alone AFR storage is required the cost would be much higher.

C.6 TRANSPORTATION UNIT COST

The transportation unit costs were also taken from the NASAP data base, but were adjusted in some areas. First, the costs were prorated with respect to the time of the expense. Second, it was assumed that the terminal disposal

site was as far from the reactor as the reprocessing plant and that the spent fuel shipping cost was the same for once-through or reprocessing cycles.

C.7 FABRICATION UNIT COSTS

The unit costs for fabrication were taken from the NASAP data base. The contact O&M cost is the value given for the LEU(5) once-through fuel and is used in our study for LEU, HEU (no ^{233}U)-thorium, and MEV (20% ^{235}U , no ^{233}U)-thorium fuels. The remote operation, contact maintenance cost is the value given for the LEU(5)U-Pu portioned recycle fuel and is used in our study for UO_2 - PuO_2 and ThO_2 - PuO_2 mixed-oxide fuels. The remote operation, remote maintenance cost is given for LEU(5) spiked recycle and DU(3)-Th recycle DU(3) and is used in our study for spiked fuels and fuels containing ^{233}U .

C.8 REPROCESSING UNIT COSTS

The unit costs for reprocessing are also taken from the NASAP data base. The low values were used to obtain the fuel cycle costs for the low estimate in Table C.1, while the high values were used to obtain costs for the high estimate. The variation in these costs caused the range of isotope prices given near the top of Table C.2. All of the other costs are also subject to some variation, but the values selected are useful for a comparison of the various cycles studied.

APPENDIX D

FUEL MATERIAL COMPOSITIONS

APPENDIX D

FUEL MATERIAL COMPOSITIONS

Fuel composition has been calculated for each of the fuels considered in the nine fuel cycles. Tables D.1 through D.9 present the initial and discharge heavy metal compositions for each of the fuels.

In each table the fuels are arranged by type; for example, all the Pu-Th fuels in a particular cycle are grouped together. The numbers (e.g., 2.2, 3.2) under the fuel type loadings refer to the particular fuel and correspond to the labeling used in Appendix A. The fuel type headings are in a somewhat abbreviated form; the following list explains the abbreviations.

LEU	Low-enriched uranium oxide fuel
ReLEU	Low-enriched uranium oxide fuel containing recycled uranium
DU-Th	Denatured uranium-thorium oxide fuel
ReDU-Th	Denatured uranium-thorium oxide fuel containing recycled uranium
HEU-Th	High-enriched uranium-thorium oxide fuel
Pu-Th	Plutonium-thorium oxide fuel
PuO ₂ -UO ₂	Plutonium-uranium oxide fuel

The first seven columns of each table contain the heavy metal weight fractions for thorium, uranium, plutonium, ²³⁷Np, ²⁴¹Am, ²⁴³Am, and ²⁴²Cm. The next ten columns show the weight fraction breakdown of the various isotopes of each element for uranium and plutonium.

At the end of the detailed composition Tables D.1 through D.9, the discharge exposures and energy contributions for each material are given in Table D.10. The base exposure for the LEU fuel is 30,000 MWd/MTM. The variation in exposure is due to the variation in fuel density when a constant amount of energy is extracted from the fuel.

TABLE D.1. Initial Fuel Compositions: Weight Fractions LEU, PuO₂ in Partially Depleted UO₂

Fuel Cycle A																	
	Th	U	Pu	²³⁷ Np	²⁴¹ Am	²⁴³ Am	²⁴² Cm	Isotopic Weight Fractions in U					Isotopic Weight Fractions in Pu				
								²³³	²³⁴	²³⁵	²³⁶	²³⁸	²³⁸	²³⁹	²⁴⁰	²⁴¹	²⁴²
LEU																	
1.1	--	1.000	--	--	--	--	--	--	3.-4	0.030	--	0.970	--	--	--	--	--
2.1	--	1.000	--	--	--	--	--	--	3.-4	0.031	0.003	0.966	--	--	--	--	--
3.1	--	1.000	--	--	--	--	--	--	4.-4	0.031	0.004	0.964	--	--	--	--	--
4.1	--	1.000	--	--	--	--	--	--	4.-4	0.031	0.005	0.963	--	--	--	--	--
5.1	--	1.000	--	--	--	--	--	--	5.-4	0.032	0.005	0.963	--	--	--	--	--
6.1	--	1.000	--	--	--	--	--	--	5.-4	0.032	0.006	0.962	--	--	--	--	--
PuO ₂ -UO ₂																	
2.2	--	0.952	0.048	--	3.-4 ^(a)	--	--	--	3.-5	0.005	--	0.995	0.013	0.595	0.232	0.119	0.042
3.2	--	0.942	0.058	--	4.-4	--	--	--	3.-5	0.005	--	0.995	0.023	0.522	0.258	0.138	0.060
4.2	--	0.936	0.063	--	4.-4	--	--	--	3.-5	0.005	--	0.995	0.030	0.483	0.263	0.145	0.074
5.2	--	0.927	0.072	--	0.001	--	--	--	3.-5	0.005	--	0.995	0.036	0.457	0.273	0.148	0.086
6.2	--	0.923	0.077	--	0.001	--	--	--	3.-5	0.005	--	0.995	0.040	0.443	0.275	0.148	0.094

Discharge Fuel Compositions																	
LEU																	
1.1	--	0.990	0.010	5.-4	5.-5	7.-5	2.-5	--	2.-4	0.008	0.004	0.988	0.013	0.587	0.229	0.130	0.041
2.1	--	0.989	0.010	0.001	5.-5	7.-5	2.-5	--	2.-4	0.009	0.006	0.985	0.019	0.583	0.225	0.128	0.040
3.1	--	0.989	0.010	0.001	5.-5	7.-5	2.-5	--	2.-4	0.009	0.008	0.983	0.023	0.590	0.222	0.127	0.038
4.1	--	0.989	0.010	0.001	5.-5	7.-5	2.-5	--	3.-4	0.009	0.008	0.982	0.024	0.589	0.222	0.127	0.038
5.1	--	0.989	0.010	0.001	5.-5	6.-5	2.-5	--	3.-4	0.010	0.008	0.982	0.024	0.591	0.220	0.127	0.038
6.1	--	0.989	0.010	0.001	5.-5	6.-5	2.-5	--	3.-4	0.010	0.009	0.981	0.026	0.591	0.220	0.126	0.037
PuO ₂ -UO ₂																	
2.2	--	0.959	0.039	2.-4	0.001	0.001	2.-4	--	2.-5	0.003	0.001	0.997	0.025	0.432	0.287	0.176	0.080
3.2	--	0.950	0.047	2.-4	0.001	0.001	2.-5	--	2.-5	0.003	0.001	0.997	0.034	0.403	0.290	0.179	0.095
4.2	--	0.945	0.052	2.-4	0.001	0.001	2.-4	--	2.-5	0.003	5.-4	0.997	0.040	0.384	0.291	0.178	0.107
5.2	--	0.937	0.060	2.-4	0.001	0.002	2.-4	--	2.-5	0.003	5.-4	0.996	0.045	0.372	0.291	0.176	0.115
6.2	--	0.933	0.064	1.-4	0.001	0.002	3.-4	--	2.-5	0.003	5.-4	0.996	0.048	0.364	0.291	0.174	0.122

(a) 3.-4 means 3.0×10^{-4} etc.

TABLE D.2. Initial Fuel Compositions: Weight Fractions LEU, Pu-Th, ^{233}U in $^{238}\text{UO}_2$

Fuel Cycle 8																	
	Th	U	Pu	237_Np	241_Am	243_Am	242_Cm	Isotopic Weight Fractions in U					Isotopic Weight Fractions in Pu				
								233	234	235	236	238	238	239	240	241	242
LEU																	
1.1	--	1.000	--	--	--	--	--	--	3.-4	0.030	--	0.970	--	--	--	--	--
2.1	--	1.000	--	--	--	--	--	--	2.-4	0.031	0.003	0.966	--	--	--	--	--
3.1	--	1.000	--	--	--	--	--	--	4.-4	0.034	0.004	0.961	--	--	--	--	--
4.1	--	1.000	--	--	--	--	--	--	4.-4	0.032	0.005	0.962	--	--	--	--	--
5.1	--	1.000	--	--	--	--	--	--	5.-4	0.032	0.005	0.962	--	--	--	--	--
6.1	--	1.000	--	--	--	--	--	--	5.-4	0.033	0.004	0.962	--	--	--	--	--
Pu-Th																	
2.2	0.925	--	0.075	--	5.-5	--	--	--	--	--	--	--	0.013	0.595	0.232	0.119	0.042
3.2	0.907	--	0.092	--	0.001	--	--	--	--	--	--	--	0.022	0.494	0.274	0.148	0.062
4.2	0.897	--	0.102	--	0.001	--	--	--	--	--	--	--	0.027	0.465	0.288	0.149	0.070
5.2	0.882	--	0.117	--	0.001	--	--	--	--	--	--	--	0.033	0.419	0.305	0.158	0.084
6.2	0.864	--	0.135	--	0.001	--	--	--	--	--	--	--	0.037	0.399	0.313	0.157	0.093
233_U/UO2																	
3.3	--	1.000	--	--	--	--	--	0.023	0.001	0.002	8.-6	0.973	--	--	--	--	--
4.3	--	1.000	--	--	--	--	--	0.022	0.003	0.001	3.-4	0.973	--	--	--	--	--
5.3	--	1.000	--	--	--	--	--	0.023	0.004	0.002	4.-4	0.972	--	--	--	--	--
6.3	--	1.000	--	--	--	--	--	0.023	0.005	0.002	7.-4	0.970	--	--	--	--	--

TABLE D.2. (contd)

Fuel Cycle B																	
Discharge Fuel Compositions																	
	Th	U	Pu	²³⁷ Np	²⁴¹ Am	²⁴³ Am	²⁴² Cm	Isotopic Weight Fractions in U					Isotopic Weight Fractions in Pu				
								²³³	²³⁴	²³⁵	²³⁶	²³⁸	²³⁸	²³⁹	²⁴⁰	²⁴¹	²⁴²
LEU																	
1.1	--	0.990	0.010	5.-4	5.-5	7.-5	2.-5	--	2.-4	0.008	0.004	0.988	0.013	0.587	0.229	0.130	0.041
2.1	--	0.989	0.010	0.001	5.-5	7.-5	2.-5	--	2.-4	0.009	0.006	0.985	0.019	0.588	0.225	0.128	0.040
3.1	--	0.989	0.010	0.001	4.-5	6.-5	2.-5	--	3.-4	0.001	0.008	0.981	0.023	0.602	0.215	0.125	0.035
4.1	--	0.989	0.010	0.001	5.-5	7.-5	2.-5	--	3.-4	0.010	0.009	0.982	0.025	0.590	0.221	0.127	0.038
5.1	--	0.989	0.010	0.001	5.-5	6.-5	2.-5	--	3.-4	0.010	0.009	0.981	0.025	0.591	0.220	0.126	0.037
6.1	--	0.989	0.010	0.001	4.-5	6.-5	2.-5	--	3.-4	0.010	0.008	0.982	0.024	0.598	0.217	0.125	0.036
Pu-Th																	
2.2	0.937	0.015	0.046	1.-6	0.001	0.001	2.-4	0.932	0.060	0.008	3.-4	--	0.025	0.332	0.338	0.212	0.093
3.2	0.921	0.014	0.062	1.-6	0.001	0.002	3.-4	0.937	0.056	0.007	2.-4	--	0.034	0.298	0.348	0.210	0.109
4.2	0.912	0.014	0.071	1.-6	0.001	0.002	3.-4	0.939	0.054	0.007	2.-4	--	0.040	0.290	0.353	0.203	0.113
5.2	0.897	0.014	0.085	1.-6	0.001	0.002	3.-4	0.942	0.052	0.006	2.-4	--	0.045	0.273	0.359	0.198	0.124
6.2	0.879	0.014	0.103	9.-7	0.002	0.002	3.-4	0.946	0.049	0.005	2.-4	--	0.047	0.276	0.362	0.187	0.128
²³³ U/UO ₂																	
3.3	--	0.990	0.010	6.-5	4.-5	7.-5	4.-4	0.005	0.002	0.001	4.-4	0.991	2.-4	0.587	0.241	0.130	0.042
4.3	--	0.991	0.009	1.-4	4.-5	7.-5	1.-5	0.005	0.003	0.001	0.001	0.990	3.-4	0.582	0.243	0.131	0.044
5.3	--	0.990	0.010	1.-4	4.-5	7.-5	1.-5	0.005	0.004	0.002	0.001	0.989	4.-4	0.586	0.241	0.130	0.043
6.3	--	0.990	0.010	2.-4	4.-5	7.-5	1.-5	0.005	0.004	0.002	0.001	0.987	5.-4	0.595	0.236	0.128	0.040

TABLE D.3. Initial Fuel Compositions: Weight Fractions LEU, Pu-Th, ReDU-Th

Fuel Cycle C																	
	Th	U	Pu	237 Np	241 Am	243 Am	242 Cm	Isotopic Weight Fractions in U					Isotopic Weight Fractions in Pu				
								233	234	235	236	238	238	239	240	241	242
LEU																	
1.1	--	1.000	--	--	--	--	--	--	3.-4	0.030	--	0.970	--	--	--	--	--
2.1	--	1.000	--	--	--	--	--	--	4.-4	0.031	0.003	0.966	--	--	--	--	--
3.1	--	1.000	--	--	--	--	--	--	5.-4	0.032	0.005	0.963	--	--	--	--	--
4.1	--	1.000	--	--	--	--	--	--	0.001	0.032	0.006	0.961	--	--	--	--	--
5.1	--	1.000	--	--	--	--	--	--	0.001	0.032	0.007	0.960	--	--	--	--	--
6.1	--	1.000	--	--	--	--	--	--	0.001	0.033	0.007	0.959	--	--	--	--	--
Pu-Th																	
2.2	0.925	--	0.075	--	4.-4	--	--	--	--	--	--	--	0.013	0.594	0.232	0.120	0.042
3.2	0.907	--	0.092	--	0.001	--	--	--	--	--	--	--	0.022	0.493	0.274	0.149	0.062
4.2	0.893	--	0.106	--	0.001	--	--	--	--	--	--	--	0.030	0.443	0.292	0.158	0.077
5.2	0.881	--	0.118	--	0.001	--	--	--	--	--	--	--	0.036	0.414	0.302	0.159	0.088
6.2	0.871	--	0.128	--	0.001	--	--	--	--	--	--	--	0.041	0.396	0.309	0.158	0.096
ReDU-Th																	
3.3	0.722	0.278	--	--	--	--	--	0.118	0.008	0.003	4.-5	0.871	--	--	--	--	--
4.3	0.718	0.282	--	--	--	--	--	0.117	0.013	0.004	4.-4	0.866	--	--	--	--	--
5.3	0.712	0.288	--	--	--	--	--	0.115	0.017	0.005	0.001	0.861	--	--	--	--	--
6.3	0.707	0.293	--	--	--	--	--	0.114	0.021	0.007	0.002	0.857	--	--	--	--	--

TABLE D.3. (contd)

Fuel Cycle C																	
Discharge Fuel Compositions																	
	Th	U	Pu	237Np	241Am	243Am	242Cm	Isotopic Weight Fractions in U					Isotopic Weight Fractions in Pu				
								233	234	235	236	238	238	239	240	241	242
LEU																	
1.1	--	0.990	0.010	5.-4	5.-5	7.-5	2.-5	--	2.-4	0.008	0.004	0.988	0.013	0.587	0.229	0.130	0.041
2.1	--	0.989	0.010	0.001	5.-5	7.-5	2.-5	--	2.-4	0.009	0.006	0.985	0.019	0.589	0.224	0.128	0.040
3.1	--	0.989	0.010	0.001	5.-5	6.-5	2.-5	--	3.-4	0.010	0.008	0.982	0.024	0.590	0.221	0.127	0.033
4.1	--	0.989	0.010	0.001	5.-5	6.-5	2.-5	--	3.-4	0.010	0.009	0.981	0.026	0.592	0.219	0.126	0.037
5.1	--	0.989	0.010	0.001	5.-5	6.-5	2.-5	--	4.-4	0.010	0.010	0.980	0.028	0.592	0.218	0.125	0.036
6.1	--	0.989	0.010	0.001	4.-5	6.-5	2.-5	--	4.-4	0.010	0.010	0.979	0.029	0.593	0.217	0.125	0.036
Pu-Th																	
2.2	0.938	0.014	0.046	1.-6	0.001	0.001	2.-4	0.928	0.063	0.008	3.-4	--	0.025	0.331	0.338	0.213	0.093
3.2	0.921	0.014	0.061	1.-6	0.001	0.001	3.-4	0.937	0.056	0.007	2.-4	--	0.034	0.297	0.348	0.211	0.110
4.2	0.908	0.014	0.075	1.-6	0.001	0.002	3.-4	0.937	0.056	0.007	2.-4	--	0.042	0.280	0.352	0.204	0.121
5.2	0.897	0.013	0.086	1.-6	0.002	0.002	3.-4	0.939	0.054	0.006	2.-4	--	0.048	0.270	0.355	0.197	0.129
6.2	0.887	0.013	0.096	9.-7	0.002	0.002	3.-4	0.941	0.053	0.006	2.-4	--	0.053	0.265	0.357	0.191	0.135
ReDU-Th																	
3.3	0.726	0.270	0.004	2.-5	3.-5	2.-5	2.-6	0.083	0.017	0.005	0.001	0.894	0.002	0.632	0.189	0.142	0.036
4.3	0.722	0.274	0.004	4.-5	2.-5	2.-5	6.-6	0.080	0.020	0.006	0.001	0.892	0.003	0.636	0.186	0.141	0.034
5.3	0.717	0.279	0.004	7.-5	2.-5	2.-5	6.-6	0.080	0.023	0.008	0.002	0.887	0.004	0.640	0.184	0.139	0.032
6.3	0.711	0.284	0.004	1.-4	2.-5	2.-5	6.-6	0.079	0.025	0.009	0.003	0.884	0.006	0.643	0.183	0.137	0.031

TABLE D.4. Initial Fuel Compositions: Weight Fractions LEU, Pu-Th, ReHEU-Th

Fuel Cycle D																	
	Th	U	Pu	237Np	241Am	243Am	242Cm	Isotopic Weight Fractions in U					Isotopic Weight Fractions in Pu				
								233	234	235	236	238	238	239	240	241	242
LEU																	
1.1	--	1.000	--	--	--	--	--	--	3.-4	0.030	--	0.970	--	--	--	--	--
2.1	--	1.000	--	--	--	--	--	--	4.-4	0.031	0.003	0.966	--	--	--	--	--
3.1	--	1.000	--	--	--	--	--	--	5.-4	0.032	0.005	0.963	--	--	--	--	--
4.1	--	1.000	--	--	--	--	--	--	0.001	0.032	0.006	0.961	--	--	--	--	--
5.1	--	1.000	--	--	--	--	--	--	0.001	0.032	0.007	0.960	--	--	--	--	--
6.1	--	1.000	--	--	--	--	--	--	0.001	0.033	0.007	0.959	--	--	--	--	--
Pu-Th																	
2.2	0.925	--	0.075	--	4.-4	--	--	--	--	--	--	--	0.013	0.594	0.232	0.120	0.042
3.2	0.906	--	0.093	--	0.001	--	--	--	--	--	--	--	0.022	0.494	0.274	0.148	0.062
4.2	0.890	--	0.109	--	0.001	--	--	--	--	--	--	--	0.032	0.436	0.296	0.158	0.078
5.2	0.878	--	0.121	--	0.001	--	--	--	--	--	--	--	0.038	0.413	0.304	0.157	0.088
6.2	0.868	--	0.131	--	0.001	--	--	--	--	--	--	--	0.043	0.394	0.312	0.155	0.096
ReHEU-Th																	
3.3	0.966	0.034	--	--	--	--	--	0.932	0.060	0.008	3.-4	--	--	--	--	--	--
4.3	0.964	0.036	--	--	--	--	--	0.872	0.103	0.021	0.003	--	--	--	--	--	--
5.3	0.962	0.038	--	--	--	--	--	0.822	0.136	0.035	0.007	--	--	--	--	--	--
6.3	0.960	0.040	--	--	--	--	--	0.785	0.156	0.046	0.013	--	--	--	--	--	--

TABLE D.4. (contd)

Fuel Cycle D

Discharge Fuel Compositions

	Th	U	Pu	237 Np	241 Am	243 Am	242 Cm	Isotopic Weight Fractions in U					Isotopic Weight Fractions in Pu				
								233	234	235	236	238	238	239	240	241	242
LEU																	
1.1	--	0.990	0.010	5.-4	3.-5	7.-5	9.-6	--	2.-4	0.008	0.004	0.988	0.013	0.587	0.229	0.130	0.042
2.1	--	0.989	0.010	0.001	3.-5	7.-5	9.-6	--	2.-4	0.009	0.006	0.985	0.019	0.589	0.225	0.128	0.040
3.1	--	0.989	0.010	0.001	3.-4	7.-5	9.-6	--	3.-4	0.010	0.008	0.982	0.024	0.590	0.221	0.127	0.038
4.1	--	0.989	0.010	0.001	3.-5	6.-5	9.-6	--	3.-4	0.010	0.009	0.981	0.026	0.592	0.219	0.126	0.037
5.1	--	0.989	0.010	0.001	3.-5	6.-5	9.-6	--	4.-4	0.010	0.010	0.980	0.028	0.592	0.218	0.125	0.036
6.1	--	0.989	0.010	0.001	3.-5	6.-5	9.-6	--	4.-4	0.010	0.010	0.979	0.030	0.593	0.217	0.125	0.036
Pu-Th																	
2.2	0.938	0.015	0.046	1.-6	0.001	0.001	2.-4	0.932	0.060	0.008	3.-4	--	0.025	0.331	0.338	0.213	0.093
3.2	0.920	0.014	0.063	1.-6	0.001	0.002	3.-4	0.937	0.056	0.007	2.-4	--	0.035	0.299	0.348	0.209	0.109
4.2	0.904	0.014	0.078	1.-6	0.002	0.002	3.-4	0.941	0.053	0.006	2.-4	--	0.044	0.278	0.354	0.202	0.121
5.2	0.893	0.014	0.089	1.-6	0.002	0.002	3.-4	0.943	0.051	0.006	1.-4	--	0.050	0.272	0.357	0.194	0.123
6.2	0.883	0.014	0.099	1.-6	0.002	0.002	4.-4	0.944	0.050	0.006	1.-4	--	0.055	0.265	0.359	0.188	0.133
ReHEU-Th																	
3.3	0.971	0.029	4.-6	2.-5	--	--	--	0.784	0.168	0.041	0.006	--	1.000	--	--	--	--
4.3	0.969	0.031	8.-6	4.-5	--	--	--	0.741	0.192	0.055	0.012	--	1.000	--	--	--	--
5.3	0.968	0.032	1.-5	6.-5	--	--	--	0.705	0.209	0.066	0.020	--	1.000	--	--	--	--
6.3	0.966	0.034	2.-5	9.-5	--	--	--	0.678	0.220	0.074	0.028	--	1.000	--	--	--	--

TABLE D.5. Initial Fuel Compositions: Weight Fractions HEU-Th

Fuel Cycle E																	
								Isotopic Weight Fractions in U					Isotopic Weight Fractions in Pu				
	Th	U	Pu	²³⁷ Np	²⁴¹ Am	²⁴³ Am	²⁴² Cm	233	234	235	236	238	238	239	240	241	242
HEU-Th																	
1.1-6.1	0.957	0.043	--	--	--	--	--	--	0.009	0.931	--	0.060	--	--	--	--	--
2.2	0.950	0.050	--	--	--	--	--	0.404	0.048	0.337	0.138	0.074	--	--	--	--	--
3.2	0.944	0.056	--	--	--	--	--	0.450	0.087	0.201	0.176	0.086	--	--	--	--	--
4.2	0.938	0.062	--	--	--	--	--	0.434	0.108	0.169	0.192	0.097	--	--	--	--	--
5.2	0.934	0.066	--	--	--	--	--	0.412	0.120	0.160	0.202	0.107	--	--	--	--	--
6.2	0.930	0.070	--	--	--	--	--	0.391	0.127	0.154	0.211	0.117	--	--	--	--	--

Discharge Fuel Compositions																	
HEU-Th																	
1.1-6.1	0.963	0.036	1.-4	0.001	3.-8	--	1.-8	0.404	0.048	0.337	0.138	0.074	0.942	0.042	0.008	0.008	--
2.2	0.957	0.042	2.-4	0.001	5.-8	1.-7	2.-8	0.469	0.102	0.147	0.192	0.091	0.938	0.036	0.010	0.010	0.005
3.2	0.951	0.047	3.-4	0.001	8.-8	2.-7	3.-8	0.445	0.130	0.108	0.212	0.105	0.936	0.036	0.012	0.012	0.004
4.2	0.946	0.052	3.-4	0.001	9.-8	2.-7	3.-8	0.415	0.143	0.101	0.223	0.117	0.939	0.038	0.010	0.010	0.003
5.2	0.943	0.056	3.-4	0.001	1.-7	2.-7	5.-8	0.391	0.148	0.101	0.231	0.129	0.934	0.039	0.012	0.012	0.003
6.2	0.939	0.060	4.-4	0.001	1.-7	2.-7	5.-8	0.369	0.151	0.100	0.239	0.141	0.935	0.041	0.011	0.011	0.003

TABLE D.6. Initial Fuel Compositions: Weight Fractions Denatured Urania - Thoria, Pu Discard

Fuel Cycle F																	
	Th	U	Pu	237Np	241Am	243Am	242Cm	Isotopic Weight Fractions in U					Isotopic Weight Fractions in Pu				
								233	234	235	236	238	238	239	240	241	242
DU-Th																	
1.1-6.1	0.768	0.232	--	--	--	--	--	--	--	0.200	--	0.800	--	--	--	--	--
DU-Th																	
2.2-6.2	0.688	0.312	--	--	--	--	--	0.057	0.005	0.084	0.023	0.831	--	--	--	--	--
3.3-6.3	0.555	0.445	--	--	--	--	--	0.060	0.010	0.039	0.031	0.861	--	--	--	--	--
4.4-6.4	0.353	0.647	--	--	--	--	--	0.047	0.013	0.021	0.032	0.887	--	--	--	--	--
5.5-6.5	0.103	0.897	--	--	--	--	--	0.033	0.013	0.013	0.031	0.909	--	--	--	--	--
Discharge Fuel Compositions																	
DU-Th																	
1.1-6.1	0.773	0.222	0.004	0.001	1.-5	2.-5	4.-6	0.057	0.005	0.084	0.023	0.831	0.045	0.620	0.165	0.139	0.031
ReDU-Th																	
2.2-6.2	0.693	0.301	0.005	0.001	2.-5	2.-5	7.-6	0.060	0.010	0.039	0.031	0.861	0.064	0.621	0.164	0.125	0.025
3.3-6.3	0.558	0.434	0.006	0.001	2.-5	2.-5	7.-6	0.048	0.013	0.021	0.032	0.887	0.061	0.644	0.163	0.112	0.020
4.4-6.4	0.353	0.637	0.008	0.002	3.-5	2.-5	8.-6	0.033	0.013	0.013	0.031	0.909	0.056	0.667	0.161	0.100	0.016
5.5-6.5	0.100	0.887	0.010	0.001	3.-5	2.-5	9.-6	0.018	0.013	0.009	0.030	0.930	0.052	0.669	0.168	0.096	0.015

TABLE D.7. Initial Fuel Compositions: Weight Fractions Denatured Urania-Thoria, Pu-Th Recycle

	Fuel Cycle G																
	Th	U	Pu	237Np	241Am	243Am	242Cm	Isotopic Weight Fractions in U					Isotopic Weight Fractions in Pu				
								233	234	235	236	238	238	239	240	241	242
DU-Th																	
1.1-6.1	0.768	0.232	--	--	--	--	--	--	--	0.200	--	0.800	--	--	--	--	--
Pu-Th																	
2.2	0.938	--	0.061	--	4.-4	--	--	--	--	--	--	--	0.027	0.645	0.168	0.127	0.032
3.2	0.927	--	0.073	--	4.-4	--	--	--	--	--	--	--	0.054	0.533	0.213	0.145	0.051
4.2	0.918	--	0.082	--	0.001	--	--	--	--	--	--	--	0.065	0.491	0.230	0.151	0.063
5.2	0.907	--	0.092	--	0.001	--	--	--	--	--	--	--	0.076	0.428	0.252	0.167	0.077
6.2	0.903	--	0.097	--	0.001	--	--	--	--	--	--	--	0.079	0.442	0.247	0.149	0.082
ReDU-Th																	
2.3	0.688	0.312	--	--	--	--	--	0.057	0.005	0.084	0.023	0.831	--	--	--	--	--
3.3	0.731	0.269	--	--	--	--	--	0.069	0.006	0.083	0.023	0.819	--	--	--	--	--
3.4	0.555	0.445	--	--	--	--	--	0.060	0.010	0.039	0.031	0.861	--	--	--	--	--
4.3	0.737	0.263	--	--	--	--	--	0.073	0.006	0.083	0.023	0.816	--	--	--	--	--
4.4	0.605	0.395	--	--	--	--	--	0.071	0.012	0.037	0.031	0.849	--	--	--	--	--
4.5	0.353	0.647	--	--	--	--	--	0.048	0.013	0.021	0.032	0.887	--	--	--	--	--
5.3	0.744	0.256	--	--	--	--	--	0.075	0.006	0.082	0.023	0.813	--	--	--	--	--
5.4	0.616	0.384	--	--	--	--	--	0.074	0.013	0.036	0.031	0.846	--	--	--	--	--
5.5	0.425	0.575	--	--	--	--	--	0.056	0.015	0.020	0.032	0.876	--	--	--	--	--
5.6	0.103	0.897	--	--	--	--	--	0.033	0.013	0.013	0.031	0.909	--	--	--	--	--
6.3	0.744	0.256	--	--	--	--	--	0.076	0.006	0.082	0.023	0.813	--	--	--	--	--
6.4	0.624	0.376	--	--	--	--	--	0.076	0.013	0.036	0.031	0.844	--	--	--	--	--
6.5	0.440	0.560	--	--	--	--	--	0.058	0.016	0.020	0.032	0.874	--	--	--	--	--
6.6	0.182	0.818	--	--	--	--	--	0.039	0.016	0.014	0.031	0.900	--	--	--	--	--
Pu-U																	
6.7	--	0.956	0.043	--	3.-4	--	--	0.018	0.012	0.009	0.030	0.930	0.079	0.443	0.247	0.149	0.082

TABLE D.7. (contd)

Fuel Cycle G																	
Discharge Fuel Compositions																	
	Th	U	Pu	²³⁷ Np	²⁴¹ Am	²⁴³ Am	²⁴² Cm	Isotopic Weight Fractions in U					Isotopic Weight Fractions in Pu				
								²³³	²³⁴	²³⁵	²³⁶	²³⁸	²³⁸	²³⁹	²⁴⁰	²⁴¹	²⁴²
<u>OU-Th</u>																	
1.1-6.1	0.773	0.222	0.004	0.001	1.-5	2.-5	4.-6	0.057	0.005	0.084	0.023	0.831	0.027	0.631	0.168	0.142	0.032
<u>Pu-Th</u>																	
2.2	0.952	0.015	0.032	1.-6	0.001	0.001	2.-4	0.926	0.065	0.009	4.-4	0	0.048	0.312	0.312	0.225	0.103
3.2	0.940	0.014	0.043	1.-6	0.001	0.001	2.-4	0.930	0.061	0.008	3.-4	0	0.080	0.280	0.311	0.215	0.114
4.2	0.929	0.014	0.054	1.-6	0.001	0.002	3.-4	0.933	0.059	0.008	3.-4	0	0.086	0.251	0.292	0.254	0.116
5.2	0.921	0.014	0.061	1.-5	0.001	0.002	3.-4	0.936	0.057	0.007	2.-4	0	0.102	0.242	0.314	0.208	0.135
6.2	0.918	0.014	0.065	1.-6	0.001	0.002	3.-4	0.936	0.056	0.007	2.-4	0	0.103	0.256	0.310	0.196	0.135
<u>ReOU-Th</u>																	
2.3	0.693	0.301	0.005	0.001	2.-5	2.-5	7.-6	0.060	0.010	0.039	0.031	0.861	0.064	0.621	0.164	0.125	0.025
3.3	0.733	0.261	0.004	0.001	2.-5	2.-5	6.-6	0.071	0.012	0.037	0.031	0.849	0.065	0.508	0.168	0.131	0.029
3.4	0.558	0.434	0.006	0.001	2.-5	2.-5	7.-6	0.048	0.013	0.021	0.032	0.887	0.061	0.644	0.163	0.112	0.021
4.3	0.743	0.252	0.004	0.001	2.-5	2.-5	6.-6	0.074	0.013	0.036	0.031	0.846	0.063	0.588	0.164	0.128	0.029
4.4	0.608	0.384	0.006	0.001	2.-5	2.-5	7.-6	0.056	0.016	0.020	0.032	0.876	0.060	0.624	0.161	0.113	0.021
4.5	0.353	0.637	0.008	0.002	3.-5	2.-5	8.-6	0.033	0.013	0.013	0.031	0.909	0.056	0.667	0.161	0.100	0.016
5.3	0.749	0.246	0.004	0.001	2.-5	2.-5	6.-6	0.076	0.013	0.036	0.030	0.844	0.066	0.603	0.169	0.132	0.030
5.4	0.619	0.374	0.006	0.001	2.-5	2.-5	7.-6	0.058	0.016	0.020	0.032	0.874	0.061	0.636	0.165	0.116	0.022
5.5	0.426	0.565	0.007	0.002	3.-5	2.-5	7.-6	0.039	0.016	0.014	0.031	0.900	0.057	0.664	0.161	0.102	0.016
5.6	0.100	0.889	0.010	0.001	3.-5	2.-5	9.-6	0.018	0.013	0.009	0.030	0.930	0.052	0.669	0.168	0.096	0.015
6.3	0.749	0.246	0.004	0.001	2.-5	2.-5	6.-6	0.076	0.013	0.036	0.030	0.844	0.066	0.603	0.169	0.132	0.030
6.4	0.628	0.365	0.006	0.001	2.-5	2.-5	7.-6	0.060	0.017	0.020	0.032	0.872	0.061	0.634	0.165	0.117	0.022
6.5	0.441	0.550	0.007	0.002	3.-5	2.-5	7.-6	0.041	0.017	0.014	0.031	0.898	0.056	0.663	0.161	0.103	0.017
6.6	0.179	0.809	0.009	0.002	3.-5	1.-5	8.-6	0.024	0.015	0.010	0.030	0.921	0.053	0.678	0.161	0.094	0.014
<u>Pu-U</u>																	
6.7	--	0.950	0.040	0.009	0.001	0.001	2.-4	0.009	0.011	0.008	0.028	0.944	0.075	0.407	0.251	0.165	0.103

TABLE D.8. Initial Fuel Compositions: Weight Fractions Denatured Urania-Thoria, PuO₂-UO₂ Recycle

	Fuel Cycle H																
	Th	U	Pu	237Np	241Am	243Am	242Cm	Isotopic Weight Fractions in U					Isotopic Weight Fractions in Pu				
								233	234	235	236	238	238	239	240	241	242
DU-Th																	
1.1-6.1	0.768	0.232	--	--	--	--	--	--	--	0.200	--	0.800	--	--	--	--	--
PuO2-UO2																	
2.2	--	0.956	0.043	--	3.-4	--	--	--	--	0.004	--	0.996	0.038	0.633	0.169	0.128	0.032
3.2	--	0.947	0.053	--	3.-4	--	--	--	--	0.004	--	0.996	0.051	0.561	0.207	0.134	0.047
4.2	--	0.941	0.058	--	4.-4	--	--	--	--	0.004	--	0.996	0.056	0.538	0.210	0.138	0.058
5.2	--	0.938	0.061	--	4.-4	--	--	--	--	0.004	--	0.996	0.057	0.504	0.230	0.145	0.064
6.2	--	0.932	0.068	--	5.-4	--	--	--	--	0.003	--	0.997	0.060	0.492	0.238	0.139	0.071
6.7	--	0.963	0.037	--	3.-4	--	--	0.018	0.013	0.009	0.030	0.930	0.050	0.492	0.238	0.139	0.071
ReDU-Th																	
2.3-6.3	0.688	0.312	--	--	--	--	--	0.057	0.005	0.084	0.023	0.831	--	--	--	--	--
3.4-6.4	0.555	0.445	--	--	--	--	--	0.060	0.010	0.039	0.031	0.861	--	--	--	--	--
4.5-6.5	0.353	0.647	--	--	--	--	--	0.048	0.013	0.021	0.032	0.877	--	--	--	--	--
5.6-6.6	0.103	0.897	--	--	--	--	--	0.033	0.013	0.013	0.031	0.909	--	--	--	--	--

TABLE D.8. (contd)

Fuel Cycle H

Discharge Fuel Compositions

DU-Th	Th	U	Pu	237Np	241Am	243Am	242Cm	Isotopic Weight Fractions in U					Isotopic Weight Fractions in Pu				
								233	234	235	236	238	238	239	240	241	242
1.1-6.1	0.773	0.222	0.004	0.001	1.-5	2.-5	4.-6	0.057	0.005	0.084	0.023	0.831	0.027	0.631	0.168	0.142	0.032
PuO2-UO2																	
2.2	--	0.964	0.034	1.-4	5.-4	0.001	2.-4	--	--	0.002	4.-4	0.998	0.042	0.451	0.262	0.170	0.075
3.2	--	0.956	0.042	1.-4	0.001	0.001	2.-4	--	--	0.002	4.-4	0.998	0.056	0.423	0.266	0.171	0.084
4.2	--	0.950	0.048	1.-4	0.001	0.001	2.-4	--	--	0.002	4.-4	0.998	0.060	0.409	0.269	0.170	0.090
5.2	--	0.948	0.050	1.-4	0.001	0.001	2.-4	--	--	0.002	4.-4	0.998	0.063	0.397	0.270	0.173	0.096
6.2	--	0.941	0.056	1.-4	0.001	0.001	2.-4	--	--	0.002	3.-4	0.998	0.065	0.393	0.273	0.169	0.100
6.7	--	0.955	0.034	0.009	5.-4	0.001	1.-4	0.009	0.011	0.008	0.028	0.944	0.059	0.438	0.249	0.163	0.091
ReDU-Th																	
2.3-6.3	0.693	0.301	0.005	0.001	2.-5	2.-5	7.-6	0.060	0.010	0.039	0.031	0.861	0.064	0.621	0.164	0.125	0.025
3.4-6.4	0.558	0.434	0.006	0.001	2.-5	2.-5	7.-6	0.048	0.013	0.021	0.032	0.887	0.061	0.644	0.163	0.112	0.021
4.5-6.5	0.353	0.637	0.008	0.002	3.-5	2.-5	8.-6	0.033	0.013	0.013	0.031	0.909	0.056	0.667	0.161	0.100	0.016
5.6-6.6	0.100	0.889	0.010	0.001	3.-5	2.-5	9.-6	0.018	0.013	0.009	0.030	0.930	0.052	0.669	0.168	0.096	0.015

TABLE D.9. Initial Fuel Compositions: Weight Fractions HEU-Th, LEU, ReLEU

	Fuel Cycle I																
	Th	U	Pu	²³⁷ Np	²⁴¹ Am	²⁴³ Am	²⁴² Cm	Isotopic Weight Fractions in U					Isotopic Weight Fractions in Pu				
								²³³	²³⁴	²³⁵	²³⁶	²³⁸	²³⁸	²³⁹	²⁴⁰	²⁴¹	²⁴²
<u>HEU-Th</u>																	
1.1-6.1	0.957	0.043	--	--	--	--	--	--	0.009	0.931	--	0.060	--	--	--	--	--
<u>LEU</u>																	
1.2-2.2	--	1.00	--	--	--	--	--	--	3.-4	0.030	--	0.970	--	--	--	--	--
<u>ReLEU</u>																	
2.3	--	1.00	--	--	--	--	--	0.011	0.002	0.017	0.008	0.962	--	--	--	--	--
3.3	--	1.00	--	--	--	--	--	0.014	0.003	0.015	0.012	0.956	--	--	--	--	--
4.3	--	1.00	--	--	--	--	--	0.014	0.004	0.017	0.016	0.950	--	--	--	--	--
5.3	--	1.00	--	--	--	--	--	0.014	0.005	0.017	0.019	0.945	--	--	--	--	--
6.3	--	1.00	--	--	--	--	--	0.015	0.005	0.016	0.022	0.941	--	--	--	--	--
<u>Discharge Fuel Compositions</u>																	
<u>HEU-Th</u>																	
1.1-6.1	0.953	0.036	1.-4	0.001	3.-8	--	1.-8	0.404	0.043	0.337	0.138	0.074	0.942	0.042	0.008	0.008	--
<u>LEU</u>																	
1.2-2.2	--	0.990	0.010	5.-4	3.-5	7.-5	9.-6	--	2.-4	0.008	0.004	0.988	0.013	0.587	0.229	0.130	0.041
<u>ReLEU</u>																	
2.3	--	0.989	0.010	0.001	4.-5	6.-5	1.-5	0.003	0.002	0.005	0.009	0.981	0.004	0.597	0.231	0.128	0.039
3.3	--	0.988	0.010	0.002	4.-5	6.-5	1.-5	0.003	0.003	0.005	0.012	0.976	0.006	0.604	0.227	0.126	0.037
4.3	--	0.987	0.010	0.003	4.-5	6.-5	1.-5	0.004	0.004	0.006	0.016	0.971	0.007	0.613	0.221	0.124	0.034
5.3	--	0.987	0.010	0.003	4.-5	5.-5	1.-5	0.004	0.004	0.007	0.019	0.966	0.009	0.618	0.218	0.123	0.032
6.3	--	0.986	0.010	0.004	4.-5	5.-5	1.-5	0.004	0.005	0.007	0.022	0.962	0.010	0.623	0.215	0.121	0.031

TABLE D.10. Discharge Exposure and Energy Contribution

Material	Exposure (MWd/MTHM)	Energy Fraction (%)	Material	Exposure (MWd/MTHM)	Energy Fraction (%)
<u>A</u> 1.1	30,000	100	<u>C</u> (contd)		
2.1	30,000	80.6	2.2	32,530	13.6
2.2	29,930	19.4	3.1	30,000	79.0
3.1	30,000	75.3	3.2	32,452	15.9
3.2	29,920	24.7	3.3	32,032	5.1
4.1	30,000	71.4	4.1	30,000	74.0
4.2	29,910	28.6	4.2	32,391	16.4
5.1	30,000	71.3	4.3	32,019	9.6
5.2	29,900	28.7	5.1	30,000	71.0
6.1	30,000	70.0	5.2	32,340	16.5
6.2	29,900	30.0	5.3	32,002	12.5
			6.1	30,000	69.1
<u>B</u> 1.1	30,000	100	6.2	32,298	16.4
2.1	30,000	86.5	6.3	31,987	14.5
2.2	32,530	13.5			
3.1	30,000	77.2	<u>D</u> 1.1	30,000	100
3.2	32,450	15.7	2.1	30,000	86.4
3.3	30,000	7.1	2.2	32,530	13.6
4.1	30,000	73.3	3.1	29,999	78.6
4.2	32,410	18.1	3.2	32,447	15.6
4.3	30,000	8.6	3.3	32,778	5.7
5.1	30,000	71.4	4.1	30,000	74.5
5.2	32,340	17.3	4.2	32,376	15.1
5.3	30,000	11.3	4.3	32,771	10.4
6.1	30,000	72.6	5.1	30,000	71.2
6.2	32,260	16.4	5.2	32,328	15.4
6.3	30,000	11.0	5.3	32,765	13.4
			6.1	30,000	69.0
<u>C</u> 1.1	30,000	100	6.2	32,284	15.3
2.1	30,000	86.4	6.3	32,760	15.7

TABLE D.10. (contd)

Material		Exposure (MWd/MTHM)	Energy Fraction (%)	Material		Exposure (MWd/MTHM)	Energy Fraction (%)
<u>E</u>	1.1	32,749	100	<u>G</u>	1.1	32,124	100
	2.1	32,749	31.6		2.1	32,124	26.4
	2.2	32,727	68.4		2.2	32,557	6.0
	3.1	32,749	32.3		2.3	31,885	67.6
	3.2	32,708	67.7		3.1	32,124	27.7
	4.1	32,749	32.3		3.2	32,505	8.0
	4.2	32,692	67.7		3.3	32,013	21.1
	5.1	32,749	31.9		3.4	31,497	43.2
	5.2	32,679	68.1		4.1	32,124	27.5
	6.1	32,749	32.3		4.2	32,471	9.5
<u>F</u>	6.2	32,667	67.7		4.3	32,032	22.7
	1.1	32,164	100		4.4	31,642	13.1
	2.1	32,164	32.4		4.5	30,923	27.2
	2.2	31,930	67.6		5.1	32,124	26.9
	3.1	32,164	34.6		5.2	32,426	9.5
	3.2	32,164	22.1		5.3	32,052	23.2
	3.3	31,543	43.3		5.4	31,672	14.1
	4.1	32,164	35.3		5.5	31,123	8.2
	4.2	31,930	23.4		5.6	30,235	18.1
	4.3	31,543	14.0		6.1	32,124	29.7
	4.4	30,962	27.3		6.2	32,410	3.1
	5.1	32,164	34.2		6.3	32,053	22.8
	5.2	31,930	23.9		6.4	31,698	14.4
	5.3	31,543	15.0		6.5	31,167	8.9
	5.4	30,962	8.9		6.6	30,449	5.3
	5.5	30,281	18.1		6.7	29,905	15.8
	6.1	32,164	46.2	<u>H</u>	1.1	32,124	100
	6.2	31,930	23.1		2.1	32,124	24.7
	6.3	31,543	15.3		2.2	29,900	7.7
	6.4	30,962	9.5		2.3	31,885	67.6
	6.5	30,281	5.9		3.1	32,124	28.2

TABLE D.10. (contd)

<u>Material</u>	<u>Exposure (MWd/MTHM)</u>	<u>Energy Fraction (%)</u>	<u>Material</u>	<u>Exposure (MWd/MTHM)</u>	<u>Energy Fraction (%)</u>
<u>H</u> (contd)			<u>I</u> 1.1	32,750	46.1
3.2	29,888	11.8	1.2	30,000	53.9
3.3	31,885	16.7	2.1	32,750	42.7
3.4	31,497	43.3	2.2	30,000	5.5
4.1	32,124	28.2	2.3	30,000	52.8
4.2	29,879	14.8	3.1	32,750	43.9
4.3	31,885	19.1	3.3	30,000	56.1
4.4	31,497	10.7	4.1	32,750	45.3
4.5	30,923	27.2	4.3	30,000	54.7
5.1	32,124	25.6	5.1	32,750	46.7
5.2	29,877	18.3	5.3	30,000	53.3
5.3	31,885	19.1	6.1	32,750	48.1
5.4	31,497	12.2	6.3	30,000	51.9
5.5	30,923	6.7			
5.6	30,236	18.1			
6.1	32,124	26.8			
6.2	29,868	15.7			
6.3	31,885	17.3			
6.4	31,497	12.2			
6.5	30,923	7.7			
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