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**SYSTEMS ANALYSIS OF AN ADVANCED NUCLEAR FUEL CYCLE
BASED ON A MODIFIED UREX+3c PROCESS**

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ACRONYMS

ABTR	Advanced Burner Test Reactor
ADU	Ammonium Diuranate
AFC	Alternative Fuel Cycle
AGNS	Allied-General Nuclear Services
ALARA	As Low As Reasonably Achievable
DOE	U.S. Department of Energy
DOT	U.S. Department of Transportation
DRU	Depleted Recycled Uranium
DU	Depleted Uranium
ENR	<i>Engineering News Record</i>
ERU	Enriched Recycled Uranium
EU	Enriched Uranium
FR	Fast Burner Reactor
GTCC	Greater Than Class C (waste)
GWe	Gigawatt-electrical
GWY	Gigawatt-year
HLW	High Level Waste
kgU	Kilograms Uranium
kW	Kilowatt
LLW	Low Level Waste
LWR	Light Water Reactor
MA	Minor Actinides (Np, Am, Cm)
MOX	Mixed Oxide Fuel (UO ₂ - PuO ₂)
mR/hr	Millirem per hour
MTIHM	Metric Tons of Initial Heavy Metal
MTU	Metric Tons Uranium
MW	Megawatt
MWD	Megawatt-day
MWe	Megawatt-electric
NFS	Nuclear Fuel Services Inc.
NRC	U.S. Nuclear Regulatory Commission
NU	Natural Uranium

EXECUTIVE SUMMARY

The research described in this report was performed under a grant from the U.S. Department of Energy (DOE) to describe and compare the merits of two advanced alternative nuclear fuel cycles -- named by this study as the “UREX+3c fuel cycle” and the “Alternative Fuel Cycle” (AFC). Both fuel cycles were assumed to support 100 1,000 MWe light water reactor (LWR) nuclear power plants operating over the period 2020 through 2100, and the fast reactors (FRs) necessary to burn the plutonium and minor actinides generated by the LWRs. These fuel cycles are described diagrammatically in Figures ES-1 and ES-2 at the end of this summary.

Reprocessing in both fuel cycles is assumed to be based on the UREX+3c process reported in earlier work by the DOE. Conceptually, the UREX+3c process provides nearly complete separation of the various components of spent nuclear fuel in order to enable recycle of reusable nuclear fuel materials, and the storage, conversion, transmutation and/or disposal of other recovered components. Output of the process contains substantially all of the plutonium, which is recovered as a 5:1 uranium/plutonium mixture, in order to discourage plutonium diversion. Mixed oxide (MOX) fuel for recycle in LWRs is made using this 5:1 U/Pu mixture plus appropriate makeup uranium. A second process output contains all of the recovered uranium except the uranium in the 5:1 U/Pu mixture. The several other process outputs are various waste streams, including a stream of minor actinides that are stored until they are consumed in future FRs.

For this study, the UREX+3c fuel cycle, is assumed to recycle only the 5:1 U/Pu mixture to be used in LWR MOX fuel and to use depleted uranium (tails) for the makeup uranium. This fuel cycle is assumed not to use the recovered uranium output stream but to discard it instead. On the other hand, the AFC is assumed to recycle both the 5:1 U/Pu mixture and all of the recovered uranium. In this case, the recovered uranium is reenriched with the level of enrichment being determined by the amount of recovered plutonium and the combined amount of the resulting MOX.

The study considered two sub-cases within each of the two fuel cycles in which the uranium and plutonium from the first generation of MOX spent fuel (i) would not be recycled to produce a second generation of MOX for use in LWRs or (ii) would be recycled to produce a

second generation of MOX fuel for use in LWRs. The study also investigated the effects of recycling MOX spent fuel multiple times in LWRs.

The study assumed that both fuel cycles would store and then reprocess spent MOX fuel that is not recycled to produce a next generation of LWR MOX fuel and would use the recovered products to produce FR fuel. The study further assumed that FRs would begin to be brought on-line in 2043, eleven years after recycle begins in LWRs, when products from 5-year cooled spent MOX fuel would be available. Fuel for the FRs would be made using the uranium, plutonium, and minor actinides recovered from MOX. For the cases where LWR fuel was assumed to be recycled one time, the 1st generation of MOX spent fuel was used to provide nuclear materials for production of FR fuel. For the cases where the LWR fuel was assumed to be recycled two times, the 2nd generation of MOX spent fuel was used to provide nuclear materials for production of FR fuel. The number of FRs in operation was assumed to increase in successive years until the rate that actinides were recovered from permanently discharged spent MOX fuel equaled the rate the actinides were consumed by the operating fleet of FRs.

To compare the two fuel cycles, the study analyzed recycle of nuclear fuel in LWRs and FRs and determined the radiological characteristics of irradiated nuclear fuel, nuclear waste products, and recycle nuclear fuels. It also developed a model to simulate the dynamics of flows of nuclear materials that could occur between and within the major systems of the two advanced nuclear fuel cycles over 81 years beginning in 2020 and ending in 2100. Simulations projected the dynamics of uses and flows of uranium, plutonium, and minor actinides as these nuclear fuel materials were produced and consumed in a fleet of 100 1,000 MWe LWRs and in FRs. The model also included recycle and reuse of extant inventories of LWR spent nuclear fuel. The results of the simulations allowed comparisons of the two fuel cycles from the standpoints of (i) cost, (ii) non-proliferation impacts, (iii) occupational and public radiological health, (iv) quantities of radioactive wastes generated, and (v) long-term sustainability. Results of the research also provide insights regarding:

- Strategies for, disadvantages of, and limitations of multiple recycling of uranium and plutonium from spent MOX fuel in LWRs
- Costs and benefits of reenriching and reusing uranium recovered by reprocessing in LWRs
- Effects of using uranium, plutonium, and minor actinides from LWR spent fuels to produce fuel for FRs

- Net rates of consumption (burning) in FRs of actinide elements produced in LWRs
- The potential to reverse increases of inventories of LWR spent fuel through reprocessing and recycling in an advanced nuclear fuel cycle
- Interdependencies of and interactions among the different systems of an advanced nuclear fuel cycle -- and the flows of nuclear materials between these systems.

Major findings, discoveries, and conclusions of the research include the following:

- (1) If plutonium and uranium is recycled more than one time in LWRs, a strategy of co-reprocessing MOX spent fuel with UO_2 spent fuel (or blending the products of reprocessing) would off-set disadvantages resulting from degraded quality in the isotopic compositions of uranium and plutonium that would occur otherwise if the MOX spent fuel was reprocessed separately. Using the co-reprocessing strategy, the composition of MOX nuclear fuel would stabilize after about the fourth recycle. This result would apply for both the UREX+3c fuel cycle and the AFC.
- (2) Reenriching and reusing uranium recovered in reprocessing LWR spent fuel (as would be the case for the AFC):
 - (a) Can reduce the nuclear fuel cycle's demand for natural uranium as well as the amount of mill tailings generated by about 10 percent, and
 - (b) Would result in a significant reduction of the radiotoxicity of uranium discarded by the fuel cycle.
- (3) For both the UREX+3c fuel cycle and the AFC, introduction of nuclear fuel reprocessing in 2031 at a rate of 3,000 MTIHM* per year and FRs beginning in 2043, would stop the continuing increase in, and could result in a substantial reduction of, the inventory of LWR spent nuclear fuel.
- (4) The net rate of consumption of actinide elements in an FR having a conversion ratio of 0.74 would be about 2.5 percent per year. The greatest rate of net consumption for long-lived actinides would be about 8 percent per year for the Np^{237} isotope. Consumption rates would be greater if the reactor conversion ratio was less.
- (5) Using makeup uranium and plutonium provided by reprocessing LWR MOX spent fuel, the composition of FR fuel would stabilize (reach an equilibrium) after about 4 recycles.
- (6) The amount of minor actinides available for producing FR fuel from twice recycled LWR MOX spent fuel could be as high as 7.2 percent of the heavy metal for the UREX+3c fuel cycle and 8.3 percent of the heavy metal for the AFC. The amount of minor actinides that could be incorporated into FR fuel may be limited

* MTIHM = metric tons of contained heavy metal prior to irradiation in a reactor.

to less than these amounts. Thus, consumption of minor actinides in an FR may require use of targets in axial or radial blankets in the reactor.

- (7) Savings in fuel cycle costs resulting from the recycle of uranium from the reprocessing of spent LWR fuel (AFC case) over a fuel cycle that does not employ such recycle (UREX+3c fuel cycle), are not realized until the price of natural uranium concentrates exceeds about \$90/lb U_3O_8 . Thus, other factors such as conservation of natural resources and disposal of recovered uranium would be more important considerations than would the cost.
- (8) The differences between the cost of implementing the UREX+3c fuel cycle and the AFC are small compared with the cost of implementing either of the fuel cycles. The total cost of implementing either of the fuel cycles would be of the order of \$1-trillion over the 2020-2100 period used in this study. (This does not include the capital costs of the LWRs and FRs, nor the revenues received from the electricity produced.)
- (9) The AFC may be slightly more vulnerable to proliferation than the UREX+3c fuel cycle -- but not to the extent that it shouldn't otherwise be considered for use.
- (10) The two fuel cycles produced about the same amounts of low level waste (LLW) - between 7.1 and 7.3 million cubic ft.-- with the AFC having a slight advantage because its net use of natural uranium is less. For perspective, the LLW produced by non-reactor processes and systems associated with recycle and reuse of nuclear fuel in these fuel cycles would be about 80% of that produced in operations of the 100 LWRs.
- (11) Because the AFC produces greater amounts of MOX recycle fuel for LWRs, than does the UREX+3c fuel cycle, the AFC would produce 14 to 16 percent more transuranium (TRU) waste. The fuel cycles would produce between 10 and 12 million cubic feet of TRU waste over the 2020-2100 period. The annual rate of production is about the same as the rate that TRU wastes are currently being disposed in DOE's Waste Isolation Pilot Plant (WIPP) near Carlsbad, New Mexico.
- (12) Because both fuel cycles would employ the same amount of reprocessing, the amount of high-level radioactive waste and fuel assembly hulls and associated wastes would be about the same. The combined volume of these wastes would be about 55 percent of the volume of the originating LWR spent fuel. This result suggests that the amount of space required for disposal of the high level waste (HLW) and fuel assembly hulls in a geologic repository would be about 55 percent of that required for the disposal of the originating spent nuclear fuel assemblies. Because these wastes would contain less than 1 percent of the long-lived actinides present in spent fuel, the radiotoxicity for disposal would be correspondingly less.

- (13) There is little difference in the consequences for worker and public radiological health between the UREX+3c fuel cycle and the AFC. The two fuel cycles would require construction and operation of comparable new fuel cycle facilities such as reprocessing, MOX fuel fabrication, fast reactors, fast reactor fuel reprocessing and fuel fabrication, uranium conversion, and waste management. The AFC would require installation of the additional capability to reenrich recovered uranium but this capability, when compared to other processes and systems required for the fuel cycles, would not present significant risks for the radiological health of workers or the public.
- (14) Both fuel cycles enhance the sustainability of nuclear electric power generation by improving the efficiency of utilization of uranium resources and reducing the toxicity of nuclear wastes that would be produced. Both fuel cycles would provide a foundation of infrastructure and experience for the evolution of sustained nuclear power generation, ultimately using the energy reserves contained in extant resources of depleted uranium and spent nuclear fuel.

Mostly, differences in the performance and impacts of the two fuel cycles are those associated with reuse of recovered uranium and the makeup uranium that is used to produce MOX fuel. These differences provide an advantage to the AFC for the use of natural uranium resources and for the disposal of uranium that is excess to the fuel cycle. Conversely, the differences provide an advantage to the UREX+3c fuel cycle regarding the amount of TRU wastes that are produced, the cost of MOX fuel used in the fuel cycle, and the need for dedicated facilities or capabilities to reenrich recovered uranium.

FIGURE ES-1
NUCLEAR FUEL CYCLE FLOWSHEET INVOLVING TOTAL RECYCLE OF PU ONLY
IN A 5:1 MIX WITH RECOVERED URANIUM
(UREX+3c Fuel Cycle)

FIGURE ES-2
NUCLEAR FUEL CYCLE FLOWSHEET INVOLVING RECYCLE OF ALL RECOVERED U + PU
 (AFC)

1.0 INTRODUCTION

This report describes research performed under a grant from the Department of Energy to develop comparisons of the merits of two advanced nuclear fuel cycles -- the UREX+3c fuel cycle and the Alternative Fuel Cycle (AFC).

1.1 BACKGROUND

The nuclear energy research program of the U.S. Department of Energy (DOE) has been investigating approaches for promoting the evolutionary development of the commercial nuclear fuel cycle, which presently provides about 20 percent of domestic electric power production in the U.S. The objectives for development include:

- (1) Securing reliable, economical, safe, and environmentally sustainable domestic nuclear energy for the future.
- (2) Promoting safe, environmentally acceptable, and sustainable use of nuclear power in other countries.
- (3) Promoting secure and safeguarded use by all countries of radioactive source and special nuclear material in the production of nuclear generated electric power.
- (4) Addressing and establishing safe and publically acceptable approaches for the disposition of radioactive wastes from nuclear electric power production.

With these objectives in mind, the DOE has sponsored an international initiative to develop an advanced nuclear fuel cycle that has the potential to achieve these goals. In the course of the resulting research, DOE has developed advanced technologies for recycling of used nuclear fuel from light water nuclear electric power reactors (LWRs) while providing for protection of weapons useable materials from diversion and misuse. The research has also addressed the challenge of managing long-lived radioactive actinide elements by developing concepts for fast reactors (FRs) (and related systems) that would consume these materials to produce energy.

The research is in its formative stages and thus is open to investigation of alternatives that might provide improvements to the concepts envisioned or insights for future implementation. The study presented here is one such investigation of alternatives. This study investigated

alternative implementations of the nuclear fuel cycle based on the UREX+3c process and variations thereto.

1.2 PURPOSE OF THE STUDY

Conceptually, the UREX+3c process provides one of the most complete separations of the various components of spent nuclear fuel in order to enable recycle of the recovered nuclear fuel materials, and the storage, conversion, transmutation and/or disposal of the other recovered components. One of the process outputs contains substantially all of the plutonium, which is recovered as a 5:1 uranium/plutonium (U:Pu) mixture, in order to discourage plutonium diversion. All of the mixed oxide (MOX) fuel for recycle is made using this 5:1 U/Pu mixture plus appropriate makeup uranium. A second process output contains all of the recovered uranium except the uranium in the 5:1 U/Pu mixture. The several other process outputs are various waste streams, including a stream of minor actinides that are stored until they are consumed in future FRs.

The purpose of the research performed under this study was to identify, analyze and compare the attributes of two alternative UREX+3c-based fuel cycle systems that start MOX recycle by substituting MOX fuel for some of the UO_2 reload fuel required by a large base of 100 operating LWRs. The first of these systems (hereinafter referred to as the UREX+3c fuel cycle) is assumed to recycle only the 5:1 U/Pu mixture to be used in LWR MOX fuel, uses depleted uranium (tails) for the makeup uranium, and disposes of the uranium in the recovered uranium output stream. The second of these systems (hereinafter referred to as the AFC) is assumed to recycle both the 5:1 U/Pu mixture and all of the recovered uranium. In this case, the recovered uranium is re-enriched with the level of enrichment being determined by the amount of recovered plutonium and the combined amount of the resulting MOX. As sub-cases within each of the two alternative fuel cycles, it was assumed that spent first generation MOX fuel would be recycled* once or twice in LWRs. After discharge from LWRs, the spent MOX that is not recycled would be stored for future reprocessing and used to produce FR fuel. FRs are introduced after recycle begins in LWRs, when 5-year cooled spent MOX fuel is available, and

* Whenever in this report it is stated that a particular type of fuel is recycled, or similar terms are used, it means that the uranium and/or plutonium contained in the spent fuel is being recycled.

at a level needed to consume the spent MOX permanently discharged from the LWRs. The FRs also work off the inventories of LWR-generated minor actinides.

Thus, the study results as reported herein consist of the following:

- (1) Establishment of Baseline Information for Systems Evaluation (Section 3.0)
 - (a) Conceptual Design of UREX +3c and AFC Systems (Section 3.1)
 - (b) Basic Fuel Characteristics & Study Assumptions (Section 3.2)
 - (c) Change in Isotopic Ratios of Uranium During Reenrichment of Recovered U (Section 3.3)
 - (d) Spent UO_2 and MOX Fuel Characteristics After Repeated Recycle in LWRs (Section 3.4)
 - (e) Spent FR Fuel Characteristics After Repeated Recycle in FRs (Section 3.5)
- (2) Analysis of Systems Aspects of Reprocessing & Recycle (Section 4.0)
- (3) Comparative Analysis and Evaluation of Systems
 - (a) Differences in the Costs of the Systems (Section 5.0)
 - (b) Differences in the Non-Proliferation Aspects of the Systems (Section 6.0)
 - (c) Differences in Potential Radiological Health Impacts for the Systems (Section 7.0)
 - (d) Differences in Waste Quantities & Types for the Systems (Section 8.0)
 - (e) Differences in the Sustainability of the Systems (Section 9.0)
- (4) Related Systems Studies (Section 10.0)

A comparison of the actual accomplishments with the goals and objectives of the study has been included in Appendix F.

2.0 STUDY APPROACH

To conduct an evaluation of the relative merits of the UREX+3c fuel cycle and the AFC it was necessary to assemble or develop a number of analysis tools, basic data, system concepts, and overall concepts of operations for the fuel cycles.

A fundamental tool of the analysis was the ORIGEN computer program developed by the Oak Ridge National Laboratory. This program, along with distributed data sets was essential and was used to evaluate production, depletion, and post-irradiation decay of radioisotopes in LWR and FR fuels. In addition, the analysis team developed a tool to evaluate the enrichment and depletion of uranium isotopes in a centrifuge enrichment cascade. This tool is described in Appendix A of this report. The Excel[®] computer software developed by Microsoft Corporation was a central resource for quickly and efficiently analyzing the flows and balances of fuel cycle materials. The Excel[®] software was used to calculate the relationships of material flows and balances in a plethora of evaluations including: utilization and composition of fuel cycle materials, spent nuclear fuel inventories, costs, production of wastes, and source terms for evaluating radiological health consequences. The MicroShield[®] computer program was used to estimate worker doses for facilities where fuel cycle materials would be processed.

Data used in the analyses were collected from many sources; these data included:

- Nuclear data distributed with the ORIGEN and MicroShield[®] computer programs
- Technical descriptions of LWR and FR (Advanced Burner Test Reactor, ABTR) nuclear fuel and nuclear reactors
- Technical descriptions of nuclear fuel reprocessing and recycle facilities and systems
- Technical descriptions of spent nuclear fuel transportation casks
- Cost data such as historic records of cost for development and construction of the Allied-General Nuclear Services reprocessing plant in Barnwell, South Carolina and the Nuclear Fuel Services reprocessing plant in West Valley, New York
- Nuclear industry estimates for volumes and quantities of low-level radioactive waste, TRU wastes, and high level radioactive wastes that would be produced by nuclear fuel cycle facilities
- Radiological dose conversion factors published by the U.S. Environmental Protection Agency

The analysis adopted recently published and historic descriptions or examples to characterize the major systems of the alternative nuclear fuel cycles evaluated and compared by

the study. The concepts included ones for FRs, state-of-the-art nuclear fuel reprocessing facilities, MOX and FR fuel production facilities, and recycled uranium enrichment facilities.

Last, the analysis developed spreadsheet models to conduct simulations of 81 years of material flows for each of the alternative nuclear fuel cycles studied. These simulations were based on the following key assumptions:

- (1) 100 LWRs (or an equivalent number of replacement LWRs) would continue to operate at a power level of 1,000 MWe each for the remainder of the century.
- (2) Extant inventories of LWR spent nuclear fuel, estimated to be about 80,000 MTIHM in 2020 increasing at a rate of 1,840 MTIHM per year, would be available for reprocessing and recycle.
- (3) The burnup of spent UO_2 reactor fuel would be 60,000 MWD/MTIHM.
- (4) LWR spent nuclear fuel reprocessing at a rate of 3,000 MTIHM per year would begin in 2031 and would continue uninterrupted. The youngest spent nuclear fuel more than 5 years cooled would be reprocessed.
- (5) The U:Pu product of the UREX+3c separations process would be used immediately to produce MOX fuel, which would be burned in LWRs.
- (6) The initial enrichment, burnup and use of MOX nuclear fuel in LWRs would be the equivalent to those for UO_2 nuclear fuel.
- (7) FRs would begin to come on-line in 2043 when TRU would become available from reprocessing spent MOX fuel. The conversion ratio for FRs would be about 0.74.
- (8) The number of FRs in operation would increase by 1 per year if LWR spent fuel was recycled only one time, and by 1 every 2 years if LWR spent fuel was recycled two times. The number of FRs would not increase for years following the year in which the plutonium and uranium recovered from MOX spent fuel was the same as the amount of recovered plutonium and uranium consumed in the operating FRs.
- (9) Spent FR fuel would be cooled for one year, reprocessed, recycled and augmented by plutonium and uranium recovered from LWR spent MOX fuel, then immediately re-inserted into FRs.

The material flows and balances that were determined in the simulations provided the information needed to evaluate differences in cost, non-proliferation, potential radiological health impacts, waste quantities, and sustainability of the UREX+3c fuel cycle and the AFC.

Initially, the analysis assumed that the UREX+3c fuel cycle would recycle uranium and plutonium recovered from LWR spent fuel as MOX to fuel LWRs two times, while the AFC would recycle the recovered uranium and plutonium only once -- after which both fuel cycles would reprocess MOX spent fuel to recover uranium, plutonium, and minor actinides to produce fast reactor fuel. However, in the simulations this difference in the number of times LWR spent fuel was recycled resulted in the UREX+3c system installing 23 fast reactors during the 81 year simulation and the AFC system installing 43 of the reactors over the same time period. This made it difficult to develop and present an easy to understand side-by-side comparison of the two systems. The amounts of resources required, waste produced, sizes of facilities, and amount of electricity produced would differ substantially -- and a significant reason for the differences could be viewed as a somewhat arbitrary selection of the number of times spent fuel would be recycled in LWRs.

Thus, to improve the ease of understanding of fundamental differences between the UREX+3c and AFC systems, the analysis approach was changed. The revised approach analyzed both systems for recycle of uranium and plutonium as MOX to LWRs either once or twice -- followed by reprocessing the last generation of MOX to recover uranium, plutonium, and minor actinides and produce FR fuel. This allowed comparison of the effects of both one and two recycles of recovered uranium and plutonium to LWRs by both fuel cycles, as well as a comparison of a system that recycles all of the reprocessed uranium and one that does not -- with both systems employing the same number of LWRs and FRs.

Thus, four fuel cycle simulations were developed for a fleet of 100 LWRs operating over 81 years (2020 to 2100) -- two for the UREX+3c fuel cycle and two for the AFC, and these form the basis for the various comparisons of the systems that are made in this report.

3.0 BASELINE INFORMATION FOR COMPARATIVE EVALUATION OF SYSTEMS

3.1 DESCRIPTION OF CONCEPTUAL DESIGN OF SYSTEMS

This section sets forth a general description of the two fuel cycles examined, and the processes involved in each.

3.1.1 UREX+3c Fuel Cycle

The UREX+3c fuel cycle is shown in Figure 3-1 (Ref. 1). It basically involves UO_2 fuel being fabricated from enriched uranium (5% U^{235}) with the depleted uranium being converted to a stable form (U_3O_8 or UF_4) for long term storage. After irradiation in a LWR, the spent fuel is reprocessed to separately recover a 5:1 U:Pu mixture^{*}, the excess uranium contained in the spent UO_2 fuel, and minor actinides (Np, Am, and Cm). The excess uranium is subsequently disposed and the other recovered materials are recycled in one of two ways, as follows:

- (1) Recycled to MOX fuel for LWRs -- the U:Pu mixture is combined with extant makeup depleted uranium from enrichment and fabricated into MOX fuel for recycle to LWRs. In this study, both one and two such recycles were considered.
- (2) Recycled to metallic fuel for FRs -- after recycling of MOX to LWRs, the U:Pu mixture from reprocessing is combined with the minor actinides recovered during reprocessing (of both UO_2 and MOX fuel) and fabricated into either the initial cores of FRs or reloads.

After irradiation in a FR, the spent FR fuel is reprocessed to separately recover a U:Pu mixture, and combined with makeup RU/Pu/MA to produce fuel for successive recycles into FRs.

3.1.2 Alternative Fuel Cycle

The AFC is shown in Figure 3-2. The principal difference between it and the UREX+3c fuel cycle is that it reenriches the excess uranium recovered from reprocessing of spent UO_2 and MOX fuel destined for recycle to LWRs. This also requires that the depleted uranium from enriching recovered uranium (UF_6) be converted to uranium oxide. Since the recycle uranium involved has higher concentration of U^{234} and U^{236} , it may be necessary to both enrich the recovered uranium and convert the resulting depleted uranium in facilities that are separate from the same operations involving virgin enriched and depleted uranium.

^{*} Plutonium is not recovered in pure form, but as a 5:1 mixture with uranium to reduce its attractiveness to proliferation.

FIGURE 3-1
NUCLEAR FUEL CYCLE FLOWSHEET INVOLVING TOTAL RECYCLE OF PU ONLY
IN A 5:1 MIX WITH RECOVERED URANIUM
(UREX+3c Fuel Cycle)

FIGURE 3-2
NUCLEAR FUEL CYCLE FLOWSHEET INVOLVING RECYCLE OF ALL RECOVERED U + PU
(AFC)

3.1.3 Separations Processes

A general flowsheet for reprocessing of spent UO_2 , MOX and FR fuels and the disposal of reprocessing wastes is shown in Figure 3-3. This flowsheet is the same for both the UREX+3c fuel cycle and the AFC, except that the quantities involved will be slightly different.

Spent LWR fuel is first subjected to voloxidation for recovery of tritium, followed by dissolution in nitric acid preparatory to feeding to the UREX +3c aqueous separations process. During dissolution and clarification, xenon, krypton, carbon dioxide and iodine are released; iodine is subsequently converted to a high level waste form. The dissolved fuel then is fed into the basic separations process (UREX) out of which comes four streams: (i) a technetium stream, (ii) a 5:1 mixture, respectively, of recovered uranium and recovered plutonium, (iii) a pure recovered uranium stream, and (iv) the raffinates containing the MAs and fission products. The hulls from the dissolution process, sludge from the clarification of the dissolver solution, and technetium recovered from UREX, are alloyed or compacted for geological disposal.

The uranium/plutonium mixture is subsequently diluted with makeup uranium and fabricated into MOX fuel for recycle into thermal reactors (presumably LWRs). The remaining uranium is either reenriched or converted to a stable solid and stored and/or otherwise disposed of.

The raffinate from UREX is fed to another separations process (TRUEX) that separates it into 2 streams: a stream containing fission products except lanthanides, and a stream containing lanthanide fission products and minor actinides. The lanthanide-free fission product stream is fed to a third separations process (FPEX) where the cesium, strontium, barium and rubidium are separated from the remaining fission products. The stream from the TRUEX process containing the lanthanide fission products and minor actinides is fed to a fourth separations process (TALSPEAK) to separate these. The cesium etc. stream is then converted to a form suitable for long term storage (~300 years), probably as a silicate or an oxide. All remaining fission products are converted into a form suitable for long term storage and disposal (probably by vitrification) and the minor actinides are stored for later fabrication into fuel for a FR.

FIGURE 3-3
BASIC UREX+3c SEPARATIONS PROCESS

3.1.4 LWR Fuel Fabrication Process

General flowsheets for the fabrication of LWR MOX fuel are included in Figures 3-4, 3-5 and 3-6. These flowsheets are the same for both the UREX+3c fuel cycle and the AFC, except that the quantities involved may be different.

Figure 3-4 covers both the conversion of reenriched uranium (in the AFC) and makeup depleted uranium (in the UREX+3c fuel cycle) from UF_6 to a uranyl nitrate solution. The process steps are identical, and include dissolution of UF_6 in water to form a UO_2F_2 solution, followed by precipitation of the uranium as ammonium diuranate (ADU), filtration to recover the ADU -- followed by drying and calcination of the ADU to produce U_3O_8 . The U_3O_8 is then dissolved in nitric acid.

Figure 3-5 covers the conversion of RU^*/Pu nitrate solution from the basic UREX+3c separations process in Figure 3-3, and the enriched RU or depleted RU from Figure 3-4, into fuel pellets for LWR MOX fuel. In the UREX+3c fuel cycle, RU/Pu and DU in nitrate solution are blended to obtain the desired isotope concentrations for MOX fuel manufacture. In the AFC, RU/Pu and the enriched RU are blended to obtain the desired isotope concentrations. Thereafter, all process steps are the same. Plutonium and uranium are precipitated from the blended nitrate solutions as an ADU/plutonium hydroxide mixture, which is then dried and subsequently calcined in a reducing atmosphere to produce UO_2 - PuO_2 powder. This powder is then milled and classified, pressed into pellets, the pellets sintered at high temperature under a hydrogen atmosphere, and inspected. Pellets having diameters higher than specifications, but that otherwise meet required specification, are centerless ground to dimensions. Other non-specification pellets are recycled to the calcination and reduction step in the process.

Figure 3-6 covers the loading of the UO_2 - PuO_2 pellets into tubing and the assembly of the MOX fuel. Tubing is received from an outside supplier after which it is inspected, cleaned and prepared for loading, and one end-plug welded in place. Other assembly components are also received from outside suppliers, inspected, and cleaned and prepared for assembly operation. The UO_2 - PuO_2 pellets are then loaded into the tubing, the remaining end cap inserted and welded into place, and the resulting fuel rod leak tested and autoclaved. The loaded tubes are then inserted into the assembly skeleton and the end fittings put in place -- followed by a detailed final inspection of the completed assembly.

* RU = uranium recovered from reprocessing

FIGURE 3-4
LWR MOX FUEL FABRICATION PROCESS
 Conversion of UF₆ Containing Either RU or DU to a Nitrate Solution*

*May Require Separate Facilities Due to Differences in Radioactivity of DU and Reenriched RU

FIGURE 3-5
LWR MOX FUEL FABRICATION PROCESS
 Pellet Manufacture*

*May Require Separate Facilities Due to Differences in Radioactivity of DU and Reenriched RU

FIGURE 3-6
LWR MOX FUEL FABRICATION PROCESS
Pellet Loading and Bundle Assembly*

*May Require Separate Facilities Due to Differences in Radioactivity of DU and Reenriched RU

3.1.5 FR Fuel Fabrication Process

General flowsheets for the fabrication of metallic FR fuel are included in Figures 3-7 and 3-8. These flowsheets are the same for both the UREX+3c fuel cycle and the AFC, except that the quantities involved may be different.

Figure 3-7 covers the conversion of RU/Pu and minor actinide nitrates into metal melt stock. The nitrate solutions are blended to the required isotopic concentration. The blended solutions are then co-precipitated as a mixture of ammonium diuranate and TRU hydroxides -- followed by drying, calcination, and reduction to a U/Pu/MA* dioxide mixture. This dioxide mixture is then hydrofluorinated to produce a U/Pu/MA tetrafluoride mixture, which is blended with calcium metal, calcium chloride and iodine and heated in a thermite-type bomb to yield a metal button of U/Pu/MA. The slag from the bomb reduction, along with other scrap materials produced in the processes described in both Figures 3-7 and 3-8, is processed by dissolution and solvent extraction to recover U/Pu/MA for recycle to the process.

Figure 3-8 covers the fabrication of U/Pu/MA/Zr metal into fuel rods, and assembly of the FR fuel. Tubing is received from outside suppliers after which it is inspected, cleaned and prepared for loading, and one end-plug welded in place. Other assembly components are also received from outside suppliers, inspected, and cleaned and prepared for assembly operations. The U/Pu/MA metal is melted in an induction furnace, where zirconium metal is added to form a U/Pu/MA/Zr alloy. The composition of the alloy is adjusted during induction melting. The plug in the bottom of the melting crucible is removed and the molten alloy is allowed to drain into an ingot mold. After cooling, the ingot is cropped and the croppings returned to the melt crucible for a subsequent melt. The alloy ingot is then extruded into rods which, after inspection, are cut to size and loaded into the tubing. Sodium gap filler is added and the remaining end-cap is inserted into the open end of the tube and welded. The resulting fuel pin is then tested and autoclaved. The fuel pins are then inserted into the assembly skeleton and the end fittings put in place -- followed by a detailed final inspection of the completed fuel assembly.

* MA = minor actinides

FIGURE 3-7
FR FUEL FABRICATION PROCESS
Metal Production

FIGURE 3-8
FR FUEL FABRICATION PROCESS
Rod Loading and Bundle Assembly

3.2 BASIC SPENT FUEL CHARACTERISTICS AND STUDY ASSUMPTIONS

In order to make a comparative evaluation of the UREX +3c fuel cycle and the AFC, it was first necessary to identify the basic characteristics of the nuclear fuel to be used in the systems and other assumptions applicable for the study.

3.2.1 LWR and LWR Fuel Assumptions

The basic characteristics of the existing and forecast inventory of spent nuclear fuel (SNF) from LWRs in the U.S. have been included in Appendix A for reference. However, in order to simplify the comparison process, it was decided to make the systems comparisons based on the assumptions that the UO_2 first introduced to the reactor could all be represented by pressurized water reactor (PWR) fuel, and that all such fuel would be the Westinghouse 17x17 type. A summary of the PWR fuel designs for both UO_2 and MOX fuel that were assumed for this study is shown in Table 3-1. Other study assumptions involving LWRs are listed below:

- (1) Average plant capacity is 1,000 MWe.
- (2) Average power level is 40 MW/MTIHM.
- (3) Average burnup is 60,000 MWD/MTIHM.
- (4) Core loading is 82.8 MTIHM (180 assys @ 0.46 kg/assy).
- (5) Refueling cycle is every 18 months (1/3 of core replaced).
- (6) Average residence of fuel in reactor is 4.5 years (1,500 full power days).
- (7) Total Pu content of MOX fuel will not be greater than 10%.
- (8) In order to compensate for increases in the U^{236} content of recycled uranium, additional fissile material will be added to recycled fuel, as appropriate.
- (9) Cooling time after discharge of fuel from reactor prior to transport and reprocessing is 5 years.
- (10) Transportation, reprocessing, and MOX fabrication occurs in 6th year after discharge of fuel from reactor
- (11) Recycled fuel is inserted in a reactor at beginning of 7th year after discharge of contained recycled uranium and plutonium from the reactor

TABLE 3-1
PWR FUEL DESIGN DATA

Description	UO₂ Fuel^a	MOX Fuel^a
Equivalent Fissile Content ^b	5%	5%
No. Rods Per Assembly	264	264
No. Water Holes	25	25
Rod Pitch (cm)	1.260	1.260
Assembly Pitch (cm)	21.42	-
Fuel Rod Data		
Fuel Density (g/cm ³)	10.41	10.41
Pellet Diameter (cm)	0.7844	0.819
Clad Material	Zircaloy-2	Zircaloy-2
Clad ID (cm)	0.8001	-
Clad OD (cm)	0.9144	0.950
Fuel Temperature (°K)	1000	900
Clad Temperature (°K)	620	640
Guide Tube Data		
Guide Tube Material	Zircaloy-2	Zircaloy-2
Inner Radius (cm)	0.5614	-
Outer Radius (cm)	0.6020	-
Moderator Data		
Density (g/cm ³)	0.6700	0.720
Average Boron Concentration (ppm)	650	600
Moderator Temperature (°K)	600	577

^aWestinghouse 17x17 array

^bAdjusted as required to offset effects of U²³⁶

3.2.2 FR and FR Fuel Assumptions

Identification of FR fuel (as well as the FR technology) to be used for this study presented a significant problem inasmuch as there is little actual experience with FRs in the U.S. -- compared with the vast amount of experience with LWRs and LWR fuel. After extensive reviews of information on FRs, it was decided to use the fuel and fuel cycle description set forth in the Advanced Burner Test Reactor (ABTR) Preconceptual Design Report (Ref. 2) that was issued in the fall of 2006 by Argonne National Laboratory. The ABTR is a 250 MWt (95 MWe) sodium cooled reactor having a conversion ratio of about 0.72. For the purposes of this study it was decided to scale this reactor up to 421 MWt (160 MWe) and use this as one of nine reactor modules in a single power plant (3789 MWt/1440 MWe) similar to that described by Cockey (Ref. 3). This system is based on the Power Reactor Innovative Small Module (PRISM) concept

originated by General Electric Co. The plant consists (conceptually) of nine reactor modules arranged in 3 power blocks, each consisting of three reactor modules and a single dedicated steam generator and turbine generator. The core of a single reactor module was scaled linearly from the ABTR design, which should result in conservatively high loadings and discharges of fuel. A summary of FR designs that were used for this study is shown in Table 3-2. Other study assumptions involving FRs are listed below:

- (1) Average plant capacity is 1,440 MWe.
- (2) Average power level is 66 MW/MTIHM.
- (3) Average burnup is 97.2 GWD/MTIHM.
- (4) Core loading (single module) is 6.782 MTIHM.
- (5) Refueling cycle is every 4 months (1/12 of core replaced).
- (6) Average residence of fuel in reactor is 4 years.
- (7) Cooling time after discharge of fuel from reactor prior to transport and reprocessing is 1 year.
- (8) Transportation, reprocessing, and recycle fuel fabrication occurs in 2nd year after discharge of spent fuel from reactor
- (9) Recycled fuel is inserted in the reactor at beginning of 3rd year after discharge of contained plutonium from the reactor
- (10) Makeup plutonium and uranium will be added to that recovered during reprocessing of spent FR fuel as necessary to meet the fuel requirements of the FRs.

TABLE 3-2
FAST BURNER REACTOR FUEL DESIGN DATA

Description	Specification
Heavy Metal Content	
TRU Content	21%
U Content	79%
No. Rods Per Assembly	217
Rod Composition	U-TRU-Zr (10%) Metal Alloy
Fuel Rod Data	
Fuel Density (g/cm ³)	17.64
Fuel Diameter (cm)	0.603
Clad Material	HT9
Clad ID (cm)	0.696
Clad OD (cm)	0.800
Wire Wrap Diameter (cm)	0.103
Assembly Data	
Overall Length (cm)	328
Assembly Pitch (cm)	14.598
Inter-Assembly Gap (cm)	0.400

3.2.3 Common Assumptions

Following are the assumptions used in this study that are common to both LWR and FR fuel cycles:

- (1) Tails assay in uranium enrichment activities is 0.25% U²³⁵.
- (2) Process loss in conversion, reprocessing, and MOX fuel fabrication is 0.5% for each.
- (3) No adjustments in fissile content will be made for the presence of plutonium isotopes that are neutron absorbers because such isotopes are converted into fissile isotopes during irradiation:

3.3 METHODOLOGY FOR DETERMINING THE CHANGE IN ISOTOPIC RATIOS OF URANIUM DURING REENRICHMENT OF RECOVERED URANIUM

A methodology was established for calculating the relative abundances of uranium isotopes (U^{234} , U^{235} , U^{236} and U^{238}) in the uranium product of a gas centrifuge cascade that uses the uranium recovered from spent fuel as the feed material. This methodology is described in Appendix B to this report. The purpose here was to determine the amount of U^{236} in enriched recovered uranium. This allowed estimation of the amount of additional, offsetting, U^{235} enrichment that would be needed to provide enrichment equivalent to 5% enrichment of natural uranium. An additional purpose was to determine the amount of U^{234} that, because it is a source of ionizing radiation, would be present in enriched recycle uranium that could adversely affect fabrication of recycle reactor fuel.

3.4 SPENT FUEL CHARACTERISTICS AFTER USE IN LWRs

Calculations were made to determine the changes in the concentrations of uranium and plutonium isotopes, minor actinides, lanthanides, and other fission products resulting from multiple recycling of plutonium and uranium in LWRs -- 5 years after discharge from the reactor. These were accomplished using the ORIGEN-ARP computer code and the methodology described in Appendix B for calculating abundances of various uranium isotopes in uranium that was recycle from SNF and then subsequently enriched. The purpose of these calculations was to be able to determine (i) the extent to which reactivity has to be added to the plutonium for effective recycle, (ii) the amounts of fission products and minor actinides that have to be dealt with in waste solidification and repository disposal, (iii) the amounts of minor actinides that might otherwise have to be fabricated into fuel for a FR, and (iv) to be able to assess the radiological safety aspects of the UREX +3c fuel cycle and the AFC.

3.4.1 Characteristics When Spent UO_2 and MOX Fuel are Reprocessed and Recycled Together

One set of calculations was made assuming companion amounts of spent UO_2 and MOX discharged from LWRs at the same time fuels were reprocessed and recycled together. These calculations were made for each of 5 different alternative sources of makeup uranium (i.e., that which had to be added to the 5:1 mixture of U:Pu to appropriately dilute the fissile content to the

level required for LWRs) -- and for up to 4 generations of recycle. These scenarios were as follows:

- Scenario #1** U and Pu mix from reprocessing is blended with depleted uranium to produce MOX fuel (5% equivalent fissile content)
- Scenario #2** U and Pu mix from reprocessing is blended with enriched recycled uranium to produce MOX fuel (5% equivalent fissile content)
- Scenario #3** U and Pu mix from reprocessing is blended with recycled uranium to produce MOX fuel (5% equivalent fissile content)
- Scenario #4** U and Pu mix from reprocessing is blended with natural uranium to produce MOX fuel (5% equivalent fissile content)
- Scenario #5** U and Pu mix from reprocessing is blended with natural uranium to produce MOX fuel (5% equivalent fissile content), with excess recycle uranium being enriched for use in UO₂ fuel

Scenario #1 is the situation relevant to the UREX+3c fuel cycle and Scenario #2 is the situation relevant to the AFC. The other scenarios were included to demonstrate the impact of possible alternative situations. The results of the calculations are included in Tables C-1 through C-15 of Appendix C. Some of the key characteristics are summarized in Tables 3-3 through 3-6. The following summarizes the conclusions drawn from the foregoing tables and Appendix C:

- (1) From all of the results reported above, there does not appear to be a technical impediment to multiple recycling of plutonium and uranium in LWRs as long as the spent MOX fuel is reprocessed and recycled together with the companion amounts of spent UO₂ fuel. This results in the dilution of plutonium and uranium that has been recycled multiple times with that which has not yet been recycled, or has been recycled fewer times. The foregoing is based on the assumption that the amount of any fissile material that must be added to the recycled MOX fuel to achieve the required level of reactivity can be effected economically.
- (2) Successive coprocessing and recycle of MOX and UO₂ fuel over 4 generations results in:
 - (a) A reduction in the U²³⁵ content of the recovered uranium of 17-20 percent from that of spent UO₂ fuel, except where enriched recycle uranium is used as makeup uranium (blend stock) for LWR MOX fuel (Scenario #2), or the recovered uranium is enriched and blended with previously

- unirradiated uranium for use as UO_2 fuel (Scenario #5) -- where there is little change. (See Table 3-3 and Appendix C).
- (b) A reduction in U^{236} content of the recovered uranium of 6-17 percent from that of spent UO_2 fuel, except where enriched recycle uranium is used as makeup uranium for LWR MOX fuel (Scenario #2), or the recovered uranium is enriched and blended with previously unirradiated uranium for use in UO_2 fuel (Scenario #5) -- where there is an increase of up to 48 percent. (See Table 3-3 and Appendix C).
 - (c) A reduction in the fissile content of the recovered plutonium of 18-19 percent from that of UO_2 fuel, except where enriched recycle uranium is used as makeup uranium for LWR MOX fuel (Scenario #2), or the recovered uranium is enriched and blended with previously unirradiated uranium for use in UO_2 fuel (Scenario #5). (See Table 3-4 and Appendix C).
 - (d) Little change in the Np content of the spent fuel, except where recycle uranium is used as makeup uranium for LWR MOX fuel. If the recycled uranium is used as makeup for MOX fuel (Scenario #3), there is an increase in the Np content of about 18 percent from that of UO_2 fuel. If the recycled uranium is enriched and used as makeup fissile material (Scenario #2), there is an increase in the Np content of about 220% from that of UO_2 fuel. (See Table 3-5 and Appendix C).
 - (e) A 300 percent increase in the Am content of the spent fuel, and about a 630 percent increase in the Cm content thereof for all scenarios studied. (See Table 3-5 and Appendix C).
- (3) The concentrations of a majority of the fission products in the spent fuel either decreased or didn't change much for the various scenarios over the 4 generations of recycle. The only ones that increased over 25% were (i) Cd (37+%) and Ag (29%), which together constitute about 0.5% of all fission products, and (ii) Pd (20-35%) which accounts for about 5% of all fission products -- and these didn't vary much over the various scenarios. (See Appendix C.)
 - (4) The weight of MOX fuel that was required to supplement UO_2 fuel in a reactor loading was generally highest for Scenario #2, and lowest for Scenario #1. (See Table 3-6.) Correspondingly, the companion amount of UO_2 fuel needing to be loaded with MOX fuel was the least for Scenario #2, and the greatest for Scenario #1.
 - (5) Because of the presence of significant amounts of the U^{236} isotope, using enriched recycle uranium to produce LWR fuel requires the fissile enrichment of the fuel be greater than that for fuel produced from natural uranium. For fuel designed to achieve a burnup of 60,000 MWD/MTIHM, the fissile enrichment must be increased by about 0.3 percent (from 5.0 percent to 5.3 percent) if the concentration of U^{236} in the fuel is at 1 percent. The amount of increased enrichment that is required is reduced when enriched recycled uranium (enriched

to only about 3 percent) is blended with recovered plutonium to produce MOX fuel.

TABLE 3-3
PERCENT U^{235} AND U^{236} IN URANIUM RECOVERED FROM REPROCESSING OF
SUCCESSIVE GENERATIONS OF SPENT UO_2 AND MOX FUEL UNDER VARIOUS
SCENARIOS OF BLEND STOCK USAGE WHEN 1 MTIHM AS SPENT UO_2 AND MOX
FUEL ARE REPROCESSED TOGETHER^a

	Scenario #1	Scenario #2	Scenario #3	Scenario #4	Scenario #5
Blend Stock ^{b,c}	DU	ERU	RU	NU	RU/ERU ^d
Percent U^{235} in Recovered U From LWR Irradiation of					
UO_2 Fuel	0.83	0.83	0.83	0.83	0.83
1 st Generation MOX Fuel	0.74	0.83	0.74	0.74	0.82
2 nd Generation MOX Fuel	0.69	0.86	0.71	0.71	0.79
3 rd Generation MOX Fuel	0.67	0.85	0.70	0.69	ND ^e
4 th Generation MOX Fuel	0.66	0.83	0.69	0.68	ND ^e
Change From UO_2 Fuel	-20.5%	-	-16.9%	-18.1%	ND ^e
Percent U^{236} in Recovered U From LWR Irradiation of					
UO_2 Fuel	0.78	0.78	0.78	0.78	0.78
1 st Generation MOX Fuel	0.70	1.02	0.76	0.69	1.02
2 nd Generation MOX Fuel	0.67	1.11	0.74	0.67	1.08
3 rd Generation MOX Fuel	0.66	1.15	0.73	0.66	ND ^e
4 th Generation MOX Fuel	0.65	1.15	0.73	0.65	ND ^e
Change From UO_2 Fuel	-16.7%	+47.4%	-6.4%	-16.7%	ND ^e

^a60,000 MWD/MTIHM burnup @ 40 MW/MTIHM, 5 yrs after discharge from reactor

^bRU = Recycled U; NU = Natural U; DU = Depleted U; ERU = Enriched Recycled U

^cAssumes Pu is recovered in a 5:1 mixture of RU:Pu and has to be blended with additional U (blend stock) to lower the fissile content to 5% fissile equivalent in the mixture for recycle to LWRs.

^dRU is used as blend stock with remaining RU being enriched for use in making UO_2 fuel

^eND = Not Determined

TABLE 3-4
PERCENT Pu²³⁹ AND Pu²⁴¹ IN PLUTONIUM RECOVERED FROM REPROCESSING OF
SUCCESSIVE GENERATIONS OF SPENT UO₂ AND MOX FUEL UNDER VARIOUS
SCENARIOS OF BLEND STOCK USAGE WHEN 1 MTIHM AS SPENT UO₂ AND MOX
FUEL ARE REPROCESSED TOGETHER^a

	Scenario #1	Scenario #2	Scenario #3	Scenario #4	Scenario 5
Blend Stock ^{b,c}	DU	ERU	RU	NU	RU/ERU ^d
Percent Pu ²³⁹ in Recovered Pu From LWR Irradiation of					
UO ₂ Fuel	51.14	51.14	51.14	51.14	51.14
1 st Generation MOX Fuel	44.20	42.60	43.89	44.03	43.89
2 nd Generation MOX Fuel	41.34	39.63	41.07	41.23	40.91
3 rd Generation MOX Fuel	39.96	38.32	39.70	39.87	ND ^e
4 th Generation MOX Fuel	39.09	37.60	38.89	39.18	ND ^e
Change From UO ₂ Fuel	-23.6%	-26.5%	-24.0%	-23.4%	ND ^e
Percent Pu ²⁴¹ in Recovered Pu From LWR Irradiation of					
UO ₂ Fuel	12.20	12.20	12.20	12.20	12.20
1 st Generation MOX Fuel	13.03	12.71	12.98	13.05	12.85
2 nd Generation MOX Fuel	13.10	12.54	13.01	13.07	12.78
3 rd Generation MOX Fuel	13.00	12.38	12.88	12.96	ND ^e
4 th Generation MOX Fuel	12.92	12.21	12.77	12.88	ND ^e
Change From UO ₂ Fuel	+5.9%	nil	+4.7%	+5.6%	ND ^e

^a60,000 MWD/MTIHM burnup @ 40 MW/MTIHM, 5 yrs after discharge from reactor

^bRU = Recycled U; NU = Natural U; DU = Depleted U; ERU = Enriched Recycled U

^cAssumes Pu is recovered in a 5:1 mixture of RU:Pu and has to be blended with additional U (blend stock) to lower the fissile content to 5% fissile equivalent in the mixture for recycle to LWRs.

^dRU is used as blend stock with remaining RU being enriched for use in making UO₂ fuel

^eND = Not Determined

TABLE 3-5
WEIGHT OF Np, Am AND Cm IN SUCCESSIVE GENERATIONS OF SPENT UO₂ AND MOX FUEL UNDER VARIOUS SCENARIOS OF BLEND STOCK USAGE WHEN 1 MTIHM AS SPENT UO₂ AND MOX FUEL ARE REPROCESSED TOGETHER^a

	Scenario #1	Scenario #2	Scenario #3	Scenario #4	Scenario #5
Blend Stock ^{b,c}	DU	ERU	RU	NU	RU/ERU ^d
kg/MTIHM of Np in Spent Fuel From LWR Irradiation of					
UO ₂ Fuel	0.942	0.942	0.942	0.942	0.942
1 st Generation MOX Fuel	0.922	1.600	1.060	0.916	1.380
2 nd Generation MOX Fuel	0.910	1.880	1.090	0.910	1.530
3 rd Generation MOX Fuel	0.908	2.020	1.110	0.910	ND ^e
4 th Generation MOX Fuel	0.909	2.070	1.110	0.911	ND ^e
Change From UO ₂ Fuel	-3.5%	+219.7%	+17.8%	-3.3%	ND ^e
kg/MTIHM of Am in Spent Fuel From LWR Irradiation of					
UO ₂ Fuel	0.811	0.811	0.811	0.811	0.811
1 st Generation MOX Fuel	1.494	1.465	1.492	1.493	1.478
2 nd Generation MOX Fuel	1.969	1.924	1.961	1.958	1.934
3 rd Generation MOX Fuel	2.290	2.230	2.280	2.270	ND ^e
4 th Generation MOX Fuel	2.480	2.420	2.480	2.470	ND ^e
Change From UO ₂ Fuel	+305.8%	+298.4%	+305.8%	+304.6%	ND ^e
kg/MTIHM of Cm in Spent Fuel From LWR Irradiation of					
UO ₂ Fuel	0.139	0.139	0.139	0.139	0.139
1 st Generation MOX Fuel	0.381	0.402	0.377	0.396	0.387
2 nd Generation MOX Fuel	0.596	0.601	0.596	0.608	0.590
3 rd Generation MOX Fuel	0.762	0.768	0.578	0.775	ND ^e
4 th Generation MOX Fuel	0.873	0.881	0.868	0.887	ND ^e
Change From UO ₂ Fuel	+628.1%	+633%	+624.5%	+638.1%	ND ^e

^a60,000 MWD/MTIHM burnup @ 40 MW/MTIHM, 5 yrs after discharge from reactor

^bRU = Recycled U; NU = Natural U; DU = Depleted U; ERU = Enriched Recycled U

^cAssumes Pu is recovered in a 5:1 mixture of RU:Pu and has to be blended with additional U (blend stock) to lower the fissile content to 5% fissile equivalent in the mixture for recycle to LWRs.

^dRU is used as blend stock with remaining RU being enriched for use in making UO₂ fuel

^eND = Not Determined

TABLE 3-6
WEIGHT OF MOX FUEL PRODUCED IN SUCCESSIVE RECYCLES OF PU & U
UNDER VARIOUS SCENARIOS OF BLEND STOCK USAGE WHEN 1 MTIHM AS SPENT
UO₂ AND MOX FUEL ARE REPROCESSED TOGETHER

Description	Blend Stock for Producing MOX Fuel ^{a,b}	Weight ^c (kg/MTIHM) of LWR MOX Fuel from Reprocessing & Recycle of			
		UO ₂ Fuel	1 st Generation MOX Fuel	2 nd Generation MOX Fuel	3 rd Generation MOX Fuel
Scenario #1	DU	168	227	256	274
Scenario #2	ERU	244	279	315	334
Scenario #3	RU	177	232	262	280
Scenario #4	NU	178	236	266	284
Scenario #5	RU/ERU ^d	178	240	ND ^e	ND ^e

^aRU = Recycled U; NU = Natural U; DU = Depleted U; ERU = Enriched Recycled U

^bAssumes Pu is recovered in a 5:1 mixture of RU:Pu and has to be blended with additional U (blend stock) to lower the fissile content to 5% fissile equivalent in the mixture for recycle to LWRs.

^cFresh UO₂ fuel needs to be added to the weights shown in the table to produce 1 MTHM for recycle

^dRU is used as blend stock with remaining RU being enriched for use in making UO₂ fuel

^eND = Not Determined

3.4.2 Characteristics When Spent UO₂ and MOX Fuel are Reprocessed and Recycled Separately

A second set of calculations was made assuming the spent UO₂ and MOX fuel were reprocessed and recycled separately. In this case the calculations were made for only Scenarios #1 and #2, and for only 2 generations of LWR recycle. The purpose here was to determine the impact of separately reprocessing the two kinds of fuel on plutonium and uranium quality. Calculations were made only for 2 generations of LWR recycle because the quality of the plutonium recycled dropped below the range that the available cross-section data used by the ORIGEN program is considered to be reliable. The results of these calculations are included in Tables D-1 through D-6 of Appendix D. Some of the key characteristics are summarized in Tables 3-7 through 3-9.

TABLE 3-7
PERCENT U^{235} AND U^{236} IN URANIUM RECOVERED FROM REPROCESSING OF
SUCCESSIVE GENERATIONS OF LWR MOX FUEL WHEN UO_2 AND MOX ARE
REPROCESSED SEPARATELY^{a,b,c}

	Scenario #1	Scenario #2
Blend Stock ^d	DU	ERU
Percent U^{235} in Recovered U from LWR Irradiation of:		
UO_2 Fuel	0.83	0.83
1 st Generation MOX	0.31	0.85
2 nd Generation MOX	0.24	0.87
Change from UO_2 Fuel	-71.1%	+4.8%
Percent U^{236} in Recovered U from LWR Irradiation of:		
UO_2 Fuel	0.78	0.78
1 st Generation MOX	0.32	1.78
2 nd Generation MOX	0.27	2.06
Change from UO_2 Fuel	-65.4%	+164.1%

^aAll MOX is reprocessed together

^b60,000 MWD/MTIHM burnup @ 40 MW/MTIHM, 5 yrs. after discharge from reactor

^cAssumes Pu is recovered in a 5:1 mixture of RU:Pu and has to be blended with additional U (blend stock) to lower the fissile content to 5% fissile equivalent in the mixture for recycle to LWRs.

^dDU = Depleted U; ERU = Enriched Recycled U

TABLE 3-8
PERCENT Pu²³⁹ AND Pu²⁴¹ IN PLUTONIUM RECOVERED FROM REPROCESSING OF
SUCCESSIVE GENERATIONS OF LWR MOX FUEL WHEN UO₂ AND MOX ARE
REPROCESSED SEPARATELY^{a,b,c}

	Scenario #1	Scenario #2
Blend Stock ^d	DU	ERU
Percent Pu ²³⁹ in Recovered Pu from LWR Irradiation of:		
UO ₂ Fuel	51.14	51.14
1 st Generation MOX	35.54	33.40
2 nd Generation MOX	33.85	31.71
Change from UO ₂ Fuel	-33.8%	-38.0%
Percent Pu ²⁴¹ in Recovered Pu from LWR Irradiation of:		
UO ₂ Fuel	12.20	12.20
1 st Generation MOX	14.00	13.29
2 nd Generation MOX	13.83	12.79
Change from UO ₂ Fuel	+13.4%	+4.8%

^aAll MOX is reprocessed together

^b60,000 MWD/MTIHM burnup @ 40 MW/MTIHM, 5 yrs. after discharge from reactor

^cAssumes Pu is recovered in a 5:1 mixture of RU:Pu and has to be blended with additional U (blend stock) to lower the fissile content to 5% fissile equivalent in the mixture for recycle to LWRs.

^dDU = Depleted U; ERU = Enriched Recycled U

TABLE 3-9
WEIGHT OF Np, Am AND Cm IN SUCCESSIVE GENERATIONS OF
SPENT UO₂ AND MOX FUEL WHEN UO₂ AND MOX ARE
REPROCESSED SEPARATELY^{a,b,c}
(Grams Per Metric Ton Initial Heavy Metal)

	Scenario #1	Scenario #2
Blend Stock ^d	DU	ERU
kg/MTIHM of Np in Spent Fuel From Irradiation of:		
UO ₂ Fuel	942	942
1 st Generation MOX	827	3,643
2 nd Generation MOX	720	4,406
Change from UO ₂ Fuel	-23.6%	+468%
kg/MTIHM of Am in Spent Fuel From Irradiation of:		
UO ₂ Fuel	811	811
1 st Generation MOX	4,881	3,496
2 nd Generation MOX	5,969	4,822
Change from UO ₂ Fuel	+736%	+595%
kg/MTIHM of Cm in Spent Fuel From Irradiation of:		
UO ₂ Fuel	139	139
1 st Generation MOX	1,583	1,221
2 nd Generation MOX	2,133	1,757
Change from UO ₂ Fuel	+1,535%	+1,264%

^aAll MOX is reprocessed together

^b60,000 MWD/MTIHM burnup; 40 MW/MTIHM, 5 yrs. after discharge

^cAssumes Pu is recovered in a 5:1 mixture of RU:Pu and has to be blended with additional U (blend stock) to lower the fissile content to 5% fissile equivalent in the mixture for recycle to LWRs.

^dDU = Depleted U; ERU = Enriched Recycled U

The following summarizes the conclusions drawn from the foregoing tables and Appendix D:

- (1) It is likely that multiple recycle of recovered uranium and plutonium would be limited if spent UO_2 and MOX fuels are reprocessed separately and the subsequent U:Pu products are not co-mingled. This conclusion contrasts with the major conclusion of the discussion in Section 3.4.1 wherein spent UO_2 and MOX fuels discharged from LWRs were assumed to be intermingled when they are reprocessed.
 - (a) After two recycles, the fraction of the fissile isotopes of plutonium that would be contained in spent fuel would decrease from greater than 63 percent in spent UO_2 fuel to 45-48 percent in second generation spent MOX fuel. (See Tables D-1 and D-2). If the U^{235} enrichment of uranium blend stock is 0.25 weight percent (U^{235}) and the fissile fraction of the plutonium is less than 47.5 percent, the amount of plutonium needed to provide 5 weight percent fissile content in MOX fuel would exceed a technical limit of 10 weight percent. This could be further exacerbated if additional enrichment was needed to offset neutron absorption by the high fraction of non-fissile plutonium isotopes.
 - (b) Although not insurmountable, increased concentrations of americium and curium isotopes in spent MOX fuel originating from separately reprocessed MOX fuel, would exacerbate technical challenges for transportation of the spent fuel.
 - (c) For the AFC, in which excess uranium recovered by reprocessing is re-enriched for use in the next generation MOX fuel, the concentration of U^{236} in even the first generation MOX is more than twice that of the U^{235} . (See Table D-2). This could further compound the reduced quality of the recycled plutonium for use in the next generation of MOX fuel.
- (2) Successive recycle of MOX fuel that has been segregated from UO_2 spent fuel when reprocessed over 2 generations results in:
 - (a) A reduction in the U^{235} content of the recovered uranium of about 71 percent from that of spent UO_2 fuel, and an increase in the U^{235} content of about 5 percent, for the UREX+3c fuel cycle and AFC, respectively. (See Table 3-7).
 - (b) A reduction in the U^{236} content of the recovered uranium of about 65 percent from that of spent UO_2 fuel, and an increase in the U^{236} content of about 164 percent, for the UREX+3c fuel cycle and AFC, respectively. (See Table 3-7).
 - (c) A reduction in the Pu^{239} content of the recovered plutonium of about 34 percent and 38 percent from that of spent UO_2 fuel, for the UREX+3c fuel cycle and AFC, respectively. (See Table 3-8).
 - (d) An increase in the Pu^{241} content of the recovered plutonium of about 13 percent and 5 percent from that of spent UO_2 fuel, for the UREX+3c fuel cycle and AFC, respectively. (See Table 3-8).

- (e) A reduction in the Np content of the spent MOX fuel by about 24 percent from that of spent UO₂ fuel, and an increase in such content of about 468 percent, for the UREX+3c fuel cycle and AFC, respectively. (See Table 3-9).
 - (f) An increase in the Am content of the spent MOX fuel by 736 percent and 595 percent, and an increase in the Cm content thereof by 1535 percent and 1264 percent, for the UREX+3c fuel cycle and AFC, respectively. (See Table 3-9).
- (3) The concentrations of a majority of the fission products in the spent MOX fuel either decreased or didn't change much for the UREX+3c fuel cycle and the AFC over 2 generations of recycle. However, the ones that did increase, did so by substantially higher percentages than those experienced when UO₂ and MOX fuel were reprocessed together (because of the dilution effect that resulted from coreprocessing). Nevertheless, these increases in concentrations of most of such isotopes were applied to very low concentrations to begin with.
 - (4) In view of the foregoing and the relatively favorable results obtained in Section 3.4.1 when all spent LWR is reprocessed and recycled together, there does not appear to be any advantage, and may be disadvantages, to separate reprocessing of spent LWR UO₂ and MOX fuels when the recovered products thereof are intended for recycle to LWRs. However, separate reprocessing is not disadvantageous where the recovered products are intended for recycle to FRs.

3.5 SPENT FUEL CHARACTERISTICS AFTER USE IN FRs

Calculations were made to determine the changes in the concentrations of uranium and plutonium isotopes, minor actinides, lanthanides, and other fission products that originate from LWR spent fuel resulting from multiple recycling of plutonium and uranium in FRs -- 2 years after discharge from the reactor. These were accomplished using the ORIGEN computer code. The purpose of these calculations was to be able to determine (i) the extent to which fissile material has to be added for effective recycle, (ii) the amounts of fission products and minor actinides that have to be dealt with in waste solidification and repository disposal, (iii) the amounts of minor actinides that might otherwise have to be fabricated into fuel for a FR, and (iv) to be able to assess the radiological safety aspects of the UREX +3c fuel cycle and the AFC. The results of the calculations are included in Tables 3-10 and 3-11, and Appendix E. The following summarizes the conclusions drawn from the foregoing tables and Appendix E.

- (1) For both the UREX+3c and AFC scenarios, where spent MOX fuel from LWRs is used to produce initial fuel for a FR and to replenish material consumed in successive recycles, cycle-to-cycle variations in the relative concentrations of

uranium and plutonium in the FR fuel are small. This is the case whether the FR fuel is produced using once or twice recycled MOX fuel from LWRs. Equilibrium conditions, where the concentrations of uranium and plutonium in recycle FR fuel from one cycle to the next are nearly the same, basically is achieved in 4 cycles.

- (2) For both scenarios, the fraction of minor actinides (Np, Am, and Cm) in spent FR fuel ranges from about 6 percent for spent startup core fuel to about 3.5 percent for spent 4th generation recycle fuel. In large part this change is the result of the FRs effectiveness in consuming Np and Am isotopes. The concentration of these minor actinides in the 4th generation spent FR fuel declines by only about 5 percent for both fuel cycle scenarios and for use of the uranium and plutonium from reprocessed MOX fuel that has been recycled either once or twice in LWRs. This small decline in the 4th generation recycle suggests that the equilibrium concentration of the minor actinides in spent FR fuel is about 3 percent. (See Appendix E)
- (3) Successive recycle of plutonium, minor actinides, and uranium over 4 generations of recycle to FRs results in:
 - (a) Less than a 1 percent increase in the uranium content of the spent FR fuel. Also, there appears to be little difference in the uranium content of the spent fuel when the FR fuel is derived from one or two recycles of MOX fuel in LWRs, or whether it originates from the UREX+3c fuel cycle or the AFC. (See Table 3-10).
 - (b) About a 6 percent increase in the plutonium content of the spent fuel for the UREX+3c fuel cycle where the plutonium originates from a single recycle of MOX in LWRs, and a 9 percent increase when the plutonium originates from two such recycles. Under the same circumstances the AFC runs about 17-20 percent higher. For each generation, the increase in plutonium is greater than that for the UREX+3c fuel cycle. (See Table 3-10). The principal reason for this is that the amounts of fissionable U²³⁵ and Np²³⁷ from LWR spent fuel from the AFC are greater than for the UREX+3c fuel cycle. Fissioning of these isotopes reduces depletion of plutonium in the FR fuel when compared to that for the UREX+3c fuel cycle.
 - (c) About a 50 percent decrease in the Np content of the spent fuel for all cases. However, as discussed in Section 3.4.1, the AFC produces significantly higher amounts of Np than does the UREX+3c fuel cycle, and these increase from generation to generation (59 percent higher, rising to 69 percent, when the initial fuel is made from MOX that has undergone a single recycle in LWRs -- and 72 percent higher, rising to 85 percent, when such fuel results from two recycles of MOX in LWRs). The relative difference in the Np content of the spent FR fuel between a single recycle of MOX fuel in LWRs, and two such recycles, remains relatively constant as the generations proceed, amounting to about 5 percent and 15 percent,

- respectively, for the UREX+3c fuel cycle and the AFC, after 4 generations. (See Table 3-11).
- (d) About a 37 percent decrease in the Am content of the spent fuel, where the initial fuel is made from MOX fuel that has undergone a single recycle in LWRs, and a 43 percent decrease when such fuel results from two recycles of MOX fuel in LWRs. The AFC produces somewhat less Am than does the UREX+3c fuel cycle (6-7 percent less in the case of a single recycle of MOX fuel and 8-10 percent less in the case of two recycles). The differences in the Am content of the spent FR fuel for a single recycle of MOX fuel, and two such recycles, diminish as the generations proceed, amounting to about 10 percent and 7 percent, respectively, for the UREX+3c fuel cycle and the AFC, after 4 generations. (See Table 3-11).
 - (e) A 3-4 percent decrease in the Cm content of the spent fuel, where the initial fuel is made from MOX fuel that has undergone a single recycle in LWRs, and a 14-15 percent decrease when such fuel results from two recycles of MOX fuel in LWRs. The AFC produces somewhat less Cm than does the UREX+3c fuel cycle (1-3 percent less in the case of a single recycle of MOX fuel and 5-7 percent less in the case of two recycles). The differences in the Cm content of the spent FR fuel for a single recycle of MOX fuel, and two such recycles, diminish as the generations proceed, amounting to about 25 percent and 20 percent, respectively, for the UREX+3c fuel cycle and the AFC, after 4 generations. (See Table 3-11).
- (4) The uranium, plutonium, and minor actinide content of spent FR fuel nears equilibrium by the end of the 4th recycle of fuel. While the cycle-to-cycle changes in the minor actinide content diminishes over 4 recycles, there is still a 7 percent, 4 percent, and 2-4 percent decline between the 3rd and 4th recycles for Np, Am, and Cm, respectively.
 - (5) The fission product content of spent reactor fuel doesn't change much over 4 recycles of fuel, and probably won't change much thereafter. No individual elements increased as much as 5 percent, and most were well below 1 percent -- for all cases considered.

TABLE 3-10

TOTAL URANIUM AND PLUTONIUM CONTENT OF SUCCESSIVE GENERATIONS OF SPENT FAST REACTOR FUEL ^a

U	Grams Uranium Per 565 kg Initial Heavy Metal					% Change Over	
	Initial FR Fuel	1 st Generation FR Fuel	2 nd Generation FR Fuel	3 rd Generation FR Fuel	4 th Generation FR Fuel	4 Generations	Last 2 Generations
Single Recycle of MOX in LWRs							
Scenario #1 -- UREX+3c Fuel Cycle	400,587	401,954	402,798	403,122	403,220	+0.7%	+0.02%
Scenario #2 -- AFC	398,904	401,311	402,339	402,738	402,954	+1.0%	+0.05%
Change From Scenario #1	-0.42%	-0.15%	-0.11%	-0.10%	-0.07%		
Two Recycles of MOX in LWRs							
Scenario #1 -- UREX+3c Fuel Cycle	399,295	401,469	402,452	402,891	402,994	+0.9%	+0.03%
Scenario #2 -- AFC	397,612	400,786	402,088	402,557	402,744	+1.3%	+0.29%
Change From Scenario #1	-0.42%	-0.17%	-0.09%	-0.08%	-0.06%		
Change from Single Recycle							
Scenario #1 -- UREX+3c Fuel Cycle	-0.32%	-0.12%	-0.09%	-0.06%	-0.06%		
Scenario #2 -- AFC	-0.32%	-0.13%	-0.06%	-0.04%	-0.06%		
Pu	Grams Plutonium Per 565 kg Initial Heavy Metal					% Change Over	
	Initial FR Fuel	1 st Generation FR Fuel	2 nd Generation FR Fuel	3 rd Generation FR Fuel	4 th Generation FR Fuel	4 Generations	Last 2 Generations
Single Recycle of MOX in LWRs							
Scenario #1 -- UREX+3c Fuel Cycle	84,368	87,089	88,519	89,319	89,806	+6.4%	+0.55%
Scenario #2 -- AFC	82,275	85,332	86,947	87,872	88,432	+7.5%	+0.64%
Change From Scenario #1	-2.5%	-2.0%	-1.7%	-1.6%	-1.5%		
Two Recycles of MOX in LWRs							
Scenario #1 -- UREX+3c Fuel Cycle	80,793	84,416	86,331	87,415	88,076	+9.0%	+0.76%
Scenario #2 -- AFC	78,071	82,272	84,470	85,708	86,464	+10.8%	+0.88%
Change From Scenario #1	-3.4%	-2.5%	-2.2%	-2.0%	-1.8%		
Change from Single Recycle							
Scenario #1 -- UREX+3c Fuel Cycle	-4.2%	-3.1%	-2.5%	-2.1%	-1.9%		
Scenario #2 -- AFC	-5.1%	-3.6%	-2.9%	-2.5%	-2.2%		

^aOne reactor of a 9-unit plant requires 565 kg IHM per refueling

TABLE 3-11
TOTAL NEPTUNIUM, AMERICIUM AND CURIUM CONTENT OF SUCCESSIVE GENERATIONS OF
FAST REACTOR FUEL^a

Np	Grams Neptunium Per 565 kg Initial Heavy Metal					% Change Over	
	Initial FR Fuel	1 st Generation FR Fuel	2 nd Generation FR Fuel	3 rd Generation FR Fuel	4 th Generation FR Fuel	4 Generations	Last 2 Generations
Single Recycle of MOX in LWRs							
Scenario #1 -- UREX+3c Fuel Cycle	6,633	4,792	3,855	3,374	3,123	-52.9%	-7.4%
Scenario #2 -- AFC	10,560	7,765	6,375	5,665	5,284	-50.0%	-6.7%
Change From Scenario #1	+59.2%	+62.0%	+65.4%	+67.9%	+69.2%		
Two Recycles of MOX in LWRs							
Scenario #1 -- UREX+3c Fuel Cycle	7,220	5,067	4,033	3,524	3,266	-54.8%	-7.3%
Scenario #2 -- AFC	12,450	8,936	7,284	6,472	6,049	-51.4%	-6.5%
Change From Scenario #1	+72.4%	+76.4%	+80.6%	+83.7%	+85.2%		
Change from Single Recycle							
Scenario #1 -- UREX+3c Fuel Cycle	+8.8%	+5.7%	+4.6%	+4.4%	+4.6%		
Scenario #2 -- AFC	+17.9%	+15.1%	+14.3%	+14.2%	+14.5%		
Am	Grams Americium Per 565 kg Initial Heavy Metal					% Change Over	
	Initial FR Fuel	1 st Generation FR Fuel	2 nd Generation FR Fuel	3 rd Generation FR Fuel	4 th Generation FR Fuel	4 Generations	Last 2 Generations
Single Recycle of MOX in LWRs							
Scenario #1 -- UREX+3c Fuel Cycle	11,813	9,520	8,306	7,700	7,399	-37.4%	-3.9%
Scenario #2 -- AFC	11,039	8,867	7,755	7,213	6,947	-37.1%	-3.7%
Change From Scenario #1	-6.6%	-6.9%	-6.6%	-6.3%	-6.1%		
Two Recycles of MOX in LWRs							
Scenario #1 -- UREX+3c Fuel Cycle	14,366	10,955	9,316	8,542	8,170	-43.1%	-4.4%
Scenario #2 -- AFC	13,097	9,944	8,466	7,781	7,459	-43.0%	-4.1%
Change From Scenario #1	-8.8%	-9.2%	-9.1%	-8.9%	-8.7%		
Change from Single Recycle							
Scenario #1 -- UREX+3c Fuel Cycle	+21.6%	+15.1%	+12.2%	+10.9%	+10.4%		
Scenario #2 -- AFC	+18.6%	+12.1%	+9.2%	+7.9%	+7.4%		
Cm	Grams Curium Per 565 kg Initial Heavy Metal					% Change Over	
	Initial FR Fuel	1 st Generation FR Fuel	2 nd Generation FR Fuel	3 rd Generation FR Fuel	4 th Generation FR Fuel	4 Generations	Last 2 Generations
Single Recycle of MOX in LWRs							
Scenario #1 -- UREX+3c Fuel Cycle	4,370	4,457	4,413	4,335	4,258	-2.6%	-1.8%
Scenario #2 -- AFC	4,315	4,371	4,314	4,240	4,157	-3.7%	-2.0%
Change From Scenario #1	-1.3%	-1.9%	-2.2%	-2.2%	-2.4%		
Two Recycles of MOX in LWRs							
Scenario #1 -- UREX+3c Fuel Cycle	6,171	5,991	5,730	5,496	5,308	-14.0%	-3.4%
Scenario #2 -- AFC	5,849	5,641	5,380	5,153	4,974	-15.0%	-3.5%
Change From Scenario #1	-5.2%	-5.8%	-6.1%	-6.2%	-6.3%		
Change from Single Recycle							
Scenario #1 -- UREX+3c Fuel Cycle	+41.2%	+34.4%	+29.8%	+26.8%	+24.7%		
Scenario #2 -- AFC	+35.6%	+29.1%	+24.7%	+21.5%	+19.7%		

^aOne reactor of a 9-unit plant requires 565 kg IHM per refueling

3.6 REFERENCES FOR SECTION 3.0

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- (3) C. L. Cockey, Actinide Transmutation in the Advanced Liquid Metal Reactor (ALMR), International Atomic Energy Agency, Proceedings of a Specialists Meeting on the Use of Fast Reactors for Actinide Transmutation, Obninsk, Russian Federation, IAEA-TECDOC-693, September 22-24, 1992

4.0 ANALYSIS OF THE SYSTEMS ASPECT OF REPROCESSING AND RECYCLE

There are two conceptual ways to approach the comparative evaluation of alternative nuclear fuel cycle systems, (i) on an individual reactor basis, or (ii) on the basis of an overall fleet of reactors. These are discussed more fully in the following.

4.1 REACTOR-BY-REACTOR BASIS

In this case it is assumed that the plutonium, and at least part of the uranium recovered from the spent fuel of an individual reactor, is recycled to the reactor from which it originated. This can be accomplished in two different ways -- one of which is to reprocess all spent fuel discharged from the reactor (both UO₂ and MOX fuel) together, and the other is to reprocess spent UO₂ fuel and spent MOX fuel separately.

In either case, if one considers a PWR that is just starting up, it will take a nominal 10.5 years before any recycle fuel can be charged to the reactor assuming:

- (1) One-third of the core is discharged every 18 months (i.e., average residence time in the reactor is 4.5 years)
- (2) The spent fuel is cooled 5 years after discharge before transport to the reprocessing facility can commence
- (3) Transport, reprocessing and fabrication of MOX fuel will be accomplished in the 6th year, and
- (4) Recycle of MOX to the reactor will commence in the 7th year.

Moreover, it will not be until the beginning of the 21st year of operation that the first recycle of MOX fuel will be available for use in the reactor, the beginning of the 31.5th year of operation that the second recycle of MOX fuel will be available for use in the reactor, the beginning of the 42nd year of operation that the third recycle of MOX fuel will be available for use in the reactor, and the beginning of the 52.5th year of operation that the fourth recycle of MOX fuel will be available for use in the reactor. This is illustrated in Figures 4-1 and 4-2.

FIGURE 4-1
ILLUSTRATION OF THE RECYCLE PATTERN FOR PU RECOVERED FROM REPROCESSING AS MOX FUEL IN A SINGLE 1,000 MWE PWR --
NO SEPARATION OF UO₂ FUEL AND MOX FUELS IN SUCCESSIVE RECYCLES

FIGURE 4-2
ILLUSTRATION OF THE RECYCLE PATTERN FOR PU RECOVERED FROM REPROCESSING AS MOX FUEL IN A SINGLE 1,000 MWE PWR --
SEPARATION OF SPENT UO₂ FUEL AND MOX FUEL IN SUCCESSIVE RECYCLES

It would seem that making comparative evaluations of the cost, safeguards, waste management, radiological health, and sustainability aspects of the AFC vs. the UREX+3c fuel cycle for an individual reactor would be an appropriate approach -- because this would represent the conditions under which a utility owner of a LWR would contract for reprocessing and recycle services. However, such an approach fails to recognize that, in an overall system as such exists currently in the U.S., there is a large backlog of spent fuel which is available for recycle and that the existing fleet of LWRs is constantly generating more spent fuel every year. While this raises many questions regarding how such a system recycle would be managed in actual practice, it was decided that, for the purposes of this study, the comparative evaluations of alternative fuel cycles would be made based on a 100 LWR system -- because this considered the overall system, rather than just a single LWR plant.

4.2 EXISTING INSTALLED FLEET OF REACTORS

Four fuel cycle simulations were developed for a 100 LWR fleet operating over 81 years (2020 to 2100) -- two for the AFC, and two for the UREX+3c fuel cycle (one each for a single MOX recycle to LWRs and the other for two such MOX recycles). Key assumptions for the models included:

- (1) 100 1,000 MWe LWR plants operate for the full term of the simulation, or are replaced by new LWRs.
- (2) The composition, initial enrichment, irradiation, and post discharge decay characteristics of all LWR spent fuel of the same type (i.e., UO₂, 1st generation MOX and 2nd generation MOX) are the same as those described in Appendix C of this report. No modification is made to account for the age, initial characteristics, or use of the extant inventories of UO₂ LWR spent fuel.
- (3) All extant inventories of UO₂ spent fuel are available on demand for reprocessing and recycle.
- (4) Reprocessing of spent LWR fuel begins in 2031 at a rate of 3,000 MTIHM per year. The rate is reduced to 1,840 MTIHM per year in the year the simulation brings the consumption of transuranic elements in FRs into balance with their production in LWRs.
- (5) In fuel cycle simulations that do not use spent 1st generation MOX fuel to produce 2nd generation MOX fuel for LWRs, FRs are installed at a rate of 1 new reactor per year beginning in 2043, the year after 1st generation MOX spent fuel is first

available as a source of uranium and transuranium elements for FR fuel. No new reactors are installed in years following the year the simulation brings the consumption of transuranium elements in FRs into balance with their production in LWRs.

- (6) In fuel cycle simulations that use spent 1st generation MOX fuel to produce 2nd generation MOX fuel in LWRs, FRs are installed at a rate of 1 new reactor every 2 years beginning in 2043, the year after the 1st generation MOX fuel is first available as a source of uranium and transuranium elements for FR fuel. Beginning in 2053 and thereafter, 2nd generation spent MOX fuel is used as the source of uranium and transuranium elements for FR fuel. No new reactors are installed following the year the simulation brings the consumption of transuranic elements in FRs into balance with their production in LWRs.
- (7) FR spent fuel is reprocessed at the same rate that the fuel was discharged from the reactors the year before.
- (8) Recycled FR fuel is produced the same year its parent FR and LWR MOX spent fuels are reprocessed and is loaded into a fast reactor at the beginning of the next year.
- (9) The isotopic compositions of the uranium and transuranium elements in startup FR fuel are those of the source of the spent fuel from LWRs. The composition of fuel material used to replenish depleted uranium and transuranium elements when recycling FR spent fuel is also that of the source LWR spent fuel.
- (10) The composition of all recycle FR fuel is that of 4th recycle.

Detailed results of the simulations are presented in Tables 4-1 through 4-4. An explanation of the information presented in Tables 4-1 through 4-4 follows the tables. Table 4-5 presents an overview comparison of the simulations of the UREX+3c fuel cycle and the AFC for 1 and 2 recycles of plutonium in LWRs.

Initial observations and conclusions that can be drawn from the results of the simulations include the following:

- (1) Reprocessing LWR spent fuel and recycling of recovered plutonium in LWR reactors will be essential elements of any advanced fuel cycle wherein long-half-life actinide elements produced in the LWRs are consumed in fast burner reactors.
- (2) Reprocessing (on an industrial scale) LWR spent fuel beginning in 2031 (earlier if possible) and recycling of recovered plutonium in LWR reactors would establish a foundation for introducing fast burner reactors in the first half of the century.

- (3) Although reprocessing would be required, if plutonium is not recycled in LWRs (not simulated), about 100 FRs would be needed to consume actinides at the rate produced in 100 LWRs. If the plutonium is recycled one time only, about 43 FRs would be required. If two generations of recycle fuel are used in the LWRs, only 23 FRs would be needed to consume the actinides at the net rate of production in 100 LWRs.
- (4) Based on reprocessing at a rate of 3,000 MTIHM per year beginning in 2031 and steady introduction of new FRs beginning in 2043, equilibrium states for the fuel cycles could be achieved before the end of the century (2086 - 2090) and the net inventory of LWR spent nuclear fuel could be reduced to 30,000 - 36,000 MTIHM or about half of that in 2010.
- (5) Recycle of plutonium in LWRs one time would reduce the consumption of virgin uranium fuel by between 20 percent for the UREX+3c fuel cycle and 27 percent for the AFC, which also reenriches and recycles recovered uranium.
- (6) Recycle of plutonium in LWRs two times would reduce the consumption of virgin uranium fuel by between 24 percent for the UREX+3c fuel cycle and 32 percent for the AFC.
- (7) Regardless of the simulation, for fuel cycle scenarios that deploy FRs, the system-wide inventory of transuranic elements would stabilize in the middle of the century at about half of what would exist if 100 LWRs continued to operate without recycle until the end of the century.
- (8) The net electric power that would be produced from 2020 through 2100 for (i) 100 LWRs with no recycle, (ii) UREX+3c or AFC with 1 recycle of plutonium in LWRs, and (iii) UREX+3c fuel cycle or AFC with 2 recycles of plutonium in LWRs would be as shown below:

Fuel Cycle Scenario	Electric Power Production 2020-2100 (GWE^a years)	Electric Power Production in 2100 (GWE^a)
100 LWRs with no recycle	8,000	100
UREX+3c or AFC with 1 Pu recycle in LWRs	10,260	162 ^b
UREX+3c or AFC with 2 Pu recycles in LWRs	9,160	133 ^c

^aGigawatts Electric

^bLong term sustainable equilibrium with 100 LWRs and 43 fast reactors

^cLong term sustainable equilibrium with 100 LWRs and 23 fast reactors

TABLE 4-1
UREX+3c FUEL CYCLE
81-YEAR SIMULATION OF 100 LWR SYSTEM MASS FLOW SPREADSHEET FOR ONE RECYCLE OF MOX IN LWRs^a

(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)	(18)	(19)	(20)	(21)	(22)	(23)
	Fuel Loaded Into LWRs (MTIHM)			Fuel Discharged From LWRs (MTIHM)			LWR Fuel Reprocessed to Produce (MTIHM)				LWR Spent Fuel Inventories (MTIHM)			Fast Reactors		Fast Reactor Fuel Produced (MTIHM)			System-Wide TRU Inventory (MTIHM)			
Year	UO2	MOX-1	MOX-2	UO2	MOX-1	MOX-2	MOX-1	MOX-2	FR InitCor	Recycle	UO2 Inv	MOX-1 Inv	MOX-2 Inv	New	Operating	First Core	Recycle	Reprocessed	LWR Fuel	FR Fuel	Process Wastes	System-Wide
2020	1840	-	-	-	-	-	-	-	-	-	80080	-	-	-	-	-	-	-	1217	-	-	1217
2021	1840	-	-	1840	-	-	-	-	-	-	81920	-	-	-	-	-	-	-	1244	-	-	1244
2022	1840	-	-	1840	-	-	-	-	-	-	83760	-	-	-	-	-	-	-	1270	-	-	1270
2023	1840	-	-	1840	-	-	-	-	-	-	85600	-	-	-	-	-	-	-	1297	-	-	1297
2024	1840	-	-	1840	-	-	-	-	-	-	87440	-	-	-	-	-	-	-	1324	-	-	1324
2025	1840	-	-	1840	-	-	-	-	-	-	89280	-	-	-	-	-	-	-	1350	-	-	1350
2026	1840	-	-	1840	-	-	-	-	-	-	91120	-	-	-	-	-	-	-	1377	-	-	1377
2027	1840	-	-	1840	-	-	-	-	-	-	92960	-	-	-	-	-	-	-	1403	-	-	1403
2028	1840	-	-	1840	-	-	-	-	-	-	94800	-	-	-	-	-	-	-	1430	-	-	1430
2029	1840	-	-	1840	-	-	-	-	-	-	96640	-	-	-	-	-	-	-	1456	-	-	1456
2030	1840	-	-	1840	-	-	-	-	-	-	98480	-	-	-	-	-	-	-	1483	-	-	1483
2031	1840	-	-	1840	-	-	3000	-	-	-	97320	-	-	-	-	-	-	-	1466	-	-	1467
2032	1334	506	-	1840	-	-	3000	-	-	-	96160	-	-	-	-	-	-	-	1583	-	1	1583
2033	1334	506	-	1840	-	-	3000	-	-	-	95000	-	-	-	-	-	-	-	1566	-	1	1567
2034	1334	506	-	1840	-	-	3000	-	-	-	93840	-	-	-	-	-	-	-	1549	-	2	1551
2035	1334	506	-	1840	-	-	3000	-	-	-	92680	-	-	-	-	-	-	-	1532	-	2	1534
2036	1334	506	-	1840	-	-	3000	-	-	-	91520	-	-	-	-	-	-	-	1516	-	3	1518
2037	1334	506	-	1334	506	-	3000	-	-	-	89854	506	-	-	-	-	-	-	1520	-	3	1523
2038	1334	506	-	1334	506	-	3000	-	-	-	88188	1012	-	-	-	-	-	-	1525	-	3	1529
2039	1334	506	-	1334	506	-	3000	-	-	-	86523	1517	-	-	-	-	-	-	1530	-	4	1534
2040	1334	506	-	1334	506	-	3000	-	-	-	84857	2023	-	-	-	-	-	-	1535	-	4	1539
2041	1334	506	-	1334	506	-	3000	-	-	-	83191	2529	-	-	-	-	-	-	1540	-	5	1545
2042	1334	506	-	1334	506	-	2809	-	191	-	81716	2844	-	-	-	61	-	-	1537	-	5	1542
2043	1366	474	-	1334	506	-	2761	-	239	-	80290	3110	-	1	1	76	-	-	1523	18	6	1547
2044	1375	465	-	1334	506	-	2713	-	287	-	78911	3329	-	1	2	92	-	-	1513	36	7	1556
2045	1383	457	-	1334	506	-	2707	-	287	6	77538	3542	-	1	3	92	15	15	1503	54	8	1565
2046	1384	456	-	1334	506	-	2701	-	287	12	76171	3749	-	1	4	92	31	31	1495	72	8	1575
2047	1385	455	-	1334	506	-	2695	-	287	18	74811	3949	-	1	5	92	46	46	1487	90	9	1586
2048	1386	454	-	1366	474	-	2689	-	287	24	73489	4111	-	1	6	92	61	61	1477	108	10	1594
2049	1387	453	-	1375	465	-	2683	-	287	30	72181	4259	-	1	7	92	76	76	1466	126	11	1603
2050	1388	452	-	1383	457	-	2676	-	287	37	70887	4393	-	1	8	92	92	92	1455	144	12	1610
2051	1389	451	-	1384	456	-	2670	-	287	43	69600	4520	-	1	9	92	107	107	1443	162	13	1618
2052	1390	450	-	1385	455	-	2664	-	287	49	68320	4640	-	1	10	92	122	122	1431	179	14	1625
2053	1391	449	-	1386	454	-	2658	-	287	55	67048	4752	-	1	11	92	137	137	1419	197	15	1631
2054	1392	448	-	1387	453	-	2652	-	287	61	65782	4858	-	1	12	92	153	153	1406	215	16	1638
2055	1393	447	-	1388	452	-	2646	-	287	67	64524	4956	-	1	13	92	168	168	1393	233	17	1644
2056	1394	446	-	1389	451	-	2640	-	287	73	63273	5047	-	1	14	92	183	183	1380	251	19	1650
2057	1395	445	-	1390	450	-	2634	-	287	79	62029	5131	-	1	15	92	198	198	1367	269	20	1656
2058	1396	444	-	1391	449	-	2628	-	287	85	60792	5208	-	1	16	92	214	214	1353	287	21	1661
2059	1397	443	-	1392	448	-	2622	-	287	91	59562	5278	-	1	17	92	229	229	1339	305	22	1666
2060	1398	442	-	1393	447	-	2615	-	287	98	58340	5340	-	1	18	92	244	244	1325	323	24	1671
2061	1399	441	-	1394	446	-	2609	-	287	104	57124	5396	-	1	19	92	259	259	1310	341	25	1676
2062	1400	440	-	1395	445	-	2603	-	287	110	55916	5444	-	1	20	92	275	275	1295	359	26	1680
2063	1401	439	-	1396	444	-	2597	-	287	116	54715	5485	-	1	21	92	290	290	1280	377	28	1684
2064	1402	438	-	1397	443	-	2591	-	287	122	53520	5520	-	1	22	92	305	305	1264	395	29	1688
2065	1403	437	-	1398	442	-	2585	-	287	128	52333	5547	-	1	23	92	320	320	1248	413	31	1692
2066	1404	436	-	1399	441	-	2579	-	287	134	51154	5566	-	1	24	92	336	336	1232	431	32	1695
2067	1405	435	-	1400	440	-	2573	-	287	140	49981	5579	-	1	25	92	351	351	1216	449	34	1698
2068	1406	434	-	1401	439	-	2567	-	287	146	48815	5585	-	1	26	92	366	366	1199	467	35	1701
2069	1407	433	-	1402	438	-	2561	-	287	152	47657	5583	-	1	27	92	381	381	1182	485	37	1703
2070	1408	432	-	1403	437	-	2555	-	287	159	46505	5575	-	1	28							

TABLE 4-2
ALTERNATIVE FUEL CYCLE
81-YEAR SIMULATION OF 100 LWR SYSTEM MASS FLOW SPREADSHEET FOR ONE RECYCLE OF MOX IN LWRs^a

(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)	(18)	(19)	(20)	(21)	(22)	(23)	
	Fuel Loaded Into LWRs (MTIHM)			Fuel Discharged From LWRs (MTIHM)			LWR Fuel Reprocessed to Produce (MTIHM)				LWR Spent Fuel Inventories (MTIHM)			Fast Reactors		Fast Reactor Fuel Produced (MTIHM)			System-Wide TRU Inventory (MTIHM)				
Year	UO2	MOX-1	MOX-2	UO2	MOX-1	MOX-2	MOX-1	MOX-2	FR InitCor	Recycle	UO2 Inv	MOX-1 Inv	MOX-2 Inv	New	Operating	First Core	Recycle	Reprocessed	LWR Fuel	FR Fuel	Process Wastes	System-Wide	
2020	1840	-	-	-	-	-	-	-	-	-	80080	-	-	-	-	-	-	-	-	1217	-	-	1217
2021	1840	-	-	1840	-	-	-	-	-	-	81920	-	-	-	-	-	-	-	-	1244	-	-	1244
2022	1840	-	-	1840	-	-	-	-	-	-	83760	-	-	-	-	-	-	-	-	1270	-	-	1270
2023	1840	-	-	1840	-	-	-	-	-	-	85600	-	-	-	-	-	-	-	-	1297	-	-	1297
2024	1840	-	-	1840	-	-	-	-	-	-	87440	-	-	-	-	-	-	-	-	1324	-	-	1324
2025	1840	-	-	1840	-	-	-	-	-	-	89280	-	-	-	-	-	-	-	-	1350	-	-	1350
2026	1840	-	-	1840	-	-	-	-	-	-	91120	-	-	-	-	-	-	-	-	1377	-	-	1377
2027	1840	-	-	1840	-	-	-	-	-	-	92960	-	-	-	-	-	-	-	-	1403	-	-	1403
2028	1840	-	-	1840	-	-	-	-	-	-	94800	-	-	-	-	-	-	-	-	1430	-	-	1430
2029	1840	-	-	1840	-	-	-	-	-	-	96640	-	-	-	-	-	-	-	-	1456	-	-	1456
2030	1840	-	-	1840	-	-	-	-	-	-	98480	-	-	-	-	-	-	-	-	1483	-	-	1483
2031	1840	-	-	1840	-	-	3000	-	-	-	97320	-	-	-	-	-	-	-	-	1466	-	-	1467
2032	1105	735	-	1840	-	-	3000	-	-	-	96160	-	-	-	-	-	-	-	-	1581	-	1	1582
2033	1105	735	-	1840	-	-	3000	-	-	-	95000	-	-	-	-	-	-	-	-	1564	-	1	1566
2034	1105	735	-	1840	-	-	3000	-	-	-	93840	-	-	-	-	-	-	-	-	1548	-	2	1549
2035	1105	735	-	1840	-	-	3000	-	-	-	92680	-	-	-	-	-	-	-	-	1531	-	2	1533
2036	1105	735	-	1840	-	-	3000	-	-	-	91520	-	-	-	-	-	-	-	-	1514	-	3	1517
2037	1105	735	-	1105	735	-	3000	-	-	-	89625	735	-	-	-	-	-	-	-	1519	-	3	1522
2038	1105	735	-	1105	735	-	3000	-	-	-	87730	1470	-	-	-	-	-	-	-	1524	-	3	1527
2039	1105	735	-	1105	735	-	3000	-	-	-	85835	2205	-	-	-	-	-	-	-	1529	-	4	1533
2040	1105	735	-	1105	735	-	3000	-	-	-	83940	2940	-	-	-	-	-	-	-	1534	-	4	1538
2041	1105	735	-	1105	735	-	3000	-	-	-	82045	3675	-	-	-	-	-	-	-	1539	-	5	1543
2042	1105	735	-	1105	735	-	2746	-	254	-	80404	4156	-	-	-	61	-	-	-	1536	-	5	1541
2043	1167	673	-	1105	735	-	2682	-	318	-	78827	4573	-	1	1	76	-	0	1521	18	6	1544	
2044	1183	657	-	1105	735	-	2619	-	381	-	77313	4927	-	1	2	92	-	15	1511	36	6	1554	
2045	1198	642	-	1105	735	-	2611	-	381	8	75808	5272	-	1	3	92	15	31	1502	54	7	1563	
2046	1200	640	-	1105	735	-	2603	-	381	16	74310	5610	-	1	4	92	31	46	1495	72	8	1574	
2047	1202	638	-	1105	735	-	2595	-	381	24	72820	5940	-	1	5	92	46	61	1487	90	8	1586	
2048	1204	636	-	1167	673	-	2587	-	381	32	71401	6199	-	1	6	92	61	76	1478	108	9	1595	
2049	1206	634	-	1183	657	-	2579	-	381	40	70005	6435	-	1	7	92	76	92	1468	126	10	1603	
2050	1208	632	-	1198	642	-	2571	-	381	48	68633	6647	-	1	8	92	92	107	1457	144	11	1611	
2051	1210	630	-	1200	640	-	2563	-	381	56	67271	6849	-	1	9	92	107	122	1446	162	11	1619	
2052	1212	628	-	1202	638	-	2555	-	381	64	65919	7041	-	1	10	92	122	137	1434	179	12	1626	
2053	1214	626	-	1204	636	-	2547	-	381	72	64576	7224	-	1	11	92	137	153	1423	197	13	1633	
2054	1216	624	-	1206	634	-	2539	-	381	80	63244	7396	-	1	12	92	153	168	1411	215	14	1640	
2055	1218	622	-	1208	632	-	2531	-	381	88	61921	7559	-	1	13	92	168	183	1398	233	15	1647	
2056	1220	620	-	1210	630	-	2523	-	381	96	60609	7711	-	1	14	92	183	198	1386	251	16	1653	
2057	1222	618	-	1212	628	-	2515	-	381	104	59307	7853	-	1	15	92	198	214	1373	269	17	1659	
2058	1224	616	-	1214	626	-	2507	-	381	112	58014	7986	-	1	16	92	214	229	1360	287	18	1665	
2059	1226	614	-	1216	624	-	2499	-	381	120	56731	8109	-	1	17	92	229	244	1346	305	19	1670	
2060	1228	612	-	1218	622	-	2491	-	381	128	55459	8221	-	1	18	92	244	259	1332	323	20	1676	
2061	1230	610	-	1220	620	-	2483	-	381	136	54196	8324	-	1	19	92	259	275	1318	341	21	1681	
2062	1232	608	-	1222	618	-	2475	-	381	144	52944	8416	-	1	20	92	275	290	1304	359	23	1685	
2063	1234	606	-	1224	616	-	2467	-	381	152	51701	8499	-	1	21	92	290	305	1289	377	24	1690	
2064	1236	604	-	1226	614	-	2459	-	381	160	50468	8572	-	1	22	92	305	320	1274	395	25	1694	
2065	1238	602	-	1228	612	-	2451	-	381	168	49245	8635	-	1	23	92	320	336	1259	413	26	1698	
2066	1240	600	-	1230	610	-	2443	-	381	176	48032	8688	-	1	24	92	336	351	1243	431	28	1702	
2067	1242	598	-	1232	608	-	2435	-	381	184	46830	8730	-	1	25	92	351	366	1227	449	29	1705	
2068	1244	596	-	1234	606	-	2427	-	381	192	45637	8763	-	1	26	92	366	381	1211	467	30	1708	
2069	1245	595	-	1236	604	-	2419	-	381														

TABLE 4-3
UREX+3c FUEL CYCLE
81-YEAR SIMULATION OF 100 LWR SYSTEM MASS FLOW SPREADSHEET FOR TWO RECYCLES OF MOX IN LWRs^a

(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)	(18)	(19)	(20)	(21)	(22)	(23)
	Fuel Loaded Into LWRs (MTIHM)			Fuel Discharged From LWRs (MTIHM)			LWR Fuel Reprocessed to Produce (MTIHM)				LWR Spent Fuel Inventories (MTIHM)			Fast Reactors		Fast Reactor Fuel Produced (MTIHM)			System-Wide TRU Inventory (MTIHM)			
Year	UO2	MOX-1	MOX-2	UO2	MOX-1	MOX-2	MOX-1	MOX-2	FR InitCor	Recycle	UO2 Inv	MOX-1 Inv	MOX-2 Inv	New	Operating	First Core	Recycle	Reprocessed	LWR Fuel	FR Fuel	Process Wastes	System-Wide
2020	1840	-	-	-	-	-	-	-	-	-	80080	-	-	-	-	-	-	-	1217	-	0	1217
2021	1840	-	-	1840	-	-	-	-	-	-	81920	-	-	-	-	-	-	-	1244	-	0	1244
2022	1840	-	-	1840	-	-	-	-	-	-	83760	-	-	-	-	-	-	-	1270	-	0	1270
2023	1840	-	-	1840	-	-	-	-	-	-	85600	-	-	-	-	-	-	-	1297	-	0	1297
2024	1840	-	-	1840	-	-	-	-	-	-	87440	-	-	-	-	-	-	-	1324	-	0	1324
2025	1840	-	-	1840	-	-	-	-	-	-	89280	-	-	-	-	-	-	-	1350	-	0	1350
2026	1840	-	-	1840	-	-	-	-	-	-	91120	-	-	-	-	-	-	-	1377	-	0	1377
2027	1840	-	-	1840	-	-	-	-	-	-	92960	-	-	-	-	-	-	-	1403	-	0	1403
2028	1840	-	-	1840	-	-	-	-	-	-	94800	-	-	-	-	-	-	-	1430	-	0	1430
2029	1840	-	-	1840	-	-	-	-	-	-	96640	-	-	-	-	-	-	-	1456	-	0	1456
2030	1840	-	-	1840	-	-	-	-	-	-	98480	-	-	-	-	-	-	-	1483	-	0	1483
2031	1840	-	-	1840	-	-	3000	-	-	-	97320	-	-	-	-	-	-	-	1466	-	0	1467
2032	1335	506	-	1840	-	-	3000	-	-	-	96160	-	-	-	-	-	-	-	1583	-	1	1583
2033	1335	506	-	1840	-	-	3000	-	-	-	95000	-	-	-	-	-	-	-	1566	-	1	1567
2034	1335	506	-	1840	-	-	3000	-	-	-	93840	-	-	-	-	-	-	-	1549	-	2	1551
2035	1335	506	-	1840	-	-	3000	-	-	-	92680	-	-	-	-	-	-	-	1532	-	2	1534
2036	1335	506	-	1840	506	-	3000	-	-	-	91520	-	-	-	-	-	-	-	1515	-	3	1518
2037	1335	506	-	1335	506	-	3000	-	-	-	89855	506	-	-	-	-	-	-	1520	-	3	1523
2038	1335	506	-	1335	506	-	3000	-	-	-	88189	1011	-	-	-	-	-	-	1525	-	3	1529
2039	1335	506	-	1335	506	-	3000	-	-	-	86524	1517	-	-	-	-	-	-	1530	-	4	1534
2040	1335	506	-	1335	506	-	3000	-	-	-	84858	2022	-	-	-	-	-	-	1535	-	4	1539
2041	1335	506	-	1335	506	-	3000	-	-	-	83193	2528	-	-	-	-	-	-	1540	-	5	1544
2042	1335	506	-	1335	506	-	804	2100	96	-	81976	2583	-	-	-	31	-	-	1525	6	5	1537
2043	1229	136	476	1335	506	-	780	2100	120	-	80784	2615	-	0.5	0.5	38	-	-	1555	17	6	1579
2044	1233	132	476	1335	506	-	757	2100	143	-	79616	2623	-	0.5	1	46	-	-	1538	28	7	1572
2045	1237	127	476	1335	506	-	754	2100	143	2	78450	2629	-	0.5	1.5	46	8	8	1520	37	7	1565
2046	1237	127	476	1335	506	-	752	2100	143	5	77287	2632	-	0.5	2	46	15	15	1504	46	8	1558
2047	1237	127	476	1335	136	-	749	2100	143	7	76126	2633	-	0.5	2.5	46	23	23	1487	55	9	1551
2048	1238	126	476	1229	132	476	747	2100	143	10	74861	2261	476	0.5	3	46	31	31	1478	64	10	1552
2049	1238	126	476	1233	127	476	745	2100	143	12	73603	1883	952	0.5	3.5	46	38	38	1470	73	10	1553
2050	1239	125	476	1237	127	476	742	2100	143	14	72351	1499	1428	0.5	4	46	46	46	1461	82	11	1554
2051	1239	125	476	1237	127	476	740	2100	143	17	71102	1112	1903	0.5	4.5	46	53	53	1451	91	12	1555
2052	1239	125	476	1237	126	476	738	2100	143	19	69856	722	2379	0.5	5	46	61	61	1442	101	13	1555
2053	1240	124	476	1238	126	476	2302	563	114	21	68324	618	2720	0.5	5.5	46	69	69	1436	110	13	1559
2054	1325	388	128	1238	125	476	2316	546	114	24	66792	652	3058	0.5	6	46	76	76	1403	119	14	1536
2055	1326	390	124	1239	125	476	2330	530	114	26	65260	688	3394	0.5	6.5	46	84	84	1404	128	15	1547
2056	1327	393	120	1239	125	476	2330	528	114	29	63731	724	3727	0.5	7	46	92	92	1406	137	15	1558
2057	1328	393	120	1239	124	476	2329	526	114	31	62204	760	4058	0.5	7.5	46	99	99	1407	146	16	1570
2058	1328	392	119	1240	388	476	2328	525	114	33	60679	796	4387	0.5	8	46	107	107	1408	155	17	1581
2059	1329	392	119	1325	390	128	2328	523	114	36	59241	1096	4365	0.5	8.5	46	114	114	1403	165	18	1585
2060	1329	392	118	1326	393	124	2327	521	114	38	57807	1398	4337	0.5	9	46	122	122	1398	174	19	1590
2061	1330	392	118	1327	393	120	2326	520	114	40	56376	1703	4303	0.5	9.5	46	130	130	1392	183	19	1594
2062	1330	392	118	1328	392	120	2325	518	114	43	54948	2008	4266	0.5	10	46	137	137	1386	192	20	1599
2063	1331	392	117	1328	392	119	2325	516	114	45	53522	2314	4226	0.5	10.5	46	145	145	1381	201	21	1603
2064	1331	392	117	1329	392	119	1227	1611	114	48	52284	2434	4183	0.5	11	46	153	153	1367	210	22	1599
2065	1268	207	365	1329	392	118	1215	1621	114	50	51050	2553	4138	0.5	11.5	46	160	160	1378	219	23	1621
2066	1268	205	367	1330	392	118	1203	1631	114	52	49821	2670	4090	0.5	12	46	168	168	1364	228	24	1617
2067	1268	203	370	1330	392	118	1201	1631	114	55	48595	2787	4039	0.5	12.5	46	175	175	1350	238	25	1613
2068	1268	202	370																			

TABLE 4-4
ALTERNATIVE FUEL CYCLE
81-YEAR SIMULATION OF 100 LWR SYSTEM MASS FLOW SPREADSHEET FOR TWO RECYCLES OF MOX IN LWRs^a

(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)	(18)	(19)	(20)	(21)	(22)	(23)
	Fuel Loaded Into LWRs (MTIHM)			Fuel Discharged From LWRs (MTIHM)			LWR Fuel Reprocessed to Produce (MTIHM)				LWR Spent Fuel Inventories (MTIHM)			Fast Reactors		Fast Reactor Fuel Produced (MTIHM)			System-Wide TRU Inventory (MTIHM)			
Year	UO2	MOX-1	MOX-2	UO2	MOX-1	MOX-2	MOX-1	MOX-2	FR InitCor	Recycle	UO2 Inv	MOX-1 Inv	MOX-2 Inv	New	Operating	First Core	Recycle	Reprocessed	LWR Fuel	FR Fuel	Process Wastes	System-Wide
2020	1840	-	-	-	-	-	-	-	-	-	80080	-	-	-	-	-	-	-	1217	-	-	1217
2021	1840	-	-	1840	-	-	-	-	-	-	81920	-	-	-	-	-	-	-	1244	-	-	1244
2022	1840	-	-	1840	-	-	-	-	-	-	83760	-	-	-	-	-	-	-	1270	-	-	1270
2023	1840	-	-	1840	-	-	-	-	-	-	85600	-	-	-	-	-	-	-	1297	-	-	1297
2024	1840	-	-	1840	-	-	-	-	-	-	87440	-	-	-	-	-	-	-	1324	-	-	1324
2025	1840	-	-	1840	-	-	-	-	-	-	89280	-	-	-	-	-	-	-	1350	-	-	1350
2026	1840	-	-	1840	-	-	-	-	-	-	91120	-	-	-	-	-	-	-	1377	-	-	1377
2027	1840	-	-	1840	-	-	-	-	-	-	92960	-	-	-	-	-	-	-	1403	-	-	1403
2028	1840	-	-	1840	-	-	-	-	-	-	94800	-	-	-	-	-	-	-	1430	-	-	1430
2029	1840	-	-	1840	-	-	-	-	-	-	96640	-	-	-	-	-	-	-	1456	-	-	1456
2030	1840	-	-	1840	-	-	-	-	-	-	98480	-	-	-	-	-	-	-	1483	-	-	1483
2031	1840	0	-	1840	-	-	3000	-	-	-	97320	-	-	-	-	-	-	-	1466	-	-	1467
2032	1105	735	-	1840	-	-	3000	-	-	-	96160	-	-	-	-	-	-	-	1583	-	1	1584
2033	1105	735	-	1840	-	-	3000	-	-	-	95000	-	-	-	-	-	-	-	1566	-	1	1568
2034	1105	735	-	1840	-	-	3000	-	-	-	93840	-	-	-	-	-	-	-	1550	-	2	1551
2035	1105	735	-	1840	-	-	3000	-	-	-	92680	-	-	-	-	-	-	-	1533	-	2	1535
2036	1105	735	-	1840	-	-	3000	-	-	-	91520	-	-	-	-	-	-	-	1516	-	3	1519
2037	1105	735	-	1105	735	-	3000	-	-	-	89625	735	-	-	-	-	-	-	1521	-	3	1524
2038	1105	735	-	1105	735	-	3000	-	-	-	87730	1470	-	-	-	-	-	-	1526	-	3	1530
2039	1105	735	-	1105	735	-	3000	-	-	-	85835	2205	-	-	-	-	-	-	1531	-	4	1535
2040	1105	735	-	1105	735	-	3000	-	-	-	83940	2940	-	-	-	-	-	-	1536	-	4	1541
2041	1105	735	-	1105	735	-	3000	-	-	-	82045	3675	-	-	-	0	-	-	1541	-	5	1546
2042	1105	735	-	1105	735	-	773	2100	127	-	80792	3769	-	-	-	31	-	-	1528	6	5	1539
2043	1065	189	586	1105	735	-	741	2100	159	-	79570	3831	-	0.5	0.5	38	-	-	1556	17	6	1579
2044	1073	182	586	1105	735	-	709	2100	191	-	78380	3861	-	0.5	1	46	-	-	1539	28	7	1573
2045	1080	174	586	1105	735	-	707	2100	191	3	77193	3889	-	0.5	1.5	46	8	8	1522	37	7	1566
2046	1081	173	586	1105	735	-	704	2100	191	5	76008	3914	-	0.5	2	46	15	15	1506	46	8	1560
2047	1082	172	586	1105	735	-	701	2100	191	8	74827	3936	-	0.5	2.5	46	23	23	1489	55	9	1553
2048	1082	172	586	1065	189	586	699	2100	191	11	73607	3410	586	0.5	3	46	31	31	1482	64	9	1556
2049	1083	171	586	1073	182	586	696	2100	191	13	72398	2874	1172	0.5	3.5	46	38	38	1475	73	10	1558
2050	1084	171	586	1080	174	586	694	2100	191	16	71199	2327	1758	0.5	4	46	46	46	1467	82	11	1561
2051	1084	170	586	1081	173	586	691	2100	191	18	70004	1777	2344	0.5	4.5	46	53	53	1460	91	12	1563
2052	1085	169	586	1082	172	586	688	2100	191	21	68812	1223	2930	0.5	5	46	61	61	1452	101	13	1565
2053	1085	169	586	1082	172	586	2308	541	127	24	67178	1263	3364	0.5	5.5	46	69	69	1455	110	13	1578
2054	1124	565	151	1083	171	586	2327	519	127	26	65541	1307	3796	0.5	6	46	76	76	1424	119	14	1557
2055	1125	570	145	1084	171	586	2347	497	127	29	63903	1356	4226	0.5	6.5	46	84	84	1427	128	15	1570
2056	1126	575	139	1084	170	586	2346	495	127	32	62267	1405	4653	0.5	7	46	92	92	1430	137	15	1582
2057	1127	575	138	1085	169	586	2345	493	127	34	60635	1454	5077	0.5	7.5	46	99	99	1433	146	16	1595
2058	1128	575	138	1085	169	586	2345	491	127	37	59005	1502	5498	0.5	8	46	107	107	1435	155	17	1608
2059	1129	574	137	1124	565	151	2344	489	127	40	57415	1948	5482	0.5	8.5	46	114	114	1431	165	18	1613
2060	1129	574	136	1125	570	145	2343	487	127	42	55829	2399	5457	0.5	9	46	122	122	1426	174	18	1619
2061	1130	574	136	1126	575	139	2342	485	127	45	54247	2855	5424	0.5	9.5	46	130	130	1422	183	19	1624
2062	1131	574	135	1127	575	138	2341	484	127	47	52668	3311	5387	0.5	10	46	137	137	1417	192	20	1629
2063	1131	574	135	1128	575	138	2341	482	127	50	51091	3768	5347	0.5	10.5	46	145	145	1412	201	21	1634
2064	1132	573	134	1129	574	137	1204	1615	127	53	49796	3947	5304	0.5	11	46	153	153	1398	210	22	1630
2065	1094	295	451	1129	574	136	1188	1629	127	55	48507	4122	5257	0.5	11.5	46	160	160	1411	219	23	1653
2066	1094	291	455	1130	574	136	1172	1643	127	58	47225	4294	5208	0.5	12	46	168	168	1398	228	24	1650
2067	1095	287	458	1131	574	135	1170	1642	127	61	45946	4466	5155	0.5	12.5	46	175	175	1384	238	25	1646
2068	1095	287	458</																			

EXPLANATION OF TABLES 4-1 THROUGH 4-4

- (A) Rows: A row in the table represents a calendar year in the system mass flow simulation. The simulation begins in the year 2021 and proceeds in 1-year steps through the year 2100. Initial conditions for the simulation are listed in the row for calendar year 2020.
- (B) Columns:
- (1) Column 1: Calendar year in the simulation
 - (2) Columns 2 through 4: Fuel Loaded Into LWRs in the specified calendar year (MTIHM). The amounts are totals for one-hundred, 1,000 MWe plants.
 - (a) Column 2: UO₂ fuel. The amount of UO₂ fuel loaded into LWRs. Equals 1,840 MTIHM less the amounts of MOX-1 (Column 3) and MOX-2 (Column 4) loaded in the same year.
 - (b) Column 3: MOX-1 (1st Generation MOX fuel). The number of metric tons of 1st generation MOX fuel loaded into LWRs in a specified year. Derived from the amount of UO₂ reprocessed to produce MOX-1 the year before (Column 8).
 - (c) Column 4: MOX-2 (2nd Generation MOX fuel). The number of metric tons of 2nd generation MOX fuel loaded into LWRs in a specified year. Derived from the amount of UO₂ and MOX-1 co-reprocessed to produce MOX-2 the year before (Column 9).
 - (3) Columns 5 through 7: Fuel Discharged From LWRs in the specified calendar year (MTIHM). The amounts are totals for one-hundred, 1,000 MWe plants.
 - (a) Column 5: UO₂ fuel discharged. The number of metric tons of UO₂ fuel discharged from LWRs in a specified year is set equal to the amount of UO₂ fuel loaded into LWRs 5 years before (Column 2).
 - (b) Column 6: MOX-1 fuel discharged. The number of metric tons of 1st generation MOX fuel discharged from LWRs in a specified year is set equal to the amount of MOX-1 fuel loaded into LWRs 5 years before (Column 3).
 - (c) Column 7: MOX-2 fuel discharged. The number of metric tons of 2nd generation MOX fuel discharged from LWRs in a specified year is set equal to the amount of MOX-2 fuel loaded into LWRs 5 years before (Column 4).
 - (4) Columns 8 through 11: LWR Fuel Reprocessed in the specified calendar year (MTIHM). The amounts are totals for a reprocessing annual throughput capacity of 3,000 MTIHM of LWR spent nuclear fuel. For the year 2086 and all following years, when (in the simulation) production of TRU in LWRs would be equal to consumption of TRU in fast reactors, the simulation reduces the throughput capacity of reprocessing facilities to a sustaining level of 1,840 MTIHM.
 - (a) Column 8: Spent UO₂ fuel reprocessed to provide products used to produce 1st generation MOX LWR fuel.
 - (b) Column 9: **(Only if 2nd generation MOX fuel is used in LWRs)** The amount of 1st generation MOX and UO₂ spent fuel co-reprocessed to produce 2nd generation MOX fuel. Derived from the amount of 1st generation MOX spent fuel discharged from LWRs 5 years before.

- (c) Column 10: Spent 1st or 2nd generation MOX fuel reprocessed to produce initial core fuel for new fast reactors.
 - (d) Column 11: Spent 1st or 2nd generation MOX fuel reprocessed to produce make-up fuel for fast reactors.
- (5) Columns 12, 13, and 14: LWR Spent Fuel Inventories (MTIHM). The amounts in each of these columns increase in direct proportion to discharges of spent fuel from LWR reactors and decrease in like proportion to the amount of each type of spent fuel that is reprocessed in an LWR spent fuel reprocessing plant. The inventory of LWR spent fuel in 2020, 80,080 MTIHM, is based on the reported inventory of 58,000 MTIHM in 2008 with an annual increase of 1,840 MTIHM in each year from 2009 through 2020, inclusive.
- (a) Column 12: UO₂ Inventory. The total inventory of UO₂ spent nuclear fuel in the specified year. This inventory is the sum of the prior year's UO₂ inventory and the amount of UO₂ spent fuel discharged in the current year (Column 5) less the amount of UO₂ spent fuel reprocessed or co-reprocessed in the current year.
 - (b) Column 13: MOX-1 Inventory. The total inventory of 1st generation MOX spent nuclear fuel in the specified year. This inventory is the sum of the prior year's inventory and the amount of 1st generation MOX spent fuel discharged in the current year (Column 6) less reprocessing of 1st generation MOX spent fuel to produce 2nd generation MOX LWR fuel or to provide U and TRU for fast reactor fuel in the current year.
 - (c) Column 14: MOX-2 Inventory. The total inventory of 2nd generation MOX spent nuclear fuel in the specified year. This inventory is the sum of the prior year's inventory and the amount of 2nd generation MOX spent fuel discharged in the current year (Column 7) less reprocessing of 2nd generation MOX spent fuel to provide U and TRU for fast reactor fuel (Columns 10 and 11).
- (6) Columns 15 and 16: Fast Reactors. These two columns respectively list the number of new fast reactors introduced into the simulation in the specified year and the total number of fast reactors in operation in the year.
- (a) Column 15: New Fast Reactors. Specifies the number of fast reactors the simulation assumes will begin operation in the specified year. The number of fast reactors placed into operation, and the years of initial operation, are input to the simulation with the objectives to achieve reasonable balances of material flows and to arrive at sustained steady-states for material flows and inventories. (The addition of 0.5 of a reactor every year in Tables 4-3 and 4-4 is used to smooth out the effects of actually adding one fast reactor every 2 years.)
 - (b) Column 16: Operating Fast Reactors. Specifies the number of fast reactors in operation in the specified year. The simulation assumes that once a fast reactor is placed into operation it would continue to operate over the 80-year term of the simulation, or otherwise be replaced by a new fast reactor.
- (7) Columns 17 through 19: Fast Reactor Fuel Produced and Reprocessed. These three columns list the amounts of fast reactor fuel produced and recycled in the specified year to satisfy requirements for new fast reactor cores and for recycle of fuel for operating fast reactors.
- (a) Column 17: First Cores. For the simulation, the core of a 1,440 MWe fast reactor is assumed to contain 61 MTIHM. In addition, the simulation assumes a fast reactor will require fuel for one-quarter of a new reactor core (15.25 MTIHM) in each of the two

years following the startup year. The simulation assumes that fuel for a fast reactor would be produced the year before it is loaded into the fast reactor.

- (b) Column 18: Recycle Fuel. For the simulation, one-quarter (1/4) of the core of a 1,440 MWe fast reactor (15.25 MTIHM) is assumed to be refueled each year. The simulation assumes that recycle fast reactor fuel would be produced from fuel discharged 2 years before and reprocessed and refabricated the year before it was loaded into the reactor.
 - (c) Column 19: Reprocessed Fuel. For the simulation, one-quarter (1/4) of the core of a 1,440 MWe fast reactor (15.25 MTIHM) is assumed to be discharged each year. The simulation assumes that reprocessing of spent fast reactor fuel would occur one year after discharge from the reactor.
- (8) Columns 20 through 23: System TRU Inventory. The first three of these four columns list inventories for each year of the simulation of the three major fuel-cycle locations of TRU -- LWR fuel (in-reactors and spent), fast reactors, and process wastes. The fourth column lists the total system-wide inventory of TRU for each year of the simulation.

TABLE 4-5
COMPARISON OF KEY RESULTS FROM 100 LWR SYSTEM FLOWSHEETS^a

	Scenario #1 (UREX+3c Fuel Cycle)	Scenario #2 (AFC)	Scenario #1 (UREX+3c Fuel Cycle)	Scenario #2 (AFC)
Makeup U in MOX Fuel for LWRs	DU Blend	Enriched RU	DU Blend	Enriched RU
No. Recycles of MOX in LWRs	1 Recycle	1 Recycle	2 Recycles	2 Recycles
No. LWR Reactors	100	100	100	100
UO ₂ Fuel (MTU)	120,461	109,187	113,340	101,228
LWR Fuel Reprocessing (MTIHM/yr)	3,000 2031 thru 2085 1,840 thereafter	3,000 2031 thru 2085 1,840 thereafter	3,000 2031 thru 2089 1,840 thereafter	3,000 2031 thru 2089 1,840 thereafter
No. Fast Reactors	0 in 2042 1 new per year 2043 thru 2085 43 Total 2085 thru 2100	0 in 2042 1 new per year 2043 thru 2085 43 Total 2085 thru 2100	0 in 2042 1 new per 2 yrs 2043 thru 2088 23 Total 2088 thru 2100	0 in 2042 1 new per 2 yrs 2043 thru 2088 23 Total 2088 thru 2100
Fast Reactor Fuel Reprocessing (MTIHM/yr)	0 in 2044 increasing to 656 in 2087 656 2087 thru 2100	0 in 2044 increasing to 656 in 2087 656 2087 thru 2100	0 in 2044 increasing to 351 in 2090 351 2090 thru 2100	0 in 2044 increasing to 351 in 2090 351 2090 thru 2100
LWR Spent Fuel Inventories				
<u>Initial in 2020</u>				
UO ₂	80,080	80,080	80,080	80,080
MOX-1	0	0	0	0
MOX-2	0	0	0	0
Total	80,080	80,080	80,080	80,080
<u>Peak</u>				
UO ₂	98,480	98,480	98,480	98,480
MOX-1	5,732	9,580	4,614	7,078
MOX-2	0	0	7,880	10,459
Total	104,212	108,060	110,974	116,017
<u>In 2100</u>				
UO ₂	28,948	25,100	18,020	13,412
MOX-1	5,732	9,580	3,998	6,183
MOX-2	0	0	7,880	10,459
Total	34,680	34,680	29,898	30,054
Total Electricity Output of System (GWE-yr)	10,261	10,261	9,160	9,160
MWyr/MTU over 80 yrs	85	94	81	90
In 2100	103	110	89	96
(No recycle = 54)				
System-wide TRU Inventory (MTHM)				
In 2020	1,217	1,217	1,217	1,217
In 2100	1,741	1,765	1,648	1,715

^aSee Tables 4-1 through 4-4.

5.0 COMPARATIVE COSTS OF THE UREX+3c FUEL CYCLE AND THE AFC

5.1 INTRODUCTION

A comparative evaluation was made of the total fuel cycle costs estimated to be incurred by 100 LWRs and the FRs needed to consume TRUs from the LWRs, over the period 2020 through 2100, by both the UREX+3c fuel cycle and the AFC, and using both one and two recycles of MOX. These fuel cycle costs included the following elements:

- (1) Virgin UO_2 Fuel (including the mining and milling, conversion, enrichment, and fuel fabrication components)
- (2) DU Conversion (conversion of depleted UF_6 from enrichment of virgin uranium to U_3O_8 for storage and/or disposal)
- (3) Reprocessing (including (i) the separation of uranium, uranium/plutonium mixtures, and technetium, (ii) partitioning of Cs, Sr, Ba, Rb, minor actinides, lanthanide fission products, and all other fission products, (iii) waste vitrification/solidification, and (iv) conversion of excess recovered uranium.)
- (4) MOX Fuel Fabrication (assuming that fabrication would be performed in hot cells, principally for safeguards purposes)
- (5) RU Enrichment (applicable to AFC cases only)
- (6) Fabrication of Initial FR Cores (assuming that initial cores would be fabricated in a MOX type facility modified for metallic fuel)
- (7) Reprocessing and Refabrication of FR Recycle Fuel (assuming that recycle FR fuel would be refabricated in facilities that could accommodate higher levels of radiation than MOX fabrication facilities)
- (8) Disposal of excess recovered uranium and depleted recycled uranium
- (9) LLW Disposal (covers cost of LLW disposal that is not otherwise included in the cost estimated for other fuel cycle operations, and does not include LLW from reactor operations)
- (10) TRU Disposal (covers all fuel cycle TRU waste disposal)
- (11) HLW Disposal (covers disposal of solidified fission product waste and compacted hulls, end fittings, dissolver sludge and technetium)

The foregoing costs did not include the cost of transportation of materials between processing locations, but such is not expected to exceed one percent of the total fuel cycle costs.

5.2 COST BASES USED FOR THE STUDY

5.2.1 General

The basic approach that was followed to calculate the comparative costs of the AFC and UREXC+3c fuel cycles was to:

- (1) Use market prices to the extent available
- (2) Perform estimates for all or part of the costs for areas of the fuel cycle that were particularly meaningful to the comparisons
- (3) Establish estimated costs for activities based on their similarity to activities for which costs were known
- (4) Otherwise utilize cost estimates set forth in INL/EXT-07-12107, Rev. 1 (Ref. 1)
- (5) Reduce all costs to unit costs, so they could be applied to the appropriate material flows set forth in Tables 4-1 through 4-4 and Tables 8-2 through 8-5 of this report.

The following sections describe the costs that were developed or otherwise used.

5.2.2 Enriched Uranium Costs

Market prices reported by U_xC Consulting were \$46/lb U₃O₈ (5/4/09) for yellowcake, \$8.50/kgU (4/27/09) for conversion to UF₆, and \$163/kg SWU (4/27/09) for enrichment. (Ref. 2) Tails assay was assumed to be 0.25 percent. Allowing for 0.5 percent loss in both conversion and fuel fabrication, this brings the cost of 5 percent enriched uranium to \$2,645/kgU. In addition, the sensitivity of this cost to increases in the prices for yellowcake was determined for \$125/lb and \$200/lb U₃O₈ -- which resulted in an increase in the cost of 5 percent enriched uranium to \$4,766/kgU and \$6,831/kgU, respectively.

5.2.3 UO₂ Fuel Fabrication Costs

The following prices for UO₂ fuel fabrication costs were reported in INL/EXT-07-12107, Rev. 1. (Ref. 1)

TABLE 5-1
PRICES FOR UO₂ FUEL FABRICATION

Source	Year	Reference Value		Esc @ 3% Per Year ^a
Energy Resources Int'l.	2007	\$207	(PWR)	\$220
		\$276	(BWR)	293
NECDB	1988	220	(High BU)	409
OECD	1994	275		428
James & Williams	1999	180	(PWR)	242
Bunn	2003	250		299
MIT	2003	275		328
Delene & Williams	2000	270		352
				\$2,571
Average				\$321

^aCPI and ENR City Cost Index increased by 3% between 1988 and 2009

For the purposes of this study, the two oldest prices were dropped, resulting in an average of \$289/kgU for the remaining prices. This was rounded upward to \$300/kgU. This, together with the costs for 5 percent enriched uranium as set forth in Section 5.2.2, produced the following unit costs for finished UO₂ fuel assemblies, as a function of varying prices for yellowcake:

Yellowcake Price (\$/lb U ₃ O ₈)	Total Cost for Finished Fuel Assy (\$/kgU)
\$46	\$2,945
125	5,066
200	7,131

5.2.4 Reprocessing

The costs for reprocessing of UO₂ and MOX fuels were developed using cost information from the 1,500 MTU/year capacity Allied-General Nuclear Services (AGNS) reprocessing plant project at Barnwell, SC as a base, and modifying these costs to include the addition and elimination of facilities as required for the UREX+3c fuel cycle and the AFC. The cost of reprocessing spent UO₂ and MOX fuel was assumed to be the same.

5.2.4.1 Reprocessing Capital Costs

In 1981, AGNS estimated that the capital expenditures for the Barnwell plant had totaled \$216,817,000 (excluding interest during construction and preoperational expenses). (See Table 5-2.) Escalating this from the year incurred to 2009 using the Engineering News Record (ENR) Construction Cost Index produced an estimated 2009 cost of \$944-million. (See Table 5-3.)

Also in 1981, AGNS estimated that it would cost an additional \$721.4-million (in early 1981 dollars) to complete the plant in a manner required by the then-applicable regulations. Escalating these costs to 2009 using the ENR Construction Cost Index (Ref. 3) from 1980 produced an estimated additional 2009 cost of \$1,918-million (see Table 5-4) -- for a total of \$2,862-million.

Table 5-5 shows the combined costs of the AGNS facilities plus a 30 percent contingency to cover the more rigorous regulatory regime that is in place in 2009 compared to that of the 1970-1980 period, and adjustments to the AGNS costs to adapt its plant to UREX+3c and AFC systems. These adjustments included:

- (1) Facilities for the conversion to UF_6 of uranium recovered from spent LWR fuel were included for the AFC case only.
- (2) Facilities for the conversion of plutonium to PuO_2 were excluded from both the UREX+3c fuel cycle and the AFC since these fuel cycles produced a 5:1 mixture of U:Pu and the cost of its conversion to MOX is included separately under MOX fuel fabrication.
- (3) Facilities for the separation of (i) strontium, cesium, barium, and rubidium, (ii) minor actinides, (iii) lanthanides, and (iv) other fission products -- and the alloying and compacting of dissolver hulls, sludge, and technetium were included for both the UREX+3c and AFC cases. The basic AGNS separations process included voloxidation and tritium recovery as well as spent fuel dissolution and separation of the hulls and fines from the dissolver solution. It was reasoned that these steps, coupled with the recovery of technetium and the purification steps necessary for the recovered U and U:Pu mixture, represent about 75 percent of the cost of the overall separations activity. Since the input flows to TRUEX and FPEX were assumed to be about the same as the basic UREX separation process, it was assumed that these facilities would each cost about 25 percent of the cost of the basic separations facilities. Since TALSPEAK would likely involve an input stream with lesser flow, it was assumed that these facilities would cost about 15 percent of the cost of the basic separations facilities. With regard to alloying and compacting facilities, much of this was provided for by AGNS in its estimated for "waste management facilities". Thus, it was assumed that any additional facilities

that were required would cost about 15 percent of the AGNS waste management facilities (which included liquid high level waste tanks, solid waste management, and volume reduction).

- (4) Facilities for the conversion of uranium to U_3O_8 were included for both the UREX+3c and AFC cases. These were estimated to cost about half of what AGNS had estimated for UF_6 conversion facilities.

All of the foregoing would bring the total cost of the 1,500 MTU/year reprocessing facility to \$3,910-million for the UREX+3c and \$4,042-million for the AFC case.

Table 5-6 shows the estimated capital spending pattern and schedule of preoperational expenses over a 10-year design, license and construction period. Preoperational expenses for a reprocessing plant to support the UREX+3c and AFC cases were estimated by JAI at \$315-million and \$326-million, respectively. Interest during construction on capital expenditures and on preoperational expenses were estimated at \$1,046-million and \$1,080-million, respectively (at 5%/year interest). This brought the total capital cost of a 1,500 MTU/year reprocessing facility upgraded to meet current regulatory requirements, and to include additional separations processes as aforementioned, to a total cost of \$5,271-million and \$5,448-million, respectively, for the UREX+3c and AFC cases.

TABLE 5-2
INITIAL EXPENDITURES FOR CONSTRUCTION OF THE AGNS BARNWELL, SC
REPROCESSING PLANT^a
(\$000)

	Separations	Conversion of U to UF ₆	CIP ^b ;Waste & Pu Conversion	Total	Preoperating Expenses	
					Net of Tax Benefits	Interest Net of Tax Benefits
Capital Expenditures ^c						
1970	\$583			\$583	\$1,384	\$60
1971	11,176	\$112		11,288	1,441	162
1972	30,536	671		31,207	1,470	272
1973	45,582	1,386	\$752	47,720	1,832	429
1974	51,927	7,843	8,610	68,380	2,870	744
1975	26,833	10,756	9,289	46,878	5,547	1,205
1976	2,727	621	2,656	6,004	7,156	1,747
1977	1,572	80	103	1,755	7,766	2,407
1978	3,069	(74)	7	3,002	466	3,187
Total	\$174,005	\$21,395	\$21,417	\$216,817	\$29,932	\$10,213
Interest on Capital Expenditures ^d						
1970	25			25		
1971	465	4		469		
1972	1,981	32		2,013		
1973	5,080	113	28	5,221		
1974	10,646	547	445	11,638		
1975	15,376	1,442	1,298	18,116		
1976	16,770	1,931	1,809	20,510		
1977	17,573	2,032	1,987	21,592		
1978	20,865	2,389	2,341	25,595		
Total	\$88,781	\$8,490	\$7,908	\$105,179		
Total Capital Expenditures & Interest Thereon				\$321,996		
Preoperating Expenses Net of Tax Benefits				\$29,932		
Preoperating Expense Interest on Net of Tax Benefit				\$10,213		
Grand Total				\$362,141		

^aLetter, Brian Forrow (Allied Chemical Corp.) to Edgar Tharpe (International Research and Appraisal Co.), March 23, 1981(Ref. 4)

^bConstruction was in progress through 1978

^cIncludes land, land improvements, property plant and equipment, before book depreciation, plus spare parts and supplies, in separations & UF₆ columns.

^dAverage yearly AA industrial bond rate: 1970, 8.65%; 1971, 7.51%; 1972, 7.20%; 1973, 7.52%; 1974, 8.77%; 1975, 8.97%; 1976, 8.32%; 1977, 7.97%; 1978, 8.68%

TABLE 5-3
AGNS PLANT COST -- INITIAL CAPITAL EXPENDITURES ADJUSTED TO 2009 DOLLARS
 (\$000)

	ENR Cost Index ^a	Separations		Conversion of U to UF ₆		Pu Conversion		Waste Management		Total Esc to 2009
		Annual	Esc to 2009	Annual	Esc to 2009	Annual	Esc to 2009	Annual	Esc to 2009	
1970	1,381	\$583	\$3,631	\$0	\$0	\$0	\$0	\$0	\$0	\$3,631
1971	1,581	11,176	60,793	112	609	0	0	0	0	61,402
1972	1,753	30,536	149,806	671	3,292	0	0	0	0	153,098
1973	1,895	45,582	206,863	1,386	6,290	376	1,706	376	1,706	216,565
1974	2,020	51,927	221,075	7,843	33,391	4,305	18,328	4,305	18,328	291,122
1975	2,212	26,833	104,324	10,756	41,818	4,645	18,059	4,644	18,055	182,256
1976	2,401	2,727	9,768	621	2,224	1,328	4,757	1,328	4,757	21,506
1977	2,576	1,572	5,248	80	267	51	170	52	174	5,859
1978	2,776	3,069	9,508	(74)	(229)	4	12	3	9	9,300
Totals		174,005	\$771,016	21,395	\$87,662	10,709	\$43,032	10,708	\$43,029	\$944,739

^aENR Construction Cost Index for 2009 was 8,600.

TABLE 5-4
ESTIMATED ADDITIONAL EXPENDITURES FOR COMPLETION OF THE AGNS
BARNWELL, SC REPROCESSING PLANT^a
(\$-Millions)

	Amount in Early 1981 Dollars	Amount Escalated to 2009^b	Estimated Time to Construct
High Level Waste Solidification	\$150	\$399	6.5 yrs
Pu Conversion Facility	150	399	5.5 yrs
Densified Spent Fuel Storage	31.4	83	3 yrs
Additional Separations/UF ₆ Facilities	50	133	4.5 yrs
Safeguards/Site/Utilities/Support Facilities	115	306	4.5 yrs
Kr Recovery and Storage	90	239	5 yrs
On-Site Waste Management (solid waste storage, volume reduction, storage)	135	359	3.5 yrs ^c
Total	\$721.4	\$1,918	

^aLetter J.J. Jernigan (AGNS) to J. A. McBride (JAI Corp.), April 3, 1981 (Ref. 5)

^bUsing ENR Construction Cost Index from 1980 to 2009

$$\left(\frac{8,600}{3,237}\right) = 2,657$$

^cAverage for the facilities involved

TABLE 5-5
TOTAL ESTIMATED CAPITAL COSTS FOR A 1,500 MTU/YR REPROCESSING
FACILITY WHICH INCLUDES THE SEPARATIONS PROCESSES NECESSARY TO
IMPLEMENT THE UREX+3c FUEL CYCLE AND AFC^a

Facilities	\$-Millions ^b		
	AGNS Costs & Estimates Esc. To 2009	Contingency (30%)	Total 2009 Cost
AGNS Facilities			
Densification of Spent Fuel Storage Facilities	\$83	25	108
Separations Facilities	889	267	1,156
Conversion of U to UF ₆ Facilities (AFC Only)	0(102)	0(30)	0(132)
Conversion of Pu to PuO ₂ Facilities ^c	0	0	0
Waste Management Facilities	402	121	523
High Level Waste Solidification Facilities ^d	399	120	519
Krypton Recovery and Storage Facilities	239	72	311
Safeguards/Site/Utilities/ Support Facilities	306	92	398
Subtotals	\$2,318(2,420)	\$697 (\$727)	\$3,015 (\$3,147)
Additional Facilities Required			
TRUEX Process Facilities (25% of Separations)			289
FPEX Process Facilities (25% of Separations)			289
TALSPEAK Process Facilities (15% of Separations)			173
Alloying Facilities (15% of Waste Management)			78
Conversion of U to U ₃ O ₈ Facilities (UREX & AFC)			66
Totals			\$3,910(\$4,042)

^aExcludes MOX fabrication, FR fuel fabrication and reprocessing, and repository disposal of wastes

^bFigures apply to both the UREX+3c and AFC cases except where the values for the AFC case are different and shown in parentheses

^cConversion of Pu to PuO₂ not applicable for UREX+3c or AFC case. Instead, a 5:1 mixture of U:Pu is converted into MOX in the MOX fuel fabrication process

^dThis should cover conversion of Sr, Cs, Ba, Rb to immobilized form, plus the vitrification of lanthanides and other remaining fission products

TABLE 5-6
CAPITAL EXPENDITURES SCHEDULE, PREOPERATIONAL EXPENSES AND
INTEREST THEREON FOR A 1,500 MTU/YR REPROCESSING PLANT
 (\$-Millions)

Year	Capital Expenditure Pattern		Preoperational Expenses		Cumulative		Interest During Construction and Preop @ 5%/yr	
	UREX+3c System	AFC System	UREX+3c System	AFC System	UREX+3c System	AFC System	UREX+3c System	AFC System
1	\$125	\$125	\$10	\$10	\$135	\$135	\$7	\$7
2	250	267	15	16	400	418	20	21
3	250	260	20	21	670	699	34	35
4	450	450	25	26	1,145	1,175	57	59
5	500	525	25	26	1,670	1,726	84	86
6	600	625	25	26	2,295	2,377	115	119
7	600	625	25	26	2,920	3,028	146	151
8	500	525	30	31	3,450	3,584	173	179
9	450	450	70	72	3,970	4,106	199	205
10	185	190	70	72	4,225	4,368	211	218
Totals	\$3,910	\$4,042	\$315	\$326	\$20,880	\$21,616	\$1,046	\$1,080

5.2.4.2 Reprocessing Operating Costs

Operating expenses were estimated for the AGNS reprocessing plant as modified to include the additional necessary facilities described in Table 5-5. Two separate sets of estimates were prepared -- one for the UREX+3c fuel cycle and one for the AFC. The principal difference between the two was that the reprocessing plant for the AFC included facilities for conversion of recovered uranium to UF₆, conversion of depleted recycled uranium to U₃O₈, and conversion of excess recovered uranium to U₃O₈, where that for the UREX+3c fuel cycle only included the latter. The following describes how the operating costs were developed.

- (1) **Salaries and Wages** -- The numbers of personnel in operations and support activities were estimated for the various operations to proceed efficiently with a minimum of downtime. Salaries were estimated based on the assumption that somewhat higher-than-normal qualifications were desirable and thus, salaries were generally higher than normally expected. Following is a breakdown of personnel requirements and correspondingly total salaries and wages:

TABLE 5-7
SUMMARY OF ANNUAL COSTS FOR REPROCESSING SALARY & WAGES^a
 (\$-Millions, 2009)

Activity	UREX+3c		AFC	
	No.	Amount	No.	Amount
General Management	12	\$2	12	\$2
Operations	607	34	705	40
Maintenance	262	15	289	16
Technical	330	19	340	20
Administration	324	16	325	16
Total	1,535	\$86	1,671	\$94

^a1,500 MTIHM/year capacity plant

- (2) Benefits -- Benefits were taken at 40 percent of direct salaries and wages, and include time off (vacations, holidays, sick days, etc.), FICA taxes, medical and hospitalization insurance, disability insurance, life insurance, pension plan contributions, etc.
- (3) Chemicals -- Used Nuclear Fuel Services, Inc. West Valley plant cost estimates (Ref. 6) escalated to 2009 dollars and scaled upward to include higher AGNS plant capacity and required additional to AGNS facility to be able to service the UREX+3c fuel cycle and the AFC.
- (4) Electricity -- Used AGNS report (NMS-10, 9/77) on level of electric service required and adjusted for additional needed facilities. (Ref. 7) Applied DOE Energy Information Administration average industrial rates for Virginia. (Ref. 8)
- (5) Operating Supplies -- Estimated operating supplies based on the assumption that they would amount to \$100/day for operators and foremen; \$125/day for maintenance, foremen and instrument technicians; and \$25/day for all others. Following is a summary of these costs:

TABLE 5-8
SUMMARY OF ANNUAL COSTS FOR REPROCESSING OPERATING SUPPLIES^a
 (\$-Millions, 2009)

Activity	UREX+3c		AFC	
	No.	Amount	No.	Amount
Operators & Foremen	519	\$12	608	\$14
Maint Mech & Foremen & Instrument Techs	186	5	212	6
All Others	830	5	851	5
Total	1,535	\$22	1,671	\$25

^a1,500 MTIHM/year capacity plant

- (6) Maintenance Spare Parts and Equipment -- Based on Nuclear Fuel Services' estimates that 8.5 percent of the value of the capital cost of installed equipment (excluding preoperational expenses and interest during construction) would be required to provide spare parts and equipment for maintaining a reprocessing facility. (Ref. 6) Assuming that the cost of equipment was 35 percent of the capital cost and that the NFS figure of 8.5 percent is appropriate, the spare parts and equipment costs would be:

$$\begin{array}{ll} \text{UREX+3c} & (\$3,910)(0.35)(0.085) = \$116\text{-million/year} \\ \text{AFC} & (\$4,042)(0.35)(0.085) = \$120\text{-million/year} \end{array}$$

- (7) Property and Liability Insurance -- Nuclear insurers were contacted for input on property and liability insurance costs. Based on these discussions and the paucity of risk experience with reprocessing plants, coupled with a prospective limitation on the ceiling of coverage, it was concluded that a \$3-million/year premium for property insurance and an \$8-million/year premium for liability insurance would be appropriate.
- (8) Property Taxes -- Based on county taxes in Calvert County, Maryland for utility assets of \$2.23/\$100 valuation, where the Calvert Cliffs Nuclear Power Station is located. (Ref. 9) This was rounded downward to \$2/\$100 valuation because Calvert County's tax rates are generally higher than tax rates in other rural areas of the U.S. Thus, the property taxes were estimated to be:

$$\begin{array}{ll} \text{UREX+3c} & (\$3,910)(\$2/100 \text{ val}) = \$78\text{-million} \\ \text{AFC} & (\$4,042)(\$2/100 \text{ val}) = \$81\text{-million} \end{array}$$

The following table summarizes the annual operating expenses for a 1,500 MTIHM reprocessing plant.

TABLE 5-9
SUMMARY OF ANNUAL OPERATING EXPENSES FOR A REPROCESSING PLANT^a
(\$-Millions, 2009)

Description	UREX+3c	AFC
Salaries & Wages	\$86	\$94
Benefits (40%)	34	38
Chemicals	17	18
Electricity	21	21
Supplies	22	25
Maint Parts & Equipment	116	120
Insurance	11	11
Property Taxes	78	81
Other ^b	12	13
Total	\$397	\$421

^a1,500 MTIHM/year capacity plant

^b30% contingency on chemicals & supplies

5.2.4.3 Reprocessing Unit Costs

The unit costs for reprocessing include the following:

- (1) Operating expenses (as set forth in Section 5.2.4.2)
- (2) Depreciation (on a 30-year straight line basis)
- (3) Return on equity investment of 20 percent/year (after taxes)
- (4) Corporate income taxes of 42 percent of annual return on equity
- (5) Interest on debt of 7 percent/year.

In order to calculate the latter three, it was assumed that the reprocessing plant would be privately owned by a consortium of utility companies and would be financed by a mixture of debt (55 percent) and equity (45 percent).

In order to determine the uniform after-tax return (R_a) which is required to be realized annually for an activity to produce a predetermined percentage return on the outstanding unrecovered equity investment (i) over the depreciable life of the facility (n), with the full amount of the equity investment (c) being recovered through straight-line depreciation during such period, the following formula applies:

$$\frac{c}{n} \left(\frac{1}{1+i} + \frac{1}{(1+i)^2} + \cdots + \frac{1}{(1+i)^n} \right) + R_a \left(\frac{1}{1+i} + \frac{1}{(1+i)^2} + \cdots + \frac{1}{(1+i)^n} \right) = c$$

This basically states that the sum of the discounted cash flow from operations (at the stipulated rate of return) must equal the initial investment.

The expression shown in parentheses are geometric progressions which can be summed using the following formula:

$$Sum = a + ar + ar^2 + \cdots + ar^{n-1} = a \left(\frac{1-r^n}{1-r} \right)$$

Where a is the first term of the series, and r is the common factor between the terms in the series $\left(\frac{1}{1+i} \right)$.

Subtracting a and adding ar^n to the formula, then substituting 1 in the formula for a and $\frac{1}{1+i}$ for r , and reducing the result to its simplest form gives the following:

$$Sum = \left(\frac{1 - \frac{1}{(1+i)^n}}{i} \right)$$

The sum set forth by the foregoing is the same as the formula used for determination of the present worth of a dollar payable periodically.

Substituting *Sum* for the geometric progressions shown in the first formula above gives the following:

$$\frac{c}{n} Sum + R_a Sum = c$$

$$R_a \frac{c - \frac{c}{n} Sum}{Sum} = c \left[\frac{n - Sum}{(n)(Sum)} \right]$$

Therefore, it can be concluded that by multiplying the amount of the equity investment by

$$\frac{n - Sum}{(n)(Sum)}$$

will give the annual after-tax return that will be required to yield the desired rate of return (*i*) over the depreciable life (*n*).

The uniform interest which is required to be realized annually in order to produce a predetermined percentage interest on the outstanding unrecovered debt over the depreciable life of the debt, with the full amount of the debt being recovered through straight-line depreciation during such period, is determined similarly.

Corporate taxes of 42 percent of annual pre-tax return was included, which consisted of the 35 percent rate for Federal corporate income taxes and 7 percent for state corporate income taxes.

The following Table 5-10 summarizes reprocessing costs for LWR spent fuel (both UO₂ and MOX) for both the UREX+3c fuel cycle and the AFC.

TABLE 5-10
SUMMARY OF ANNUAL LWR REPROCESSING COSTS (UO₂ & MOX)
 (\$-Millions, 2009)

Description	UREX+3c	AFC
Operating Expenses	\$397	\$421
Depreciation (30 yr S/L)	176	182
Return on 45% of Investment (@20%/yr After-Tax Return)	398	411
$\frac{30 - 4.978}{(30)(4.978)} = 0.1676$ $(.1676)(2372) = 398 \text{ (UREX+3c)}$ $(.1676)(2452) = 411 \text{ (AFC)}$		
Corporate Income Taxes (@ 42% of Before Tax Profit)	288	298
$\frac{398}{.58} - 398 = 288 \text{ (UREX + 3c)}$ $\frac{411}{.58} - 411 = 298 \text{ (AFC)}$		
Interest on Debt of 55% of Investment (@ 7%/yr)	137	142
$\frac{30 - 12.409}{(30)(12.409)} = 0.0473$ $(.0473)(2899) = 137 \text{ (UREX+3c)}$ $(.0473)(2996) = 142 \text{ (AFC)}$		
Totals	\$1,396	\$1,454
Unit Cost	\$931/kgHM	\$969kgHM

5.2.5 MOX Fuel Fabrication

The costs for fabrication of MOX fuel for LWRs were determined by using the very limited capital cost information available and estimating the operating costs involved.

5.2.5.1 MOX Fabrication Capital Costs

INL/EXT-07-12107, Rev. 1 (Ref. 1) mentioned two different situations, a 120 MTIHM/yr plant at a 2005 cost of \$1.2-billion, and the Savannah River Site-MOX Fuel Fabrication Facility at a 2007 cost of \$2.5-billion. These were escalated using the ENR Construction Cost Index (Ref. 3), and averaged, to give a cost of \$2-billion for a 120 MTHM/yr plant.

Now the amount of MOX fuel fabrication capacity required varies with the scenario involved in Tables 4-1, 4-2, 4-3 and 4-4 of Section 4.0 of this report, as follows:

- UREX+3c using 1 recycle requires a capacity of 500 MTIHM/year for MOX fuel
- AFC using 1 recycle requires a capacity of 750 MTIHM/year for MOX fuel
- UREX+3c using 2 recycles requires a capacity of 600 MTIHM/year for MOX fuel
- AFC using 2 recycles requires a capacity of 800 MTIHM/year for MOX fuel

Therefore, the \$2-billion 120 MTIHM/yr. plant was scaled upward to each of the foregoing capacities using the 0.6 power factor^a, as follows:

<u>UREX+3c/1 Recycle</u>	<u>UREX+3c/2 Recycles</u>
$\left(\frac{500}{120}\right)^{0.6} (\$2\text{-billion}) = \$4,709\text{-million}$	$\left(\frac{600}{120}\right)^{0.6} (\$2\text{-billion}) = \$5,253\text{-million}$
<u>AFC/1 Recycle</u>	<u>AFC/2 Recycles</u>
$\left(\frac{750}{120}\right)^{0.6} (\$2\text{-billion}) = \$6,006\text{-million}$	$\left(\frac{800}{120}\right)^{0.6} (\$2\text{-billion}) = \$6,243\text{-million}$

Tables 5-11 and 5-12 show the estimated spending patterns and schedule of preoperational expenses over a 10-year design, licensing and construction period for the four facility capacities involved. Preoperational expenses were based on the assumption that they would amount to 10 percent of capital expenditures. These brought the total capital costs of the facilities to those shown in Table 5-13.

^a This power factor is used by industry to estimate the relative cost of facilities having different throughput capacities. (Refs. 10, 11, 12)

TABLE 5-11
CAPITAL EXPENDITURES SCHEDULE, PREOPERATIONAL EXPENSES AND
INTEREST THEREON FOR A MOX FUEL FABRICATION PLANT
SINGLE RECYCLE OF MOX FUEL IN LWRs
 (\$-Millions)

Year	Capital Expenditure Pattern		Preoperational Expenses ^a		Cumulative		Interest During Construction and Preop @ 5%/yr	
	UREX+3c System ^b	AFC System ^c	UREX+3c System ^b	AFC System ^c	UREX+3c System ^b	AFC System ^c	UREX+3c System ^b	AFC System ^c
1	\$151	\$192	\$15	\$19	\$166	\$211	\$8	\$11
2	301	384	30	38	497	633	25	32
3	301	384	30	39	828	1,056	41	53
4	542	691	54	69	1,424	1,816	71	91
5	603	769	60	77	2,087	2,662	104	133
6	720	919	72	91	2,879	3,672	144	184
7	720	919	72	92	3,671	4,683	184	234
8	603	769	60	77	4,334	5,529	217	276
9	542	691	54	69	4,930	6,289	247	314
10	226	288	23	29	5,179	6,606	259	330
Totals	\$4,709	\$6,006	\$470	\$600	\$25,995	\$33,157	\$1,300	\$1,658

^a10% of capital expenditure^b500 MTIHM capacity plant^c750 MTIHM capacity plant

TABLE 5-12
CAPITAL EXPENDITURES SCHEDULE, PREOPERATIONAL EXPENSES AND
INTEREST THEREON FOR A MOX FUEL FABRICATION PLANT
TWO RECYCLES OF MOX FUEL IN LWRs
 (\$-Millions)

Year	Capital Expenditure Pattern		Preoperational Expenses ^a		Cumulative		Interest During Construction and Preop @ 5%/yr	
	UREX+3c System ^b	AFC System ^c	UREX+3c System ^b	AFC System ^c	UREX+3c System ^b	AFC System ^c	UREX+3c System ^b	AFC System ^c
1	\$168	\$200	\$17	\$20	\$185	\$220	\$9	\$11
2	336	400	34	40	555	660	28	33
3	336	400	34	40	925	1,100	46	55
4	604	718	60	72	1,589	1,890	79	95
5	672	799	67	80	2,328	2,769	116	138
6	804	955	80	95	3,212	3,819	161	191
7	804	955	80	95	4,096	4,869	205	244
8	672	799	67	80	4,835	5,748	242	287
9	604	718	60	72	5,499	6,538	275	327
10	253	299	25	30	5,777	6,767	289	338
Totals	\$5,253	\$6,243	\$524	\$624	\$29,001	\$34,380	\$1,450	\$1,719

^a10% of capital expenditure^b600 MTIHM capacity plant^c800 MTIHM capacity plant

TABLE 5-13
SUMMARY OF CAPITAL COSTS FOR MOX FUEL FABRICATION FACILITIES
 (\$-Millions, 2009)

	UREX+3c 1 Recycle (500 MTIHM)	AFC 1 Recycle (750 MTIHM)	UREX+3c 2 Recycles (600 MTIHM)	AFC 2 Recycles (800 MTIHM)
Facility Cost	\$4,709	\$6,006	\$5,253	\$6,243
Pre-Op Cost	470	600	524	624
Int. During Construction	1,300	1,658	1,450	1,719
Totals	\$6,479	\$8,264	\$7,227	\$8,586

5.2.5.2 MOX Fabrication Operating Costs

Operating expenses were estimated for each of the four capacities of plants described in Section 5.2.5.1, as described by the following:

- (1) Salaries and Wages -- The numbers of personnel in operations and support activities were estimated for the various operations to proceed efficiently with a minimum of downtime. Salaries were estimated based on the assumption that somewhat higher-than-normal qualifications were desirable and thus, salaries were generally higher than normally expected. Following is a breakdown of personnel requirements and correspondingly total salaries and wages:

TABLE 5-14
SUMMARY OF ANNUAL COSTS FOR MOX FABRICATION SALARY & WAGES
 (\$-Millions, 2009)

Activity	UREX+3c 1 Recycle (500 MTIHM)		AFC 1 Recycle (750 MTIHM)		UREX+3c 2 Recycles (600 MTIHM)		AFC 2 Recycles (800 MTIHM)	
	No.	Amount	No.	Amount	No.	Amount	No.	Amount
General Management	12	\$2	12	\$2	12	\$2	12	\$2
Operations	301	17	390	22	337	19	408	23
Maintenance	158	9	199	11	174	10	207	12
Technical	253	15	296	18	269	16	304	18
Administration	365	18	372	18	368	18	374	18
Total	1,089	\$61	1,269	\$71	1,160	\$65	1,305	\$73

- (2) Benefits -- Benefits were taken at 40 percent of direct salaries and wages, and include time off (vacations, holidays, sick days, etc.), FICA taxes, medical and hospitalization insurance, disability insurance, life insurance, pension plan contributions, etc.
- (3) Hardware -- Rough estimates were made of the cost of Zircaloy 4 tubing, tubing end caps, upper and bottom end fittings and spacer grids, plus a 30 percent

contingency allowance. These amounted to about \$70/kgIHM (about 23 percent of the cost of UO₂ fuel fabrication) -- which seemed reasonable.

- (4) Chemicals -- Determined the quantities required for principal chemicals (ammonia, hydrogen, miscellaneous chemicals) and applied estimated current market prices to them.
- (5) Operating Supplies -- Estimated operating supplies based on the assumption that they would amount to \$100/day for operators and foremen; \$125/day for maintenance, foremen and instrument technicians; and \$25/day for all others -- as shown below

TABLE 5-15
SUMMARY OF ANNUAL MOX FABRICATION OPERATING SUPPLIES
(2009)

Activity	UREX+3c 1 Recycle (500 MTIHM)		AFC 1 Recycle (750 MTIHM)		UREX+3c 2 Recycles (600 MTIHM)		AFC 2 Recycles (800 MTIHM)	
	No.	Amount	No.	Amount	No.	Amount	No.	Amount
Operators & Foremen 500 MT - 192+44 750 MT - 325 (interpolated) 600 MT - 271 (interpolated) 800 MT - 295+48	236	\$5,428	325	\$7,575	271	\$6,223	343	\$7,889
Maint Mech, Foremen & Instr 500 MT - 86+17 750 MT - 144 (interpolated) 600 MT - 119 (interpolated) 800 MT - 121+31	103	2,961	144	4,140	119	3,421	152	4,370
All Other 500 MT - 1,089-339 750 MT - (interpolated) 600 MT - (interpolated) 800 MT - 1,305-495	750	4,313	800	4,600	770	4,428	810	4,658
Total	1,089	\$12,702	1,269	\$16,215	1,160	\$14,082	1,305	\$16,917

- (6) Maintenance Spare Parts and Equipment -- Based on Nuclear Fuel Services' estimates that 8.5 percent of the value of the capital cost of installed equipment (excluding preoperational expenses and interest during construction) would be required to provide spare parts and equipment for maintaining a reprocessing facility and assuming that this would also apply to at least a semi-remotely operated fabrication plant for MOX fuel. (Ref. 6) Assuming that the cost of equipment was 35 percent of the capital cost and that the NFS figure of 8.5 percent is appropriate, then the spare parts and equipment costs would be:

UREX+3c/1 Recycle (4,709-million)(0.35)(0.085) = \$140-million/year
(500 MTIHM Capacity)

AFC/1 Recycle (\$6,006-million)(0.35)(0.085) = \$179-million/year
(750 MTIHM Capacity)

UREX+3c/2 Recycles (\$5,253-million)(0.35)(0.085) = \$156-million/year
(600 MTIHM Capacity)

AFC/2 Recycles (\$6,243-million)(0.35)(0.085) = \$186-million/year
(800 MTIHM Capacity)

(7) Electricity -- Estimated kW installed capacity as follows:

UREX+3c/1 Recycle
(500 MTIHM Capacity) 9,500 kW

AFC/1 Recycle
(750 MTIHM Capacity) 12,000 kW

UREX+3c/2 Recycles
(600 MTIHM Capacity) 10,500 kW

AFC/2 Recycles
(800 MTIHM Capacity) 12,500 kW

Applied DOE Energy Information Administration average industrial rates for Virginia. (Ref. 8)

(8) Property and Liability Insurance -- Nuclear insurers were contacted for input on property and liability insurance costs. Based on these discussions and the paucity of risk experience with MOX fuel fabrication plants, coupled with a prospective limitation on the ceiling of coverage, it was concluded that a \$3-million/year premium for property insurance and an \$6-million/year premium for liability insurance would be appropriate.

(9) Property Taxes -- Based on county taxes in Calvert County, Maryland for utility assets of \$2.23/\$100 valuation, where the Calvert Cliffs Nuclear Power Station is located. (Ref. 9) This was rounded downward to \$2/\$100 valuation because Calvert County's tax rates are generally higher than tax rates in other rural areas of the U.S. Thus, the property taxes were estimated to be:

UREX+3c/1 Recycle
(500 MTIHM Capacity) (\$4,709-million)(\$2/100 val) = \$94-million

AFC/1 Recycle
(750 MTIHM Capacity) (\$6,006-million)/(\$2/100 val) = \$120-million

UREX+3c/2 Recycles
(600 MTIHM Capacity) (\$5,253-million)/(\$2/100 val) = \$105-million

AFC/2 Recycles
(800 MTIHM Capacity) (\$6,243-million)/(\$2/100 val) = \$125-million

The following table summarizes the annual operating expenses for the various capacities of fuel fabrication facilities used in this study.

TABLE 5-16
SUMMARY OF ANNUAL OPERATING EXPENSES FOR MOX FABRICATION
FACILITIES
(\$-Millions, 2009)

Description	UREX+3c 1 Recycle (500 MTIHM)	AFC 1 Recycle (750 MTIHM)	UREX+3c 2 Recycles (600 MTIHM)	AFC 2 Recycles (800 MTIHM)
Salaries & Wages	\$61	\$71	\$65	\$73
Benefits (40%)	24	28	26	29
Hardware	34	52	42	55
Chemicals	4	6	5	6
Supplies	13	16	14	17
Maint Parts & Equipment	140	179	156	186
Electricity	6	7	6	7
Insurance	6	6	6	6
Property Taxes	94	120	105	125
Other ^a	5	7	6	7
Total	\$387	\$492	\$431	\$512

^a30% of chemicals and supplies

5.2.5.3 MOX Fabrication Unit Costs

The unit costs for MOX fabrication include the following:

- (1) Operating expenses (as set forth in Section 5.2.5.2)
- (2) Depreciation (on a 30-year straight line basis)
- (3) Return on equity investment of 20 percent/year (after taxes)
- (4) Corporate income taxes of 42 percent of annual return on equity
- (5) Interest on debt of 7 percent/year.

In order to calculate the latter three, it was assumed that the MOX fabrication plant would be privately owned by a consortium of utility companies and would be financed by a mixture of debt (55 percent) and equity (45 percent).

The uniform annual after-tax return and the uniform annual interest rate that is required to meet return and interest requirements were determined as described in Section 5.2.4.3.

Corporate taxes of 42 percent of annual pre-tax return was included which consisted of the 35 percent rate for Federal corporate income taxes plus 7 percent for state corporate income taxes.

The following Table 5-17 summarizes the estimated costs for MOX fuel fabrication for the various capacity facilities required by this study.

TABLE 5-17
SUMMARY OF ANNUAL MOX FUEL FABRICATION COSTS
(\$-Millions, 2009)

Description	UREX+3c 1 Recycle (500 MTIHM)	AFC 1 Recycle (750 MTIHM)	UREX+3c 2 Recycles (600 MTIHM)	AFC 2 Recycles (800 MTIHM)
Operating Expenses	\$387	\$492	\$431	\$512
Depreciation (30 yr S/L)	216	275	241	286
Return on 45% of investment (@20%/yr After-Tax Return)	489	623	545	648
$\frac{30 - 4.978}{(30)(4.978)} = 0.1676$ (.1676)(2,916) = 489 (.1676)(3,719) = 623 (.1676)(3,252) = 545 (.1676)(3,864) = 648				
Corporate Income Taxes (@ 42% of Before Tax Profit)	354	451	395	469
$\frac{489}{.58} - 489 = 354$ $\frac{623}{.58} - 623 = 451$ $\frac{545}{.58} - 545 = 395$ $\frac{648}{.58} - 648 = 469$				
Interest on Debt of 55% of Investment (@ 7%/yr)	169	215	188	223
$\frac{30 - 12.409}{(30)(12.409)} = 0.0473$ (.0473)(3,563) = 169 (.0473)(4,545) = 215 (.0473)(3,975) = 188 (.0473)(4,722) = 223				
Totals	\$1,615	\$2,056	\$1,800	\$2,138
Unit Cost	\$3,230/kgHM	\$2,741/kgHM	\$3,000/kgHM	\$2,673/kgHM

5.2.6 Conversion of Depleted UF₆ to U₃O₈

The unit cost of \$10/kgU (2004 dollars) for conversion and disposal of depleted UF₆ from enrichment of virgin uranium as shown in INL/EXT-07-12107, Rev. 1 (Ref. 1) was used after escalation by the CPI from 2007 to 2009. This produced a unit cost for the conversion of \$10.40/kgU.

These costs were based on Uranium Disposal Services estimates of \$2.80/kgUF₆ (\$4.14/kgU) for conversion, and \$5/kgU₃O₈ (\$5.90/kgU) for disposal, both in 2007 dollars, and escalated to 2009 using the CPI.

5.2.7 Enrichment of Recycled U

It was assumed that the enrichment of recycled uranium would involve a premium of about 50% over that of enrichment of virgin uranium, or about \$245/kgSWU (2009 dollars). DeRoo and Parsons of MIT (Ref. 13) estimated that this premium would amount to 10%, but the 50% figure was used in order to be conservatively high since it was expected that there would need to be dedicated facilities or equipment lines that were appropriately shielded in order to enrich recovered uranium.

5.2.8 Fabrication of Initial Cores of FR Fuel

5.2.8.1 FR Initial Core Fabrication Capital Cost

The amount of FR fuel (initial cores) that were needed to support the disposal of TRU from 100 LWRs operating over 81 years was the same for both the UREX+3c fuel cycle and the AFC, for both a single recycle of MOX to LWRs and two such recycles. However, a single recycle in both cases required a FR fuel fabrication capacity of 92 MTIHM/yr. and two recycles in both cases required a FR fuel fabrication capacity of 46 MTIHM/yr.

It was assumed that the fabrication of the initial cores of FR fuel, which would be metallic Pu/U/Zr rods with sodium in the space between the rods and the clad for heat transfer efficiency, would involve a 25 percent higher degree of difficulty than fabrication of MOX fuel.

Therefore, the \$2-billion 120 MTIHM/yr. plant described in Section 5.2.5.1 for MOX fuel was scaled downward to each of the foregoing required capacities using the 0.6 power factor, as follows:

Single Recycle

$$\left(\frac{92}{120}\right)^{0.6} (\$2\text{-billion}) = \$1,705\text{-million}$$

Two Recycles

$$\left(\frac{46}{120}\right)^{0.6} (\$2\text{-billion}) = \$1,125\text{-million}$$

Using estimated spending patterns and schedules of preoperational expenses over a 10-year design, licensing and construction period for the two facility capacities involved brought the total capital costs to \$2,347-million and \$1,549-million for the 92 and 46 MTIHM/yr. capacity facilities, respectively, as shown in the following Table F-18.

TABLE 5-18
SUMMARY OF CAPITAL COSTS FOR FABRICATION OF INITIAL FR CORES
 (\$-Millions, 2009)

Description	UREX+3c and AFC 1 Recycle (92 MTIHM)	UREX+3c and AFC 2 Recycles (46 MTIHM)
Facility Cost	\$1,705	\$1,125
Pre-Operational Cost	171	113
Interest During Construction	471	311
Totals	\$2,347	\$1,549

Increasing these for the 25 percent added difficulty of fabricating the high plutonium content metallic fuel over that of MOX produced total capital costs of \$2,934-million and \$1,936-million for the 92 and 46 MTIHM/yr. facilities, respectively.

5.2.8.2 FR Initial Core Fabrication Operating Costs

The operating costs for FR fabrication (initial core) were obtained by linearly extrapolating the operating costs for MOX fuel fabrication facilities from Table 5-16 to the capacities required for FR fuel fabrication, and increasing the result by 25 percent to adjust for the added difficulty in fabricating, as previously mentioned.

5.2.8.3 Unit Costs for Fabrication of Initial FR Cores

The unit costs for FR fuel fabrication include all of the same elements as described for MOX fuel fabrication in Section 5.2.5.3. The following Table 5-19 summarizes the estimated costs for FR fuel fabrication for the two capacity facilities required by this study. This clearly is an approximation, but its use impacts the UREX+3c fuel cycle and the AFC the same and, thus, it does not impact a comparison of costs for the two fuel cycle systems. However, it does impact the comparison of costs for the number of MOX recycles involved in each system.

TABLE 5-19
SUMMARY OF ANNUAL FUEL FABRICATION COSTS FOR INITIAL FR CORES
(\$-Millions, 2009)

Description	UREX+3c and AFC 1 MOX Recycle (92 MTIHM)	UREX+3c and AFC 2 MOX Recycles (46 MTIHM)
Operating Expenses	\$260	\$234
Depreciation (30 yr S/L)	98	65
Return on 45% of investment (@20%/yr After-Tax Return) $\frac{30 - 4.978}{(30)(4.978)} = 0.1676$ $(.1676)(1,320) = 221$ $(.1676)(871) = 146$	221	146
Corporate Income Taxes (@ 42% of Before Tax Profit) $\frac{221}{.58} - 221 = 160$ $\frac{146}{.58} - 146 = 106$	160	106
Interest on Debt of 55% of Investment (@ 7%/yr) $\frac{30 - 12.409}{(30)(12.409)} = 0.0473$ $(.0473)(1,614) = 76$ $(.0473)(1,065) = 50$	76	50
Totals	\$815	\$601
Unit Cost	\$8,859/kgHM	\$13,065/kgHM

5.2.9 Reprocessing and Refabrication of FR Fuel

This activity was assumed to involve electrochemical processing of FR metallic fuel (head end, separations and FR fuel fabrication only -- not waste product packaging, storage or disposal), plus refabrication of FR fuel. To the unit cost of \$10,000/kgIHM (2007 dollars) for targets containing significant level of actinides (Np, Am, Cm) from INL/EXT-07-12107, Rev. 1 (Ref. 1), 10 percent was added for waste handling, and the result escalated to 2009 dollars using the CPI. This gave a total unit cost of \$11,330/kgIHM. This is really a soft number, but its use impacts the UREX+3x fuel cycle and the AFC the same and, thus, it does not impact a comparison of costs for the two fuel cycle systems. However, it does impact the comparison of costs for the number of MOX recycles involved in each system.

5.2.10 Disposal of Excess U Recovered in Reprocessing and Depleted U from Reenrichment

It was assumed that these excess uranium streams would be converted to U_3O_8 and the cost therefor has been included in other line items. It was also assumed that the U_3O_8 would have to be disposed of in a geologic repository. Therefore, the unit costs for disposal used were taken from INL/EXT-07-12107, Rev. 1, and amounted to \$72/kgU (2009 dollars) for geologic disposal, etc. (Ref. 1) Escalating these costs to 2009, gave a unit cost of \$75/kgU for disposal.

These costs are probably high, but have little effect on the overall fuel cycle costs.

5.2.11 Low Level Waste Disposal

The State of South Carolina FY-2009 pricing schedule for Class A waste at the Barnwell, SC LLW disposal facility of \$297/cu ft. plus a surcharge of \$6/cu ft. was used in this study. (Ref. 14) It was adjusted for one year of escalation to produce a unit cost of \$312/cu ft. for LLW disposal.

5.2.12 Transuranium Waste Disposal

Funding at the DOE's Waste Isolation Pilot Plant (WIPP) for TRU waste disposal has average a little over \$200-million annually for the past three years. (Ref. 15) Assuming this pattern is a continuation of past practices, WIPP would have been funded for about \$2,000-million since its startup in 1999. Also during this period, WIPP has emplaced a total of 2,177,798 cu ft. TRU waste in its geologic repository. (Ref. 16) This averages about \$918/cu ft.

of TRU waste. This unit cost was increased by about 25 percent to allow for recovery of capital related costs, resulting in a total unit cost of \$1,150/cu ft.

5.2.13 Disposal of High Level Waste

The Nuclear Waste Policy Act (NWPA) of 1982 and its amendments established that owners of spent fuel and high level wastes would pay for the disposal thereof by DOE at a rate of 1 mill/kWH of electricity generated by nuclear power plants -- basically as such generation occurred. While this rate covered spent LWR fuel and high level waste, it was assumed for the purposes of this study that such would also apply to spent FR fuel and high level waste.

However, this fee was based on receiving mostly spent LWR fuel, which amounts to about 300 cu ft./GWY, whereas reprocessing results in vitrified wastes of about 72 cu ft./GWY (which includes Cs, Sr, Ba and Rb), and hulls and end fittings of about 94 cu ft./GWY. This represents only about 55 percent of the volume of spent fuel. Thus, it was assumed for the purposes of this study that the cost of waste disposal from the LWR spent fuel reprocessing step of the UREX+3c fuel cycle and AFC would only be about 55 percent of the cost of spent fuel disposal, or \$0.00055/kWH. This is a conservatively high cost inasmuch as it has the potential to be reduced by as much as 50 percent by separating the technetium from the other wastes, thereby allowing the latter to be disposed of separately and not consume so much repository capacity.

Now it is understood that there are certain fixed costs that do not vary with the volume of waste emplaced. However, since in the UREX+3c fuel cycle and AFC (i) the bulk of the fission products (Cs, Sr, etc.) are being stored above ground until they decay to innocuous levels, thus, reducing the volume of vitrified waste substantially, (ii) protection against UO₂ oxidation in the spent fuel isn't necessary, and (iii) substantially simpler receipt, processing and handling facilities would be required for the waste -- it was believed that the use of a cost that was determined as a function of volume, would give an appropriate approximation of the expected cost.

In applying the foregoing described fee to the amount of electricity generated, it was assumed that LWRs would operate at a 90 percent plant factor (recent experience in US) and that FRs would operate at a 100 percent plant factor (a lower plant factor would increase the number of FRs required, but not the total fuel cycle quantities or costs). This means that LWRs would

experience a cost of \$4.336-million each reactor year, and that FRs would experience a cost of \$6.939-million each reactor year, for high level waste disposal.

Here again, the HLW disposal cost impacts the UREX+3c fuel cycle and the AFC the same and, thus, it does not impact a comparison of costs for the two fuel cycle systems.

However, it does impact the comparison of costs for the number of MOX recycles involved in each system.

5.3 UNIT COSTS

The unit costs used to calculate the total fuel cycle costs of the UREX+3c fuel cycle and the AFC, for both one and two recycles of MOX fuel in LWRs, were determined as described in Section 5.2. The results of this determination are summarized in Table 5-20.

TABLE 5-20
UNIT COSTS USED FOR DEVELOPING FUEL CYCLE COST COMPARISONS FOR THE
UREX+3c FUEL CYCLE AND THE AFC
(\$, 2009)

Description	UREX+3c 1 Recycle	AFC 1 Recycle	UREX+3c 2 Recycles	AFC 2 Recycles
UO ₂ Fuel (\$/kgU)				
\$46/lb U ₃ O ₈	\$2,945	\$2,945	\$2,945	\$2,945
\$125/lb U ₃ O ₈	\$5,066	\$5,066	\$5,066	\$5,066
\$200/lb U ₃ O ₈	\$7,131	\$7,131	\$7,131	\$7,131
LWR Fuel Reprocessing (\$/kgIHM)	\$931	\$969	\$931	\$969
MOX Fuel Fabrication (\$/kgIHM)	\$3,230	\$2,741	\$3,000	\$2,673
DU Conversion to U ₃ O ₈ (\$/kgU)	\$10.40	\$10.40	\$10.40	\$10.40
RU Reenrichment (\$/kgSWU)	N/A ^a	\$245	N/A ^a	\$245
Initial FR Core Fabrication (\$/kgIHM)	\$8,859	\$8,859	\$13,065	\$13,065
Reprocessing & Refabrication FR Fuel (\$/kgIHM)	\$11,330	\$11,330	\$11,330	\$11,330
Excess Recovered U and Depleted Recycled U Disposal (\$/kgIHM)	\$75	\$75	\$75	\$75
LLW Disposal (\$/cu ft.)	\$312	\$312	\$312	\$312
TRU Disposal (\$/cu ft.)	\$1,150	\$1,150	\$1,150	\$1,150
HLW Disposal (mill/kWh)	\$0.55	\$0.55	\$0.55	\$0.55

^aN/A = Not Applicable

5.4 COMPARATIVE FUEL CYCLE COSTS

Using the unit costs set forth in Table 5-20, and the relevant material flows included in Section 4.0 (Tables 4-1 through 4-4), and Section 8 (Tables 8-2 through 8-5), the total fuel cycle costs were calculated for the four cases studied -- both on a current dollar and present worth basis. The results are shown in Table 5-21. Detailed cost calculations are included in Tables 5-22 through 5-37 at the end of this Section 5.0.

It is generally good business practice to discount future cash flows. The principle here is that a dollar received today is worth more than a dollar received in the future --because the former can be put in the bank and earn interest. Therefore, future year costs are generally discounted to present value using the discount rate that reflects the real value of money (i.e., the ratio of the value of money to the inflation rate). Now, there are some who don't practice discounting of the cost of nuclear activities that are similar to activities covered in this report because of likely future regulatory changes that can cause costs to increase in the future above those estimated at the present. However, the cost of possible future regulatory changes cannot be determined and should be dealt with when they are under consideration and/or implemented rather than speculating today what, when, and how extensive they might be. Thus, for the purpose of this study, a conservatively low discount rate of 2 percent/year real value of money was used in connection with the cost comparisons.

TABLE 5-21
COMPARISON OF ESTIMATED FUEL CYCLE COSTS FOR 100 LWRs AND ASSOCIATED FAST REACTORS
OVER 81 YEARS^a

Activity	Total Cost (\$-Millions, 2009)				Total Cost Discounted @ 2%/Year (\$-Millions)			
	UREX+3c 1 Recycle	AFC 1 Recycle	UREX+3c 2 Recycles	AFC 2 Recycles	UREX+3c 1 Recycle	AFC 1 Recycle	UREX+3c 2 Recycles	AFC 2 Recycles
UO ₂ Fuel (\$46/lb U ₃ O ₈)	\$ 354,758	\$ 321,557	\$ 333,786	\$ 298,115	\$ 146,398	\$ 133,775	\$ 140,032	\$ 126,891
UO ₂ Fuel (\$125/lb U ₃ O ₈)	610,257	553,143	574,181	512,819	251,834	230,120	240,884	218,279
UO ₂ Fuel (\$200/lb U ₃ O ₈)	859,009	778,615	808,228	721,854	354,487	323,922	339,072	307,254
Reprocessing	179,311	186,629	183,630	191,126	66,020	68,715	66,933	69,665
MOX Fuel Fabrication	93,168	110,242	108,156	129,021	35,015	41,743	39,178	47,119
DU Conversion (U ₃ O ₈)	11,661	10,570	10,972	9,799	4,812	4,397	4,603	4,171
RU Enrichment	-	21,758	-	24,369	-	8,239	-	8,968
Fabrication of Initial FR Cores	34,877	34,877	27,512	27,512	12,216	12,216	9,395	9,395
Repro & Refab of FR Recycle Fuel	260,200	260,200	133,212	133,212	65,048	65,048	33,098	33,098
Excess Recovered U & Depleted Recycled U Disposal	12,013	9,944	12,255	9,908	4,444	3,658	4,487	3,623
LLW Disposal	3,120	2,965	3,192	3,016	1,150	1,092	1,165	1,100
TRU Disposal	12,249	13,924	11,255	13,066	4,081	4,728	3,859	4,534
HLW Disposal	46,161	46,161	40,788	40,788	17,024	17,024	15,644	15,644
TOTAL (\$46/lb U₃O₈)	\$1,007,518	\$1,018,827	\$ 864,758	\$ 879,932	\$ 356,208	\$ 360,635	\$ 318,394	\$ 324,208
TOTAL (\$125/lb U₃O₈)	\$1,263,017	\$1,250,413	\$1,105,153	\$1,094,636	\$ 461,644	\$ 456,980	\$ 419,246	\$ 415,596
TOTAL (\$200/lb U₃O₈)	\$1,511,769	\$1,475,885	\$1,339,200	\$1,303,671	\$ 564,297	\$ 550,782	\$ 517,434	\$ 504,571

^aExcludes investment in FRs, cost of LLW disposal for LWRs, revenues from electric power generation, and O+M costs for LWRs and FRs

The cost comparisons set forth in Table 5-21 show that the AFC is more costly (on a discounted basis) than the UREX+3c fuel cycle by about 1.4 percent when MOX is recycled once in each fuel cycle, and is about 1.9 percent more costly when MOX is recycled twice in each fuel cycle -- when the price of natural uranium concentrates is at \$46/lb U_3O_8 . However, when this price increases to \$125/lb U_3O_8 , the AFC is less costly by 1.1 percent when MOX is recycled once and by 0.7 percent when MOX is recycled twice. Even increasing the price of concentrates to \$200/lb U_3O_8 only gains the AFC a 2.3 percent advantage over the UREX+3c fuel cycle for either recycle scenario.

There are several opportunities to reduce the fuel cycle costs shown in Table 5-21. For example:

- (1) Generally the cost of MOX fabrication, the fabrication of the initial cores of fast reactors, and the reprocessing and refabrication of FR fuels, may be brought down as more experience is gained by the newer generation of engineers and scientists working in demonstration projects. Most of those experienced in related technologies and projects have retired since reprocessing and recycle was abandoned in the U.S. in 1977 because of proliferation concerns. Thus, the construction and operation of pilot and demonstration facilities in these areas is an absolute necessity to bring U.S. technology current with or ahead of the current world state-of-the-art.
- (2) The MOX fuel fabrication costs in Table 5-21 seem high even for a totally remotely operated processing system. This is influenced to a large extent by the recent cost experience of DOE in its design and construction of a first-of-a-kind MOX fuel fabrication facility for disposition of weapons plutonium. It is highly likely that a MOX fuel fabrication facility owned by private industry could be built at a substantially lower cost.
- (3) The fabrication of FR recycle fuel and initial FR cores might be performed in the same facility thus significantly reducing the unit costs involved.
- (4) From our knowledge, the reprocessing of spent FR fuel has not had extensive study from a cost standpoint, and many technical details remain to be identified. Improvements in the knowledge of the costs of facilities, once the technical details are in the process of being firmed up, can be expected to result in substantially lower costs.
- (5) The costs for disposal of depleted recycled uranium in the AFC were based on the conservative assumption that such would have to be accomplished in a geologic repository. This is probably not realistic inasmuch as U^{234} contained in the input to enrichment would move toward the enriched uranium stream in the enrichment

process and, thus, would not represent an additional radiological health hazard for the depleted uranium. This would reduce the disposal costs for this material by about 95 percent.

- (6) As mentioned in Section 5.2.13, the HLW disposal costs have the potential for being reduced by as much as 50 percent by not including technetium with hulls, end fittings, etc.

5.5 CONCLUSIONS

From the foregoing sections, it can be seen that:

- (1) There aren't any savings in fuel cycle costs resulting from the recycle of uranium from the reprocessing of spent LWR fuel (AFC case) over a fuel cycle that does not employ such recycle (UREX+3c fuel cycle), until the price of natural uranium concentrates exceeds about \$90/lb U_3O_8 . Thus, other factors such as conservation of natural resources and disposal of recovered uranium would be more important considerations than would be cost.
- (2) The cost of UO_2 fuel, fabrication of MOX fuel, the reprocessing of spent LWR fuel, and reprocessing and refabrication of recycle FR fuel are the main cost drivers (by magnitude) among the fuel cycle cost elements examined. The cost of fabrication of initial FR cores, and spent FR fuel reprocessing and refabrication don't have an impact on the comparison of costs for the UREX+3c fuel cycle and AFC systems.
- (3) Recycling of MOX fuel twice to LWRs results in 8-11 percent lower fuel cycle costs than does a single recycle of MOX fuel in both the UREX+3c fuel cycle and the AFC.

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TABLE 5-22
COSTS OF UO₂ FUEL
FOR 100 LWRs OVER 81 YEARS
(*\$46/lb U₃O₈ YELLOWCAKE*)

Year	UREX+3c -- 1 MOX Recycle				AFC -- 1 MOX Recycle				UREX+3c -- 2 MOX Recycles				AFC -- 2 MOX Recycles			
	Quantity (MTU)	Unit Price (\$/kgU, 2009)	Total Cost (\$-Millions, 2009)	Total Cost Discounted @ 2%/Year (\$-Millions)	Quantity (MTU)	Unit Price (\$/kgU, 2009)	Total Cost (\$-Millions, 2009)	Total Cost Discounted @ 2%/Year (\$-Millions)	Quantity (MTU)	Unit Price (\$/kgU, 2009)	Total Cost (\$-Millions, 2009)	Total Cost Discounted @ 2%/Year (\$-Millions)	Quantity (MTU)	Unit Price (\$/kgU, 2009)	Total Cost (\$-Millions, 2009)	Total Cost Discounted @ 2%/Year (\$-Millions)
2020	1,840	\$ 2,945	\$ 5,419	\$ 4,358	1,840	\$ 2,945	\$ 5,419	\$ 4,358	1,840	\$ 2,945	\$ 5,419	\$ 4,358	1,840	\$ 2,945	\$ 5,419	\$ 4,358
2021	1,840	2,945	5,419	4,273	1,840	2,945	5,419	4,273	1,840	2,945	5,419	4,273	1,840	2,945	5,419	4,273
2022	1,840	2,945	5,419	4,189	1,840	2,945	5,419	4,189	1,840	2,945	5,419	4,189	1,840	2,945	5,419	4,189
2023	1,840	2,945	5,419	4,107	1,840	2,945	5,419	4,107	1,840	2,945	5,419	4,107	1,840	2,945	5,419	4,107
2024	1,840	2,945	5,419	4,026	1,840	2,945	5,419	4,026	1,840	2,945	5,419	4,026	1,840	2,945	5,419	4,026
2025	1,840	2,945	5,419	3,947	1,840	2,945	5,419	3,947	1,840	2,945	5,419	3,947	1,840	2,945	5,419	3,947
2026	1,840	2,945	5,419	3,870	1,840	2,945	5,419	3,870	1,840	2,945	5,419	3,870	1,840	2,945	5,419	3,870
2027	1,840	2,945	5,419	3,794	1,840	2,945	5,419	3,794	1,840	2,945	5,419	3,794	1,840	2,945	5,419	3,794
2028	1,840	2,945	5,419	3,720	1,840	2,945	5,419	3,720	1,840	2,945	5,419	3,720	1,840	2,945	5,419	3,720
2029	1,840	2,945	5,419	3,647	1,840	2,945	5,419	3,647	1,840	2,945	5,419	3,647	1,840	2,945	5,419	3,647
2030	1,840	2,945	5,419	3,575	1,840	2,945	5,419	3,575	1,840	2,945	5,419	3,575	1,840	2,945	5,419	3,575
2031	1,840	2,945	5,419	3,505	1,840	2,945	5,419	3,505	1,840	2,945	5,419	3,505	1,840	2,945	5,419	3,505
2032	1,334	2,945	3,929	2,492	1,105	2,945	3,254	2,064	1,335	2,945	3,930	2,492	1,105	2,945	3,254	2,064
2033	1,334	2,945	3,929	2,443	1,105	2,945	3,254	2,023	1,335	2,945	3,930	2,443	1,105	2,945	3,254	2,023
2034	1,334	2,945	3,929	2,395	1,105	2,945	3,254	1,984	1,335	2,945	3,930	2,396	1,105	2,945	3,254	1,984
2035	1,334	2,945	3,929	2,348	1,105	2,945	3,254	1,945	1,335	2,945	3,930	2,349	1,105	2,945	3,254	1,945
2036	1,334	2,945	3,929	2,302	1,105	2,945	3,254	1,907	1,335	2,945	3,930	2,302	1,105	2,945	3,254	1,907
2037	1,334	2,945	3,929	2,257	1,105	2,945	3,254	1,869	1,335	2,945	3,930	2,257	1,105	2,945	3,254	1,869
2038	1,334	2,945	3,929	2,213	1,105	2,945	3,254	1,832	1,335	2,945	3,930	2,213	1,105	2,945	3,254	1,832
2039	1,334	2,945	3,929	2,169	1,105	2,945	3,254	1,797	1,335	2,945	3,930	2,170	1,105	2,945	3,254	1,797
2040	1,334	2,945	3,929	2,127	1,105	2,945	3,254	1,761	1,335	2,945	3,930	2,127	1,105	2,945	3,254	1,761
2041	1,334	2,945	3,929	2,085	1,105	2,945	3,254	1,727	1,335	2,945	3,930	2,085	1,105	2,945	3,254	1,727
2042	1,334	2,945	3,929	2,044	1,105	2,945	3,254	1,693	1,335	2,945	3,930	2,045	1,105	2,945	3,254	1,693
2043	1,366	2,945	4,024	2,052	1,167	2,945	3,438	1,753	1,229	2,945	3,618	1,845	1,065	2,945	3,136	1,599
2044	1,375	2,945	4,048	2,024	1,183	2,945	3,484	1,742	1,233	2,945	3,630	1,815	1,073	2,945	3,159	1,579
2045	1,383	2,945	4,072	1,996	1,198	2,945	3,529	1,730	1,237	2,945	3,642	1,785	1,080	2,945	3,182	1,560
2046	1,384	2,945	4,075	1,958	1,200	2,945	3,535	1,699	1,237	2,945	3,643	1,751	1,081	2,945	3,183	1,530
2047	1,385	2,945	4,078	1,921	1,202	2,945	3,541	1,668	1,237	2,945	3,644	1,717	1,082	2,945	3,185	1,501
2048	1,386	2,945	4,081	1,885	1,204	2,945	3,547	1,638	1,238	2,945	3,646	1,684	1,082	2,945	3,187	1,472
2049	1,387	2,945	4,084	1,850	1,206	2,945	3,552	1,609	1,238	2,945	3,647	1,652	1,083	2,945	3,189	1,444
2050	1,388	2,945	4,087	1,815	1,208	2,945	3,558	1,580	1,239	2,945	3,648	1,620	1,084	2,945	3,191	1,417
2051	1,389	2,945	4,090	1,780	1,210	2,945	3,564	1,551	1,239	2,945	3,649	1,588	1,084	2,945	3,193	1,390
2052	1,390	2,945	4,093	1,747	1,212	2,945	3,570	1,523	1,239	2,945	3,650	1,558	1,085	2,945	3,195	1,363
2053	1,391	2,945	4,096	1,714	1,214	2,945	3,576	1,496	1,240	2,945	3,651	1,528	1,085	2,945	3,197	1,338
2054	1,392	2,945	4,099	1,681	1,216	2,945	3,581	1,469	1,325	2,945	3,901	1,600	1,124	2,945	3,309	1,357
2055	1,393	2,945	4,102	1,650	1,218	2,945	3,587	1,443	1,326	2,945	3,905	1,570	1,125	2,945	3,313	1,332
2056	1,394	2,945	4,105	1,618	1,220	2,945	3,593	1,417	1,327	2,945	3,909	1,541	1,126	2,945	3,317	1,308
2057	1,395	2,945	4,108	1,588	1,222	2,945	3,599	1,391	1,328	2,945	3,910	1,512	1,127	2,945	3,319	1,283
2058	1,396	2,945	4,111	1,558	1,224	2,945	3,604	1,366	1,328	2,945	3,912	1,482	1,128	2,945	3,322	1,259
2059	1,397	2,945	4,114	1,528	1,226	2,945	3,610	1,341	1,329	2,945	3,913	1,454	1,129	2,945	3,324	1,235
2060	1,398	2,945	4,117	1,500	1,228	2,945	3,616	1,317	1,329	2,945	3,915	1,426	1,129	2,945	3,326	1,211
2061	1,399	2,945	4,120	1,471	1,230	2,945	3,622	1,293	1,330	2,945	3,916	1,399	1,130	2,945	3,328	1,188
2062	1,400	2,945	4,123	1,444	1,232	2,945	3,628	1,270	1,330	2,945	3,918	1,372	1,131	2,945	3,330	1,166
2063	1,401	2,945	4,126	1,416	1,234	2,945	3,633	1,247	1,331	2,945	3,919	1,345	1,131	2,945	3,332	1,144
2064	1,402	2,945	4,129	1,390	1,236	2,945	3,639	1,225	1,331	2,945	3,921	1,319	1,132	2,945	3,334	1,122
2065	1,403	2,945	4,132	1,363	1,238	2,945	3,645	1,202	1,268	2,945	3,735	1,232	1,094	2,945	3,222	1,063
2066	1,404	2,945	4,135	1,338	1,240	2,945	3,651	1,181	1,268	2,945	3,734	1,208	1,094	2,945	3,223	1,042
2067	1,405	2,945	4,138	1,312	1,242	2,945	3,656	1,159	1,268	2,945	3,733	1,184	1,095	2,945	3,224	1,022
2068	1,406	2,945	4,141	1,287	1,244	2,945	3,662	1,138	1,268	2,945	3,735	1,161	1,095	2,945	3,225	1,003
2069	1,407	2,945	4,144	1,263	1,245	2,945	3,668	1,118	1,269	2,945	3,736	1,139	1,096	2,945	3,227	984
2070	1,408	2,945	4,147	1,239	1,247	2,945	3,674	1,098	1,269	2,945	3,737	1,117	1,097	2,945	3,229	965
2071	1,409	2,945	4,150	1,216	1,249	2,945	3,679	1,078	1,269	2,945	3,739	1,095	1,097	2,945	3,231	947
2072	1,410	2,945	4,153	1,193	1,251	2,945	3,685	1,058	1,270	2,945	3,740	1,074	1,098	2,945	3,233	929
2073	1,411	2,945	4,156	1,170	1,253	2,945	3,691	1,039	1,270	2,945	3,741	1,053	1,099	2,945		

TABLE 5-23
COSTS OF UO₂ FUEL
FOR 100 LWRs OVER 81 YEARS
(\$125/lb U₃O₈ YELLOWCAKE)

Year	UREX+3c -- 1 MOX Recycle				AFC -- 1 MOX Recycle				UREX+3c -- 2 MOX Recycles				AFC -- 2 MOX Recycles			
	Quantity (MTU)	Unit Price (\$/kgU, 2009)	Total Cost (\$-Millions, 2009)	Total Cost Discounted @ 2%/Year (\$-Millions)	Quantity (MTU)	Unit Price (\$/kgU, 2009)	Total Cost (\$-Millions, 2009)	Total Cost Discounted @ 2%/Year (\$-Millions)	Quantity (MTU)	Unit Price (\$/kgU, 2009)	Total Cost (\$-Millions, 2009)	Total Cost Discounted @ 2%/Year (\$-Millions)	Quantity (MTU)	Unit Price (\$/kgU, 2009)	Total Cost (\$-Millions, 2009)	Total Cost Discounted @ 2%/Year (\$-Millions)
2020	1,840	\$ 5,066	\$ 9,321	\$ 7,497	1,840	\$ 5,066	\$ 9,321	\$ 7,497	1,840	\$ 5,066	\$ 9,321	\$ 7,497	1,840	\$ 5,066	\$ 9,321	\$ 7,497
2021	1,840	5,066	9,321	7,350	1,840	5,066	9,321	7,350	1,840	5,066	9,321	7,350	1,840	5,066	9,321	7,350
2022	1,840	5,066	9,321	7,206	1,840	5,066	9,321	7,206	1,840	5,066	9,321	7,206	1,840	5,066	9,321	7,206
2023	1,840	5,066	9,321	7,064	1,840	5,066	9,321	7,064	1,840	5,066	9,321	7,064	1,840	5,066	9,321	7,064
2024	1,840	5,066	9,321	6,926	1,840	5,066	9,321	6,926	1,840	5,066	9,321	6,926	1,840	5,066	9,321	6,926
2025	1,840	5,066	9,321	6,790	1,840	5,066	9,321	6,790	1,840	5,066	9,321	6,790	1,840	5,066	9,321	6,790
2026	1,840	5,066	9,321	6,657	1,840	5,066	9,321	6,657	1,840	5,066	9,321	6,657	1,840	5,066	9,321	6,657
2027	1,840	5,066	9,321	6,526	1,840	5,066	9,321	6,526	1,840	5,066	9,321	6,526	1,840	5,066	9,321	6,526
2028	1,840	5,066	9,321	6,399	1,840	5,066	9,321	6,399	1,840	5,066	9,321	6,399	1,840	5,066	9,321	6,399
2029	1,840	5,066	9,321	6,273	1,840	5,066	9,321	6,273	1,840	5,066	9,321	6,273	1,840	5,066	9,321	6,273
2030	1,840	5,066	9,321	6,150	1,840	5,066	9,321	6,150	1,840	5,066	9,321	6,150	1,840	5,066	9,321	6,150
2031	1,840	5,066	9,321	6,029	1,840	5,066	9,321	6,029	1,840	5,066	9,321	6,029	1,840	5,066	9,321	6,029
2032	1,334	5,066	6,759	4,286	1,105	5,066	5,598	3,550	1,335	5,066	6,761	4,287	1,105	5,066	5,598	3,550
2033	1,334	5,066	6,759	4,202	1,105	5,066	5,598	3,480	1,335	5,066	6,761	4,203	1,105	5,066	5,598	3,480
2034	1,334	5,066	6,759	4,120	1,105	5,066	5,598	3,412	1,335	5,066	6,761	4,121	1,105	5,066	5,598	3,412
2035	1,334	5,066	6,759	4,039	1,105	5,066	5,598	3,345	1,335	5,066	6,761	4,040	1,105	5,066	5,598	3,345
2036	1,334	5,066	6,759	3,960	1,105	5,066	5,598	3,280	1,335	5,066	6,761	3,961	1,105	5,066	5,598	3,280
2037	1,334	5,066	6,759	3,882	1,105	5,066	5,598	3,215	1,335	5,066	6,761	3,883	1,105	5,066	5,598	3,215
2038	1,334	5,066	6,759	3,806	1,105	5,066	5,598	3,152	1,335	5,066	6,761	3,807	1,105	5,066	5,598	3,152
2039	1,334	5,066	6,759	3,731	1,105	5,066	5,598	3,090	1,335	5,066	6,761	3,732	1,105	5,066	5,598	3,090
2040	1,334	5,066	6,759	3,658	1,105	5,066	5,598	3,030	1,335	5,066	6,761	3,659	1,105	5,066	5,598	3,030
2041	1,334	5,066	6,759	3,587	1,105	5,066	5,598	2,970	1,335	5,066	6,761	3,587	1,105	5,066	5,598	2,970
2042	1,334	5,066	6,759	3,516	1,105	5,066	5,598	2,912	1,335	5,066	6,761	3,517	1,105	5,066	5,598	2,912
2043	1,366	5,066	6,922	3,531	1,167	5,066	5,913	3,016	1,229	5,066	6,224	3,174	1,065	5,066	5,394	2,751
2044	1,375	5,066	6,963	3,482	1,183	5,066	5,992	2,996	1,233	5,066	6,245	3,122	1,073	5,066	5,433	2,717
2045	1,383	5,066	7,004	3,434	1,198	5,066	6,071	2,976	1,237	5,066	6,265	3,071	1,080	5,066	5,473	2,683
2046	1,384	5,066	7,009	3,369	1,200	5,066	6,081	2,923	1,237	5,066	6,267	3,012	1,081	5,066	5,476	2,632
2047	1,385	5,066	7,015	3,305	1,202	5,066	6,091	2,870	1,237	5,066	6,269	2,954	1,082	5,066	5,479	2,582
2048	1,386	5,066	7,020	3,243	1,204	5,066	6,101	2,818	1,238	5,066	6,271	2,897	1,082	5,066	5,483	2,533
2049	1,387	5,066	7,025	3,182	1,206	5,066	6,111	2,768	1,238	5,066	6,273	2,841	1,083	5,066	5,486	2,485
2050	1,388	5,066	7,030	3,121	1,208	5,066	6,121	2,718	1,239	5,066	6,275	2,786	1,084	5,066	5,489	2,437
2051	1,389	5,066	7,035	3,063	1,210	5,066	6,131	2,669	1,239	5,066	6,277	2,732	1,084	5,066	5,493	2,391
2052	1,390	5,066	7,041	3,005	1,212	5,066	6,141	2,621	1,239	5,066	6,279	2,680	1,085	5,066	5,496	2,345
2053	1,391	5,066	7,046	2,948	1,214	5,066	6,151	2,573	1,240	5,066	6,281	2,628	1,085	5,066	5,499	2,301
2054	1,392	5,066	7,051	2,892	1,216	5,066	6,161	2,527	1,325	5,066	6,710	2,753	1,124	5,066	5,692	2,335
2055	1,393	5,066	7,056	2,838	1,218	5,066	6,171	2,482	1,326	5,066	6,717	2,701	1,125	5,066	5,699	2,292
2056	1,394	5,066	7,061	2,784	1,220	5,066	6,180	2,437	1,327	5,066	6,724	2,651	1,126	5,066	5,707	2,250
2057	1,395	5,066	7,067	2,732	1,222	5,066	6,190	2,393	1,328	5,066	6,727	2,600	1,127	5,066	5,710	2,207
2058	1,396	5,066	7,072	2,680	1,224	5,066	6,200	2,350	1,328	5,066	6,729	2,550	1,128	5,066	5,714	2,165
2059	1,397	5,066	7,077	2,629	1,226	5,066	6,210	2,307	1,329	5,066	6,732	2,501	1,129	5,066	5,717	2,124
2060	1,398	5,066	7,082	2,580	1,228	5,066	6,220	2,266	1,329	5,066	6,734	2,453	1,129	5,066	5,721	2,084
2061	1,399	5,066	7,087	2,531	1,230	5,066	6,230	2,225	1,330	5,066	6,737	2,406	1,130	5,066	5,725	2,044
2062	1,400	5,066	7,093	2,483	1,232	5,066	6,240	2,185	1,330	5,066	6,739	2,359	1,131	5,066	5,728	2,005
2063	1,401	5,066	7,098	2,436	1,234	5,066	6,250	2,145	1,331	5,066	6,742	2,314	1,131	5,066	5,732	1,967
2064	1,402	5,066	7,103	2,390	1,236	5,066	6,260	2,106	1,331	5,066	6,744	2,270	1,132	5,066	5,735	1,930
2065	1,403	5,066	7,108	2,345	1,238	5,066	6,270	2,068	1,268	5,066	6,424	2,119	1,094	5,066	5,543	1,829
2066	1,404	5,066	7,114	2,301	1,240	5,066	6,280	2,031	1,268	5,066	6,423	2,078	1,094	5,066	5,544	1,793
2067	1,405	5,066	7,119	2,257	1,242	5,066	6,290	1,994	1,268	5,066	6,422	2,036	1,095	5,066	5,545	1,758
2068	1,406	5,066	7,124	2,215	1,244	5,066	6,300	1,958	1,268	5,066	6,424	1,997	1,095	5,066	5,548	1,725
2069	1,407	5,066	7,129	2,173	1,245	5,066	6,310	1,923	1,269	5,066	6,427	1,959	1,096	5,066	5,552	1,692
2070	1,408	5,066	7,134	2,132	1,247	5,066	6,320	1,888	1,269	5,066	6,429	1,921	1,097	5,066	5,555	1,660
2071	1,409	5,066	7,140	2,092	1,249	5,066	6,329	1,854	1,269	5,066	6,431	1,884	1,097	5,066	5,559	1,628
2072	1,410	5,066	7,145	2,052	1,251	5,066	6,339	1,821	1,270	5,066	6,433	1,848	1,098	5,066	5,562	1,597
2073	1,411	5,066	7,150	2,013	1,253	5,066	6,349	1,788	1,270	5,066	6,435	1,812	1,099	5,066	5,565	1,567
2074	1,412															

TABLE 5-24
COSTS OF UO₂ FUEL
FOR 100 LWRs OVER 81 YEARS
(\$200/lb U₃O₈ YELLOWCAKE)

Year	UREX+3c -- 1 MOX Recycle				AFC -- 1 MOX Recycle				UREX+3c -- 2 MOX Recycles				AFC -- 2 MOX Recycles			
	Quantity (MTU)	Unit Price (\$/kgU, 2009)	Total Cost (\$-Millions, 2009)	Total Cost Discounted @ 2%/Year (\$-Millions)	Quantity (MTU)	Unit Price (\$/kgU, 2009)	Total Cost (\$-Millions, 2009)	Total Cost Discounted @ 2%/Year (\$-Millions)	Quantity (MTU)	Unit Price (\$/kgU, 2009)	Total Cost (\$-Millions, 2009)	Total Cost Discounted @ 2%/Year (\$-Millions)	Quantity (MTU)	Unit Price (\$/kgU, 2009)	Total Cost (\$-Millions, 2009)	Total Cost Discounted @ 2%/Year (\$-Millions)
2020	1,840	\$ 7,131	\$ 13,121	\$ 10,553	1,840	\$ 7,131	\$ 13,121	\$ 10,553	1,840	\$ 7,131	\$ 13,121	\$ 10,553	1,840	\$ 7,131	\$ 13,121	\$ 10,553
2021	1,840	7,131	13,121	10,346	1,840	7,131	13,121	10,346	1,840	7,131	13,121	10,346	1,840	7,131	13,121	10,346
2022	1,840	7,131	13,121	10,143	1,840	7,131	13,121	10,143	1,840	7,131	13,121	10,143	1,840	7,131	13,121	10,143
2023	1,840	7,131	13,121	9,944	1,840	7,131	13,121	9,944	1,840	7,131	13,121	9,944	1,840	7,131	13,121	9,944
2024	1,840	7,131	13,121	9,749	1,840	7,131	13,121	9,749	1,840	7,131	13,121	9,749	1,840	7,131	13,121	9,749
2025	1,840	7,131	13,121	9,558	1,840	7,131	13,121	9,558	1,840	7,131	13,121	9,558	1,840	7,131	13,121	9,558
2026	1,840	7,131	13,121	9,371	1,840	7,131	13,121	9,371	1,840	7,131	13,121	9,371	1,840	7,131	13,121	9,371
2027	1,840	7,131	13,121	9,187	1,840	7,131	13,121	9,187	1,840	7,131	13,121	9,187	1,840	7,131	13,121	9,187
2028	1,840	7,131	13,121	9,007	1,840	7,131	13,121	9,007	1,840	7,131	13,121	9,007	1,840	7,131	13,121	9,007
2029	1,840	7,131	13,121	8,830	1,840	7,131	13,121	8,830	1,840	7,131	13,121	8,830	1,840	7,131	13,121	8,830
2030	1,840	7,131	13,121	8,657	1,840	7,131	13,121	8,657	1,840	7,131	13,121	8,657	1,840	7,131	13,121	8,657
2031	1,840	7,131	13,121	8,487	1,840	7,131	13,121	8,487	1,840	7,131	13,121	8,487	1,840	7,131	13,121	8,487
2032	1,334	7,131	9,514	6,033	1,105	7,131	7,880	4,997	1,335	7,131	9,516	6,035	1,105	7,131	7,880	4,997
2033	1,334	7,131	9,514	5,915	1,105	7,131	7,880	4,899	1,335	7,131	9,516	5,917	1,105	7,131	7,880	4,899
2034	1,334	7,131	9,514	5,799	1,105	7,131	7,880	4,803	1,335	7,131	9,516	5,800	1,105	7,131	7,880	4,803
2035	1,334	7,131	9,514	5,685	1,105	7,131	7,880	4,709	1,335	7,131	9,516	5,687	1,105	7,131	7,880	4,709
2036	1,334	7,131	9,514	5,574	1,105	7,131	7,880	4,616	1,335	7,131	9,516	5,575	1,105	7,131	7,880	4,616
2037	1,334	7,131	9,514	5,465	1,105	7,131	7,880	4,526	1,335	7,131	9,516	5,466	1,105	7,131	7,880	4,526
2038	1,334	7,131	9,514	5,358	1,105	7,131	7,880	4,437	1,335	7,131	9,516	5,359	1,105	7,131	7,880	4,437
2039	1,334	7,131	9,514	5,253	1,105	7,131	7,880	4,350	1,335	7,131	9,516	5,254	1,105	7,131	7,880	4,350
2040	1,334	7,131	9,514	5,150	1,105	7,131	7,880	4,265	1,335	7,131	9,516	5,151	1,105	7,131	7,880	4,265
2041	1,334	7,131	9,514	5,049	1,105	7,131	7,880	4,181	1,335	7,131	9,516	5,050	1,105	7,131	7,880	4,181
2042	1,334	7,131	9,514	4,950	1,105	7,131	7,880	4,099	1,335	7,131	9,516	4,951	1,105	7,131	7,880	4,099
2043	1,366	7,131	9,744	4,970	1,167	7,131	8,324	4,245	1,229	7,131	8,761	4,468	1,065	7,131	7,593	3,872
2044	1,375	7,131	9,802	4,901	1,183	7,131	8,435	4,218	1,233	7,131	8,790	4,395	1,073	7,131	7,648	3,824
2045	1,383	7,131	9,859	4,833	1,198	7,131	8,546	4,189	1,237	7,131	8,819	4,323	1,080	7,131	7,704	3,777
2046	1,384	7,131	9,866	4,742	1,200	7,131	8,560	4,114	1,237	7,131	8,822	4,240	1,081	7,131	7,708	3,705
2047	1,385	7,131	9,874	4,652	1,202	7,131	8,574	4,040	1,237	7,131	8,824	4,158	1,082	7,131	7,713	3,634
2048	1,386	7,131	9,881	4,565	1,204	7,131	8,588	3,967	1,238	7,131	8,827	4,078	1,082	7,131	7,718	3,565
2049	1,387	7,131	9,888	4,478	1,206	7,131	8,602	3,896	1,238	7,131	8,830	3,999	1,083	7,131	7,722	3,497
2050	1,388	7,131	9,896	4,394	1,208	7,131	8,616	3,826	1,239	7,131	8,833	3,922	1,084	7,131	7,727	3,431
2051	1,389	7,131	9,903	4,311	1,210	7,131	8,630	3,757	1,239	7,131	8,836	3,846	1,084	7,131	7,731	3,365
2052	1,390	7,131	9,910	4,229	1,212	7,131	8,644	3,689	1,239	7,131	8,839	3,772	1,085	7,131	7,736	3,301
2053	1,391	7,131	9,918	4,150	1,214	7,131	8,658	3,622	1,240	7,131	8,842	3,699	1,085	7,131	7,741	3,239
2054	1,392	7,131	9,925	4,071	1,216	7,131	8,672	3,557	1,325	7,131	9,446	3,875	1,124	7,131	8,013	3,287
2055	1,393	7,131	9,932	3,994	1,218	7,131	8,686	3,493	1,326	7,131	9,455	3,803	1,125	7,131	8,023	3,226
2056	1,394	7,131	9,940	3,919	1,220	7,131	8,700	3,430	1,327	7,131	9,465	3,732	1,126	7,131	8,033	3,167
2057	1,395	7,131	9,947	3,845	1,222	7,131	8,714	3,368	1,328	7,131	9,469	3,660	1,127	7,131	8,038	3,107
2058	1,396	7,131	9,954	3,772	1,224	7,131	8,728	3,307	1,328	7,131	9,472	3,590	1,128	7,131	8,043	3,048
2059	1,397	7,131	9,962	3,701	1,226	7,131	8,742	3,248	1,329	7,131	9,476	3,521	1,129	7,131	8,048	2,990
2060	1,398	7,131	9,969	3,631	1,228	7,131	8,756	3,189	1,329	7,131	9,479	3,453	1,129	7,131	8,053	2,933
2061	1,399	7,131	9,976	3,563	1,230	7,131	8,770	3,132	1,330	7,131	9,483	3,386	1,130	7,131	8,058	2,877
2062	1,400	7,131	9,984	3,495	1,232	7,131	8,784	3,075	1,330	7,131	9,486	3,321	1,131	7,131	8,063	2,823
2063	1,401	7,131	9,991	3,429	1,234	7,131	8,798	3,020	1,331	7,131	9,490	3,257	1,131	7,131	8,068	2,769
2064	1,402	7,131	9,998	3,365	1,236	7,131	8,812	2,965	1,331	7,131	9,494	3,195	1,132	7,131	8,073	2,717
2065	1,403	7,131	10,006	3,301	1,238	7,131	8,826	2,912	1,268	7,131	9,043	2,983	1,094	7,131	7,803	2,574
2066	1,404	7,131	10,013	3,239	1,240	7,131	8,840	2,859	1,268	7,131	9,041	2,924	1,094	7,131	7,804	2,524
2067	1,405	7,131	10,020	3,177	1,242	7,131	8,854	2,807	1,268	7,131	9,040	2,867	1,095	7,131	7,805	2,475
2068	1,406	7,131	10,028	3,117	1,244	7,131	8,868	2,757	1,268	7,131	9,043	2,811	1,095	7,131	7,810	2,428
2069	1,407	7,131	10,035	3,059	1,245	7,131	8,882	2,707	1,269	7,131	9,046	2,757	1,096	7,131	7,815	2,382
2070	1,408	7,131	10,042	3,001	1,247	7,131	8,895	2,658	1,269	7,131	9,049	2,704	1,097	7,131	7,820	2,337
2071	1,409	7,131	10,050	2,944	1,249	7,131	8,909	2,610	1,269	7,131	9,052	2,652	1,097	7,131	7,824	2,292
2072	1,410	7,131	10,057	2,888	1,251	7,131	8,923	2,563	1,270	7,131	9,055	2,601	1,098	7,131	7,829	2,249
2073	1,411	7,131	10,064	2,834	1,253	7,131	8,937	2,517	1,270	7,131	9,059					

TABLE 5-25
COSTS OF REPROCESSING LWR FUEL (UO₂ AND MOX)
FOR 100 LWRs OVER 81 YEARS
(Single Recycle)

Year	UREX+3c/1 Recycle								AFC/1 Recycle							
	LWR Fuel Reprocessed to Produce (MTIHM)					Unit Price (\$/kgU, 2009)	Total Cost (\$-Millions, 2009)	Total Cost Discounted @ 2%/Year (\$-Millions)	LWR Fuel Reprocessed to Produce (MTIHM)					Unit Price (\$/kgU, 2009)	Total Cost (\$-Millions, 2009)	Total Cost Discounted @ 2%/Year (\$-Millions)
	MOX-1	MOX-2	FR InitCor	Recycle	Total				MOX-1	MOX-2	FR InitCor	Recycle	Total			
2020	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2021	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2022	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2023	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2024	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2025	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2026	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2027	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2028	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2029	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2030	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2031	3,000	-	-	-	3,000	\$ 931	\$ 2,793	\$ 1,807	3,000	-	-	-	3000	\$ 969	\$ 2,907	\$ 1,880
2032	3,000	-	-	-	3,000	931	2,793	1,771	3,000	-	-	-	3000	969	2,907	1,843
2033	3,000	-	-	-	3,000	931	2,793	1,736	3,000	-	-	-	3000	969	2,907	1,807
2034	3,000	-	-	-	3,000	931	2,793	1,702	3,000	-	-	-	3000	969	2,907	1,772
2035	3,000	-	-	-	3,000	931	2,793	1,669	3,000	-	-	-	3000	969	2,907	1,737
2036	3,000	-	-	-	3,000	931	2,793	1,636	3,000	-	-	-	3000	969	2,907	1,703
2037	3,000	-	-	-	3,000	931	2,793	1,604	3,000	-	-	-	3000	969	2,907	1,670
2038	3,000	-	-	-	3,000	931	2,793	1,573	3,000	-	-	-	3000	969	2,907	1,637
2039	3,000	-	-	-	3,000	931	2,793	1,542	3,000	-	-	-	3000	969	2,907	1,605
2040	3,000	-	-	-	3,000	931	2,793	1,512	3,000	-	-	-	3000	969	2,907	1,573
2041	3,000	-	-	-	3,000	931	2,793	1,482	3,000	-	-	-	3000	969	2,907	1,543
2042	2,809	-	191	-	3,000	931	2,793	1,453	2,746	-	254	-	3000	969	2,907	1,512
2043	2,761	-	239	-	3,000	931	2,793	1,425	2,682	-	318	-	3000	969	2,907	1,483
2044	2,713	-	287	-	3,000	931	2,793	1,397	2,619	-	381	-	3000	969	2,907	1,454
2045	2,707	-	287	6	3,000	931	2,793	1,369	2,611	-	381	8	3000	969	2,907	1,425
2046	2,701	-	287	12	3,000	931	2,793	1,342	2,603	-	381	16	3000	969	2,907	1,397
2047	2,695	-	287	18	3,000	931	2,793	1,316	2,595	-	381	24	3000	969	2,907	1,370
2048	2,689	-	287	24	3,000	931	2,793	1,290	2,587	-	381	32	3000	969	2,907	1,343
2049	2,683	-	287	30	3,000	931	2,793	1,265	2,579	-	381	40	3000	969	2,907	1,317
2050	2,676	-	287	37	3,000	931	2,793	1,240	2,571	-	381	48	3000	969	2,907	1,291
2051	2,670	-	287	43	3,000	931	2,793	1,216	2,563	-	381	56	3000	969	2,907	1,265
2052	2,664	-	287	49	3,000	931	2,793	1,192	2,555	-	381	64	3000	969	2,907	1,241
2053	2,658	-	287	55	3,000	931	2,793	1,169	2,547	-	381	72	3000	969	2,907	1,216
2054	2,652	-	287	61	3,000	931	2,793	1,146	2,539	-	381	80	3000	969	2,907	1,192
2055	2,646	-	287	67	3,000	931	2,793	1,123	2,531	-	381	88	3000	969	2,907	1,169
2056	2,640	-	287	73	3,000	931	2,793	1,101	2,523	-	381	96	3000	969	2,907	1,146
2057	2,634	-	287	79	3,000	931	2,793	1,080	2,515	-	381	104	3000	969	2,907	1,124
2058	2,628	-	287	85	3,000	931	2,793	1,058	2,507	-	381	112	3000	969	2,907	1,102
2059	2,622	-	287	91	3,000	931	2,793	1,038	2,499	-	381	120	3000	969	2,907	1,080
2060	2,615	-	287	98	3,000	931	2,793	1,017	2,491	-	381	128	3000	969	2,907	1,059
2061	2,609	-	287	104	3,000	931	2,793	997	2,483	-	381	136	3000	969	2,907	1,038
2062	2,603	-	287	110	3,000	931	2,793	978	2,475	-	381	144	3000	969	2,907	1,018
2063	2,597	-	287	116	3,000	931	2,793	959	2,467	-	381	152	3000	969	2,907	998
2064	2,591	-	287	122	3,000	931	2,793	940	2,459	-	381	160	3000	969	2,907	978
2065	2,585	-	287	128	3,000	931	2,793	921	2,451	-	381	168	3000	969	2,907	959
2066	2,579	-	287	134	3,000	931	2,793	903	2,443	-	381	176	3000	969	2,907	940
2067	2,573	-	287	140	3,000	931	2,793	886	2,435	-	381	184	3000	969	2,907	922
2068	2,567	-	287	146	3,000	931	2,793	868	2,427	-	381	192	3000	969	2,907	904
2069	2,561	-	287	152	3,000	931	2,793	851	2,419	-	381	200	3000	969	2,907	886
2070	2,555	-	287	159	3,000	931	2,793	835	2,411	-	381	208	3000	969	2,907	869
2071	2,548	-	287	165	3,000	931	2,793	818	2,403	-	381	216	3000	969	2,907	852
2072	2,542	-	287	171	3,000	931	2,793	802	2,395	-	381	224	3000	969	2,907	835
2073	2,536	-	287	177	3,000	931	2,793	786	2,387	-	381	232	3000	969	2,907	819
2074	2,530	-	287	183	3,000	931	2,793	771	2,379	-	381	240	3000	969	2,907	802
2075	2,524	-	287	189	3,000	931	2,793	756	2,371	-	381	248	3000	969	2,907	787
2076	2,518	-	287	195	3,000	931	2,793	741	2,363	-	381	256	3000	969	2,907	771
2077	2,512	-	287	201	3,000	931	2,793	727	2,355	-	381	264	3000	969	2,907	756
2078	2,506	-	287	207	3,000	931	2,793	712	2,347	-	381	272	3000	969	2,907	741
2079	2,500	-	287	213	3,000	931	2,793	698	2,339	-	381	280	3000	969	2,907	727
2080	2,494	-	287	220	3,000	931	2,793	685	2,331	-	381	288	3000	969	2,907	713
2081	2,487	-	287	226	3,000	931	2,793	671	2,323	-	381	296	3000	969	2,907	699
2082	2,481	-	287	232	3,000	931	2,793	658	2,315	-	381	304	3000	969	2,907	685
2083	2,475	-	287	238	3,000	931	2,793	645	2,307	-	381	312	3000	969	2,907	671
2084	2,469	-	287	244	3,000	931	2,793	632	2,299	-	381	320	3000	969	2,907	658
2085	2,654	-	96	250	3,000	931	2,793	620	2,545	-	127	328	3000	969	2,907	645
2086	1,536	-	48	256	1,840	931	1,713	373	1,440	-	64	336	1840	969	1,783	388
2087	1,578	-	0	262	1,840	931	1,713	366	1,496	-	0	344	1840	969	1,783	380
2088	1,578	-	0	262	1,840	931	1,713	358	1,496	-	0	344	1840	969	1,783	373
2089	1,578	-	0	262	1,840	931	1,713	351	1,496	-	0	344	1840	969	1,783	366
2090	1,578	-	0	262	1,840	931	1,713	344	1,496	-	0	344	1840	969	1,783	359
2091	1,578	-	0	262	1,840	931	1,713	338	1,496	-	0	344	1840	969	1,783	352
2092	1,578	-	0	262	1,840	931	1,713	331	1,496	-	0	344	1840	969	1,783	345
2093	1,578	-	0	262	1,840	931	1,713	325	1,496	-	0	344	1840	969	1,783	338
2094	1,578	-	0	262	1,840	931	1,713	318	1,496	-	0	344	1840	969	1,783	331
2095	1,578	-	0	262	1,840	931	1,713	312	1,496	-	0	344	1840	969	1,783	325
2096	1,578	-	0	262	1,840	931	1,713	306	1,496	-	0	344	1840	969	1,783	318
2097	1,578	-	0	262	1,840	931	1,713	300	1,496	-	0	344	1840	969	1,783	312
2098	1,578	-	0	262	1,840	931	1,713	294	1,496	-	0	344	1840	969	1,783	306
2099	1,578	-	0	262	1,840	931	1,713	288	1,496	-	0	344	1840	969	1,783	300
2100	1,578	-	0	262	1,840	931	1,713	283	1,496	-	0	344	1840	969	1,783	294
	171,084		12,338	9,178			\$179,311	\$ 66,020	164,160		16,398	12,042		\$186,629	\$ 68,715	

TABLE 5-26
COSTS OF REPROCESSING LWR FUEL (UO₂ AND MOX)
FOR 100 LWRs OVER 81 YEARS
(Two Recycles)

Year	UREX+3C/2 Recycles								AFC/2 Recycles							
	LWR Fuel Reprocessed to Produce (MTIHM)					Unit Price (\$/kgU, 2009)	Total Cost (\$-Millions, 2009)	Total Cost Discounted @ 2%/Year (\$-Millions)	LWR Fuel Reprocessed to Produce (MTIHM)					Unit Price (\$/kgU, 2009)	Total Cost (\$-Millions, 2009)	Total Cost Discounted @ 2%/Year (\$-Millions)
	MOX-1	MOX-2	FR InitCor	Recycle	Total				MOX-1	MOX-2	FR InitCor	Recycle	Total			
2020	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2021	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2022	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2023	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2024	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2025	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2026	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2027	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2028	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2029	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2030	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2031	3,000	-	-	-	3,000	\$ 931	\$ 2,793	\$ 1,807	3,000	-	-	-	3000	\$ 969	\$ 2,907	1,880
2032	3,000	-	-	-	3,000	931	2,793	1,771	3,000	-	-	-	3000	969	2,907	1,843
2033	3,000	-	-	-	3,000	931	2,793	1,736	3,000	-	-	-	3000	969	2,907	1,807
2034	3,000	-	-	-	3,000	931	2,793	1,702	3,000	-	-	-	3000	969	2,907	1,772
2035	3,000	-	-	-	3,000	931	2,793	1,669	3,000	-	-	-	3000	969	2,907	1,737
2036	3,000	-	-	-	3,000	931	2,793	1,636	3,000	-	-	-	3000	969	2,907	1,703
2037	3,000	-	-	-	3,000	931	2,793	1,604	3,000	-	-	-	3000	969	2,907	1,670
2038	3,000	-	-	-	3,000	931	2,793	1,573	3,000	-	-	-	3000	969	2,907	1,637
2039	3,000	-	-	-	3,000	931	2,793	1,542	3,000	-	-	-	3000	969	2,907	1,605
2040	3,000	-	-	-	3,000	931	2,793	1,512	3,000	-	-	-	3000	969	2,907	1,573
2041	3,000	-	-	-	3,000	931	2,793	1,482	3,000	-	-	-	3000	969	2,907	1,543
2042	804	2,100	96	-	3,000	931	2,793	1,453	773	2,100	127	-	3000	969	2,907	1,512
2043	780	2,100	120	-	3,000	931	2,793	1,425	741	2,100	159	-	3000	969	2,907	1,483
2044	757	2,100	143	-	3,000	931	2,793	1,397	709	2,100	191	-	3000	969	2,907	1,454
2045	754	2,100	143	2	3,000	931	2,793	1,369	707	2,100	191	3	3000	969	2,907	1,425
2046	752	2,100	143	5	3,000	931	2,793	1,342	704	2,100	191	5	3000	969	2,907	1,397
2047	749	2,100	143	7	3,000	931	2,793	1,316	701	2,100	191	8	3000	969	2,907	1,370
2048	747	2,100	143	10	3,000	931	2,793	1,290	699	2,100	191	11	3000	969	2,907	1,343
2049	745	2,100	143	12	3,000	931	2,793	1,265	696	2,100	191	13	3000	969	2,907	1,317
2050	742	2,100	143	14	3,000	931	2,793	1,240	694	2,100	191	16	3000	969	2,907	1,291
2051	740	2,100	143	17	3,000	931	2,793	1,216	691	2,100	191	18	3000	969	2,907	1,265
2052	738	2,100	143	19	3,000	931	2,793	1,192	688	2,100	191	21	3000	969	2,907	1,241
2053	2,302	563	114	21	3,000	931	2,793	1,169	2,308	541	127	24	3000	969	2,907	1,216
2054	2,316	546	114	24	3,000	931	2,793	1,146	2,327	519	127	26	3000	969	2,907	1,192
2055	2,330	530	114	26	3,000	931	2,793	1,123	2,347	497	127	29	3000	969	2,907	1,169
2056	2,330	528	114	29	3,000	931	2,793	1,101	2,346	495	127	32	3000	969	2,907	1,146
2057	2,329	526	114	31	3,000	931	2,793	1,080	2,345	493	127	34	3000	969	2,907	1,124
2058	2,328	525	114	33	3,000	931	2,793	1,058	2,345	491	127	37	3000	969	2,907	1,102
2059	2,328	523	114	36	3,000	931	2,793	1,038	2,344	489	127	40	3000	969	2,907	1,080
2060	2,327	521	114	38	3,000	931	2,793	1,017	2,343	487	127	42	3000	969	2,907	1,059
2061	2,326	520	114	40	3,000	931	2,793	997	2,342	485	127	45	3000	969	2,907	1,038
2062	2,325	518	114	43	3,000	931	2,793	978	2,341	484	127	47	3000	969	2,907	1,018
2063	2,325	516	114	45	3,000	931	2,793	959	2,341	482	127	50	3000	969	2,907	998
2064	1,227	1,611	114	48	3,000	931	2,793	940	1,204	1,615	127	53	3000	969	2,907	978
2065	1,215	1,621	114	50	3,000	931	2,793	921	1,188	1,629	127	55	3000	969	2,907	959
2066	1,203	1,631	114	52	3,000	931	2,793	903	1,172	1,643	127	58	3000	969	2,907	940
2067	1,201	1,631	114	55	3,000	931	2,793	886	1,170	1,642	127	61	3000	969	2,907	922
2068	1,199	1,630	114	57	3,000	931	2,793	868	1,167	1,642	127	63	3000	969	2,907	904
2069	1,197	1,630	114	59	3,000	931	2,793	851	1,165	1,641	127	66	3000	969	2,907	886
2070	1,195	1,629	114	62	3,000	931	2,793	835	1,163	1,641	127	69	3000	969	2,907	869
2071	1,193	1,629	114	64	3,000	931	2,793	818	1,161	1,640	127	71	3000	969	2,907	852
2072	1,191	1,628	114	67	3,000	931	2,793	802	1,159	1,640	127	74	3000	969	2,907	835
2073	1,189	1,628	114	69	3,000	931	2,793	786	1,157	1,639	127	76	3000	969	2,907	819
2074	1,188	1,627	114	71	3,000	931	2,793	771	1,155	1,638	127	79	3000	969	2,907	802
2075	1,953	859	114	74	3,000	931	2,793	756	1,948	843	127	82	3000	969	2,907	787
2076	1,960	850	114	76	3,000	931	2,793	741	1,957	832	127	84	3000	969	2,907	771
2077	1,966	842	114	78	3,000	931	2,793	727	1,965	820	127	87	3000	969	2,907	756
2078	1,965	840	114	81	3,000	931	2,793	712	1,964	819	127	90	3000	969	2,907	741
2079	1,964	839	114	83	3,000	931	2,793	698	1,963	817	127	92	3000	969	2,907	727
2080	1,963	838	114	86	3,000	931	2,793	685	1,962	816	127	95	3000	969	2,907	713
2081	1,962	837	114	88	3,000	931	2,793	671	1,961	814	127	98	3000	969	2,907	699
2082	1,961	835	114	90	3,000	931	2,793	658	1,960	813	127	100	3000	969	2,907	685
2083	1,960	834	114	93	3,000	931	2,793	645	1,958	811	127	103	3000	969	2,907	671
2084	1,958	833	114	95	3,000	931	2,793	632	1,957	810	127	105	3000	969	2,907	658
2085	1,957	831	114	97	3,000	931	2,793	620	1,956	808	127	108	3000	969	2,907	645
2086	1,419	1,367	114	100	3,000	931	2,793									

TABLE 5-27
COSTS FOR MOX FUEL FABRICATION
FOR 100 LWRs OVER 81 YEARS
(Single Recycle)

Year	UREX+3c/1 Recycle						AFC/1 Recycle					
	MOX Fuel Fabrication (MTIHM)			Unit Price (\$/kgU, 2009)	Total Cost (\$-Millions, 2009)	Total Cost Discounted @ 2%/Year (\$-Millions)	MOX Fuel Fabrication (MTIHM)			Unit Price (\$/kgU, 2009)	Total Cost (\$-Millions, 2009)	Total Cost Discounted @ 2%/Year (\$-Millions)
	MOX-1	MOX-2	Total				MOX-1	MOX-2	Total			
2020	-	-	-	-	-	-	-	-	-	-	-	-
2021	-	-	-	-	-	-	-	-	-	-	-	-
2022	-	-	-	-	-	-	-	-	-	-	-	-
2023	-	-	-	-	-	-	-	-	-	-	-	-
2024	-	-	-	-	-	-	-	-	-	-	-	-
2025	-	-	-	-	-	-	-	-	-	-	-	-
2026	-	-	-	-	-	-	-	-	-	-	-	-
2027	-	-	-	-	-	-	-	-	-	-	-	-
2028	-	-	-	-	-	-	-	-	-	-	-	-
2029	-	-	-	-	-	-	-	-	-	-	-	-
2030	-	-	-	-	-	-	-	-	-	-	-	-
2031	506	-	506	\$ 3,230	\$ 1,634	\$ 1,057	735	-	735	\$ 2,741	\$ 2,015	\$ 1,303
2032	506	-	506	3,230	1,634	1,036	735	-	735	2,741	2,015	1,278
2033	506	-	506	3,230	1,634	1,016	735	-	735	2,741	2,015	1,253
2034	506	-	506	3,230	1,634	996	735	-	735	2,741	2,015	1,228
2035	506	-	506	3,230	1,634	976	735	-	735	2,741	2,015	1,204
2036	506	-	506	3,230	1,634	957	735	-	735	2,741	2,015	1,180
2037	506	-	506	3,230	1,634	938	735	-	735	2,741	2,015	1,157
2038	506	-	506	3,230	1,634	920	735	-	735	2,741	2,015	1,134
2039	506	-	506	3,230	1,634	902	735	-	735	2,741	2,015	1,112
2040	506	-	506	3,230	1,634	884	735	-	735	2,741	2,015	1,090
2041	506	-	506	3,230	1,634	867	735	-	735	2,741	2,015	1,069
2042	474	-	474	3,230	1,530	796	673	-	673	2,741	1,844	959
2043	465	-	465	3,230	1,504	767	657	-	657	2,741	1,801	919
2044	457	-	457	3,230	1,477	739	642	-	642	2,741	1,759	879
2045	456	-	456	3,230	1,474	723	640	-	640	2,741	1,753	859
2046	455	-	455	3,230	1,471	707	638	-	638	2,741	1,748	840
2047	454	-	454	3,230	1,468	691	636	-	636	2,741	1,742	821
2048	453	-	453	3,230	1,464	676	634	-	634	2,741	1,737	802
2049	452	-	452	3,230	1,461	662	632	-	632	2,741	1,732	784
2050	451	-	451	3,230	1,458	647	630	-	630	2,741	1,726	766
2051	450	-	450	3,230	1,454	633	628	-	628	2,741	1,721	749
2052	449	-	449	3,230	1,451	619	626	-	626	2,741	1,716	732
2053	448	-	448	3,230	1,448	606	624	-	624	2,741	1,710	716
2054	447	-	447	3,230	1,444	592	622	-	622	2,741	1,705	699
2055	446	-	446	3,230	1,441	579	620	-	620	2,741	1,699	683
2056	445	-	445	3,230	1,438	567	618	-	618	2,741	1,694	668
2057	444	-	444	3,230	1,434	554	616	-	616	2,741	1,689	653
2058	443	-	443	3,230	1,431	542	614	-	614	2,741	1,683	638
2059	442	-	442	3,230	1,428	530	612	-	612	2,741	1,678	623
2060	441	-	441	3,230	1,424	519	610	-	610	2,741	1,673	609
2061	440	-	440	3,230	1,421	507	608	-	608	2,741	1,667	595
2062	439	-	439	3,230	1,418	496	606	-	606	2,741	1,662	582
2063	438	-	438	3,230	1,414	485	604	-	604	2,741	1,656	569
2064	437	-	437	3,230	1,411	475	602	-	602	2,741	1,651	556
2065	436	-	436	3,230	1,408	464	600	-	600	2,741	1,646	543
2066	435	-	435	3,230	1,404	454	598	-	598	2,741	1,640	531
2067	434	-	434	3,230	1,401	444	596	-	596	2,741	1,635	518
2068	433	-	433	3,230	1,398	435	595	-	595	2,741	1,630	507
2069	432	-	432	3,230	1,394	425	593	-	593	2,741	1,624	495
2070	431	-	431	3,230	1,391	416	591	-	591	2,741	1,619	484
2071	430	-	430	3,230	1,388	407	589	-	589	2,741	1,613	473
2072	429	-	429	3,230	1,384	398	587	-	587	2,741	1,608	462
2073	428	-	428	3,230	1,381	389	585	-	585	2,741	1,603	451
2074	427	-	427	3,230	1,378	380	583	-	583	2,741	1,597	441
2075	426	-	426	3,230	1,375	372	581	-	581	2,741	1,592	431
2076	425	-	425	3,230	1,371	364	579	-	579	2,741	1,587	421
2077	423	-	423	3,230	1,368	356	577	-	577	2,741	1,581	411
2078	422	-	422	3,230	1,365	348	575	-	575	2,741	1,576	402
2079	421	-	421	3,230	1,361	340	573	-	573	2,741	1,570	393
2080	420	-	420	3,230	1,358	333	571	-	571	2,741	1,565	384
2081	419	-	419	3,230	1,355	326	569	-	569	2,741	1,560	375
2082	418	-	418	3,230	1,351	318	567	-	567	2,741	1,554	366
2083	417	-	417	3,230	1,348	311	565	-	565	2,741	1,549	358
2084	416	-	416	3,230	1,345	305	563	-	563	2,741	1,544	350
2085	448	-	448	3,230	1,445	321	623	-	623	2,741	1,709	379
2086	259	-	259	3,230	836	182	353	-	353	2,741	967	211
2087	266	-	266	3,230	859	183	367	-	367	2,741	1,005	214
2088	266	-	266	3,230	859	180	367	-	367	2,741	1,005	210
2089	266	-	266	3,230	859	176	367	-	367	2,741	1,005	206
2090	266	-	266	3,230	859	173	367	-	367	2,741	1,005	202
2091	266	-	266	3,230	859	169	367	-	367	2,741	1,005	198
2092	266	-	266	3,230	859	166	367	-	367	2,741	1,005	194
2093	266	-	266	3,230	859	163	367	-	367	2,741	1,005	190
2094	266	-	266	3,230	859	160	367	-	367	2,741	1,005	187
2095	266	-	266	3,230	859	156	367	-	367	2,741	1,005	183
2096	266	-	266	3,230	859	153	367	-	367	2,741	1,005	179
2097	266	-	266	3,230	859	150	367	-	367	2,741	1,005	176
2098	266	-	266	3,230	859	147	367	-	367	2,741	1,005	172
2099	266	-	266	3,230	859	145	367	-	367	2,741	1,005	169
2100	266	-	266	3,230	859	142	367	-	367	2,741	1,006	166
	28,845		28,845		\$ 93,168	\$ 35,015	40,220		40,220		\$110,242	\$ 41,743

TABLE 5-28
COSTS FOR MOX FUEL FABRICATION
FOR 100 LWRs OVER 81 YEARS
(Two Recycles)

Year	UREX+3c/2 Recycles						AFC/2 Recycles					
	MOX Fuel Fabrication (MTIHM)			Unit Price (\$/kgU, 2009)	Total Cost (\$-Millions, 2009)	Total Cost Discounted @ 2%/Year (\$-Millions)	MOX Fuel Fabrication (MTIHM)			Unit Price (\$/kgU, 2009)	Total Cost (\$-Millions, 2009)	Total Cost Discounted @ 2%/Year (\$-Millions)
	MOX-1	MOX-2	Total				MOX-1	MOX-2	Total			
2020	-	-	-	-	-	-	-	-	-	-	-	-
2021	-	-	-	-	-	-	-	-	-	-	-	-
2022	-	-	-	-	-	-	-	-	-	-	-	-
2023	-	-	-	-	-	-	-	-	-	-	-	-
2024	-	-	-	-	-	-	-	-	-	-	-	-
2025	-	-	-	-	-	-	-	-	-	-	-	-
2026	-	-	-	-	-	-	-	-	-	-	-	-
2027	-	-	-	-	-	-	-	-	-	-	-	-
2028	-	-	-	-	-	-	-	-	-	-	-	-
2029	-	-	-	-	-	-	-	-	-	-	-	-
2030	-	-	-	-	-	-	-	-	-	-	-	-
2031	506	-	506	\$ 3,000	\$ 1,517	\$ 981	735	-	735	\$ 2,673	\$ 1,965	1,271
2032	506	-	506	3,000	1,517	962	735	-	735	2,673	1,965	1,246
2033	506	-	506	3,000	1,517	943	735	-	735	2,673	1,965	1,221
2034	506	-	506	3,000	1,517	924	735	-	735	2,673	1,965	1,198
2035	506	-	506	3,000	1,517	906	735	-	735	2,673	1,965	1,174
2036	506	-	506	3,000	1,517	888	735	-	735	2,673	1,965	1,151
2037	506	-	506	3,000	1,517	871	735	-	735	2,673	1,965	1,128
2038	506	-	506	3,000	1,517	854	735	-	735	2,673	1,965	1,106
2039	506	-	506	3,000	1,517	837	735	-	735	2,673	1,965	1,085
2040	506	-	506	3,000	1,517	821	735	-	735	2,673	1,965	1,063
2041	506	-	506	3,000	1,517	805	735	-	735	2,673	1,965	1,043
2042	136	476	611	3,000	1,834	954	189	586	775	2,673	2,072	1,078
2043	132	476	607	3,000	1,822	929	182	586	767	2,673	2,051	1,046
2044	127	476	603	3,000	1,810	905	174	586	760	2,673	2,031	1,015
2045	127	476	603	3,000	1,809	887	173	586	759	2,673	2,029	995
2046	127	476	603	3,000	1,808	869	172	586	758	2,673	2,027	974
2047	126	476	602	3,000	1,806	851	172	586	758	2,673	2,025	954
2048	126	476	602	3,000	1,805	834	171	586	757	2,673	2,024	935
2049	125	476	601	3,000	1,804	817	171	586	756	2,673	2,022	916
2050	125	476	601	3,000	1,803	800	170	586	756	2,673	2,020	897
2051	125	476	601	3,000	1,802	784	169	586	755	2,673	2,019	879
2052	124	476	600	3,000	1,800	768	169	586	755	2,673	2,017	861
2053	388	128	515	3,000	1,546	647	565	151	716	2,673	1,915	801
2054	390	124	514	3,000	1,542	633	570	145	715	2,673	1,911	784
2055	393	120	513	3,000	1,538	619	575	139	714	2,673	1,907	767
2056	393	120	512	3,000	1,537	606	575	138	713	2,673	1,905	751
2057	392	119	512	3,000	1,535	593	575	138	712	2,673	1,904	736
2058	392	119	511	3,000	1,534	581	574	137	711	2,673	1,902	721
2059	392	118	511	3,000	1,532	569	574	136	711	2,673	1,900	706
2060	392	118	510	3,000	1,531	557	574	136	710	2,673	1,898	691
2061	392	118	510	3,000	1,529	546	574	135	709	2,673	1,896	677
2062	392	117	509	3,000	1,528	535	574	135	709	2,673	1,894	663
2063	392	117	509	3,000	1,526	524	573	134	708	2,673	1,892	649
2064	207	365	572	3,000	1,716	577	295	451	746	2,673	1,993	671
2065	205	367	572	3,000	1,716	566	291	455	746	2,673	1,993	658
2066	203	370	572	3,000	1,717	555	287	458	745	2,673	1,993	644
2067	202	370	572	3,000	1,716	544	287	458	745	2,673	1,991	631
2068	202	369	571	3,000	1,714	533	286	458	744	2,673	1,989	618
2069	202	369	571	3,000	1,713	522	286	458	743	2,673	1,987	606
2070	201	369	571	3,000	1,712	511	285	458	743	2,673	1,985	593
2071	201	369	570	3,000	1,710	501	285	458	742	2,673	1,984	581
2072	201	369	570	3,000	1,709	491	284	457	741	2,673	1,982	569
2073	200	369	569	3,000	1,708	481	283	457	741	2,673	1,980	558
2074	200	369	569	3,000	1,707	471	283	457	740	2,673	1,978	546
2075	329	195	524	3,000	1,571	425	477	235	712	2,673	1,904	515
2076	330	193	523	3,000	1,569	416	479	232	711	2,673	1,902	505
2077	331	191	522	3,000	1,566	407	482	229	710	2,673	1,899	494
2078	331	190	522	3,000	1,565	399	481	228	710	2,673	1,897	484
2079	331	190	521	3,000	1,563	391	481	228	709	2,673	1,895	474
2080	331	190	521	3,000	1,562	383	481	228	708	2,673	1,893	464
2081	331	190	520	3,000	1,560	375	480	227	708	2,673	1,891	455
2082	330	189	520	3,000	1,559	367	480	227	707	2,673	1,889	445
2083	330	189	519	3,000	1,557	360	480	226	706	2,673	1,888	436
2084	330	189	519	3,000	1,556	352	480	226	705	2,673	1,886	427
2085	330	188	518	3,000	1,555	345	479	226	705	2,673	1,884	418
2086	239	310	549	3,000	1,647	358	343	380	723	2,673	1,933	421
2087	238	311	549	3,000	1,646	351	340	382	723	2,673	1,931	412
2088	250	312	561	3,000	1,684	352	359	384	743	2,673	1,986	415
2089	253	312	564	3,000	1,693	347	364	384	747	2,673	1,998	410
2090	60	311	371	3,000	1,114	224	84	383	468	2,673	1,250	251
2091	60	311	371	3,000	1,114	220	85	383	468	2,673	1,250	247
2092	60	311	371	3,000	1,114	215	85	383	468	2,673	1,250	242
2093	60	311	371	3,000	1,114	211	85	383	468	2,673	1,250	237
2094	60	311	371	3,000	1,114	207	85	382	468	2,673	1,250	232
2095	61	311	371	3,000	1,114	203	85	382	468	2,673	1,250	228
2096	61	310	371	3,000	1,114	199	86	382	468	2,673	1,250	223
2097	124	225	349	3,000	1,048	183	181	273	454	2,673	1,215	213
2098	125	224	349	3,000	1,047	180	183	271	454	2,673	1,214	208
2099	117	235	352	3,000	1,056	178	170	286	456	2,673	1,219	205
2100	117	235	352	3,000	1,056	174	170	286	456	2,673	1,219	201
	19,011	17,041	36,052		\$108,156	\$ 39,178	27,437	20,832	48,268		\$129,021	\$ 47,119

TABLE 5-29
COSTS FOR CONVERSION OF VIRGIN DEPLETED URANIUM TO U₃O₈ AND DISPOSAL
FOR 100 LWRs OVER 81 YEARS

Year	UREX+3c/1 Recycle				AFC/1 Recycle				UREX+3c/2 Recycles				AFC/2 Recycles			
	DU Quantity (MTU)	Unit Price (\$/kgU, 2009)	Total Cost (\$-Millions, 2009)	Total Cost Discounted @ 2%/Year (\$-Millions)	DU Quantity (MTU)	Unit Price (\$/kgU, 2009)	Total Cost (\$-Millions, 2009)	Total Cost Discounted @ 2%/Year (\$-Millions)	DU Quantity (MTU)	Unit Price (\$/kgU, 2009)	Total Cost (\$-Millions, 2009)	Total Cost Discounted @ 2%/Year (\$-Millions)	DU Quantity (MTU)	Unit Price (\$/kgU, 2009)	Total Cost (\$-Millions, 2009)	Total Cost Discounted @ 2%/Year (\$-Millions)
2020	17,127	\$ 10.40	\$ 178	\$ 143	17,127	\$ 10.40	\$ 178	\$ 143	17,127	\$ 10.40	\$ 178	\$ 143	17,127	\$ 10.40	\$ 178	\$ 143
2021	17,127	10.40	178	140	17,127	10.40	178	140	17,127	10.40	178	140	17,127	10.40	178	140
2022	17,127	10.40	178	138	17,127	10.40	178	138	17,127	10.40	178	138	17,127	10.40	178	138
2023	17,127	10.40	178	135	17,127	10.40	178	135	17,127	10.40	178	135	17,127	10.40	178	135
2024	17,127	10.40	178	132	17,127	10.40	178	132	17,127	10.40	178	132	17,127	10.40	178	132
2025	17,127	10.40	178	130	17,127	10.40	178	130	17,127	10.40	178	130	17,127	10.40	178	130
2026	17,127	10.40	178	127	17,127	10.40	178	127	17,127	10.40	178	127	17,127	10.40	178	127
2027	17,127	10.40	178	125	17,127	10.40	178	125	17,127	10.40	178	125	17,127	10.40	178	125
2028	17,127	10.40	178	122	17,127	10.40	178	122	17,127	10.40	178	122	17,127	10.40	178	122
2029	17,127	10.40	178	120	17,127	10.40	178	120	17,127	10.40	178	120	17,127	10.40	178	120
2030	17,127	10.40	178	118	17,127	10.40	178	118	17,127	10.40	178	118	17,127	10.40	178	118
2031	17,127	10.40	178	115	17,127	10.40	178	115	17,127	10.40	178	115	17,127	10.40	178	115
2032	12,419	10.40	129	82	10,285	10.40	107	68	12,422	10.40	129	82	10,285	10.40	107	68
2033	12,419	10.40	129	80	10,285	10.40	107	67	12,422	10.40	129	80	10,285	10.40	107	67
2034	12,419	10.40	129	79	10,285	10.40	107	65	12,422	10.40	129	79	10,285	10.40	107	65
2035	12,419	10.40	129	77	10,285	10.40	107	64	12,422	10.40	129	77	10,285	10.40	107	64
2036	12,419	10.40	129	76	10,285	10.40	107	63	12,422	10.40	129	76	10,285	10.40	107	63
2037	12,419	10.40	129	74	10,285	10.40	107	61	12,422	10.40	129	74	10,285	10.40	107	61
2038	12,419	10.40	129	73	10,285	10.40	107	60	12,422	10.40	129	73	10,285	10.40	107	60
2039	12,419	10.40	129	71	10,285	10.40	107	59	12,422	10.40	129	71	10,285	10.40	107	59
2040	12,419	10.40	129	70	10,285	10.40	107	58	12,422	10.40	129	70	10,285	10.40	107	58
2041	12,419	10.40	129	69	10,285	10.40	107	57	12,422	10.40	129	69	10,285	10.40	107	57
2042	12,419	10.40	129	67	10,285	10.40	107	56	12,422	10.40	129	67	10,285	10.40	107	56
2043	12,719	10.40	132	67	10,865	10.40	113	58	11,436	10.40	119	61	9,911	10.40	103	53
2044	12,794	10.40	133	67	11,010	10.40	115	57	11,473	10.40	119	60	9,983	10.40	104	52
2045	12,869	10.40	134	66	11,155	10.40	116	57	11,511	10.40	120	59	10,056	10.40	105	51
2046	12,879	10.40	134	64	11,173	10.40	116	56	11,515	10.40	120	58	10,062	10.40	105	50
2047	12,888	10.40	134	63	11,191	10.40	116	55	11,518	10.40	120	56	10,068	10.40	105	49
2048	12,898	10.40	134	62	11,210	10.40	117	54	11,522	10.40	120	55	10,074	10.40	105	48
2049	12,907	10.40	134	61	11,228	10.40	117	53	11,526	10.40	120	54	10,080	10.40	105	47
2050	12,917	10.40	134	60	11,246	10.40	117	52	11,530	10.40	120	53	10,086	10.40	105	47
2051	12,926	10.40	134	59	11,264	10.40	117	51	11,533	10.40	120	52	10,092	10.40	105	46
2052	12,936	10.40	135	57	11,283	10.40	117	50	11,537	10.40	120	51	10,098	10.40	105	45
2053	12,946	10.40	135	56	11,301	10.40	118	49	11,541	10.40	120	50	10,104	10.40	105	44
2054	12,955	10.40	135	55	11,319	10.40	118	48	12,329	10.40	128	53	10,459	10.40	109	45
2055	12,965	10.40	135	54	11,337	10.40	118	47	12,342	10.40	128	52	10,472	10.40	109	44
2056	12,974	10.40	135	53	11,356	10.40	118	47	12,355	10.40	128	51	10,485	10.40	109	43
2057	12,984	10.40	135	52	11,374	10.40	118	46	12,359	10.40	129	50	10,492	10.40	109	42
2058	12,993	10.40	135	51	11,392	10.40	118	45	12,364	10.40	129	49	10,498	10.40	109	41
2059	13,003	10.40	135	50	11,410	10.40	119	44	12,369	10.40	129	48	10,505	10.40	109	41
2060	13,013	10.40	135	49	11,429	10.40	119	43	12,373	10.40	129	47	10,511	10.40	109	40
2061	13,022	10.40	135	48	11,447	10.40	119	43	12,378	10.40	129	46	10,518	10.40	109	39
2062	13,032	10.40	136	47	11,465	10.40	119	42	12,383	10.40	129	45	10,525	10.40	109	38
2063	13,041	10.40	136	47	11,483	10.40	119	41	12,387	10.40	129	44	10,531	10.40	110	38
2064	13,051	10.40	136	46	11,502	10.40	120	40	12,392	10.40	129	43	10,538	10.40	110	37
2065	13,060	10.40	136	45	11,520	10.40	120	40	11,803	10.40	123	40	10,185	10.40	106	35
2066	13,070	10.40	136	44	11,538	10.40	120	39	11,802	10.40	123	40	10,187	10.40	106	34
2067	13,080	10.40	136	43	11,556	10.40	120	38	11,800	10.40	123	39	10,188	10.40	106	34
2068	13,089	10.40	136	42	11,575	10.40	120	37	11,804	10.40	123	38	10,194	10.40	106	33
2069	13,099	10.40	136	42	11,593	10.40	121	37	11,808	10.40	123	37	10,201	10.40	106	32
2070	13,108	10.40	136	41	11,611	10.40	121	36	11,812	10.40	123	37	10,207	10.40	106	32
2071	13,118	10.40	136	40	11,629	10.40	121	35	11,816	10.40	123	36	10,213	10.40	106	31
2072	13,127	10.40	137	39	11,648	10.40	121	35	11,820	10.40	123	35	10,219	10.40	106	31
2073	13,137	10.40	137	38	11,666	10.40	121	34	11,824	10.40	123	35	10,225	10.40	106	30
2074	13,147	10.40	137	38	11,684	10.40	122	34	11,828	10.40	123	34	10,232	10.40	106	29
2075	13,156	10.40	137	37	11,702	10.40	122	33	11,832	10.40	123	33	10,238	10.40	106	29
2076	13,166	10.40	137	36	11,721	10.40	122	32	12,251	10.40	127	34	10,496	10.40	109	29
2077	13,175	10.40	137	36	11,739	10.40	122	32	12,260	10.40	127	33	10,505	10.40	109	28
2078	13,185	10.40	137	35	11,757	10.40	122	31	12,268	10.40	128	33	10,515	10.40	109	28</

TABLE 5-30
COSTS FOR ENRICHMENT OF RECYCLE URANIUM
FOR 100 LWRs OVER 81 YEARS
(AFC Cases Only)

Year	AFC/1 Recycle					AFC/2 Recycles				
	RU Feed (MTU)	Separative Work (MTSWU/ MTU)	Unit Price (\$/SWU, 2009)	Total Cost (\$-Millions, 2009)	Total Cost Discounted @ 2%/Year (\$-Millions)	RU Feed (MTU)	Separative Work (MTSWU/ MTU)	Unit Price (\$/SWU, 2009)	Total Cost (\$-Millions, 2009)	Total Cost Discounted @ 2%/Year (\$-Millions)
2020	-	-	-	-	-	-	-	-	-	-
2021	-	-	-	-	-	-	-	-	-	-
2022	-	-	-	-	-	-	-	-	-	-
2023	-	-	-	-	-	-	-	-	-	-
2024	-	-	-	-	-	-	-	-	-	-
2025	-	-	-	-	-	-	-	-	-	-
2026	-	-	-	-	-	-	-	-	-	-
2027	-	-	-	-	-	-	-	-	-	-
2028	-	-	-	-	-	-	-	-	-	-
2029	-	-	-	-	-	-	-	-	-	-
2030	-	-	-	-	-	-	-	-	-	-
2031	2,568	0.632	\$ 245	\$ 398	\$ 257	2,568	0.632	\$ 245	\$ 398	\$ 257
2032	2,568	0.632	245	398	252	2,568	0.632	245	398	252
2033	2,568	0.632	245	398	247	2,568	0.632	245	398	247
2034	2,568	0.632	245	398	242	2,568	0.632	245	398	242
2035	2,568	0.632	245	398	238	2,568	0.632	245	398	238
2036	2,568	0.632	245	398	233	2,568	0.632	245	398	233
2037	2,568	0.632	245	398	228	2,568	0.632	245	398	228
2038	2,568	0.632	245	398	224	2,568	0.632	245	398	224
2039	2,568	0.632	245	398	220	2,568	0.632	245	398	220
2040	2,568	0.632	245	398	215	2,568	0.632	245	398	215
2041	2,568	0.632	245	398	211	2,568	0.632	245	398	211
2042	2,350	0.632	245	364	189	2,392	0.632	245	370	193
2043	2,296	0.632	245	356	181	2,365	0.632	245	366	187
2044	2,242	0.632	245	347	174	2,338	0.632	245	362	181
2045	2,235	0.632	245	346	170	2,335	0.632	245	362	177
2046	2,228	0.632	245	345	166	2,333	0.632	245	361	174
2047	2,221	0.632	245	344	162	2,331	0.632	245	361	170
2048	2,214	0.632	245	343	158	2,329	0.632	245	361	167
2049	2,207	0.632	245	342	155	2,326	0.632	245	360	163
2050	2,200	0.632	245	341	151	2,324	0.632	245	360	160
2051	2,194	0.632	245	340	148	2,322	0.632	245	360	156
2052	2,187	0.632	245	339	145	2,320	0.632	245	359	153
2053	2,180	0.632	245	338	141	2,421	0.632	245	375	157
2054	2,173	0.632	245	336	138	2,420	0.632	245	375	154
2055	2,166	0.632	245	335	135	2,418	0.632	245	374	151
2056	2,159	0.632	245	334	132	2,416	0.632	245	374	147
2057	2,153	0.632	245	333	129	2,414	0.632	245	374	144
2058	2,146	0.632	245	332	126	2,412	0.632	245	373	142
2059	2,139	0.632	245	331	123	2,409	0.632	245	373	139
2060	2,132	0.632	245	330	120	2,407	0.632	245	373	136
2061	2,125	0.632	245	329	118	2,405	0.632	245	372	133
2062	2,118	0.632	245	328	115	2,403	0.632	245	372	130
2063	2,111	0.632	245	327	112	2,401	0.632	245	372	128
2064	2,105	0.632	245	326	110	2,362	0.632	245	366	123
2065	2,098	0.632	245	325	107	2,359	0.632	245	365	121
2066	2,091	0.632	245	324	105	2,357	0.632	245	365	118
2067	2,084	0.632	245	323	102	2,354	0.632	245	365	116
2068	2,077	0.632	245	322	100	2,352	0.632	245	364	113
2069	2,070	0.632	245	321	98	2,350	0.632	245	364	111
2070	2,063	0.632	245	320	95	2,348	0.632	245	364	109
2071	2,057	0.632	245	318	93	2,346	0.632	245	363	106
2072	2,050	0.632	245	317	91	2,343	0.632	245	363	104
2073	2,043	0.632	245	316	89	2,341	0.632	245	362	102
2074	2,036	0.632	245	315	87	2,339	0.632	245	362	100
2075	2,029	0.632	245	314	85	2,362	0.632	245	366	99
2076	2,022	0.632	245	313	83	2,360	0.632	245	365	97
2077	2,016	0.632	245	312	81	2,358	0.632	245	365	95
2078	2,009	0.632	245	311	79	2,356	0.632	245	365	93
2079	2,002	0.632	245	310	77	2,354	0.632	245	364	91
2080	1,995	0.632	245	309	76	2,352	0.632	245	364	89
2081	1,988	0.632	245	308	74	2,349	0.632	245	364	87
2082	1,981	0.632	245	307	72	2,347	0.632	245	363	86
2083	1,974	0.632	245	306	71	2,345	0.632	245	363	84
2084	1,968	0.632	245	305	69	2,343	0.632	245	363	82
2085	2,178	0.632	245	337	75	2,341	0.632	245	362	80
2086	1,233	0.632	245	191	42	2,321	0.632	245	359	78
2087	1,281	0.632	245	198	42	2,318	0.632	245	359	77
2088	1,281	0.632	245	198	41	2,388	0.632	245	370	77
2089	1,281	0.632	245	198	41	2,404	0.632	245	372	76
2090	1,281	0.632	245	198	40	1,427	0.632	245	221	44
2091	1,281	0.632	245	198	39	1,427	0.632	245	221	44
2092	1,281	0.632	245	198	38	1,427	0.632	245	221	43
2093	1,281	0.632	245	198	38	1,427	0.632	245	221	42
2094	1,281	0.632	245	198	37	1,427	0.632	245	221	41
2095	1,281	0.632	245	198	36	1,427	0.632	245	221	40
2096	1,281	0.632	245	198	35	1,427	0.632	245	221	39
2097	1,281	0.632	245	198	35	1,440	0.632	245	223	39
2098	1,281	0.632	245	198	34	1,440	0.632	245	223	38
2099	1,281	0.632	245	198	33	1,438	0.632	245	223	37
2100	1,281	0.632	245	198	33	1,438	0.632	245	223	37
	140,521			\$ 21,758	\$ 8,239	157,384			\$ 24,369	\$ 8,968

TABLE 5-31
COSTS FOR FABRICATION OF INITIAL CORES FOR FAST REACTORS
USED TO CONSUME TRUs FROM LWRs
FOR 100 LWRs OVER 81 YEARS

Year	UREX+3c and AFC/1 Recycle				UREX+3c and AFC/2 Recycles			
	Initial FR Cores (MTIHM)	Unit Price (\$/kgIHM, 2009)	Total Cost (\$-Millions, 2009)	Total Cost Discounted @ 2%/Year (\$-Millions)	Initial FR Cores (MTIHM)	Unit Price (\$/kgIHM, 2009)	Total Cost (\$-Millions, 2009)	Total Cost Discounted @ 2%/Year (\$-Millions)
2020	-	-	-	-	-	-	-	-
2021	-	-	-	-	-	-	-	-
2022	-	-	-	-	-	-	-	-
2023	-	-	-	-	-	-	-	-
2024	-	-	-	-	-	-	-	-
2025	-	-	-	-	-	-	-	-
2026	-	-	-	-	-	-	-	-
2027	-	-	-	-	-	-	-	-
2028	-	-	-	-	-	-	-	-
2029	-	-	-	-	-	-	-	-
2030	-	-	-	-	-	-	-	-
2031	-	-	-	-	-	-	-	-
2032	-	-	-	-	-	-	-	-
2033	-	-	-	-	-	-	-	-
2034	-	-	-	-	-	-	-	-
2035	-	-	-	-	-	-	-	-
2036	-	-	-	-	-	-	-	-
2037	-	-	-	-	-	-	-	-
2038	-	-	-	-	-	-	-	-
2039	-	-	-	-	-	-	-	-
2040	-	-	-	-	-	-	-	-
2041	-	-	-	-	-	-	-	-
2042	61	\$ 8,859	\$ 541	\$ 281	31	\$ 13,065	\$ 399	\$ 207
2043	76	8,859	676	345	38	13,065	498	254
2044	92	8,859	811	406	46	13,065	598	299
2045	92	8,859	811	398	46	13,065	598	293
2046	92	8,859	811	390	46	13,065	598	287
2047	92	8,859	811	382	46	13,065	598	282
2048	92	8,859	811	375	46	13,065	598	276
2049	92	8,859	811	367	46	13,065	598	271
2050	92	8,859	811	360	46	13,065	598	266
2051	92	8,859	811	353	46	13,065	598	260
2052	92	8,859	811	346	46	13,065	598	255
2053	92	8,859	811	339	46	13,065	598	250
2054	92	8,859	811	333	46	13,065	598	245
2055	92	8,859	811	326	46	13,065	598	241
2056	92	8,859	811	320	46	13,065	598	236
2057	92	8,859	811	314	46	13,065	598	231
2058	92	8,859	811	307	46	13,065	598	227
2059	92	8,859	811	301	46	13,065	598	222
2060	92	8,859	811	295	46	13,065	598	218
2061	92	8,859	811	290	46	13,065	598	214
2062	92	8,859	811	284	46	13,065	598	209
2063	92	8,859	811	278	46	13,065	598	205
2064	92	8,859	811	273	46	13,065	598	201
2065	92	8,859	811	268	46	13,065	598	197
2066	92	8,859	811	262	46	13,065	598	193
2067	92	8,859	811	257	46	13,065	598	190
2068	92	8,859	811	252	46	13,065	598	186
2069	92	8,859	811	247	46	13,065	598	182
2070	92	8,859	811	242	46	13,065	598	179
2071	92	8,859	811	238	46	13,065	598	175
2072	92	8,859	811	233	46	13,065	598	172
2073	92	8,859	811	228	46	13,065	598	168
2074	92	8,859	811	224	46	13,065	598	165
2075	92	8,859	811	220	46	13,065	598	162
2076	92	8,859	811	215	46	13,065	598	159
2077	92	8,859	811	211	46	13,065	598	156
2078	92	8,859	811	207	46	13,065	598	153
2079	92	8,859	811	203	46	13,065	598	150
2080	92	8,859	811	199	46	13,065	598	147
2081	92	8,859	811	195	46	13,065	598	144
2082	92	8,859	811	191	46	13,065	598	141
2083	92	8,859	811	187	46	13,065	598	138
2084	92	8,859	811	184	46	13,065	598	135
2085	31	8,859	270	60	46	13,065	598	133
2086	15	8,859	135	29	46	13,065	598	130
2087	-	8,859	-	-	46	13,065	598	128
2088	-	8,859	-	-	15	13,065	199	42
2089	-	8,859	-	-	8	13,065	100	20
2090	-	8,859	-	-	-	13,065	-	-
2091	-	8,859	-	-	-	13,065	-	-
2092	-	8,859	-	-	-	13,065	-	-
2093	-	8,859	-	-	-	13,065	-	-
2094	-	8,859	-	-	-	13,065	-	-
2095	-	8,859	-	-	-	13,065	-	-
2096	-	8,859	-	-	-	13,065	-	-
2097	-	8,859	-	-	-	13,065	-	-
2098	-	8,859	-	-	-	13,065	-	-
2099	-	8,859	-	-	-	13,065	-	-
2100	-	8,859	-	-	-	13,065	-	-
	3,937		\$ 34,877	\$ 12,216	2,106		\$ 27,512	\$ 9,395

TABLE 5-32
COSTS FOR REPROCESSING AND REFABRICATION OF FAST REACTOR FUEL
USED TO CONSUME TRUs FROM LWRs
FOR 100 LWRs OVER 81 YEARS

Year	UREX+3c and AFC/1 Recycle				UREX+3c and AFC/2 Recycles			
	Recycled FR Fuel (MTIHM)	Unit Price (\$/kgIHM, 2009)	Total Cost (\$-Millions, 2009)	Total Cost Discounted @ 2%/Year (\$-Millions)	Recycled FR Fuel (MTIHM)	Unit Price (\$/kgIHM, 2009)	Total Cost (\$-Millions, 2009)	Total Cost Discounted @ 2%/Year (\$-Millions)
2020	-	-	-	-	-	-	-	-
2021	-	-	-	-	-	-	-	-
2022	-	-	-	-	-	-	-	-
2023	-	-	-	-	-	-	-	-
2024	-	-	-	-	-	-	-	-
2025	-	-	-	-	-	-	-	-
2026	-	-	-	-	-	-	-	-
2027	-	-	-	-	-	-	-	-
2028	-	-	-	-	-	-	-	-
2029	-	-	-	-	-	-	-	-
2030	-	-	-	-	-	-	-	-
2031	-	-	-	-	-	-	-	-
2032	-	-	-	-	-	-	-	-
2033	-	-	-	-	-	-	-	-
2034	-	-	-	-	-	-	-	-
2035	-	-	-	-	-	-	-	-
2036	-	-	-	-	-	-	-	-
2037	-	-	-	-	-	-	-	-
2038	-	-	-	-	-	-	-	-
2039	-	-	-	-	-	-	-	-
2040	-	-	-	-	-	-	-	-
2041	-	-	-	-	-	-	-	-
2042	-	-	-	-	-	-	-	-
2043	-	-	-	-	-	-	-	-
2044	-	-	-	-	-	-	-	-
2045	15	\$ 11,330	\$ 173	\$ 85	8	\$ 11,330	\$ 86	\$ 42
2046	31	11,330	346	166	15	11,330	173	83
2047	46	11,330	519	244	23	11,330	259	122
2048	61	11,330	692	319	31	11,330	346	160
2049	76	11,330	864	392	38	11,330	432	196
2050	92	11,330	1,037	461	46	11,330	519	230
2051	107	11,330	1,210	527	53	11,330	605	263
2052	122	11,330	1,383	590	61	11,330	692	295
2053	137	11,330	1,556	651	69	11,330	778	326
2054	153	11,330	1,729	709	76	11,330	864	355
2055	168	11,330	1,902	765	84	11,330	951	382
2056	183	11,330	2,075	818	92	11,330	1,037	409
2057	198	11,330	2,248	869	99	11,330	1,124	434
2058	214	11,330	2,420	917	107	11,330	1,210	459
2059	229	11,330	2,593	964	114	11,330	1,297	482
2060	244	11,330	2,766	1,008	122	11,330	1,383	504
2061	259	11,330	2,939	1,050	130	11,330	1,470	525
2062	275	11,330	3,112	1,090	137	11,330	1,556	545
2063	290	11,330	3,285	1,127	145	11,330	1,642	564
2064	305	11,330	3,458	1,164	153	11,330	1,729	582
2065	320	11,330	3,631	1,198	160	11,330	1,815	599
2066	336	11,330	3,804	1,230	168	11,330	1,902	615
2067	351	11,330	3,976	1,261	175	11,330	1,988	630
2068	366	11,330	4,149	1,290	183	11,330	2,075	645
2069	381	11,330	4,322	1,317	191	11,330	2,161	659
2070	397	11,330	4,495	1,343	198	11,330	2,248	672
2071	412	11,330	4,668	1,367	206	11,330	2,334	684
2072	427	11,330	4,841	1,390	214	11,330	2,420	695
2073	443	11,330	5,014	1,412	221	11,330	2,507	706
2074	458	11,330	5,187	1,432	229	11,330	2,593	716
2075	473	11,330	5,360	1,451	237	11,330	2,680	725
2076	488	11,330	5,532	1,468	244	11,330	2,766	734
2077	504	11,330	5,705	1,484	252	11,330	2,853	742
2078	519	11,330	5,878	1,499	259	11,330	2,939	750
2079	534	11,330	6,051	1,513	267	11,330	3,026	756
2080	549	11,330	6,224	1,526	275	11,330	3,112	763
2081	565	11,330	6,397	1,537	282	11,330	3,198	769
2082	580	11,330	6,570	1,548	290	11,330	3,285	774
2083	595	11,330	6,743	1,557	298	11,330	3,371	779
2084	610	11,330	6,916	1,566	305	11,330	3,458	783
2085	626	11,330	7,088	1,574	313	11,330	3,544	787
2086	641	11,330	7,261	1,581	320	11,330	3,631	790
2087	656	11,330	7,434	1,586	328	11,330	3,717	793
2088	656	11,330	7,434	1,555	336	11,330	3,804	796
2089	656	11,330	7,434	1,525	343	11,330	3,890	798
2090	656	11,330	7,434	1,495	351	11,330	3,976	800
2091	656	11,330	7,434	1,466	351	11,330	3,976	784
2092	656	11,330	7,434	1,437	351	11,330	3,976	769
2093	656	11,330	7,434	1,409	351	11,330	3,976	754
2094	656	11,330	7,434	1,381	351	11,330	3,976	739
2095	656	11,330	7,434	1,354	351	11,330	3,976	724
2096	656	11,330	7,434	1,327	351	11,330	3,976	710
2097	656	11,330	7,434	1,301	351	11,330	3,976	696
2098	656	11,330	7,434	1,276	351	11,330	3,976	682
2099	656	11,330	7,434	1,251	351	11,330	3,976	669
2100	656	11,330	7,434	1,226	351	11,330	3,976	656
	22,966		\$ 260,200	\$ 65,048	11,757		\$ 133,212	\$ 33,098

TABLE 5-33
COSTS OF DISPOSAL OF EXCESS RECOVERED URANIUM AND DEPLETED RECYCLED URANIUM
FOR 100 LWRs OVER 81 YEARS

Year	UREX+3c/1 Recycle				AFC/1 Recycle				UREX+3c/2 Recycles				AFC/2 Recycles			
	RUOX to Dispository (MTU)	Unit Price (\$/cu ft, 2009)	Total Cost (\$-Millions, 2009)	Total Cost Discounted @ 2%/Year (\$-Millions)	RUOX + DRUO ₂ to Dispository (MTU)	Unit Price (\$/cu ft, 2009)	Total Cost (\$-Millions, 2009)	Total Cost Discounted @ 2%/Year (\$-Millions)	RUOX to Dispository (MTU)	Unit Price (\$/cu ft, 2009)	Total Cost (\$-Millions, 2009)	Total Cost Discounted @ 2%/Year (\$-Millions)	RUOX + DRUO ₂ to Dispository (MTU)	Unit Price (\$/cu ft, 2009)	Total Cost (\$-Millions, 2009)	Total Cost Discounted @ 2%/Year (\$-Millions)
2020	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2021	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2022	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2023	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2024	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2025	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2026	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2027	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2028	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2029	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2030	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2031	2,568	\$ 75	\$ 193	\$ 125	2,055	\$ 75	\$ 154	\$ 100	2,568	\$ 75	\$ 193	\$ 125	2,055	\$ 75	\$ 154	\$ 100
2032	2,568	75	193	122	2,055	75	154	98	2,568	75	193	122	2,055	75	154	98
2033	2,568	75	193	120	2,055	75	154	96	2,568	75	193	120	2,055	75	154	96
2034	2,568	75	193	117	2,055	75	154	94	2,568	75	193	117	2,055	75	154	94
2035	2,568	75	193	115	2,055	75	154	92	2,568	75	193	115	2,055	75	154	92
2036	2,568	75	193	113	2,055	75	154	90	2,568	75	193	113	2,055	75	154	90
2037	2,568	75	193	111	2,055	75	154	89	2,568	75	193	111	2,055	75	154	89
2038	2,568	75	193	108	2,055	75	154	87	2,568	75	193	108	2,055	75	154	87
2039	2,568	75	193	106	2,055	75	154	85	2,568	75	193	106	2,055	75	154	85
2040	2,568	75	193	104	2,055	75	154	83	2,568	75	193	104	2,055	75	154	83
2041	2,568	75	193	102	2,055	75	154	82	2,568	75	193	102	2,055	75	154	82
2042	2,526	75	189	99	2,061	75	155	80	2,467	75	185	96	1,991	75	149	78
2043	2,516	75	189	96	2,062	75	155	79	2,462	75	185	94	1,991	75	149	76
2044	2,505	75	188	94	2,064	75	155	77	2,457	75	184	92	1,992	75	149	75
2045	2,504	75	188	92	2,064	75	155	76	2,456	75	184	90	1,992	75	149	73
2046	2,503	75	188	90	2,064	75	155	74	2,456	75	184	89	1,992	75	149	72
2047	2,501	75	188	88	2,064	75	155	73	2,455	75	184	87	1,992	75	149	70
2048	2,500	75	188	87	2,065	75	155	72	2,455	75	184	85	1,992	75	149	69
2049	2,499	75	187	85	2,065	75	155	70	2,454	75	184	83	1,993	75	149	68
2050	2,497	75	187	83	2,065	75	155	69	2,454	75	184	82	1,993	75	149	66
2051	2,496	75	187	81	2,065	75	155	67	2,453	75	184	80	1,993	75	149	65
2052	2,495	75	187	80	2,065	75	155	66	2,453	75	184	79	1,993	75	149	64
2053	2,493	75	187	78	2,066	75	155	65	2,507	75	188	79	2,026	75	152	64
2054	2,492	75	187	77	2,066	75	155	64	2,507	75	188	77	2,026	75	152	62
2055	2,491	75	187	75	2,066	75	155	62	2,507	75	188	76	2,027	75	152	61
2056	2,489	75	187	74	2,066	75	155	61	2,506	75	188	74	2,027	75	152	60
2057	2,488	75	187	72	2,066	75	155	60	2,506	75	188	73	2,027	75	152	59
2058	2,487	75	187	71	2,066	75	155	59	2,505	75	188	71	2,027	75	152	58
2059	2,486	75	186	69	2,067	75	155	58	2,504	75	188	70	2,026	75	152	56
2060	2,484	75	186	68	2,067	75	155	56	2,504	75	188	68	2,026	75	152	55
2061	2,483	75	186	66	2,067	75	155	55	2,503	75	188	67	2,026	75	152	54
2062	2,482	75	186	65	2,067	75	155	54	2,502	75	188	66	2,026	75	152	53
2063	2,480	75	186	64	2,067	75	155	53	2,502	75	188	64	2,026	75	152	52
2064	2,479	75	186	63	2,068	75	155	52	2,460	75	184	62	1,989	75	149	50
2065	2,478	75	186	61	2,068	75	155	51	2,458	75	184	61	1,989	75	149	49
2066	2,476	75	186	60	2,068	75	155	50	2,457	75	184	60	1,988	75	149	48
2067	2,475	75	186	59	2,068	75	155	49	2,457	75	184	58	1,988	75	149	47
2068	2,474	75	186	58	2,068	75	155	48	2,456	75	184	57	1,988	75	149	46
2069	2,472	75	185	57	2,068	75	155	47	2,455	75	184	56	1,988	75	149	45
2070	2,471	75	185	55	2,069	75	155	46	2,455	75	184	55	1,987	75	149	45
2071	2,470	75	185	54	2,069	75	155	45	2,454	75	184	54	1,987	75	149	44
2072	2,468	75	185	53	2,069	75	155	45	2,453	75	184	53	1,987	75	149	43
2073	2,467	75	185	52	2,069	75	155	44	2,453	75	184	52	1,987	75	149	42
2074	2,466	75	185	51	2,069	75	155	43	2,452	75	184	51	1,987	75	149	41
2075	2,464	75	185	50	2,070	75	155	42	2,480	75	186	50	2,012	75	151	41
2076	2,463	75	185	49	2,070	75	155	41	2,480	75	186	49	2,012	75	151	40
2077	2,462	75	185	48	2,070	75	155	40	2,480	75	186	48	2,012	75	151	39
2078	2,460	75	185	47	2,070	75	155	40	2,479	75	186	47	2,012	75	151	38
2079	2,459	75	184	46	2,070	75	155	39	2,478	75	186	46	2,012	75	151	38
2080	2,458	75	184	45	2,071	75	155	38	2,478	75	186	46	2,012	75	151	37
2081	2,456	75	184	44	2,071	75	155	37	2,477	75	186	45	2,012	75	151	36
2082	2,455	75	184	43	2,071	75	155	37	2,477	75	186	44	2,011	75	151	36
2083	2,454	75	184	43	2,071	75	155	36	2,476	75	186	43	2,011	75	151	35
2084	2,452	75	184	42	2,071	75	155	35	2,475	75	186	42	2,011	75	151	34
2085	2,493	75	187	42	2,066	75	155	34	2,475	75	186	41	2,011	75	151	33
2086	1,509	75	113	25	1,270	75	95	21	2,454	75	184	40	1,993	75	149	33
2087	1,518	75	114	24	1,268	75	95	20	2,453	75	184	39	1,993	75	149	32
2088	1,518	75	114	24	1,268	75	95	20	2,474	75	186	39	1,999	75	150	31
2089	1,518	75	114	23	1,268	75	95	20	2,479	75	186	38	2,000	75	150	31
2090	1,518	75	114	23	1,268	75	95	19	1,491	75	112	22	1,207	75	91	18
2091	1,518	75	114	22	1,268	75	95	19	1,491	75	112	22	1,207	75	91	18
2092	1,518	75	114	22	1,268	75	95	18	1,491	75	112	22	1,207	75	91	17
2093	1,518	75	114	22	1,268	75	95	18	1,491	75	112	21	1,207	75	91	17
2094	1,518	75	114	21	1,268	75	95	18	1,491	75	112	21	1,207	75	91	17
2095	1,518	75	114	21	1,268	75	95	17	1,491	75	112	20	1,207	75	91	16
2096	1,518	75	114	20	1,268	75	95	17	1,491	75	112	20	1,207	75	91	16
2097	1,518	75	114	20	1,268	75	95	17	1,505	75	113	20	1,220	75	91	16
2098	1,518	75	114	20	1,268	75	95	16	1,505	75	113	19	1,220	75	91	16
2099	1,518	75	114	19	1,268	75	95	16	1,504	75	113	19	1,218	75	91	15
2100	1,518	75	114	19	1,268	75	95	16	1,503	75	113	19	1,218	75	91	15
	160,175		\$ 12,013	\$ 4,444	132,590		\$ 9,944	\$ 3,658	163,402		\$ 12,255	\$ 4,487	132,110		\$ 9,908	\$ 3,623

TABLE 5-34
COSTS FOR LLW DISPOSAL
FOR 100 LWRs AND ASSOCIATED FAST REACTORS OVER 81 YEARS

Year	UREX+3c/1 Recycle				AFC/1 Recycle				UREX+3c/2 Recycles				AFC/2 Recycles			
	Total LLW (cu ft)	Unit Price (\$/cu ft, 2009)	Total Cost (\$-Millions, 2009)	Total Cost Discounted @ 2%/Year (\$-Millions)	Total LLW (cu ft)	Unit Price (\$/cu ft, 2009)	Total Cost (\$-Millions, 2009)	Total Cost Discounted @ 2%/Year (\$-Millions)	Total LLW (cu ft)	Unit Price (\$/cu ft, 2009)	Total Cost (\$-Millions, 2009)	Total Cost Discounted @ 2%/Year (\$-Millions)	Total LLW (cu ft)	Unit Price (\$/cu ft, 2009)	Total Cost (\$-Millions, 2009)	Total Cost Discounted @ 2%/Year (\$-Millions)
2020	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2021	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2022	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2023	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2024	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2025	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2026	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2027	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2028	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2029	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2030	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2031	157,083	\$ 312	\$ 49	\$ 32	147,851	\$ 312	\$ 46	\$ 30	157,083	\$ 312	\$ 49	\$ 32	147,851	\$ 312	\$ 46	\$ 30
2032	157,083	312	49	31	147,851	312	46	29	157,083	312	49	31	147,851	312	46	29
2033	157,083	312	49	30	147,851	312	46	29	157,083	312	49	30	147,851	312	46	29
2034	157,083	312	49	30	147,851	312	46	28	157,083	312	49	30	147,851	312	46	28
2035	157,083	312	49	29	147,851	312	46	28	157,083	312	49	29	147,851	312	46	28
2036	157,083	312	49	29	147,851	312	46	27	157,083	312	49	29	147,851	312	46	27
2037	157,083	312	49	28	147,851	312	46	26	157,083	312	49	28	147,851	312	46	26
2038	157,083	312	49	28	147,851	312	46	26	157,083	312	49	28	147,851	312	46	26
2039	157,083	312	49	27	147,851	312	46	25	157,083	312	49	27	147,851	312	46	25
2040	157,083	312	49	27	147,851	312	46	25	157,083	312	49	27	147,851	312	46	25
2041	157,083	312	49	26	147,851	312	46	24	157,083	312	49	26	147,851	312	46	24
2042	156,333	312	49	25	147,957	312	46	24	155,272	312	48	25	146,695	312	46	24
2043	156,145	312	49	25	147,984	312	46	24	155,178	312	48	25	146,708	312	46	23
2044	155,957	312	49	24	148,010	312	46	23	155,084	312	48	24	146,721	312	46	23
2045	155,934	312	49	24	148,014	312	46	23	155,075	312	48	24	146,723	312	46	22
2046	155,910	312	49	23	148,017	312	46	22	155,066	312	48	23	146,724	312	46	22
2047	155,886	312	49	23	148,020	312	46	22	155,056	312	48	23	146,725	312	46	22
2048	155,862	312	49	22	148,024	312	46	21	155,047	312	48	22	146,726	312	46	21
2049	155,838	312	49	22	148,027	312	46	21	155,038	312	48	22	146,727	312	46	21
2050	155,814	312	49	22	148,030	312	46	21	155,028	312	48	21	146,728	312	46	20
2051	155,790	312	49	21	148,034	312	46	20	155,019	312	48	21	146,729	312	46	20
2052	155,766	312	49	21	148,037	312	46	20	155,010	312	48	21	146,730	312	46	20
2053	155,742	312	49	20	148,040	312	46	19	155,986	312	49	20	147,329	312	46	19
2054	155,718	312	49	20	148,044	312	46	19	155,985	312	49	20	147,338	312	46	19
2055	155,694	312	49	20	148,047	312	46	19	155,984	312	49	20	147,347	312	46	18
2056	155,670	312	49	19	148,050	312	46	18	155,972	312	49	19	147,345	312	46	18
2057	155,646	312	49	19	148,054	312	46	18	155,961	312	49	19	147,342	312	46	18
2058	155,622	312	49	18	148,057	312	46	18	155,950	312	49	18	147,339	312	46	17
2059	155,599	312	49	18	148,060	312	46	17	155,938	312	49	18	147,337	312	46	17
2060	155,575	312	49	18	148,064	312	46	17	155,927	312	49	18	147,334	312	46	17
2061	155,551	312	49	17	148,067	312	46	16	155,915	312	49	17	147,332	312	46	16
2062	155,527	312	49	17	148,070	312	46	16	155,904	312	49	17	147,329	312	46	16
2063	155,503	312	49	17	148,074	312	46	16	155,893	312	49	17	147,326	312	46	16
2064	155,479	312	49	16	148,077	312	46	16	155,131	312	48	16	146,670	312	46	15
2065	155,455	312	49	16	148,080	312	46	15	155,112	312	48	16	146,658	312	46	15
2066	155,431	312	48	16	148,084	312	46	15	155,093	312	48	16	146,647	312	46	15
2067	155,407	312	48	15	148,087	312	46	15	155,080	312	48	15	146,643	312	46	15
2068	155,383	312	48	15	148,090	312	46	14	155,068	312	48	15	146,640	312	46	14
2069	155,359	312	48	15	148,094	312	46	14	155,056	312	48	15	146,637	312	46	14
2070	155,335	312	48	14	148,097	312	46	14	155,044	312	48	14	146,633	312	46	14
2071	155,311	312	48	14	148,100	312	46	14	155,032	312	48	14	146,630	312	46	13
2072	155,288	312	48	14	148,104	312	46	13	155,020	312	48	14	146,627	312	46	13
2073	155,264	312	48	14	148,107	312	46	13	155,007	312	48	14	146,623	312	46	13
2074	155,240	312	48	13	148,110	312	46	13	154,995	312	48	13	146,620	312	46	13
2075	155,216	312	48	13	148,114	312	46	13	155,508	312	49	13	147,074	312	46	12
2076	155,192	312	48	13	148,117	312	46	12	155,501	312	49	13	147,077	312	46	12
2077	155,168	312	48	13	148,120	312	46	12	155,495	312	49	13	147,080	312	46	12
2078	155,144	312	48	12	148,124	312	46	12	155,483	312	49	12	147,077	312	46	12
2079	155,120	312	48	12	148,127	312	46	12	155,472	312	49	12	147,074	312	46	11
2080	155,096	312	48	12	148,130	312	46	11	155,460	312	49	12	147,072	312	46	11
2081	155,072	312	48	12	148,134	312	46	11	155,448	312	48	12	147,069	312	46	11
2082	155,048	312	48	11	148,137	312	46	11	155,437	312	48	11	147,066	312	46	11
2083	155,024	312	48	11	148,140	312	46	11	155,425	312	48	11	147,063	312	46	11
2084	155,000	312	48	11	148,144	312	46	10	155,413	312	48	11	147,060	312	46	10
2085	155,727	312	49	11	148,041	312	46	10	155,402	312	48	11	147,057	312	46	10
2086	95,152	312	30	6	90,849	312	28	6	155,023	312	48	11	146,734	312	46	10
2087	95,316	312	30	6	90,826	312	28	6	155,007	312	48	10	146,727	312	46	10
2088	95,316	312	30	6	90,826	312	28	6	155,391	312	48	10	146,838	312	46	10
2089	95,316	312	30	6	90,826	312	28	6	155,479	312	49	10	146,864	312	46	9
2090	95,316	312	30	6	90,826	312	28	6	94,828	312	30	6	89,722	312	28	6
2091	95,316	312	30	6	90,826	312	28	6	94,829	312	30	6	89,722	312	28	6
2092	95,316	312	30	6	90,826	312	28	5	94,829	312	30	6	89,723	312	28	5
2093	95,316	312	30	6	90,826	312	28	5	94,830	312	30	6	89,723	312	28	5
2094	95,316	312	30	6	90,826	312	28	5	94,830	312	30	5	89,724	312	28	5
2095	95,316	312	30	5	90,826	312	28	5	94,831	312	30	5	89,724	312	28	5
2096	95,316	312	30	5	90,826	312	28	5	94,831	312	30	5	89,725	312	28	5
2097	95,316	312	30	5	90,826	312	28	5	95,089	312	30	5	89,949	312	28	5
2098	95,316	312	30	5	90,826	312	28	5	95,092	312	30	5	89,953	312	28	5
2099	95,316	312	30	5	90,826	312	28	5	95,059	312	30	5	89,922	312	28	5
2100	95,316	312	30	5	90,826	312	28	5	95,051	312	30	5	89,914	312	28	5
	10,000,325		\$ 3,120	\$ 1,150	9,503,907		\$ 2,965	\$ 1,092	10,229,887		\$ 3,192	\$ 1,165	9,666,759		\$ 3,016	\$ 1,100

TABLE 5-35
COSTS FOR TRU WASTE DISPOSAL
FOR 100 LWRs AND ASSOCIATED FAST REACTORS OVER 81 YEARS

Year	UREX+3c/1 Recycle				AFC/1 Recycle				UREX+3c/2 Recycles				AFC/2 Recycles			
	Total TRU (cu ft)	Unit Price (\$/cu ft, 2009)	Total Cost (\$-Millions, 2009)	Total Cost Discounted @ 2%/Year (\$-Millions)	Total TRU (cu ft)	Unit Price (\$/cu ft, 2009)	Total Cost (\$-Millions, 2009)	Total Cost Discounted @ 2%/Year (\$-Millions)	Total TRU (cu ft)	Unit Price (\$/cu ft, 2009)	Total Cost (\$-Millions, 2009)	Total Cost Discounted @ 2%/Year (\$-Millions)	Total TRU (cu ft)	Unit Price (\$/cu ft, 2009)	Total Cost (\$-Millions, 2009)	Total Cost Discounted @ 2%/Year (\$-Millions)
2020	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2021	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2022	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2023	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2024	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2025	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2026	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2027	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2028	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2029	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2030	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2031	113,665	\$ 1,150	\$ 131	\$ 85	143,007	\$ 1,150	\$ 164	\$ 106	113,627	\$ 1,150	\$ 131	\$ 85	143,007	\$ 1,150	\$ 164	\$ 106
2032	113,665	1,150	131	83	143,007	1,150	164	104	113,627	1,150	131	83	143,007	1,150	164	104
2033	113,665	1,150	131	81	143,007	1,150	164	102	113,627	1,150	131	81	143,007	1,150	164	102
2034	113,665	1,150	131	80	143,007	1,150	164	100	113,627	1,150	131	80	143,007	1,150	164	100
2035	113,665	1,150	131	78	143,007	1,150	164	98	113,627	1,150	131	78	143,007	1,150	164	98
2036	113,665	1,150	131	77	143,007	1,150	164	96	113,627	1,150	131	77	143,007	1,150	164	96
2037	113,665	1,150	131	75	143,007	1,150	164	94	113,627	1,150	131	75	143,007	1,150	164	94
2038	113,665	1,150	131	74	143,007	1,150	164	93	113,627	1,150	131	74	143,007	1,150	164	93
2039	113,665	1,150	131	72	143,007	1,150	164	91	113,627	1,150	131	72	143,007	1,150	164	91
2040	113,665	1,150	131	71	143,007	1,150	164	89	113,627	1,150	131	71	143,007	1,150	164	89
2041	113,665	1,150	131	69	143,007	1,150	164	87	113,627	1,150	131	69	143,007	1,150	164	87
2042	117,350	1,150	135	70	142,847	1,150	164	85	131,090	1,150	151	78	152,068	1,150	175	91
2043	118,272	1,150	136	69	142,807	1,150	164	84	131,551	1,150	151	77	152,048	1,150	175	89
2044	119,193	1,150	137	69	142,767	1,150	164	82	132,012	1,150	152	76	152,028	1,150	175	87
2045	121,264	1,150	139	68	144,719	1,150	166	82	133,062	1,150	153	75	153,046	1,150	176	86
2046	123,334	1,150	142	68	146,670	1,150	169	81	134,112	1,150	154	74	154,065	1,150	177	85
2047	125,405	1,150	144	68	148,621	1,150	171	81	135,162	1,150	155	73	155,083	1,150	178	84
2048	127,476	1,150	147	68	150,573	1,150	173	80	136,212	1,150	157	72	156,102	1,150	180	83
2049	129,547	1,150	149	67	152,524	1,150	175	79	137,262	1,150	158	71	157,120	1,150	181	82
2050	131,617	1,150	151	67	154,475	1,150	178	79	138,311	1,150	159	71	158,139	1,150	182	81
2051	133,688	1,150	154	67	156,427	1,150	180	78	139,361	1,150	160	70	159,157	1,150	183	80
2052	135,759	1,150	156	67	158,378	1,150	182	78	140,411	1,150	161	69	160,176	1,150	184	79
2053	137,829	1,150	159	66	160,330	1,150	184	77	130,668	1,150	150	63	156,392	1,150	180	75
2054	139,900	1,150	161	66	162,281	1,150	187	77	131,593	1,150	151	62	157,313	1,150	181	74
2055	141,971	1,150	163	66	164,232	1,150	189	76	132,518	1,150	152	61	158,235	1,150	182	73
2056	144,041	1,150	166	65	166,184	1,150	191	75	133,556	1,150	154	61	159,245	1,150	183	72
2057	146,112	1,150	168	65	168,135	1,150	193	75	134,593	1,150	155	60	160,256	1,150	184	71
2058	148,183	1,150	170	65	170,086	1,150	196	74	135,631	1,150	156	59	161,266	1,150	185	70
2059	150,253	1,150	173	64	172,038	1,150	198	74	136,668	1,150	157	58	162,277	1,150	187	69
2060	152,324	1,150	175	64	173,989	1,150	200	73	137,706	1,150	158	58	163,287	1,150	188	68
2061	154,395	1,150	178	63	175,940	1,150	202	72	138,743	1,150	160	57	164,298	1,150	189	67
2062	156,465	1,150	180	63	177,892	1,150	205	72	139,781	1,150	161	56	165,308	1,150	190	67
2063	158,536	1,150	182	63	179,843	1,150	207	71	140,818	1,150	162	56	166,319	1,150	191	66
2064	160,607	1,150	185	62	181,794	1,150	209	70	150,012	1,150	173	58	172,272	1,150	198	67
2065	162,677	1,150	187	62	183,746	1,150	211	70	151,137	1,150	174	57	173,350	1,150	199	66
2066	164,748	1,150	189	61	185,697	1,150	214	69	152,261	1,150	175	57	174,428	1,150	201	65
2067	166,819	1,150	192	61	187,648	1,150	216	68	153,307	1,150	176	56	175,444	1,150	202	64
2068	168,889	1,150	194	60	189,600	1,150	218	68	154,353	1,150	178	55	176,460	1,150	203	63
2069	170,960	1,150	197	60	191,551	1,150	220	67	155,400	1,150	179	54	177,476	1,150	204	62
2070	173,031	1,150	199	59	193,502	1,150	223	66	156,446	1,150	180	54	178,493	1,150	205	61
2071	175,101	1,150	201	59	195,454	1,150	225	66	157,492	1,150	181	53	179,509	1,150	206	60
2072	177,172	1,150	204	59	197,405	1,150	227	65	158,538	1,150	182	52	180,525	1,150	208	60
2073	179,243	1,150	206	58	199,356	1,150	229	65	159,584	1,150	184	52	181,541	1,150	209	59
2074	181,313	1,150	209	58	201,308	1,150	232	64	160,630	1,150	185	51	182,557	1,150	210	58
2075	183,384	1,150	211	57	203,259	1,150	234	63	155,967	1,150	179	49	180,113	1,150	207	56
2076	185,455	1,150	213	57	205,210	1,150	236	63	156,952	1,150	180	48	181,082	1,150	208	55
2077	187,526	1,150	216	56	207,162	1,150	238	62	157,937	1,150	182	47	182,050	1,150	209	54
2078	189,596	1,150	218	56	209,113	1,150	240	61	158,977	1,150	183	47	183,062	1,150	211	54
2079	191,667	1,150	220	55	211,064	1,150	243	61	160,018	1,150	184	46	184,075	1,150	212	53
2080	193,738	1,150	223	55	213,016	1,150	245	60	161,058	1,150	185	45	185,087	1,150	213	52
2081	195,808	1,150	225	54	214,967	1,150	247	59	162,098	1,150	186	45	186,099	1,150	214	51
2082	197,879	1,150	228	54	216,918	1,150	249	59	163,138	1,150	188	44	187,111	1,150	215	51
2083	199,950	1,150	230	53	218,870	1,150	252	58	164,178	1,150	189	44	188,123	1,150	216	50
2084	202,020	1,150	232	53	220,821	1,150	254	58	165,218	1,150	190	43	189,135	1,150	218	49
2085	200,406	1,150	230	51	222,932	1,150	256	57	166,258	1,150	191	42	190,147	1,150	219	49
2086	157,604	1,150	181	39	169,628	1,150	195	42	171,295	1,150	197	43	193,581	1,150	223	48
2087	158,754	1,150	183	39	171,619	1,150	197	42	172,377	1,150	198	42	194,627	1,150	224	48
2088	158,754	1,150	183	38	171,619	1,150	197	41	171,191	1,150	197	41	194,430	1,150	224	47
2089	158,754	1,150	183	37	171,619	1,150	197	40	171,668	1,150	197	40	195,134	1,150	224	46
2090	158,754	1,150	183	37	171,619	1,150	197	40	128,209	1,150	147	30	140,543	1,150	162	33
2091	158,754	1,150	183	36	171,619	1,150	197	39	128,204	1,150	147	29	140,539	1,150	162	32
2092	158,754	1,150	183	35	171,619	1,150	197	38	128,198	1,150	147	28	140,535	1,150	162	31
2093	158,754	1,150	183	35	171,619	1,150	197	37	128,192	1,150	147	28	140,532	1,150	162	31
2094	158,754	1,150	183	34	171,619	1,150	197	37	128,187	1,150	147	27	140,528	1,150	162	30
2095	158,754	1,150	183	33	171,619	1,150	197	36	128,181	1,150	147	27	140,525	1,150	162	29
2096	158,754	1,150	183	33	171,619	1,150	197	35	128,176	1,150	147	26	140,521	1,150	162	29
2097	158,754	1,150	183	32	171,619											

TABLE 5-36
COSTS OF HIGH LEVEL WASTE DISPOSAL
FOR 100 LWRs OVER 81 YEARS
(All Cases)

Year	LWR Electric Generation ^a (kWH- Millions)	NWPA Fee ^b (\$/kWH, 2009)	Total Cost (\$-Millions, 2009)	Total Cost Discounted @ 2%/Year (\$-Millions)
2020	788,400	\$ 0.00055	\$ 434	\$ 349
2021	788,400	0.00055	434	342
2022	788,400	0.00055	434	335
2023	788,400	0.00055	434	329
2024	788,400	0.00055	434	322
2025	788,400	0.00055	434	316
2026	788,400	0.00055	434	310
2027	788,400	0.00055	434	304
2028	788,400	0.00055	434	298
2029	788,400	0.00055	434	292
2030	788,400	0.00055	434	286
2031	788,400	0.00055	434	280
2032	788,400	0.00055	434	275
2033	788,400	0.00055	434	270
2034	788,400	0.00055	434	264
2035	788,400	0.00055	434	259
2036	788,400	0.00055	434	254
2037	788,400	0.00055	434	249
2038	788,400	0.00055	434	244
2039	788,400	0.00055	434	239
2040	788,400	0.00055	434	235
2041	788,400	0.00055	434	230
2042	788,400	0.00055	434	226
2043	788,400	0.00055	434	221
2044	788,400	0.00055	434	217
2045	788,400	0.00055	434	213
2046	788,400	0.00055	434	208
2047	788,400	0.00055	434	204
2048	788,400	0.00055	434	200
2049	788,400	0.00055	434	196
2050	788,400	0.00055	434	193
2051	788,400	0.00055	434	189
2052	788,400	0.00055	434	185
2053	788,400	0.00055	434	181
2054	788,400	0.00055	434	178
2055	788,400	0.00055	434	174
2056	788,400	0.00055	434	171
2057	788,400	0.00055	434	168
2058	788,400	0.00055	434	164
2059	788,400	0.00055	434	161
2060	788,400	0.00055	434	158
2061	788,400	0.00055	434	155
2062	788,400	0.00055	434	152
2063	788,400	0.00055	434	149
2064	788,400	0.00055	434	146
2065	788,400	0.00055	434	143
2066	788,400	0.00055	434	140
2067	788,400	0.00055	434	137
2068	788,400	0.00055	434	135
2069	788,400	0.00055	434	132
2070	788,400	0.00055	434	130
2071	788,400	0.00055	434	127
2072	788,400	0.00055	434	125
2073	788,400	0.00055	434	122
2074	788,400	0.00055	434	120
2075	788,400	0.00055	434	117
2076	788,400	0.00055	434	115
2077	788,400	0.00055	434	113
2078	788,400	0.00055	434	111
2079	788,400	0.00055	434	108
2080	788,400	0.00055	434	106
2081	788,400	0.00055	434	104
2082	788,400	0.00055	434	102
2083	788,400	0.00055	434	100
2084	788,400	0.00055	434	98
2085	788,400	0.00055	434	96
2086	788,400	0.00055	434	94
2087	788,400	0.00055	434	93
2088	788,400	0.00055	434	91
2089	788,400	0.00055	434	89
2090	788,400	0.00055	434	87
2091	788,400	0.00055	434	85
2092	788,400	0.00055	434	84
2093	788,400	0.00055	434	82
2094	788,400	0.00055	434	81
2095	788,400	0.00055	434	79
2096	788,400	0.00055	434	77
2097	788,400	0.00055	434	76
2098	788,400	0.00055	434	74
2099	788,400	0.00055	434	73
2100	788,400	0.00055	434	72
	63,860,400		\$ 35,123	\$ 14,209

^a7,884-million kWH/year/reactor; 100 1,000 MWe reactors; 90% plant factor
^b55% of the 1-mill/kWH fee stipulated by NWPA, due to reduced volume of waste

TABLE 5-37
COSTS OF HIGH LEVEL WASTE DISPOSAL FOR FAST REACTORS USED TO CONSUME TRUs FROM LWRs
FOR 100 LWRs OVER 81 YEARS

Year	UREX+3c AND AFC/1 Recycle					UREX+3c AND AFC/2 Recycles				
	Cum. No. Operating FRs	FR Electric Generation ^a (kWH- Millions)	NWPA Fee ^b (\$/kWH, 2009)	Total Cost (\$-Millions, 2009)	Total Cost Discounted @ 2%/Year (\$-Millions)	Cum. No. Operating FRs	FR Electric Generation ^a (kWH- Millions)	NWPA Fee ^b (\$/kWH, 2009)	Total Cost (\$-Millions, 2009)	Total Cost Discounted @ 2%/Year (\$-Millions)
2020	-	-	-	-	-	-	-	-	-	-
2021	-	-	-	-	-	-	-	-	-	-
2022	-	-	-	-	-	-	-	-	-	-
2023	-	-	-	-	-	-	-	-	-	-
2024	-	-	-	-	-	-	-	-	-	-
2025	-	-	-	-	-	-	-	-	-	-
2026	-	-	-	-	-	-	-	-	-	-
2027	-	-	-	-	-	-	-	-	-	-
2028	-	-	-	-	-	-	-	-	-	-
2029	-	-	-	-	-	-	-	-	-	-
2030	-	-	-	-	-	-	-	-	-	-
2031	-	-	-	-	-	-	-	-	-	-
2032	-	-	-	-	-	-	-	-	-	-
2033	-	-	-	-	-	-	-	-	-	-
2034	-	-	-	-	-	-	-	-	-	-
2035	-	-	-	-	-	-	-	-	-	-
2036	-	-	-	-	-	-	-	-	-	-
2037	-	-	-	-	-	-	-	-	-	-
2038	-	-	-	-	-	-	-	-	-	-
2039	-	-	-	-	-	-	-	-	-	-
2040	-	-	-	-	-	-	-	-	-	-
2041	-	-	-	-	-	-	-	-	-	-
2042	-	-	-	-	-	-	-	-	-	-
2043	1	12,614	\$ 0.00055	\$ 7	\$ 4	0.5	6,307	\$ 0.00055	\$ 3	\$ 2
2044	2	25,229	0.00055	14	7	1.0	12,614	0.00055	7	3
2045	3	37,843	0.00055	21	10	1.5	18,922	0.00055	10	5
2046	4	50,458	0.00055	28	13	2.0	25,229	0.00055	14	7
2047	5	63,072	0.00055	35	16	2.5	31,536	0.00055	17	8
2048	6	75,686	0.00055	42	19	3.0	37,843	0.00055	21	10
2049	7	88,301	0.00055	49	22	3.5	44,150	0.00055	24	11
2050	8	100,915	0.00055	56	25	4.0	50,458	0.00055	28	12
2051	9	113,530	0.00055	62	27	4.5	56,765	0.00055	31	14
2052	10	126,144	0.00055	69	30	5.0	63,072	0.00055	35	15
2053	11	138,758	0.00055	76	32	5.5	69,379	0.00055	38	16
2054	12	151,373	0.00055	83	34	6.0	75,686	0.00055	42	17
2055	13	163,987	0.00055	90	36	6.5	81,994	0.00055	45	18
2056	14	176,602	0.00055	97	38	7.0	88,301	0.00055	49	19
2057	15	189,216	0.00055	104	40	7.5	94,608	0.00055	52	20
2058	16	201,830	0.00055	111	42	8.0	100,915	0.00055	56	21
2059	17	214,445	0.00055	118	44	8.5	107,222	0.00055	59	22
2060	18	227,059	0.00055	125	45	9.0	113,530	0.00055	62	23
2061	19	239,674	0.00055	132	47	9.5	119,837	0.00055	66	24
2062	20	252,288	0.00055	139	49	10.0	126,144	0.00055	69	24
2063	21	264,902	0.00055	146	50	10.5	132,451	0.00055	73	25
2064	22	277,517	0.00055	153	51	11.0	138,758	0.00055	76	26
2065	23	290,131	0.00055	160	53	11.5	145,066	0.00055	80	26
2066	24	302,746	0.00055	167	54	12.0	151,373	0.00055	83	27
2067	25	315,360	0.00055	173	55	12.5	157,680	0.00055	87	27
2068	26	327,974	0.00055	180	56	13.0	163,987	0.00055	90	28
2069	27	340,589	0.00055	187	57	13.5	170,294	0.00055	94	29
2070	28	353,203	0.00055	194	58	14.0	176,602	0.00055	97	29
2071	29	365,818	0.00055	201	59	14.5	182,909	0.00055	101	29
2072	30	378,432	0.00055	208	60	15.0	189,216	0.00055	104	30
2073	31	391,046	0.00055	215	61	15.5	195,523	0.00055	108	30
2074	32	403,661	0.00055	222	61	16.0	201,830	0.00055	111	31
2075	33	416,275	0.00055	229	62	16.5	208,138	0.00055	114	31
2076	34	428,890	0.00055	236	63	17.0	214,445	0.00055	118	31
2077	35	441,504	0.00055	243	63	17.5	220,752	0.00055	121	32
2078	36	454,118	0.00055	250	64	18.0	227,059	0.00055	125	32
2079	37	466,733	0.00055	257	64	18.5	233,366	0.00055	128	32
2080	38	479,347	0.00055	264	65	19.0	239,674	0.00055	132	32
2081	39	491,962	0.00055	271	65	19.5	245,981	0.00055	135	33
2082	40	504,576	0.00055	278	65	20.0	252,288	0.00055	139	33
2083	41	517,190	0.00055	284	66	20.5	258,595	0.00055	142	33
2084	42	529,805	0.00055	291	66	21.0	264,902	0.00055	146	33
2085	43	542,419	0.00055	298	66	21.5	271,210	0.00055	149	33
2086	43	542,419	0.00055	298	65	22.0	277,517	0.00055	153	33
2087	43	542,419	0.00055	298	64	22.5	283,824	0.00055	156	33
2088	43	542,419	0.00055	298	62	23.0	290,131	0.00055	160	33
2089	43	542,419	0.00055	298	61	23.0	290,131	0.00055	160	33
2090	43	542,419	0.00055	298	60	23.0	290,131	0.00055	160	32
2091	43	542,419	0.00055	298	59	23.0	290,131	0.00055	160	31
2092	43	542,419	0.00055	298	58	23.0	290,131	0.00055	160	31
2093	43	542,419	0.00055	298	57	23.0	290,131	0.00055	160	30
2094	43	542,419	0.00055	298	55	23.0	290,131	0.00055	160	30
2095	43	542,419	0.00055	298	54	23.0	290,131	0.00055	160	29
2096	43	542,419	0.00055	298	53	23.0	290,131	0.00055	160	28
2097	43	542,419	0.00055	298	52	23.0	290,131	0.00055	160	28
2098	43	542,419	0.00055	298	51	23.0	290,131	0.00055	160	27
2099	43	542,419	0.00055	298	50	23.0	290,131	0.00055	160	27
2100	43	542,419	0.00055	298	49	23.0	290,131	0.00055	160	26
				\$ 11,038	\$ 2,815				\$ 5,665	\$ 1,435

^aCumulative number of operating FRs times 12,614-million kWH/reactor year.
^b55% of the 1-mill/kWH fee stipulated by NWPA, due to reduced volume of waste and reduced handling.

6.0 DIFFERENCES IN THE NON-PROLIFERATION ASPECTS OF THE UREX+3c FUEL CYCLE AND THE AFC

6.1 INTRODUCTION

The U.S. Nuclear Regulatory Commission (NRC) categorizes nuclear materials into source materials, special nuclear material, and byproduct material, as follows:

- (1) Source Material (SM) is defined as
 - (a) Uranium, thorium, or any combination thereof, in any physical or chemical form, or
 - (b) Ores that contain 0.05 percent or more uranium, thorium, or any combination thereof
- (2) Special Nuclear Material (SNM) is defined as
 - (a) Plutonium, U^{233} , or uranium enriched in U^{233} or U^{235} , or
 - (b) Any material artificially enriched by any of the foregoing
- (3) Byproduct Material is defined as
 - (a) Any radioactive material (except SNM) yielded in or made radioactive by exposure to radiation incident to the process of producing or using SNM
 - (b) Any of the following:
 - (i) Discrete source of Ra^{226} that is produced for a commercial, medical or research activity
 - (ii) Material that has been made radioactive by use of a particle accelerator
 - (iii) Material that is produced for a commercial, medical or research activity
 - (c) Any discrete source of naturally occurring radioactive material other than source material that
 - (a) NRC in conjunction with other agencies determines would pose a threat similar to that posed by a discrete source of Ra^{226} to public safety, defense or security
 - (b) Is produced for a commercial, medical or research activity

Moreover, NRC further categorizes SNM into three safeguards categories, according to its potential for direct use in a clandestine fissile explosive, as follows:

Category I - Strategic SNM (SSNM) -- SSNM includes SNM in any combination in a quantity of:

- (1) 2.0 kg or more of plutonium or U^{233} , or
- (2) 5 kg or more of U^{235} (in ≥ 20 percent enriched U), or
- (3) 5 kg or more in any combination determined by the following equation:

$$\text{grams} = (\text{grams } U^{235}) + 2.5(\text{grams } U^{233} + \text{grams } Pu)$$

(The foregoing is referred to as a formula quantity.)

Category II - SNM of Moderate Strategic Significance -- SNM of moderate strategic significance includes:

- (1) Less than a formula quantity of SSNM, but
- (2) More than 1,000 grams U^{235} (in ≥ 20 percent enriched U), or
- (3) More than 500 grams U^{233} or Pu, or
- (4) A combination of more than 1,000 grams determined by the following equation:

$$\text{grams} = (\text{grams } U^{235}) + 2(\text{grams } U^{233} + \text{grams } Pu)$$

- (5) 10,000 grams or more U^{235} (in ≥ 10 percent to < 20 percent enriched U)

Category III - SNM of Low Strategic Significance -- SNM of low strategic significance includes:

- (1) Less than an amount of SNM of moderate strategic significance, but
- (2) More than 15 grams U^{235} (in ≥ 20 percent enriched U), or
- (3) More than 15 grams U^{233} or Pu, or
- (4) A combination of 15 grams determined by the following equation:

$$\text{grams} = (\text{grams } U^{235} + \text{grams } U^{233} + \text{grams } Pu)$$

- (5) Less than 10,000 grams but more than 1,000 grams U^{235} contained (in ≥ 10 percent to < 20 percent enriched U), or
- (6) 10,000 grams or more U^{235} (in above natural but < 10 percent enriched U).

Further, NRC provides requirements for the protection of SM in excess of 150 lb in any calendar year as follows:

Source Material -- An effective kilogram of source material is defined by NRC as:

- (1) 10,000 kg of source material uranium in which U^{235} is greater than 0.5 weight percent of the total uranium present, and
- (2) 20,000 kg of any other source material.

Thus, in the exercise of its regulatory jurisdiction, the NRC recognizes the need to safeguard all varieties of SNM, through both physical protection and timely detection from theft or diversion from use in the unauthorized production of nuclear weapons -- and to appropriately protect

source material as well, but to a much less rigorous extent in view of the lesser threat that it represents. The categories of SNM, coupled with the definition of an effective kilogram of source material, effectively identify the relative attractiveness of source and special nuclear material for theft or diversion. The regulatory requirements for physical protection, and for nuclear material control and accountability, become more robust with increasing attractiveness of the materials involved.

6.2 COMPARATIVE ANALYSIS OF SAFEGUARDS REQUIREMENTS

Using the NRC requirements as a base, a comparison was made of the level of safeguards requirements (both material control and accounting, and physical protection) that are imposed by NRC on U.S. commercial nuclear fuel cycle facilities such as those represented by the UREX+3c fuel cycle and the AFC. This comparison is shown in Table 6-1.

From the table it can be seen that there is little difference in the safeguards requirements for the various steps of the UREX+3c fuel cycle and the AFC, because most of the steps are basically the same. The single difference is that the UREX+3c fuel cycle does not recycle uranium recovered from reprocessing, while the AFC does so. This means that the AFC has an additional enrichment step (for the recovered uranium) and, moreover, must convert and dispose of the resultant depleted uranium. Therefore, the AFC is slightly more vulnerable to proliferation than the UREX+3c fuel cycle. This would be mitigated significantly if the enrichment was conducted in a separate line at enrichment facilities used to enrich virgin uranium (which is likely).

Another way of looking at the prospective comparative vulnerability of the different steps in the fuel cycles is on the basis of the total system throughputs of the nuclear materials involved in each over the 2020-2100 period. This comparison is shown in Table 6-2.

From Table 6-2 it can be seen that:

- (1) Except for wastes, the largest quantities involved in each of the fuel cycles are represented by the processing steps used for producing UO₂ fuel for LWRs -- and which are about 10 percent less for the AFC than for the UREX+3c fuel cycle.
- (2) The most significant differential in quantities of strategic nuclear materials in the UREX+3c fuel cycle and the AFC are in:
 - (a) MOX fuel fabrication -- where the quantities of MOX to be processed were 33 to 38 percent greater in the AFC than in the UREX+3c fuel cycle, because the use of enriched recovered uranium to provide part of the

fissile content of MOX fuel results in the production of larger amounts of MOX.

- (b) Enrichment of recovered uranium -- where the quantities are only experienced in the AFC
- (c) Disposal of excess recovered uranium -- where the quantities of recovered uranium to be processed were 87 to 93 percent less in the AFC.

TABLE 6-1
COMPARISON OF PREVAILING NRC SAFEGUARDS CATEGORIES
OF NUCLEAR MATERIALS PROCESSED IN
THE UREX+3c FUEL CYCLE AND THE AFC

Activity	Level of Safeguards Required by NRC			
	1 MOX Recycle		2 MOX Recycles	
	UREX+3c	AFC	UREX+3c	AFC
Mining & Milling	SM	SM	SM	SM
Conversion, to UF ₆	SM	SM	SM	SM
Enrichment, ≤5%	SNM-III	SNM-III	SNM-III	SNM-III
DU Conversion, to U ₃ O ₈ (MTDU)	SM	SM	SM	SM
UO ₂ Fuel Fabrication	SNM-III	SNM-III	SNM-III	SNM-III
Reprocessing	SNM-I	SNM-I	SNM-I	SNM-I
MOX Fuel Fabrication	SNM-I	SNM-I	SNM-I	SNM-I
Recycled U Enrichment, ~3%	N/A ^a	SNM-III	N/A ^a	SNM-III
Initial FR Core Fabrication	SNM-I	SNM-I	SNM-I	SNM-I
Reprocessing & Refabrication of FR Recycle Fuel	SNM-I	SNM-I	SNM-I	SNM-I
Disposal of Excess RU	SNM-III	SNM-III	SNM-III	SNM-III
Disposal of Depleted RU	N/A ^a	SM	N/A ^a	SM
LLW Disposal	EX	EX	EX	EX
TRU Disposal	EX	EX	EX	EX
HLW Disposal	EX	EX	EX	EX

^aN/A = Not Applicable

Legend:

SNM-I -- Special Nuclear Material Category I

SNM-III -- Special Nuclear Material Category III

SM -- Source Material

EX -- Exempt from Category Requirements because of (i) very low concentrations, or (ii) high levels of associated radioactivity.

TABLE 6-2
COMPARISON OF THE ESTIMATED TOTAL QUANTITIES PROCESSED FOR THE
THE UREX+3c FUEL CYCLE AND THE AFC OVER THE PERIOD 2020 THRU 2100
 (Thousands of Indicated Units)

Activity	Total Quantities					
	1 MOX Recycle			2 MOX Recycles		
	UREX+3c	AFC	% Change ^a	UREX+3c	AFC	% Change ^a
Mining & Milling (MTNU)	1,242	1,127	-9.3	1,168	1,043	-10.7
Conversion, to U ₆ (MTNU)	1,242	1,127	-9.3	1,168	1,043	-10.7
Enrichment, 5% (MTEU)	121	110	-9.1	114	102	-10.5
DU Conversion, to UO ₂ (MTDU)	1,121	1,016	-9.4	1,055	942	-10.7
UO ₂ Fuel Fabrication (MTEU)	120	109	-9.2	113	101	-10.6
Reprocessing (MTIHM)	193	193	0	197	197	0
MOX Fuel Fabrication (MTIHM)	29	40	+37.9	36	48	+33.3
Recycled U Enrichment, ~3% (MTEU)	0	141	+	0	157	+
Initial FR Core Fabrication	4	4	0	2	2	0
Reprocessing & Refabrication of FR Recycle Fuel (MTIHM)	23	23	0	12	12	0
Disposal of Excess RU (MTRU)	160	20	-87.5	163	7	-92.9
Disposal of Depleted RU	0	112	+	0	125	+
LLW Disposal (cu ft.)	37,700	34,613	-8.2	36,294	32,945	-9.2
TRU Disposal (cu ft.)	10,651	12,108	+13.7	9,787	11,362	+16.1
HLW Disposal (MTIHM)	119	119	0	116	116	0

^a% difference of quantities in the AFC compared with those of the UREX+3c fuel cycle

Legend:

Special Nuclear Material Category I

Special Nuclear Material Category III

Source Material

Exempt from Category Requirements because of (i) very low concentrations, or (ii) high levels of associated radioactivity.

NU = Natural U
 DU = Depleted U
 EU = Enriched U

IHM = Initial Heavy Metal
 RU = Recovered U
 DRU = Depleted Recycled U

6.3 CONCLUSIONS

As a result of the analysis set forth in Section 6.2, it can be concluded that the AFC may be slightly more vulnerable to proliferation than the UREX+3c fuel cycle -- but not to the extent that it shouldn't otherwise be considered for use.

7.0 DIFFERENCES IN POTENTIAL RADIOLOGICAL HEALTH IMPACTS OF THE UREX+3c FUEL CYCLE AND THE AFC

This section presents an assessment of the comparative impacts to occupational and public radiological health (exposure to ionizing radiation) for the UREX+3c fuel cycle and the AFC for normal operations and accidents. The characteristics of the two fuel cycle systems are described elsewhere in this report. Section 3.0 presents baseline information and Section 4.0 presents analyses of reprocessing and recycle that are used in the comparisons presented here. In addition, information used in this section is also taken from Section 8.0, which discusses differences in the characteristics and quantities of radioactive wastes, and from Section 10.2, which discusses effects on transportation of spent nuclear fuel of multiple recycle of plutonium in LWRs.

Because the designs of many of the facilities that would be needed for the two fuel cycles are conceptual and thus lack detail regarding systems and operations, the analysis herein is at a corresponding overview level. Nonetheless, because occupational and public radiation exposures are controlled by the mixtures of radionuclides encountered (referred to in this study as the source terms), general comparisons can be made by identifying, analyzing and comparing the radionuclide mixtures. The resulting comparisons of the consequences to the radiological health of workers and the public of implementing the fuel cycles provide insights that are broadly useful.

Specifically, this section presents a general assessment of the extent to which shielded facilities are required for the following:

- Conversion of uranium recovered from reprocessing to U_3O_8 , with storage (or disposal) of excess (or surplus) uranium in a repository
- Conversion of uranium recovered from reprocessing to UF_6
- Re-enrichment of uranium recovered from reprocessing, and the conversion of the resulting depleted recycle uranium to U_3O_8
- Fabrication of MOX fuel
- Fabrication of FR fuel (uranium, plutonium, minor actinides, and zirconium metal)
- Packaging and transportation of the products from the items above.

7.1 BACKGROUND, AND ANALYSIS APPROACH

The UREX+3c fuel cycle and the AFC employ nuclear fuel reprocessing and recycling and advanced fast-spectrum nuclear power reactors. As a caution to the reader, it is important to consider the status (in 2009) of environmental and nuclear safety regulations that would govern these fuel cycle facilities and associated activities. For example, the NRC licensing process for commercial nuclear fuel reprocessing facilities has not been exercised or updated since 1976. While domestic regulations and guidance for licensing other nuclear facilities, such as light water nuclear power reactors or low-level waste disposal facilities, have been updated, the regulations and guidance specific to nuclear fuel reprocessing and fast reactors have not. This means that there is regulatory process uncertainty associated with licensing these kinds of advanced nuclear fuel cycle facilities. Nonetheless, it is safe to assume, and it is therefore assumed herein, that updated guidance and regulation that would be implemented would be consistent with established regulations for other commercial nuclear facilities.

Although there are similarities to existing nuclear fuel cycle facilities, several of the facilities considered for the UREX+3c fuel cycle and AFC are unique, either with their design, operation, radionuclide source terms, or regulation; these merit mention here and include:

- (1) Fuel reprocessing facilities. In the past, the U.S. had substantial industrial-scale experience, both in plutonium reprocessing for defense purposes, and in programs to develop commercial nuclear reactors that would “breed” plutonium for use in commercial nuclear energy systems. However, there has been no commercial nuclear fuel reprocessing activity in the United States since April 7, 1977. And, in the early 1990’s the DOE ceased both defense reprocessing and the development of breeder reactors. As a consequence, there are policy, technology, and regulatory gaps that would need to be filled before a reprocessing facility could be designed, constructed, and operated. The basic assumption for this study is that the policy will change, the technology can be obtained (even if from off shore), and that the regulatory process will expand to cover design, construction, and operation of a present-day commercial reprocessing facility.
- (2) Uranium reenrichment facilities. Currently, only newly mined (natural) uranium is considered for enrichment in the U.S. Since this uranium has not been irradiated in a nuclear reactor, it does not contain significant amounts of U^{232} , U^{234} , or U^{236} , or trace amounts of fission products (such as Tc^{99}), or actinides. However recycled uranium does contain these contaminants which can complicate the enrichment process. In enrichment, the lighter uranium isotopes in recovered uranium will concentrate in the enriched product along with U^{235} . Trace contaminants such as Tc^{99} and actinide isotopes will generally be trapped in

the feed and initial stages of an enrichment plant adding to radioactive wastes and worker exposures. For this study, it is assumed that re-enrichment can occur, either in current facilities, or in dedicated facilities used solely for this purpose.

- (3) Mixed oxide fuel fabrication facilities. A consortium, Shaw AREVA MOX Services, LLC, under contract to the National Nuclear Security Administration is designing and building a MOX Fuel Fabrication Facility at the Savannah River Site (SRS) in South Carolina. The facility will remove impurities from surplus nuclear-weapons plutonium and blend the product with uranium to produce MOX fuel and LWR fuel assemblies. These assemblies will be irradiated in domestic commercial nuclear power reactors for power production. The NRC will license and oversee the facility. This facility is conceptually similar to the MOX fuel production facility envisioned in this study for both of the fuel cycles considered; key differences are the amounts and characteristics of the plutonium that would be processed, and the presence of significant amounts of minor actinide radioisotopes.

For other fuel cycle facilities, including mining/milling/conversion, uranium enrichment, UO₂ fuel fabrication, and operations at LWRs, except for the quantities of materials that would be processed, no appreciable differences are expected between the two alternative fuel cycles and current fuel cycles. Thus, the radiological health impacts of these facilities are not evaluated herein.

In addition, this analysis does not consider the health and safety impacts of fuel reprocessing facilities that would be associated with the two fuel cycles. The source terms encountered with the recovery of uranium, plutonium, and minor actinide elements and the generation of fission product, low-level, and transuranic wastes would be largely the same for the two fuel cycles and would involve similar mixtures of fission products, fissile material, and minor actinides. The ALARA design considerations to limit worker and public radiation doses would also be essentially the same. For workers these features would include significant shielding in key process and waste handling locations for the fission product photon emitters, limiting batch sizes to reduce overall dose rates, and remote or automated process control systems to eliminate worker exposures. For protection of the public, specific effluent controls would be tailored to the radionuclides encountered and accident prevention or mitigation features would be included. These, and other ALARA design features, are consistent with the design of currently operating fuel reprocessing facilities world-wide. Thus, for reprocessing, the comparison of the UREX+3c and the AFC would be null and not material to the overall comparison of health impacts of the two fuel cycles.

Figures 7-1 through 7-4, which are discussed in greater detail in the remainder of this section, describe the flows of nuclear fuel materials for the two fuel cycles and two LWR fuel recycle scenarios. Both fuel cycles generally consider use of the plutonium and uranium, recovered from spent MOX fuel after either one or two recycles of spent fuel in LWRs, to fuel FRs. The minor actinides recovered during reprocessing of both LWR and FR spent fuel are assumed to be consumed in FRs in both fuel cycles.

The source terms identified on the flow sheets are provided as the activity of each radionuclide per mass of total material resulting from processing one MTIHM. These sources are derived from ORIGEN computer analysis of the fuel in each nuclear fuel cycle. For spent fuel reprocessing steps, the materials are assumed to be used promptly in the subsequent enrichment, fuel production, or conversion of uranium to be sent to a depository. Thus decay products, notably those from the decay of U^{232} , would be of minor importance.

FIGURE 7-1

URANIUM MATERIAL BALANCES AND SOURCE TERMS FOR THE UREX+3c FUEL CYCLE WITH ONE MOX RECYCLE

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FIGURE 7-3
URANIUM MATERIAL BALANCES AND SOURCE TERMS FOR THE UREX+3c FUEL CYCLE WITH TWO MOX RECYCLES

FIGURE 7-4
URANIUM MATERIAL BALANCES AND SOURCE TERMS FOR THE AFC WITH TWO MOX RECYCLES

The source terms were translated into the activity per radionuclide in each mixture providing the basis for a relative comparison of potential radiation dose rates that would be encountered and for considering requirements for shielding at the various fuel cycle facilities. Also considered were the potential impacts of radionuclides produced from the decay of U^{232} .

The MicroShield[®] (Ref. 1) computer code was used to estimate the relative photon dose rates to workers from each source for the applicable fuel cycle facilities. The relative importance of neutron radiation was determined by inspection of radionuclide compositions.

The risk for members of the public of exposure to radioactive materials and ionizing radiation (dose) from facilities and operations for the two fuel cycles would be principally from airborne releases and consequent inhalation of radionuclides. The analysis evaluated doses to the public by considering differences among the facilities and fuel cycles for the relative materials at risk. This allowed comparison of potential impacts for each facility without having to analyze design basis accidents.

Transportation impacts were evaluated against regulatory requirements for packaging and shipping.

7.2 SOURCE TERMS

The facilities for which source terms are developed are identified in Figures 7-1 through 7-4 above. The source terms are translated into units of activity and are shown in Tables 7-1 through 7-14 as the far right hand column titled "Total Curies."

As shown in Figure 7-1, there are five source terms for the UREX+3c fuel cycle with one recycle of spent fuel. The first source term -- Source Term 1 -- which is listed in Table 7-1, identifies the amounts and radioactivity of uranium isotopes in excess uranium (856 kg) recovered from reprocessing 1 MTIHM of UO_2 spent fuel. The source's radioactivity is used to assess the radiological health impacts of converting the recovered uranium to U_3O_8 for storage or disposal.

TABLE 7-1
SOURCE TERM 1, UREX+3c FUEL CYCLE WITH 1 AND 2 MOX RECYCLES:
RADIONUCLIDE SOURCES IN RU (EXCESS FROM UO₂ REPROCESSING) FOR
CONVERSION TO U₃O₈^a

Radionuclide	Fraction of total U	Half-life (d)	Atomic Mass	Total Curies
U ²³²	2.81E-09	2.63E+04	232	5.14E-02
U ²³⁴	2.33E-04	8.93E+07	234	1.25E+00
U ²³⁵	8.28E-03	2.57E+11	235	1.53E-02
U ²³⁶	7.74E-03	8.55E+09	236	4.29E-01
U ²³⁸	9.84E-01	1.62E+12	238	2.85E-01
Total	1.00E+00			2.03E+00

^aBased on 0.856 metric tons of excess recovered uranium (RU) from reprocessing 1 MTIHM of uranium spent fuel.

Source Term 2, listed in Table 7-2, is for plutonium, uranium, and americium radionuclides recovered from spent UO₂ fuel reprocessing for production of 168.5 kg of first generation of MOX (MOX-1) LWR fuel.

Source Term 3, listed in Table 7-3, is for 104.3 kg of excess uranium recovered from MOX-1 spent fuel. This excess uranium is converted to U₃O₈ for storage or disposal. Reprocessing of the MOX-1 spent fuel in this case is done to recover uranium, plutonium, and minor actinides for production of FR fuel.

Source Term 4, shown in Table 7-4, represents 54.3 kg of uranium, neptunium, plutonium, curium, and americium sources for fabrication of nuclear fuel for first cores of fast reactors. This source term comes from the products of reprocessing MOX-1 and the minor actinides recovered from the preceding step of reprocessing UO₂ spent fuel, and is adjusted linearly to 565 kg as a base.

Source Term 5, the last source term for the UREX+3c fuel cycle with one recycle of spent fuel, is listed in Table 7-5. This source term includes 430 kg of uranium, neptunium, plutonium, curium, and americium sources for fabrication of recycle fuel for fast reactor cores, and is adjusted linearly to 565 kg as a base. The difference between this source term and the one presented in Table 7-4 is that this source term includes the uranium, plutonium, and minor actinides from reprocessed MOX-1 and minor actinides recovered from the preceding step of reprocessing UO₂ spent fuel as well as the uranium, plutonium and minor actinides recovered from reprocessing/recycling spent FR fuel.

TABLE 7-2
SOURCE TERM 2, UREX+3c FUEL CYCLE WITH 1 AND 2 MOX RECYCLES: SOURCES
FOR PRODUCTION OF FIRST GENERATION MOX LWR FUEL^a

Radionuclide	Kilograms	Percent	Half-life (d)	Atomic Mass	Total Curies
Pu ²³⁸	4.70E-01	0.28	3.20E+04	238	8.05E+03
Pu ²³⁹	6.39E+00	3.78	8.79E+06	239	3.98E+02
Pu ²⁴⁰	3.06E+00	1.81	2.39E+06	240	6.98E+02
Pu ²⁴¹	1.45E+00	0.86	5.26E+03	241	1.50E+05
Pu ²⁴²	1.05E+00	0.62	1.37E+08	242	4.15E+00
U ²³²	4.38E-07	0.00	2.63E+04	232	3.76E-02
U ²³⁴	1.46E-02	0.01	8.93E+07	234	9.10E-02
U ²³⁵	1.18E+00	0.70	2.57E+11	235	2.56E-03
U ²³⁶	4.84E-01	0.29	8.55E+09	236	3.13E-02
U ²³⁸	1.54E+02	91.34	1.62E+12	238	5.22E-02
Am ²⁴¹	7.23E-02	0.04	1.58E+05	241	2.48E+02
Total	1.69E+02	100			1.59E+05

^aBased on 0.1685 MTIHM

TABLE 7-3
SOURCE TERM 3, UREX+3c FUEL CYCLE WITH 1 MOX RECYCLE: SOURCES FOR
CONVERSION OF RECOVERED URANIUM (EXCESS FROM MOX-1 REPROCESSING)
TO U₃O₈^a

Radionuclide	Fraction of Total U	Half-life (d)	Atomic Mass	Total Curies
U ²³²	5.90E-09	2.63E+04	232	1.31E-02
U ²³⁴	2.23E-04	8.93E+07	234	1.45E-01
U ²³⁵	3.08E-03	2.57E+11	235	6.95E-04
U ²³⁶	3.23E-03	8.55E+09	236	2.18E-02
U ²³⁸	9.94E-01	1.62E+12	238	3.51E-02
Total	1.00E+00			2.16E-01

^aBased on 0.1043 metric tons of excess recovered uranium (RU) from reprocessing MOX-1 spent fuel.

TABLE 7-4
SOURCE TERM 4, UREX+3c FUEL CYCLE WITH 1 MOX RECYCLE: SOURCES FOR FAST REACTOR FIRST CORE
FABRICATION^{a,b}

Radionuclide	Grams from 0.1685 MTIHM of MOX-1	Fraction of Total Mass	Fractional Curies	Grams in 54.3 kg of FR Fuel	Fraction of Total Mass	Fractional Curies	Grams in 565 kg of FR Fuel	Fraction of Total Mass	Total Curies ^c
U ²³²	8.68E-04	5.46E-09	1.17E-07	2.53E-04	4.66E-09	9.97E-08	2.63E-03	4.65E-09	5.63E-02
U ²³⁴	3.29E+01	2.07E-04	1.30E-06	9.58E+00	1.76E-04	1.10E-06	9.97E+01	1.76E-04	6.22E-01
U ²³⁵	4.53E+02	2.85E-03	6.18E-09	1.32E+02	2.43E-03	5.26E-09	1.37E+03	2.42E-03	2.96E-03
U ²³⁶	4.77E+02	3.00E-03	1.95E-07	1.39E+02	2.56E-03	1.66E-07	1.45E+03	2.57E-03	9.38E-02
U ²³⁸	1.46E+05	9.18E-01	3.12E-07	4.26E+04	7.85E-01	2.66E-07	4.43E+05	7.84E-01	1.50E-01
Total U ^d	1.47E+05	9.25E-01		4.29E+04	7.90E-01		4.46E+05	7.89E-01	
Np ²³⁷	1.07E+03	6.73E-03	4.76E-06	1.07E+03	1.97E-02	1.39E-05	1.11E+04	1.96E-02	7.80E+00
Pu ²³⁸	4.34E+02	2.73E-03	4.69E-02	4.34E+02	7.99E-03	1.37E-01	4.51E+03	7.98E-03	7.74E+04
Pu ²³⁹	2.95E+03	1.86E-02	1.16E-03	2.95E+03	5.43E-02	3.38E-03	3.07E+04	5.43E-02	1.91E+03
Pu ²⁴⁰	2.48E+03	1.56E-02	3.56E-03	2.48E+03	4.57E-02	1.04E-02	2.58E+04	4.57E-02	5.88E+03
Pu ²⁴¹	1.17E+03	7.36E-03	7.58E-01	1.17E+03	2.15E-02	2.22E+00	1.21E+04	2.14E-02	1.25E+06
Pu ²⁴²	1.28E+03	8.05E-03	3.19E-05	1.28E+03	2.36E-02	9.29E-05	1.33E+04	2.35E-02	5.24E+01
Total Np+Pu	9.38E+03	5.90E-02		9.38E+03	1.73E-01		9.75E+04	1.73E-01	
Am ²⁴¹	9.26E+02	5.82E-03	2.00E-02	9.26E+02	1.71E-02	5.85E-02	9.63E+03	1.70E-02	3.31E+04
Am ²⁴³	6.93E+02	4.36E-03	8.70E-04	6.93E+02	1.28E-02	2.54E-03	7.21E+03	1.28E-02	1.44E+03
Cm ²⁴³	2.52E+00	1.58E-05	8.22E-04	2.52E+00	4.64E-05	2.40E-03	2.62E+01	4.64E-05	1.36E+03
Cm ²⁴⁴	3.53E+02	2.22E-03	1.80E-01	3.53E+02	6.50E-03	5.26E-01	3.67E+03	6.50E-03	2.97E+05
Cm ²⁴⁵	4.29E+01	2.70E-04	4.65E-05	4.29E+01	7.90E-04	1.36E-04	4.46E+02	7.89E-04	7.68E+01
Cm ²⁴⁶	4.60E+00	2.89E-05	8.91E-06	4.60E+00	8.47E-05	2.60E-05	4.79E+01	8.48E-05	1.47E+01
Total Am+Cm	2.02E+03	1.27E-02		2.02E+03			2.10E+04	3.72E-02	
Overall Total	1.58E+05			5.43E+04			5.65E+05		1.67E+06

^aBased on 0.1685 MTIHM of LWR MOX-1 spent fuel, where 54.3 kg of fast reactor fuel is produced from the products of reprocessing 168.5 kg of LWR MOX-1 spent fuel.

^bIncludes minor actinides from reprocessing 1 MTIHM of UO₂ spent fuel.

^cThis mixture is used for the analysis of Source Term 4 impacts.

^dAll of the uranium is not used -- 104.3 kg is excess from the product of reprocessing 168.5 kg of MOX-1 LWR spent fuel.

TABLE 7-5
SOURCE TERM 5, UREX+3c FUEL CYCLE WITH 1 MOX RECYCLE: SOURCES FOR FAST REACTOR RECYCLE CORE
FABRICATION^{a,b}

Radionuclide in Spent Fuel	(Grams)	Fraction of Total Mass	Fractional Curies	(Grams)	Fraction of Total Mass	Fractional Curies	Grams in 565 kg of FR Fuel	Fraction of Total Mass	Total Curies ^c
U ²³²	8.68E-04	5.46E-09	1.17E-07	7.61E-02	1.77E-07	3.79E-06	1.00E-01	1.77E-07	2.14E+00
U ²³⁴	3.29E+01	2.07E-04	1.29E-06	5.70E+02	1.33E-03	8.28E-06	7.49E+02	1.33E-03	4.68E+00
U ²³⁵	4.53E+02	2.85E-03	6.16E-09	2.01E+02	4.68E-04	1.01E-09	2.64E+02	4.68E-04	5.71E-04
U ²³⁶	4.77E+02	3.00E-03	1.94E-07	1.06E+03	2.47E-03	1.60E-07	1.40E+03	2.47E-03	9.04E-02
U ²³⁸	1.46E+05	9.20E-01	3.12E-07	3.38E+05	7.86E-01	2.66E-07	4.44E+05	7.86E-01	1.50E-01
Total U ^d	1.47E+05	9.26E-01		3.40E+05	7.90E-01		4.46E+05	7.90E-01	
Np ²³⁷	1.07E+03	6.73E-03	4.74E-06	3.61E+03	8.40E-03	5.92E-06	4.75E+03	8.40E-03	3.34E+00
Pu ²³⁸	4.34E+02	2.73E-03	4.68E-02	4.85E+03	1.13E-02	1.93E-01	6.37E+03	1.13E-02	1.09E+05
Pu ²³⁹	2.95E+03	1.86E-02	1.15E-03	3.67E+04	8.54E-02	5.31E-03	4.82E+04	8.54E-02	3.00E+03
Pu ²⁴⁰	2.48E+03	1.56E-02	3.55E-03	2.15E+04	4.99E-02	1.14E-02	2.82E+04	4.99E-02	6.44E+03
Pu ²⁴¹	1.17E+03	7.33E-03	7.56E-01	3.35E+03	7.79E-03	8.02E-01	4.40E+03	7.79E-03	4.53E+05
Pu ²⁴²	1.28E+03	8.07E-03	3.18E-05	9.23E+03	2.15E-02	8.46E-05	1.21E+04	2.15E-02	4.78E+01
Total Np+Pu	9.38E+03	5.90E-02		7.92E+04	1.84E-01		1.04E+05	1.84E-01	
Am ²⁴¹	9.26E+02	5.82E-03	2.00E-02	3.29E+03	7.65E-03	2.62E-02	4.32E+03	7.65E-03	1.48E+04
Am ²⁴³	6.93E+02	4.36E-03	8.67E-04	4.13E+03	9.60E-03	1.91E-03	5.42E+03	9.60E-03	1.08E+03
Cm ²⁴²	0.00E+00	0.00E+00	0.00E+00	6.75E+00	1.57E-05	5.20E-02	8.87E+00	1.57E-05	2.94E+04
Cm ²⁴³	2.52E+00	1.59E-05	8.20E-04	2.11E+01	4.90E-05	2.53E-03	2.77E+01	4.90E-05	1.43E+03
Cm ²⁴⁴	3.53E+02	2.22E-03	1.80E-01	2.91E+03	6.77E-03	5.48E-01	3.82E+03	6.77E-03	3.10E+05
Cm ²⁴⁵	4.29E+01	2.70E-04	4.64E-05	5.84E+02	1.36E-03	2.33E-04	7.67E+02	1.36E-03	1.32E+02
Cm ²⁴⁶	4.60E+00	2.90E-05	8.88E-06	1.46E+02	3.40E-04	1.04E-04	1.92E+02	3.40E-04	5.88E+01
Total Am+Cm	2.02E+03	1.27E-02		1.11E+04	2.58E-02		1.46E+04	2.58E-02	
Overall Total	1.59E+05			4.30E+05			5.65E+05		9.28E+05

^aBased on producing 0.430 MTIHM of fast reactor fuel using makeup derived from reprocessing LWR MOX-1.

^bUsing the amount of actinides from 168.5 MTIHM of MOX-1.

^cThis mixture is used for the analysis of Source Term 5 impacts.

^dAll of the uranium is not used -- 111 kg is excess from the product of reprocessing 168.5 kg of MOX-1 LWR spent fuel.

Seven source terms are identified for the AFC with one spent fuel reprocessing step, as shown in Figure 7-2. Source Term 6, listed in Table 7-6, includes 856 kg of uranium isotopes recovered from reprocessing UO_2 spent fuel. This source term is the same as that listed in Table 7-1 except that it is for the conversion of excess uranium to UF_6 for re-enrichment.

Source Term 7, listed in Table 7-7, is for 171 kg of enriched recycled uranium product used for the fabrication of MOX-1 fuel.

Source Term 8, shown in Table 7-8, is for 244.5 kg of plutonium, uranium, and americium radionuclides from LWR fuel reprocessing and re-enriched recovered uranium for production of MOX-1 LWR fuel.

Source Term 9, listed in Table 7-9, represents 685 kg of uranium isotopes in depleted recycled uranium (DRU) tails from enrichment of recovered uranium. This DRU would be converted to U_3O_8 for storage or disposal.

Source Term 10, listed in Table 7-10, is for the 169.7 kg of excess uranium recovered from MOX-1 spent fuel that is not needed for production of FR fuel. This excess uranium is converted to U_3O_8 for storage or disposal. Reprocessing of the MOX-1 spent fuel in this case is done to recover uranium, plutonium, and minor actinides for production of FR fuel.

Source Term 11, listed in Table 7-11, represents 59 kg of uranium, neptunium, plutonium, curium, and americium sources for fabrication of first cores for FRs. This source term comes from the products of reprocessing MOX-1 and the minor actinides recovered from the preceding step of reprocessing UO_2 spent fuel, and is adjusted linearly to 565 kg as a base.

Source Term 12, the last source term for the AFC with one recycle of spent fuel, is listed in Table 7-12. This source term includes 479 kg of uranium, neptunium, plutonium, curium, and americium sources for fabrication of FR cores, and is adjusted linearly to 565 kg as a base. The difference between this source term and the one presented in Table 7-11 is that this source term includes the uranium, plutonium, and minor actinides from reprocessed MOX-1 and minor actinides recovered from the preceding step of reprocessing UO_2 spent fuel as well as the uranium, plutonium and minor actinides recovered from reprocessing/recycling spent FR fuel.

TABLE 7-6
SOURCE TERM 6, AFC WITH 1 AND 2 MOX RECYCLES: SOURCES FOR RU FEED
CONVERTED TO UF₆ FOR ENRICHMENT^a

Radionuclide	Fraction of total U	Half-life (d)	Atomic Mass	Total Curies
U ²³²	2.81E-09	2.63E+04	232	5.14E-02
U ²³⁴	2.33E-04	8.93E+07	234	1.25E+00
U ²³⁵	8.28E-03	2.57E+11	235	1.53E-02
U ²³⁶	7.74E-03	8.55E+09	236	4.29E-01
U ²³⁸	9.84E-01	1.62E+12	238	2.85E-01
Total	1.00E+00			2.03E+00

^aBased on 0.856 metric tons of recovered uranium (RU) from reprocessing 1 MTIHM uranium spent fuel.

TABLE 7-7
SOURCE TERM 7, AFC WITH 1 AND 2 MOX RECYCLES: SOURCES FOR RE-
ENRICHED RU PRODUCT^a

Radionuclide	Fraction of total U	Half-life (d)	Atomic Mass	Total Curies
U ²³²	1.40E-08	2.63E+04	232	5.13E-02
U ²³⁴	1.00E-03	8.93E+07	234	1.07E+00
U ²³⁵	3.15E-02	2.57E+11	235	1.16E-02
U ²³⁶	2.40E-02	8.55E+09	236	2.65E-01
U ²³⁸	9.45E-01	1.62E+12	238	5.47E-02
Total	1.00E+00			1.45E+00

^aBased on 0.171 metric tons of reenriched excess recovered (RU) uranium from reprocessing 1 MTIHM uranium spent fuel.

TABLE 7-8
SOURCE TERM 8, AFC WITH 1 AND 2 MOX RECYCLES: SOURCES FOR
PRODUCTION OF FIRST GENERATION MOX LWR FUEL^a

Radionuclide	Kilograms	Percent	Half-life (d)	Atomic Mass	Total Curies
Pu ²³⁸	4.70E-01	0.28	3.20E+04	2.38E+02	1.17E+04
Pu ²³⁹	6.39E+00	3.79	8.79E+06	2.39E+02	5.77E+02
Pu ²⁴⁰	3.06E+00	1.82	2.39E+06	2.40E+02	1.01E+03
Pu ²⁴¹	1.45E+00	0.86	5.26E+03	2.41E+02	2.17E+05
Pu ²⁴²	1.05E+00	0.63	1.37E+08	2.42E+02	6.01E+00
U ²³²	2.56E-06	0.00	2.63E+04	2.32E+02	5.45E-02
U ²³⁴	1.85E-01	0.08	8.93E+07	2.34E+02	1.15E+00
U ²³⁵	5.87E+00	2.40	2.57E+11	2.35E+02	1.27E-02
U ²³⁶	4.52E+00	1.85	8.55E+09	2.36E+02	2.93E-01
U ²³⁸	2.22E+02	90.58	1.62E+12	2.38E+02	7.51E-02
Am ²⁴¹	7.23E-02	0.04	1.58E+05	2.41E+02	3.59E+02
Total	2.45E+02	100			2.31E+05

^aBased on 0.2445 MTIHM.

TABLE 7-9
SOURCE TERM 9, AFC WITH 1 AND 2 MOX RECYCLES: SOURCES FOR RU
ENRICHMENT TAILS CONVERTED TO U₃O₈^a

Radionuclide	Fraction of Total U	Half-life (d)	Atomic Mass	Total Curies
U ²³²	1.00E-10	2.63E+04	232	1.47E-03
U ²³⁴	4.00E-05	8.93E+07	234	1.71E-01
U ²³⁵	2.50E-03	2.57E+11	235	3.71E-03
U ²³⁶	3.70E-03	8.55E+09	236	1.64E-01
U ²³⁸	9.94E-01	1.62E+12	238	2.31E-01
Total	1.00E+00			5.73E-01

^aBased on 0.685 metric tons of depleted recovered uranium (RU).

TABLE 7-10
SOURCE TERM 10, AFC WITH 1 MOX RECYCLE: SOURCES FOR CONVERSION OF
RU (EXCESS FROM MOX-1 REPROCESSING) TO U₃O₈^a

Radionuclide	Fraction of Total U	Half-life (d)	Atomic Mass	Total Curies
U ²³²	1.53E-08	2.63E+04	232	5.55E-02
U ²³⁴	5.87E-04	8.93E+07	234	6.23E-01
U ²³⁵	8.55E-03	2.57E+11	235	3.14E-03
U ²³⁶	1.78E-02	8.55E+09	236	1.95E-01
U ²³⁸	9.73E-01	1.62E+12	238	5.60E-02
Total	1.00E+00			9.33E-01

^aBased on 0.1697 metric tons excess recovered uranium (RU) from reprocessing MOX-1 spent fuel.

TABLE 7-11
SOURCE TERM 11, AFC WITH 1 MOX RECYCLE: SOURCES FOR FAST REACTOR FIRST CORE FABRICATION^{a, b}

Radionuclide	Grams from 0.2455 MTIHM of MOX-1	Fraction of Total Mass	Fractional Curies	Grams in 59.4 kg of FR Fuel	Fraction of Total Mass	Fractional Curies	Grams in 565 kg of FR Fuel	Fraction of Total Mass	Total Curies ^c
U ²³²	3.31E-03	1.45E-08	3.10E-07	7.18E-04	1.21E-08	2.59E-07	6.83E-03	1.21E-08	1.46E-01
U ²³⁴	1.27E+02	5.55E-04	3.47E-06	2.76E+01	4.64E-04	2.90E-06	2.62E+02	4.64E-04	1.64E+00
U ²³⁵	1.85E+03	8.09E-03	1.75E-08	4.01E+02	6.76E-03	1.46E-08	3.82E+03	6.76E-03	8.25E-03
U ²³⁶	3.86E+03	1.69E-02	1.09E-06	8.37E+02	1.41E-02	9.12E-07	7.96E+03	1.41E-02	5.15E-01
U ²³⁸	2.11E+05	9.20E-01	3.12E-07	4.57E+04	7.69E-01	2.60E-07	4.34E+05	7.69E-01	1.47E-01
Total U ^d	2.17E+05	9.46E-01		4.69E+04	7.90E-01		4.46E+05	7.90E-01	
Np ²³⁷	1.82E+03	7.93E-03	5.59E-06	1.82E+03	3.06E-02	2.16E-05	1.73E+04	3.06E-02	1.22E+01
Pu ²³⁸	7.64E+02	3.34E-03	5.72E-02	7.64E+02	1.29E-02	2.21E-01	7.27E+03	1.29E-02	1.25E+05
Pu ²³⁹	2.88E+03	1.26E-02	7.81E-04	2.88E+03	4.84E-02	3.01E-03	2.74E+04	4.84E-02	1.70E+03
Pu ²⁴⁰	2.41E+03	1.05E-02	2.40E-03	2.41E+03	4.06E-02	9.25E-03	2.29E+04	4.06E-02	5.23E+03
Pu ²⁴¹	1.09E+03	4.74E-03	4.89E-01	1.09E+03	1.83E-02	1.88E+00	1.03E+04	1.83E-02	1.06E+06
Pu ²⁴²	1.44E+03	6.27E-03	2.47E-05	1.44E+03	2.42E-02	9.52E-05	1.37E+04	2.42E-02	5.38E+01
Total Np+Pu	1.04E+04	4.54E-02		1.04E+04	1.75E-01		9.88E+04	1.75E-01	
Am ²⁴¹	9.04E+02	3.95E-03	1.35E-02	9.03E+02	1.52E-02	5.22E-02	8.59E+03	1.52E-02	2.95E+04
Am ²⁴³	7.47E+02	3.26E-03	6.49E-04	7.47E+02	1.26E-02	2.50E-03	7.10E+03	1.26E-02	1.41E+03
Cm ²⁴³	2.53E+00	1.11E-05	5.71E-04	2.53E+00	4.26E-05	2.20E-03	2.41E+01	4.26E-05	1.24E+03
Cm ²⁴⁴	3.84E+02	1.68E-03	1.36E-01	3.84E+02	6.47E-03	5.24E-01	3.66E+03	6.47E-03	2.96E+05
Cm ²⁴⁵	4.10E+01	1.79E-04	3.08E-05	4.10E+01	6.91E-04	1.19E-04	3.90E+02	6.91E-04	6.72E+01
Cm ²⁴⁶	5.33E+00	2.33E-05	7.15E-06	5.33E+00	8.97E-05	2.75E-05	5.07E+01	8.97E-05	1.55E+01
Total Am+Cm	2.08E+03	9.10E-03		2.08E+03	3.51E-02		1.98E+04	3.51E-02	
Overall Total	2.29E+05			5.94E+04			5.65E+05		1.52E+06

^aBased on 0.2455 metric tons of LWR MOX-1 spent fuel, where 59.4 kg of fast reactor fuel is produced from the products of reprocessing 245.5 kg of LWR MOX-1 spent fuel.

^bIncludes minor actinides from reprocessing 1 MTIHM of UO₂ spent fuel.

^cThis mixture is used for the analysis of Source Term 11 impacts.

^dAll of the uranium is not used -- 169.7 kg is excess from the product of reprocessing 245.5 kg of MOX-1 LWR spent fuel.

TABLE 7-12
SOURCE TERM 12, AFC WITH 1 MOX RECYCLE: SOURCES FOR FAST REACTOR RECYCLE CORE FABRICATION^{a, b}

Radionuclide in Spent Fuel	(Grams)	Fraction of Total Mass	Fractional Curies	(Grams)	Fraction of Total Mass	Fractional Curies	Grams in 565 kg of FR Fuel	Fraction of Total Mass	Total Curies ^c
U ²³²	3.31E-03	1.45E-08	3.10E-07	1.36E-01	2.83E-07	6.06E-06	1.60E-01	2.83E-07	3.42E+00
U ²³⁴	1.27E+02	5.55E-04	3.47E-06	9.07E+02	1.89E-03	1.18E-05	1.07E+03	1.89E-03	6.67E+00
U ²³⁵	1.85E+03	8.09E-03	1.75E-08	5.79E+02	1.21E-03	2.62E-09	6.83E+02	1.21E-03	1.48E-03
U ²³⁶	3.86E+03	1.69E-02	1.09E-06	6.01E+03	1.26E-02	8.13E-07	7.09E+03	1.26E-02	4.59E-01
U ²³⁸	2.11E+05	9.20E-01	3.12E-07	3.71E+05	7.74E-01	2.62E-07	4.38E+05	7.74E-01	1.48E-01
Total U ^d	2.17E+05	9.46E-01		3.78E+05	7.90E-01		4.46E+05	7.90E-01	
Np ²³⁷	1.82E+03	7.93E-03	5.59E-06	6.57E+03	1.37E-02	9.67E-06	7.75E+03	1.37E-02	5.46E+00
Pu ²³⁸	7.64E+02	3.34E-03	5.72E-02	7.17E+03	1.50E-02	2.57E-01	8.46E+03	1.50E-02	1.45E+05
Pu ²³⁹	2.88E+03	1.26E-02	7.81E-04	3.98E+04	8.32E-02	5.17E-03	4.70E+04	8.32E-02	2.92E+03
Pu ²⁴⁰	2.41E+03	1.05E-02	2.40E-03	2.20E+04	4.60E-02	1.05E-02	2.60E+04	4.60E-02	5.93E+03
Pu ²⁴¹	1.09E+03	4.74E-03	4.89E-01	3.28E+03	6.84E-03	7.04E-01	3.86E+03	6.84E-03	3.98E+05
Pu ²⁴²	1.44E+03	6.27E-03	2.47E-05	1.00E+04	2.09E-02	8.24E-05	1.18E+04	2.09E-02	4.66E+01
Total Np+Pu	1.04E+04	4.54E-02		8.89E+04	1.86E-01		1.05E+05	1.86E-01	
Am ²⁴¹	9.04E+02	3.95E-03	1.35E-02	3.22E+03	6.73E-03	2.31E-02	3.80E+03	6.73E-03	1.31E+04
Am ²⁴³	7.47E+02	3.26E-03	6.49E-04	4.48E+03	9.35E-03	1.86E-03	5.28E+03	9.35E-03	1.05E+03
Cm ²⁴²	2.53E+00	1.11E-05	3.66E-02	6.57E+00	1.37E-05	4.54E-02	7.75E+00	1.37E-05	2.57E+04
Cm ²⁴³	3.84E+02	1.68E-03	8.68E-02	2.07E+01	4.32E-05	2.23E-03	2.44E+01	4.32E-05	1.26E+03
Cm ²⁴⁴	4.10E+01	1.79E-04	1.45E-02	3.17E+03	6.62E-03	5.36E-01	3.74E+03	6.62E-03	3.03E+05
Cm ²⁴⁵	5.33E+00	2.33E-05	4.00E-06	6.27E+02	1.31E-03	2.25E-04	7.40E+02	1.31E-03	1.27E+02
Cm ²⁴⁶	4.60E+00	2.01E-05	6.17E-06	1.67E+02	3.48E-04	1.07E-04	1.96E+02	3.48E-04	6.05E+01
Total Am+Cm	2.09E+03	9.12E-03		1.17E+04	2.44E-02		1.38E+04	2.44E-02	
Overall Total	2.29E+05			4.79E+05			5.65E+05		8.96E+05

^aBased on producing 0.479 metric tons of fast reactor fuel using makeup derived from reprocessing LWR MOX-1 spent fuel.

^bUsing the amount of actinides from 0.2445 metric ton of MOX-1 fuel.

^cThis mixture is used for the analysis of Source Term 12 impacts.

^dAll of the uranium is not used -- 176 kg is excess from the product of reprocessing 245.5 kg of MOX-1 LWR spent fuel.

Five source terms (in addition to Source Terms 1 and 2) are identified for the UREX+3c fuel cycle with two recycles of spent fuel, as shown in Figure 7-3. Source Terms 1 and 2 (listed in Tables 7-1 and 7-2) for the UREX+3c fuel cycle with one recycle of LWR spent fuel are the same for the UREX+3c fuel cycle with two recycles of spent fuel.

Source Term 13, listed in Table 7-13, identifies the amounts and radioactivity of 818 kg of uranium isotopes in excess uranium recovered from reprocessing 1 MTIHM of UO_2 (0.8315 MTU) plus MOX-1 (0.1685 MTIHM) spent fuel. The source's radioactivity is used to assess the radiological health impacts of converting the recovered uranium to U_3O_8 for storage or disposal.

Source Term 14, listed in Table 7-14, is for 226.3 kg of plutonium, uranium, and americium radionuclides recovered from co-reprocessing UO_2 and MOX-1 spent fuel and used to produce the second generation of MOX LWR fuel (MOX-2).

Source Term 15, listed in Table 7-15, is for 123.4 kg of uranium that is recovered from MOX-2 spent fuel that is not needed for production of FR fuel. Reprocessing of the MOX-2 spent fuel in this case is done to recover uranium, plutonium, and minor actinides for production of FR fuel. The excess uranium is converted to U_3O_8 for storage or disposal.

Source Term 16, listed in Table 7-16, represents 92 kg of uranium, neptunium, plutonium, curium, and americium sources for fabrication of nuclear fuel for first cores of FRs. This source term comes from the products of reprocessing MOX-2 and the minor actinides recovered from the preceding steps of reprocessing UO_2 and UO_2 /MOX-1 spent fuels, and is adjusted linearly to 565 kg as a base.

Source Term 17, the last source term for the AFC with two recycles of LWR spent fuel, is listed in Table 7-17. This source term includes 741 kg of uranium, neptunium, plutonium, curium, and americium sources for producing recycle FR fuel, and is adjusted linearly to 565 kg as a base. The difference between this source term and the one presented in Table 7-16 is that this source term includes the uranium, plutonium, and minor actinides from reprocessed MOX-2 and minor actinides recovered from the preceding steps of reprocessing UO_2 and UO_2 /MOX-1 spent fuels as well as the uranium, plutonium and minor actinides recovered from reprocessing/recycling spent FR fuel.

TABLE 7-13
SOURCE TERM 13, UREX+3c FUEL CYCLE WITH 2 MOX RECYCLES:
RADIONUCLIDE SOURCES IN RU (EXCESS FROM UO₂ REPROCESSING) FOR
CONVERSION TO U₃O₈^a

Radionuclide	Fraction of Total U	Half-life (d)	Atomic Mass	Total Curies
U ²³²	3.33E-09	2.63E+04	232	5.83E-02
U ²³⁴	2.31E-04	8.93E+07	234	1.18E+00
U ²³⁵	7.44E-03	2.57E+11	235	1.32E-02
U ²³⁶	7.01E-03	8.55E+09	236	3.71E-01
U ²³⁸	9.85E-01	1.62E+12	238	2.73E-01
Total	1.00E+00			1.89E+00

^aBased on 0.818 metric tons of excess recovered uranium (RU) from reprocessing 1 MTIHM uranium spent fuel.

TABLE 7-14
SOURCE TERM 14, UREX+3c FUEL CYCLE WITH 2 MOX RECYCLES: SOURCES FOR
PRODUCTION OF SECOND GENERATION MOX LWR FUEL^a

Radionuclide	Kilograms	Percent	Half-life (d)	Atomic Mass	Total Curies
Pu ²³⁸	8.20E-01	0.36	3.20E+04	2.38E+02	1.40E+04
Pu ²³⁹	8.27E+00	3.65	8.79E+06	2.39E+02	5.14E+02
Pu ²⁴⁰	5.02E+00	2.21	2.39E+06	2.40E+02	1.14E+03
Pu ²⁴¹	2.31E+00	1.02	5.26E+03	2.41E+02	2.38E+05
Pu ²⁴²	2.16E+00	0.95	1.37E+08	2.42E+02	8.49E+00
U ²³²	1.29E-07	0.00	2.63E+04	2.32E+02	2.76E-03
U ²³⁴	2.16E-02	0.01	8.93E+07	2.34E+02	1.35E-01
U ²³⁵	9.82E-01	0.43	2.57E+11	2.35E+02	2.12E-03
U ²³⁶	6.56E-01	0.29	8.55E+09	2.36E+02	4.23E-02
U ²³⁸	2.06E+02	91.02	1.62E+12	2.38E+02	6.97E-02
Am ²⁴¹	1.24E-01	0.05	1.58E+05	2.41E+02	4.25E+02
Total	2.26E+02	100			2.54E+05

^aBased on 0.2263 MTIHM.

TABLE 7-15
SOURCE TERM 15, UREX+3c FUEL CYCLE WITH 2 MOX RECYCLES: SOURCES FOR
CONVERSION OF RU (EXCESS FROM MOX-2 REPROCESSING) TO U_3O_8 ^a

Radionuclide	Fraction of total U	Half-life (d)	Atomic Mass	Total Curies
U^{232}	6.90E-09	2.63E+04	232	1.83E-02
U^{234}	1.28E-04	8.93E+07	234	9.87E-02
U^{235}	1.99E-03	2.57E+11	235	5.31E-04
U^{236}	2.88E-03	8.55E+09	236	2.30E-02
U^{238}	9.95E-01	1.62E+12	238	4.16E-02
Total	1.00E+00			1.82E-01

^aBased on 0.1234 metric tons of excess recovered uranium (RU) from reprocessing MOX-2 spent fuel.

TABLE 7-16
SOURCE TERM 16, UREX+3c FUEL CYCLE WITH 2 MOX RECYCLES: SOURCES FOR FAST REACTOR
FIRST CORE FABRICATION^{a, b}

Radionuclide	Grams from 0.2263 MTIHM of MOX-1	Fraction of Total Mass	Fractional Curies	Grams in 92 kg of FR Fuel	Fraction of Total Mass	Fractional Curies	Grams in 565 kg of FR Fuel	Fraction of Total Mass	Total Curies ^c
U ²³²	1.35E-03	6.28E-09	1.34E-07	5.01E-04	5.45E-09	1.17E-07	3.08E-03	5.45E-09	6.61E-02
U ²³⁴	5.32E+01	2.47E-04	1.55E-06	1.97E+01	2.14E-04	1.34E-06	1.21E+02	2.14E-04	7.57E-01
U ²³⁵	3.90E+02	1.82E-03	3.93E-09	1.45E+02	1.57E-03	3.40E-09	8.89E+02	1.57E-03	1.92E-03
U ²³⁶	5.66E+02	2.63E-03	1.70E-07	2.10E+02	2.28E-03	1.48E-07	1.29E+03	2.28E-03	8.36E-02
U ²³⁸	1.95E+05	9.07E-01	3.07E-07	7.23E+04	7.86E-01	2.66E-07	4.44E+05	7.86E-01	1.50E-01
Total U ^d	1.96E+05	9.12E-01		7.27E+04	7.90E-01		4.46E+05	7.89E-01	
Np ²³⁷	2.03E+03	9.42E-03	6.64E-06	2.03E+03	2.20E-02	1.55E-05	1.24E+04	2.20E-02	8.76E+00
Pu ²³⁸	7.17E+02	3.34E-03	5.72E-02	7.17E+02	7.80E-03	1.34E-01	4.40E+03	7.80E-03	7.57E+04
Pu ²³⁹	4.29E+03	1.99E-02	1.24E-03	4.29E+03	4.66E-02	2.90E-03	2.63E+04	4.66E-02	1.64E+03
Pu ²⁴⁰	3.68E+03	1.71E-02	3.90E-03	3.68E+03	4.00E-02	9.11E-03	2.26E+04	4.00E-02	5.15E+03
Pu ²⁴¹	1.75E+03	8.13E-03	8.37E-01	1.75E+03	1.90E-02	1.96E+00	1.07E+04	1.90E-02	1.11E+06
Pu ²⁴²	2.24E+03	1.04E-02	4.11E-05	2.24E+03	2.44E-02	9.60E-05	1.38E+04	2.44E-02	5.42E+01
Total Np+Pu	1.47E+04	6.84E-02		1.47E+04	1.60E-01		9.03E+04	1.60E-01	
Am ²⁴¹	2.00E+03	9.32E-03	3.20E-02	2.00E+03	2.18E-02	7.47E-02	1.23E+04	2.18E-02	4.22E+04
Am ²⁴³	1.61E+03	7.50E-03	1.49E-03	1.61E+03	1.75E-02	3.49E-03	9.91E+03	1.75E-02	1.97E+03
Cm ²⁴³	5.81E+00	2.70E-05	1.40E-03	5.82E+00	6.32E-05	3.27E-03	3.57E+01	6.32E-05	1.85E+03
Cm ²⁴⁴	8.68E+02	4.04E-03	3.27E-01	8.68E+02	9.44E-03	7.64E-01	5.33E+03	9.44E-03	4.32E+05
Cm ²⁴⁵	1.16E+02	5.40E-04	9.29E-05	1.16E+02	1.26E-03	2.17E-04	7.13E+02	1.26E-03	1.23E+02
Cm ²⁴⁶	1.20E+01	5.56E-05	1.71E-05	1.20E+01	1.30E-04	3.99E-05	7.35E+01	1.30E-04	2.25E+01
Total Am+Cm	4.62E+03	2.15E-02		4.62E+03	5.02E-02		2.84E+04	5.02E-02	
Overall Total	2.15E+05			9.20E+04			5.65E+05		1.67E+06

^aBased on 0.2263 MTIHM of LWR MOX-2 spent fuel, where 92 kg of fast reactor fuel is produced from the products of reprocessing 226.3 kg of LWR MOX-2 spent fuel, with 123.4 kg of uranium excess.

^bIncludes minor actinides from reprocessing 1 MTIHM of UO₂ spent fuel and co-reprocessing 1 MTIHM of UO₂/MOX-1 spent fuel.

^cThis mixture is used for the analysis of Source Term 16 impacts.

^dAll of the uranium is not used -- 123.4 kg is excess from the product of reprocessing 226.3 kg of MOX-1 LWR spent fuel.

TABLE 7-17
SOURCE TERM 17, UREX+3c FUEL CYCLE WITH 2 MOX RECYCLES: SOURCES FOR FAST REACTOR RECYCLE
CORE FABRICATION^{a,b}

Radionuclide in Spent Fuel	(Grams)	Fraction of Total Mass	Fractional Curies	(Grams)	Fraction of Total Mass	Fractional Curies	Grams in 565 kg of FR fuel	Fraction of Total Mass	Total Curies ^c
U ²³²	1.35E-03	6.28E-09	1.34E-07	1.31E-01	1.77E-07	3.78E-06	1.00E-01	1.77E-07	2.14E+00
U ²³⁴	5.32E+01	2.47E-04	1.55E-06	1.06E+03	1.43E-03	8.94E-06	8.12E+02	1.44E-03	5.07E+00
U ²³⁵	3.90E+02	1.81E-03	3.92E-09	2.50E+02	3.37E-04	7.30E-10	1.90E+02	3.36E-04	4.11E-04
U ²³⁶	5.66E+02	2.63E-03	1.70E-07	1.59E+03	2.15E-03	1.39E-07	1.21E+03	2.14E-03	7.85E-02
U ²³⁸	1.95E+05	9.07E-01	3.07E-07	5.82E+05	7.85E-01	2.66E-07	4.44E+05	7.86E-01	1.50E-01
Total U ^d	1.96E+05	9.12E-01		5.85E+05	7.89E-01		4.46E+05	7.89E-01	
Np ²³⁷	2.03E+03	9.44E-03	6.65E-06	6.60E+03	8.91E-03	6.28E-06	5.03E+03	8.90E-03	3.54E+00
Pu ²³⁸	7.17E+02	3.33E-03	5.72E-02	8.83E+03	1.19E-02	2.04E-01	6.73E+03	1.19E-02	1.15E+05
Pu ²³⁹	4.29E+03	2.00E-02	1.24E-03	6.18E+04	8.34E-02	5.18E-03	4.71E+04	8.34E-02	2.93E+03
Pu ²⁴⁰	3.68E+03	1.71E-02	3.90E-03	3.45E+04	4.66E-02	1.06E-02	2.63E+04	4.65E-02	5.99E+03
Pu ²⁴¹	1.75E+03	8.14E-03	8.39E-01	5.18E+03	6.99E-03	7.20E-01	3.95E+03	6.99E-03	4.07E+05
Pu ²⁴²	2.24E+03	1.04E-02	4.10E-05	1.59E+04	2.15E-02	8.45E-05	1.21E+04	2.14E-02	4.77E+01
Total Np+Pu	1.47E+04	6.84E-02		1.33E+05	1.79E-01		1.01E+05	1.79E-01	
Am ²⁴¹	2.00E+03	9.30E-03	3.19E-02	6.28E+03	8.48E-03	2.91E-02	4.79E+03	8.48E-03	1.64E+04
Am ²⁴³	1.61E+03	7.49E-03	1.49E-03	8.43E+03	1.14E-02	2.26E-03	6.43E+03	1.14E-02	1.28E+03
Cm ²⁴²	0.00E+00	0.00E+00	0.00E+00	1.25E+01	1.69E-05	5.59E-02	9.51E+00	1.68E-05	3.15E+04
Cm ²⁴³	5.81E+00	2.70E-05	1.40E-03	4.18E+01	5.64E-05	2.92E-03	3.19E+01	5.65E-05	1.65E+03
Cm ²⁴⁴	8.68E+02	4.04E-03	3.27E-01	6.39E+03	8.62E-03	6.98E-01	4.87E+03	8.62E-03	3.94E+05
Cm ²⁴⁵	1.16E+02	5.40E-04	9.28E-05	1.33E+03	1.79E-03	3.09E-04	1.02E+03	1.81E-03	1.75E+02
Cm ²⁴⁶	1.20E+01	5.58E-05	1.71E-05	3.59E+02	4.84E-04	1.49E-04	2.74E+02	4.85E-04	8.42E+01
Total Am+Cm	4.62E+03	2.15E-02		2.28E+04	3.08E-02		1.74E+04	3.08E-02	
Overall Total	2.15E+05			7.41E+05			5.64E+05		9.77E+05

^aBased on producing 0.741 MTIHM of fast reactor fuel using makeup derived from reprocessing LWR MOX-2.

^bUsing the amount of actinides from 0.2263 MTIHM of MOX-2 and minor actinides from reprocessing 1 MTIHM of UO₂ spent fuel and co-reprocessing 1 MTIHM of UO₂/MOX-1 spent fuel.

^cThis mixture is used for the analysis of Source Term 17 impacts.

^dAll of the uranium is not used -- 134 kg is excess from the product of reprocessing 226.3 kg of MOX-2 LWR spent fuel.

Finally, seven additional source terms are identified for the AFC with two MOX reprocessing steps, as shown in Figure 7-4. Four Source Terms, 6, 7, 8, and 9 (Tables 7-6, through 7-9) are the same as defined for the AFC with one recycle of LWR spent fuel.

Source Term 18, listed in Table 7-18, represents 824 kg of uranium isotopes recovered from co-reprocessing spent UO_2 and MOX-1 fuels that are feed to reenrichment of recovered uranium. This reenriched recovered uranium is blended with the U:Pu product of reprocessing to produce second generation MOX (MOX-2) LWR fuel.

Source Term 19, listed in Table 7-19, is for 171 kg of enriched recycled uranium.

Source Term 20, listed in Table 7-20, is for 278.8 kg of plutonium, uranium, and americium radionuclides recovered from co-reprocessing UO_2 and MOX-1 LWR spent fuel including the uranium isotopes from blending these materials with reenriched recovered uranium to produce MOX-2 LWR fuel.

Source Term 21, listed in Table 7-21, is for 653 kg of uranium isotopes in DRU from reenrichment of the RU recovered from co-processing UO_2 and MOX-1 spent fuel. This source term is for conversion of the DRU to U_3O_8 for storage or disposal.

Source Term 22, listed in Table 7-22, is for 164.2 kg of uranium that is recovered from reprocessing MOX-2 spent fuel but is not needed for the production of FR fuel. Reprocessing of the MOX-2 fuel is done in this case to recover uranium, plutonium, and minor actinides from the LWR spent fuel for use to produce FR fuel. The excess uranium is converted to U_3O_8 for storage or disposal.

Source Term 23, listed in Table 7-23, represents 101 kg of uranium, neptunium, plutonium, curium, and americium radionuclides for production of first cores for FRs. This source term comes from the products of reprocessing MOX-2 and minor actinides recovered from the preceding steps of reprocessing UO_2 and UO_2 /MOX-1 spent fuels, and is adjusted linearly to 565 kg as a base.

Source Term 24, the last source term for the fourth scenario, listed in Table 7-24. This source term also includes 809 kg of uranium, neptunium, plutonium, curium, and americium sources for producing FR fuel, and is adjusted linearly to 565 kg as a base. The difference between this source term and the one presented in Table 7-23 is that this source term includes the uranium, plutonium, and minor actinides from reprocessed MOX-2 and minor actinides recovered from the preceding steps of reprocessing UO_2 and UO_2 /MOX-1 spent fuels as well as

the uranium, plutonium and minor actinides recovered from reprocessing/recycling spent FR fuel.

TABLE 7-18
SOURCE TERM 18, AFC WITH 2 MOX RECYCLES: SOURCES FOR RU FEED TO ENRICHMENT FROM UO₂/MOX-1 CO-REPROCESSING FOR CONVERSION TO UF₆^a

Radionuclide	Fraction of total U	Half-life (d)	Atomic Mass	Total Curies
U ²³²	5.27E-09	2.63E+04	232	9.31E-02
U ²³⁴	3.17E-04	8.93E+07	234	1.63E+00
U ²³⁵	8.35E-03	2.57E+11	235	1.48E-02
U ²³⁶	1.01E-02	8.55E+09	236	5.41E-01
U ²³⁸	9.82E-01	1.62E+12	238	2.74E-01
Total	1.00E+00			2.55E+00

^aBased on 0.824 metric tons of recovered uranium (RU) from co-reprocessing 1 MTIHM UO₂/MOX-1 spent fuel.

TABLE 7-19
SOURCE TERM 19, AFC WITH 2 MOX RECYCLES: SOURCES FOR RU ENRICHMENT PRODUCT^a

Radionuclide	Fraction of total U	Half-life (d)	Atomic Mass	Total Curies
U ²³²	2.45E-08	2.63E+04	232	8.96E-02
U ²³⁴	1.32E-03	8.93E+07	234	1.41E+00
U ²³⁵	3.07E-02	2.57E+11	235	1.14E-02
U ²³⁶	3.02E-02	8.55E+09	236	3.35E-01
U ²³⁸	9.40E-01	1.62E+12	238	5.44E-02
Total	1.00E+00			1.90E+00

^aBased on 0.171 metric tons of reenriched recovered uranium (RU) from co-reprocessing 1 MTIHM UO₂/MOX-1 spent fuel.

TABLE 7-20
SOURCE TERM 20, AFC WITH 2 MOX RECYCLES: SOURCES FOR PRODUCTION OF
SECOND GENERATION MOX LWR FUEL^a

Radionuclide	Kilograms	Percent	Half-life (d)	Atomic Mass	Total Curies
Pu ²³⁸	1.12E+00	0.40	3.20E+04	238	1.92E+04
Pu ²³⁹	7.73E+00	2.77	8.79E+06	239	4.80E+02
Pu ²⁴⁰	4.73E+00	1.70	2.39E+06	240	1.08E+03
Pu ²⁴¹	2.18E+00	0.78	5.26E+03	241	2.25E+05
Pu ²⁴²	2.24E+00	0.80	1.37E+08	242	8.81E+00
U ²³²	4.67E-06	0.00	2.63E+04	232	1.00E-01
U ²³⁴	2.52E-01	0.09	8.93E+07	234	1.58E+00
U ²³⁵	5.98E+00	2.14	2.57E+11	235	1.29E-02
U ²³⁶	6.06E+00	2.17	8.55E+09	236	3.93E-01
U ²³⁸	2.48E+02	89.09	1.62E+12	238	8.42E-02
Am ²⁴¹	1.18E-01	0.04	1.58E+05	241	4.04E+02
Total	2.79E+02	100			2.46E+05

^aBased on 0.2788 MTIHM.

TABLE 7-21
SOURCE TERM 21, AFC WITH 2 MOX RECYCLES: SOURCES FOR CONVERSION OF
TAILS TO U₃O₈ WHERE THE TAILS ORIGINATED FROM REENRICHMENT OF RU FOR
USE IN PRODUCTION OF MOX-2 FUEL^a

Radionuclide	Fraction of total U	Half-life (d)	Atomic Mass	Total Curies
U ²³²	2.00E-10	2.63E+04	232	2.80E-03
U ²³⁴	5.00E-05	8.93E+07	234	2.04E-01
U ²³⁵	2.50E-03	2.57E+11	235	3.53E-03
U ²³⁶	4.88E-03	8.55E+09	236	2.06E-01
U ²³⁸	9.93E-01	1.62E+12	238	2.20E-01
Total	1.00E+00			6.38E-01

^aBased on 0.653 metric tons of recovered uranium (RU) from reprocessing 1 MTIHM UO₂/MOX-1 spent fuel.

TABLE 7-22
SOURCE TERM 22, AFC WITH 2 MOX RECYCLES: SOURCES FOR CONVERSION OF
RU (EXCESS FROM MOX-2 REPROCESSING) TO U₃O₈^a

Radionuclide	Fraction of total U	Half-life (d)	Atomic Mass	Total Curies
U ²³²	2.09E-08	2.63E+04	232	7.34E-02
U ²³⁴	7.96E-04	8.93E+07	234	8.16E-01
U ²³⁵	9.73E-03	2.57E+11	235	3.45E-03
U ²³⁶	2.03E-02	8.55E+09	236	2.15E-01
U ²³⁸	9.70E-01	1.62E+12	238	5.39E-02
Total	1.00E+00			1.16E+00

^aBased on 0.1624 metric tons of excess recovered uranium (RU) from reprocessing MOX-2 spent fuel.

TABLE 7-23
SOURCE TERM 23, AFC WITH 2 MOX RECYCLES: SOURCES FOR FAST REACTOR FIRST CORE FABRICATION^{a, b}

Radionuclide	Grams from 0.2788 MTIHM of MOX-1	Fraction of Total Mass	Fractional Curies	Grams in 101.2 kg of FR Fuel	Fraction of Total Mass	Fractional Curies	Grams in 278.8 kg of FR Fuel	Fraction of Total Mass	Total Curies ^c
U ²³²	5.09E-03	1.92E-08	4.11E-07	1.67E-03	1.65E-08	3.54E-07	9.33E-03	1.65E-08	2.00E-01
U ²³⁴	1.94E+02	7.31E-04	4.57E-06	6.36E+01	6.30E-04	3.94E-06	3.55E+02	6.29E-04	2.22E+00
U ²³⁵	2.28E+03	8.61E-03	1.86E-08	7.49E+02	7.42E-03	1.60E-08	4.18E+03	7.41E-03	9.04E-03
U ²³⁶	4.94E+03	1.87E-02	1.21E-06	1.62E+03	1.61E-02	1.04E-06	9.07E+03	1.60E-02	5.88E-01
U ²³⁸	2.36E+05	8.90E-01	3.02E-07	7.75E+04	7.67E-01	2.60E-07	4.33E+05	7.66E-01	1.46E-01
Total U ^d	2.43E+05	9.18E-01		7.99E+04	7.92E-01		4.46E+05	7.90E-01	
Np ²³⁷	3.72E+03	1.41E-02	9.90E-06	3.73E+03	3.69E-02	2.60E-05	2.08E+04	3.68E-02	1.46E+01
Pu ²³⁸	1.32E+03	4.99E-03	8.55E-02	1.32E+03	1.31E-02	2.24E-01	7.38E+03	1.31E-02	1.27E+05
Pu ²³⁹	4.10E+03	1.55E-02	9.62E-04	4.10E+03	4.06E-02	2.53E-03	2.29E+04	4.06E-02	1.42E+03
Pu ²⁴⁰	3.48E+03	1.31E-02	2.99E-03	3.48E+03	3.44E-02	7.84E-03	1.94E+04	3.44E-02	4.42E+03
Pu ²⁴¹	1.65E+03	6.23E-03	6.42E-01	1.65E+03	1.63E-02	1.68E+00	9.22E+03	1.63E-02	9.49E+05
Pu ²⁴²	2.33E+03	8.79E-03	3.46E-05	2.33E+03	2.31E-02	9.09E-05	1.30E+04	2.30E-02	5.12E+01
Total Np+Pu	1.66E+04	6.27E-02		1.66E+04	1.64E-01		9.28E+04	1.64E-01	
²⁴¹ Am	1.91E+03	7.19E-03	2.47E-02	1.91E+03	1.89E-02	6.47E-02	1.06E+04	1.88E-02	3.65E+04
²⁴³ Am	1.69E+03	6.38E-03	1.27E-03	1.69E+03	1.68E-02	3.34E-03	9.45E+03	1.67E-02	1.88E+03
²⁴³ Cm	5.65E+00	2.13E-05	1.10E-03	5.65E+00	5.60E-05	2.89E-03	3.16E+01	5.58E-05	1.63E+03
²⁴⁴ Cm	9.12E+02	3.44E-03	2.79E-01	9.13E+02	9.04E-03	7.32E-01	5.10E+03	9.02E-03	4.12E+05
²⁴⁵ Cm	1.09E+02	4.11E-04	7.06E-05	1.09E+02	1.08E-03	1.85E-04	6.08E+02	1.08E-03	1.05E+02
²⁴⁶ Cm	1.27E+01	4.78E-05	1.47E-05	1.27E+01	1.25E-04	3.85E-05	7.07E+01	1.25E-04	2.17E+01
Total Am+Cm	4.64E+03	1.75E-02		4.64E+03	4.59E-02		2.59E+04	4.58E-02	
Overall Total	2.65E+05			1.01E+05			5.65E+05		1.53E+06

^aBased on 0.2788 MTIHM of LWR MOX-2 spent fuel, where 101.2 kg of fast reactor fuel is produced from the products of reprocessing 278.8 kg of LWR MOX-2 spent fuel.

^bIncludes minor actinides from reprocessing 1 MTIHM of UO₂ spent fuel, and co-reprocessing 1 MTIHM of UO₂/MOX-1 spent fuel.

^cThis mixture is used for the analysis of Source Term 23 impacts.

^dAll of the uranium is not used -- 164.2kg is excess from the product of reprocessing 278.8 kg of MOX-2 LWR spent fuel.

TABLE 7-24
SOURCE TERM 24, AFC WITH 2 MOX RECYCLES: SOURCES FOR FAST REACTOR RECYCLE CORE FABRICATION^{a, b}

Radionuclide in Spent Fuel	(Grams)	Fraction of Total Mass	Fractional Curies	(Grams)	Fraction of Total Mass	Fractional Curies	Grams in 565 kg of FR Fuel	Fraction of Total Mass	Total Curies ^c
U ²³²	5.09E-03	1.92E-08	4.11E-07	2.72E-01	3.36E-07	7.20E-06	1.90E-01	3.36E-07	4.07E+00
U ²³⁴	1.94E+02	7.31E-04	4.57E-06	1.75E+03	2.17E-03	1.35E-05	1.22E+03	2.17E-03	7.63E+00
U ²³⁵	2.28E+03	8.61E-03	1.86E-08	1.08E+03	1.33E-03	2.88E-09	7.52E+02	1.33E-03	1.63E-03
U ²³⁶	4.94E+03	1.87E-02	1.21E-06	1.15E+04	1.43E-02	9.22E-07	8.05E+03	1.43E-02	5.21E-01
U ²³⁸	2.36E+05	8.90E-01	3.02E-07	6.25E+05	7.72E-01	2.62E-07	4.36E+05	7.72E-01	1.48E-01
Total U ^d	2.43E+05	9.18E-01		6.39E+05	7.90E-01		4.46E+05	7.90E-01	
Np ²³⁷	3.72E+03	1.41E-02	9.90E-06	1.28E+04	1.59E-02	1.12E-05	8.96E+03	1.59E-02	6.33E+00
Pu ²³⁸	1.32E+03	4.99E-03	8.55E-02	1.33E+04	1.65E-02	2.83E-01	9.31E+03	1.65E-02	1.60E+05
Pu ²³⁹	4.10E+03	1.55E-02	9.62E-04	6.57E+04	8.12E-02	5.05E-03	4.59E+04	8.12E-02	2.85E+03
Pu ²⁴⁰	3.48E+03	1.31E-02	2.99E-03	3.43E+04	4.23E-02	9.64E-03	2.39E+04	4.23E-02	5.45E+03
Pu ²⁴¹	1.65E+03	6.23E-03	6.42E-01	4.97E+03	6.15E-03	6.33E-01	3.47E+03	6.15E-03	3.58E+05
Pu ²⁴²	2.33E+03	8.79E-03	3.46E-05	1.59E+04	1.97E-02	7.75E-05	1.11E+04	1.97E-02	4.38E+01
Total Np+Pu	1.66E+04	6.27E-02		1.47E+05	1.82E-01		1.03E+05	1.82E-01	
Am ²⁴¹	1.91E+03	7.19E-03	2.47E-02	5.94E+03	7.34E-03	2.52E-02	4.15E+03	7.34E-03	1.42E+04
Am ²⁴³	1.69E+03	6.38E-03	1.27E-03	8.63E+03	1.07E-02	2.12E-03	6.02E+03	1.07E-02	1.20E+03
Cm ²⁴²	5.65E+00	2.13E-05	7.06E-02	1.17E+01	1.45E-05	4.80E-02	8.19E+00	1.45E-05	2.71E+04
Cm ²⁴³	9.12E+02	3.44E-03	1.78E-01	3.95E+01	4.88E-05	2.52E-03	2.76E+01	4.88E-05	1.42E+03
Cm ²⁴⁴	1.09E+02	4.11E-04	3.32E-02	6.56E+03	8.11E-03	6.57E-01	4.58E+03	8.11E-03	3.71E+05
Cm ²⁴⁵	1.27E+01	4.78E-05	8.22E-06	1.35E+03	1.67E-03	2.87E-04	9.44E+02	1.67E-03	1.62E+02
Cm ²⁴⁶	1.10E+01	4.14E-05	1.27E-05	3.64E+02	4.50E-04	1.38E-04	2.54E+02	4.50E-04	7.80E+01
Total Am+Cm	4.65E+03	1.75E-02		2.29E+04	2.83E-02		1.60E+04	2.83E-02	
Overall Total	2.65E+05			8.09E+05			5.65E+05		9.41E+05

^aBased on producing 0.809 MTIHM of fast reactor fuel using makeup derived from reprocessing LWR MOX-2 spent fuel.

^bUsing the amount of actinides from 0.2785 MTIHM of MOX-2 and minor actinides from reprocessing 1 MTIHM of UO₂ spent fuel and co-reprocessing 1 MTIHM of UO₂/MOX-1 spent fuel.

^cThis mixture is used for the analysis of Source Term 24 impacts.

^dAll of the uranium is not used -- 173.7kg is excess from the product of reprocessing 278.8 kg of MOX-2 LWR spent fuel.

Table 7-25 correlates the twenty-four source terms listed in this section to the fuel cycle scenarios being analyzed, and the fuel cycle facilities where the sources would apply.

TABLE 7-25
CORRELATION OF THE RADIOLOGICAL SOURCE TERMS BY SOURCE TERM
NUMBER, TO FUEL CYCLE SCENARIO AND TYPES OF FACILITY

Type of Facility^a	Source Terms for UREX+3c Fuel Cycle: 1 Recycle	Source Terms for AFC: 1 Recycle	Source Terms for UREX+3c Fuel Cycle: 2 Recycles	Source Terms for AFC: 2 Recycles
U Product Conversion to U ₃ O ₈ for Storage or Disposal	1 3	9 10	1 13 15	9 21 22
Conversion of Reprocessed Uranium to UF ₆ for Re-Enrichment		6		6 18
Re-Enrichment of Recovered Uranium		7		7 19
MOX 1 or 2 Fuel Fabrication	2	8	2 14	8 20
Fast Reactor Fuel Fabrication	4 5	11 12	16 17	23 24

^aSources for nuclear fuel reprocessing are not included because they would be largely the same for the alternatives evaluated.

7.3 ANALYTICAL METHODS

This section describes the methods used to evaluate the potential for radiological health impacts to workers and to the public for the UREX+3c fuel cycle and the AFC, and for recycling LWR spent fuel one and two times in the fuel cycles.

7.3.1 Methods to Estimate Worker Radiological Health Impacts -- MicroShield[®] Analysis

As discussed above, the relative photon radiation doses to workers from the nuclear fuel cycle facilities considered in this study are estimated using the MicroShield[®] (Ref. 1) computer program. MicroShield[®] is a simple point kernel shielding code used for comprehensive photon/gamma ray shielding and dose assessment. It is widely used for designing shields, estimating source strength from radiation measurements, and analyzing radiation detector performance. This code can be used to evaluate facility designs, determine equipment shielding requirements, and to evaluate detector responses for different radiation detectors and shielding combinations.

For this analysis, photon radiation source terms input to the MicroShield[®] program were defined in terms of the activity by radionuclide in each mixture with no delay between reprocessing and other fuel cycle steps. The analysis used a standardized geometry, assuming one metric ton of each source term, for each fuel cycle facility. In this manner, the relative doses from each source term, for each facility, was determined, and through inspection of the various results, a general assessment of the extent to which shielded facilities would be required were made.

The standardized geometry considered in this study for the MicroShield[®] analysis is one metric ton of material in a simple cylinder, about 21.3 cm (8.4 inches) high, with a radius of 21.3 cm (8.4 inches). This cylinder is surrounded by a steel wall cladding, with a thickness of 0.318 cm (1/8 inch), which approximates conditions for small storage containers or pipes containing source materials. This is a reasonable standardized geometry since it should shield alpha and beta radiation (consistent with industrial designs and experience), while allowing the relative photon contributions to be evaluated.

7.3.2 Methods to Evaluate Public Radiological Health Impacts

Estimates of the radiological impacts to members of the public during both normal operation and for typical accident situations were made by considering the potential for inhalation doses resulting from airborne releases of the various source terms for the facilities considered. Although the quantities of radioactive materials will vary within a facility by process location, it is assumed that the radionuclide mixtures will not vary within a facility. This means that the relative doses per unit mixture for the source terms identified in this study can be used to evaluate the relative doses that might arise from those mixtures, and serve as the basis for a general assessment of the potential radiological impacts to members of the public. Current designs for nuclear facilities typically reduce offsite doses through a rigorous design process that includes as low as reasonably achievable (ALARA) evaluations for each component and process within a facility. Measures to mitigate or eliminate effluents to air and water have become standard in facility design to the point that the dose to maximally exposed individuals is typically on the order of a few millirem per year, or less. However, the relative comparison of source terms based on the potential for inhalation will help rank the potential effort involved in mitigating those doses and the potential importance of design basis accidents.

The method used to estimate inhalation doses for the source terms is to simply multiply the activity concentration of each radionuclide in each source term, by the inhalation dose conversion factors for each radionuclide, without considering the effects of dilution with downwind transport to the point of a receptor^{*}. The inhalation dose conversion factors used are those published by the U.S. Environmental Protection Agency in Federal Guidance Report No. 11 (Ref. 2), which are consistent with the current regulatory basis in the U.S. These dose conversion factors are shown in Table 7-26.

^{*} The reader is cautioned not to use the results obtained by multiplying dose conversion factors and concentrations of radionuclides in source terms to assess risks of radiological dose to members of the public. As discussed herein, the products of multiplying dose conversion factors by the concentrations of the radionuclides in each source term and combining the results are used only to provide a gauge of the radiological hazard associated therewith relative to the radiological hazards of other source terms and absent any mitigations. This gauge is used to support assessment of the scope and extent of measures that would be needed to safely handle and process the materials represented by the respective source terms and thus the relative cost of the associated facilities. Thus, the radiological hazard of a source term for FR fuel, which would contain significant concentrations of plutonium and minor actinides, is about 1-million times that of recycled uranium if mitigating measures are not considered. However, with required environmental controls in place, processing recycled uranium would present a very low radiological risk to members of the public. And, for processing of FR fuel, substantial and extensive mitigating measures would be effected to prevent releases of radioactive materials to the environment where public health and safety could be placed at risk. Thus, although the relative hazard of the two sources differs substantially, the risks to public health and safety associated with their handling and processing would not be.

TABLE 7-26
INHALATION DOSE CONVERSION FACTORS FROM FEDERAL GUIDANCE REPORT
NO. 11

Radionuclide	Inhalation Dose Conversion Factor (Sv/Bq)
U ²³²	3.85E-05
U ²³⁴	3.58E-05
U ²³⁵	3.23E-05
U ²³⁶	3.39E-05
U ²³⁸	3.20E-05
Np ²³⁷	1.46E-04
Pu ²³⁸	1.06E-04
Pu ²³⁹	1.16E-04
Pu ²⁴⁰	1.16E-04
Pu ²⁴¹	2.23E-06
Pu ²⁴²	1.11E-04
Am ²⁴¹	1.20E-04
Am ²⁴³	1.19E-04
Cm ²⁴²	4.67E-06
Cm ²⁴³	8.30E-05
Cm ²⁴⁴	7.60E-05
Cm ²⁴⁵	1.23E-04
Cm ²⁴⁶	1.22E-04

A summary of the source terms, in units of curies per unit mass, and the dose conversion factors for the radionuclides in those source terms, by type of facility, are shown in Tables 7-27 through 7-31.

TABLE 7-27
ACTIVITY CONCENTRATION SOURCE TERMS AND DOSE CONVERSION FACTORS FOR FACILITIES TO CONVERT
URANIUM INTO U₃O₈^a

Radionuclide	Dose Conversion Factor	Source Term 1	Source Term 3	Source Term 13	Source Term 15	Source Term 9	Source Term 10	Source Term 21	Source Term 22
	Sv/Bq	Curies per Unit Mass	Curies per Unit Mass	Curies per Unit Mass	Curies per Unit Mass	Curies per Unit Mass	Curies per Unit Mass	Curies per Unit Mass	Curies per Unit Mass
U ²³²	3.85E-05	6.01E-08	1.26E-07	7.13E-08	1.48E-07	2.14E-09	3.27E-07	4.28E-09	4.47E-07
U ²³⁴	3.58E-05	1.46E-06	1.39E-06	1.44E-06	8.00E-07	2.50E-07	3.67E-06	3.12E-07	4.97E-06
U ²³⁵	3.23E-05	1.79E-08	6.66E-09	1.61E-08	4.30E-09	5.41E-09	1.85E-08	5.41E-09	2.10E-08
U ²³⁶	3.39E-05	5.01E-07	2.09E-07	4.54E-07	1.86E-07	2.39E-07	1.15E-06	3.16E-07	1.31E-06
U ²³⁸	3.20E-05	3.33E-07	3.37E-07	3.34E-07	3.37E-07	3.37E-07	3.30E-07	3.36E-07	3.28E-07

^aUnit mass = 1 gram

TABLE 7-28
ACTIVITY CONCENTRATION SOURCE TERMS AND DOSE CONVERSION FACTORS FOR CONVERSION OF
REPROCESSED URANIUM TO UF₆ FOR ENRICHMENT^a

Radionuclide	Dose Conversion Factor	Source Term 6	Source Term 18
	Sv/Bq	Curies per Unit Mass	Curies per Unit Mass
U ²³²	3.85E-05	6.01E-08	1.13E-07
U ²³⁴	3.58E-05	1.46E-06	1.98E-06
U ²³⁵	3.23E-05	1.79E-08	1.80E-08
U ²³⁶	3.39E-05	5.01E-07	6.56E-07
U ²³⁸	3.20E-05	3.33E-07	3.32E-07

^aUnit mass = 1 gram

TABLE 7-29
ACTIVITY CONCENTRATION SOURCE TERMS AND DOSE CONVERSION FACTORS
FOR REENRICHMENT WITH CONVERSION OF DEPLETED URANIUM TO U₃O₈^a

Radionuclide	Dose Conversion Factor	Source Term 7	Source Term 19
	Sv/Bq	Curies per Unit Mass	Curies per Unit Mass
U ²³²	3.85E-05	3.00E-07	5.24E-07
U ²³⁴	3.58E-05	6.25E-06	8.25E-06
U ²³⁵	3.23E-05	6.81E-08	6.64E-08
U ²³⁶	3.39E-05	1.55E-06	1.96E-06
U ²³⁸	3.20E-05	3.20E-07	3.18E-07

^aUnit mass = 1 gram

TABLE 7-30
ACTIVITY CONCENTRATION SOURCE TERMS AND DOSE CONVERSION FACTORS
FOR MOX 1 OR 2 FUEL FABRICATION^a

Radionuclide	Dose Conversion Factor	Source Term 2	Source Term 14	Source Term 8	Source Term 20
	Sv/Bq	Curies per Unit Mass	Curies per Unit Mass	Curies per Unit Mass	Curies per Unit Mass
Pu ²³⁸	1.06E-04	4.78E-02	6.20E-02	4.78E-02	6.89E-02
Pu ²³⁹	1.16E-04	2.36E-03	2.27E-03	2.36E-03	1.72E-03
Pu ²⁴⁰	1.16E-04	4.14E-03	5.04E-03	4.14E-03	3.87E-03
Pu ²⁴¹	2.23E-06	8.88E-01	1.05E+00	8.88E-01	8.07E-01
Pu ²⁴²	1.11E-04	2.46E-05	3.75E-05	2.46E-05	3.16E-05
U ²³²	3.85E-05	2.23E-07	1.22E-08	2.23E-07	3.59E-07
U ²³⁴	3.58E-05	5.40E-07	5.96E-07	4.71E-06	5.66E-06
U ²³⁵	3.23E-05	1.52E-08	9.37E-09	5.18E-08	4.64E-08
U ²³⁶	3.39E-05	1.86E-07	1.87E-07	1.20E-06	1.41E-06
U ²³⁸	3.20E-05	3.10E-07	3.08E-07	3.07E-07	3.02E-07
Am ²⁴¹	1.20E-04	1.47E-03	1.88E-03	1.47E-03	1.45E-03

^aUnit mass = 1 gram

TABLE 7-31
ACTIVITY CONCENTRATION SOURCE TERMS AND DOSE CONVERSION FACTORS FOR FAST REACTOR FUEL
FABRICATION^a

Radionuclide	Dose Conversion Factor	Source Term 4	Source Term 5	Source Term 16	Source Term 17	Source Term 11	Source Term 12	Source Term 23	Source Term 24
	Sv/Bq	Curies per Unit Mass	Curies per Unit Mass	Curies per Unit Mass	Curies per Unit Mass	Curies per Unit Mass	Curies per Unit Mass	Curies per Unit Mass	Curies per Unit Mass
U ²³²	3.85E-05	9.96E-08	3.79E-06	1.17E-07	3.79E-06	2.59E-07	6.06E-06	3.54E-07	7.20E-06
U ²³⁴	3.58E-05	1.10E-06	8.28E-06	1.34E-06	8.98E-06	2.90E-06	1.18E-05	3.93E-06	1.35E-05
U ²³⁵	3.23E-05	5.24E-09	1.01E-09	3.40E-09	7.27E-10	1.46E-08	2.62E-09	1.60E-08	2.88E-09
U ²³⁶	3.39E-05	1.66E-07	1.60E-07	1.48E-07	1.39E-07	9.12E-07	8.13E-07	1.04E-06	9.22E-07
U ²³⁸	3.20E-05	2.66E-07	2.66E-07	2.66E-07	2.66E-07	2.60E-07	2.62E-07	2.59E-07	2.62E-07
Np ²³⁷	1.46E-04	1.38E-05	5.92E-06	1.55E-05	6.27E-06	2.16E-05	9.67E-06	2.59E-05	1.12E-05
Pu ²³⁸	1.06E-04	1.37E-01	1.93E-01	1.34E-01	2.04E-01	2.21E-01	2.57E-01	2.24E-01	2.83E-01
Pu ²³⁹	1.16E-04	3.38E-03	5.31E-03	2.90E-03	5.18E-03	3.01E-03	5.17E-03	2.52E-03	5.05E-03
Pu ²⁴⁰	1.16E-04	1.04E-02	1.14E-02	9.11E-03	1.06E-02	9.25E-03	1.05E-02	7.83E-03	9.64E-03
Pu ²⁴¹	2.23E-06	2.21E+00	8.02E-01	1.96E+00	7.20E-01	1.88E+00	7.04E-01	1.68E+00	6.33E-01
Pu ²⁴²	1.11E-04	9.27E-05	8.46E-05	9.60E-05	8.44E-05	9.52E-05	8.24E-05	9.07E-05	7.75E-05
Am ²⁴¹	1.20E-04	5.85E-02	2.62E-02	7.47E-02	2.91E-02	5.22E-02	2.31E-02	6.46E-02	2.52E-02
Am ²⁴³	1.19E-04	2.54E-03	1.91E-03	3.49E-03	2.27E-03	2.50E-03	1.86E-03	3.33E-03	2.12E-03
Cm ²⁴²	4.67E-06	2.40E-03	5.20E-02	0.00E+00	5.57E-02	0.00E+00	4.54E-02	0.00E+00	4.80E-02
Cm ²⁴³	8.30E-05	5.26E-01	2.53E-03	3.27E-03	2.92E-03	2.20E-03	2.23E-03	2.89E-03	2.52E-03
Cm ²⁴⁴	7.60E-05	1.36E-04	5.48E-01	7.64E-01	6.98E-01	5.24E-01	5.36E-01	7.30E-01	6.57E-01
Cm ²⁴⁵	1.23E-04	2.60E-05	2.33E-04	2.17E-04	3.10E-04	1.19E-04	2.25E-04	1.85E-04	2.87E-04
Cm ²⁴⁶	1.22E-04	5.85E-02	1.04E-04	3.99E-05	1.49E-04	2.75E-05	1.07E-04	3.84E-05	1.38E-04

^aUnit mass = 1 gram

7.3.3 Methods to Estimate Radiological Health Impacts of Transportation

The analysis evaluated differences in the impacts of transporting the nuclear materials for the UREX+3c fuel cycle and the AFC by comparing the source terms for the materials that would be transported and projecting how materials having these source terms would affect requirements for their packaging and safe transportation. This analysis was based on, and extrapolated from, present-day regulatory requirements of the NRC and U.S. Department of Transportation (DOT), and industry practices that result in the safe transport of nuclear materials.

7.4 MODELING RESULTS

The modeling results for this study are presented in this section for the source terms identified in Section 7.2, by type of facility, using the analytical methods defined in Section 7.3. The results include:

- The photon doses to workers for each source term for a standard geometry, and
- The inhalation doses to members of the public for each source term.

This section presents the basic modeling results. An evaluation and discussion of dose results for facility workers and the public is set forth in Section 7.5.

7.4.1 Estimated Worker Radiological Health Impacts

Worker radiological impacts are estimated based on the relative photon radiation doses to workers from the nuclear fuel cycle facilities from the source terms using the MicroShield[®] computer program, as discussed in Section 7.3. Table 7-32 provides the estimated radiation doses, for each source term, for the two fuel cycles and two LWR spent fuel recycling scenarios. Table 7-33 provides the source term doses for each type of facility considered, for the UREX+3c fuel cycle and the AFC and for recycling LWR spent fuel one and two times. These results represent the radiation dose rate, as described in Section 7.3.1 above, at 1 meter in air from a standard source configuration for one metric ton of each source term.

TABLE 7-32
MICROSHIELD[®] MODELING RESULTS
(mR/hr, 1 M from a 1 Metric Ton Source)

Source Term (Table Number)	UREX+3c Fuel Cycle: 1 Recycle (Figure 7-1)	AFC: 1 Recycle (Figure 7-2)	UREX+3c Fuel Cycle: 2 Recycles (Figure 7-3)	AFC: 2 Recycles (Figure 7-4)
1	1.2E-03		1.2E-03	
2	1.5E-01		1.5E-01	
3	4.5E-04			
4	1.9E+02			
5	2.0E+02			
13			1.1E-03	
14			1.9E-01	
15			2.9E-04	
16			2.6E+02	
17			2.3E+02	
6		1.2E-03		1.2E-03
7		4.6E-03		4.6E-03
8		1.6E-01		1.6E-01
9		3.6E-04		3.6E-04
10		1.3E-03		
11		1.7E+02		
12		1.7E+02		
18				1.2E-03
19				4.5E-03
20				1.5E-01
21				3.6E-04
22				1.4E-03
23				2.3E+02
24				2.0E+02

TABLE 7-33
MICROSHIELD® MODELING RESULTS FOR THE TYPES OF FACILITIES CONSIDERED

Type of Facility^a (Source Term)	UREX+3c Fuel Cycle: 1 Recycle (mR/hr)	AFC: 1 Recycle (mR/hr)	UREX+3c Fuel Cycle: 2 Recycles (mR/hr)	AFC: 2 Recycles (mR/hr)
U Product Conversion to U ₃ O ₈ for Storage or Disposal (1) (3) (13) (15) (9) (10) (21) (22)	1.2E-03 4.5E-04	3.6E-03 1.3E-03	1.2E-03 4.5E-04 1.1E-03 2.9E-04	3.6E-04 1.4E-03
Conversion of Reprocessed Uranium to UF ₆ for Re-Enrichment (6) (18)		1.2E-03		1.2E-03 1.2E-03
Re-Enrichment of recovered uranium (7) (19)		4.6E-03		4.6E-03 4.5E-03
MOX 1 or 2 Fuel Fabrication (2) (14) (8) (20)	1.5E-01	1.6E-01	1.5E-01 1.9E-01	1.6E-01 1.5E-01
Fast Reactor Fuel Fabrication (4) (5) (16) (17) (11) (12) (23) (24)	1.9E+02 2.0E+02	1.7E+02 1.7E+02	2.6E+02 2.3E+02	2.3E+02 2.0E+02

^aSources for nuclear fuel reprocessing are not included because they would be largely the same for the alternatives evaluated.

7.4.2 Estimated Public Radiological Health Impacts

Public radiological impacts are estimated from the relative doses for inhalation of each source term. These results are provided in Tables 7-34 through 7-38, organized by type of facility. Table 7-39 provides the relative public doses for each source term, for each type of facility, by fuel cycle scenario.

TABLE 7-34
PUBLIC DOSES BY SOURCE TERM AND RATIOS TO SOURCE TERM 1 FOR FACILITIES FOR URANIUM
CONVERSION TO U₃O₈

Radionuclide	Source Term 1	Source Term 3	Source Term 13	Source Term 15	Source Term 9	Source Term 10	Source Term 21	Source Term 22
	mR/yr	mR/yr	mR/yr	mR/yr	mR/yr	mR/yr	mR/yr	mR/yr
U ²³²	8.56E+03	1.79E+04	1.02E+04	2.11E+04	3.05E+02	4.66E+04	6.10E+02	6.37E+04
U ²³⁴	1.93E+05	1.84E+05	1.91E+05	1.06E+05	3.31E+04	4.86E+05	4.13E+04	6.58E+05
U ²³⁵	2.14E+03	7.96E+02	1.92E+03	5.14E+02	6.47E+02	2.21E+03	6.47E+02	2.51E+03
U ²³⁶	6.28E+04	2.62E+04	5.69E+04	2.33E+04	3.00E+04	1.44E+05	3.96E+04	1.64E+05
U ²³⁸	3.94E+04	3.99E+04	3.95E+04	3.99E+04	3.99E+04	3.91E+04	3.98E+04	3.88E+04
Total	3.06E+05	2.69E+05	2.99E+05	1.91E+05	1.04E+05	7.18E+05	1.22E+05	9.28E+05
Ratio to Source Term 1	1.00E+00	8.79E-01	9.78E-01	6.24E-01	3.40E-01	2.35E+00	3.99E-01	3.03E+00

TABLE 7-35
RELATIVE PUBLIC DOSES BY SOURCE TERM AND RATIOS TO SOURCE TERM 1 FOR UF₆ FACILITIES FOR
ENRICHMENT

Radionuclide	Source Term 1	Source Term 6	Source Term 18
	mR/yr	mR/yr	mR/yr
U ²³²	8.56E+03	8.56E+03	1.61E+04
U ²³⁴	1.93E+05	1.93E+05	2.62E+05
U ²³⁵	2.14E+03	2.14E+03	2.15E+03
U ²³⁶	6.28E+04	6.28E+04	8.23E+04
U ²³⁸	3.94E+04	3.94E+04	3.93E+04
Total	3.06E+05	3.06E+05	4.02E+05
Ratio to Source Term 1	1.00E+00	1.00E+00	1.31E+00

TABLE 7-36
RELATIVE PUBLIC DOSES BY SOURCE TERM AND RATIOS TO SOURCE TERM 1
FOR FACILITIES FOR REENRICHMENT WITH CONVERSION OF DEPLETED
URANIUM TO U₃O₈

Radionuclide	Source Term 1	Source Term 7	Source Term 19
	mR/yr	mR/yr	mR/yr
U ²³²	8.56E+03	4.27E+04	7.46E+04
U ²³⁴	1.93E+05	8.28E+05	1.09E+06
U ²³⁵	2.14E+03	8.14E+03	7.94E+03
U ²³⁶	6.28E+04	1.94E+05	2.46E+05
U ²³⁸	3.94E+04	3.79E+04	3.77E+04
Total	3.06E+05	1.11E+06	1.46E+06
Ratio to Source Term 1	1.00E+00	3.63E+00	4.77E+00

TABLE 7-37
RELATIVE PUBLIC DOSES BY SOURCE TERM AND RATIOS TO SOURCE TERM 1
FOR FACILITIES FOR FABRICATION OF MOX 1 OR 2 LWR FUEL

Radionuclide	Source Term 2	Source Term 14	Source Term 8	Source Term 20
	mR/yr	mR/yr	mR/yr	mR/yr
Pu ²³⁸	1.87E+10	2.43E+10	1.87E+10	2.70E+10
Pu ²³⁹	1.01E+09	9.74E+08	1.01E+09	7.38E+08
Pu ²⁴⁰	1.78E+09	2.16E+09	1.78E+09	1.66E+09
Pu ²⁴¹	7.33E+09	8.66E+09	7.33E+09	6.66E+09
Pu ²⁴²	1.01E+07	1.54E+07	1.01E+07	1.30E+07
U ²³²	3.18E+04	1.74E+03	3.18E+04	5.11E+04
U ²³⁴	7.15E+04	7.89E+04	6.24E+05	7.50E+05
U ²³⁵	1.82E+03	1.12E+03	6.19E+03	5.55E+03
U ²³⁶	2.33E+04	2.35E+04	1.51E+05	1.77E+05
U ²³⁸	3.67E+04	3.65E+04	3.63E+04	3.58E+04
Am ²⁴¹	6.53E+08	8.35E+08	6.53E+08	6.44E+08
Total	2.95E+10	3.70E+10	2.95E+10	3.67E+10
Ratio to Source Term 1	9.65E+04	1.21E+05	9.65E+04	1.20E+05

TABLE 7-38
RELATIVE PUBLIC DOSES BY SOURCE TERM AND RATIOS TO SOURCE TERM 1 FOR FACILITIES FOR FABRICATING
FAST REACTOR FUEL

Radionuclide	Source Term 4	Source Term 5	Source Term 16	Source Term 17	Source Term 11	Source Term 12	Source Term 23	Source Term 24
	mR/yr	mR/yr	mR/yr	mR/yr	mR/yr	mR/yr	mR/yr	mR/yr
U ²³²	1.42E+04	5.40E+05	1.67E+04	5.40E+05	3.69E+04	8.63E+05	5.04E+04	1.03E+06
U ²³⁴	1.46E+05	1.10E+06	1.77E+05	1.19E+06	3.84E+05	1.56E+06	5.21E+05	1.79E+06
U ²³⁵	6.26E+02	1.21E+02	4.06E+02	8.69E+01	1.74E+03	3.13E+02	1.91E+03	3.44E+02
U ²³⁶	2.08E+04	2.01E+04	1.86E+04	1.74E+04	1.14E+05	1.02E+05	1.30E+05	1.16E+05
U ²³⁸	3.15E+04	3.15E+04	3.15E+04	3.15E+04	3.08E+04	3.10E+04	3.07E+04	3.10E+04
Np ²³⁷	7.45E+06	3.20E+06	8.37E+06	3.39E+06	1.17E+07	5.22E+06	1.40E+07	6.05E+06
Pu ²³⁸	5.37E+10	7.57E+10	5.26E+10	8.00E+10	8.67E+10	1.01E+11	8.79E+10	1.11E+11
Pu ²³⁹	1.45E+09	2.28E+09	1.24E+09	2.22E+09	1.29E+09	2.22E+09	1.08E+09	2.17E+09
Pu ²⁴⁰	4.46E+09	4.89E+09	3.91E+09	4.55E+09	3.97E+09	4.51E+09	3.36E+09	4.14E+09
Pu ²⁴¹	1.82E+10	6.62E+09	1.62E+10	5.94E+09	1.55E+10	5.81E+09	1.39E+10	5.22E+09
Pu ²⁴²	3.81E+07	3.47E+07	3.94E+07	3.47E+07	3.91E+07	3.38E+07	3.73E+07	3.18E+07
Am ²⁴¹	2.60E+10	1.16E+10	3.32E+10	1.29E+10	2.32E+10	1.03E+10	2.87E+10	1.12E+10
Am ²⁴³	1.12E+09	8.41E+08	1.54E+09	9.99E+08	1.10E+09	8.19E+08	1.47E+09	9.33E+08
Cm ²⁴²	4.15E+07	8.99E+08	0.00E+00	9.62E+08	0.00E+00	7.84E+08	0.00E+00	8.29E+08
Cm ²⁴³	1.62E+11	7.77E+08	1.00E+09	8.97E+08	6.76E+08	6.85E+08	8.88E+08	7.74E+08
Cm ²⁴⁴	3.82E+07	1.54E+11	2.15E+11	1.96E+11	1.47E+11	1.51E+11	2.05E+11	1.85E+11
Cm ²⁴⁵	1.18E+07	1.06E+08	9.88E+07	1.41E+08	5.42E+07	1.02E+08	8.42E+07	1.31E+08
Cm ²⁴⁶	2.64E+10	4.69E+07	1.80E+07	6.73E+07	1.24E+07	4.83E+07	1.73E+07	6.23E+07
Total	2.93E+11	2.58E+11	3.25E+11	3.05E+11	2.80E+11	2.77E+11	3.43E+11	3.21E+11
Ratio to Source Term 1	9.58E+05	8.43E+05	1.06E+06	9.97E+05	9.15E+05	9.05E+05	1.12E+06	1.05E+06

TABLE 7-39
RELATIVE PUBLIC DOSE MODELING RESULTS FOR THE TYPES OF FACILITIES
CONSIDERED

Type of Facility ^a (Source Term)	UREX+3c Fuel Cycle: 1 Recycle (mR/yr)	AFC: 1 Recycle (mR/yr)	UREX+3c Fuel Cycle: 2 Recycles (mR/yr)	AFC: 2 Recycles (mR/yr)
U Product Conversion to U ₃ O ₈ for Storage or Disposal (1) (3) (13) (15) (9) (10) (21) (22)	3.1E+05 2.7E+05	1.0E+05 7.2E+05	3.1E+05 2.7E+05 3.0E+05 1.9E+05	1.0E+05 7.2E+05 1.2E+05 9.3E+05
Conversion of Reprocessed Uranium to UF ₆ for Re-Enrichment (6) (18)		3.1E+05		3.1E+05 4.0E+05
Re-Enrichment of recovered uranium (7) (19)		1.1E+06		1.1E+06 1.5E+06
MOX 1 or 2 Fuel Fabrication (2) (14) (8) (20)	3.0E+10	3.0E+10	3.0E+10 3.7E+10	3.0E+10 3.7E+10
Fast Reactor Fuel Fabrication (4) (5) (16) (17) (11) (12) (23) (24)	2.9E+11 2.6E+11	2.8E+11 2.8E+11	3.2E+11 3.0E+11	3.4E+11 3.2E+11

^aSources for nuclear fuel reprocessing are not included because they would be largely the same for the alternatives evaluated.

7.5 DOSE RESULTS FOR FACILITY WORKERS AND THE PUBLIC

This section uses the dose results in Section 7.4 to derive the relative doses to workers and members of the public for the fuel cycle facilities and source terms considered for this study.

7.5.1 Worker Dose Results

To provide an overall assessment of the comparative radiological health impacts of the UREX+3c fuel cycle and the AFC, the relative differences among the source terms are next evaluated. Since most current facilities that handle or process mixtures of uranium require little or no shielding (i.e., fuel fabrication), Source Term 1 (Table 7-1), or the radionuclides considered for conversion of excess recovered uranium into U_3O_8 for storage or disposal, serves as the comparative basis for the remainder of the source terms. Table 7-40 provides the relative dose rates for each source term, divided by the dose rate estimated for Source Term 1, organized by the four fuel cycle scenarios that were evaluated. These results are presented in Table 7-41, organized by type of facility and scenario.

TABLE 7-40
WORKER DOSE RESULTS: RATIO OF MICROSHIELD[®] MODELING RESULTS TO THE
RESULTS FOR SOURCE TERM 1, CONVERSION OF URANIUM RECOVERED FROM
UREX+3c REPROCESSING TO U₃O₈ FOR STORAGE OR DISPOSAL

Source Term (Table Number)	UREX+3c Fuel Cycle: 1 Recycle (Figure 7-1)	AFC: 1 Recycle (Figure 7-2)	UREX+3c Fuel Cycle: 2 Recycles (Figure 7-3)	AFC: 2 Recycles (Figure 7-4)
1	1.0		1.0	
2	130		130	
3	0.38			
4	160,000			
5	160,000			
13			0.90	
14			160	
15			0.25	
16			200,000	
17			210,000	
6		1.0		1.0
7		3.3		3.8
8		130		130
9		0.30		0.30
10		1.0		
11		140,000		
12		140,000		
18				1.0
19				3.7
20				130
21				0.30
22				1.2
23				190,000
24				160,000

TABLE 7-41
WORKER DOSE RESULTS: RATIO OF MICROSHIELD® MODELING RESULTS TO THE
RESULTS FOR SOURCE TERM 1, CONVERSION OF URANIUM RECOVERED FROM
UREX+3c REPROCESSING TO U₃O₈ FOR STORAGE OR DISPOSAL BY TYPE OF
FACILITY CONSIDERED

Type of Facility ^a (Source Term)	UREX+3c Fuel Cycle: 1 Recycle	AFC: 1 Recycle	UREX+3c Fuel Cycle: 2 Recycles	AFC: 2 Recycles
U Product Conversion to U ₃ O ₈ for Storage or Disposal (1) (3) (13) (15) (9) (10) (21) (22)	1.0 0.38	0.30 1.0	1.0 0.38 0.90 0.25	0.30 1.0 0.30 1.2
Conversion of Reprocessed Uranium to UF ₆ for Re-Enrichment (6) (18)		1.0		1.0 1.0
Re-Enrichment of recovered uranium (7) (19)		3.3		3.3 3.7
MOX 1 or 2 Fuel Fabrication (2) (14) (8) (20)	130	130	130 160	130 130
Fast Reactor Fuel Fabrication (4) (5) (16) (17) (11) (12) (23) (24)	160,000 160,000	140,000 140,000	200,000 210,000	190,000 160,000

^aSources for nuclear fuel reprocessing are not included because they would be largely the same for the alternatives evaluated.

An estimate is next made of the potential shielding requirements for fast reactor fuel fabrication, since this type of facility encounters the highest potential photon dose rates for workers. The dose rates for these source terms are controlled largely by the curium radionuclides (notably Cm²⁴³ and Cm²⁴⁵). However, these source terms also contains a large amount of U²³², which has been cited as an obstacle to nuclear proliferation because the intense photon radiation of its decay products. All initial calculations were conducted assuming no

radioactive decay between reprocessing and conversion of uranium for storage, re-enrichment, or production of reactor fuel. Since it takes several years for the decay products of U^{232} to reach equilibrium, and since it is reasonable to assume a delay between reprocessing and subsequent fuel cycle operations, the potential increase in dose rates from the U^{232} decay products must be considered. These decay products include:

- Th^{228} produced by alpha decay of U^{232}
- Ra^{224} produced by alpha decay of Th^{228}
- Rn^{220} produced by alpha decay of Ra^{224}
- Pb^{212} produced by alpha decay of Rn^{220}
- Bi^{212} produced by beta decay of Pb^{212}
- Po^{212} produced two thirds of the time from beta decay of Bi^{212}
- Tl^{208} produced one third of the time from alpha decay of Bi^{212} .

The source term with the highest dose rate and curie contribution of U^{232} is Source Term 24, as shown in Table 7-24. A MicroShield[®] analysis was conducted using the same source geometry and dose location as considered for the worker dose analysis, using the Source Term 24 concentrations with U^{232} and each of the decay products through Tl^{208} in equilibrium. U^{232} has a half life of 68.9 years and is typically of importance since it is a side product in the thorium fuel cycle. The resulting dose rate for this mixture was about 550 mR/hr, or about 2.8 times higher than the dose rate for Source Term 24 shown in Table 7-32 without U^{232} and its decay products. This means that, depending on batch size, when the U^{232} decay chain members are added to the source terms, the estimated doses would increase by about 2.8 (or roughly 3) times the Table 7-32 doses for Source Term 24. This change was extended to consider the potential impact of U^{232} and its decay products on the estimated dose rates. For all of the other facilities, the dose rates would be on the order of 1-2 mR/hr; which would not require the use of additional shielding as a design consideration. For all of these facilities, ALARA design considerations could limit the batch sizes processed, or restrict the location of materials and process components to optimize dose rates, without the need for additional shielding.

MicroShield[®] was next used to assess the potential impact of concrete shielding in reducing the dose rate for Source Term 24 with U^{232} and its decay products in equilibrium. With about 54 cm of concrete shield the dose rate is reduced from 550 mR/hr to a design goal of 2 mR/hr. In addition, it is quite reasonable that process batch size limitations may also be used to reduce worker doses, from an ALARA design perspective.

In addition to the production of neutrons from spontaneous fission, alpha radiation emitted from the decay of the uranium (and its decay products) and transuranic radionuclides can initiate nuclear reactions in some of the lighter elements found in a compound, transforming certain nuclides, while emitting a neutron. This is known as alpha/neutron decay. Such reactions are rather frequent in uranium hexafluoride (UF₆). The radionuclide source terms considered for this study that would most likely be of concern to alpha/neutron interactions, with the production of neutrons in excess of those from spontaneous fission, would be those associated with MOX-1 or MOX-2, and fast reactor fuel fabrication. As previously concluded, process batch size limitations or the addition of concrete shielding at key locations would also provide an effective method of controlling neutron doses. Thus, it is unlikely that alpha/neutron reactions would pose significant radiation protection issues for workers at MOX or fast reactor fuel fabrication facilities.

7.5.2 Public Dose Results

For this analysis, it is recognized that current designs for nuclear facilities typically reduce offsite doses through a rigorous design process that includes ALARA evaluations for each component and process within a facility. Measures to mitigate or eliminate effluents to air and water have become standard in facility design to the point that the dose to maximally exposed individuals is typically on the order of a few millirem per year, or less. A relative comparison of source terms based on the potential for inhalation, as a surrogate for the material at risk evaluations used in safety analyses, is provided in Table 7-42, as a ratio to the inhalation doses for Source Term 1. This relative comparison will help rank the potential effort involved in mitigating those doses and the potential importance of accidents.

TABLE 7-42
RATIO OF PUBLIC DOSE MODELING RESULTS FOR THE TYPES OF FACILITIES TO
THE RESULTS FOR SOURCE TERM 1

Type of Facility ^a (Source Term)	UREX+3c Fuel Cycle: 1 Recycle	AFC: 1 Recycle	UREX+3c Fuel Cycle: 2 Recycles	AFC: 2 Recycles
UO ₂ Conversion to U ₃ O ₈ for Storage or Disposal (1) (3) (13) (15) (9) (10) (21) (22)	1.0 0.88		1.0 0.8 0.98 0.62	
Conversion of Reprocessed Uranium to UF ₆ for Re-Enrichment (6) (18)		1.0 1.3		1.0 1.3
Re-Enrichment of recovered uranium (7) (19)		3.6		3.6 4.8
MOX 1 or 2 Fuel Fabrication (2) (14) (8) (20)	96,000	96,000	96,000 121,000	96,000 120,000
Fast Reactor Fuel Fabrication (4) (5) (16) (17) (11) (12) (23) (24)	935,000 843,000	920,000 900,000	1,100,000 1,000,000	1,100,000 1,100,000

^aSources for nuclear fuel reprocessing are not included because they would be largely the same for the alternatives evaluated.

7.5.3 Transportation Impacts

Potential transportation impacts are evaluated considering transportation of the source terms associated with each of the facilities considered in this study and current regulations regarding the transportation of radioactive materials. Under these regulations, the radioactive materials considered in the source terms of this study would be shipped as Class 7 material, and require specific labeling based on the radiation level at the surface of the package and the transport index. The transport index is a dimensionless number placed on the label of a package

to designate the degree of control to be exercised by the carrier during transportation. It is determined by multiplying the maximum radiation level (in millisieverts per hour) at 1 m from the external surface of a package by 100. As the transport index increases, additional labeling and packaging requirements must be used. The lowest transport index is 0, with dose rate limits at the surface of a package of less than 0.5 mR/hr. Typical unirradiated LWR fuel would meet this criterion. The next transportation indices are limited as follows:

- (1) Transport Index 1, greater than 0.5 mR/hr, but less than or equal to 50 mR/hr (Yellow-II label category),
- (2) Transport Index >1 but < 10, greater than 50 mR/hr, but less than or equal to 200 mR/hr (Yellow – III label category), and
- (3) Transport Index > 10, greater than 200 mR/hr, but less than or equal to 1,000 mR/hr (Yellow-III label category requiring exclusive use shipment).

In addition, radioactive shipments are limited by the total quantity of materials, considering the aggregate of radioactive materials present. Type A quantities are those materials where the aggregate radioactivity does not exceed a specified value (A_1) for special form radioactive materials, or a separate value (A_2) for normal form radioactive materials, where these specified values are given in 40 CFR 173.435. What this simply means is that, in addition to dose rate limitations, the mixture of radionuclides in a shipment can also determine the need for packaging requirements. There is also a special definition of unirradiated uranium, which imposes limitations on the presence of plutonium, U^{236} , and fission products as related to each gram of U^{235} . Type B materials are those materials that exceed the Type A quantities.

Of most concern to this study is the shipment of uranium recovered from reprocessing, MOX fuels, and FR fuels, and the associated transportation requirements. The following observations and conclusions are made:

- Limitations imposed on the presence of U^{236} in uranium mixtures, not to exceed more than 0.005 gram of U^{236} per gram of U^{235} , indicate that none of the uranium processing source terms would qualify as unirradiated uranium. This means that containers approved for shipment of Type B materials will likely be required.
- Because of the presence of transuranic and minor actinide radionuclides, shipment of materials associated with MOX or FR fuels will require containers approved

for shipment of Type B materials, both from dose rate and radionuclide content perspectives.

Current Type B shipping containers are for shipment of radioactive waste and spent nuclear fuel. There are numerous types of waste shipment containers that could be used to ship quantities of U_3O_8 in various configurations, so the focus of this analysis is on shipment of MOX or FR fuels. Spent fuel shipment casks have been designed, certified, and licensed by the NRC for shipment of spent fuel. These casks have been used to ship highly radioactive components for radioactive waste disposal because of their volume and shielding properties. Many current spent fuel casks are designed for rail transport, to maximize the payload. Typical casks have provisions for the removal of decay heat from spent reactor fuel, and several used neutron absorbing materials to limit neutron dose rates at the exterior of the cask. For this study it is assumed that fresh MOX or FR fuels can be safely shipped in current design spent fuel containers or ones that would be similar (see Section 10.2 of this report).

7.6 CONCLUSIONS

This section evaluates the comparative radiological health impacts of the UREX+3c fuel cycle and the AFC. For facility workers, the results provide a general assessment of the extent to which shielded facilities were required. For members of the public, the assessment considered the need for special design features beyond those found at current U_3O_8 conversion facilities to protect against the potential for airborne releases of radioactive materials during routine operations and accidents. Finally, a general assessment was conducted of potential transportation impacts for shipping materials from the facilities considered.

Health and safety impacts of fuel reprocessing facilities that would be associated with the two fuel cycles, which would be largely the same, were not evaluated. Reprocessing plants would provide significant shielding in key process and waste handling locations for the fission product photon emitters, limit batch sizes to reduce overall dose rates, and remote or automated process control systems to eliminate worker exposures. For protection of the public, plants would include significant effluent controls that would be tailored to the radionuclides encountered and accident prevention and mitigation features. These, and other ALARA design features, are consistent with the design of currently operating fuel reprocessing facilities world-

wide. Thus, for reprocessing, the comparison of the UREX+3c fuel cycle and the AFC would be null and not material to the overall comparison of health impacts of the two fuel cycles.

7.6.1 Occupational Radiological Health

Based on the above analysis, the following conclusions are made regarding the comparison of impacts to occupational radiological health for the UREX+3c fuel cycle and the AFC:

- (1) The dose ratios by type of facility are generally consistent, within about a factor of two to four, across all scenarios.
- (2) The dose ratios for all of the source terms for conversion of uranium recovered from reprocessing into U_3O_8 for storage at depository are all within about a factor of four, and are on the order of 10^{-4} to 10^{-3} mR/hr. For purposes of this analysis, this is close agreement, and because of the low dose rates, it can be concluded that these facilities should require minimal shielding for photons in their initial designs.
- (3) The dose rates for conversion of reprocessed uranium to UF_6 are equal to those for uranium conversion facilities, indicating that there is no appreciable difference in this process compared to U_3O_8 conversion facilities, and no additional shielding requirements would be necessary in their initial designs to protect workers.
- (4) The reenrichment facilities encounter source terms that are four times the dose rate from Source Term 1. This would indicate that the doses are still quite low and that ALARA design considerations, such as limiting batch sizes, may all that is needed to maintain worker doses during facility operations.
- (5) The MOX 1 or 2 fuel fabrication facilities encounter source terms that are over 100 times the dose rate from Source Term 1; however, these dose rates are still on the order of 0.1 mR/hr. At most, this may require application of special ALARA design features, which may include limiting batch sizes and providing some additional shielding to protect workers in key process or material storage locations.
- (6) The FR fuel fabrication facilities encounter source terms that are over 100,000 times the dose rate from Source Term 1 with estimated dose rates on the order of 100 mR/hr. Depending on batch size and equipment configurations, these facilities will require special ALARA design features to meet an ALARA design target of 2 mR/hr for locations where workers have frequent access. These features would include shielding to protect workers in key process or material storage locations.

- (7) It was estimated that about 54 cm of concrete would be required to reduce the dose rate for fast reactor fuel fabrication facilities, when U^{232} and its decay products are present in equilibrium, to a design goal of 2 mR/hr. Additional ALARA design considerations might include limiting process batch sizes.
- (8) For the other facilities considered in this study, including U^{232} and its decay products in equilibrium in the source terms would increase the dose rates to 1-2 mR/hr, which would not require the use of additional shielding.
- (9) For all of the facilities considered, ALARA design considerations could limit the batch sizes processed, or restrict the location of materials and process components to optimize dose rates.
- (10) The radionuclide source terms considered for this study that would most likely be of concern to alpha/neutron interactions, with the production of neutrons in excess of those from spontaneous fission, would be those associated with MOX 1 or 2, and FR fuel fabrication. As previously concluded, the addition of concrete shielding at key locations would be required, which would also provide an effective method of controlling neutron doses. Thus, it is unlikely that alpha/neutron reactions would pose significant radiation protection issues for workers.

7.6.2 Public Radiological Health

Conclusions regarding the relative impacts of the UREX+3c fuel cycle and the AFC on public health are as follows:

- (1) The public dose ratios by type of facility are generally consistent, within about a factor of two to three, across all scenarios.
- (2) The potential for public dose was about equal for all of the source terms for conversion of uranium recovered from reprocessing into U_3O_8 for storage and it can be concluded that these facilities should need minimal requirements for effluent control or accident prevention or mitigation beyond those required in current facility designs.
- (3) The potential public doses for facilities that would convert reprocessed uranium to UF_6 are nearly equal to those for U_3O_8 conversion facilities.
- (4) The reenrichment facilities encounter source terms that may produce up to four times the public dose from U_3O_8 conversion facilities. This would indicate that additional ALARA design considerations would be applied including additional effluent control or accident prevention or mitigation features to maintain public doses during facility operations or accidents, compared with current uranium enrichment facilities.

- (5) The MOX fuel fabrication facilities encounter source terms that may produce public doses over 100,000 times the public doses expected from U_3O_8 conversion facilities. This is caused by the presence of plutonium and minor actinides in the source term, and their importance to public inhalation exposures. These facilities would definitely require special design features, including additional effluent controls and accident prevention or mitigation features to maintain public doses during facility operations or accidents, comparable to current uranium fuel fabrication facilities. However, the requirements would be similar to those being designed by AREVA MOX Services, LLC for the MOX Fuel Fabrication Facility at the Savannah River Site in South Carolina.
- (6) The FR fuel fabrication facilities encounter source terms potentially produce over 1,000,000 times the public dose from U_3O_8 conversion facilities because of the presence of transuranic radionuclides. These facilities would also require special ALARA design features, including additional effluent controls and accident prevention or mitigation features to maintain public doses during facility operations or accidents, comparable to current uranium fuel fabrication facilities.

7.6.3 Transportation Impacts

Conclusions regarding the transportation impacts of the UREX+3c fuel cycle and the AFC are as follows:

- (1) Of most concern to this study is the shipment of uranium recovered from reprocessing MOX fuels and FR fuels, and the associated transportation requirements.
- (2) Limitations imposed on the presence of U^{236} in uranium mixtures, not to exceed more than 0.005 gram of U^{236} per gram of U^{235} , indicate that none of the uranium processing source terms would qualify as unirradiated uranium. This means that containers approved for shipment of Type B materials will likely be required.
- (3) Because of the presence of transuranic and minor actinide radionuclides, shipment of materials associated with MOX nuclear fuel and FR fuels will require containers approved for shipment of Type B materials, both from dose rate and radionuclide content perspectives. Given the duration of the nuclear fuel cycles considered in this study, it may be worth the investment to design shipping containers for the shipment of unirradiated MOX and FR fuel.
- (4) There are numerous types of waste shipment containers that could be used to ship quantities of U_3O_8 in various configurations.
- (5) Many current spent fuel casks are designed for rail transport. Typical casks have provisions for the removal of decay heat from spent reactor fuel and neutron

absorbing materials to limit neutron dose rates at the exterior of the cask. Spent MOX or FR fuels could be safely shipped in Type B casks similar in design to current spent fuel casks.

7.7 REFERENCES FOR SECTION 7.0

- (1) Grove Engineering, MicroShield[®], Version 8, Rockville, Maryland, 2008
<http://www.radiationsoftware.com/mshield.html>.
- (2) K. F. Eckerman, A. B. Wolbarst, and A. C. B. Richardson, Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion, Prepared for the U.S. Environmental Protection Agency by Oak Ridge National Laboratory, Oak Ridge, TN, 1988

8.0 DIFFERENCES IN WASTE TYPES AND QUANTITIES FOR THE UREX+3c FUEL CYCLE AND THE AFC

This section describes the differences in the quantities and characteristics of radioactive wastes that would be produced by the UREX+3c fuel cycle and AFC illustrated in Figures 3-1 and 3-2, respectively. Table 8-1 lists quantities of the wastes that were estimated by simulating 81 years of operation of the fuel cycle systems and based on both 1 and 2 recycles of the uranium and/or plutonium recovered from LWR spent fuel. Year-by-year details of the simulations of waste production for the UREX+3c fuel cycle and the AFC, each for scenarios wherein uranium and/or plutonium is assumed to be recycled one and two times, are presented in Tables 8-2 through 8-5. These latter four tables are companions to the fuel cycle simulations presented in Tables 4-1 through 4-4 in Section 4.0.

8.1 WASTES FROM URANIUM MINING AND MILLING AND REACTOR FUEL PRODUCTION

The largest differences in the amounts of wastes between the two fuel cycle systems and the two scenarios for recycling LWR spent fuel are in the amounts of wastes from uranium mining and milling, and uranium conversion and fuel production. Wastes from mining and milling range from about 9.7 billion cubic feet for the UREX+3c fuel cycle where LWR spent fuel is recycled one time, to 8.1 billion cubic feet (16 percent less) for the AFC where LWR spent fuel is recycled two times. Where the number of times LWR spent fuel is recycled is the same, the amounts of mine and mill tailings for the AFC are about 10 percent less than for the UREX+3c fuel cycle. For the UREX+3c fuel cycle and the AFC, the amounts of uranium mine and mill tailings are, respectively, 5.9 percent and 7.3 percent less where LWR spent fuel is recycled one time versus two times. Reenrichment of recycle uranium in the AFC is the reason the amounts of mining, milling, and uranium fuel fabrication wastes are less than for the UREX+3c fuel cycle system.

LLW from uranium conversion and fuel production ranges from about 46.3 million cubic feet for the UREX+3c fuel cycle where LWR spent fuel is recycled one time to about 38.9 million cubic feet (16 percent less) for the AFC where LWR spent fuel is recycled two times.

TABLE 8-1
SUMMARY OF SOLID RADIOACTIVE WASTES FROM FOUR ADVANCED FUEL
CYCLE SCENARIOS

Fuel Cycle Activity	Waste Type	Fuel Cycle Scenario			
		UREX+3c/ 1 Recycle	AFC/ 1 Recycle	UREX+3c/ 2 Recycles	AFC/ 2 Recycles
Uranium Mining and Milling	Mine and Mill Tailings (cu ft.)	9,715,441,522	8,806,168,913	9,141,117,391	8,164,258,260
Uranium Fuel Production	Uranium Conversion LLW (cu ft.)	22,913,777	20,769,266	21,559,239	19,255,326
	UO ₂ Fuel Fabrication LLW (cu ft.)	2,618,717	2,373,630	2,463,913	2,200,609
	Depleted Uranium Conversion LLW (cu ft.)	20,755,359	18,812,850	19,528,408	17,441,526
	Depleted Uranium Oxide (MTU)	1,121,254	1,016,316	1,054,969	942,227
LWR Spent Fuel Reprocessing	LLW (cu ft.)	7,117,826	7,117,826	7,289,304	7,289,304
	TRU (cu ft.)	3,140,217	3,140,217	3,215,870	3,215,870
	Hulls/Tc ⁹⁹ /Dissolver Sludge (metric tons SNF equiv)	60,348	60,348	61,802	61,802
	Kr ⁸⁵ /H ³ /C ¹⁴ (metric tons)	6	6	6	6
	Sr/Cs/Ba/Rb (metric tons)	1,886	1,884	1,931	1,930
	HLW (vitrified) (metric tons SNF equiv)	46,224	46,224	47,338	47,338
	RU Conversion Wastes (cu ft.)	2,882,499	362,444	2,940,583	119,558
	RUO _x Disposition (metric tons)	160,175	20,140	163,402	6,644
RU Re-enrichment	DRU Conversion Wastes, LLW (cu ft.)	N/A	2,023,638	N/A	2,257,897
	Depleted RUO _x (metric tons)	N/A	112,449	N/A	125,467
LWR MOX Production	TRU Wastes (cu ft.)	3,692,682	5,148,830	4,604,336	6,179,283
FR Fuel Production	TRU Wastes (cu ft.)	3,444,040	3,444,040	1,774,765	1,774,765
FR Spent Fuel Reprocessing	TRU (cu ft.)	374,438	374,438	191,697	191,697
	Hulls/Tc ⁹⁹ /Process Solids (metric tons LWR SNF equiv)	7,196	7,196	3,684	3,684
	Kr ⁸⁵ /H ³ (metric tons)	14	15	8	8
	Sr/Cs/Ba/Rb (metric tons)	396	396	203	203
	HLW (metric tons LWR SNF equiv)	5,512	5,512	2,822	2,822

Source: Values in this table were determined using information contained in (Ref. 1).

TABLE 8-2
UREX+3c FUEL CYCLE
ANNUAL PRODUCTION OF RADIOACTIVE WASTES FOR ONE RECYCLE OF MOX IN LWRs

Year	Uranium Enrichment				LLW from LWR Spent Fuel Reprocessing, LWR and FR reactor ops. & RU Reenrichment							TRU Wastes from LWR & FR Spent Fuel Reprocessing & LWR MOX & FR Reactor Fuel Fabrication					High Level Radioactive Wastes from LWR Spent Fuel Reprocessing			High Level Radioactive Wastes from FR Spent Fuel Reprocessing			High Level Radioactive Wastes from UREX Fuel Cycle with 1 Pu Recycle		
	UO ₂ Reactor Fuel @ 5% Enrichment (metric tons U)	Natural U Feed to Enrichment (metric tons U)	DU Enrichment Tails (metric tons U)	RU Feed to Enrichment (metric tons U)	RUOx to Depository (metric tons U)	DRUOx to Depository (metric tons U)	RU Conversion		Reprocess LLW (cu. ft.)	Reactor Ops LLW (cu. ft.)	Fuel Cycle LLW (cu. ft.)	TRU from LWR Spent Fuel Reprocessing (cu. ft.)	TRU from FR Spent Fuel Reprocessing (cu. ft.)	TRU from LWR MOX Fuel Fab (cu. ft.)	TRU from FR Fuel Fab (cu. ft.)	Fuel Cycle TRU (cu. ft.)	Kr ⁸⁵ /H ³ from LWR Spent Fuel (metric tons)	HLW + Hulls/Tc ⁹⁹ /Dissolver Sludge from LWR Spent Fuel (metric ton SNF Equiv)	Sr/Cs/Ba/Rb from LWR Spent Fuel (metric tons)	Kr ⁸⁵ /H ³ from FR Spent Fuel (metric tons)	HLW + Hulls/Tc ⁹⁹ /Dissolver Sludge from FR Spent Fuel (metric ton SNF Equiv)	Sr/Cs/Ba/Rb from FR Spent Fuel (metric tons)	Kr ⁸⁵ /H ³ from UREX+3c Fuel Cycle (metric tons)	HLW + Hulls/Tc ⁹⁹ /Dissolver Sludge from UREX+3c Fuel Cycle (metric ton SNF Equiv)	Sr/Cs/Ba/Rb from UREX+3c Fuel Cycle (metric tons)
							Waste LLW (cu. ft.)	DRU Conversion Waste LLW (cu. ft.)																	
2020	1,840	18,967	17,127	0	0	0	0	0	0	470,000	470,000	0	0	0	0	0	0.00	0	0	0.00	0	0	0.00	0	0
2021	1,840	18,967	17,127	0	0	0	0	0	0	470,000	470,000	0	0	0	0	0	0.00	0	0	0.00	0	0	0.00	0	0
2022	1,840	18,967	17,127	0	0	0	0	0	0	470,000	470,000	0	0	0	0	0	0.00	0	0	0.00	0	0	0.00	0	0
2023	1,840	18,967	17,127	0	0	0	0	0	0	470,000	470,000	0	0	0	0	0	0.00	0	0	0.00	0	0	0.00	0	0
2024	1,840	18,967	17,127	0	0	0	0	0	0	470,000	470,000	0	0	0	0	0	0.00	0	0	0.00	0	0	0.00	0	0
2025	1,840	18,967	17,127	0	0	0	0	0	0	470,000	470,000	0	0	0	0	0	0.00	0	0	0.00	0	0	0.00	0	0
2026	1,840	18,967	17,127	0	0	0	0	0	0	470,000	470,000	0	0	0	0	0	0.00	0	0	0.00	0	0	0.00	0	0
2027	1,840	18,967	17,127	0	0	0	0	0	0	470,000	470,000	0	0	0	0	0	0.00	0	0	0.00	0	0	0.00	0	0
2028	1,840	18,967	17,127	0	0	0	0	0	0	470,000	470,000	0	0	0	0	0	0.00	0	0	0.00	0	0	0.00	0	0
2029	1,840	18,967	17,127	0	0	0	0	0	0	470,000	470,000	0	0	0	0	0	0.00	0	0	0.00	0	0	0.00	0	0
2030	1,840	18,967	17,127	0	0	0	0	0	0	470,000	470,000	0	0	0	0	0	0.00	0	0	0.00	0	0	0.00	0	0
2031	1,840	18,967	17,127	0	2,568	0	46,214	0	110,870	470,000	627,083	48,913	0	64,752	0	113,665	0.09	1,660	30	0.00	0	0	0.09	1,660	30
2032	1,334	13,753	12,419	0	2,568	0	46,214	0	110,870	470,000	627,083	48,913	0	64,752	0	113,665	0.09	1,660	30	0.00	0	0	0.09	1,660	30
2033	1,334	13,753	12,419	0	2,568	0	46,214	0	110,870	470,000	627,083	48,913	0	64,752	0	113,665	0.09	1,660	30	0.00	0	0	0.09	1,660	30
2034	1,334	13,753	12,419	0	2,568	0	46,214	0	110,870	470,000	627,083	48,913	0	64,752	0	113,665	0.09	1,660	30	0.00	0	0	0.09	1,660	30
2035	1,334	13,753	12,419	0	2,568	0	46,214	0	110,870	470,000	627,083	48,913	0	64,752	0	113,665	0.09	1,660	30	0.00	0	0	0.09	1,660	30
2036	1,334	13,753	12,419	0	2,568	0	46,214	0	110,870	470,000	627,083	48,913	0	64,752	0	113,665	0.09	1,660	30	0.00	0	0	0.09	1,660	30
2037	1,334	13,753	12,419	0	2,568	0	46,214	0	110,870	470,000	627,083	48,913	0	64,752	0	113,665	0.09	1,660	30	0.00	0	0	0.09	1,660	30
2038	1,334	13,753	12,419	0	2,568	0	46,214	0	110,870	470,000	627,083	48,913	0	64,752	0	113,665	0.09	1,660	30	0.00	0	0	0.09	1,660	30
2039	1,334	13,753	12,419	0	2,568	0	46,214	0	110,870	470,000	627,083	48,913	0	64,752	0	113,665	0.09	1,660	30	0.00	0	0	0.09	1,660	30
2040	1,334	13,753	12,419	0	2,568	0	46,214	0	110,870	470,000	627,083	48,913	0	64,752	0	113,665	0.09	1,660	30	0.00	0	0	0.09	1,660	30
2041	1,334	13,753	12,419	0	2,568	0	46,214	0	110,870	470,000	627,083	48,913	0	64,752	0	113,665	0.09	1,660	30	0.00	0	0	0.09	1,660	30
2042	1,334	13,753	12,419	0	2,526	0	45,463	0	110,870	470,000	626,333	48,913	0	60,623	7,814	117,350	0.09	1,660	29	0.00	0	0	0.09	1,660	29
2043	1,366	14,085	12,719	0	2,516	0	45,276	0	110,870	476,768	632,913	48,913	0	59,591	9,768	118,272	0.09	1,660	29	0.00	0	0	0.09	1,660	29
2044	1,375	14,168	12,794	0	2,505	0	45,088	0	110,870	483,536	639,493	48,913	0	58,559	11,721	119,193	0.09	1,660	29	0.00	0	0	0.09	1,660	29
2045	1,383	14,252	12,869	0	2,504	0	45,064	0	110,870	490,304	646,238	48,913	249	58,427	13,675	121,264	0.09	1,660	29	0.01	8	0	0.10	1,668	30
2046	1,384	14,262	12,879	0	2,503	0	45,040	0	110,870	497,072	652,982	48,913	498	58,296	15,628	123,334	0.09	1,660	29	0.02	17	1	0.11	1,677	30
2047	1,385	14,273	12,888	0	2,501	0	45,016	0	110,870	503,840	659,726	48,913	746	58,164	17,582	125,405	0.09	1,660	29	0.03	25	1	0.12	1,685	30
2048	1,386	14,283	12,898	0	2,500	0	44,992	0	110,870	510,608	666,470	48,913	995	58,032	19,535	127,476	0.09	1,660	29	0.04	34	1	0.13	1,694	30
2049	1,387	14,294	12,907	0	2,499	0	44,968	0	110,870	517,376	673,214	48,913	1,244	57,901	21,489	129,547	0.09	1,660	29	0.05	42	1	0.14	1,702	31
2050	1,388	14,305	12,917	0	2,497	0	44,944	0	110,870	524,144	679,958	48,913	1,493	57,769	23,442	131,617	0.09	1,660	29	0.06	51	2	0.15	1,711	31
2051	1,389	14,315	12,926	0	2,496	0	44,920	0	110,870	530,912	686,702	48,913	1,742	57,638	25,396	133,688	0.09	1,660	29	0.06	59	2	0.15	1,719	31
2052	1,390	14,326	12,936	0	2,495	0	44,896	0	110,870	537,680	693,446	48,913	1,990	57,506	27,349	135,759	0.09	1,660	29	0.07	68	2	0.16	1,728	31
2053	1,391	14,336	12,946	0	2,493	0	44,873	0	110,870	544,448	700,190	48,913	2,239	57,374	29,303	137,829	0.09	1,660	29	0.08	76	2	0.17	1,736	32
2054	1,392	14,347	12,955	0	2,492	0	44,849	0	110,870																

TABLE 8-3
AFC
ANNUAL PRODUCTION OF RADIOACTIVE WASTES FOR ONE RECYCLE OF MOX IN LWRs

Year	Uranium Enrichment				LLW from LWR Spent Fuel Reprocessing, LWR and FR reactor ops. & RU Reenrichment							Reactor Fuel Fabrication					High Level Radioactive Wastes from LWR Spent Fuel Reprocessing			High Level Radioactive Wastes from FR Spent Fuel Reprocessing			High Level Radioactive Wastes from AFC Fuel Cycle with 1 Pu Recycle		
	UO ₂ Reactor Fuel @ 5% Enrichment (metric tons U)	Natural U Feed to Enrichment (metric tons U)	DU Enrichment Tails (metric tons U)	RU Feed to Enrichment (metric tons U)	RUOx to Depository (metric tons U)	DRUOx to Depository (metric tons U)	RU Conversion		Reprocess LLW (cu. ft.)	Reactor Ops LLW (cu. ft.)	Fuel Cycle LLW (cu. ft.)	TRU from LWR Spent Fuel Reprocessing (cu. ft.)	TRU from FR Spent Fuel Reprocessing (cu. ft.)	TRU from LWR MOX Fuel Fab (cu. ft.)	TRU from FR Fuel Fab (cu. ft.)	Fuel Cycle TRU (cu. ft.)	Kr ⁸⁵ /H ³ from LWR Spent Fuel (metric tons)	HLW + Hulls/Tc ⁹⁹ / Dissolver Sludge from LWR Spent Fuel (metric ton SNF Equiv)	Sr/Cs/Ba/Rb from LWR Spent Fuel (metric tons)	Kr ⁸⁵ /H ³ from FR Spent Fuel (metric tons)	HLW + Hulls/Tc ⁹⁹ / Dissolver Sludge from FR Spent Fuel (metric ton SNF Equiv)	Sr/Cs/Ba/Rb from FR Spent Fuel (metric tons)	Kr ⁸⁵ /H ³ from UREX+3c Fuel Cycle (metric tons)	HLW + Hulls/Tc ⁹⁹ / Dissolver Sludge from UREX+3c Fuel Cycle (metric ton SNF Equiv)	Sr/Cs/Ba/Rb from UREX+3c Fuel Cycle (metric tons)
							Waste LLW (cu. ft.)	Conversion Waste LLW (cu. ft.)																	
2020	1,840	18,967	17,127	0	0	0	0	0	0	470,000	470,000	0	0	0	0	0	0.00	0	0	0.00	0	0	0.00	0	0
2021	1,840	18,967	17,127	0	0	0	0	0	0	470,000	470,000	0	0	0	0	0	0.00	0	0	0.00	0	0	0.00	0	0
2022	1,840	18,967	17,127	0	0	0	0	0	0	470,000	470,000	0	0	0	0	0	0.00	0	0	0.00	0	0	0.00	0	0
2023	1,840	18,967	17,127	0	0	0	0	0	0	470,000	470,000	0	0	0	0	0	0.00	0	0	0.00	0	0	0.00	0	0
2024	1,840	18,967	17,127	0	0	0	0	0	0	470,000	470,000	0	0	0	0	0	0.00	0	0	0.00	0	0	0.00	0	0
2025	1,840	18,967	17,127	0	0	0	0	0	0	470,000	470,000	0	0	0	0	0	0.00	0	0	0.00	0	0	0.00	0	0
2026	1,840	18,967	17,127	0	0	0	0	0	0	470,000	470,000	0	0	0	0	0	0.00	0	0	0.00	0	0	0.00	0	0
2027	1,840	18,967	17,127	0	0	0	0	0	0	470,000	470,000	0	0	0	0	0	0.00	0	0	0.00	0	0	0.00	0	0
2028	1,840	18,967	17,127	0	0	0	0	0	0	470,000	470,000	0	0	0	0	0	0.00	0	0	0.00	0	0	0.00	0	0
2029	1,840	18,967	17,127	0	0	0	0	0	0	470,000	470,000	0	0	0	0	0	0.00	0	0	0.00	0	0	0.00	0	0
2030	1,840	18,967	17,127	0	0	0	0	0	0	470,000	470,000	0	0	0	0	0	0.00	0	0	0.00	0	0	0.00	0	0
2031	1,840	18,967	17,127	2,568	0	2,055	0	36,982	110,870	470,000	617,851	48,913	0	94,094	0	143,007	0.09	1,660	30	0.00	0	0	0.09	1,660	30
2032	1,105	11,390	10,285	2,568	0	2,055	0	36,982	110,870	470,000	617,851	48,913	0	94,094	0	143,007	0.09	1,660	30	0.00	0	0	0.09	1,660	30
2033	1,105	11,390	10,285	2,568	0	2,055	0	36,982	110,870	470,000	617,851	48,913	0	94,094	0	143,007	0.09	1,660	30	0.00	0	0	0.09	1,660	30
2034	1,105	11,390	10,285	2,568	0	2,055	0	36,982	110,870	470,000	617,851	48,913	0	94,094	0	143,007	0.09	1,660	30	0.00	0	0	0.09	1,660	30
2035	1,105	11,390	10,285	2,568	0	2,055	0	36,982	110,870	470,000	617,851	48,913	0	94,094	0	143,007	0.09	1,660	30	0.00	0	0	0.09	1,660	30
2036	1,105	11,390	10,285	2,568	0	2,055	0	36,982	110,870	470,000	617,851	48,913	0	94,094	0	143,007	0.09	1,660	30	0.00	0	0	0.09	1,660	30
2037	1,105	11,390	10,285	2,568	0	2,055	0	36,982	110,870	470,000	617,851	48,913	0	94,094	0	143,007	0.09	1,660	30	0.00	0	0	0.09	1,660	30
2038	1,105	11,390	10,285	2,568	0	2,055	0	36,982	110,870	470,000	617,851	48,913	0	94,094	0	143,007	0.09	1,660	30	0.00	0	0	0.09	1,660	30
2039	1,105	11,390	10,285	2,568	0	2,055	0	36,982	110,870	470,000	617,851	48,913	0	94,094	0	143,007	0.09	1,660	30	0.00	0	0	0.09	1,660	30
2040	1,105	11,390	10,285	2,568	0	2,055	0	36,982	110,870	470,000	617,851	48,913	0	94,094	0	143,007	0.09	1,660	30	0.00	0	0	0.09	1,660	30
2041	1,105	11,390	10,285	2,568	0	2,055	0	36,982	110,870	470,000	617,851	48,913	0	94,094	0	143,007	0.09	1,660	30	0.00	0	0	0.09	1,660	30
2042	1,105	11,390	10,285	2,350	180	1,881	3,240	33,848	110,870	470,000	617,957	48,913	0	86,120	7,814	142,847	0.09	1,660	29	0.00	0	0	0.09	1,660	29
2043	1,167	12,032	10,865	2,296	225	1,837	4,050	33,064	110,870	476,768	624,752	48,913	0	84,127	9,768	142,807	0.09	1,660	29	0.00	0	0	0.09	1,660	29
2044	1,183	12,193	11,010	2,242	270	1,794	4,860	32,281	110,870	483,536	631,546	48,913	0	82,133	11,721	142,767	0.09	1,660	29	0.00	0	0	0.09	1,660	29
2045	1,198	12,353	11,155	2,235	276	1,788	4,962	32,182	110,870	490,304	638,318	48,913	249	81,882	13,675	144,719	0.09	1,660	29	0.01	8	0	0.10	1,668	30
2046	1,200	12,374	11,173	2,228	281	1,783	5,064	32,083	110,870	497,072	645,089	48,913	498	81,631	15,628	146,670	0.09	1,660	29	0.02	17	1	0.11	1,677	30
2047	1,202	12,394	11,191	2,221	287	1,777	5,166	31,985	110,870	503,840	651,860	48,913	746	81,380	17,582	148,621	0.09	1,660	29	0.03	25	1	0.12	1,685	30
2048	1,204	12,414	11,210	2,214	293	1,772	5,268	31,886	110,870	510,608	658,632	48,913	995	81,129	19,535	150,573	0.09	1,660	29	0.04	34	1	0.13	1,694	30
2049	1,206	12,434	11,228	2,207	298	1,766	5,370	31,788	110,870	517,376	665,403	48,913	1,244	80,878	21,489	152,524	0.09	1,660	29	0.05	42	1	0.14	1,702	31
2050	1,208	12,454	11,246	2,200	304	1,761	5,472	31,689	110,870	524,144	672,174	48,913	1,493	80,628	23,442	154,475	0.09	1,660	29	0.06	51	2	0.15	1,711	31
2051	1,210	12,475	11,264	2,194	310	1,755	5,574	31,590	110,870	530,912	678,946	48,913	1,742	80,377	25,396	156,427	0.09	1,660	29	0.07	59	2	0.16	1,719	31
2052	1,212	12,495	11,283	2,187	315	1,750	5,676	31,492	110,870	537,680	685,717	48,913	1,990	80,126	27,349	158,378	0.09	1,660	29	0.08	68	2	0.17	1,728	31
2053	1,214	12,515	11,301	2,180	321	1,744	5,778	31,393	110,870	544,448	692,488	48,913	2,239	79,875	29,303	160,330	0.09	1,660	29	0.09	76</				

TABLE 8-4
UREX+3c FUEL CYCLE
ANNUAL PRODUCTION OF RADIOACTIVE WASTES FOR TWO RECYCLES OF MOX IN LWRs

Year	Uranium Enrichment				LLW from LWR Spent Fuel Reprocessing, LWR and FR reactor ops. & RU Reenrichment										TRU Wastes from LWR & FR Spent Fuel Reprocessing & LWR MOX & FR Reactor Fuel Fabrication					High Level Radioactive Wastes from LWR Spent Fuel Reprocessing			High Level Radioactive Wastes from FR Spent Fuel Reprocessing			High Level Radioactive Wastes from UREX Fuel Cycle with 2 Pu Recycles		
	UO ₂ Reactor Fuel @ 5% Enrichment (metric tons U)	Natural U Feed to Enrichment (metric tons U)	DU Enrichment Tails (metric tons U)	RU Feed to Enrichment (metric tons U)	RUOx to Depository (metric tons U)	DRUOx to Depository (metric tons U)	RU Conversion		Reprocess LLW (cu. ft.)	Reactor Ops LLW (cu.ft.)	Fuel Cycle LLW (cu.ft.)	TRU from LWR Spent Fuel Reprocessing (cu. ft.)	TRU from FR Spent Fuel Reprocessing (cu. ft.)	TRU from LWR MOX Fuel Fab (cu. ft.)	TRU from FR Fuel Fab (cu. ft.)	Fuel Cycle TRU (cu. ft.)	Kr ⁸⁵ /H ³ from LWR Spent Fuel (metric tons)	HLW + Hulls/Tc ⁹⁹ / Dissolver Sludge from LWR Spent Fuel (metric ton SNF Equiv)	Sr/Cs/Ba/Rb from LWR Spent Fuel (metric tons)	Kr ⁸⁵ /H ³ from FR Spent Fuel (metric tons)	HLW + Hulls/Tc ⁹⁹ / Dissolver Sludge from FR Spent Fuel (metric ton SNF Equiv)	Sr/Cs/Ba/Rb from FR Spent Fuel (metric tons)	Kr ⁸⁵ /H ³ from UREX+3c Fuel Cycle (metric tons)	HLW + Hulls/Tc ⁹⁹ / Dissolver Sludge from UREX+3c Fuel Cycle (metric ton SNF Equiv)	Sr/Cs/Ba/Rb from UREX+3c Fuel Cycle (metric tons)			
							Waste LLW (cu. ft.)	DRU Conversion Waste LLW (cu. ft.)																				
2020	1,840	18,967	17,127	0	0	0	0	0	0	470,000	470,000	0	0	0	0	0	0.00	0	0	0.00	0	0	0.00	0	0			
2021	1,840	18,967	17,127	0	0	0	0	0	0	470,000	470,000	0	0	0	0	0	0.00	0	0	0.00	0	0	0.00	0	0			
2022	1,840	18,967	17,127	0	0	0	0	0	0	470,000	470,000	0	0	0	0	0	0.00	0	0	0.00	0	0	0.00	0	0			
2023	1,840	18,967	17,127	0	0	0	0	0	0	470,000	470,000	0	0	0	0	0	0.00	0	0	0.00	0	0	0.00	0	0			
2024	1,840	18,967	17,127	0	0	0	0	0	0	470,000	470,000	0	0	0	0	0	0.00	0	0	0.00	0	0	0.00	0	0			
2025	1,840	18,967	17,127	0	0	0	0	0	0	470,000	470,000	0	0	0	0	0	0.00	0	0	0.00	0	0	0.00	0	0			
2026	1,840	18,967	17,127	0	0	0	0	0	0	470,000	470,000	0	0	0	0	0	0.00	0	0	0.00	0	0	0.00	0	0			
2027	1,840	18,967	17,127	0	0	0	0	0	0	470,000	470,000	0	0	0	0	0	0.00	0	0	0.00	0	0	0.00	0	0			
2028	1,840	18,967	17,127	0	0	0	0	0	0	470,000	470,000	0	0	0	0	0	0.00	0	0	0.00	0	0	0.00	0	0			
2029	1,840	18,967	17,127	0	0	0	0	0	0	470,000	470,000	0	0	0	0	0	0.00	0	0	0.00	0	0	0.00	0	0			
2030	1,840	18,967	17,127	0	0	0	0	0	0	470,000	470,000	0	0	0	0	0	0.00	0	0	0.00	0	0	0.00	0	0			
2031	1,840	18,967	17,127	0	2,568	0	46,214	0	110,870	470,000	627,083	48,913	0	64,714	0	113,627	0.09	1,660	30	0.00	0	0	0.09	1,660	30			
2032	1,335	13,756	12,422	0	2,568	0	46,214	0	110,870	470,000	627,083	48,913	0	64,714	0	113,627	0.09	1,660	30	0.00	0	0	0.09	1,660	30			
2033	1,335	13,756	12,422	0	2,568	0	46,214	0	110,870	470,000	627,083	48,913	0	64,714	0	113,627	0.09	1,660	30	0.00	0	0	0.09	1,660	30			
2034	1,335	13,756	12,422	0	2,568	0	46,214	0	110,870	470,000	627,083	48,913	0	64,714	0	113,627	0.09	1,660	30	0.00	0	0	0.09	1,660	30			
2035	1,335	13,756	12,422	0	2,568	0	46,214	0	110,870	470,000	627,083	48,913	0	64,714	0	113,627	0.09	1,660	30	0.00	0	0	0.09	1,660	30			
2036	1,335	13,756	12,422	0	2,568	0	46,214	0	110,870	470,000	627,083	48,913	0	64,714	0	113,627	0.09	1,660	30	0.00	0	0	0.09	1,660	30			
2037	1,335	13,756	12,422	0	2,568	0	46,214	0	110,870	470,000	627,083	48,913	0	64,714	0	113,627	0.09	1,660	30	0.00	0	0	0.09	1,660	30			
2038	1,335	13,756	12,422	0	2,568	0	46,214	0	110,870	470,000	627,083	48,913	0	64,714	0	113,627	0.09	1,660	30	0.00	0	0	0.09	1,660	30			
2039	1,335	13,756	12,422	0	2,568	0	46,214	0	110,870	470,000	627,083	48,913	0	64,714	0	113,627	0.09	1,660	30	0.00	0	0	0.09	1,660	30			
2040	1,335	13,756	12,422	0	2,568	0	46,214	0	110,870	470,000	627,083	48,913	0	64,714	0	113,627	0.09	1,660	30	0.00	0	0	0.09	1,660	30			
2041	1,335	13,756	12,422	0	2,568	0	46,214	0	110,870	470,000	627,083	48,913	0	64,714	0	113,627	0.09	1,660	30	0.00	0	0	0.09	1,660	30			
2042	1,335	13,756	12,422	0	2,467	0	44,402	0	110,870	470,000	625,272	48,913	0	78,270	3,907	131,090	0.09	1,660	29	0.00	0	0	0.09	1,660	29			
2043	1,229	12,664	11,436	0	2,462	0	44,309	0	110,870	473,384	628,562	48,913	0	77,754	4,884	131,551	0.09	1,660	29	0.00	0	0	0.09	1,660	29			
2044	1,233	12,706	11,473	0	2,457	0	44,215	0	110,870	476,768	631,852	48,913	0	77,239	5,861	132,012	0.09	1,660	29	0.00	0	0	0.09	1,660	29			
2045	1,237	12,748	11,511	0	2,456	0	44,205	0	110,870	480,152	635,227	48,913	124	77,187	6,837	133,062	0.09	1,660	29	0.00	4	0	0.09	1,664	29			
2046	1,237	12,752	11,515	0	2,456	0	44,196	0	110,870	483,536	638,602	48,913	249	77,136	7,814	134,112	0.09	1,660	29	0.01	8	0	0.10	1,668	30			
2047	1,237	12,756	11,518	0	2,455	0	44,187	0	110,870	486,920	641,976	48,913	373	77,085	8,791	135,162	0.09	1,660	29	0.01	13	0	0.10	1,673	30			
2048	1,238	12,760	11,522	0	2,455	0	44,177	0	110,870	490,304	645,351	48,913	498	77,033	9,768	136,212	0.09	1,660	29	0.02	17	1	0.11	1,677	30			
2049	1,238	12,764	11,526	0	2,454	0	44,168	0	110,870	493,688	648,726	48,913	622	76,982	10,744	137,262	0.09	1,660	29	0.02	21	1	0.11	1,681	30			
2050	1,239	12,768	11,530	0	2,454	0	44,159	0	110,870	497,072	652,100	48,913	746	76,931	11,721	138,311	0.09	1,660	29	0.03	25	1	0.12	1,685	30			
2051	1,239	12,772	11,533	0	2,453	0	44,149	0	110,870	500,456	655,475	48,913	871	76,880	12,698	139,361	0.09	1,660	29	0.03	30	1	0.12	1,690	30			
2052	1,239	12,776	11,537	0	2,453	0	44,140	0	110,870	503,840	658,850	48,913	995	76,828	13,675	140,411	0.09	1,660	29	0.04	34	1	0.13	1,694	30			
2053	1,240	12,781	11,541	0	2,507	0	45,116	0	110,870	507,224	663,210	48,913	1,120	65,984	14,651	130,668	0.09	1,660	29	0.04	38	1	0.14	1,698	31			
2054	1,325	13,654	12,329	0	2,507	0	45,115	0	110,870	510,608	666,593																	

TABLE 8-5
AFC
ANNUAL PRODUCTION OF RADIOACTIVE WASTES FOR TWO RECYCLES OF MOX IN LWRs

Year	Uranium Enrichment				LLW from LWR Spent Fuel Reprocessing, LWR and FR reactor ops. & RU Reenrichment							TRU Wastes from LWR & FR Spent Fuel Reprocessing & LWR MOX & FR Reactor Fuel Fabrication					High Level Radioactive Wastes from LWR Spent Fuel Reprocessing			High Level Radioactive Wastes from FR Spent Fuel Reprocessing			High Level Radioactive Wastes from AFC Fuel Cycle with 2 Pu Recycles		
	UO ₂ Reactor Fuel @ 5% Enrichment	Natural U Feed to Enrichment	DU Enrichment Tails	RU Feed to Enrichment	RUOx to Depository	DRUOx to Depository	Conversion Waste LLW	DRU Conversion Waste LLW	Reprocess LLW	Reactor Ops LLW	Fuel Cycle LLW	TRU from LWR Spent Fuel Reprocessing	TRU from FR Spent Fuel Reprocessing	TRU from LWR MOX Fuel Fab	TRU from FR Fuel Fab	Fuel Cycle TRU	Kr ⁸⁵ /H ³ from LWR Spent Fuel	HLW + Hulls/Tc ⁹⁹ / Dissolver Sludge from LWR Spent Fuel	Sr/Cs/Ba/Rb from LWR Spent Fuel	Kr ⁸⁵ /H ³ from FR Spent Fuel	HLW + Hulls/Tc ⁹⁹ / Dissolver Sludge from FR Spent Fuel	Sr/Cs/Ba/Rb from FR Spent Fuel	Kr ⁸⁵ /H ³ from UREX+3c Fuel Cycle	HLW + Hulls/Tc ⁹⁹ / Dissolver Sludge from UREX+3c Fuel Cycle	Sr/Cs/Ba/Rb from UREX+3c Fuel Cycle
	(metric tons U)	(metric tons U)	(metric tons U)	(metric tons U)	(metric tons U)	(metric tons U)	(cu. ft.)	(cu. ft.)	(cu. ft.)	(cu. ft.)	(cu. ft.)	(cu. ft.)	(cu. ft.)	(cu. ft.)	(cu. ft.)	(cu. ft.)	(metric tons)	(metric ton SNF Equiv)	(metric tons)	(metric tons)	(metric ton SNF Equiv)	(metric tons)	(metric tons)	(metric tons)	(metric ton SNF Equiv)
2020	1,840	18,967	17,127	0	0	0	0	0	0	0	470,000	470,000	0	0	0	0	0.00	0	0	0.00	0	0	0.00	0	0.00
2021	1,840	18,967	17,127	0	0	0	0	0	0	0	470,000	470,000	0	0	0	0	0.00	0	0	0.00	0	0	0.00	0	0.00
2022	1,840	18,967	17,127	0	0	0	0	0	0	0	470,000	470,000	0	0	0	0	0.00	0	0	0.00	0	0	0.00	0	0.00
2023	1,840	18,967	17,127	0	0	0	0	0	0	0	470,000	470,000	0	0	0	0	0.00	0	0	0.00	0	0	0.00	0	0.00
2024	1,840	18,967	17,127	0	0	0	0	0	0	0	470,000	470,000	0	0	0	0	0.00	0	0	0.00	0	0	0.00	0	0.00
2025	1,840	18,967	17,127	0	0	0	0	0	0	0	470,000	470,000	0	0	0	0	0.00	0	0	0.00	0	0	0.00	0	0.00
2026	1,840	18,967	17,127	0	0	0	0	0	0	0	470,000	470,000	0	0	0	0	0.00	0	0	0.00	0	0	0.00	0	0.00
2027	1,840	18,967	17,127	0	0	0	0	0	0	0	470,000	470,000	0	0	0	0	0.00	0	0	0.00	0	0	0.00	0	0.00
2028	1,840	18,967	17,127	0	0	0	0	0	0	0	470,000	470,000	0	0	0	0	0.00	0	0	0.00	0	0	0.00	0	0.00
2029	1,840	18,967	17,127	0	0	0	0	0	0	0	470,000	470,000	0	0	0	0	0.00	0	0	0.00	0	0	0.00	0	0.00
2030	1,840	18,967	17,127	0	0	0	0	0	0	0	470,000	470,000	0	0	0	0	0.00	0	0	0.00	0	0	0.00	0	0.00
2031	1,840	18,967	17,127	2,568	0	2,055	0	36,982	110,870	470,000	617,851	48,913	0	94,094	0	143,007	0.09	1,660	30	0.00	0	0	0.09	1,660	30
2032	1,105	11,390	10,285	2,568	0	2,055	0	36,982	110,870	470,000	617,851	48,913	0	94,094	0	143,007	0.09	1,660	30	0.00	0	0	0.09	1,660	30
2033	1,105	11,390	10,285	2,568	0	2,055	0	36,982	110,870	470,000	617,851	48,913	0	94,094	0	143,007	0.09	1,660	30	0.00	0	0	0.09	1,660	30
2034	1,105	11,390	10,285	2,568	0	2,055	0	36,982	110,870	470,000	617,851	48,913	0	94,094	0	143,007	0.09	1,660	30	0.00	0	0	0.09	1,660	30
2035	1,105	11,390	10,285	2,568	0	2,055	0	36,982	110,870	470,000	617,851	48,913	0	94,094	0	143,007	0.09	1,660	30	0.00	0	0	0.09	1,660	30
2036	1,105	11,390	10,285	2,568	0	2,055	0	36,982	110,870	470,000	617,851	48,913	0	94,094	0	143,007	0.09	1,660	30	0.00	0	0	0.09	1,660	30
2037	1,105	11,390	10,285	2,568	0	2,055	0	36,982	110,870	470,000	617,851	48,913	0	94,094	0	143,007	0.09	1,660	30	0.00	0	0	0.09	1,660	30
2038	1,105	11,390	10,285	2,568	0	2,055	0	36,982	110,870	470,000	617,851	48,913	0	94,094	0	143,007	0.09	1,660	30	0.00	0	0	0.09	1,660	30
2039	1,105	11,390	10,285	2,568	0	2,055	0	36,982	110,870	470,000	617,851	48,913	0	94,094	0	143,007	0.09	1,660	30	0.00	0	0	0.09	1,660	30
2040	1,105	11,390	10,285	2,568	0	2,055	0	36,982	110,870	470,000	617,851	48,913	0	94,094	0	143,007	0.09	1,660	30	0.00	0	0	0.09	1,660	30
2041	1,105	11,390	10,285	2,568	0	2,055	0	36,982	110,870	470,000	617,851	48,913	0	94,094	0	143,007	0.09	1,660	30	0.00	0	0	0.09	1,660	30
2042	1,105	11,390	10,285	2,392	90	1,901	1,620	34,205	110,870	470,000	616,695	48,913	0	99,248	3,907	152,068	0.09	1,660	29	0.00	0	0	0.09	1,660	29
2043	1,065	10,975	9,911	2,365	113	1,879	2,025	33,814	110,870	473,384	620,092	48,913	0	98,251	4,884	152,048	0.09	1,660	29	0.00	0	0	0.09	1,660	29
2044	1,073	11,056	9,983	2,338	135	1,857	2,430	33,422	110,870	476,768	623,489	48,913	0	97,254	5,861	152,028	0.09	1,660	29	0.00	0	0	0.09	1,660	29
2045	1,080	11,136	10,056	2,335	137	1,855	2,464	33,389	110,870	480,152	626,875	48,913	124	97,172	6,837	153,046	0.09	1,660	29	0.01	4	0	0.10	1,664	29
2046	1,081	11,143	10,062	2,333	139	1,854	2,497	33,357	110,870	483,536	630,260	48,913	249	97,089	7,814	154,065	0.09	1,660	29	0.01	8	0	0.10	1,668	29
2047	1,082	11,149	10,068	2,331	141	1,852	2,531	33,324	110,870	486,920	633,645	48,913	373	97,006	8,791	155,083	0.09	1,660	29	0.02	13	0	0.11	1,673	30
2048	1,082	11,156	10,074	2,329	142	1,850	2,564	33,292	110,870	490,304	637,030	48,913	498	96,924	9,768	156,102	0.09	1,660	29	0.02	17	1	0.11	1,677	30
2049	1,083	11,163	10,080	2,326	144	1,848	2,598	33,259	110,870	493,688	640,415	48,913	622	96,841	10,744	157,120	0.09	1,660	29	0.03	21	1	0.12	1,681	30
2050	1,084	11,169	10,086	2,324	146	1,846	2,632	33,227	110,870	497,072	643,800	48,913	746	96,758	11,721	158,139	0.09	1,660	29	0.03	25	1	0.12	1,685	30
2051	1,084	11,176	10,092	2,322	148	1,845	2,665	33,194	110,870	500,456	647,185	48,913	871	96,676	12,698	159,157	0.09	1,660	29	0.04	30	1	0.13	1,690	30
2052	1,085	11,182	10,098	2,320	150	1,843	2,699	33,162	110,870	503,840	650,570	48,913	995	96,593	13,675	160,176	0.09	1,660	29	0.04	34	1	0.13	1,694	30
2053	1,085	11,189	10,104	2,421	92	1,934	1,653	34,807	110,870	507,224	654,553	48,913	1,120	91,708	14,651	156,392	0.09	1,660	29	0.05	38	1			

8.2 LOW LEVEL RADIOACTIVE WASTES (LLW)

Because the cumulative amount of LWR spent fuel reprocessed is about the same in the fuel cycle simulations conducted for this study, the amounts of LLW from reprocessing are about the same -- 7.1 to 7.3 million cubic feet. Slight differences are a result of differences in the year that the simulations transition from increasing the number of FRs in succeeding years to a steady-state where the number of FRs remains constant from year to year. The simulations of the fuel cycles that assume LWR spent fuel would be recycled one time make the transition to the steady-state status in 2086; for the simulations where the spent fuel is recycled twice, the transition occurs in 2088. At about the same time the transition occurs, the simulation reduced the rate of reprocessing of LWR spent fuel from 3,000 MTIHM per year to 1,840 MTIHM per year -- the rate that LWR spent fuel is produced in the LWR reactors. Thus, the amounts of LLW for the simulations where LWR spent fuel is assumed to be recycled twice is about 2 percent greater than for those where LWR spent fuel is assumed to be recycled only one time.

Overall, the simulations estimate that the amounts of LLW from the two fuel cycles and for 1 and 2 recycles of LWR spent fuel would range from 56.3 million cubic feet for the UREX+3c fuel cycle with 1 recycle, to 48.6 million cubic feet (13.6 percent less) for the AFC with 2 recycles.

8.3 TRU WASTES

The amounts of TRU wastes from the simulations range from about 9.8 million cubic feet for the UREX+3c fuel cycle with 2 recycles of LWR spent fuel, to 12.1 (23 percent more) million cubic feet for the AFC with 1 recycle. The amounts of TRU wastes differ for two reasons. First, as discussed in other sections of this report, in the AFC, reenriched recovered uranium is used to provide makeup uranium for production of MOX fuel. Use of the reenriched recovered uranium produces a greater amount of MOX fuel compared to the amount produced in the UREX+3c fuel cycle, which uses depleted uranium for makeup. The greater amount of MOX in the AFC leads to a corresponding greater amount (about 39 percent more for 1 recycle LWR spent fuel and about 34 percent more if the spent fuel is recycled two times) of TRU waste from MOX fuel fabrication activities. The TRU waste from MOX fuel fabrication accounts for about 35 percent of fuel cycle TRU for the UREX+3c fuel cycle with 1 recycle, and about 54 percent of the fuel cycle TRU wastes for the AFC with 2 recycles.

The second reason the amounts of TRU wastes differ among the simulations is that the number of FRs differ. More FRs are needed to consume plutonium and minor actinide elements produced in LWRs if the LWR spent fuel is recycled only one time, than are needed if the LWR spent fuel is recycled twice. For the simulations conducted for this study, the achievement of a steady-state situation (where the amounts of plutonium and minor actinides being produced in LWRs are in balance with the amounts consumed in the FRs) required 43 FRs* for 1 recycle, and 23 fast reactors for 2 recycles of LWR spent fuel. Thus, there is a difference of almost a factor of 2 (between the cases for 1 and 2 recycles of LWR spent fuel) in the amounts of FR fuel that would need to be produced and recycled. This gives rise to a corresponding difference of almost a factor of 2 in the amounts of TRU waste that would arise from FR fuel production and recycling. Among the simulations, the TRU wastes from FR fuel production and recycling ranges from 3.8 million cubic feet for the cases where LWR spent fuel is assumed to be recycled only one time, to 2 million cubic feet if it is assumed to be recycled 2 times.

Last, as discussed in Section 8.2, there is little difference among the scenarios in the amounts of LWR spent fuel that is reprocessed. Thus, there is little variation in the amount of TRU waste from reprocessing LWR spent fuel. TRU from LWR spent fuel reprocessing, which ranges from 3.1 to 3.2 million cubic feet, constitutes between 26 and 33 percent of the TRU wastes projected by the fuel cycle simulations.

8.4 FUEL CLADDING HULLS AND HLW

The simulations also estimated the aggregate volume of spent fuel cladding hulls and associated fuel assembly metal components; dissolver sludge (from LWR spent fuel reprocessing) and process solids (from both LWR and FR spent fuel reprocessing); and Tc⁹⁹. For purposes of the analysis here, the volume of these wastes was assumed to be 94 cubic feet per gigawatt-year of electric power production -- a value commonly used (Ref. 1) to estimate the volume of these wastes from LWR spent nuclear fuel reprocessing and assumed herein also to be applicable for reprocessing of FR spent fuel.

* For simplicity, the number of fast reactors is based on a 100 percent plant utilization factor. If a more realistic (lower) utilization factor were used the number of fast reactors in the simulations would have been greater -- but, the amount of fast reactor fuel recycled would have been about the same.

Even though the amount of heat produced would be substantially less, spent fuel hulls and the associated wastes would likely be disposed in a mined geologic repository in packages similar in dimension and weight to repository spent fuel waste packages. As a measure of disposal equivalence, the analysis compared the volume of spent fuel hulls and associated wastes to that of the originating spent fuel assemblies. The resulting fraction is 0.31 when 300 cubic feet is used for the annual volume of spent fuel discharged per gigawatt-year of electric power production. Thus, for disposal in a mined geologic repository, the hulls and associated materials from 1 MTIHM of spent nuclear fuel were assumed to be equivalent to 0.31 MTIHM of spent nuclear fuel assemblies. A similar analysis was done to determine a corresponding measure of equivalence for vitrified high level radioactive waste from reprocessing. For vitrified HLW, the analysis used a volume of 72 cubic feet per gigawatt-year of electric power produced by nuclear plants (Ref. 1). Thus, for HLW, the fractional equivalence to spent fuel assemblies determined in the same manner as for spent fuel cladding hulls is 0.24. Thus, the analysis assumed that disposal of hulls and associated materials plus HLW wastes from reprocessing 1 MTIHM of LWR spent fuel in a mined geologic repository would be equivalent to disposing 0.55 metric tons (initial heavy metal) of spent fuel assemblies. Said differently, on a volume basis, the geologic repository would accommodate almost two times the amount of wastes from a gigawatt-year of nuclear power production if the spent fuel is reprocessed. For the simulations conducted for this study, the hulls, process solids, associated wastes, and vitrified HLW from the fuel cycles range from 115,600 to 119,300 MTIHM equivalent of spent fuel. This amount of waste arises from reprocessing 209,000 MTIHM of LWR and FR spent fuel when LWR spent fuel is recycled one time and 215,600 MTIHM when recycled two times. The differences are the result of the different number of FRs required (see above discussion) and the resulting total amount of reactor fuel that would be reprocessed.

8.5 EXCESS URANIUM

Both the UREX+3c fuel cycle and the AFC would generate excess recovered uranium from LWR spent fuel reprocessing. Here, the differences between the two fuel cycle alternatives are noteworthy. The analysis assumes that the UREX+3c fuel cycle would convert the excess recovered uranium to a stable solid form (U_3O_8) and dispose of it as Class B, C, or possibly greater than Class C (GTCC) LLW (Ref. 2). In contrast, the AFC would reenrich the excess

recovered uranium and reuse the enriched product in LWR MOX fuel. The analysis assumes that the tails from this reenrichment would be converted to a stable solid form and disposed of as Class B or C LLW (because U^{232} , U^{234} and U^{236} would be depleted in the tails). The quantities of the uranium that would be disposed are listed in Table 8-6. Because about 20 percent of the excess recovered uranium would end up in the reenriched recovered uranium in the AFC, the total amount of depleted recycled uranium that would be disposed as LLW would be about 20 percent less than the amount of excess recovered uranium that would be disposed in the UREX+3c fuel cycle. Table 8-6 also identifies the process origins of the excess recovered uranium for the two fuel cycles and for the alternatives of 1 and 2 recycles of LWR spent fuel. As shown, all the excess recovered uranium from the UREX+3c fuel cycle originates from LWR reprocessing, whereas reprocessing is the origin of only a relatively small amount of excess recovered uranium from the AFC if LWR spent fuel is recycled 1 time and even less if the LWR spent fuel is recycled two times. The remainder of the excess recovered uranium from the AFC is the depleted recycled uranium from reenrichment.

TABLE 8-6
QUANTITIES OF RECOVERED URANIUM FROM THE UREX+3c FUEL CYCLE
AND AFC

Originating Process	Fuel Cycle (MTU)			
	UREX+3c/ 1 Recycle	AFC/ 1 Recycle	UREX+3c 2 Recycles	AFC/ Recycles
UO ₂ Reprocessing	146,448	N/A	96,567	N/A
Reenrichment of RU After UO ₂ Reprocessing	N/A	112,449	N/A	76,701
UO ₂ + MOX-1 Co-reprocessing	N/A	N/A	61,526	N/A
Reenrichment of RU After UO ₂ + MOX-1 Co-reprocessing	N/A	N/A	N/A	48,766
MOX-1 Reprocessing	13,727	20,140	1,014	1,490
MOX-2 Reprocessing	N/A	N/A	4,329	5,158
Total Excess RU & RDU	160,175	132,589	163,436	132,115

Table 8-7 presents the isotopic compositions of the various recovered uranium and depleted recycled uranium streams from the two fuel cycles for the process origins described above in Tables 8-2 through 8-5. What is important about the compositions of these streams is that the presence of significant amounts of U^{232} , U^{234} , and U^{236} (and probably trace quantities of transuranium isotopes) in these materials make it likely that they would be disposed as Class B, C, or GTCC LLW. Thus, this analysis assumes the excess recovered uranium would be disposed as Class B, C, or GTCC LLW after it had been converted to a stable solid form.

For the AFC, excess recovered uranium originates from segregated processing of MOX spent fuel to provide uranium, plutonium, and minor actinides for use in FR fuel. This excess recovered uranium has high concentrations of U^{236} that would be likely to make it unattractive for reenrichment for further use. Also, for the UREX+3c fuel cycle, the very low concentrations of U^{235} in the excess recovered uranium from reprocessed MOX spent fuel would likely make it unattractive for reenrichment for further use. The comparatively high enrichments of excess recovered uranium from the UREX+3c fuel cycle from reprocessing UO_2 spent fuel and co-processing UO_2 and MOX spent fuel calls its disposal into question (see Section 10.1 of this report).

TABLE 8-7
COMPOSITION OF RECOVERED URANIUM FOR THE UREX+3c FUEL CYCLE
AND AFC

Uranium Isotope	Originating Process	Fraction of Total in Excess Uranium			
		UREX+3c/ 1 Recycle	AFC/ 1 Recycle	UREX+3c/ 2 Recycles	AFC/ 2 Recycles
U ²³²	UO ₂ Reprocessing RU Reenrichment	2.81E-09	1.00E-10	2.81E-09	1.00E-10
	UO ₂ + MOX-1 Co- reprocessing RU Reenrichment	N/A	N/A	3.33E-09	2.00E-10
	MOX-1 Reprocessing	5.90E-09	1.53E-08	5.90E-09	1.53E-08
	MOX-2 Reprocessing	N/A	N/A	6.9E-09	2.09E-08
U ²³⁴	UO ₂ Reprocessing RU Reenrichment	2.33E-04	4.00E-05	2.33E-04	4.00E-05
	UO ₂ + MOX-1 Co- reprocessing RU Reenrichment	N/A	N/A	2.31E-04	5.00E-05
	MOX-1 Reprocessing	2.23E-04	5.87E-04	2.23E-04	5.87E-04
	MOX-2 Reprocessing	N/A	N/A	1.28E-04	7.96E-04
U ²³⁵	UO ₂ Reprocessing RU Reenrichment	8.28E-03	2.50E-03	8.28E-03	2.50E-03
	UO ₂ + MOX-1 Co- reprocessing RU Reenrichment	N/A	N/A	7.44E-03	2.5E-03
	MOX-1 Reprocessing	3.08E-03	8.55E-03	3.08E-03	8.55E-03
	MOX-2 Reprocessing	N/A	N/A	1.99E-03	9.37E-03
U ²³⁶	UO ₂ Reprocessing RU Reenrichment	7.74E-03	3.70E-03	7.74E-03	3.70E-03
	UO ₂ + MOX-1 Co- reprocessing RU Reenrichment	N/A	N/A	7.01E-03	4.88E-03
	MOX-1 Reprocessing	3.23E-03	1.78E-02	3.23E-03	1.78E-02
	MOX-2 Reprocessing	N/A	N/A	2.88E-03	2.03E-02
U ²³⁸	UO ₂ Reprocessing RU Reenrichment	9.84E-01	9.94E-01	9.84E-01	9.94E-01
	UO ₂ + MOX-1 Co- reprocessing RU Reenrichment	N/A	N/A	9.85E-01	9.93E-01
	MOX-1 Reprocessing	9.94E-01	9.73E-01	9.94E-01	9.73E-01
	MOX-2 Reprocessing	N/A	N/A	9.95E-01	9.70E-01

8.6 CONCLUSIONS

Over the 81 year simulation used in this study, and for both 1 and 2 recycles of LWR spent fuel, the AFC produced about 10 percent less (~700 million cubic feet less) uranium mine and mill tailings than the UREX+3c fuel cycle. This is because the AFC reenriches and reuses the excess recovered uranium from reprocessing, where the UREX+3c fuel cycle does not. The AFC, therefore, uses less UO_2 fuel in LWRs than does the UREX+3c fuel cycle.

Based on the 81 year simulations, the AFC would produce between 8 and 9 percent less LLW than the UREX+3c fuel cycle. Recycling LWR spent fuel two times in the fuel cycles would result in 4 to 5 percent less LLW than recycling the spent fuel once. These estimates do not consider the amount of LLW that would be produced by nuclear power plant operations and do not include the disposal of excess recovered uranium or depleted recycled uranium from the two fuel cycles. If disposed as LLW, the latter materials would increase the amount (cu ft.) of LLW by about 1 percent.

The AFC would produce between 14 and 16 percent more cubic feet of TRU waste than would the UREX+3c fuel cycle. The principal difference is in the amount of TRU waste from fabricating LWR MOX fuel, which accounts for between 35 and 54 percent of this waste from the fuel cycles. Use of reenriched recovered uranium by the AFC results in the production of more MOX fuel compared to the amount produced for the UREX+3c fuel cycle. Because there would be fewer FRs and thus less FR fuel reprocessed and recycled by the two fuel cycles, recycling LWR spent fuel two times versus one time would result in 6 to 8 percent less TRU waste.

Because the fuel cycle simulations used the same amount of reprocessing for the two fuel cycles (slight difference for 2 recycles of LWR spent fuel compared to 1 recycle), the amounts of spent fuel cladding hulls, dissolver sludge and fines, other processing solids (possibly including Tc^{99}) did not differ. Because they would contain small amounts of TRU and long lived fission products (including Tc^{99}), these wastes would be disposed in a geologic repository. Compaction or possibly alloying would reduce the volume of this waste stream for both fuel cycles to 31 percent (possibly less) of the volume of the originating spent fuel assemblies.

For the same reason -- that is, the amounts of reprocessing would be the same -- the amounts of vitrified HLW would be the same for the two fuel cycles and would amount to about

24 percent of the volume of the originating spent fuel assemblies. These wastes would also be disposed in a geologic repository.

Fuel assembly hulls and HLW from both fuel cycles would produce low levels of heat compared to what would be produced by the originating spent nuclear fuel because:

- (1) More than 99 percent of the TRU isotopes (which generate significant amounts of heat) from spent fuel would not be in the wastes but would instead be used in fuel for LWR and FRs
- (2) The high-heat, cesium and strontium fission products would be removed for storage to allow these fission products to decay until their heat levels are low enough for disposal.

There would be about 20 percent more unused (excess) recovered uranium from the UREX+3c fuel cycle than depleted recycled uranium plus recovered uranium from the AFC. Both would need to be converted to a stable solid form for long term storage or disposal. Because of its radiotoxicity, excess recovered uranium from the UREX+3c fuel cycle (and a relatively small amount from the AFC) might be classified as GTCC LLW. Because reenrichment would remove or deplete most of the isotopes of concern, it is possible that the depleted recycled uranium from the AFC could be disposed in a near-surface repository as Class B or C LLW. However, some wastes from the reenrichment facility might contain TRU.

8.7 REFERENCES FOR SECTION 8.0

- (1) Institute of Nuclear Materials Management, Waste Management Tutorial, Waste Management Technical Division, February 17, 2009 (updated values to be published).
- (2) G. D. Del Cul et al, Analysis of the Reuse of Uranium Recovered from the Reprocessing of Commercial LWR Spent Fuel, Oak Ridge National Laboratory, ORNL/TM-2007/207, January 2009

9.0 DIFFERENCES IN THE SUSTAINABILITY OF THE UREX+3c FUEL CYCLE AND THE AFC

This section presents a comparative analysis of the sustainability of the UREX+3c fuel cycle and the AFC. The metrics for sustainability that are used are based on a contemporary description provided by Clark* in 2006 (Ref. 1). These metrics are:

- (1) Natural Resources
 - (a) Consumption of natural resources compared to resulting benefits
 - (b) Consumption of natural resources compared to availability of the resources
- (2) Wastes
 - (a) Environmental toxicity, quantity, and persistence of waste products compared to associated benefits
 - (b) Consumption of waste treatment and disposal resources compared to associated benefits
- (3) Societal Acceptance/Tolerance
 - (a) Level of ongoing societal risk compared to associated societal benefits
- (4) Economic Resources
 - (a) Consumption of economic resources compared to resulting benefits
 - (b) Consumption of economic resources compared to availability of the resources
- (5) Human Resources
 - (a) Consumption of human resources compared to resulting benefits
 - (b) Consumption of human resources compared to the availability of the resources
- (6) Cumulative impacts that affect sustainability

9.1 NATURAL RESOURCES

Over the 81 year period of the simulations described in Section 4.0, both the UREX+3c fuel cycle and the AFC, each of which has 100 LWRs, would improve use of natural resources by consuming less natural uranium and producing more electric power than would a once-through fuel cycle with the same 100 LWRs. Both advance fuel cycle systems would similarly

* “Sustainability applies demand management of all goods and services by promoting reduced consumption, using renewable resources where possible, and encouraging practices that minimize resource intensity while maximizing resource productivity.”

use less uranium resources if plutonium was recycled two times versus one time to produce MOX fuel for the LWRs. Over the 81 years, an AFC in which MOX was recycled two times in the LWRs would use the least amount of uranium resources -- 32 percent less than a once-through fuel cycle -- but, because there would be fewer FRs, it would also produce about 11 percent less electric power than if the MOX were recycled only one time. Also over the 81 years, an AFC that recycles MOX in LWRs only one time would produce the most electric power per metric ton of uranium that was consumed -- 94 MWe-years per metric ton uranium; this is about 10 percent greater than that for a UREX+3c fuel cycle with one recycle of MOX in LWRs and about 4 percent greater than an AFC with two recycles of MOX in LWRs. For perspective, the value for a once-through LWR fuel cycle would be 54 MWe-years per metric ton uranium fuel. At the end of the 81 year period of the simulation, an AFC with 43 FRs in which plutonium was recycled once in LWRs would produce electric power at a rate of 110 MWe-years per metric ton of LWR uranium fuel.

Conversely, both fuel cycles would consume raw materials to construct and operate major new facilities -- reprocessing plants and FRs. The AFC would also require raw materials to construct and operate a plant for enriching recovered uranium. Both systems would use lesser amounts of raw materials for the construction and operation of FRs if MOX was recycled two times versus one time in LWRs; but they would be likely to use similar amounts of raw materials to construct reprocessing plants and plants to produce MOX and FR fuels. Continuing operation of the existing once-through LWR fuel cycle would only use raw materials for those operations and for the construction of new plants and facilities, as may be needed to replace ones retired from service because of age. The UREX+3c fuel cycle and the AFC are similarly assumed to replace plants and facilities that are retired from service.

The amounts of long-lived radioactive wastes requiring packaging would be similar for the two fuel cycles. For either fuel cycle, as simulated, if MOX was recycled once in LWRs, versus two times, there would be greater amounts of high level radioactive wastes because there would be more FRs and greater amounts of electric power would be produced. Nonetheless, compared to the once-through nuclear fuel cycle, about half as much high-value raw materials would be needed for geologic disposal and isolation of the high level radioactive wastes.

Sustained, long term operation of the UREX+3c fuel cycle or the AFC would require consumption of raw materials to periodically replace major facilities that would be retired from

service because of age. For both systems, the number of FRs that would need to be maintained and periodically replaced would be less if MOX was recycled two times in the systems' LWRs. The UREX+3c fuel cycle system would have a slight advantage because the facility used in the AFC system to reenrich recovered uranium would need to be maintained.

None of the resources that would be needed to construct and operate the facilities in the either fuel-cycle are scarce or at risk of being depleted. Depletion of uranium reserves would be significantly less, compared to the once-through uranium fuel cycle, if either the UREX+3c fuel cycle or AFC were implemented. And, the long-term sustained use of uranium resources by either of the fuel cycles would be about one-half of that used (per MWe-year) by a once-through fuel cycle (see Table 4-5).

9.2 WASTES

As simulated, over 81 years both the UREX+3c fuel cycle and the AFC, each of which has 100 LWRs, would significantly reduce the nearly 60,000 MTIHM of legacy inventory of spent nuclear fuel that is expected to exist in 2010. Also, a UREX+3c fuel cycle or AFC would eliminate the need to dispose of 165,000 MTU of additional spent nuclear fuel that would accumulate from 2010 to 2100 from 100 LWRs in a once-through fuel cycle.

Both of the advanced nuclear fuel cycles would produce HLW, TRU, and LLW that would need to be managed. The amounts of these wastes would depend on whether the fuel cycle employed one or two recycles of MOX in LWRs. There would be about 6 percent less of the wastes if the MOX was recycled two times because there would be fewer FRs, less nuclear fuel to recycle, and less electric power (also about 11 percent less) generated. Either advanced fuel cycle system would unavoidably discard a small mass (about 1 percent) of the TRU elements it produced as radioactive waste requiring isolation in a geologic repository. This can be compared to a once-through fuel cycle where the spent nuclear fuel containing all of the TRU elements produced in LWRs is a waste. However, in both of the UREX+3c fuel cycle and the AFC, reprocessing and fabrication of nuclear fuel containing plutonium and actinide elements would produce significant amounts of low level TRU wastes that would require geologic disposal.

The amounts of high level radioactive waste containing the low-heat fission products, high-heat-producing cesium-strontium wastes, and fuel assembly hardware and cladding hulls

separated from LWR and FR spent nuclear fuel in reprocessing plants would be approximately proportional to the total amount of electric power produced. Although, because the burnup achieved in FR fuel is greater, the amounts of fuel assembly hardware and cladding hulls would be somewhat less per MWe-year for fuel cycles that had more FRs. Nonetheless, principally because they would burn more nuclear fuel and produce more electric power, over the 81 year simulation the UREX+3c fuel cycle and the AFC that employ only one recycle of MOX in LWRs produced about 11 percent more high level radioactive waste than would the same systems in which MOX was recycled two times.

Nuclear fuel reprocessing and new fuel production facilities would be constructed and operated for either of the advanced fuel cycle systems. FR nuclear power plants would also be constructed for both systems. The AFC would require construction and operation of a recovered uranium enrichment facility. At the ends of their economic lives, the reprocessing, MOX, and FR fuel facilities would be decontaminated and decommissioned producing low level and TRU wastes as well as industrial wastes. Likewise, at the end of its economic life the AFC facility used to enrich recovered uranium would also have to be decontaminated and decommissioned producing low level radioactive, possibly TRU, and industrial wastes.

Last, the AFC would discard lesser amounts of recovered uranium that would, on a per ton basis, have less radiotoxicity than would the UREX+3c fuel cycle. The UREX+3c fuel cycle would discard about 93 percent of the uranium from the spent LWR fuel if MOX was recycled one time in LWRs and about 91 percent if the MOX was recycled two times. The AFC would discard about 83 percent of the uranium from the spent fuel if MOX was recycled once and 79 percent if the MOX was recycled two times. The AFC would consume less virgin uranium than would the UREX+3c fuel cycle for either the one or two recycles of MOX.

The composition of uranium discarded by the UREX+3c fuel cycle would include between 0.8 weight percent (for 1 recycle) and 0.7 weight percent (for 2 recycles) for each of the U^{235} and U^{236} isotopes, about 0.02 weight percent of the U^{234} isotope, and $2.8-6.9 \times 10^{-7}$ weight percent U^{232} as well as trace amounts of TRU and fission products. After conversion to a stable solid form, this recovered uranium would be discarded as Class B, C, or possibly GTCC LLW (Ref. 2). For the AFC, the composition of depleted recycled uranium that would be discarded as tails from reenrichment would be about 0.25 weight percent for the U^{235} , 0.4 weight percent for U^{236} , 0.004-0.005 weight percent for U^{234} , and $1-2 \times 10^{-8}$ weight percent for U^{232} . After

conversion to a suitable solid, these tails would likely be disposed as Class B or C LLW. Between 5 and 15 percent of uranium that would be discarded by the AFC would be non-enriched recovered uranium. The U^{235} content of this recovered uranium would be between 0.85 and 0.95 weight percent, but the recovered uranium would also contain between 1.8 and 2 weight percent U^{236} , between 0.06 and 0.08 weight percent U^{234} , and between 1.5 and 2.1×10^{-6} weight percent U^{232} as well as trace amounts of TRU and fission product isotopes (see Figures 7-2 and 7-4). After conversion to a suitable solid, this last category of recovered uranium would likely be disposed as GTCC.

9.3 SOCIETAL ACCEPTANCE/TOLERANCE

Because both advanced fuel cycles require nearly equal investment in, and introduction of, reprocessing, production of recycle nuclear fuels, introduction of FRs, and advanced approaches for the management of high level radioactive wastes, neither appears to have an advantage or disadvantage regarding initial and sustained acceptance and tolerance by society. Both systems would require substantial shifts in national energy policy regarding utilization of plutonium in the nuclear fuel cycle, management of radioactive wastes, and introduction of major new technologies for producing electric power in FRs. The AFC would make more effective use of uranium resources -- a possible advantage -- but would require a facility to enrich recycled uranium -- a possible disadvantage. Both systems, as simulated, would consume the legacy spent nuclear fuel from the presently practiced once-through fuel cycle and would substantially reduce the long-term inventories of these materials. In addition, societal acceptance of both systems would benefit from the fact that both provide sustainable and productive consumption of minor actinide elements -- a waste product of LWR nuclear power reactors -- as well as improvements for managing other radioactive wastes from nuclear power. Both systems would offer the societal benefit of providing increased production of electric power by non-greenhouse gas producing sources.

9.4 ECONOMIC RESOURCES

Both the UREX+3c fuel cycle and the AFC would require substantial economic investments in nuclear technology, equipment, facilities, and human resources and neither appears to have a significant advantage or disadvantage with regard to the other. Developing,

demonstrating, and implementing either system would be likely to require substantial financial underwriting by the federal government -- because of the past precedents of changes in Federal Government policy that can threaten the economic viability of the facilities involved. Assuming federal government investment, the scope and duration of financial investments would require that either system be shown to offer reasonable potential for long-term, sustained, and sustainable benefits to national energy security and the national economy. Both have the potential to provide economic advantages but, given the associated economic risks, either would need to be further developed and demonstrated before its economic long-term viability could be assured.

9.5 HUMAN RESOURCES

Neither the UREX+3c fuel cycle nor the AFC has an advantage over the other in the development and utilization of human resources. Both would require substantial investments in the development and training of people and in the maintenance of a complex institutional system that would involve:

- Supply and protection of nuclear materials
- Operations and maintenance of nuclear fuel reprocessing, fuel fabrication, and electric power plants
- Nuclear plant and equipment safety and oversight
- Physical security for plants, equipment, and materials.

The human resources would be employed in a sustained and diverse array of jobs including:

- Facility construction
- Equipment fabrication
- Facility and equipment maintenance
- Radioactive waste management
- Nuclear, chemical, and other engineering disciplines
- Nuclear plant operations
- Physical security
- Nuclear materials management and accountability
- Quality assurance
- Information technology; law and regulatory compliance
- Management, and other jobs having requirements ranging from specialized technical, business, and administrative skills to unskilled labor.

However, neither advanced fuel cycle would require a complexity of human institutions or command use of human resources at a level that would tax the capability of domestic education and training resources. Both systems would promote advanced training and knowledge that would have extended benefits for sustained national economic health.

9.6 CUMULATIVE EFFECTS

Because they both make productive use of byproducts of nuclear fission of uranium in LWRs, both fuel cycle systems present the potential to reduce long-term cumulative effects of generating electric power using the energy from nuclear fission reactions. As simulated, both systems present the potential to substantially reduce the accumulating amounts of stored LWR spent nuclear fuel. Both systems would make 100-fold reductions in accumulations (for disposal) of long-lived minor actinide elements that have significant effects on the size and performance of a mined geologic repository, and both would limit the systemwide accumulation of plutonium and minor actinide elements to 50 percent of what would accumulate over the 81 year simulation if electric power was produced in 100 LWRs with a once-through fuel cycle.

Both systems present the potential to significantly reduce the accumulation of environmental impacts of mining, milling, and enriching virgin uranium and disposing of the depleted uranium residue. The AFC would further reduce the use of virgin uranium and would also reduce the toxicity of recovered uranium that was discarded.

Both systems make significant reductions in the amounts of plutonium that would have the potential to be diverted in the future for use in nuclear explosive devices. Both systems would also provide incentives for sustained and sustainable positive institutional and self protection controls on limited inventories of plutonium that would have substantial value as energy resources.

Both systems would accumulate risks of chronic health effects arising from low levels of ionizing radiation from nuclear plants, radioactive materials transportation, and disposal activities. Both would increase the amounts of radioactive fission products that would need to be managed and safely disposed. The amounts of the long-life fission products Tc^{99} (25 grams per MWe-year) and I^{129} (5 grams per MWe-year) would be approximately proportional to the amount of electric power produced (MWe-years) by nuclear fission reactors. The amount of

electric power produced, which for the simulations would be comparable for the two fuel-cycle systems, would be greater if MOX was recycled one time versus two times in LWRs.

By offsetting requirements for electric power generation by fossil fueled sources, both systems offer the potential to substantially and sustainably reduce the production of green-house gasses.

Both systems present the prospect of providing a base for the evolutionary introduction of fast breeder reactors to replace LWRs and FRs and thereby provide access to the substantial energy resource available in extant inventories of depleted uranium.

9.7 CONCLUSIONS

Both the UREX+3c fuel cycle and the AFC could contribute to the long-term sustainability of nuclear electric power production in the U.S. Each offers advantages for improving the utilization of natural resources, managing radioactive wastes, promoting societal acceptance of nuclear power, stewarding economic resources, promoting the development of human resources, and reducing the cumulative effects of nuclear power and energy utilization more generally. Each also presents challenges to long term sustainability that would need to be addressed. Factors that would affect the sustainability of both systems have a high degree of uncertainty. These factors include:

- (1) Energy policies of the U.S. government
- (2) The need for substantial technical development of key systems
- (3) The need for large investments in the infrastructure of plant, equipment, materials, knowledge, human resources, and regulatory practice
- (4) The future cost to industry and other consumers of energy from non-nuclear sources
- (5) The future national and international demand for energy
- (6) The future cost of natural resources such as uranium

Because of these uncertainties and because the two fuel cycles investigated by this study are very similar in scope, neither has a clear advantage with regard to long-term sustainability when the two are compared. The UREX+3c fuel cycle does not require facilities used by the AFC to reenrich recovered uranium but requires capacity for disposal (potentially GTCC) of the excess recovered uranium containing significant amounts of U^{232} , U^{234} , and U^{236} isotopes. The AFC consumes less natural uranium and thus produces less mine and mill tailings than does a

UREX+3c fuel cycle but the AFC produces more TRU waste requiring a greater disposal capacity. Both fuel cycles make more sustainable use of uranium resources if LWR spent fuel is recycled twice. And, both fuel cycles provide a nearly 2-fold reduction in the amount of high-level radioactive waste compared to spent nuclear fuel assemblies. FRs employed by both fuel cycles would sustainably consume the transuranic elements produced in LWRs. Last, development, demonstration, and deployment of either the UREX+3c fuel cycle or the AFC would establish a foundation for continued and sustainable development and production of nuclear power using extant resources of depleted uranium.

9.8 REFERENCES FOR SECTION 9.0

- (1) D. Clark, A Rough Guide to Ethical Living, London: Penguin. ISBN 9781843537922
- (2) G. D. Del Cul et. al., Analysis of the Reuse of Uranium Recovered from the Reprocessing of Commercial LWR Spent Fuel, Oak Ridge National Laboratory, ORNL/TM-2007/207, January 2009

10.0 OTHER SYSTEM STUDIES

10.1 ALTERNATIVES FOR DISPOSITIONING EXCESS RECOVERED URANIUM

This section discusses alternatives for the disposition of recovered uranium that would be in excess of that contained in the 5:1 U:Pu product of the UREX+3c separations process. The discussion addresses the amounts, characteristics, and issues associated with disposition of this excess recovered uranium for both the UREX+3c fuel cycle and AFC investigated by this study. Also included is discussion of the disposition of depleted recycled uranium following enrichment of recovered uranium in the AFC.

10.1.1 Fast Reactor Fuel

There is no excess recovered uranium from reprocessing of FR spent fuel because FRs consume uranium. This analysis assumes replacement (or makeup) uranium would come from the recovered uranium from reprocessing LWR MOX spent fuel.

10.1.2 Overview of Recovered Uranium Disposition Alternatives

Table 10-1 lists the amounts and isotopic makeup of the recovered uranium from reprocessing for the two fuel cycle cases. Figures 10-1 through 10-4 illustrate the flows of uranium for the two cases. As illustrated in the figures, Table 10-1 describes the recovered uranium for the following two LWR recycle scenarios:

- (1) MOX-1: 1st generation spent MOX is not recycled in LWRs and is subsequently used to produce FR fuel
- (2) MOX-2: 1st generation spent MOX is recycled to produce 2nd generation MOX which, after use in LWRs, is used to produce FR fuel.

For the UREX+3c and the AFC cases, the U:Pu (5:1) product of reprocessing contains only part of the recovered uranium. As shown in the table and illustrated in Figures 10-1 and 10-3 for the UREX+3c case, 0.856 metric tons of recovered uranium remains (or is excess) following reprocessing of 1 MTIHM of UO₂ spent fuel. For the AFC case illustrated in Figures 10-2 and 10-4, 0.685 MT of depleted recycled uranium would remain. However, for the AFC, the depleted recycled uranium is enrichment tails, in which lighter uranium isotopes are substantially depleted.

TABLE 10-1
CHARACTERISTICS OF RECOVERED URANIUM FROM THE UREX+3c FUEL CYCLE
AND THE AFC

Characteristics of RU		Fuel-Cycle Case			
		UREX+3c/ 1 Recycle	AFC/ 1 Recycle	UREX+3c/ 2 Recycles	AFC/ 2 Recycles
RU From Reprocessing Spent UO ₂ Fuel	MTRU	0.856	0.685	0.856	0.685
	U ²³² (%)	2.8E-07	1.0E-08	2.8E-07	1.0E-08
	U ²³⁴ (%)	2.3E-02	4.0E-03	2.3E-02	4.0E-03
	U ²³⁵ (%)	8.3E-01	2.5E-01	8.3E-01	2.5E-01
	U ²³⁶ (%)	7.7E-01	3.7E-01	7.7E-01	3.7E-01
	U ²³⁸ (%)	9.84E+01	9.94E+01	9.84E+01	9.94E+01
RU From Co-Reprocessing 1 st Generation MOX and UO ₂ Spent Fuel	MTRU	N/A	N/A	0.818	0.653
	U ²³² (%)			3.3E-07	2.0E-08
	U ²³⁴ (%)			2.3E-02	5.0E-02
	U ²³⁵ (%)			7.4E-01	2.5E-01
	U ²³⁶ (%)			7.0E-01	4.9E-01
	U ²³⁸ (%)			9.85E+01	9.93E+01
Excess RU From MOX-2 Reprocessing	MTRU	0.108	0.174	0.128	0.170
	U ²³² (%)	5.9E-07	1.5E-06	6.9E-07	2.1E-06
	U ²³⁴ (%)	2.2E-02	5.9E-02	1.3E-02	8.0E-02
	U ²³⁵ (%)	3.1E-01	8.6E-01	2.0E-01	9.4E-01
	U ²³⁶ (%)	3.2E-01	1.8E+00	2.9E-01	2.0E+00
	U ²³⁸ (%)	9.94E+01	9.73E+01	9.95E+01	9.70E+01

RU = Recovered Uranium

TABLE 10-2
ASTM STANDARD C787 MAXIMUM CONCENTRATIONS OF URANIUM NUCLIDES IN
RECOVERED URANIUM (%)

Nuclide	ASTM C787 (Recycle Uranium)	ASTM C996 (Enriched Recycle Uranium)
U ²³²	5.0E-07	5.0E-06
U ²³⁴	4.8E-02	2.0E-01
U ²³⁶	8.4E-01	-

Source: See (Ref. 1 and Ref. 2)

FIGURE 10-1
URANIUM MATERIAL BALANCES FOR THE UREX+3C FUEL CYCLE WITH ONE MOX RECYCLE

FIGURE 10-2
URANIUM MATERIAL BALANCES FOR THE AFC WITH ONE MOX RECYCLE

FIGURE 10-3
URANIUM MATERIAL BALANCES FOR THE UREX+3C FUEL CYCLE WITH TWO MOX RECYCLES

FIGURE 10-4
URANIUM MATERIAL BALANCES FOR THE AFC WITH TWO MOX RECYCLES

Also illustrated in Figures 10-2 and 10-4, the product of enrichment is about 0.171 metric tons of reenriched recovered uranium, which is used along with the 5:1 U:Pu product from reprocessing UO_2 to produce 1st generation LWR MOX fuel.

The excess recovered uranium from the UREX+3c fuel cycle, and the excess recovered uranium plus depleted recycled uranium from the AFC, are set aside for future disposition possibly as Class B, C, or GTCC LLW, although classification as GTCC is unlikely for the depleted recycled uranium.

When LWR spent fuel is recycled a second time and 1 MTIHM of companion amounts of UO_2 and 1st generation MOX spent fuel are co-reprocessed, as described in earlier sections of this report, the amount of excess recovered uranium for the UREX+3c case is 0.818 MTU and the amount of depleted recycled uranium for the AFC is 0.653 MTU. The excess recovered uranium and depleted recycled uranium from second recycling of LWR spent fuel would also be set aside for future disposition as discussed above.

For both the UREX+3c and AFC cases, the analysis assumed excess recovered uranium from reprocessing spent MOX fuel to recover uranium and TRU for production of FR fuel would not be recycled. For the UREX+3c case, low concentrations of U^{235} , and for the AFC, high concentrations of U^{236} , make economic reuse of the recovered uranium unlikely except possibly for future fast breeder reactors or alternative enrichment technologies.

10.1.3 Recovered Uranium from the UREX+3c Fuel Cycle

As shown, recovered uranium from recycling LWR spent fuels in a UREX+3c fuel cycle contains U^{235} at concentrations greater than that of natural uranium, which suggests it has residual energy value. However, even though it is within the specification limits of ASTM Standard C787 (Ref. 1) for reprocessed material (Table 10-2), the U^{236} content of this uranium detracts from its potential for reuse in LWRs principally because enrichment plants may be reluctant to introduce the impurity into their cascades. Thus, dedicated enrichment facilities may be required. As a consequence, the UREX+3c case assumes that the excess recovered uranium would be set aside (converted to a stable form and stored) until a final disposition (possibly as LLW or in CANDU reactor fuel) is determined. (In Section 5.0 of this report on costs, it was assumed that this recovered uranium would be disposed of.)

An alternative disposition for this excess recovered uranium could be to reenrich it in dedicated facilities and blend the enriched product with enriched virgin uranium. Using this approach, the UREX+3c fuel cycle's use of enriched uranium from virgin uranium could be reduced by 7.5 percent (for once recycled LWR fuel) or 9.5 percent (for twice recycled LWR fuel). It would be necessary to slightly increase (0.15 percent) the initial enrichment of the resulting blended uranium fuel to compensate for the neutron absorption by U^{236} from the recovered uranium. Table 10-3 lists the composition of uranium that could be derived from blending enriched virgin uranium and enriched recovered uranium for the UREX+3c cases.

TABLE 10-3
CHARACTERISTICS OF BLENDED ENRICHED VIRGIN URANIUM AND ENRICHED
RECOVERED URANIUM FOR THE UREX+3c CASE

Characteristics of Blend		Fuel-Cycle Scenario	
		UREX+3c/1 Recycle	UREX+3c/2 Recycles
Enriched Virgin U Plus Enriched Recovered U 1 st LWR Fuel Recycle	U^{232} (%)	2.9E-07	2.9E-07
	U^{234} (%)	2.7E-02	2.7E-02
	U^{235} (%)	5.15E+00	5.15E+00
	U^{236} (%)	6.0E-01	6.0E-01
	U^{238} (%)	9.4E+01	9.4E+01
Enriched Virgin U Plus Enriched Recovered U 2 nd LWR Fuel Recycle	U^{232} (%)	N/A	3.6E-07
	U^{234} (%)		2.8E-02
	U^{235} (%)		5.15E+00
	U^{236} (%)		5.4E-01
	U^{238} (%)		9.4E+01

As is the case for the AFC, reenriching excess recovered uranium from the UREX+3c fuel cycle would produce depleted recycled uranium enrichment tails. The amount and radiotoxicity of this depleted recycled uranium would be significantly less than that of the excess recovered uranium. Nonetheless, as discussed above for the depleted recycled uranium from the AFC, because of U^{232} and U^{234} concentrations, (and possibly trace quantities of TRU isotopes) it is likely that these tails would be disposed as Class B or C low-level radioactive waste (Ref. 3). Except for the possibility of use in breeder reactors, the depleted recycled uranium would not have economic value and would be converted to a stable solid form for disposition or long term storage.

Processing MOX spent fuel to recover uranium and TRU for use in FR fuel would also produce excess recovered uranium that would need to be dispositioned. As shown in Table 10-1, this recovered uranium would have low U^{235} enrichments (0.3 percent and 0.2 percent) making its reuse in LWRs uneconomical. A possible exception could be use as makeup uranium for production of LWR MOX fuel (in shielded production facilities) where the presence of U^{232} in relatively high concentrations might not be a detriment. Otherwise, this recovered uranium would be converted to a stable solid form for disposition or long term storage.

10.1.4 Excess Recovered Uranium from the AFC

In contrast to the UREX+3c fuel cycle, and notwithstanding the relatively high concentration of U^{236} , the AFC reenriches the excess recovered uranium for use in making MOX fuel for recycle to LWRs. The depleted recycled uranium from this enrichment activity would have very low economic value except possibly in a future fuel cycle that incorporates breeder reactors. The amount of depleted recycled uranium from the AFC is about 20 percent less than the excess recovered uranium from the UREX+3c fuel cycle and, because the concentrations of U^{232} and U^{234} are much less, the radiotoxicity of the depleted recycled uranium is much less than that of recovered uranium.

The depleted recycled uranium from the AFC described here would be converted to a stable solid form for disposition or long-term storage. As discussed in the preceding section, it is unlikely that the depleted recycled uranium could be disposed as Class A low-level radioactive waste, which is the proposed disposition path for tails from enriching virgin uranium.

As for the UREX+3c case, reprocessing MOX spent fuel in the AFC to recover uranium and TRU elements for FR fuel would produce excess recovered uranium that would need to be dispositioned. Differing from the UREX+3c case, as shown in Table 10-1, this excess recovered uranium from the AFC would have U^{235} enrichment that exceeds that of natural uranium (0.86 percent and 0.94 percent) making its economic reuse a possibility. Because there would be a relatively small amount, this recovered uranium might be reenriched (in a dedicated enrichment facility) then (as discussed above) blended with enriched natural uranium for use in UO_2 fuel for LWRs. Such reuse could reduce the amount of natural uranium consumed by the AFC by a few percent. Table 10-4 lists the characteristics of the reenriched recovered uranium and Table 10-5 lists those characteristics after the reenriched recovered uranium has been blended with virgin

enriched uranium. As shown, the blended product is similar to that described above for the alternative of blending reenriched recovered uranium for the UREX+3c case.

TABLE 10-4
CHARACTERISTICS OF ENRICHED RECOVERED URANIUM FROM FR FUEL
PRODUCTION FOR THE AFC

Characteristics of Blend		Fuel-Cycle Case	
		AFC/1 Recycle	AFC/2 Recycles
Excess Recovered Uranium From MOX Reprocessing	U ²³² (%)	9.0E-06	1.0E-05
	U ²³⁴ (%)	4.0E-01	4.9E-01
	U ²³⁵ (%)	5.10E+00	5.15E+00
	U ²³⁶ (%)	8.2E+00	8.6E+00
	U ²³⁸ (%)	8.6E+01	8.6E+01

TABLE 10-5
CHARACTERISTICS OF BLENDED ENRICHED VIRGIN URANIUM AND RECOVERED
URANIUM FOR THE AFC

Characteristics of Blend		Fuel-Cycle Scenario	
		AFC/1 Recycle	AFC/2 Recycles
Enriched Virgin Uranium Plus Enriched Recovered Uranium From MOX Reprocessing	U ²³² (%)	4.5E-07	5.8E-07
	U ²³⁴ (%)	2.0E-02	2.9E-02
	U ²³⁵ (%)	5.10E+00	5.15E+00
	U ²³⁶ (%)	4.1E-01	5.0E-01
	U ²³⁸ (%)	9.4E+01	9.4E+01

Conversely, the very high concentrations of U²³², U²³⁴, and U²³⁶ call such use into question.

The tails from reenriching the recovered uranium would not have economic value and would be converted to a stable solid form for disposition or long term storage. As discussed above, it is likely that these tails would be disposed as Class B or C low-level radioactive waste.

10.2 EFFECTS OF MULTIPLE RECYCLE OF PLUTONIUM ON LWR SPENT FUEL TRANSPORTATION

The design, fabrication, and use of packaging (shipping casks) for the transport of LWR spent nuclear fuel and the security of shipments of the material are regulated by the NRC (Ref. 4 and 5). The safety of the various modes of transport that could be used to move spent fuel is regulated by the DOT (Ref. 6). The performance standards established by the regulations of

these two agencies specified in 10 CFR 71, 10 CFR 73, and 49 CFR are the same regardless of whether UO_2 or MOX spent fuel is packaged and transported.

However, differences in the isotopic composition and resulting decay characteristics of the UO_2 and MOX spent fuels could result in substantial differences in the capacities and other design characteristics of the spent fuel shipping casks that would be used. Table 10-6 presents a comparison of 5-year cooled UO_2 and MOX spent fuels for those characteristics where the difference would significantly affect the shielding and heat dissipation features of shipping casks. The table compares the characteristics for the UREX+3c fuel cycle and the AFC. Shown for each fuel cycle are characteristics for UO_2 and two generations of MOX spent fuel where (i) spent MOX is assumed to be co-processed with companion UO_2 spent fuel, and (ii) where the 2nd generation MOX is derived from 1st generation MOX that has been segregated from UO_2 spent fuel for reprocessing.

The gamma-ray source per MTIHM is essentially the same for the six types of spent fuel. This result is expected because gamma ray sources mostly derive from decay of fission products, the amounts of which are approximately proportional to the burnup of the spent fuel -- the burnup for each of the six spent fuels is 60,000 (MWD/MTIHM). This suggests that the thickness of the heavy-metal shield of a transportation cask would be about the same for the six types of spent fuel.

However, the difference in the thickness of a casks' neutron shielding could be significant. The thickness of neutron shielding would depend on whether MOX spent fuel assemblies could be placed in interior locations and surrounded by UO_2 spent fuel, which would be possible in a cask that has a capacity of 9 or more fuel assemblies. Nonetheless, to ensure that the external dose rate following an accident would not exceed the regulatory limit of 1 rem, it could be necessary to limit the amount of MOX spent fuel in a cask.

TABLE 10-6
CHARACTERISTICS OF SPENT NUCLEAR FUEL FROM A UREX+3C FUEL CYCLE
AND AN AFC THAT SIGNIFICANTLY AFFECT THE DESIGN OF SPENT FUEL
TRANSPORT CASKS

Fuel Cycle	Type of Spent Fuel	Reprocessing Strategy	Gamma Source (photons/sec per MTIHM)	Neutron Source (neutrons/sec per MTIHM)	Heat Source (watts per MTIHM)
UREX+3c & AFC	UO ₂	N/A	1.85 E+16	1.46 E+09	3.52 E+03
UREX+3c	1 st Generation MOX (MOX-1)	From reprocessed UO ₂	1.84 E+16	1.55 E+10	8.40 E+03
	2 nd Generation MOX (MOX-2)	From co-processed MOX-1 & UO ₂	1.85 E+16	2.10 E+10	1.02 E+04
	2 nd Generation MOX (MOX-2)	From reprocessed MOX-1	1.86 E+16	2.93 E+10	1.29 E+04
AFC	1 st Generation MOX (MOX-1)	From reprocessed UO ₂	1.81 E+16	1.22 E+10	7.76 E+03
	2 nd Generation MOX (MOX-2)	From co-processed MOX-1 & UO ₂	1.81 E+16	1.78 E+10	1.01 E+04
	2 nd Generation MOX (MOX-2)	From reprocessed MOX-1	1.82 E+16	2.76 E+10	1.49 E+04

When analyzed for the hypothetical accidents specified by the NRC regulations, safety analyses have historically made the conservative assumption that the neutron shielding would be lost. For casks that are designed to contain high burnup UO₂ spent fuel, these analyses often calculate after-accident neutron dose rates external to a cask as high as several hundred millirem per hour (Ref. 7). This conservative approach to analysis of cask shielding would probably be followed for the design and certification of casks to transport spent first-generation MOX spent fuel. As a consequence, because the neutron source strength is ten-fold greater, casks used to ship MOX spent fuel are likely to have capacities that are significantly less than those of present-day spent fuel casks designed for UO₂ spent fuel.

As shown in the table, this challenge for the design of transportation casks for MOX spent fuel would be about 40 percent greater for 2nd generation MOX spent fuel (compared to 1st generation MOX spent fuel) and about 100 percent greater if the 2nd generation MOX fuel is produced from plutonium recovered only from 1st generation MOX spent fuel.

Efficient removal of heat from the decay of radioactive isotopes in spent fuel is another requirement for the design of a spent fuel cask. The design limit of present-day, state-of-the-art casks for transporting LWR spent fuels is about 25 kW. Using this limit, a cask for transporting the 5-year cooled UO₂ spent fuel described above would contain no more than about 15 PWR fuel assemblies. For 1st generation MOX spent fuel, the limit would be about 7 assemblies for the AFC and 6 assemblies for the UREX+3c fuel cycle. For 2nd generation MOX spent fuel the limits would be about 5 assemblies for both fuel cycles. If the 2nd generation MOX was derived from reprocessed MOX spent fuel, the limits would be about 4 fuel assemblies.

The reduced capacity of the shipping casks might be somewhat offset by designing casks to transport MOX and UO₂ spent fuel assemblies together. But, transportation of significant amounts of MOX LWR spent fuel for either of the advanced nuclear fuel cycles will require a new generation of spent fuel casks that will have less capacity than those in current use.

10.3 ESTIMATION OF THE ENERGY VALUE OF ACTINIDE ISOTOPES IN FR TARGETS

The analysis presented in Section 3.0 assumes the minor actinide elements neptunium, americium, and curium would be combined with plutonium and uranium recovered from spent LWR MOX fuel to produce nuclear fuel for FRs. Because all of the minor actinide isotopes have high values of η at the energy of neutrons in an advanced FR (Ref. 8), they would contribute to the amount of energy produced in the fuel by nuclear fission and to sustaining the nuclear chain reaction in the reactor. The amount of energy produced by fissioning of minor actinides would be approximately the same as that from fissioning plutonium-239, about 200 Mev per fission, and proportional to the amounts of each that was consumed (or burned) in the reactor.

However, the fractional amounts of minor actinides that accumulate in LWR spent nuclear fuel, which would be the major source of fissionable material for an UREX+3c fuel cycle or AFC, could exceed technical limits for FR fuel. Currently this limit is thought to be about 3 percent of the uranium and TRU content. The analysis presented in Section 3.0 estimates that the fraction of minor actinides in the initial cores and early recycles of FR fuel could be as high as 8.3 percent if all of the minor actinides from the LWR fuel were included. And, the fractional

amount of minor actinides is highest if the LWR fuel is recycled more than once before being used to produce FR fuel.

The analysis also showed that the fraction of the minor actinides would decrease slowly with successive recycles of fuel in a FR and, after about the fourth recycle, would reach an equilibrium level of about 3 percent. If the technical limit of 3 percent is not increased, in order to consume the minor actinides that would be initially available from LWR fuel before a fleet of FRs was in operation, an alternative to incorporation in the FR fuel could be necessary. A possible approach that has been suggested would be to produce minor actinide targets that could possibly be placed in blankets surrounding the fuel-assembly cores of the FRs. The target assemblies would produce power (by fissioning minor actinide isotopes) as well as provide for consumption of the minor actinides. An engineering and economic analysis would be needed to determine whether to incorporate some (say a 3 percent fraction of the uranium and TRU content) of the minor actinides in the FR fuel and the remainder in target assemblies, or to only use plutonium and uranium in the FR fuel and locate all of the minor actinides in targets. Factors such as heat generation, radiation from the minor actinides, rate of consumption, as well as reactor performance considerations would influence the outcome of this analysis.

10.4 REFERENCES FOR SECTION 10.0

- (1) ASTM C787-06, Standard Specification for Uranium Hexafluoride for Enrichment, ASTM International, West Conshohocken, PA, 2006, www.astm.org.
- (2) ASTM C996 - 04e1, Standard Specification for Uranium Hexafluoride Enriched to Less Than 5 % ²³⁵U, ASTM International, West Conshohocken, PA, 2004, www.astm.org.
- (3) G. D. Del Cul et al, Analysis of the Reuse of Uranium Recovered from the Reprocessing of Commercial LWR Spent Fuel, Oak Ridge National Laboratory, ORNL/TM-2007/207, January 2009
- (4) 10 CFR 71, Packaging and Transportation of Radioactive Material
- (5) 10 CFR 73, Physical Protection of Plants and Materials
- (6) 49 CFR, Transportation

- (7) Holtec International, Inc., Safety Analysis Report on the HI-STAR 180 Package, Rev. 2, Supp. A., March 9, 2009
- (8) H. Nifenecker, S. David, J.M. Loiseaux, and A. Giorno, Hybrid Nuclear Reactors, IN2P3-UJF, France, December 19, 1998

APPENDIX A

SPENT NUCLEAR FUEL CHARACTERISTICS

APPENDIX A

SPENT NUCLEAR FUEL CHARACTERISTICS

The purpose of this Appendix is to summarize the quantities and principal characteristics of the spent UO_2 fuel that has been and will be discharged by the 14 shutdown and 104 operating LWRs that exist in the U.S. as of 2009. Spent UO_2 fuel assemblies are the predominant candidate feed inputs to the front end of the potential future chemical processing facilities, regardless of the extent of MOX recycling. The fuel assembly designs are characterized by reference to the different reactor designs by the four principal designers of the of the original US LWRs, Westinghouse, General Electric, Combustion Engineering, and Babcock & Wilcox. Although the latter two of these four have been absorbed into other organizations, the basic fuel designs used in all of the 104 currently operating reactors will necessarily continue in use because of the dimensional and nuclear constraints imposed by each particular original reactor core design.

A basic premise of the discussion in this section is that all 104 operating US reactors will receive 20-year operating license extensions, for a total of 60 years of operations, and will continue to operate at high capacity factors, in the range of 90%, until the end of their licensed operating periods. The cumulative discharges by 2055, when the last of the 104 reactors completes its 60th year of operation, are estimated to total about 450,000 fuel assemblies, with an original uranium content of 130,000 metric tons of uranium. It is also noted that there are active utility projects potentially leading to a significant number of new reactor startups. However, the prospective spent fuel discharges from these new reactors have not been included in this work, simply because there is ample variety and scope for optimizing the potential recovery of spent fuel residual value and reducing spent fuel disposal costs, within the known historic discharges and relatively predictable future discharges of the currently-operating reactors.

The various types of individual fuel assemblies make up the input feed stream to the processing/recycle system. Once the assemblies have been disassembled and dissolved, thereby losing their identity, their quantities and characteristics are thereafter quantified by their original MTU content, their cooling age since discharge, and their discharge burnups. The latter characteristics can then be used to determine their isotopic content and related radiation and decay heat characteristics. The remainder of this section summarizes key characteristics of the

total inventory of discharged spent UO_2 fuel as a function of time, without reflecting removals of any part of that inventory for processing and recycle.

Figure A-1 summarizes the total (cumulative) assemblies discharged by US commercial power reactors as a function of time, as well as the distribution of assemblies among the principal fuel designs. The two primary boiling water reactor (BWR) fuel designs are at the top of the distribution. The pressurized water reactor (PWR) fuel types are shown below the two BWR types, except for the very bottom small category which covers a mix of small, early BWR and PWR fuel types. A more detailed discussion of the principal fuel designs is provided below. It should be noted that there are somewhat more BWR assemblies than PWR assemblies. However, because the BWR assemblies typically contain only about 40% of the uranium in a PWR assembly, there are more PWR MTUs than BWR MTUs in the total inventory, as will be shown shortly. The historic data goes through 2002, when the most recent Energy Information Administration's RW-859 survey of US reactors was completed. At that point, about 150,000 spent fuel assemblies had been discharged. Also note the projected total of about 450, 000 spent fuel assemblies in 2055, when the last of the 104 current reactors reaches the end of its 60-year operating license. Projections after 2002 were made by the DOE Office of Civilian Radioactive Waste Management's management and operating contractor.

The following summarizes the connection between the various operating and shut-down reactor designs of the four original reactor system vendors, and the fuel design type designations that will be used in this section for characterizing historic and projected future discharges of civilian spent fuel.

Reactor Supplier	Reactor Type	Number of Reactors		Operating Reactor Fuel Types	Shutdown Reactor Fuel Types
		Operating	Shutdown		
Westinghouse	PWR	48	6	W14,15,17,17ST	W17, P
General Electric	BWR	35	4	G2,G4	G2, B
Combustion Engineering	PWR	14	1	C14, 16, 16sp	C14
Babcock & Wilcox	PWR	7	2	B15	B15, P
Other (Allis Chal)	BWR		1		B
Total		104	14		

The principal physical difference between the fuel assemblies of PWR and BWR is in the width of the square assembly cross-section, somewhat less than 9 inches square for PWRs (to accommodate a 14x14 to 17x17 array of fuel rods) as compared to somewhat less than 6 inches square for BWRs (to accommodate a 7x7 to 10x10 array of fuel rods). Each fuel rod holds a stack of uranium dioxide (UO₂) pellets about 12 feet long, which with end caps is typically 13 to 14 feet long. The whole fuel assembly, with its array of fuel rods held together with fuel rod spacer grids and end fittings is typically 14 to 15 feet long. BWR assemblies weigh on the order of 600 pounds, and PWR assemblies weigh on the order of 1400 pounds. The uranium content of the typical assemblies with 12-foot UO₂ pellet stacks is 0.17 to 0.18 MTU for BWR assemblies and 0.42 to 0.48 MTU for PWR assemblies. The PWR fuel design identifier for Babcock & Wilcox is B15, designating B&W's fuel design which has a 15x15 fuel rod array. The W17ST design is Westinghouse's 17x17 rod array design for the two South Texas reactors, which have a 14-foot stack of UO₂ pellets, rather than the 12-foot pellet stack in the W14, W15 and W17 designs for other Westinghouse reactors and most other PWRs and BWRs. Similarly, Combustion Engineering's C16sp design is somewhat longer than the older C16 and C14 designs. For the General Electric BWR designs, the G2 and G4 assembly identifiers are associated with an earlier and a more recent GE core design, respectively, with the newer G4 fuel assembly being 4 inches longer than the older G2 fuel assembly. The shutdown reactors include some earlier reactor designs which are designated as B and P types (for early BWR and PWR designs) in order to categorize a variety of considerably shorter fuel rod and assembly designs. Within each of the design identifiers cited above, there can be several sub-variants. One of the more important recent sub-variants is the growing use of natural uranium fuel pellets, rather than enriched uranium pellets in the top and bottom 6 to 12 inches of the fuel rods. This sub-variant can be important to the chemical processing of spent fuel because of the differences in the discharged average composition of the recovered uranium and plutonium that it causes.

Figure A-2 shows MTU content of the discharged UO₂ for both BWRs and PWRs, as a function of time. As noted earlier, once the original assemblies have been disassembled and dissolved, the principal measure of quantity becomes total MTUs, or BWR MTUs and PWR MTUs, when it is necessary to separately determine characteristics of BWR and PWR process streams. Note that although there are more BWR assemblies than PWR assemblies (Figure A-1), there are more PWR MTU discharges than BWR MTU discharges. There were 47,000 actual

MTU discharges reported at the end of 2002, with a projected total of 130,000 MTU discharges from the shutdown and current operating reactors by 2055, of which 45,000 MTU are from BWRs and 85,000 MTU are from PWRs.

Figure A-3 shows the average BWR and PWR burnups in GWD/MTU as a function of discharge year. This data plus age at the time of processing or handling is needed to determine additional characteristics of a process stream, such as isotopic content, radioactivity or decay heat levels. Note the steady increase in burnups up to the 55 GWD/MTU range in the 2030 to 2040 time frame. The subsequent decrease is due to the growing number of reactors that begin shutting down at the end of their assumed 60-year license limit, at which time they each discharge a full core of fuel with an average burnup of about 2/3 of their normal burnup discharge level, thereby reducing the overall average burnup, in part because there are progressively less and less reactors still operating.

Figure A-4 shows the cooling age distribution of discharged spent fuel inventory, by year. For any fixed year, the age increases approximately linearly as you go from the upper limit to the lower limit within each age band. This age figure presents a very regular pattern because cooling age is, by definition, linear with time.

Figure A-5 shows the total plutonium isotopic content of the total UO_2 spent fuel inventory, by year, including the effects of Pu^{241} and Pu^{238} decay, assuming no removals of UO_2 fuel from inventory. These data quantify the large amount, and the relatively high quality of plutonium (61-68% fissile) contained in discharged UO_2 spent fuel.

Figure A-6 shows the distribution of decay heat from the total inventory of discharged UO_2 spent fuel that has more than five years of cooling, by year. For any fixed year, the decay heat decreases approximately linearly as you go from the upper limit to the lower limit within each decay heat band. These data show that if the reprocessing plant intends to focus on utilizing 5-year-cooled fuel (that is, along the uppermost part of the uppermost decay heat band), it will have to use very-high-heat transport casks capable of handling 4 to 7 kW/MTU fuel, in order to deliver the fuel to the reprocessing plant. These data can also be used to estimate the decay heat of discharged fuel and the related repository underground usage that would have been required, as compared to the underground disposal area needed if that same fuel were reprocessed, with only a much-reduced amount requiring ultimate disposal.

FIGURE A-1
TOTALS AND DISTRIBUTION OF ASSEMBLY TYPES AND DESIGNS BY YEAR

FIGURE A-2
TOTAL HISTORIC AND PROJECTED SNF DISCHARGES BY FUEL TYPE
(MTU-THOUSANDS)

FIGURE A-3
BWR AND PWR AVERAGE DISCHARGE BURNUPS BY YEAR OF DISCHARGE

FIGURE A-4
AGE DISTRIBUTION OF DISCHARGED SNF INVENTORY BY YEAR

FIGURE A-5
TOTAL PLUTONIUM ISOTOPE INVENTORIES IN SNF DISCHARGES BY YEAR INCLUDING Pu^{241} & Pu^{238} DECAY
(WITHOUT RECYCLE)

FIGURE A-6
DECAY HEAT DISTRIBUTION OF DISCHARGED SNF INVENTORIES OLDER THAN FIVE YEARS
(KW/MTU)

APPENDIX B

**A METHOD FOR CALCULATING ABUNDANCES OF
U-234, U-235, AND U-236 IN THE ENRICHED PRODUCT OF
A CENTRIFUGE CASCADE**

Summary:

This paper presents, and provides validation of, a method for calculating the relative abundances of uranium isotopes U-234, U-235, and U-236 in the uranium product of a centrifuge cascade. The method was developed for its JAI Advanced Fuel Cycle Initiative project to allow calculation of the distribution of uranium isotopes in the enriched product when recycle uranium is used as the feed material to a cascade.

The analysis presented in the following sections is based on principles described by Benedict (1981) of the functioning of a gas enrichment centrifuge to enrich the U-235 isotope in uranium. The analysis extends the principles to allow calculation of the effect of a centrifuge for the U-234 and U-236 isotopes, which are prominent in recycle uranium discharged from nuclear power reactors.

The analysis then uses the results for a single centrifuge to develop a general method for determining the relative abundances of the U-234, U-235, U-236, and U-238 isotopes that would be in an enriched product when the feed to a cascade of centrifuges is recycled uranium. Also presented are comparisons of the results calculated using the method developed here to results presented by others including, for example, the IAEA.

Analysis:*Enrichment Factors for a Single Centrifuge*

Eq. 14.158 (Benedict 1981) presents the ratio of the pressure, or density, of a gas between an interior radius r and the outer wall of a gas centrifuge.

$$\frac{p(r)}{p(a)} = \frac{\rho(r)}{\rho(a)} = \exp \left[- \left(\frac{mv_a^2}{2RT} \right) \left(1 - \frac{r^2}{a^2} \right) \right] \quad (\text{Eq. 1})$$

Where :

- $p(r)$ pressure of the gas at radial position r measured from the centrifuge's centerline
- $p(a)$ pressure of the gas at $r = a$, the centrifuge's cylindrical interior wall
- $\rho(r)$ density of the gas at radial position r measured from the centrifuge's centerline
- $\rho(a)$ density of the gas at the centrifuge's cylindrical interior wall
- m is the molecular weight of the gas
- v_a is the velocity of the cylindrical interior wall of the centrifuge
- R is the universal gas constant
- T is the temperature of the gas

This equation may also be used to determine the ratios of the partial pressure for each of the components of a gas comprised of more than one species.

$$\frac{p_i(r)}{p_i(a)} = \frac{\rho_i(r)}{\rho_i(a)} = \exp \left[- \left(\frac{m_i v_a^2}{2RT} \right) \left(1 - \frac{r^2}{a^2} \right) \right] \quad (\text{Eq. 2})$$

Where:

- i designates the i^{th} component of the gas

The total pressure of the gas at each position r including at the centrifuge wall (a) is the sum of the partial pressures of the constituent species of the gas at these locations.

$$P(r) = p_1(r) + p_2(r) + p_3(r) + p_4(r) + \dots + p_n(r) = \sum p_i(r) \quad (\text{Eq. 3})$$

Where:

- $P(r)$ is the total pressure of gas at position r
- $p_n(r)$ is the partial pressure of the n^{th} gas species at position r
- n is the number of species in the mixture of gasses inside the centrifuge

Also:

$$p_i(r) = p_i(a) \times \exp \left[- \left(m_i v_a^2 / 2RT \right) \left(1 - r^2 / a^2 \right) \right] \quad (\text{Eq. 4})$$

And:

$$p_1(a) = P(a) - (p_2(a) + p_3(a) + p_4(a) + \dots p_n(a)) = x_1(a) \times P(a) \quad (\text{Eq. 5})$$

$$p_2(a) = P(a) - (p_1(a) + p_3(a) + p_4(a) + \dots p_n(a)) = x_2(a) \times P(a) \quad (\text{Eq. 6})$$

$$p_3(a) = P(a) - (p_1(a) + p_2(a) + p_4(a) + \dots p_n(a)) = x_3(a) \times P(a) \quad (\text{Eq. 7})$$

$$p_4(a) = P(a) - (p_1(a) + p_2(a) + p_3(a) + p_5(a) + \dots p_n(a)) = x_4(a) \times P(a) \quad (\text{Eq. 8})$$

etc.

Where:

$x_1(a), x_2(a), x_3(a), x_4(a) \dots x_n(a)$ are the ratios of the partial pressures of the gas's constituent species to the total gas pressure at the centrifuge wall

$$\sum x_i(r) = 1 \text{ for all values of } r \text{ between and including } r = 0 \text{ and } r = a \quad (\text{Eq. 9})$$

Thus:

$$p_i(r) = x_i(a) \times P(a) \times \exp \left[- \left(m_i v_a^2 / 2RT \right) \left(1 - r^2 / a^2 \right) \right] = x_i(r) \times P(r) \quad (\text{Eq. 10})$$

And:

$$\left(x_i(r) \times P(r) \right) / \left(x_i(a) \times P(a) \right) = \exp \left[- \left(m_i v_a^2 / 2RT \right) \left(1 - r^2 / a^2 \right) \right] \quad (\text{Eq. 11})$$

Then for any two gas constituents i and j :

$$\left(x_i(a) \times x_j(r) \right) / \left(x_i(r) \times x_j(a) \right) = \exp \left[\left((m_i - m_j) v_a^2 / 2RT \right) \left(1 - r^2 / a^2 \right) \right] \quad (\text{Eq. 12})$$

Using the terminology of Benedict, Pigford, and Levi (Benedict 1981)

$$\alpha_{ij}(r, a) = \left(x_i(a) \times x_j(r) \right) / \left(x_i(r) \times x_j(a) \right) \quad (\text{Eq. 13})$$

Values of $\alpha_{ij}(r, a)$ can be calculated for any r , a , and molecular weights m_i and m_j . Table 1 below presents values of $\alpha_{ij}(r, a)$ for a representative centrifuge for uranium hexafluoride where the uranium is comprised of U^{234} , U^{235} , U^{236} , and U^{238} isotopes. The table also provides a comparison of values of $\alpha_{ij}(r, a)$, where the respective i^{th} and j^{th} isotopes are U^{238} and U^{235} , to values reported by Benedict, Pigford, and Levi.

Table 1.

Values of $\exp \left[\left((m_i - m_j) v_a^2 / 2RT \right) \left(1 - r^2 / a^2 \right) \right]$ for an Uranium Enrichment Centrifuge

r	r/a	m_i	m_j	$\exp \left[\left((m_i - m_j) v_a^2 / 2RT \right) \left(1 - r^2 / a^2 \right) \right]$	Benedict, Pigford, and Levi*
0.45	0.9	352 (238)	350 (236)	1.012262	
		352 (238)	349 (235)	1.018449	1.018
		352 (238)	348 (234)	1.024675	
		350 (236)	349 (235)	1.006112	
		350 (236)	348 (234)	1.012262	
		349 (235)	348 (234)	1.006112	
0.40	0.8	352 (238)	350 (236)	1.023361	
		352 (238)	349 (235)	1.035245	1.035
		352 (238)	348 (234)	1.047267	
		350 (236)	349 (235)	1.011613	
		350 (236)	348 (234)	1.023361	
		349 (235)	348 (234)	1.011613	
R = 8.3145E+3 MPa-Liter / K°-mole					
T = 300 °K					
$a = 0.5$ meter					
$v_a = 400$ m / sec					

* See Table 14.13, p. 857, Benedict 1981.

Next:

$$x_i(r) = \left(x_i(a) / x_j(a) \right) \times \left(1 / \alpha_{ij}(r, a) \right) \times x_j(r) \quad (\text{Eq. 14})$$

$$x_i(a) = \left(x_i(r) / x_j(r) \right) \times \alpha_{ij}(r, a) \times x_j(a) \quad (\text{Eq. 15})$$

$$\sum_i x_i(a) = \sum_i x_i(r) = 1 \quad (\text{Eq. 9})$$

If values for the terms $x_i(r)/x_j(r)$ are known for a gas at a radial location r in a centrifuge, then the values of $x_i(a)$ can be calculated. Conversely, if the values of $x_i(a)/x_j(a)$ are known for the gas at the centrifuge wall, then the values of $x_i(r)$ can be calculated.

$$x_i(a) = 1 / \left[1 + \sum_{i \neq j} \left\{ \left(x_j(r) / x_i(r) \right) \times \left(1 / \alpha_{ij}(r, a) \right) \right\} \right] \quad (\text{Eq. 16})$$

$$x_i(r) = 1 / \left[1 + \sum_{i \neq j} \left(x_j(a) / x_i(a) \right) \times \alpha_{ij}(r, a) \right] \quad (\text{Eq. 17})$$

Table 2 and Table 3 present the values of $x_i(a)$ and $x_i(0.8a)$ where the values of $x_i(r)/x_j(r)$ for the intermediate radial location $r/a = 0.9$ in a centrifuge are assumed to equal the values of x_i/x_j of uranium gas constituents bled into the centrifuge. Table 2 presents data for recycle uranium having an initial enrichment of 4.5 wt% U^{235} and burnup of 48 GWd/t HM. Table 3 presents like data for recycle uranium having an initial enrichment of 4.1 wt% U^{235} and burnup of 60 GWd/t HM. The tables also present the resulting enrichment factors for three uranium isotopes. Both sets of data assume that the temperature of UF_6 gas in the centrifuge is 300 °K.

The procedure used to calculate the values in the tables is as follows:

1. Set the values of $x_i(0.9a)$ equal to the fractional abundance of each isotope in the uranium feed to the centrifuge.
2. Calculate values of $x_i(a)$ using (i) the above equation (Eq. 16), (ii) values for $x_i(0.9a)$ from Step 1, and (iii) values of $\alpha_{ij}(0.9a, a)$ presented in Table 1 of this calculation.
3. Calculate values for $x_i(0.8a)$ using (i) the above equation (Eq. 17), (ii) values of $x_i(a)$ calculated in Step 2, and (iii) values of $\alpha_{ij}(0.8a, a)$ presented in Table 1 of this calculation.
4. Calculate enrichment factors for each isotope by dividing the values of $x_i(0.8a)$ by the values of $x_i(0.9a)$, the second set of values being the fractional abundances for each isotope in the uranium feed to the centrifuge.

This procedure incorporates two key assumptions: (i) the relative abundances of isotopes at a single radial location in the centrifuge (e.g., 90% of the radius of the centrifuge's inside wall) correspond to the relative abundances of the uranium feed and (ii) the product stream is represented by the abundances of isotopes in gas at the radial location that is 80% of the wall radius. Table 4 shows the sensitivity of the enrichment factors to the first of these assumptions. The results presented in the table show that calculated values of enrichment factors are not

sensitive to variations in the relative abundances of isotopes in material fed to the centrifuge. The relative insensitivity is valid provided the range of enrichments is limited. This is the case for low enriched uranium used in light-water-reactor nuclear fuel. Therefore, one can expect the first assumption to lead only to small errors in the calculated values for a centrifuge's enrichment factors.

Table 2. Abundance of Uranium Isotopes at Three Radial Locations in a Gas Centrifuge and Estimated Enrichment Factors for Recycle U (4.5%, 48 GWd/t HM)

Uranium Isotope	Fraction in Feed* (wt%)	Fraction at $r = 0.9a$ (wt%)	Fraction at $r = a$ (wt%)	Fraction at $r = 0.8a$ (wt%)	Enrichment Factor f
U^{238}	0.982558	0.982558	0.982833	0.9823082	0.999746
U^{236}	0.00622	0.00622	0.006146	0.006286263	1.010707
U^{235}	0.010988	0.010988	0.010792	0.011166809	1.016233
U^{234}	0.000234	0.000234	0.000228	0.000238728	1.021789

* Based on values presented in Table 7, p. 16, IAEA 2007 normalized to 1 MT U.

Table 3. Abundance of Uranium Isotopes at Three Radial Locations in a Gas Centrifuge and Estimated Enrichment Factors for Recycle U (4.1%, 60 GWd/t HM)

Uranium Isotope	Fraction in Feed* (wt%)	Fraction at $r = 0.9a$ (wt%)	Fraction at $r = a$ (wt%)	Fraction at $r = 0.8a$ (wt%)	Enrichment Factor f
U^{238}	0.988348743	0.988348743	0.988522	0.988191481	0.999841
U^{236}	0.006148763	0.006148763	0.006075	0.006215191	1.010803
U^{235}	0.005326041	0.005326041	0.00523	0.005413014	1.01633
U^{234}	0.000176452	0.000176452	0.000172	0.000180314	1.021886

* Based on values presented in Table 7, p. 16, IAEA 2007 normalized to 1 MT U.

Table 4. Comparison of Calculated Values for Enrichment Factors for Different Isotopic Compositions of Uranium at a Radial Location Equal to $0.9a$ in a UF_6 Gas Centrifuge

Uranium Isotope	Enrichment Factor for RU (4.5%, 48 GWd/t HM) (Table 2)	Enrichment Factor for RU (4.1%, 60 GWd/t HM) (Table 3)	Enrichment Factor Difference Absolute / %	Feed Difference Absolute / %
U^{238}	0.999746	0.999841	9.5E-05 / 0.0095	0.005791 / 0.5894
U^{236}	1.010707	1.010803	9.6E-05 / 0.0095	-7.1E-05 / -1.145
U^{235}	1.016233	1.01633	9.7E-05 / 0.0095	-0.00566 / -51.529
U^{234}	1.021789	1.021886	9.7E-05 / 0.0095	-5.8E-05 / -24.593

If the isotopic abundances in the product stream correspond to gas at a different radial location than 80% of the wall radius, the enrichment factors will be either higher (for a radius less than 80% of the wall radius) or lower (for a radius that is greater than 80%). Table 5 shows the enrichment factors for the case presented in Table 2 but where the isotopic abundances in the product is represented by gas at a radial location 70% of the wall radius.

Table 6 shows the effect on the relative abundances of isotopes in the uranium product of two enrichment cascades where one product stream is assumed to be represented by gas at 80% of the centrifuge's wall radius and the other assumes that the product stream is represented by gas at 70% of the wall radius. The analysis assumes that the first cascade uses centrifuges that have the enrichment factors presented in Table 2 above; the enrichment factors used for the alternative cascade are those presented in Table 5. Both assume the enrichment of uranium tails is 0.2 wt% and the average gas temperature in the centrifuge is 300 °K. The results are shown to be insensitive to the assumption regarding the radius at which the gas in the centrifuge represents the product gas.

Table 5. Abundance of Uranium Isotopes at Three Radial Locations in a Gas Centrifuge and Estimated Enrichment Factors for Recycle U (4.5%, 48 GWd/t HM)

Uranium Isotope	Fraction in Feed* (wt%)	Fraction at $r = 0.9a$ (wt%)	Fraction at $r = a$ (wt%)	Fraction at $r = 0.7a$ (wt%)	Enrichment Factor f
U ²³⁸	0.982558	0.982558	0.982833	0.982084264	0.9995179
U ²³⁶	0.00622	0.00622	0.006146	0.006345932	1.0202464
U ²³⁵	0.010988	0.010988	0.010792	0.011326115	1.0307713
U ²³⁴	0.000234	0.000234	0.000228	0.000243689	1.0414047

* Based on values presented in Table 7, p. 16, IAEA 2007 normalized to 1 MT U.

Table 6. Comparison of Relative Abundances of Isotopes from Enrichment Cascades Where U Product is Represented by Gas at 80% and at 70% of the Centrifuge Wall Radius

Uranium Isotope	Fraction in Feed to Cascade (wt%)	Stage Enrichment Factor f $r = 80\%$	Fraction in Cascade Product $r = 80\%$	Stage Enrichment Factor f $r = 70\%$	Fraction in Cascade Product $r = 70\%$	Comparison of Product Abundances % Difference
U ²³⁸	0.982558	0.999746	0.959373	0.9995179	0.959151	- 0.023
U ²³⁶	0.00622	1.010707	0.02353636	1.0202464	0.02366102	0.53
U ²³⁵	0.010988	1.016233	0.04992157	1.0307713	0.05000743	0.17
U ²³⁴	0.000234	1.021789	0.00115891	1.0414047	0.00116242	0.30
No. of Cascade Stages		295		157		

U tails – 0.2 wt% U²³⁵

Enrichment of Uranium Isotopes in a Centrifuge Cascade

The following procedure may used to determine the abundances of each of a multiple of uranium isotopes at each stage in a cascade of enrichment machines.

1. Identify the abundances of isotopes in the feed to the enrichment cascade and the total mass of uranium feed (e.g., 1,000,000 grams). (NOTE: Assumes that the enrichment cascade is in steady-state continuous operation with a uniform, continuous feed and withdrawal of product and tails.)

2. Follow the procedure described in the preceding section to determine enrichment factors for each isotope. This will require selecting an average temperature for UF₆ gas in the cascade. Wood (1996) uses a temperature of 340 °K (Table 1, p. 1204).
3. Establish a U²³⁵ enrichment for tails that will be discarded from the cascade.
4. Calculate the number of enrichment stages between the feed stage and the tails stage.

The average enrichment of the U²³⁵ isotope in the first stage of the cascade is:

$$e_1^{235} = f_{235} \times e_{tails}^{235} \quad (Eq. 18)$$

Where:

e_1^{235} is the U²³⁵ average enrichment in the tails stage of the cascade
 f_{235} is the single stage enrichment factor for U²³⁵
 e_{tails}^{235} is the U²³⁵ enrichment in the cascade's tails

Then:

$$e_{feed}^{235} = (f_{235})^N \times e_{tails}^{235} \quad (Eq. 19)$$

$$N = \left[\ln \left(e_{feed}^{235} / e_{tails}^{235} \right) \right] \times \left[1 / \ln (f_{235}) \right] \quad (Eq. 20)$$

Where:

e_{feed}^{235} is the U²³⁵ enrichment of the uranium feed to the cascade
 f_{235} is the single-stage enrichment factor for U²³⁵
 e_{tails}^{235} is the U²³⁵ enrichment of uranium tails that will be discharged from the cascade
 N is the number of enrichment stages in the cascade between the tails stage and the feed stage plus 1 (the tails stage).

5. Calculate the net quantity of uranium that passes from one stage in the cascade to the next higher stage.

The mass balance for total uranium in the n^{th} and $n+1^{st}$ stages of a cascade is:

$$U_{n+1} = U_n - U_{n,tails} \quad (Eq. 21)$$

Where:

U_n is quantity of uranium contained in the n^{th} stage of the cascade
 U_{n+1} is the net quantity of uranium passed to the $n + 1^{st}$ stage in the cascade

$U_{n_{tails}}$ is the net quantity of uranium from the n^{th} stage in the cascade discharged in the cascade's tails stream

The mass balance for the U^{235} isotope in the n^{th} and $n+1^{st}$ stages of a cascade is:

$$U_{n+1}^{235} = e_{n+1}^{235} \times U_{n+1} = e_n^{235} \times U_n - e_{tails}^{235} \times U_{n_{tails}} \quad (Eq. 22)$$

Where :

e_n^{235} and e_{n+1}^{235} are the respective levels of enrichment for the U^{235} isotope in the n^{th} and $n+1^{st}$ stages of the cascade

Thus:

$$U_{n_{tails}} = U_n \times \left(\left(e_n^{235} \times f_{235} \right) - e_n^{235} \right) / \left(\left(e_n^{235} \times f_{235} \right) - e_{tails}^{235} \right) \quad (Eq. 23)$$

$$U_{n+1} = U_n \times \left(1 - \left(\left(e_n^{235} \times f_{235} \right) - e_n^{235} \right) / \left(\left(e_n^{235} \times f_{235} \right) - e_{tails}^{235} \right) \right) \quad (Eq. 24)$$

6. Calculate the net quantity of each uranium isotope that passes from one stage in the cascade to the next higher stage.

$$U_{n+1}^{235} = e_n^{235} \times U_n - e_{tails}^{235} \times U_{n_{tails}} \quad (Eq. 22)$$

The quantities of the other isotopes that pass to the next higher stage are determined in three steps:

- (1) Determine the amount of uranium tails for the stage using *equation 23* above.
- (2) Determine the amounts of each isotope contained in the stage's tails.
- (3) Perform a mass balance by subtracting the quantities discharged in tails from the amounts available in the stage to calculate the net quantities passed to the next higher cascade stage.

The enrichment of each isotope contained in cascade tails that derive from a stage is determined as follows:

$$e_{i_{feed}} = f_i^N \times e_{i_{tails}} \quad (Eq. 25)$$

Where:

$e_{i_{feed}}$ is the enrichment of the i^{th} uranium isotope in uranium fed to the cascade
 f_i is the single-stage enrichment factor for the i^{th} uranium isotope

$e_{i_{tails}}$ is the enrichment of the i^{th} uranium isotope in the cascade's tails
 N is the number of cascade stages between the feed stage and the tails stage plus 1

Thus:

$$e_{i_{tails}} = e_{i_{feed}} / f_i^N \quad (\text{Eq. 26})$$

The amount of each isotope that is passed from one stage in the enrichment cascade to the next higher stage is therefore:

$$U_{i_{n+1}} = e_{i_n} \times U_n - e_{i_{tails}} \times U_{n_{tails}} \quad (\text{Eq. 27})$$

Where:

e_{i_n} is the enrichment of the i^{th} uranium isotope in the n^{th} cascade stage

7. Calculate the enrichment of each isotope in the $n+1$ cascade stage as follows:

$$e_{i_{n+1}} = U_{i_{n+1}} / U_n \quad (\text{Eq. 28})$$

Analysis Results

Tables 7, 8, 9 and 10 present the results of the above analysis for cases that use initial uranium enrichments and cascade parameters listed by others. The tables also provide comparisons of the results to the corresponding results reported by the other analysts and observers. All cases analyzed assume that centrifuges are 1 meter in diameter, operate at 340 °K, have a wall velocity of 400 meters per second, and that product is collected at 80% of the centrifuge wall's inside radius. Results are presented for U^{238} , U^{236} , U^{235} , and U^{234} isotopes. Calculation spreadsheets for the cases are presented for the cases in Attachments A, B, C, and D.

Table 7. U Isotope Composition for Re-Enriched Recycled Uranium at U²³⁵ Content 3.25wt% and 5.0wt%

Isotopes	Feed Material ¹	Calculated Product U ²	IAEA 2007 ^{1,3} 3.25wt%	Calculated Product U ²	IAEA 2007 ¹ 5.0wt%
U ²³⁸	98.64%	95.5%	95.25%	93.18%	93.23%
U ²³⁶	0.42%	1.19%	1.09%	1.72%	1.55%
U ²³⁵	0.92%	3.26%	3.25%	4.98%	5.0%
U ²³⁴	0.021%	0.086%	0.085%	0.134%	0.133%

1 Data taken from Table 19, p. 33 of IAEA 2007.

2 Values assume cascade tails at 0.35wt% U²³⁵.

3 Percentages do not add to 100%.

Table 8. U Isotope Composition for Re-Enriched Spent Reactor Uranium at U-235 Content 3.00wt%

Isotopes	Feed Material ¹	Calculated Product U ²	Calculated by Wood ^{2,3}
U ²³⁸	98.68	95.82	95.89
U ²³⁶	0.4	1.09	1.027
U ²³⁵	0.9	3.01	3.00
U ²³⁴	0.02	0.076	0.077

1 Data taken from Table 1, p. 1204, Wood 1996.

2 Values based on cascade tails at 0.3wt% U²³⁵ from Wood 1996, p. 1204.

3 Data taken from Table 2, p. 1206, Wood 1996.

Table 9. U Isotope Composition for Re-Enriched Recycled Uranium at U-235 Content 4.05wt%

Isotopes	Feed Material ¹	Calculated Product U ²	IAEA 2007 ¹
U ²³⁸	98.819%	94.574%	94.652%
U ²³⁶	0.343%	1.28%	1.207%
U ²³⁵	0.822%	4.05%	4.047%
U ²³⁴	0.016%	0.094%	0.094%

1 Data taken from Table 34, p. 79 of IAEA 2007.

2 Values assume cascade tails at 0.35wt% U²³⁵.

Table 10. U Isotope Composition for Re-Enriched Natural Uranium at U-235 Content 2.207wt% and 3.42wt%

Isotopes	Feed Material ¹	Calculated Product U ²	IAEA 2007 ¹ 2.207wt%	Calculated Product U ²	IAEA 2007 ¹ 3.42wt%
U ²³⁸	99.2765%	94.574%	94.652%	96.5381	96.55%
U ²³⁶	0.0%	0.0%	0.0%	0.0%	0.0%
U ²³⁵	0.718%	2.200%	2.207%	3.43%	3.42%
U ²³⁴	0.0055%	0.0198%	0.019%	0.0319%	0.030%

1 Data taken from Table 5, p. 7 of IAEA 2007.

2 Values assume cascade tails at 0.35wt% U²³⁵.

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Attachment A

Calculation Spreadsheets for

Table 7. U Isotope Composition for Re-Enriched Recycled Uranium at U²³⁵ Content 3.25wt% and 5.0wt%

Isotopes	Feed Material ¹	Calculated Product U ²	IAEA 2007 ^{1,3} 3.25wt%	Calculated Product U ²	IAEA 2007 ¹ 5.0wt%
U ²³⁸	98.64%	95.5%	95.25%	93.18%	93.23%
U ²³⁶	0.42%	1.19%	1.09%	1.72%	1.55%
U ²³⁵	0.92%	3.26%	3.25%	4.98%	5.0%
U ²³⁴	0.021%	0.086%	0.085%	0.134%	0.133%

1 Data taken from Table 19, p. 33 of IAEA 2007.

2 Values assume cascade tails at 0.35wt% U²³⁵.

3 Percentages do not add to 100%.

Enrichment Factor Calculator for Centrifuges

wall radius		sample radius	isotope masses		alpha	Feed (IAEA 2007)		
a	r	m _i	m _j	$\alpha_{ij} = \exp [((m_i - m_j) * v_a^2 / 2RT) * (1 - r^2 / a^2)]$	$x_i(r_0)$	$x_{238}(r_0)$	$x_i(a)$	$x_j(a)$
0.5	0.5	238	236	1	0.00021	0.98639	0.00021	$x_i(a) = 1 / (1 + \text{SUM}_{j \neq i} ((1 / \alpha_{ij}(r,a)) * (x_j(r) / x_i(r))))$
0.5	0.5	238	235	1	0.0092		0.0092	$x_j(r) = 1 / (1 + \text{SUM}_{i \neq j} ((\alpha_{ij}(r,a)) * (x_i(a) / x_j(a))))$
0.5	0.5	238	234	1	0.0042		0.0042	
0.5	0.5	236	235	1			0.98639	
0.5	0.5	236	234	1				
0.5	0.5	235	234	1				
					Assumed value of x for radius r		Calculated value of x for radius a	
					$x_i(r)$	U_{238}	$x_i(a)$	
				0.999761	0.98639	U_{238}	0.98662563	
0.5	0.45	238	236	1.010812	0.00021	U_{236}	0.0002078	
0.5	0.45	238	235	1.016261	0.0092	U_{235}	0.00905495	
0.5	0.45	238	234	1.02174	0.0042	U_{234}	0.00411162	
0.5	0.45	236	235	1.005391				
0.5	0.45	236	234	1.010812				
					Calculated value of x for radius r		enrichment factor	
0.5	0.45	235	234	1.005391	$x_j(r) = \alpha_{ij}(a,r) * (x_i(a) / x_j(a))$		$e = x_i(r) / x_j(a)$	
					Calculated value of x at r given x at a			
					0.986175627	U_{238}	0.98662563	0.9997827
0.5	0.4	238	236	1.020584	0.000211984	U_{236}	0.0002078	1.0094488
0.5	0.4	238	235	1.031035	0.009331714	U_{235}	0.00905495	1.0143168
0.5	0.4	238	234	1.041593	0.004280675	U_{234}	0.00411162	1.0192083
0.5	0.4	236	235	1.01024				
0.5	0.4	236	234	1.020584				
0.5	0.4	235	234	1.01024				
R	8.31E+03 (g-m ² /sec ²)/(g-mole*K°)							
T	340 °K							
v _a	400 m/sec							

0.9994875 1.00944875 1.01431678 1.019208286															U-238 enrichment factor U-236 enrichment factor U-235 enrichment factor U-234 enrichment factor															U _{n+1} = U _n * (1 - (e _n 235 * (f ₂₃₆ - 1)) / (e _n 235 * f ₂₃₆ - e235-tails)) T _n = U _n - U _{n+1} e235-tails = 0.0035														
Stage	U feed	wt% 236	g 236	wt% 235	g 235	wt% 234	g 234	Tails (0.2 wt% 235)	0.00	Cum Tails	wt% 238	g 238	Total g	MUF	Stage	stages from e235-tails	tails 234	grams 234 in tails	grams 234 in product	tails 235	grams 235 in tails	grams 235 in product	tails 236	grams 236 in tails	grams 236 in product	grams 238 in tails	grams 238 in product																	
1	1,000,000.00	0.42	4200	0.92	9200	0.021	210	0.00	0.0	0.98639	986390	1000000	0.00	1	68.77895	0.005674	1.013966	208.9806	0.35	62.87756	9137.122	0.219957	39.51526	4160.485	17,861.61	968,528.39																		
2	982,034.98	0.4236	4160.485	0.930427	9137.122	0.02128	208.9806	17,965.02	17,965.02	0.986176	968,528.39	1000000	0.00	2	69.57475	0.005664	0.996129	207.9845	0.35	61.55977	9075.563	0.220219	38.73321	4121.752	17,487.22	951,041.18																		
3	964,446.48	0.42737	4121.752	0.934103	9075.563	0.021565	207.9845	17,588.50	35,553.52	0.985961	951,041.18	1000000	0.00	3	70.37347	0.005653	0.973485	207.011	0.35	60.27489	9015.288	0.220485	37.9706	4083.781	17,122.18	933,919.00																		
4	947,225.08	0.431131	4083.781	0.951758	9051.288	0.021854	207.011	17,221.40	52,774.92	0.985747	933,919.00	1000000	0.00	4	71.17507	0.005642	0.951416	206.0596	0.35	59.02197	8956.266	0.220755	37.22686	4046.554	16,766.22	917,152.78																		
5	930,361.66	0.434944	4046.554	0.962665	8966.266	0.022148	206.0596	16,863.42	69,638.34	0.985533	917,152.78	1000000	0.00	5	71.97951	0.005631	0.929905	205.1297	0.35	57.80008	8898.466	0.221029	36.50143	4010.053	16,419.08	900,733.70																		
6	913,847.35	0.43881	4010.053	0.973737	8898.466	0.022447	205.1297	16,514.31	86,152.65	0.985319	900,733.70	1000000	0.00	6	72.78675	0.00562	0.908938	204.2208	0.35	56.60831	8841.857	0.221307	35.79377	3974.259	16,080.49	884,653.21																		
7	897,673.55	0.442729	3974.259	0.984975	8841.857	0.02275	204.2208	16,173.80	102,326.45	0.985104	884,653.21	1000000	0.00	7	73.59676	0.005609	0.888497	203.3323	0.35	55.44558	8786.412	0.221589	35.10338	3939.155	15,750.22	868,902.99																		
8	881,831.89	0.446701	3939.155	0.996382	8786.412	0.023058	203.3323	15,841.66	118,168.11	0.98489	868,902.99	1000000	0.00	8	74.40949	0.005597	0.868569	202.4637	0.35	54.31169	8732.1	0.221875	34.42975	3904.726	15,428.02	853,474.98																		
9	866,314.27	0.450729	3904.726	1.00796	8732.1	0.023371	202.4637	15,517.63	133,685.73	0.984678	853,474.98	1000000	0.00	9	75.22491	0.005586	0.849138	201.6146	0.35	53.20519	8678.895	0.222165	33.77239	3870.953	15,113.66	838,361.32																		
10	851,112.78	0.454811	3870.953	1.019711	8678.895	0.023688	201.6146	15,201.48	148,887.22	0.984462	838,361.32	1000000	0.00	10	76.04298	0.005574	0.830191	200.7844	0.35	52.12549	8626.769	0.222459	33.13084	3837.823	14,806.91	823,554.41																		
11	836,219.79	0.458949	3837.823	1.031639	8626.769	0.024011	200.7844	14,893.00	163,780.21	0.984234	823,554.41	1000000	0.00	11	76.86366	0.005563	0.811713	199.9727	0.35	51.07183	8575.697	0.222757	32.50465	3805.318	14,507.56	809,046.85																		
12	821,627.84	0.463144	3805.318	1.043745	8575.697	0.024339	199.9727	14,591.95	178,372.16	0.984034	809,046.85	1000000	0.00	12	77.68691	0.005551	0.793692	199.179	0.35	50.04349	8525.654	0.22306	31.89338	3773.424	14,215.41	794,831.44																		
13	807,329.70	0.467396	3773.424	1.056031	8525.654	0.024671	199.179	14,298.14	192,670.30	0.983821	794,831.44	1000000	0.00	13	78.51271	0.005539	0.776115	198.4028	0.35	49.03973	8476.614	0.223362	31.29661	3742.128	13,930.24	780,901.20																		
14	793,318.35	0.471706	3742.128	1.068501	8476.614	0.025009	198.4028	14,011.35	206,681.65	0.983607	780,901.20	1000000	0.00	14	79.34101	0.005527	0.758897	197.6439	0.35	48.05987	8428.554	0.223677	30.71391	3711.414	13,651.86	767,249.34																		
15	779,586.96	0.476074	3711.414	1.081156	8428.554	0.025352	197.6439	13,731.39	220,413.04	0.983393	767,249.34	1000000	0.00	15	80.17178	0.005515	0.742244	196.9016	0.35	47.10324	8381.451	0.223991	30.14491	3681.269	13,380.08	753,869.27																		
16	766,128.89	0.480503	3681.269	1.094	8381.451	0.025701	196.9016	13,458.07	233,871.11	0.983179	753,869.27	1000000	0.00	16	81.00499	0.005503	0.725925	196.1757	0.35	46.16919	8335.282	0.22431	29.58992	3651.68	13,114.71	740,754.55																		
17	752,937.69	0.484991	3651.68	1.107035	8335.282	0.026055	196.1757	13,191.20	247,062.31	0.98296	740,754.55	1000000	0.00	17	81.84059	0.00549	0.710003	195.4657	0.35	45.25709	8290.025	0.224633	29.04653	3622.633	12,845.57	727,898.97																		
18	740,007.09	0.489454	3622.633	1.120263	8290.025	0.026414	195.4657	12,930.60	259,992.91	0.982752	727,898.97	1000000	0.00	18	82.67856	0.005478	0.694467	194.7712	0.35	44.36633	8245.658	0.224961	28.51623	3594.117	12,602.52	715,296.45																		
19	727,331.00	0.494152	3594.117	1.133687	8245.658	0.026779	194.7712	12,678.10	272,669.00	0.982538	715,296.45	1000000	0.00	19	83.51886	0.005466	0.679306	194.0919	0.35	43.49633	8202.162	0.225292	27.99824	3566.119	12,355.35	702,941.10																		
20	714,903.48	0.498825	3566.119	1.14731	8202.162	0.027149	194.0919	12,427.52	285,096.52	0.982325	702,941.10	1000000	0.00	20	84.36146	0.005454	0.664511	193.4277	0.35	42.64652	8159.516	0.225628	27.48914	3538.627	12,113.92	690,827.19																		
21	702,718.76	0.503562	3538.627	1.161135	8159.516	0.027526	193.4277	12,184.72	297,281.24	0.982111	690,827.19	1000000	0.00	21	85.20633	0.005441	0.650068	192.7774	0.35	41.81633	8117.699	0.225968	26.99759	3511.629	11,878.06	678,948.13																		
22	690,771.23	0.508364	3511.629	1.175165	8117.699	0.027908	192.7774	11,947.52	309,228.77	0.981898	678,948.13	1000000	0.00	22	85.96304	0.005428	0.635972	192.1414	0.35	41.00524	8076.694	0.226313	26.51428	3485.115	11,647.63	667,301.50																		
23	679,055.45	0.51323	3485.115	1.189401	8076.694	0.028295	192.1414	11,715.78	320,944.55	0.981685	667,301.50	1000000	0.00	23	86.80274	0.005416	0.622212	191.5192	0.35	40.21273	8036.481	0.226661	26.04191	3459.073	11,422.48	655,879.02																		
24	667,566.10	0.518162	3459.073	1.203848	8036.481	0.028689	191.5192	11,489.35	332,433.90	0.981471	655,879.02	1000000	0.00	24	87.75421	0.005403	0.608779	190.9104	0.35	39.4383	7997.043	0.																						

Attachment B

Calculation Spreadsheets for

Table 8. U Isotope Composition for Re-Enriched Spent Reactor Uranium at U-235 Content 3.00wt%

Isotopes	Feed Material ¹	Calculated Product U ²	Calculated by Wood ^{2,3}
U ²³⁸	98.68	95.82	95.89
U ²³⁶	0.4	1.09	1.027
U ²³⁵	0.9	3.01	3.00
U ²³⁴	0.02	0.076	0.077

1 Data taken from Table 1, p. 1204, Wood 1996.

2 Values based on cascade tails at 0.3wt% U²³⁵ from Wood 1996, p. 1204.

3 Data taken from Table 2, p. 1206, Wood 1996.

Enrichment Factor Calculator for Centrifuges

wall radius	sample radius	isotope masses		alpha	Feed			
a	r	m _i	m _j	$\alpha_{ij} = \exp [((m_i - m_j) \cdot v_a^2 / 2RT) \cdot (1 - r^2 / a^2)]$	x _i (r ₀)	x ₂₃₈ (r ₀)	(Wood 1996)	
0.5	0.5	238	236	1	0.0002	0.9868	0.0002	$x_i(a) = 1 / (1 + \text{SUM}_{i \neq j} ((1 / \alpha_{ij}(r,a)) \cdot (x_j(r) / x_i(r))))$
0.5	0.5	238	235	1	0.009		0.009	$x_i(r) = 1 / (1 + \text{SUM}_{i \neq j} ((\alpha_{ij}(r,a)) \cdot (x_j(a) / x_i(a))))$
0.5	0.5	238	234	1	0.004		0.004	
0.5	0.5	236	235	1			0.9868	
0.5	0.5	236	234	1				
0.5	0.5	235	234	1				
					Assumed value of x for radius r		Calculated value of x for radius a	
					x _i (r)		x _i (a)	
				0.999769	0.9868	U ₂₃₈	0.98702826	
0.5	0.45	238	236	1.010812	0.0002	U ₂₃₆	0.00019791	
0.5	0.45	238	235	1.016261	0.009	U ₂₃₅	0.00885804	
0.5	0.45	238	234	1.02174	0.004	U ₂₃₄	0.00391579	
0.5	0.45	236	235	1.005391				
0.5	0.45	236	234	1.010812				
					Calculated value of x for radius r			
0.5	0.45	235	234	1.005391			$x_j(r) = \alpha_{ij}(a,r) \cdot (x_i(a) / x_j(a))$	enrichment factor
					Calculated value of x at r given x at a			$e = x_j(r) / x_j(a)$
					0.986592334	U ₂₃₈	0.98702826	0.9997896
0.5	0.4	238	236	1.020584	0.000201891	U ₂₃₆	0.00019791	1.0094557
0.5	0.4	238	235	1.031035	0.009128914	U ₂₃₅	0.00885804	1.0143238
0.5	0.4	238	234	1.041593	0.004076861	U ₂₃₄	0.00391579	1.0192153
0.5	0.4	236	235	1.01024				
0.5	0.4	236	234	1.020584				
0.5	0.4	235	234	1.01024				
R	8.31E+03 (g·m ² /sec ²)/(g·mole·K°)							
T	340 °K							
v _a	400 m/sec							

0.999789556
1.009455704
1.014323767
1.019215307

U-238 enrichment factor
U-236 enrichment factor
U-235 enrichment factor
U-234 enrichment factor

$$U_{n+1} = U_n * (1 - (e_{n235} * (f_{236} - 1)) / (e_{n235} * f_{236} - e235\text{-tails}))$$
$$T_n = U_n - U_{n+1}$$
$$e235\text{-tails} = 0.003$$

Stage	U feed	wt% 236	g 236	wt% 235	g 235	wt% 234	g 234	Tails (0.2 wt% 235)	Cum Tails	wt% 238	g 238	Total g	MUF	Stage	stages from e235-tails	tails 234	grams 234 in tails	grams 234 in product	tails 235	grams 235 in tails	grams 235 in product	tails 236	grams 236 in tails	grams 236 in product	grams 238 in tails	grams 238 in product
1	1,000,000.00	0.4	4000	0.9	9000	0.02	200	0.00		0.9868	986800	1000000	0.00	1	78.10457	0.004523	0.817898	199.1821	0.3	54.24974	8945.75	0.19179	34.68185	3965.318	17,993.50	968,806.50
2	981,916.75	0.403834	3965.318	0.91105	8945.75	0.020285	199.1821	18,083.25	18,083.25	0.986648	968,806.50	1000000	0.00	2	78.96479	0.004513	0.798608	198.3835	0.3	53.08809	8892.662	0.192067	33.98827	3931.33	17,608.15	951,198.35
3	946,901.78	0.407721	3931.33	0.922264	8840.705	0.020574	197.6037	17,696.03	53,779.28	0.986494	951,198.35	1000000	0.00	3	79.82718	0.004503	0.779837	197.6037	0.3	51.95684	8840.705	0.192348	33.21268	3898.017	17,232.90	933,965.45
5	929,950.10	0.415653	3865.363	0.945196	8789.85	0.021167	196.8421	16,951.68	70,049.90	0.98618	917,098.05	1000000	0.00	4	80.69171	0.004493	0.76157	196.8421	0.3	50.85503	8789.85	0.192633	32.65449	3865.363	16,867.41	917,098.05
6	913,356.19	0.419699	3833.35	0.956918	8740.069	0.02147	196.0983	16,593.91	86,643.81	0.986019	900,586.67	1000000	0.00	5	81.55584	0.004482	0.743791	196.0983	0.3	49.78174	8740.069	0.192921	32.01317	3833.35	16,511.37	900,586.67
7	897,110.83	0.423801	3801.961	0.968814	8691.332	0.021778	195.3718	16,245.36	102,889.17	0.985856	884,422.17	1000000	0.00	6	82.42704	0.004472	0.726484	195.3718	0.3	48.73607	8691.332	0.193213	31.38819	3801.961	16,164.51	884,422.17
8	881,205.11	0.427957	3771.182	0.980886	8643.615	0.022209	194.6622	15,905.72	118,794.89	0.985691	868,595.65	1000000	0.00	7	83.29777	0.004462	0.709635	194.6622	0.3	47.71717	8643.615	0.193509	30.77902	3771.182	15,826.52	868,595.65
9	865,630.37	0.43217	3740.997	0.993136	8596.891	0.022408	193.9689	15,574.74	134,369.63	0.985523	853,098.52	1000000	0.00	8	84.1705	0.004451	0.69323	193.9689	0.3	46.72421	8596.891	0.193809	30.18519	3740.997	15,497.13	853,098.52
10	850,378.25	0.43644	3711.391	1.005568	8551.135	0.02273	193.2917	15,252.13	149,621.75	0.985353	837,922.43	1000000	0.00	9	85.0452	0.00444	0.677254	193.2917	0.3	45.75638	8551.135	0.194112	29.56622	3711.391	15,176.09	837,922.43
11	835,440.60	0.440767	3682.349	1.018184	8506.322	0.023057	192.63	14,937.64	164,559.40	0.98518	823,059.30	1000000	0.00	10	85.92184	0.00443	0.661696	192.63	0.3	44.81292	8506.322	0.194419	29.04164	3682.349	14,863.13	823,059.30
12	820,809.57	0.445153	3653.858	1.030986	8462.429	0.02339	191.9835	14,631.03	179,190.43	0.985005	808,501.30	1000000	0.00	11	86.80037	0.004419	0.645642	191.9835	0.3	43.8931	8462.429	0.19473	28.49103	3653.858	14,558.00	808,501.30
13	806,477.51	0.449598	3625.904	1.043976	8419.433	0.023727	191.3517	14,332.06	193,522.49	0.984827	794,240.82	1000000	0.00	12	87.68079	0.004408	0.63178	191.3517	0.3	42.99618	8419.433	0.195045	27.95395	3625.904	14,260.48	794,240.82
14	792,432.02	0.454102	3598.474	1.057158	8377.311	0.024069	190.7343	14,040.50	207,562.98	0.984647	780,270.50	1000000	0.00	13	88.56304	0.004397	0.617399	190.7343	0.3	42.12149	8377.311	0.195363	27.43	3598.474	13,970.33	780,270.50
15	778,680.90	0.458667	3571.556	1.070534	8336.043	0.024417	190.1309	13,756.12	221,319.10	0.984464	766,583.17	1000000	0.00	14	89.4471	0.004386	0.603387	190.1309	0.3	41.26835	8336.043	0.195686	26.91878	3571.556	13,687.32	766,583.17
16	765,202.20	0.463294	3545.136	1.084106	8295.607	0.02477	189.5412	13,478.71	234,797.80	0.984278	753,171.91	1000000	0.00	15	90.33295	0.004375	0.589732	189.5412	0.3	40.43612	8295.607	0.196012	26.4199	3545.136	13,411.26	753,171.91
17	751,994.13	0.467983	3519.293	1.097879	8255.982	0.025128	188.9647	13,208.06	248,005.87	0.984049	740,029.98	1000000	0.00	16	91.22055	0.004364	0.576425	188.9647	0.3	39.62418	8255.982	0.196342	25.93302	3519.293	13,141.93	740,029.98
18	739,050.15	0.472734	3493.745	1.111853	8227.115	0.025462	188.4013	12,943.98	260,949.85	0.983899	727,150.86	1000000	0.00	17	92.10988	0.004353	0.563456	188.4013	0.3	38.83194	8227.115	0.196676	25.45777	3493.745	12,877.15	727,150.86
19	726,363.88	0.47755	3468.751	1.126032	8179.092	0.025862	187.8505	12,686.27	273,636.12	0.983706	714,528.19	1000000	0.00	18	93.00091	0.004342	0.550813	187.8505	0.3	38.05882	8179.092	0.197014	24.99377	3468.751	12,622.67	714,528.19
20	713,929.13	0.48234	3444.21	1.14042	8141.787	0.026237	187.312	12,434.75	286,070.87	0.983508	702,155.82	1000000	0.00	19	93.8936	0.004331	0.538489	187.312	0.3	37.30425	8141.787	0.197356	24.54075	3444.21	12,372.52	702,155.82
21	701,739.89	0.487376	3420.112	1.155018	8105.22	0.026617	186.7855	12,189.23	298,260.11	0.98331	690,027.78	1000000	0.00	20	94.78794	0.004319	0.526473	186.7855	0.3	36.5677	8105.22	0.197702	24.09836	3420.112	12,128.04	690,027.78
22	689,790.34	0.492388	3396.446	1.16983	8069.371	0.027004	186.2707	11,949.55	310,209.66	0.983108	678,138.26	1000000	0.00	21	95.68389	0.004308	0.514756	186.2707	0.3	35.84865	8069.371	0.198052	23.66629	3396.446	11,889.52	678,138.26
23	678,074.81	0.497467	3373.201	1.184858	8034.224	0.027431	185.7674	11,715.53	321,925.19	0.982903	666,481.62	1000000	0.00	22	96.58144	0.004296	0.503331	185.7674	0.3	35.1466	8034.224	0.198405	23.24425	3373.201	11,656.64	666,481.62
24	666,587.79	0.502615	3350.669	1.200106	7999.763	0.027795	185.2752	11,487.02	333,412.21	0.982695	655,052.38	1000000	0.00	23	97.48055	0.004285	0.492188	185.2752	0.3	34.46105	7999.763	0.198763	22.83195	3350.669	11,429.23	655,052.38
25	655,323.94	0.507831	3327.94	1.215578	7969.972	0.028199	184.7939	11,263.85	344,676.06	0.982484	643,845.24	1000000	0.00	24	98.3812	0.004273	0.48132	184.7939	0.3	33.79155	7969.972	0.199123	22.42911	3327.94		

Attachment C

Calculation Spreadsheets for

Table 9. U Isotope Composition for Re-Enriched Recycled Uranium at U-235 Content 4.05wt%

Isotopes	Feed Material ¹	Calculated Product U ²	IAEA 2007 ¹
U ²³⁸	98.819%	94.574%	94.652%
U ²³⁶	0.343%	1.28%	1.207%
U ²³⁵	0.822%	4.05%	4.047%
U ²³⁴	0.016%	0.094%	0.094%

1 Data taken from Table 34, p. 79 of IAEA 2007.

2 Values assume cascade tails at 0.35wt% U²³⁵.

Enrichment Factor Calculator for Centrifuges

wall radius	sample radius	isotope masses		alpha	Feed			
a	r	m _i	m _j	$\alpha_{ij} = \exp [((m - m_i) \cdot v_a^2 / 2RT) \cdot (1 - \tilde{r}^2 / a^2)]$	$x_i(r_0)$	$x_{238}(r_0)$	(IAEA 2007)	
0.5	0.5	238	236	1	0.00016	0.98819	0.00016	$x_i(a) = 1 / (1 + \text{SUM}_{i \neq j} ((1 / \alpha_{ij}(r,a)) \cdot (x_i(r) / x_i(r))))$
0.5	0.5	238	235	1	0.00822		0.00822	$x_i(r) = 1 / (1 + \text{SUM}_{i \neq j} ((\alpha_{ij}(r,a)) \cdot (x_i(a) / x_i(a))))$
0.5	0.5	238	234	1	0.00343		0.00343	
0.5	0.5	236	235	1			0.98819	
0.5	0.5	236	234	1				
0.5	0.5	235	234	1				
				Assumed value of x for radius r		Calculated value of x for radius a		
				$x_i(r)$		$x_i(a)$		
				0.999794	0.98819	U ₂₃₈	0.98839383	
0.5	0.45	238	236	1.010812	0.00016	U ₂₃₆	0.00015832	
0.5	0.45	238	235	1.016261	0.00822	U ₂₃₅	0.00809014	
0.5	0.45	238	234	1.02174	0.00343	U ₂₃₄	0.00335771	
0.5	0.45	236	235	1.005391				
0.5	0.45	236	234	1.010812				
					Calculated value of x for radius r			
0.5	0.45	235	234	1.005391			$x_j(r) = \alpha_{ij}(a,r) \cdot (x_i(a) / x_j(a))$	enrichment factor
					Calculated value of x at r given x at a			$e = x_i(r) / x_j(a)$
					0.988004564	U ₂₃₈	0.98839383	0.9998123
0.5	0.4	238	236	1.020584	0.000161517	U ₂₃₆	0.00015832	1.0094787
0.5	0.4	238	235	1.031035	0.008337931	U ₂₃₅	0.00809014	1.0143469
0.5	0.4	238	234	1.041593	0.003495988	U ₂₃₄	0.00335771	1.0192385
0.5	0.4	236	235	1.01024				
0.5	0.4	236	234	1.020584				
0.5	0.4	235	234	1.01024				
R	8.31E+03 (g-m ² /sec ²)/(g-mole*K°)							
T	340 °K							
v _a	400 m/sec							

0.999812348
1.009478716
1.01434689
1.019238541

U-238 enrichment factor
U-236 enrichment factor
U-235 enrichment factor
U-234 enrichment factor

$$U_{n+1} = U_n * (1 - (e_n * 235 * (f_{235} - 1)) / (e_n * 235 * f_{235} - e235 * tails))$$
$$T_n = U_n - U_{n+1}$$

e235-tails= 0.0035

	U feed	wt% 236	g 236	wt% 235	g 235	wt% 234	g 234	Tails (0.2 wt% 235)	Cum Tails	wt% 238	g 238	Total g	MUF	Stage	stages from e235-tails 234	grams 234 in tails	grams 234 in product	tails 235	grams 235 in tails	grams 235 in product	tails 236	grams 236 in tails	grams 236 in product	grams 238 in tails	grams 238 in product	
1	1,000,000.00	0.343	3430	0.822	8220	0.016	160	0.00		0.98819	988190	1000000	0.00	1	60.69769	0.005033	0.936435	159.0636	0.35	65.12536	8154.875	0.193466	35.98877	3394.001	18.50518	969.684.82
2	981.392.76	0.345835	3394.001	0.830949	8154.875	0.016208	159.0636	18,607.24	18,607.24	0.98807	969.684.82	1000000	0.00	2	61.64124	0.005024	0.9144	158.1492	0.35	63.69708	8091.178	0.193665	35.2485	3358.756	18.099.31	951.585.51
3	963.193.59	0.34871	3358.756	0.840036	8028.871	0.016419	158.1492	18,199.16	36,806.41	0.987948	951.585.51	1000000	0.00	3	62.22815	0.005016	0.892948	157.2562	0.35	62.30617	8028.871	0.193868	34.51189	3324.244	17.704.05	933.881.46
4	945.931.83	0.351626	3324.244	0.849624	8028.871	0.016634	157.2562	17,801.76	54,608.17	0.987825	933.881.46	1000000	0.00	4	62.99839	0.005008	0.872063	156.3842	0.35	60.95151	7967.92	0.194073	33.79735	3290.446	17.319.10	916.562.36
5	927.977.11	0.354583	3290.446	0.858633	7967.92	0.016852	156.3842	17,414.72	72,022.89	0.987699	916.562.36	1000000	0.00	5	63.77189	0.004999	0.851728	155.5324	0.35	59.632	7908.288	0.194282	33.10128	3257.345	16.944.13	899.618.23
6	910.939.39	0.357581	3257.345	0.868146	7908.288	0.017074	155.5324	17,037.71	89,060.61	0.987572	899.618.23	1000000	0.00	6	64.54683	0.00499	0.831926	154.7005	0.35	58.34683	7849.941	0.194495	32.42314	3224.922	16.578.85	883.039.38
7	894.268.95	0.360621	3224.922	0.877805	7849.941	0.017299	154.7005	16,670.45	105,731.05	0.987443	883.039.38	1000000	0.00	7	65.32855	0.004982	0.812642	153.8879	0.35	57.0942	7752.847	0.19471	31.76238	3193.16	16.222.96	866.816.42
8	877.956.32	0.363704	3193.16	0.887612	7752.847	0.017528	153.8879	15,963.97	138,007.65	0.987178	850.940.24	1000000	0.00	8	66.11161	0.004973	0.793859	153.004	0.35	55.87389	7736.973	0.194929	31.11848	3162.041	15.876.18	850.940.24
9	861.992.35	0.366829	3162.041	0.897569	7736.973	0.01776	153.004	15,624.20	153,631.85	0.987043	850.940.24	1000000	0.00	9	66.89778	0.004964	0.775564	152.3184	0.35	54.6847	7682.289	0.195152	30.49082	3131.55	15.538.25	835.401.99
10	846.368.15	0.369999	3131.55	0.907677	7682.289	0.017997	152.3184	15,281.06	168,924.91	0.986960	820.193.10	1000000	0.00	10	67.687	0.004955	0.757741	151.5607	0.35	53.5257	7628.763	0.195378	29.87923	3101.671	15.208.89	820.193.10
11	831.075.09	0.373212	3101.671	0.917939	7628.763	0.018237	151.5607	14,970.28	183,895.23	0.986767	805.305.23	1000000	0.00	11	68.47925	0.004946	0.740377	150.8203	0.35	52.39599	7576.367	0.195607	29.72838	3072.386	14.887.86	805.305.23
12	816.104.81	0.37647	3072.386	0.928357	7576.367	0.018481	150.8203	14,655.63	198,550.98	0.986626	790.730.33	1000000	0.00	12	69.27447	0.004936	0.723459	150.0969	0.35	51.2947	7525.072	0.19584	28.70157	3043.687	14.574.91	790.730.33
13	801.448.18	0.379773	3043.687	0.938933	7525.072	0.018728	150.0969	14,348.86	212,899.68	0.986482	776.460.53	1000000	0.00	13	70.07262	0.004927	0.706974	149.3899	0.35	50.22011	7474.851	0.196076	28.13469	3015.552	14.269.80	776.460.53
14	787.100.32	0.383122	3015.552	0.949669	7474.851	0.018988	149.3899	14,049.74	226,949.42	0.986337	762.488.23	1000000	0.00	14	70.87367	0.004918	0.690908	148.609	0.35	49.1741	7425.677	0.196316	27.58188	2987.97	13.972.30	762.488.23
15	773.050.58	0.386517	2987.97	0.960568	7425.677	0.019235	148.609	13,758.05	240,707.47	0.986189	748.006.05	1000000	0.00	15	71.67577	0.004908	0.675251	148.0237	0.35	48.15318	7377.524	0.196559	27.04272	2960.927	13.682.18	748.006.05
16	759.292.53	0.389959	2960.927	0.971631	7377.524	0.019495	148.0237	12,451.85	254,181.24	0.986039	735.406.81	1000000	0.00	16	72.48429	0.004898	0.659989	147.3637	0.35	47.15751	7330.366	0.196806	26.51681	2934.41	13.399.24	735.406.81
17	745.818.95	0.393448	2934.41	0.982861	7330.366	0.019759	147.3637	12,136.10	259,377.15	0.985887	722.283.54	1000000	0.00	17	73.28379	0.004889	0.645112	146.7186	0.35	46.18635	7284.18	0.197056	26.00376	2908.407	13.123.27	722.283.54
18	732.622.85	0.396986	2908.407	0.994261	7284.18	0.020299	146.7186	12,025.43	280,302.58	0.985733	709.429.49	1000000	0.00	18	74.10602	0.004879	0.630600	146.088	0.35	45.239	7238.941	0.19731	25.50321	2882.903	12.854.06	709.429.49
19	719.697.42	0.400572	2882.903	1.005831	7238.941	0.020929	146.088	12,925.14	292,963.94	0.985576	696.838.07	1000000	0.00	19	74.92095	0.004869	0.616469	145.4715	0.35	44.31476	7194.628	0.197568	25.01479	2857.889	12.591.41	696.838.07
20	707.036.06	0.404207	2857.889	1.017576	7194.628	0.020575	145.4715	12,681.36	292,963.94	0.985418	684.502.92	1000000	0.00	20	75.73855	0.004859	0.602682	144.8699	0.35	43.41298	7151.213	0.197829	24.53815	2833.351	12.351.15	684.502.92
21	694.632.35	0.407892	2833.351	1.029496	7151.213	0.020855	144.8699	12,403.71	302,367.65	0.985256	672.417.73	1000000	0.00	21	76.55876	0.004849	0.589237	144.2796	0.35	42.53301	7108.68	0.198094	24.07295	2809.278	12.085.09	672.417.73
22	682.480.06	0.411628	2809.278	1.041595	7108.68	0.02114	144.2796	12,152.29	317,519.94	0.985093	660.576.77	1000000	0.00	22	77.38157	0.004839	0.576126	143.7035	0.35	41.67424	7067.006	0.198362	23.61887	2785.659	11.841.06	660.576.77
23	670.573.14	0.415415	2785.659	1.053875	7067.006	0.02143	143.7035	11,966.93	329,426.86	0.984927	648.973.00	1000000	0.00	23	78.20693	0.004828	0.563338	143.1402	0.35	40.83606	7026.17	0.198635	23.17559	2762.483	11.602.87	648.973.00
24	658.905.69	0.419253	2762.483	1.066339	7026.17	0.021724	143.1402	11,667.45	341,094.31	0.984758	637.603.53	1000000	0.00	24	79.0348	0.004818	0.550865	142.5893	0.35	40.01789	6986.152	0.19891	22.74729	2739.74	11.370.37	637.603.53
25	647.472.01	0.423144	2739.74	1.078989	6986.152	0.022022	142.5893	11,433.68	352,527.92	0.984588	626.460.12	1000000	0.00	25	79.86516	0.004807	0.538698	142.0506	0.35	39.21918	6946.933	0.19919	22.32019	2717.42	11.104.20	626.460.12
26	636.266.53	0.4270887																								

Attachment D

Calculation Spreadsheets for

Table 10. U Isotope Composition for Re-Enriched Natural Uranium at U-235 Content 2.207wt% and 3.42wt%

Isotopes	Feed Material ¹	Calculated Product U ²	IAEA 2007 ¹ 2.207wt%	Calculated Product U ²	IAEA 2007 ¹ 3.42wt%
U ²³⁸	99.2765%	94.574%	94.652%	96.5381	96.55%
U ²³⁶	0.0%	0.0%	0.0%	0.0%	0.0%
U ²³⁵	0.718%	2.200%	2.207%	3.43%	3.42%
U ²³⁴	0.0055%	0.0198%	0.019%	0.0319%	0.030%

1 Data taken from Table 5, p. 7 of IAEA 2007.

2 Values assume cascade tails at 0.35wt% U²³⁵.

Enrichment Factor Calculator for Centrifuges

wall radius	sample radius	isotope masses		alpha	Feed			
a	r	m _i	m _j	$\alpha_{ij} = \exp [((m_j - m_i) * v_a^2 / 2RT) * (1 - r^2 / a^2)]$	x _i (r ₀)	x ₂₃₈ (r ₀)	(IAEA 2007)	
0.5	0.5	238	236	1	0.000055	0.992765	0.000055	$x_i(a) = 1 / (1 + \text{SUM}_{i \neq j} ((1 / \alpha_{ij}(r,a)) * (x_i(r) / x_j(r))))$
0.5	0.5	238	235	1	0.00718		0.00718	$x_i(r) = 1 / (1 + \text{SUM}_{i \neq j} ((\alpha_{ij}(r,a)) * (x_i(a) / x_j(a))))$
0.5	0.5	238	234	1	1E-09		0.000000001	
0.5	0.5	236	235	1			0.992764999	
0.5	0.5	236	234	1				
0.5	0.5	235	234	1				
					Assumed value of x for radius r	Calculated value of x for radius a		
					x _i (r)	x _i (a)		
				0.999885	0.992764999	U ₂₃₈	0.992879653	
0.5	0.45	238	236	1.010812	0.000055	U ₂₃₆	5.4418E-05	
0.5	0.45	238	235	1.016261	0.00718	U ₂₃₅	0.007065928	
0.5	0.45	238	234	1.02174	0.000000001	U ₂₃₄	9.78835E-10	
0.5	0.45	236	235	1.005391				
0.5	0.45	236	234	1.010812				
0.5	0.45	235	234	1.005391				
					Calculated value of x at r given x at a	Calculated value of x for radius r		
							$x_i(r) = \alpha_{ij}(a,r) * (x_i(a) / x_j(a))$	enrichment factor
								e = x _i (r) / x _i (a)
					0.99266086	U ₂₃₈	0.992879653	0.9998951
0.5	0.4	238	236	1.020584	5.55259E-05	U ₂₃₆	5.4418E-05	1.0095623
0.5	0.4	238	235	1.031035	0.007283613	U ₂₃₅	0.007065928	1.0144308
0.5	0.4	238	234	1.041593	1.01932E-09	U ₂₃₄	9.78835E-10	1.0193229
0.5	0.4	236	235	1.01024				
0.5	0.4	236	234	1.020584				
0.5	0.4	235	234	1.01024				
R	8.31E+03 (g-m ² /sec ²)/(g-mole*K°)							
T	340 °K							
v _a	400 m/sec							

0.999895102
1.00956227
1.014430848
1.019322903U-238 enrichment factor
U-236 enrichment factor
U-235 enrichment factor
U-234 enrichment factor

$$U_{n+1} = U_n * (1 - (e_{u235} * (f_{236} - 1))) / (e_{u235} * f_{236} - e235\text{-tails})$$
$$T_n = U_n - U_{n+1}$$

e235-tails = 0.0035

	U feed	wt% 236	g 236	wt% 235	g 235	wt% 234	g 234	Tails (0.2 wt% 235)	Cum Tails	wt% 238	g 238	Total g	MUF	Stage	stages from e235-tails	tails 234	grams 234 in tails	grams 234 in product	tails 235	grams 235 in tails	grams 235 in product	tails 236	grams 236 in tails	grams 236 in product	tails 237	grams 237 in tails	grams 237 in product	tails 238	grams 238 in tails	grams 238 in product				
1	1,000,000.00	1E-07	0.001	0.718	7180	0.0055	55	0.00	0.992765	992765	1000000	0.00																						
2	980,389.65	1.01E-07	0.000988	0.725361	7111.364	0.005568	54.59253	19,610.35	19,610.35	0.992691	973,223.70	1000000	0.00	1	50.862	0.002078	0.407468	54.59253	0.35	68.63621	7111.364	6.16E-08	1.21E-05	0.000988	19,541.30	973.223.70	0.35	67.03274	7044.331	6.17E-08	1.18E-05	0.000976	19,084.78	954.138.92
3	961,237.44	1.02E-07	0.000976	0.73284	7044.331	0.005638	54.19511	19,152.21	38,762.56	0.992615	954,138.92	1000000	0.00	2	51.57795	0.002075	0.397419	54.19511	0.35	65.47383	7044.331	6.17E-08	1.15E-05	0.000965	18,209.45	935,497.97	0.35	63.95807	6914.899	6.18E-08	1.13E-05	0.000953	18,640.90	917,288.57
4	942,530.64	1.02E-07	0.000965	0.740438	6978.857	0.005709	53.80746	18,706.81	57,469.36	0.992539	935,497.97	1000000	0.00	3	52.29788	0.002072	0.387651	53.80746	0.35	63.95807	6914.899	6.18E-08	1.13E-05	0.000953	18,640.90	917,288.57	0.35	63.95807	6914.899	6.18E-08	1.13E-05	0.000953	18,640.90	917,288.57
5	924,256.90	1.03E-07	0.000953	0.748158	6914.899	0.005781	53.42931	18,273.73	75,743.10	0.992461	917,288.57	1000000	0.00	4	53.02176	0.002069	0.378154	53.42931	0.35	62.48409	6914.899	6.18E-08	1.11E-05	0.000942	17,789.75	899,498.83	0.35	62.48409	6914.899	6.18E-08	1.11E-05	0.000942	17,789.75	899,498.83
6	906,404.30	1.04E-07	0.000942	0.756	6852.415	0.005854	53.06309	17,852.60	93,595.70	0.992381	899,498.83	1000000	0.00	5	53.74954	0.002066	0.368921	53.06309	0.35	62.48409	6852.415	6.18E-08	1.1E-05	0.000942	17,789.75	899,498.83	0.35	61.05058	6791.364	6.19E-08	1.08E-05	0.000931	17,381.61	882,117.22
7	888,961.28	1.05E-07	0.000931	0.763966	6791.364	0.005928	52.70044	17,443.02	111,038.72	0.992301	882,117.22	1000000	0.00	6	54.48116	0.002064	0.359944	52.70044	0.35	61.05058	6791.364	6.19E-08	1.08E-05	0.000931	17,381.61	882,117.22	0.35	59.65624	6731.708	6.22E-08	1.06E-05	0.000921	16,984.63	885,132.58
8	871,916.64	1.06E-07	0.000921	0.772059	6731.708	0.006004	52.34923	17,044.64	128,083.36	0.992219	865,132.58	1000000	0.00	7	55.21658	0.002061	0.351213	52.34923	0.35	59.65624	6731.708	6.22E-08	1.06E-05	0.000921	16,984.63	885,132.58	0.35	58.29986	6673.408	6.22E-08	1.03E-05	0.000911	16,598.46	848,534.12
9	855,259.54	1.06E-07	0.000911	0.780279	6673.408	0.006081	52.00651	16,657.10	144,740.46	0.992136	848,534.12	1000000	0.00	8	55.95576	0.002058	0.342723	52.00651	0.35	58.29986	6673.408	6.22E-08	1.03E-05	0.000911	16,598.46	848,534.12	0.35	56.98025	6616.428	6.21E-08	1.01E-05	0.0009	16,222.76	832,311.37
10	838,979.47	1.07E-07	0.0009	0.788628	6616.428	0.006159	51.67204	16,280.07	161,020.53	0.992052	832,311.37	1000000	0.00	9	56.78963	0.002056	0.334464	51.67204	0.35	56.98025	6616.428	6.21E-08	1.01E-05	0.0009	16,222.76	832,311.37	0.35	54.46755	6506.732	6.21E-08	0.989E-06	0.000891	15,857.19	816,454.18
11	823,066.25	1.08E-07	0.000891	0.797109	6560.732	0.006238	51.34561	15,913.21	176,933.75	0.991967	816,454.18	1000000	0.00	10	57.44516	0.002051	0.32643	51.34561	0.35	54.46755	6506.732	6.21E-08	0.989E-06	0.000891	15,857.19	816,454.18	0.35	54.46755	6506.732	6.21E-08	0.989E-06	0.000891	15,857.19	816,454.18
12	807,510.04	1.09E-07	0.000881	0.805722	6506.285	0.006319	51.027	15,556.21	192,489.96	0.99188	800,952.73	1000000	0.00	11	58.19529	0.002048	0.318615	51.027	0.35	53.23069	6453.054	6.22E-08	0.97E-06	0.000871	15,155.23	785,797.50	0.35	52.04704	6401.007	6.23E-08	0.927E-06	0.000862	14,818.23	770,979.27
13	792,301.27	1.1E-07	0.000871	0.81447	6453.054	0.006401	50.71599	15,208.77	207,698.73	0.991791	785,797.50	1000000	0.00	12	58.94988	0.002045	0.31101	50.71599	0.35	52.04704	6401.007	6.23E-08	0.927E-06	0.000862	14,818.23	770,979.27	0.35	50.89478	6350.113	6.24E-08	0.907E-06	0.000853	14,490.77	756,489.09
14	777,430.69	1.11E-07	0.000862	0.823354	6401.007	0.006484	50.41238	14,870.58	222,569.31	0.991702	770,979.27	1000000	0.00	13	59.70619	0.002042	0.303611	50.41238	0.35	50.89478	6350.113	6.24E-08	0.907E-06	0.000853	14,490.77	756,489.09	0.35	49.77296	6300.34	6.24E-08	0.888E-06	0.000844	14,490.77	756,489.09
15	762,889.32	1.12E-07	0.000853	0.832377	6350.113	0.006569	50.11597	14,541.37	237,110.68	0.991611	756,489.09	1000000	0.00	14	60.46686	0.002038	0.29641	50.11597	0.35	49.77296	6300.34	6.24E-08	0.888E-06	0.000844	14,490.77	756,489.09	0.35	48.86065	6251.659	6.25E-08	0.869E-06	0.000835	13,859.79	728,458.51
16	748,668.47	1.13E-07	0.000844	0.841539	6300.34	0.006655	49.82656	14,220.85	251,331.53	0.991518	742,318.31	1000000	0.00	15	61.23095	0.002035	0.289402	49.82656	0.35	48.86065	6251.659	6.25E-08	0.869E-06	0.000835	13,859.79	728,458.51	0.35	47.81694	6204.042	6.26E-08	0.851E-06	0.000827	13,556.95	714,901.57
17	734,759.72	1.14E-07	0.000835	0.850844	6251.659	0.006743	49.54398	13,908.76	265,240.28	0.991424	728,458.51	1000000	0.00	16	61.99842	0.002032	0.282851	49.54398	0.35	47.81694	6204.042	6.26E-08	0.851E-06	0.000827	13,556.95	714,901.57	0.35	46.58096	6157.461	6.26E-08	0.849E-06	0.000819	13,262.00	701,639.57
18	721,154.88	1.15E-07	0.000827	0.860293	6204.042	0.006822	49.26804	13,604.84	282,153.97	0.991322	714,901.57	1000000	0.00	17	62.76922	0.002028	0.27594	49.26804	0.35	46.58096	6157.461	6.26E-08	0.849E-06	0.000819	13,262.00	701,639.57	0.35	45.57188	6111.889	6.27E-08	0.816E-06	0.00081	12,974.70	688,664.87
19	707,846.03	1.16E-07	0.000819	0.869887	6157.461	0.006902	48.99857	13,308.85	292,153.97	0.991232	701,639.57	1000000	0.00	18	63.5433	0.002025	0.269475	48.99857	0.35	45.57188	6111.889	6.27E-08	0.816E-06	0.00081	12,974.70	688,664.87	0.35	44.58887	6067.3	6.28E-08	0.78E-06	0.000802	12,694.83	675,970.04
20	694,825.50	1.17E-07	0.00081	0.879629	6111.889	0.007014	48.73539	13,020.54	305,174.50	0.991134	688,664.87	1000000	0.00	19	64.30263	0.002021	0.263181	48.73539	0.35	44.58887	6067.3	6.28E-08	0.78E-06	0.0										

APPENDIX C

**ORIGEN RESULTS FOR SPENT UO_2 FUEL AND SUCCESSIVE GENERATIONS
OF MOX FUEL FROM LWRs WHEN THEY ARE REPROCESSED AND RECYCLED
TOGETHER**

TABLE C-1
ACTINIDE CONCENTRATIONS IN SUCCESSIVE GENERATIONS OF SPENT UO₂ AND MOX FUEL FROM LWRs^{a,b}
SCENARIO #1 -- U+PU MIX FROM REPROCESSING BLENDED W/DU
TO PRODUCE MOX (5.0% EQUIVALENT FISSILE)

Category	Grams Per Metric Ton Initial Heavy Metal				Percent Abundance in Category					
	UO ₂ Fuel	1 st Generation MOX Fuel	2 nd Generation MOX Fuel	3 rd Generation MOX Fuel	4 th Generation MOX Fuel	UO ₂ Fuel	1 st Generation MOX Fuel	2 nd Generation MOX Fuel	3 rd Generation MOX Fuel	4 th Generation MOX Fuel
Uranium										
U-234	214	211	219	227	233	0.02	0.02	0.02	0.02	0.03
U-235	7,610	6,780	6,280	6,100	5,990	0.83	0.74	0.69	0.67	0.66
U-236	7,114	6,400	6,070	5,930	5,850	0.78	0.70	0.67	0.66	0.65
U-238	903,758	898,000	895,000	893,000	891,000	98.37	98.54	98.62	98.65	98.66
	918,696	911,391	907,569	905,257	903,073	100.00	100.00	100.00	100.00	100.00
Plutonium										
Pu-238	475	831	1,090	1,260	1,380	3.78	4.42	4.85	5.10	5.29
Pu-239	6,424	8,310	9,280	9,870	10,200	51.14	44.20	41.34	39.96	39.09
Pu-240	3,072	5,040	6,070	6,660	7,020	24.45	26.81	27.04	26.96	26.91
Pu-241	1,532	2,450	2,940	3,210	3,370	12.2	13.03	13.10	13.00	12.92
Pu-242	1,059	2,170	3,070	3,700	4,120	8.43	11.54	13.67	14.98	15.79
	12,562	18,801	22,450	24,700	26,090	100.00	100.00	100.00	100.00	100.00
Minor Actinides										
Np	942	922	910	908	909	49.79	32.96	26.19	22.93	21.33
Am	811	1,494	1,969	2,290	2,480	42.86	53.42	56.66	57.83	58.19
Cm	139	381	596	762	873	7.35	13.62	17.15	19.24	20.48
	1,892	2,797	3,475	3,960	4,262	100.00	100.00	100.00	100.00	100.00
TOTAL	933,150	932,989	933,494	933,917	933,425					

^aAssumes UO₂ and MOX fuel are reprocessed together to dilute impact of changes in non-fissile actinides.

^b60,000 MWD/MTIHM burnup @ 40 MW/MTIHM - 5 yrs. after discharge from LWR

TABLE C-2
ACTINIDE CONCENTRATIONS IN SUCCESSIVE GENERATIONS OF SPENT UO₂ AND MOX FUEL FROM LWRs^{a,b}
SCENARIO #2 -- U+PU MIX FROM REPROCESSING BLENDED W/ERU
TO PRODUCE MOX (5.0% EQUIVALENT FISSILE)

Category	Grams Per Metric Ton Initial Heavy Metal					Percent Abundance in Category				
	UO ₂ Fuel	1 st Generation MOX Fuel	2 nd Generation MOX Fuel	3 rd Generation MOX Fuel	4 th Generation MOX Fuel	UO ₂ Fuel	1 st Generation MOX Fuel	2 nd Generation MOX Fuel	3 rd Generation MOX Fuel	4 th Generation MOX Fuel
Uranium										
U-234	214	289	349	392	420	0.02	0.03	0.04	0.04	0.05
U-235	7,610	7,610	7,780	7,670	7,530	0.83	0.83	0.86	0.85	0.83
U-236	7,114	9,260	10,100	10,400	10,400	0.78	1.02	1.11	1.15	1.15
U-238	903,758	895,000	889,000	886,000	884,000	98.37	98.12	97.99	97.96	97.97
	918,696	912,159	907,229	904,462	902,350	100.00	100.00	100.00	100.00	100.00
Plutonium										
Pu-238	475	1,130	1,670	2,070	2,320	3.78	6.22	7.59	8.46	8.92
Pu-239	6,424	7,740	8,760	9,380	9,780	51.14	42.60	39.83	38.32	37.60
Pu-240	3,072	4,750	5,710	6,290	6,640	24.45	26.14	25.95	25.69	25.53
Pu-241	1,532	2,310	2,760	3,030	3,180	12.20	12.71	12.54	12.38	12.23
Pu-242	1,059	2,240	3,100	3,710	4,090	8.43	12.33	14.09	15.15	15.72
	12,562	18,170	22,000	24,480	26,010	100.00	100.00	100.00	100.00	100.00
Minor Actinides										
Np	942	1,600	1,880	2,020	2,070	49.79	46.14	42.68	40.26	38.54
Am	811	1,465	1,924	2,230	2,420	42.86	42.26	43.68	44.44	45.06
Cm	139	402	601	768	881	7.35	11.60	13.64	15.30	16.40
	1,892	3,467	4,405	5,018	5,371	100.00	100.00	100.00	100.00	100.00
TOTAL	933,150	933,796	933,634	933,960	933,731					

^aAssumes UO₂ and MOX fuel are reprocessed together to dilute impact of changes in non-fissile actinides.

^b60,000 MWD/MTIHM burnup @ 40 MW/MTIHM - 5 yrs. after discharge from LWR

TABLE C-3
ACTINIDE CONCENTRATIONS IN SUCCESSIVE GENERATIONS OF SPENT UO₂ AND MOX FUEL FROM LWRs^{a,b}
SCENARIO #3 -- U+PU MIX FROM REPROCESSING BLENDED W/RU
TO PRODUCE MOX (5.0% EQUIVALENT FISSILE)

Category	Grams Per Metric Ton Initial Heavy Metal					Percent Abundance in Category				
	UO ₂ Fuel	1 st Generation MOX Fuel	2 nd Generation MOX Fuel	3 rd Generation MOX Fuel	4 th Generation MOX Fuel	UO ₂ Fuel	1 st Generation MOX Fuel	2 nd Generation MOX Fuel	3 rd Generation MOX Fuel	4 th Generation MOX Fuel
Uranium										
U-234	214	223	238	251	258	0.02	0.02	0.03	0.03	0.03
U-235	7,610	6,780	6,480	6,320	6,220	0.83	0.74	0.71	0.70	0.69
U-236	7,114	6,900	6,760	6,670	6,620	0.78	0.76	0.74	0.73	0.73
U-238	903,758	898,005	894,000	892,000	890,000	98.37	98.48	98.52	98.54	98.55
	918,696	911,908	907,478	905,241	903,098	100.00	100.00	100.00	100.00	100.00
Plutonium										
Pu-238	475	897	1,210	1,420	1,570	3.78	4.79	5.39	5.74	5.98
Pu-239	6,424	8,220	9,220	9,830	10,200	51.14	43.89	41.07	39.70	38.89
Pu-240	3,072	5,000	6,030	6,620	6,990	24.45	26.7	26.86	26.74	26.65
Pu-241	1,532	2,430	2,920	3,190	3,350	12.20	12.98	13.01	12.88	12.77
Pu-242	1,059	2,180	3,070	3,700	4,120	8.43	11.64	13.67	14.94	15.71
	12,562	18,727	22,450	24,760	26,230	100.00	100.00	100.00	100.00	100.00
Minor Actinides										
Np	942	1,060	1,090	1,110	1,110	49.79	36.19	29.89	26.76	24.90
Am	811	1,492	1,961	2,280	2,480	42.86	50.94	53.77	54.97	55.63
Cm	139	377	596	758	868	7.35	12.87	16.34	18.27	19.47
	1,892	2,929	3,647	4,148	4,458	100.00	100.00	100.00	100.00	100.00
TOTAL	933,150	933,564	933,575	934,149	933,786					

^a Assumes UO₂ and MOX fuel are reprocessed together to dilute impact of changes in non-fissile actinides.

^b 60,000 MWD/MTIHM burnup @ 40 MW/MTIHM - 5 yrs. after discharge from LWR

TABLE C-4
ACTINIDE CONCENTRATIONS IN SUCCESSIVE GENERATIONS OF SPENT UO₂ AND MOX FUEL FROM LWRs^{a,b}
SCENARIO #4 -- U+PU MIX FROM REPROCESSING BLENDED W/NU
TO PRODUCE MOX (5.0% EQUIVALENT FISSILE)

Category	Grams Per Metric Ton Initial Heavy Metal					Percent Abundance in Category				
	UO ₂ Fuel	1 st Generation MOX Fuel	2 nd Generation MOX Fuel	3 rd Generation MOX Fuel	4 th Generation MOX Fuel	UO ₂ Fuel	1 st Generation MOX Fuel	2 nd Generation MOX Fuel	3 rd Generation MOX Fuel	4 th Generation MOX Fuel
Uranium										
U-234	214	211	219	227	233	0.02	0.02	0.02	0.02	0.03
U-235	7,610	6,720	6,430	6,270	6,180	0.83	0.74	0.71	0.69	0.68
U-236	7,114	6,330	6,070	5,940	5,860	0.78	0.69	0.67	0.66	0.65
U-238	903,758	899,000	895,000	893,000	891,000	98.37	98.55	98.60	98.63	98.64
	918,696	912,261	907,719	905,437	903,273	100.00	100.00	100.00	100.00	100.00
Plutonium										
Pu-238	475	823	1,080	1,250	1,360	3.78	4.42	4.87	5.13	5.27
Pu-239	6,424	8,200	9,150	9,720	10,100	51.14	44.03	41.23	39.87	39.18
Pu-240	3,072	4,990	5,980	6,550	6,890	24.45	26.79	26.95	26.87	26.73
Pu-241	1,532	2,430	2,900	3,160	3,320	12.2	13.05	13.07	12.96	12.88
Pu-242	1,059	2,180	3,080	3,700	4,110	8.43	11.71	13.88	15.17	15.94
	12,562	18,623	22,190	24,380	25,780	100.00	100.00	100.00	100.00	100.00
Minor Actinides										
Np	942	916	910	910	911	49.79	32.66	26.18	23.00	21.35
Am	811	1,493	1,958	2,270	2,470	42.86	53.23	56.33	57.40	57.87
Cm	139	396	608	775	887	7.35	14.11	17.49	19.60	20.78
	1,892	2,805	3,476	3,955	4,268	100.00	100.00	100.00	100.00	100.00
TOTAL	933,150	933,689	933,385	933,772	933,321					

^aAssumes UO₂ and MOX fuel are reprocessed together to dilute impact of changes in non-fissile actinides.

^b60,000 MWD/MTIHM burnup @ 40 MW/MTIHM - 5 yrs. after discharge from LWR

TABLE C-5
ACTINIDE CONCENTRATIONS IN SUCCESSIVE GENERATIONS OF SPENT UO₂ AND MOX FUEL FROM LWRs^{a,b}
SCENARIO #5 -- U+PU MIX FROM REPROCESSING BLENDED W/RU
TO PRODUCE MOX (5.0% EQUIVALENT FISSILE) WITH EXCESS RU ENRICHED FOR USE IN UO₂ FUEL

Category	Grams Per Metric Ton Initial Heavy Metal					Percent Abundance in Category				
	UO ₂ Fuel	1 st Generation MOX Fuel	2 nd Generation MOX Fuel	3 rd Generation MOX Fuel	4 th Generation MOX Fuel	UO ₂ Fuel	1 st Generation MOX Fuel	2 nd Generation MOX Fuel	3 rd Generation MOX Fuel	4 th Generation MOX Fuel
Uranium										
U-234	214	281	335			0.02	0.03	0.04		
U-235	7,610	7,490	7,190			0.83	0.82	0.79		
U-236	7,114	9,300	9,800			0.78	1.02	1.08		
U-238	903,758	894,000	889,000			98.37	98.13	98.09		
	918,696	911,071	906,325			100.00	100.00	100.00	0.00	0.00
Plutonium										
Pu-238	475	1,080	1,540			3.78	5.71	6.77		
Pu-239	6,424	8,300	9,310			51.14	43.89	40.91		
Pu-240	3,072	4,960	5,980			24.45	26.23	26.27		
Pu-241	1,532	2,430	2,910			12.2	12.85	12.78		
Pu-242	1,059	2,140	3,020			8.43	11.32	13.27		
	12,562	18,910	22,760			100.00	100.00	100.00	0.00	0.00
Minor Actinides										
Np	942	1,380	1,530			49.79	42.53	37.74		
Am	811	1,478	1,934			42.86	45.55	47.71		
Cm	139	387	590			7.35	11.92	14.55		
	1,892	3,245	4,054			100.00	100.00	100.00	0.00	0.00
TOTAL	933,150	933,226	933,139							

^aAssumes UO₂ and MOX fuel are reprocessed together to dilute impact of changes in non-fissile actinides.

^b60,000 MWD/MTIHM burnup @ 40 MW/MTIHM - 5 yrs. after discharge from LWR

TABLE C-6
MAJOR FISSION PRODUCT CONCENTRATIONS IN SUCCESSIVE GENERATIONS OF
SPENT UO₂ AND MOX FUEL FROM LWRs^{a,b}
SCENARIO #1 -- U + PU MIX FROM REPROCESSING BLENDED W/DU
TO PRODUCE MOX (5.0% EQUIVALENT FISSILE)

	Grams Per Metric Ton Initial Heavy Metal				
	UO ₂ Fuel	1 st Generation MOX Fuel	2 nd Generation MOX Fuel	3 rd Generation MOX Fuel	4 th Generation MOX Fuel
Technetium	1,350	1,340	1,330	1,330	1,330
Iodine	355	377	383	386	387
Krypton	633	587	567	558	552
Total	2,338	2,304	2,280	2,274	2,269
Strontium	1,429	1,314	1,267	1,244	1,231
Cesium	4,691	4,786	4,821	4,837	4,853
Barium	3,118	3,077	3,067	3,057	3,046
Rubidium	622	573	552	544	538
Total	9,860	9,750	9,707	9,682	9,668
Lanthanum	2,200	2,160	2,140	2,130	2,130
Cerium	4,325	4,255	4,225	4,205	4,205
Praseodymium	2,010	1,970	1,960	1,950	1,940
Neodinium	7,332	7,156	7,086	7,048	7,029
Promethium	61	61	61	61	61
Samarium	1,396	1,418	1,425	1,429	1,432
Europium	255	270	276	278	281
Gadolinium	260	266	269	270	271
Total	17,839	17,556	17,442	17,371	17,349
Zirconium	6,464	6,075	5,914	5,834	5,782
Molybdenum	6,069	5,963	5,901	5,880	5,860
Ruthenium	4,196	4,290	4,328	4,348	4,347
Palladium	2,738	3,071	3,201	3,260	3,292
Xenon	9,653	9,504	9,416	9,357	9,327
Total	29,120	28,903	28,760	28,679	28,608
Other	2,780	2,869	2,897	2,910	2,916
GRAND TOTAL	61,937	61,382	61,086	60,916	60,810

^aAssumes UO₂ and MOX fuel are reprocessed together

^b60,000 MWD/MTIHM burnup @ 40 MW/MTIHM - 5 yrs. after discharge

TABLE C-7
MAJOR FISSION PRODUCT CONCENTRATIONS IN SUCCESSIVE GENERATIONS OF
SPENT UO₂ AND MOX FUEL FROM LWRs^{a,b}

*SCENARIO #2 -- U +PU MIX FROM REPROCESSING BLENDED W/ERU
 TO PRODUCE MOX (5.0% EQUIVALENT FISSILE)*

	Grams Per Metric Ton Initial Heavy Metal				
	UO ₂ Fuel	1 st Generation MOX Fuel	2 nd Generation MOX Fuel	3 rd Generation MOX Fuel	4 th Generation MOX Fuel
Technetium	1,350	1,340	1,340	1,330	1,330
Iodine	355	380	382	386	387
Krypton	633	582	570	559	553
Total	2,338	2,302	2,292	2,275	2,270
Strontium	1,429	1,309	1,280	1,254	1,238
Cesium	4,691	4,780	4,819	4,837	4,853
Barium	3,118	3,071	3,049	3,037	3,028
Rubidium	622	569	556	545	539
Total	9,860	9,729	9,704	9,673	9,658
Lanthanum	2,200	2,160	2,140	2,130	2,120
Cerium	4,325	4,255	4,235	4,205	4,195
Praseodymium	2,010	1,970	1,960	1,950	1,940
Neodinium	7,332	7,145	7,094	7,043	7,002
Promethium	61	62	62	62	62
Samarium	1,396	1,420	1,424	1,428	1,427
Europium	255	270	273	276	277
Gadolinium	260	266	263	264	264
Total	17,839	17,548	17,451	17,358	17,287
Zirconium	6,464	6,024	5,903	5,809	5,787
Molybdenum	6,069	5,952	5,890	5,859	5,838
Ruthenium	4,196	4,286	4,292	4,301	4,312
Palladium	2,738	3,086	3,147	3,210	3,247
Xenon	9,653	9,476	9,378	9,319	9,279
Total	29,120	28,824	28,610	28,498	28,463
Other	2,780	2,870	2,886	2,899	2,908
GRAND TOTAL	61,937	61,273	60,943	60,703	60,586

^aAssumes UO₂ and MOX fuel are reprocessed together

^b60,000 MWD/MTIHM burnup @ 40 MW/MTIHM - 5 yrs. after discharge

TABLE C-8
MAJOR FISSION PRODUCT CONCENTRATIONS IN SUCCESSIVE GENERATIONS OF
SPENT UO₂ AND MOX FUEL FROM LWRs^{a,b}
SCENARIO #3 -- U + PU MIX FROM REPROCESSING BLENDED W/RU
TO PRODUCE MOX (5.0% EQUIVALENT FISSILE)

	Grams Per Metric Ton Initial Heavy Metal				
	UO ₂ Fuel	1 st Generation MOX Fuel	2 nd Generation MOX Fuel	3 rd Generation MOX Fuel	4 th Generation MOX Fuel
Technetium	1,350	1,340	1,330	1,330	1,330
Iodine	355	377	383	385	387
Krypton	633	585	568	559	553
Total	2,338	2,302	2,281	2,274	2,270
Strontium	1,429	1,310	1,269	1,246	1,232
Cesium	4,691	4,786	4,811	4,838	4,844
Barium	3,118	3,077	3,056	3,055	3,045
Rubidium	622	571	554	544	538
Total	9,860	9,744	9,690	9,683	9,659
Lanthanum	2,200	2,160	2,140	2,130	2,130
Cerium	4,325	4,255	4,225	4,205	4,205
Praseodymium	2,010	1,970	1,960	1,950	1,940
Neodinium	7,332	7,156	7,088	7,037	7,026
Promethium	61	61	61	61	61
Samarium	1,396	1,418	1,424	1,428	1,430
Europium	255	270	275	278	279
Gadolinium	260	268	268	269	269
Total	17,839	17,558	17,441	17,358	17,340
Zirconium	6,464	6,061	5,914	5,844	5,782
Molybdenum	6,069	5,953	5,901	5,880	5,859
Ruthenium	4,196	4,300	4,327	4,337	4,336
Palladium	2,738	3,084	3,192	3,249	3,280
Xenon	9,653	9,503	9,406	9,347	9,317
Total	29,120	28,901	28,740	28,657	28,574
Other	2,780	2,866	2,892	2,910	2,912
GRAND TOTAL	61,937	61,371	61,044	60,882	60,755

^aAssumes UO₂ and MOX fuel are reprocessed together

^b60,000 MWD/MTIHM burnup @ 40 MW/MTIHM - 5 yrs. after discharge

TABLE C-9
MAJOR FISSION PRODUCT CONCENTRATIONS IN SUCCESSIVE GENERATIONS OF
SPENT UO₂ AND MOX FUEL FROM LWRs^{a,b}

*SCENARIO #4 -- U +PU MIX FROM REPROCESSING BLENDED W/NU
 TO PRODUCE MOX (5.0% EQUIVALENT FISSILE)*

	Grams Per Metric Ton Initial Heavy Metal				
	UO ₂ Fuel	1 st Generation MOX Fuel	2 nd Generation MOX Fuel	3 rd Generation MOX Fuel	4 th Generation MOX Fuel
Technetium	1,350	1,340	1,330	1,330	1,330
Iodine	355	378	383	386	388
Krypton	633	585	568	558	552
Total	2,338	2,303	2,281	2,274	2,270
Strontium	1,429	1,309	1,266	1,244	1,231
Cesium	4,691	4,786	4,820	4,836	4,851
Barium	3,118	3,078	3,066	3,056	3,046
Rubidium	622	571	552	542	537
Total	9,860	9,744	9,704	9,678	9,665
Lanthanum	2,200	2,160	2,140	2,130	2,130
Cerium	4,325	4,255	4,225	4,205	4,205
Praseodymium	2,010	1,970	1,960	1,950	1,940
Neodinium	7,332	7,156	7,088	7,038	7,029
Promethium	61	61	61	61	61
Samarium	1,396	1,418	1,426	1,429	1,432
Europium	255	270	275	279	279
Gadolinium	260	269	269	270	271
Total	17,839	17,559	17,444	17,362	17,347
Zirconium	6,464	6,050	5,902	5,822	5,780
Molybdenum	6,069	5,953	5,901	5,880	5,860
Ruthenium	4,196	4,300	4,338	4,347	4,347
Palladium	2,738	3,093	3,205	3,262	3,295
Xenon	9,653	9,503	9,416	9,357	9,327
Total	29,120	28,899	28,762	28,668	28,609
Other	2,780	2,869	2,897	2,910	2,916
GRAND TOTAL	61,937	61,374	61,088	60,892	60,807

^aAssumes UO₂ and MOX fuel are reprocessed together

^b60,000 MWD/MTIHM burnup @ 40 MW/MTIHM - 5 yrs. after discharge

TABLE C-10
MAJOR FISSION PRODUCT CONCENTRATIONS IN SUCCESSIVE GENERATIONS OF
SPENT UO₂ AND MOX FUEL FROM LWRs^{a,b}

*SCENARIO #5 -- U +PU MIX FROM REPROCESSING BLENDED WITH RU TO PRODUCE MOX
 (5% EQUIVALENT FISSILE) WITH EXCESS RU ENRICHED FOR USE IN UO₂ FUEL*

	Grams Per Metric Ton Initial Heavy Metal		
	UO ₂ Fuel	1 st Generation MOX Fuel	2 nd Generation MOX Fuel
Technetium	1,350	1,340	1,340
Iodine	355	376	382
Krypton	633	591	572
Total	2,338	2,307	2,294
Strontium	1,429	1,324	1,280
Cesium	4,691	4,811	4,837
Barium	3,118	3,073	3,060
Rubidium	622	577	558
Total	9,860	9,785	9,735
Lanthanum	2,200	2,160	2,140
Cerium	4,325	4,265	4,235
Praseodymium	2,010	1,980	1,960
Neodinium	7,332	7,164	7,094
Promethium	61	62	62
Samarium	1,396	1,418	1,425
Europium	255	268	273
Gadolinium	260	259	261
Total	17,839	17,576	17,450
Zirconium	6,464	6,081	5,941
Molybdenum	6,069	5,961	5,899
Ruthenium	4,196	4,266	4,303
Palladium	2,738	3,030	3,147
Xenon	9,653	9,468	9,381
Total	29,120	28,806	28,671
Other	2,780	2,868	2,894
GRAND TOTAL	61,937	61,342	61,044

^aAssumes UO₂ and MOX fuel are reprocessed together

^b60,000 MWD/MTIHM burnup @ 40 MW/MTIHM - 5 yrs. after discharge

TABLE C-11
FISSION PRODUCT CONCENTRATIONS IN SUCCESSIVE GENERATIONS OF
SPENT UO₂ AND MOX FUEL FROM LWRs^{a,b}
SCENARIO #1 -- U + PU MIX FROM REPROCESSING BLENDED W/DU
TO PRODUCE MOX (5.0% EQUIVALENT FISSILE)

	Grams Per Metric Ton Initial Heavy Metal				
	UO ₂ Fuel	1 st Generation	2 nd Generation	3 rd Generation	4 th Generation
		MOX Fuel	MOX Fuel	MOX Fuel	MOX Fuel
Br-81	37	31	28	27	27
Se-82	59	56	55	54	53
Kr-83	66	64	62	62	62
Kr-84	213	198	192	189	187
Kr-85	31	26	24	23	22
Rb-85	186	171	165	163	161
Kr-86	323	299	289	284	281
Rb-87	436	402	387	381	377
Sr-88	601	554	535	525	520
Y-89	803	737	709	696	688
Sr-90	828	760	732	719	711
Zr-90	164	136	127	122	119
Zr-91	1,060	979	947	932	923
Zr-92	1,150	1,080	1,050	1,030	1,030
Zr-93	1,250	1,140	1,100	1,080	1,060
Zr-94	1,390	1,330	1,300	1,290	1,280
Mo-95	1,320	1,280	1,260	1,250	1,240
Zr-96	1,450	1,410	1,390	1,380	1,370
Mo-96	109	103	101	100	100
Mo-97	1,450	1,420	1,400	1,390	1,380
Mo-98	1,490	1,470	1,460	1,460	1,460
Tc-99	1,350	1,340	1,330	1,330	1,330
Mo-100	1,700	1,690	1,680	1,680	1,680
Ru-100	278	271	269	268	267
Ru-101	1,390	1,390	1,390	1,390	1,380
Ru-102	1,490	1,510	1,520	1,520	1,520
Rh-103	742	815	843	857	865
Ru-104	1,030	1,110	1,140	1,160	1,170
Pd-104	542	534	528	523	520
Pd-105	724	812	846	862	871
Ru-106	8	9	9	10	10
Pd-106	699	800	841	859	870
Pd-107	413	493	525	540	549
Pd-108	270	325	348	359	364
Ag-109	136	161	170	174	176
Pd-110	90	107	113	117	118
Cd-110	89	123	137	144	148
Cd-111	43	53	57	59	60
Cd-112		8	11	12	13
Cd-114		7	9	10	11
Sn-126	34	37	38	38	38
I-127	81	85	86	87	87
Te-128	157	163	165	165	165
I-129	274	292	297	299	300

	Grams Per Metric Ton Initial Heavy Metal				
	UO ₂ Fuel	1 st Generation	2 nd Generation	3 rd Generation	4 th Generation
		MOX Fuel	MOX Fuel	MOX Fuel	MOX Fuel
Te-130	680	678	675	673	672
Xe-131	633	644	646	647	647
Xe-132	2,130	2,130	2,120	2,110	2,110
Cs-133	1,930	1,920	1,910	1,910	1,910
Xe-134	2,770	2,740	2,720	2,710	2,700
Cs-134	49	48	48	48	48
Ba-134	346	342	341	340	339
Cs-135	762	868	913	939	955
Xe-136	4,120	3,990	3,930	3,890	3,870
Ba-136	68	70	72	73	73
Cs-137	1,950	1,950	1,950	1,940	1,940
Ba-137	344	345	344	344	344
Ba-138	2,360	2,320	2,310	2,300	2,290
La-139	2,200	2,160	2,140	2,130	2,130
Ce-140	2,300	2,270	2,260	2,250	2,250
Pr-141	2,010	1,970	1,960	1,950	1,940
Ce-142	2,020	1,980	1,960	1,950	1,950
Nd-142	55	51	48	49	49
Nd-143	1,280	1,290	1,300	1,300	1,300
Ce-144	5	5	5	5	5
Nd-144	2,530	2,400	2,340	2,320	2,300
Nd-145	1,140	1,120	1,110	1,100	1,100
Nd-146	1,340	1,300	1,290	1,280	1,280
Pm-147	61	61	61	61	61
Sm-147	289	290	290	290	290
Nd-148	663	660	659	658	658
Sm-148	294	297	298	299	300
Nd-150	324	335	339	341	342
Sm-150	545	547	546	546	545
Sm-151		8	12	14	16
Sm-152	201	201	201	201	201
Eu-153	220	230	233	234	235
Sm-154	67	75	78	79	80
Eu-154	35	40	43	44	46
Gd-156	216	218	219	219	219
Gd-158	44	48	50	51	52
Total	61,937	61,382	61,086	60,915	60,810
Rounding & Minor Isotopes	263	18	14	(15)	90
TOTAL	62,200	61,400	61,100	60,900	60,900

Downward or Flat Trend

Upward Trend

^aAssumes UO₂ and MOX fuel are reprocessed together

^b60,000 MWD/MTIHM burnup @ 40 MW/MTIHM - 5 yrs. after discharge

TABLE C-12
FISSION PRODUCT CONCENTRATIONS IN SUCCESSIVE GENERATIONS OF
SPENT UO₂ AND MOX FUEL FROM LWRs^{a,b}
SCENARIO #2 -- U +PU MIX FROM REPROCESSING BLENDED W/ERU
TO PRODUCE MOX (5.0% EQUIVALENT FISSILE)

	Grams Per Metric Ton Initial Heavy Metal				
	UO ₂ Fuel	1 st	2 nd	3 rd	4 th
		Generation MOX Fuel	Generation MOX Fuel	Generation MOX Fuel	Generation MOX Fuel
Br-81	37	28	27	25	25
Se-82	59	56	55	54	53
Kr-83	66	64	63	63	62
Kr-84	213	197	193	189	187
Kr-85	31	23	22	21	21
Rb-85	186	170	166	163	161
Kr-86	323	298	292	286	283
Rb-87	436	399	390	382	378
Sr-88	601	552	540	529	523
Y-89	803	734	717	702	693
Sr-90	828	757	740	725	715
Zr-90	164	149	118	112	141
Zr-91	1,060	975	955	937	926
Zr-92	1,150	1,070	1,050	1,040	1,030
Zr-93	1,250	1,110	1,080	1,050	1,040
Zr-94	1,390	1,320	1,310	1,290	1,280
Mo-95	1,320	1,280	1,260	1,250	1,240
Zr-96	1,450	1,400	1,390	1,380	1,370
Mo-96	109	102	100	99	98
Mo-97	1,450	1,410	1,390	1,380	1,380
Mo-98	1,490	1,470	1,460	1,460	1,450
Tc-99	1,350	1,340	1,340	1,330	1,330
Mo-100	1,700	1,690	1,680	1,670	1,670
Ru-100	278	267	263	262	262
Ru-101	1,390	1,390	1,380	1,380	1,380
Ru-102	1,490	1,510	1,510	1,510	1,510
Rh-103	742	817	839	854	862
Ru-104	1,030	1,110	1,130	1,140	1,150
Pd-104	542	530	518	513	510
Pd-105	724	815	834	851	861
Ru-106	8	9	9	9	10
Pd-106	699	804	824	844	856
Pd-107	413	499	517	533	542
Pd-108	270	331	343	354	361
Ag-109	136	164	169	173	176
Pd-110	90	107	111	115	117
Cd-110	89	124	131	138	143
Cd-111	43	53	56	58	59
Cd-112		10	11	13	14
Cd-114		9	10	11	12
Sn-126	34	37	37	38	38
I-127	81	85	85	86	86
Te-128	157	163	163	164	165
I-129	274	295	297	300	301

	Grams Per Metric Ton Initial Heavy Metal				
	UO ₂ Fuel	1 st Generation	2 nd Generation	3 rd Generation	4 th Generation
		MOX Fuel	MOX Fuel	MOX Fuel	MOX Fuel
Te-130	680	675	671	669	668
Xe-131	633	646	648	649	649
Xe-132	2,130	2,120	2,100	2,100	2,100
Cs-133	1,930	1,920	1,920	1,910	1,910
Xe-134	2,770	2,730	2,720	2,700	2,690
Cs-134	49	48	47	47	47
Ba-134	346	338	335	334	334
Cs-135	762	862	912	940	956
Xe-136	4,120	3,980	3,910	3,870	3,840
Ba-136	68	68	70	70	71
Cs-137	1,950	1,950	1,940	1,940	1,940
Ba-137	344	345	344	343	343
Ba-138	2,360	2,320	2,300	2,290	2,280
La-139	2,200	2,160	2,140	2,130	2,120
Ce-140	2,300	2,270	2,260	2,250	2,240
Pr-141	2,010	1,970	1,960	1,950	1,940
Ce-142	2,020	1,980	1,970	1,950	1,950
Nd-142	55	50	49	48	47
Nd-143	1,280	1,300	1,310	1,310	1,310
Ce-144	5	5	5	5	5
Nd-144	2,530	2,380	2,340	2,300	2,280
Nd-145	1,140	1,120	1,110	1,110	1,100
Nd-146	1,340	1,300	1,290	1,280	1,270
Pm-147	61	62	62	62	62
Sm-147	289	293	294	294	293
Nd-148	663	660	658	656	655
Sm-148	294	295	296	297	297
Nd-150	324	335	337	339	340
Sm-150	545	548	545	544	543
Sm-151		9	12	15	16
Sm-152	201	200	200	200	199
Eu-153	220	231	232	233	234
Sm-154	67	75	77	78	79
Eu-154	35	39	41	43	43
Gd-156	216	218	214	214	214
Gd-158	44	48	49	50	50
Total	61,937	61,273	60,943	60,703	60,586
Rounding & Minor Isotopes	263	27	(43)	(3)	14
TOTAL	62,200	61,300	60,900	60,700	60,600

	Downward or Flat Trend
	Upward Trend

^aAssumes UO₂ and MOX fuel are reprocessed together

^b60,000 MWD/MTIHM burnup @ 40 MW/MTIHM - 5 yrs. after discharge

TABLE C-13
FISSION PRODUCT CONCENTRATIONS IN SUCCESSIVE GENERATIONS OF
SPENT UO₂ AND MOX FUEL FROM LWRs^{a,b}
SCENARIO #3 -- U +PU MIX FROM REPROCESSING BLENDED W/RU
TO PRODUCE MOX (5.0% EQUIVALENT FISSILE)

	Grams Per Metric Ton Initial Heavy Metal				
	UO ₂ Fuel	1 st Generation	2 nd Generation	3 rd Generation	4 th Generation
		MOX Fuel	MOX Fuel	MOX Fuel	MOX Fuel
Br-81	37	30	28	27	26
Se-82	59	56	55	54	53
Kr-83	66	64	63	62	62
Kr-84	213	198	192	189	187
Kr-85	31	25	24	23	22
Rb-85	186	171	166	163	161
Kr-86	323	298	289	285	282
Rb-87	436	400	388	381	377
Sr-88	601	552	535	526	520
Y-89	803	734	710	697	689
Sr-90	828	758	734	720	712
Zr-90	164	135	126	121	118
Zr-91	1,060	976	948	933	924
Zr-92	1,150	1,080	1,050	1,040	1,030
Zr-93	1,250	1,140	1,100	1,080	1,060
Zr-94	1,390	1,330	1,300	1,290	1,280
Mo-95	1,320	1,280	1,260	1,250	1,240
Zr-96	1,450	1,400	1,390	1,380	1,370
Mo-96	109	103	101	100	99
Mo-97	1,450	1,410	1,400	1,390	1,380
Mo-98	1,490	1,470	1,460	1,460	1,460
Tc-99	1,350	1,340	1,330	1,330	1,330
Mo-100	1,700	1,690	1,680	1,680	1,680
Ru-100	278	271	268	267	266
Ru-101	1,390	1,390	1,390	1,380	1,380
Ru-102	1,490	1,510	1,520	1,520	1,520
Rh-103	742	816	842	856	864
Ru-104	1,030	1,120	1,140	1,160	1,160
Pd-104	542	535	526	521	518
Pd-105	724	815	844	860	868
Ru-106	8	9	9	10	10
Pd-106	699	804	838	856	866
Pd-107	413	496	524	538	547
Pd-108	270	327	347	358	363
Ag-109	136	161	169	173	176
Pd-110	90	107	113	116	118
Cd-110	89	124	136	143	147
Cd-111	43	53	56	58	59
Cd-112		8	11	12	13
Cd-114		7	9	10	11
Sn-126	34	37	38	38	38
I-127	81	85	86	86	87
Te-128	157	163	164	165	165
I-129	274	292	297	299	300

	Grams Per Metric Ton Initial Heavy Metal				
		1 st	2 nd	3 rd	4 th
	UO ₂ Fuel	Generation MOX Fuel	Generation MOX Fuel	Generation MOX Fuel	Generation MOX Fuel
Te-130	680	677	674	673	671
Xe-131	633	643	646	647	647
Xe-132	2,130	2,130	2,120	2,110	2,110
Cs-133	1,930	1,920	1,910	1,910	1,900
Xe-134	2,770	2,740	2,720	2,710	2,700
Cs-134	49	48	48	48	48
Ba-134	346	342	340	339	338
Cs-135	762	866	913	940	956
Xe-136	4,120	3,990	3,920	3,880	3,860
Ba-136	68	70	72	72	73
Cs-137	1,950	1,950	1,940	1,940	1,940
Ba-137	344	345	344	344	344
Ba-138	2,360	2,320	2,300	2,300	2,290
La-139	2,200	2,160	2,140	2,130	2,130
Ce-140	2,300	2,270	2,260	2,250	2,250
Pr-141	2,010	1,970	1,960	1,950	1,940
Ce-142	2,020	1,980	1,960	1,950	1,950
Nd-142	55	51	50	49	48
Nd-143	1,280	1,290	1,300	1,300	1,300
Ce-144	5	5	5	5	5
Nd-144	2,530	2,390	2,340	2,310	2,300
Nd-145	1,140	1,120	1,110	1,100	1,100
Nd-146	1,340	1,300	1,290	1,280	1,280
Pm-147	61	61	61	61	61
Sm-147	289	290	290	290	290
Nd-148	663	660	659	658	657
Sm-148	294	297	298	299	299
Nd-150	324	335	339	340	341
Sm-150	545	547	546	545	545
Sm-151		8	12	14	16
Sm-152	201	201	201	201	200
Eu-153	220	230	233	234	235
Sm-154	67	75	77	79	80
Eu-154	35	40	42	44	44
Gd-156	216	219	218	218	218
Gd-158	44	49	50	51	51
Total	61,937	61,359	61,044	60,878	60,755
Rounding & Minor Isotopes	263	(59)	356	22	45
TOTAL	62,200	61,300	61,400	60,900	60,800

	Downward or Flat Trend
	Upward Trend

^aAssumes UO₂ and MOX fuel are reprocessed together

^b60,000 MWD/MTIHM burnup @ 40 MW/MTIHM - 5 yrs. after discharge

TABLE C-14
FISSION PRODUCT CONCENTRATIONS IN SUCCESSIVE GENERATIONS OF
SPENT UO₂ AND MOX FUEL FROM LWRs^{a,b}
SCENARIO #4 -- U +PU MIX FROM REPROCESSING BLENDED W/NU
TO PRODUCE MOX (5.0% EQUIVALENT FISSILE)

	Grams Per Metric Ton Initial Heavy Metal				
	UO ₂ Fuel	1 st Generation	2 nd Generation	3 rd Generation	4 th Generation
		MOX Fuel	MOX Fuel	MOX Fuel	MOX Fuel
Br-81	37	30	28	27	26
Se-82	59	56	55	54	53
Kr-83	66	64	63	62	62
Kr-84	213	198	192	189	187
Kr-85	31	25	24	23	22
Rb-85	186	171	165	162	161
Kr-86	323	298	289	284	281
Rb-87	436	400	387	380	376
Sr-88	601	552	534	525	520
Y-89	803	733	709	696	689
Sr-90	828	757	732	719	711
Zr-90	164	135	125	120	117
Zr-91	1,060	975	947	932	923
Zr-92	1,150	1,080	1,050	1,030	1,030
Zr-93	1,250	1,130	1,090	1,070	1,060
Zr-94	1,390	1,330	1,300	1,290	1,280
Mo-95	1,320	1,280	1,260	1,250	1,240
Zr-96	1,450	1,400	1,390	1,380	1,370
Mo-96	109	103	101	100	100
Mo-97	1,450	1,410	1,400	1,390	1,380
Mo-98	1,490	1,470	1,460	1,460	1,460
Tc-99	1,350	1,340	1,330	1,330	1,330
Mo-100	1,700	1,690	1,680	1,680	1,680
Ru-100	278	271	269	267	267
Ru-101	1,390	1,390	1,390	1,390	1,380
Ru-102	1,490	1,510	1,520	1,520	1,520
Rh-103	742	816	843	857	865
Ru-104	1,030	1,120	1,150	1,160	1,170
Pd-104	542	536	528	523	521
Pd-105	724	817	847	862	871
Ru-106	8	9	9	10	10
Pd-106	699	806	841	860	870
Pd-107	413	498	526	541	549
Pd-108	270	329	349	359	365
Ag-109	136	162	170	174	176
Pd-110	90	107	114	117	119
Cd-110	89	125	137	144	148
Cd-111	43	53	57	59	60
Cd-112		8	11	12	13
Cd-114		7	9	11	11
Sn-126	34	37	38	38	38
I-127	81	85	86	87	87
Te-128	157	164	165	165	165
I-129	274	293	297	299	301

	Grams Per Metric Ton Initial Heavy Metal				
	UO ₂ Fuel	1 st Generation	2 nd Generation	3 rd Generation	4 th Generation
		MOX Fuel	MOX Fuel	MOX Fuel	MOX Fuel
Te-130	680	678	675	673	672
Xe-131	633	643	646	647	647
Xe-132	2,130	2,130	2,120	2,110	2,110
Cs-133	1,930	1,920	1,910	1,910	1,910
Xe-134	2,770	2,740	2,720	2,710	2,700
Cs-134	49	48	48	48	48
Ba-134	346	343	340	340	339
Cs-135	762	865	912	938	953
Xe-136	4,120	3,990	3,930	3,890	3,870
Ba-136	68	70	72	72	73
Cs-137	1,950	1,950	1,950	1,940	1,940
Ba-137	344	345	344	344	344
Ba-138	2,360	2,320	2,310	2,300	2,290
La-139	2,200	2,160	2,140	2,130	2,130
Ce-140	2,300	2,270	2,260	2,250	2,250
Pr-141	2,010	1,970	1,960	1,950	1,940
Ce-142	2,020	1,980	1,960	1,950	1,950
Nd-142	55	51	50	49	49
Nd-143	1,280	1,290	1,300	1,300	1,300
Ce-144	5	5	5	5	5
Nd-144	2,530	2,390	2,340	2,310	2,300
Nd-145	1,140	1,120	1,110	1,100	1,100
Nd-146	1,340	1,300	1,290	1,280	1,280
Pm-147	61	61	61	61	61
Sm-147	289	290	290	290	290
Nd-148	663	660	659	658	658
Sm-148	294	297	298	299	300
Nd-150	324	335	339	341	342
Sm-150	545	547	547	546	546
Sm-151		8	12	14	16
Sm-152	201	201	201	201	200
Eu-153	220	230	233	235	235
Sm-154	67	75	78	79	80
Eu-154	35	40	42	44	44
Gd-156	216	220	219	219	219
Gd-158	44	49	50	51	52
Total	61,937	61,361	61,088	60,892	60,807
Rounding & Minor Isotopes	263	39	(88)	8	93
TOTAL	62,200	61,400	61,000	60,900	60,900

Downward or Flat Trend
Upward Trend

^aAssumes UO₂ and MOX fuel are reprocessed together

^b60,000 MWD/MTIHM burnup @ 40 MW/MTIHM - 5 yrs. after discharge

TABLE C-15
FISSION PRODUCT CONCENTRATIONS IN SUCCESSIVE GENERATIONS OF
SPENT UO₂ AND MOX FUEL FROM LWRs^{a,b}
SCENARIO #5 -- U +PU MIX FROM REPROCESSING BLENDED W/RU TO PRODUCE
MOX (5.0% EQUIVALENT FISSION) WITH EXCESS RU ENRICHED FOR USE IN UO₂ FUEL

	Grams Per Metric Ton Initial Heavy Metal		
	UO ₂ Fuel	1 st Generation	2 nd Generation
		MOX Fuel	MOX Fuel
Br-81	37	30	28
Se-82	59	56	55
Kr-83	66	65	64
Kr-84	213	199	193
Kr-85	31	26	24
Rb-85	186	173	167
Kr-86	323	301	291
Rb-87	436	404	391
Sr-88	601	558	540
Y-89	803	742	716
Sr-90	828	766	740
Zr-90	164	136	126
Zr-91	1,060	985	955
Zr-92	1,150	1,080	1,060
Zr-93	1,250	1,140	1,100
Zr-94	1,390	1,330	1,310
Mo-95	1,320	1,280	1,260
Zr-96	1,450	1,410	1,390
Mo-96	109	101	99
Mo-97	1,450	1,420	1,400
Mo-98	1,490	1,470	1,460
Tc-99	1,350	1,340	1,340
Mo-100	1,700	1,690	1,680
Ru-100	278	267	264
Ru-101	1,390	1,390	1,390
Ru-102	1,490	1,500	1,510
Rh-103	742	817	844
Ru-104	1,030	1,100	1,130
Pd-104	542	523	515
Pd-105	724	806	837
Ru-106	8	9	9
Pd-106	699	788	825
Pd-107	413	487	517
Pd-108	270	321	342
Ag-109	136	159	167
Pd-110	90	105	111
Cd-110	89	121	134
Cd-111	43	52	56
Cd-112		8	11
Cd-114		7	9
Sn-126	34	37	37
I-127	81	85	86
Te-128	157	163	164
I-129	274	291	296

	Grams Per Metric Ton Initial Heavy Metal		
	UO ₂ Fuel	1 st Generation	2 nd Generation
		MOX Fuel	MOX Fuel
Te-130	680	676	673
Xe-131	633	648	651
Xe-132	2,130	2,120	2,110
Cs-133	1,930	1,930	1,920
Xe-134	2,770	2,740	2,720
Cs-134	49	48	47
Ba-134	346	338	335
Cs-135	762	883	930
Xe-136	4,120	3,960	3,900
Ba-136	68	70	71
Cs-137	1,950	1,950	1,940
Ba-137	344	345	344
Ba-138	2,360	2,320	2,310
La-139	2,200	2,160	2,140
Ce-140	2,300	2,270	2,260
Pr-141	2,010	1,980	1,960
Ce-142	2,020	1,990	1,970
Nd-142	55	50	48
Nd-143	1,280	1,320	1,320
Ce-144	5	5	5
Nd-144	2,530	2,380	2,330
Nd-145	1,140	1,120	1,110
Nd-146	1,340	1,300	1,290
Pm-147	61	62	62
Sm-147	289	295	295
Nd-148	663	660	659
Sm-148	294	296	297
Nd-150	324	334	337
Sm-150	545	545	544
Sm-151		8	12
Sm-152	201	200	200
Eu-153	220	228	231
Sm-154	67	74	77
Eu-154	35	40	42
Gd-156	216	212	212
Gd-158	44	47	49
Total	61,937	61,342	61,044
Rounding & Minor Isotopes	263	58	356
TOTAL	62,200	61,400	61,400

	Downward or Flat Trend
	Upward Trend

^aAssumes UO₂ and MOX fuel are reprocessed together

^b60,000 MWD/MTIHM burnup @ 40 MW/MTIHM - 5 yrs. after discharge

APPENDIX D

ORIGEN RESULTS FOR SPENT UO_2 AND MOX FUEL FROM LWRs
WHEN SUCCESSIVE GENERATIONS OF MOX ARE REPROCESSED AND
RECYCLED TOGETHER -- BUT SEPARATE FROM SPENT UO_2 FUEL

TABLE D-1
ACTINIDE CONCENTRATIONS OF SPENT UO₂ AND MOX FUEL FROM LWRs WHEN SUCCESSIVE
GENERATIONS OF MOX ARE REPROCESSED AND RECYCLED TOGETHER -- BUT SEPARATE
FROM SPENT UO₂ FUEL^{a,b}

*SCENARIO #1 -- U+PU MIX FROM REPROCESSING BLENDED W/DU
TO PRODUCE MOX (5.0% EQUIVALENT FISSILE)*

Category	UO ₂ Fuel (Grams HM/ MTIHM)	1 st Generation MOX Fuel (Grams HM/ 0.168 MTIHM)	2 nd Generation MOX Fuel (Grams HM/ 0.225 MTIHM)	Percent Abundance in Category		
				UO ₂ Fuel	1 st Generation MOX Fuel	2 nd Generation MOX Fuel
Uranium						
U-234	214	33	53	0.02	0.02	0.03
U-235	7,610	455	473	0.83	0.31	0.24
U-236	7,114	480	518	0.78	0.32	0.27
U-238	903,758	147,000	194,291	98.37	99.35	99.46
	918,696	147,968	195,335	100.00	100.00	100.00
Plutonium						
Pu-238	475	436	713	3.78	5.22	5.58
Pu-239	6,424	2,970	4,323	51.14	35.54	33.85
Pu-240	3,072	2,490	3,725	24.45	29.80	29.17
Pu-241	1,532	1,170	1,766	12.20	14.00	13.83
Pu-242	1,059	1,290	2,244	8.43	15.44	17.57
	12,562	8,356	12,771	100.00	100.00	100.00
Minor Actinides						
Np	942	139	162	49.79	11.35	8.16
Am	811	820	1,343	42.86	66.94	67.66
Cm	139	266	480	7.35	21.71	24.18
	1,892	1,225	1,985	100.00	100.00	100.00
TOTAL	933,150	157,549	210,091			

^a60,000 MWD/MTIHM burnup; 40 MW/MTIHM

^b5 yrs after discharge from LWRs

TABLE D-2
ACTINIDE CONCENTRATIONS IN SPENT UO₂ AND MOX FUEL FROM LWRs WHEN SUCCESSIVE
GENERATIONS OF MOX ARE REPROCESSED AND RECYCLED TOGETHER -- BUT SEPARATE
FROM SPENT UO₂ FUEL^{a,b}

SCENARIO #2 -- U+PU MIX FROM REPROCESSING BLENDED W/ERU
TO PRODUCE MOX (5.0% EQUIVALENT FISSILE)

Category	UO ₂ Fuel (Grams HM/ MTIHM)	1 st Generation MOX Fuel (Grams HM/ 0.244 MTIHM)	2 nd Generation MOX Fuel (Grams HM/ 0.276 MTIHM)	Percent Abundance in Category		
				UO ₂ Fuel	1 st Generation MOX Fuel	2 nd Generation MOX Fuel
Uranium						
U-234	214	128	196	0.02	0.06	0.08
U-235	7,610	1,860	2,102	0.83	0.85	0.87
U-236	7,114	3,880	4,994	0.78	1.78	2.06
U-238	903,758	212,000	234,663	98.37	97.31	96.99
	918,696	217,868	241,955	100.00	100.00	100.00
Plutonium						
Pu-238	475	754	1,341	3.78	8.71	10.34
Pu-239	6,424	2,890	4,112	51.14	33.40	31.71
Pu-240	3,072	2,420	3,505	24.45	27.96	27.03
Pu-241	1,532	1,150	1,658	12.20	13.29	12.79
Pu-242	1,059	1,440	2,351	8.43	16.64	18.13
	12,562	8,654	12,967	100.00	100.00	100.00
Minor Actinides						
Np	942	889	1,216	49.79	43.58	40.10
Am	811	853	1,331	42.86	41.81	43.90
Cm	139	298	485	7.35	14.61	16.00
	1,892	2,040	3,032	100.00	100.00	100.00
TOTAL	933,150	228,562	257,954			

^a60,000 MWD/MTIHM burnup; 40 MW/MTIHM

^b5 yrs after discharge from LWRs

TABLE D-3
MAJOR FISSION PRODUCT CONCENTRATIONS IN SPENT UO₂ AND MOX FUEL FROM LWRs
WHEN SUCCESSIVE GENERATIONS OF MOX ARE REPROCESSED AND RECYCLED TOGETHER --
BUT SEPARATE FROM SPENT UO₂ FUEL^{a,b}
SCENARIO #1 -- U +PU MIX FROM REPROCESSING BLENDED W/DU TO PRODUCE MOX
(5% EQUIVALENT FISSILE)

	Grams Per Metric Ton Initial Heavy Metal		
	UO ₂ Fuel	1 st Generation MOX Fuel	2 nd Generation MOX Fuel
Technetium	1,350	1,310	1,293
Iodine	355	488	476
Krypton	633	351	342
Total	2,338	2,149	2,111
Strontium	1,429	750	724
Cesium	4,691	5,298	5,284
Barium	3,118	2,905	2,871
Rubidium	622	333	324
Total	9,860	9,286	9,204
Lanthanum	2,200	1,982	1,960
Cerium	4,325	3,935	3,884
Praseodymium	2,010	1,815	1,787
Neodinium	7,332	6,327	6,244
Promethium	61	65	62
Samarium	1,396	1,524	1,520
Europium	255	345	342
Gadolinium	260	304	298
Total	17,839	16,298	16,098
Zirconium	6,464	4,190	4,076
Molybdenum	6,069	5,423	5,329
Ruthenium	4,196	4,810	4,782
Palladium	2,738	4,732	4,738
Xenon	9,653	8,732	8,547
Total	29,120	27,887	27,471
Other	2,780	3,310	3,280
GRAND TOTAL	61,937	58,929	58,164

^aAssumes UO₂ and MOX fuel are reprocessed separately

^b60,000 MWD/MTIHM burnup @ 40 MW/MTIHM - 5 yrs. after discharge

TABLE D-4
MAJOR FISSION PRODUCT CONCENTRATIONS IN SPENT UO₂ AND MOX FUEL FROM LWRs
WHEN SUCCESSIVE GENERATIONS OF MOX ARE REPROCESSED AND RECYCLED TOGETHER --
BUT SEPARATE FROM SPENT UO₂ FUEL^{a,b}
SCENARIO #2 -- U +PU MIX FROM REPROCESSING BLENDED W/ERU TO PRODUCE MOX
(5% EQUIVALENT FISSILE)

	Grams Per Metric Ton Initial Heavy Metal		
	UO ₂ Fuel	1 st Generation MOX Fuel	2 nd Generation MOX Fuel
Technetium	1,350	1,328	1,308
Iodine	355	459	453
Krypton	633	426	406
Total	2,338	2,213	2,167
Strontium	1,429	943	899
Cesium	4,691	5,090	5,145
Barium	3,118	2,922	2,888
Rubidium	622	410	388
Total	9,860	9,365	9,319
Lanthanum	2,200	2,033	2,000
Cerium	4,325	4,049	3,978
Praseodymium	2,010	1,861	1,826
Neodinium	7,332	6,574	6,457
Promethium	61	66	65
Samarium	1,396	1,492	1,496
Europium	255	320	322
Gadolinium	260	287	275
Total	17,839	16,680	16,420
Zirconium	6,464	4,721	4,540
Molybdenum	6,069	5,549	5,442
Ruthenium	4,196	4,578	4,558
Palladium	2,738	4,172	4,210
Xenon	9,653	8,934	8,681
Total	29,120	27,955	27,431
Other	2,780	3,152	3,156
GRAND TOTAL	61,937	59,365	58,493

^aAssumes UO₂ and MOX fuel are reprocessed separately

^b60,000 MWD/MTIHM burnup @ 40 MW/MTIHM - 5 yrs. after discharge

TABLE D-5
FISSION PRODUCT CONCENTRATIONS OF UO₂ AND MOX FUEL FROM LWRs
WHEN SUCCESSIVE GENERATIONS OF MOX ARE REPROCESSED TOGETHER --
BUT SEPARATE FROM SPENT UO₂ FUEL^{a,b}

(Grams Per Metric Ton Initial Heavy Metal)

SCENARIO #1 -- U +PU MIX FROM REPROCESSING BLENDED W/DU
TO PRODUCE MOX (5.0% EQUIVALENT FISSILE)

	UO ₂ Fuel	1 st Generation MOX Fuel	2 nd Generation MOX Fuel
Br-81	37	-	0
Se-82	59	42	40
Kr-83	66	8	49
Kr-84	213	125	120
Kr-85	31	0	0
Rb-85	186	101	98
Kr-86	323	179	173
Rb-87	436	232	227
Sr-88	601	321	311
Y-89	803	411	396
Sr-90	828	429	413
Zr-90	164	-	0
Zr-91	1,060	607	587
Zr-92	1,150	726	702
Zr-93	1,250	595	578
Zr-94	1,390	1,042	1013
Mo-95	1,320	1,065	1040
Zr-96	1,450	1,220	1196
Mo-96	109	77	76
Mo-97	1,450	1,226	1204
Mo-98	1,490	1,399	1378
Tc-99	1,350	1,310	1293
Mo-100	1,700	1,655	1631
Ru-100	278	238	236
Ru-101	1,390	1,399	1382
Ru-102	1,490	1,631	1622
Rh-103	742	1,179	1182
Ru-104	1,030	1,530	1529
Pd-104	542	494	471
Pd-105	724	1,250	1253
Ru-106	8	12	13
Pd-106	699	1,304	1311
Pd-107	413	893	902
Pd-108	270	601	609
Ag-109	136	280	284
Pd-110	90	190	191
Cd-110	89	292	298
Cd-111	43	101	93
Cd-112		48	44
Cd-114		42	40
Sn-126	34	54	49

	UO ₂ Fuel	1 st Generation MOX Fuel	2 nd Generation MOX Fuel
I-127	81	107	102
Te-128	157	190	187
I-129	274	381	373
Te-130	680	673	658
Xe-131	633	696	689
Xe-132	2,130	2,107	2067
Cs-133	1,930	1,893	1867
Xe-134	2,770	2,571	2538
Cs-134	49	48	44
Ba-134	346	321	320
Cs-135	762	1,393	1436
Xe-136	4,120	3,357	3253
Ba-136	68	83	84
Cs-137	1,950	1,964	1938
Ba-137	344	351	342
Ba-138	2,360	2,149	2124
La-139	2,200	333	1960
Ce-140	2,300	2,143	2116
Pr-141	2,010	1,815	1787
Ce-142	2,020	1,792	1764
Nd-142	55	36	31
Nd-143	1,280	228	1351
Ce-144	5	-	4
Nd-144	2,530	1,750	1711
Nd-145	1,140	1,006	991
Nd-146	1,340	1,143	1129
Pm-147	61	65	62
Sm-147	289	298	298
Nd-148	663	649	644
Sm-148	294	310	311
Nd-150	324	387	387
Sm-150	545	554	547
Sm-151		48	53
Sm-152	201	202	200
Eu-153	220	280	276
Sm-154	67	113	111
Eu-154	35	65	67
Gd-156	216	232	227
Gd-158	44	71	71
Total	61,937	56,111	58,156
Rounding & Minor Isotopes	263	3,235	19,791
TOTAL	62,200	59,345	77,946

	Downward or Flat Trend
	Upward Trend

^aAssumes UO₂ and MOX fuel are reprocessed separately.

^b60,000 MWD/MTHM Burnup @ 40 MW/MTHM - 5 Yrs. after discharge

TABLE D-6
FISSION PRODUCT CONCENTRATIONS IN SPENT UO₂ AND MOX FUEL FROM LWRs^{a,b}
WHEN SUCCESSIVE GENERATIONS OF MOX ARE REPROCESSED TOGETHER --
BUT SEPARATE FROM SPENT UO₂ FUEL^{a,b}

(Grams Per Metric Ton Initial Heavy Metal)

SCENARIO #2 -- U +PU MIX FROM REPROCESSING BLENDED W/ERU
TO PRODUCE MOX (5.0% EQUIVALENT FISSILE)

	UO ₂ Fuel	1st Generation MOX Fuel	2 nd Generation MOX Fuel
Br-81	37	0	0
Se-82	59	45	43
Kr-83	66	57	54
Kr-84	213	148	141
Kr-85	31	0	0
Rb-85	186	123	116
Kr-86	323	221	210
Rb-87	436	287	272
Sr-88	601	402	384
Y-89	803	525	496
Sr-90	828	541	514
Zr-90	164	102	69
Zr-91	1,060	730	696
Zr-92	1,150	832	801
Zr-93	1,250	656	634
Zr-94	1,390	1,127	1,094
Mo-95	1,320	1,135	1,105
Zr-96	1,450	1,275	1,246
Mo-96	109	82	76
Mo-97	1,450	1,258	1,236
Mo-98	1,490	1,418	1,395
Tc-99	1,350	1,328	1,308
Mo-100	1,700	1,656	1,630
Ru-100	278	234	228
Ru-101	1,390	1,389	1,373
Ru-102	1,490	1,566	1,558
Rh-103	742	1,049	1,080
Ru-104	1,030	1,377	1,388
Pd-104	542	496	460
Pd-105	724	1,098	1,116
Ru-106	8	12	11
Pd-106	699	1,131	1,149
Pd-107	413	766	786
Pd-108	270	520	533
Ag-109	136	250	254
Pd-110	90	160	167
Cd-110	89	234	243
Cd-111	43	86	87
Cd-112		41	40
Cd-114		37	36
Sn-126	34	45	47

	UO ₂ Fuel	1st Generation MOX Fuel	2 nd Generation MOX Fuel
I-127	81	98	98
Te-128	157	180	181
I-129	274	361	355
Te-130	680	660	649
Xe-131	633	684	685
Xe-132	2,130	2,074	2,036
Cs-133	1,930	1,914	1,888
Xe-134	2,770	2,607	2,565
Cs-134	49	45	43
Ba-134	346	311	308
Cs-135	762	1,172	1,279
Xe-136	4,120	3,570	3,395
Ba-136	68	70	76
Cs-137	1,950	1,959	1,935
Ba-137	344	348	344
Ba-138	2,360	2,193	2,159
La-139	2,200	2,033	2,000
Ce-140	2,300	2,189	2,156
Pr-141	2,010	1,861	1,826
Ce-142	2,020	1,857	1,819
Nd-142	55	37	33
Nd-143	1,280	1,365	1,373
Ce-144	5	4	4
Nd-144	2,530	1,926	1,851
Nd-145	1,140	1,049	1,033
Nd-146	1,340	1,176	1,152
Pm-147	61	66	65
Sm-147	289	307	308
Nd-148	663	652	645
Sm-148	294	295	301
Nd-150	324	369	370
Sm-150	545	557	543
Sm-151		37	43
Sm-152	201	197	199
Eu-153	220	266	264
Sm-154	67	98	101
Eu-154	35	53	58
Gd-156	216	225	214
Gd-158	44	61	62
Total	61,937	59,365	58,493
Rounding & Minor Isotopes	263	471	7,491
TOTAL	62,200	59,836	65,984

Downward or Flat Trend

Upward Trend

^a Assumes UO₂ and MOX fuel are reprocessed separately.

^b 60,000 MWD/MTHM Burnup @ 40 MW/MTHM - 5 Yrs. after discharge

APPENDIX E

**ORIGEN RESULTS FOR SPENT FR FUEL FOR
SUCCESSIVE GENERATIONS OF RECYCLE**

TABLE E-1
ACTINIDE CONCENTRATIONS IN SPENT FAST REACTOR FUEL WHERE FUEL IS MADE FROM U+Pu
AFTER ONE RECYCLE OF MOX FUEL IN LWRs^{a,b,c,d}

SCENARIO #1 -- U + Pu MIX FROM LWR REPROCESSING BLENDED W/DU TO PRODUCE MOX FOR LWRs (5% EQUIVALENT FISSILE)

Category	Grams Per 565 kg Initial Heavy Metal					Percent Abundance in Category				
	Initial FR Fuel	1 st Generation FR Fuel	2 nd Generation FR Fuel	3 rd Generation FR Fuel	4 th Generation FR Fuel	Initial FR Fuel	1 st Generation FR Fuel	2 nd Generation FR Fuel	3 rd Generation FR Fuel	4 th Generation FR Fuel
Uranium										
U-234	333	544	677	746	773	0.08	0.13	0.17	0.19	0.19
U-235	412	189	133	120	119	0.10	0.05	0.03	0.03	0.03
U-236	1,342	1,321	1,288	1,256	1,228	0.34	0.33	0.32	0.31	0.31
U-238	398,500	399,900	400,700	401,000	401,100	99.48	99.49	99.48	99.47	99.47
	400,587	401,954	402,798	403,122	403,220	100.00	100.00	100.00	100.00	100.00
Plutonium										
Pu-238	7,042	7,152	6,517	5,861	5,369	8.35	8.21	7.36	6.56	5.98
Pu-239	37,500	41,810	43,850	44,800	45,220	44.45	48.00	49.54	50.16	50.35
Pu-240	22,950	23,430	24,310	25,210	25,980	27.20	26.91	27.46	28.23	28.93
Pu-241	4,746	3,197	2,892	2,898	2,967	5.62	3.67	3.27	3.24	3.30
Pu-242	12,130	11,500	10,950	10,550	10,270	14.38	13.21	12.37	11.81	11.44
	84,368	87,089	88,519	89,319	89,806	100.00	100.00	100.00	100.00	100.00
Minor Actinides										
Np	6,633	4,792	3,855	3,374	3,123	29.07	25.53	23.26	21.9	21.13
Am	11,813	9,520	8,306	7,700	7,399	51.77	50.72	50.11	49.97	50.06
Cm	4,370	4,457	4,413	4,335	4,258	19.16	23.75	26.63	28.13	28.81
	22,816	18,769	16,574	15,409	14,780	100.00	100.00	100.00	100.00	100.00
TOTAL	507,771	507,812	507,891	507,850	507,806					

^aFR fuel content is 21% TRU, 79% U

^b565 kgIHM is discharged from a single reactor unit every 4 months (of 9 reactor units in the plant)

^c97,770 MWD/MTIHM burnup @ 66.6 MW/MTIHM -- 2 years after discharge from FR

^dFR = fast reactor

TABLE E-2
ACTINIDE CONCENTRATIONS IN SPENT FAST REACTOR FUEL WHERE FUEL IS MADE FROM U+Pu
AFTER ONE RECYCLE OF MOX FUEL IN LWRs^{a,b,c,d}

SCENARIO #2 -- U + Pu MIX FROM LWR REPROCESSING BLENDED W/ERU TO PRODUCE MOX FOR LWRs (5% EQUIVALENT FISSILE)

Category	Grams Per 565 kg Initial Heavy Metal					Percent Abundance in Category				
	Initial FR Fuel	1 st Generation FR Fuel	2 nd Generation FR Fuel	3 rd Generation FR Fuel	4 th Generation FR Fuel	Initial FR Fuel	1 st Generation FR Fuel	2 nd Generation FR Fuel	3 rd Generation FR Fuel	4 th Generation FR Fuel
Uranium										
U-234	561	818	974	1,052	1,081	0.14	0.20	0.24	0.26	0.27
U-235	1,129	503	327	279	266	0.28	0.13	0.08	0.07	0.07
U-236	7,014	6,790	6,538	6,307	6,107	1.76	1.69	1.62	1.57	1.51
U-238	390,200	393,200	394,500	395,100	395,500	97.82	97.98	98.06	98.1	98.15
	398,904	401,311	402,339	402,738	402,954	100.00	100.00	100.00	100.00	100.00
Plutonium										
Pu-238	9,566	9,345	8,456	7,633	7,032	11.63	10.95	9.72	8.69	7.95
Pu-239	35,830	40,660	42,960	44,040	44,520	43.55	47.65	49.41	50.12	50.34
Pu-240	20,660	21,260	22,310	23,360	24,240	25.11	24.91	25.66	26.58	27.41
Pu-241	4,109	2,797	2,571	2,609	2,694	4.99	3.28	2.96	2.97	3.05
Pu-242	12,110	11,270	10,650	10,230	9,946	14.72	13.21	12.25	11.64	11.25
	82,275	85,332	86,947	87,872	88,432	100.00	100.00	100.00	100.00	100.00
Minor Actinides										
Np	10,560	7,765	6,375	5,665	5,284	40.75	36.97	34.56	33.09	32.24
Am	11,039	8,867	7,755	7,213	6,947	42.60	42.22	42.05	42.14	42.39
Cm	4,315	4,371	4,314	4,240	4,157	16.65	20.81	23.39	24.77	25.37
	25,914	21,003	18,444	17,118	16,388	100.00	100.00	100.00	100.00	100.00
TOTAL	507,093	507,646	507,730	507,728	507,774					

^aFR fuel content is 21% TRU, 79% U

^b565 kgIHM is discharged from a single reactor unit every 4 months (of 9 reactor units in the plant)

^c97,770 MWD/MTIHM burnup @ 66.6 MW/MTIHM -- 2 years after discharge from FR

^dFR = fast reactor

TABLE E-3
ACTINIDE CONCENTRATIONS IN SPENT FAST REACTOR FUEL WHERE FUEL IS MADE FROM U+Pu
AFTER TWO RECYCLES OF MOX FUEL IN LWRs^{a,b,c,d}
SCENARIO #1 -- U + Pu MIX FROM LWR REPROCESSING BLENDED W/DU TO PRODUCE MOX FOR LWRs (5% EQUIVALENT FISSILE)

Category	Grams Per 565 kg Initial Heavy Metal					Percent Abundance in Category				
	Initial FR Fuel	1 st Generation FR Fuel	2 nd Generation FR Fuel	3 rd Generation FR Fuel	4 th Generation FR Fuel	Initial FR Fuel	1 st Generation FR Fuel	2 nd Generation FR Fuel	3 rd Generation FR Fuel	4 th Generation FR Fuel
Uranium										
U-234	370	598	738	807	831	0.09	0.15	0.18	0.20	0.21
U-235	261	129	101	97	98	0.07	0.03	0.03	0.02	0.02
U-236	1,164	1,142	1,113	1,087	1,065	0.29	0.29	0.28	0.27	0.26
U-238	397,500	399,600	400,500	400,900	401,000	99.55	99.53	99.51	99.51	99.51
	399,295	401,469	402,452	402,891	402,994	100.00	100.00	100.00	100.00	100.00
Plutonium										
Pu-238	7,839	7,834	7,022	6,249	5,694	9.70	9.28	8.13	7.15	6.47
Pu-239	35,830	40,840	43,190	44,270	44,740	44.35	48.38	50.03	50.64	50.80
Pu-240	20,610	21,410	22,620	23,760	24,690	25.51	25.36	26.20	27.18	28.03
Pu-241	4,124	2,792	2,589	2,646	2,742	5.10	3.31	3.00	3.03	3.11
Pu-242	12,390	11,540	10,910	10,490	10,210	15.34	13.67	12.64	12.00	11.59
	80,793	84,416	86,331	87,415	88,076	100.00	100.00	100.00	100.00	100.00
Minor Actinides										
Np	7,220	5,067	4,033	3,524	3,266	26.01	23.02	21.14	20.07	19.51
Am	14,366	10,955	9,316	8,542	8,170	51.76	49.77	48.83	48.64	48.79
Cm	6,171	5,991	5,730	5,496	5,308	22.23	27.21	30.03	31.29	31.7
	27,757	22,013	19,079	17,562	16,744	100.00	100.00	100.00	100.00	100.00
TOTAL	507,845	507,898	507,862	507,868	507,814					

^aFR fuel content is 21% TRU, 79% U

^b565 kgIHM is discharged from a single reactor unit every 4 months (of 9 reactor units in the plant)

^c97,770 MWD/MTIHM burnup @ 66.6 MW/MTIHM -- 2 years after discharge from FR

^dFR = fast reactor

TABLE E-4
ACTINIDE CONCENTRATIONS IN SPENT FAST REACTOR FUEL WHERE FUEL IS MADE FROM U+Pu
AFTER TWO RECYCLES OF MOX FUEL IN LWRs^{a,b,c,d}
SCENARIO #2 -- U + Pu MIX FROM LWR REPROCESSING BLENDED W/ERU TO PRODUCE MOX FOR LWRs (5% EQUIVALENT FISSILE)

Category	Grams Per 565 kg Initial Heavy Metal					Percent Abundance in Category				
	Initial FR Fuel	1 st Generation FR Fuel	2 nd Generation FR Fuel	3 rd Generation FR Fuel	4 th Generation FR Fuel	Initial FR Fuel	1 st Generation FR Fuel	2 nd Generation FR Fuel	3 rd Generation FR Fuel	4 th Generation FR Fuel
Uranium										
U-234	665	948	1,117	1,198	1,225	0.17	0.24	0.28	0.30	0.31
U-235	1,202	541	357	307	294	0.30	0.13	0.09	0.08	0.07
U-236	7,945	7,697	7,414	7,152	6,925	2.00	1.92	1.84	1.77	1.72
U-238	387,800	391,600	393,200	393,900	394,300	97.53	97.71	97.79	97.85	97.90
	397,612	400,786	402,088	402,557	402,744	100.00	100.00	100.00	100.00	100.00
Plutonium										
Pu-238	10,830	10,530	9,450	8,486	7,800	13.87	12.80	11.19	9.90	9.02
Pu-239	34,050	39,640	42,280	43,490	44,020	43.61	48.18	50.05	50.74	50.91
Pu-240	18,060	19,050	20,450	21,750	22,800	23.13	23.16	24.21	25.38	26.37
Pu-241	3,571	2,422	2,280	2,363	2,475	4.58	2.94	2.70	2.76	2.86
Pu-242	11,560	10,630	10,010	9,619	9,369	14.81	12.92	11.85	11.22	10.84
	78,071	82,272	84,470	85,708	86,464	100.00	100.00	100.00	100.00	100.00
Minor Actinides										
Np	12,450	8,936	7,284	6,472	6,049	39.65	36.44	34.47	33.35	32.73
Am	13,097	9,944	8,466	7,781	7,459	41.72	40.55	40.07	40.1	40.36
Cm	5,849	5,641	5,380	5,153	4,974	18.63	23.01	25.46	26.55	26.91
	31,396	24,521	21,130	19,406	18,482	100.00	100.00	100.00	100.00	100.00
TOTAL	507,079	507,579	507,688	507,671	507,690					

^aFR fuel content is 21% TRU, 79% U

^b565 kgIHM is discharged from a single reactor unit every 4 months (of 9 reactor units in the plant)

^c97,770 MWD/MTIHM burnup @ 66.6 MW/MTIHM -- 2 years after discharge from FR

^dFR = fast reactor

TABLE E-5
ACTINIDE CONCENTRATIONS IN SUCCESSIVE GENERATIONS OF FAST REACTOR FUEL WHERE FUEL IS MADE FROM U + Pu AFTER ONE RECYCLE IN LWRs^{a,b}
SCENARIO #1 -- U + Pu MIX FROM LWR REPROCESSING BLENDED W/DU TO PRODUCE MOX FOR LWRs (5.0% EQUIVALENT FISSILE)
(Grams/565 kg Initial Heavy Metal)

(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)	(18)	(19)	(20)	(21)	(22)	(23)	(24)	Percent Abundance in Category for Initial & Reload Fuel					
Initial FR Core				1st Recycle					2nd Recycle					3rd Recycle					4th Recycle										
Initial Fuel from Spent MOX-1 Spent Fuel Increase (Decrease)				Makeup from Spent MOX-1 Recycle FR Fuel Spent FR Fuel Increase (Decrease)					Makeup from Spent MOX-1 Recycle FR Fuel Spent FR Fuel Increase (Decrease)					Makeup from Spent MOX-1 Recycle FR Fuel Spent FR Fuel Increase (Decrease)					Makeup from Spent MOX-1 Recycle FR Fuel Spent FR Fuel Increase (Decrease)					Initial FR Core	1st Recycle	2nd Recycle	3rd Recycle	4th Recycle	
CR = 0.78				CR = 0.76					CR = 0.75					CR = 0.74					CR = 0.74										
U-234	100	333	233	3	11	341	544	202	5	11	549	677	128	7	11	681	746	65	7	11	749	773	24	0.02%	0.08%	0.12%	0.15%	0.17%	
U-235	1374	412	-963	4	153	561	189	-371	2	149	336	133	-204	1	146	278	120	-158	1	145	264	119	-146	0.31%	0.13%	0.08%	0.06%	0.06%	
U-236	1447	1342	-105	13	161	1490	1321	-169	13	157	1465	1288	-177	13	154	1429	1256	-173	13	153	1397	1228	-169	0.32%	0.33%	0.33%	0.32%	0.31%	
U-238	443429	398500	-44900	3985	49443	443958	399900	-44100	3999	48099	444000	400700	-43300	4007	47270	443963	401000	-43000	4010	46950	443940	401100	-42800	99.35%	99.46%	99.47%	99.47%	99.46%	
Totals	446350	400587	-45734	4006	49769	446350	401954	-44438	4019	48416	446350	402797	-43553	4028	47581	446350	403122	-43266	4031	47260	446350	403219	-43091	100.00%	100.00%	100.00%	100.00%	100.00%	
Np-237	11136	6633	-4507	66	1177	7743	4792	-2951	48	1300	6044	3855	-2189	39	1371	5188	3374	-1814	34	1405	4745	3123	-1622	100.00%	100.00%	100.00%	100.00%	100.00%	
Pu-238	4513	7042	2529	70	477	7448	7152	-296	72	527	7607	6517	-1090	65	556	7007	5861	-1147	59	569	6372	5369	-1003	5.22%	8.04%	7.90%	7.13%	6.41%	
Pu-239	30712	37500	6790	375	3245	40370	41810	1440	418	3585	44977	43850	-1130	438	3781	47193	44800	-2390	448	3875	48227	45220	-3010	35.51%	43.57%	46.70%	48.02%	48.55%	
Pu-240	25779	22950	-2830	229	2724	25445	23430	-2010	234	3009	26205	24310	-1890	243	3174	27241	25210	-2030	252	3252	28210	25980	-2230	29.81%	27.46%	27.21%	27.72%	28.40%	
Pu-241	12133	4746	-7384	47	1282	5981	3197	-2784	32	1416	4581	2892	-1689	29	1494	4357	2898	-1459	29	1531	4400	2967	-1433	14.03%	6.45%	4.76%	4.43%	4.43%	
Pu-242	13348	12130	-1220	121	1410	13419	11500	-1920	115	1558	12943	10950	-1990	110	1643	12484	10550	-1930	106	1684	12128	10270	-1860	15.43%	14.48%	13.44%	12.70%	12.21%	
Totals	86485	84368	-2115	844	9139	92663	87089	-5570	871	10096	96314	88519	-7789	885	10648	98282	89319	-8956	893	10911	99337	89806	-9536	100.00%	100.00%	100.00%	100.00%	100.00%	
Am-241	9632	6175	-3457	62	1018	7131	4430	-2701	44	1124	5510	3541	-1969	35	1186	4691	3140	-1551	31	1215	4324	2979	-1345	57.19%	52.92%	48.37%	45.56%	44.36%	
Am-243	7209	5638	-1571	56	762	6343	5090	-1253	51	841	5881	4765	-1116	48	888	5605	4560	-1045	46	909	5424	4420	-1004	42.81%	47.08%	51.63%	54.44%	55.64%	
Totals	16840	11813	-5028	118	1779	13474	9520	-3954	95	1966	11391	8306	-3085	83	2073	10296	7700	-2596	77	2125	9748	7399	-2349	100.00%	100.00%	100.00%	100.00%	100.00%	
Cm-242	0	19	19	0	0	19	14	-5	0	0	13	10	-3	0	0	10	9	-1	0	0	9	8	0	0.00%	0.40%	0.28%	0.21%	0.18%	
Cm-243	26	39	13	0	3	41	35	-6	0	3	38	29	-9	0	3	32	25	-7	0	3	28	22	-5	0.63%	0.86%	0.77%	0.65%	0.57%	
Cm-244	3668	3621	-47	36	388	3972	3594	-378	36	428	3986	3498	-488	35	452	3915	3396	-519	34	463	3825	3307	-518	87.58%	83.30%	81.33%	80.14%	79.35%	
Cm-245	446	610	164	6	47	651	693	42	7	52	738	718	-20	7	55	766	718	-48	7	56	767	708	-59	10.65%	13.66%	15.07%	15.68%	15.91%	
Cm-246	48	81	33	1	5	85	121	36	1	6	125	158	32	2	6	162	188	26	2	6	192	213	20	1.14%	1.79%	2.56%	3.31%	3.99%	
Totals	4189	4370	182	44	443	4769	4457	-312	45	489	4901	4413	-488	44	516	4885	4335	-549	43	528	4820	4258	-563	100.00%	100.00%	100.00%	100.00%	100.00%	
Grand Total	565000	507771	-57202	5078	62307	565000	507812	-57225	5078	62267	565000	507890	-57104	5079	62189	565000	507850	-57181	5078	62229	565000	507805	-57161						
TRU	118650	107184	-11468	1072	12538	118650	105858	-12787	1059	13851	118650	105093	-13551	1051	14608	118650	104728	-13915	1047	14969	118650	104586	-14070	100.00%	100.00%	100.00%	100.00%	100.00%	
Pu	86485	84368	-2115	844	9139	92663	87089	-5570	871	10096	96314	88519	-7789	885	10648	98282	89319	-8956	893	10911	99337	89806	-9536	72.89%	78.10%	81.17%	82.83%	83.72%	
Np, Am, Crr	32165	22816	-9353	228	3399	25987	18769	-7217	188	3755	22336	16574	-5762	166	3960	20368	15409	-4959	154	4058	19313	14780	-4534	27.11%	21.90%	18.83%	17.17%	16.28%	
U	446350	400587	-45734	4006	49769	446350	401954	-44438	4019	48416	446350	402797	-43553	4028	47581	446350	403122	-43266	4031	47260	446350	403219	-43091	79.00%	79.00%	79.00%	79.00%	79.00%	
TRU	118650	107184	-11468	1072	12538	118650	105858	-12787	1059	13851	118650	105093	-13551	1051	14608	118650	104728	-13915	1047	14969	118650	104586	-14070	21.00%	21.00%	21.00%	21.00%	21.00%	
Totals	565000	507771	-57202	5078	62307	565000	507812	-57225	5078	62267	565000	507890	-57104	5079	62189	565000	507850	-57181	5078	62229	565000	507805	-57161						
																								Np, Am, Cm	5.69%	4.60%	3.95%	3.61%	3.42%

^aHeavy metal composition of unirradiated fuel -- 21% TRU, 79% U
^bAssumes process losses of 1% (for reprocessing and fuel fabrication)

TABLE E-6
ACTINIDE CONCENTRATIONS IN SUCCESSIVE GENERATIONS OF FAST REACTOR FUEL WHERE FUEL IS MADE FROM U + Pu AFTER ONE RECYCLE IN LWRs^{a,b}
SCENARIO #2 -- U + Pu MIX FROM LWR REPROCESSING BLENDED W/ERU TO PRODUCE MOX FOR LWRs (5.0% EQUIVALENT FISSILE)
(Grams/565 kg Initial Heavy Metal)

(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)	(18)	(19)	(20)	(21)	(22)	(23)	(24)	Percent Abundance in Category for Initial & Reload Fuel					
	Initial FR Core			1st Recycle					2nd Recycle					3rd Recycle					4th Recycle					Initial FR Core	1st Recycle	2nd Recycle	3rd Recycle	4th Recycle	
	Initial Fuel from Spent MOX-1	Spent Fuel	Increase (Decrease)	Process Loss	Makeup from Spent MOX-1	Recycle FR Fuel	Spent FR Fuel	Increase (Decrease)	Process Loss	Makeup from Spent MOX-1	Recycle FR Fuel	Spent FR Fuel	Increase (Decrease)	Process Loss	Makeup from Spent MOX-1	Recycle FR Fuel	Spent FR Fuel	Increase (Decrease)	Process Loss	Makeup from Spent MOX-1	Recycle FR Fuel	Spent FR Fuel	Increase (Decrease)						
	CR = 0.8					CR = 0.77					CR = 0.75					CR = 0.74					CR = 0.74								
U-234	262	561	299	-6	30	585	818	233	-8	29	839	974	135	-10	28	993	1052	59	-11	28	1069	1081	12	0.06%	0.13%	0.19%	0.22%	0.24%	
U-235	3817	1129	-2688	-11	440	1558	503	-1055	-5	419	917	327	-590	-3	411	735	279	-456	-3	407	683	266	-417	0.86%	0.35%	0.21%	0.16%	0.15%	
U-236	7958	7014	-944	-70	917	7861	6790	-1071	-68	875	7597	6538	-1059	-65	856	7329	6307	-1022	-63	849	7093	6107	-986	1.78%	1.76%	1.70%	1.64%	1.59%	
U-238	434313	390200	-44113	-3902	50048	436346	393200	-43146	-3932	47729	436997	394500	-42497	-3945	46739	437294	395100	-42194	-3951	46355	437504	395500	-42004	97.30%	97.76%	97.90%	97.97%	98.02%	
Total	446350	398904	-47446	-3989	51435	446350	401311	-45039	-4013	49052	446350	402339	-44011	-4023	48034	446350	402738	-43612	-4027	47639	446350	402954	-43396	100.00%	100.00%	100.00%	100.00%	100.00%	
Np-237	17280	10560	-6720	-106	1681	12135	7765	-4370	-78	1948	9636	6375	-3261	-64	2084	8396	5665	-2731	-57	2142	7751	5284	-2467	100.00%	100.00%	100.00%	100.00%	100.00%	
Pu-238	7266	9566	2300	-96	707	10177	9345	-832	-93	819	10071	8456	-1615	-85	876	9248	7633	-1615	-76	901	8457	7032	-1425	8.91%	11.39%	10.75%	9.64%	8.71%	
Pu-239	27358	35830	8472	-358	2662	38133	40660	2527	-407	3085	43338	42960	-378	-430	3300	45831	44040	-1791	-440	3392	46991	44520	-2471	33.55%	42.66%	46.26%	47.78%	48.39%	
Pu-240	22942	20660	-2282	-207	2232	22685	21260	-1425	-213	2587	23634	22310	-1324	-223	2768	24855	23360	-1495	-234	2844	25971	24240	-1731	28.13%	25.38%	25.23%	25.91%	26.75%	
Pu-241	10326	4109	-6217	-41	1005	5073	2797	-2276	-28	1164	3933	2571	-1362	-26	1246	3791	2609	-1182	-26	1280	3863	2694	-1169	12.66%	5.67%	4.20%	3.95%	3.98%	
Pu-242	13660	12110	-1550	-121	1329	13318	11270	-2048	-113	1540	12698	10650	-2048	-107	1648	12191	10650	-1961	-102	1694	11821	9946	-1875	16.75%	14.90%	13.56%	12.71%	12.17%	
Total	81552	82275	723	-823	7934	89386	85332	-4054	-853	9196	93674	86947	-6727	-869	9838	95915	87872	-8043	-879	10111	97104	88432	-8672	100.00%	100.00%	100.00%	100.00%	100.00%	
Am-241	8592	5457	-3135	-55	836	6238	3872	-2366	-39	969	4802	3098	-1704	-31	1037	4104	2766	-1338	-28	1065	3804	2642	-1162	54.74%	50.08%	45.52%	42.87%	41.86%	
Am-243	7105	5582	-1523	-56	691	6217	4995	-1222	-50	801	5746	4657	-1089	-47	857	5468	4447	-1021	-44	881	5283	4305	-978	45.26%	49.92%	54.48%	57.13%	58.14%	
Total	15697	11039	-4658	-110	1527	12456	8867	-3589	-89	1770	10548	7755	-2793	-78	1894	9571	7213	-2358	-72	1946	9087	6947	-2140	100.00%	100.00%	100.00%	100.00%	100.00%	
Cm-242	0	17	17	0	0	17	12	-5	0	0	12	9	-3	0	0	9	8	-1	0	0	8	7	0	0.00%	0.36%	0.25%	0.19%	0.16%	
Cm-243	24	35	11	0	2	37	31	-6	0	3	33	25	-8	0	3	28	22	-6	0	3	24	20	-5	0.58%	0.79%	0.69%	0.59%	0.52%	
Cm-244	3656	3595	-61	-36	356	3915	3535	-380	-35	412	3912	3426	-486	-34	441	3833	3320	-513	-33	453	3740	3228	-512	88.72%	83.78%	81.64%	80.39%	79.43%	
Cm-245	390	587	197	-6	38	619	674	55	-7	44	712	700	-12	-7	47	740	699	-41	-7	48	740	688	-52	9.47%	13.26%	14.85%	15.51%	15.71%	
Cm-246	51	81	30	-1	5	85	119	34	-1	6	123	154	31	-2	6	159	192	34	-2	6	196	214	18	1.23%	1.81%	2.57%	3.33%	4.17%	
Total	4121	4315	194	-43	401	4673	4371	-302	-44	465	4792	4314	-478	-43	497	4768	4240	-528	-42	511	4709	4157	-552	100.00%	100.00%	100.00%	100.00%	100.00%	
Grand Total	565000	507093	-57907	-5071	62978	565000	507646	-57354	-5076	62430	565000	507730	-57270	-5077	62347	565000	507728	-57272	-5077	62349	565000	507774	-57226						
TRU	118650	108189	-10461	-1082	11543	118650	106335	-12315	-1063	13379	118650	105391	-13259	-1054	14313	118650	104990	-13660	-1050	14710	118650	104820	-13830	100.00%	100.00%	100.00%	100.00%	100.00%	
Pu	81552	82275	723	-823	7934	89386	85332	-4054	-853	9196	93674	86947	-6727	-869	9838	95915	87872	-8043	-879	10111	97104	88432	-8672	68.73%	75.34%	78.95%	80.84%	81.84%	
Np, Am, Cm	37098	25914	-11184	-259	3609	29264	21003	-8261	-210	4183	24976	18444	-6532	-184	4475	22735	17118	-5616	-171	4599	21546	16388	-5158	31.27%	24.66%	21.05%	19.16%	18.16%	
U	446350	398904	-47446	-3989	51435	446350	401311	-45039	-4013	49052	446350	402339	-44011	-4023	48034	446350	402738	-43612	-4027	47639	446350	402954	-43396	79.00%	79.00%	79.00%	79.00%	79.00%	
TRU	118650	108189	-10461	-1082	11543	118650	106335	-12315	-1063	13379	118650	105391	-13259	-1054	14313	118650	104990	-13660	-1050	14710	118650	104820	-13830	21.00%	21.00%	21.00%	21.00%	21.00%	
																								Np, Am, Cm	6.57%	5.18%	4.42%	4.02%	3.81%

^aHeavy metal composition of unirradiated fuel -- 21% TRU, 79% U
^bAssumes process losses of 1% (for reprocessing and fuel fabrication)

TABLE E-7
ACTINIDE CONCENTRATIONS IN SUCCESSIVE GENERATIONS OF FAST REACTOR FUEL WHERE FUEL IS MADE FROM U + Pu AFTER TWO RECYCLES IN LWRs^{a,b}
SCENARIO #1 -- U + Pu MIX FROM LWR REPROCESSING BLENDED W/DU TO PRODUCE MOX FOR LWRs (5.0% EQUIVALENT FISSILE)

(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)	(18)	(19)	(20)	(21)	(22)	(23)	(24)	Percent Abundance in Category for Initial & Reload Fuel				
Initial FR Core				1st Recycle					2nd Recycle					3rd Recycle					4th Recycle					Initial FR	1st	2nd	3rd	4th
Initial Fuel from Spent MOX-2 Spent Fuel Increase (Decrease)				Process Losses	Makeup from Spent MOX-2	Recycle FR Fuel	Spent FR Fuel	Increase (Decrease)	Process Losses	Makeup from Spent MOX-2	Recycle FR Fuel	Spent FR Fuel	Increase (Decrease)	Process Losses	Makeup from Spent MOX-2	Recycle FR Fuel	Spent FR Fuel	Increase (Decrease)	Process Losses	Makeup from Spent MOX-2	Recycle FR Fuel	Spent FR Fuel	Increase (Decrease)	Core	Recycle	Recycle	Recycle	Recycle
CR = 0.81				CR = 0.77					CR = 0.75					CR = 0.75					CR = 0.74									
U-234	121	370	249	4	14	380	598	218	6	13	605	738	133	7	13	744	807	63	8	13	812	831	19	0.03%	0.09%	0.14%	0.17%	0.18%
U-235	889	261	-628	3	102	360	129	-231	1	97	225	101	-125	1	95	195	97	-98	1	95	190	98	-93	0.20%	0.08%	0.05%	0.04%	0.04%
U-236	1288	1164	-124	12	147	1300	1142	-158	11	141	1272	1113	-159	11	138	1240	1087	-153	11	137	1213	1065	-148	0.29%	0.29%	0.28%	0.28%	0.27%
U-238	444052	397500	-46600	3975	50786	444311	399600	-44700	3996	48644	444248	400500	-43700	4005	47676	444171	400900	-43300	4009	47244	444135	401000	-43100	99.49%	99.54%	99.53%	99.51%	99.50%
Totals	446350	399294	-47104	3992	51049	446350	401470	-44870	4015	48895	446350	402452	-43851	4025	47923	446350	402891	-43488	4029	47489	446350	402993	-43321	100.00%	100.00%	100.00%	100.00%	100.00%
Np-237	12442	7220	-5220	72	1173	8321	5067	-3254	51	1393	6409	4033	-2377	40	1499	5492	3524	-1968	35	1544	5033	3266	-1767	100.00%	100.00%	100.00%	100.00%	100.00%
Pu-238	4405	7839	3434	78	415	8176	7834	-342	78	493	8249	7022	-1227	70	531	7482	6249	-1233	62	547	6733	5694	-1039	5.66%	9.36%	8.94%	7.89%	7.00%
Pu-239	26327	35830	9500	358	2482	37954	40840	2890	408	2948	43379	43190	-190	432	3172	45930	44270	-1660	443	3267	47094	44740	-2350	33.82%	43.46%	47.00%	48.43%	48.95%
Pu-240	22599	20610	-1990	206	2130	22534	21410	-1120	214	2530	23726	22620	-1110	226	2723	25116	23760	-1360	238	2804	26327	24690	-1640	29.03%	25.81%	25.71%	26.48%	27.37%
Pu-241	10734	4124	-6606	41	1012	5095	2792	-2303	28	1202	3966	2589	-1377	26	1293	3856	2646	-1210	26	1332	3952	2742	-1209	13.79%	5.83%	4.30%	4.07%	4.11%
Pu-242	13775	12390	-1390	124	1299	13565	11540	-2020	115	1542	12967	10910	-2060	109	1660	12460	10490	-1970	105	1709	12094	10210	-1880	17.70%	15.53%	14.05%	13.14%	12.57%
Totals	77841	80793	2948	808	7338	87323	84416	-2895	844	8716	92288	86331	-5964	863	9378	94845	87415	-7433	874	9659	96200	88076	-8118	100.00%	100.00%	100.00%	100.00%	100.00%
Am-241	12307	7112	-5198	71	1160	8201	4783	-3418	48	1378	6113	3728	-2385	37	1483	5173	3292	-1881	33	1527	4786	3132	-1654	55.41%	50.26%	45.85%	43.48%	42.69%
Am-243	9905	7254	-2651	73	934	8115	6172	-1943	62	1109	7219	5588	-1631	56	1193	6725	5250	-1475	53	1229	6427	5038	-1389	44.59%	49.74%	54.15%	56.52%	57.31%
Totals	22212	14366	-7849	144	2094	16316	10955	-5361	110	2487	13332	9316	-4016	93	2676	11899	8542	-3356	85	2756	11213	8170	-3043	100.00%	100.00%	100.00%	100.00%	100.00%
Cm-242	0	23	23	0	0	23	15	-8	0	0	15	11	-4	0	0	11	10	-1	0	0	10	9	0	0.00%	0.34%	0.23%	0.17%	0.15%
Cm-243	36	50	14	1	3	53	42	-11	0	4	45	33	-12	0	4	37	28	-9	0	4	32	25	-7	0.58%	0.80%	0.69%	0.58%	0.51%
Cm-244	5333	5072	-261	51	503	5524	4785	-739	48	597	5334	4491	-843	45	642	5089	4253	-836	43	662	4872	4072	-800	86.64%	82.57%	80.57%	79.33%	78.53%
Cm-245	713	901	188	9	67	959	968	9	10	80	1038	966	-72	10	86	1042	938	-104	9	89	1017	906	-111	11.59%	14.34%	15.69%	16.25%	16.40%
Cm-246	73	125	52	1	7	130	181	51	2	8	187	229	42	2	9	236	267	32	3	9	274	296	23	1.19%	1.95%	2.83%	3.67%	4.41%
Totals	6155	6171	16	62	580	6690	5991	-699	60	689	6621	5730	-890	57	742	6414	5496	-919	55	764	6205	5308	-896	100.00%	100.00%	100.00%	100.00%	100.00%
Grand Total	565000	507845	-57208	5078	62234	565000	507899	-57079	5079	62180	565000	507861	-57098	5079	62217	565000	507867	-57164	5078	62212	565000	507813	-57145					
Pu	77841	80793	2948	808	7338	87323	84416	-2895	844	8716	92288	86331	-5964	863	9378	94845	87415	-7433	874	9659	96200	88076	-8118	65.61%	73.60%	77.78%	79.94%	81.08%
Np, Am, Cm	40809	27757	-13053	278	3847	31327	22013	-9314	220	4569	26363	19079	-7283	191	4916	23805	17562	-6243	176	5064	22450	16744	-5706	34.39%	26.40%	22.22%	20.06%	18.92%
Total TRU	118650	108550	-10105	1085	11185	118650	106429	-12209	1064	13285	118650	105410	-13247	1054	14294	118650	104977	-13676	1050	14723	118650	104820	-13824	100.00%	100.00%	100.00%	100.00%	100.00%
U	446350	399294	-47104	3992	51049	446350	401470	-44870	4015	48895	446350	402452	-43851	4025	47923	446350	402891	-43488	4029	47489	446350	402993	-43321	79.00%	79.00%	79.00%	79.00%	79.00%
TRU	118650	108550	-10105	1085	11185	118650	106429	-12209	1064	13285	118650	105410	-13247	1054	14294	118650	104977	-13676	1050	14723	118650	104820	-13824	21.00%	21.00%	21.00%	21.00%	21.00%
Totals	565000	507845	-57208	5078	62234	565000	507899	-57079	5079	62180	565000	507861	-57098	5079	62217	565000	507867	-57164	5078	62212	565000	507813	-57145	100.00%	100.00%	100.00%	100.00%	100.00%

^aHeavy metal composition of unirradiated fuel -- 21% TRU, 79% U
^bAssumes process losses of 1% (for reprocessing and fuel fabrication)

Np, Am, Cm
7.22% 5.54% 4.67% 4.21% 3.97%

TABLE E-8
ACTINIDE CONCENTRATIONS IN SUCCESSIVE GENERATIONS OF FAST REACTOR FUEL WHERE FUEL IS MADE FROM U + Pu AFTER TWO RECYCLES IN LWRs^{a,b}
SCENARIO #2 -- U + Pu MIX FROM LWR REPROCESSING BLENDED W/ERU TO PRODUCE MOX FOR LWRs (5.0% EQUIVALENT FISSILE)

(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)	(18)	(19)	(20)	(21)	(22)	(23)	(24)	Percent Abundance in Category for Initial & Reload Fuel					
	Initial FR Core			1st Recycle					2nd Recycle					3rd Recycle					4th Recycle					Initial FR Core	1st Recycle	2nd Recycle	3rd Recycle	4th Recycle	
	Initial Fuel from Spent MOX	Spent Fuel	Increase (Decrease)	Process Loss	Makeup from Spent MOX-2	Recycle FR Fuel	Spent FR Fuel	Increase (Decrease)	Process Loss	Makeup from Spent MOX-2	Recycle FR Fuel	Spent FR Fuel	Increase (Decrease)	Process Loss	Makeup from Spent MOX-2	Recycle FR Fuel	Spent FR Fuel	Increase (Decrease)	Process Loss	Makeup from Spent MOX-2	Recycle FR Fuel	Spent FR Fuel	Increase (Decrease)						
	CR = 0.82					CR = 0.78					CR = 0.76					CR = 0.75					CR = 0.75								
U-234	355	665	310	-7	42	700	948	248	-9	39	978	1117	139	-11	38	1144	1198	54	-12	38	1224	1225	1	0.08%	0.16%	0.22%	0.26%	0.27%	
U-235	4184	1202	-2982	-12	494	1684	541	-1143	-5	465	1000	357	-643	-4	453	806	307	-499	-3	448	752	294	-458	0.94%	0.38%	0.22%	0.18%	0.17%	
U-236	9067	7945	-1122	-79	1071	8936	7697	-1239	-77	1007	8627	7414	-1213	-74	981	8321	7152	-1169	-72	971	8052	6925	-1127	2.03%	2.00%	1.93%	1.86%	1.80%	
U-238	432744	387800	-44944	-3878	51107	435029	391600	-43429	-3916	48061	435745	393200	-42545	-3932	46811	436079	393900	-42179	-3939	46361	436322	394300	-42022	96.95%	97.46%	97.62%	97.70%	97.75%	
Total	446350	397612	-48738	-3976	52714	446350	400786	-45564	-4008	49572	446350	402088	-44262	-4021	48283	446350	402557	-43793	-4026	47818	446350	402744	-43606	100.00%	100.00%	100.00%	100.00%	100.00%	
Np-237	20800	12450	-8350	-125	1802	14127	8936	-5191	-89	2266	11112	7284	-3828	-73	2473	9684	6472	-3212	-65	2557	8964	6049	-2915	100.00%	100.00%	100.00%	100.00%	100.00%	
Pu-238	7382	10830	3448	-108	639	11361	10530	-831	-105	804	11229	9450	-1779	-95	878	10233	8486	-1747	-85	908	9309	7800	-1509	10.26%	13.60%	12.58%	11.10%	9.93%	
Pu-239	22915	34050	11135	-341	1985	35694	39640	3946	-396	2496	41740	42280	540	-423	2724	44581	43490	-1091	-435	2817	45872	44020	-1852	31.85%	42.74%	46.75%	48.36%	48.96%	
Pu-240	19426	18060	-1366	-181	1683	19562	19050	-512	-191	2116	20976	20450	-526	-205	2309	22555	21750	-805	-218	2388	23921	22800	-1121	27.00%	23.42%	23.49%	24.47%	25.53%	
Pu-241	9218	3571	-5647	-36	798	4334	2422	-1912	-24	1004	3402	2280	-1122	-23	1096	3353	2363	-990	-24	1133	3473	2475	-998	12.81%	5.19%	3.81%	3.64%	3.71%	
Pu-242	13011	11560	-1451	-116	1127	12571	10630	-1941	-106	1417	11941	10010	-1931	-100	1547	11457	9619	-1838	-96	1600	11122	9369	-1753	18.08%	15.05%	13.37%	12.43%	11.87%	
Total	71952	78071	6119	-781	6233	83523	82272	-1251	-823	7838	89287	84470	-4817	-845	8554	92179	85708	-6471	-857	8846	93697	86464	-7233	100.00%	100.00%	100.00%	100.00%	100.00%	
Am-241	10645	6184	-4461	-62	922	7044	4131	-2913	-41	1160	5249	3228	-2021	-32	1266	4461	2869	-1592	-29	1309	4149	2748	-1401	52.98%	47.90%	43.62%	41.42%	40.78%	
Am-243	9448	6913	-2535	-69	818	7662	5813	-1849	-58	1029	6784	5238	-1546	-52	1123	6309	4912	-1397	-49	1162	6024	4711	-1313	47.02%	52.10%	56.38%	58.58%	59.22%	
Total	20093	13097	-6996	-131	1740	14706	9944	-4762	-99	2189	12033	8466	-3567	-85	2389	10770	7781	-2989	-78	2470	10173	7459	-2714	100.00%	100.00%	100.00%	100.00%	100.00%	
Cm-242	0	20	20	0	0	20	13	-7	0	0	13	10	-3	0	0	9	8	-1	0	0	8	8	0	0.00%	0.32%	0.21%	0.16%	0.14%	
Cm-243	32	44	12	0	3	46	36	-10	0	3	39	28	-11	0	4	32	24	-8	0	4	28	22	-6	0.54%	0.73%	0.63%	0.53%	0.47%	
Cm-244	5095	4836	-259	-48	441	5229	4520	-709	-45	555	5030	4226	-804	-42	606	4789	3995	-794	-40	626	4581	3822	-759	87.77%	83.08%	80.90%	79.61%	78.78%	
Cm-245	608	833	226	-8	53	878	905	27	-9	66	962	904	-58	-9	72	967	878	-89	-9	75	944	847	-97	10.47%	13.94%	15.47%	16.07%	16.23%	
Cm-246	71	116	46	-1	6	121	168	46	-2	8	174	212	39	-2	8	219	248	29	-2	9	254	275	21	1.22%	1.93%	2.79%	3.63%	4.37%	
Total	5805	5849	44	-58	503	6294	5641	-652	-56	632	6217	5380	-837	-54	690	6017	5153	-863	-52	714	5815	4974	-842	100.00%	100.00%	100.00%	100.00%	100.00%	
Grand Total	565000	507079	-57921	-5071	62992	565000	507580	-57421	-5076	62496	565000	507688	-57312	-5077	62388	565000	507671	-57329	-5077	62406	565000	507690	-57310						
Pu	71952	78071	6119	-781	6233	83523	82272	-1251	-823	7838	89287	84470	-4817	-845	8554	92179	85708	-6471	-857	8846	93697	86464	-7233	60.64%	70.39%	75.25%	77.69%	78.97%	
Np, Am, Crr	46698	31396	-15302	-314	4045	35127	24521	-10606	-245	5087	29363	21130	-8233	-211	5552	26471	19406	-7065	-194	5741	24953	18482	-6472	39.36%	29.61%	24.75%	22.31%	21.03%	
Total TRU	118650	109467	-9183	-1095	10278	118650	106793	-11857	-1068	12925	118650	105600	-13050	-1056	14106	118650	105114	-13536	-1051	14587	118650	104946	-13704	100.00%	100.00%	100.00%	100.00%	100.00%	
U	446350	397612	-48738	-3976	52714	446350	400786	-45564	-4008	49572	446350	402088	-44262	-4021	48283	446350	402557	-43793	-4026	47818	446350	402744	-43606	79.00%	79.00%	79.00%	79.00%	79.00%	
TRU	118650	109467	-9183	-1095	10278	118650	106793	-11857	-1068	12925	118650	105600	-13050	-1056	14106	118650	105114	-13536	-1051	14587	118650	104946	-13704	21.00%	21.00%	21.00%	21.00%	21.00%	
Totals	565000	507079	-57921	-5071	62992	565000	507580	-57421	-5076	62496	565000	507688	-57312	-5077	62388	565000	507671	-57329	-5077	62406	565000	507690	-57310	100.00%	100.00%	100.00%	100.00%	100.00%	
																								Np, Am, Cm	8.27%	6.22%	5.20%	4.69%	4.42%

^aHeavy metal composition of unirradiated fuel -- 21% TRU, 79% U
^bAssumes process losses of 1% (for reprocessing and fuel fabrication)

EXPLANATION OF TABLES E-5 THROUGH E-8

These tables present the isotopic composition of heavy metal elements in an initial or startup core, and in four successive recycle generations of fast reactor fuel. The table shows the generation-to-generation changes in isotopic composition of recycle fuel, with the initial startup fuel and the subsequent makeup fuel being produced using uranium, plutonium and minor actinides obtained from reprocessing spent LWR fuel

Organization of the table.

(A) Rows: A row of values in the table identifies:

- (1) The number of grams of each radionuclide in the heavy metal (uranium and transuranium elements) comprising an initial 565 kg loading, or equal reload, of fast reactor fuel produced from the products of reprocessing spent LWR fuels, or
- (2) The total number of grams of each heavy metal element (uranium, neptunium, plutonium, americium, and curium) comprising an initial 565 kg loading, or equal reload, of fast reactor fuel produced from the products of reprocessing spent LWR fuels, or
- (3) The sum of the grams of neptunium, americium, and curium isotopes in the initial, or reload, fuels, or
- (4) The sum of the grams of all transuranium (neptunium, plutonium, americium, and curium) isotopes in the initial, or reload, fuels.

(B) Columns:

- (1) Column 1: The entries in this column define the rows in the table. Each entry identifies a heavy metal isotope, element, or collection of elements that would be constituents of fast reactor fuel produced from the products of reprocessing spent LWR fuels.
- (2) Columns 2 through 4: Initial Fast Reactor Core. These three columns list the amounts of heavy metal isotopes, elements, and combinations thereof that would constitute 565 kg of initial, or startup core fast reactor fuel; the corresponding spent fuel following discharge from a fast reactor after 4 years of use; and the associated amounts of the isotopes, elements, and combinations thereof consumed.
 - (a) Column 2: Initial Fuel from Spent MOX. This column lists the amounts of heavy-metal isotopes, elements, and combinations thereof that comprise unirradiated, initial-core fuel for a fast reactor; this fuel assumed to be produced from the U, Pu and minor actinide products of reprocessing LWR spent fuel.
 - (b) Column 3: Spent Fuel. This column lists the amounts of heavy metal isotopes, elements, and combinations thereof that remain in the initial-core fast reactor fuel following irradiation in a fast reactor at a power level of 66.6 mega-watts (MW) per metric-ton of initial heavy metal (MTIHM) for 4 years (1460 days) to achieve a burnup of 97,200 MWD/MTIHM). The amounts were determined by calculation using the ORIGEN computer program.
 - (c) Column 4: Increase (Decrease). This column lists changes (production and depletion) in the amounts of heavy metal isotopes, elements, and combinations thereof in 565 kg of fast reactor fuel following 4 years of power production use in a fast reactor. The amounts in each row listed in this column are the differences determined by subtracting the amounts listed in the corresponding rows in Column 2 (Initial Fuel) from those amounts listed in the same rows of Column 3 (Spent Fuel).

- (3) Columns 5 through 9: 1st Recycle. These columns list information characterizing the recovery, reconstitution, and reuse of 565 kg MTIHM of spent fast reactor fuel from the startup core of a fast reactor.
- (a) Column 5: Process Loss. This column lists the amounts of radionuclides, elements, and combinations thereof that would be lost in reprocessing and recycling 565 kg MTIHM of spent fuel from the initial fast reactor core. The analysis assumed that 0.5% of the constituents of spent fast reactor fuel would be entrained in process wastes in reprocessing and that an additional 0.5% would be lost to waste streams from the processes that would be used to produce recycle fast reactor fuel.
 - (b) Column 6: Makeup from Spent MOX. This column lists the amounts of radionuclides, elements, and combinations thereof, derived from the products of reprocessing LWR spent fuel, that would need to be added to the materials recovered from fast reactor spent fuel (Column 3 less Column 5) to produce 565 kg of recycle (reconstituted) fast reactor fuel. The analysis assumed uranium, plutonium, and minor actinides recovered from LWR spent fuel would be used to replenish (makeup) the heavy metal components of the fast reactor fuels. The amounts of uranium and transuranium elements in the makeup fuel are those needed to compensate for the amounts depleted (consumed) by a fast reactor in producing power and those amounts subsequently lost to processing waste streams. Small differences between the amount of uranium consumed in the reactor plus process losses (add corresponding values from Columns 4 and 5) and the amount of uranium in makeup fuel may be observed. These differences result from numerical rounding in performing the calculations and are not significant. Similar small and non-significant differences may be observed in corresponding totals for the amount of transuranium elements.
 - (c) Column 7: Recycle Fast Reactor (FR) Fuel. This column lists the amounts of radionuclides, elements, and combinations thereof that result from adding the like components of makeup fuel listed in Column 6 and the materials recovered from fast reactor spent fuel (Column 3 less Column 5). When combined, the amounts of radionuclides for these two components -- makeup fuel and recycled fast reactor fuel -- establish the composition of unirradiated 1st recycle fast reactor fuel.
 - (d) Columns 8: Spent Fast Reactor (FR) Fuel. This column lists the amounts of heavy metal isotopes, elements, and combinations thereof that remain in 1st recycle fast reactor fuel described in Column 7 following irradiation in a fast reactor at a power level of 66.6 mega-watts (MW) per metric-ton of initial heavy metal (MTIHM) for 4 years (1460 days) to achieve a burnup of 97,200 MWD/MTIHM). The amounts were determined by calculation using the ORIGEN computer program.
 - (e) Column 9: Increase (Decrease). This column lists changes (production and depletion) in the amounts of heavy metal isotopes, elements, and combinations thereof in 565 kg of 1st recycle fast reactor fuel following 4 years of power production use in a fast reactor. The amounts in each row listed in the column are the differences determined by subtracting the amounts listed in the corresponding rows in Column 7 from those amounts listed in the same rows in Column 8.

- (4) Columns 10 through 24: 2nd Recycle, 3rd Recycle, and 4th Recycle. These columns list information characterizing the recovery, reconstitution, and reuse of 565 kg MTIHM of spent fast reactor fuel in 2nd, 3rd, and 4th recycles of the fuel.
- (a) Column 10, 15, and 20: Process Loss. These columns list the amounts of radionuclides, elements, and combinations thereof that would be lost in reprocessing and recycling 565 kg MTIHM of spent fuel from 1st, 2nd, and 3rd recycle fast reactor cores. The analysis assumed that 0.5% of the constituents of spent fast reactor fuel would be entrained in process wastes in reprocessing and that an additional 0.5% would be lost to waste streams from the processes that would be used to produce the next generation of recycle fast reactor fuel.
 - (b) Column 11, 16, and 21: Makeup from Spent MOX. These columns list the amounts of radionuclides, elements, and combinations thereof, derived from the products of reprocessing LWR spent fuel, that would need to be added to the materials recovered from fast reactor spent fuel (respective amounts in Columns 8, 13, and 18 less amounts listed in Columns 10, 15, and 20) to produce 565 kg of recycle (reconstituted) fast reactor fuel. The analysis assumed uranium, plutonium, and minor actinides recovered from LWR spent fuel would be used to replenish (makeup) the heavy metal components of the fast reactor fuels. The amounts of uranium and transuranium elements in the makeup fuel are those needed to compensate for the amounts depleted (consumed) by a fast reactor in producing power and those amounts subsequently lost to processing wastes. Small differences between the amount of uranium consumed in the reactor plus process losses (e.g., add corresponding values from Columns 9 and 10) and the amount of uranium in makeup fuel (e.g., value in Column 11) may be observed. These differences result from numerical rounding in performing the calculations and are not significant. Similar small and non-significant differences may be observed in corresponding totals for the amount of transuranium elements.
 - (c) Column 12, 17, and 22: Recycle Fast Reactor (FR) Fuel. These columns list the amounts of radionuclides, elements, and combinations thereof that result from adding the like components of makeup fuel listed in Columns 11, 16, and 21 and the materials recovered from fast reactor spent fuel (respectively, the values in Columns 8, 13, and 18 less process losses listed in Columns 10, 15, and 20). When combined, the amounts of radionuclides for these two components – makeup fuel and recycled fast reactor fuel – establish the compositions of unirradiated 2nd, 3rd, and 4th recycle fast reactor fuel.
 - (d) Columns 13, 18, and 23: Spent Fast Reactor (FR) Fuel. These three columns list the amounts of heavy metal isotopes, elements, and combinations thereof that remain in 2nd, 3rd, and 4th recycle of fast reactor fuel, respectively described in Columns 12, 17, and 22, following irradiation in a fast reactor at a power level of 66.6 mega-watts (MW) per metric-ton of initial heavy metal (MTIHM) for 4 years (1460 days) to achieve a burnup of 97,200 MWD/MTIHM). The amounts were determined by calculation using the ORIGEN computer program.
 - (e) Column 14, 19, and 24: Increase (Decrease). These columns list changes (production and depletion) in the amounts of heavy metal isotopes, elements, and combinations thereof, respectively, in 565 kg of 2nd, 3rd, and 4th recycle fast reactor fuel following 4 years of power-production use in a fast reactor. The amounts in each row listed in these columns are the differences determined by subtracting the respective amounts listed in the corresponding rows in Columns 12, 17, and 22 from those amounts listed in the same rows in Columns 13, 18, and 23.

TABLE E-9

MAJOR FISSION PRODUCT CONCENTRATIONS IN SUCCESSIVE GENERATIONS OF FAST REACTOR FUEL WHERE FUEL IS MADE FROM U-Pu AFTER ONE RECYCLE IN LWRs^{a,b}*SCENARIO #1 -- U+Pu MIX FROM LWR REPROCESSING BLENDED W/DU TO PRODUCE MOX FOR LWRs (5.0% EQUIVALENT FISSILE)*

Fission Product	(Grams Per 565 kg Initial Heavy Metal)					% Change from Initial			
	Initial FR Fuel	1 st Generation FR Fuel	2 nd Generation FR Fuel	3 rd Generation FR Fuel	4 th Generation FR Fuel	1 st Generation FR Fuel	2 nd Generation FR Fuel	3 rd Generation FR Fuel	4 th Generation FR Fuel
Technetium	1,304	1,314	1,318	1,320	1,320	0.77%	1.07%	1.21%	1.23%
Iodine	486	497	502	504	505	2.22%	3.10%	3.49%	3.79%
Krypton	339	343	343	343	342	1.11%	1.21%	1.06%	0.93%
Total	2,129	2,154	2,163	2,166	2,167	1.15%	1.56%	1.72%	1.76%
Strontium	730	733	733	731	730	0.44%	0.37%	0.19%	0.05%
Cesium	6,310	6,353	6,375	6,387	6,393	0.69%	1.02%	1.20%	1.33%
Barium	2,294	2,298	2,299	2,299	2,299	0.19%	0.22%	0.24%	0.23%
Rubidium	323	324	324	324	324	0.57%	0.58%	0.47%	0.35%
Total	9,656	9,709	9,730	9,741	9,746	0.55%	0.77%	0.88%	0.94%
Lanthanum	1,913	1,901	1,899	1,898	1,898	-0.63%	-0.74%	-0.79%	-0.78%
Cerium	3,677	3,688	3,689	3,687	3,685	0.30%	0.32%	0.27%	0.21%
Praseodymium	1,705	1,709	1,712	1,713	1,714	0.23%	0.41%	0.47%	0.53%
Neodmium	5,828	5,789	5,776	5,770	5,767	-0.66%	-0.89%	-1.00%	-1.03%
Promethium	337	335	335	335	334	-0.33%	-0.48%	-0.57%	-0.62%
Samarium	1,581	1,572	1,567	1,564	1,562	-0.57%	-0.91%	-1.13%	-1.25%
Europium	231	229	228	227	227	-1.14%	-1.56%	-1.77%	-1.87%
Gadolinium	139	136	135	135	135	-1.98%	-2.54%	-2.69%	-2.68%
Total	15,410	15,360	15,341	15,328	15,322	-0.33%	-0.45%	-0.54%	-0.57%
Zirconium	4,168	4,203	4,208	4,207	4,204	0.84%	0.96%	0.93%	0.87%
Molybdenum	5,009	5,054	5,069	5,074	5,075	0.91%	1.19%	1.30%	1.33%
Ruthenium	4,923	4,940	4,955	4,963	4,967	0.34%	0.64%	0.81%	0.90%
Palladium	3,979	3,912	3,899	3,899	3,899	-1.68%	-2.07%	-2.06%	-2.01%
Xenon	7,255	7,299	7,316	7,323	7,326	0.61%	0.84%	0.93%	0.98%
Total	25,334	25,408	25,446	25,467	25,472	0.29%	0.44%	0.52%	0.55%
Other	3,521	3,537	3,545	3,529	3,530	0.48%	0.69%	0.24%	0.27%
Grand Total	56,049	56,167	56,225	56,231	56,237	0.21%	0.31%	0.32%	0.33%

^aHeavy metal composition of unirradiated fuel -- 21% TRU, 79% U^bAssumes process losses of 1% (for reprocessing and fuel fabrication)

TABLE E-10

MAJOR FISSION PRODUCT CONCENTRATIONS IN SUCCESSIVE GENERATIONS OF FAST REACTOR FUEL WHERE FUEL IS MADE FROM U-Pu AFTER ONE RECYCLE IN LWRs^{a,b}*SCENARIO #2 -- U+Pu MIX FROM LWR REPROCESSING BLENDED W/ERU TO PRODUCE MOX FOR LWRs (5.0% EQUIVALENT FISSILE)*

Fission Product	Grams Per 565 kg Initial Heavy Metal					% Change from Initial			
	Initial FR Fuel	1 st Generation FR Fuel	2 nd Generation FR Fuel	3 rd Generation FR Fuel	4 th Generation FR Fuel	1 st Generation FR Fuel	2 nd Generation FR Fuel	3 rd Generation FR Fuel	4 th Generation FR Fuel
Technetium	1,308	1,318	1,321	1,322	1,323	0.76%	0.99%	1.07%	1.15%
Iodine	483	496	501	503	504	2.52%	3.62%	4.10%	4.30%
Krypton	382	380	379	377	376	-0.50%	-0.93%	-1.30%	-1.56%
Total	2,173	2,194	2,200	2,202	2,203	0.93%	1.24%	1.33%	1.37%
Strontium	768	757	751	748	746	-1.46%	-2.19%	-2.60%	-2.86%
Cesium	6,291	6,340	6,364	6,377	6,384	0.77%	1.16%	1.36%	1.47%
Barium	2,298	2,301	2,301	2,301	2,301	0.12%	0.11%	0.13%	0.13%
Rubidium	338	335	333	331	331	-1.15%	-1.71%	-2.07%	-2.28%
Total	9,696	9,732	9,749	9,758	9,762	0.37%	0.54%	0.64%	0.68%
Lanthanum	1,914	1,902	1,899	1,898	1,898	-0.63%	-0.78%	-0.84%	-0.84%
Cerium	3,703	3,706	3,704	3,701	3,697	0.07%	0.01%	-0.05%	-0.16%
Praseodymium	1,710	1,711	1,713	1,715	1,715	0.06%	0.18%	0.29%	0.29%
Neodimium	5,861	5,811	5,792	5,785	5,780	-0.86%	-1.17%	-1.30%	-1.38%
Promethium	338	336	336	335	335	-0.41%	-0.59%	-0.71%	-0.77%
Samarium	1,573	1,569	1,564	1,561	1,559	-0.28%	-0.57%	-0.77%	-0.90%
Europium	227	226	226	225	225	-0.29%	-0.45%	-0.57%	-0.63%
Gadolinium	133	133	133	133	133	-0.55%	-0.56%	-0.51%	-0.48%
Total	15,459	15,393	15,366	15,353	15,343	-0.43%	-0.60%	-0.68%	-0.75%
Zirconium	4,273	4,269	4,262	4,255	4,250	-0.08%	-0.25%	-0.41%	-0.54%
Molybdenum	5,044	5,075	5,087	5,091	5,090	0.62%	0.85%	0.93%	0.93%
Ruthenium	4,848	4,895	4,919	4,930	4,937	0.96%	1.45%	1.69%	1.84%
Palladium	3,841	3,827	3,832	3,838	3,844	-0.38%	-0.25%	-0.10%	0.06%
Xenon	7,247	7,296	7,316	7,323	7,325	0.68%	0.95%	1.05%	1.08%
Total	25,253	25,362	25,415	25,437	25,446	0.43%	0.64%	0.73%	0.76%
Other	3,493	3,520	3,531	3,536	3,538	0.80%	1.11%	1.24%	1.30%
Grand Total	56,074	56,201	56,261	56,286	56,292	0.23%	0.33%	0.38%	0.39%

^aHeavy metal composition of unirradiated fuel -- 21% TRU, 79% U^bAssumes process losses of 1% (for reprocessing and fuel fabrication)

TABLE E-11

MAJOR FISSION PRODUCT CONCENTRATIONS IN SUCCESSIVE GENERATIONS OF FAST REACTOR FUEL WHERE FUEL IS MADE FROM U-Pu AFTER TWO RECYCLES IN LWRs^{a,b}SCENARIO #1 -- U+Pu MIX FROM LWR REPROCESSING BLENDED W/DU TO PRODUCE MOX FOR LWRs (5.0% EQUIVALENT FISSION)

Fission Product	Grams Per 565 kg Initial Heavy Metal					% Change from Initial			
	Initial FR Fuel	1 st Generation FR Fuel	2 nd Generation FR Fuel	3 rd Generation FR Fuel	4 th Generation FR Fuel	1 st Generation FR Fuel	2 nd Generation FR Fuel	3 rd Generation FR Fuel	4 th Generation FR Fuel
Technetium	1,296	1,310	1,315	1,317	1,318	1.08%	1.47%	1.62%	1.70%
Iodine	482	495	500	503	504	2.63%	3.75%	4.23%	4.44%
Krypton	363	368	369	368	368	1.44%	1.57%	1.43%	1.29%
Total	2,141	2,173	2,184	2,188	2,189	1.49%	2.00%	2.18%	2.24%
Strontium	727	732	732	731	730	0.73%	0.69%	0.52%	0.39%
Cesium	6,271	6,328	6,357	6,372	6,379	0.91%	1.36%	1.60%	1.72%
Barium	2,295	2,300	2,301	2,300	2,300	0.21%	0.25%	0.23%	0.21%
Rubidium	321	324	324	324	323	0.88%	0.96%	0.89%	0.76%
Total	9,614	9,684	9,713	9,726	9,732	0.73%	1.03%	1.17%	1.23%
Lanthanum	1,911	1,901	1,898	1,898	1,898	-0.52%	-0.68%	-0.68%	-0.68%
Cerium	3,677	3,692	3,693	3,690	3,688	0.42%	0.44%	0.35%	0.30%
Praseodymium	1,702	1,708	1,711	1,713	1,714	0.35%	0.53%	0.65%	0.71%
Neodmium	5,837	5,797	5,784	5,777	5,773	-0.68%	-0.91%	-1.02%	-1.09%
Promethium	337	336	336	335	335	-0.33%	-0.50%	-0.62%	-0.71%
Samarium	1,598	1,585	1,577	1,572	1,569	-0.81%	-1.30%	-1.60%	-1.78%
Europium	236	232	231	230	229	-1.46%	-2.13%	-2.49%	-2.70%
Gadolinium	141	138	137	137	136	-2.28%	-2.98%	-3.29%	-3.47%
Total	15,439	15,390	15,366	15,352	15,343	-0.32%	-0.47%	-0.56%	-0.62%
Zirconium	4,145	4,190	4,198	4,198	4,195	1.08%	1.28%	1.27%	1.22%
Molybdenum	4,981	5,036	5,055	5,061	5,065	1.11%	1.48%	1.62%	1.70%
Ruthenium	4,902	4,925	4,943	4,953	4,959	0.49%	0.84%	1.06%	1.18%
Palladium	4,002	3,927	3,911	3,910	3,910	-1.87%	-2.27%	-2.31%	-2.30%
Xenon	7,218	7,275	7,298	7,307	7,313	0.79%	1.11%	1.23%	1.32%
Total	25,248	25,354	25,405	25,429	25,443	0.42%	0.62%	0.72%	0.77%
Other	3,567	3,580	3,586	3,587	3,586	0.37%	0.52%	0.55%	0.52%
Grand Total	56,009	56,181	56,254	56,282	56,293	0.31%	0.44%	0.49%	0.51%

^aHeavy metal composition of unirradiated fuel -- 21% TRU, 79% U^bAssumes process losses of 1% (for reprocessing and fuel fabrication)

TABLE E-12

MAJOR FISSION PRODUCT CONCENTRATIONS IN SUCCESSIVE GENERATIONS OF FAST REACTOR FUEL WHERE FUEL IS MADE FROM U-Pu AFTER TWO RECYCLES IN LWRs^{a,b}*SCENARIO #2 -- U+Pu MIX FROM LWR REPROCESSING BLENDED W/ERu TO PRODUCE MOX FOR LWRs (5.0% EQUIVALENT FISSION)*

Fission Product	Grams Per 565 kg Initial Heavy Metal					% Change from Initial			
	Initial FR Fuel	1 st Generation FR Fuel	2 nd Generation FR Fuel	3 rd Generation FR Fuel	4 th Generation FR Fuel	1 st Generation FR Fuel	2 nd Generation FR Fuel	3 rd Generation FR Fuel	4 th Generation FR Fuel
Technetium	1,302	1,315	1,319	1,321	1,321	1.00%	1.31%	1.46%	1.46%
Iodine	480	494	500	502	503	2.96%	4.19%	4.75%	4.96%
Krypton	386	383	381	379	378	-0.74%	-1.34%	-1.74%	-2.04%
Total	2,168	2,192	2,200	2,203	2,203	1.12%	1.47%	1.62%	1.61%
Strontium	777	762	756	752	750	-1.89%	-2.77%	-3.23%	-3.51%
Cesium	6,254	6,316	6,347	6,363	6,372	0.99%	1.49%	1.75%	1.89%
Barium	2,299	2,303	2,302	2,301	2,301	0.16%	0.13%	0.11%	0.10%
Rubidium	342	337	334	333	332	-1.50%	-2.18%	-2.57%	-2.83%
Total	9,672	9,717	9,739	9,750	9,755	0.47%	0.70%	0.81%	0.87%
Lanthanum	1,913	1,901	1,899	1,898	1,898	-0.63%	-0.73%	-0.78%	-0.78%
Cerium	3,711	3,715	3,711	3,707	3,703	0.12%	0.01%	-0.11%	-0.19%
Praseodymium	1,709	1,711	1,714	1,715	1,716	0.12%	0.29%	0.35%	0.41%
Neodimium	5,877	5,822	5,801	5,793	5,789	-0.93%	-1.28%	-1.42%	-1.49%
Promethium	339	337	336	336	336	-0.41%	-0.62%	-0.77%	-0.86%
Samarium	1,584	1,578	1,572	1,567	1,564	-0.36%	-0.75%	-1.03%	-1.21%
Europium	229	228	227	227	226	-0.33%	-0.62%	-0.85%	-0.98%
Gadolinium	134	134	134	134	133	-0.34%	-0.43%	-0.51%	-0.57%
Total	15,494	15,426	15,394	15,376	15,366	-0.44%	-0.65%	-0.76%	-0.83%
Zirconium	4,285	4,278	4,268	4,261	4,255	-0.16%	-0.39%	-0.56%	-0.69%
Molybdenum	5,030	5,066	5,080	5,084	5,085	0.72%	0.98%	1.06%	1.10%
Ruthenium	4,809	4,871	4,901	4,916	4,924	1.28%	1.91%	2.22%	2.39%
Palladium	3,818	3,815	3,823	3,830	3,835	-0.08%	0.13%	0.30%	0.44%
Xenon	7,213	7,275	7,300	7,310	7,315	0.86%	1.21%	1.34%	1.41%
Total	25,156	25,305	25,372	25,400	25,415	0.59%	0.86%	0.97%	1.03%
Other	3,499	3,531	3,541	3,545	3,545	0.91%	1.20%	1.31%	1.32%
Grand Total	55,988	56,172	56,246	56,273	56,284	0.33%	0.46%	0.51%	0.53%

^aHeavy metal composition of unirradiated fuel -- 21% TRU, 79% U^bAssumes process losses of 1% (for reprocessing and fuel fabrication)

APPENDIX F

**COMPARISON OF ACTUAL ACCOMPLISHMENTS WITH THE GOALS AND
OBJECTIVES OF THE STUDY**

APPENDIX F
COMPARISON OF ACTUAL ACCOMPLISHMENTS WITH THE GOALS AND OBJECTIVES OF THE STUDY

Proposed Project Activity	Report Section References	
(1) Determine important characteristics of key systems and components in the reference UREX+3c fuel cycle and the AFC.		
(a) Determine quantity of spent LWR U and spent MOX fuels that would be discharged annually from LWRs collectively producing 100 GWe in an AFC.	4.2	
(b) Prepare process flow sheet and facility(ies) description for UREX+3c Process systems for (i) the reference UREX+3c process; (ii) batch processing of spent U and spent MOX LWR fuel and burned FR fuel; (iii) separated process systems for U LWR spent fuel, MOX LWR spent fuel, and burned FR fuel	3.1 4.0	
(c) Describe (i) unirradiated and spent UO ₂ LWR (PWR and BWR) fuel [initial MTU (kg), initial enrichment (wt% U ²³⁵), burnup (GWd/MTIHM), post-irradiation isotopic abundances]; (ii) unirradiated and spent 1st cycle MOX LWR fuel [initial MTHM (kg) {initial U (kg) + initial Pu (kg)}, initial enrichment (wt% U ²³⁵ , Pu ²³⁹ , Pu ²⁴¹), burnup, post-irradiation isotopic abundances]; (iii) unirradiated and spent n-cycle MOX LWR fuel [initial MTHM (kg) {initial U (kg) + initial Pu (kg)}, initial enrichment (wt% U ²³⁵ , Pu ²³⁹ , Pu ²⁴¹), burnup, post-irradiation isotopic abundances]	3.2 3.4 Appendix C Appendix D	
(d) Describe unirradiated and burned U-Pu FR fuel [Initial MTHM (kg) {initial U (kg) + initial Pu (kg)}, initial enrichment (wt% U ²³⁵ , Pu ²³⁸ , Pu ²³⁹ , Pu ²⁴⁰ , Pu ²⁴¹ , Pu ²⁴²), burnup (GWd/MTIHM), post-irradiation isotopic abundances]	3.2 3.5	Appendix E
(e) Describe uranium (i) virgin-U used in LWR fuels [enrichment (wt% U ²³⁵), U ²³⁴ (ppm), ASTM specification requirements]; (ii) characteristics of uranium in spent LWR U fuel [average enrichment (wt% U ²³⁵), variation of discharge enrichment (wt% U ²³⁵), average U ²³⁴ and U ²³⁶ and variations (ppm)]; (iii) characteristics of uranium in spent LWR MOX fuel (1 st cycle and n-cycle) [average enrichment (wt% U ²³⁵), variation of discharge enrichment (wt% U ²³⁵), average U ²³⁴ and U ²³⁶ and variations (ppm)]	3.2 3.3 3.4	Appendix A Appendix C Appendix D
(f) Describe alternatives for converting excess uranium from the UREX process (U-oxide for storage or disposal, UF ₆ for use in reactor fuel or for additional processing)	3.1 10.1	
(g) Describe alternative processes for producing a standardized recovered uranium product(s) for use in LWR fuel. (Product blending and/or dilution and enrichment)	3.1 10.1	
(h) Determine energy value of actinide isotopes in FR targets.	10.3	
(i) Determine the number of FRs each having a capacity of 1 GWe that would be required to consume actinides and plutonium derived from spent LWR fuel.	4.2	
(2) Conduct systems analyses to compare attributes of the UREX+3c fuel cycle and the AFC.		
(a) Conduct comparisons of significant characteristics of the systems and associated major components of the UREX+3c fuel cycle and the AFC:	5.0 6.0 7.0	8.0 9.0

Proposed Project Activity	Report Section References
<ul style="list-style-type: none"> For LWR fuel design: assess the effects of (i) differences in Pu isotopic make-up for 1st, 2nd, and nth recycle of Pu; (ii) variability of enrichment in reuse U; and (iii) U²³⁶ in reuse U 	3.2 Appendix A 3.3 Appendix C 3.4 Appendix D
<ul style="list-style-type: none"> For LWR fuel fabrication: assess the effects of (i) differences in amounts of Am²⁴¹ and Pu²³⁸ for 1st, 2nd, and nth recycle Pu used in MOX; and (ii) U²³² and U²³⁴ in reuse U 	3.2 Appendix A 3.3 Appendix C 3.4 Appendix D
<ul style="list-style-type: none"> For spent LWR fuel transportation: compare the effects of (i) 1st recycle Pu in spent MOX LWR fuels; and (ii) nth recycle Pu in spent MOX LWR fuels 	10.2
<ul style="list-style-type: none"> For UREX+3c process systems: compare the effects of (i) separately processing spent U LWR fuel, spent MOX LWR fuel, and burned FR fuel using batch processing; and (ii) separately processing spent U LWR fuel, spent MOX LWR fuel, and burned FR fuel using independent process lines 	3.1 Appendix C 3.4 Appendix D 3.5 Appendix E 4.2
<ul style="list-style-type: none"> For fission product (FP) wastes: assess the effects of separately processing spent U LWR, spent MOX LWR, and burned FR fuels 	4.2 8.0 5.0
<ul style="list-style-type: none"> For FR systems: compare the effects of (i) 1st, 2nd, and nth LWR recycle of Pu in LWRs on amounts of actinide isotopes (Np/Am/Cm) requiring FR targets and associated impacts on FR requirements; and (ii) 1st, 2nd, and nth LWR recycle of Pu in LWRs on amount of Pu from LWRs used in FR fuel/targets and associated impacts on FR requirements 	3.1 3.5 4.2 Appendix E
<ul style="list-style-type: none"> For Pu inventory management: compare the effects of 1st, 2nd, and nth LWR recycle of Pu on requirements for storage, transport, and protection of U:Pu from the UREX process 	4.2 6.0
<ul style="list-style-type: none"> For recovery and reuse of the excess U: compare the (i) consequences of disposing excess U from UREX; (ii) effects and feasibility of direct reuse (no additional processing) of excess U from UREX in LWR MOX fuel; (iii) effects of selection and blending of excess U from UREX to produce a standard recycle U product for use in LWR MOX fuel; and (iv) effects of re-enriching recycle U to produce a low enrichment (e.g., 2.4 wt% U²³⁵) U product for use in LWR MOX fuel. 	5.0 9.0 10.1
(b) Collect the results of the above comparisons in an integrated systems analysis of the UREX+3c fuel cycle and the AFC	
<ul style="list-style-type: none"> Perform health and safety evaluations to assess system-wide effects on radiological dose to workers and the public from normal operations and accidents 	7.0
<ul style="list-style-type: none"> Perform safeguards and security evaluations of the effects of the modified AFC on transportation security for SNM, SNM inventory security, safeguards, and security of nuclear facilities 	6.0
<ul style="list-style-type: none"> Assess radioactive waste management consequences for the (i) volume and classification of low-level radioactive waste (LLW); (ii) volume and classification of transuranic wastes (TRU); (iii) management of Cs/Sr high-level radioactive waste (Cs/Sr-HLW); (iv) disposal of reactor fuel-assembly hulls and dissolver sludge from chemical processing; (v) disposal of long-half-life-fission-products HLW; and (vi) disposal of residual actinide-isotopes HLW or TRU 	8.0

Proposed Project Activity	Report Section References
<ul style="list-style-type: none"> Analyze the economic consequences of the modified AFC (use information from the “Advanced Fuel Cycle Cost Basis”, April 2007, and other sources as available): (i) exclude capital, operating, and D&D costs for LWRs but include incremental costs associated with use of MOX fuel; (ii) estimate development and demonstration costs for AFC processes and systems; (iii) estimate capital costs of AFC facilities and equipment; (iv) estimate operating costs of AFC facilities and equipment; (v) estimate costs of materials and services for AFC; and (vi) estimate decommissioning costs for AFC facilities and equipment 	5.0
<ul style="list-style-type: none"> Analyze resource utilization and sustainability including estimate of (i) electric power generated per ton of natural uranium used; (ii) high-level radioactive waste repository space required per 100 GWe-yr; (iii) electric power generated per dollar of initial investment in R&D, plant, and equipment; (iv) electric power generated per dollar spent on operations, materials, and services; and (v) D&D costs per 100 GWe-yr 	9.0
<ul style="list-style-type: none"> Assess the affordability of the modified AFC by comparing the schedule of expenditures that would be required for implementation to revenues and savings that would accrue over the full life cycle. Expenditures to be estimated encompass: (i) research, development and demonstration; (ii) design, licensing, and construction; (iii) operations and maintenance; (iv) materials, supplies, and services; (v) waste management and disposition; (vi) decommissioning; and (vii) charges on expended funds. Revenues and savings include: (i) sales of LWR MOX fuel to utilities; (ii) Nuclear Waste Fund (NWF) receipts for management of LWR spent fuel; and (iii) receipts from FR electric power generation. 	5.0
(3) Prepare a report that describes the work performed and the results obtained.	This is the Final Report