Managing Wastes With and Without Plutonium Separation

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SUMMARY

This study examines whether reprocessing and plutonium recycle will make radioactive waste management more effective and economical. It compares the wastes generated in three alternative nuclear fuel cycles. The first cycle is low-enriched uranium in once-through mode (LEU-OT), which is the choice followed by the great majority of the civilian nuclear reactor operators in the world. The second cycle is mixed-oxide fuel in once-through mode (MOX-OT), which reprocessing-pursuing countries currently prefer. The third cycle is self-generating recycle (SGR) where plutonium is reprocessed and recycled repeatedly in the reactor throughout its operating life. Although current cost picture and the cost trends make SGR unlikely, it is included so that one can see its advantages and disadvantages for potential use in the future.

This study compares wastes from alternative fuel cycles that generate the same amount of electricity. Also, as opposed to focusing only on the back-end waste, this study estimates the wastes at various radioactive levels generated in every fuel cycle step: uranium mining and milling, conversion, enrichment, fabrication, reactor operations, waste storage, reprocessing, high-level waste vitrification, and spent fuel encapsulation. It also uses total waste disposition cost as a proxy for evaluating whether reprocessing "eases waste management"-- the cheaper are the sum of the costs of conditioning and disposal of wastes generated in these steps, the "easier" is the waste managed.

For high-level waste (HLW) and spent fuel, this study estimates their heat generation which is the factor that most affects their disposition costs. For low-level waste (LLW), intermediate-level waste (ILW) and mill tailings, the key cost driver is the waste volume, which is also provided by this study. Finally, it estimates and compares the total waste disposition costs of the three fuel cycles -- LEU-OT, MOX-OT, and SGR.

The study arrived at the following findings:

- o A plutonium fuel cycle generates about 20-30% less in the waste volume of mill tailings, but 5-10% more low-level waste and 90%-150% more intermediate-level waste relative to the current uranium once-through fuel cycle.
  - -- In the MOX-OT case, the tailings are 23% less than LEU-OT, LLW is 6%-8% more, and the ILW is 88% to 115% more.
  - -- In the SGR case, the tailings are 32% less, LLW is 5%-9% more, and the ILW is 123% to 149% more.

- o A plutonium fuel cycles generates from 7% less to 44% more heat from high-level waste and spent fuel than that from the spent fuel of the current uranium once-through fuel cycle.
  - -- In the MOX-OT case, the HLW and MOX-OT spent fuel generate 4% to 10% more heat than the LEU-OT spent fuel.

-- In the SGR case, the HLW generates from 7% less heat to 44% more heat than LEU-OT spent fuel, depending rather sensitively on when and how (equivalent energy density or average power density) heat is compared.

- The total waste disposition cost for a plutonium fuel cycle is higher than that of the current uranium once-through fuel cycle by 20% to 25%.
  - -- In the MOX-OT case, the cost is higher than the LEU-OT case by 21%-25%.
  - -- In the SGR case, the same cost is higher by 20%-22%.

- If the total waste disposition cost is used as a proxy for quantifying the easiness or difficulty in managing waste, this study found that the reprocessing and the use of plutonium actually make waste management more difficult.

- There is a potential to reduce the disposal cost in a repository by aging the spent fuel and high level waste longer in temporary storage in order to make them cooler.

This study found that a more attractive way to lower the waste disposition cost is to determine the optimal age at which the spent fuel or HLW should generally be transferred to the repository. The selection will be based on the tradeoff of the cost of interim storage versus the cost of final repository storage, which can depend sensitively on heat and thus the age of the HLW or spent fuel. The study recommends that such an optimal timing be studied and implemented; by taking into account also other factors such as the spread in the discharged-fuel's ages and characteristics and the availability of interim and permanent storage.

The United States and other countries need to question the often-quoted statement that the reprocessing of spent fuel and the recycling of plutonium will ease radioactive waste management problems. They should not proceed with plutonium activities based on this unsubstantiated benefit.

CONTENTS

I. INTRODUCTION

This study examines whether reprocessing will make radioactive waste management more effective and economical. Although "the United States does not encourage the civil use of plutonium,"[1] the U.S. policy can change in the future. There have been reprocessing advocates in the nuclear industry, the Congress and other parts of the government that urge the U.S. to use reprocessing as a vital step of waste management and to use plutonium as fuel. Also, countries such as Japan, France, United Kingdom and Russian often express the view that reprocessing helps waste management, and this is a key reason why they favor reprocessing. If reprocessing does not help, then in conjunction with the continuing dim economic prospect of plutonium use, the rationale for plutonium commercialization would be weakened.

This study compares the wastes generated in three alternative nuclear fuel cycles. The first cycle is low-enriched uranium in once-through mode (LEU-OT), which is the choice followed by the great majority of the civilian nuclear reactor operators in the world. The second cycle is mixed-oxide fuel in once-through mode (MOX-OT), which reprocessing-pursuing countries currently prefer. The third cycle is self-generating recycle (SGR) where plutonium is reprocessed and recycled repeatedly in the reactor throughout its operating life. Although current cost picture and the cost trends make SGR unlikely, it is included so that one can see its advantages and disadvantages for potential use in the future.

As opposed to focusing only on the back-end waste, this study estimates the waste generated in every step from cradle to grave of the fuel cycle, and it compares cycles with and without plutonium separation. It uses total waste disposition cost as a proxy for evaluating whether reprocessing "eases waste management"-- the cheaper are the sum of the costs of conditioning and disposal of wastes generated in these steps, the "easier" is the waste managed.

For high-level waste (HLW) and spent fuel, this study estimates their heat generation which is the factor that most affects their disposition costs. For completeness, this report also gives the radioactive waste types and their volumes. For low-level waste (LLW), intermediate-level waste (ILW) and mill tailings, the key cost driver is the waste volume, which is also provided by this study. Finally, it estimates and compares the total waste disposition costs of the three fuel cycles -- LEU-OT, MOX-OT, and SGR.

[1] *President's Nonproliferation and Export Control Policy: Fact Sheet*, September 27, 1993.

II. WASTE MANAGEMENT

A nuclear fuel cycle consists of various steps. Table 1 shows the typical steps of a once-thru cycle and a cycle where the spent fuel is reprocessed and the plutonium recycled. The world's nuclear power plants run predominately on a once-through cycle, in which no plutonium is used as fuel and the spent fuel will eventually be disposed of without first separating the plutonium or uranium. There are a number of fuel cycles that use reprocessing. Shown in Table 1 is one that uses plutonium in part of the reactor's core and that plutonium and uranium are extracted before the high level waste is placed in permanent repository.

Table 1: Fuel Cycle Steps of Two Typical Nuclear Fuel Cycles

STEP	Once-Thru	Reprocess
Mining	*	*
Milling	*	*
Conversion to UF_6	*	*
Enrichment	*	*
UO_2 fabrication	*	*
MOX fabrication		*
Power reactor operation	*	*
Spent fuel storage	*	*
Reprocessing		*
Pu, HLW storage		*
HLW vitrification		*
Spent fuel encapsulation	*	
Final disposal	*	*

The fuel cycle begins when uranium is mined from the ground. During milling operations, uranium (U_3O_8) or yellow cake is removed from the ore by chemical and physical means. The ore residues containing chemical effluents and natural radioactivity particularly radon are called mill tailings. Radioactive waste is also generated during conversion, enrichment and fabrication. For example, there are scrap materials still consisting plutonium and uranium, and enrichment tailings containing depleted uranium. During and after power reactor operations, radioactivity remains in three sources. First is the fission products resulting from nuclear fission taken place in the reactor. Typical long-life nuclide fragments with the highest radioactivity are Cesium-137, Strontium-90, and their daughters Barium-137 and Yttrium-90. Also generated are tritium and noble gases such as Krypton-85. The second source is actinides which are uranium and transuranic (TRU) elements mainly neptunium, plutonium, americium and curium. The third source of radioactivity is activation products such as those resulting from neutron irradiation of

chemical additives (e.g. boron in neutron poison), structural material (e.g. nickel in control rod and cobalt in reactor vessel), and impurities (e.g. nitrogen in oxide fuel). Thus, many radioactive elements of different intensities and half-lives are generated throughout the nuclear fuel cycle.

The great variety and complexity of radioactive elements preclude the possibility of managing waste according to these elements' types and amounts. Rather, radioactive waste is classified into five categories which can be handled, stored and disposed of differently:

Spent fuel	discharged, irradiated fuel (before plutonium/uranium separation or no separation).
High level waste	primary stream of waste (liquid or its solidified form) after reprocessing.[2] Still contains minor actinides[3] and traces of uranium and plutonium.
Inter. level waste	waste contaminated with alpha-emitting transuranic radionuclides with half-lives greater than 20 years and a total concentration of such radionuclides in excess of 0.1 Curie per metric ton (Ci/t) of waste at the time of assay.
Mill tailings	ore residues from milling after uranium is extracted.
Low level waste[4]	none of the above.

The category of intermediate level waste (ILW) is the same as DOE's transuranic waste.[5] We prefer the term ILW, because it is a more common term used internationally. The United Kingdom uses similar classification, except that the ILW is above 0.108 Ci/t (i.e. 4 GBq/t or 4 billion disintegrations per second per metric ton), instead of 0.1 Ci/t alpha, or that the ILW is above 12 GBq/t beta and gamma.[6] Also, there is an added

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[2] The waste forms include liquid, sludge, salt cake, slurry, calcine, precipitate, zeolite, glass and capsules of separated strontium and cesium. Oak Ridge National Laboratory, *Integrated Data Base Report--1995: U.S. Spent Nuclear Fuel and Radioactive Waste Inventories, Projections, and Characteristics*, DOE/RW-0006, Rev. 12, December 1996, p. 6.

[3] Minor actinides are actinides other than uranium and plutonium.

[4] Depleted uranium (DPU) from enrichment is in this category, although its waste management cost may differ from those of other LLWs. In this study, we will estimate the DPU waste management cost separately from those of the rest of the LLWs.

[5] The definition used by DOE for transuranic waste appears in Oak Ridge National Laboratory, *Integrated Data Base Report--1995: U.S. Spent Nuclear Fuel and Radioactive Waste Inventories, Projections, and Characteristics*, DOE/RW-0006, Rev. 12, December 1996, p. 61. The transuranic waste or ILW does not include spent fuel and HLW.

[6] Mike Smith and Nigel Mote, *Identification of Reprocessing Waste Quantities and Characteristics*, December 1997, p.13. The article also cites that the ILW from reprocessing at THORP has alpha and beta/gamma activities of 120 and 38,000 GBq/t respectively.

category, very low level waste, in the U.K. classification. Because the VLLW can be safely disposed of with ordinary refuse,[7] we will not consider this category further. France defines LLW as having alpha activity less than 0.1 Ci/t.[8]

The U.S. also uses four classes to distinguish non-HLW solid radioactive wastes.[9] Class A contains very low levels of long-lived and short-lived radionuclides and can be buried near surface. Class B has higher radioactivity, but can still be buried near surface. Class C has yet higher radioactivity and requires more rigorous waste form to assure stability. These three classes can generally be treated as LLW, although there are exceptions in some countries. For example, some countries might classify those U.S. class C wastes that require remote handling as ILW. Finally, U.S. also uses a category called greater than Class C (GTCC), which has radioactivity higher than Classes A, B and C and probably requires deep geologic disposal. For waste containing transuranics, the definition of GTCC waste is essentially the same as DOE's transuranic waste. It is, however, possible to have GTCC waste that contains little or no transuranic isotopes. This material should also be considered ILW. In any case, little "low transuranic" GTCC is actually generated.

Waste management is to protect workers, the public and the environment from radioactive exposure and other hazards of these wastes. One key measure of the easiness of waste management is the cost of preparing and disposing the waste so that exposure is kept below an acceptable level. The costs for disposing spent fuel (SF) and high level waste (HLW) are driven more by the heat contained in the waste than its volume. On the other hand, as a cost driver, waste volume[10] is more important than heat content in disposing of mill tailings, low level waste (LLW), and intermediate level waste (ILW). In the subsequent chapters, we will provide the waste volumes for all waste types, although the readers are cautioned not to simply add up these volumes and use the sum as a proxy for waste disposition cost, because the key cost driver for SF and HLW is not the volume, but heat. In Chapter V, the heat of SF and HLW will be compared. Moreover, one should not even add up volumes other than SF and HLW, because these other waste categories still have different waste disposition costs. What one can do is to sum the volumes within each waste category, LLW or ILW etc, because their waste disposition characteristics are similar.

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[7] P. D. Wilson, *The Nuclear Fuel Cycle: From Ore to Waste*, Oxford University Press, 1996, p. 162.

[8] Mike Smith and Nigel Mote, *Identification of Reprocessing Waste Quantities and Characteristics*, December 1997, p.13. See also J.L. Ma et al, "Development of Advanced Device for Low Level Waste Assay Based on 14 Mev Neutron Interrogation," Nuclear Materials Management, 35th Annual Meeting Proceedings, Volume XXIII, Institute of Nuclear Materials Management, p.762. The article says that, in France, the acceptance threshold for the final package (apparently LLW) in a surface storage is that the "alpha emitters" (i.e. U, Pu and Am) have an activity, evaluated 300 years later, lower than 0.1 Ci/t.

[9] Robert McKee, Michael White and Kenneth Ames, "Recycle Waste Stream Analysis," Pacific Northwest National Laboratory, June 1997, pp. 7-8. See their report for a more detailed discussion of the four waste classes.

[10] To be precise, this is the compacted waste volume. It is the volume that has been reduced by chemical or physical means with reasonable economic expenses.

III. METHODOLOGY

This chapter describes how wastes from alternative nuclear fuel cycles will be compared. Two common shortcomings should be avoided. First is an incomplete comparison. One only compares the waste before and after the plutonium and uranium are separated. Of necessity, the spent fuel (before or without separation) is more radioactive and hotter than the HLW (waste after separation), because taking something out has to reduce both radioactivity and heat. This approach inevitably leads to the conclusion that reprocessing always took away some radioactivity and heat and could only "help" waste management. The problem of this approach is that the reprocessing process itself generates waste. Also, other steps of the fuel cycle such as MOX fabrication also generates waste. An equitable comparison would require the tallying of all wastes generated during the full fuel cycle. A second shortcoming in methodology is to add up all the waste volume arisings regardless of radioactive levels. Again, since the largest waste volume comes from mill tailings, a reprocessed cycle with less mining and milling would have much less waste volume and thus under this methodology would seem to help waste management. This comparison is inappropriate, because some wastes such as mill tailings, although voluminous can be taken care of cheaply. One should add up the waste disposition costs of handling wastes in different categories, but not to add up the waste volumes themselves.

The methodology adopted for this study will include all wastes generated throughout the fuel cycles. While the waste volumes, radioactivity and/or heat will be indicated for various waste arisings, we will eventually use the waste disposition cost as a proxy to indicate the "ease" of waste management. If the total disposition cost of the wastes from fuel cycle A is lower than that of fuel cycle B, it is an indicator that it is easier to manage the cycle A's waste. Other proxies to measure the easiness of waste management, which are not included here, are the risk and level of worker and public exposures.

The purpose of a civilian power generating fuel cycle is to generate energy, and waste is an unavoidable consequence of the process. It is both appropriate and convenient to measure the resulting waste per unit of electricity generated. Thus, the waste volume would be in units of m^3/GWe-yr; and waste disposition cost in units of $/GWe-yr.[11]

For this study, we will make two comparisons:

A. Once-through low-enriched uranium fuel cycle (LEU-OT) versus once-through mixed-oxide fuel cycle (MOX-OT), and

[11] Gigawatt (electric)-year. A typical 1.3 GWe power plant operating at 80% capacity factor would generate 1.04 GWe-year of electricity.

B. Once-through low-enriched uranium fuel cycle (LEU-OT) versus self-generating recycle (SGR).

Comparison A

- In the once-through LEU case, the fuel is low-enriched uranium, and the spent fuel is encapsulated and disposed of in a repository, without first recovering the plutonium and uranium.

- In the once-through MOX case, an average of 14% of the fuel is in MOX, and the remaining 86% will continue to be LEU. The MOX spent fuel will be disposal of without plutonium or uranium recovery, while the LEU spent fuel will be reprocessed for plutonium and uranium first before final disposal. Depleted uranium is used in MOX fuel and reprocessed uranium, in LEU fuel. The total uranium savings are 23%. The enrichment savings are 13%.

Comparison A will compare the wastes from once-through LEU and once-through MOX. In the latter, the recovered plutonium and uranium are used only once in refueling. The MOX spent fuel will be disposed of without extracting its plutonium and uranium. As uranium price remains low, plutonium reprocessing and fabrication costs stay high, and plutonium's fuel value in thermal reactor declines after repeated irradiation, it is more economical to use plutonium that has been in the reactor only once. In fact, once-through MOX is becoming the recycle mode of choice in France, one of the most ardent countries for plutonium reprocessing. This calls for a comparison of the two once-through fuel cycles, especially when once-through MOX mode might continue indefinitely in countries favoring plutonium recycling.

Comparison B

- In the self-generating recycle case, an equilibrium state has been reached in which the power reactor uses and only uses all the plutonium from the reactor's previous cycle for a subsequent fuel reload. The HLW will be vitrified and placed in a repository. MOX fuel is used in 24% of the reactor core, and the remaining 76% core will continue to use LEU fuel.[12] As in the MOX-OT case, we assume that depleted uranium is used in the MOX fuel, resulting in further savings in natural uranium. Reprocessed uranium is also assumed to be used in LEU fuel for yet more uranium savings. The total uranium savings due to the use of reprocessed plutonium, depleted uranium, and reprocessed uranium are 32%. The savings in enrichment are 23%, which are lower than 32%, because the enrichment level for the reprocessed uranium fuel needs to be about 10% higher in order to compensate for the U^{236} neutronic penalty.

[12] Many numbers in this report are not rounded, so that the calculations are easier to trace.

Based on the current nuclear costs and trends, MOX-OT is more likely than SGR. But, this study will still compare SGR with LEU-OT, because it is a traditional comparison and some plutonium proponents still hope that some day plutonium will be recycled more than once in either thermal reactors or fast reactors. Comparison B can shed some light on such a future of multiple times of plutonium recycle. Since the early days of nuclear power, many people have envisioned that fissile materials would be recovered from spent fuel and used in new fuel. This repeated plutonium recycling would be viable, when uranium price were to rise rapidly, the costs of reprocessing and fabrication plutonium-containing fuel declined significantly, and the proliferation risk were low.

For all three alternative fuel cycles, the fuel is assumed to have a burnup of 42.5 GWD/MTHM[13] and the reactor, a 33% thermal efficiency. These are typical characteristics of a modern light water reactor. Twenty-six metric tons of uranium are required to generate 1 GWe-yr of electricity.

[13] Gigawatt days (thermal) per metric ton of heavy metal (i.e. uranium and plutonium).

IV. FRONT-END AND REACTOR-OPERATION WASTES

In the next four chapters, we will tally the wastes generated in the three alternative fuel cycles-- once-through LEU, once-through MOX and self-generating recycle. This chapter covers wastes from mining and milling, conversion, enrichment, fabrication and reactor operations.

Mining and Milling

In mining operations, rocks are moved in order to access the uranium ore. These rocks contain little uranium and are not milled. They are often returned to the pits and not considered as waste materials.

On the other hand, conventional milling operations do generate large volume of tailings. In 1996, 82% of total Western milling was conventional.[14] The tailings volume depends most critically on the ore grade. NAC's analysis yields tailings volumes range from 50 to 800 m^3/MTHM; with Canadian productions from 50-250 m^3/MTHM and non-Canadian productions from 250-800 m^3/MTHM. Canadian uranium is generally of higher grade and its milling results in lower tailings volume. In 1996, 42% of the uranium production came from Canada and had an average ore grade of 1.32%, while the remaining 58% of the non-Canadian production had a grade of only 0.21%. An unconventional mining method, in-situ leach (ISL), can also be used in extracting uranium but generating little tailings. In 1996, 18% of the total Western milling was ISL.

Based on the above data for 1996, one can determine a weighted average tailings volume. About 20% (ISL) of the milling generates no tailings. Of the remaining 80% (conventional milling), 40% of which came from Canada and generates 50-250 m^3/MTHM of tailings volume and 60% of which from non-Canada generates 250-800 m^3/MTHM. These numbers give a weighted average of 300 m^3/MTHM.[15]

A typical 957 MWe nuclear power plant operating at 75% capacity factor requires a reload of 18.4 MTHM of 3.7%-enriched uranium annually. The plant generates 0.72 GWe-yr of electricity in a year. Thus, 1 GWe-yr would require 26 MTHM of 3.7% U^{235} with burnup to 42,500 MWD/MTHM. With a tails assay of 0.3% U^{235}, the feed-to-product ratio is 8.3, or 8.3 MT of natural uranium to produce 1 MT of 3.7% U^{235}.[16]

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[14] The estimate of mill tailings volume is based on analysis by Mike Smith and Nigel Mote of NAC International, September 1997.

[15] 0.2*0+0.8*0.4*(50+250)/2+0.8*0.6*(250+800)/2=300

[16] This tails assay of 0.3% is higher than the historic one, because higher grade ore has been found and used and uranium price has been low in recent years. Moreover, reprocessing would not be economical until a much higher uranium price is reached. But at a higher uranium price, the optimal tail assay would be lower, and a lower assay would lead to some uranium savings even for the once-through LEU. This

Therefore, the mill tailings volume is 65,000 m^3/GWe-yr.[17] This figure is consistent with the 64,400 m^3/GWe-yr derived from a Nuclear Energy Agency report.[18] On the other hand, our figure is considerably higher than the 20,000 m^3/GWe-yr given in a BNFL article.[19] If much higher grade uranium is used in the future, the waste volume can be brought down to 40,000 m^3/GWe-yr.[20] On the other hand, the Department of Energy provided a figure of 118,000 m^3/GWe-yr apparently based on very low grade ore.[21] In this study, we will use a value of 65,000 m^3/GWe-yr.

When plutonium and uranium are reprocessed from spent fuel and used to refuel the reactor, less (virgin) natural uranium would be required. The amount of mill tailings would also be reduced correspondingly. In the case of self-generating recycle, 24% of the reactor core uses MOX fuel, and the use of reprocessed plutonium and depleted uranium would reduce natural uranium requirement by 24%.[22] If reprocessed uranium is also used in the LEU fuel, the reduction would be increased to 32%.[23] Economically, the reprocessed uranium might not be used at all, because the added cost in handling the more radioactive reprocessed uranium and the higher level of enrichment might be more than the savings in uranium. In this study, we however use 32% as the reference case for self-

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study uses a high 0.3% tails assay to ensure that the benefits of reprocessing is not underestimated in this regard.

[17] 300*26*8.3=64,740~65,000.

[18] *The Economics of the Nuclear Fuel Cycle*, Nuclear Energy Agency, Organisation for Economic Co-Operation and Development, 1994, p. 34. The report lists that 1,800 m^3 of ore is required to generate $3.5x10^8$ kwhr of electricity. This translates into 45,056 m^3/GWe-yr. Smith and Mote observed that the ore density in the report was assumed to be 2 gm/cm^3 or 17.7 ft^3/MT. Yet, mill tailings have a density of 25.3 ft^3/MT. Adjusting for the difference in density, one gets 64,400 m^3/GWe-yr, which is consistent with the 65,000 m^3/GWe-yr used here.

[19] D.M. Beaumont et al, "The Environmental Benefits of MOX Recycle," British Nuclear Fuel Limited, in Proceedings of the Fifth International Conference on Radioactive Waste Management and Environmental Remediation, Volume 1, September 3-7, 1995, p. 521. The article assumes an uranium ore grade of 0.4% and a density of 2 gm/cm^3. The lower grade and density are far from sufficient to account for the large difference between 20,000 and 64,400 m^3/GWe-yr.

[20] This is based on a projection of 60% of U_3O_8 from 0.3% ore, 25% from 10% ore and 15% from in situ leaching. "Nuclear Fuel Cycle Project: Waste Volume Data for Direct Disposal and Once-Through MOX Cycles," Don Habib, Project Performance Corporation, October 22, 1997. This projection assuming better uranium grades is relatively more favorable to the LEU-OT case than the MOX-OT and SGR cases.

[21] U.S. Department of Energy, *Integrated Data Base for 1992: U.S. Spent Fuel and Radioactive Waste Inventories, Projections, and Characteristics*, Oak Ridge National Laboratory, DOE/RW-0006, Rev. 8, 1992. The DOE number also appears on Table 22.1 of Raymond Murray, *Nuclear Energy*, 4th Edition, Reed Educational and Professional Publishing Ltd, p. 300.

[22] The self-generating recycle is based on characteristics given in *Plutonium Fuel: An Assessment*, Nuclear Energy Agency, OECD, 1989, p. 43. The equilibrium core has 28.8% MOX fuel and 71.2% LEU fuel, if natural uranium is used in the MOX fuel. For this study, we assume that depleted uranium is used in the MOX fuel. Then, we estimate that the equilibrium core has 24% MOX fuel and 76% LEU fuel.

[23] This estimate is based on that reprocessed uranium has an enriched level of 0.68% of U^{235}. Also, a 10% higher enrichment or 4.1% U^{235} would be required to compensate for the U^{236} neutronic penalty in the reprocessed uranium. Since it takes 10 kg of 0.68% U^{235} to make 1 kg of 4.1% U^{235} with tails assay of 0.3%, the use of reprocessed uranium would produce a 10% savings in LEU fuel. In the self-generating recycle, 76% of the fuel is LEU and thus the savings are 7.6%. The combined savings of using reprocessed plutonium, depleted uranium and reprocessed uranium are 32% (i.e. 24% plus 7.6%).

generating recycle to ensure that the benefits of recycle are not understated. In other words, the mill tailings for the SGR case are 32% less.

In the once-through MOX (MOX-OT) case, the uranium savings due to the use of reprocessed plutonium and depleted uranium are 14%.[24] If reprocessed uranium is also used in LEU reloads, the uranium savings would be increased to 23%. Thus, the mill tailings for the MOX-OT case are 23% less.

Since mill tailings for all three alternative fuel cycles are often disposal of when the milling plant and site are closed down, we include the decommissioning wastes for milling here.

Conversion Waste

The conversion to uranium hexafluoride (UF_6) starts with the dissolution of the yellowcake (U_3O_8) in nitric acid, and the filtering and treating the solution with chemical solvents. The resulting uranyl nitrate will be more than 99.95% pure. It is then converted to uranium oxide and to volatile UF_6 for the enrichment process. During these processes, wastes are generated in the forms of dry active waste, crushed drums and scrap metal, calcium fluoride sludge, mixed waste, fluorinator reactor ash and raffinate sludge. Currently, there is only one operating commercial uranium conversion plant in the U.S. The facility is operated by Converdyn and is located in Metropolis, Illinois. It generates 32 m^3 of LLW/GWe-yr (See Table 2). Data at a large commercial conversion plant in Gore, Oklahoma, which ceased operations in 1992 and is now undergoing decommissioning, indicated considerably larger amount of LLW waste, 112 m^3/GWe-yr. The amount of conversion waste depends on what processing procedures are used and how the wastes are treated and compacted. DOE provided a value of 10.4 m^3/GWe-yr, which perhaps is based on less stringent environmental regulations.[25] For this study, we use a range of 32-112 m^3 of LLW/GWe-yr for the LEU once-through fuel cycle. For the other cases-- MOX-OT and SGR, the proportional reductions are the same as indicated in the subsection on Mining and Milling.

Enrichment Wastes

Two types of waste are generated during enrichment. Of much larger volume is the depleted uranium hexafluoride (DUF_6), left in the enrichment process. It can be chemically converted to the stable U_3O_8 for disposal or reused. The volume is estimated at 32 m^3/GWe-yr.[26] The other waste type includes a variety of solid and liquid materials

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[24] Based on our ORIGEN2 runs discussed in Chapter V.

[25] U.S. Department of Energy, *Integrated Data Base for 1992: U.S. Spent Fuel and Radioactive Waste Inventories, Projections, and Characteristics*, Oak Ridge National Laboratory, DOE/RW-0006, Rev. 8, 1992. The DOE number also appears on Table 22.1 of Raymond Murray, *Nuclear Energy*, 4th Edition, Reed Educational and Professional Publishing Ltd, p. 300.

[26] "Nuclear Fuel Cycle Project: Waste Volume Data for Direct Disposal and Once-Through MOX Cycles," Don Habib, Project Performance Corporation, October 22, 1997.

contaminated by uranium, such as air filters, wipes, personnel protective equipment, wastewater sludge and laboratory waste. It amounts to 7 m^3/GWe-yr. Thus, the total amount of enrichment waste is 39 m^3 of LLW/GWe-yr in the once-through LEU case. On the other hand, DOE provided a very low figure of 3.5 m^3/GWe-yr.[27] This study uses a range of 3-40 m^3/GWe-yr.

The enrichment waste volumes for the MOX-OT and SGR cases are 13% and 23% less respectively.

Table 2: Radioactive Wastes From Uranium Conversion Operations

	Waste Volume in m^3/GWe-yr	
Waste Type	Plant at Metropolis, IL	Plant at Gore, OK
Dry active waste	0.9-2.2	5.2
Crushed drums & scrap metal	0.44	0.04
Calcium fluoride sludge	24	32
Mixed waste	0.01	no data
Fluorinator reactor ash	6.9	0
Raffinate sludge	0	75
Total	~32	112

Source: "Uranium Conversion Wastes and Costs," Don Habib, Project Performance Corporation, July 16, 1997.

Fabrication Waste

Fuel fabrication typically includes the sintered pelletization of UO_2 and/or PuO_2, and its cladding into fuel rods. The wastes are the discarded processing equipment and materials contaminated with uranium and plutonium.[28] During 1996, five commercial UO_2 fuel fabrication plants operated in the U.S. with a total capacity of 3,900 MTHM/yr. Waste data are available at the two largest plants in Columbia, South Carolina (1,150 MTHM/yr) and Wilmington, North Carolina (1,200 MTHM/yr). The wastes generated are 3 and 9 m^3 of LLW/GWe-yr respectively at these two plants.[29] This is the range used

[27] U.S. Department of Energy, *Integrated Data Base for 1992: U.S. Spent Fuel and Radioactive Waste Inventories, Projections, and Characteristics*, Oak Ridge National Laboratory, DOE/RW-0006, Rev. 8, 1992. The DOE number also appears on Table 22.1 of Raymond Murray, *Nuclear Energy*, 4th Edition, Reed Educational and Professional Publishing Ltd, p. 300.

[28] There are also effluents that are vented to the environment. This study does not include off-gas streams in the fabrication and other steps of the fuel cycle.

[29] "Nuclear Fuel Cycle Project: Waste Volume Data for Direct Disposal and Once-Through MOX Cycles," Don Habib, Project Performance Corporation, October 22, 1997.

in this study for UO_2 fabrication waste.[30] For the MOX-OT and SGR cases, the UO_2 fabrication waste volumes are 14% and 24% less respectively.

As to MOX fabrication waste, we derive a value of 505 m^3 of ILW and 191 m^3 of LLW from MOX fabrication plant's annual operation.[31] Because the plant has a throughput capacity of 150 MT/yr,[32] the waste volumes would be 87 m^3 of ILW/GWe-yr and 33 m^3 of LLW/GWe-yr.

Since MOX-OT uses only 14% MOX fuel in the reactor core, the corresponding values are 13 m^3 of ILW/GWe-yr and 5 m^3 of LLW/GWe-yr. For SGR using 24% MOX fuel, the values are 21 m^3 of ILW/GWe-yr and 8 m^3 of LLW/GWe-yr.

Reactor Operation Waste

Most of the radioactive waste generated during reactor operations remains in the irradiated fuel. Staying outside of the fuel, there is a small amount of radioactive waste resulting from neutron activation of structural materials, corrosion products and chemical additives. Fuel cladding failure also allows a minute fraction of the oxide fuel leaks into the primary coolant or the fuel storage pool coolant. These dissolved or suspended solids are removed from the liquid streams before they are reused or discharged, and form the waste of reactor operations.

The World Association of Nuclear Operators (WANO) publishes various performance indicators. One of them is the volume of low-level solid radioactive waste, which measures such waste generated during reactor operations. For pressurized power reactors worldwide, the wastes were 71, 50 and 59 m^3/unit for 1994, 95 and 96

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[30] Because we expect the waste volume for UO_2 fabrication to be on a down trend, our range does not include the high value of 87 m^3/GWe-yr shown in U.S. Department of Energy, *Integrated Data Base for 1992: U.S. Spent Fuel and Radioactive Waste Inventories, Projections, and Characteristics*, Oak Ridge National Laboratory, DOE/RW-0006, Rev. 8, 1992. The DOE number also appears on Table 22.1 of Raymond Murray, *Nuclear Energy*, 4th Edition, Reed Educational and Professional Publishing Ltd, p. 300.

[31] Nuclear Materials and Reconfiguration Technology Programs, Los Alamos National Laboratory, *New Mixed Oxide Fuel Fabrication Facility Data Report for the Fissile Material Disposition Program Programmatic Environmental Impact Statement.* On page 67 of the report, a table provides both the generated and post-treated waste volumes. Each entry is, however, given in terms of less than, as opposed to equal to, a certain quantity. DOE uses these less-than quantities as if they were equal-to quantities; apparently for the reason that, although the volumes are indicated as less than certain amounts, they are probably close to those amounts. (DOE's Office of Fissile Materials Disposition, *Storage and Disposition of Weapons-Usable Fissile Materials Final Programmatic Environmental Impact Statement*, Volume III, December 1996, p. E-165.) We follow DOE's assumption and replace the less than sign with an equal sign. On the other hand, we use the post-treated quantities, instead of the generated quantities appearing in the DOE report, because the treated waste volumes are the more relevant ones in this study. Further, we add up Los Alamos' 630 yd^3 of transuranic solid waste and 30 yd^3 of mixed transuranic solid waste as ILW; and its 200 yd^3 of low-level solid waste and 50 yd^3 of mixed low-level solid waste as LLW. Los Alamos does not find much liquid ILW and LLW after these wastes are treated. We have also converted the waste volumes from yd^3 to m^3.

[32] Ibid. p. 4.

respectively.[33] The average was 60 m^3/unit. To convert to m^3/GWe-yr, one needs to know the average GWe-yr in each PWR unit during those years. Smith and Mote found that to be 0.7 GWe-yr per unit.[34] Thus, the waste was 86 m^3/GWe-yr (i.e. 60 m^3/unit times unit/0.7 GWe-yr). U.S. power reactor operations during 1993-1995 show somewhat larger numbers for LLW, 162 to 145 and to 137 m^3/GWe-yr.[35] These larger numbers are probably caused by the larger volume of LLW generated in a boiling water reactor, as opposed to a PWR. The limited data indicates the possibility of a downward trend.[36] This trend should be reinforced in the U.S. by the skyrocketing rise of the LLW disposal cost, because the higher is the disposal cost, the more is the incentive to reduce the waste volume. We use a range of 86-130 m^3 of LLW/GWe-yr in this study.

The same WANO report did not provide a value for ILW. On the other hand, Beaumont et al give a number of 50 m^3/GWe-yr for ILW, but use a considerably larger number of 200 m^3/GWe-yr for LLW.[37] Because the WANO number reflects the actual data, the Beaumont numbers might have been too high. Thus, we calculated a range for ILW based on the same ratio of the WANO and Beaumont numbers for LLW and ILW. The range of ILW for reactor operations used in this study is 22 to 33 m^3/GWe-yr (i.e. 86*50/200 to 130*50/200).

For MOX-OT and SGR fuel cycles, the waste volumes during reactor operations should be very similar.

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[33] World Association of Nuclear Operators, "1996 Performance Indicators."

[34] They found the total annual electricity generation for PWRs during those years from "World Nuclear Performance," McGraw Hill Nuclear Publications. The data indicated that the mean annual PWR generation for the population reported by WANO was 0.7 GWe-yr/unit (only reactors in Czech Republic and mainland China were not reported in the McGraw Hill publication.). Thus, the waste is 86 m^3/GWe-yr (i.e. 60/0.7).

[35] Don Habib, October 22, 1997.

[36] The downward trend is reinforced by the data appearing in an older report. DOE showed 165.7 and 466.4 m^3 of LLW/GWe-yr for PWR and BWR respectively in U.S. Department of Energy, *Integrated Data Base for 1992: U.S. Spent Fuel and Radioactive Waste Inventories, Projections, and Characteristics*, Oak Ridge National Laboratory, DOE/RW-0006, Rev. 8, 1992. The DOE numbers also appear on Table 22.1 of Raymond Murray, *Nuclear Energy*, 4th Edition, Reed Educational and Professional Publishing Ltd, p. 300.

[37] D.M. Beaumont et al, "The Environmental Benefits of MOX Recycle," British Nuclear Fuel Limited, in Proceedings of the Fifth International Conference on Radioactive Waste Management and Environmental Remediation, Volume 1, September 3-7, 1995, p. 521.

V. TECHNICAL CHARACTERISTICS AND THE DISPOSAL OF SPENT FUEL AND HLW

Reactor spent fuel contains by far the most radioactivity of any waste in the nuclear fuel cycle. This radioactivity is the result of a wide variety of isotopes, many of which are long-lived. These isotopes must be disposed of in such a way so as to prevent their reaching the environment for many thousands of years. Many countries, including the United States, intend to place this waste in stable geologic formations to ensure safe disposal. The spent fuel can be disposed of directly or it can be first reprocessed and the high level waste (HLW) from the reprocessing solidified and disposed of.[38] In the past, a waste comparison was often made between the directly disposed spent fuel and this HLW which resulted from the reprocessing of LEU fuel. However, this ignored the ultimate fate of the recovered plutonium. It had been thought that this plutonium would be used in breeder reactors but delays and cancellations in breeder reactor programs around the world make this increasing unlikely. Therefore if the plutonium is to be used at all, it would be as a fuel in LWRs but even here the fuel cycle will probably be different than was envisioned 20 years ago.

Comparison A: Once-Through MOX Case vs. Once-Through LEU Case

Twenty years ago, self-generating recycle was the way that it was conceived plutonium would be recycled in LWRs. In this cycle all of the fuel from the LWR is reprocessed and the recovered plutonium recycled into that reactor. Since LWRs are not breeder reactors, the resulting MOX fuel is not enough to provide all of the fuel for the LWR and additional LEU fuel is required. After several recycles, an equilibrium is reached where the consumption of plutonium in the MOX part of the core is matched by the production of plutonium in the LEU part of the core so that the amount of plutonium recovered at the end of each cycle is equal to the amount used in the MOX fuel at the beginning of each cycle. At this point roughly 30 percent of the core is MOX and the rest LEU fuel.

Now, however, a once-through MOX cycle now seems more likely. In this cycle the plutonium recovered from the reprocessing of LEU fuel is made into MOX fuel which is burned in an LWR. The resulting MOX spent fuel is not reprocessed but is disposed of. The reasons for not reprocessing the MOX spent fuel have to do with how the plutonium changes each time it is recycled and with the availability of plutonium for recycling. Each successive time that plutonium is recycled in an LWR there is a decrease in the percentage of Pu 239 and a buildup in the percentage of other isotopes (Pu 238, Pu 240, Pu 241, Pu 242). These other isotopes decrease the plutonium's neutronic value and make fuel fabrication more difficult due to increased radiation. These physics properties of plutonium have, of course, not changed. But twenty years ago it was

[38] In Chapter VI, we will discuss the ILW and LLW generated by reprocessing.

thought that nuclear power would expand quickly and that plutonium would be in short supply. Now, however, there is a vast oversupply of plutonium from the reprocessing of LEU fuel.[39] There is no incentive to reprocess spent MOX fuel to recover second-generation plutonium when one can more easily use already separated first-generation plutonium. Over the past several years this new reality has become increasingly apparent and even the French now say that recovered MOX will not be reprocessed.[40]

For our once-through MOX case we will assume that the MOX burning reactor uses a core which is 30 percent MOX and 70 percent LEU fuel. The MOX spent fuel is disposed of and the LEU fuel is reprocessed and the recovered plutonium is recycled into the MOX burning reactor. Such a reactor will run a significant plutonium deficit. In order to calculate how much plutonium is in the spent fuel (as well as the heat output from the spent fuel and HLW), we have used the ORIGEN2 computer code.[41] Except for standard cases, ORIGEN2 users must supply the input characteristics to the program. Our MOX fuel is made from depleted uranium and we use a 5.31 percent plutonium content.[42] Our LEU fuel is 3.2 percent enriched. Both fuels have a burnup of 33,000 MWD/T.[43] Each metric ton (initial heavy metal) of fresh MOX fuel therefore requires 53.1 kg of plutonium. A proportionate amount of LEU spent fuel (2.33 MT, i.e. 70/30) from the MOX burning reactor contains only 21.6 kg of plutonium. To make up the deficit of 31.5 kg of plutonium will require plutonium obtained by reprocessing LEU fuel from LEU burning reactors. In this case 3.59 MT of LEU spent fuel (3.2 percent enriched, burned to 33,000 MWD/T) from a LEU burning reactor is needed to provide the 31.5 kg of plutonium required. The ratio 3.59/(1 +2.33) provides the number of LEU burning reactors needed to support one MOX burning reactor. This ratio

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[39] See for example, *Problems Concerning the Accumulation of Separated Plutonium*, IAEA-TECDOC-765, Report of an Advisory Group meeting held in Vienna, 26–29 April 1993, International Atomic Energy Agency, Vienna, September 1994.

[40] Statement of the French Industry Secretary, Christian Pierret. See "France to Stop MOX Expansion, Will Seek HLW Disposal Options," *Nucleonics Week*, Vol. 38, No. 43, October 23, 1997, p. 9. This is in conformity with the plans of the French utility, Electricite de France. See: Bernard Esteve, "Aval Du Cycle Du Combustible Nucleaire Positionnement D'Electricite De France A Moyen Et Long Terme," *Global 1995—International Conference on Evaluation of Emerging Nuclear Fuel Cycle Systems*, American Nuclear Society Topical Meeting, September 11–14, 1995, pp. 63–72.

[41] For a description of the model, see: A.G. Croff, "ORIGEN2: A Versatile Computer Code for Calculating the Nuclide Compositions and Characteristics of Nuclear Materials," *Nuclear Technology*, Vol. 62, 1983, p. 335.

[42] Our MOX fuel is based on *Plutonium Fuel—An Assessment*, Nuclear Energy Agency, Organization for Economic Co-operation and Development, Paris, 1989, p. 50.

[43] Typical fuel burnup is now around 42,500 MWD/T, and this is the burnup used in the other chapters of this report. However, ORIGEN2 does not provide cross section libraries for this burnup but only for 33,000 MWD/T and 50,000 MWD/T. We ran our once-through MOX vs. once-through LEU comparison using a 42,500 MWD/T burnup twice—once using ORIGEN2's 33,000 MWD/T cross section library and once using the 50,000 MWD/T cross section library. The difference between these two runs was about 5 percent. Rather than introduce this additional source of error, we have used a fuel burnup of 33,000 MWD/T in this chapter using the 33,000 MWD/T cross section library. Since our results in this chapter hinge on the ratio between our ORIGEN2 runs, rather than their absolute values, this should provide more accurate results. In any case, the difference is not large. If one takes the average of the two 42,500 MWD/T runs and compares it to our 33,000 MWD/T case, the difference is less than 5 percent.

equals 1.08 which means that slightly more than one LEU burning reactor is needed to support each MOX burning reactor. Therefore for each metric ton of MOX fuel burned and disposed of, there will be 2.33 MT of LEU fuel burned in the MOX reactor and reprocessed and 3.59 MT of LEU fuel burned in a LEU burning reactor and reprocessed. All of the reprocessed LEU fuel will produce HLW which will have to be disposed of. For our once-through LEU case we assume that 1 MT of LEU fuel (3.2 percent enriched, burned to 33,000 MWD/T) is disposed of. In order to normalize our once-through MOX case to the same energy output we divided the quantities discussed above by 6.93 (i.e. 1 + 2.33 +3.59). Therefore our once-through MOX case will consist of the direct disposal of 0.144 MT (1/6.93) of MOX fuel plus the disposal of the HLW resulting from the reprocessing of 0.336 MT (2.33/6.93) of LEU spent fuel burned in the MOX burning reactor plus the disposal of the HLW resulting from the reprocessing of 0.518 MT (3.59/6.93) of LEU spent fuel burned in a LEU burning reactor.[44]

Waste comparison between spent fuel and HLW is often made in terms of volume.[45] A metric ton of spent fuel occupies about 1.0 m^3 whereas the HLW from the reprocessing of 1 MT of LEU fuel occupies about 0.15 m^3. For our once-through MOX case the volume is a little higher because part of the waste is the directly disposed MOX but for the waste from an average MT of fuel (defined in the previous paragraph), the waste volume is around 0.2 m^3. Such a comparison clearly favors reprocessing but how relevant is it? In a geologic repository the heat generated by the directly disposed spent fuel and HLW will be what determines the actual space that the waste occupies.[46] This results from the requirement to limit the temperature of the waste packages and the surrounding rock so as to limit corrosion of the waste packages, maintain mechanical stability, control the moisture content of the rock and air, and limit mineralogical changes to the rock that might change its ability to impede the migration of radionuclides[47]. This need to space out the spent fuel and HLW means that their effective volume in the repository will be much greater than their actual volume.

We have made some ORIGEN2 runs in order to explore the waste heat implications of our cases. For the once-through MOX case three runs are needed, one for each of that case's components.[48] These results are shown in Figure 1. The heat is measured in watts and the results are shown starting 2 years after fuel reprocessing and

[44] This case of one MOX burning reactor (with 30 percent MOX and 70 percent LEU fuel) and 1.08 LEU burning reactor is equivalent to the case of a single reactor with 14 percent MOX and 86 percent LEU fuel. The latter case is used to calculate the waste volumes generated in various steps of the nuclear fuel cycle.

[45] M. Viala and M. Salivatores, "The Spin Program," *Global 1995*, op. cit., p. 126.

[46] *Nuclear Wastes—Technologies for Separations and Transmutation*, National Research Council, National Academy Press, Washington D.C., 1996, p. 315.

[47] *Mined Geologic Disposal System Advanced Conceptual Design Report*, Volume II of IV, B00000000-01717-5705-00027 REV 00, TRW Environmental Safety Systems Inc., Las Vegas, Nevada, March 1996, p. 8–36.

[48] The LEU from the MOX reactor and the LEU from the LEU reactor each require a separate ORIGEN2 run.

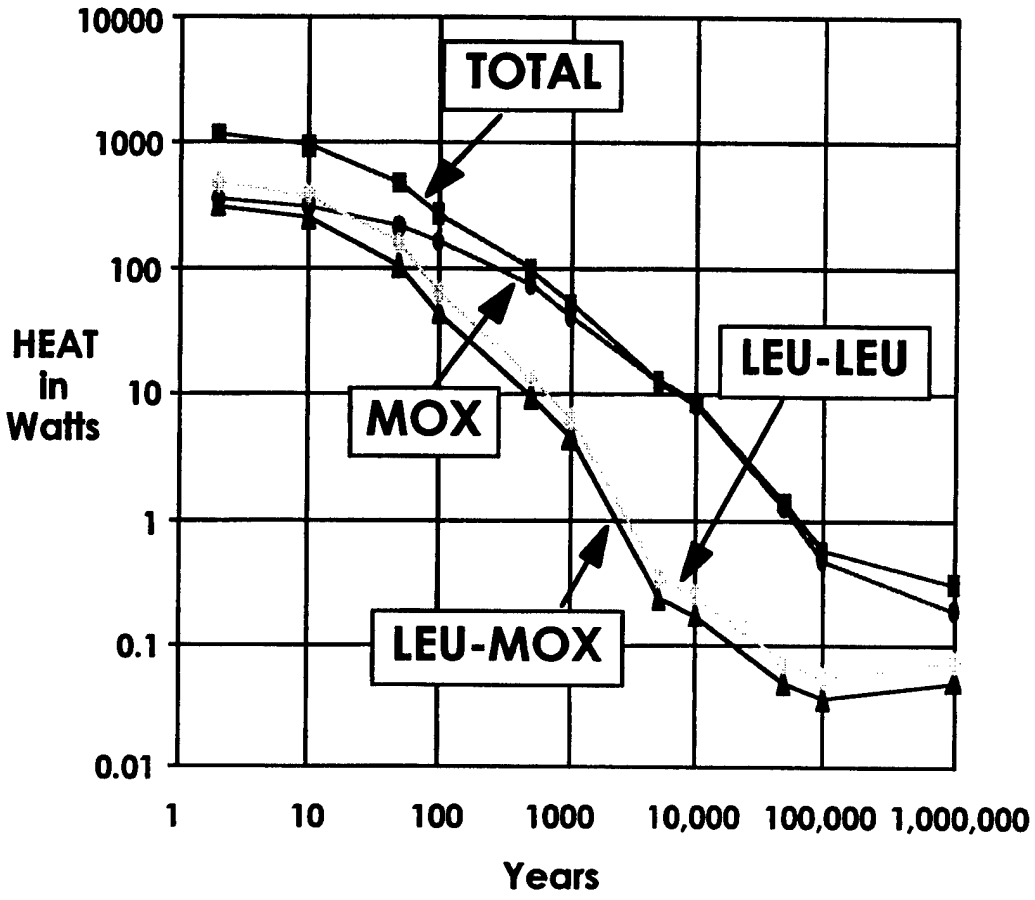

**Figure 1—Heat Output: Once-Through MOX Case—
Directly Disposed MOX + HLW From LEU Reprocessing**

continues for 1,000,000 years.[49] Figure 2 shows the heat output of the once-through LEU case with that of the once-through MOX case. The use of a log scale makes it difficult to see the magnitude of the difference between the two cases, especially during the first 1,000 years. Therefore, Figure 3 shows the ratio of the heat output of the once-through MOX case compared to the once-through LEU case. Wherever the ratio is greater than 1.00, the once-through MOX case has a greater heat output. From Figure 3 one can see that for the first 500 years the once-through MOX case has a slightly higher heat output and that after that time the once-through LEU case has the higher output.

There are no geologic repositories for spent fuel or HLW disposal operating anywhere in the world. This fact makes it difficult to say exactly over what time interval

[49] The fuel is reprocessed 10 years after discharge from the reactor. The directly disposed LEU and MOX spent fuel are aged by this same amount.

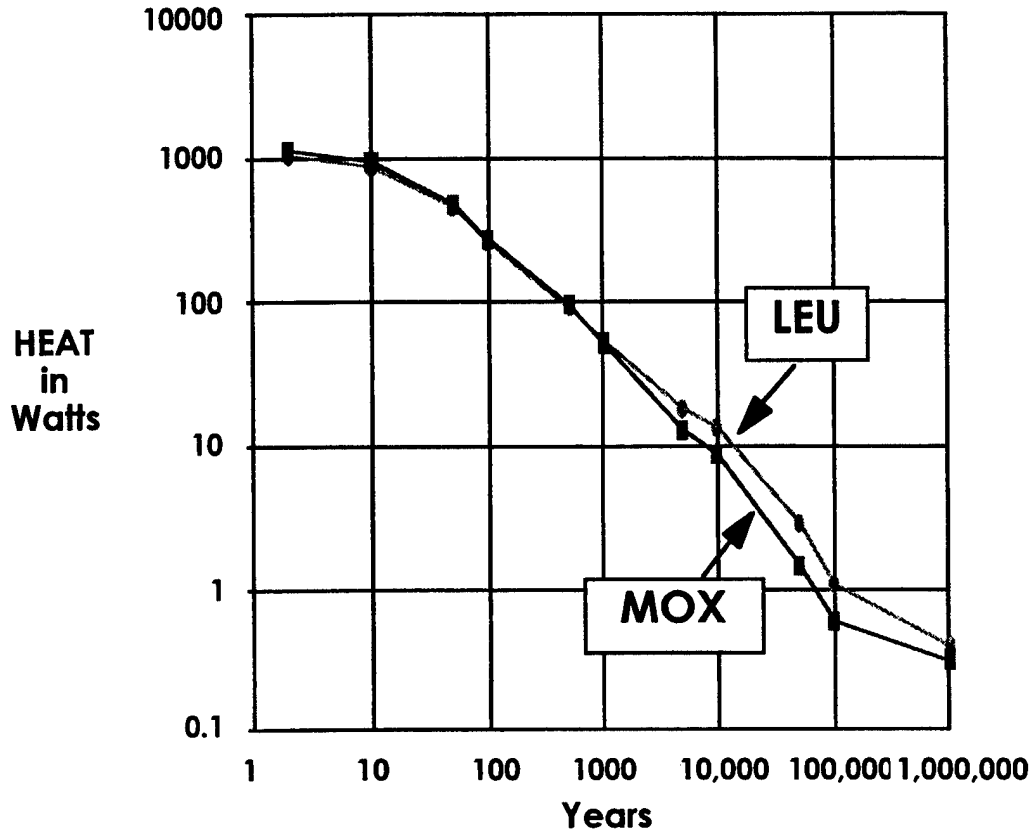

Figure 2—Heat Output: Once-Through MOX Case Compared with Once-Through LEU Case

the waste heat will be important. However, studies have generally indicated that some portion of the interval between 10 years and 1,000 years after reactor discharge is likely to be the relevant one for determining waste spacing based on heat output. Several strategies have been purposed for using the waste heat to determine the waste spacing in the repository but, as Figure 3 shows, during this time the heat output from these two cases is quite similar with the once-through MOX case being slightly higher. Therefore the choice of strategy will not strongly influence which case takes up more repository space. One measure being considered to account for the waste heat at the Yucca Mountain repository is to space the waste according to its integrated heat during the time interval of 10 years to 1,000 years after fuel discharge.[50] If this measure were to be used then the waste from the once-through MOX case would take up 4 percent more space

[50] This is termed "Equivalent Energy Density" (EED). See: *Repository Thermal Loading Management Analysis*, B00000000-01717-0200-00135 REV 00, TRW Environmental Safety Systems Inc.; Las Vegas, NV, June 1997, pp. 20–57.

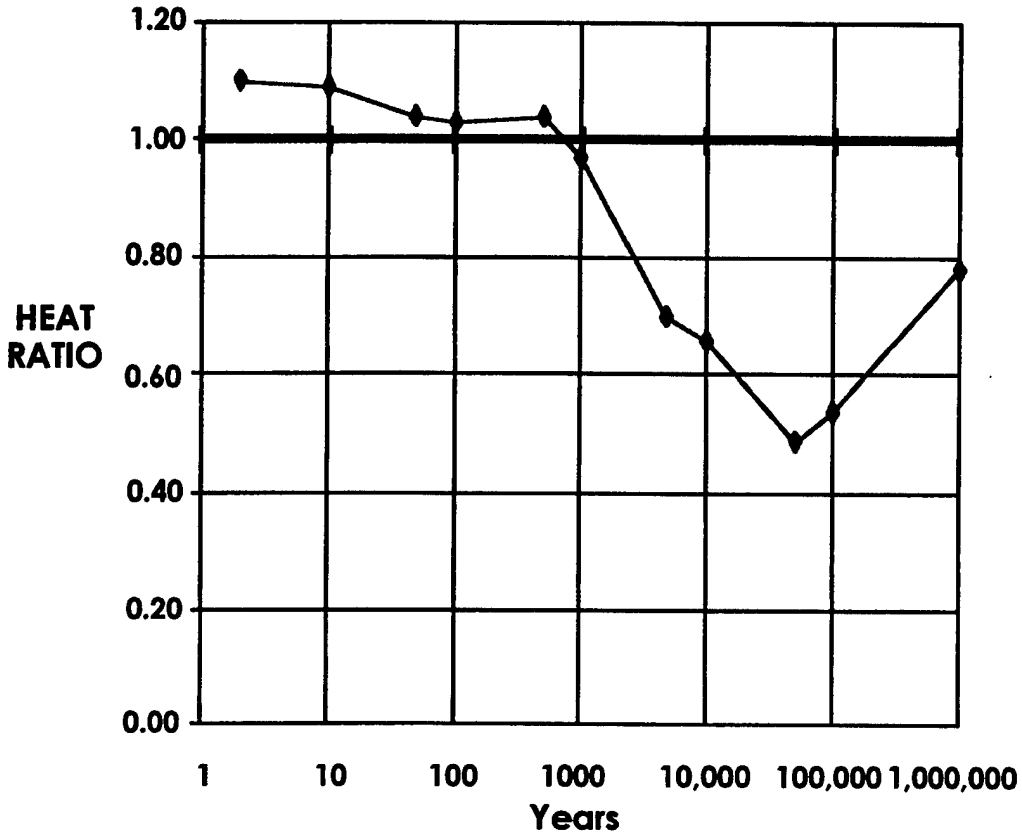

Figure 3—Heat Ratio: Once-Through MOX Case Divided by Once-Through LEU Case

than the once-through LEU case. Given the various uncertainties in this calculation, it would be safe to say that the heat output of the two cases is roughly equal and therefore the effective volume of the two wastes in a repository are also about equal.

In order to determine the reason for this rough equivalence in heat output between the two cases, we have extracted from the ORIGEN2 results the key isotopes that are generating the heat. Table 3 shows the results for 60 years after fuel discharge which is 50 years after fuel reprocessing. The first column lists, in order of importance, the heat generated by the seven isotopes that are producing almost all of the heat in the once-through LEU case. The third column lists the heat produced by these same isotopes for the once-through MOX case. The second column lists the differences between columns one and three. As can be seen from the table the same seven isotopes are the key ones for both cases. The increase in Am 241 and Cm 244 in the once-through MOX case is

Table 3

Once-Through MOX Case Compared with Once-Through LEU Case—
Heat Output (Watts) at 50 Years After Reprocessing

	Once-Thru LEU		Once-Thru MOX
Am 241	129	+23	152
Cs 137*	126	0	126
Sr 90*	118	-9	109
Pu 238	52	-6	46
Pu 240	15	-2	13
Pu 239	10	-6	4
Cm 244	8	+17	25
All Others	3	+1	4
	461	+18	479 (+4%)

* Includes daughter products

Table 4

Once-Through MOX Case Compared with Once-Through LEU Case—
Heat Output (Watts) at 500 Years After Reprocessing

	Once-Thru LEU		Once-Thru MOX
Am 241	67	+10	77
Pu 240	14	-1	13
Pu 239	10	-6	4
All Others	2	+1	3
	93	+4	97 (+4%)

enough to compensate for the decrease in plutonium and Sr 90, leading to the small difference in heat output between the two cases.[51]

Table 4 shows the same results but for 500 years after fuel reprocessing. Now there are only three isotopes of importance. The increase in Am 241 in the once-through MOX case again compensates for the decrease in plutonium.

These results show the importance of following the plutonium recovered from reprocessing until its disposal. The decrease in plutonium using a once-through MOX case is far less than the 99+ percent that is usually assumed in comparisons between HLW and spent fuel. And the use of a plutonium-based fuel leads to the buildup of Am and Cm isotopes that compensate for the reduction in plutonium achieved by MOX burning. As a result the heat output of the waste from the once-through MOX case is the same if not slightly higher than that from the once-through LEU case. This means that in a geologic repository that the HLW and spent fuel waste from the once-through MOX case are likely to require essentially the same volume for disposal as the spent fuel from the once-through LEU case.

Comparison B: SGR case vs. Once-Through LEU case

Although we think the once-through MOX case is more likely, we have also done a comparison between a self-generating recycle (SGR) case and a once-through LEU case. As we stated in chapter II, we think that it is important to look at the SGR case since it was, until recently, considered to be the standard case for recycling plutonium in LWRs and this case also provides insight into the waste implications of multiple plutonium recycles in LWRs.

In this chapter our SGR case will look at an equilibrium SGR cycle where 32.4 percent of the core is MOX and the other 67.6 percent is LEU fuel.[52] The MOX is 5.0 percent plutonium (61.3 percent fissile) and is made using natural uranium. The LEU is 3.2 percent enriched and both fuels are burned to 33,000 MWD/T. An average metric ton of fuel from this cycle is 0.324 MT of MOX and 0.674 MT of LEU. In this cycle both the MOX and LEU fuel is reprocessed and the resulting HLW is disposed of. This case is compared to our once-through LEU case, which we used in the previous section. One MT of 3.2 percent enriched fuel is burned to 33,000 MWD/T and the spent fuel is directly disposed of.

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[51] The Sr 90 is reduced because Pu 239's fission yield of that isotope is less than half that of U 235.

[52] Our equilibrium SGR case is based on: Blomeke, J.O., C.W. Kee, and J.P. Nichols, *Projections of Radioactive Wastes to be Generated by the U.S. Nuclear Power Industry*, Oak Ridge National Laboratory, ORNL-TM-3965, February 1974, pp. 9, 24. In the estimates of waste volumes in this study, we assume a similar SGR case but using depleted instead of natural uranium in MOX fuel. The MOX would then be 24 percent of core using a burnup of 42,500 MWD/T.

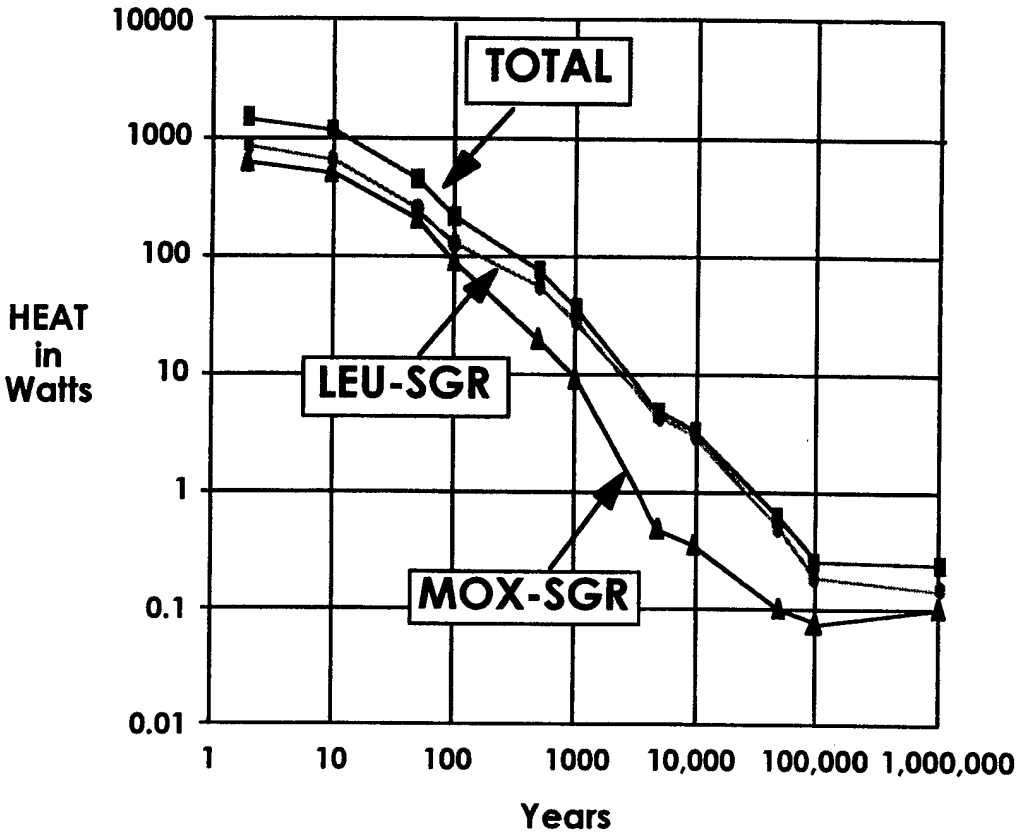

Figure 4—Heat Output: SGR Case—HLW From MOX Reprocessing + HLW From LEU Reprocessing

As in the previous section we have made ORIGEN2 runs in order to examine the waste heat resulting from the HLW and spent fuel from these two cases. Figure 4 shows the results of the two runs needed for the SGR case and their total. Note that even though the HLW from the MOX is only about one-half the quantity of the HLW from the LEU, the heat produced by the MOX HLW is always greater than that of the LEU HLW and starting about 500 years after fuel reprocessing it is the source of almost all of the waste heat. This again shows the importance of looking at a complete plutonium cycle.

Figure 5 compares the heat output of the SGR case with that of the once-through LEU case. The SGR case clearly has the higher heat output during the first fifty years and the once-through LEU case is clearly higher after the first fifty years. In order to make the differences between the two cases more apparent, Figure 6 shows the ratio of the heat output of the SGR case compared with that of the once-through LEU case. Wherever the ratio is greater than 1.00 the SGR case has a greater heat output. The heat ratio of these two cases is rather different than that of the once-through MOX case and the once-through LEU case (Figure 3). Two years after fuel reprocessing, the SGR case is 42 percent hotter. By fifty years it is only 1 percent hotter and at 100 years it is 16 percent cooler.

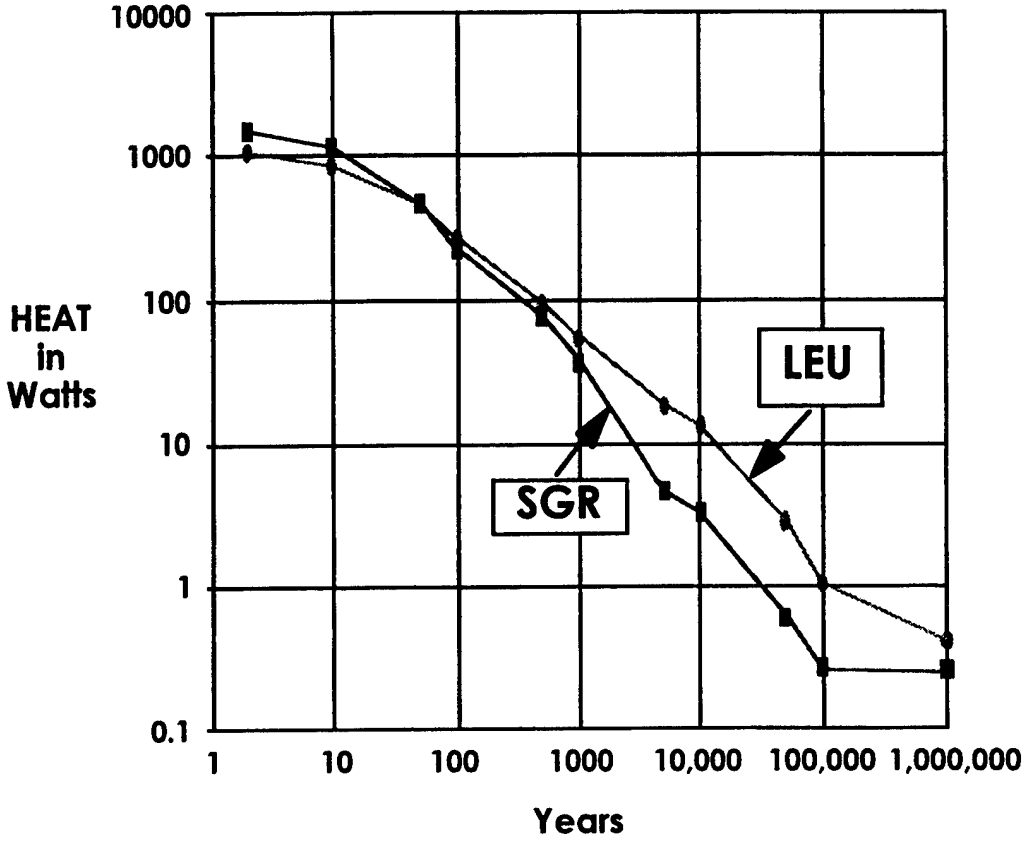

Figure 5—Heat Output—SGR Case Compared with Once-Through LEU Case

As stated above, there are several strategies that could be used at a geologic repository for determining waste spacing based on waste heat.[53] If the waste is emplaced based on the integrated heat over the interval 10 years after reactor discharge to 1,000 years after reactor discharge then the HLW from the SGR case will take up 7 percent less space in the repository than will the spent fuel from the once-through LEU case. If the waste is emplaced based on its heat output at the time of emplacement (Areal Power Density) then the HLW from the SGR case could take up anywhere from 42 percent more space to 16 percent less space depending on when waste disposal took place. As with our comparison of the once-through MOX case with the once-through LEU case, we feel that the uncertainties are large enough that essentially the SGR waste and the once-through LEU waste will take up approximately the same space in the repository.[54]

[53] *Repository Thermal Loading Management Analysis*, op. cit., p. 20.

[54] Unlike the once-through MOX vs. once-through LEU comparison, the change in the ratio of SGR heat to once-through LEU heat means that for a specific strategy it is possible for either the SGR case or the once-through LEU case to be significantly hotter.

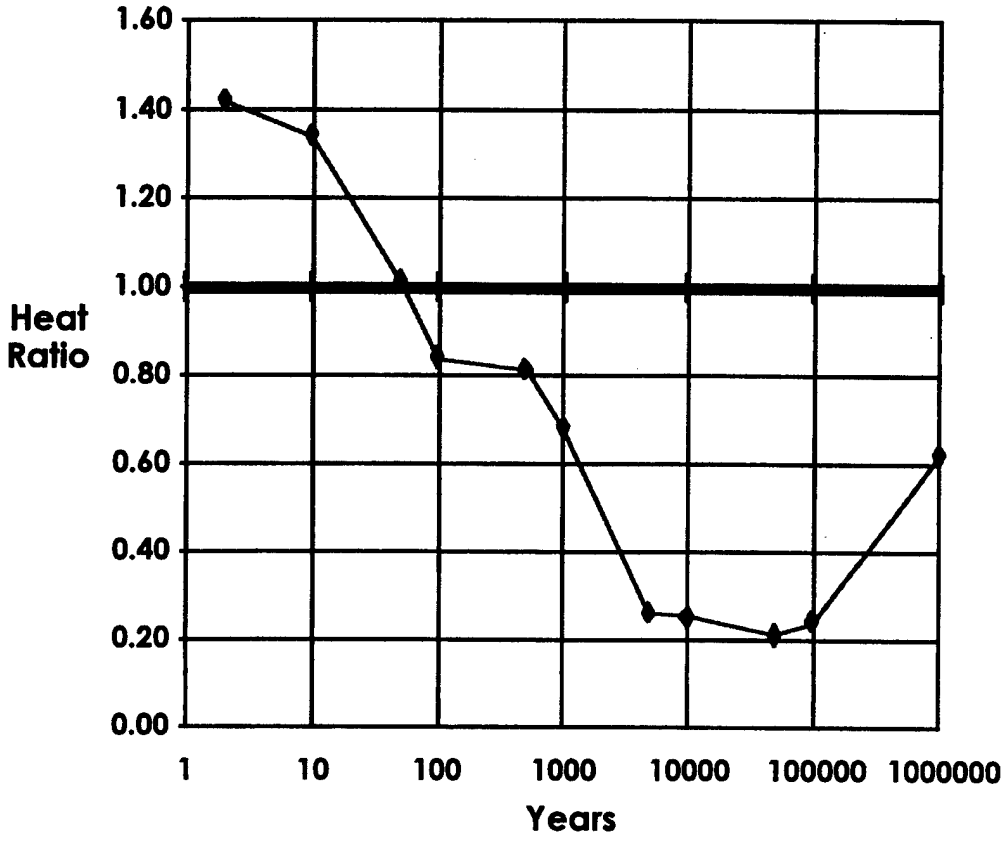

Figure 6—Heat Ratio: SGR Case Divided by Once-Through LEU Case

As previously, in order to determine why the ratio in Figure 6 changes the way it does, we have extracted from the ORIGEN2 results the key isotopes that are generating the heat. Table 5 shows the results for 10 years after fuel reprocessing. The first column lists, in order of importance, the heat generated by the eight isotopes that are producing almost all of the heat in the once-through LEU case. The third column lists the heat produced by these same isotopes for the SGR case. The second column lists the differences between columns one and three. One can see that the SGR case leads to a very large increase (almost a factor of 12) in the amount of Cm 244 generated. This increase, along with a smaller increase in Am 241 is enough to more than compensate for the near elimination of plutonium and the reduction in Sr 90. Much more Cm 244 is produced in the SGR case than in the once-through MOX case because the equilibrium recycle plutonium used in the SGR case has a significantly higher percentage of Pu 242 than the first generation recycle plutonium used in our once-through MOX case.

Table 6 shows the same results but for 50 years after fuel reprocessing. Now the Cm 244 (18 year half life) has decayed significantly and the difference in Am 241 has decreased due to the decay of Pu 241 in the spent fuel in the once-through LEU case. As a result the heat output from the two cases is almost identical. Table 7 shows the

Table 5

Self-Generating Recycle Case Compared with Once-Through LEU Case—
Heat Output (Watts) at 10 Years After Reprocessing

	Once-Thru LEU		SGR
Cs 137*	317	0	317
Sr 90*	307	-52	255
Am 241	90	+59	149
Pu 238	72	-71	1
Cm 244	35	+374	409
Eu 154	16	+4	20
Pu 240	15	-14	1
Pu 239	10	-10	0
All Others	10	+6	16
	872	+296	1,168 (+34%)

* Includes daughter products

Table 6

Self-Generating Recycle Case Compared with Once-Through LEU Case—
Heat Output (Watts) at 50 Years After Reprocessing

	Once-Thru LEU		SGR
Am 241	129	+12	141
Cs 137*	126	0	126
Sr 90*	118	-19	99
Pu 238	52	-51	1
Pu 240	15	-14	1
Pu 239	10	-10	0
Cm 244	8	+80	88
All Others	3	+5	8
	461	+3	464 (+1%)

* Includes daughter products

Table 7

Self-Generating Recycle Case Compared with Once-Through LEU Case—Heat Output (Watts) at 500 Years After Reprocessing

	Once-Thru LEU		SGR
Am 241	67	+1	68
Pu 240	14	-13	1
Pu 239	10	-10	0
All Others	2	+4	6
	93	-18	75 (-19%)

results at 500 years after fuel reprocessing. Now the Cm 244 has decayed away and the amounts of Am 241 are almost the same. As a result the waste from the SGR case is 19 percent cooler than that from the once-through LEU case. The reason that it is not lower is that the Am 241 is still providing the preponderance of the heat.

As with the once-through MOX case, these results show the importance of following the plutonium through its successive recycles. Even though the plutonium is theoretically never disposed of, in the SGR case it still has an important impact on the HLW by the generation of higher actinides, especially Cm 244. The fact that the relatively short-lived Cm 244 is a major source of heat in the SGR case means that interim storage of the HLW would have a greater benefit than the interim storage of the LEU spent fuel. Whether such storage would be cost effective would, of course, depend on the cost of the storage and the saving in repository costs.

VI. BACK-END WASTES

This chapter provides waste volumes at various radioactive levels (Spent fuel, HLW, ILW and LLW) from the fuel cycle's back-end. It should be emphasized that the key cost driver for the first two, spent fuel and HLW, is not waste volume but heat. We, however, provide these volumes as well for the purpose of completeness.

For the LEU once-through, spent fuel is assumed to be discharged into an on-site water cooling pool. At some point in the future, it will be encapsulated and placed in a permanent repository. This study does not assume that the spent fuel be first transferred to an interim site for storage before final disposal.[55]

In the MOX-OT, spent fuel from an earlier cycle is sent to a reprocessing plant for fissile material extraction. The reprocessed plutonium and uranium are used in the fabrication of MOX and LEU fuels, which are charged into the reactor. After irradiation in the reactor, the MOX spent fuel is discharged into an on-site storage pool and eventually encapsulated and disposed of in a repository. The LEU spent fuel is reprocessed, and the resulting HLW is vitrificated and disposed of in the same repository.

In the SGR, spent fuel is first placed in a cooling pool. It is then transferred to a reprocessing plant for plutonium and uranium extraction. The plutonium and uranium are then transferred to fuel fabrication plant, while the HLW is stored awaiting vitrification and final disposal.

Waste from Spent Fuel Storage, Encapsulation and Disposal

A multiple-purpose canister containing 21 assemblies of spent fuel has an encapsulated volume of 9.02 m^3 per 9.68 MTHM. This translates into 26 m^3 of spent fuel/GWe-yr.[56] COGEMA estimated that the LEU spent fuel volume to be 1.5 m^3/MTU.[57] At 42.5 GWD/MTU, the volume is translated into 37 m^3/GWe-yr. BNFL also gives a similar figure of 40 m^3/GWe-yr. We, however, consider both the COGEMA and BNFL numbers are too high, especially when older U.S. number indicated only 13.7 m^3/GWe-yr, even lower than the modern value of 26 m^3/GWe-yr.[58] Therefore, we use 26 m^3/GWe-yr as the reference value for spent fuel in this study.

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[55] If, in addition to storing spent fuel in a water pool, dry casks are used for interim storage, there will be waste generated during interim storage. The waste from dry cask storage is estimated to be 6 m^3 of LLW/GWe-yr. Don Habib, Project Performance Corporation, October 22, 1997.

[56] Don Habib, October 22, 1997. The maximum MPC loading of 21 PWR assemblies is based on a 40 GWD/MTHM. Habib assumed that the MPC is proportionally somewhat larger for a slightly higher burnup of 42.5 GWD/MTHM.

[57] Appears on p. 34, *The Economics of the Nuclear Fuel Cycle*, Nuclear Energy Agency, OECD, 1994.

[58] U.S. Energy Research & Development Administration (predecessor to the Department of Energy), *Alternative for Managing Wastes From Reactors and Post-Fission Operations in the LWR Fuel Cycle*, May 1976., Volume 1, p. 1.3. A value of 13.7 m^3/GWe-yr was given as the volume of the spent fuel assembly casks. Moreover, this volume was for considerably lower fuel burnup (probably near 30

When spent fuel is in storage or in the process of packaging in casks or canisters, ILW and LLW can be generated. Unfortunately, there is little modern data on these wastes. An old DOE report gave 1.6 m^3/GWe-yr as the volume of ILW generated during storage in a water pool.[59] DOE did not give the corresponding volume of LLW, apparently because only ILW but little LLW is generated during storage. There is a recent projection of the volume of waste generated during packaging in a MPC. The value is 0.2 m^3 of LLW/GWe-yr.[60] COGEMA presented a figure of 0.35 m^3 of ILW/MTU and 1.4 m^3 of LLW/MTU.[61] They translate into 9 m^3 of ILW/GWe-yr and 36 m^3 of LLW/GWe-yr . BNFL gives a smaller, but still similar, figure of 6 m^3 of ILW/GWe-yr, but provides no estimate of LLW. We have already discussed that BNFL probably has overestimated the waste volumes generated during reactor operations for the once-through fuel cycle as well as for recycle.[62] It is likely that COGEMA has also overestimated the waste volumes for spent fuel storage, encapsulation and disposal. This study uses 2 m^3 of ILW/GWe-yr and 0.2 m^3 of LLW/GWe-yr for the LEU once-through.

As shown in Chapter V, MOX spent fuel has a heat rate at the time of encapsulation that can be two to three times that of LEU spent fuel. But, in order not to bias against MOX-OT, we assume that the handling of the MOX spent fuel generates the same amounts of ILW and LLW wastes as that of LEU spent fuel.

These waste volumes will apply to the LEU-OT and MOX-OT cases, but not to the SGR case where spent fuel is reprocessed instead. The waste volumes of SGR's back-end are shown in the next subsection.

Waste from Reprocessing, Storage, Conditioning and Disposal

Table 8 summarizes various estimates of radioactive wastes generated from reprocessing, storage, conditioning (including vitrification in the case of HLW) and final disposal. The HLW volumes clustered within a range of 2-4 m^3/GWe-yr regardless reprocessors. Similarly, the ILW volumes are in the range of 20-45 m^3/GWe-yr, while

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MWD/MTU). At the modern burnup of 42.5 GWD/MTU, the waste volume should be even considerably lower than 13.7 m^3/GWe-yr.

[59] U.S. Energy Research & Development Administration (predecessor to the Department of Energy), *Alternative for Managing Wastes From Reactors and Post-Fission Operations in the LWR Fuel Cycle*, May 1976., Volume 1, p. 2.40. The report gave 1.6 m^3/GWe-yr with 4 ci/m^3 of radioactivity resulting from activation and fission products. The radioactivity level for LLW is 100 nanocurie/gm or about 0.2 ci/m^3. Since 4 ci/m^3 is higher than the LLW radioactive level, we classify the waste as ILW.

[60] Based on a projection of roughly 1,000 ft^3 of LLW produced from loading and unloading 3,000 MT of spent fuel, Don Habib arrives at 0.0094 m^3/MTHM or 0.24 m^3/GWe-yr. The projection appears in *Multi-Purpose Canister System Evaluation: A Systems Engineering Approach*, Office of Civilian Radioactive Waste Management, U.S. Department of Energy, September 1994, p. 4-5.

[61] *The Economics of the Nuclear Fuel Cycle*, Nuclear Energy Agency, OECD, 1994. The report gives 0.2 m^3/MTU (metric ton of uranium) as the volume of ILW and 0.15 m^3/MTU as that of solid waste, which we also treat as ILW. The volume of LLW given is 1.4 m^3/MTU.

[62] See the discussion in the subsection on Reactor Operation Waste in Chapter IV.

those of LLW in 70-95 m^3/GWe-yr.[63] These numbers will be applied to the SGR case. For the MOX-OT case, since only 86% of the fuel will be reprocessed, the numbers will be adjusted downward by multiplying a factor of 0.86. These numbers will be zeroes for the LEU-OT case, where no reprocessing takes place.

Table 8: Wastes From Reprocessing, Storage, Conditioning and Disposal

		----------Waste Volume in m^3/GWe-yr----------		
Source of Data	Reference	HLW	ILW	LLW
BNFL	1	2.2	29.6	78
COGEMA*	1&2	4	20	70
WAK**	1	2.5		
PNNL***	3	2	45	13
BNFL****	4	3	32	95
Used in this study		2-4	20-45	70-95

* For HLW and ILW, we use the data (for the end of 1995) in COGEMA's own publication (reference 2). But since it does not provide data for LLW, we use reference 1 for LLW. The HLW consists of glassified waste, while ILW consists of wastes listed in reference 2 as Concrete: hulls and end-fittings and Grout Concrete: technological waste.

** Wiederaufarbeitungs Analge Karlsruhe, Germany

*** Pacific Northwest National Laboratory

**** The reference gives 2.9, 31.1 and 92.4 m^3/GWe-yr for HLW, ILW and LLW respectively. These values are, however, corresponding to the reprocessing of 25.3 MT of spent fuel. We adjust them proportionally for our case of 26 MT.

Reference 1: Mike Smith and Nigel Mote, "Identification of Reprocessing Waste Quantities and Characteristics," NAC International, December 1997.

Reference 2: COGEMA, Nuclear Materials Reprocessing and Recycling, June 1995.

Reference 3: Robert McKee, Michael White and Kenneth Ames, "Recycle Waste Stream Analysis," Pacific Northwest National Laboratory, June 1997.

Reference 4: D.M. Beaumont et al, "The Environmental Benefits of MOX Recycle," British Nuclear Fuel Limited, in Proceedings of the Fifth International Conference on radioactive Waste Management and Environmental Remediation, Volume 1, September 3-7, 1995, pp. 519-523.

[63] PNNL's number of 13 m^3 of LLW/GWe-yr is considerably lower than the other estimates, perhaps due to the use of some Japanese data from Mitsubishi Materials Corporation, as opposed to European data.

VII. WASTES FROM FACILITY DECOMMISSIONING

In addition to routine operations, wastes are generated during facility decommissioning. We allocate the waste volumes to any specific fuel cycle as follows. First, determine the volumes of waste generated during the decommissioning process. Second, determine the total material throughput during the lifetime of the plant. Third, estimate the amount of electricity that has been generated using this amount of material. Finally, divide the waste volumes by the amount of electricity generated to obtain m^3/GWe-yr.

Conversion Plant Decommissioning

The ongoing decommissioning of the UF_6 conversion plant in Gore, Oklahoma is projected to generate 70,000 m^3 of additional LLW. Assuming 20 years of operation at 90% of the plant's rated capacity, one obtains a waste volume of 92 m^3/GWe-yr for the LEU-OT case.[64] In the MOX-OT and SGR cases, the volume is reduced proportionally depending on the percentage of natural uranium savings, 23% and 32%.

Enrichment Plant Decommissioning

Louisiana Energy Services proposed an enrichment center, which will contain 40,000 centrifuges and can produce annually 318 MTU of 3.7% LEU with a tails assay of 0.3%.[65] Over 30 years of plant operation, 9540 MTU will be produce. LES estimated that the decommissioning waste is merely 100 m^3 in total or levelized to 0.01 m^3/MTU or 0.27 m^3/GWe-yr. This figure is consistent with the actual decommissioning projects in Capenhurst, UK and Almelo in the Netherlands. The Capenhurst plant, however, allowed 99+ percent of 160,000 m^3 of the materials decontaminated and reclassified as nonradioactive material. If merely 2,000 m^3 of the material remains as waste in the LES case, the waste volume would be about 5 m^3 of LLW/GWe-yr in the case of LEU-OT. This study uses the larger number, 5 m^3 of LLW/GWe-yr, allowing the possibility that considerably more waste than 0.27 m^3/GWe-yr is generated. The waste volumes for MOX-OT and SGR are 13% and 23% less respectively.

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[64] Don Habib, October 22, 1997.

[65] Don Habib, October 22, 1997, and "Uranium Enrichment Wastes and Costs," June 10, 1997.

UO_2 Fabrication Plant Decommissioning

Table 9 shows the data for the decommissioning of four UO_2 fabrication plants. The average waste volume is 6 m^3 of LLW/GWe-yr for the case of LEU(OT).[66] The volumes are 14% and 24% less for the MOX-OT and SGR cases respectively.

Table 9: Decommissioning Wastes From UO_2 Fuel Fabrication Plant

Plant Location	Decommissioning LLW Volume (m^3)	Estimated Lifetime Production (MTU)	Waste Volume (m^3/GWe-yr)
Columbia, SC	12,300	31,000	10
Wilmington, NC	7,020	32,000	5.7
Richland, WA	4,240	19,000	5.8
Hematite, MO	1,610	12,000	3.5
Used in this study			6

Source: Don Habib, October 22, 1997

MOX Fabrication Plant Decommissioning

The only data available to us is from an old report by Pacific Northwest National Laboratory.[67] It estimated the decommissioning would generate 3 m^3 of ILW/GWe-yr and 1.6 m^3 of LLW/GWe-yr. But for every GWe-yr, only 14% originated from MOX fuel in the MOX-OT case. Thus, the waste volumes for the MOX-OT case are 0.4 m^3 of ILW/GWe-yr and 0.2 m^3 of LLW/GWe-yr. Similarly, the waste volumes for the SGR using 24% MOX fuel are 0.7 m^3 of ILW/GWe-yr and 0.4 m^3 of LLW/GWe-yr.

Reactor Decommissioning

The same PNNL report gives a waste volume of 175 m^3 of LLW/GWe-yr for a PWR.[68] Recently, DOE provides a value of 6,992 m^3 for a 1,175 MWe PWR.[69] This translates into 230 m^3 of LLW/GWe-yr.[70] This much more recent data is used in this study. The data in the same DOE report leads to a waste volume of 9 m^3 of ILW/GWe-

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[66] We also have data for a fifth plant at Apollo, PA. The waste volume is 64 m^3/GWe-yr. This is not included in computing the average, because Apollo plant predated most of other plants and was used for operations other than UO_2 fabrication.

[67] *Alternatives for Managing Wastes From Reactors and Post-Fission Operations in the LWR Fuel Cycle*, Volume 1, May 1976, p. 1.9. Report prepared at the request of the Division of Nuclear Fuel Cycle and Production, U.S. Energy Research & Development Administration (predecessor of DOE), ERDA-76-43.

[68] *Alternatives for Managing Wastes From Reactors and Post-Fission Operations in the LWR Fuel Cycle*, Volume 1, May 1976, pp. 1.3 and 1.5.

[69] NUREG/CR-5884. [give full reference].

[70] Don Habib, October 1997.

yr. These waste volumes are the same for all three cases--LEU(OT), MOX(OT) and SGR.

Reprocessing Plant Decommissioning

We can only found some old data from the above PNNL report, which gives 0.8 m^3 of ILW/GWe-yr and 5 m^3 of LLW/GWe-yr.[71] These waste volumes are applicable to the SGR case, but are reduced by 14% in the MOX-OT case.

[71] *Alternatives for Managing Wastes From Reactors and Post-Fission Operations in the LWR Fuel Cycle*, Volume 1, May 1976, pp. 1.6 and 1.8.

VIII. A LIFE-CYCLE LOOK AT WASTE GENERATION AND DISPOSAL

In this chapter, we will first compare the waste volumes and heat determined in previous chapters for alternative nuclear fuel cycles. Comparison A focuses on the comparison of LEU once-through versus MOX once-through, and comparison B, on LEU once-through versus self-generating recycle. Then, we will go through comparisons A and B again, but focusing on waste disposition costs instead of waste volumes and heat.

Waste Heat and Volumes

Generally, there are large uncertainties in the waste volumes. For example, the decommissioning volumes hinge on the stringency of the prevailing environmental regulations. The more strict are the regulations, the less soil and material can be considered as successfully decontaminated, and the more waste results. Stringent regulations will also result in larger waste volumes during routine operations. Also, if one is willing to spend more money, one can further reduce or compact the waste volumes. Finally, as states levy large fees for the disposal of LLW, processors would likely reduce the waste volumes in order to lower the disposal fees. In spite of all these uncertainties, our comparisons should still be meaningful, because it is quite probable that factors would affect all waste volumes in similar ways. When environmental regulations are tight, all wastes volumes are likely to take the higher values, regardless of fuel cycles and of fuel cycle steps. This allows us to compare alternative fuel cycles assuming all waste volumes are taking the high estimates of waste volumes. Similarly, we can make comparisons based on low estimates of all waste volumes.

Comparison A: Once-Thru LEU Versus Once-Thru MOX

Table 10 summarizes numbers from previous chapters that are most relevant to the determination of waste disposition cost. For spent fuel (SF) and high level waste (HLW), the relative heat is used as a proxy, which is the amount of heat relative or normalized to that of spent fuel in the LEU-OT case. In the case of LEU-OT, the amount of heat in its LEU spent fuel is, by definition, 1. In the case of MOX-OT, it consists of 14% MOX spent fuel and 86% of HLW. Chapter V shows that the relative heat for the combined SF/HLW varies from 1.04 to 1.1. The value depends on whether the comparison of instantaneous heat in the LEU-OT and MOX-OT cases occurs at 60 or 10 years after fuel discharged from reactors.[72] At 60 years, the instantaneous heat output in the MOX-OT case is 4% higher than that of LEU-OT, while at 10 years, the former is 10% higher. The instantaneous heat is a relevant measure, if spacings among spent fuel or HLW canisters in

[72] Or 50 years or immediately after reprocessing, because this study assumes that, if fuel is reprocessed, it will be reprocessed 10 years after fuel discharge.

the repository are determined by the average power density (APD).[73] In other words, canisters whether containing SF or HLW are placed in the repository so that they generate the same amount of power (heat per unit time) per unit area at the time of disposal. Thus, the number of canisters per unit area or the spacings among canisters are determined by the instantaneous heat. On this basis, if spent fuel and HLW are disposed of 60 years after reactor discharge, MOX-OT case would require 4% more area in the repository. Since a repository has a capacity constraint and, in the long run, multiple repositories are likely to be required to deal with permanent waste storage, the MOX-OT fuel cycle that takes 4% more area for waste disposal would result in 4% higher disposal cost. It should, however, be emphasized that this disposal cost is only a component of the total waste disposal costs, which also includes the costs of disposing mill tailings, LLW and ILW.

Table 10: Waste Volumes and Heat: LEU-OT versus MOX-OT (First column for each case in heat relative to LEU-OT spent fuel and the rest in m^3/GWe-yr)

	LEU-OT				MOX-OT			
	SF	ILW	LLW	Tailings	SF/HLW	ILW	LLW	Tailings
Mining & milling		0	0	65,000		0	0	50,060
Conversion to UF_6		0	32-112	0		0	25-86	
Enrichment		0	3-40	0		0	3-35	
Fabrication, UO_2		0	3-9	0		0	2.6-7.7	
Fabrication, MOX		0	0	0		13	4.8	
Reactor operations		22-33	86-130	0		22-33	86-130	
Spent fuel stor. & encap.		2	0.2	0		0.3	0.03	
Reprocess. & vitrification		0	0	0		17-39	60-81	
Decom, UF_6 conver. plant		0	92	0		0	71	
Decom, enrich. plant		0	5	0		0	4.4	
Decom, UO_2 fab. plant		0	6	0		0	5.1	
Decom, MOX fab. plant		0	0	0		0.4	0.2	
Decom, reactor		9	230	0		9	230	
Decom, reprocess. plant		0	0	0		0.7	4.3	
Total	1	33-44	457-624	65,000	1.04-1.1	62-95	495-660	50,060
Volume change					.04-.1	29-51	38-36	(14,940)**
Volume change in %					4%-10%	88%-115%	8%-6%*	(23%)

* The 8% corresponds to the numbers above it on the left, while the 6%, to numbers on the right.
** Numbers in parentheses are negative.

Other methods have also been proposed in determining the canister spacings in a repository. Another method is called the equivalent energy density (EED), which measures the heat generated over a period of time per unit area. The use of 1,000 years as the period of time (starting at 10 years after fuel discharge from the reactor) has been mentioned and is used in this study. The relative heat determined by this method for the MOX-OT case is 4% higher than that of the LEU(OT) case.

[73] See, for example, Thomas Buscheck and John Nitao, Lawrence Livermore National Laboratory, "The Impact of Thermal Loading on Repository Performance at Yucca Mountain," High Level Radioactive Waste Management, Proceedings of the Third International Conference, American Nuclear Society and American Society of Civil Engineers, Volume 1, 1992, p. 1003.

To have an overall picture of waste generation, one still needs to look at wastes other than SF and HLW. We have pointed out that, for less radioactive wastes, waste volumes are more relevant in determining the waste disposition cost than waste heat. Table 7 lists the waste volumes measuring in m^3/GWe-yr for both LEU(OT) and MOX(OT) cases. The waste volumes for ILW in the MOX-OT case are 88% to 115% higher than those in the LEU(OT) case. The disposition cost of ILW for MOX-OT is also likely to be that much higher. The waste volumes for the LLW in MOX-OT are 6% to 8% higher, so are the likely LLW disposition cost. On the other hand, MOX-OT generates 23% less mill tailings than LEU(OT) does.

One cannot, however, simply add up all the waste volumes, because they have different disposal cost. We will address the cost issue in the next section, after a similar discussion on waste heat and volumes for Comparison B.

Comparison B: Once-Thru LEU-OT Versus Self Generating Recycle

The SGR generates considerably more instantaneous heat, if its HLW is placed in the repository 10 years after reactor discharge and if the method of APD is used. The heat under that disposal strategy is 44% more than that from LEU(OT). This is shown in Table 11. On the other hand, if the HLW in the SGR case and the spent fuel in the LEU(OT) case are both placed in the repository 60 years after discharge, the heat from HLW is only 1% higher. If EED (integrated over 1,000 years starting 10 years after fuel discharged from the reactor) is used, the SGR generates 7% less heat than that from LEU spent fuel. Regardless of HLW and SF, this brings out an interesting issue concerning how old the waste should be before it is placed in the repository. The cooler is the waste, the more waste can be placed in a given unit area in the repository. The tradeoff is saving repository cost versus saving pre-disposal storage cost in on-site water pool and/or off-site facility. Perhaps, some countries have focused too much attention on attempting to use reprocessing to lower the waste disposition cost. A more fruitful approach may be to examine delaying placement in repository and prolonging temporary storage could lower the overall waste disposition cost, as older and cooler spent fuel and HLW would allow their canisters be much more closely packed and their disposal cost reduced. Since repository is very expensive, it is quite possible that the lower of the disposal cost would well compensate for the much cheaper temporary storage cost. This important issue should be examined.

As in Comparison A, SGR generates 123% to 149% more ILW, and 5 to 9% more LLW than the LEU(OT). On the other hand, SGR generates 32% less mill tailings. This assumes that both plutonium and uranium are reprocessed and reused, and that depleted uranium is also used. But even if reprocessed plutonium is reused, reprocessed uranium might not be, because it has higher radioactivity than does natural uranium which makes it more difficult to handle and there is the U^{236} neutronic penalty. Then without the use of reprocessed uranium, the reduction in mill tailings in the SGR case would be less or about 24%. In the next section, we will discuss the implications of waste heat and volumes on waste disposition cost.

Table 11: Comparison of Waste Volumes: LEU-OT versus SGR (First column for each case in heat relative to LEU-OT spent fuel and the rest in m^3/GWe-yr)

	LEU-OT				SGR			
	SF	HLW	LLW	Tailings	HLW	HLW	LLW	Tailings
Mining & milling		0	0	65,000		0	0	44,390
Conversion to UF_6		0	32-112	0		0	22-76	
Enrichment		0	3-40	0		0	2.3-31	
Fabrication, UO_2		0	3-9	0		0	2.3-6.8	
Fabrication, MOX		0	0	0		21	8	
Reactor operations		22-33	86-130	0		22-33	86-130	
Spent fuel stor. & encap.		2	0.2	0		0	0	
Reprocess. & vitrification		0	0	0		20-45	70-95	
Decom, UF_6 conver. plant		0	92	0		0	63	
Decom, enrich. plant		0	5	0		0	3.9	
Decom, UO_2 fab. plant		0	6	0		0	4.6	
Decom, MOX fab. plant		0	0	0		0.7	0.4	
Decom, reactor		9	230	0		9	230	
Decom, reprocess. plant		0	0	0		0.8	5	
Total	1	33-44	457-624	65,000	0.93-1.44	74-110	497-654	44,390
Volume change					(.07)-.44	41-66	40-30	(20,610)**
Volume change in %					(7%)-44%	123%-149%	9%-5%*	(32%)

* The 9% corresponds to the numbers above it on the left, while the 5%, to numbers on the right.
** Numbers in parentheses are negative.

Comparison of Total Waste Disposition Cost

One way to quantify the ease or difficulty of waste management is to measure the disposition cost of all the wastes in a given nuclear fuel cycle. The higher is the waste disposition cost, the more "difficult" it is to manage the fuel cycle's wastes. This subsection will first discuss the cost of disposing of wastes in different categories. It then applies these costs to the two comparisons A and B.

Disposition Cost of Mill Tailings

It is often during milling plant decommissioning that the on-site mill tailings are prepared for final disposition. Thus, it is more convenient to include the decommissioning waste and cost in this category. Based on milling plant decommissioning data from U.S., Canadian and Australian sites, Habib recommends the use of \$850/MT U_3O_8 as the reference value and \$130 to \$4,200/MT U_3O_8 as the wide range.[74] Using a value of 200 m^3 of mill tailings per MT U_3O_8, 43,356 m^3 of mill tailings per GWe-yr and other factors, he arrived at \$156,000 GWe-yr. This translates into \$3.6/m^3 of mill tailings. Similarly, the range corresponds to \$0.55 to \$18/m^3 of mill tailings.

Disposition Cost of LLW

[74] Don Habib, "Uranium Mining and Milling: Waste Volumes and Decommissioning Costs," August 7, 1997.

In 1995, 56% of the commercially-disposed low-level waste volumes was sent to Envirocare facility, 13% sent to U.S. Ecology and the remaining 31% to Barnwell. Based on the cost data available from the last two sites, Kate Huggins developed three disposition scenarios and arrived at a disposal cost range of \$79 to \$382/ft^3 of LLW.[75] The range translates into \$2,800 to \$13,500/m^3 of LLW disposed. This range is consistent with the rapid rise of LLW disposal cost recently.[76]

Although we have classified depleted uranium (DPU) from enrichment as LLW, the cost of its disposition can be considerably higher than that of disposing other LLW of equal volume. Since an underestimate of this cost would favor the LEU(OT), we will segregate the DPU from the rest of the LLW and use a higher cost estimate for DPU. The cost involves the conversion of depleted uranium hexafluoride to the chemically stable U_3O_8, and its burial in a shallow or mined burial site. Habib provided a cost range of \$1.4 million to \$2.2 million/GWe-yr for disposing DPU.[77] The DPU disposal cost will be explicitly shown in Table 12. To avoid double counting, we will exclude the DPU waste volume and cost in calculating the cost of LLW disposal. The LLW (excluding DPU) disposal cost will also be shown on a separate line directly below the DPU line in Table 9.

Disposition Cost of ILW

There is no cost data for the final disposal of ILW. The lower bound is that of LLW, and the upper bound, that of spent fuel. The range for LLW is \$2,800 to \$13,500/m^3, while the reference value for LEU(OT) spent fuel is \$337,000/m^3.[78] In this study, we assume that ILW disposal cost is only twice that of LLW, or 1/12 to 1/60 that of spent fuel. This relatively low cost of ILW benefits the MOX(OT) and SGR fuel cycles, as they have larger volumes of ILW than LEU(OT) has.

Disposition Cost of SF or HLW

The disposition cost of spent fuel has two key components--encapsulation and final disposal in repository. The major cost of encapsulation is the cost of the container such as the multi-purpose canister. Kate Huggins found that the MPC cost ranges from \$4.27 to

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[75] Kate Huggins, Project Performance Corporation, "Low-Level Waste Disposal Prices," September 29, 1997.

[76] Originally, the Barnwell sites in South Carolina charged a perpetual maintenance fee of only \$0.08/ft^3 in the early 1970s. It was increased to \$2.8/ft^3 in the early 1990s. [Virgil Autry, "Agreement State Regulatory Aspects of Low Level Radioactive Waste," Nuclear Materials Management, 35th Annual Meeting Proceedings, Volume XXIII, Institute of Nuclear Materials Management, p.735.] But, recently, the Illinois LLW disposal project estimates that its disposal facility will cost from \$195 to 600/ft^3 or \$7,000 to \$21,000/m3. [Tom Harrison, "Illinois LLW Disposal Project on Hold due to Cost Concerns," Nucleonics Week, April 10, 1997, pp. 13-14.]

[77] Don Habib, "Uranium Enrichment Wastes and Costs," June 10, 1997. Habib also found an additional charge of about \$1,000/GWe-yr for the disposal of other operation waste such as air filters, wipes, personnel protective equipment, wastewater sludges and laboratory waste contaminated with uranium. But, this charge is negligible compared with the DPU disposal cost of \$1.4 to 2.2 million/GWe-yr.

[78] This is based on the typical charge of 1 mill/kwhr that the utility pays for the final disposal of spent fuel. Other number used in the conversion is 26 m^3 of spent fuel/GWe-yr.

$10.27 billions for 63,000 metric tons of LEU spent fuel.[79] This translates into $1.8 to $4.2 millions/GWe-yr or 0.20 to 0.48 mill/kwhr, and this is the range used in this study. There are additional costs of placing spent fuel into the MPCs and of transportation. But, they should be small relative to the MPC cost. A cost of $26.55 billions has been allocated as the total system life cycle cost of the repository for 84,000 metric tons of LEU spent fuel.[80] Thus, the LEU spent fuel disposal cost used in this study is $8.2 millions/GWe-yr or 0.94 mill/kwhr. The cost for both encapsulation and final disposal of LEU spent fuel would be 1.1 to 1.4 mills/kwhr. This is somewhat higher, but not much higher, than the 1 mill/kwhr that utilities pay DOE to take care of their spent fuel.

The equivalent steps for HLW are vitrification and final disposal. Smith and Mote provide a cost range of $330,000 to $530,000/MTHM for conditioning of HLW, ILW and LLW; the vitrification of HLW; and subsequent interim storage of conditioned wastes before return to the customer.[81] With 26 MTHM per GWe-yr, this range corresponds to $8.58 to 13.78 million/GWe-yr or 1.0 to 1.6 mill/kwhr. They further reported that the HLW vitrification accounts for 60% of the above cost-- $5.1 to $8.3 million/GWe-yr or 0.6 to 1.0 mill/kwhr.[82] This is the cost range used in this study. Strictly speaking, the vitrification, as well as the encapsulation, cost would depend on the heat in the HLW and the spent fuel. In one extreme, one may assume that the canister accounts for most of the cost and that the amount of waste in each canister is limited by the same amount of heat. Then, the vitrification or encapsulation cost would be directly proportional to heat. In the other extreme, the canister does not account for most of the cost and that it can be redesigned with changes in size and wall thickness so that the canister cost is not sensitive to heat in the waste. In this situation, the vitrification or encapsulation cost may change little with heat content. Because we do not know at this time which extreme is closer to the reality and we do not want to select the case which is generally more favorable to LEU-OT, we thus choose the case where the vitrification or encapsulation cost is independent of heat. While this cost is assumed to be not sensitive to heat, we however assume that the cost of placing HLW or spent fuel in a repository is proportional to heat. In the EED case with 1,000 years, the HLW repository cost for SGR is 7% lower than the SF repository cost, because the HLW has 7% less integrated heat. In the APD case with 10 years after fuel discharge, the HLW repository cost for SGR is 44% higher, while the APD case with 60 years, the cost is 1% higher.

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[79] Kate Huggins, "Commercial SNF--Storage and Disposal Canister Costs," May 15, 1997.

[80] U.S Department of Energy, *Analysis of the Total System Life Cycle Cost of the Civilian Radioactive Waste Management Program*, DOE/RW-0479, September 1995, p.39.

[81] Mike Smith and Nigel Mote, *Identification of Reprocessing Waste Quantities and Characteristics*, December 1997, p.15. They gave a cost range of $280,000 to $450,000/MTHM. But, one should use the prices (including profit) instead which is 18% higher (i.e. $1.6 million total reprocessing price/$1.36 million total reprocessing cost).

[82] One could have included the other 40% of the cost, because the 40% cost is used to condition ILW and LLW and is part of the waste disposition cost. In this study, because we have only one data source for the vitrification cost, we only use 60% of the cost in order to guard against the possibility that the estimate we use is too high by a factor of up to two.

As to the MOX-OT case, we continue to assume that the vitrification and encapsulation cost is independent of heat as in the LEU-OT and SGR cases above. Also, the encapsulation cost is the same for the same amount of LEU-OT or MOX-OT spent fuel. On the other hand, the repository cost is proportional to heat. Relative to heat in LEU-OT spent fuel, the heat from the same weight of MOX-OT HLW is 51%, 8% or 33% less in EED (1,000 yrs), APD (10 yrs), or APD (60 years) respectively. The heat from the same amount of MOX-OT spent fuel is, however, 320%, 130% and 220% higher than LEU-OT spent fuel. We will use these same, lower and higher percentages to determine the costs of vitrification/encapsulation and of permanent storage in a repository.

Comparison of Total Waste Disposition Costs

Table 12 summarizes our estimates of the waste disposition costs for the three alternative fuel cycles under study. In the reference case, the disposition cost of wastes generated throughout the full MOX-OT fuel cycle is higher than that for the LEU(OT) fuel cycle by 23%. The range is 21% to 25%. As to SGR, the waste disposition cost in the reference case is 21% higher than that of the LEU(OT), and the range is 20% to 22%. Thus, whether one uses reprocessed plutonium once or multiple times, the total waste disposition cost is higher than that of not reprocessing at all by twenty-some percent. If the total waste disposition cost is used as a proxy for the easiness or difficulty in managing waste, we found that reprocessing does not help ease waste management.

Table 12 numbers are based on the assumption that canister spacings in the repository are driven by the equivalent energy density (EED) integrated over 1,000 years. The spacings are important, because they affect the number of spent fuel and waste canisters that can be placed in the repository of a given size and thus the disposal cost. Other methods have been proposed for determining the spacings. We have also calculated the waste disposition costs using the average power density. One set of calculations was made with the assumption that spent fuel and HLW will be placed in the repository 10 years after fuel discharged from the reactor. Another set was made assuming 60 years. In the first set, the waste disposition cost for MOX-OT is 26% higher than that of LEU(OT) with a range of 23% to 30%. In the second set, the waste disposition cost for MOX-OT is 23% higher with a range of 21% to 26%. In the first set for SGR, the waste disposition cost is 44% higher with a range of 37% to 54%. The SGR cost in the second set is 25% higher than that of LEU-OT with a range of 22% to 27%. Thus, if APD is used instead of EED, the waste disposition costs for single or multiple use of plutonium is even higher than those used in our reference case above. Using APD would reinforce our conclusion that disposing of waste from plutonium recycle is more costly.

Table 12: Waste Disposition Costs in $millions/GWe-yr

	Reference			Range		
	LWR(OT)	MOX-OT	SGR	LWR(OT)	MOX-OT	SGR
Mill tailings	0.23	0.18	0.16	.04-1.16	.03-0.89	.02-0.79
DPU from enrichment*	1.80	1.57	1.39	1.40-2.20	1.22-1.92	1.08-1.70
LLW (except DPU)	4.58	4.90	4.90	1.27-7.89	1.38-8.44	1.39-8.41
ILW	0.69	1.45	1.69	0.18-1.19	.35-2.55	0.41-2.96
HLW vit. &/or SF encap.**	3.00	6.17	6.71	1.80-4.20	4.67-7.68	5.15-8.27
HLW &/or SF in reposit.***	8.20	8.47	7.63	8.20-8.20	8.47-8.47	7.63-7.63
TOTAL	18.50	22.75	22.47	12.89-24.83	16.11-29.95	15.68-29.75
CHANGE FROM LEU(OT)		23%	21%		25%-21%#	22%-20%

* Depleted uranium

** HLW vitrification and/or spent fuel encapsulation

*** HLW and/or spent fuel in repository

The 25% corresponds to the numbers above it on the left, while the 21%, to numbers on the right.

IX. FINDINGS AND RECOMMENDATIONS

This study compares the wastes generated in three alternative nuclear fuel cycles. The first cycle is low-enriched uranium in once-through mode (LEU-OT), which is the choice followed by the great majority of the civilian nuclear reactor operators in the world. The second cycle is mixed-oxide fuel in once-through mode (MOX-OT), which reprocessing-pursuing countries currently prefer. The third cycle is self-generating recycle (SGR) where plutonium is reprocessed and recycled repeatedly in the same reactor throughout its operating life. Although current cost picture and the cost trends make SGR unlikely, we included it in our examination so that one can see its advantages and disadvantages. We have tallied the wastes generated throughout the full nuclear life cycle, not just those during reprocessing and at the backend of the fuel cycle.

We arrived at the following findings:

- o Plutonium fuel cycles generate 20-30% less in the waste volume of mill tailings, but 5-10% more low-level waste and 90%-150% more intermediate-level waste relative to the current uranium once-through fuel cycle.

    -- In the MOX-OT case, the tailings are 23% less than LEU-OT, LLW is 6%-8% more, and the ILW is 88% to 115% more.

    -- In the SGR case, the tailings are 32% less, LLW is 5%-9% more, and the ILW is 123% to 149% more.

- o Plutonium fuel cycles generate from 7% less to 44% more heat from high-level waste and spent fuel than that from the spent fuel of the current uranium once-through fuel cycle.

    -- In the MOX-OT case, the HLW and spent fuel generate 4% to 10% more heat than the LEU-OT spent fuel.

    -- In the SGR case, the HLW generates from 7% less heat to 44% more heat than LEU-OT spent fuel, depending rather sensitively on when and how (equivalent energy density or average power density) heat is compared.

- o The total waste disposition cost for a plutonium fuel cycle is higher than that of current uranium once-through fuel cycle by 20% to 25%.

    -- In the MOX-OT case, the cost is higher than the LEU-OT case by 21%-25%.

    -- In the SGR case, the same cost is higher by 20%-22%.

- o If the total waste disposition cost is used as a proxy for quantifying the easiness or difficulty in managing waste, this study found that the reprocessing and the use of plutonium actually make waste management more difficult.

- o There is a potential to reduce the disposal cost in a repository by aging the spent fuel and high level waste longer in temporary storage in order to make them cooler.

We found that a more attractive way to lower the waste disposition cost is to determine the optimal age at which the spent fuel or HLW should generally be transferred to the repository. The selection will be based on the tradeoff of the cost of interim storage versus the cost of final repository storage, which can depend sensitively on heat and thus the age of the HLW or spent fuel. We recommend that such an optimal timing be studied and implemented; by taking into account also other factors such as the spread in the discharged-fuel's ages and characteristics, the mix of civilian and military spent fuel/waste, and the availability of interim and permanent storage.

The United States and other countries need to question the often-quoted statement that the reprocessing of spent fuel and the recycling of plutonium will ease radioactive waste management problems. They should not proceed with plutonium activities based on this unsubstantiated benefit.