

Performance Summary of Advanced Nuclear Fuel Cycles

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SUMMARY

The purpose of this report is to summarize the performance data received and the analyses done for the alternatives addressed in the Programmatic Environmental Impact Statement (PEIS) for the Global Nuclear Energy Partnership (GNEP) program. The results were used as the basis of comparison between the alternatives in the PEIS.

The alternative nuclear fuel cycles considered in the PEIS were selected as potential representative options for satisfying the GNEP purpose and need statement, addressing the requirements to reduce the proliferation risk of using nuclear power (arising from the creation of fissile material such as Pu-239 from U-238 during irradiation) and to reduce the environmental impact from spent fuel (partly due to the creation of higher actinide isotopes from uranium during irradiation, partly due to fission products). Eight alternatives had been selected for the PEIS and have been evaluated, as follows:

1. No-action alternative – continue with “once-through” use of nuclear fuel in light-water reactors (LWRs) as is done today, with direct disposal of spent LWR fuel.
2. HWR alternative – replace today’s LWRs with heavy-water reactors (HWRs), with direct disposal of spent HWR fuel.
3. HTGR alternative – replace today’s LWRs with high-temperature gas-cooled reactors (HTGRs), with direct disposal of spent HTGR fuel.
4. Thorium /uranium alternative – replace use of enriched uranium fuel with a mixture of thorium and uranium in LWRs, with direct disposal of all spent fuel.
5. Repeated thermal recycle of plutonium alternative – use LWRs to recycle plutonium, so that there is essentially no disposal of plutonium in the environment. All spent fuel is processed to recover the plutonium, with disposal of high-level processing wastes.
6. DUPIC alternative – continue with use of LWRs, and use a thermal/mechanical process to reconstitute the fuel for use in an HWR, followed by direct disposal of the spent HWR fuel.
7. Fast reactor recycle alternative – continue with use of LWRs, and use fast reactors to recycle the transuranic (TRU) elements (plutonium, neptunium, americium, and curium). All spent fuel is processed to recover the TRU, with disposal of high-level processing wastes.
8. Fast and thermal reactor recycle alternative – continue with use of LWRs, use LWRs for limited recycle of plutonium and use fast reactors for recycle of the TRU elements. All spent fuel is processed to recover the TRU, with disposal of high-level processing wastes.

Performance data have been collected for all of these alternatives, with the data for each alternative adjusted for power production of 100 GWe-yr per year. The data included uranium requirements, enrichment needs, spent fuel amounts, and generation of radioactive waste. Various performance metrics have been calculated for these alternatives, allowing a comparison of the impact on waste management and infrastructure requirements. The following general conclusions can be made:

1. Alternate once-through strategies provide little or no benefit to waste management, on the order of a factor of 2-3.
2. Recycle strategies that result in the disposal of spent fuel or a significant part of the TRU elements provide small waste management benefits, on the order of a factor of 2-3.
3. For large waste management benefits, it is essential to recover all of the TRU elements to keep them out of the waste stream. In the PEIS, the specific examples of such alternatives show the potential benefit of using fast reactors to recycle the TRU elements.
4. For all alternatives, geologic disposal capability is required.

5. For alternatives that require spent fuel processing, the amount of HLW, LLW-GTCC, and LLW can vary widely depending on processing and operational details, potential waste reduction technologies, and the potential for waste hazard reduction prior to disposal. All wastes will require a disposal path, whether geologic disposal or near-surface burial.

CONTENTS

SUMMARY	v
ACRONYMS	xi
1. INTRODUCTION	1
2. ALTERNATIVE NUCLEAR FUEL CYCLES	1
3. PERFORMANCE DATA	3
4. WASTE MANAGEMENT METRICS	3
5. SUMMARY OF RESULTS	4
Appendix A—Generation of Summary Performance Data	11
Appendix B—Estimates of Waste Volumes for SNF and Processing Wastes	29
Appendix C—Radiotoxicity Reduction – Time to Decay to Level of Natural Uranium.....	51

FIGURES

A-1. Thorium and PWR Decay Heat (from Ref. A6).	20
A-2. “Material Flow of the Whole DUPIC Fuel Cycle,” Figure 10 from Ref. A9.	24
C-1. Normalized radiotoxicity as a function of time after discharge for spent LWR fuel.....	52
C-2. Normalized radiotoxicity as a function of time after discharge for spent HWR fuel.	53
C-3. Normalized radiotoxicity as a function of time after discharge for spent HTGR fuel.....	53
C-4. Normalized radiotoxicity as a function of time after discharge for spent thorium/uranium fuel.	54
C-5. Normalized radiotoxicity as a function of time after discharge for the HLW from processing spent LWR/LWR (MOX-U-Pu) fuel.	54
C-6. Normalized radiotoxicity as a function of time after discharge for the HLW from processing spent LWR and fast reactor fuel.	55
C-7. Normalized radiotoxicity as a function of time after discharge for the HLW from processing spent LWR and fast reactor fuel.	56

TABLES

1. Comparative Summary of the Programmatic Alternatives—Steady-State 100 GWe-yr per year Production Scenario.....	7
A-1. “Transuranic Mass Flows and Thermal Load Reduction Factors”, Table 1 from Reference A1 (Note: Reference Case and Option Notations are for Reference A1, not for the PEIS Cases; Thermal Load Reduction Factors in this Table not used for PEIS.).....	13

A-2.	“Summary of Uranium Resource Requirements,” Table 2 from Reference A1. (Note: Reference Case and Option Notations are for Reference A1, not for the PEIS Cases.).....	14
A-3.	“Mass Flow Data for GNEP PEIS,” Table 1 from Reference A3.	16
A-4.	Normalized Decay Heat, Watt/GWe-d (from Ref. A2).....	17
A-5.	Normalized Decay Heat, Watt/GW _e -d (from Ref. A4).....	18
A-6.	Thorium Fuel Characteristics (from Ref. A5).	20
A-7.	“Mass Flow Data for GNEP PEIS,” Table 1 from Ref. A7.....	22
A-8.	“Pu Isotopic Composition in Various Spent Fuels,” Table 14 from Ref. A9.	24
B-1.	Estimates of Waste Volumes from Operations for Aqueous Separations Processing of Spent LWR Fuel, 60 GWd/MITHM Burnup. (Data from Table 10 in Reference 1.).....	32
B-2.	Estimates of Waste Volumes from Operation of a Transmutation Fuel Fabrication Facility for Fast Reactor Oxide Fuel. (Data from Table 7 in Reference 2.)	33
B-3.	Estimates of Waste Volumes from Operations for Aqueous Separations Processing of Spent Fast Reactor Fuel, 250 GWd/MTIHM Burnup. (Data from Table 8 in Reference 3.).....	33
B-4.	Estimates of Waste Volumes from Aqueous Separations Processing of Spent LWR Fuel, 51 GWd/MTIHM Burnup. (Data from Table 2 in Reference 4.)	34
B-5.	Estimates of Waste Volumes from Electrochemical Separations Processing of Spent Fast Reactor Fuel, 107 GWd/MTIHM Burnup. (Data from Table 3 in Reference 4.)	34
B-5a.	Estimates of Waste Volumes from Aqueous Separations Processing of Spent Fast Reactor Fuel, 107 GWd/MTIHM Burnup. (Data derived from Tables 4 and 5.).....	35
B-6.	Upper Bound Estimate for Annual Waste Generation from Aqueous Separations Processing of Spent LWR Fuel, 45 GWd/MTIHM burnup.....	37
B-7.	Lower Bound Estimate for Annual Waste Generation from Aqueous Separations Processing of Spent LWR Fuel, 45 GWd/MTIHM burnup.....	38
B-8.	Upper Bound Estimate for Annual Waste Generation from Aqueous Separations Processing of Spent LWR Fuel, 51 GWd/MTIHM burnup.....	40
B-9.	Upper Bound Estimates for Annual Waste Generation from Fabrication of Fast Reactor Oxide Transmutation Fuel.	41
B-10.	Upper Bound Estimates of Annual Waste Generation from Aqueous Separations Processing of Spent Fast Reactor Fuel, 107 GWd/MTIHM burnup.....	41
B-11.	Lower Bound Estimate for Annual Waste Generation from Aqueous Separations Processing of Spent LWR Fuel, 51 GWd/MTIHM burnup.....	42
B-12.	Lower Bound Estimates of Annual Waste Generation from Aqueous and Electrochemical Separations Processing of Spent Fast Reactor Fuel, 107 GWd/MTIHM burnup.....	43
B-13.	Upper Bound Estimate for Annual Waste Generation from Aqueous Separations Processing of Spent LWR Fuel, 51 GWd/MTIHM burnup.....	44
B-14.	Upper Bound Estimate for Annual Waste Generation from Aqueous Separations Processing of Spent LWR Pu-MOX Fuel, 50 GWd/MTIHM burnup.....	45

B-15.	Upper Bound Estimates of Annual Waste Generation from Fabrication of Fast Reactor Oxide Transmutation Fuel.	45
B-16.	Upper Bound Estimates of Annual Waste Generation from Aqueous Separations Processing of Spent Fast Reactor Fuel, 105 GWd/MTIHM burnup.....	46
B-17.	Lower Bound Estimate for Annual Waste Generation from Aqueous Separations Processing of Spent LWR Fuel, 51 GWd/MTIHM burnup.....	47
B-18.	Lower Bound Estimate for Annual Waste Generation from Aqueous Separations Processing of Spent LWR Pu-MOX Fuel, 50 GWd/MTIHM burnup.	47
B-19.	Lower Bound Estimates of Annual Waste Generation from Aqueous and Electrochemical Separations Processing of Spent Fast Reactor Fuel, 105 GWd/MTIHM burnup.....	48
B-20.	Summary of Annual Waste Generation for GNEP PEIS Alternatives.	48

ACRONYMS

AFCI	Advanced Fuel Cycle Initiative
CFTC	Consolidated Fuel Treatment Center
DUPIIC	Direct Use of PWR Spent Fuel in CANDU Reactors
EAS	Engineering Alternatives Study
FR	Fast-neutron reactor
GNEP	Global Nuclear Energy Partnership
GTCC	greater-than-Class-C waste
HLW	high-level waste
HTGRs	high-temperature gas-cooled graphite-moderated reactors
HWRs	heavy-water-moderated reactors
ILW	intermediate-level waste
LEU	Low-Enriched Uranium
LLW	Low-Level Waste
LWRs	light-water-moderated reactors
MTHM	metric tons of heavy metal
MTIHM	metric tons of initial heavy metal
NGNP	next generation nuclear plant
PEIS	Programmatic Environmental Impact Statement
Pu-MOX	plutonium and uranium oxide
SNF	spent nuclear fuel
ThOX/UOX	thorium-based oxide
TRU	transuranic
UOX	enriched-uranium oxide

PERFORMANCE SUMMARY OF ADVANCED NUCLEAR FUEL CYCLES

1. INTRODUCTION

The purpose of this report is to summarize the performance data collected and the analyses done for the alternatives addressed in the Programmatic Environmental Impact Statement (PEIS) for the Global Nuclear Energy Partnership (GNEP) program. The results were used as the basis of comparison between the alternatives in the PEIS.

2. ALTERNATIVE NUCLEAR FUEL CYCLES

The alternative nuclear fuel cycles considered in the PEIS were selected as potential options for satisfying the GNEP purpose and need statement. Based on the requirements to reduce the proliferation risk of using nuclear power (arising from the creation of fissile material such as Pu-239 from U-238 during irradiation) and to reduce the environmental impact from spent fuel (partly due to the creation of higher actinide isotopes from uranium during irradiation, partly due to fission products), the options fall in one of four general categories:

1. Once-through use of uranium-based fuel in thermal-neutron reactors (different moderation or fuel type may reduce the proliferation risk and environmental impact by changing isotopic composition of spent fuel)
2. Once-through use of at least some non-uranium-based fuel in thermal-neutron reactors (reduced uranium content may reduce the proliferation risk and environmental impact by changing the isotopic and elemental composition of spent fuel)
3. One or more recycles in thermal-neutron reactors of actinide elements recovered from spent fuel (essentially eliminating one or more of the higher actinide elements from the wastes destined for disposal may reduce proliferation risk and environmental impact)
4. One or more recycles in fast-neutron reactors of actinide elements recovered from spent fuel (essentially eliminating the higher actinide elements from the wastes destined for disposal may reduce proliferation risk and environmental impact).

For the purposes of the PEIS, the following specific examples had been selected as being representative of options in each of these categories:

1. **No-action** (an example in Category 1): the no-action alternative is a continuation of present practice, which is represented as the “once-through” use of enriched-uranium oxide (UOX) fuel in light-water-moderated reactors (LWRs), where the fuel is irradiated in the reactor until it reaches the desired burnup, storage of the discharged spent fuel for an extended period of time (typically 25 years or more to allow for cooling and radioactive decay), followed by disposal of the spent fuel in an engineered repository in a geologic environment.
2. **HWR alternative** (an example in Category 1): “once-through” use of enriched-uranium oxide (UOX) fuel in heavy-water-moderated reactors (HWRs), where the fuel is irradiated in the reactor until it reaches the desired burnup, storage of the discharged spent fuel for an extended period of time (typically 25 years or more to allow for cooling and radioactive decay), followed by disposal of the spent fuel in an engineered repository in a geologic environment.

3. **HTGR alternative** (an example in Category 1): “once-through” use of enriched-uranium oxide (UOX) fuel in high-temperature gas-cooled graphite-moderated reactors (HTGRs), where the fuel is irradiated in the reactor until it reaches the desired burnup, then storage of the discharged spent fuel for an extended period of time (typically 25 years or more to allow for cooling and radioactive decay), followed by disposal of the spent fuel in an engineered repository in a geologic environment.
4. **Thorium alternative** (an example in Category 2): “once-through” use of enriched uranium (UOX) and thorium-uranium oxide (ThOX/UOX) fuel in LWRs (enriched uranium is still included in the fuel to support the fission process), where the fuels are irradiated in the reactor until they reach the desired burnups, storage of the discharged spent fuels for an extended period of time (typically 25 years or more to allow for cooling and radioactive decay), followed by disposal of the spent fuels in an engineered repository in a geologic environment.
5. **LWR/LWR (MOX-U-Pu) alternative** (an example in Category 3): while in general this could be one or more recycles of plutonium by using mixed plutonium and uranium oxide (Pu-MOX) fuel in LWRs, in the PEIS the specific example of continuous, or repeated, recycle of plutonium as Pu-MOX in LWRs is used for the purposes of evaluating the benefits to proliferation risk and environmental impact, since the benefits would be maximized with no disposal of any spent fuel. All discharged spent fuel is processed (typically by chemical means) to recover the uranium and plutonium, with the other transuranic (TRU) elements (neptunium, americium, and curium) and the fission products being in the processing waste, with separation of cesium and strontium for separate decay storage. The recovered plutonium and uranium are used for fabrication of part of the new LWR fuel. Enriched uranium is used for the rest of the new LWR fuel, since uranium enrichment is needed to support the fission process in the reactor. The processing waste is stored for an extended period of time (depending on the time of processing, with the total time from discharge to disposal typically 25 years or more to allow for cooling and radioactive decay), followed by disposal of the high-level processing waste (HLW) in an engineered repository in a geologic environment, with other classes of radioactive waste that were created as a result of processing and recycling also requiring appropriate disposal.
6. **LWR/HWR (DUPIC) alternative** (an example in Category 3): use of enriched-uranium oxide (UOX) fuel in LWRs, followed by thermal and mechanical processing of the spent LWR fuel to recover the fuel for reconstitution into new fuel for “once-through” use in HWRs. A small part of the fission product inventory is vaporized during processing, recovered from the off-gas stream, and is in the processing waste along with the cladding and other assembly hardware. After achieving the desired burnup, the discharged HWR spent fuel is stored for an extended period of time along with the processing waste (typically 25 years or more total time from discharge to disposal to allow for cooling and radioactive decay), followed by appropriate disposal of all processing wastes and the spent HWR fuel in an engineered repository in a geologic environment.
7. **Fast Reactor Recycle alternative** (an example in Category 4): use of enriched-uranium oxide (UOX) fuel in LWRs, followed by separations processing to recover the actinide elements (uranium and TRU). The processing waste contains fission products, hardware, and process-loss amounts of the actinides. The actinides recovered from processing the spent LWR fuel are used as part of the contents of new fuel for the fast-neutron reactors. The fast-neutron reactor example in the PEIS uses a TRU conversion ratio of about 0.5, (ratio of TRU created to TRU destroyed during irradiation in the reactor), which means that new TRU material from spent LWR fuel is needed for each recycle in the fast reactor. Spent fast reactor fuel is also processed to recover the actinides, which are used as part of the contents of new fast reactor fuel, so that the TRU elements are indefinitely recycled in fast reactors, consuming TRU with each recycle. Processing waste contains fission products, hardware, and process-loss amounts of actinides, is stored for an extended period of time (depending on the time of processing, with the total time from discharge to disposal typically 25 years to allow for cooling and radioactive decay), and is followed by disposal of HLW in an engineered repository in a geologic environment and disposal of other classes of wastes as appropriate.

8. **Thermal & Fast Recycle alternative** (an example in Category 4): use of enriched-uranium oxide fuel in LWRs, followed by separations processing to recover the actinide elements (uranium and TRU). The processing waste contains fission products, hardware, and process-loss amounts of the actinides. The plutonium recovered from processing spent LWR fuel is used as part of the contents of new Pu-MOX LWR fuel for one recycle in LWRs. The other actinide elements are separated and stored for use in the fast reactor. The spent Pu-MOX LWR fuel is processed to recover the actinide elements, with the processing waste containing fission products, hardware, and process-loss amounts of the actinides. All of the actinide elements recovered from processing LWR spent fuels, both enriched uranium and Pu-MOX, are used as part of the contents of new fuel for the fast-neutron reactor. The fast reactor has a conversion ratio of about 0.5, so that all TRU elements are indefinitely recycled in fast reactors, consuming TRU with each recycle. Processing waste containing fission products, hardware, and processing loss amounts of actinides, is stored for an extended period of time (depending on the time of processing, with the total time from discharge to disposal typically 25 years to allow for cooling and radioactive decay), and is followed by disposal of HLW in an engineered repository in a geologic environment and disposal of other classes of wastes as appropriate.

The first four alternatives are examples of “once-through” use of nuclear fuel, while the second four alternatives are examples of various methods of recycling nuclear fuel. All of the alternatives require an engineered repository in a geologic environment for the disposal of spent fuel, processing wastes, or both.

3. PERFORMANCE DATA

For the comparison of these alternatives, performance data for relevant parameters were required. These included uranium resource requirements, enrichment needs, net production of TRU, and anticipated waste quantities requiring disposal. Many of the alternatives had been studied previously as part of the DOE Advanced Fuel Cycle Initiative (AFCI) program, although some new studies were performed to complete the information needed in the PEIS, while for others the information was obtained from open-literature sources. The performance data were received from a number of contributors, as listed in the references for Appendix A. The performance data were modified as needed to account for differences in analysis assumptions so that all of the summary data information in Table 1 would have a uniform basis. The required analyses are provided in Appendices A, B, and C.

4. WASTE MANAGEMENT METRICS

Several waste management metrics were selected for comparing the alternatives, including:

1. TRU Disposal (MT per 100 years of operation)
2. Mass of spent nuclear fuel (SNF) to Repository, MTIHM/yr
3. Volume of spent nuclear fuel (SNF) to Repository, m³/yr
4. Volume of Processing Wastes Classified as high-level waste (HLW) to a Repository, m³/yr
5. Volume of Greater-Than-Class-C waste (GTCC) Waste, m³/yr
6. Volume of Low-Level waste (LLW) Waste, m³/yr
7. Thermal Load Reduction Factor (relative to No-Action alternative)
8. Radiotoxicity Reduction – Time to Decay to Natural Uranium Ore Radiotoxicity.

The calculations for the waste masses and volumes listed in Table 1 are given in Appendix B for each alternative, where appropriate. As stated in one of the Table footnotes, for those alternatives that include processing and recycling of the spent fuel, the total losses are assumed to be 0.1%, and include spent fuel processing losses, fuel fabrication losses, and losses from all other activities where such losses may occur. The method for obtaining the radiotoxicity estimates and the results shown in Table 1 is shown in Appendix C.

5. SUMMARY OF RESULTS

As can be seen from Table 1, the performance results allow a comparison of the impact of each of the alternatives with respect to the current no-action alternative, the 'once-through' use of enriched uranium fuel in LWRs, as follows:

Common to all alternatives -

1. Uranium enrichment, although some alternatives require higher LEU enrichment levels than are currently available commercially.
2. Fabrication of reactor fuel, although in some cases this may have to be done remotely, depending on the hazard presented by the fuel constituents.
3. For the alternatives requiring spent fuel processing, HLW, LLW-GTCC, and LLW are created, with volume estimates covering a wide range in each case.
4. Geologic disposal facilities required for SNF, HLW, or both, depending on the alternative.

Specific to each alternative -

HWR alternative as compared to the no-action alternative –

- There is the need to add heavy water production facilities for this alternative.
- Uranium resource requirement is essentially the same.
- The mass of SNF to be disposed is about 2.4 times higher, but the volume of SNF is about the same. TRU content of the SNF is about 35% higher.
- Essentially no change in thermal load for geologic disposal.
- Essentially no change in long-lived radiotoxicity of the disposed spent fuel.

HTGR alternative as compared to the no-action alternative –

- New fuel fabrication technology for HTGR fuel needs to be implemented.
- Uranium resource requirement is essentially the same, although the required LEU enrichment level is higher than commercially available today, at 14%, due to the higher fuel burnup.
- The mass of SNF to be disposed is lower partly due to the higher fuel burnup, but the volume of SNF is higher by about a factor of 2.6 if the fuel compacts can be separated from the hexagonal prismatic blocks, or substantially higher by a factor of 15 if this process is not successful. TRU content is 57% of that for spent LWR fuel.
- The thermal load for geologic disposal is lower by a factor of about 1.4.
- The long-lived radiotoxicity is lower by a factor of about 2-3.

Thorium/Uranium alternative as compared to the no-action alternative –

- A supply of thorium is required, along with implementation of fabrication of thorium oxide fuel.
- Uranium resource requirement is essentially the same, although the required LEU enrichment levels are higher than commercially available today, at 19.9% and 12.2%.
- The mass and volume of SNF to be disposed is about 50% lower, reflecting the higher burnup of the spent fuel. However, the TRU content is only about 28% due to the use of thorium/uranium fuel.
- The thermal load for geologic disposal is estimated to be lower by a factor of 2.
- Although initially lower, the radiotoxicity is higher after 20,000 years by a factor of 2-3.

Repeated thermal recycle of plutonium alternative –

- Remote fabrication of the recycle fuel is likely to be required. Processing of the spent fuel needs to be implemented.
- Uranium resource requirement is 15% lower due to the use of recycled plutonium, but enriched uranium at 4.6% enrichment is still needed.
- There is no SNF to be disposed, only wastes from the processing of the spent fuel. The mass of the TRU elements in the HLW is only 30% that of the no-action alternative.
- The thermal load for geologic disposal is lower by a factor of 1.8.
- Although initially higher, the radiotoxicity of the HLW is lower than the LWR SNF by a factor of 4 by the time the radiotoxicity is reduced to that of the original natural uranium ore.

DUPIC alternative –

- Remote fabrication of the HWR fuel will be required. Processing of the spent LWR fuel also needs to be implemented.
- Uranium resource requirement is 35% lower.
- The spent HWR fuel is directly disposed with slightly less mass but only about 35% of the volume of the LWR SNF. The wastes, including HLW, from the thermal/mechanical processing also requires geologic disposal. Taken together, the total volume requiring geologic disposal appears to be essentially unchanged from the no-action alternative. The TRU content is about 50% that of the no-action alternative.
- The thermal load for geologic disposal is lower by a factor estimated at about 1.6.
- The radiotoxicity is estimated to be the same as, or somewhat lower, than the spent LWR fuel in the no-action alternative.

Fast reactor recycle alternative –

- Remote fabrication of fast reactor fuel is required. Processing of spent LWR fuel and spent fast reactor fuel needs to be implemented.
- Uranium resource requirement is almost 40% lower.
- There is no disposal of spent fuel, only wastes from processing all spent fuel. The volume of HLW is less than the volume of SNF, the amount determined by the details of processing, operation, and other factors. The TRU content disposed is about 0.3% of the amount in spent LWR fuel.
- The thermal load for geologic disposal is lower by a factor of 235.
- The radiotoxicity is substantially lower, by a factor of 100-1000 after several hundred years.

Fast and thermal recycle alternative –

- Fabrication of PuMOX LWR fuel is required. Remote fabrication of fast reactor fuel is required. Processing of spent LWR fuel and spent fast reactor fuel needs to be implemented.
- Uranium resource requirement is 35% lower.
- There is no disposal of spent fuel, only wastes from processing all spent fuel. The volume of HLW is less than the volume of SNF, the amount determined by the details of processing and operation. The TRU content disposed is about 0.3% of the amount in spent LWR fuel.
- The thermal load for geologic disposal is lower by a factor of 235.
- The radiotoxicity is substantially lower, by a factor of 100-1000 after several hundred years.

There are several general conclusions that can be drawn from these comparisons:

1. Alternate “once-through” strategies provide little or no benefit to waste management, on the order of a factor of 2-3, due to the content of the spent fuel to be disposed.
2. Recycle strategies that result in the disposal of spent fuel or a significant part of the TRU elements provide small waste management benefits, on the order of a factor of 2-3.
3. For large waste management benefits, it is essential to recycle all of the TRU elements to keep them out of the waste stream. In the PEIS, the specific examples of such alternatives show the potential benefit of using fast reactors to recycle the TRU elements.
4. For all alternatives, geologic disposal capability is required.
5. For alternatives that require spent fuel processing, the amount of HLW, LLW-GTCC, and LLW can vary widely depending on processing and operational details, potential waste reduction technologies, and the potential for waste hazard reduction prior to disposal. All wastes will require a disposal path, whether geologic disposal or near-surface burial.

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Table 1. Comparative Summary of the Programmatic Alternatives—Steady-State 100 GWe-yr per year Production Scenario.

Case Description	No Action (Once-Through Fuel Cycle)	HWR or HTGR Alternative (Once-Through Fuel Cycle)		Thorium Alternative (Once-Through Fuel Cycle)	Thermal Recycle Alternative		Thermal /Fast Recycle Alternative (CR=0.5)	
		All HWR	All HTGR		Option 1-- LWR/LWR (MOX-U-Pu)	Option 2— LWR/HWR (DUPIC)	Fast Reactor Recycle Only	Thermal & Fast Recycle Option
Reactor Power Production (GWe-yr/yr) ^a								
LWR-UOX or HWR-UOX or HTGR-UOX	100 (LWR)	100 (HWR)	100 (HTGR)	0	0	72.7 (LWR)	62.1 (LWR)	64.7 (LWR)
LWR-MOX-U-Pu, or LWR-HWR	0	0	0	0	100 (LWR)	27.3 (HWR)	0	7.6 (LWR)
Fast-Neutron Reactor (FR)	0	0	0	0	0	0	37.9 (FR)	27.7 (FR)
LWR-ThOX/UOX	0	0	0	100 (LWR)	0	0	0	0
Other Facilities Required								
Enrichment Facility	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Fuel Fabrication Facility	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Heavy Water Production Facility	No	Yes	No	No	No	Yes	No	No
Nuclear Fuel Recycling Center	No	No	No	No	Yes	Yes	Yes	Yes
Uranium (Natural and Low-Enriched Uranium [LEU]) or Thorium Resource Requirement (Annual)								
Natural U Feed (MT/yr)	19600	21400	22800	19600	16500	12800	12200	12700
LEU (MT/yr)	2170	5300	770	410/80	1660	1800	1350	1400
LEU Enrichment	4.4	2.1	14.0	19.9/12.2	4.6	3.5	4.4	4.4
Nat. Thorium (MT/yr)	0	0	0	535	0	0	0	0

Table 1. (continued).

Case Description	No Action (Once-Through Fuel Cycle)	HWR or HTGR Alternative (Once-Through Fuel Cycle)		Thorium Alternative (Once-Through Fuel Cycle)	Thermal Recycle Alternative		Thermal/Fast Recycle Alternative (CR=0.5)	
		All HWR	All HTGR		Option 1-- LWR/LWR (MOX-U-Pu)	Option 2-- LWR/HWR (DUPIC)	Fast Reactor Recycle Only	Thermal & Fast Recycle Option
SNF/ HLW/TRU Production/Cs/Sr storage/Recovered U storage (Annual)								
Fuel Burn-up at Discharge (GWd/MTIHM)	51	21	100	149 (UOX) 75 (ThOX)	45	35 (UOX) 15 (HWR)	51 (LWR) 107 (FR)	51 (LWR) 50 (LWR – MOX/Pu) 105 (FR)
Net TRU Production (MT/yr)	28	38	16	7.8	8.3	15	0.10	0.11
Amount of TRU to waste (MT/yr)	28	38	16	7.8	8.3	15	0.10	0.11
Mass of SNF to repository (MTIHM/yr) ^b	2170	5300	770	1025	0	1800	0	0
Cs/Sr to decay storage (MT/yr)	0	0	0	0	12	0	12	12
Recovered U to Storage (MT/yr)	0	0	0	0	2250	0	1250	1230
Waste Management Metrics								
Volume of SNF to repository, m ³ /yr	975	1060	2600 – 14940 ^c	460	0	360	0	0
Volume of Processing Wastes Classified as HLW to repository, LB; UB, (m ³ /yr) ^d	0	0	0	0	32; 870	ND ^e ; 300-800	26-62; 920	27-55; 905
Volume of GTCC Waste from processing, LB; UB (m ³ /yr) ^d	0	0	0	0	170; 6,690	120; ND ^e	195-210; 6840	186-196; 6,575
Volume of Low-Level Waste from processing; LB: UB (m ³ /yr) ^d	0	0	0	0	17; 26,000	ND ^e	59-209; 38,160	49-161; 34,200
Thermal Load Reduction Factor (relative to No Action)	1	0.9	1.4	2.0	1.8	1.6	235	235
Radiotoxicity Reduction –Time to Decay to Natural U Ore Radiotoxicity (Yr)	240,000	255,000	85,000	525,000	55,000	ND ^e	375	400

Table 1. (continued).

CF = Capacity Factor; ND= No Data Available

a 100 GWe-yr per year is the assumed power production. Since reactor capacity factors (i.e., the percentage of time that the reactor is producing power) are less than 100%, the installed capacity of the reactors must be greater than 100 GWe. Typical values of reactor capacity factors are 90% or higher for LWRs and other thermal reactors, and are about 80-85% for fast reactors.

b Mass listed is only for the remaining heavy metal and the fission products in the spent fuel; no hardware or cladding is included

c The lower value represents the volume of the fuel compacts after separation from the graphite hexagonal prismatic blocks; the higher value represents the volume of spent fuel assuming that the fuel compacts are still in the graphite hexagonal prismatic blocks.

d Lower values represent the “lower bound” (LB) estimates of waste by considering waste from the spent fuel only, with no consideration of wastes from operations, maintenance, etc. The LB estimate considers potential volume reductions associated with advanced waste forms, decay storage to reduce hazard, and potential reclassification of wastes so that classification and disposal requirements are based on hazard instead of origin. Upper values represent the “upper bound” (UB) and are estimated using projections based on existing technologies and operating experience, and all fission products including gases are considered to be HLW with no allowance for advanced waste management approaches like decay storage of wastes with shorter-lived hazards prior to disposal.

e Based on references, an estimate for HLW production is 0.35 m³/yr per TWhe, or 300 m³/yr for 100 GWe-yr per year; however, this value includes HLW from operating the processing plant, and represents more than just the materials originating in the fuel. As such, this value is not compatible with the HLW values listed for the other alternatives that only consider HLW originating with the spent fuel. There is also a value for ILW of 0.55 m³ per terawatt-hr in the references, which may include some material that would be classified as HLW in the United States, and would result in another 482.13 m³, so the total is conservatively estimated at 789 m³, rounded to a range of 300-800 m³/yr for 100 GWe-yr/yr. References also provided an estimate for the LLW at 5.91 m³ per terawatt-hr. This would result in LLW waste generation of $876.6 \times 5.91 = 5180$ m³/yr. Since it is not known whether this represents an upper or lower bound, or represents an expectation based on recent results, this value is not listed in the Table. Although no radiotoxicity data was available for the DUPIC approach, given that all of the fission products and actinides are sent to the repository as with the current once-through approach, and that the direct disposal of spent PWR fuel and spent HWR fuel have similar transient radiotoxicity characteristics, it is likely that the time required for the radiotoxicity of the disposed spent HWR fuel from DUPIC to decrease to that of natural uranium ore to be similar as well, i.e., on the order of 100,000 – 300,000 years.

Appendix A

Generation of Summary Performance Data

Generation of Summary Performance Data

The source of the summary data is provided in this section. Where necessary to provide a consistent basis, adjustments to the data are discussed. Much of the source data is taken from studies where the installed capacity was the basis for analysis, rather than the actual production. Since a valid comparison between alternatives requires each alternative to generate the same amount of power, selected as 100 GWe-yr per year, the source data are modified as follows to obtain the data required in Table 1.

A-1. No-Action Fuel Cycle Alternative – Once-through LWRs

For this case, the source data is from Ref. A1. Tables 1 and 2 from Ref. A1 are reproduced here as Tables A-1 and A-2. The LWR data was calculated for an installed capacity of 100 GWe with a capacity factor of 0.9, resulting in an electricity production of 90 GWe-yr per year, so the following data were used, with the calculations for adjustments as given in the following:

Discharge Burnup (GWd/MTIHM)	51
LWR-UOX Power Production (GWe-yr/yr)	100
(Assuming 1000 MWe per LWR, at 90 % capacity factor, generation of 100 GWe-yr per year would require 111 LWRs, rounded to ~110 reactors in Table 4.7-1)	
LEU Requirement (MT/yr)	$1953/0.9 = 2170$
LEU U-235 Enrichment	4.4%
Natural U Feed (MT/yr)	
Uranium tails enrichment was assumed to be 0.2% in the reference for this calculation; all other cases such as HWR, HTGR, CORAIL, etc. used 0.25%. The natural uranium requirements have been adjusted for this case for a uranium tails enrichment of 0.25%, i.e., the natural uranium requirement of 16121 MT/yr is increased to 17657 MT/yr. Accounting for capacity factor, this becomes:	
	$17657/0.9 = 19619$, rounded to 19600.
Mass of SNF to Repository (MTIHM/yr)	2170
Mass of Processing Wastes Classified as HLW to Repository (MT/yr)	0
Cs/Sr to Decay Storage (MT/yr)	0
Recovered U to Storage (MT/yr)	0
TRU Production (MT/yr)	$25.3/0.9 = 28.1$, rounded to 28
Amount of TRU to Waste (MT/yr)	28.1, rounded to 28
TRU Disposal per 100 Years (MT)	$28.1 \times 100 = 2810$, rounded to 2800
Thermal Load Reduction Factor	1.0; No-action alternative is the basis

Table A-1. “Transuranic Mass Flows and Thermal Load Reduction Factors”, Table 1 from Reference A1 (Note: Reference Case and Option Notations are for Reference A1, not for the PEIS Cases; Thermal Load Reduction Factors in this Table not used for PEIS.)

Case Description	Once-Through	Reference Approach UREX+1a			Alternative 1 U-Pu Co-extraction/MCX			Alternative 2 UREX+2/UREX+3/UREX+4			
		Fast Rx. Recycle CR=0.25	Fast Rx Recycle CR=0.5	Fast Rx Recycle CR=1.1	LWR Recycle Only	2-Tier: LWR & Fast Rx Recycle	Fast Rx. Recycle only	2-Tier: Am/Cm Disposal	2-Tier Am/Cm Recovery	2-Tier Cm Storage	Fast Rx. Only Cm Storage
Case No.	---	RA1, with varying CR values			1A	1B1	1B2	2A	2B	2C1	2C2
Reactor Power, GWe											
LWR-UOX	100	70	60	0	90	63	60	67	60	61	60
LWR-MOX	0	0	0	0	10	7	0	6	6	6	0
Fast Reactor	0	30	40	100	0	30	40	27	34	34	40
TRU Prod. in UOX, MT/y	25.3	17.8	15.2	0	22.7	16.0	15.2	17.0	15.3	15.3	15.3
LWR MOX											
Charge, MT/y					20.4	14.3		16.1	14.4	14.5	
Discharge, MT/y					13.8	9.7		13.5	12.1	12.1	
Fast Reactors											
Charge, MT/y		75.8	92.6	134.2		81.3	92.6	65.0	88.7	89.8	99.2
Discharge, MT/y		58.0	77.4	144.8		70.0	77.4	54.8	75.8	76.9	83.9
TRU to Waste, MT/y	25.3	0.08	0.09	0.14	15.0	0.10	0.09	4.28	0.10	0.10	0.10
Thermal Load Reduction Factor for Repository Waste (relative to once-through fuel cycle)	1	165	160	200	1.1	145	150	3.3	148	148	150

Table A-2. "Summary of Uranium Resource Requirements," Table 2 from Reference A1. (Note: Reference Case and Option Notations are for Reference A1, not for the PEIS Cases.)

Case	Once-Through	Reference Approach			U-Pu Co-extraction		UREX+2/+3/+4			
		FR Recycle CR=0.25	<u>FR Recycle CR=0.5</u>	FR Recycle CR=1.1	LWR Recycle Only	2-tier LWR&FR Recycle	Am/Cm Disposal 2-Tier	Am/Cm Recovery 2-Tier	Cm Storage 2-Tier	Cm Storage FR only
Reactor Power, GWe										
LWR-UOX	100	70	<u>60</u>	0	90	63	67	60	60	60
LWR-MOX	0	0	<u>0</u>	0	10	7	6	6	6	0
Fast Reactor	0	30	<u>40</u>	100	0	30	27	34	34	40
LWR-UOX										
LEU, MT/y	1953	1372	<u>1171</u>	0	1750	1231	1311	1178	1181	1177
Nat. U Feed, MT/y	16121	11320	<u>9664</u>	0	14441	10163	10820	9722	9747	9712
DU Unused, MT/y	14168	9949	<u>8493</u>	0	12691	8931	9509	8544	8566	8535
MOX DU Charge, MT/y					187	130	106	96	95	0
FR DU Charge, MT/y		7	<u>18</u>	94		13	12	15	15	17
Total U Req., MT/y	16121	11327	<u>9682</u>	94	14628	10306	10938	9833	9857	9729

A-2. HWR Once-Through Fuel Cycle Alternative

For this case, the source data is from Ref. A2 and A3, identified as ACR-700. (It should be noted that for the metrics being calculated on system requirements and waste management, estimates using ACR-1000 are similar.) Table 1 from Ref. A3 is reproduced in this report as Table A-3. The HWR data in the reference used a capacity factor of 0.9 for a production capacity of 90 GWe-yr per year. The following data were used, with the calculations for adjustments as given in the following:

Discharge Burnup (GWd/MTIHM)	20.5 (HWR-UOX), rounded to 21
HWR-UOX Power Production (GWe-yr/yr)	100 (HWR)
(Analyses were done for 680 MWe per HWR; at 90% capacity factor, 163.4 reactors are required for 100 GWe-yr per year generation, rounded to ~165)	
LEU Requirement (MTHM/yr)	$4800/0.9 = 5333$, rounded to 5300
LEU U-235 Enrichment	2.10%
Natural U Feed (MTHM/yr)	$19263/0.9 = 21403$, rounded to 21400
Mass of SNF to Repository (MTIHM/yr)	$4800/0.9 = 5333$, rounded to 5300
Mass of Processing Wastes Classified as HLW to Repository (MT/yr)	0
Cs/Sr to Decay Storage (MT/yr)	0
Recovered U to Storage (MT/yr)	0
TRU Production (MT/yr)	$33.8/0.9 = 37.6$, rounded to 38
Amount of TRU to Waste (MT/yr)	37.6, rounded to 38
TRU Disposal per 100 Years (MT)	$37.6 \times 100 = 3760$, rounded to 3800
Thermal Load Reduction Factor	~0.88, based on the integrated decay heat estimate from 100-1500 years in Table A-4 (Ref. A2), rounded to 0.9

Table A-3. "Mass Flow Data for GNEP PEIS," Table 1 from Reference A3.

Case Description	No Action (Once Through Open Fuel Cycle)	GNEP Recycle (CR=0.5)	Thorium (Once Through Open Fuel Cycle)	Continuous Thermal Recycle Alternative (7 cycles for LWR/LWR) (1 cycle for LWR/HWR)		HWR/HTGR Open Fuel Cycle	
				All LWR/LWR (MOX-TRU)	All LWR/HWR (DUPIC)	All HWR	All HTGR
Reactor Power (GWe)							
LWR-UOX or HWR-UOX or HTGR-UOX				0		100	100
LWR-MOX/TRU or LWR-HWR				100		0	0
Fast Reactor				0		0	0
LWR-ThUOX				0		0	0
Uranium or Thorium Resource Requirement (Annual)							
Natural U Feed (MT/yr)				16,346		19,263	20,561
LEU, 3-5% (MT/yr) or 1% (HWR)				1,495		4,800	689 ^{a)}
LEU, 19,9% (MT/yr)				0		0	0
Thorium-232 (MT/yr)				0		0	0
Depleted Uranium Unused (MT/yr)				14,295		14,463	19,728
SNF/ HLW/TRU Production/Cs/Sr storage/Recycled U storage (Annual)							
TRU Production in UOX (MT/yr)				0		33.8	14.0
Amount of TRU to waste (MT/yr)				0.14 ^{b)}		33.8	14.0
Amount SNF to repository (MT/yr)				0		4,800	689
Amount HLW to repository (MT/yr)				98.3		0	0
Cs/Sr to decay storage (MT/yr)				1.1		0	0
Recycled U to Storage				1,958		0	0

Table A-3. (continued).

Case Description	No Action (Once Through Open Fuel Cycle)	GNEP Recycle (CR=0.5)	Thorium (Once Through Open Fuel Cycle)	Continuous Thermal Recycle Alternative (7 cycles for LWR/LWR) (1 cycle for LWR/HWR)		HWR/HTGR Open Fuel Cycle	
				All LWR/LWR (MOX-TRU)	All LWR/HWR (DUPIC)	All HWR	All HTGR
TRU Inventories							
In UOX LWRs or UOX HWRs or UOX HTGRs				0.0		31.7	20.9
In MOX-TRU LWRs				573		0	0
In Fast Reactors				0		0	0
In ThUOX LWRs				0		0	0
In SNF Storage				0.0		949	378
TRU Disposal (per 100 years of operation)				14.1 ^{e)}		3,376	1,398
Thermal Load Reduction Factor (relative to No Action Alternative)							

Table A-4. Normalized Decay Heat, Watt/GWe-d (from Ref. A2)

		Discharge	1	5	10	30	100	500	1000	1500	Integral
ACR-700 20.5 GWD/t Burnup	FP	271249.1	1055.1	145.5	82.2	46.4	8.9	0.0	0.0	0.0	385.8
	HM	16718.4	29.3	12.0	13.4	16.2	15.7	8.8	5.4	3.9	10663.0
	FP+HM	288691.8	1097.7	157.6	95.7	62.6	24.6	8.8	5.4	3.9	11048.8
PWR 33 GWD/t Burnup	FP	184573.0	893.4	145.0	82.8	45.8	8.7	0.0	0.0	0.0	374.5
	HM	10881.5	46.1	17.6	18.6	20.2	17.8	9.0	5.2	3.6	10647.0
	FP+HM	195454.5	939.5	162.6	101.3	66.0	26.5	9.0	5.2	3.6	11021.5
PWR 50 GWD/t Burnup	FP	121090.9	740.0	143.3	82.4	44.8	8.5	0.0	0.0	0.0	364.4
	HM	7418.2	66.8	27.4	26.7	24.2	17.9	7.9	4.5	3.0	9401.1
	FP+HM	128509.1	806.8	170.6	109.1	69.0	26.3	7.9	4.5	3.0	9765.5

A-3. HTGR Once-Through Fuel Cycle Alternative

For this case, the source data is from Ref. A3 and A4, where this case is identified as NGNP. The HTGR data as listed in Table A-3 were calculated with a capacity factor of 0.9 for a production capacity of 90 GWe-yr per year. The following data were used, with the calculations for adjustments as given in the following:

Discharge Burnup (GWd/MTIHM)	99.9 (HTGR-UOX), rounded to 100
HTGR-UOX Power Production (GWe-yr/yr)	100 (HTGR)
(Analyses were done for 600 MWth, at 47.7% thermal efficiency, which gives 286.2 MWe per reactor, rounded to 285 MWe in the Table; for 100 GWe-yr per year generation, at 90% capacity factor, 388.2 reactors are required, rounded to ~390.)	
LEU Requirement (MT/yr)	$689/0.9 = 765.7$, rounded to 770
LEU U-235 Enrichment	14.0%
Natural U Feed (MT/yr)	$20561/0.9 = 22845$, rounded to 22800
Mass of SNF to Repository (MTIHM/yr)	$689/0.9 = 766$, rounded to 770
Mass of Processing Waste Classified as HLW to Repository (MT/yr)	0
Cs/Sr to Decay Storage (MT/yr)	0
Recovered U to Storage (MT/yr)	0
TRU Production (MT/yr)	$14/0.9 = 15.6$, rounded to 16
Amount of TRU to Waste (MT/yr)	16
TRU Disposal per 100 Years (MT)	$15.6 \times 100 = 1560$
Thermal Load Reduction Factor	1.44, based on the integrated decay heat estimated from 100-1500 years in Table A-5 (Ref. A4), rounded to 1.4

Table A-5. Normalized Decay Heat, Watt/GW_e-d (from Ref. A4).

		Discharge	1	5	10	30	100	500	1000	1500	Integral
NGNP 100 GWD/t Burnup	FP	116531.6	545.4	91.9	58.0	33.7	6.4	0.0	0.0	0.0	276.5
	HM	5227.4	35.9	10.4	11.2	12.8	11.5	5.6	3.0	1.8	6485.3
	FP+HM	121778.0	581.3	102.3	69.3	46.4	17.9	5.6	3.0	1.8	6761.8
PWR 33 GWD/t Burnup	FP	184573.0	893.4	145.0	82.8	45.8	8.7	0.0	0.0	0.0	374.5
	HM	10881.5	46.1	17.6	18.6	20.2	17.8	9.0	5.2	3.6	10647.0
	FP+HM	195454.5	939.5	162.6	101.3	66.0	26.5	9.0	5.2	3.6	11021.5
PWR 50 GWD/t Burnup	FP	121090.9	740.0	143.3	82.4	44.8	8.5	0.0	0.0	0.0	364.4
	HM	7418.2	66.8	27.4	26.7	24.2	17.9	7.9	4.5	3.0	9401.1
	FP+HM	128509.1	806.8	170.6	109.1	69.0	26.3	7.9	4.5	3.0	9765.5

A-4. Thorium Once-Through Fuel Cycle Alternative

For this case, the source data is from Ref. A5, using the thorium seed-blanket approach. The spreadsheet calculation is reproduced in this report as Table A-6. The thorium analyses were done on a different basis than the other cases, so it was necessary to normalize the results based on results for an LWR once-through case using the same analysis method, which showed a natural uranium requirement of 20,257 MT/yr instead of 19620 MT/yr, or a ratio of 0.969.

Discharge Burnup (GWd/MTIHM)	149 (UOX LWR pins) 75 (UOX/ThOX LWR pins)
LWR-ThUOX Power Production (GWe-yr/yr)	100 (UOX/ThOX LWR)
The LWR is the same as for the current no-action alternative, LWRs of 1000 MWe each, 90% capacity factor, so that the number of reactors needed is the same, rounded to ~110.	
LEU Requirement @ 19.9% (MT/yr)	$423.4 \times 0.969 = 410.3$, rounded to 410
LEU Requirement @ 12.2% (MT/yr)	$82.5 \times 0.969 = 79.9$, rounded to 80
Total LEU Requirement (MT/yr)	$410.3 + 79.9 = 490.2$, rounded to 490
Natural U Feed (MT/yr)	$20186 \times 0.969 = 19560$, rounded to 19600
Natural Thorium (MT/yr)	$552.1 \times 0.969 = 535$
Mass of SNF to Repository (MTIHM/yr)	$1058 \times 0.969 = 1025$
Mass of Processing Waste Classified as	
HLW to Repository (MT/yr)	0
Cs/Sr to Decay Storage (MT/yr)	0
Recovered U to Storage (MT/yr)	0
TRU Production (MT/yr)	$8.0 \times 0.969 = 7.75$, rounded to 7.8
Amount of TRU to Waste (MT/yr)	$8.0 \times 0.969 = 7.75$, rounded to 7.8
TRU Disposal per 100 Years (MT)	$7.75 \times 100 = 775$, rounded to 780
Thermal Load Reduction Factor	1.97, based on the integrated decay heat estimate from 100-1500 years, Figure A-1 (Ref. A6), rounded to 2.0

Table A-6. Thorium Fuel Characteristics (from Ref. A5).

		LWR	seed	RTF blanket	seed+blanket
Power Output	GW(e)	100			100
	GW(th)	303.03	172.73	130.30	303.03
Average Burnup	Gw(th)*days/MT	51	149	75	
HM(U)	MT/days	5.942	1.159	0.226	1.385
HM(Th)				1.512	1.512
	days/year	365.25	365.25	365.25	
HM(U)	MT/y	2170.2	423.4	82.5	505.9
HM(Th)				552.1	552.1
enrichment	w/o	4.553	19.9	12.2	
NU enrichment	w/o	0.711	0.711	0.711	
tail	w/o	0.25	0.25	0.25	
Feed Factor		9.334	42.625	25.922	
Waste Factor		8.334	41.625	24.922	
NU	MT/y	20257.0	18047.9	2138.4	20186.3
Natural Th	MT/y			552.1	552.1
DU	MT/y	18086.82	17624.5	2055.9	19680.4

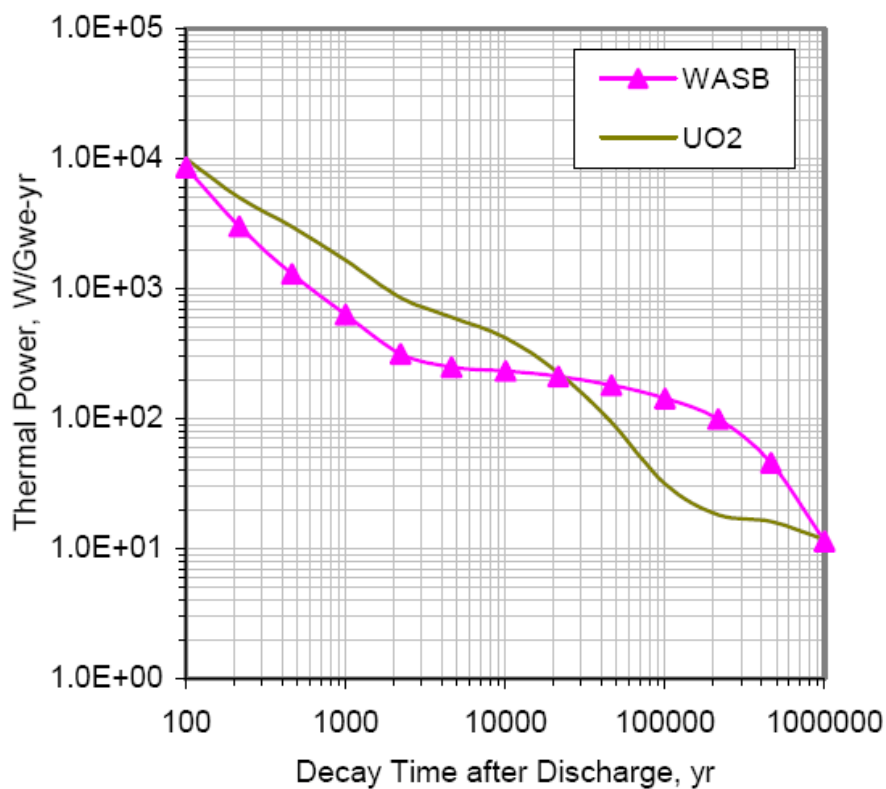


Figure A-1. Thorium and PWR Decay Heat (from Ref. A6).

A-5. Thermal Reactor Recycle Alternative, Option 1, LWR/LWR (MOX-U-Pu)

For this case, the source data is from Ref. A7 and A8, where this case is identified as CORAIL-Pu. The LWR data was calculated with a capacity factor of 0.9 for a production capacity of 90 GWe-yr per year. The following data were used, as shown in Table A-7, with the calculations for adjustments as indicated:

Discharge Burnup (GWd/MTIHM)	45
LWR-MOX-U-Pu Power Production (GWe-yr/yr)	100
(The reactors in this case are the same as for the current no-action alternative, LWRs of 1000 MWe each, 90% capacity factor, so the number of reactors needed is the same ~110.)	
LEU Requirement (MT/yr)	1495/0.9 = 1661, rounded to 1660
(Note: The LEU requirement only reflects the enriched uranium required for the UOX fuel, which is approximately 2/3 of the fuel required. The remaining fuel is Pu-MOX, fabricated from the recovered plutonium and either natural, depleted, or recovered uranium.)	
LEU U-235 Enrichment	4.6%
Natural U Feed (MT/yr)	14807/0.9 = 16452, rounded to 16500
Mass of SNF to Repository (MTIHM/yr)	0
Cs/Sr to Decay Storage (MT/yr)	10.8/0.9 = 12
Recovered U to Storage (MT/yr)	2023/0.9 = 2248, rounded to 2250
TRU Production (MT/yr)	7.5/0.9 = 8.3
Amount of TRU to Waste (MT/yr)	7.5/0.9 = 8.3
TRU Disposal per 100 Years (MT)	8.3 × 100 = 830
Thermal Load Reduction Factor	1.82, rounded to 1.8

Another alternative for thermal recycle presented in Table A-7 is identified as CORAIL-TRU. It should be noted that this case only examines the possibility of continuous recycle of all of the actinides in thermal reactors such as LWRs from a nuclear physics standpoint. In reality, the creation of higher actinide isotopes during recycle, such as californium and berkelium, with very high radiation, would make this option extremely difficult to achieve in practice due to the much higher shielding requirements for all activities, including handling, transport, processing, and fuel fabrication, much more so than for any of the other alternatives being discussed. Estimates of the radiation for the CORAIL-TRU case have been generated, showing that while the gamma radiation for CORAIL-TRU is comparable to that for spent LWR fuel, the neutron radiation is more than three orders of magnitude higher than spent LWR fuel (Ref. A8), and one to two orders of magnitude than for spent fast reactor fuel, depending on the fast reactor TRU conversion ratio.

Photon and Neutron Sources and Decay Heat per Initial Heavy Metal Metric Ton									
	UOX		CORAIL-TRU		ABR CR=0.5		ABR CR=1.0		
	Discharge	10 year	Discharge	10 year	Discharge	10 year	Discharge	10 year	
Gamma (W)	5.45E+05	6.20E+02	5.34E+05	5.15E+02	1273972	1.47E+03	883770	8.07E+02	
Decay heat (W)	1.98E+06	2.04E+03	2.00E+06	1.22E+04	4311670	2.31E+04	3032268	4.20E+03	
Neutron (sec)	1.64E+09	7.19E+08	9.08E+12	7.16E+11	1.09E+11	6.36E+10	8.87E+09	4.71E+09	
Ratio compared to UOX									
Gamma			0.98	0.83	2.34	2.37	1.62	1.30	
Decay heat			1.01	5.98	2.17	11.37	1.53	2.06	
Neutron			5551.34	995.69	66.86	88.46	5.42	6.54	
Gamma (MeV/sec)									
Actinide	2.01E+17	3.44E+12	1.01E+17	4.61E+13					
FP	3.20E+18	3.87E+15	3.23E+18	3.17E+15					

Table A-7. "Mass Flow Data for GNEP PEIS," Table 1 from Ref. A7.

	CORAIL-TRU	CORAIL-Pu	UOX/MOX
Reactor Power (GWe)			
LWR-UOX or HWR-UOX or HTGR-UOX	0	0	90.21
LWR-MOX/TRU or LWR-HWR	100	100	9.79
Fast Reactor	0		
LWR-ThUOX	0		
Uranium or Thorium Resource Requirement (Annual)			
Natural U Feed (MT/yr)	16,346	14,807	15,288
LEU, 3-5% (MT/yr) or 1% (HWR)	1,495	1,495	1,780
LEU, 19,9% (MT/yr)	0		
Thorium-232 (MT/yr)	0		
Depleted Uranium Unused (MT/yr)	14,295	12,674	13,508
SNF/ HLW/TRU Production/Cs/Sr storage/Recycled U storage (Annual)			
TRU Production in UOX (MT/yr)	0	0	23.3
Amount of TRU to waste (MT/yr)	0.14 ^{a)}	7.5	17.1
Amount SNF to repository (MT/yr)	0	0	193.2
Amount HLW to repository (MT/yr)	88.5	89.6	86.4
Cs/Sr to decay storage (MT/yr)			Use Table 3.6-1 value
Recycled U to Storage	1958	2023	1491
TRU Inventories (MT)			
In UOX LWRs or UOX HWRs or UOX HTGRs	0.0	0	53
In MOX-TRU LWRs	573	260	85
In Fast Reactors	0		
In ThUOX LWRs	0		
In SNF Storage	0.0	0	435
TRU Disposal (per 100 years of operation)	14.1 ^{b)}	752	1,706
Thermal Load Reduction Factor (relative to No Action Alternative)			
Thermal Load Reduction Factor (relative to No Action Alternative)	82.2	1.8	1.1

A-6. Thermal Reactor Recycle Alternative, Option 2 - LWR/HWR (DUPIC)

For this case, the source data is from Ref. A9, where this case is identified as DUPIC, or the Direct Use of Spent PWR Fuel In CANDUs. The following data were used, with the calculations for adjustments as indicated:

Discharge Burnup (GWd/MTIHM)	35 (LWR-UOX) 15 (HWR-UOX)
LWR-UOX Power Production (GWe-yr/yr)	72.7 (UOX)
	(Assuming the same LWRs as in the no-action alternative, 1000 MWe, 90% capacity factor, but lower discharge burnup, this requires 80.8 reactors, rounded to ~80)
HWR-UOX Power Production (GWe-yr/yr)	27.3 (HWR)
	(Assuming the HWRs have about the same power level as in the once-through HWR alternative; 680 MWe, 90% capacity factor, but lower discharge burnup, this requires 44.6 reactors, rounded to ~45)
LEU Requirement (MT/yr)	1800
	(It should be noted that as shown in Fig. A2, 18 MT of LEU is used to produce 0.727 GWe-yr. If the thermal efficiency is about 34% for the PWR, this would require producing 2.138 GWth-yr, or 780 GWth-days of energy. To produce this much thermal energy with 18 MT of uranium would appear to require a burnup of about 43 GWd/MTIHM, not 35. Similarly, for the HWR, 17.84 MT of spent UOX fuel is used to produce 0.273 GWe-yr. This would require a burnup of about 16 GWd/MTIHM, close to the value of 15 given in the reference. The potential impact from these potential discrepancies should be considered when developing conclusions about the DUPIC approach.)
LEU U-235 Enrichment	3.5%
Natural U Feed (MT/yr)	128 MT uranium per GWe-yr (Fig. A2) 12800 MT/yr for 100 GWe-yr/yr
Mass of SNF to Repository (MTIHM/yr)	1800
Cs/Sr to Decay Storage (MT/yr)	0
Recycled U to Storage (MT/yr)	0
TRU Production (MT/yr)	7962 g (Pu)/MTHM (Table A-8) 17.66 MTHM per GWe-yr gives 0.14 MT(Pu), or 14.06 MT(Pu)/yr for 100 GWe-yr/yr
	(This is only plutonium content in the spent fuel. To estimate production of TRU, it is noted that Pu is about 96% of TRU production at 20.5 GWd/MT in an HWR starting from UOX fuel [Ref. A2], and is similar to 33 GWd/MT spent PWR fuel. Percentage Pu decreases with burnup, down to about 92% for 50 GWd/MT spent PWR fuel, so the Pu percentage in 50 GWd/MT total burnup spent fuel (35 GWd/MT in PWR followed by 15 GWd/MT in HWR) is estimated to be similar to 50 GWd/MT spent PWR fuel. This would result in an estimated at 15.3 MT/yr. If 14.06 MT of Pu are created, accounting for the other higher actinides would result in a total TRU production of about 15 MT/yr.)
Amount of TRU to Waste (MT/yr)	15
TRU Disposal per 100 Years (MT)	$15 \times 100 = 1500$
Thermal Load Reduction Factor	

Ref. A10, states that “the decay-heat characteristics of the higher burnup spent CANDU-DUPIC fuel are similar to that of spent PWR fuel ...”, and is shown in Fig. 6 of Ref. A10. Given that the spent PWR fuel in the DUPIC case in Ref. A10 is 35 GWd/MT, and the total burnup is 56 GWd/MT in Ref. A10, for the same decay heat, then on a per energy basis this would imply a thermal load reduction factor in the vicinity of 1.6 ($56/35 = 1.6$).

Figure A-2. "Material Flow of the Whole DUPIC Fuel Cycle," Figure 10 from Ref. A9.

Table A-8. "Pu Isotopic Composition in Various Spent Fuels," Table 14 from Ref. A9.

A-7. Thermal/Fast Recycle Alternative (CR=0.50): Fast Reactor Recycle Only

For this case, the source data is also from Ref. A1, with data as reproduced in Tables A-1 and A-2. The LWR data was calculated with a capacity factor of 0.9, while the fast reactor data was calculated with a capacity factor of 0.82. With 59.9 GWe of LWR capacity at 0.9 capacity factor, the LWRs make 53.9 GWe of power. The 40.1 GWe of fast reactor (FR) capacity at 0.82 capacity factor provides 32.9 GWe of power, for a total of 86.8 GWe. The following data were used, with the calculations for adjustments as indicated:

Discharge Burnup (GWd/MTIHM)	51 (LWR) 107 (FR)
LWR-UOX Power Production (GWe-yr/yr)	53.9/86.8 = 62.1 (LWR)
(These are the same reactors used in the no-action alternative, 1000 MWe each, so 62 GWe-yr/yr requires 69 reactors, rounded to ~70.)	
FR Power Production (GWe-yr/yr)	32.9/86.8 = 37.9 (FR)
(The fast reactor analysis was performed for an 840 MWth design with a thermal efficiency of 38%, for 319.2 MWe per reactor, capacity factor of 82%; 38 GWe-yr/yr would then require 145 fast reactors.)	
LEU Requirement (MT/yr)	1171/0.868 = 1349, rounded to 1350
LEU U-235 Enrichment	4.4%
Natural U Feed (MT/yr)	9664/0.868 = 11134, rounded to 11100
(Uranium tails enrichment was assumed to be 0.2% for this calculation in Ref. A1; all other cases such as HWR, HTGR, CORAIL, etc. used 0.25%. The natural uranium requirement is increased to 12200 for 0.25% uranium tails enrichment for consistency.)	
Mass of SNF to Repository (MTIHM/yr)	0
Cs/Sr to Decay Storage (MT/yr)	12
(Cs/Sr amount is given as 3.9 MT/yr in Ref. A1, which only includes Sr-90 and Cs-137. Accounting for the other isotopes to estimate the elemental masses, based on isotopic percentages for spent PWR fuel and spent fast reactor fuel, the result is about 11.7 MT/yr, rounded to 12.)	
Recovered U to Storage (MT/yr)	1080/0.868 = 1244, rounded to 1250
Spent Fast Reactor Fuel (MTIHM/yr)	290/0.868 = 334, rounded to 335 MTIHM/yr, metallic fuel
TRU Production (MT/yr)	0.09/0.868 = 0.104, rounded to 0.10
Amount of TRU to Waste (MT/yr)	0.09/0.868 = 0.104, rounded to 0.10
TRU Disposal per 100 Years (MT)	0.104 × 100 = 10.4, rounded to 10
Thermal Load Reduction Factor	235.3, rounded to 235, from Ref. A11

A-8. Thermal/Fast Recycle Alternative (CR=0.50): Thermal & Fast Recycle Option

For this case, the source data is also from Ref. A1, with data as reproduced in Tables A-1 and A-2. The case selected is for the “U-Pu Co-extraction/MCX” dual-tier data with actinide recovery from LWR spent fuel, but only one recycle of plutonium as Pu-MOX in the LWR, with the rest of the TRU going to the fast reactor for recycle. The spent Pu-MOX fuel from the LWR is also processed with all TRU going to the fast reactor. The LWR data was calculated with a capacity factor of 0.9, the Pu-MOX LWR data was calculated with a capacity factor of 0.9, and the fast reactor data was calculated with a capacity factor of 0.82. With 63.0 GWe of LWR capacity at 0.9 capacity factor, the LWRs using UOX will produce 56.7 GWe of power. The 7.35 GWe of Pu-MOX LWR capacity at 0.9 capacity factor will produce 6.615 GWe of power. The 29.65 GWe of FR capacity at 0.82 capacity factor produce 24.313 GWe of power, for a total power generation of 87.628 GWe, rounded to 87.6 GWe. The following data were used, with the calculations for adjustments as indicated:

Discharge Burnup (GWd/MTIHM)	51 (LWR) 50 (Pu-MOX LWR) 105 (FR)
LWR-UOX Power Production (GWe-yr/yr)	56.7/87.6 = 64.7 (LWR)
(These reactors are the same as those used in the no-action alternative, 1000 MWe each, so for 64.7 GWe-yr, 72 1000 MWe LWRs at 90% capacity factor are needed.)	
LWR Pu-MOX Power Production (GWe-yr/yr)	6.6/87.6 = 7.55 (Pu-MOX LWR), rounded to 7.6
(These reactors are the same as those used for UOX fuel, so for 7.6 GWe-yr/yr, 8.4 1000 MWe LWRs at 90% capacity using Pu-MOX would be needed, rounded to 8.)	
FR Power Production (GWe-yr/yr)	24.3/87.6 = 27.7 (FR)
(The fast reactors are 319.2 MWe each, so to generate 27.7 GWe-yr/yr, 106 fast reactors at 82% capacity factor are needed, rounded to ~ 105.)	
Natural U Feed (MT/yr)	10163/0.876 = 11601, rounded to 11600
(Uranium tails enrichment was assumed to be 0.2% for this calculation in Ref. A1; all other cases such as HWR, HTGR, CORAIL, etc. used 0.25%. The natural uranium requirement is increased to 12700 for 0.25% uranium tails enrichment for consistency.)	
LEU Enrichment (MT/yr)	1231/0.876 = 1405, rounded to 1400
LEU U-235 Enrichment	4.4%
Mass of SNF to Repository (MTIHM/yr)	0
Cs/Sr to Decay Storage (MT/yr)	
(Cs/Sr amount given as 4.0 MT/yr in Ref. A1, which only includes Sr-90 and Cs-137. Accounting for the other isotopes to estimate the elemental masses, based on isotopic percentages for spent PWR fuel and spent fast reactor fuel, the result is about 12 MT/yr.)	
Recycled U to Storage (MT/yr)	1080/0.876 = 1233, rounded to 1230
Spent Fast Reactor Fuel (MTIHM/yr)	220/0.876 = 251, rounded to 250 MTIHM/yr, oxide fuel
TRU Production (MT/yr)	0.10/0.876 = 0.114, rounded to 0.11
Amount of TRU to Waste (MT/yr)	0.10/0.876 = 0.114, rounded to 0.11
TRU Disposal per 100 Years (MT)	0.114 × 100 = 11.4, rounded to 11

A-9. References for Appendix A

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Appendix B

Estimates of Waste Volumes for SNF and Processing Wastes

Estimates of Waste Volumes for SNF and Processing Wastes

For all of the alternatives, including the no-action alternative and the proposed GNEP action, wastes requiring geologic disposal or other forms of isolation will be generated, including spent fuel and processing wastes. For the 100 GWe-yr per year assumed in Table 1, masses and volumes of spent fuel are calculated for those cases involving direct disposal of spent fuel, while masses of the processing wastes are calculated for those cases that involve reprocessing and recycling, where the estimates are obtained from the expected neutronic performance and the assumed efficiency of the separations plants. For those cases where reprocessing and recycling are used, an upper bound is calculated using data for the Engineering Alternatives Studies for Separations (Ref. B1), the transmutation fuel fabrication facility (Ref. B2), and the fast reactor spent nuclear fuel recycling facility (Ref. B3), where waste volumes for all of the processes have been estimated. The upper bound estimates reflect limitations associated with existing waste form technologies as well as waste classification based on the origin of the waste materials, regardless of the hazard. A lower bound is calculated (where sufficient data is available) using the masses reflecting only the content of the spent fuel being processed, not any other wastes that could be generated in support of spent fuel processing, while taking into consideration the potential beneficial impact of advanced waste forms, decay storage, and waste reclassification based on the hazard of the waste rather than its origin.

B-1. SNF Volumes

Spent PWR fuel has a volume of about 0.45 m^3 per MTIHM of spent fuel, and is used to represent the expected volume from any alternative involving direct disposal of spent LWR fuel. For the no-action alternative, with 2170 MTIHM of spent fuel being generated per year, this would result in a volume of $2170 \times 0.45 = 976.5 \text{ m}^3$, which is rounded to 975 m^3 in Table 1. Similarly, for the thorium option, since the same LWR subassembly design is used, the volume per MT should be reasonably similar, although the density of the thorium fuel may not be exactly the same as the uranium oxide fuel. The result is that the SNF disposal volume for the once-through thorium option is estimated $1025 \times 0.45 = 461.3 \text{ m}^3$, rounded to 460 m^3 .

The HWR assemblies are of a different size and shape than PWR assemblies, with typical dimensions being 10.24 cm in diameter and 49.53 cm long (volume of 4077 cm^3 , or 0.004077 m^3), with 19.2 kg of uranium. (Ref. B4) This would result in a volume of about 0.212 m^3 per MTIHM. As a result, for the HWR once-through case, 5300 MTIHM/yr would be $5300 \times 0.212 = 1123.6 \text{ m}^3/\text{yr}$, rounded to 1125 m^3 . For the spent HWR fuel from the DUPIC approach, this would be $1770 \times 0.212 = 375.2 \text{ m}^3$, rounded to 375 m^3 .

For the HTGR, the example in the PEIS uses a number of "TRISO" fuel particles formed into a short cylindrical "fuel compact". The fuel compacts are then loaded into hexagonal prismatic blocks of graphite, which are then loaded into the reactor. For the example of NGNP used in the PEIS as the HTGR, the fuel particles are loaded at 776 grams U-235 per block at 14% enrichment [B5], or 5.54 kg of uranium per block. The fuel compacts in each block have a radius of 0.6225 cm and a length of 4.93 cm, for a volume of 6.0 cm^3 . Each block contains 210 fuel rods of 15 compacts each, for a total of 3150 compacts per block. So, the 3150 compacts have a total volume of 18900 cm^3 and a fuel mass of 5.54 kg. This converts to $0.0189 \text{ m}^3/0.00554 \text{ MTIHM}$, or $3.4 \text{ m}^3/\text{MTIHM}$. For the HTGR case, then the 770 MTIHM/yr results in $2618 \text{ m}^3/\text{yr}$ of spent fuel compacts, rounded to 2600 m^3 . This value does not include the graphite block that holds the fuel compacts, as it is expected the fuel compacts will be separated from the blocks prior to disposal. However, physically separating the fuel compacts from the blocks is an untested procedure and for conservatism, an estimate is also made assuming that the compacts remain in the graphite blocks, using data from Ref. B6 as follows:

The active core has 102 fuel columns in rings 6 (30), 7 (36), and 8 (36).

Rings 1-5 are graphite only, 1 (1), 2 (6), 3 (12), 4 (18), 5 (24), totaling 61 columns

Rings 9 and 10 are graphite only, 8 (6), 9 (48), 10 (42), totaling 96 columns

Each column is 10 blocks high, each block is 79.3 cm high, 36 cm flat-to-flat.

∴ length of block side = 20.77 cm = 0.2077 m

∴ block cross-sectional area = $2.59808 \text{ side}^2 = 0.112 \text{ m}^2$

∴ block volume = $0.112 \text{ m}^2 \times 0.793 \text{ m} = 0.0888 \text{ m}^3$

∴ total block volume for fuel elements = $0.0888 \text{ m}^3 \times 102 \times 10 = 90.6 \text{ m}^3$ per reactor.

There are 5.54 kg of uranium fuel per block.

Number of blocks = $770 \text{ MTIHM/yr} \times 1000 \text{ kg per MT} / 5.54 \text{ kg per block} = 138989 \text{ blocks/yr}$.

At 0.0888 m^3 per block, this is 12342 m^3 of graphite fuel blocks per year, rounded to $12340 \text{ m}^3/\text{yr}$.

In the table, if the fuel compacts were separated from the graphite blocks after discharge from the reactor, the graphite blocks could either be part of the SNF, or they could be GTCC or LLW, depending on the level of contamination or activation. It is not known how often the non-fuel graphite blocks would need to be replaced, and they are not included in this calculation.

B-2. Waste Generation

This section summarizes the waste generation estimates for the recycle alternatives in the PEIS for GNEP. All contents of the spent fuel assembly can be represented by grouping into uranium, TRU (plutonium and the minor actinides Np, Am, and Cm), fission products, and irradiated metals from cladding and hardware. Since waste generation quantities for processing and fabrication facilities that have not been designed or built are subject to large uncertainties, upper and lower bounds have been estimated. Previous experience can be used to provide an 'upper bound', as such estimates may not account for advances in processing technology, equipment, operations, maintenance, waste form optimization, or waste minimization. A 'lower bound' is obtained by assuming that the processing of the spent fuel is essentially 'perfect', i.e., the only wastes generated are those for the waste forms containing the contents of the spent fuel, and there are no additional wastes created as a result of the operation and maintenance of the processing plants, such as job control wastes and damaged or contaminated equipment, etc. in the lower bound waste estimates. The lower bound estimates also account for the hazard of each waste at the projected time of disposal and classifies the waste on this basis, rather than using another approach such as the origin of the waste materials as been done in the past, regardless of the hazard. As a consequence, for the fuel fabrication plants, no wastes would be generated in the lower bound case, as all materials would be placed in the new reactor fuel. The degree to which a given processing or fabrication plant is closer to the upper or lower bound would depend on the extent to which waste generation can be minimized by processing technology, waste form development, plant design, operation, and maintenance.

B-2.1 Facility Waste Generation Estimates

The data for obtaining the upper and lower bound estimates for waste generation from the processing and fabrication facilities was taken from the reference documents. The first three references provide data for the upper bound, since that was the intended purpose of these documents. Reference B7 only considers the wastes originating in the spent fuel, so that data was used for the lower bound.

B-2.1.1 Upper Bound Estimates

The total amount of waste is listed in each category, and includes all of the wastes from operating the processing plant (job control waste, contaminated equipment, etc.) in addition to the spent fuel wastes. In these three reports, the spent fuel has an assumed burnup of 60 GWd/MTIHM for LWR fuel and

250 GWd/MTIHM for fast reactor fuel, so waste generation numbers have been adjusted for burnup for each alternative discussed in the PEIS, as appropriate. It is useful to note that borosilicate glass is used for all fission products when aqueous processing is used, from Figure 7 in Reference B1, reflecting the use of existing conventional technologies for developing the waste estimates, and resulting in upper bound estimates for waste volumes. Cs/Sr wastes are assumed to be part of HLW, and are included in the HLW values for the upper bound. Waste estimates are based on the operation of existing facilities, and as such would also represent upper bounds from plant operations, and reflect current regulations as to waste classification. All fission gases are also considered HLW for these upper bound estimates.

Table B-1 shows the waste generation estimates for a facility processing spent LWR fuel with 60 GWd/MTIHM burnup using aqueous separations with TRU recovery and recycling.[B1] Table B-2 shows the waste generation estimates for a facility that fabricates oxide TRU fast reactor fuel from the materials recovered in the separations facilities.[B2] Table B-3 shows the waste generation estimates for a facility using aqueous separations processing for spent fast reactor fuel, again with TRU recovery and recycling.[B3] While there is also an estimate for electrochemical processing of spent LWR fuel in Reference 1, there are no estimates for using electrochemical separations processing of spent fast reactor fuel in Reference 3, so these cases were not included in the estimates of an upper bound for waste generation. This lack of data should not be an issue at this time, since if electrochemical processing was determined to be desirable for reasons other than waste management, and if waste generation proved to be greater than the current upper bound estimate in this document, then the environmental impact would have to be evaluated at the time that justification for that choice was required.

Table B-1. Estimates of Waste Volumes from Operations for Aqueous Separations Processing of Spent LWR Fuel, 60 GWd/MITHM Burnup. (Data from Table 10 in Reference 1.)

Waste Category	800 MTHM/year Facility Annual Waste Generation
Low Level*	
Solid (m ³)	7,802
Mixed Low-level	
Solid (m ³)	32
Greater Than Class C (GTCC)	
Solid (m ³)	1,229
Mixed Solid (m ³)	77
HLW	
Solid (m ³)	377
Hazardous*	
Solid (m ³)	93
Nonhazardous	
Liquid (L)	248,000,000
Solid (m ³)	16,463
* Liquid LLW (solvents, etc.) estimated at 2156 L/yr (2.156 m ³ /yr) will be treated on site to result in solid LLW for shipment and disposal. Liquid hazardous waste estimated at 100 L/yr (0.1 m ³ /yr) will be similarly treated. No liquid LLW or hazardous liquid waste will be produced requiring long-term storage or disposal.	

Table B-2. Estimates of Waste Volumes from Operation of a Transmutation Fuel Fabrication Facility for Fast Reactor Oxide Fuel. (Data from Table 7 in Reference 2.)

Waste Category	100 MTHM/year Facility Annual Waste Generation
Low Level*	
Solid (m ³)	2,367
Mixed Low-level	
Solid (m ³)	18
Greater Than Class C (GTCC)	
Solid (m ³)	500
Hazardous*	
Solid (m ³)	14.3
Nonhazardous	
Liquid (L)	55,300,000
Solid (m ³)	19,500
* Liquid LLW (solvents, etc.) estimated at 1000 L/yr (1.0 m ³ /yr) will be treated on site to result in solid LLW for shipment and disposal. Liquid hazardous waste estimated at 33 L/yr (0.033 m ³ /yr) will be similarly treated. No liquid LLW or hazardous liquid waste will be produced requiring long-term storage or disposal.	

Table B-3. Estimates of Waste Volumes from Operations for Aqueous Separations Processing of Spent Fast Reactor Fuel, 250 GWd/MTIHM Burnup. (Data from Table 8 in Reference 3.)

Waste Category	100 MTHM/year Facility Annual Waste Generation
Low Level*	
Solid (m ³)	5,010
Mixed Low-level	
Solid (m ³)	50
Greater Than Class C (GTCC)	
Solid (m ³)	874
Mixed Solid (m ³)	10
HLW	
Solid (m ³)	267
Hazardous*	
Solid (m ³)	40
Nonhazardous	
Liquid (L)	223,000,000
Solid (m ³)	17,200
* Liquid LLW (solvents, etc.) estimated at 340 L/yr (0.34 m ³ /yr) will be treated on site to result in solid LLW for shipment and disposal. Liquid hazardous waste estimated at 88 L/yr (0.088 m ³ /yr) will be similarly treated. No liquid LLW or hazardous liquid waste will be produced requiring long-term storage or disposal.	

B-2.1.2 Lower Bound Estimates

The GNEP Integrated Waste Management Strategy (IWMS), Reference B7, has recently been completed and provides estimates of waste volumes for the separations process waste streams based solely on the content of the spent fuel, with the potential for advanced waste form development and for changes in waste classification based on the resulting waste hazard. Low and high waste volume estimates are provided for using aqueous separations to process LWR spent fuel, Table B-4, and for using electrochemical processing for fast reactor spent fuel, Table B-5, with TRU recovery and recycling in both cases. The waste volume estimates are consistent with the values used for the waste forms assumed in the analyses in References B1-B3, although some of the wastes may be in a less-hazardous waste category in the IWMS, based on potential waste reclassification justified by lower hazard. Using the lower values for waste volume estimates from Reference B7 provides a lower bound for the waste streams for these cases, although not all waste streams are represented in the IWMS since facility operations were not considered. It is also important to recognize that these estimates are only for the waste forms, and do not include any of the potential effects of packaging requirements, such as decay heat management, shielding, etc.

Table B-4. Estimates of Waste Volumes from Aqueous Separations Processing of Spent LWR Fuel, 51 GWd/MTIHM Burnup. (Data from Table 2 in Reference 4.)

Waste Stream	Waste Form	Waste Volume, Low, m ³ /MTIHM	Waste Classification
Hydrogen	Grout	7.1×10^{-5}	Decay/LLW ^a
Iodine	Encapsulated Zeolite	2.3×10^{-3}	GTCC
Krypton	Gas	2.9×10^{-3}	Decay ^a
Carbon	Grouted Ca ₂ CO ₃	1.9×10^{-2}	GTCC
Cs/Sr	Glass/Ceramic	4.0×10^{-3}	Decay/HLW/LLW ^b
Tc/UDS/FP	Metal Alloy	6.2×10^{-3}	HLW ^c /GTCC
Lanthanide FP	Glass	6.6×10^{-3}	HLW ^c /GTCC
Cladding Hulls	Metal	3.8×10^{-2}	HLW/GTCC ^d
Assembly Hardware	Metal	8.6×10^{-3}	GTCC
a Assumed to be LLW in this report for consistency with References B1-B3 b Assumed to be LLW after sufficient decay reduces radionuclide content c Assumed to be HLW in this report for consistency with References B1-B3 d Assumed to be GTCC in this report for consistency with References B1-B3			

Table B-5. Estimates of Waste Volumes from Electrochemical Separations Processing of Spent Fast Reactor Fuel, 107 GWd/MTIHM Burnup. (Data from Table 3 in Reference 4.)

Waste Stream	Waste Form	Waste Volume, m ³ /MTIHM	Waste Classification
Metals/FP	Metal Alloy	1.27×10^{-1}	HLW ^a /GTCC
Cs/Sr	Glass-bonded Mineral	5.97×10^{-1}	HLW/GTCC ^b
Lanthanide FP	Glass	1.24×10^{-2}	HLW ^a /GTCC
Stainless Steel Hardware	Metal	3.18×10^{-1}	GTCC
a Assumed to be HLW for consistency with Reference B1 b Cs/Sr waste form may contain sufficient iodine to prevent decay to LLW. Assumed further development removes this limitation and will be able to decay to LLW			

Examination of the data in Tables B-4 and B-5 shows that electrochemical processing of spent fast reactor fuel is expected to result in about 10 times the volume of HLW as compared to aqueous processing of spent LWR fuel. Even accounting for the difference in burnup, it appears that at this time, electrochemical processing is expected to create more HLW than aqueous processing. The Cs/Sr waste stream resulting from electrochemical processing would have 150 times the volume of aqueous processing due to the inclusion of other process chemicals in the waste form and the relatively low loading density of the waste form. The difference for the hardware waste stream is due to the design of the fast reactor assemblies, and is not due to the processing technology. Since either processing technology can be used, in order to generate lower bounds for all recycle alternatives on a consistent basis, lower bound estimates were obtained for the processing of spent fast reactor fuel by using aqueous separations based on the relative burnup, as shown in Table B-5a. All of the fission product masses were multiplied by $107/51 = 2.10$ to reflect the higher burnup. The cladding hulls and hardware were retained without change, as their masses should be proportional to the metric tons of fuel processed, not the burnup.

Table B-5a. Estimates of Waste Volumes from Aqueous Separations Processing of Spent Fast Reactor Fuel, 107 GWd/MTIHM Burnup. (Data derived from Tables 4 and 5.)

Waste Stream	Waste Form	Waste Volume, Low, m ³ /MTIHM	Waste Classification
Hydrogen	Grout	1.5×10^{-4}	Decay/LLW ^a
Iodine	Encapsulated Zeolite	4.8×10^{-3}	GTCC
Krypton	Gas	6.1×10^{-3}	Decay ^a
Carbon	Grouted Ca ₂ CO ₃	4.0×10^{-2}	GTCC
Cs/Sr	Glass/Ceramic	8.4×10^{-3}	Decay/HLW/LLW ^b
Tc/UDS/FP	Metal Alloy	1.3×10^{-2}	HLW ^c /GTCC
Lanthanide FP	Glass	1.4×10^{-2}	HLW ^c /GTCC
Cladding Hulls	Metal	3.18×10^{-1}	HLW/GTCC ^d
Assembly Hardware	Metal		GTCC

a Assumed to be LLW in this report for consistency with References B1-B3
b Assumed to be LLW after sufficient decay reduces radionuclide content
c Assumed to be HLW in this report for consistency with References B1-B3
d Assumed to be GTCC in this report for consistency with References B1-B3

B-3. Waste Generation for the GNEP PEIS Recycle Alternatives

In this section, the upper and lower bound waste estimates for each alternative are discussed in detail. All waste estimates are performed for 100 GWe-yr per year of electricity generation.

B-3.1 Thermal Reactor Recycle Alternative, Option 1 – LWR/LWR (MOX-U-Pu)

The characteristics of this alternative, and the spent fuel and waste streams are as follows:

1. Some fuel is made from recycled plutonium (PuMOX, about 1/3) while the remainder (UOX, about 2/3) is made from enriched uranium, all for use in LWRs.
2. All spent fuel is chemically processed, with all fission products, minor actinides (Np, Am, Cm), and irradiated metals going to the waste stream for disposal as HLW, GTCC or LLW.
3. Plutonium/uranium is recovered and fabricated into new LWR PuMOX fuel for recycle.
4. The remainder of the recovered uranium is stored for either future use or eventual disposal as LLW.
5. Only processing and possibly some fabrication wastes go to geologic disposal.

The wastes that need to be evaluated for this alternative are:

- Fission products – to HLW
- Minor actinides (Np, Am, Cm) – to HLW
- Irradiated Metals – to GTCC
- Uranium – to storage for reuse or potential disposal as LLW.

It should be noted that the plutonium loss to the waste stream is targeted at 0.1%, which is a negligible mass in comparison to the other components of the waste streams and is not included in these estimates.

B-3.1.1 Upper Bound Estimates

Using Table B-1, it is possible to develop an upper bound estimate for the wastes associated with this alternative including all operations wastes using aqueous processing. The data will need to be modified due to the lower burnup of the spent fuel for this alternative, 45 GWd/MTIHM vs. the 60 GWd/MTIHM assumed in Reference B1. The spent fuel produced per year is about 2460 MTIHM/year for 100 GWe-yr per year of electricity (33% LWR thermal efficiency).

As shown in Table B-4, with the TRU recovered and recycled, the HLW is dominated by fission products, which are essentially a linear function of burnup. Since the values in Table B-1 are for 60 GWd/MTIHM burnup, processing fuel with 45 GWd/MTIHM burnup would produce approximately $(45/60) = 75\%$ of the fission products, and 75% of the HLW amount listed in Table B-1 per MTIHM.

The thermal reactor recycle alternative only recycles plutonium, so including the minor actinides in the waste stream with the fission products will increase the mass of the HLW waste stream by about 10%. However, since all of the fission products are placed in a glass waste form for the upper bound calculations, the glass volume is dominated by the effect of the low solubility of the transition metal fission products.[B7, B8] In this case, the increase caused by the addition of the minor actinides to the glass is expected to be a few percent at most, and most likely no change in volume at all. As a result, the upper bound HLW estimate was not changed further to account for the addition of the minor actinides. The other waste classes are essentially determined by the metric tons being processed, i.e., fuel assembly hardware wastes would be proportional to the number of tons processed, etc., and are part of the GTCC waste. The LLW is from operations and would also be proportional to the tons processed, and does not include any of the recovered uranium. As a result, for all waste categories except HLW, the numbers from Table B-1 are used unaltered, as listed in Table B-6. Uranium recovered from this process is estimated to be 2250 MTHM/yr, with a volume of about 1020 m³/yr requiring storage, assuming no recovered uranium is reused in fabrication of the Pu-MOX fuel.

There will be also an increase in the wastes from LWR fuel fabrication for this alternative due to the larger amount required per year, 2460 MTHM/yr as compared to 2170 MTHM/yr for the no-action alternative, although it is complicated by fabricating less enriched uranium fuel but adding the manufacture of Pu-MOX fuel. The amount of enriched uranium fuel fabrication decreases from 2170 to 1660 MTHM/yr, but no credit is taken for the resulting decrease in waste generation. There is fabrication of about 800 MTHM/yr of Pu-MOX fuel.

Table B-6. Upper Bound Estimate for Annual Waste Generation from Aqueous Separations Processing of Spent LWR Fuel, 45 GWd/MTIHM burnup.

Waste Category	800 MTHM/year Facility, 60 GWd/MTIHM	800 MTHM/year Facility, 45 GWd/MTIHM	100 GWe-yr/yr
			2460 MTHM/year, 45 GWd/MTIHM
Low Level^a			
Solid (m ³ /yr)	7,802	7,802	24,000
Mixed Low-level			
Solid (m ³ /yr)	32	32	100
Greater Than Class C (GTCC)			
Solid (m ³ /yr)	1,229	1,229	3,775
Mixed Solid (m ³ /yr)	77	77	240
HLW			
Solid (m ³ /yr)	377	283	870 ^b
Hazardous^a			
Solid (m ³ /yr)	93	93	285
Nonhazardous			
Liquid (L/yr)	248,000,000	248,000,000	760,000,000

a. Liquid LLW (solvents, etc.) will be treated on site to result in solid LLW for shipment and disposal. Liquid hazardous waste will also be solidified. No liquid LLW or hazardous liquid waste will be produced requiring long-term storage or disposal.

b. The HLW estimate is 510 m³ for most fission products in glass and Tc/UDS in a metal alloy with TRU losses. Addition of minor actinides to the waste stream produces a negligible increase in glass volume due to the volume being dominated by low solubility transition metal fission products.

Some insight on waste generation from Pu-MOX fuel fabrication can be gained from the Pu-MOX facility currently under construction at the Savannah River Site.[B9] With the use of weapons plutonium, the amount of GTCC (TRU) waste is projected to be about 234 m³/yr (Table 4.11, Reference B9) while producing between 70 (4.8% Pu) and 150 (2.3% Pu) MTHM of Pu-MOX per year. Using the higher plutonium concentration, which is close to that required for this alternative, this would imply about $(800/70) \times 234 = 2675$ m³/yr of GTCC from Pu-MOX fuel fabrication. It is not known if this should be considered conservative or not at this time. Similarly for the LLW, about 176 m³/yr is expected from the MOX facility, which would be about 2000 m³/yr for this alternative. These estimates are included in the waste estimates for the upper bound in Table B-20. For the lower bound waste estimates, since all maintenance wastes, job control wastes, etc. are ignored, the lower bound for waste generation would be zero, and this is reflected in the lower bound values in Table B-20.

B-3.1.2 Lower Bound Estimates

A lower bound for wastes from reprocessing spent LWR fuel can be estimated using Table B-4, as if there were no additional wastes produced as a result of operating the processing facility, i.e., processing is 'perfect', with wastes representing only what is required to put the materials from the spent fuel into a waste form, and there is no job control waste, contaminated equipment to be disposed, additional packaging, etc. The lower bound also recognizes the potential reclassification of wastes that would reflect the hazard of the waste rather than the origin of the materials in the waste. The adjustment for approximating the HLW volume is $45/51 = 0.88$, since Table B-4 is based on processing spent LWR fuel with 51 GWd/MTIHM burnup. Not all categories of waste generation are created for the lower bound, since many of the waste streams only arise as a result of actual processing operations in the facility, such as maintenance and repair.

Table B-7. Lower Bound Estimate for Annual Waste Generation from Aqueous Separations Processing of Spent LWR Fuel, 45 GWd/MTIHM burnup.

Waste Category	1 MTIHM	100 GWe-yr/yr
		2460 MTIHM/year
Low Level		
Solid (m ³ /yr)	7.0×10^{-3}	17
Greater Than Class C (GTCC)		
Solid (m ³ /yr)	6.8×10^{-2}	170
HLW		
Solid (m ³ /yr)	1.32×10^{-2} ^a	32 ^a

a. The estimate for the HLW fission products, partly in glass (6.6×10^{-3} m³) and partly in metal (6.2×10^{-3} m³) = 1.28×10^{-2} m³, is 1.13×10^{-2} m³ after multiplication by the factor of 0.88 to adjust for burnup. Addition of the minor actinides to the glass containing the lanthanide fission products is estimated to increase the glass volume in this case by about 33%, from 6.6×10^{-3} m³ up to 8.8×10^{-3} m³, which makes the total 1.32×10^{-2} m³ after adjusting for burnup.

The adjustment to the HLW amount in this case to take into account the minor actinides in the waste stream is much larger, since only the lanthanides are placed in the glass waste form, and the glass volume is much smaller. Overall, the lower HLW volume in Table B-7 reflects the changes from only using a glass waste form in Table B-6 to using a combination of metal and glass waste forms, as discussed in Reference B7, as well as the omission of all operations-related wastes. The much lower value for GTCC in Table B-7 as compared to Table B-6 also reflects the impact of GTCC wastes arising from operations in the plant due to contamination by radioactive materials from the spent fuel. The same is true for the LLW estimate, where relatively little LLW would be created directly from the spent fuel. All of the Cs/Sr waste is assumed to be decayed to LLW and is included in the LLW total.

B-3.2 Thermal Reactor Recycle Alternative, Option 2 – LWR/HWR (DUPIC)

The characteristics of this alternative, and the spent fuel and waste streams are as follows:

1. Standard LWR enriched uranium fuel (UOX) is used in LWRs as is done today
2. All spent LWR fuel is processed with a thermal/mechanical method
3. Volatile fission products and irradiated metals are recovered during processing and are sent to the waste stream for disposal as HLW, GTCC or LLW
4. No other separations occur, with the product of processing (all uranium, TRU, non-volatile fission products) being fabricated into new fuel for an HWR.
5. After once-through use in the HWR, the spent HWR fuel is sent to disposal
6. Spent HWR fuel, and possibly some processing and fabrication wastes, go to geologic disposal

The wastes that need to be evaluated for this alternative are:

- Volatile fission products – to HLW
- Irradiated Metals – to GTCC

These wastes are in addition to the spent HWR fuel.

Information on this alternative is not as detailed as it is for the other alternatives. Reference B10 gives an HLW volume of 0.35 m^3 per terawatt-hr, although 80% of this is Cs and Ru. Since the Cs would not be available separately in this case, it is included in the HLW. For 1 GWe-yr, there are $365.25 \times 24 = 8766$ hours per year, or 8766 GWe-hrs per year per GWe, or 8.766 terawatt-hours per GWe per year. For 100 GWe-yr, there would be 876.6 TWhe. The corresponding HLW volume is $0.35 \times 876.6 = 307 \text{ m}^3$, rounded to 300 m^3 . There is also a value for ILW of 0.55 m^3 per terawatt-hr, which may include some material that would be classified as HLW in the United States, and would result in another 482.13 m^3 , so the total is conservatively estimated at 789 m^3 , rounded to a range of $300\text{-}800 \text{ m}^3/\text{yr}$ for 100 GWe-yr/yr. This is in addition to the mass and volume of spent HWR fuel which is estimated at about 1800 MTIHM/yr, or $360 \text{ m}^3/\text{yr}$.

The processing of the spent PWR fuel will also generate cladding and hardware waste that is sent to disposal, and this material may be classified as GTCC waste. Approximately 1800 MTIHM/yr of spent PWR fuel would be processed. This would result in 500 MT/yr of cladding and hardware waste. The volume would be estimated at $122 \text{ m}^3/\text{yr}$, rounded to $120 \text{ m}^3/\text{yr}$, and would be considered a lower bound estimate as it does not include any other wastes from processing operations.

The LLW may also be estimated based on a value of 5.91 m^3 per terawatt-hr. [B10] This would result in LLW waste generation of $876.6 \times 5.91 = 5180 \text{ m}^3/\text{yr}$. It is not known whether this represents an upper or lower bound, or represents an expectation based on recent results. For the purposes of this report, it is noted but not included in Table 20.

As in the first alternative, no estimate is given for any reduction in waste that may arise from fabrication of the initial LWR fuel, although less is needed in this alternative compared to the no-action case, 1800 MTHM/yr as compared to 2170 MTHM/yr.

B-3.3 Thermal/Fast Recycle Alternative (CR=0.50): Fast Reactor Recycle Only

The characteristics of this alternative, and the spent fuel and waste streams are as follows:

1. Standard LWR enriched uranium fuel (UOX) is used in LWRs as is done today
2. All spent LWR fuel is chemically processed, with all fission products and irradiated metals going to the waste stream for disposal as HLW, GTCC, or LLW
3. The TRU/uranium from spent LWR fuel is recovered and fabricated into new fast reactor fuel for recycle
4. The remainder of the recovered uranium from spent LWR fuel is stored either for future use or eventual disposal as LLW
5. The TRU is repeatedly recycled in the fast reactor
6. All spent fast reactor fuel is chemically processed to recover the TRU/uranium for recycle, while the fission products and irradiated metals go to the waste stream for disposal as HLW, GTCC, or LLW
7. The TRU/uranium from spent fast reactor fuel is recovered and fabricated into new fast reactor fuel for recycle
8. The remainder of the recovered uranium from spent fast reactor fuel is stored either for future use or for eventual disposal as LLW
9. Only processing and possibly some fabrication wastes go to geologic disposal

The wastes that need to be evaluated are:

- Fission products (from spent LWR fuel and spent fast reactor fuel) – to HLW
- Irradiated Metals – to GTCC

- Uranium – to storage for reuse or potential disposal as LLW.

Following the same logic as for the first alternative, using Tables B-1, B-2, and B-3 will allow development of an upper bound estimate for the wastes associated with this alternative. Power production is 62.1% from the LWRs and 37.9% from the fast reactors.

B-3.3.1 Upper Bound Estimates

The spent LWR fuel produced per year is 1350 MTIHM/year (33% LWR thermal efficiency). Since all TRU is recovered and recycled in this case, the HLW is dominated by fission products, which are a function of burnup. Since the values in Table B-1 are for 60 GWd/MTHM burnup, processing fuel with 51 GWd/MTHM burnup would produce approximately $(51/60) = 85\%$ of the HLW amount listed in Table B-1. However, the other waste classes are likely to be determined by the metric tons being processed, i.e., fuel assembly hardware, wastes proportional to the number of tons processed, etc. In those cases, the numbers from Table B-1 would be used unaltered, as shown in Table B-8. The LLW values do not include any of the recovered uranium. The uranium recovered from processing the spent LWR fuel would be about 1250 MTHM/yr, with a volume of 570 m³ requiring storage, assuming that it is not used as part of fast reactor fuel.

Table B-8. Upper Bound Estimate for Annual Waste Generation from Aqueous Separations Processing of Spent LWR Fuel, 51 GWd/MTIHM burnup.

Waste Category	800 MTIHM/year Facility, 60 GWd/MTIHM	800 MTIHM/year Facility, 51 GWd/MTIHM	100 GWe-yr/yr 1350 MTIHM/year, 51 GWd/MTIHM
	Low Level^a		
Solid (m ³ /yr)	7,802	7,802	13,170
Mixed Low-level			
Solid (m ³ /yr)	32	32	54
Greater Than Class C (GTCC)			
Solid (m ³ /yr)	1,229	1,229	2,075
Mixed Solid (m ³ /yr)	77	77	130
HLW			
Solid (m ³ /yr)	377	320	540
Hazardous^a			
Solid (m ³ /yr)	93	93	155
Nonhazardous			
Liquid (L/yr)	248,000,000	248,000,000	418,500,000
Solid (m ³ /yr)	16,463	16,463	27,800

a. Liquid LLW (solvents, etc.) will be treated on site to result in solid LLW for shipment and disposal. Liquid hazardous waste will also be solidified. No liquid LLW or hazardous liquid waste will be produced requiring long-term storage or disposal.

New fast reactor fuel is fabricated and all spent fast reactor fuel is processed, which for this case is about 335 MTIHM/yr (38% fast reactor thermal efficiency). Using Table B-2 for the fuel fabrication facility (even though this case uses metallic fuel for the fast reactor, there is no data for that process; it is assumed that the waste production will be similar to that for fabricating oxide fuel, or one could view this alternative as being done with oxide fuel for the purposes of estimating waste generation), the result is as shown in Table B-9.

Table B-9. Upper Bound Estimates for Annual Waste Generation from Fabrication of Fast Reactor Oxide Transmutation Fuel.

Waste Category	100 MTHM/year Facility	100 GWe-yr/yr
		335 MTHM per year
Low Level^a		
Solid (m ³ /yr)	2,367	7,929
Mixed Low-level		
Solid (m ³ /yr)	18	60
Greater Than Class C (GTCC)		
Solid (m ³ /yr)	500	1675
Hazardous^a		
Solid (m ³ /yr)	14.3	47.9
Nonhazardous		
Liquid (L/yr)	55,300,000	185,260,000
Solid (m ³ /yr)	19,500	65,300

a. Liquid LLW (solvents, etc.) will be treated on site to result in solid LLW for shipment and disposal. Liquid hazardous waste will also be solidified. No liquid LLW or hazardous liquid waste will be produced requiring long-term storage or disposal.

Similarly, for processing the spent fast reactor fuel, the net result can be calculated using Table B-3, which is for a spent fast reactor fuel processing facility. The HLW is adjusted to account for the discharge burnup of 107 GWd/MTIHM instead of 250 GWd/MTIHM, a factor of about 0.43, as shown in Table B-10. The recovered uranium is used in the fabrication of new fast reactor fuel, so no additional uranium goes to storage or waste from this process.

Table B-10. Upper Bound Estimates of Annual Waste Generation from Aqueous Separations Processing of Spent Fast Reactor Fuel, 107 GWd/MTIHM burnup.

Waste Category	100 MTIHM/year Facility, 250 GWd/MTIHM	100 MTIHM/year Facility, 107 GWd/MTIHM	100 GWe-yr/yr
			335 MTIHM/year, 107 GWd/MTIHM
Low Level^a			
Solid (m ³ /yr)	5,010	5,010	16,780
Mixed Low-level			
Solid (m ³ /yr)	50	50	168
Greater Than Class C (GTCC)			
Solid (m ³ /yr)	874	874	2930
Mixed Solid (m ³ /yr)	10	10	34
HLW			
Solid (m ³ /yr)	267	114	380
Hazardous^a			
Solid (m ³ /yr)	40	40	134
Nonhazardous			
Liquid (L/yr)	223,000,000	223,000,000	747,000,000
Solid (m ³ /yr)	17,200	17,200	57,600

a. Liquid LLW (solvents, etc.) will be treated on site to result in solid LLW for shipment and disposal. Liquid hazardous waste will also be solidified. No liquid LLW or hazardous liquid waste will be produced requiring long-term storage or disposal.

The contribution from each type of facility is added (Tables B-8, B-9, and B-10) for each waste class to obtain the total for this alternative as shown in Table B-20. The reduction in any waste from LWR fuel fabrication for this alternative due to the lower amount required per year, 1350 MTIHM/yr as compared to 2170 MTIHM/yr for the no-action alternative, is not reflected in these estimates.

B-3.3.2 Lower Bound Estimates

The lower bound estimates are obtained using the same logic as for the first alternative. Starting with the processing of spent LWR fuel, the waste volume estimates from Table B-4 are used. No adjustment is required for burnup in this case, since the spent PWR fuel for this alternative also has a burnup of 51 GWd/MTIHM. Cs/Sr is included in the LLW total, as it is assumed that the Cs/Sr waste has been allowed to decay sufficiently such that the remaining hazard would be consistent with waste classification for LLW. The results are shown in Table B-11.

Table B-11. Lower Bound Estimate for Annual Waste Generation from Aqueous Separations Processing of Spent LWR Fuel, 51 GWd/MTIHM burnup.

Waste Category	1 MTIHM	100 GWe-yr/yr
		1350 MTIHM/year
Low Level		
Solid (m ³ /yr)	7.0×10^{-3}	9
Greater Than Class C (GTCC)		
Solid (m ³ /yr)	6.8×10^{-2}	90
HLW		
Solid (m ³ /yr)	1.28×10^{-2}	17

The lower bound waste generation estimate for the transmutation fuel processing facility is zero, as it was for the Pu-MOX fuel fabrication facility in the first alternative, since all wastes from that process are the result of losses during operations, job control wastes, maintenance, etc.

Obtaining an estimate for the lower bound for the processing of the spent fast reactor fuel is more complicated. The IWMS only has data for electrochemical processing and the resulting waste forms. As noted above, estimates for this processing technology are currently higher than those for aqueous processing, and this needs to be considered when comparing the waste generation from the recycle alternatives since either processing technology could be used. In this case, a range is calculated, using data from Tables B-5 and B-5a, as shown in Table B-12.

For the LLW and HLW, the lower value in the range is associated with aqueous processing and the higher value is for electrochemical processing, while for GTCC, the lower value is for electrochemical processing. As before, the Cs/Sr waste is assumed to be decayed such that it could be reclassified as LLW before disposal based on the remaining hazard. The total for the lower bound is obtained by adding the contributions from both Table B-11 and Table B-12, as shown in Table B-20.

Table B-12. Lower Bound Estimates of Annual Waste Generation from Aqueous and Electrochemical Separations Processing of Spent Fast Reactor Fuel, 107 GWd/MTIHM burnup.

Waste Category	1 MTIHM	100 GWe-yr/yr
		335 MTIHM/year
Low Level		
Solid (m ³ /yr)	$(1.5 - 5.97) \times 10^{-1}$	50-200
Greater Than Class C (GTCC)		
Solid (m ³ /yr)	$(3.18 - 3.63) \times 10^{-1}$	105-120
HLW		
Solid (m ³ /yr)	$(0.27 - 1.39) \times 10^{-1}$	9-45

B-3.4 Thermal/Fast Recycle Alternative (CR=0.50): Thermal & Fast Reactor Recycle Option

The characteristics of this alternative, and the spent fuel and waste streams are as follows:

1. Standard LWR enriched uranium fuel (UOX) is used in LWRs as is done today
2. All spent LWR fuel is chemically processed, with all fission products and irradiated metals going to the waste stream for disposal as HLW, GTCC, or LLW
3. The TRU is recovered as plutonium/uranium for use in LWRs and as minor actinides/uranium for recycle in fast reactors
4. The remainder of the recovered uranium from spent LWR fuel is stored either for future use or eventual disposal as LLW
5. The plutonium/uranium from spent LWR fuel is fabricated into new LWR PuMOX fuel for once-through use in LWRs
6. The spent LWR PuMOX fuel is chemically processed, with all fission products and irradiated metals going to the waste stream for disposal as HLW, GTCC, or LLW
7. The TRU/uranium from the spent LWR PuMOX fuel is recovered for recycle in fast reactors
8. The remainder of the recovered uranium from spent LWR PuMOX fuel is stored either for future use or eventual disposal as LLW
9. The minor actinides/uranium (from spent LWR fuel) and the TRU/uranium (from spent LWR PuMOX fuel) are fabricated into new fuel for the fast reactor
10. The TRU is repeatedly recycled in the fast reactor
11. All spent fast reactor fuel is chemically processed to recover the TRU/uranium for recycle, while the fission products and irradiated metals go to the waste stream for disposal as HLW, GTCC, or LLW
12. The TRU/uranium from spent fast reactor fuel is recovered and fabricated into new fast reactor fuel for recycle
13. The remainder of the recovered uranium from spent fast reactor fuel is stored either for future use or for eventual disposal as LLW
14. Only processing and possibly some fabrication wastes go to geologic disposal.

The wastes that need to be evaluated are:

- Fission products (from spent LWR fuel, spent LWR PuMOX, and spent fast reactor fuel) – to HLW
- Irradiated Metals – to GTCC
- Uranium – to storage for reuse or potential disposal as LLW.

Following the same logic as for the first alternative, using Tables B-1, B-2, and B-3 will allow development of an upper bound estimate for the wastes associated with this alternative. Power production is 64.7% LWR, 7.6% LWR PuMOX, and 27.7% fast reactor.

B-3.4.1 Upper Bound Estimates

The LWR spent fuel produced per year is 1400 MTIHM/year (33% LWR thermal efficiency). Since all TRU is recovered from the spent LWR fuel and recycled in this case, the HLW is dominated by fission products, which are a function of burnup. However, the TRU is partitioned into plutonium for one recycle in LWRs as Pu-MOX, and the minor actinides are set aside for recycle in the fast reactors. Since the values in Table B-1 are for 60 GWd/MTIHM burnup, processing fuel with 51 GWd/MTIHM burnup would produce approximately $(51/60) = 85\%$ of the HLW amount listed in Table B-1. However, the other waste classes are likely to be determined by the metric tons being processed, i.e., fuel assembly hardware, wastes proportional to the number of tons processed, etc. In those cases, the numbers from Table B-1 would be used unaltered, as shown in Table B-13. The LLW values do not include any of the recovered uranium. The uranium recovered from processing the spent LWR fuel would be about 1310 MTHM/yr, with a volume of 590 m³ requiring storage, assuming that it is not used as part of fast reactor fuel.

The recovered plutonium is fabricated into Pu-MOX. As with the first alternative, an estimate of the wastes from the fabrication step can be obtained from the data provided to support the construction of the new MOX plant at Savannah River Site.[B9] For this alternative, about 165 MTHM/yr of Pu-MOX fuel needs to be fabricated (33% LWR thermal efficiency). About 234 m³ of GTCC is made for 70 MTHM of Pu-MOX, so this alternative would produce 560 m³/yr. Similarly, for the LLW, approximately 420 m³/yr would be generated.

Table B-13. Upper Bound Estimate for Annual Waste Generation from Aqueous Separations Processing of Spent LWR Fuel, 51 GWd/MTIHM burnup.

Waste Category	800 MTIHM/year Facility, 60 GWd/MTIHM	800 MTIHM/year Facility, 51 GWd/MTIHM	100 GWe-yr/yr
			1400 MTIHM/year, 51 GWd/MTIHM
Low Level^a			
Solid (m ³ /yr)	7,802	7,802	13,650
Mixed Low-level			
Solid (m ³ /yr)	32	32	56
Greater Than Class C (GTCC)			
Solid (m ³ /yr)	1,229	1,229	2,150
Mixed Solid (m ³ /yr)	77	77	135
HLW			
Solid (m ³ /yr)	377	320	560
Hazardous^a			
Solid (m ³ /yr)	93	93	165
Nonhazardous			
Liquid (L/yr)	248,000,000	248,000,000	434,000,000
Solid (m ³ /yr)	16,463	16,463	28,800

a. Liquid LLW (solvents, etc.) will be treated on site to result in solid LLW for shipment and disposal. Liquid hazardous waste will also be solidified. No liquid LLW or hazardous liquid waste will be produced requiring long-term storage or disposal.

The spent LWR Pu-MOX fuel must also be processed, with the HLW adjusted for the burnup of 50 GWd/MTIHM for the data from Table B-1. All of the TRU is recovered for use in the fast reactor, so the data from Table B-1 should provide a reasonable approximation for the waste generation, even though LWR Pu-MOX is being processed, since the spent fuel composition relevant to waste production is similar to spent LWR fuel, especially for the fission products. As a result, waste production is assumed to be the same whether LWR Pu-MOX processing occurs in the same facility as processing of spent LWR fuel or in a separate facility, as shown in Table B-14. The uranium recovered from processing the spent LWR Pu-MOX fuel would be about 150 MTHM/yr, with a volume of 70 m³ requiring storage, assuming that it is not used as part of fast reactor fuel.

Table B-14. Upper Bound Estimate for Annual Waste Generation from Aqueous Separations Processing of Spent LWR Pu-MOX Fuel, 50 GWd/MTIHM burnup

Waste Category	800 MTIHM/year Facility, 60 GWd/MTIHM	800 MTIHM/year Facility, 50 GWd/MTIHM	100 GWe-yr/yr 165 MTIHM/year, 50 GWd/MTIHM
	Low Level^a		
Solid (m ³ /yr)	7,802	7,802	1,610
Mixed Low-level			
Solid (m ³ /yr)	32	32	7
Greater Than Class C (GTCC)			
Solid (m ³ /yr)	1,229	1,229	253
Mixed Solid (m ³ /yr)	77	77	16
HLW			
Solid (m ³ /yr)	377	314	65
Hazardous^a			
Solid (m ³ /yr)	93	93	19
Nonhazardous			
Liquid (L/yr)	248,000,000	248,000,000	51,000,000
Solid (m ³ /yr)	16,463	16,463	3,395

a. Liquid LLW (solvents, etc.) will be treated on site to result in solid LLW for shipment and disposal. Liquid hazardous waste will also be solidified. No liquid LLW or hazardous liquid waste will be produced requiring long-term storage or disposal.

New fast reactor fuel is fabricated and the spent fast reactor fuel is processed, which for this case is about 250 MTIHM/yr (38% fast reactor thermal efficiency). Using Table B-2 for the fuel fabrication facility upper bound waste estimate, the result is as shown in Table B-15.

Similarly, for processing the spent fast reactor fuel, the upper bound can be calculated using Table B-3, which is for a spent fast reactor fuel processing facility. The HLW is adjusted to account for the discharge burnup of 105 GWd/MTIHM instead of 250 GWd/MTIHM, while the other waste volumes are proportional to metric tons processed. The result is shown in Table B-16.

Table B-15. Upper Bound Estimates of Annual Waste Generation from Fabrication of Fast Reactor Oxide Transmutation Fuel.

Waste Category	100 MTHM/year Facility	100 GWe-yr/yr
		250 MTHM per year
Low Level^a		
Solid (m ³ /yr)	2,367	5,920
Mixed Low-level		
Solid (m ³ /yr)	18	45
Greater Than Class C (GTCC)		
Solid (m ³ /yr)	500	1250
Hazardous^a		
Solid (m ³ /yr)	14.3	36
Nonhazardous		
Liquid (L/yr)	55,300,000	138,250,000
Solid (m ³ /yr)	19,500	48,750

a. Liquid LLW (solvents, etc.) will be treated on site to result in solid LLW for shipment and disposal. Liquid hazardous waste will also be solidified. No liquid LLW or hazardous liquid waste will be produced requiring long-term storage or disposal.

Table B-16. Upper Bound Estimates of Annual Waste Generation from Aqueous Separations Processing of Spent Fast Reactor Fuel, 105 GWd/MTIHM burnup.

Waste Category	100 MTIHM/year Facility, 250 GWd/MTIHM	100 MTIHM/year Facility, 105 GWd/MTIHM	100 GWe-yr/yr
			250 MTIHM/year, 105 GWd/MTIHM
Low Level^a			
Solid (m ³ /yr)	5,010	5,010	12,530
Mixed Low-level			
Solid (m ³ /yr)	50	50	125
Greater Than Class C (GTCC)			
Solid (m ³ /yr)	874	874	2185
Mixed Solid(m ³ /yr)	10	10	25
HLW			
Solid (m ³ /yr)	267	112	280
Hazardous^a			
Solid (m ³ /yr)	40	40	100
Nonhazardous			
Liquid (L/yr)	223,000,000	223,000,000	557,500,000
Solid (m ³ /yr)	17,200	17,200	43,000

a. Liquid LLW (solvents, etc.) will be treated on site to result in solid LLW for shipment and disposal. Liquid hazardous waste will also be solidified. No liquid LLW or hazardous liquid waste will be produced requiring long-term storage or disposal.

The contribution from each type of facility is added (Tables B-13, B-14, B-15, and B-16, and the Pu-MOX fuel fabrication estimate) for each waste class to obtain the total upper bound for this alternative. The reduction in any waste from LWR fuel fabrication for this alternative due to the lower amount required per year, 1400 MTIHM/yr for LWR fuel, as compared to 2170 MTIHM/yr for the no-action alternative, is not reflected in these waste estimates.

B-3.4.2 Lower Bound Estimates

The lower bound estimates are obtained using the same logic as for the first alternative. Starting with processing of spent LWR fuel, the waste volume estimates from Table B-4 are used. No adjustment is required for burnup in this case, since the spent PWR fuel for this alternative also has a burnup of 51 GWd/MTIHM. The results are shown in Table B-17.

Table B-17. Lower Bound Estimate for Annual Waste Generation from Aqueous Separations Processing of Spent LWR Fuel, 51 GWd/MTIHM burnup.

Waste Category	1 MTIHM	100 GWe-yr/yr
		1400 MTIHM/year
Low Level		
Solid (m ³ /yr)	7.0×10^{-3}	10
Greater Than Class C (GTCC)		
Solid (m ³ /yr)	6.8×10^{-2}	95
HLW		
Solid (m ³ /yr)	1.28×10^{-2}	18

As discussed previously, the lower bound waste generation estimate for both the LWR Pu-MOX fuel fabrication facility and the transmutation fuel processing facility is zero, since all wastes from that process are the result of losses during operations, job control losses, maintenance, etc. The lower bound estimated for processing the spent LWR Pu-MOX fuel is estimated in the same manner as for the spent LWR fuel. Cs/Sr waste is included in the LLW total, since it is assumed that the Cs/Sr waste has been allowed to decay. The result is shown in Table B-18.

Table B-18. Lower Bound Estimate for Annual Waste Generation from Aqueous Separations Processing of Spent LWR Pu-MOX Fuel, 50 GWd/MTIHM burnup.

Waste Category	1 MTIHM	100 GWe-yr/yr
		165 MTIHM/year
Low Level		
Solid (m ³ /yr)	7.0×10^{-3}	1
Greater Than Class C (GTCC)		
Solid (m ³ /yr)	6.8×10^{-2}	11
HLW		
Solid (m ³ /yr)	1.28×10^{-2}	2

Following the logic described above for the third alternative, both aqueous and electrochemical processing needs to be considered for the processing of the spent fast reactor fuel. As noted above, waste estimates for electrochemical processing technology are currently higher than those for aqueous processing, and this needs to be considered when comparing the waste generation from the recycle alternatives since either processing technology could be used. In this case, a range is calculated, using data from Tables B-5 and B-5a, as shown in Table B-19.

Table B-19. Lower Bound Estimates of Annual Waste Generation from Aqueous and Electrochemical Separations Processing of Spent Fast Reactor Fuel, 105 GWd/MTIHM burnup.

Waste Category	1 MTIHM	100 GWe-yr/yr
		250 MTIHM/year
Low Level		
Solid (m ³ /yr)	$(1.5 - 5.97) \times 10^{-1}$	38-150
Greater Than Class C (GTCC)		
Solid (m ³ /yr)	$(3.18 - 3.63) \times 10^{-1}$	80-90
HLW		
Solid (m ³ /yr)	$(0.27 - 1.39) \times 10^{-1}$	7-35

The lower value in the range for LLW and HLW are for aqueous processing, while the lower value of GTCC is for electrochemical processing. The total for the lower bound is obtained by adding the contributions from Tables B-17, B-18 and B-19, as shown in Table B-20.

B-4. Summary

For each of the recycle alternatives in the GNEP PEIS, upper and lower bounds for waste production have been generated. As can be seen in Table B-20, there is a large difference between the lower and upper bounds. For LLW and GTCC, this difference is mainly due to the estimated wastes from operations of processing and fabrication facilities, based on past experience, although waste form loading can also be an issue. For HLW, the difference is caused more by the effects of advanced waste forms as compared to conventional waste forms, as well as potential reclassification of some of the wastes. Where a particular facility would be within each range depends on the extent to which existing performance can be improved upon, reducing waste production from operations, maintenance, and similar activities.

Uranium storage requirements have also been obtained for the uranium recovered from processing spent fuel. The amounts range from 1020 m³/yr for the thermal recycle alternative, to 570 m³/yr for the thermal/fast recycle case with fast reactor recycle, and to 660 m³/yr for the thermal/fast recycle case with both thermal and fast recycle.

All of the upper bound estimates are based on assumed facility sizes, staffing levels and rates of operation.[B1, B2, B3] While they are typical for the processing of spent LWR fuel, with processing plants of that size in operation today, there are no processing plants operating for the processing of spent fast reactor fuel, or for the fabrication of transmutation fuel. The values chosen in References B2 and B3 are believed to be representative based on the engineering judgment of the contributors to those reports. Given the relatively large, 'upper bound', nature of the waste production estimates, it is likely that these estimates will bound waste production, even if the real facilities are of a significantly different size, or are operated in a manner that is substantially different than assumed in the reports.

The lower bound estimates for waste volume are based on the wastes that would arise solely from the spent fuel contents, and do not include any of the wastes associated with operating the facilities or for the effects of packaging, shielding, etc. that may be required. The lower bounds also reflect the potential for reclassification of wastes, such as the possibility that Cs/Sr waste may satisfy the regulation for LLW if sufficient decay time is provided before disposal. As such, the lower bounds represent not only a technological limit, but changes in waste classification as well, where the wastes would be classified based on hazard rather than origin.

At this stage, there are still unanswered questions, particularly about the electrochemical processing technologies and the associated waste production. The bounds provided in Table B-20 are representative of what would be expected from the technologies existing today, where more detailed information was available. It is also fair to say that if a desired processing technology were to produce waste amounts in excess of those listed in Table B-20, then the environmental impacts of that technology would have to be evaluated at the time that justification for that choice was required.

Table B-20. Summary of Annual Waste Generation for GNEP PEIS Alternatives.

Waste Category	Estimated Bounds for Waste Generation per 100 GWe-yr/yr								
	Option 1 LWR/LWR MOX-U-Pu		Option 2 LWR/HWR DUPIC*		Thermal/Fast Recycle: Fast Reactor Recycle Only		Thermal/Fast Recycle: Thermal and Fast Reactor Recycle		
	Lower	Upper	Lower	Upper	Lower	Upper	Lower	Upper	
Low Level									
Solid (m ³ /yr)	17	26,000	-	-	59-209	38,160	49-161	34,200	
Greater Than Class C (GTCC)									
Solid (m ³ /yr)	170	6,690	120	-	195-210	6,840	186-196	6,575	
HLW									
Solid (m ³ /yr)	32	870	-	300-800	26-62	920	27-55	905	

* The waste volumes for DUPIC do not include the direct disposal of the spent HWR fuel.

B-5. References

- B1. "Engineering Alternative Studies for Separations NEPA Data Input Report," EAS-Q-NEP-G-00001, June 2008 Revision 4.
- B2. "Transmutation Fuel Fabrication Facility NEPA Data Input Report," EAS-Q-NEP-G-00003, April 2008 Revision 2.
- B3. "Fast Reactor Spent Nuclear Fuel Recycling Facility NEPA Data Input Report," EAS-Q-NEP-G-00004, June 2008 Revision 2.
- B4. Ellis, R. J., "CANDU Reactors: An Overview and Heavy Water Requirements," Oak Ridge National Laboratory, September 7, 2007.
- B5. Sterbentz, J. W., et al, "Reactor Physics Parametric and Depletion Studies in Support of TRISO Particle Fuel Particle Specification for the Next Generation Nuclear Plant," INEEL/EXT-04-02331, September 2004.
- B6. Kim, T. K., "Evaluation of Spent Nuclear Fuel Characteristics of Next Generation Nuclear Plant (NGNP)," ANL Intra-Laboratory Memo, October 5, 2004.
- B7. "Global Nuclear Energy Partnership Integrated Waste Management Strategy," GNEP-WAST-WAST-AI-RT-2008-000214, March 2008.
- B8. "Property Data for Simulated Americium/Curium Glasses," PNNL-13009, J. D. Vienna, et al, September 1999.

- B9. “Environmental Impact Statement on the Construction and Operation of a Proposed Mixed Oxide Fuel Fabrication Facility at the Savannah River Site, South Carolina,” NUREG-1767, January 2005.
- B10. “Advanced Nuclear Fuel Cycles and Radioactive Waste Management,” OECD/NEA No. 5990, 2006.

Appendix C

Radiotoxicity Reduction – Time to Decay to Level of Natural Uranium

Radiotoxicity Reduction – Time to Decay to Level of Natural Uranium

The radiotoxicity of the SNF or processing wastes placed in geologic disposal provides a measure of the environmental hazard from these materials. There are several potential exposure pathways; in this summary, the ingestion pathway was used. Since radiotoxicity varies greatly with time, a useful metric for comparing one fuel cycle approach to another is to compare the time required for the radiotoxicity to decay to the level of the natural uranium ore that was used as the source material for the fuel.

C-1. No-Action Fuel Cycle Alternative – Once-Through LWRs

The normalized radiotoxicity for the direct disposal of spent LWR fuel is given in Figure C-1 [C1], where the radiotoxicity is normalized to that of the natural uranium ore, which has a constant radiotoxicity per ton in nature. The time at which the normalized radiotoxicity equals 1.0 indicates when the SNF is as radiotoxic as the natural uranium was that was used to make the fuel.

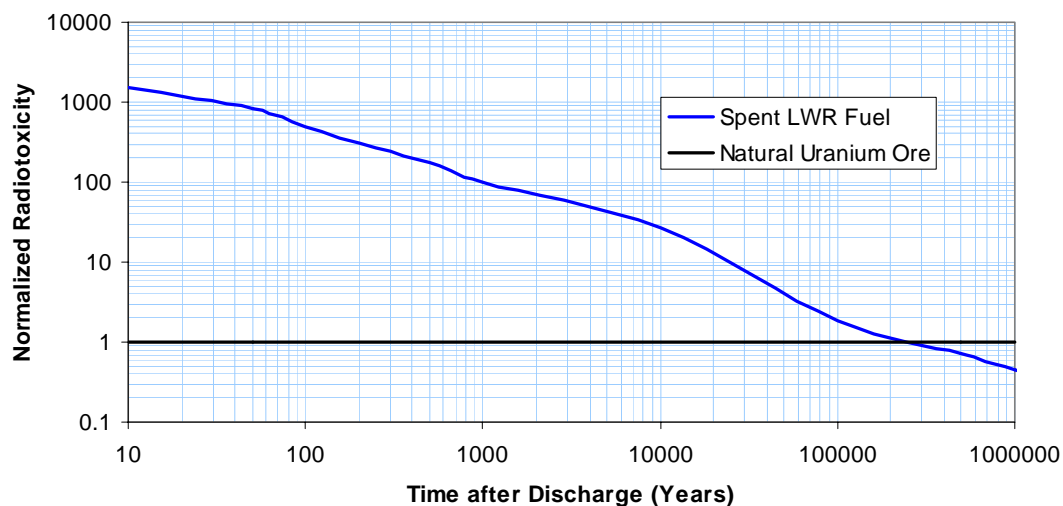


Figure C-1. Normalized radiotoxicity as a function of time after discharge for spent LWR fuel.

A graphical measure of the intersection of the normalized radiotoxicity for spent LWR fuel and the natural uranium ore using the fitted curve gives 240,000 years, which is the value listed in Table 1.

C-2. HWR Once-Through Fuel Cycle Alternative

The normalized radiotoxicity for spent HWR fuel is shown in Figure C-2. [C1]

The radiotoxicity of the spent HWR fuel is quite similar to that of the spent LWR fuel. A graphical measure of the intersection of the normalized radiotoxicity for spent HWR fuel and the natural uranium ore using the fitted curve gives 255,000 years, which is the value listed in Table 1.

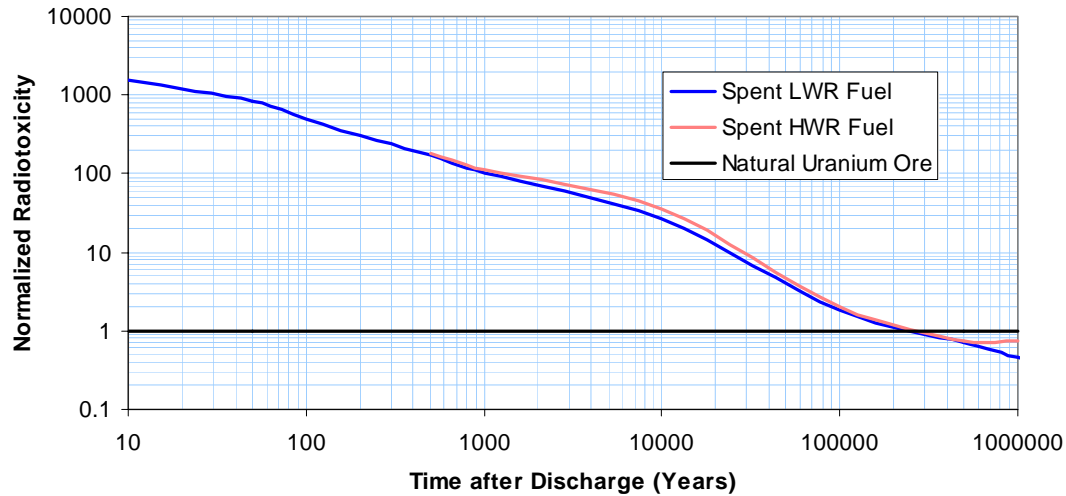


Figure C-2. Normalized radiotoxicity as a function of time after discharge for spent HWR fuel.

C-3. HTGR Once-Through Fuel Cycle Alternative

The normalized radiotoxicity for spent HTGR fuel is shown in Figure C-3. [C1]

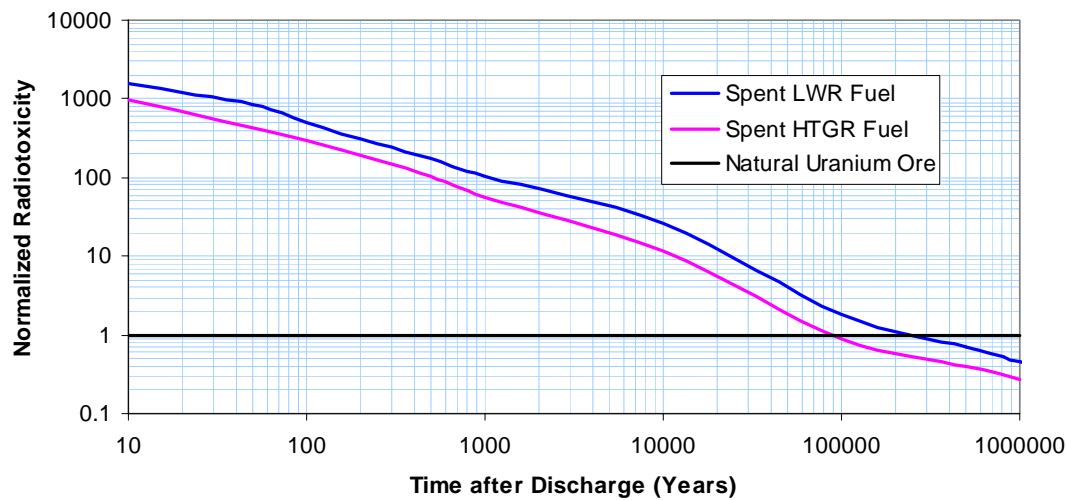


Figure C-3. Normalized radiotoxicity as a function of time after discharge for spent HTGR fuel.

The radiotoxicity of the spent HTGR fuel is lower than the radiotoxicity of spent LWR fuel at all times, by about a factor of 2-3. As a consequence, the radiotoxicity of the spent HTGR fuel becomes equal to that of the natural uranium ore at an earlier time. A graphical measure of the intersection of the normalized radiotoxicity for spent HTGR fuel and the natural uranium ore using the fitted curve gives 85,000 years, which is the value listed in Table 1.

C-4. Thorium Once-Through Fuel Cycle Alternative

The radiotoxicity for spent thorium/uranium fuel is shown in Figure C-4. [C2]

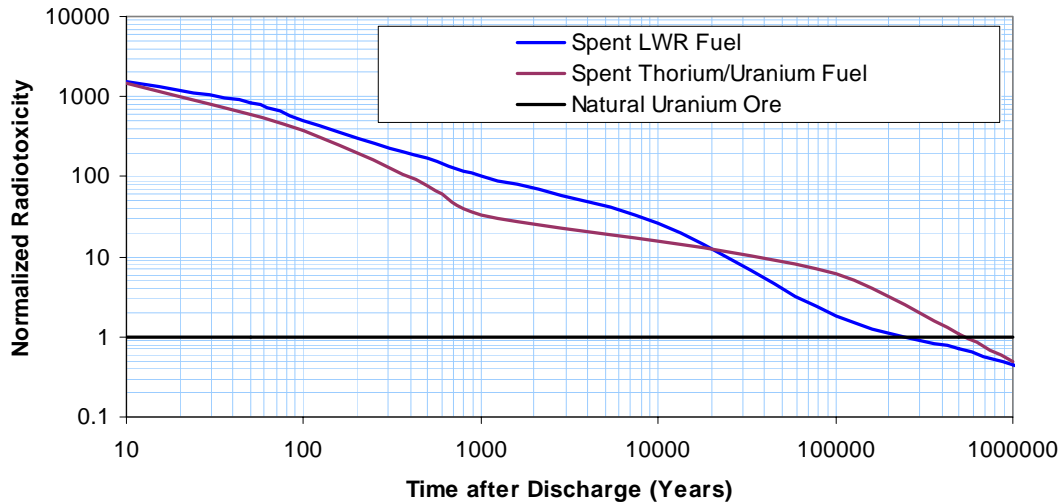


Figure C-4. Normalized radiotoxicity as a function of time after discharge for spent thorium/uranium fuel.

The thorium/uranium radiotoxicity is initially lower than the radiotoxicity of spent LWR fuel, but is higher than the radiotoxicity of spent PWR fuel after about 20,000 years due to the different decay products in the spent fuel. It can be estimated that the radiotoxicity of the thorium option shown is equal to the radiotoxicity of the natural uranium ore at about 525,000 years, which is listed in Table 1.

C-5. Thermal Reactor Recycle Alternative, Option 1 - LWR/LWR (MOX-U-Pu)

The normalized radiotoxicity for the HLW from the LWR/LWR (MOX-U-Pu) recycle option is shown in Figure C-5. [C1] Graphically, the radiotoxicity of the HLW equals that of the natural uranium ore at about 55,000 years, and is listed in Table 1.

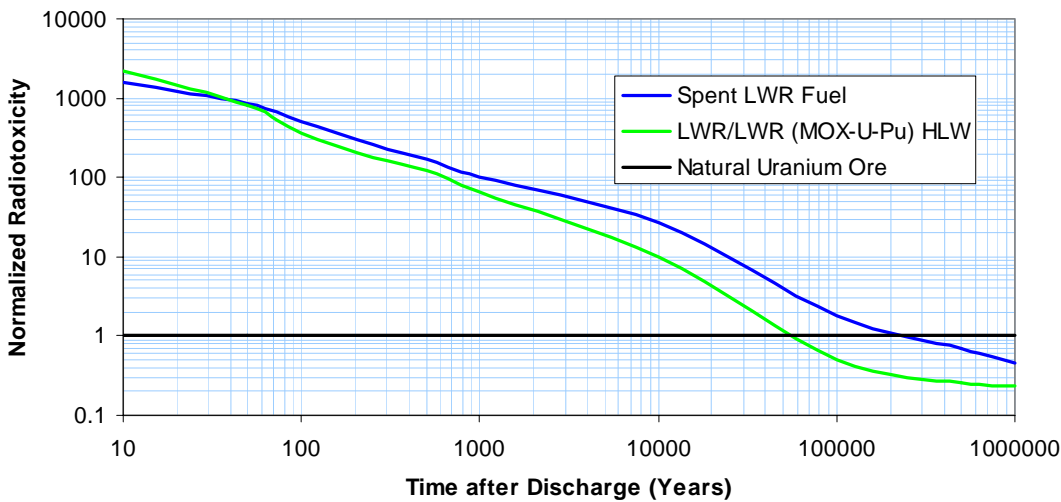


Figure C-5. Normalized radiotoxicity as a function of time after discharge for the HLW from processing spent LWR/LWR (MOX-U-Pu) fuel.

C-6. Comments on Thermal Reactor Recycle Alternative, Option 2 – LWR/HWR (DUPIC)

Although no radiotoxicity data was available for the DUPIC approach, given that all of the fission products and actinides are sent to the repository as with the current once-through approach, and that the direct disposal of spent PWR fuel and spent HWR fuel have similar transient radiotoxicity characteristics, it is likely that the time required for the radiotoxicity of the disposed spent HWR fuel from DUPIC to decrease to that of natural uranium ore to be similar as well, i.e., on the order of 100,000 – 300,000 years.

C-7. Thermal/Fast Recycle Alternative (CR=0.50): Fast Reactor Recycle Only

The normalized radiotoxicity for the HLW from the processing of spent LWR fuel, recycle of all of the TRU in fast reactors, and indefinite processing of the spent fast reactor fuel and recycle of the TRU (the fast reactor recycle only option) is shown in Figure C-6. [C3]

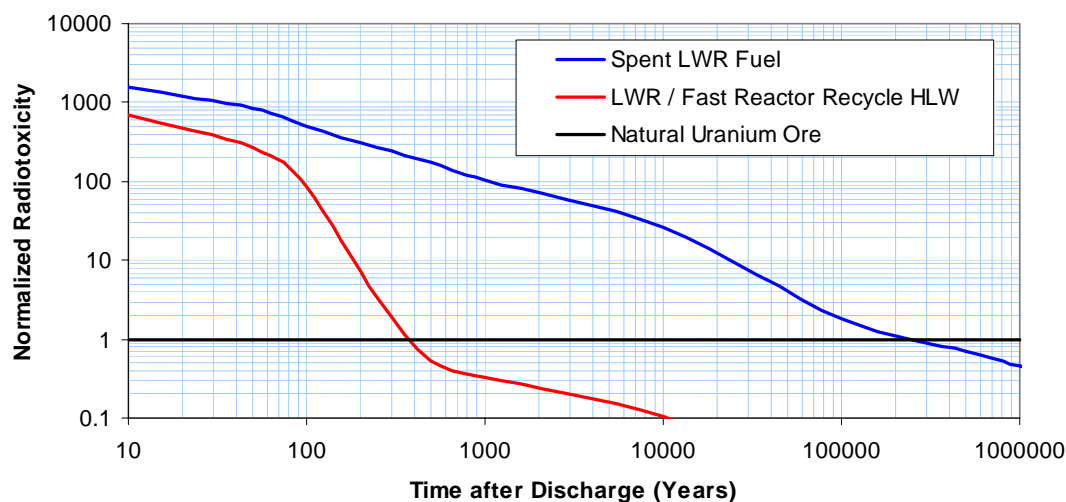


Figure C-6. Normalized radiotoxicity as a function of time after discharge for the HLW from processing spent LWR and fast reactor fuel.

Graphically, the radiotoxicity of the HLW equals that of the natural uranium ore at about 375 years, and is listed in Table 1.

C-8. Thermal/Fast Recycle Alternative (CR=0.50): Thermal & Fast Reactor Recycle Option

The normalized radiotoxicity for the HLW from the processing of spent LWR fuel, recycle of the recovered plutonium for one recycle in an LWR, processing of the recycle spent LWR fuel, with the minor actinides (MA) from the spent LWR fuel processing and all of the TRU from the recycle spent LWR fuel processing being recycled in fast reactors, and indefinite processing of the spent fast reactor fuel and recycle of the TRU (the thermal and fast reactor recycle option) is shown in Figure C-7. [C3]

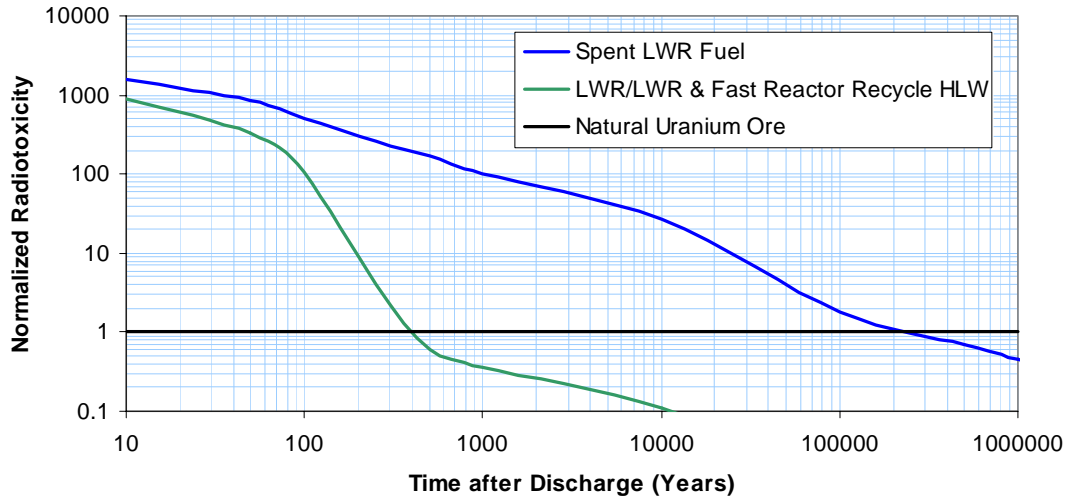


Figure C-7. Normalized radiotoxicity as a function of time after discharge for the HLW from processing spent LWR and fast reactor fuel.

Graphically, the radiotoxicity of the HLW equals that of the natural uranium ore at about 400 years, and is listed in Table 1.

C-9. References

- C1. Kim, T. K., and Taiwo, T. A., Radiotoxicity Data, Argonne National Laboratory, email communication, October 16, 2007.
- C2. Todosow, M., Radiotoxicity Data, Brookhaven National Laboratory, October 23, 2007.
- C3. Kim, T. K., and Taiwo, T. A., Radiotoxicity Data, Argonne National Laboratory, email communication, October 25, 2007.