# *Pros and Cons Analysis of HALEU Utilization in Example Fuel Cycles*

## Nuclear Fuel Cycle and Supply Chain

***Prepared for***

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***Systems Analysis and Integration***

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# Executive Summary

The Systems Analysis and Integration campaign assessed the pros and cons of high-assay low-enriched uranium (HALEU) utilization in advanced reactors and associated fuel cycles. The assessment was done for three example fuel cycles at equilibrium states: once-through, limited recycle, and continuous recycle (CR) starting with HALEU. Front- and back-end fuel cycle parameters and the Levelized Cost of Fuel (LCF), which is the Levelized Cost of Electricity excluding reactor cost, of the three example fuel cycles were calculated using a single Analysis Example Reactor. The pros and cons of HALEU utilization were assessed by normalizing the fuel cycle parameters and LCF to a unit of electricity generation (GWe-year) and comparing them with a Basis of Comparison. In this study, a sodium-cooled reactor with sodium-bonded metallic fuel having a burnup of ~100 GWd/t was used as the Analysis Example Reactor because its technology readiness level is high, and the burnup and fuel enrichment are in the middle of those ranges of advanced reactor concepts that are under development. The current once-through Light Water Reactors (OT-LWRs) with <5% low-enriched uranium and 50 GWd/t burnup were used as the Basis of Comparison. In addition, a series of sensitivity analyses was conducted by varying burnup, enrichment, fuel forms, and reactor types to capture the design variations in two once-through Advanced Reactor Demonstration Program (ARDP) reactors, Natrium with sodium-free metallic fuel having a burnup of ~150 GW/t and Xe-100 with Tristructural-Isotropic (TRISO) pebble fuel having a burnup of ~168 GWd/t.

**Once-through fuel cycle**

- Figure E.1 shows the fuel cycle performance and LCFs of the once-through fuel cycle relative to the Basis of Comparison. The x-axis denotes fuel cycle parameters, and the y-axis indicates their ratios to the Basis of Comparison. For comparison purposes, the values of the two above-mentioned ARDP reactors are included in the figure.
- The normalized front-end fuel cycle parameters, such as natural uranium (NU) requirement and Separative Work Units (SWUs), increase as the fuel enrichment increases but decrease as the burnup increases. Thus, the pros of the front-end fuel cycle with HALEU compete with the cons depending on the relative change in fuel enrichment versus achieved burnup. Figure E.1 shows that the charge fuel mass of the Analysis Example Reactor is smaller, but other front-end fuel cycle parameters are larger than those of the Basis of Comparison because the burnup is not increased as much as the increase in fuel enrichment. The fuel enrichment of the Analysis Example Reactor is elevated by a factor of four compared to that of the Basis of Comparison, but the burnup is elevated by only a factor of two. The front-end LCF is also higher than that of the Basis of Comparison, owing in part to the more stringent security requirements imposed on HALEU enrichment and fuel fabrication facilities.
- In contrast, the back-end fuel cycle parameters of the Analysis Example Reactor are generally better than those of the Basis of Comparison. The increased burnup of the Analysis Example Reactor with HALEU reduces the discharge fuel (DF) mass, which requires less interim storage and less geologic-disposal volume. However, the back-end LCF of the Analysis Example Reactor is higher than that of the Basis of Comparison because the sodium-bonded metallic fuel was assumed to undergo electrometallurgical treatment to remove the bond sodium before disposal.
- The sensitivity study shows that the front-end fuel cycle parameters of the two ARDP reactors, Natrium and Xe-100, are generally improved compared to the Analysis Example Reactor and closer to the Basis of Comparison for NU demand and SWUs, while LCF performance is mixed.
- For maximizing the pros and mitigating the cons in the front-end fuel cycle, R&D efforts on further extension of fuel burnup and affordable TRISO fuel fabrication are indicated.
- For the back-end fuel cycle parameters and LCFs, the two ARDP reactors show the same trend for the waste mass and decay heat, which are lower than those of the Basis of Comparison, but they have different directions with respect to the disposal volume and LCF because the disposal volume and

LCF are also affected by fuel types. Natrium's disposal volume is reduced relative to that of the Basis of Comparison, but Xe-100 generates a greater disposal volume, owing to the high volume fraction of graphite and matrix materials in pebble fuel. The back-end LCF of Natrium is significantly reduced because the sodium-free metallic fuel does not need additional treatment before disposal. Xe-100's TRISO/pebble fuel is also suitable for direct disposal without further treatment. However, the LCF contribution is increased significantly because of the large disposal volume.

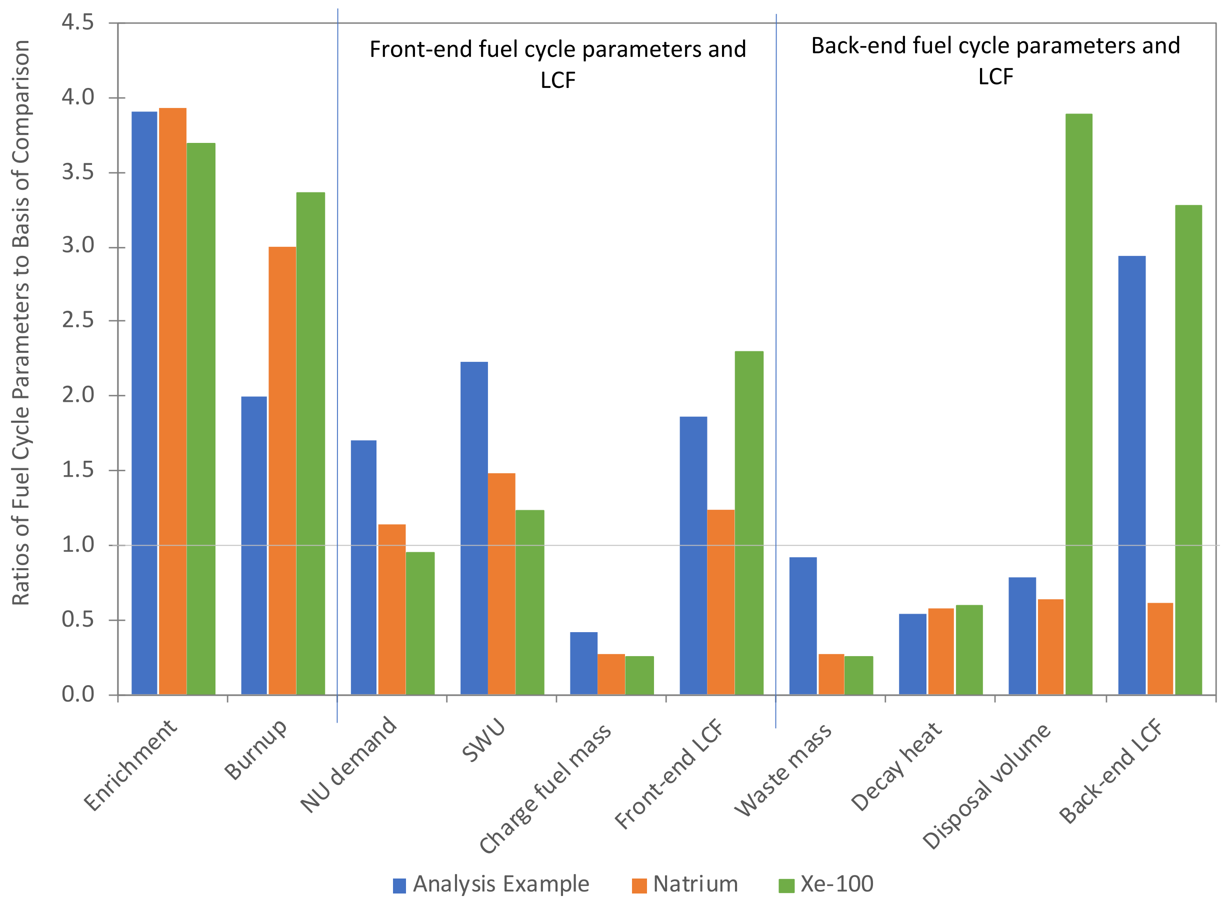


Figure E.1 Relative comparison of fuel cycle parameters and LCFs of once-through fuel cycles with HALEU to those of Basis of Comparison

**Limited- and continuous-recycle fuel cycles**

- The pros and cons of HALEU utilization in limited-recycle fuel cycles were assessed using a particular limited fuel cycle consisting of two stages. The first stage is an advanced reactor with HALEU fuel, and the second stage is an LWR utilizing the recovered uranium from the first-stage HALEU DF. For this particular limited-recycle system, overall front- and back-end fuel cycle parameters were improved compared with the once-through fuel cycle with HALEU because about 50% of total electricity is generated using the recovered uranium.
- In a continuous-recycle fuel cycle, the reactor is sustainable without HALEU. The HALEU fuel is only needed to start the core and a few initial cycles (5 reload cycles in the present work). Thus, the required HALEU during the reactor lifetime is small, and as a result, the pros and cons of HALEU

utilization in a continuous-recycle fuel cycle are minimal. However, HALEU permits the initiation of fuel cycle evolution to CR, which was identified as a “most promising” fuel cycle in the fuel cycle evaluation and screening study [Wigeland et al. 2014].

# Acknowledgments

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# Abbreviations

| | |
|---|---|
| ABR | Advanced Burner Reactor |
| ANL | Argonne National Laboratory |
| ARDP | Advanced Reactor Demonstration Program |
| BWR | Boiling Water Reactor |
| Conv. LR | Conventional Limited Recycle Fuel Cycle |
| CR | Continuous Recycle |
| DF | Discharge Fuel |
| DOD | U.S. Department of Defense |
| DOE | U.S. Department of Energy |
| DPC | Dual Purpose Canister |
| DU | Depleted Uranium |
| EMT | Electrometallurgical Treatment |
| E&S | Evaluation and Screening |
| FCCI | Fuel-Cladding Chemical Interaction |
| FHR | Fluoride-salt-cooled High-temperature Reactor |
| FOAK | First-of-a-kind |
| FP | Fission Product |
| GFR | Gas-cooled Fast Reactor |
| HALEU | High-Assay Low Enriched Uranium |
| HLW | High-Level Waste |
| HM | Heavy Metal |
| HTGR | High-Temperature Gas-cooled Reactor |
| LCF | Levelized Cost of Fuel |
| LEU | Low Enriched Uranium |
| LFR | Lead-cooled Fast reactor |
| LR-RU | Limited Recycle with Recovered Uranium |
| LWR | Light Water Reactor |
| MA | Minor Actinide |
| MOX | Mixed Oxide |
| MR | Microreactor |
| MSFR | Molten Salt Fast Reactor |
| MSR | Molten Salt Reactor |
| NASA | National Aeronautics and Space Administration |
| NOAK | Nth-of-a-kind |
| NRC | U.S. Nuclear Regulatory Commission |
| NU | Natural Uranium |
| NWTRB | Nuclear Waste Technical Review Board |
| OT-HALEU | Once-through Fuel Cycle with HALEU |
| OT-LWR | Once-through Light Water Reactor |
| PWR | Pressurized Water Reactor |
| RU | Recovered Uranium |
| SA&I | Systems Analysis and Integration |
| SFR | Sodium-cooled Fast Reactor |
| SMR | Small Modular Reactor |
| SNF | Spent Nuclear Fuel |
| SWU | Separative Work Unit |
| TRISO | Tristructural-Isotropic |
| TRU | Transuranics |
| TU | Tail Uranium |

| | |
|---|---|
| UFD | Used Fuel Disposition |
| UOX | Uranium Oxide |
| w/o | Weight percent |

# SYSTEMS ANALYSIS AND INTEGRATION CAMPAIGN

# PROS AND CONS ANALYSIS OF HALEU UTILIZATION IN EXAMPLE FUEL CYCLES

## 1. Introduction

Various advanced reactor concepts and associated nuclear fuel cycles have been proposed recently by industry, universities, and national laboratories. Most of them utilize 10%–19.75% enriched uranium, aiming for higher burnup, compact core design, improved thermal efficiency with higher operating temperature, etc. As a trade-off, however, the utilization of high-assay low enriched uranium (HALEU) increases natural uranium (NU) demand and requires enrichment in facilities with more stringent security. Generally, it is expected that HALEU utilization would be beneficial to the back-end fuel cycle but detrimental to the front-end fuel cycle, and the net impact on the overall nuclear fuel cycle performance is dependent on the fuel enrichment, burnup, and fuel forms adopted by the various advanced reactor concepts. It is noted that "HALEU" denotes a uranium having an assay greater than 5.0 weight percent (w/o) and less than 20 w/o of U-235. However, the present work only considers HALEU having the assay of U-235 in the range of 10–19.75 w/o.[1]

In this study, the Systems Analysis and Integration (SA&I) campaign assessed the pros and cons of HALEU utilization in advanced reactor concepts through a systematic evaluation of the fuel cycle performance parameters and Levelized Cost of Fuel (LCF), which is the Levelized Cost of Electricity excluding reactor cost. First, the SA&I campaign collected the existing advanced rector concepts and identified three example fuel cycles associated with advanced reactor concepts with HALEU. A series of studies of the fuel cycle transition from the current nuclear fleet to the example fuel cycles is required to calculate the mass flow data needed for fuel cycle parameter estimates. However, in the present study, both front- and back-end fuel cycle parameters and LCF of the example fuel cycles were evaluated using the mass flow data at the equilibrium state. For consistent and quantitative assessment, the fuel cycle parameters were calculated using a single Analysis Example Reactor concept that has roughly the average burnup and enrichment of the advanced reactor concepts reviewed in the present study.

The advanced reactor concepts adopt various reactor types and fuel forms, and the resulting fuel cycle parameters could be spread over a wide range. Thus, a series of sensitivity analyses were conducted in the present study to capture the impact of the variation in the parameters of the advanced reactor designs, which include two once-through Advanced Reactor Demonstration Program (ARDP) reactors: Natrium, with sodium-free metallic fuel having a burnup of 150 GW/t, and Xe-100, with Tristructural-Isotropic (TRISO) pebble fuel having a burnup of ~168 GWd/t.

The resulting fuel cycle parameters were normalized to a unit of electricity generation (GWe-year) for comparison. Finally, the pros and cons, including the potential R&D needs for maximizing the pros and mitigating the cons, were derived by comparing normalized fuel cycle parameters to a Basis of

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[1] In a legal document such as the Energy Act of 2020, "high-assay low enriched uranium" (HA-LEU or HALEU) means uranium having an assay of U-235 greater than 5.0 w/o and less than 20.0 w/o. It is noted that advanced reactors considered in the present work utilize 10–19.75% enriched uranium fuels, and a facility handling those fuels is subject to more stringent security requirements because uranium having an assay of U-235 greater than 10 w/o and less than 20 w/o is classified as Category-II in accordance with the categories of special nuclear materials or facilities by the U.S. Nuclear Regulatory Commission (NRC).

Comparison, which is the current Once-through Light Water Reactor (OT-LWR) with <5% low enriched uranium (LEU) fuel and 50 GWd/t burnup.

The rationales for the selection of the three example nuclear fuel cycles and the Analysis Example Reactor concept are discussed in Section 2. In Section 3, the front-end and back-end fuel cycle parameters and LCFs of the three example fuel cycles are calculated, and the results are compared with the Basis of Comparison. The sensitivity study performed to capture the impacts of advanced-reactor variation on the fuel cycle parameters is discussed in Section 4. The pros and cons are discussed in Section 5, including the potential R&D needed to maximize the pros and mitigate the cons. The conclusions are provided in Section 6. Finally, various assumptions and computation models used in the present study are summarized in the Appendices.

# 2. Example Fuel Cycles and Associated Analysis Example Reactor Concepts

## 2.1 Example Nuclear Fuel Cycles with HALEU

Various advanced reactor concepts are currently under development in the United States, and the total number of advanced reactor concepts has been increasing with the investment of private capital and various initiatives of government agencies (DOE, DOD, NASA, etc.). The advanced reactor concepts being developed in the United States and Canada are listed in Appendix A. The list includes LWR-based small modular reactors (SMRs), microreactors (MRs), and non-LWR advanced reactor concepts but excludes evolutionary LWRs. It is noted that the list might not reflect the actual status of advanced reactor concepts because the reactor concepts have been evolving and, in some cases, only limited non-proprietary information is available.

The nuclear fuel cycles of the listed advanced reactor concepts are tallied in Figure 2.1. The dominant nuclear fuel cycle is once-through, followed by continuous recycle (CR). Most advanced reactor concepts based on the once-through fuel cycle use HALEU fuels with an enrichment range of 10%–19.75%. In comparison, several SMRs based on LWR technologies utilize conventional LEU fuel (< 5%). Evolutionary LWRs and accident-tolerant fuels use 5–10% enriched fuels, but the evolutionary LWRs and fuel concepts are not considered here because the present study focuses on HALEU utilization in advanced reactor concepts. The CR reactor concepts and once-through breed-and-burn reactor concepts require NU or depleted uranium (DU) at an equilibrium state, but they also need HALEU for starting the reactors and for the initial reactor cycles when the bred fissile is insufficient.

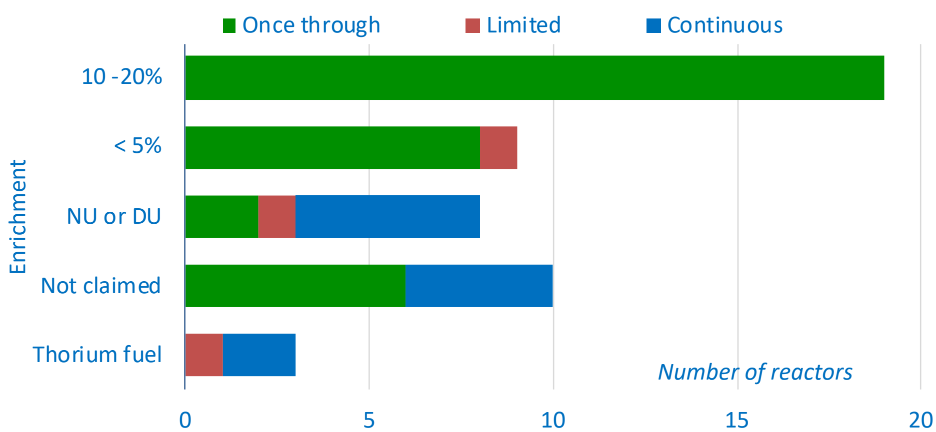


Figure 2.1 Uranium enrichments of fuels and associated fuel cycles for advanced reactors listed in Appendix A

Using the collected information on the advanced reactor concepts and associated fuel cycles, the pros and cons of HALEU utilization were assessed for the following three example fuel cycles:

- Once-through fuel cycle in an advanced reactor with HALEU,
- Two-stage limited recycle—the first stage is an advanced reactor with HALEU and the second stage is an LWR with down-blended recovered uranium (RU) from the first-stage discharge fuel (DF)—and
- CR of U/Pu (or U/TRU) at equilibrium state in an advanced reactor. The initial core starts with HALEU.

The limited-recycle fuel cycle evaluated in this report consists of two stages. The first stage is an advanced reactor with HALEU, and the second stage is an advanced LWR with <5% LEU fuel using down-blended RU from the first-stage DF. The DF from the second stage and non-recovered materials from the first stage are placed in interim storage and disposed of in a geologic repository without further recycling.

The continuous-recycle (CR) fuel cycle does not need enriched uranium in the equilibrium state, but HALEU fuel is required for the initial core and goes through a few reload cycles until sufficient Pu (or transuranics, TRU) is bred.

## 2.2 Analysis Example Reactors with HALEU

For assessing the pros and cons of HALEU utilization, a single Analysis Example Reactor for each fuel cycle was identified among the proposed advanced reactor concepts. Even though a single Analysis Example Reactor was used for three fuel cycle scenarios, the fuel compositions were differentiated depending on fuel cycles. The selection of the Analysis Example Reactor associated with the three fuel cycles and their design parameters is discussed in this section, and the detailed design information for the Analysis Example Reactor is summarized in Appendix B.

### 2.2.1 Analysis Example Reactor for Once-through Fuel Cycle

Figure 2.2 shows reactor-type tallies of the once-through advanced reactor concepts with HALEU collected in the present study. Fuels with TRISO particles and metallic fuels are dominant in the once-through thermal and fast reactor concepts, respectively. For instance, thermal reactors such as the Fluoride-salt-cooled High-temperature Reactors (FHR) and High-Temperature Gas-cooled Reactors (HTGR) use a TRISO fuel containing HALEU kernels. In contrast, fast reactors such as the Lead-cooled Fast Reactor (LFR) and the Sodium-cooled Fast Reactor (SFR) use metallic U-Zr alloy fuels with HALEU. Both thermal and fast MR (less than 50 MWe) concepts using TRISO and metallic fuels have been proposed.

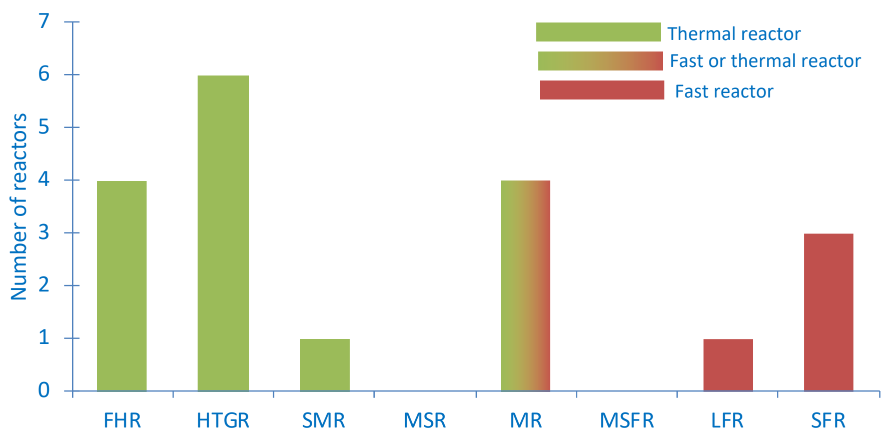


Figure 2.2 Once-through advanced reactor concepts with HALEU in the U.S. and Canada

The primary design parameters of the once-through advanced reactor concepts are summarized in Table 2.1. They are grouped into thermal reactor concepts with TRISO fuels and fast reactor concepts with metallic fuels. Reactors in the same group have comparable reactor physics characteristics and fuel cycle performance because the reactors have similar neutron spectrums and fuel forms. For comparison purposes, a conventional LWR (Pressurized Water Reactor and Boiling Water Reactor) has been added to the table, and the Analysis Example Reactor selected for the HALEU utilization assessment is included in the last column.

The fast reactor concepts require HALEU to compensate for their high neutron leakage rates and structural materials' parasitic absorptions. Thermal reactor concepts with TRISO fuels need HALEU to increase the burnup. Both thermal and fast MR concepts use HALEU fuels to achieve the desired cycle length by compensating for a high neutron leakage rate from a small-size core.

Among various reactor types utilizing HALEU, an SFR was selected as the once-through advanced reactor concept because an SFR (Natrium) is one of the reactor types to be demonstrated through the ARDP initiative, and the remaining U-235 content in the DF is adequate for the limited-recycle fuel cycle considered in Section 2.2.2. As shown in Appendix A, various advanced reactor types are under development. A series of sensitivity studies was conducted to capture the impact of reactor type variation on the fuel cycle parameters, and the results are compared in Section 4.4.

Table 2.1 Primary reactor design parameters of once-through advanced reactor concepts

| | Conventional LWR | Thermal reactor concepts with TRISO fuel | Fast reactor concepts with metallic fuel | Analysis Example Reactor |
|---|---|---|---|---|
| Reactor type | PWR, BWR | FHR, HTGR, MR | LFR, SFR | SFR |
| Coolant | Light water | Gas, salt | Liquid metal | Sodium |
| Electric power output, MWe [a)] | Large | Small | Small, medium | Medium |
| Thermal efficiency, % | ~33 | 40–50 | ~40 | 40 |
| Fuel form | $UO_2$ | TRISO with UCO, $UO_2$, UN kernels | U-Zr metal | U-Zr metal |
| Driver fuel fissile content, % | ~5 | 12–19.9 | 13.5–19.5 | 16.4 |
| Avg. discharge burnup, GWd/t | 50–60 | 120–180 | 80–150 | ~100 |
| DF content, %<br>- U-234/U<br>- U-235/U<br>- U-236/U<br>- U-238/U<br>- U/HM [b)]<br>- Pu/HM<br>- MA/HM | <br>0.02<br>0.78<br>0.58<br>97.2<br>98.6<br>1.26<br>0.1 | XE-100<br>- U-235/U ratio is < 1.5% | Natrium<br>- U-235/U ratio is 5–10% | <br>0.14<br>9.2<br>1.8<br>88.8<br>94.5<br>5.3<br>0.2 |

a) Power rates: small < 300 MWe, medium 300–1000 MWe, large > 1000 Mwe
b) HM = heavy metal

The DF compositions are compared in Table 2.1. One of the important differences between the thermal and fast advanced reactor concepts is the remaining U-235 content in the DFs. X-Energy public information indicates that more than 90% of the U-235 is depleted in the Xe-100 reactor before the fuel is discharged from the core [Mulder and Boyes 2020], and the remaining U-235 content (i.e., U-235 mass relative to the total uranium mass) in DF is ~1.5%. However, for fast reactor concepts such as the Analysis Example Reactor or Natrium, the remaining U-235 content is higher than that of the charge fuel in conventional LWRs [Schloss and Hejzlar 2021].

The Analysis Example Reactor design parameters were obtained by revising the advanced burner reactor concept [Kim et al. 2008] such that the driver fuel enrichment and discharge burnup are in the middle of the enrichment and burnup ranges of once-through advanced reactor concepts. The driver-fuel uranium enrichment is 16.4%, and the discharge burnup is 96.7 GWd/t. In the DF, the remaining U-235 content is ~9.2%, and the mass ratio of Pu to the total heavy metal (HM) is 5.3%. Both values are much higher than those of the LWR DF.

It is noted that U-Zr metallic fuel has been demonstrated and qualified up to an average burnup of 100 GWd/t [Crawford et al. 2007]. Thus, the Analysis Example Reactor was designed to have a discharge burnup of ~100 GWd/t. However, because a further extension of the discharge burnup is planned in the two demonstration reactor concepts (Natrium and Xe-100), the impacts of the higher burnup on the fuel cycle parameters are discussed in Section 4.

### 2.2.2 Analysis Example Reactor for Limited Recycle

Various limited fuel cycle options have been proposed, mostly focusing on the reuse of recovered U/Pu from LWR DFs where the Pu provided most of the recovered fissile content. The fuel cycle performances of those limited fuel cycles were previously assessed in the nuclear fuel cycle evaluation and screening (E&S) study [Wigeland et al. 2014]. In the present study, the focus in the limited fuel cycle system is on the reuse of the RU because unlike typical LWR fuels, the discharge fuel from an advanced fast reactor (such as Natrium) with HALEU fuel has a high U-235 content.

Figure 2.3 is an overview of the two-stage limited-recycle fuel cycle option considered in the present study. The first stage is an advanced reactor with HALEU. The RU (or RU/Pu) from the DF is sent to the second stage, while the minor actinides (MAs) and fission products (FPs) are to be disposed of in a geologic repository (Δ). The second stage is a PWR, with uranium oxide (UOX) fuel made with the RU from the first-stage DF. The DF from the second stage is sent to a repository without further reprocessing.

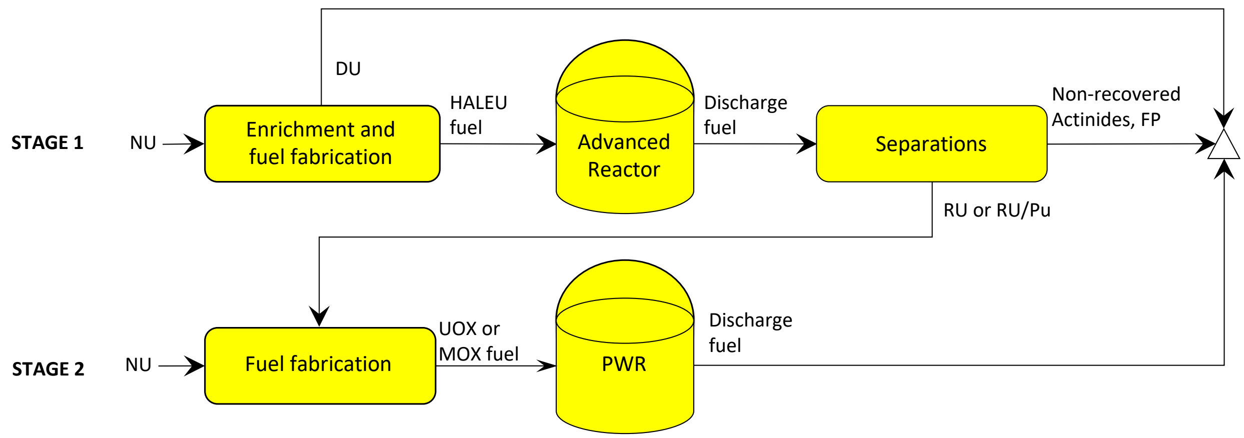


Figure 2.3 Overview of two-stage limited-recycle fuel cycle with HALEU

The motivation for the limited-recycle fuel cycle is to recycle the remaining higher U-235 content in HALEU DF. Table 2.1 shows that the remaining U-235 content in the HALEU DF is 9.2%, which is higher than the enrichment of current LWR fresh fuels. Thus, the second-stage PWR fuel can be fabricated by down-blending RU with NU. A similar limited-recycle fuel cycle in LWRs has been studied [OECD/NEA 2002], in which the recovered Pu from the first-stage LWR DF is recycled in the second-stage LWR in the form of U-Pu mixed oxide (MOX) fuel [Kim 2013]. This limited-recycle fuel cycle is called "*conventional limited-recycle fuel cycle (Conv. LR*)" in the present report.

Both Ru and U/Pu from the HALEU DF can be recycled in the second stage, but the present study focuses on the recycling of the RU only; the fuel cycle performance parameters, including those of the Conv. LR, are provided in Table 2.2. Detailed information on the limited-recycle fuel cycle is provided in Appendix B.

The down-blending ratio of the second-stage PWR fuel was iteratively searched to have a discharge burnup of ~50 GWd/t, which is the average discharge burnup of a conventional PWR with 4.2% LEU UOX fuel. The resulting uranium enrichment is 4.4%, while the blending ratio is 43.8%. It is noted that the second-stage PWR would require a slightly higher enrichment compared to the conventional PWR because the RU contains U-236, which acts as a poison. U-236 is created from the U-235 (n,$\gamma$) reaction and the $\alpha$-decay of Pu-240, and U-236 content in the DF is proportional to burnup and U-235 content. Table 2.1 shows that the U-236 content in the HALEU DF is ~1.8%.

Table 2.2 Comparison of fuel cycle parameters of two limited-recycle fuel cycles

| | Conventional limited-recycle fuel cycle from LWR to LWR (Conv. LR) | | Limited-recycle fuel cycle from advanced reactor to LWR (LR-RU) | |
|---|---|---|---|---|
| | Stage 1 | Stage 2 | Stage 1 | Stage 2 |
| Reactor type | LWR | LWR | SFR | LWR |
| Charge fuel fissile | LEU | LEU + Pu | HALEU | RU |
| Fuel form | UOX | UOX and MOX | HALEU metal | UOX |
| Recycling fuel fraction in core | N/A | 30%[a)] | N/A | 100% |
| Recovered material | U/Pu | - | RU | - |
| Blending ratio with NU [b)] | - | 7–12% | - | 43.8% |
| Charge fuel uranium enrichment | 4.2% | ~0.7% | 16.4% | 4.4% |
| Avg. discharge burnup, GWd/t | ~50 | ~50 | ~100 | ~50 |

a) MOX fuel fraction in the core is limited to 30% or less, owing to reactivity control issues (see Appendix B).
b) Blending ratio indicates the mass fraction of recovered uranium or plutonium relative to the total HM in the fresh fuel.

In the second stage of the Conv. LR, the MOX fuel assembly fraction in the core is limited to 30%, and the Pu content in the MOX fuel is limited to 12% because of a reduced shutdown margin and a positive void reactivity feedback (see Appendix B). When recycling the RU from the HALEU reactor DF, however, those limitations are avoided because the second-stage fuel is UOX fuel, not MOX fuel. Thus, *the current LWR fleet can be used as the second-stage reactor of the limited-recycle system without technical and operational limitations*.

### 2.2.3 Analysis Example Reactor for Continuous Recycling

Advanced reactor concepts aiming at a CR fuel cycle require NU or DU as the external make-up feed at a break-even equilibrium state. However, those reactors require HALEU (or another external fissile supply) during the transition from the start-up core to the break-even equilibrium core. Figure 2.4 shows the tally of CR advanced reactor concepts, among the advanced reactor concepts considered in the present study, that require HALEU during the transition period. All four—the Gas-cooled Fast Reactor (GFR), Molten-salt Fast Reactor (MSFR), LFR, and SFR—are fast reactors. The thermal molten-salt reactor (MSR) concept has also been proposed as a CR nuclear system, but it was not considered in the present study because the thermal MSR requires a 5% LEU fuel during the transition.

The primary design parameters of the CR advanced reactors are summarized in Table 2.3 by grouping them into three fast reactor types: i.e., gas-cooled, molten-salt, and liquid-metal cooled fast reactors. The last column denotes the Analysis Example Reactor used to assess the HALEU utilization. Except for the start-up core fissile, all data in the table are obtained at the equilibrium state.

It is noted that the Analysis Example Reactor was designed to have a break-even breeding ratio by recycling RU and TRU without blanket fuels at an equilibrium state. The detailed physics analysis from the start-up core to the break-even equilibrium core has been made unnecessary in the present study by assuming that the Analysis Example Reactor can evolve to the break-even equilibrium core within a few cycles. Under the Global Nuclear Energy Partnership program, Kim et al. (2008) demonstrated the evolution of a fissile breeding ratio from low to break-even in a single reactor by changing fuel designs. For quantitative comparison of the HALEU utilization, it was assumed that the initial core is fully loaded with HALEU fuels and one-fifth of the core is replaced during each reload cycle. HALEU fuel is used for the first five cycles. Then, the recovered U/Pu fuel is used as the reloading fuel from the sixth cycle to the end of reactor lifetime (which is assumed to be 80 years). A lifetime-averaged HALEU requirement was then calculated and used in the present study. Detailed information is provided in Appendix B.

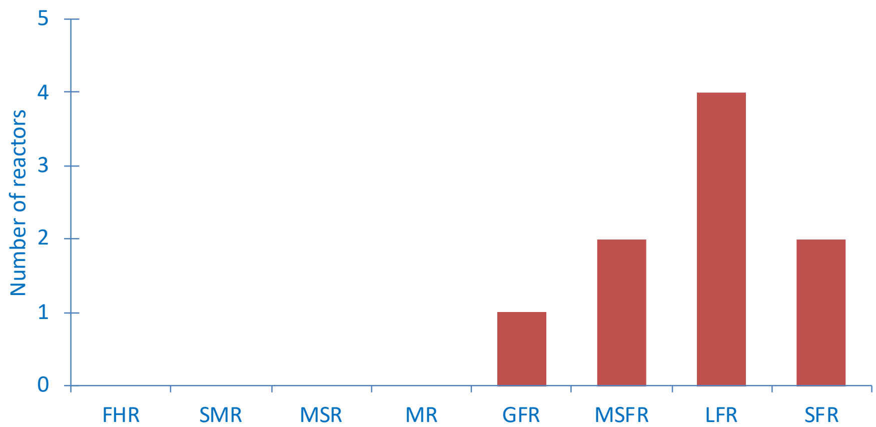


Figure 2.4 Tally of recycling fuel cycle advanced reactors requiring HALEU in the start-up core and initial reload cycles

Table 2.3 Primary reactor design parameters of continuous-recycle advanced reactor concepts

| | Gas-cooled fast reactor | Molten-salt fast reactor | Liquid-metal cooled fast reactor | Analysis Example Reactor |
|---|---|---|---|---|
| Reactor type | GFR | MSFR | LFR, SFR | SFR |
| Coolant | Gas | Chloride salt | Liquid metal | Sodium |
| Electric power output, MWe | < 600 | < 600 | < 600 | < 600 |
| Thermal efficiency, % | ~50 | ~45 | ~40 | 40 |
| Fuel form | TRISO with UCO, $UO_2$, UN kernels | Chloride salt | U-Pu-Zr, U-TRU-Zr, (U,TRU)N | U-TRU-Zr metal (no blanket) |
| External make-up feed at equilibrium cycle | NU or DU | NU or DU | NU or DU | NU |
| Fissile of start-up core | HALEU (12%) | Not claimed | HALEU (10–15%) | HALEU (13.5%) |
| Avg. discharge burnup, GWd/t | ~150 | Not claimed | 80–110 | ~100 |
| Discharge fuel content, %<br>- U-234/U<br>- U-235/U<br>- U-236/U<br>- U-238/U<br>- U/HM<br>- Pu/HM<br>- MA/HM | Not known | Not known | Not known | 0.009<br>0.9<br>0.2<br>98.9<br>84.7<br>14.7<br>0.6 |

# 3. Assessment of Pros and Cons of HALEU Utilization

This section evaluates the fuel cycle performance parameters of three example fuel cycles using Analysis Example Reactors. The primary design parameters of the three example fuel cycles are summarized in Table 3.1. For comparison purposes, the current once-through fuel cycle based on an LWR fleet with <5% LEU (OT-LWR), which is the Basis of Comparison in the present study, is also provided in the table.

Table 3.1 Comparison of fuel cycle data of example fuel cycles with HALEU

| | Once-through in LWR | Once-through in advanced reactor | Continuous recycle with HALEU start-up | Limited recycle | |
|---|---|---|---|---|---|
| | | | | 1st stage | 2nd stage |
| Analysis Example Reactor type | PWR | SFR | SFR | SFR | PWR |
| Fuel form | $UO_2$ | U-Zr metal [a)] | U-Pu-Zr metal | U-Zr metal | $UO_2$ |
| Capacity factor | 90% | 90% | 90% | 90% | 90% |
| Thermal efficiency | 33.3% | 40% | 40% | 40% | 33.3% |
| Charge material/ U-235 content | LEU/4.21% | HALEU/16.4% | RU+Pu/0.71 (HALEU/13.5%)[b)] | HALEU/16.4% | 56%(NU)44%(RU) /4.4% |
| Burnup, GWd/t | ~50 | ~100 | ~100 | ~100 | ~50 |
| Power sharing, % | - | - | - | 54.1 | 45.9 |
| HM mass per assembly, MT | 0.442 | 0.106 | 0.104 | 0.106 | 0.442 |

a) Impact of sodium bond is discussed in Section 4.2.
b) HALEU in parentheses indicates the fuel needed initially to start the reactor.

Sodium-bonded metallic fuel is used for the three example fuel cycles because it has been developed and qualified for commercial use. However, it is also recognized that a sodium-free metallic fuel is under development and is being considered as the future fuel for SFRs. Generally, both sodium-bonded and sodium-free fuels have the same front-end fuel cycle performance but different back-end fuel cycle performance because enough information does not current exist to evaluate the performance of direct disposal of sodium-bonded SNF [UFD 2014], and treatment to remove the bond sodium was assumed in this report to be consistent with past practice with EBR-II fuel [DOE 2000]. The impacts of sodium-bonded and sodium-free metallic fuels on the back-end fuel cycle are discussed in Section 4.2.

The fuel cycle parameters are normalized to the unit of electricity generation (GWe-year) at the equilibrium state. The material loss during fuel fabrication and reprocessing was ignored for simplification in comparing fuel cycle parameters. The fuel cycle cost data were obtained from the Advanced Fuel Cycle Cost Basis Report [Dixon et al. 2017], but the missing cost data (the HALEU enrichment cost, etc.) were assumed by combining the existing data with engineering judgments. The detailed fuel cycle cost data are summarized in Appendix C.

## 3.1 Front-end Fuel Cycle Parameters

The front-end fuel cycle consists of four stages:

1. Uranium mining and milling to generate yellowcake ($U_3O_8$),
2. Conversion to uranium hexafluoride ($UF_6$),
3. Uranium enrichment and deconversion, and
4. Fuel fabrication.

It is expected that the technologies and infrastructures currently available for uranium mining, milling, and conversion would be applicable to HALEU. However, new facilities or infrastructure designed and licensed for HALEU enrichment and fuel fabrication is needed because the uranium enrichment of HALEU is higher than the enrichment level achievable at the existing commercial enrichment facilities, and various HALEU fuel forms and designs are different from those of the current LWR fleet fuel.

Figure 3.1 shows the tallies of the fuel forms and enrichments of advanced reactor concepts discussed in this report. There is a notable variation in fuel forms. The TRISO fuels dominate in the once-through fuel cycles, followed by metallic fuels. The primary design purpose of the once-through advanced reactor concepts with TRISO and metallic fuels is to extend the discharge burnup and provide higher thermal efficiency. Oxide and molten-salt fuels with < 5% enriched uranium are used in LWR-based SMRs and thermal MSR concepts. Various fuel forms with NU are used as the external make-up feed of the CR fast reactors.

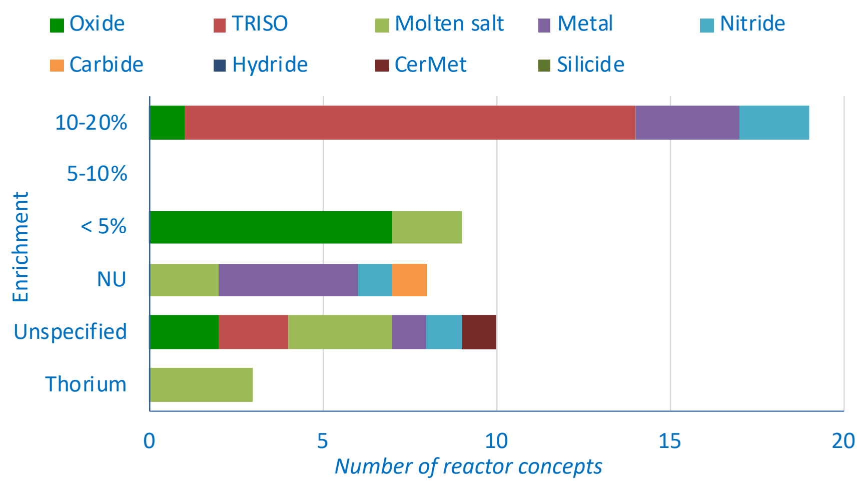


Figure 3.1 Advanced fuel forms and uranium enrichments

In the following section, NU demand, DU production, required SWUs for uranium enrichment, and required fuel fabrication mass are evaluated.

### 3.1.1 Natural Uranium Demand

Figure 3.2 shows the NU demand to support the three example fuel cycles. The DU assay was assumed to be 0.25% in the calculations. The NU demand increases with the fuel enrichment but decreases with burnup. Thus, the NU demand will be comparable to the Basis of Comparison when the burnup increases as much as the enrichment increases (while accounting for differences in thermal efficiency). However, the once-through fuel cycle with HALEU (OT-HALEU) requires a factor of 1.7 higher NU compared to the Basis of Comparison. This requirement indicates that the increase in burnup of the Analysis Example Reactor is not as great as the increase in the enrichment: i.e., the enrichment of the HALEU fuel is a factor of 3.9 higher compared to the Basis of Comparison, but its burnup is only a factor of 2 higher.

The CR fuel cycle requires a small mass of NU (or DU) as make-up feed and the HALEU to fill the initial core and reload for a few cycles (5 cycles in the present work) initially. For the limited recycle with recovered uranium (LR-RU) fuel cycle, the NU demand is comparable to that of the Basis of Comparison (OT-LWR) because the second-stage LWR uses the down-blended RU rather than external enriched uranium.

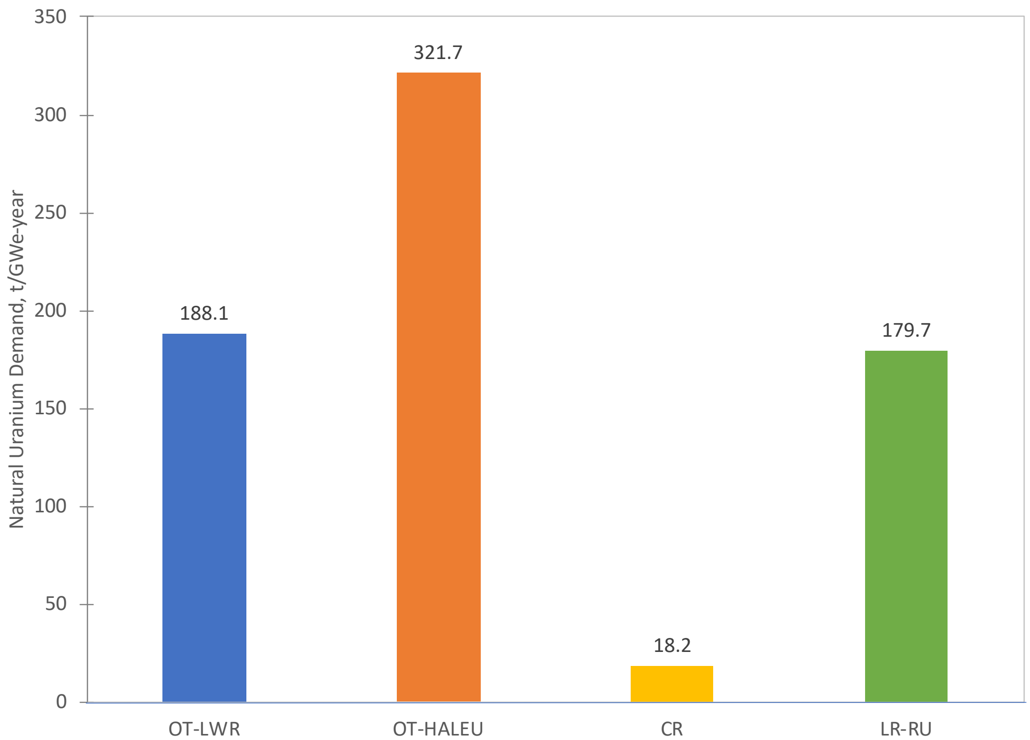


Figure 3.2 Natural uranium demand for example fuel cycles

### 3.1.2 Enrichment

#### *3.1.2.1 Enrichment System for Commercial-scale Production of HALEU*

The current uranium enrichment facilities have been licensed for up to 5% enrichment for the commercial LWR fleet. The enrichment range is likely to evolve to 5–10% for accident-tolerant fuels and further extension of the conventional LWR burnup. Higher-enrichment (20% or higher) uranium has been fabricated on a small scale for defense purposes or special reactors such as test reactors.

In accordance with the classification of fuel facilities by the U.S. Nuclear Regulatory Commission (NRC), a facility handling more than 10 kg of 10%- to 19.7 %-enriched uranium is classified as a ***Category-II*** or moderate strategic significance facility, while a fuel facility with less than 10% enrichment is classified as a ***Category-III*** or low strategic significance facility [NRC 2020]. An enrichment facility in Category II requires more stringent security requirements than the current commercial enrichment facilities in Category III. Category-II enrichment would therefore be expected to incur a higher cost. Thus, to save costs, the future HALEU enrichment infrastructure is anticipated to minimize enrichment in Category-II facilities.

Future enrichment infrastructure could take on different configurations. Figure 3.3 identifies three general configurations that could be envisioned. The most straightforward configuration is the first, where all enrichment from NU to HALEU is performed in a Category-II facility with a single-step enrichment infrastructure. This configuration maximizes the fraction of SWUs in Category-II facilities. The next is a two-step enrichment infrastructure consisting of a commercial Category-III facility (< 5%) and a Category-II facility (10–19.75%). The third is a three-step enrichment infrastructure, which consists of the current commercial Category-III facility (<5%), an intermediate Category-III facility to produce 5–10% enriched uranium, and the Category-II facility. In the two- or three-step enrichment infrastructures, the enriched uranium produced from a lower-enrichment facility is used in a higher-enrichment facility as feed uranium, and the tail uranium (TU) discharged from a higher-enrichment facility is used in a lower-enrichment facility as additional feed uranium. The multi-step enrichment infrastructure would reduce the size of the Category-II facility and the associated impact of the higher-security facility on total enrichment costs. The impacts of the envisioned enrichment infrastructure are discussed in the next section, and the cost variations associated with different enrichment infrastructures are compared in Appendix C.

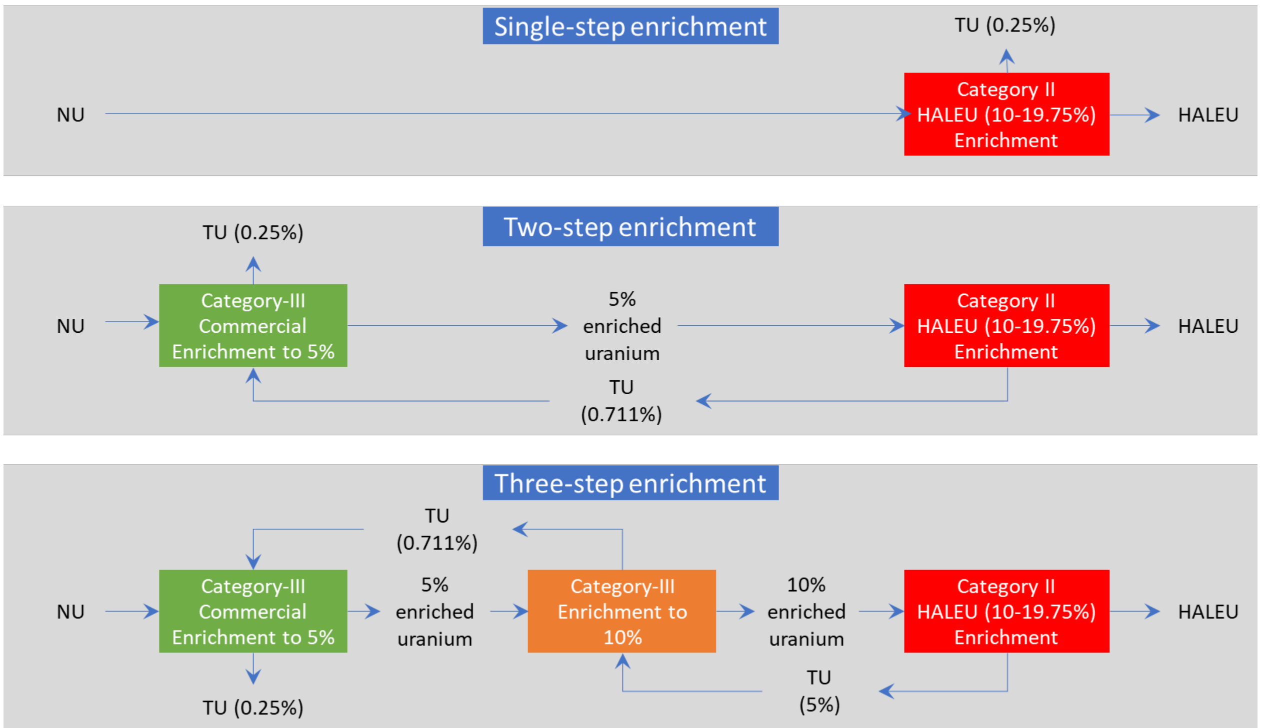


Figure 3.3 Overview of potential future HALEU enrichment infrastructures

#### *3.1.2.2 Enrichment and Deconversion Efforts*

The enrichment efforts for the three example fuel cycles are compared in Table 3.2. The enrichment effort was measured by the SWUs per unit of electricity generation, and the total DU produced from the enrichment is also provided in the table.

The Basis of Comparison (OT-LWR) requires an enrichment effort of 137.2 t-SW/GWe-year. The OT-HALEU requires a much higher enrichment effort (306.2 t-SW/GWe-year) regardless of the number of enrichment stages. However, SWUs in the Category-II facility decrease significantly in the multi-stage enrichment systems. SWUs in the Category-II facility are 40.0 t-SW/GWe-year and 4.5 t-SW/GWe-year in two- and three-stage enrichment infrastructures, respectively, which are 13.1% or 1.5% of the total SWUs. This finding reveals that the required capacity of the Category-II enrichment facilities would be reduced if the cascade enrichment infrastructures were utilized, as shown in Figure 3.3.

The CR fuel cycle requires an enrichment effort of 15.7 t-SW/GWe-year because HALEU is needed to start the reactor initially. The total SWUs of the limited fuel cycle decrease to 165.6 t-SW/GWe-year because of the fractional energy generated in the second-stage advanced LWR without additional enriched uranium. A similar trend is observed in the LR-RU: i.e., SWUs in the Category-II facility are about 13.1% and 1.5% of the total SWUs for two- and three-stage infrastructure, respectively.

To dispose of the DU (~0.25%) produced in the enrichment facilities, a deconversion of depleted $UF_6$ to $U_3O_8$ or $UF_4$ is required because $UF_6$ is soluble and produces hazardous chemical and radiological exposures. The production of DU from the three fuel cycles is quantified in Table 3.2. The DU production rates are proportional to the SWUs. Compared to the Basis of Comparison, the DU production rate from the once-through fuel cycle is a factor of two higher, but it is comparable or much smaller for the limited-recycle or CR fuel cycle. The current deconversion facilities can be utilized by extending their capacity if needed.

Table 3.2 Comparison of SWUs and DU

<table>
<tr><th colspan="2" rowspan="2"></th><th rowspan="2">NU demand, t/GWe-year</th><th colspan="4">SWUs, t-SW/GWe-year</th><th rowspan="2">DU production, t/GWe-year</th></tr>
<tr><th><5%</th><th>5–10%</th><th>10–19.75%</th><th>Total</th></tr>
<tr><td colspan="2">OT-LWR</td><td>188.1</td><td>137.2</td><td></td><td></td><td>137.2</td><td>166.2</td></tr>
<tr><td rowspan="3">OT-HALEU</td><td>Single-stage</td><td rowspan="3">321.7</td><td></td><td></td><td>306.2</td><td rowspan="3">306.2</td><td rowspan="3">312.5</td></tr>
<tr><td>Two-stage</td><td>266.2</td><td></td><td>40.0</td></tr>
<tr><td>Three-stage</td><td>266.2</td><td>35.6</td><td>4.5</td></tr>
<tr><td rowspan="3">CR[a)]</td><td>Single-stage</td><td rowspan="3">14.0</td><td></td><td></td><td>11.8</td><td rowspan="3">15.7</td><td rowspan="3">16.3</td></tr>
<tr><td>Two-stage</td><td>13.9</td><td></td><td>1.8</td></tr>
<tr><td>Three-stage</td><td>13.9</td><td>1.7</td><td>0.1</td></tr>
<tr><td rowspan="3">LR-RU</td><td>Single-stage</td><td rowspan="3">179.9</td><td></td><td></td><td>165.6</td><td rowspan="3">165.6</td><td rowspan="3">169.1</td></tr>
<tr><td>Two-stage</td><td>144.0</td><td></td><td>21.7</td></tr>
<tr><td>Three-stage</td><td>144.0</td><td>19.2</td><td>2.4</td></tr>
</table>

a) HALEU is required to start the reactor and provide fissile for a few initial cycles.

### 3.1.3 Charge Fuel Mass

The charge fuel masses per unit of electricity generation for the example fuel cycles in the equilibrium state are compared in Figure 3.4. In the figure, the reprocessed fuel means the fuel fabricated using the RU and plutonium with the make-up feed with natural (or depleted) uranium. Compared to the Basis of Comparison (OT-LWR), the advanced reactor concepts require smaller charge fuel masses because the burnup is a factor of two higher. The charge fuel masses of the OT-HALEU and the CR fuel cycle (CR-U/TRU) are about 9.1 t/GWe-year, and the LR-RU requires 15.0 t/GWe-year. The charge fuel masses would be reduced further by increasing the burnup or thermal efficiency. The impact of the higher burnup is discussed in Section 4.1.

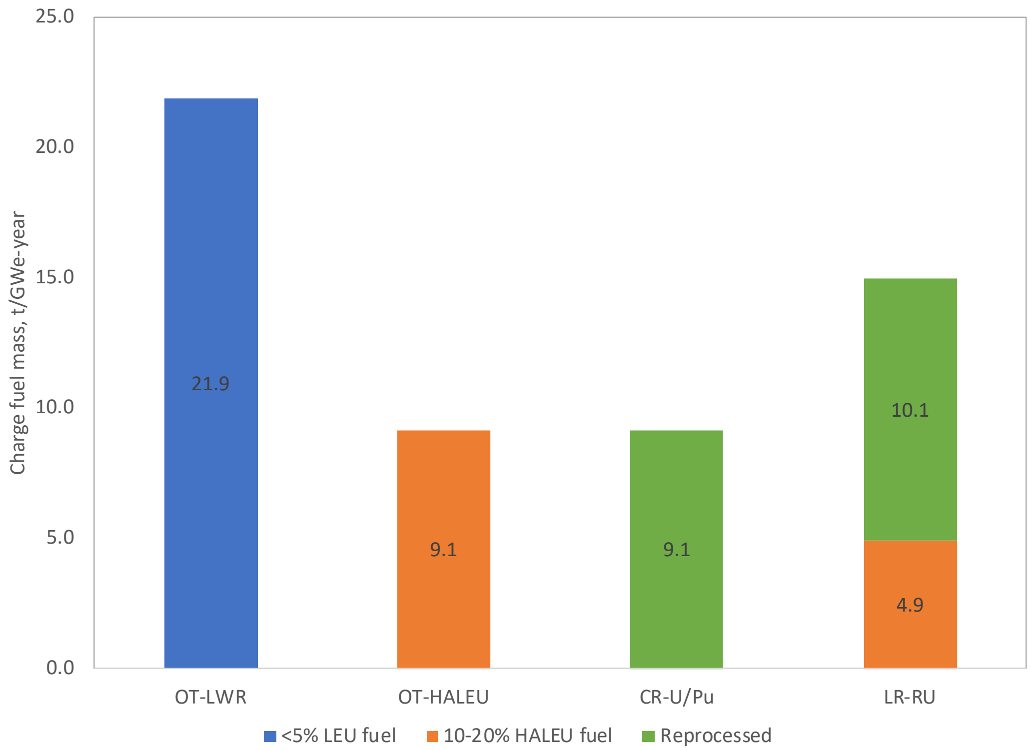


Figure 3.4 Charge fuel masses of example fuel cycles

## 3.2 Back-end Fuel Cycle Parameters

The back-end fuel cycle consists of three stages:

1. Interim storage of DF,
2. Recycling and fabrication of used nuclear fuel if necessary, and
3. Disposal of radioactive wastes.

This section discusses the DF characteristics (mass and radiotoxicity), mass of high-level waste (HLW), and mass of fuel fabrication using the recovered actinides in limited-recycle and CR fuel cycles. It is noted that the material losses during the fuel reprocessing and fuel fabrication were ignored in the present study for relative comparison purposes between fuel cycles. However, about 1–2% material losses are expected in the back-end fuel cycle of the limited-recycle and CR fuel cycles.

### 3.2.1 Impact of Discharge Fuel on Interim Storage

The mass, radioactivity, and decay heat of the DFs from the example fuel cycles are estimated to assess the interim storage impact. The DF masses are the same as the charge fuel masses shown in Figure 3.4, and the radioactivity and decay heat per assembly at the time of charge and discharge, and five years after discharge, are compared in Figure 3.5 and Figure 3.6, respectively. Because the interim storage capacity is dictated by the radioactivity and decay heat per DF assembly, the values in the figures are the radioactivity and decay heat per DF assembly.

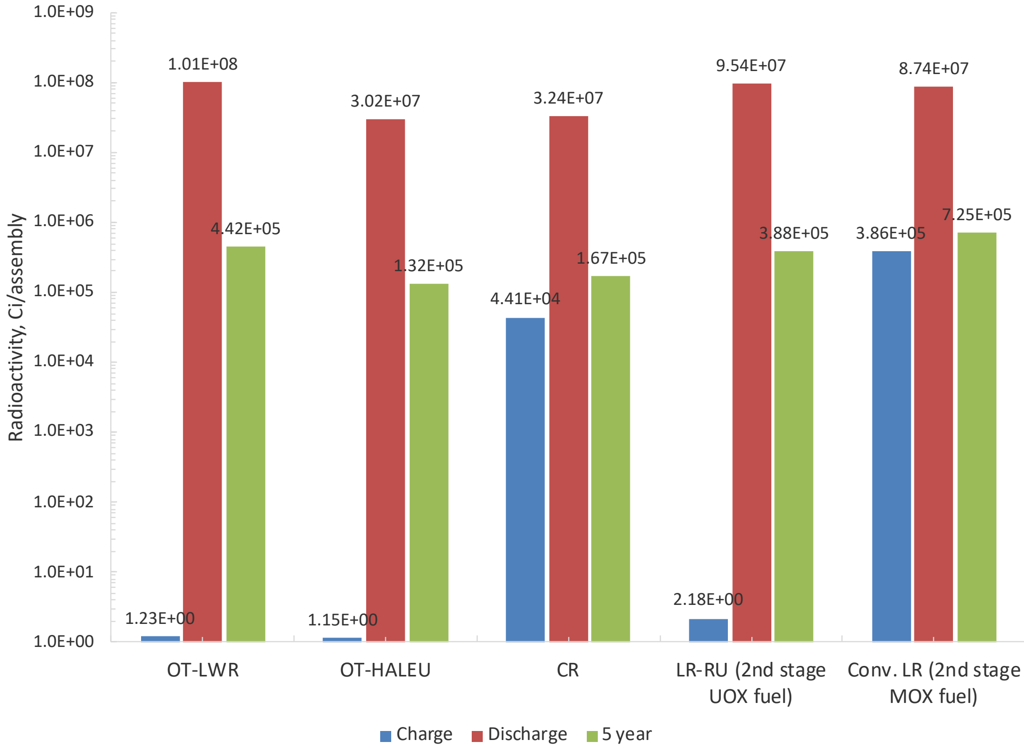


Figure 3.5 Comparison of radioactivity per assembly for example fuel cycles

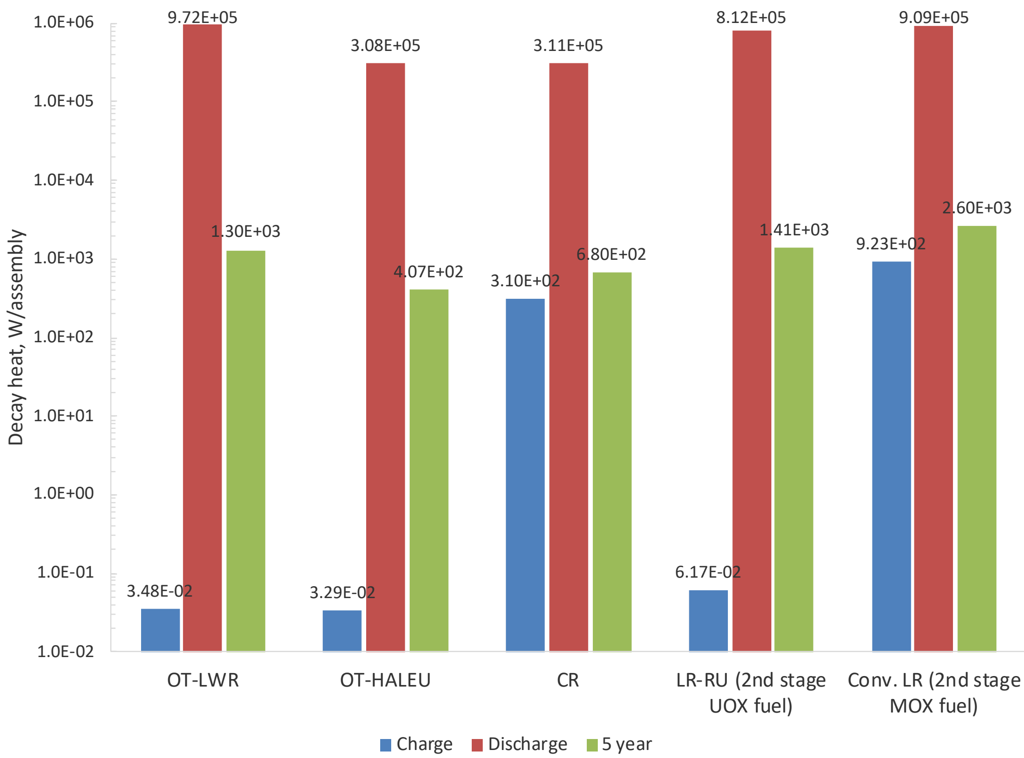


Figure 3.6 Decay heat comparison for example fuel cycles

For limited recycle, the radioactivity and decay heat of the second-stage DF are provided because the first-stage DF data are identical to those for the once-through fuel cycles. The post-irradiation cooling time in interim storage is dependent on the design of the disposal waste canister and repository requirements. To standardize the comparisons in the present study, it was assumed that the DFs are stored in interim storage for five years, then sent to the disposal site or reprocessing facilities. For comparison purposes, the Basis of Comparison (OT-LWR) and the Conv. LR are also shown in the figures.

Both radioactivity and decay heat show the same trend. For the charge fuels, the OT-HALEU and the LR-RU have low radioactivity and decay heat because the charge fuels consist of uranium (no plutonium). However, the charge fuels of the CR and Conv. LR fuel cycles have higher radioactivity and decay heat because of plutonium in the charge fuel. The discharge burnup of the advanced reactors is a factor of 2 higher than that of the Basis of Comparison, but its HM mass per assembly is a factor of 4 smaller (see Table 3.1). Therefore, DFs from advanced reactors have lower radioactivity and decay heat per discharge assembly compared to the Basis of Comparison. This observation indicates that the current interim storage and packaging technologies are adequate to hold the DF assemblies from the advanced reactor with HALEU.

### 3.2.2 Recycling and Fuel Fabrication of Used Nuclear Fuels

Among the three example fuel cycles, two fuel cycles require the reprocessing of used nuclear fuels: CR and limited recycle. In Table 3.3, the fuel compositions of the LR-RU are compared with the Conv. LR. In the limited recycle considered in the present study, the U-235 content in the first-stage HALEU DF is 9.2%, and it is down-blended to an enrichment of about 4.4% for the second-stage LWR fuel. The RU from the electrochemical pyroprocess is in a metallic form. The recovered metal can be stored but should be converted to $UF_6$ or $U_3O_8$ for down-blending with NU (or DU) [IAEA 2007]. The current conversion facility can be used to convert the RU, and there is no technical barrier to down-blending the RU with NU (or DU).

Table 3.3 Fuel composition of limited-recycle fuel cycles

| | Conventional limited-recycle fuel cycle from LWR to LWR (Conv. LR) | | | | Limited-recycle fuel cycle from advanced reactor to LWR (LR-RU) | | | |
|---|---|---|---|---|---|---|---|---|
| | Stage 1 | | Stage 2 | | Stage 1 | | Stage 2 | |
| Reactor type | LWR | | LWR | | SFR | | LWR | |
| Fuel form | UOX | | MOX | | U-Zr | | UOX | |
| Power sharing, % | 90.8 | | 9.2 | | 54.1 | | 45.9 | |
| Burnup, GWd/t | 50.0 | | 50.0 | | 100.0 | | 50.0 | |
| Charge fuel, t/GWe-yr | 10.9 | | 10.9 | | 4.9 | | 10.1 | |
| Content, % | Charge | Discharge | Charge | Discharge | Charge | Discharge | Charge | Discharge |
| U-234/U | 0.04 | 0.02 | 0.02 | 0.02 | 0.16 | 0.14 | 0.06 | 0.05 |
| U-235/U | 4.20 | 0.71 | 0.71 | 0.40 | 16.34 | 9.20 | 4.4 | 0.7 |
| U-236/U | - | 0.59 | - | 0.07 | - | 1.81 | 0.8 | 1.3 |
| U/HM | 100.0 | 98.5 | 90.2 | 99.51 | 100.0 | 94.5 | 100.0 | 94.5 |
| Pu/HM | - | 1.34 | 9.80 | 6.54 | - | 5.27 | - | 1.34 |

The second-stage LWR fuel of the limited recycle is similar in fuel composition to the UOX fuel of the current LWR fleet, except for the presence of U-236 and the slightly higher U-235 content. U-236 is bred in the first stage primarily from a U-235 (n,$\gamma$) reaction and $\alpha$-decay of Pu-240, but its radiological impact during fuel fabrication is insignificant because of its long half-life, and its daughter (Th-232) also has very low activity.

The U-232 content in the HALEU DF has not been traced in the reactor performance analysis because its impact on neutronics performance is negligible. U-232 is created by alpha-decay of Pu-236 (half-life of 2.85 years), and Pu-236 is approximately proportional to the initial content of U-235 and U-236. The U-232 content in a LWR DF is in the range of 20–30 ppb and increases during post-irradiation cooling [Arthur 1977]. The U-232 content is small, but it may affect the reprocessing of used fuel or fuel fabrication using the RU because the U-232 decay chain contains a strong gamma emitter (Tl-208). It is expected that the U-232 content in the HALEU DF is higher than that in the LWR DF because of higher U-235 content in the HALEU fuels. The isotopic impact on reprocessing and non-proliferation will be evaluated in the future.

### 3.2.3 Radioactive Wastes for Disposal

In the present study, the spent nuclear fuel (SNF) and HLW generated from advanced reactors or reprocessing facilities are considered as the radioactive wastes for permanent geologic disposal.

Analysis Example Reactors with the three example fuel cycles use sodium-bonded metallic fuels. It is noted that the sodium-bonded metallic fuel was selected because it is currently qualified up to an average burnup of ~100 GWd/t and is licensable [Crawford et al. 2007]. A sodium-free fuel is currently under development by the Advanced Fuel Campaign [Richardson 2019] and TerraPower [Neider and Hejzlar 2021] to achieve high burnup and enable direct disposal of the SNF.

Per [UFD 2014], “Enough information does not current exist to evaluate the performance of direct disposal of sodium-bonded SNF in any geologic disposal concept. This waste type may require treatment regardless of the disposal concept.” DOE decided to treat the sodium-bonded metallic fuels discharged from EBR-II using electrometallurgical treatment (EMT) [DOE 2000]. The recovered materials and byproducts from EMT are considered HLW, including the salt waste containing FPs and non-recovered actinides and metal waste containing cladding and Zr in the fuel. The detailed data and calculation models for SNF production and HLW generation are provided in Appendix D.

The radioactive waste production rates and disposal volumes are summarized in Table 3.4. The disposal volume is defined by the total volume of waste canisters needed to accommodate all the SNF or HLW, which is consistent with the waste volume calculated by the Used Fuel Disposition (UFD) campaign [UFD 2014]. The UFD campaign estimated that 32 PWR assemblies could be accommodated in a single Dual-Purpose Canister (DPC).

For the Basis of Comparison, the disposal volume is ~25 $m^3$ for disposal of 21.9 t of SNF using a 5.13-m-tall DPC with a 2-m outer diameter, which is consistent with the UFD estimation. The DF mass of the OT-HALEU and CR reactors is 58% smaller compared to the Basis of Comparison, but its disposal volume is about 19% smaller because the HLW volume was increased by the EMT process. The limited recycle generates HLW and SNF from the first and second stages, respectively. Total radioactive waste mass and volume are comparable to those of the Basis of Comparison.

The radioactive waste mass and disposal volume of sodium-bonded metallic fuel and sodium-free metallic fuel are compared in Table 3.5. In the comparison, burnup was assumed to be 100 GWd/t for both fuels. The sodium-bonded fuel produces more radioactive waste mass because all byproducts and salt from the EMT process are considered as HLW. However, the sodium-bonded fuels result in a 20% smaller disposal volume. It is noted that the fuel assemblies and fuel pins of sodium-bonded fuel are chopped prior to the EMT process, while the whole fuel assemblies of sodium-free fuel are packed into the waste canister for direct disposal of SNF. Thus, the EMT products of sodium-bonded fuel can be packed into a smaller canister compared to a whole SNF assembly of the sodium-free fuel. The container design is related to the requirements of temperature, radiotoxicity, criticality issues, and the post-irradiation cooling time of the DF, but a redesign of the EMT product container has not been attempted because it is beyond the scope of this study.

Table 3.4 SNF and HLW mass and disposal volume production rates for sodium-bonded fuels

| | OT-LWR | OT-HALEU | CR | LR-RU | |
|---|---|---|---|---|---|
| | | | | 1st stage | 2nd stage |
| Discharge fuel, t/GWe-year | 21.9 | 9.1 | 9.1 | 4.9 | 10.9 |
| SNF or HLW generation rate, t/GWe-year | | | | | |
| SNF mass | 21.9 | - | - | - | 10.9 |
| HLW mass a) | - | 20.1 | 20.1 | 10.9 | - |
| Volume for disposal, $m^3$/GWe-year | | | | | |
| SNF b) | 25.0 | - | - | - | 11.5 |
| HLW c) | - | 19.7 | 19.7 | 10.6 | - |
| Total, $m^3$/GWe-year | 25.0 | 19.7 | 19.7 | 21.5 | |

a) HLW mass production rate of 2.20 t per treatment of 1 metric ton metallic fuel [UFD 2014].
b) DPC volume is 1.14 $m^3$ to accommodate one metric ton of PWR SNF [UFD 2014].
c) ANL canister volume is 0.98 $m^3$ to accommodate one metric ton of HLW [UFD 2014].

Table 3.5 Comparison of radioactive waste production rates between sodium-bonded and sodium-free fuels

| | Sodium-bonded fuel | Sodium-free fuel |
|---|---|---|
| Waste type | HLW – products from EMT | SNF – DF assemblies |
| Waste mass, t/GWe-year | | |
| - OT-HALEU | 20.1 | 9.1 |
| - CR | 20.1 | 9.1 |
| - LR-RU, 1st stage only | 10.9 | 4.9 |
| Disposal volume, $m^3$/GWe-year | | |
| - OT-HALEU | 19.7 | 24.0 |
| - CR | 19.7 | 24.5 |
| - LR-RU, 1st stage only | 10.6 | 13.0 |

## 3.3 Levelized Cost of Fuel

The fuel cycle costs of the example fuel cycles are compared in this section. The fuel cycle cost was calculated by assuming that all fuel cycle facilities are operating on a commercial scale at an equilibrium steady state. There will be significant fuel-cycle cost variation from the first-of-a-kind (FOAK) to the nth-of-a-kind (NOAK) deployments; the cost variations from FOAK to NOAK are not considered in the present study, For a relative comparison of the fuel cycle costs of the example fuel cycles, all reactor-related costs were excluded and only fuel-cycle-related costs were considered. These costs are collectively referred to as the Levelized Cost of Fuel (LCF).

The LCFs of the three fuel cycles were calculated using the unit cost and the distribution data taken from the latest Advanced Fuel Cycle Cost Basis Report [Dixon et al. 2017]. The results were calibrated with 2021 dollars ($2021) by adjusting the inflation rate. For missing data, the cost was developed by using engineering judgment. For instance, the missing enrichment cost in a Category-II enrichment facility was assumed by applying an engineering-judged multiplier to the enrichment cost in a current commercial enrichment facility. The detailed cost data are provided in Appendix C, and the impacts of the multiplication factors are discussed in Section 4.3.The mean LCF values calculated by a random sampling technique are compared in Figure 3.7 and the itemized LCF values are provided in

Table 3.6.

Compared to the Basis of Comparison, the total LCF of the OT-HALEU is a factor of 2.2 higher. The front-end fuel cycle cost is higher by a factor of ~2 because of increased NU demand and SWUs. A smaller back-end fuel cycle cost was expected because of the smaller DF mass (from higher burnup). However, the back-end fuel cycle cost increases by a factor of ~3 due to the cost of separating bond sodium from the metallic fuel through the EMT process. The impact of sodium-free metal fuel on the back-end fuel cycle cost is discussed in Section 4.2. The LCF of the CR is about 10% smaller than that of the Basis of Comparison, which is the uncertainty of the unit costs (see Appendix C). The back-end fuel cycle cost is a factor of ~3 higher, but the front-end fuel cycle cost is reduced significantly. The LCF of the LR-RU is about 40% higher compared to the Basis of Comparison because of the increased front-end fuel cycle cost.

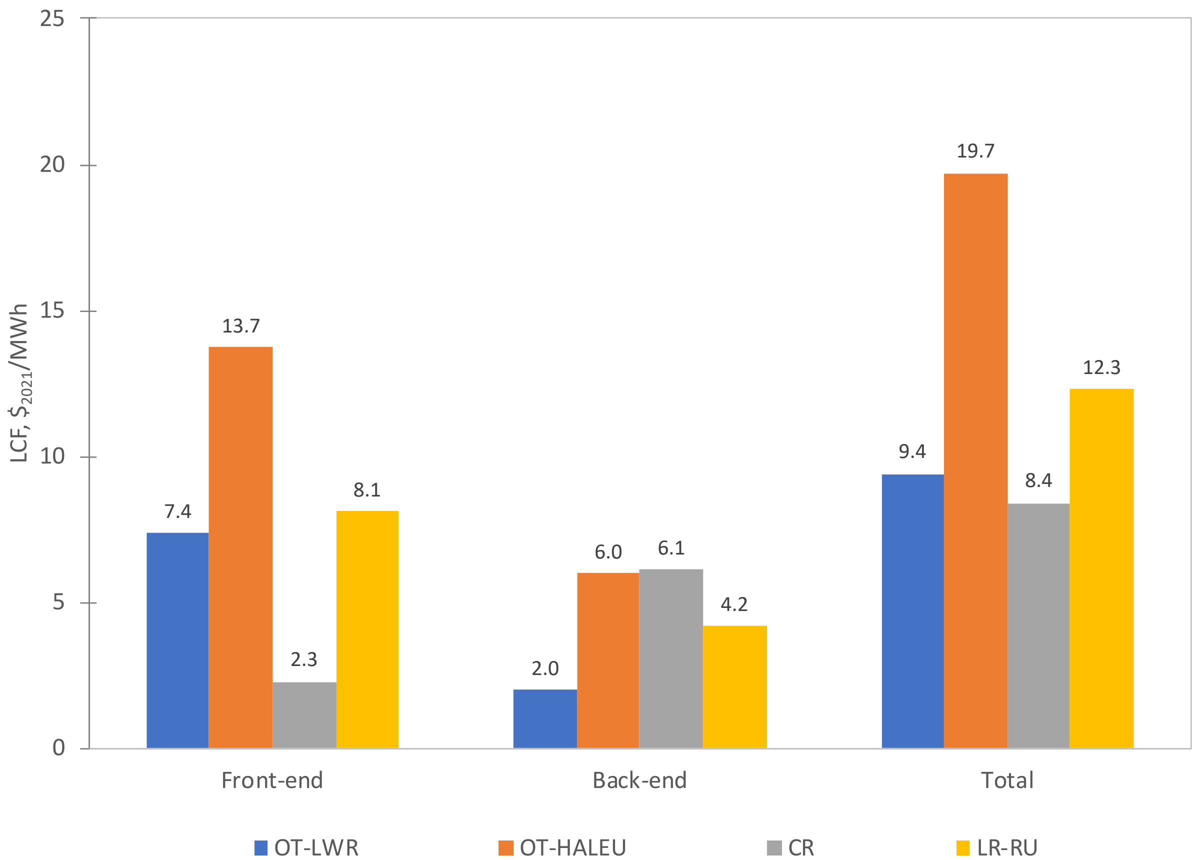


Figure 3.7 Comparison of Levelized Cost of Fuel for example fuel cycles

Table 3.6 Breakdown of LCF contributors

| | | | OT-LWR | OT-HALEU | CR | LR-RU |
|---|---|---|---|---|---|---|
| Front-end, $K/GWe-year | Natural uranium | | 3.40 | 5.82 | 0.31 | 3.25 |
| | Conversion | | 0.32 | 0.54 | 0.03 | 0.30 |
| | Enrichment | <5% enrichment facility | 2.23 | 4.33 | 0.23 | 2.34 |
| | | 5–10% enrichment facility [a)] | - | 0.79 | 0.04 | 0.43 |
| | | 10–19.75% enrichment facility [b)] | - | 0.17 | 0.005 | 0.09 |
| | Deconversion | | 0.14 | 0.26 | 0.01 | 0.14 |
| | DU disposition | | 0.14 | 0.26 | 0.01 | 0.14 |
| | UOX fuel fabrication with LEU (<5%) | | 1.14 | - | - | - |
| | Metal fuel fabrication with HALEU | | - | 1.56 | - | 0.84 |
| | Metal fuel fabrication with RU/Pu | | - | - | 1.66 | - |
| | UOX fuel fabrication with RU | | - | - | - | 0.57 |
| Back-end, $K/GWe-year | Down-blending [c)] | | - | - | - | 0.02 |
| | EMT processing & waste forms | | - | 3.09 | 3.09 | 1.67 |
| | LWR SNF packaging | | 0.36 | - | - | 0.16 |
| | Advanced reactor SNF packaging | | - | 0.26 | 0.26 | 0.14 |
| | EMT-derived HLW packaging | | - | 2.05 | 2.15 | 1.11 |
| | LWR SNF disposal | | 1.67 | - | - | 0.77 |
| | EMT-derived HLW disposal | | - | 0.60 | 0.62 | 0.32 |
| Total, $K/GWe-year | | | 9.4 | 19.7 | 8.4 | 12.3 |

a) Unit enrichment cost ($/kg-SW) was multiplied by a factor of 1.2 to the unit cost of <5% enrichment.
b) Unit enrichment cost ($/kg-SW) was multiplied by a factor of 2.0 to the unit cost of <5% enrichment.
c) Unit enrichment cost ($/kg-HM) was multiplied by a factor of 2.0 to the unit cost of deconversion.

## 3.4 Cumulative Fuel Cycle Parameters to Support Decarbonization

The SA&I campaign has projected that the U.S. electricity generation capacity will achieve a net-zero carbon emissions economy by 2050 [Dixon et al. 2021]. Assuming that the decarbonization effort results in electrification of whole end-use sectors, including power, residential, industrial, and transportation, by 2050, the total electricity demand in the United States was projected to increase by a factor of two compared to current electricity generation. In the projection, most of the fossil-based energy capacity decreases significantly. It is replaced by low-carbon-emission energies, and the nuclear energy capacity grows to ~250 GWe by 2050, which is about 24% of total electricity generation in 2050.

The projected nuclear energy capacity by 2050 is plotted in Figure 3.8 [Kim et al. 2022]. The green color denotes the legacy LWR capacity assuming the reactor lifetimes are extended to 80 years, and the orange color indicates the capacity addition attributable to new reactors. Assuming an average capacity factor of 90%, the accumulated electricity generation from 2021 to 2050 would be 2,482 GWe-year and 1,286 GWe-year from the legacy LWRs and new reactors, respectively. The new reactors could be advanced LWRS, LWR-based SMRs, non-LWR-based advanced reactors such as ARDP reactors, etc. In the present study, it was assumed that 50% of new reactors are non-LWR-based advanced reactors with HALEU. Thus, 643 GWe-year of electricity will be generated by non-LWR-based advanced reactors with HALEU fuels, and the remaining 643 GWe-year will be generated by LWR-based reactors with <5% LEU fuels.

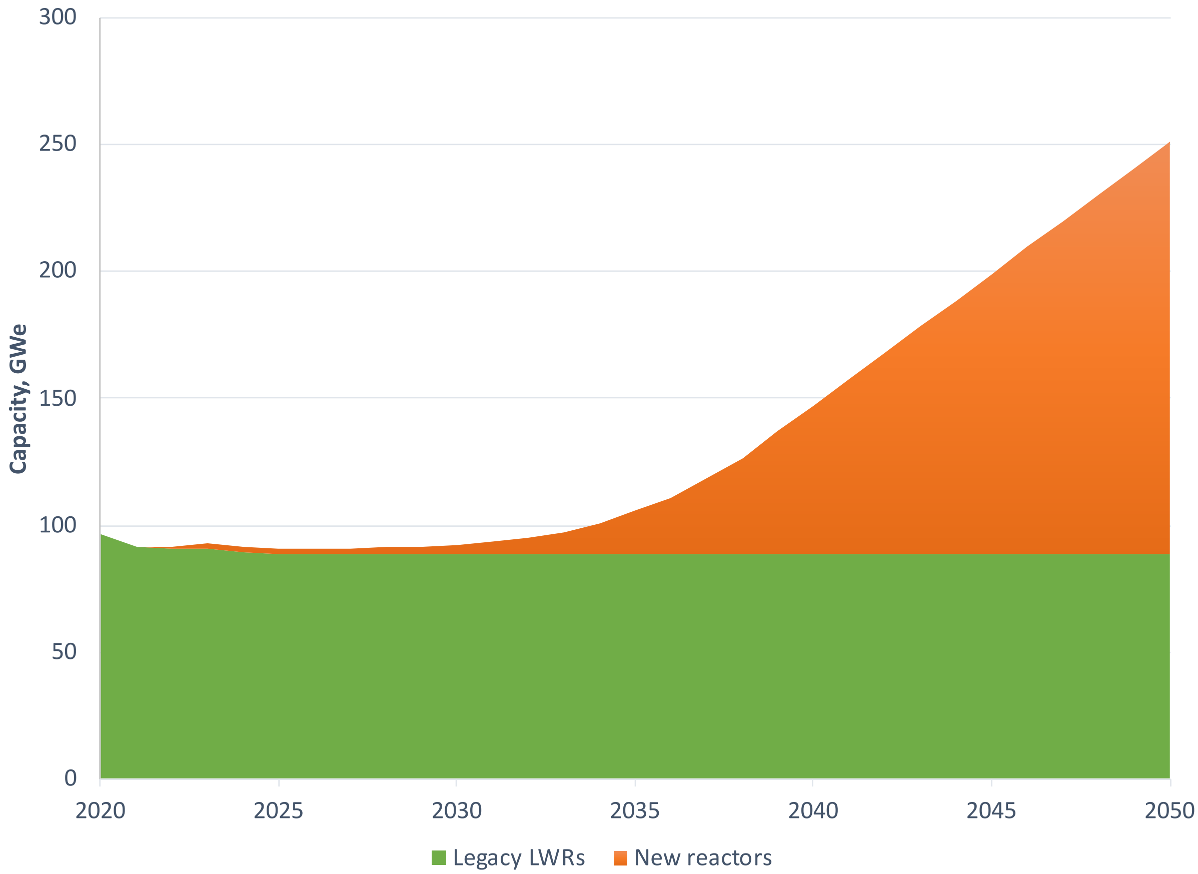


Figure 3.8 Projection of nuclear energy capacity to support non-zero-emissions economy by 2050

Based on the nuclear capacity projection, the fuel cycle parameters were calculated *only for the electricity generation by new reactors (orange color in Fig. 3.8)*. To generate the additional 1,286 GWe-year of electricity from 2021 to 2050, the required HALEU masses are calculated as follows:

$$M_{cum}^{HALEU} = m_{per\ electric}^{HALEU} \times E_{cum}^{new\ reactor},$$

where $m_{per\ electric}^{HALEU}$ is the HALEU mass per unit of electricity generation (t/GWe-yr) and $E_{cum}^{new\ reactor}$ is the cumulative electricity generation (GWe-year) from 2021 to 2050. For the OT-HALEU and LR-RU fuel cycles, the HALEU masses per unit of electricity generation are provided in Figure 3.4. For the CR fuel cycle, HALEU is required initially to start the CR reactors. The average HALEU mass required from 2021 to 2050 is 0.56 t/GWe-year. The required HALEU masses are 5,867 t, 378 t, and 3,174 t for the OT-HALEU, CR, and LR-RU fuel cycles, respectively. The other fuel cycle parameters accumulated by 2050 are compared in Figure 3.9.

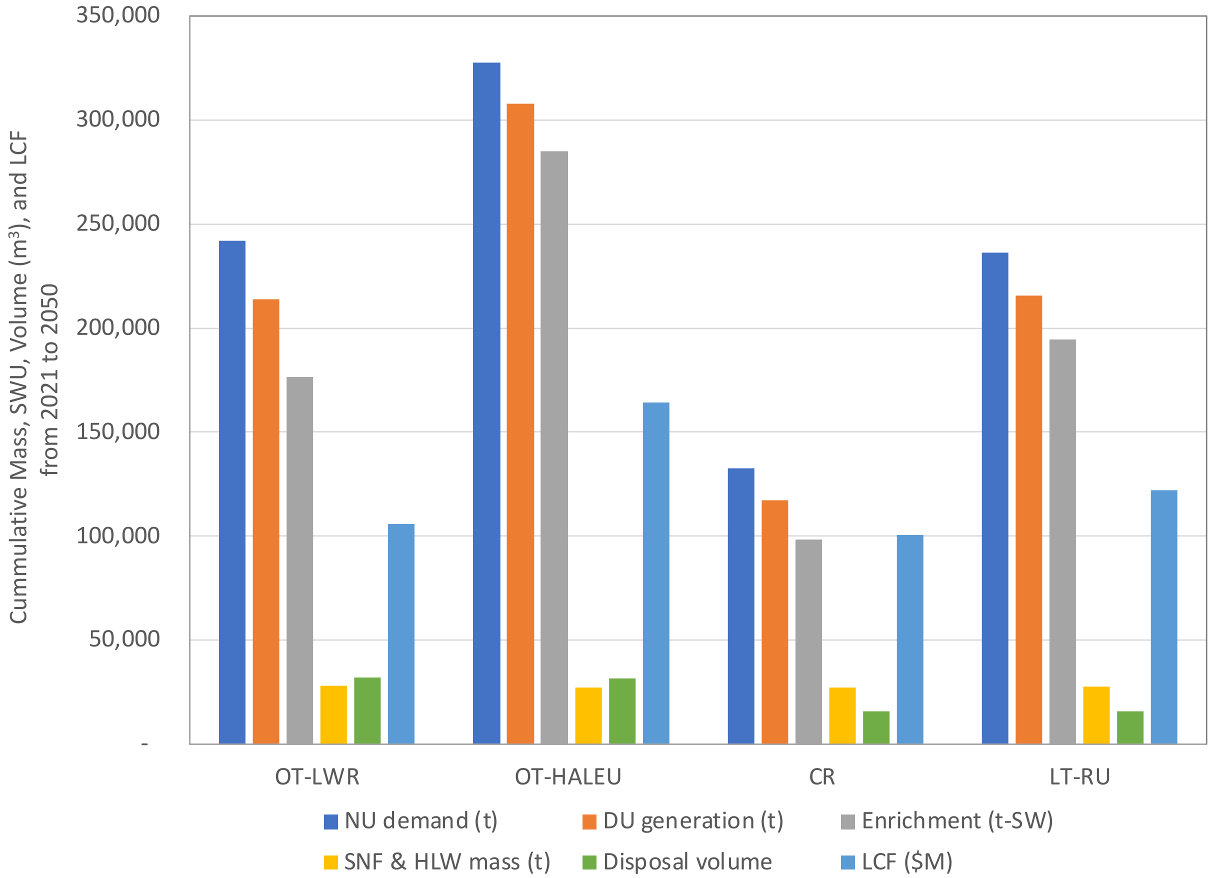


Figure 3.9 Fuel cycle parameters for new reactors

# 4. Sensitivity Study of Once-through Fuel Cycle Reactor Parameters

Figure 4.1 shows the fuel cycle parameters of the OT-HALEU relative to the Basis of Comparison. The front-end and back-end fuel cycle LCFs are a factor of two and three higher, respectively, compared to the Basis of Comparison. The primary reasons for the higher LCFs are the higher NU and SWU demand, and additional treatment of the DF before disposal. Thus, to reduce the overall LCFs, design variations are needed to reduce the NU and SWU demand and treatments of the DF. The impacts of design variations are evaluated through a sensitivity study. In particular, the impacts of variations of burnup, enrichment, and fuel form on the OT-HALEU are discussed in this section.

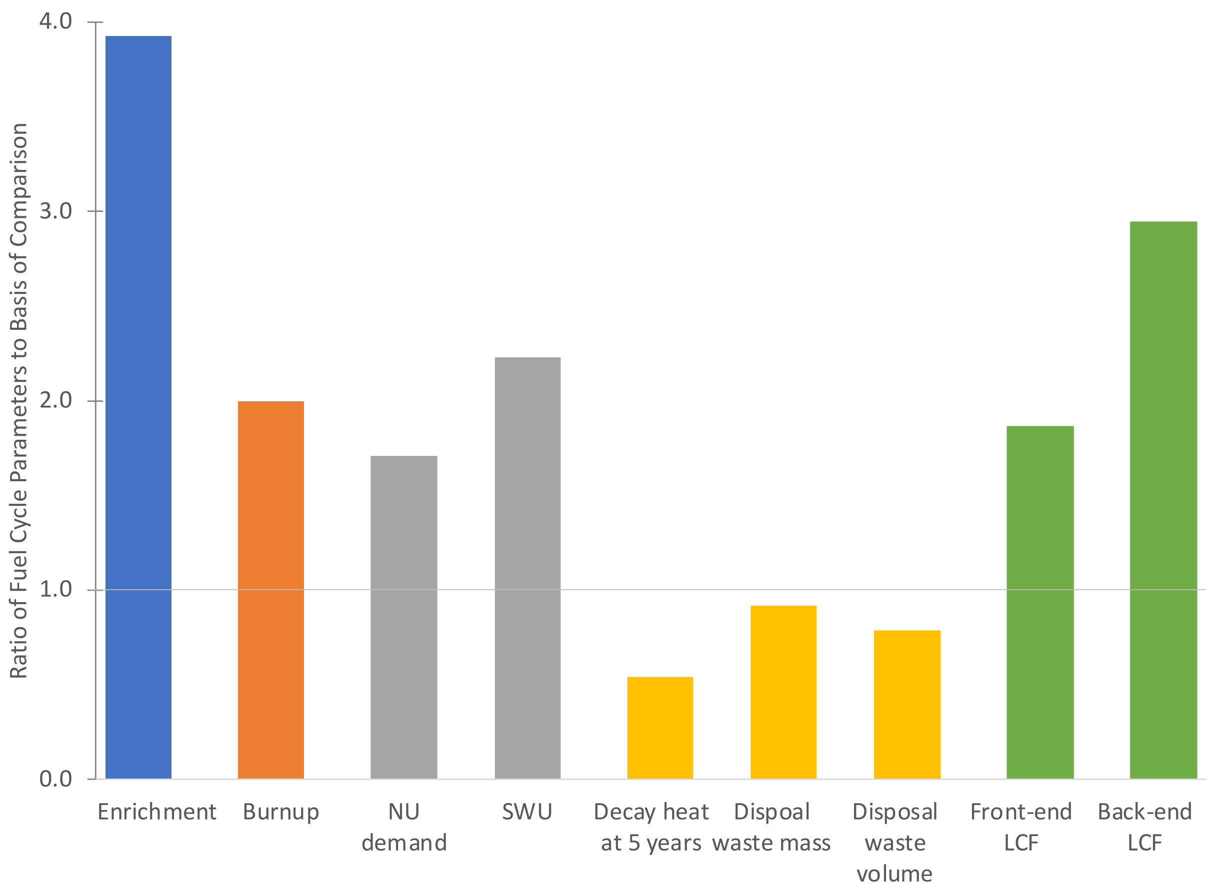


Figure 4.1 Ratios of once-through fuel cycle parameters for Analysis Example Reactor to the Basis of Comparison

## 4.1 Variation of Enrichment and Burnup

The larger NU and SWU demands relative to the fuel burnup are the primary reasons for the higher front-end LCF of the OT-HALEU. The NU demand and SWUs can be reduced by increasing burnup or decreasing enrichment. The discharge burnup and required uranium enrichment vary depending on reactor types and associated fuel cycles. For instance, the thermal reactor fuel is discharged when the excess reactivity is exhausted with depletion of initially loaded fissile material. For achieving a higher burnup, the initial enrichment in thermal reactor fuels can be raised, but advanced cladding materials would also be required to avoid cladding failures. In particular, a robust cladding form such as TRISO particles can achieve a high burnup by increasing fuel enrichment. For fast reactors, the fuel is discharged when the cladding reaches a fluence limit, while significant fissile content remains in the DF. The fast reactor fuel burnup can be increased further with advanced cladding, but an additional increase in the fuel enrichment is not needed because fissile materials are bred during irradiation.

To assess the relationship between burnup and fuel enrichment, an NU demand line equivalent to the Basis of Comparison was calculated and plotted in Figure 4.2. The NU demand of all pairs of burnup and enrichment on the equivalent line is the same as that of the Basis of Comparison, which is 188.1 t/GWe-year with 4.2% LEU fuel and a discharge burnup of 50 GWd/t in LWRs. For instance, the NU demand of 15% HALEU fuel is equal to that of the Basis of Comparison fuel when its burnup is ~155 GWd/t. In Figure 4.2, the NU demand is smaller or larger than the Basis of Comparison when the burnup is above or below the equivalent line, respectively. The enrichment and burnup of Natrium, XE-100, and the Analysis Example Reactor are plotted in Figure 4.2. The Xe-100 is on the equivalent line, but Natrium and the Analysis Example Reactor are below the line. Thus, the NU demand of Xe-100 is comparable to the Basis of Comparison, but Natrium and the Analysis Example Reactor require more NU.

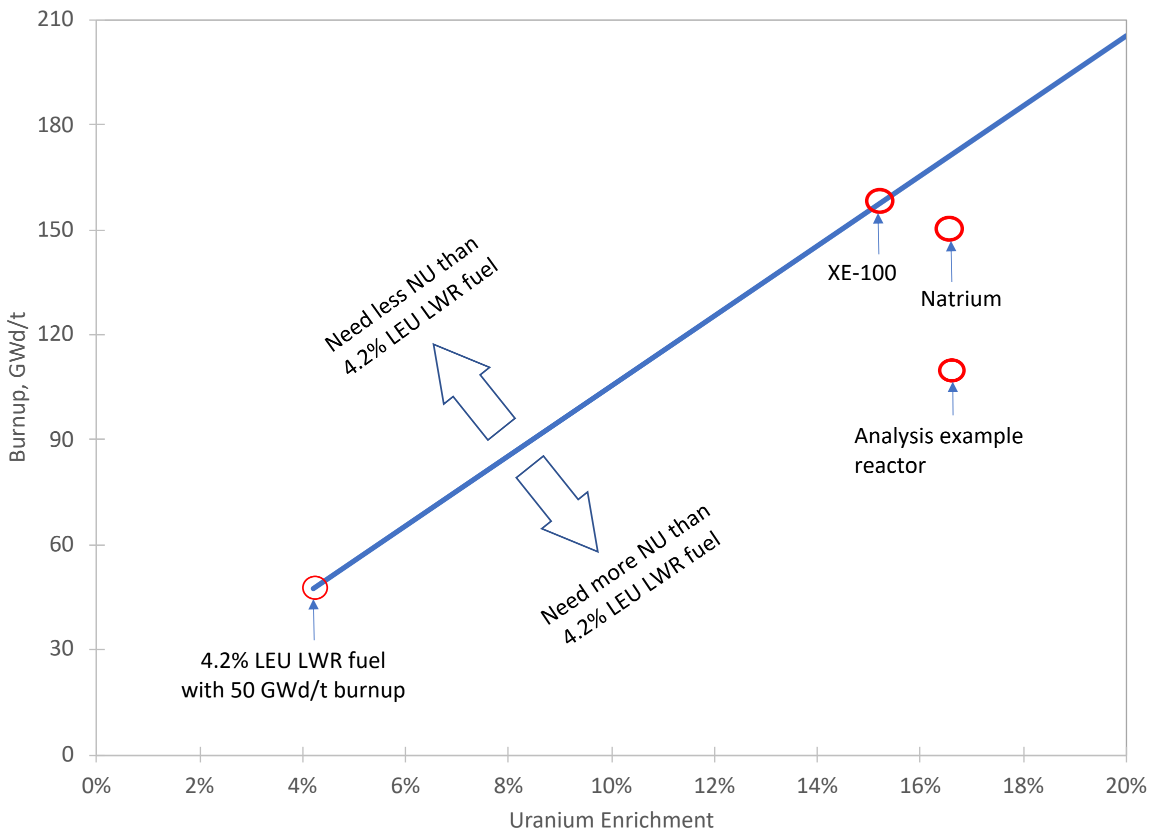


Figure 4.2 NU demand equivalent line for once-through fuel cycle

The fuel cycle parameters of the once-through fuel cycles with different burnups are compared in Table 4.1. The second and third columns are the Basis of Comparison and the OT-HALEU, respectively. In the last column, the burnup is increased to 171 GWd/t without changing the enrichment until the required NU demand of the Analysis Example Reactor is equal to $UO_2$ fuel in the LWR. The resulting NU demand is equal to the Basis of Comparison, and other fuel cycle parameters are improved significantly. *This result indicates the need to increase burnup further for advanced reactor concepts with HALEU to reduce the fuel cycle parameters and LCF.* For the metallic fuel, the cladding integrity is degraded through constituent migration, fuel-cladding chemical interaction (FCCI), wastage of cladding, etc., as burnup increases. There are various ways to increase the metallic-fuel burnup (smaller smeared fuel density in a sodium-free annular shape, coating or liner on the inner cladding surface, alloy addition to reduce FCCI, etc.). However, even though the NU demand is the same, the total fuel-cycle LCF is still 36% higher than the Basis of Comparison because of the increased SWUs and additional treatment of the sodium-bonded metallic fuel in the back-end fuel cycle. In Table 2.1, U-235 and Pu contents in the discharge fuel experiencing 100 GWd/t burnup are 9.2% and 5.3%, respectively. The U-235 content will be smaller in the higher-burnup DF, but the Pu content will be larger. Thus, recovering uranium or U/Pu from the once-through fast reactor fuel and recycling it in the current LWR fleet would be attractive for the fuel cycle performance of the limited-recycle fuel cycle, as discussed in Section 3.

Table 4.1 Comparison of once-through fuel cycle parameters for different burnups

| Reactor | | Basis of Comparison (LWR) | Analysis Example Reactor with different burnups | |
|---|---|---|---|---|
| | | | Conventional-burnup fuel | Higher-burnup fuel |
| Fuel form | | UOX | Metal with sodium bond | |
| Charge fuel fissile enrichment, % | | 4.2 | 16.4 | 16.4 |
| Burnup, GWd/t | | 50.0 | 100.0 | 171.0 |
| Fissile content in discharge fuel, % | | ~1.9 % | ~14.5% | Not calculated |
| Front-end | NU demand, t/GWe-year | 188.1 | 321.7 | 188.1 |
| | SWUs, t-SW/GWe-year | 137.2 | 306.2 | 179.1 |
| | Charge fuel, t/GWe-year | 21.9 | 9.1 | 5.3 |
| | LCF, $\$_{2021}$/MWh | 7.4 | 14.5 | 8.5 |
| Back-end | SNF or HLW mass, t/GWe-year | 21.9 | 20.1 | 11.7 |
| | Disposal volume, $m^3$/GWe-year | 25.0 | 19.7 | 11.5 |
| | LCF, $\$_{2021}$/MWh | 2.0 | 6.0 | 4.3 |
| Total LCF, $\$_{2021}$/MWh | | 9.4 | 20.5 | 12.8 |

## 4.2 Variation of Fuel Forms

The impact of fuel forms on the once-through fuel cycle parameters is compared in Table 4.2. The fuel enrichment and burnup of the Analysis Example Reactor were kept, but the fuel forms were changed to sodium-free metallic fuel [Carmack 2009] and pebble-bed fuel with TRISO kernels. The NU demand and enrichment efforts were the same regardless of the fuel forms, but the total fuel-cycle LCF was affected by the fuel form. The fuel fabrication costs of the sodium-bonded metallic fuel and TRISO fuels were obtained from the Advanced Fuel Cycle Cost Basis Report [Dixon et al. 2017]. It was assumed that the fuel fabrication cost of the sodium-free metallic fuel is the same as that of the sodium-bonded metallic fuel. On the back end, the sodium-bonded fuel is assumed to be treated to separate the bond sodium, followed by disposal of the radioactive waste, but sodium-free and TRISO spent fuels are assumed to be directly disposed.

Compared to the sodium-bonded metallic fuel, the sodium-free metallic fuel has the same front-end fuel cycle LCF but a factor-of-three smaller back-end fuel cycle LCF because it does not include additional treatment to separate sodium. The TRISO fuel has higher front-end and back-end fuel cycle LCFs because of the higher fuel fabrication cost and large disposal volume.

Table 4.2 Comparison of once-through fuel cycle parameters for different fuel forms

| Reactor | | LWR | SFR | | HTGR |
|---|---|---|---|---|---|
| Fuel | | UOX | Sodium-bonded metallic fuel | Sodium-free metallic fuel | Pebble with TRISO |
| Enrichment, % | | 4.2 | 16.4 | 16.4 | 16.4 |
| Burnup, GWd/t | | 50.0 | 100.0 | 100.0 | 100.0 |
| Front-end | NU demand, t/GWe-year | 188.1 | 321.7 | 321.7 | 321.7 |
| | SWUs, t-SW/GWe-year | 137.2 | 306.2 | 306.2 | 306.2 |
| | Charge fuel, t/GWe-year | 21.9 | 9.1 | 9.1 | 9.1 |
| | LCF, $\$_{2021}$/MWh | 7.4 | 14.5 | 14.5 | 30.2 |
| Back-end | SNF or HLW mass, t/GWe-year | 21.9 | 20.1 | 9.1 | 9.1 |
| | Disposal volume, $m^3$/GWe-year | 25.0 | 19.7 | 23.9 | 163.5 |
| | LCF, $\$_{2021}$/MWh | 2.0 | 6.0 | 1.9 | 11.2 |
| Total LCF, $\$_{2021}$/MWh | | 9.4 | 20.5 | 16.4 | 41.4 |

## 4.3 Enrichment Cost in Category-II Facility

The enrichment cost data for <5% LEU in the current commercial Category-III facilities is provided in the Advanced Fuel Cycle Cost Basis Report [Dixon et al. 2017]. However, the enrichment costs for enriching up to 10% in the Category-III facilities and 10–19.75% in the Category-II facilities are not available.[2] In Section 3.3, the unit enrichment costs were assumed to be a factor of 1.2 and 2 higher for 5–10% and 10–19.75% enrichment, respectively, compared to the enrichment cost in the current commercial Category-III facility. In particular, a factor-of-2 cost increase was assumed for the 10–19.75% enrichment because it should be done in a Category-II facility with stringent physical security requirements (a tighter criticality control constraint, more shielding, etc.). Because the enrichment costs used in Section 3.3 are somewhat arbitrarily assumed, the impact of the enrichment cost escalation in the Category-II facility was assessed.

The overall enrichment costs of the OT-HALEU are compared in Figure 4.3 for the single-step, two-step, and three-step enrichment schemes. The SWU values for the three schemes are provided in Table 3.2. The x-axis indicates the unit cost multiplication factor: i.e., the unit enrichment cost in the Category-II facility is a factor of 2, 4, 6, 8, and 10 higher than the <5% LEU enrichment cost in the Category-III facility. The y-axis indicates the total enrichment cost of the OT-HALEU. For the single-step enrichment scheme, the enrichment cost increases as the Category-II enrichment premium increases. However, for the two- or three-step enrichment scheme, the enrichment premium in the Category-II facility produces a minor increase in the total enrichment cost because of the small fraction of SWUs in the Category-II facility (see Table 3.2). *This finding indicates that enriching uranium in a multi-step enrichment scheme is one of the options to reduce the enrichment cost of the HALEU fuels.*

[2] Both costs for enriching uranium up to 10% in the Category-III facilities and 10–19.75% in the Category-II facilities will be available in the next release of the Advanced Fuel Cycle Cost Basis Report.

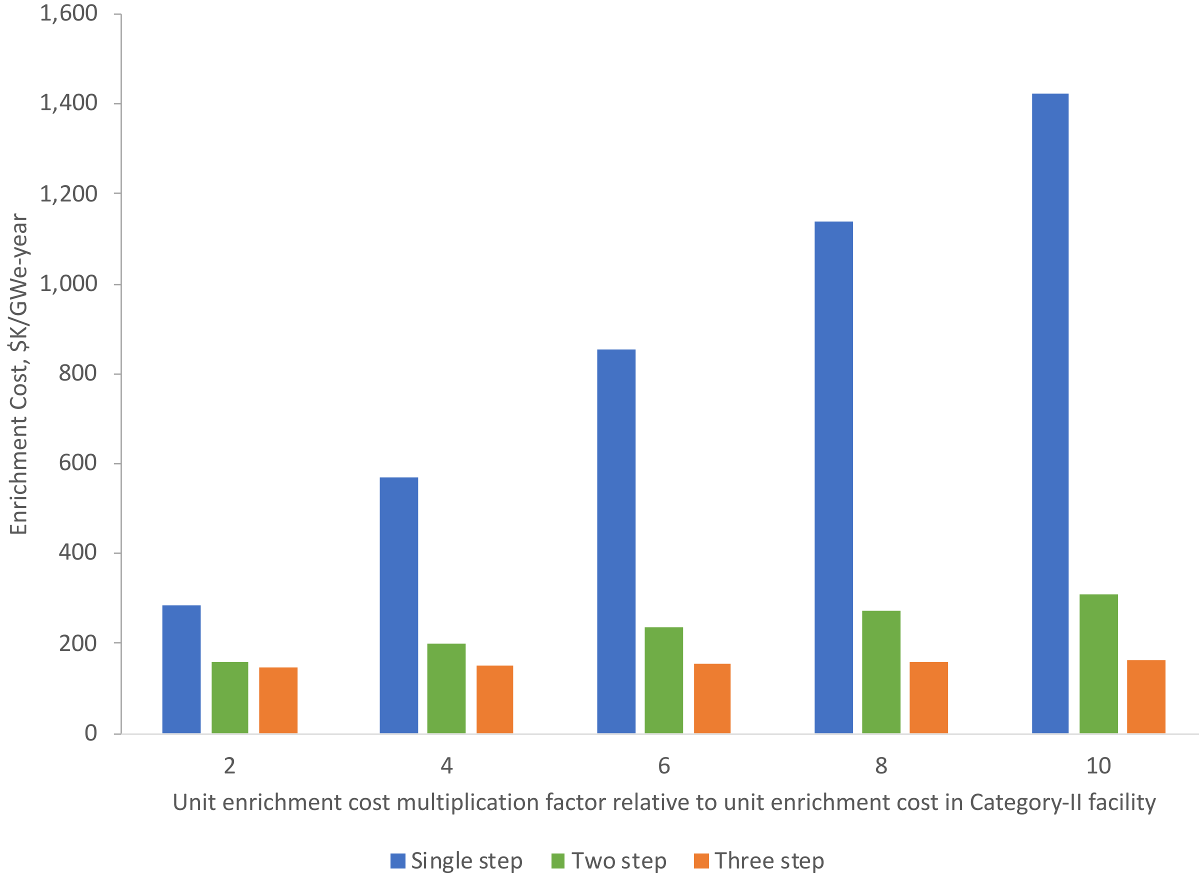


Figure 4.3 Enrichment cost in Category-II facility based on single-, two-, and three-stage options

## 4.4 Advanced Reactor Types

The HTGR and the SFR are the dominant thermal and fast advanced reactor concepts, respectively (see Figure 2.2), and the two ARDP demonstration reactors, Xe-100 and Natrium, are an HTGR and an SFR, respectively. To compare the integral effect of enrichment, burnup, and fuel forms, the fuel cycle parameters are compared for three reactor types: the Analysis Example Reactor with sodium-bonded metallic fuel, Natrium with sodium-free metallic fuel, and Xe-100 with pebbles containing TRISO particles.

The comparison is provided in Table 4.3. The Natrium concept has a front-end and back-end fuel cycle cost lower than that of the Analysis Example Reactor, owing to higher burnup and direct disposal of SNF, but higher than the Basis of Comparison. For the Xe-100, the NU demand and enrichment effort are smaller than those of the Basis of Comparison, but its front-end fuel cycle cost is higher because of the higher cost of TRISO fuel fabrication. The back-end fuel cycle cost is also higher than for other reactor concepts because of the higher disposal volume.

Table 4.3 Comparison of once-through fuel cycle parameters for different reactor types

| Reactor and fuel | | LWR, UOX | Analysis Example Reactor, sodium bond metallic fuel | Natrium, sodium free metallic fuel | Xe-100. Pebble with TRISO |
|---|---|---|---|---|---|
| Enrichment, % | | 4.2 | 16.4 | 16.4 | 15.5 |
| Burnup, GWd/t | | 50.0 | 100.0 | 150.0 | 168.3 |
| Front-end | NU demand, t/GWe-year | 188.1 | 321.7 | 214.4 | 179.4 |
| | SWUs, t-SW/GWe-year | 137.2 | 306.2 | 204.1 | 169.6 |
| | Charge fuel, t/GWe-year | 21.9 | 9.1 | 6.1 | 5.4 |
| | LCF, $\$_{2021}$/MWh | 7.4 | 14.5 | 9.7 | 17.4 |
| Back-end | SNF or HLW mass, t/GWe-year | 21.9 | 20.1 | 6.1 | 5.4 |
| | Disposal volume, $m^3$/GWe-year | 25.0 | 19.7 | 16.0 | 97.1 |
| | LCF, $\$_{2021}$/MWh | 2.0 | 6.0 | 1.2 | 6.6 |
| Total LCF, $\$_{2021}$/MWh | | 9.4 | 20.5 | 10.9 | 24.0 |

# 5. Pros and Cons and Future R&D Needs

## 5.1 Summary of Pros and Cons of HALEU Utilization

The pros and cons of HALEU utilization in advanced reactors are summarized in Table 5.1. The fuel cycle performance parameters of advanced reactors with HALEU have been assessed for three example fuel cycles (once-through, CR started with HALEU, and limited recycle). By comparing the fuel cycle performance parameters with the Basis of Comparison (once-through LWR), the pros and cons are identified.

Table 5.1 Pros and cons of HALEU utilization in advanced reactors

| Pros | Cons |
|---|---|
| • Allows advanced reactor design with advanced coolants (sodium, lead, molten salt, etc.) for improvement of thermal efficiency and incorporation of inherent safety features<br>• Allows compact or leaky core (such as MR) design having relatively longer cycle length<br>• Expansion of cycle length or discharge burnup<br>• Smaller charge fuel and DF masses<br>• Recovered uranium is usable in LWRs<br>• Permits initiation of CR fuel cycle without Pu | • Higher NU and SWU demands<br>• Requirement for Category-II enrichment facility<br>• Need for secure transportation of HALEU<br>• Higher cost of fuel fabrication |

The prime benefit of using 10% - 19.75% HALEU is to allow advanced reactor designs with advanced coolant materials and a compact and leaky core. For instance, sodium- or lead-cooled fast reactors have been proposed and are under development because of various attractive features of reactor performance (ability to breed fissile and burn waste, higher thermal efficiency with higher operating temperature, excellent inherent safety features, etc.). However, fast reactors cannot achieve criticality without a certain amount of fissile (such as HALEU or recovered Pu) in the charge fuel. In particular, the HALEU is absolutely needed for once-through fast reactors. Similarly, a compact and leaky reactor, such as an MR, requires HALEU to maintain criticality for a reasonably long cycle length (e.g., a few years) without refueling.

In addition, most advanced reactor concepts summarized in Section 2.1 use HALEU to achieve longer cycle length, higher burnup, and other desirable design attributes. The increased burnup (or longer cycle length) with HALEU leads to lower charge fuel and DF masses, which result in less fuel fabrication and may affect radioactive-waste management.

The cons listed in Table 5.1 are related to higher enrichments of HALEU. The NU and SWU demands are proportional to the uranium enrichment of HALEU, and a higher enrichment incurs additional costs for enrichment and fuel fabrication in Category-II facilities and transportation in a secure manner. Thus, the pros and cons are dependent on the achievable burnup and required enrichment of HALEU fuel. For the identified pros to be dominant, burnup should be sufficiently high with a relatively low-enriched HALEU fuel. Otherwise, the cons are dominant. The NU demand equivalent line (see Figure 4.2) illustrates this relationship. For instance, the NU demand per unit of electricity generation with Natrium's 16.5% HALEU fuel is equivalent to the Basis of Comparison when the burnup increases to 171 GWd/t. Similar curves can be generated for other metrics of interest.

The RU from the HALEU used fuel can be reused in LWRs because it contains high uranium fissile content. Typically, the U-235 content in the HALEU used fuel is higher than the uranium enrichment of LWRs. Thus, by down-blending with NU, the RU from the HALEU used fuel can be reused in LWRs. Even though the RU contains U-236 (which acts as a poison in LWRs), the down-blended RU can be used in an LWR without major technical issues. The detailed physics parameters are provided in Appendix B.

In the E&S study [Wigeland et al. 2014), a CR U/TRU fuel cycle in a fast reactor is identified as one of the most promising fuel cycle options. Thus, the nuclear fuel cycle may evolve to a CR option from the current once-through. A fuel cycle transition could occur after recycling technologies are fully developed; or the use of HALEU allows another option, which does not need Pu for the initial start of the fast reactor and evolves a once-through fuel cycle into one of the most promising options. Because various fast reactors are under development and one of them (Natrium) will be demonstrated through ARDP, the latter option would be practical. Thus, an additional pro of HALEU utilization is the initiation of conditions for a CR fuel cycle option even though the recycling technologies are being deferred.

## 5.2 Future R&D for Enhancement of HALEU Utilization

As discussed in the previous section, the pros of HALEU utilization in advanced reactors are related to the increased burnup, while the cons are related to the increased cost for enrichment, fuel fabrication, and transportation. In addition, the specific fuel forms adopted by advanced reactors increase the back-end fuel cycle LCFs. Thus, R&D for enhancement of HALEU utilization should be focused on increasing burnup and decreasing cost. The potential R&D items are listed below.

- **Advanced cladding technology for metallic fuel**: For the HALEU binary metallic fuel, the achievable burnup is dependent on cladding life, which is affected by factors such as cladding wastage by FCCI and fuel-cladding mechanical interaction. Figure 4.2 shows the relationship between target burnup for efficient fissile utilization and HALEU enrichment. Advanced cladding technology should be developed to extend the burnup into the desirable burnup range.
- **Sodium-free metallic fuel**: Enough information does not current exist to evaluate the performance of direct disposal of sodium-bonded SNF in any geologic disposal concept [UFD 2014]. The assumed additional treatment of used fuel to separate bond sodium increases the processing cost and disposal volume. A sodium-free metallic fuel would reduce the disposal volume and back-end fuel cycle cost.
- **Affordable HTGR fuel fabrication technolog**y: According to the Advanced Fuel Cycle Cost Basis Report, the fabrication cost of gas-cooled reactor particle fuels is much higher than that of other fuel forms (UOX, metallic fuel, etc.). As a result, the HTGR front-end LCF is about 2–3 times higher than that of other reactor types. Affordable HTGR fuel fabrication technology is needed to reduce the overall LCF of HTGRs.
- **Reduction of disposal volume of HTGR SNFs**: Particle fuels perform well in a repository because of the multiple levels of encapsulation provided by the fuel form. However, owing to the small volumetric power density, the SNF volume of HTGRs is much larger than for other reactor types. To reduce the disposal volume while maintaining the fuel performance as a waste form, compact canister options to accommodate discharged pebbles should be explored.
- **Enrichment infrastructure, fuel fabrication facility, and transportatio**n: For enrichment of HALEU and fuel fabrication, Category-II facilities are needed. In addition, HALEU fuel requires extra transportation security compared to LWR fuels. Overall, the more stringent security requirements for enrichment, fuel fabrication, and transportation will increase the cost. As discussed in Section 3.1, affordable and integral infrastructure for HALEU enrichment, fuel fabrication, and transportation is needed.

# 6. Conclusions

Various advanced reactor concepts and associated nuclear fuel cycles are under development through current reactor development programs and initiatives. Most of them utilize 10–19.75% high-assay low enriched uranium (HALEU) to improve fuel cycle performance and fuel cycle economics by increasing burnup and other design attributes (compact core design, higher thermal efficiency, etc.). On the other hand, HALEU utilization requires additional costs for higher enrichment and treatment of HALEU in facilities with more stringent security requirements.

The Systems Analysis and Integration (SA&I) campaign assessed the pros and cons of HALEU utilization in advanced reactor concepts through a systematic evaluation of fuel cycle performance parameters and the Levelized Cost of Fuel (LCF), which is the Levelized Cost of Electricity excluding reactor cost. First, the SA&I campaign identified three example fuel cycles associated with advanced reactor concepts using HALEU:

- Once-through fuel cycle in advanced reactors with HALEU (OT-HALEU),
- Limited-recycle fuel cycle based on a two-stage system (LT-RU). The first stage is an advanced fast rector with HALEU and the second stage is an LWR with the recovered uranium (RU) from the first-stage used nuclear fuel, and
- Continuous recycle of U/Pu in advanced reactors, started with HALEU (CR).

Then, front- and back-end fuel cycle parameters and LCF of the example fuel cycles were evaluated using a single Analysis Example Reactor concept. A sodium-cooled fast reactor (SFR) having a burnup of ~100 GWd/t with sodium-bonded metallic fuel was selected as the Analysis Example Reactor because its technology readiness level is high, and the burnup and fuel enrichment are in the middle of those ranges of various advanced reactor concepts that are currently under development. Additionally, a series of sensitivity analyses were conducted by varying burnup, enrichment, fuel forms, and reactor types to capture the design variations in two once-through ARDP reactors, Natrium with sodium-free metallic fuel having a burnup of ~150 GW/t and Xe-100 with TRISO pebble fuel having a burnup of ~168 GWd/t. The resulting fuel cycle parameters were normalized to the unit of electricity generation for comparison. Finally, the pros and cons, including the potential R&D for maximizing the pros and mitigating the cons, were identified by comparing normalized fuel cycle parameters to the Basis of Comparison, which is the current once-through LWRs with <5% low-enriched uranium (LEU) fuel and 50 GWd/t burnup.

For the front-end fuel cycle parameters, such as natural uranium (NU) demand and Separative Work Units (SWUs), the pros of HALEU utilization derive from the increased burnup. The advanced reactor core was designed to be more compact, and the increased burnup reduces the charge fuel mass and fuel fabrication mass. The cons derive from the higher enrichment of HALEU. The higher NU demand and SWUs, including the more stringent security requirements of enrichment and fuel fabrication facilities, increase the front-end fuel cycle LCF. Thus, the pros are predominant in advanced reactor concepts if the burnup is sufficiently high with a relatively low-enriched HALEU. On the other hand, the cons are predominant in advanced reactor concepts if the burnup is insufficiently high.

The evaluated front-end fuel cycle parameters and LCF of the Analysis Example Reactor show that the OT-HALEU has worse front-end performance and LCF than the Basis of Comparison, owing to the insufficiently high burnup. For the Analysis Example Reactor, the required enrichment (16.4%) is a factor of four higher than that of the Basis of Comparison, but the achieved burnup (~100 GWd/t) is only a factor of two higher. Further increase in the burnup is required to beat the front-end fuel cycle performance and LCF of the Basis of Comparison. As an indicator of the boundary between the pros and cons, an NU demand equivalent line (see Figure 4.2) is provided in this report.

For two ARDP reactors, the front-end fuel parameters and LCF are improved compared to the Analysis Example Reactor but worse than the Basis of Comparison because the burnup (~150 GWd/t for Natrium) is still insufficient and TRISO/pebble fuel fabrication (for Xe-100) is expensive. Thus, to maximize the pros and mitigate the cons, further improvement of fuel burnup and reduction of TRISO fuel fabrication costs are needed.

The pros of the back-end fuel cycle parameters (such as interim storage, reprocessing of used fuel, and radioactive waste disposal) are also related to the increased burnup with HALEU. The increased burnup reduces the discharge fuel (DF) mass, which requires less interim storage and reprocessing if needed. However, waste disposal volume and back-end LCF are affected by other factors, including the fuel forms adopted by the advanced reactor concepts.

In the present study, the Analysis Example Reactor uses a sodium-bonded metallic fuel. Removal of bond sodium from the DF is assumed before disposal, consistent with the disposition of EBR-II fuel [DOE 2000]. In both limited and continuous recycle, the bond sodium is removed during reprocessing. Other fuels (uranium oxide, pebbles, etc.) are assumed to be suitable for direct disposal without additional treatment. In the present study, electrometallurgical treatment (EMT) was considered for the removal of bond sodium from the metallic fuel. The recovered materials and byproducts from EMT are treated as high-level waste (HLW), increasing the back-end LCF.

In a sensitivity study, the impacts of two ARDP reactor types and fuel forms were evaluated. Natrium will be demonstrated using a sodium-bonded metallic fuel, but the commercial version of Natrium will use a sodium-free metallic fuel. Thus, the cons of the sodium-bonded metallic fuel will be eliminated in the commercial version of Natrium. Xe-100 uses TRISO/pebble fuel. Owing to the robust technology for encapsulating fission products, the fuel can achieve very high burnup and may undergo direct disposal without additional treatment, but encapsulation increases the disposal volume significantly. Overall, the back-end fuel cycle LCF of Natrium was below that of the Basis of Comparison, but the back-end fuel cycle LCF of Xe-100 was higher by a factor of three.

Other pros of HALEU utilization include the reuse of the high U-235 content in the HALEU used fuel in a conventional LWR and the potential for initiation of further fuel cycle evolution. The pros and cons of HALEU utilization in limited-recycle fuel cycles were assessed using a particular limited-recycle fuel cycle consisting of a two-stage system. The first stage is an advanced reactor with HALEU fuel, and the second stage is an LWR with the recovered uranium (RU) from the first-stage HALEU DF. The overall front- and back-end fuel cycle parameters were improved compared with the OT-HALEU because about 50% of total electricity was generated using the RU.

In a continuous-recycle fuel cycle, the reactor is sustainable without HALEU. The HALEU fuel is only needed to start the core and a few initial cycles (5 reload cycles in the present work) until sufficient Pu or TRU is bred. Thus, the amount of HALEU required during the reactor lifetime is small, and the pros and cons of HALEU utilization in a continuous-recycle fuel cycle are minimal. However, HALEU permits the initiation of fuel cycle evolution to CR, which was identified as a promising fuel cycle in the fuel evaluation and screening study [Wigeland et al. 2014].

In conclusion, most of the advanced reactor concepts under development utilize HALEU. The pros and cons of HALEU utilization in the advanced reactor concepts are mainly dependent on the HALEU enrichment and the achievable burnup. The evaluations indicate that the burnup of once-through advanced reactors with HALEU is still insufficiently high to beat the front-end fuel cycle performance of the current LWR fleet with respect to LCF. Several fuel cycle parameters, such as disposal volume and LCFs, are also affected by fuel forms. To maximize the pros and mitigate the cons, R&D is needed on further extension of discharge burnup, removal of waste-management hurdles (such as sodium bonds), the multi-step enrichment infrastructure for enriching uranium, affordable TRISO fuel fabrication technologies, etc.

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# Appendix A. Advanced Reactor Concepts

Various advanced reactor concepts currently under development in the United States and Canada are listed in Table A. 1. The list includes small modular reactor (SMR) concepts, microreactor concepts, and non-LWR (light water reactor) advanced reactor concepts, but excludes evolutionary LWRs. It is noted that the list might miss the most recently proposed reactor concepts or not reflect the latest status of the other advanced reactor concepts because the reactor concepts have been evolving and, in some cases, only limited non-proprietary information is available.

Table A. 1 List of Advanced Reactor Concepts

| Reactor type [a)] | Reactor Name | Proponent | Fuel | Fissile |
|---|---|---|---|---|
| ADS | ADSMS | Texas A&M | Molten Salt | TRU |
| | GEMSTAR (MuSTAR) | Virginia Tech | Molten salt | NU |
| FHR | FHR | MIT | TRISO, UCO | 15% EU |
| | KAIROS | Kairos | TRISO, UCO | 19.9% EU |
| | PB-FHR | MIT & UC-Berkeley | TRISO, UCO | 19.9% EU |
| | SmAHTR | ORNL | TRISO, UCO | 19.75% EU |
| GCR | GT-MHR | GA | TRISO, UCO | 15.5% EU |
| | Hybrid | Hybrid Power | TRISO, $UO_2$ | 12.0% EU |
| | NGNP | INL, etc. | TRISO, $UO_2$ | 15.5% EU |
| | PHTR | GA | TRISO, UCO | 15.5% EU |
| | SC-HTGR | AREVA | TRISO, UCO, $UO_2$ | 19.9% EU |
| | STARCORE | Starcore Nuclear | TRISO, UCO, $UO_2$ | TBD |
| | XE-100 | X-Energy | TRISO, UCO | 15.5% |
| GFR | $EM^2$ | GA | Carbide, UC | DU |
| LFR | ENHS | UC-Berkeley | Metal, U-Pu-Zr | 13% Pu |
| | G4M | Gen4 Energy | Nitride, UN | 19.5% EU |
| | LBFR | CBCG | TBD | NU |
| | LC-E-SSTAR | LakeChime | Nitride, U-TRU-N | TBD |
| | LEADIR-PS100 | Northern Nuclear | TRISO, $UO_2$ | TBD |
| | LFR-AS-200 | Hydromine | Oxide, $(U,Pu)O_2$ | TBD |
| | STAR | ANL/LLNL | Nitride, (U,TRU)N | 17.0% TRU |
| MR | eVinci | Westinghouse | Oxide, $UO_2$ | TBD |
| | Holos | HoloseGen | TRISO, $UO_2$ | 19.9% EU |
| | MMR | USNC | TRISO, UCO/$UO_2$/UN | 19.9% EU |
| | Oklo | Oklo | Metal, U-Zr | TBD |
| | Radiant | Radiant | TRISO, UCO | 19.8% EU |
| | Ubattery | URENCO | TRISO, $UO_2$ | 19.9% EU |
| MSFR | Elysium | Elysium | Chloride Salt | TBD |
| | MCFR | TerraPower | Chloride Salt | TBD |
| | SSR | Moltex Energy | Chloride Salt | TBD |
| MSR | IMSR | Terrestrial | Molten salt | 4% EU |
| | LFTR | Flibe Energy | Molten Salt | U-233 |
| | ThorCon | Martingale Inc. | Molten Salt | LEU/Th |
| | Thorenco | Thorenco | Molten Salt | U-233 |
| OCR | ORSN | MIT | Oxide, $UO_2$ | 5.0% EU |
| SFR | AFR-100 | ANL | Metal, U-Zr | 13.5% EU |
| | ARC-100 | ARC LLC | Metal, U-Zr | 10–15% EU |
| | Natrium | TerraPower | Metal, U-Zr | 16.5% EU |

| Reactor type [a)] | Reactor Name | Proponent | Fuel | Fissile |
|---|---|---|---|---|
| | PRISM | GE | Metal, U-Pu-Zr | 20.0% Pu |
| | Seed & Blanket | UC-Berkeley | Metal, U-Zr | NU |
| | TWR | TerraPower | Metal, U-Zr | NU |
| SMR | IRIS | Westinghouse | Oxide, $UO_2$ | 4.95% EU |
| | mPower | Generation mPower | Oxide, $UO_2$ | 5% EU |
| | NuScale | NuScale | Oxide, $UO_2$ | 4.95% EU |
| | RADIX | Radix Power | Hydride, $UZrH_{1.x}$ | 5-20 % EU |
| | SMART | Dunedin | CerMet | TBD |
| | SMR | Westinghouse | Oxide, $UO_2$ | 5.0% EU |
| | SMR-160 (HI-SMUR) | Holtec | Oxide, $UO_2$ | 4.95% EU |
| | VSBWR | GE-Hitachi | Oxide, $UO_2$ | 5.0% EU |

a) Reactor types: **ADS**=Accelerator Driven System, **FHR**=Fluoride-salt cooled High temperature Reactor, **GCR**=Gas-Cooled (thermal) Reactor, **GFR**= Gas-cooled Fast Reactor, **LFR**=Lead or Lead-cooled or Lead bismuth-eutectic Fast Reactor, **MR**=Micro Reactor, **MSFR**=Molten Salt Fast Reactor, **MSR**=Molten Salt Reactor, **OCR**=Organic Coolant Reactor, **SFR**=Sodium-cooled Fast Reactor, **SMR**=Small Modular Reactor.

# Appendix B. Analysis Example Reactors

## B-1. Analysis Example Reactors Using Once-through and Recycling Fuel Cycles

A sodium-cooled fast reactor (SFR) was selected as the Analysis Example Reactor concept for once-through and recycling advanced reactor concepts because the SFR (Natrium) is one of the reactor types that will be demonstrated through the Advanced Reactor Demonstration Program. The overall technology readiness level is relatively high compared to other advanced reactor types. However, to capture the impact of reactor type variation, a series of sensitivity studies was conducted, and the results are compared in Section 4.4.

The Analysis Example Reactor design parameters were obtained by revising the advanced burner reactor (ABR) concept [Kim et al. 2008]. The driver fuel enrichment and discharge burnup are similar to those of the advanced reactor concepts listed in Appendix A. The radial core configuration of the Analysis Example Reactor is shown in Figure B. 1, and the major design parameters are given in Table B. 1.

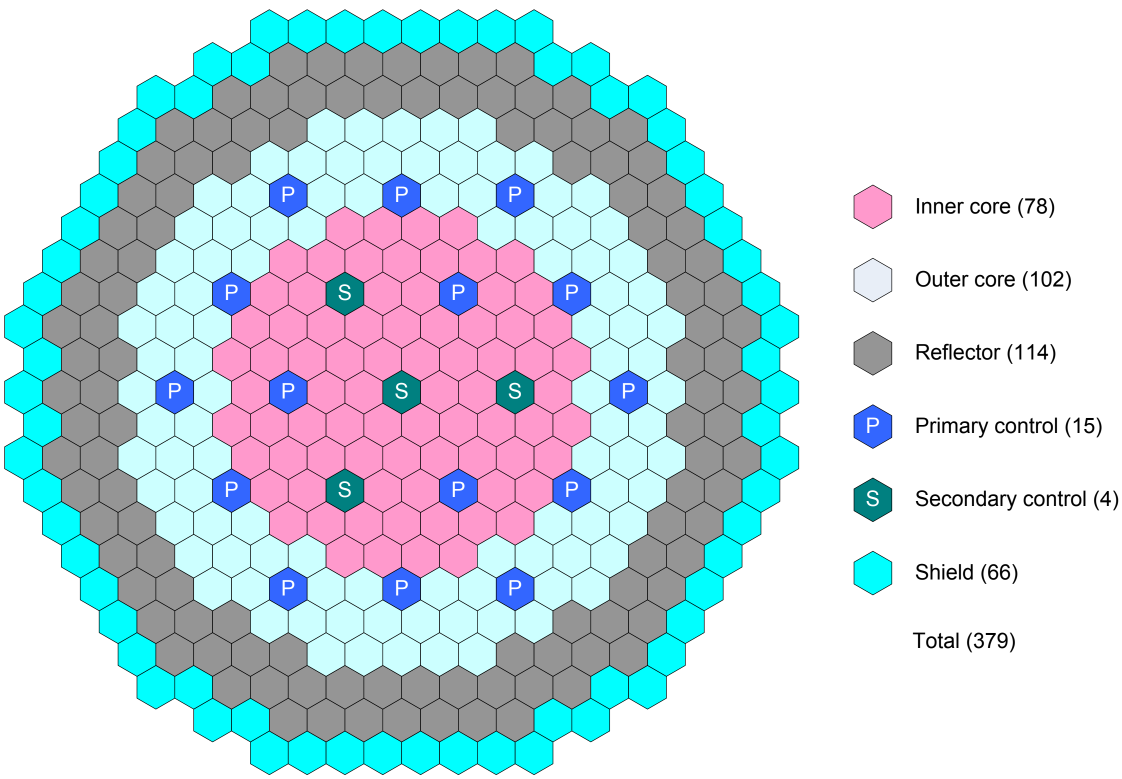


Figure B. 1 Radial core configuration of Analysis Example Reactor for once-through and recycling concepts

For the once-through Analysis Example Reactor, the discharge burnup of 96.7 GWd/t was achieved using a U-Zr metallic fuel with 16.5% HALEU. The remaining U-235 content in the DF is ~9.2%, and the mass ratio of Pu to the total heavy metal (HM) is 5.3%. Both values are much higher than those of the LWR DF. Thus, the Analysis Example Reactor for the once-through fuel cycle is used as the reactor concept for the first-stage analysis of the limited-recycle fuel cycle in Section 2.2.2.

For the Analysis Example Reactor for the recycling fuel cycle, the transuranics (TRU) recovered from the DF were continuously recycled as the fissile material of the charge fuel. HALEU was used as a make-up feed if the recovered TRU was insufficient during the initial 5 reloading cycles. The total HALEU mass required to fill the initial core and reloading cores was divided by the reactor lifetime (which was assumed to be 80 years for all reactor types) to evaluate the annual HALEU mass in the recycling concept. This particular fuel management scheme is adopted in this analysis to model the transition from an OT-HALEU to a recycling fuel cycle. The discharge burnup of 104.0 GWd/t was achieved using a U-TRU-Zr metallic fuel with a fissile content of ~15.3%. The remaining U-235 content in the DF is very small, while the Pu content increases to 14.7%.

Table B. 1 Design parameters of Analysis Example Reactor for once-through and recycling concepts

| | ABR | Analysis Example Reactor | |
|---|---|---|---|
| | | Once-through | Recycling |
| Fuel volume fraction, % | 29.2 | 34.3 | 34,3 |
| Active core height, cm | 81.3 | 95.0 | 95.0 |
| Fuel pin diameter, cm | 0.755 | 0.808 | 0.808 |
| Fuel residence time, month | 4 × 12 | 5 × 14 | 5 × 13 |
| Power density, W/cm$^3$ | 303 | 303 | 303 |
| Specific power density, kW/kg | 73.2 | 52.5 | 53.4 |
| Fresh fuel content, % | | | |
| - U-235/U | 0.25 | 16.4 | 0.25 |
| - TRU/HM | 22.1 | - | 22.1 |
| Discharge burnup, MWd/kg | 93 | 96.7 | 104.0 |
| Discharge fuel content, % | | | |
| - U-234/U | ~0.0 | 0.14 | 0.009 |
| - U-235/U | 1.2 | 9.2 | 0.9 |
| - U-236/U | 0.02 | 1.8 | 0.2 |
| - U-238/U | 99.9 | 88.8 | 98.9 |
| - U/HM | 74.5 | 94.5 | 84.7 |
| - Pu/HM | 25.1 | 5.3 | 14.7 |
| - MA/HM | 0.04 | 0.2 | 0.6 |

## B-2. Analysis Example of Limited-recycle Fuel Cycle

The overall concept of the two-stage limited-recycle fuel cycle is depicted in Figure B. 2 (shown previously as Figure 2.3). The first stage is identical to the Analysis Example Reactor for the once-through fuel cycle, which is an SFR with 16.4% HALEU. The recovered uranium (RU) or both RU and plutonium (RU/Pu) from the DF are sent to the second stage, while the unrecovered actinides and fission products (FPs) are sent to a repository (Δ). The DF composition of the first stage is given in Table 2.2. The U-235 content in the RU is 9.2%, which is higher than the uranium enrichment of the second-stage PWR fuel. The second stage is a Pressurized Water Reactor (PWR) with uranium oxide (UOX) fuel made with the RU from the first stage. The DF from the second stage is sent to a repository without further reprocessing.

It is noted that a similar two-stage limited recycle has been studied [OECD/NEA 2002] using PWRs in both the first and second stages. The first-stage PWR uses <5% LEU fuel, and the recovered Pu from the first stage is recycled in the second-stage PWR in the form of U-Pu mixed oxide (MOX) fuel. The conventional two-stage limited recycle has several design limitations, which include limitations on the total number of MOX assemblies and Pu content. Typically, the number of MOX fuel assemblies in the second-stage PWR is limited to 30% of the total number of fuel assemblies, and the Pu content is limited to 12% of the total HM because a high content of Pu reduces the shutdown margin and results in a positive void reactivity feedback.

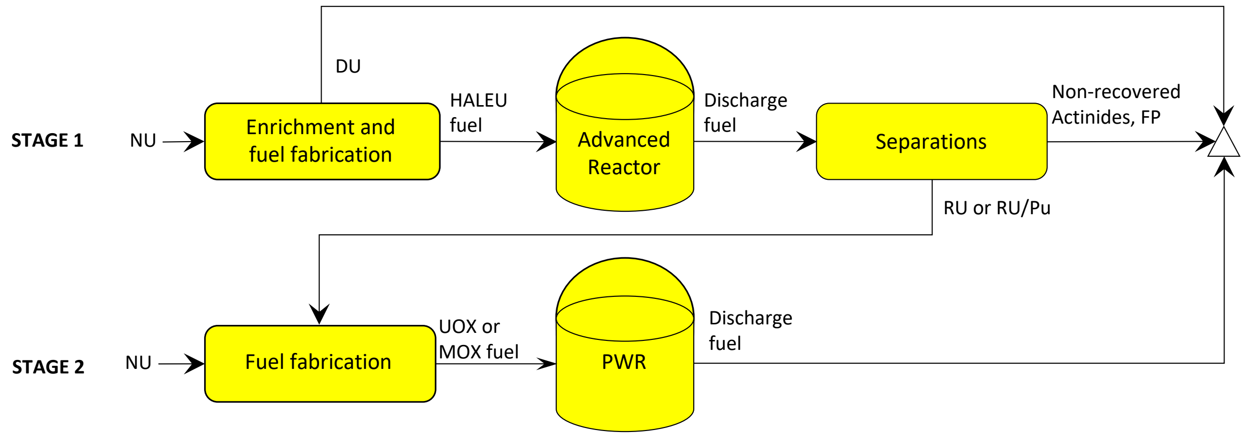


Figure B. 2 Overview of two-stage limited-recycle fuel cycle with HALEU

In the present study, the reactor performance parameters of the two-stage limited fuel cycle were calculated using the TRITON code by assuming that the second-stage PWR adopts a standard Westinghouse 17x17 assembly. The down-blending ratio of RU with NU was iteratively searched for a value giving a discharge burnup of about 50 GWd/t. For comparison purposes, the reactor performance parameters of both RU and Pu (RU/Pu) recycle were also calculated. The down-blending ratios, defined by the ratio of the RU mass (or RU+Pu mass) to the total HM mass, are provided in Table B. 2. Figure B. 3 shows the k-infinite value of the PWR assemblies with the down-blended fuels, compared with the UOX fuel assembly with 4.0% enriched uranium and the MOX fuel assembly with 10% Pu content. To investigate any design limitations in the two-stage limited-recycle fuel cycle considered in the present study, additional physics parameters, such as the Doppler constant, boron coefficient (reactivity change per soluble boron variation), and moderator temperature coefficient, were evaluated, and the results are shown in Figure B. 4, Figure B. 5, and Figure B. 6, respectively.

Table B. 2 Down-blending ratios of second-stage PWR fuels

| Recycling material | Down-blending ratio | Fissile content | Discharge burnup |
|---|---|---|---|
| RU | 43.8% | 4.4 % | 49.6 GWd/t |
| RU/Pu | 29.3% | 4.5 % | 50.3 GWd/t |

Generally, the down-blended fuels have similar reactor performance parameters compared to the conventional UOX fuel, while the MOX fuel has quite different reactor performance parameters. The resonance peaks of Pu isotopes (Pu-239, Pu-240, and Pu-242) in epithermal and thermal energy harden the neutron spectrum. The hardened spectrum of the MOX fuel reduces the control rod worth and increases the moderator temperature coefficient. Thus, the MOX fuel reduces the shutdown margin and causes a positive void reactivity feedback, and the Pu loading is limited in the conventional two-stage limited recycle. However, the RU (or RU/Pu) down-blended fuel can be used in the second-stage PWR without the design limitations that are present in the conventional two-stage limited recycle.

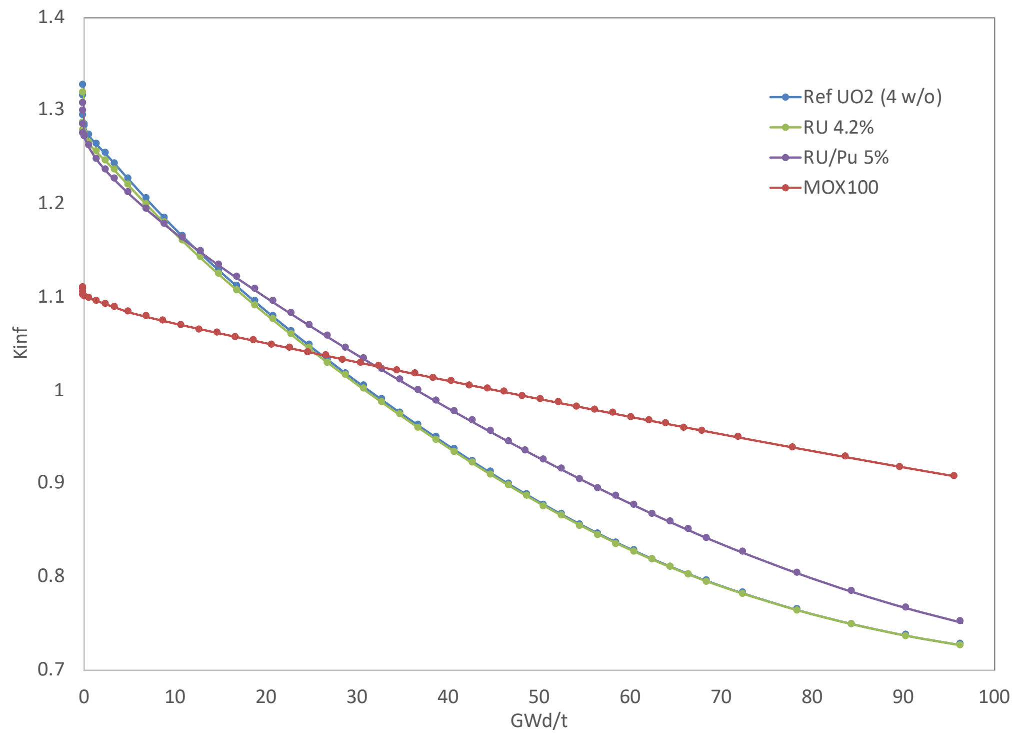


Figure B. 3 Neutron multiplication factor in infinite lattice assembly

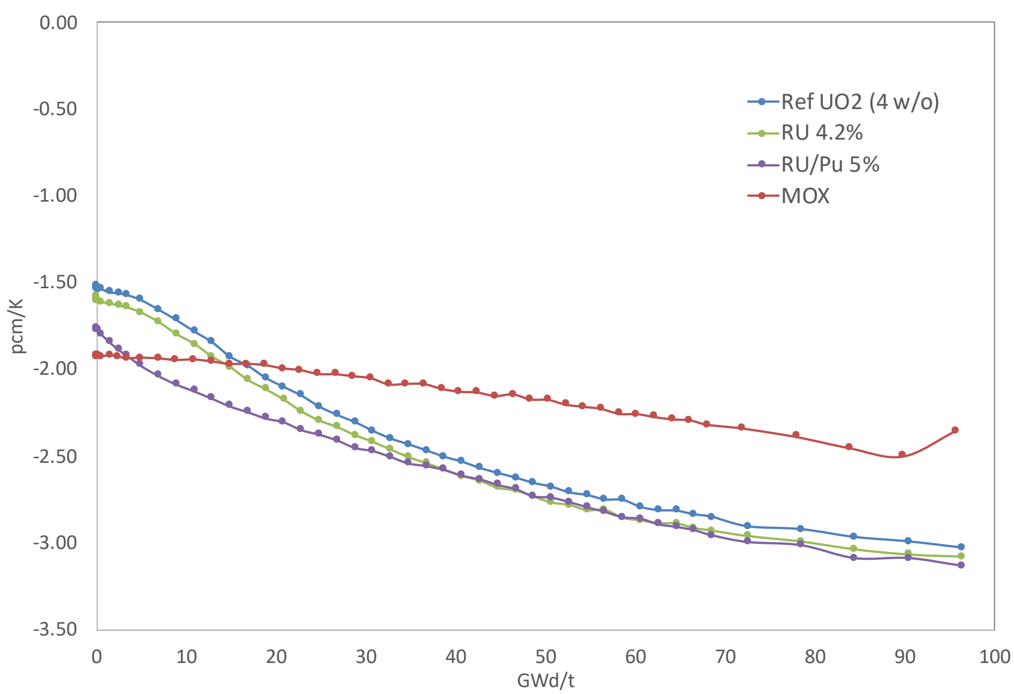


Figure B. 4 Doppler constant of PWR fuel assembly with down-blended fuels

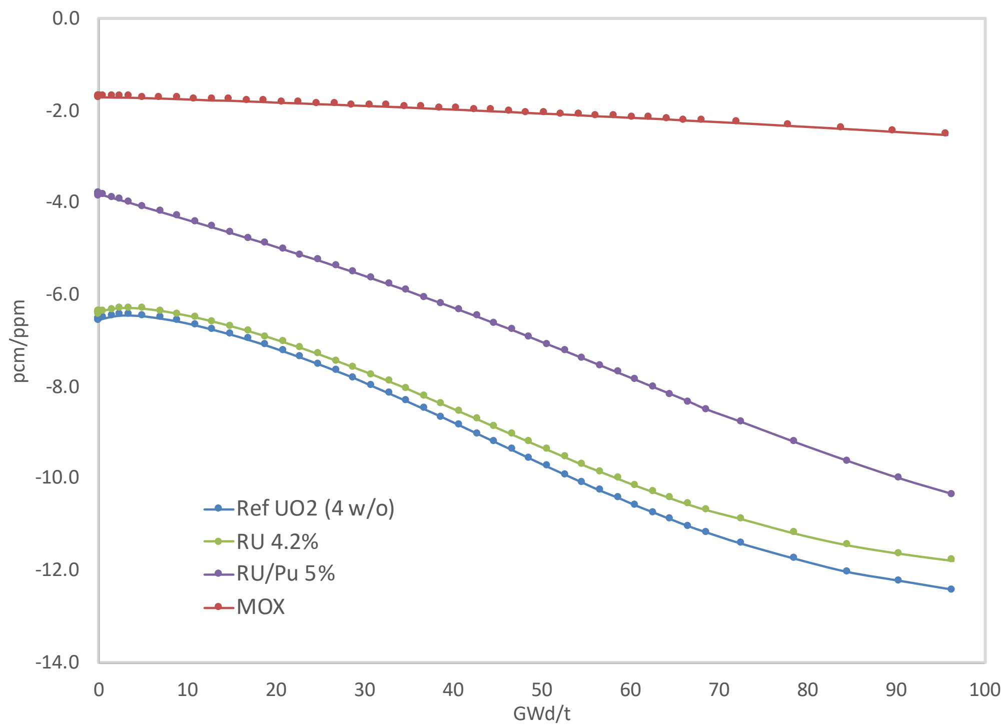


Figure B. 5 Soluble boron coefficient of PWR fuel assembly with down-blended fuels

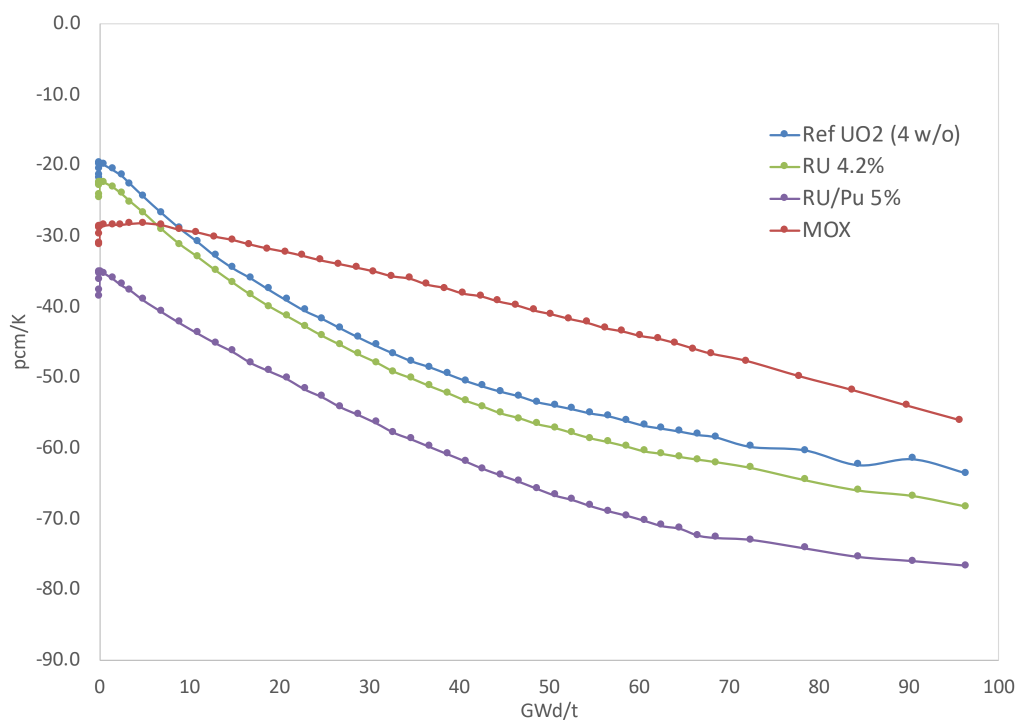


Figure B. 6 Moderator temperature coefficient of PWR fuel assembly with down-blended fuels

# Appendix C. Fuel Cycle Cost Data

The Levelized Costs of Fuel (LCFs) were calculated using the unit cost from the latest Advanced Fuel Cycle Cost Basis Report [Dixon et al. 2017], and the results were calibrated with 2021 dollars ($2021) by adjusting the inflation rate. In the Advanced Fuel Cycle Cost Basis Report, the unit costs are provided in terms of low, mean, high, and mode values with the cost probability distributions. For the relative comparison to the Basis of Comparison, the LCF was calculated using the mean values without considering the discounting rate. Table C. 1 shows the unit cost data used in the present study for calculating LCFs. The cost items highlighted in yellow indicate the missing data in the latest Advanced Fuel Cycle Cost Basis Report because of the lack of unit cost data. In the present study, the missing unit cost data were assumed by applying an engineering-judged multiplier to the base unit cost data.

Table C. 2 shows the multipliers of the missing unit cost items along with the base cost items. For instance, the HALEU enrichment unit cost was obtained by multiplying the base cost item of LEU (<5%) enrichment unit cost by 2.0.

Table C. 1 Unit cost data

| Item | Unit | Mean |
|---|---|---:|
| Natural Uranium | $/kg U | 158.5 |
| Conversion | $/kg U | 14.8 |
| Enrichment (<5%) | $/kg SW | 142.5 |
| Enrichment (5% to <10%) | $/kg SW | 194.9 |
| Enrichment (10% to <20%) | $/kg SW | 324.9 |
| Deconversion | $/kg U | 7.4 |
| DU Disposal | $/kg U | 7.4 |
| PWR UOX Fabrication | $/kg U | 457.1 |
| SFR HALEU Metallic Fabrication | $/kg U | 1,493.4 |
| SFR RU/Pu Metallic Fabrication | $/kg HM | 1,596.0 |
| PWR UOX Fabrication (RU) | $/kg U | 495.9 |
| Gas-Cooled Reactor Particle Fuel | $/kg/U | 16,530.0 |
| Down-blending | $/kg HM | 14.8 |
| EMT Fuel Processing & Waste Forms | $/kg HM | 2,964.0 |
| PWR DF Packaging | $/kg HM | 143.6 |
| HTGR DF Packaging | $/kg HM | 215.5 |
| SFR DF Packaging | $/kg HM | 245.6 |
| EMT-derived HLW Packaging | $/kg FP | 19,596.6 |
| PWR UOX SNF Disposal | $/kg HM | 669.2 |
| SFR SNF disposal | $/kg HM | 428.0 |
| Pebble SNF disposal | $/kg HM | 2,605.1 |
| HLW Disposal | $/kg FP | 5,700.0 |

Table C. 2 Cost multipliers of missing unit cost items

| Adjusted cost item | Base cost item | Multiplier |
|---|---|---|
| LEU (5% to <10%) | LEU (<5%) | 1.20 |
| LEU (10% to <20%) | LEU (<5%) | 2.00 |
| Pebble packaging | PWR packaging | 1.50 |
| SFR packaging | PWR packaging | 1.50 |
| Down-blending (RU+NU) | Deconversion | 2.00 |

The LCFs of four example fuel cycles were calculated by multiplying the unit cost data by the fuel cycle performance parameters. Table C. 3Table C. 3 shows the fuel cycle performance data associated with generating one unit of electricity for one year (GWe-year), including the calculated LCFs.

Table C. 3 Fuel cycle performance data for LCF estimation

| Cost item | Unit | OT-LWR | OT-HALEU | CR | LR-RU |
|---|---|---|---|---|---|
| Energy Generated | GWe-yr | 1.00 | 1.00 | 1.00 | 1.00 |
| Natural Uranium | kg U | 188,121 | 321,651 | 16,899 | 179,663 |
| Conversion | kg U | 188,121 | 321,651 | 16,899 | 179,663 |
| Enrichment (<5%) | kg SW | 137,244 | 266,159 | 13,891 | 143,992 |
| Enrichment (5% to <10%) | kg SW | - | 35,563 | 1,694 | 19,240 |
| Enrichment (10% to <20%) | kg SW | - | 4,458 | 131 | 2,412 |
| Deconversion | kg U | 166,221 | 312,526 | 16,311 | 169,077 |
| DU Disposal | kg U | 166,221 | 312,526 | 16,311 | 169,077 |
| PWR UOX Fabrication | kg U | 21,900 | | | - |
| SFR HALEU Metallic Fabrication | kg U | - | 9,125 | - | 4,937 |
| SFR RU/Pu Metallic Fabrication | kg iHM | - | - | 9,125 | - |
| PWR UOX Fabrication (RU) | kg U | - | - | - | 10,062 |
| Down-blending | kg iHM | | | | 10,062 |
| EMT Fuel Processing & Waste Forms | kg iHM | - | 9,125 | 9,125 | 4,937 |
| PWR DF Packaging | kg iHM | 21,900 | - | - | 10,062 |
| SFR DF Packaging | kg iHM | - | 9,125 | 9,125 | 4,937 |
| EMT-derived HLW Packaging | kg FP | - | 915 | 960 | 495 |
| PWR UOX SNF Disposal | kg iHM | 21,900 | - | - | 10,062 |
| HLW Disposal | kg FP | - | 915 | 960 | 495 |
| Total LCF | $/GWe-year | 9.4 | 19.7 | 8.4 | 12.3 |

# Appendix D. Disposal Volumes of Spent Nuclear Fuel and High-level Wastes

The spent nuclear fuels (SNFs) and high-level wastes (HLWs) generated from advanced reactors or reprocessing facilities are considered the radioactive wastes for permanent disposal. In the present study, the discharge fuel (DF) assembly to be disposed of directly without any further treatment or reprocessing was considered SNF, while the non-recovered radioactive waste and byproducts from the treatment or reprocessing of the used fuel were considered as HLW.

Usually, the fuel assembly discharged from the once-through fuel cycle is considered SNF. However, [UFD 2014] indicated, "Enough information does not currently exist to evaluate the performance of direct disposal of sodium-bonded SNF in any geologic disposal concept. This waste type may require treatment regardless of the disposal concept." DOE decided to treat the sodium-bonded metallic fuel from EBR-II using electrometallurgical treatment (EMT) [DOE 2000]. This approach is also used in this analysis. The recovered materials and byproducts from EMT are considered as HLW; these include the salt-waste-containing fission products and non-recovered actinides and metal-waste-containing cladding and Zr in the fuel.

In the present study, the disposal volumes of SNF and HLW were calculated using the method used by the U. S. Nuclear Waste Technical Review Board (NWTRB) in the development of recommendations for how to move the nation's nuclear waste management program forward [NWTRB 2021]. The NWTRB calculated the disposal volume as the total volume of canisters needed to accommodate the SNF assemblies. In the Used Fuel Disposition (UFD) campaign, it was assumed that the SNF is disposed of using the DPC, and the HLW discharged from EMT is disposed of using the canister designed by Argonne [UFD 2014].

Table D. 1 shows the design parameters of the DPC used to evaluate options for permanent geologic disposal of SNF and HLW [UFD 2014]. The DPC can accommodate 32 PWR assemblies. The internal space for the PWR SNF assemblies is 6.24 $m^3$ (=32 x volume of 17x17 Westinghouse assembly [0.195 $m^3$]). Assuming that the SNF is stored in on-site spent fuel storage for five years before it is moved, a single DPC can hold a thermal energy of 41.6 kW (32 x decay heat of PWR assembly after 5-year post-irradiation cooling [1.3 kW/assembly]).

Table D. 1 Design parameters of DPC

| Parameter | Value |
|---|---|
| Diameter, m | 2.0 |
| Height, m | 5.0 |
| Volume, $m^3$/DPC | 16.1 |
| No. PWR assemblies per DPC | 32 |
| Thermal energy, kW/DPC | 41.6 [a)] |
| Space for SNF in DPC, $m^3$ | 6.24 [b)] |

a) Equivalent thermal energy to the decay heat of 32 PWR SNF assemblies with 50 GWd/t burnup after 5-year post-irradiation cooling.
b) Volume of 32 PWR assemblies.

A commercial canister for the HLW recovered from EMT has not yet been developed. Thus, the HLW production rate and canister information were obtained from the EBR-II and FFTF metallic fuel treatment [UFD 2014]. Table D. 2 shows the HLW generation from the EBR-II and FFTF metallic fuel treatment. Two types of HLW are created in the EMT process, i.e., salt (ceramic waste form) and metal (metallic waste form) waste. From the treatment of a unit metric ton of heavy metal (HM), about 1.98 metric tons of salt waste and 0.23 metric ton of metal waste are generated. Table D. 3 shows the design parameters of the HLW canister. The salt and metal wastes are stored in cylinder and disk forms, respectively. Then, both cylinders and metal disks are loaded into the HLW canister.

Table D. 2 HLW generation from EBR-II and FFTF metallic fuel treatment

| Parameter | Value | |
|---|---|---|
| EBR-II used fuel inventory, HMMT a) | 25.8 | |
| | Salt waste form | Metal waste form |
| Total HLW generation from EMT, t | 50.96 | 5.85 |
| Normalized HLW generation, t/HMMT | 1.98 | 0.23 |

a) HMMT = heavy metal metric ton.

Table D. 3 Design parameters of HLW canister

| Parameter | | Salt waste form | Metal waste form |
|---|---|---|---|
| Sub-components | Form | Cylinder | Disk |
| | Diameter, m | 0.5 | 0.381 |
| | Length or thickness, m | 1.0 | 0.127 |
| | Mass, t-HLW/cylinder or disk | 0.4 | 0.012 |
| Canister | Diameter (internal/outer), m | 0.58/0.61 | |
| | Length (internal/outer), m | 2.50/3.00 | |
| | Volume (internal/outer), $m^3$ | 0.67/0.88 | |
| | Capacity,<br>- No. cylinders or disks/canister<br>- Ton HLW/canister | <br>2<br>0.8 | <br>8<br>0.10 |

The number of SNF assemblies accommodated in a single DPC is limited by factors such as maximum temperature (or decay heat), available internal space to accommodate fuel assemblies (pebbles), and criticality control. In the present study, the number of SNF assemblies accommodated in a single DPC was determined by loading SNF assemblies into a single DPC until either total SFR assembly volume or decay heat equaled the volume or the decay heat of 32 PWR assemblies.[3] Then, the total DPC volume to accommodate a unit ton of SNF was calculated.

The resulting SNF assemblies per DPC and the disposal DPC volume after generation of unit electricity are provided in Table D. 4. The sodium-free metallic fuel assembly can be treated as SNF because it is suitable for direct disposal. Owing to higher electricity generation per unit initial HM mass, the DPC can accommodate more SNF and the SNF disposal volume is smaller than the disposal volume of the PWR SNF. For the HTGR, the disposal volume of all pebbles is a factor of ~3.9 larger than that of PWR SNF because the pebble contains a large fraction of carbon. The disposal volume of HLW generated from the sodium-bonded metallic fuel processed with EMT is provided in Table D. 5 . Compared to the direct

[3] The fuel assembly dimensions of Natrium reactor are different those of a PWR: i.e., assembly has a hexagonal shape with smaller pitch and taller length. Thus, the current DPC design is designed to accommodate PWR assemblies and a new DPC design is required to accommodate Natrium assemblies. Tentatively, in the present work, the internal volume of the new DPC design is assumed to be the same to the internal volume of the current DPC.

disposal of sodium-free metallic fuel, the sodium-bonded metallic fuel produces about 23% higher volume (19.7 vs. 16.0 $m^3$/GWe-year) for disposal.

Table D. 4 SNF disposal volume

| | PWR | SFR with sodium-free fuel | HTGR with pebble |
|---|---|---|---|
| Fuel (HM+FP) mass, t/assembly or t/pebble | 0.442 | 0.106 | 7.0E-6 [a)] |
| Assembly or pebble volume, $m^3$ | 0.195 | 0.108 | 4.86E-5 [b)] |
| Decay heat at 5 years, kW/assembly | 1.30 | 0.41 | N/A |
| SNF assemblies or pebbles per DPC, #/DPC | 32 | 58 | 128,500 [c)] |
| DPC volume per unit ton SNF, $m^3$/t-SNF | 1.14 | 2.62 | 17.92 |
| SNF production, t/GWe-year | 21.9 | 6.1 | 5.4 |
| Volume, $m^3$/GWe-year | 25.0 | 16.0 | 97.1 |

a) Mass per pebble.
b) Volume of a pebble.
c) Number of pebbles.

Table D. 5 HLW disposal volume

| | SFR with sodium-bonded fuel |
|---|---|
| Fuel (HM+FP) mass, t/assembly | 0.104 |
| Assembly volume, $m^3$ | 0.108 |
| HLW generation, t/GWe-year<br>- Salt waste<br>- Metal waste | <br>18.0<br>2.1 |
| No. HLW canisters/GWe-year | 23 |
| Volume, $m^3$/GWe-year | 19.7 |