Report, General

A Feasibility Study on the Recycling of Used CANDU Fuel

Research and Development

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1.   EXECUTIVE SUMMARY

1.1   Mandate

This is a technical report in response to the Ontario Ministry of Energy's (MoE) Request for Bids Contract awarded to Canadian Nuclear Laboratories (CNL) for a high level feasibility study to investigate advanced nuclear recycling options for Used CANDU Fuel in Ontario.

The MoE's primary focus is to understand the potential challenges/benefits of recycling Canada's forecasted inventory of used nuclear fuel[1] (UNF) from CANDU natural uranium (NU) reactors. Natural Resources Canada (NRCan) also participated in the study to gain a better understanding of the potential for recycling CANDU UNF, as well as the implications if such an approach were to be considered. CNL sub-contracted National Nuclear Laboratory Ltd (NNL) to support this analysis.

Sections 2, 3 and 4 of this report provide an overview of used nuclear fuel in Canada, the options under consideration, and reprocessing technologies for used fuel recycling. Sections 5 to Section 12 analyze the options based on eight evaluation criteria that were mutually agreed to by the MoE, NRCan and CNL. The results of this analysis are summarized in Section 13, which is followed by appendices that provide additional technical details to support the findings of the report.

From the recommendations in the initial bid provided by CNL, the MoE defined the scope of the study to include a timeframe needed to recycle all CANDU UNF bundles, taking into account the following:

- Electrical power generated by recycling these nuclear used fuel bundles;

- Overnight Capital Cost and life cycle cost for the whole recycle system;

- Volume of high level waste removed by the recycle system, and volume of low and intermediate level waste produced and stored in a deep geologic repository (DGR);

- Other parameters in the Request for Bids (RFB) – e.g., environmental impact; degree of proliferation resistance etc.; and

- Other social economic benefits – jobs, supply chain, innovation for advanced technologies to retain critical skills in Canada, potential exportable technologies for Canada and Ontario.

As a result, the feasibility study is a 150 year study period from 2015 to 2165, based on a Canadian inventory of 103,000 metric tonnes of CANDU UNF, and three reprocessing and recycling scenarios for consideration. On the proposed recycling system options, MoE and NRCan requested that the study adopt CNL's bid proposal to analyze both reprocessing and

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[1] Used nuclear fuel as defined in this report is synonymous with nuclear fuel waste as defined in the Canadian Nuclear Fuel Waste Act.

recycling of Canada's UNF inventory through CANDU reactors and sodium-cooled fast reactors (SFRs) summarized in Table 1-1.

1.2   Option Description

CANDU (CANadian Deutrium Uranium) is a Canadian designed reactor system, with twenty-two reactors built and operated in Canada and eleven internationally. Although the reactor was intended to be fuelled using natural uranium (NU), the flexible design allows for a wide variety of fuels and can use recycled plutonium from CANDU UNF. Given Canada's extensive knowledge of CANDU technology, and the existing supply chain located in Ontario, this reactor was selected as Option 1 for analysis. Option 1 is to recycle plutonium from reprocessing CANDU UNF in order to make a new fuel to be reused with the existing fleet of 12 CANDU Reactors in Ontario (i.e., Darlington and Bruce nuclear stations).

The second technology selected for analysis was the SFR, which has had limited international experience, though earlier pioneering development occurred in Russia, France and Japan and is still under development through the Generation IV International Forum.[2] Unlike the commercial reactors operating today, an SFR operates in the fast spectrum (high energy neutrons from the fission process), which has benefits for "burning" transuranic materials (i.e., long-lived radiotoxic elements which are present in UNF), such as plutonium, compared to other technologies. For this reason, many experts have proposed that SFRs could improve the characteristics of the waste prior to disposal – meaning reducing waste volume and radiotoxicity. Two SFR technologies were selected for analysis: Option 2 considers a large (1,500 MWe) SFR based on the ASTRID (Advanced Sodium Technological Reactor for Industrial Demonstration[3]) reactor prototype, and Option 3 uses a smaller (380 MWe) SFR that was based on the PRISM (Power Reactor Innovative Small Module[4]) reactor concept. Three variations of Options 2 and 3 were included in the analysis to address different deployment schemes from a burn-once cycle through the SFR to a closed cycle through the SFR consisting of repeated recycling of SFR nuclear used fuel.

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[2] The Generation IV International Forum (GIF) is a co-operative international endeavour by 13 countries (including Canada) which was set up to carry out the research and development needed to establish the feasibility and performance capabilities of the next generation nuclear energy systems.

[3] The ASTRID reactor is a Commissariat à l'énergie atomique et l'énergie alternative (CEA) design.

[4] The PRISM reactor is a GE Hitachi design.

Table 1-1
Options to be Analyzed

Recycling System Option	Reactor Techology for Recycling Used CANDU Fuel	Used Fuel Reprocessing Method	Reactor Fuel Fabrication Method
Option 1	Seventeen[5] 818 MWe Refurbished CANDU	Co-Ex (Demonstrated)	MELOX (Commercialised)
Option 2a	One 1,500 MWe ASTRID SFR	GANEX (Demonstrated)	MELOX (Commercialised)
Option 2b	Two 1,500 MWe ASTRID SFRs		
Option 2c	Eight 1,500 MWe ASTRID SFRs		
Option 3a	Four 380 MWe PRISM SFRs	Pyro-Processing (Novel)	Integrated System (Novel)
Option 3b	Eight 380 MWe PRISM SFRs		
Option 3c	Forty 380 MWe PRISM SFRs		

In Options 2a, 2b, 3a and 3b, the number of reactors was defined based upon discussions with MoE & NRCan to illustrate how much recycling is achieved through the SFR burn-once options.

In Options 1, 2c and 3c the MoE and NRCan did not define the number of reactors to be analyzed, instead the system was configured to better utilize Canada's UNF including a closed fuel cycle recycling system.

Eight mutually agreed upon evaluation criteria were used to generate a high level assessment of the challenges and benefits associated with each option (Table 1-2).

Table 1-2
Evaluation Criteria

Criteria	Description	Key Measure(s)
Environmental	Impact of the reactor and supporting facilities on the environment: • Radiological discharges • Reprocessing wastes • Implications on marine environments	Volume of UNF and high level waste (HLW) produced
Safety and Licensing	Effort required and perceived barriers to licensing the required facilities: • Existing experience (in Canada, internationally) • Known safety concerns • Outstanding technical challenges	Experience and technology maturity

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[5] Eight reactors at the Bruce site and four reactors at the Darlington site were considered in this study. In order to recycle all CANDU UNF, five reactors required an additional refurbishment for a total of 17 equivalent reactors.

Criteria	Description	Key Measure(s)
Sustainability	Ability to produce electricity without drawing from Canada's existing uranium reserves.	Electricity produced from recycled material
Non-Proliferation	Barriers in place to impede the diversion or undeclared production of nuclear material. • Accessibility and usefulness for weapons manufacture • Ease of process modifications to increase the purity and/quantity of Pu produced	Inherent proliferation resistance of the reprocessing technologies and fuels
Community and Social Considerations	Analysis of the public perception of the risks and benefits, with a focus on tools for measurement and growth of community support.	N/A[6]
Costs	Estimated costs of the capital, operating and decommissioning costs for all new facilities included. • Reactor costs • Reprocessing & fabrication costs • Storage costs	Total investment measured in $/MWh produced (undiscounted)[7]
Waste Disposal	Effort required to properly dispose of all of the waste from the following streams: • UNF in storage at the end of the study period • Waste from reprocessing and fuel fabrication • Reprocessed Uranium • Decommissioning Waste	Total cost to dispose of all waste generated
Economic Impact	Analysis of the potential impact on the economy; in direct development and through supply chain growth.	GDP impact and person years of employment

The last three criteria: cost, waste disposal and economic impact produced a set of economic results for each option (Table 1-3 and Table 1-4).

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[6] Given the lack of Canadian specific societal data on the options analyzed in this study, a ranking was not possible. Further study and public surveys are needed.

[7] Costs are undiscounted given the difficulty in determining an appropriate discount rate for multiple capital investments and cashflows at different intervals over the 150 year study period for each option.

1.3   Summary of Findings

Table 1-3
Waste Disposal Costs

	ILW	RU	DGR Costs ($M)			
			UNF and RepHLW[a]		Total	
			Lower Bound	Upper Bound	Lower Bound	Upper Bound
Reference	54		24,670		24,725	
Option 1	2,080	6,633	25,698		34,411	
Option 2a	262	3,959	22,761	25,847	26,982	30,067
Option 2b	513	7,918	20,856	27,033	29,287	35,464
Option 2c	774	10,098	20,938	25,126	31,809	35,997
Option 3a	301	4,830	22,474	25,747	27,606	30,879
Option 3b	603	9,661	20,272	26,813	30,535	37,077
Option 3c	1,024	11,205	22,131	25,135	34,254	37,258
a: The range provided is based on a low and high estimate for CANDU-NU UNF in SFRs using LWR UNF and weapons PU as proxies, since CANDU-NU UNF in SFRs data was unavailable.						

The lower bound on the total disposal costs for every option for recycling CANDU UNF is greater than the reference case of placing all CANDU UNF in the DGR. While all of the SFR options may reduce the cost of the HLW DGR relative to reference system, this reduction in cost is less than the increase in costs due to the disposal of ILW and RU.

Table 1-4
Economic Results

	Electricity Generated TWh	Undiscounted Cost ($ Billion)	Unit Cost $/MWh	Person Years of Employment (Thousands)	Total GDP Impact ($ Billion)
Reference[8]	0	0	N/A	8,382	708
Option 1	2,959	463	155	11,399	963
Option 2a	671	107-146	160-217	2,238-2,861	188-241
Option 2b	1,341	215-291	160-217	4,056-5,302	342-448
Option 2c	4,024	427-680	106-169	7,556-11,686	639-989
Option 3a	680	119	176	2,445	205
Option 3b	1,359	237	175	4,448	375
Option 3c	5,436	758	139	12,988	1,100

The economic results illustrate two high level trade-offs:

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[8] The reference scenario assumes the current status quo APM DGR adopted by NWMO. Any benefit from the CANDU-NU reactors or UNF resulting from additional reactors that may be built/operated in Canada during the study period was excluded from the reference scenario and the options analysis.

- The first is the increased monetary investment required versus the increased economic impact (GDP and employment), and

- The second is the increased electricity generated versus the increased disposal costs.

The results from the remaining criteria which were analyzed all indicate that the increased risk associated with the newer and more novel approaches used in Options 2 and 3 introduce significant challenges, as illustrated in the following matrix (Figure 1-1):

Impacts	Environmental	Safety & Licensing	Sustainability	Non-Prolifeation	Cost	Waste Disposal	Economic Impact
Option 1							
Option 2a							
Option 2b							
Option 2c							
Option 3a							
Option 3b							
Option 3c							

Best Option   Worst Option

Figure 1-1 Ranking Summary

Note: The matrix shows a ranking of each option within each of the evaluation criteria. No assumptions are made about the priority of each criterion relative to one another; therefore the matrix does not represent an overall ranking of the options.

These novel approaches do however allow for a tremendous amount of power to be generated (Options 2c and 3c) without further depleting Canada's uranium reserves. This would improve the sustainability of Canada's fuel cycle while also ensuring Ontario has a reliable source of baseload electricity.

Should Ontario consider the deployment of one of the options analyzed in this study, a number of technical challenges would need to be overcome:

1. Each option requires a significant upfront investment to establish the infrastructure needed to recycle fuel (i.e., new or refurbished reactors, and fuel reprocessing and fabrication facilities. Options would also require numerous investments over a long timeframe to maintain/replace this infrastructure as it ages.

2. New waste streams would be created during reprocessing and recycling that do not currently exist in Canada. Waste owners would be required to find a long-term solution for all waste types not covered under the Nuclear Fuel Waste Act.

3. A time horizon of 150 years, ending in 2165, was used for this study. This introduced a number of issues such as; multiple refurbishments of CANDU reactors (Option 1) which

will be technologically challenging and may require additional R&D, recycling systems are evolving and it is difficult to predict how they may progress over the study period

4. SFRs have yet to achieve commercial success. Most of the fast reactors built within the past decade faced significant reliability and safety challenges that have seriously affected performance. More R&D will be required before deployment in Canada.

5. Commercial reprocessing has never been carried out in Canada. Significant knowledge and skills must be developed or introduced to ensure that reprocessing facilities could be safely installed, licensed, regulated, operated and eventually decommissioned. Note: the technologies considered in this report range from unproven to internationally demonstrated designs.

6. Recycling will increase proliferation risk compared to the once-through natural uranium fuel cycle. The Licensee (Operator) and the Canadian Nuclear Safety Commission (CNSC) would have to work closely with the International Atomic Energy Agency (IAEA) to ensure adequate safeguards are put into place.

In addition to the above-mentioned technical challenges, there are potential social and public perception challenges (i.e., social license) which are not included in the scope of this report.

If these challenges are overcome, Ontario could experience benefits related to recycling such as:

1. A reliable source of baseload nuclear electricity, without further depletion of the Canadian uranium reserves.

2. Potential for less long-lived and/or less long- term radiotoxic nuclear waste destined for disposal.

3. Growth in Ontario's nuclear knowledge and supply chain, including potential for international exports of certain components produced by the Canadian supply chain.

4. Opportunities for Canada to participate as a leader/world expert in international deployments of similar recycling systems.

As technologies develop, it is expected that there will be many opportunities to optimize the recycling system, further improving the characteristics of the final nuclear waste material requiring disposal. In addition, further studies may be undertaken on other potential recycling systems which may not require reprocessing and on technology innovations that could manage final nuclear waste material which could result in lower storage costs.

2.   CANADA'S USED NUCLEAR FUEL

2.1   Nuclear Power in Canada

To date, twenty-two power-generating CANDU nuclear reactors have been built and operated in Canada. Nineteen of these reactors are still in operation today, generating more than 18% of Canada's electricity. All nuclear power reactors in Canada operate on an open natural uranium (NU) fuel cycle, where fuel is irradiated once and then the used nuclear fuel (UNF) is storage awaiting permanent disposal. According to the latest Nuclear Waste Management Organisation (NWMO) update on UNF in Canada, there was a total of 50,000 tonnes Heavy Metal (tHM) of UNF[9] in storage at the end of June 2014 [2].

The 2013 version of Ontario's Long-Term Energy Plan (LTEP), which details the provinces electricity generation plans through to 2032, has no plans for nuclear new builds, however several refurbishments are planned. These plans have been updated since 2013. The current reactor refurbishment scheduled represent updates to the 2013 LTEP and can be found at [3] for Darlington units and [4] for Bruce units.

The NWMO provides a forecast of the total amount of CANDU UNF that will have accumulated at the end of operation of the current fleet of reactors in Canada. The low, reference and high forecasts are 69,000 tHM, 89,000 tHM and 103,000 tHM, respectively. These cases are based on the number of reactors in the current fleet that will be refurbished and excludes any UNF arising from any newly built reactors. Based on a recommendation from the MoE this study will use the high forecast of 103,000 tHM for the amount of the CANDU UNF in 2061, at which time the last of the current CANDU reactors will be offline.

2.2   The Composition of Canada's UNF

The composition of UNF from NU fuelled CANDU reactors, detailed in Table 2-1, has four main components: uranium (U), plutonium (Pu), minor actinides (MAs) and fission products (FPs).

Table 2-1
A Representative Composition of Canada's UNF at Discharge from the Reactor

Component	CANDU (wt. %)
Uranium	98.89
Plutonium	0.38
Minor Actinides	< 0.01
Fission Products	0.72

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[9] [9] Canada's UNF is in oxide from, principally UO_2. When discussing the weight of UNF it is usual to ignore the weight of the oxygen (about 11.5%) and specify only the weight of the 'heavy metal' (HM). Heavy metal consists of isotopes of uranium, plutonium, minor-actinides etc. The unit of UNF in this report would therefore be the tHM, or ktHM (tonne or kilotonne of heavy metal).

2.2.1 Uranium

Uranium is by far the largest proportion of CANDU UNF. The radioactivity of the uranium is dominated by that of ^{238}U and its decay products. However, the ^{238}U is the least radioactive component of the CANDU-NU UNF more than a hundred thousand years. The isotopic composition of this uranium is detailed in Table 2-2.

- ^{238}U, the largest component of this uranium, is a fertile isotope that is the starting material for the production of fissile plutonium isotopes when irradiated in a reactor.

- ^{235}U, the only fissile isotope in this uranium, is present in such a small proportion that this uranium cannot be used as reactor fuel unless it is enriched or supplemented with material that has a sufficiently high proportion of fissile nuclides. The proportion of ^{235}U in this uranium, by coincidence, is similar to that of ^{235}U in the depleted tails of uranium enrichment plants and therefore the uranium is, at most, as valuable as these tails.

- ^{236}U is produced via the capture of a neutron by ^{235}U and presents a hindrance to the use of this uranium as reactor fuel because it captures neutrons without yielding a fission reaction or a fissile nuclide. The presence of ^{236}U reduces the value of the uranium in Canada's UNF relative to depleted uranium (DU) tails, which do not contain a significant amount of ^{236}U.

Table 2-2
A Representative Composition of the Uranium in Canada's UNF at Discharge

	CANDU (U wt. %)
^{235}U	0.24%
^{236}U	0.07%
^{238}U	99.69%

2.2.2 Fission Products

The second largest proportion of the UNF is made up of fission products (FPs), which represent the most radioactive component of the UNF for the first few decades after discharge from the reactor. As their name suggests, fission products are nuclides that are created from the fission of nuclides in the reactor. None of the FPs are fissile or fertile, meaning that they do not readily fission, or lead to the creation of fissile nuclides in a reactor. In fact, the build up of FPs, many of which absorb neutrons, eventually limits the residence time of fuel in a reactor.

2.2.3 Plutonium

The Pu in UNF is primarily produced via the capture of neutrons by ^{238}U in the reactor and is the element in UNF that contains the highest proportion of fissile isotopes, making it the most valuable as reactor fuel. The composition of plutonium in Canada's UNF is detailed in Table 2-3. The fissile isotopes of plutonium, ^{239}Pu and ^{241}Pu, make up as much as 74% of the plutonium in the UNF, although in a practical reprocessing scheme the proportion will be less due to the

relatively short half-life of ^{241}Pu (~14 years). All the plutonium isotopes are radioactive, however, and their decay dominates the radioactivity of all the UNF components between 1000 and 100,000 years. The high proportion of ^{239}Pu makes the UNF a more attractive than average target with respect to diversion for the proliferation for nuclear weapons. However, the presence of a significant fraction of ^{240}Pu, whose decays include neutron emission, leading to pre-detonation in nuclear weapons, makes the plutonium far below what is considered 'weapons grade' and keeps it firmly in the 'reactor grade' category,

The relatively short half-life of ^{241}Pu, which decays to the MA ^{241}Am, means that the amount of fissile plutonium in Canada's UNF depends on the length of time that the UNF has been in storage. Isotope ^{239}Pu, on the other hand, has a half-life of over 20,000 years so its presence in Canada's UNF is relatively stable. In the NWMO high scenario, neglecting the ^{241}Pu, by the year 2054 Canada's entire stock of UNF will contain a total of (103 kt UNF) x (0.38% Pu/UNF) x (69.24% ^{239}Pu/Pu) = 271 t of ^{239}Pu.

Table 2-3
A Representative Composition of the Plutonium in Canada's UNF at Discharge from the Reactors

	CANDU (Pu wt. %)
^{238}Pu	0.09%
^{239}Pu	69.24%
^{240}Pu	24.93%
^{241}Pu	4.72%
^{242}Pu	1.02%

2.2.4   Minor Actinides

The final component of Canada's UNF to be discussed here are the minor actinides (MAs). The MAs are comprised primarily of isotopes of americium and curium with neutron high capture cross sections, and hence have no use as reactor fuel with respect to generating power. Although the proportion of MAs in Canada's UNF is quite small by mass, the longer-lived ones do make a significant contribution to the radiotoxicity of the UNF for over 1000 years. It is for this reason that much research and development on transmuting MAs from long-lived MAs to short-lived MAs via irradiation in a reactor has been conducted.

2.3   Deep Geological Repository

Canada's current plan, through the NWMO, is to permanently store the UNF in a deep geological repository (DGR). However, a requirement has been placed on this DGR that the UNF remains retrievable for a period of time, 300 years according to the current plan, following emplacement to allow for recycling of UNF if desired. The NWMO plan for a DGR in a suitable rock formation will be implemented over several decades.

3.   OPTIONS TO BE ANALYZED

Three recycling scenario's for the future of nuclear power in the Province of Ontario have been investigated. An additional option involving the use of molten salt reactors (MSRs) was considered briefly, and a general discussion of the advantages and disadvantages of the MSR technology is included in Appendix A. Other reprocessing options exist, which could be the focus of future study.

Three variations of options two and three were included to create a total of seven options:

- Option 1 – The CANDU UNF would be reprocessed in Ontario via the CO-EXtraction (COEX) process. Pu and a proportion of the U products would be recycled in thermal mixed oxide (MOX) fuelled CANDU reactors. Used CANDU MOX fuel would then be stored until sufficiently cool and then disposed of at a DGR along with any remaining U, immobilised fission products and intermediate level wastes from reprocessing (i.e., irradiated cladding tubes and other assembly furniture). For analysis, the Bruce and Darlington reactor sites are assumed to each contain a COEX reprocessing facility, fuel fabrication facilities (the number of which is dependent on fabrication requirements), the refurbished CANDU reactors, storage facilities and a fuel encapsulation plant.

- Option 2a – The CANDU UNF would be reprocessed in Ontario via the Grouped ActiNide EXtraction (GANEX) process. Transuranic (TRU) and U products would be recycled as MOX fuel in a 1,500MWe (MW of electrical power) sodium-cooled fast reactor (SFR) in a burner[10] configuration with a once through fuel cycle. One reactor site (Bruce or Darlington) would contain a GANEX reprocessing facility, a fuel fabrication facility, a SFR, storage facilities and a fuel encapsulation plant. Used SFRMOX fuel would be stored until sufficiently cooled in a DGR. All remaining CANDU UNF and fast reactor UNF, reprocessed U, immobilised fission products and intermediate level wastes from reprocessing would likely be disposed of in the DGR (this is true for all scenarios).

- Option 2b – The CANDU UNF would be reprocessed in Ontario via the GANEX process. TRU and U products would be recycled as MOX fuel in two 1,500MWe SFRs, both in the burner configuration with once through fuel cycles. Used SFRMOX fuel would be stored until sufficiently cool in a DGR. The Bruce and Darlington reactor sites as assumed to each contain a GANEX reprocessing facility, a fuel fabrication facility, a SFR, storage facilities and a fuel encapsulation plant for the analysis. This scenario considers reprocessing the majority of the CANDU UNF stockpile within 50-60 years.

- Option 2c – The CANDU UNF would be reprocessed in Ontario via the GANEX process. TRU and U products would be irradiated in a once-through fashion as SFRMOX in the

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[10] A burner configuration is one in which the fissile content of the UNF is less than that of the initial fuel loading. For fast reactors, the opposite effect is possible, in which case the reactor is termed a breeder. Breeding is usually affected by having 'blanket' of fertile material surround the core. Fissile material creation in the blanket then offsets burning in the core.

first generation of fast reactors comprising two 1,500 MWe 'burner' units. A second generation fleet of reactors would follow during which time SFR UNF and any remaining CANDU UNF could be continually recycled. The second generation SFR fleet (of six SFRs) has been sized such that the existing SFR UNF and CANDU UNF stockpiles would be fully utilized. The Bruce and Darlington reactor sites are assumed to each contain a CANDU UNF GANEX reprocessing facility, a fuel fabrication facility to supply fuel to the first fleet of reactors and SFRs for this analysis. A dedicated SFR reprocessing system would also be located at each site to reprocess the used SFRMOX fuel to support multiple recycle of fast reactor fuel. The SFR UNF encapsulation plants would be present at both reactor sites to encapsulate fuel remaining after reprocessing operations have ceased. This option is concerned with the closing of the fuel cycle and thus second generation reactors would be a multi-recycle system.

- Option 3a – The CANDU UNF would be reprocessed in a pyro-processing facility located at the Darlington or Bruce site, with the U and TRU metal products then recycled in four 380 MWe metal-fuelled 'burner' fast reactors. The fuel would be irradiated in a once-through fashion, the resulting metallic SFR UNF would be stored pending disposal at a DGR. It is assumed that one reactor site (Bruce or Darlington) would contain a pyroprocessing facility, a fuel fabrication facility, PRISM-type reactors, storage facilities and a fuel encapsulation plant for this analysis.

- Option 3b – The CANDU UNF would be reprocessed in a pyro-processing facilities located at the Darlington and Bruce sites, with the U and TRU metal products then recycled into eight 380 MWe metal-fuelled fast reactors, all in a burner configuration. The fuel would be irradiated in a once-though fashion, the resulting metallic SFR UNF would be stored until sufficiently cool for a DGR. It is assumed that four reactors, a pyro-processing facility, a fuel fabrication facility, storage facilities and a fuel encapsulation plant would be located at each of the sites for this analysis.

- Option 3c – The CANDU UNF would be reprocessed in pyro-processing facilities located at the Darlington and Bruce sites, with the U and TRU metal products first recycled in eight 380 MWe metal-fuelled 'burner' fast reactors. The Bruce and Darlington reactor sites would each contain a pyroprocessing facility, a fuel fabrication facility, PRISM-type reactors, storage facilities and a fuel encapsulation plant. The resulting UNF would be stored pending a second, much larger generation of 380-MWe SFRs. During this second phase, the fuel would be burned in a multi-pass fashion with a dedicated FR fuel multi-cycle recycle system present at each site to reprocess the metallic SFR UNF. The second fleet of metallic SFRs is sized such that the existing metallic SFR UNF stockpile and any remaining CANDU UNF would be fully utilized. Final cores and any remaining UNF would be stored pending disposal and then disposed of at a DGR.

By 2061, 103,000 tHM of CANDU UNF would be in storage throughout Canada; this is assumed to be the feedstock for all reprocessing options discussed [2]. Only the existing Darlington and Bruce sites are considered for new facilities that may be required as part of the seven options.

The study period will encompass 150 years.

Each option is to be assessed against a range of criteria to determine the feasibility and potential benefits. Environmental impact, economic benefit, proliferation resistance, licensability, sustainability and the timeline for implementation are just a few of the criteria that the options will be judged against in Section 5 to Section 12 of this report.

Further technical details of each of the reprocessing options are located in Appendix D.

4.   RECYCLING

4.1   Reprocessing and Recycling

Nuclear reprocessing is the separation of used fuel assemblies into various product and waste streams. For example, LWR used fuel assemblies are currently being reprocessed to produce two product streams, plutonium and uranium. During reprocessing, the used fuel assembly is dismantled and the used fuel is treated to recover useable material from non-useable or waste material. The non-useable material may be in multiple forms (e.g., high-level and intermediate-level waste) and would have to be disposed of safely. In addition, aerial and liquid effluent discharges arise as a result of reprocessing operations.

The recycling of nuclear fuel is the extraction of useable material from used fuel assemblies and the subsequent reuse of that material in reactors as reuse fuel. Recycling describes an integrated life-cycle process that results in the production of new reactor fuel from used nuclear fuel. Reprocessing is stages in the recycling lifecycle.

4.2   Why Recycle

There are a number of potential benefits that can be realized through recycling UNF:

- To achieve a reduction in NU consumption per unit generated energy, which may have economic benefits if the price of NU becomes too high.

- The earths' economically accessible supply of uranium is sufficient to run the current fleet of reactors for approximately 100-200 years. Recycling UNF and the introduction of fast reactors which can be set in breeder mode could significantly extend and enhance the sustainability of this supply.

- To achieve a reduction in the disposal requirements; which may have economic benefits if the reduction in disposal cost is significant.

Recycling UNF in reactors would generate high level waste (HLW) with characteristics that differ from CANDU UNF. The composition of HLW that is generated from recycling UNF depends on various fuel cycle options, such as the reprocessing technology, the composition of recycled fuel and the reactors that are fuelled with it. However, regardless of option, the proposed electricity production would result in a greater amount of FPs being sent to disposal than what is contained in the original UNF.

Only the U.K. and France currently operate commercial reprocessing facilities; a few additional facilities are scheduled to start as shown in Table 4-1.

Table 4-1
Current and Planned Reprocessing Facilities by Country

Country	Location	Facility	Fuel	Technology	Operation Start	Through Put (tHM/year)
China	Lanzhou		LWR	"Advanced Technology"	2020	800
France	La Hague	UP2	LWR	PUREX	1967	1000
	La Hague	UP3	LWR	PUREX	1990	1000
Japan	Tokai-mura	JAEA TRP	LWR	PUREX	1977	90
	Rokkasho-mura	JNFL RRP	LWR	Modified PUREX	2016	800
Russian Federation	Krasnoyarsk	RT2	WWER 1000	TBD	2025	1500
	Krasnoyarsk	Demonstrative Facilities	VVER-1000, RBMK	TBD	2013	100
United Kingdom	Sellafield	Magnox Reprocessing Plant	GCR	PUREX	1967	1500
	Sellafield	THORP	LWR/AGR	PUREX	1994	900

When considering UNF recycling options in Canada, due to the composition of CANDU UNF is at a disadvantage versus countries with LWR UNF. When starting with Canada's UNF, all recycling options would at the very least include the recycling of plutonium because it has the only significant quantity of fissile isotopes in CANDU UNF and because it represents most of the medium-term (1000 to 100,000 years) radiotoxicity of Canada's UNF. In addition, minor actinides (principally Am-241) are also recycled in the fast reactor scenarios.

Traditionally, MOX fuel has been fabricated by blending Pu with DU to achieve the desired fissile content in initial reactor fuel. Since Canada has no enrichment facilities, it does not have domestic DU stockpiles. The DU could be imported; however, this would increase the volume of nuclear material required to be disposed of in Canada. This could be avoided using the reprocessed uranium (RU) from the CANDU UNF instead of DU. This would present an additional challenge as the fuel would have been irradiated and therefore the separated U might require additional storage and shielding for fabrication (in addition to the requirements needed for handling Pu).

4.3 Reprocessing Technologies

4.3.1 CO-Extraction

Co-extraction (CO-EX, Figure 4-1) [5] developed by AREVA is an evolution of the existing plutonium-uranium extraction (PUREX) process, and would be used in Option 1. It utilises the existing solvent used in PUREX, however CO-EX co-extracts some U with Pu to form a combined product. This increases the proliferation resistance of the Pu product by increasing the effort required to obtain pure Pu, thus decreasing the attractiveness of the material for non-civil use [5]. The bulk of the uranium would be processed into a secondary U product (a result of the large ratio of U to Pu in UNF). For this study the required quantity of uranium is assumed to be routed to MOX production, the remainder would be stored pending a future review to determine the most suitable management route.
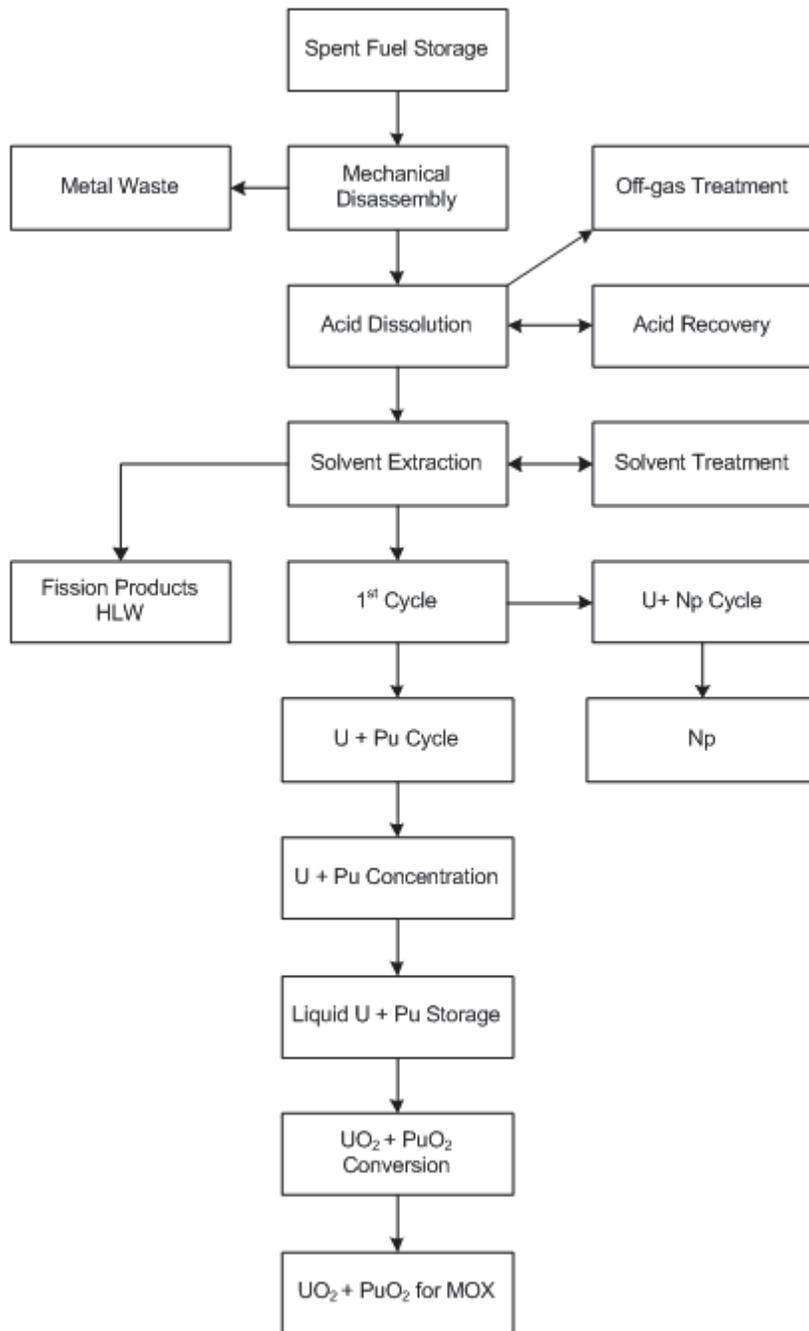
Figure 4-1 Simplified COEX Process

4.3.2   GANEX Reprocessing

The GANEX process proposed for Option 2 (Figure 4-2) [6] is an advanced aqueous reprocessing technology that has evolved from the existing PUREX process and is undergoing international research and development. As proposed, the GANEX concept flowsheet produces two major

product streams: a uranium dioxide powder and a single transuranic (TRU) product containing the plutonium and minor actinides. GANEX utilises novel solvents to extract the U as one product and the Pu and transuranics as a second product [7]. This improves the proliferation resistance of reprocessing UNF compared with the standard PUREX method since the plutonium is in a complex, impure form (it is less attractive for non-civil compared with other Pu sources.
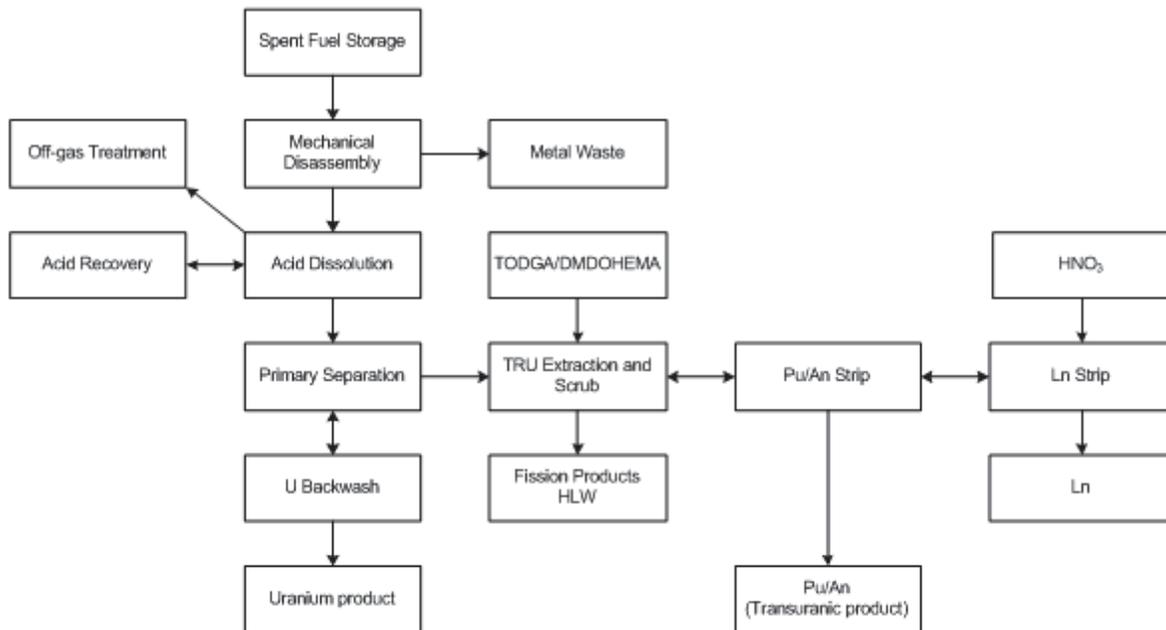
Figure 4-2 A GANEX Process Concept

4.3.3   Pyro-Processing

Pyrochemical processing (Pyro-processing) involves high-temperature reprocessing of spent fuel without utilising aqueous media (solvents), and is proposed for Option 3. Almost all commercial reprocessing facilities utilise aqueous (hydrometallurgical) processes such as PUREX. A number of methods for pyro-processing have been proposed, for example electrorefining and electrowinning; the most mature is molten chloride salt reprocessing using electrorefining, as has been used in the USA for the treatment of used fuel from the EBRII test reactor (Figure 4-3).

Pyro-processing extracts uranium, plutonium and minor actinides from UNF and is usually considered when manufacturing metallic fuel for use in fast reactors [8]. Pyro-processing utilises an electric current to dissolve the used fuel into a bath of molten salt at approximately 500°C. Typically uranium deposits onto a solid cathode and plutonium, minor actinides and some uranium deposit onto a liquid metal cathode. Once the deposition process is complete, the cathodes are removed and cleaned of residual salt. Solid metal ingots of fuel are cast and any remaining salt is recycled back into the electrorefiner [8]. Pyro-processing technology does

not require moderating material, such as water, present in aqueous reprocessing plants. As a result, criticality risk can be significantly decreased meaning material of a higher fissile content can be handled.
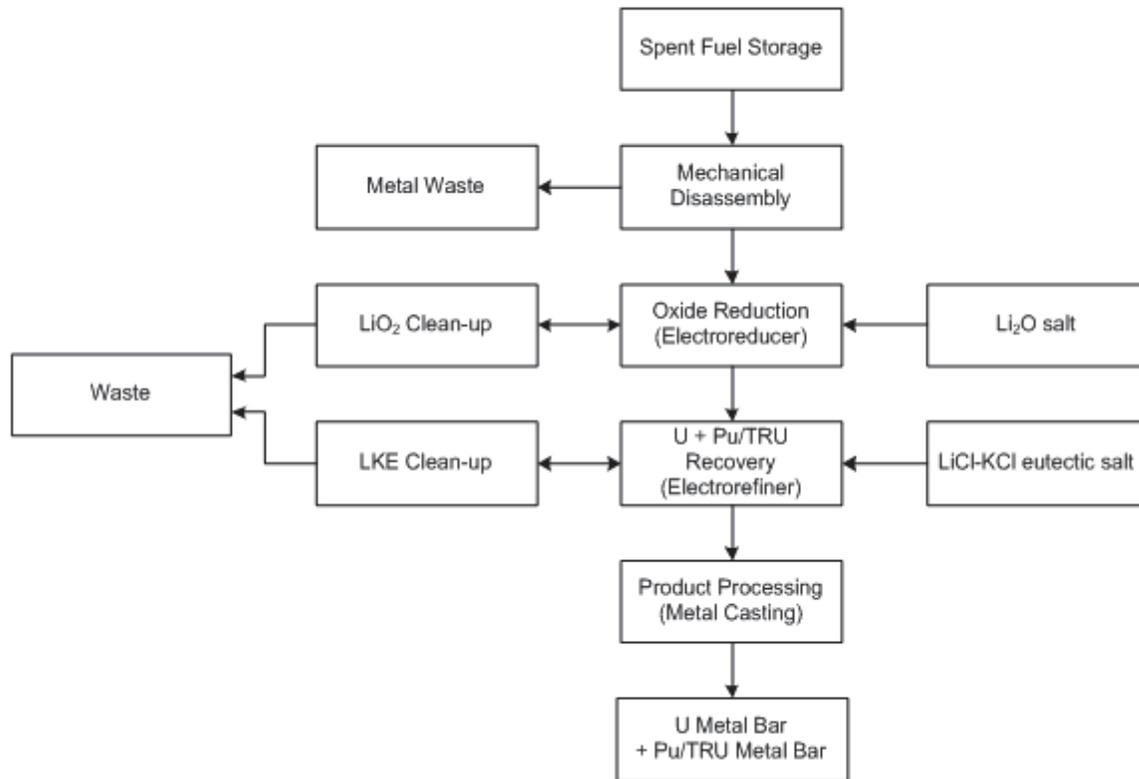
Figure 4-3 A Pyro-Processing Process Concept

4.4   Reactor Technologies

4.4.1   CANDU Reactors

CANDU stands for CANadian Deutrium Uranium and it is a Canadian designed PHWR (Presurrized Heavy Water Reactor) that uses heavy water (deuterium oxide) for moderator and coolant and natural uranium for fuel. The use of natural uranium for fuel eliminates the need for an enrichment facility and makes fuel fabrication easier and less costly. Heavy water as a moderator is highly efficient due to its low neutron absorption and affords the highest neutron economy of all commercial thermal reactors. As a result the chain reaction in the reactor is possible using only natural uranium fuel. The CANDU reactor core design allows on power refuelling which enables high capability factors.

CANDU reactors are exceptionally safe. The safety systems are independent from the rest of the plant, and each key safety component has three backups. Not only does this redundancy

increase the overall safety of the system, but it also makes it possible to test the safety system while the reactor is operating under full power.

4.4.2   Sodium-Cooled Fast Reactors

Sodium cooled fast reactors (SFRs) (Figure 4-4) [9], are part of the generation IV reactor initiative centred on deploying actinide management in fast reactor technology by 2030. The SFR operates with a fast neutron energy spectrum that burns transuranic material (TRUs), such as Pu and MAs, to a greater extent compared with thermal reactors [9]. Therefore recycling UNF in SFRs could lower the radiotoxicity and heat load in wastes.

The SFR utilises liquid sodium as a primary coolant. The system must be completely sealed and free of air and moisture due to the high chemical reactivity of sodium; providing rigorous engineering requirements, but also helping to prevent corrosion. The sodium coolant provides a high power density per volume of coolant and has a temperature of approximately 500°C when exiting the reactor [9].

The reactor can either be arranged in a pool or compact loop layout depending on reactor size. In general French, American and British designs have adopted a pool arrangement whereas Japanese designs have adopted a compact loop arrangement. It also has several passive safety features such as a long thermal response time and an intermediate sodium coolant loop between the primary coolant loop and the steam generating/ working fluid loop. The system is not significantly over-pressurised making certain accident scenarios easier manage. Large sized (600-1500MWe) reactors are recommended for U-Pu oxide fuel with centralised reprocessing. Intermediate sized (300-1500MWe) pool type reactors are recommended for either MOX or metallic fuels [9].

This reactor concept draws on 390 reactor-years of experience, making it the nearest-term deployable system for the management of actinides [10]. The CEA ASTRID (Advanced Sodium Technological Reactor for Industrial Demonstration) [11] is to be considered a reference for the SFR mentioned in Option 2. The ASTRID is a prototype reactor that will be used to qualify the safety innovations of the 4th generation of SFRs, however it has be deemed appropriate to this investigation due to its relative technical maturity.
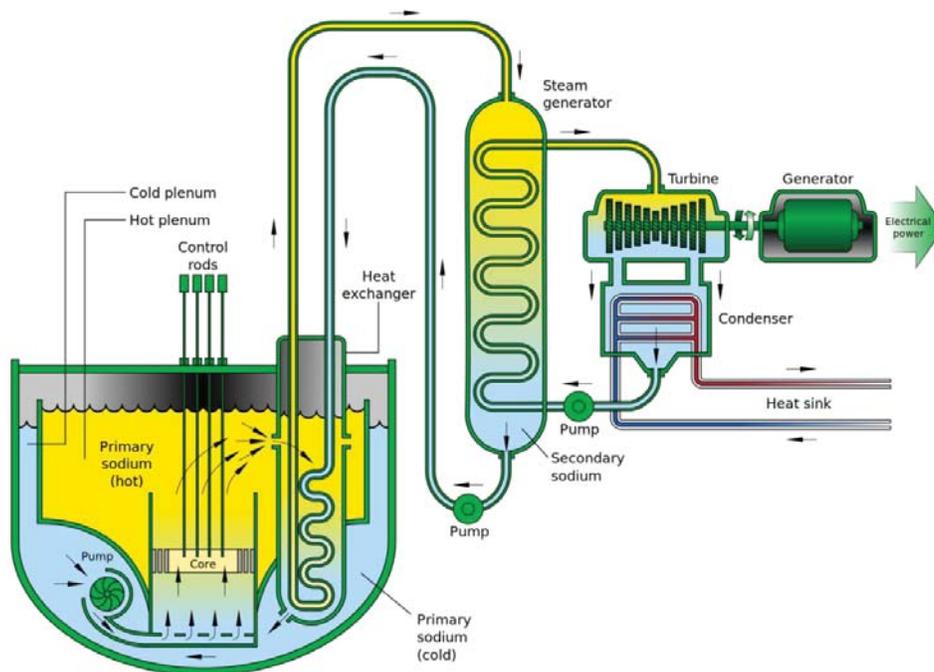
Figure 4-4 Sodium-Cooled Fast Reactor

The reactor design for Option 3 is based on the Power Reactor Innovative Small Module (PRISM) reactor is a GE Hitachi reactor concept, which has evolved from experimental reactors [12]. The reactor is a pool type metal fuelled SFR with a net 380MWe output [13]. The reactor can be grouped into reactor islands comprising of two reactors feeding a single steam generator system. The PRISM relies on passive safety systems typical of generation IV reactor concepts such as an intermediate sodium loop (to provide another barrier between the radioactive primary cooling loop and the steam loop, and minimise the potential of a water-sodium chemical reaction). The PRISM reactor concept also possesses a passive cooling system to prevent the core from overheating and melting in an emergency [14].

5.   ENVIRONMENTAL CRITERIA ANALYSIS

Each of the options would include the addition of at least one reprocessing facility and a fuel fabrication plant, either in a combined or a separate facility. The reprocessing facility would separate the majority of the fission products, from the recovered U and Pu/TRU, with the latter elements being recombined to form the new CANDU or SFR fuels. The environmental impact is assessed for each option in terms of:

- Controlled solid, liquid and gaseous effluent discharges that would be generated during reactor operations and decommissioning, including the additional ongoing wastes generated that would require repository disposal.

- Solid, liquid and gaseous effluent discharges that would be generated during reprocessing operations and decommissioning including the additional ongoing wastes generated that would require repository disposal.

- Solid, liquid and gaseous effluent discharges that would be generated during fuel fabrication operations and decommissioning including the additional ongoing wastes generated that would require repository disposal.

- Transportation.

- Potential uncontrolled releases that may lead to pollution events.

- Construction and decommissioning activities.

The potential environmental impacts associated with each of the options are discussed in the following sections and detailed in Table 5-5.

5.1   Reactor Radiological Discharges

A review of the Canadian Waste Inventory (CWI) [15] has presented some useful metrics for ongoing and decommissioning waste volumes for the nuclear fuel cycle, assumed to represent the current fleet of 22 reactors currently operating in Canada. In 2010, OPG produced 3,800 m^3 of low level waste (LLW) and 80 m^3 of intermediate level waste (ILW), from 16 reactors giving a per reactor volume of 238 m^3/y LLW and 5 m^3/y ILW. These volumes shall be used to derive the total LLW/ILW volumes from reactor operations across the options.

For the different options, ongoing wastes will be primarily based on the number of reactors and the quantity of fuel discharged through the reactor. It may therefore be pragmatic to base the total volume of ongoing LLW and ILW on an annual basis per reactor multiplied by the number of reactors that would be in operation. Ongoing LLW tends to be clean materials contaminated during operations and should therefore be produced at a consistent rate during reactor use.

A few challenges to this assumption may be:

- The quantity of fission products in fast reactors could lead to higher amounts of radioactive gases, such as ^{3}H and ^{85}Kr content.

- Decontamination of the Na coolant in a FR could generate a different volume of ILW.

- The higher percentage of stainless steel that would be used to fabricate fast reactor fuel may generate a different volume of ILW.

For the MOX CANDU UNF that would require disposal, it is possible that due to higher anticipated heat loading and higher fissile mass per assembly, there may be a smaller number of UNF bundles per disposal flask compared with CANDU-NU UNF. This is reflected in the work undertaken in the UK by Radioactive Waste Management Ltd (RWM), where the UK DGR models a single LWR MOX assembly per disposal package [16].

Decommissioning activities are generally carried out in three phases, designed to minimise operator dose. The first phase effectively defuels the reactor and its principal components, such that the reactor can then be placed into a defined period of decay storage (e.g., SAFE STORE), which can last between 30 and 60 years. The final phase is the demolition and site clearance of the remaining building structures to either de-licence the site or re-licence for alternative purposes. In terms of waste volumes produced during the three phases are:

- Phase 1 – immediately. Several hundred m^3 of LLW/ILW produced per reactor.

- Phase 2 – lasting ~65 years. Small quantities <10 m^3/y produced per reactor.

- Phase 3 – lasting ~20 years. This phase would generate the majority of the waste volume, in the range of thousands of m^3.

Decommissioning waste volumes for thermal reactors would be broadly in line with the current SAFE STORE period. Volumes of waste expected for Bruce and Darlington were provided in IAEA TE_1572 [17] and were approximately 5,000 m^3 of LLW per station (comprising a bank of four reactors). This has been increased primarily based on the CWI, which state that total decommissioning volumes from the nuclear fuel cycle were 135,000 m^3 LLW and 6,000 m^3 ILW which across a fleet of 22 reactors represents a total decommissioning volume of 6140 m^3 LLW and 273 m^3 ILW per reactor. These latter volumes shall be used to represent the decommissioning volumes for each reactor decommissioned for each option.

The differentiator here would be the type of final fuel discharged during Phase 1, the assumed period of time in SAFE STORE (i.e., overnight costs) and the delicensing criteria should contaminated land be discovered (the latter possibly as a result of uncontrolled release).

The potential for the production and management of radioactively contaminated land has not been factored into this assessment, but it is possible that increased levels of discharged radioactivity to the local waters, primarily from the reprocessing facilities, may require intervention programmes prior to final site clearance, or if environmental monitoring regimes trigger action levels [18].

5.1.1   Fabrication Wastes

Fuel fabrication facilities are required to produce the MOX fuel for Option 1, SFR MOX fuel for Option 2 and SFR metallic fuel for Option 3. Data has been published by Springfields Fuels Ltd,

in the UK, for the wastes resulting from the fabrication of uranium bearing fuels with varying levels of U enrichment. The majority of wastes arising from the Advanced Gas Cooled Reactor (AGR) and MAGNOX[11] fuel production has a very low level activity and a high U content and is currently consigned to a local landfill called Clifton Marsh which has a specific permit to manage waste with a high U content (defined as 0.2 GBq U per tonne of waste) [19]. In the past the waste has been consigned to the UK's low level waste repository.

However, it is likely that the manufacture of MOX fuel would require greater levels of management because of the Pu content, which, if high enough, can reclassify the waste as plutonium contamination material (PCM) that must be managed as an ILW waste. Wastes with Pu below the threshold are treated as LLW and there are a number of ongoing LLW streams in the UK radioactive waste inventory. It is therefore expected that the ratio of ILW to LLW would increase for facilities where mixed oxides are processed.

For MOX fabrication, [20] presents ongoing waste volume rates of 1.25 m^3/tHM LLW and 3.35 m^3/tHM ILW with decommissioning volumes of 0.6 m^3/tHM LLW and 0.12 m^3/tHM ILW.

This has been cross checked with the information in the UK RWI for Springfields fuel fabrication facility, which has 21,000 m^3 of general waste and process waste arising over a seven year period. Given that fuel production runs at a capacity of 950 tU per annum, this gives a rate of LLW produced of approximately 3.2 m^3 per tHM.

5.2   Reprocessing Wastes

Table 5-1 [19], [21] and Table 5-2 [22] summarise the radioactive waste arising from the UK's Thermal Oxide Reprocessing Facility (THORP) and La Hague reprocessing facilities in France respectively.

Table 5-1
Total Waste Produced from THORP per tHM UNF Input

	Packaged m^3 per tHM reprocessed
HLW (Thorp)	0.07
ILW (Thorp)[(a)(b)]	2.0
LLW (Thorp)[(b)]	1.3

a) Includes assembly material.

b) Operational streams only up to post operational clean out (POCO).

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[11] MAGNOX refers to 'MAGnesium-aluminium clad Oxide fuel'.

Table 5-2
Total Waste Produced from La Hague per tHM UNF Input

	Primary package m^3 per tHM reprocessed
HLW (La Hague)	0.12
ILW (La Hague)	0.43
LLW (La Hague)[a]	0.94

a)   In France, the LLW category includes short-lived ILW.

While these are useful indicators, incorporation rates can vary due to the heat properties of the fuel at the point of reprocessing. A ceiling of 2.5 kW per canister can be used to bound the number of canisters produced for each option. This will allow the assessment to develop incorporation rates for the CANDU UNF that is lower per tHM than those for the SFR UNF. While there are many factors that drive incorporation rates, this represents a reasonable approximation given the data available.

The ILW and LLW for La Hague will be used as a proxy for the Canadian ILW and LLW ongoing waste generation rates used in this assessment.

Analysis of the UK RWI shows that the decommissioning of the Prototype Fast Reactor (PFR) Reprocessing Plant is expected to generate 4,500 m^3 of LLW and 390 m^3 of ILW. These can be used on an indicative basis, per reprocessing facility used in each option to represent the decommissioning volumes, nothing that the PFR Reprocessing Plant may have reprocessed fuels with different cooling times.

5.2.1   Radioactivity in the Environment from Reactor Operations (Controlled)

For the Option 2 and 3 scenarios, liquid discharges from the reactors should be lower due to the use of the Na coolant. Aerial releases however may increase unless higher Decontamination Factors (DFs) are sought to combat the relative increase in FPs from operating a fast reactor.

5.2.2   Radioactivity in the Environment from Reprocessing Operations (Controlled)

The volumes of liquid effluent released to the sea from La Hague would not be possible to release into Lake Huron or Lake Ontario. The volumes and water residence time of these systems preclude the manner in which La Hague is operated. Therefore, the waste stream would need to be treated to a much greater degree in Ontario. It is also possible that increased releases of tritium to air or water would result in issues with the CNSC's intentions on managing tritium in drinking water. Analysis of the Bruce Power monitoring report for 2014 [18] provides a baseline for the aerial and liquid effluents discharged from the currently operating reactor site.

By design, reprocessing separates the FPs and activation products (AP) from the U, Pu and other TRU to be retained. This total FP and AP activity, if not abated, would be discharged from

site via a number of aerial or liquid effluent discharge points; aerial through stack discharges and liquid through leachate management tanks and pipelines. Abatement processes for gaseous routes typically include scrubbers and filters that are inline and part of the facility build. Abatement processes for liquid effluent routes tend to be dedicated facilities that chemically neutralise and target specific radionuclides prior to discharge of the remaining aqueous liquid from site from controlled pipelines. The recovered radioactivity is bound in solid form, encapsulated, and becomes part of the site's HLW destined for the DGR.

These processes produce a number of waste streams, both liquid and solid in nature and require differing levels of processing depending on the level of radioactivity they contain:

- HLW, typically a liquid raffinate concentrated through evaporation and subject to storage prior to its immobilisation (i.e., vitrification) and long term storage. The rate of incorporation is assumed on the basis of each 150L canister containing 2.5kW of processed waste.

- ILW is sent to encapsulation facilities for cementation and then placed in long term storage.

- LLW is sent, as appropriate, to compaction facilities for size reduction, before being sent to long term storage.

- Other contaminated solid wastes produced by the plant, for example from plant failure or maintenance activities require appropriate packaging and long term storage.

- Liquid effluents are discharged to coastal waters. The radiological and non-radiological contents of the discharged effluent are likely to require abatement steps to ensure that they fall below or within the permitted discharge limits. A number of plants undertake process steps to remove radioactive components and condition the remaining liquor to meet regulatory requirements for safe discharge.

- A number of other facilities to support the process may also be required, such as tank farms and flask storage and maintenance facilities.

Figure 5-1 and Figure 5-2 demonstrate the introduction of abatement facilities over 30 years at Sellafield in the UK, to contain and encapsulate, as solid waste, the radioactivity separated from the recovered nuclear material. Recovering radioactivity separated during reprocessing has enabled the discharged activity to fall from thousands of TBq (beta) and hundreds of TBq (alpha) to those shown in Figure 5-2. The use of wet chemistry to reprocess UNF increases the likelihood of increased liquid discharges to a lake/river. Conversely the use of dry UNF separation methods, such as pyro-processing, should facilitate abatement methods and limit the potential activity in liquid effluent discharges.

To put these reprocessing discharges into perspective against reactor discharges, the 2014 environmental monitoring report for Bruce A and B stations states a total alpha and beta activity in waterborne discharge from site as being 3 MBq and 3 GBq respectively (Figure 5-3). This is most likely tritium, not ^{137}Cs as shown for the Sellafield example below, and those levels

of ^{137}Cs would likely pose significant problems for releases to Lakes Huron and Ontario, where not only the volumes of water are much smaller, but the potential for bioaccumulation by fish is orders of magnitude greater in freshwater as opposed to seawater. For example, it is possible to trace the movements of Atlantic Salmon based on their ^{137}Cs content and interaction with the Sellafield plume. Increased ^{90}Sr would also likely cause problems in the Great Lakes as well, due to much higher bioaccumulation in fish in freshwater relative to seawater. To put these reprocessing discharges into perspective against reactor discharges, the 2014 environmental monitoring report for Bruce A and B stations state a total alpha and beta activity in waterborne discharge from site as being 3MBq and 3GBq respectively (Figure 5-3); many orders of magnitude lower than observed from wet chemical separation (Figure 5-1).
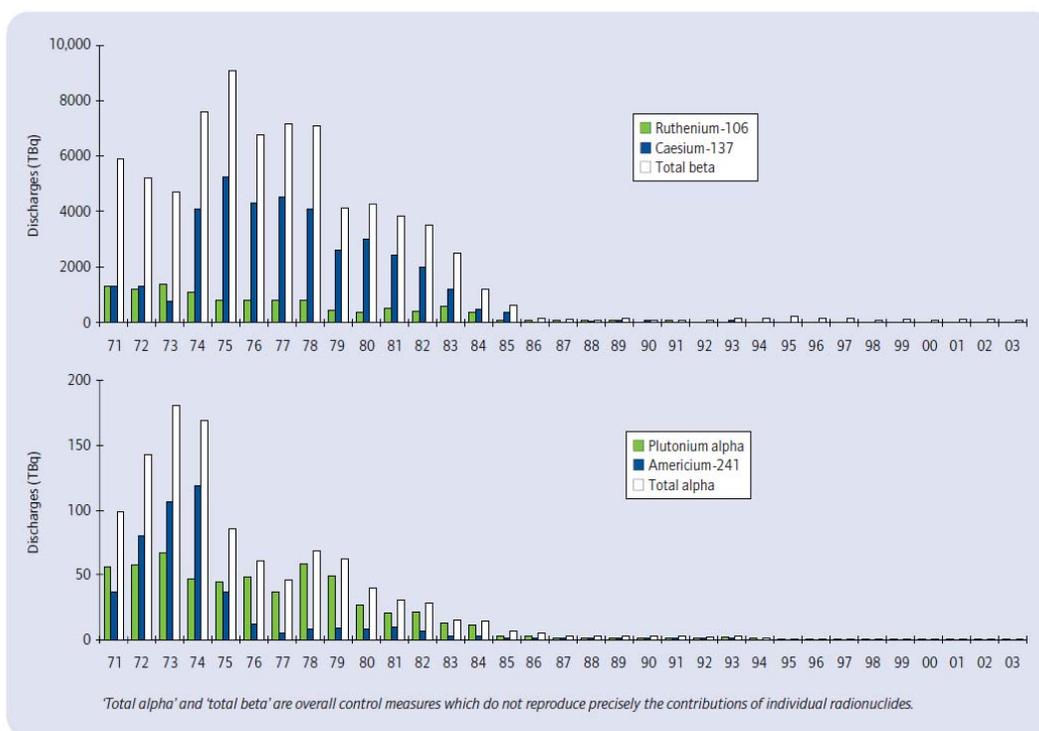
Figure 5-1 Marine Pipeline Discharges (TBq) from Sellafield between 1971 and 2003

The efficiency and success of each abatement process is typically defined by its DF. This is reliant on the feed characteristics meeting specific criteria. Changes in feed specification can lead to less than optimal DFs being achieved, leading to higher levels of activity in off-site discharges.

This part of the assessment does not have the detail to undertake a full analysis of the new UNFs that would arise and the potential DFs that would be required. Taylor [23] has shown that DFs for fission products from SFR UNF would have to be considerably higher to maintain the same levels of activity discharged.

Airborne discharges are also significant when reviewing the discharge limits for THORP (Figure 5-4) and compare with Bruce (Figure 5-5). During reprocessing, the airborne release of minor actinides and fission products occurs in contrast to no releases from typical reactor operations. There would be a significant increase in ^{85}Kr released during reprocessing, that would have to be reviewed in the context of the Canadian permit to discharge.

Radionuclide	Annual discharge (TBq)					Authorised Limit (TBq)
	1999	2000	2001	2002	2003	
Tritium	2500	2300	2600	3300	3900	25,000[b]
Carbon-14	5.8	4.6	9.5	13.0	17	20.8
Sulphur-35	0.32	0.36	0.16	0.17	0.19	
Manganese-54	0.04	0.01	0.03	0.02	0.02	
Iron-55	0.02	0.04	0.02	0.03	0.02	
Cobalt-60	0.89	1.2	1.2	0.89	0.43	13
Nickel-63	0.58	0.43	0.27	0.46	0.39	
Zinc-65	0.07	0.03	0.05	0.03	0.03	
Strontium-89	0.60	0.64	0.76	0.52	0.56	
Strontium-90	31	20	26	20	14	48
Zirconium-95	0.10	0.10	0.13	0.17	0.14	9
Niobium-95	0.08	0.09	0.14	0.25	0.16	
Technetium-99	69	44	79	85	37	90
Ruthenium-103	0.13	0.11	0.15	0.18	0.18	
Ruthenium-106	2.7	2.7	3.9	6.0	12	63
Silver-110m	0.09	0.08	0.10	0.11	0.10	
Antimony-125	7.9	7.8	13	17	23	
Iodine-129	0.48	0.47	0.63	0.73	0.55	1.6[b]
Caesium-134	0.34	0.23	0.48	0.49	0.39	6.6
Caesium-137	9.1	6.9	9.6	7.7	6.2	75
Cerium-144	0.60	0.55	0.79	0.97	0.88	8
Promethium-147	0.41	0.35	0.42	0.79	0.67	
Europium-152	0.11	0.07	0.11	0.13	0.23	
Europium-154	0.05	0.06	0.08	0.13	0.22	
Europium-155	0.04	0.05	0.07	0.10	0.19	
Neptunium-237	0.04	0.03	0.04	0.06	0.05	
Plutonium alpha	0.12[b]	0.11	0.16	0.34	0.36	0.7
Plutonium-241	2.9	3.2	4.6	10	10	27
Americium-241	0.03	0.03	0.04	0.04	0.06	0.3
Curium-242	0.003	0.003	0.006	0.02	0.01	
Curium-243+244	0.002	0.003	0.003	0.005	0.01	
Total alpha[a]	0.13	0.12	0.20	0.35	0.4	1.0
Total beta[a]	110	77	120	110	83	400
Uranium (kg)	540	610	390	440	480	2040

a. 'Total alpha' and 'total beta' are control measures relating to specified analytical determinations. They do not reproduce precisely the contributions from all individual isotopes.

b. Discharge limits applicable to a Thorp plant uranium throughput in the range 400-800 te per annum. This was achieved in every year of the period 1997-2003.

Figure 5-2 Radioactivity Discharges from the Sea Pipeline from Sellafield Site, UK

Pathway - Radionuclide	Bruce A	Bruce B
Water		
Tritium Oxide	1.94E+14	6.42E+14
C-14	1.13E+09	8.06E+09
Gross β/ γ	1.02E+09	1.99E+09
Gross β	Not applicable	Not applicable
Gross α	1.77E+06	1.49E+06
Am241	<MDL	<MDL
As76	<MDL	<MDL
Co60	7.30E+05	4.99E+08
Cr51	<MDL	<MDL
Cs136	<MDL	<MDL
Cs137	3.27E+07	<MDL
Fe59	<MDL	<MDL
Hg203	<MDL	<MDL
Mn54	<MDL	3.16E+06
Nb95	<MDL	1.30E+06
Sb124	<MDL	1.50E+07
Sr90	1.39E+06	2.94E+06
U235	<MDL	1.47E+07
Zr95	<MDL	<MDL
Minimum Detection Level = MDL		

Figure 5-3 Waterborne Discharges for Bruce A and Bruce B

Radionuclide	Thorp	
	Discharge	Limit[b]
Tritium	17	33
Carbon-14	0.11	0.65
Argon-41	-	-
Krypton-85	85,000	350,000
Sulphur-35	-	-
Cobalt-60	-	-
Strontium-90	0.014	7.8
Ruthenium-106	0.43	37
Antimony-125	-	-
Iodine-129	9.6	33
Iodine-131	-	-
Caesium-137	0.014	11
Plutonium alpha	0.0006	0.5
Plutonium-241	0.0002	13
Americium-241 +Curium-242	0.002	0.39
Total alpha	0.003	1.0
Total beta	0.02	280

Figure 5-4 2003 Airborne Discharges from THORP at Sellafield, UK

Pathway - Radionuclide	Bruce A	Bruce B
Air		
Tritium Oxide	7.51E+14	4.13E+14
Noble Gas	5.30E+13	5.25E+13
I-131	3.94E+08	4.02E+07
Particulate - Gamma Scan	3.13E+06	1.53E+07
Particulate - Gross Alpha	8.02E+05	2.26E+06
C-14	1.64E+12	1.26E+12

Figure 5-5 2014 Airborne Discharges (Bq) at Bruce, Ontario

5.2.3   Facility Construction, Operation and Decommissioning

The environmental impact associated with facility construction and decommissioning is principally associated with the increased ingress and egress of materials during the construction and demolition phases. Resources such as energy, jobs, concrete, and steel would increase with each facility, as would the hotel cost of site operations. Traffic would increase significantly during construction and demolition activities. A general comment is provided linked to the number of facilities and the number of sites such facilities would be based on.

Each facility after it is commissioned for active operations would generate LLW as ongoing wastes and upon decommissioning. Material requirements would also depend on the final set of waste products, e.g., reprocessing UNF would reduce the need for UNF disposal canisters in the DGR and more vitrified residue containers (hence more steel). The packaged material is typically associated with the level of shielding required to achieve license-status and is rated for the particular waste type.

While having additional facilities, particularly highly active liquor storage and treatment facilities, may increase the probability of spills and leaks, such events have not been considered, in terms of the generation of contaminated land or increases in on- or off-site dose against each scenario.

5.2.4   Fuel and Waste Transportation

While not directly related to the impacts associated with the operation of reactors and reprocessing facilities, it is worth considering the environmental impacts associated with the transport of nuclear and non-nuclear material up to the end of each option. The new types of UNF and the additional UNF arising from the extended programmes would not have been considered by NMWO within their programme.

Currently Canada transports approximately five shipments of UNF off site per year and it is shown [24] that this would increase significantly following the commissioning of the DGR, as the stored UNF and various wastes are transported from their site of origin.

NMWO's current programme has per annum transports of 620 single-package road shipments per year and a further 62 10-package shipments per year. The environmental impacts of the extended programmes would add vitrified residue (HLW) shipments to these programmes and many more m^3 of packaged ILW and LLW.

On-site consideration should be given to:

- The production of suitable site export facilities;

- Rail and roads built to accommodate loads of up to 130 t;

- Carbon footprint of manufacturing the flasks and transporting the cargoes; and

- Traffic management in the local vicinity.

Each scenario would also produce wastes that may require the development of a delivery strategy at the DGR, as there is be a maximum number of loads acceptable by the DGR per annum. These may influence on-site storage lifetimes and delay final site clearance activities.

5.3   Implication of Differing Marine Environments for Ontario

The Laurentian Great Lakes differ from the marine environments of the Sellafield and La Hague sites with respect to several factors that influence the retention, transport, and bioaccumulation of discharged radionuclides. Although the reactors at Bruce and Darlington are located on coastal areas where high dispersion of liquid discharges occur, the volume of water in Lakes Huron and Ontario is much smaller than that of the Atlantic Ocean or seas around the British Isles. In addition to the smaller volume of receiving water, the northern basin of Lake Huron has a water residence time of 19 years and the southern basin has a water residence time of 3 years [25]. Thus, only a relatively small fraction of releases of radionuclides to Lake Huron would be expected to be removed from the system by outflow until a new steady state with the discharges is reached. Outflow from Lake Huron is linked to Lake Ontario through Lake Erie, and thus, Lake Erie would reflect the concentrations of radionuclides leaving Lake Huron, as most water flowing through Lake Erie originates in the upstream Laurentian Great Lakes via Lake Huron.

Releases from facilities on Lake Ontario would be additive to those leaving Lake Huron, as most water in Lake Ontario also originates in upstream Laurentian Great Lakes via Lake Erie, and with a water residence time of seven years [25], would also build up through time with new releases until reaching a new steady state. The Laurentian Great Lakes then drain through the St. Lawrence River, which would contain releases to both Lakes Huron and Ontario. Lakes Huron, Erie and Ontario and the St. Lawrence River provide drinking water to many major cities including Detroit, Windsor, Cleveland, Buffalo, Toronto and Montreal. Although releases from Sellafield and La Hague are additive within the North Atlantic basin, they do not impact drinking

water. Another concern would be increased releases of tritium to Lakes Huron and Ontario, given that tritium concentrations can approach the 20 Bq L^{-1} limit proposed by the Ontario Drinking Water Advisory Council at water intakes near Bruce and Darlington.

A major difference between discharges to marine environments and freshwater environments is the greater potential for bioaccumulation of two key fission products, ^{137}Cs and ^{90}Sr. The bioaccumulation of cesium and strontium in aquatic biota are strongly, inversely dependent on the external concentration of their essential analogues, potassium and calcium, respectively. These elements are present at much lower concentrations in freshwater with respect to their concentrations in seawater, which makes freshwater ecosystems much more vulnerable than marine ecosystems to releases of ^{137}Cs and ^{90}Sr [26] [27]. Cesium also biomagnifies, increasing with each trophic level, while strontium is found mainly in fish bone [26] [27]. For example, piscivorous fish from Lake Huron have ^{137}Cs bioaccumulation factors (ratio of concentrations in fish to those in water) that average about 14,400 and those in Lake Ontario average about 4,300 [27]. Marine fish, on the other hand, would be expected to have bioaccumulation factors of about 57 [28]. Thus, the potential for bioaccumulation of ^{137}Cs in Lake Huron is more than 250-fold that of marine systems, while that of Lake Ontario is about 75-fold greater. For ^{90}Sr, bioaccumulation factors for whole fish are about 11-fold greater in Lake Huron and about 9-fold greater in Lake Ontario than in marine systems [27].

All of these factors make Lakes Huron and Ontario more sensitive and vulnerable to discharges of fission products from reactors and reprocessing facilities.

5.4   Conclusions and Options Ranking for Environmental Impacts

The use of wet chemical separation techniques would likely increase the volumes and radioactive content of controlled discharges to Great Lakes. Certain fission products are difficult to abate and the required DFs to meet discharge permits may be challenging. The Great Lakes are much more vulnerable to releases of ^{137}Cs and ^{90}Sr due to long water residence times and several orders of magnitude more bioaccumulation in fish.

The introduction of wet chemistry separation, holding tanks and extended plant lifetimes may all contribute to the possibility of uncontrolled release of radioactivity, thereby increasing the potential to create contaminated land requiring mitigation prior to delicensing.

Highly active waste liquor (raffinate) produced and stored during reprocessing represents a considerable on-site hazard, however, if allowed to decay in storage, incorporation rates can be improved, reducing the total volume of HLW produced. This is significant for some of the options.

Insufficient literature is available to appropriately differentiate between the benefits of the fast reactor systems, in terms of the fission product mix. It is therefore difficult to more precisely review aerial discharges and solid waste volumes associated with these systems.

The nuclear industry has an excellent track record of capturing and dispositioning LLW, ILW and HLW created during operations. Therefore, it is assumed that these wastes would be safely

disposed of without any adverse impacts to the environment. Without a detailed environmental study, a decision was made to rank the options based on the UNF and HLW produced, with all options considered worse than CANDU-NU reactors due to reprocessing. The mass comparison of CANDU UNF including waste and nuclear material created by each option is summarized in Table 5-3 & Table 5-4.

Table 5-3
Mass Comparison of CANDU UNF Including Waste and Nuclear Material Created by Each Option

	CANDU UNF Inventory (tHM)	Advanced Reactor UNF (tHM)	HLW Reprocessing Waste (tonnes)	RU Requiring disposal (tHM)	UNF+HLW+RU (tonnes)
Option 1	0	35,693	1,471	66,332	103,496
Option 2a	62,560	464	579	39,592	103,195
Option 2b	22,120	929	1,157	79,184	103,390
Option 2c	0	645	2,658	100,976	104,279
Option 3a	58,485	576	637	43,517	103,215
Option 3b	13,970	1,151	1,274	87,034	103,429
Option 3c	0	542	10,606	100,945	112,093

Table 5-4
Environmental Results

Impacts	Environmental
Option 1	🟩
Option 2a	🟩
Option 2b	🟩
Option 2c	🟨
Option 3a	🟩
Option 3b	🟩
Option 3c	🟥

Best Option   Worst Option

CANDU-NU

Table 5-5
Outline of Options and the Environmental Impact of their Wastes at Bruce (B) and / or Darlington (D)

	Option 1	Option 2a	Option 2b	Option 2c	Option 3a	Option 3b	Option 3c
Reprocessing Facility	**COEX (B) and (D)** — Three generations of two 700 tHM/y plants	**GANEX (B) or (D)** — Three generations of a single 600tHM/y plant	**GANEX (B) and (D)** — Three generations of two 600tHM/y plants	**GANEX (B) and (D)** — Four generations of two 600tHM/y for CANDU UNF and two generations of two 25tHM/y plants for SFR UNF	**Pyro-processing (B) or (D)** — Three generations of a single 660tHM/y plant	**Pyro-processing (B) and (D)** — Three generations of two 660tHM/y plants	**Pyro-processing (B) and (D)** — Three generations of two 675tHM/y plants for CANDU UNF and two generations of two 40 tHM/y plants for SFR UNF
Fabrication facility	**MELOX (B) and (D)** — Three generations of 180tHM/y (two at one site, one at the other)	**MELOX (B) or (D)** — Three generations of a single 7tHM/y plant	**MELOX (B) and (D)** — Three generations of two 7tHM/y plants	**MELOX (B) and (D)** — Three generations of two 7tHM/y plants for fuel from CANDU UNF and two generations of 21tHM/y plants for fuel from SFR UNF	**Intergrated (B) or (D)** — Three generations of a single 9tHM/y plant	**Intergrated (B) and (D)** — Three generations of two 9tHM/y plants	**Intergrated (B) and (D)** — Three generations of two 9tHM/y plants for fuel from CANDU UNF and two generations of two 35tHM/y plants for fuel form SFR UNF
Fuel Type	MOX	MOX	MOX	MOX	TRU metal	TRU metal	TRU metal
FR MOX Fuel Use	Once through	Once through	Once through	Once through, then multi-pass	Once through	Once through	Once through, then multi-pass
No of reactors (duration)	x12(30y), x5(20-30y)	x1(60y)	x2(60y)	x2(60y), x6(40y)	x4(60y)	x8(60y)	x8(60y),x32(45y)
Reactor Type	CANDU	ASTRID type SFR	ASTRID type SFR	ASTRID type SFR	PRISM type SFR	PRISM type SFR	PRISM type SFR
CANDU reprocessing duration (y)	73	66	65	45	67	65	47
CANDU UNF reprocessed (tHM)	102,830	40,440	80,880	103,000	44,515	89,030	103,000

	Option 1	Option 2a	Option 2b	Option 2c	Option 3a	Option 3b	Option 3c
Decay heat of CANDU UNF at reprocessing [29][12] (kW/tHM)	0.09	0.09	0.09	0.09	0.09	0.09	0.09
CANDU UNF required to be disposed (tHM)	170	62,560	22,120	0	58,485	13,970	0
MOX from CANDU UNF to be disposed (tHM)	35,523	-	-	-	-	-	-
Decay heat of SFR UNF at reprocessing [30]	-	-	-	5y cooled 3.5kW/tHM	-	-	2y cooled 15kW/tHM
SFR UNF required to be reprocessed (tHM)	-	-	-	2,256	-	-	3,829
SFR UNF required to be disposed (tHM)[13]		464	929	645	576	1,151	542
Total Fuel Reprocessed (tHM)	102,830	40,440	80,880	105,256	44,515	89,030	106,829
Total Fuel for Disposal (tHM)	35,693	63,024	23,049	645	59,061	15,121	542
Total Rep U produced (tHM)	101,464	39,903	79,805	101,632	43,923	87,847	101,632
Total Rep U used (tHM)	35,132	311	621	2,533	406	813	3,668
Total RU Requiring Disposition (tHM)	66,332	39,592	79,184	100,976	43,517	87,034	100,945
Solid wastes from recycled CANDU UNF HLW canisters [14]	3,702[15]	1,456	2912	3,708	1,603	3,205	3,708

[12] The heat associated with 50 year cooled CANDU UNF is 0.134 kW/tHM, however the majority of the decay heat from recovered radionuclides should be removed as only a small fraction of these would be incorporated into the HLW.

[13] The CANMOX UNF and the FR UNF to be disposed includes the final tHM discharged from the core during defueling operations. The cumulative core mass for each option is provided in Harris and Gregg.

[14] The incorporation rate for HLW is based on a maximum heat loading of 2.5 kW per container [30]; It should be noted that use of the stated average 2 kW/container would increase all HLW arising estimates by 20%.

	Option 1	Option 2a	Option 2b	Option 2c	Option 3a	Option 3b	Option 3c
Solid wastes from recycled FR UNF HLW[14] canisters	-	-	-	2,978	-	-	22,974
Total HLW Primary Container Volume (m³)	555	218	437	1,003	240	481	4,002
No of HLW flasks[16]	132 (431,630t)	52 (169,747t)	104 (339,495t)	239 (779,562t)	57 (186,852t)	114 (373,704t)	953 (3,111,054t)
Total HLW Flask Volume (m³)	4,560	1,793	3,586	8,213	1,970	3,940	32,821
Solid wastes from reactors ILW (m³)	3,675	300	600	1,800	1,200	2,400	9,600
Solid wastes from reactors LLW (m³)	174,563	14,250	28,500	85,500	57,000	114,000	456,000
Solid wastes from decommissioning reactors ILW (m³)	4,636	273	545	2,182	1,091	2,182	10,909
Solid wastes from decommissioning reactors LLW (m³)	104,318	6,136	12,273	49,091	24,545	49,091	245,455
Solid wastes from fuel fabrication ILW (m³) [17,18]	119,002	1,554	3,112	9,718	1,930	3,856	14,643
Solid wastes from fuel fabrication LLW (m³) [11]	45,114	589	1,180	3,684	732	1,462	5,551

15 It has been assumed that the HLW canister is consistent with that used in the UK and France, with a payload of 150 litres.

16 The flask used in the study is the CASTOR HAW 20/28, which has a TARE mass of 116 t and can accommodate up to 28 HLRW canisters [95].

17 Typically wastes generated during UO₂ fuel fabrication are LLRW. However for MOX, the Pu+Am content in the waste may increase this classification to PCM (ILRW).

18 Solid wastes from TRU metal FR fuel manufacture and decommissioning assumed to be equivalent to FR MOX manufacture for this study.

	Option 1	Option 2a	Option 2b	Option 2c	Option 3a	Option 3b	Option 3c
Solid wastes from reprocessing ILW (m^3)	44,217	17,839	34,778	45,260	19,141	38,283	45,936
Solid wastes from reprocessing LLW (m^3)	96,660	38,014	76,027	98,941	41,844	83,688	100,419
Solid wastes from decommissioning Rep & Fab facilities ILW (m^3)	5,850	2,340	4,680	7,020	2,340	4,680	6,240
Solid wastes from decommissioning Rep & Fab facilities LLW (m^3)	67,500	27,000	54,000	81,000	27,000	54,000	72,000

6.   **SAFETY AND LICENSING CRITERIA ANALYSIS**

6.1   Introduction

The CNSC has provided an estimate of the costs and timeline for licensing a fuel recycling facility and a fast reactor facility; this is included as Appendix B. The timelines provided by the CNSC are based upon 'receiving full and complete licensing submissions'. Licensing phases consist of:

- Pre-licensing of vendor design;
- License to Prepare Site (concurrent activity with an environmental assessment);
- License to Construct/Refurbish; and
- License to Operate.

As part of the construction and operation licenses a safety assessment of each facility is required. The overall safety objectives to be demonstrated by the safety assessment are [31]:

- To control the radiation exposure of people and the release of radioactive material to the environment during operational states;
- To restrict the likelihood of events that might lead to a loss of control over a nuclear reactor core, nuclear chain reaction, radioactive source, spent nuclear fuel, radioactive waste or any other source of radiation at a nuclear power plant; and
- To mitigate the consequences of such events were they to occur.

6.2   Safety and Licensing for Option 1

Option 1 uses CANDU reactors to recycle the CANDU UNF. Canada has decades of experience constructing and operating CANDU reactors, however Option 1 would require repeated refurbishments extending the operating life beyond 100 years. There are no expected barriers related to licensing the reactors technology, however some barriers associated with multiple refurbishments may arise in this option. The CO-EX reprocessing technology has not yet been used on an industrial scale. However, being based on the existing PUREX technology makes it a good candidate for licensing in Canada. The MELOX fuel fabrication technology used in this option has extensive international experience. Although Canada does not currently operate facilities of this type, the licensing work is expected to draw heavily from the international experience and therefore the barriers would be minimal.

Canada's current DGR design is based on NU fuel only. Option 1 would introduce MOX fresh fuel, reprocessing waste and MOX CANDU UNF. It is expected that all licenses and safety assessments would require resubmission to assess the impacts of the new waste types since the higher heat load and fissile mass per element would pose additional issues.

The safety assessments for Option 1 are assumed to require minimal efforts due to the extensive CANDU knowledge that exists in Canada and the CNSC, except for the DGR, which would require a new post-closure safety analysis. The largest change that must be addressed is the introduction of MOX fuel, which has been thoroughly studied at CNL.

6.3   Safety and Licensing for Option 2

All facilities are to be located on existing (licensed) reactor sites. Therefore, the site licensing is assumed to only require an updated environmental assessment for the new reprocessing and fuel fabrication facilities.

Option 2 uses SFRs to recycle the CANDU UNF. Canada has no experience in constructing and operating SFRs, however approximately 400 operating years of experience does exist internationally. Technical difficulties or safety issues associated with the SFR include:

- Material selection-crack and leakages (e.g., Phoenix, Superphoenix);
- Safety function related to control of the core reactivity and neutronic stability especially for 'burner' concepts with low U-238 contents;

- Fuel handling; and
- Sodium technology.

SFR's present some unique challenges with regard to safety and licensing. The primary concern is the interaction of the molten sodium used as coolant with the air, water, or reactor components. When exposed to air, sodium burns and when exposed to water, it explodes. This imposes additional containment constraints when compared to other reactor designs. While sodium does not present a significant risk of corrosion to steel reactor components, it does for other metal components such as the fuel cladding, which presents new material design challenges.

Canada should be able to draw upon the international experience to accelerate licensing; therefore moderate barriers to licensing are anticipated.

The GANEX reprocessing technology is not yet close to industrial scale deployment. However, being based on the existing PUREX technology makes it a reasonable candidate for licensing in Canada. The MELOX fuel fabrication technology used in this option has extensive international experience. Although Canada does not currently operate facilities of this type, the licensing work is expected to draw heavily from the international experience and therefore the barriers would be minimal.

Canada's current DGR design is based on NU fuel only. Option 2 would introduce reprocessing waste and SFRMOX UNF. It is expected that all licenses and safety assessments would have to be resubmitted to assess the impacts of the new waste types, given the much higher heat load, and fissile mass per given mass of fuel, in addition to the potential of a sodium ingress between clad and fuel pellet with failed fuel.

The safety assessments for Option 2 are assumed to require moderate efforts due to the extensive international knowledge that exists. The largest change that must be addressed is the introduction of MOX fuel with such a high Pu concentration, which has been thoroughly studied at CNL [32].

6.4   Safety and Licensing for Option 3

All facilities are to be located on existing (licensed) reactor sites. Therefore, the site licensing is assumed to only require an updated environmental assessment for the new reprocessing and fuel fabrication facilities.

Option 3 uses metallic SFRs to recycle the CANDU UNF. Canada has no experience in constructing and operating SFRs. Very little international experience exists for this reactor type; with no commercial experience. Technical difficulties and safety issues associated with the metallic SFR are at a minimum the same as those presented for Option 2. Canada would be required to create much of the commercial licensing requirements since the reactors would be first of a kind (FOAK). Therefore, each phase of the reactor licensing is expected to include major barriers that must be overcome.

The pyro-processing technology is not yet close to industrial scale deployment. A facility in Canada would be a FOAK therefore there is no operating experience available to support the licensing of this technology. Experience with the metallic fuel fabrication technology used in this option is limited only to research, with no commercial experience. Therefore the barriers are anticipated to be significant.

Canada's current DGR design is based on NU fuel only. Option 3 would introduce reprocessing waste and metallic SFR UNF. It is expected that all licenses and safety assessments would have to be redone to assess the impacts of the new waste types given the much higher heat load, fissile mass per given mass of fuel in addition to sub-optimal fuel chemistry (metallic fuel swells) and the intentional use of a sodium bond between the clad and fuel pellet to enhance thermal conductivity and therefore to allow for a larger pin pellet gap and therefore increased fuel swelling.

The safety assessments for Option 3 are assumed to require significant efforts due to the limited metallic-fuel SFR knowledge that exists as well as known reactor technical/safety

challenges that have yet to be addressed. The supporting reprocessing and fuel fabrications would also be FOAK; therefore, little information is available to accelerate the licensing and safety assessments of these facilities.

6.5   Conclusion and Options Ranking for Safety and Licensing

The fuel reprocessing technology and metallic-fuel SFR technology have not previously been licensed in Canada. Additional time and effort by both the regulator and the design authority would be required to develop the appropriate guidelines for review and approval processes for these facilities. While some credit may be taken for the licensing of these types of facilities in other parts of the world, considerable effort would still be required to ensure that appropriate analysis and information is available to support licensing activities. A lack of knowledge and expertise in Canada of both the regulator and operators of the proposed facilities would add time and cost to the licensing process.

The assessment here is necessarily high level and qualitative. A more detailed assessment is required to confirm this situation, but the general conclusions and ranking is likely to remain unchanged. A high level ranking of the options is presented in Figure 6-1 below. Option 1 is considered the easiest to license since it includes no FOAK facilities. Its main challenges are repeated refurbishments and the introduction of MOX fuel in CANDU would require further analysis before they can be used in a refurbished CANDU reactor and then stored in a DGR.

Option 2 includes a 1,500 MWe SFR which has not been deployed commercially. Therefore, Ontario would be required to complete the reactor safety and licensing without the benefit of historic data. Option 2 introduces GANEX reprocessing technology which has evolved from PUREX however has also not been deployed commercially and Option 2c requires the reprocessing of SFR MOX UNF. This would introduce additional shielding requirements in both the reprocessing and fuel fabrication facilities. Substantial rework would be required to the DGR analysis as the UNF composition and format would be different.

Option 3 would be the most challenging for Ontario. The metallic SFR, the pyro-processing, and metallic fuel fabrication facilities all have a lower technical maturity when compared to the technologies being considered in the previous options. Ontario would be pioneering much of the safety and licensing requirements for these options. The DGR would be significantly affected because the UNF would no longer be oxide fuel, which has been studied extensively at CNL. Instead, the metallic UNF, which has a higher risk of corrosion, would be placed into the DGR; creating a more complex safety case. In addition there is the potential for novel HLW, ILW and LLW waste forms from pyroprocessing.

It is important to note that all options would present increased safety and licensing risks compared to natural uranium CANDU reactors since they each require new and novel technologies to be deployed in Canada.

Figure 6-1 Safety and Licensing Results

7.   SUSTAINABILITY CRITERIA ANALYSIS

One of the main drivers to reprocess used nuclear fuel is to promote sustainable use of the finite global supply of uranium. Reprocessing UNF also reduces the volume of UNF to be sent to DGRs. Reprocessing may enhance energy security and reduce the impact of nuclear waste on the environment, relative to energy production from once-through systems over time.

By 2007 the total mass of UNF discharged from world-wide commercial power reactors was 290,000 tHM, of which 90,000 tHM was reprocessed [33]. The total mass of fuel discharged from reactors is estimated to increase to 445,000 tHM by 2020, which will exert a strain on storage facilities and emphasise the need for a final end point. Global annual reprocessing capacity is approximately 4,500 tHM UNF/year (based on oxide fuels), however actual throughput figures are much lower due to a combination of some plants not being fully operational and plants not operating at full capacity [33]. Figure 7-1 [33] illustrates the current trend in reprocessing activities. Capacity can be seen to gradually increase, however not as rapidly as the total UNF inventory.
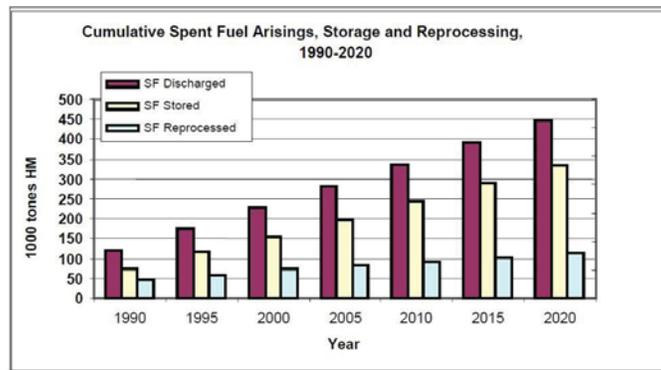
Figure 7-1 Spent Fuel Volumes and Management

At present, the total measured quantity of uranium as a known economic resource is approximately 5.9 million tonnes, enough to last about 90 years at the current rate of use in reactors of about 65 tHM U per year [34]. Reserves which are expected to exist, but have not yet been found, will supply enough uranium to last 230 years [35]. Commercial reprocessing presently reduces the amount of mined uranium needed for fuel manufacture by two – 44 tHMU/year with more capacity coming online by 2020 [34]. The inclusion of fast breeder reactor technology in conjunction with advanced reprocessing could potentially extend this supply for tens of thousands of years.

Reprocessing therefore plays a critical part in the long-term inclusion of nuclear power in the global energy mix, through efficient utilisation of uranium for thermal reactors and the provision of fuel for a generation of fast reactor technology [8].

Today, commercial reprocessing operations occur in France, Japan, the Russian Federation and the UK [33]. The reprocessing operations in the UK are near their end of life, and are scheduled for closure in the early 2020's having reached end of life after saftisfying all commercial contracts. Several countries have at some point operated plants and research facilities that have now ceased operations. Other countries are investigating the feasibility of incorporating reprocessing into their nuclear fuel cycles. Table 4-1 lists current and planned reprocessing facilities by country [33].

Options were ranked for sustainability based on the electricity they provide to the grid from recycled UNF over the study period. This assumes that if the electricity was not generated using UNF, it would have been generated using fresh uranium reactors. Therefore each MW produced from the CANDU UNF reserves reduces the Ontario demand for fresh uranium and improves the sustainability of the fuel cycle.

- Options 2a and 3a have the smallest impact, at 670 and 680 million MWh respectively, on sustainability due to their limited capacity.

- Options 2b and 3b provide a slight improvement, at 1.3 billion MWh, in sustainability with the addition of twice the amount of reactors.

- Option 1 has good sustainability, as it provides 3 billion MWhs of electricity to Ontario.

- Options 2c and 3c are the most sustainable since they recycle both the CANDU UNF and the SFR UNF producing 4 and 5.4 billion MWhs of electricity for Ontario, respectively.

This is summarized in Table 7-1 below.

Table 7-1
Sustainability Results

Impacts	Sustainability
Option 1	
Option 2a	
Option 2b	
Option 2c	
Option 3a	
Option 3b	
Option 3c	

Best Option   Worst Option

CANDU-NU

8.   NON-PROLIFERATION CRITERIA ANALYSIS

8.1   Non-Proliferation Aspects of Reprocessing

Proliferation resistance is defined as "that characteristic of a nuclear energy system that impedes the diversion or undeclared production of nuclear material or misuse of technology by States, in order to acquire nuclear weapons or other nuclear explosive devices" [36]. It is generally accepted that measures contributing to proliferation resistance are of either an "intrinsic" nature (e.g., the level of difficulty associated with converting a given nuclear material to weapons form), or an "extrinsic" nature (e.g., institutional controls such as IAEA safeguards). It is further recognized that both types of proliferation resistance are necessary; that is, no technology is "proliferation proof" – regardless of the level of intrinsic proliferation resistance that may be present, diversion of nuclear material is always possible and therefore extrinsic measures like IAEA safeguards would always be needed. The degree and complexity of safeguards, however, may vary significantly depending on a technology's intrinsic proliferation resistance features, affecting a characteristic known generally as the "safeguardability" of the technology.

The primary proliferation concern with used nuclear fuel, whether from a CANDU or any other power reactor, is the plutonium content. Although this plutonium is considered to be "reactor grade", meaning that it contains a sufficient fraction of heavier plutonium isotopes to make weapons manufacture significantly more difficult, it can still be used in a nuclear weapon. Even in a less sophisticated device that would be considered a "poor" nuclear weapon from the point of view of reliability or yield, reactor-grade plutonium would be considered attractive to some actors. Moreover, unlike uranium which must be enriched to high levels in one isotope (^{235}U) in order to be effective in a weapon, almost all isotopic mixtures of plutonium would be considered usable for this purpose (the IAEA recognizes one exception: mixtures containing a ^{238}Pu concentration greater than 80% are considered unsuitable for weapons, due to their high heat generation).

The current safeguards approach used with CANDU UNF (the Reference System) employs periodic inventory verification, using "containment and surveillance" to maintain Continuity of Knowledge between verifications, and to permit verification where direct inspection is impossible: fuel bundles are tracked within the plant and after irradiation are monitored using their radiation signal from the point they leave the reactor core to the point they enter the irradiated fuel reception bay. The irradiation fuel bay (IFB) itself is monitored visually and the fuel stacks verified by IAEA inspectors at prescribed intervals. Following a cooling period of some years in the IFB, used fuel bundles are moved to dry storage, where safeguards verification continues to be provided by IAEA seals. To a certain degree the UNF is also self protecting against theft due to its continuing emission of significant radiation. This is more of a physical security aspect than non-proliferation, however, as the latter activity is defined as involving State participation, typically with an assumption of sufficient resources to overcome the radiation barrier.

In the Reference System, the long-term plan for UNF disposition in Canada, under the Adaptive Phased Management (APM) process administered by the NWMO, is emplacement in a DGR. The APM process is currently undergoing a lengthy volunteer site selection process, and full implementation will take many decades. Safeguards within the APM process continue to be applied to the UNF, through to its emplacement within a DGR.

The safeguards approach that would be applied to any of the recycle options would depend on a number of factors, both technological (intrinsic) and institutional (extrinsic). The encompassing institutional context for any implementation within Canada will be the State-Level Safeguards concept currently being applied by the IAEA, whereby a holistic and state-specific approach is taken to assessing the level of containment, surveillance and verification required – thus increasing both the efficiency and the effectiveness of IAEA safeguards.

At the "intrinsic" or technological level, plutonium in the Reference System UNF has characteristics that make it inherently unattractive for diversion: it is created in relatively low concentration (e.g., roughly half that found in LWR UNF) and high physical dispersion, requiring

successful diversion of approximately 100 CANDU bundles to achieve a "Significant Quantity" of plutonium under the IAEA's definition (compared to about three UNF assemblies from an LWR). This feature of the Reference System increases the proliferation resistance relative to that of an LWR fuel cycle.

Features which might be considered to strengthen inherent proliferation resistance in the general case include:

1. Minimisation of the amount and quality of fissile material produced.

2. The avoidance of minimisation of the production of separated fissile material.

3. The presence of radiation fields high enough to complicate access to the nuclear materials by the State or a sub-national threat.

4. The production of nuclear materials in a form for which chemical separation of the attractive isoptopes is difficult.

5. Avoidance of the production of nuclear materials in an easily dispersed form.

Option 1 consists of refurbished CANDU units and would therefore have similar non-proliferation characteristics to that of the Reference System, in terms of the reactor technology itself. The fuel for Option 1 is MOX, which would be classified by the IAEA as Unirradiated Direct-Use material, as it would contain plutonium without concurrent self-protection from radiation from fission products (as in UNF). This is the highest proliferation risk classification for the IAEA, with accordingly the strictest requirements for verification (e.g., a nominally one-month inspection frequency). One would therefore expect the front-end fuelling operations (receiving, handling, storage, fuelling) for Option 1 to involve significantly more safeguards, including: containment, surveillance and verification in comparison to the Reference System. This may require new IAEA instrumentation, as MOX for CANDUs is not currently a fuel type monitored by the IAEA in an operational environment. At a higher level of assessment, Option 1 would ultimately contribute to a reduction in global plutonium inventory, although the extent of this would depend upon the mix of U and Pu chosen, as the U present would create more in-situ plutonium. Option 1 would also necessitate the separation of Pu from its self-protecting matrix of fission products and its packaging and transport in higher concentration than found in UNF in the Reference System – albeit not as separated or metallic product, but still in a more accessible (and thus attractive) form than in the Reference System.

Option 2 examines the use of MOX fuel in a SFR in burner mode. This represents new technology to the Canadian regulatory and operational environment and in addition the IAEA has little experience safeguarding SFR technology as (a) few have operated in anything approaching a commercial environment and (b) none have operated commercially in a non-nuclear weapons state, where it would be subject to full IAEA safeguards. There is some experience with the safeguarding of the prototype Monju facility in Japan. Safeguards would more than likely require new IAEA instrumentation for monitoring and verification. As with Option 1, this option would contribute to a reduction in global plutonium inventory, the extent of which would depend upon the mix of U and Pu. Option 2 would also necessitate the separation of Pu (along with other actinides) from its self-protecting matrix of fission products and its packaging and transport in higher concentration than found in UNF in the Reference System. This presents an increase in intrinsic proliferation risk over that of the Reference System (mitigated by an increase in appropriate safeguards and security). An additional added intrinsic proliferation risk is associated with the capacity of SFR technology for undeclared plutonium production – again, mitigated by an appropriate safeguards approach. It is possible that the end of state for Option 2c, which incorporates multi-cycle reprocessing, would have lower proliferation risk due primarily to the characteristics of the final UNF form; however, the overall proliferation risk of this Option would depend upon the aggregate pathway to this end state, requiring further assessment of each step in the option.

Option 3 examines the use of metallic fuel in an SFR-type fast neutron reactor. This represents the most advanced of the recycle options, both in the fuel type and the reactor technology and thus requires the steepest learning curve of not only the Canadian regulator and operational environment, but the IAEA as well, as there is no global experience with safeguarding this

technology (some developmental work in this regard has been carried out by KAERI in South Korea). It is certain that this technology would require new IAEA instrumentation, training and development work for monitoring and verification. As with Options 1 and 2, this option would contribute to a reduction in global plutonium inventory, the extent of which would depend upon the mix of U and Pu. Option 3 would also necessitate the separation of Pu (along with other actinides) from its self-protecting matrix of fission products and its packaging and transport in higher concentration than found in UNF in the Reference System. Additionally, the fuel form would be metallic, which from a proliferation risk viewpoint places it one step closer to usefulness for weapons manufacture and thus at a higher level of attraction. Concerns have also been expressed as to the relative ease with which process parameters may be adjusted in a pyro-processing facility in order to increase the purity of plutonium in the product. As with Option 2, this presents an increase in intrinsic proliferation risk over that of the Reference System (mitigated by an increase in appropriate safeguards and security. It is possible that the end-state of Option 3c, which incorporates multi-cycle reprocessing, would have lower proliferation risk due primarily to the characteristics of the final UNF form; however, the overall proliferation risk of this Option would depend upon the aggregate pathway to this end state, requiring further assessment of each step in the option.

In conclusion, the three options likely represent increased intrinsic proliferation risk over the Reference System, which would require additional safeguards and new IAEA monitoring and verification technology. The ranking of increased proliferation risk over the Reference System requires further assessment, but is likely to be, generally speaking, in the order of the numbering of the three options: Option 1 represents the smallest of the added risks and Option 3 the greatest. There is little international experience in the application of full safeguards to these technologies in a commercial environment. Looking more broadly, all three options have a common non-proliferation benefit related to drawing down the global inventory of plutonium; however, in Canada this plutonium exists in low concentration within radioactive UNF, with certain inherent non-proliferation benefits. The balance of overall proliferation risk is subject to a more detailed assessment.

The assessment here is necessarily high-level and qualitative; a more detailed assessment will elucidate this situation, but the general conclusion and ranking of proliferation risk would likely remain unchanged. A detailed assessment would take into account the technology features that address the intrinsic risks, as well as the extrinsic features that exist (including the broader conclusion that has been drawn by the IAEA regarding Canada's demonstrated lack of proliferation intent and the resulting Integrated Safeguards approach that this country's nuclear program operates under). Internationally accepted methodologies exist for this kind of assessment (e.g., the Proliferation Resistance and Physical Protection (PRPP) methodology developed by the Generation IV International Forum) and Canada has experts on these methodologies who have participated in their development. Additional detail for a simplified application of the PRPP methodology can be found in Appendix C.

Therefore, it is likely that all of the options would at various intermediate stages increase the degree of inherent proliferation risk relative to the current Reference System in which CANDU UNF is held in long term interim storage followed by emplacement in a DGR. All of the options introduce a vulnerability that would apply during the reprocessing operations. While there is the possibility that the final form of the fissile material in some of the options considered may be beneficial for inherent proliferation resistance, there may be an overall detriment due to vulnerabilities in the intermediate stages. A decision to proceed with reprocessing for CANDU UNF would therefore need to balance this detriment against other benefits.

8.2   Other Non-Proliferation Topics

Canada has a rich and diverse nuclear fuel cycle infrastructure and decades of experience with the development and implementation of international nuclear safeguards for its current fuel cycle. To date, this experience has not included commercial-scale UNF reprocessing, MOX-fuelled reactors, or sodium-cooled fast reactors. Internationally there is little experience with the implementation of comprehensive safeguards to commercial reprocessing and SFR reactors, as little of this activity exists in non-nuclear weapon states. Should Canada implement

one of these recycle options, this country's nuclear program, working with the federal regulator and the IAEA, would be in many ways at the forefront of development of safeguards policy and technology – in addition to the other technology and policy development that would be required. It is important that this additional development be recognized as it is often an overlooked aspect of new nuclear development, with potentially significant financial implications.

To this end, it would be crucial to the success of any of the recycle options that non-proliferation issues be taken into account at the earliest possible time in the development process. This is a practice espoused by the IAEA, known as "Safeguards by Design", with the goal of increasing both the efficiency and effectiveness of safeguards – which ultimately benefits the developer through reduction of both financial and schedule risk. This is particularly true of new or innovative technology, which often requires that new or innovative IAEA monitoring and verification technology be developed and tested before operation can begin. Internationally accepted methodologies exist for assessing technology at various levels of development, an example is detailed in Appendix C. Canada does have experience in the development and application of these methodologies, as well as the implementation of the "Safeguards by Design" concept (with positive demonstrated results).

8.3   Conclusion and Ranking for Non- Proliferation

The ranking of the seven options with regard to non-proliferation concerns is necessarily high-level and qualitative (Figure 8-1). A more detailed assessment is required for a complete analysis, but the conclusions and ranking would likely remain unchanged.

For Option 1, an assessment would be needed for the inherent proliferation resistance of the CO-EX process. If the MOX bundles are irradiated to a higher burnup than NU bundles, they would have a higher protective radiation field, which is beneficial, but this would be balanced by an overall higher Pu inventory in the UNF.

For Option 2, NNL has previously assessed the GANEX reprocess and it offers a limited improvement over CO-EX in terms of inherent proliferation resistance. SFR MOX UNF would have a higher protective radiation field and other inherent features that may result in improved proliferation resistance.

Option 3 requires the separation of Pu from its self-protecting matrix of FPs. The metallic fuel form places it one step closer to usefulness for weapons manufacture and the purity of the Pu product from pyro-processing facilities can be easily adjusted. Therefore, Option 3 presents some significant concerns with regards to proliferation resistance.

Impacts	Non-Prolifeation
Option 1	
Option 2a	
Option 2b	
Option 2c	
Option 3a	
Option 3b	
Option 3c	

Best Option   Worst Option

CANDU NU

Figure 8-1 Non-Proliferation Results

9.   **COMMUNITY AND SOCIAL CONSIDERATIONS**

The purpose of this section is to provide an overview of social and community programs that could enhance acceptance of the facilities in communities that may potentially host new or additional nuclear facilities. The focus is on social and economic benefits and related community empowerment measures that could be implemented in communities and regions hosting nuclear facilities in Ontario.

Since this report only considers siting at Bruce and Darlington, which are existing nuclear sites, some of the considerations which follow may not apply or could already be mitigated.

9.1   **Current Level of Social Acceptance**

The social acceptance of a technology plays a pivotal role in its successful introduction into society. The level of public acceptance of nuclear technology varies from one country to another and from one period in time to another, depending on social, economic, political and technological factors [37] [38]. There is no available Canadian data on the level of social acceptance associated with the options analysed in this paper. However, proponents of recycling of UNF argue that recycling to reduce the hazards of long lived waste and reducing the environmental burden on a DGR could lead to greater social acceptance. Therefore, part of the rationale for reducing waste toxicity through recycling has been for ethical and public acceptance reasons. It should be expected that some opposition to the extraction of Pu and new reprocessing waste streams is also likely.

Limited evidence from the US shows that UNF recycling may have a significant positive impact on the level of public acceptance. Several surveys [39], [40], [41] suggest that the majority of Americans supported plans to "recycle" UNF to "make more electricity and reduce the amount of nuclear waste". However, the survey did not ask respondents about their preferences for specific radioactive waste recycling technologies. Also, findings from other studies of public support for facility siting, indicate that modifying the function of a facility in a manner that addresses the initial risks, both reducing the risks and providing benefits relevant to those risks, enhances social acceptance of the facility [42]. Recycling used nuclear fuel can provide ways to more safely and efficiently manage and dispose of nuclear materials thereby reducing the relevant risks and provides high-value employment and other economic development benefits. Consequently, it is probable that recycling technologies are more likely to have a relatively high level of public acceptance than once through systems.

Confirmatory survey research should be conducted and coupled with qualitative and quantitative analysis in order to develop an understanding of the manner in which UNF recycling options and facility design variations impact public acceptance.

9.2   **Importance of the "Framing" of Social and Community Benefits**

Globally, different framing is used to refer to social and community benefits, resulting in the use of varying terminology in different programs [43]. In some countries (such as Switzerland and Germany) explicit reference is made to benefits as "compensation" or "compensatory measures", while other countries (such as Spain) strongly oppose such terminology, arguing that risk cannot be compensated for and that the risk induced by activities associated with nuclear facilities is considered comparable to other industrial activities. Therefore, the social and community benefits are referred to as "incentives for acceptance". In Japan, such programs are referred to as a "siting promotion subsidy". In Belgium, Canada and the UK, community benefits are framed as benefits and incentives aimed at enhancing local acceptance and community support for the facility. Consequently, it is important for the benefits to be framed in an appropriate manner so that they strike the right balance between mitigation of the effects of the facilities and incentives for participation.

While community benefits are potentially valuable tools in gaining local acceptance of nuclear projects, in some cases offering compensation could heighten local opposition, as some residents may view the offer as a bribe rather than a form of benefit sharing [44]. Kunreuther [45] noted that the social and community benefits mechanism must be viewed as only one of a set of policy tools dealing with facility siting. Jenkins-Smith and Kunreuther [46] conclude that

the type of safety and benefit measures and the order in that they are framed make a difference in the level of local acceptance. They suggest that benefits should be presented initially in the form of some type of non-monetary return to the community. Otherwise, and especially if cash compensation is offered after mitigation measures, the compensation is likely to be interpreted as a bribe.

Each option utilises an existing nuclear site for the reactor, reprocessing facility and fuel fabrication facility; therefore, the framework may already be established. Given the limited number of commercial reprocessing facilities, there is no clear indication of a specific factor affecting public attitudes towards the implementation of such facilities. While in Germany a project to construct a reprocessing facility was stopped in 1989 due to civil protest, the construction of the Rokkasho reprocessing plant in Japan did not face comparable social opposition [1]. For the DGR, although a site has not yet been established, the NWMO has already performed extensive public consultations, using their established framework. It is unlikely that the nuclear or DGR siting frameworks would change for any of the seven options proposed.

9.3   Types of Social and Community Benefits

A review of the literature on the use of social and community benefits in association with siting nuclear facilities and other hazardous material facilities recognises three broad types of benefits. These are: cash incentives, social benefit measures and community empowerment measures [48] [49]. It is common for communities to be offered multidimensional packages containing benefits of several different types, depending on the nature and development stage of the project. Evidence from the literature review suggests that local context matters and that the nature, dimension and scope of the various benefits are, to a large extent, determined by the social, economic and political context, as well as the needs and requirements of the host communities.

9.3.1   Cash Benefits

Cash incentives can be used to encourage a community to either become involved in a process, to allow a development to continue, or both. The exact amount of compensation can be negotiated with the host communities and surrounding areas. The cash incentives can be provided in different forms and different combinations of various forms of incentives can be used. The different forms of cash incentives include:

a. **Lump Sums -** These are payments made directly to the affected communities in order to encourage participation. In many cases there are few controls on what the money may be used for; in others conditions are attached. It is not uncommon for the payments to be made in installments, dependent upon achievement of project regulatory and informal milestones.

b. **Annual Payments -** In many cases community benefit agreements or incentive packages are structured as regular annual payments that are available to the communities. The level of payments can vary depending on certain factors, such as the volume of the waste to be stored or recycled and the status of regulatory approvals.

c. **Expert Support Cash Packages -** Support packages could be offered to assist communities to commission reviews by independent experts. This type of benefit is an important way of demonstrating transparency in the public consultations with the community during a project. In many cases these funds are paid as part of the support programs provided labelled as 'Community Empowerment'.

d. **Tax Benefits -** Special taxes can be levied on the proposed facilities and made payable to the local community as an additional incentive for involvement of the communities in the recycling program. The "nuclear tax" in France is an example of such a cash incentive.

e. **Trust Fund for Future Generations -** These are funds established with the aim of supporting the community in the long-term, in case the facility operation affects local

economic development. Such a fund is recognised as an excellent way of providing intergenerational equity in view of the long-lived nature of the potential hazard from the nuclear facilities. Funds can also be established to provide assurance to the community that there would be funds available to carry out any necessary potential remediation in the future.

9.3.2　　　　　Social Benefit Measures

Social benefit measures are measures, financial or otherwise, whose goal is to offset the impact of any stigma, perceived or actual, on the community. These measures are also aimed at providing social benefits in recognition of the communities' decision to host such facilities. Social benefits include measures such as guaranteed property values, local hiring and integrated economic development programs and facilities. Social benefits that could enhance public acceptance of nuclear facilities are as follows:

a. **Employment -** The enhanced local direct and indirect employment opportunities resulting from the development of nuclear facilities could be advanced as potential benefits designed to encourage communities to become involved. If the unemployment rate is high for the region, the addition of new jobs is vital to the communities in the region. However, it is also important to forecast the likely effect/impact on other industries in terms of local labour supply. The effects on the regional labour supply due to facilities construction and operation of new nuclear facilities could cause a regional labour shortage that could disrupt the regional communities.

    Another aspect of labour supply which must be considered is the availability of skilled labour. One indicator could be the level of unfilled vacancies in skilled occupations or occupations closely aligned to the nuclear sector. If suitably qualified workers are not available in the community, an influx of outsiders can often be seen as a major detriment. Therefore, there may be a need to train members of the local community.

b. **Local Procurement and Subcontracting -** Procurement of goods and services from local/regional suppliers are also potential social benefits associated with new nuclear facilities. It is important to note that the extent of local procurement would depend on local capabilities to supply the new nuclear facilities.

c. **Infrastructure Improvements -** It is generally recognised that development of any nuclear facilities would likely have a number of impacts upon the infrastructure a local community, especially one where no nuclear facilities have previously existed. For example, if the community hosting the facility is largely rural and does not contain an industrial infrastructure, it may be necessary to expand the existing road networks or even construct new ones. In addition, there may be a need to develop additional health and social facilities for the influx of construction and operations workers. Such impacts can generally be anticipated and mitigation measures developed to reduce their effects. Also, evidence suggests that large grants for community facilities or high tech projects which create new jobs are effective in winning local support for large scale sitting projects [50].

d. **Property Value Protection -** There is a common perception that the presence of a nuclear facility can lead to reduction in the value of house prices and reduce the overall economic profile of a region. It is therefore, not uncommon for benefit packages to include some form of property value protection, whereby funds are put aside to compensate claimants for demonstrable decreases in value. Kunreuther et al [50] found that protection of property values (a measure that responds directly to the actual impacts that may occur) was regarded by people as important in gaining public acceptance of hazardous facilities.

e. **Miscellaneous Development Projects and Facilities -** These projects and facilities refer to benefits that do not easily fit into the categories discussed earlier. Projects and facilities designed to ensure integrated economic development and benefit the community over the long term are becoming common community benefits. These projects could include the development of local support industries, specialist services

and research facilities linked to, for example, waste recycling research (a recycling centre of excellence). Such projects and facilities are likely to lead to substantial benefits to the community in terms of jobs, taxes, improvement in local services and standard of living. It is normal that such benefits only become available following local agreement to host a facility and the granting of the necessary construction permits and regulatory authorisations.

9.3.3   Measures Aimed at Empowering Communities

Community empowerment measures are designed to provide an opportunity for host communities to establish and maintain some degree of control over the siting, development and even operation of the nuclear facilities. This is especially important in cases where the community is a volunteer participant. Community empowerment measures usually include mechanisms to facilitate local involvement in decision making, local monitoring or review groups. The following community empowerment measures could be instituted.

a. **Formation of Community Partnership** – It is now becoming common for community partnerships to be established involving: local elected bodies, interest groups and local citizen groups. Such partnerships could be given the opportunity to influence the details of the program, from certain aspects of technical design to details regarding economic development projects. In many cases the local community possesses a right to withdrawal from a process, or a veto at certain defined points in the decision-making process. The extent of power the local partnership organizations have over the facilities can be negotiated with the community. Community partnerships allow a degree of ownership and control to be developed locally [51]. To ensure the institutionalisation of this process, local community partnerships should be provided with financial support to allow oversight and ensure that local views and concerns are taken into account throughout the process. Local acceptance of facilities has been shown to grow when communities are involved in the decisions making process regarding a potential facility [50]. It is also important for funding to be provided to finance public information, liaison, consultation and engagement, salaries and associated costs of partnership staff, office costs and overheads, process evaluation, etc.

b. **Capacity Building of Community Members -** This includes measures designed to allow an oversight group or other community members to become more knowledgeable about the issues involved in the operation of the facilities. This could include organisation of meetings, discussions with independent experts and visits to operating facilities. It can also assist a community to develop the capability to cope with additional demands on services that may be required.

c. **Environmental Monitoring -** Consultations would be necessary to explore the expectations of the local community with respect to environmental monitoring and then devise mechanisms through which the local community can contribute to monitoring. Empowerment measures could include funding for community environmental monitoring to measure the level of environmental pressure exerted on the local ecosystem by the nuclear facilities. Independent expertise or independent review is important in gaining the trust of the community; potentially overcoming the traditional pro and anti nuclear polarisation that exists in many communities. Local monitoring must be empowered by institutionalising local involvement over a long period of time.

d. **Public Consultation Program -** The creation and strengthening of mutual trust and relationships should be regarded as integral to the overall process. An extensive local consultation program would be important in building trust between the operators of the nuclear facilities and the local communities hosting the facilities. Consultation should be open, timely, fair and inclusive. This program should include publications on the facilities, site visits, open house days, technological exhibitions (to demonstrate how the technologies would function), science festivals and mobile exhibits to move from one location to another in the region.

The content of information required for such a program should be customised based on the phase of the decision process, distance from potential site, decision making role and cultural context. Included within this measure could be financial support to enable local people, elected representatives, national and local journalists etc. to visit similar nuclear facilities either nationally or internationally, usually as part of a proponent's 'information and education' programme. Also, the appointment of a local mediator or facilitator capable of centralising and translating information should be considered. Hybrid forums, formal and informal meetings in which scientific and societal stakeholders come to grips with complex issues should be considered.

9.4 Principles Underpinning Social and Community Benefits

Available literature shows that community benefit packages employed in other energy-related projects are underpinned by certain principles:

a. **Spreading or Staging of Payments and Benefits: -** The available literature shows that it is common for cash payments and other benefits to be made contingent on satisfactory progress in terms of various permissions and approvals [43]. These payments usually follow the various stages of community agreement and the formal regulatory milestones. The intention of this linkage is to ensure smooth development of the facilities and to assure both sides that benefits are not to be seen as separate from the process. These payments are additional to the provision of support for engagement and participation in the process.

b. **Timing of Benefits –** A community benefit scheme should provide both short and long-term payments and benefits. Payments in the form of money or social benefits should reflect the longevity of the effects of the proposed facilities. Thus, any package agreed should include trans-generational benefits to encourage long-term community support.

c. **Distinguishing Mitigatory Measures from Other Payments and Benefits -** Community and social programs should be identified separately from the mitigation of a development's direct and indirect impacts. Social and community benefits are to be seen as voluntary, supplementary to, and 'above and beyond' the mitigation efforts.

d. **Those Impacted the Most Should Benefit the Most -** The principle that the communities and individuals impacted most should benefit most, should operate across all levels of implementation; from street to neighbourhood to communities and regions. Achieving the goals of this principle requires ongoing dialogue between communities, local authorities and the facilities' developers. Pivotal to this principle is the definition of affected community. If there is a narrow definition of affected communities, then there is a risk that adjoining communities and transport communities could become alienated from the process and cause problematic delays and difficulties through objections and other actions.

9.5 Social and Community Benefits and the Impact of Local Context

The size of a social and community benefits scheme should reflect the overall scale, nature and local/ provincial significance of the proposed development and the particular local circumstances of the host communities. Some of the local factors to consider include:

- The proposed and existing community benefit fund arrangements in the area.

- The resources of communities involved.

- The population density and the size of the host community.

- The relative poverty or prosperity of the host communities as compared to other communities, in the region and to the provincial average.

- The centrality or isolation of the host communities in terms of geography and mobility.

- The recognition that some groups may require support to fully contribute to the process and discussions, while others may have established networks which can be utilised.

- The social needs of the communities.

- Existing social and community plans which can be supplemented.

- Community issues identified through other means or processes (identified through consultation with local authority and community members).

9.6   Conclusion and Options Ranking for Community and Social Considerations

Social and community benefits are offered not only to compensate for risk or impact, real or imagined, but in recognition of the community's participation in an activity that is perceived to be in the national interest. Given this trend, it is therefore equitable and consistent that social and community benefit schemes should be applied in relation to the management of the Ontario's nuclear legacy. Consequently, it is essential that the socio-economic and environmental needs of the communities that host the nuclear facilities are adequately addressed, recognising the important contribution that host communities make. The provision of community benefits and funds is an important part of ensuring that provincial/national needs for nuclear facilities are met in a way that is seen at the local level as fair and reasonable.

The transportation of UNF and HLW can be a matter of concern to local communities and the general public and represents the most exposed part of the entire fuel cycle. Considerable investment may be required to implement transport methods/systems. Public acceptance issues and protest actions related to transport may lead to time delays, increased costs and may even impact strategic choices [47].

Without survey research for Canada, ranking is not possible. All options would require similar community engagement strategies to establish support for these new technologies (at the Bruce and Darlington sites) and the modifications that would be required for the DGR.

To provide a ranking, focused studies are required to survey communities within Ontario to establish social acceptance levels for recycling systems and fast reactor technologies. Some of the issues analyzed in these studies would include:

- trade-offs between technology risk and duration required to recycle all CANDU UNF;

- issues related to transportation of UNF and other nuclear material;

- proliferation and safety concerns associated with recycling;

- affects on the DGR: time until waste is dispositioned, increased complexity of the DGR, etc.; and

- trade-off between capital costs required vs. economic benefits for Ontario.

10.   UNDISCOUNTED COST CRITERIA ANALYSIS

There are four main cost categories to be evaluated when considering the deployment of a nuclear energy system: capital expenditures, operations and maintenance (O&M) costs, decontamination and decommissioning (D&D) costs and disposal cost. This section will address the first three cost categories; disposal costs (including encapsulation and LLW treatment/storage) are addressed in Section 11. Additional costs, specifically nuclear materials transport and material handling activities were deemed too complex to include in this high level cost estimate. Single point, undiscounted indicative costs were estimated for all facilities in each of the seven reprocessing options. The facilities considered for each option were:

- reactors;
- reprocessing and associated waste plants;
- fuel fabrication plants;
- interim storage of UNF produced;
- vitrified HLW product storage; and
- Pu & U storage.

The necessary capital expenditures (CAPEX) for facilities are estimated and show the total overnight cost of each facility, including the costs of design, construction and commissioning. When no suitable source was available, the CAPEX estimates were also used to estimate D&D costs (assumed to be 30% of CAPEX) and O&M costs (assumed to be 1% and 10% for storage and plants respectively). The costs of any research and development needed to for commercialization of the facilities has not been included.

For simplicity, costs were estimated in 2015 Canadian dollars (CAN$) with no consideration for escalation, inflation or learning; similarly interest costs on the initial investment and discounting are not considered. If costs estimates were discounted, this would reduce the original investment values for all options since the value of money would be based on time and discount rate. Discounting will have a smaller effect on the recycling options where the majority of the investment made at the start (ex. Option 2a) compared to the options where the investment is more spread out (ex. Option 3c). In other words, introducing the discount rate would place greater weight on the short-term aspects of a project, favouring projects with longer duration and lower upfront costs. This could have a significant impact on the results presented in this report since investment timelines vary substantially.

10.1   Facilities Considered

According to the Cost Estimating Guidelines for Generation IV Nuclear Energy Systems, published by the Generation IV International Forum Economic Working Group [52], the uncertainty typically associated with concept screening is -50% to +100%. Given the fact that many of the technologies considered in this report are still underdevelopment, this uncertainty range will apply to all the facilities estimates below.

10.1.1   Reactors

Three reactor technologies were considered for this analysis. CANDU reactor refurbishment costs were based on the most recent estimates for the current round of refurbishments planned to be completed at Bruce and Darlington between 2020 and 2030 [53]. No adjustments to the cost estimates were made to account for the challenge of multiple refurbishments or the fact that MOX fuel would be used instead of natural uranium fuel. Therefore, this estimate is considered optimistic.

Several SFR technologies are under development and are not anticipated to be commercially available at least until 2050. There is a large degree of uncertainty in the forecasts for first of a kind (FOAK) and nth of a kind (NOAK) fast reactor concepts, which are potentially optimistic in light of current new reactor build experience. For example, the cost to build the planned European Pressurised Reactor (EPR) at the Hinkley C site in the UK (3260MWe, twin station) is forecasted in the range of ~$33B to ~$50B; although it is worth noting that there have been issues with the EPR design and licensing, putting these costs very much at a worst case extreme. For Option 2, a SFR based on an up-scaled ASTRID reactor is proposed. In 2002, the OECD-NEA reported that such a reactor could have a capital cost of $5,524M CAD [54]. It is said by some that SFR technology will be more expensive than thermal PWR technology, and while this has not been proven, the cost derived from [54] does seem very inexpensive compared to the first of a kind EPR for the UK. A nuclear expert at NNL has proposed that based on the recent experience at Hinkley C site, a 1,500 MWe SFR could cost as much as $14,415M CAD to construct. Therefore, a capital cost ranges was used, with $5,524M as the low end, and $14,415M as the high end.

For Option 3 a PRISM-type SFR is proposed. The reference used to derive the cost for a PRISM-type SFR has been authored by GE-Hitachi (GEH), the vendors of the PRISM reactor concept [55]. This GEH paper provides an estimate of $3.2B USD for a single FOAK reactor unit (i.e., half a power block of 622MWe) linked with a 50tHM/y recycling centre. This is the basis for a single 380MWe PRISM-type once the allowance for an integrated recycling centre has been removed. Therefore, a capital cost of a PRISM-type SFR has been assumed to be $3,767M in 2015 CAD$; this is considered an optimistic value, but without engineering substantiation and current international experience of operating a metal fuelled reactor commercially, are unable to suggest an alternative estimate.

Capital cost estimates for each of the options, compared to other electricity generation options are provided in Table 10-1 below.

Table 10-1: Reactor Capital Costs

	CAPEX ($M)	CAPEX ($/kWe)
CANDU Refurbishment	2,498 [53]	3,076
MOX Fuelled SFR	5,524 [54] to 14,415	3,682 to 9,610
Metal Fuelled SFR	3,767 [55]	9,913
Hinkley C site	33,000 to 50,000	9,202 to 16,871
Gen III Reactor	-	2,021 to 6,215 [56][19]
SMR(NuScale)	-	5,078 [57]

The refurbished CANDU reactor CAPEX is at the low end of the Gen III reactor cost range; this is expected since it is not a full new build. The metal fuelled SFR cost is high; this is attributed to the novel technology and small electrical capacity. The MOX-fuelled SFR CAPEX estimate has the highest degree of uncertainty. The vendor estimate quotes a cost that will be competitive with Gen III reactors. However, experts predict that the CAPEX for SFRs will be higher than LWRs due to the unique challenges of fast reactors.

The cost estimates, in millions, for a single reactor of each type are included in Table 10-2.

Table 10-2
Reactor Unit Costs

$M (CAD $_{2015}$)	CAPEX	Annual O&M Costs	D&D Costs
CANDU	2,498 [53]	$203 [53]	749
MOX Fuelled SFR	5,524 [54] to 14,415	276 [54] to 721	1,657 to 4,325
Metal Fuelled SFR	3,767 [55]	188 [55]	1,130

CANDU reactors were assumed to have an operating life of 30 years, with at least three refurbishments possible. The MOX fuelled SFR and the metal fuelled SFR were assumed to have a 60 year life based on current estimates of generation four reactors operating life [55]. For the SFR reactors, the option to refurbish the reactor was not considered. Instead, in Options 2c and 3c, where a second generation of reactors is required, new reactors were built.

These reactor cost estimates are considered high level estimates, and therefore include a high degree of uncertainty; a second and third refurbishment has never been performed on a CANDU reactor and few SFRs have been commercialized.

10.1.2   Reprocessing

Three different reprocessing technologies were considered in this study:

- COEX, developed by AREVA is a variation of PUREX that has not yet been commercialized;

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[19] The wide cost range is a result of varying country specific factors such as labour rates.

- GANEX, a variation of PUREX that has not yet been commercialized; and

- Pyro-Processing, a novel reprocessing technology that is still under development.

All reprocessing facilities were assumed to have a 25 year life based on current operating experience. If there are multiple generations of plant, it is assumed new facilities are built, rather than refurbishing existing plants. The cost estimates, scaled to the capacities required by each option, for a single reprocessing facility of each type are included in Table 10-3.

Table 10-3
Reprocessing Unit Cost

$M (CAD $_{2015}$)	CAPEX	Annual O&M Costs	D&D Costs
COEX (700 tHM/yr)	6,675 [47]	668	2,003
GANEX (25 tHM/yr)	278	28	83
GANEX (600 tHM/yr)	6,675	668	2,003
Pyro-Processing (40 tHM/yr)	228	23	68
Pyro-Processing (~660 tHM/yr)	3,764 [58]	376	1,129

a: GANEX has not yet been commercialized; therefore detailed costs are not readily available. It was assumed that costs would be slightly higher then the PUREX process, since more advanced solvents are used. Therefore, the CAPEX for the slightly smaller GANEX facility (600 tHM/yr compared to 700 tHM/yr) is assumed to be the same as the PUREX facility.

b: Pyro-processing is assumed to have the same elemental recovery percentages as GANEX.

10.1.3   Fabrication

There were two different fuel fabrication technologies considered in this study:

- MELOX, a proprietary process that has been fully-developed and commercialized by Areva to produced MOX fuels.

- Integrated pyro-processing and fabrication, which is still under development.

The fabrication cost of SFR driver fuel was estimated by the NEA to be $2,600 USD$_{2000}$/kGHM [54]. However, when compared with the £3,700 GBP$_{2015}$ /kgHM estimate for the low PU content MOX used in Option 1 [59], the SFR fabrication costs appears to be unrealistically low. Therefore, the costs for the fabrication facilities required for the fast reactor systems are based on the following formula [60], where Option 1 is the base option:

$$Cost_{Option\ X} = \left(\frac{Throughput_{Option\ X}}{Throughout_{Base\ Option}} \right)^{0.6} * Cost_{Base\ Option}$$

The 0.6 factor is used to account for the cost not scaling linearly with throughput.

It has been documented that fuels with a fissile content greater than 20% are more expensive to fabricate due to security and criticality concerns [61]. Since Pu content was not used in the cost estimation, it is likely that the costs are underestimated as the base option was a thermal MOX facility costs with a much lower Pu content then the SFR fuels.

The quantity of fuel each option requires varies significantly, therefore cost estimates for different size fabrication plants are provided. The cost estimates, in millions, for a single facility of each type are included in Table 10-4.

Table 10-4
Fabrication Unit Costs

$M (CAD $_{2015}$)	CAPEX	Annual O&M Costs	D&D Costs
MELOX for SFR MOX Fuel (7 tHM/yr)	551	55	165
MELOX for SFR MOX Fuel (21 tHM/yr)	1,064	107	319
MELOX for CANDU MOX Fuel (180 tHM/yr)	4,116 [59]	412	1,235
Integrated Pyro-Processing (9 tHM/yr)	640	64	192
Integrated Pyro-Processing (35 tHM/yr)	1,541	154	462

a: Due limited publically available data on the cost of fabrication in an integrated pyro-processing facility, the metallic SFR fuel was estimated to cost the same to fabricate as the oxide SFR fuel. This is an optimistic assumption given the increased proliferation, safety and licensing risks discussed earlier.

10.1.4   Waste Conditioning/Packaging and Storage

Each option requires four types of product storage prior to encapsulation and disposal in a DGR.

- Interim UNF storage prior to disposal in a DGR.

- Vitrified reprocessing waste product storage prior to emplacement in a DGR.

- Reprocessed uranium storage prior to emplacement in a DGR.

- Reprocessed Pu Storage prior to recycling.

The quantity required for each option varies significantly. Therefore, the costs were estimated, based on the required capacity, using a linear equation (Table 10-5) [47]. These costs are used in Section 10.2 below to estimate the cost required for each option.

Table 10-5
Waste Conditioning Unit Costs

$M (CAD $_{2015}$)	CAPEX	Annual O&M Costs	D&D Costs
Interim UNF Storage (x tHM, x<12,000)[20]	0.1872x+516.1 [47]	0.0009x+16.438 [47]	(0.1872x+516.1)*0.3
Interim UNF Storage (12,000 tHM)	2,763 [47]	27 [1]	829
Vitrified Waste Storage (x tHM)	0.1344x + 1.6537	0.0014x-0.0059	(0.1344x +1.6537)*0.3
Reprocessed U Storage (x tHM/yr)	0.0019x+0.0506	0.00002x-0.0059	(0.0019x+0.0506)*0.3
Interim Pu/TRU Storage (x tHM/yr)	9.8227x+411.83 [59]	0.2513x+6.6058 [59]	(9.8227x+411.83)*0.3

10.2   Cost Estimates for Each Option

Each of the seven options considered in the report require a different combination of the facilities presented in Section 10.1. This section will estimate the total CAPEX, O&D and D&D costs required for each option. For multiple facilities one standard cost was used and no allowances were made for efficiencies, e.g., from supply chain or construction learning from experience (LFE). The resulting estimate is conservative, however this is only a high level analysis, and an uncertainty of -50% to +100% should be applied [52].

For more information on each option refer to Appendix D.

10.2.1   Option 1 Cost Estimate

Option 1 assumes that eight CANDU reactors at Bruce and four CANDU reactors at Darlington are considered for refurbishment to accept MOX fuel. In the first generation of MOX fuelled CANDU reactors, all reactors are operated for the full 30 year life. In the second generation of MOX fuelled CANDU reactors, only five reactors are operated for between 20 and 30 years, at which time all the 103,000 tHM of CANDU UNF would be recycled.

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[20] Linear equations for estimating interim storage costs were developed from visual inspection of the figures 3.12 and 3.13 of [47] published by the OECD and then escalated to 2015 CAN $.

To support these reactors, two reprocessing facilities and three fuel fabrication facilities would be required to operate for a total of 73 years and 65 years respectively. Since these facilities have an operating life of 25 years, three generations would be required.

It was assumed that all storage facilities would be sized such that two facilities are required, except interim UNF storage. Since interim UNF storage requirements exceed 12,000 tHM, three facilities would be required.

The total CAPEX, O&M and D&D costs for Option 1 are summarized in Table 10-6 below.

Table 10-6
Option 1 Costs

Facility	Capacity	Facilities Required			Total Operating Years[21]	Total Costs ($M)		
		1st Gen	2nd Gen	3rd Gen		CAPEX	O&M	D&D
Reactor	818 MWe	12	5	0	489	42,466	99,267	12,733
Reprocessing	700 tHM/yr	2	2	2	146	40,050	97,455	12,018
Fabrication	180 tHM/yr	3	3	3	195	37,044	80,340	11,115
Interim UNF Storage	35,523 tHM[22]	3	N/A	N/A	219	8,289	5,913	2,487
Vitrified Waste Storage	278 m^3	2	N/A	N/A	142	78	54	23
Reprocessed U Storage	29,352 tHM	2	N/A	N/A	140	112	81	33
Reprocessed Pu/TRU Storage	195 tHM	2	N/A	N/A	140	4,655	7,785	1,396
					Sub-Total	132,693	290,896	39,806
					Total		463,396	
					Total MWh Generated[23]		2,980,454,902	
					Levelized Unit Cost ($/MWh)		155	

Therefore, the total cost for Option 1 is estimated at approximately $463 B or $155/MWh. A timeline for the reprocessing, fabrication and reactor facilities can be found below in Figure 10-1.

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[21] Total operating years were calculated by NNL based on the fuel requirements of the reactor.

[22] Costing assumes three 12,000 tHM facilities

[23] Based on a constant capacity factor of 85% for each reactor and 365.25 days per year.
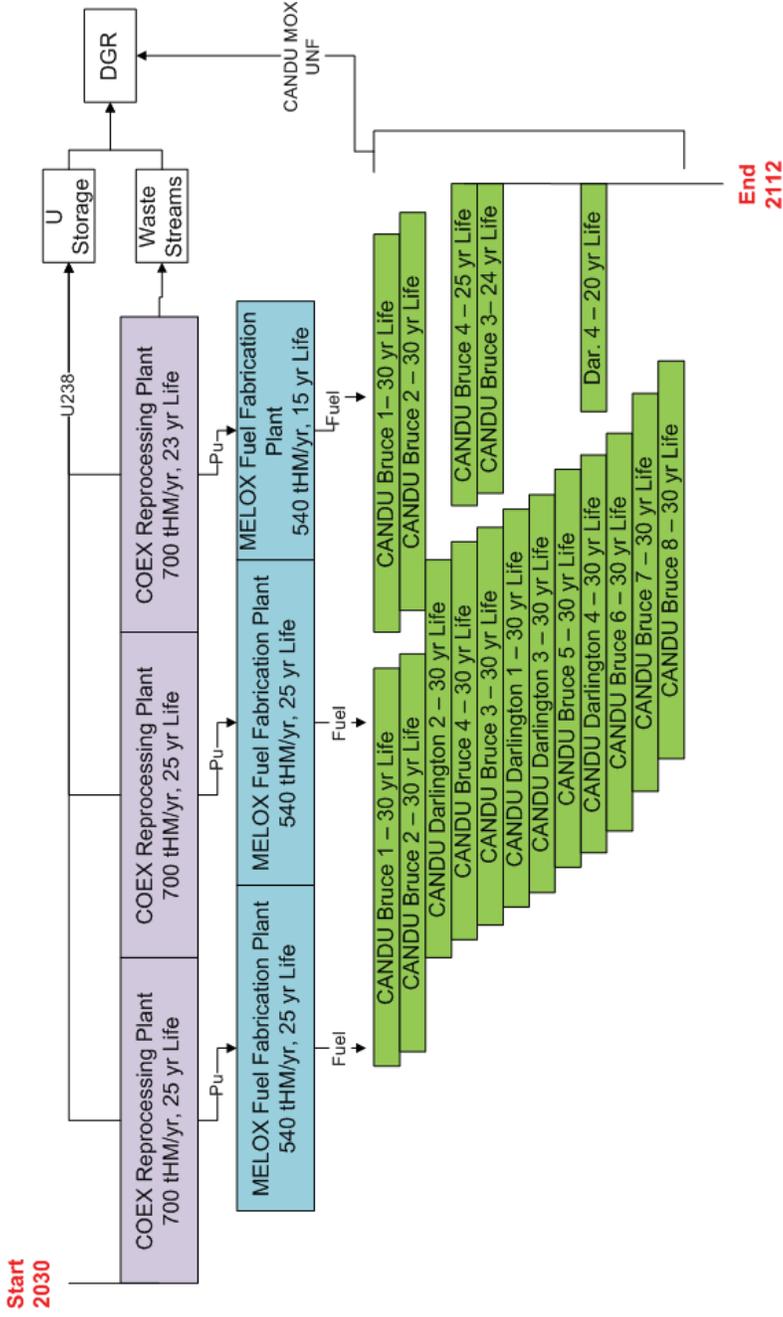
Figure 10-1 Option 1 Timeline

10.2.2   Option 2a Cost Estimate

Option 2a assumes that one MOX fuelled 1,500 MWe SFR is operated. This option would recycle only 40,440 tHM of the 103,000 tHM of CANDU UNF. It is assumed a decision could be made at the end of the SFRs life whether or not to continue using SFRs to recycle the CANDU UNF stockpile.

To support this reactor a GANEX reprocessing facility with a capacity of 600 tHM/yr, and a MELOX fuel fabrication facility with a capacity of 7 tHM/yr would be required. The reprocessing facility would be required to operate for 66 years, and the fuel fabrication facility for 65 years, therefore three generations of each facility, with a 25 year life, would be required.

It was assumed that all storage facilities would be sized such that only one facility was required.

The low and high estimates of total CAPEX, O&M and D&D costs for Option 2a are summarized in Table 10-7 and Table 10-8 below.

Table 10-7
Option 2a Low Costs

Facility	Capacity	Facilities Required			Total Operating Years	Total Costs ($M)		
		1st Gen	2nd Gen	3rd Gen		CAPEX	O&M	D&D
Reactor	1,500 MWe	1	0	0	60	5,524	16,560	1,657
Reprocessing	600 tHM/yr	1	1	1	66	20,025	44,055	6,009
Fabrication	7 tHM/yr	1	1	1	65	1,653	3,575	496
Interim UNF Storage	464 tHM	1	N/A	N/A	61	603	1,028	181
Vitrified Waste Storage	219 m^3	1	N/A	N/A	68	31	20	9
Reprocessed U Storage	35,039 tHM	1	N/A	N/A	70	67	49	20
Reprocessed Pu/TRU Storage	154 tHM	1	N/A	N/A	71	1,925	3,217	577
					Sub-Total	29,828	68,504	8,950
					Total		107,281	
					Total MWh Generated[24]		670,599,000	
					Levelized Unit Costs ($/MWh)		160	

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[24] Based on a constant capacity factor of 85% for each reactor and 365.25 days per year.

Table 10-8
Option 2a High Costs

Facility	Capacity	Facilities Required			Total Operating Years	Total Costs ($M)		
		1st Gen	2nd Gen	3rd Gen		CAPEX	O&M	D&D
Reactor	1,500 MWe	1	0	0	60	14,415	43,260	4,325
Reprocessing	600 tHM/yr	1	1	1	66	20,025	44,055	6,009
Fabrication	7 tHM/yr	1	1	1	65	1,653	3,575	496
Interim UNF Storage	464 tHM	1	N/A	N/A	61	603	1,028	181
Vitrified Waste Storage	219 m³	1	N/A	N/A	68	31	20	9
Reprocessed U Storage	35,039 tHM	1	N/A	N/A	70	67	49	20
Reprocessed Pu/TRU Storage	154 tHM	1	N/A	N/A	71	1,925	3,217	577
					Sub-Total	38,719	95,204	11,617
					Total		145,540	
					Total MWh Generated[25]		670,599,000	
					Levelized Unit Costs ($/MWh)		217	

Therefore, the low estimate of total cost for Option 2a is estimated at just over $107B or $160/MWh and the high estimate is just over $145B or $217/MWh. A timeline for the reprocessing, fabrication and reactor facilities can be found below in Figure 10-2.

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[25] Based on a constant capacity factor of 85% for each reactor and 365.25 days per year.
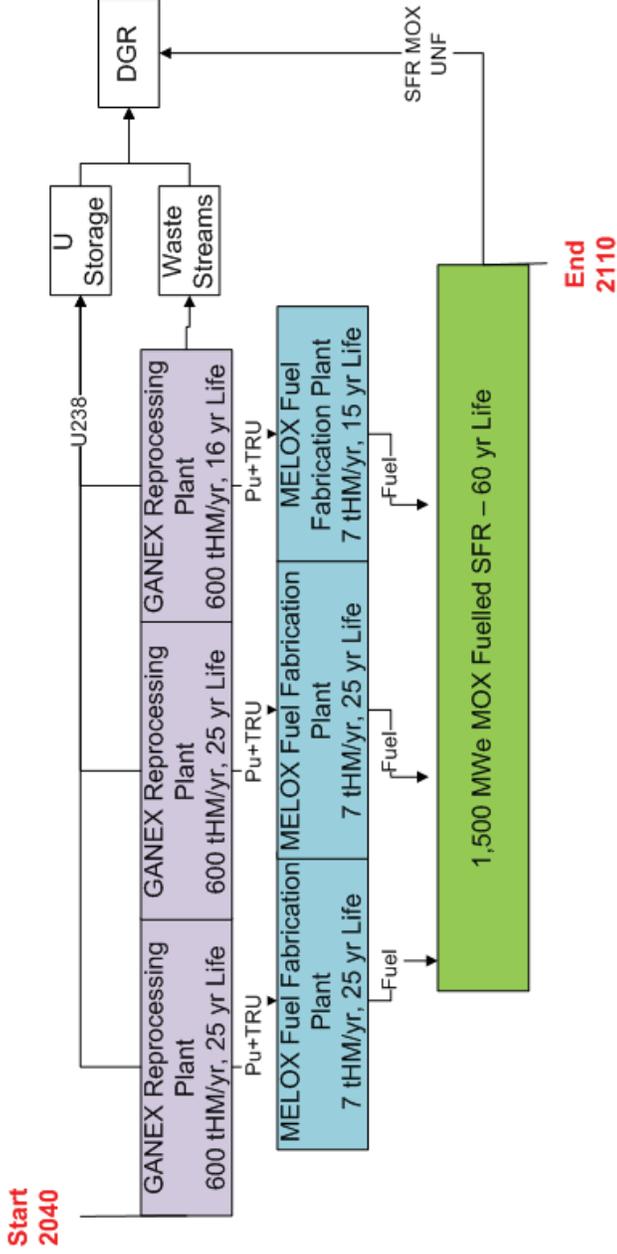
Figure 10-2 Option 2a Timeline

10.2.3   Option 2b Cost Estimate

Option 2b analyzes the scenario where two identical and independent systems would be located at each site; Bruce and Darlington. Therefore, the system in Option 2a is replicated at the other site, this would allow for more of the CANDU UNF to be recycled (80,880 tHM of CANDU UNF) before the reactors reach their end of life.

The low and high estimates for total CAPEX, O&M and D&D costs for Option 2b are summarized in Table 10-9 and Table 10-10 below.

Table 10-9
Option 2b Low Costs

Facility	Capacity	Facilities Required			Total Operating Years	Total Costs ($M)		
		1st Gen	2nd Gen	3rd Gen		CAPEX	O&M	D&D
Reactor	1,500 MWe	2	0	0	120	11,048	33,120	3,314
Reprocessing	600 tHM/yr	2	2	2	132	40,050	88,110	12,018
Fabrication	7 tHM/yr	2	2	2	130	3,306	7,150	992
Interim UNF Storage	464 tHM	2	N/A	N/A	122	1,206	2,056	362
Vitrified Waste Storage	219 m^3	2	N/A	N/A	136	62	41	19
Reprocessed U Storage	35,039 tHM	2	N/A	N/A	140	134	97	40
Reprocessed Pu/TRU Storage	154 tHM	2	N/A	N/A	142	3,839	6,416	1,152
					Sub-Total	59,4645	136,990	17,896
					Total		214,532	
					Total MWh Generated[26]		1,341,198,000	
					Levelized Unit Costs ($/MWh)		160	

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[26] Based on a constant capacity factor of 85% for each reactor and 365.25 days per year.

Table 10-10
Option 2b High Costs

Facility	Capacity	Facilities Required			Total Operating Years	Total Costs ($M)		
		1st Gen	2nd Gen	3rd Gen		CAPEX	O&M	D&D
Reactor	1,500 MWe	2	0	0	120	28,830	86,520	8,649
Reprocessing	600 tHM/yr	2	2	2	132	40,050	88,110	12,018
Fabrication	7 tHM/yr	2	2	2	130	3,306	7,150	992
Interim UNF Storage	464 tHM	2	N/A	N/A	122	1,206	2,056	362
Vitrified Waste Storage	219 m^3	2	N/A	N/A	136	62	41	19
Reprocessed U Storage	35,039 tHM	2	N/A	N/A	140	134	97	40
Reprocessed Pu/TRU Storage	154 tHM	2	N/A	N/A	142	3,839	6,416	1,152
					Sub-Total	77,427	190,390	23,231
					Total		291,049	
					Total MWh Generated[27]		1,341,198,000	
					Levelized Unit Costs ($/MWh)		217	

The low estimate of the total cost for Option 2b is estimated at just under $215B or $160/MWh and the high estimate is just over $291B or $217/MWh. A timeline for the reprocessing, fabrication and reactor facilities can be found below in Figure 10-3.

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[27] Based on a constant capacity factor of 85% for each reactor and 365.25 days per year.
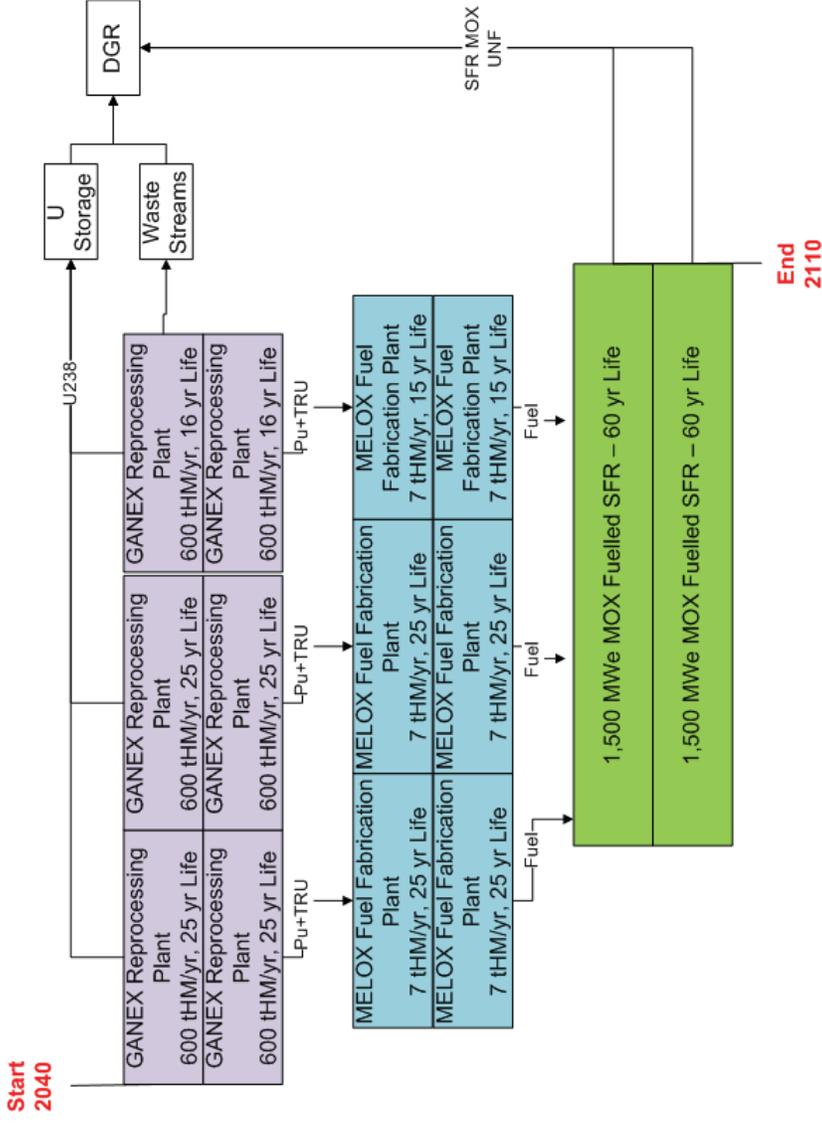
Figure 10-3 Option 2b Timeline

10.2.4   Option 2c Cost Estimate

In Option 2c, it is assumed that the first generation of reactor deployment is identical to Option 2b. However, a second generation would also be deployed to recycle the remaining CANDU UNF as well as the MOX SFR UNF to improve sustainability, increase the

amount of TRUs that are burned and also reduce the MA mass sent to the DGR. To support these reactors, two sets of reprocessing and fabrication facilities would be required.

Two GANEX reprocessing plants, each with a capacity of 600 tHM/y and two MELOX fuel fabrication facilities, each with a capacity of 7 tHM/y would be required; one set at each site. To recycle all of the CANDU UNF, the reprocessing facilities would be required to operate for 85 years, and the fabrication facility for 65 years. Therefore, four generations of reprocessing facilities and three generations of fuel fabrication facilities would be needed.

To recycle the MOX SFR UNF, a second fleet of GANEX reprocessing and MELOX fuel fabrication facilities would be required. These facilities would be co-located at the Bruce and Darlington sites. The reprocessing facilities for the MOX SFR UNF would have a much lower capacity (25 tHM/y) than the CANDU UNF reprocessing facility due to the higher concentration of Pu in the MOX SFR UNF. Two fuel fabrication facilities would be needed, each with a capacity of 21 tHM/y, to support the second generation of MOX fuelled SFRs. The reprocessing and fuel fabrication facilities would be required operate for 45 and 46 years respectively, therefore two generations would be needed.

The low and high estimates for total CAPEX, O&M and D&D costs for Option 2c are summarized in Table 10-11 and Table 10-12 below.

Table 10-11
Option 2c Low Costs

Facility	Capacity	Facilities Required				Total Operating Years	Total Costs ($M)		
		1st Gen	2nd Gen	3rd Gen	4th Gen		CAPEX	O&M	D&D
Reactor	1,500 MWe	2	6[28]	0	0	360	44,192	99,360	13,256
CANDU UNF Reprocessing	600 tHM/yr	2	2	2	2	170	53,400	113,475	16,024
SFR UNF Reprocessing	25 tHM/yr	2	2	0	0	90	1,112	2,520	332
Fabrication from CANDU UNF	7 tHM/yr	2	2	2	0	130	3,306	7,150	992
Fabrication from SFR UNF	21 tHM/yr	2	2	0	0	92	4,256	9,798	1,277
Interim UNF Storage	323 tHM	2	N/A	N/A	N/A	202	1,153	3,379	346
Vitrified Waste Storage	278 m³	2	N/A	N/A	N/A	258	78	99	23
Reprocessed U Storage	44,682 tHM	2	N/A	N/A	N/A	260	171	231	51
Reprocessed Pu/TRU Storage	533 tHM	2	N/A	N/A	N/A	260	11,285	36,510	3,385
						Sub-Total	118,953	272,522	35,687
						Total		427,161	
						Total MWh Generated[29]		4,023,594,000	
						Levelized Unit Costs ($/MWh)		106	

[28] The second generation of MOX fuelled SFRs only operates for 40 years, at which point there is insufficient Pu in the UNF to sustain further operation.

[29] Based on a constant capacity factor of 85% for each reactor and 365.25 days per year.

Table 10-12
Option 2c High Costs

| Facility | Capacity | Facilities Required | | | | Total Operating Years | Total Costs ($M) | | |
		1st Gen	2nd Gen	3rd Gen	4th Gen		CAPEX	O&M	D&D
Reactor	1,500 MWe	2	6[30]	0	0	360	115,320	259,560	34,596
CANDU UNF Reprocessing	600 tHM/yr	2	2	2	2	170	53,400	113,475	16,024
SFR UNF Reprocessing	25 tHM/yr	2	2	0	0	90	1,112	2,520	332
Fabrication from CANDU UNF	7 tHM/yr	2	2	2	0	130	3,306	7,150	992
Fabrication from SFR UNF	21 tHM/yr	2	2	0	0	92	4,256	9,798	1,277
Interim UNF Storage	323 tHM	2	N/A	N/A	N/A	202	1,153	3,379	349
Vitrified Waste Storage	278 m^3	2	N/A	N/A	N/A	258	78	99	23
Reprocessed U Storage	44,682 tHM	2	N/A	N/A	N/A	260	171	231	51
Reprocessed Pu/TRU Storage	533 tHM	2	N/A	N/A	N/A	260	11,285	36,510	3,385
						Sub-Total	190,081	432,722	57,027
						Total	679,829		
						Total MWh Generated[31]	4,023,594,000		
						Levelized Unit Costs ($/MWh)	169		

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[30] The second generation of MOX fuelled SFRs only operates for 40 years, at which point there is insufficient Pu in the UNF to sustain further operation.

[31] Based on a constant capacity factor of 85% for each reactor and 365.25 days per year.

The low estimate for the total cost for Option 2c is estimated at just over $427B or $106/MWh, and the high estimate is just over $679B or $169/MWh. A timeline for the reprocessing, fabrication and reactor facilities can be found below in Figure 10-4.
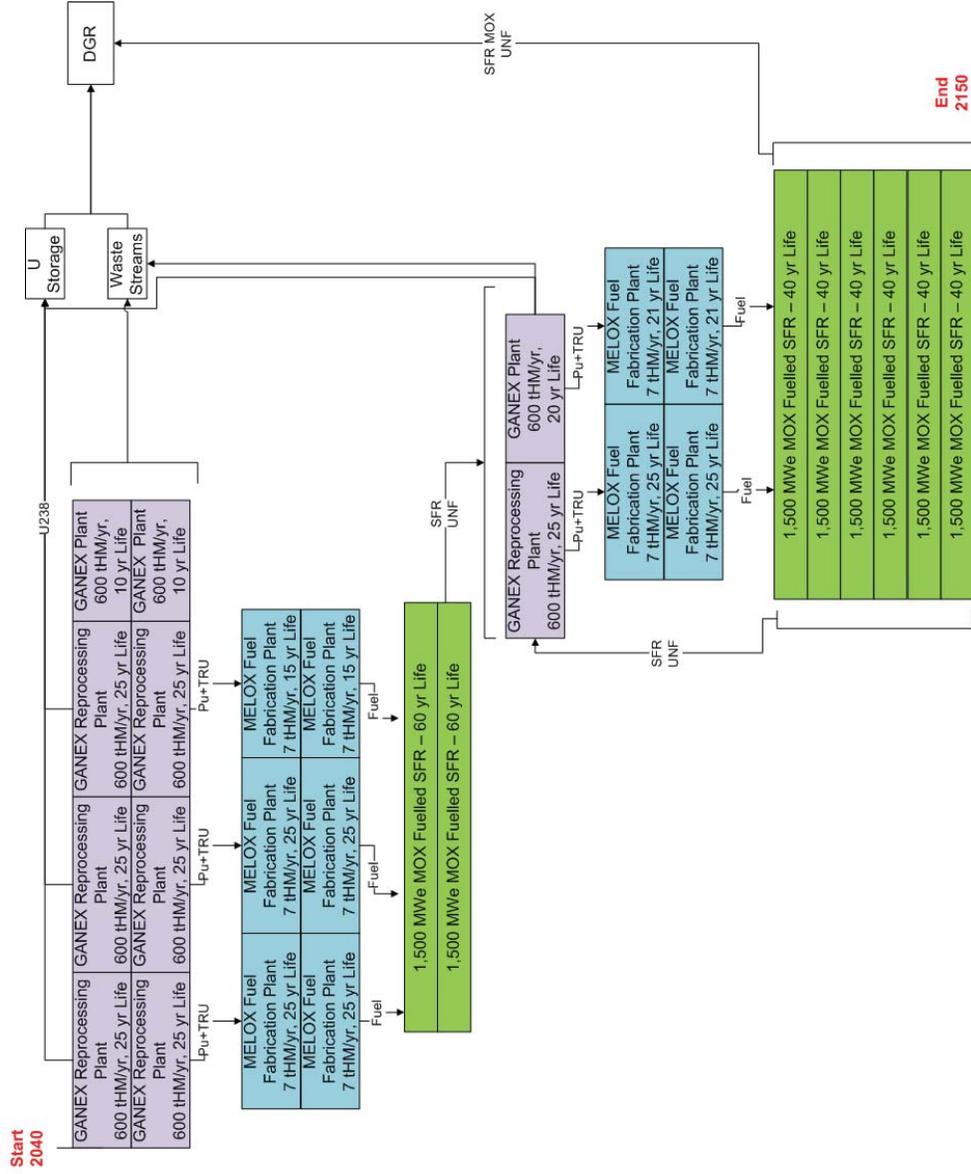
Figure 10-4 Option 2c Timeline

10.2.5 **Option 3a**

Option 3a assumes that four metallic fuelled 380 MWe SFRs are operated. This option would recycle only 44,515 tHM of the 103,000 tHM of CANDU UNF. It is assumed a decision could be made at the end of the SFRs life whether or not to continue using SFRs to recycle the CANDU UNF stockpile.

To support this reactor, a pyro-processing facility with a capacity of 660 tHM/yr and an integrated fuel fabrication facility with a capacity of 9 tHM/yr would be required. The reprocessing facility would be required to operate for 67 years, and the fuel fabrication facility for 62 years, therefore three generations of each facility, with a 25 year life, would be required.

It was assumed that all storage facilities would be sized such that only one facility was required. The total CAPEX, O&M and D&D costs for Option 3a are summarized in Table 10-13 below.

Table 10-13
Option 3a Costs

Facility	Capacity	Facilities Required			Total Operating Years	Total Costs ($M)		
		1st Gen	2nd Gen	3rd Gen		CAPEX	O&M	D&D
Reactor	380 MWe	4	0	0	240	15,068	45,180	4,520
Reprocessing	660 tHM/yr	1	1	1	67	11,292	25,226	3,387
Fabrication	9 tHM/yr	1	1	1	62	1,920	3,968	576
Interim UNF Storage	576 tHM	1	N/A	N/A	59	624	1,000	187
Vitrified Waste Storage	240 m^3	1	N/A	N/A	70	34	23	10
Reprocessed U Storage	38,513 tHM	1	N/A	N/A	71	74	54	22
Reprocessed Pu/TRU Storage	169 tHM	1	N/A	N/A	71	2,072	3,484	622
					Sub-Total	31,083	78,936	9,324
					Total		119,344	
					Total MWh Generated[32]		679,540,320	
					Levelized Unit Cost ($/MWh)		176	

Therefore, the total cost for Option 3a is estimated at just over $119B or $176/MWh. A timeline for the reprocessing, fabrication and reactor facilities can be found below in Figure 10-5.

[32] Based on a constant capacity factor of 85% for each reactor and 365.25 days per year.
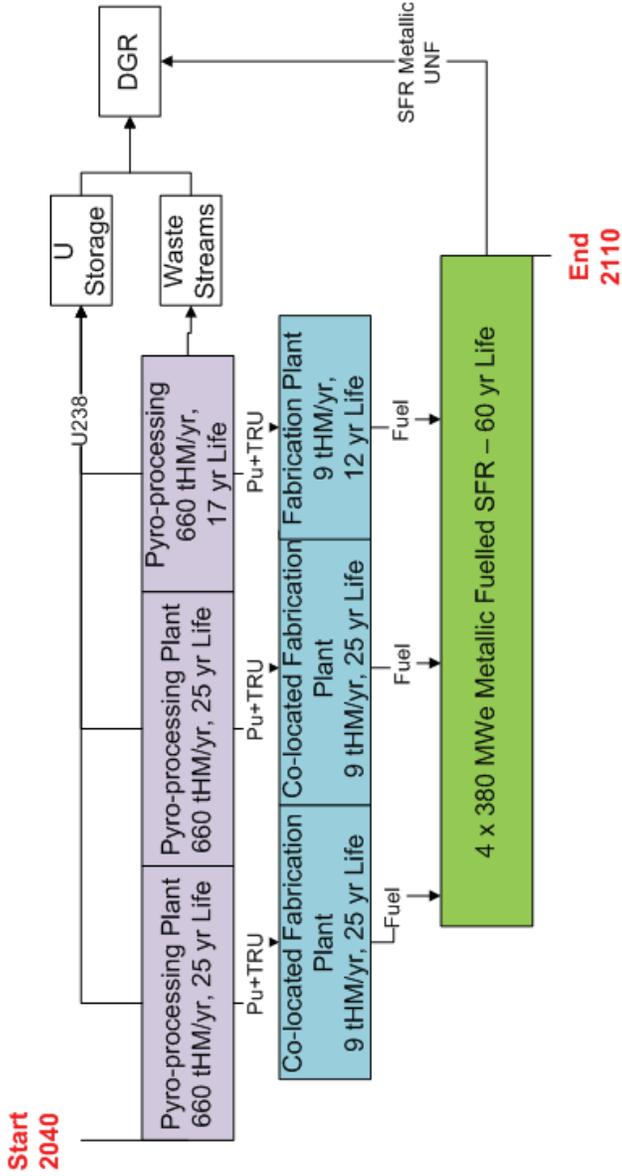
Figure 10-5 Option 3a Timeline

10.2.6   Option 3b

Option 3b analyzes the scenario where two identical and independent systems are located at each site; Bruce and Darlington. Therefore, the system in Option 3a would be replicated at the other site, this would allow for more of the CANDU UNF to be recycled (89,030 tHM of CANDU UNF) before the reactors reach their end of life.

The total CAPEX, O&M and D&D costs for Option 3b are summarized in Table 10-14 below.

Table 10-14
Option 3b Costs

Facility	Capacity	Facilities Required			Total Operating Years	Total Costs ($M)			
		1st Gen	2nd Gen	3rd Gen		CAPEX	O&M	D&D	
Reactor	380 MWe	8	0	0	480	30,136	90,360	9,041	
Reprocessing	660 tHM/yr	2	2	2	130	22,584	48,945	6,774	
Fabrication	9 tHM/yr	2	2	2	126	3,840	8,064	1,152	
Interim UNF Storage	576 tHM	2	N/A	N/A	122	1,248	2,069	374	
Vitrified Waste Storage	241 m^3	2	N/A	N/A	140	68	46	20	
Reprocessed U Storage	38,513 tHM	2	N/A	N/A	140	147	107	44	
Reprocessed Pu/TRU Storage	169 tHM	2	N/A	N/A	144	4,144	6,871	1,243	
					Sub-Total	62,167	156,462	18,649	
					Total		237,277		
					Total MWh Generated[33]		1,359,080,640		
					Levelized Unit Cost ($/MWh)		175		

The total cost for Option 3b is estimated at just over $237 B or $175/MWh. A timeline for the reprocessing, fabrication and reactor facilities can be found below in Figure 10-6.

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[33] Based on a constant capacity factor of 85% for each reactor and 365.25 days per year.
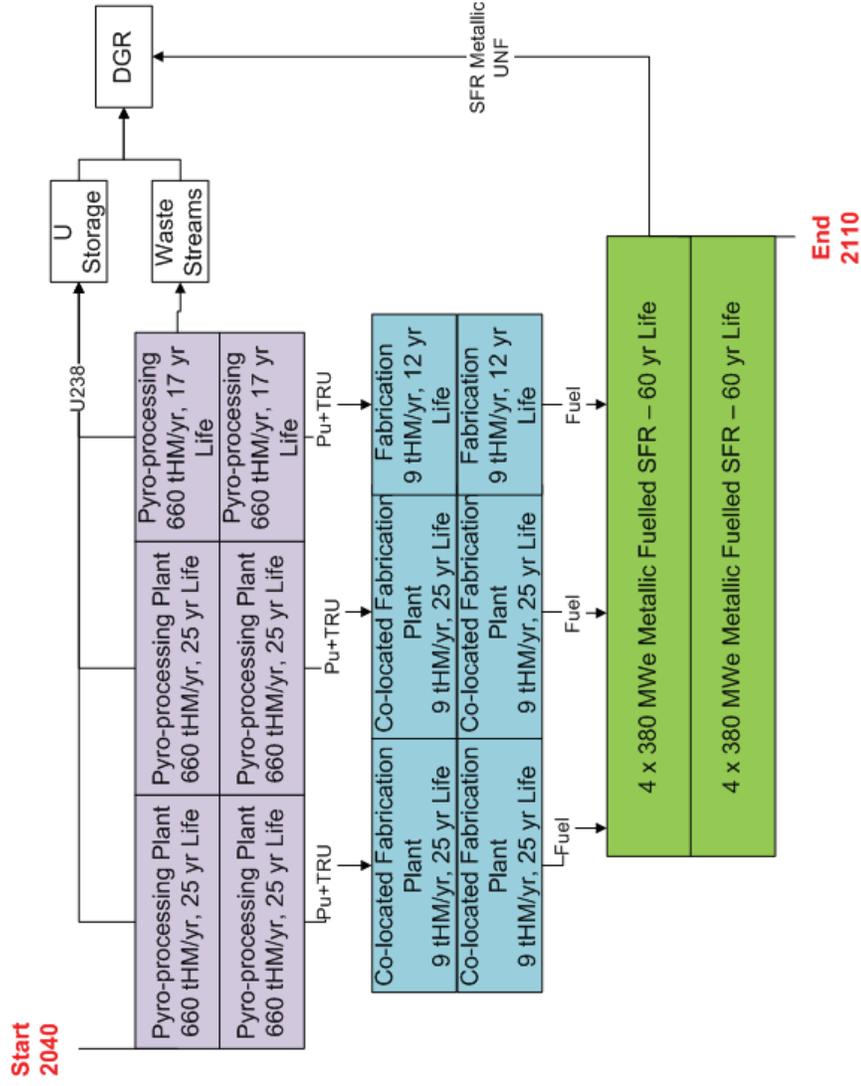
Figure 10-6 Option 3b Timeline

10.2.7   Option 3c

Option 3c assumes that the first generation of reactor deployment would be identical to Option 3b. However, a second generation would also be deployed to recycle the remaining CANDU UNF as well as the metallic SFR UNF to improve sustainability, increase the

amount of TRUs that are burned and also reduce the MA mass sent to the DGR. To support these reactors, two sets of reprocessing and fabrication facilities would be required.

Two pyro-processing plants, each with a capacity of 675 tHM/y and two integrated fuel fabrication facilities, each with a capacity of 9 tHM/y would be required; one set at each site. To recycle all of the CANDU UNF, the reprocessing facilities would be required to operate for 76 years, and the fabrication facility for 63 years. Therefore, four generations of reprocessing facilities and three generations of fuel fabrication facilities would be needed.

To recycle the metallic SFR UNF, a second fleet of pyro-processing and integrated fuel fabrication facilities would also be needed. These facilities would be located at the Bruce and the Darlington sites. The reprocessing facilities for the metallic SFR UNF would have a much lower capacity (40 tHM/y) than the CANDU UNF reprocessing facility due to the higher concentration of Pu in the metallic SFR UNF. One fuel fabrication facility would be needed at each site, with a capacity of 35 tHM/y, to support the second generation of metallic fuelled SFRs. The reprocessing and fuel fabrication facilities would be required to operate for 47 years (at each site), therefore two generations would be required.

The total CAPEX, O&M and D&D costs for Option 3c are summarized in Table 10-15 below.

Table 10-15
Option 3c Costs

Facility	Capacity	Facilities Required			Total Operating Years	Total Costs ($M)		
		1st Gen	2nd Gen	3rd Gen		CAPEX	O&M	D&D
Reactor	380 MWe	8	32[34]	0	1,920	150,680	361,440	45,204
CANDU UNF Reprocessing	675 tHM/yr	2	2	2	152	22,584	57,228	6,774
SFR UNF Reprocessing	40 tHM/yr	2	2	0	94	912	2,162	272
Fabrication from CANDU UNF	9 tHM/yr	2	2	2	126	3,840	8,064	1,152
Fabrication from SFR UNF	35 tHM/yr	2	2	0	94	6,164	14,476	1,848
Interim UNF Storage	271 tHM	2	N/A	N/A	210	1,134	3,503	340
Vitrified Waste Storage	278 m³	2	N/A	N/A	246	78	94	23
Reprocessed U Storage	44,668 tHM	2	N/A	N/A	246	171	218	51
Reprocessed Pu/TRU Storage	769 tHM	2	N/A	N/A	246	15,931	49,164	4,779
					Sub-Total	**201,493**	**496,350**	**60,444**
					Total	**758,288**		
					Total MWh Generated[35]	**5,436,322,560**		
					Levelized Unit Cost ($/MWh)	**139**		

The total cost for Option 3c is estimated at just over $758B or $139/MWh. A timeline for the reprocessing, fabrication and reactor facilities can be found below in Figure 10-7.

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[34] The second generation of metallic fuelled SFRs only operates for 45 years, at which point there is insufficient Pu in the UNF to sustain further operation.

[35] Based on a constant capacity factor of 85% for each reactor and 365.25 days per year.
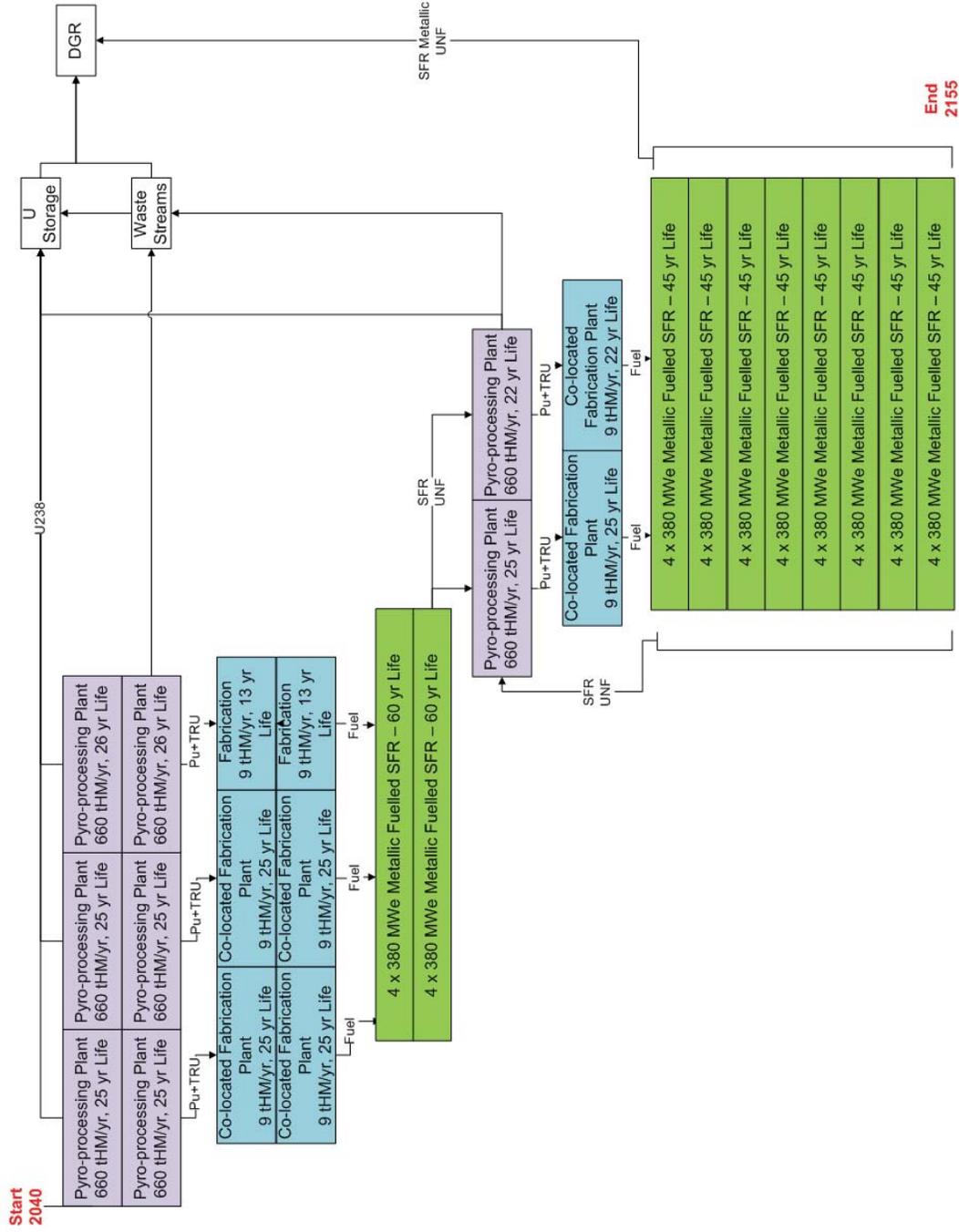
Figure 10-7 Option 3cTimeline

10.3 Assumptions and Limitations

All costing estimates have the common assumptions and limitations:

1. All technologies will be fully developed and ready for deployment when the timeline for the option starts in 2040, thus no additional research and development (R&D) costs are applicable.

2. Due to the difficulty in predicting costs for future technology accurately, a large degree of uncertainty exists. In Option 2, a range was provided for the reactor capital cost to demonstrate the potential impact of this uncertainty.

3. Where reference costs were not in Canadian dollars the following exchange rates have been used at the request of the MoE:
   a. 1GBP : 2.0592CAD
   b. 1USD : 1.2557 CAD

   These exchange rates are based on the 25-year monthly average of noon spot rates between Jan 1990 and Dec 2015 [62].

4. Plant costs have been based on 2015 CAD dollars, adjusted using the Chemical Engineering Plant Cost Index (CEPCI) where appropriate [63].

5. Any plants of which there are more than one have a standard cost and no allowances are made for efficiencies, e.g. from supply chain or construction learning from experience (LFE).

6. The single plant costs are from reference data sources, escalated using the plant cost index then converted to CAD.

7. If there are multiple generations of plant, costing will assume brand new facilities are built, rather than refurbishing existing plants (CANDU refurbishment excluded). This is based on UK experience from THORP, due to be decommissioned after its design life of 25 years of operation. There is currently no existing MOX plant in the world that has been refurbished to allow for significant design life extension.

8. Only wage inflation is taken into consideration when scaling O&M costs to 2015 levels [64] [65].

9. Decommissioning costs of plant are 30% of the total CAPEX.

10. Where O&M costs are not referenced, they have been assumed at a rate of 1% CAPEX per annum for storage plants and 10% CAPEX per annum for manufacturing plants.

10.4 Conclusion

The power that could be generated by each option, the associated total costs and the levelized cost ($/MWh) are shown in Table 10-16. Total costs were levelized over potential electricity generated in order to make a comparison between options and against the current CANDU-NU reactor. It is important to note that the duration and investment schedule of each option is

different, for a more thorough cost/benefit analysis the time value of money should be considered through discounting[36].

Table 10-16
Power Generated and Levelized Cost

	MWh	Cost ($M)	$/MWh
CANDU-NU			70 [66]
Option 1	2,980,454,902	463,396	155
Option 2a - Low	670,599,000	107,281	160
Option 2a - High	670,599,000	145,540	217
Option 2b – Low	1,341,198,000	214,532	160
Option 2b – High	1,341,198,000	291,049	217
Option 2c – Low	4,023,594,000	427,161	106
Option 2c - High	4,023,594,000	679,829	169
Option 3a	679,540,320	119,344	176
Option 3b	1,359,080,640	237,277	175
Option 3c	5,436,322,560	758,288	139

Each option has its own advantages and disadvantages with regard to cost.

- Option 1 would refurbish existing reactors; therefore the estimated cost of each reactor is less. Since, the fuel would have a low Pu content the per unit fabrication cost would be less. However, more fuel assemblies would be required per unit of energy produced making the total estimated fuel fabrication more expensive compared to the Options 2 and 3.

- The reactor in Option 2 would have a capacity of 1,500 MWe, therefore due to economies of scale even the high cost estimate has a lower capital cost per unit energy installed ($/kWe) than the CAPEX estimate for Option 3. Option 2 would also have the highest Pu content of the three reactor fuels considered. This would allow for a longer fuel residence time, reducing the number of fuel assemblies that would need to be fabricated.

- Option 3 uses smaller, 380 MWe, reactors. The lack of economies of scale produces a estimated capital cost per unit energy installed ($/kWe) that is 2.5 to 3.25 times greater than the other options estimates. However, the integrated pyro-processing and fuel fabrication facility design is intended to reduce costly and time-consuming packaging and transportation between facilities [61]. Therefore, the reprocessing costs for this option could be much reduced.

When each option is broken down further additional trends emerge.

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[36] Note: Discounting may have a significant affect on the results, since weight would be placed on short-term aspects of the project, favouring projects with longer durations and lower upfront costs.

- Options 2c and 3c would produce substantially more power than the other options by recycling the SFR UNF, which can generate a large amount of electricity per kg of UNF due to its high Pu content. In Option 3c, since the metallic fuel would not have to cool as long before it can be reprocessed, the Pu could be used more efficiently. This would increase the potential electricity produced in Option 3c to 35% more than Option 2c; further reducing its estimated cost per electricity produced ($/MWh).

- Option 1 can produce the most power out of the CANDU UNF without requiring multiple recycles. This is a result of the CANDU design, which uses heavy water as a moderator to improve neutron economy. If only one recycle was considered, Option 1 would have the lowest cost per unit energy.

- In Options 2a, 2b, 3a and 3b, the UNF from the SFRs would not be recycled, Options 2a & 2b are estimated to be less costly per unit of electricity produced than Options 3a and 3b due to economies of scale. The exception to this is when the capital cost of the reactor in Option 2a and 2b is high. In this case, the expected savings in Options 3a and 3b associated with the lower Pu content and integrated facilities, are sufficient to counteract the high capital cost per unit energy installed ($/kWe) such that Options 3a and 3b would become less costly than Options 2a and 2b.

Estimated costs for Option 1 are based on contracted prices for refurbishments as well as reprocessing and fuel fabrication operating experience. Although these costs may be optimistic based on the number of CANDU refurbishments and dated costs for supporting facilities, this estimate can be considered credible. A range was used for the reactor CAPEX estimate in Option 2 to capture the uncertainty associated with FOAK cost estimates. The reactor cost estimate for Option 3 is based on a vendor estimate, which are inherently optimistic, but without engineering substantiation or current international experience there is insufficient information to suggest an alternative estimate. Reprocessing and fuel fabrication facilities considered in these options have not been commercialized and publically available data has only studied fuels with lower Pu concentrations. As a result costs sourced for Option 2 and 3 are significantly less credible. A more detailed cost estimate should be conducted and could lead to changes in the rankings.

These results are displayed in Figure 10-8 below; results from the Option 2 low estimates are displayed on the left and high estimates on the right.

Figure 10-8 Undiscounted Cost Criteria Results

11.   DEEP GEOLOGICAL REPOSITORY IMPACTS AND COSTS

All of the seven options would introduce the following waste streams which must be accounted for in the nuclear waste management plan:

- UNF from recycled fuel that is not reprocessed by the end of the scenario,
- HLW stream from reprocessing[37] (RepHLW),
- RU,
- ILW from reprocessing and recycled fuel fabrication facilities (including decommissioning), and
- ILW from reactor operations and decommissioning.

Waste owners would be required to find a long-term solution for all waste types not covered under the Nuclear Fuel Waste Act. Since there are many regulatory and technical uncertainties with regards to this waste, it is possible that some of it may not be placed in a DGR. The safest, but probably least cost-efficient, option is to place all of this waste into a DGR. Whether all of this waste is placed in one or more DGRs depends on the organizations responsible for this waste. A challenge for the future must address who would be responsible for each of these waste streams and how they would be disposed.

11.1   DGR Post Closure Safety

The Post-closure Safety Assessment of a Used CANDU Fuel Repository [67][68] identifies the main contributors to the estimated dose rates to humans from used CANDU fuel as being the very long-lived fission and activation products such as ^{129}I, ^{135}Cs, ^{79}Se, ^{41}Ca, ^{14}C, and ^{36}Cl, as is shown in Figure 11-1. These same long-lived nuclides are found in both the UNF and RepHLW for all of the recycling options, which must be sufficiently isolated for over 10,000 years to comply with the current dose acceptance criterion used for assessment of DGRs in Canada and internationally. It must be noted that although these are the dominant contributors to the total estimated dose rate from a DGR, the level is still orders of magnitude below the natural background level and the current acceptance criterion set by the CNSC. Since they are produced from the fission and activation of material in a reactor, the total inventory of these long-lived nuclides, and therefore the total estimated dose rate from a DGR, is proportional to the total energy produced by the reactors.

The dose to humans from a DGR depends on both the ingestion radiotoxicity and solubility of the emplaced material. Although the TRUs are highly radiotoxic for thousands of years, their low solubility means that they would take significantly longer than their radiotoxic lifetime to escape the DGR. Therefore the radiotoxicity of TRUs in a DGR does not significantly affect its design with respect to containment of the emplaced material.

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[37] A mix of uranium, fission products and TRUs, the proportion of which depends on the reprocessing technology employed.

Any long term management of UNF and HLW from reprocessing would have to comply with the current dose acceptance criteria used for assessment of DGRs in Canada and internationally. The current interim dose acceptance criteria used for the safety assessment of a DGR in Canada (3×10^{-4} Sv/a) is consistent with the ICRP and IAEA recommended values (ICRP 2007, IAEA 2006). For reference the average natural background dose rate in Canada is 1.8×10^{-3} Sv/a.
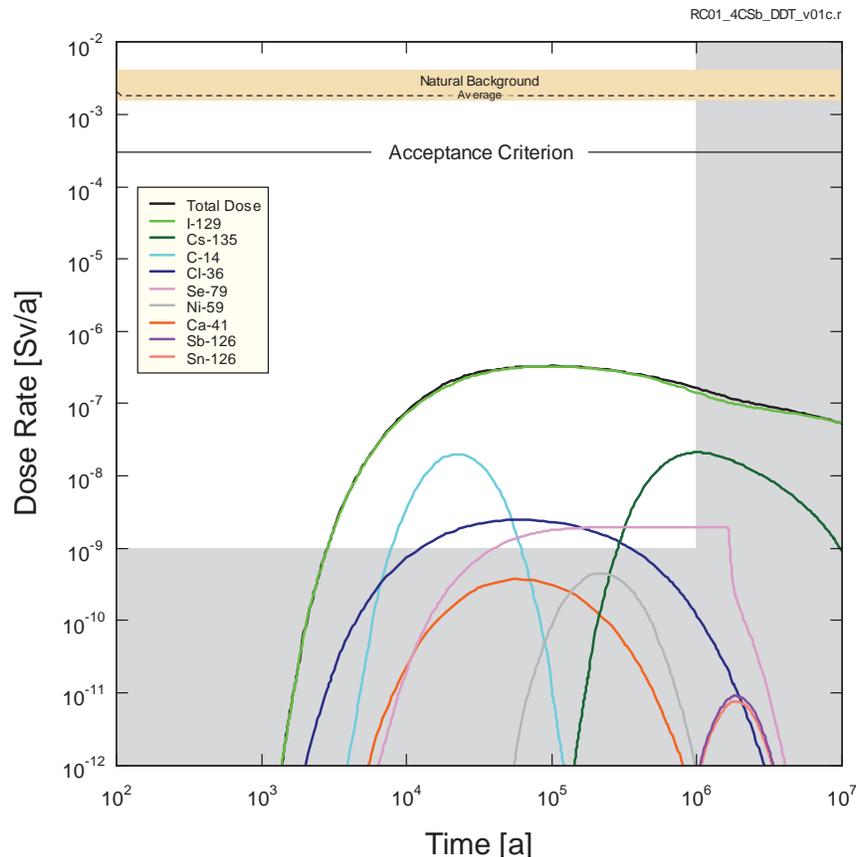
Figure 11-1 The Expected Dose Rate from a DGR [67]

With regard to the impacts of the recycling options on waste disposal, the European consensus for a 2007 study on the Impact of Partitioning, Transmutation and Waste Reduction Technologies on the Final Nuclear Waste Disposal [69] also concluded:

- For all scenarios and fuel cycles under consideration, the dominant radiological impacts are due to the soluble long-lived fission or activation products, such as [129]I or ^{14}C in the HLW and associated ILW for the different fuel cycles.

- Removal of minor actinides (MAs) has nearly no effect on the long term radiological impact under normal evolution of the repository, because MAs (Am, Cm, Np) are almost insoluble in underground waters and they migrate extremely slowly in the reducing conditions prevalent in European geological repositories.

- Particular attention should be paid to long-lived ILW, separated Uranium and release/confinement of volatile isotopes resulting from partitioning processes.

- Long lived ILW from reprocessing could become the dominant dose contributor. In addition the volume of this ILW must be taken into consideration as it is estimated to exceed the volume of the HLW.

The maximum dose at the surface of a DGR after closure has not been estimated as doing so is beyond the scope of this analysis. What can be said about the impact of recycling CANDU UNF on the surface dose of a DGR is that generating nuclear power from either NU or recycled UNF results in the production of those nuclides that are the largest contributors to surface dose from a DGR.

11.2   DGR Size

11.2.1   Impact of RepHLW and UNF on DGR Size

With respect to UNF and RepHLW, the size of a DGR depends principally on the decay power[38] of the emplaced material over the first few decades after emplacement as shown in Figure 12-1 below. This dictates the spacing between containers required in order to keep the surface temperature of the containers and the buffer material surrounding them below a threshold temperature of 100°C. The temperature limit is imposed to ensure that the buffer material used to isolate the HLW maintains its self-sealing capability, which will inhibit microbial activity near the container and retard contaminant transport. At temperatures above 120°C there is possible conversion of montmorillonite clay to illite, at which point the self-sealing capability of the buffer material may degrade.

The two short-lived[39] fission products, ^{90}Sr and ^{137}Cs, are a major source of decay power of UNF and RepHLW for over 50 years after removal from the reactor. The inventory of fission products such as ^{137}Cs and ^{90}Sr is proportional to the total energy produced. Since the fission products are captured during reprocessing and incorporated into the RepHLW form, the RepHLW is a major source of the decay power in the DGR.

The other major source of decay power of UNF are the TRUs, including ^{238}Pu, ^{241}Am, ^{242}Cm, and ^{244}Cm. The decay heat generated by CANDU MOX UNF and SFR UNF is dominated by the TRUs within 50 years after removal from the reactor, due to the loading of TRUs in fresh fuel and the additional irradiation of the TRUs resulting in the increased production of ^{238}Pu and ^{244}Cm. Therefore, CANDU MOX and SFR UNF are also major sources of decay power in a DGR.

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[38] The decay power is the radioactive property of a material that generates heat.

[39] Short-lived with respect to most of the actinide contributors to the decay heat.
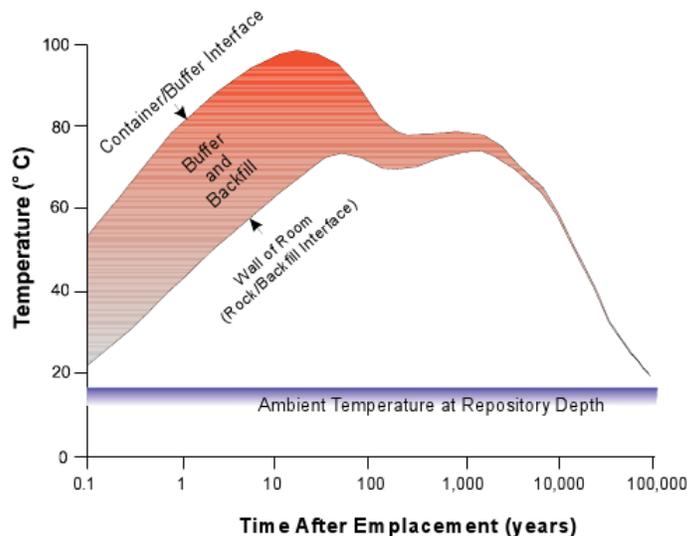
Note: Figure from McMurry et al. (2003).

Figure 11-2: Illustative Example of the Range of Temperature Variation over Time in a Place in a Room [70]

The decay power of UNF and RepHLW are shown in Table 11-1. Estimating the decay power of UNF requires knowledge of its composition, especially the composition of TRUs and FPs. Since there is no publicly available data for the resulting SFR UNF composition when recycling TRUs from CANDU-NU UNF into a SFR, bounding estimates based on recycling LWR UNF were used in this analysis. The lower bound estimate is based on applying the relative change in decay power of LWR TRUs due to irradiation in a SFR to the decay power of CANDU-NU TRUs. The upper bound estimate is based on the decay power of LWR TRUs after irradiation in a SFR. The assumption that the fast reactor TRUs are of LWR origin increases the amount of total TRUs required in a fast reactor, since the fissile content is lower than that of CANDU-NU, and also increases the predicted amount of MAs in the UNF, since there are significantly more MAs in LWR UNF. Details on the calculation of decay power are presented in Appendix E.

Table 11-1
Total Decay Power of HLW in DGR

| Option | Decay Power (MW) | | | | | Total Decay Power in DGR (MW) | |
| | RepHLW (CANDU-NU) | RepHLW (SFR) | UNF (CANDU-NU) | UNF (Recycled) | | | |
				Lower Bound	Upper Bound	Lower Bound	Upper Bound
Reference	-	-	18.5	-		18.5	18.5
1	9.2	-	< 0.1	10.3		19.6	19.6
2a	3.6	-	11.3	1.6	4.9	16.5	19.8
2b	7.3	-	4.0	3.3	9.8	14.5	21.0
2c	9.3	2.6	-	2.7	7.1	14.6	19.0
3a	4.0	-	10.5	1.7	5.1	16.2	19.7
3b	8.0	-	2.5	3.4	10.3	13.9	20.8
3c	9.3	4.6	-	1.9	5.0	15.8	18.9

In summary, the decay power of the material requiring disposal in a DGR is the limiting factor. In all recycling options the UNF from recycling is a significant source of decay power. The recycling options proposed would have the following effect on decay power:

- All options would reduce the CANDU UNF to be disposed of in a DGR by varying degrees. However, all options would also create HLW from reprocessing and UNF from recycling.

- Option 1: MOX fuel for CANDUs is composed of Pu and RU, and is not 'loaded' with MAs. Therefore, all of the MAs or FPs in the CANDU-NU UNF would be included in the HLW from reprocessing. The MOX irradiation would burn some of the plutonium but create additional FPs and MAs, thus the decay power of the material to be placed in the DGR would be higher in Option 1 than the reference scenario.

- Options 2 & 3: fuel for SFRs is comprised of TRUs and RU. The irradiation in SFRs would burn some of the plutonium but create additional FPs. It is unclear whether irradiating CANDU-NU TRUs in a SFR would reduce or increase the proportion of MAs, important contributors to the decay power of UNF. Based on a lower bounding assumption that the decay power of CANDU-NU TRUs would grow by a relatively small amount due to irradiation in a SFR, the total decay power of HLW in the SFR options may decrease relative to the reference scenario. Conversely, if the decay power of CANDU-NU TRUs was to increase up to that of LWR TRUs in a SFR then the total decay power of HLW in the SFR options may increase relative to the reference scenario.

The decay power could be further reduced by allowing all UNF and RepHLW to decay for a longer period of time prior to emplacement in a DGR. This would have the greatest impact on the decay power of RepHLW due to ^{137}Cs and ^{90}Sr being the dominant contributors to the decay power of RepHLW, which each have a short half-life relative to the TRUs. According to the INL

cost basis report [61], Separating Cs and Sr from the RepHLW and storing them for 300 years would be sufficient to reduce their radioactivity to such low levels that their ultimate decay products could be classed as LLW and placed in near-surface disposal.

11.2.2   Impact of ILW on DGR Size

The decay heat produced from the fission products incorporated into the ILW produced during reprocessing the UNF is not expected to be significant. This results in the container spacing for ILW being a function of the volume of the waste packages and the geo-mechanical limitation of the host rock in which the DGR is placed, not the impact of the thermal heat load on the integrity of the engineered barriers. International studies [69] have indicated that the volume of ILW from recycling and reprocessing would at a minimum be equivalent to UNF, with a maximum of being a factor of five greater than the UNF.

Table 11-2
Volume of ILW Produced in Each Option

| Option | ILW Volume (m^3) | | | | | |
	Reactor Operation	Reactor Decomm	Fuel Fabrication	Reprocessing	Decomm Fuel Fabrication & Reprocessing	Total
Reference	0	4636	0	0	0	4636[a]
1	3675	4636	119002	44217	5850	177380
2a	300	273	1554	17839	2340	22306
2b	600	545	3112	34778	4680	43715
2c	1800	2182	9718	45260	7020	65980
3a	1200	1091	1930	19141	2340	25702
3b	2400	2182	3856	38283	4680	51401
3c	9600	10909	14643	45936	6240	87328
[a] The ILW in the reference case is the same as the ILW from the decommissioning of reactors in Option 1.						

11.2.3   Impact of Unused RU on DGR Size

All of the options for the recycling of CANDU-NU UNF result in a stock of RU that must be managed. One option for the management of RU is to convert it to U_3O_8 and place it in a DGR [61]. If RU is to be disposed of, then it would likely be placed in a DGR due to the long-term radiotoxicity of the uranium and the trace amounts of FPs that would remain with the uranium after reprocessing. While the alternative of storing RU above ground is currently being practiced in countries that reprocess their UNF, this storage is considered temporary until such time that a decision is made to recycle or dispose of the RU. In the context of the options being analyzed in this study, it is assumed that all unused RU would be placed in the DGR.

Since RU has negligible decay power, the impact of RU on the size of the DGR only depends on the volume of the waste containers and the geo-mechanical limitation of the host rock in which the DGR is placed, not the impact of the thermal heat load on the integrity of the engineered barriers. In the absence of data on the RU containers, the mass of unused RU is used as an indicator of impact of RU on the size of the DGR for each option, which are shown in Table 11-3.

Table 11-3
The Mass of Unused RU in Each Option

Option	Unused RU (tU)
1	66,332
2a	39,592
2b	79,184
2c	100,976
3a	43,517
3b	87,034
3c	100,945

11.3   DGR Costs

The DGR costs presented in this section assume separate facilities for HLW, ILW, and RU. While it is likely that placing all waste in a single facility would be most cost-efficient, the publicly available data on the costs of ILW and RU disposal are for stand-alone facilities. Therefore, the total costs presented in this section are conservative.

11.3.1   DGR Cost for UNF and RepHLW

According to the NWMO, the total DGR cost for 3.6 million CANDU UNF bundles is $19,360M [71]. Should the Province of Ontario decide to move forward with reprocessing then the current Canadian DGR concept would have to be re-evaluated to assess the impacts of any new forms of HLW that would need disposal. The purpose of this section is to compare the impact of each CANDU UNF recycling option on the cost of a DGR for HLW in Canada. Due to the high-level nature of the analysis presented in this report, a detailed DGR cost estimate for each option is not in the scope of this work and would need to be undertaken in order to obtain a more accurate cost estimate. As such the DGR costs presented in this section should only be used to compare the potential impact of each option on HLW disposal costs.

According to an NWMO report on the financial implications of variations in DGR capacity the cost estimates for DGRs with two different capacities may be used to estimate the incremental cost (i.e., $ per CANDU UNF bundle) of a DGR in Canada [72]. The options for reprocessing and recycling CANDU UNF being analyzed in this report all require the disposal of HLW from the reprocessing of UNF, and the disposal of higher burnup UNF. In these cases, such an incremental cost is not necessarily indicative of certain impacts of these new forms of waste on the DGR cost. One such impact is the handling and encasement of material that is more

radioactive than CANDU UNF, which may require upgrades to the shielding in the fuel handling facilities and to the used fuel containers. Another such impact is the change in the size of and spacing between the underground emplacement rooms to account for the increased decay power of the emplaced materials.

One method of estimating the impact of UNF and HLW on DGR cost is to calculate the incremental DGR cost using the total decay power of all UNF and HLW. The total decay power of all emplaced material is one of the key criteria to determine the size of the DGR (as discussed in Section 11.2.1), and serves as a good proxy for the fuel handling requirements in that material with higher decay power will require upgrades to fuel handling facilities. This approach was used in the OECD NEA report on the economics of the back end of the nuclear fuel cycle [1].

The total DGR cost for a given option is calculated using Equation 11-1:

Equation 11-1: The DGR Costs

$$C_{DGR} = P_{dec} \cdot c_{inc} + c_{fix}$$

where

- P_{dec} is the total decay power of all HLW that would be placed in a repository at the end of the scenario;

- c_{inc} is the DGR cost per unit decay power; and

- c_{fix} is the fixed DGR cost.

The incremental cost of a DGR, c_{inc}, is estimated using the above equation and the two NWMO DGR cost estimates: $19,360M and $31,694M for 3.6M and 7.2M fuel bundles, respectively [71]. The total decay power, P_{dec}, for each size DGR is calculated assuming that each CANDU UNF bundle emits 3.6 W of decay power at the time that it is placed in the DGR [73]. Given the above assumptions, then the incremental DGR cost is calculated to be $952/W, and the fixed DGR cost is calculated to be $7,026M.

As was previously noted, the use of SFR UNF data for the recycling of TRU elements from LWR UNF instead of TRU elements from CANDU UNF means that the total DGR decay power and costs of Options 2 and 3 based on this data are likely over-estimates, and that reactor physics calculations would need to be performed to obtain more accurate estimates. As such, the HLW DGR costs for the SFR options are given as a range, where the lower and upper costs correspond to the lower and upper bounds on the decay power of HLW. The DGR cost for disposing of HLW for each option is shown in Table 11-3.

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Table 11-4
The HLW DGR Cost for Each Option

Option	Total DGR Decay Power (MW)		DGR Cost ($M)	
	Lower Bound	Upper Bound	Lower Bound	Upper Bound
Reference	18.5		24,670	
1	19.6		25,698	
2a	16.5	19.8	22,761	25,847
2b	14.5	21.0	20,856	27,033
2c	14.6	19.0	20,938	25,126
3a	16.2	19.7	22,474	25,747
3b	20.8	20.8	20,272	26,813
3c	15.8	19.0	22,025	25,135

11.3.2 ILW Disposal Costs

The reprocessing and recycling of CANDU UNF would result in the generation of ILW from the operation and decommissioning all of the facilities (reactors, reprocessing and fuel fabrication) that are required to implement a given CANDU UNF recycling option. The disposal of this new ILW has associated disposal costs, which would be in addition to the cost of disposing of ILW that would be generated in Canada in the absence of reprocessing CANDU UNF.

The disposal cost of ILW is taken from the NEA report on advanced fuel cycles and radioactive waste management [74], and converted to 2015 Canadian dollars. This report has the disposal cost for long-lived intermediate level waste (LILW-LL), which is assumed to correspond to ILW. These unit costs are shown in Table 11-5. Given these unit costs and the volumes of ILW, the total disposal cost for ILW for each option is shown in Table 11-6. These costs do not include the disposal costs for ILW generated by the NU fuel cycle in Canada.

Table 11-5
ILW Unit Volume Disposal Costs

Waste Type	ILW (LILW-LL)
Unit Disposal Costs ($/m^3)	$11725

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Table 11-6
ILW Disposal Costs for Each Option

Option	ILW Volume (m^3)	ILW Disposal Cost ($M)
Reference	4636	54
1	177380	2,080
2a	22306	262
2b	43715	513
2c	65980	774
3a	25702	301
3b	51401	603
3c	87328	1,024

11.3.3   Disposal Cost for Unused Reprocessed Uranium

All of the options for the recycling of CANDU UNF result in a stock of reprocessed uranium that must be managed. One option for the management of reprocessed uranium is to convert it to U_3O_8 and place it in a DGR [61]. The unit costs of converting reprocessed uranium to U_3O_8 and disposing of it in a DGR are shown in Table 11-7 [61][40]. These costs are higher than that of disposing of DU due to the higher radioactivity of RU[41]. The total costs of conversion and disposal of unused reprocessed uranium for each option are shown in Table 11-8.

Table 11-7
Unit Costs of Converting and Disposing of Reprocessed Uranium

Process Step	Aqueous RU Conversion	Pyro RU Conversion	RU Disposal
Unit Costs ($/tU)	$14,000	$25,000	$86,000

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[40] All costs are converted into 2015$ CAD.

[41] RU contains trace amounts of FPs, ^{232}U and ^{236}U, which are radioactive and are absent from DU.

Table 11-8
Total Cost of Reprocessed Uranium Conversion and Disposal for Each Option

Option	RU Mass to DGR (tU)	RU Disposal Cost ($M)
1	66,332	6,633
2a	39,592	3,959
2b	79,184	7,918
2c	100,976	10,098
3a	43,517	4,830
3b	87,034	9,661
3c	100,945	11,205

11.4   DGR Conclusions and Options Rankings

With the global experience of constructing, operating and closing a DGR being very limited, this assessment of the impacts of advanced fuel cycles, which have yet to be implemented on an industrial scale, on the disposal of ILW and HLW is very high level and based on numerous assumptions. A summary of the results are shown in Figure 11-3.

Figure 11-3 DGR Impacts Results

11.4.1   DGR Cost Conclusions

The total costs of disposing of all ILW, leftover RU, and HLW in a DGR for each option are shown in Table 11-9. The lower bound on the disposal costs for every option for recycling CANDU UNF is greater than the reference case of placing all CANDU UNF in the DGR. While all of the SFR options may reduce the cost of the HLW DGR, this reduction in cost is less than the increase in costs due to the disposal of ILW, and reprocessed uranium.

Table 11-9
Total Disposal Costs

Option	ILW	RU	Disposal Costs ($M)			
			UNF and RepHLW		Total	
			Lower Bound	Upper Bound	Lower Bound	Upper Bound
Reference	54		24,670		24,725	
1	2,080	6,633	25,698		34,411	
2a	262	3,959	22,761	25,847	26,982	30,067
2b	513	7,918	20,856	27,033	29,287	35,464
2c	774	10,098	20,938	25,126	31,809	35,997
3a	301	4,830	22,474	25,747	27,606	30,879
3b	603	9,661	20,272	26,813	30,535	37,077
3c	1,024	11,205	22,131	25,135	34,254	37,258

For the SFR options, the total disposal costs for each option are proportional to the total amount of electricity produced, with Option 3c having the highest disposal cost, followed by 2c.

All metal fuelled SFR options have higher disposal costs than their corresponding MOX fuelled SFR options. This is due to the higher amount of ILW produced and the higher conversion cost for the RU from pyro-processing relative to aqueous reprocessing.

Options 2a and 3a have the lowest disposal costs of all the recycle options since they generate the least amount of ILW and reprocessed uranium of all the options, a consequence of reprocessing the least amount of CANDU UNF. This results in lower disposal costs for ILW and RU for Options 2a and 3a.

11.4.2   DGR Post Closure Safety Conclusions

The largest contributors to the dose at the surface of a DGR were identified to be some of the long-lived FPs and products of neutron capture, not the TRUs. Since the quantities of these nuclides in the HLW and ILW are proportional to the amount of electricity generated in each option, the ranking of these options with respect to the post closure safety is also proportional to the amount of electricity produced in each option. This assumes that the long-term performance of the HLW immobilization technology would have attributes similar to that of UNF. A more detailed technical assessment would be required to estimate the impact on surface dose from a DGR for each option. It should be noted that should one of these options be pursued by the Province of Ontario, the licensing of the DGR would require that the surface dose should not exceed the dose limit set by the CNSC.

12.   ECONOMIC BENEFITS

12.1   Canada's Current Supply Chain

Canada's nuclear facility supply chain activities (Figure 12-1) consist of: facility construction, commissioning, operations, maintenance, refurbishment and decommissioning, which are carried out by hundreds of companies including SNC Lavlin, GE Hitachi, and Babcock & Wilcox. Most of these companies have a facility located in Ontario to support the existing fleet of CANDU reactors.

Canada's current nuclear fuel cycle supply chain activities are compromised of mining, milling, refining, conversion, and fuel fabrication and assembly. These activities are carried out by very few companies, including: Areva, Cameco, and GE Hitachi, spread across Ontario and Saskatchewan. Used fuel management is planned for by the NWMO.
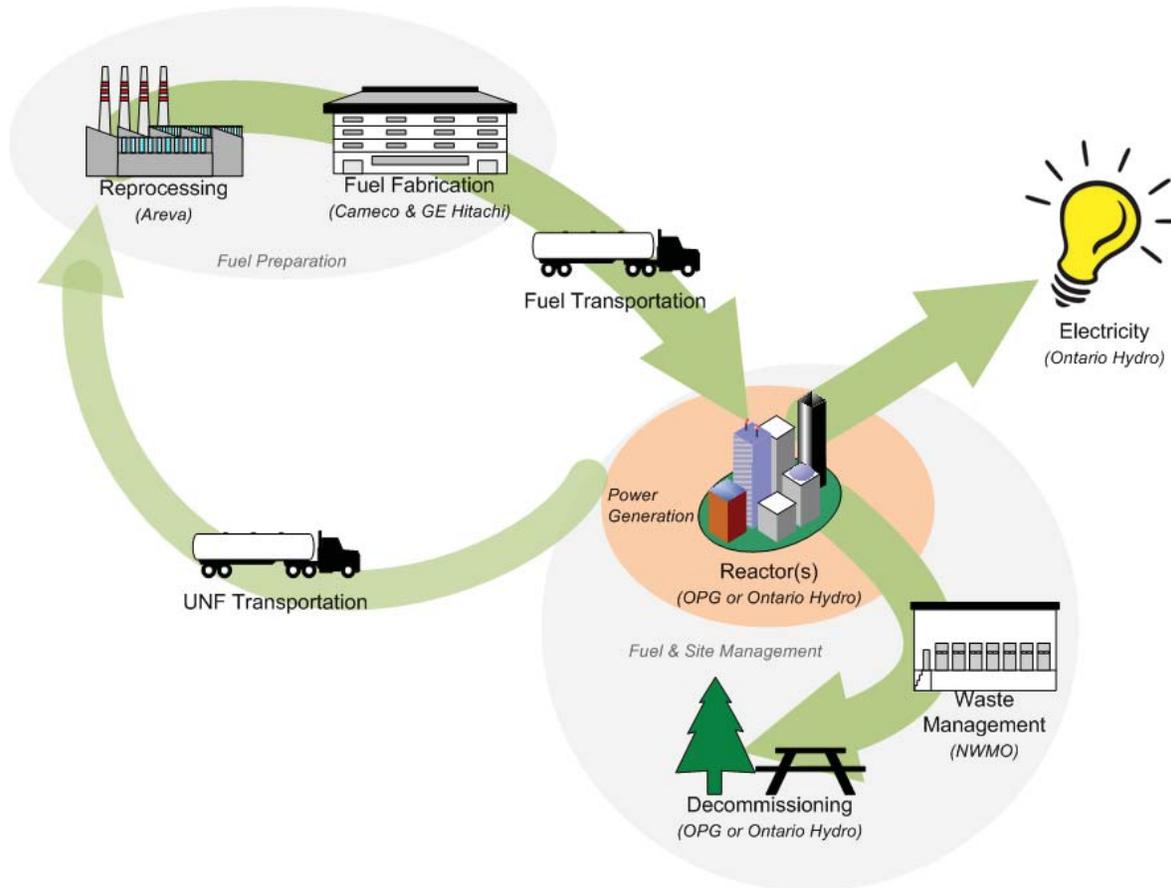
Figure 12-1 Supply Chain

This supply chain, including mining, plant operations and refurbishment projects, has an annual value of $7.9B to Canada, creating 60,000 direct and indirect jobs [75]. In addition to the

annual contributions, each international reactor sale brings $2.57B and 2,284 person years of work to Canada [76].

12.2   Analysis & Results

The investments required to pursue any of the seven options would have an impact on Ontario's economy. One way to capture this impact is to measure the change in gross domestic product (GDP) as the result of new investment. From the costs presented in Section 10 and Section 11, the total economic impact for Ontario can be calculated. This measures the amount of money spent in Ontario and is additive to the GDP. The economic impact for Ontario was calculated based on Equation 12-1 and Equation 12-2 below.

Equation 12-1 $Ontario's\ Direct\ Share\ (\$B) = Total\ Cost * Ontario's\ Market\ Share$

Equation 12-2 $Ontario's\ Indirect\ Share\ (\$B) = Ontario's\ Direct\ Share * Impact\ Factor$

where:

- Ontario's market share equals the percentage of the total cost over the lifetime of the option that is expected to be spent in Ontario; and

- Impact factor captures the stimulation that an economy would go through as the result of a direct investment; a value of 1.2 was assumed for this analysis [77].

Based on Onaran and Galanis [77], the multiplier is derived without discounting.

The results of the economic impact analysis can be found in Table 12-1, with a reference scenario where all CANDU-NU UNF is directly disposed of in the DGR without recycling.

Table 12-1
Total Economic Impact for Recycling Options ($M CAD $_{2015}$)

Options	Capital Cost	O&M Costs	D&D & DGR Costs	Total Cost	Ontario Value (Direct Impact)	Indirect Impact	Total Addition to GDP
Reference			$25,000	$25,000	$22,000	46,400	$48,400
Option 1	$132,693	$290,896	$74,217	$497,807	$438,070	$525,684	$963,753
Option 2a Low	$29,828	$68,504	$37,474	$135,806	$85,558	$102,669	$188,227
Option 2a High	$38,719	$95,204	$40,142	$174,064	$109,660	$131,593	$241,253
Option 2b Low	$59,645	$136,990	$50,272	$246,907	$155,552	$186,662	$342,213
Option 2b High	$77,427	$190,390	$55,607	$323,424	$203,757	$244,509	$448,266
Option 2c Low	$118,953	$272,522	$69,590	$461,064	$290,471	$348,565	$639,035
Option 2c High	$190,081	$432,722	$90,930	$713,732	$449,651	$539,582	$989,233
Option 3a	$31,083	$78,936	$38,567	$148,586	$93,609	$112,331	$205,940
Option 3b	$62,167	$156,462	$52,455	$271,083	$170,782	$204,939	$375,721
Option 3c	$201,493	$496,350	$96,200	$794,044	$500,248	$600,297	$1,100,545

The direct and indirect investments would also create employment in Ontario. The total person years of employment were estimated using Equation 12-3.

Equation 12-3:

$$Employment\ (person\ years) = \frac{Ontario's\ Indirect\ Share * \%\ attributed\ to\ labour}{Average\ Annual\ Salary}$$

where:

- Cost includes capital, O&M, D&D, DGR and indirect total lifetime costs associated with each option.

- The percentage attributed to labour is a measure of how much of the total cost is spent on labour at each phase:
  - Construction: 50.75% [78]
  - Operations: 53.6% [79]
  - Decommissioning: 59.85% [80]
  - Indirect: 56% [81]

- The average annual salary[42] is based on the work required at each phase:
  - Construction: $80,000/year [78] [82]
  - Operations: $105,000/year [78] [82]
  - Decommissioning: $90,000/year [83] [82]
  - Indirect: $33,000/year [84] [82]

The expected years of employement generated by each option, including the reference scenario, are summarized in Table 12-2.

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[42] Salaries from each source were escalated by an inflation factor [82] to arrive at a salary in 2015 terms.

Table 12-2
Expected Number of Person Years of Employment Over Option Life

Options	Direct				Indirect	Total Employment
	Construction	O&M	D&D and DGR	Total		
			146,300	146,300	447,842	594,142
Option 1	740,760	1,306,761	434,319	2,481,840	8,917,544	11,399,383
Option 2a Low	119,208	220,309	156,998	496,514	1,741,648	2,238,163
Option 2a High	154,741	306,176	168,173	629,091	2,232,297	2,861,387
Option 2b Low	238,377	440,560	210,613	889,550	3,166,477	4,056,027
Option 2b High	309,443	612,295	232,964	1,154,702	4,147,774	5,302,477
Option 2c Low	475,402	876,431	291,546	1,643,379	5,912,949	7,556,328
Option 2c High	759,670	1,391,634	380,950	2,532,253	9,153,305	11,685,559
Option 3a	124,227	253,857	161,576	539,660	1,905,551	2,445,211
Option 3b	248,454	503,180	219,760	971,394	3,476,524	4,447,917
Option 3c	805,281	1,596,263	403,030	2,804,574	10,183,263	12,987,837

12.2.1   Option 1 Economic Benefits

In Section 10, Option 1, had a capital cost estimate of $132,693M, an O&M cost of $290,896M and a D&D cost of $39,806M. In Section 11, the DGR costs were estimated at $34,411M. The costs for reactor refurbishing and reprocessing are approximately the same therefore since CANDU is a Canadian technology Ontario is assumed to provide 96% of the content. For reprocessing this would decline to 80% based on the AREVA proposal to build a MOX facility in the United Kingdom. Therefore, Ontario's market share is estimated to be 88%, with the remaining 12% being provided by international suppliers (primarily for imported reprocessing technologies) [85] [86]. Indirect impacts are estimated using an impact factor of 1.2 [77]. Therefore, the economic benefit of this option was calculated to be: $438,070M direct impact and $525,684M indirect impact.

For each dollar spent on capital, 50.75% is attributed to direct labour, with an average annual salary of $80,000 [78] [82]. The amount attributed to direct labour for O&M costs is slightly higher at 53.6%, with an average salary of $105,000/year [78] [79] [82]. The amount attributed to direct labour for D&D and DGR costs is 59.85% [80], with an average annual salary of $90,000 [82] [83]. The expected number of person years of employment over the lifetime of the option was calculated to be 2,481,840 years, using Equation 12-3.

Indirect jobs were assumed to have an average annual salary of $33,000, which is equal to the median Canadian salary [82] [84]. Of the $525,684M indirect dollars, 56% [81] was attributed to indirect labour, the total indirect person years of employment were estimated to be 8,917,544 (using Equation 12-3).

12.2.2 Options 2 Economic Benefits

Options 2a, 2b and 2c use reactor and reprocessing technologies that are being developed abroad. Therefore, as a conservative assumption it is assumed that, Ontario's share of the total cost would decline to 63%. Using the formulae and impact factor presented in Section 12.2, the total economic benefits, based on the costs in Section 10 and 11, for each option respectively have ranges of $188,227M to $241,253M, $342,213M to $448,266M, and $639,035M to $989,233M, while the person years of employment had 2,238,163 to 2,861,387, 4,056,027 to 5,302,477, and 7,556,328 to 11,685,559 for Options 2a, 2b and 2c respectively. Low and high cost estimates were used for Options 2a, 2b, and 2c. The low estimates are based on vendor's estimates. However, based on recent experience for reactor builds, these estimates seem overly optimistic. As a consequence, an expert valuation was provided for the high estimates (see the discussion in Section 10).

12.2.3 Options 3 Economic Benefits

Options 3a, 3b, and 3c would create similar benefit for Ontario per dollar spent as Option 2. Thus, Ontario's share of the total cost would be 63%. Using the formulae and impact factor presented in Section 12.2, the total economic benefits, based on the costs in Section 10 and 11, are $205,940M, $375,721M, and $1,100,545M and the person years of employment are 2,445,211, 4,447,917, and 12,987,837 for Options 3a, 3b and 3c respectively.

12.3 Conclusion and Options Ranking for Economic Benefits

The development of another nuclear energy system in Canada could bring substantial economic benefits. From Table 12-1 and Table 12-2 above, the economic impact and person years of employment created are directly related to the investment made for each option. Ranking the seven options based on the total economic impact and person years of employment presented above may not provide the complete picture.

For instance, 'Economies of Scope' is the ability of a company to use the same or similar production processes (or capabilities) for different products. This is one of the key measures for indicating industrial success of supply chain diversification; since it can lead to achieving a competitive cost advantage through lower unit costs. Ontario's existing nuclear capabilities are deemed suitable for an alternative nuclear system depending on their ability to achieve economies of scope. Successfully diversifying Ontario's nuclear industry will depend on how well companies in the nuclear supply chain can integrate the requirements of a new reactor technology with their current production process or capabilities. If Ontario's supply chain cannot diversify to include other reactor systems the competitive cost advantage from economies of scope would not be achieved, making it more difficult to compete with international suppliers. In addition, there is a benefit associated with choosing technologies which other countries would adopt as well. Successfully deploying FOAK technologies could lead to international sales with additional economic benefits and job creation.

Option 1 leverages existing Canadian technologies for the reactors, fuel fabrication and DGR; therefore, Ontario's supply chain can likely achieve economies of scope through technology and process synergies. The MOX fuel is currently under consideration internationally as a method to deal with stockpiles of UNF and Pu from historic weapons programs. If MOX fuel is successfully deployed in Ontario's reactors, this might generate the confidence needed to sell the technology abroad. However, for the reprocessing and fuel fabrication facilities a great amount of experience exists internationally; therefore, Ontario is unlikely to become an exporter of fuel fabrication and reprocessing technologies if Option 1 is pursued. However, the total impact and number of jobs created estimated is significant, suggesting Option 1 would have a very positive effect on the Ontario economy.

There is no commercial experience for the GANEX reprocessing or fuel fabrication required to fabricate fuels with high amounts of Pu-241 which is used in Option 2. This fuel fabrication would therefore require more automation and "hands off" work which would require commercial development. The 1,500 MWe MOX SFR technology in this option also has not yet been commercialised. If Ontario were to build all of these FOAK facilities it would develop valuable knowledge, skills and manufacturing that could then have the potential to be exported. Options 2a and 2b have a relatively low total expenditure and jobs created, decreasing the total impact of these options. Option 2c, due to the additional reactors and the recycling of SFR MOX UNF, has a higher total expenditure; increasing the total impact compared to Options 2a and 2b. However, Option 2c's economic benefit is only 60% of the economic benefit attributed to option 1.

Option 3 presents the greatest opportunity for Ontario to increase exports internationally, since neither, the fast reactor metallic fuel fabrication and reprocessing or the reactor technologies have been commercialized. In this case metallic fuel would represent a significant differentiator, however if other countries (France, Russia, United States) choose to grow nuclear fleets with oxide fuels there may be less scope for technology export. It should be noted though that there are some countries abroad that have established research and development programs that would enable them to commercialize faster than Ontario. Although these new technologies may not have the same synergies expected for Option 1, there is an opportunity for Canada to be first to the market with metallic SFRs and/or pyro-processing. This would enable the development of valuable knowledge, skills and manufacturing that could then be exported. The total GDP impact and jobs created estimates for Options 3a, 3b and 3c increase with the expected expenditure. Since Option 3c has the largest fleet of reactors of all the options considered it also has the greatest total expenditure and greatest number of jobs created. However it should be noted that for Option 3 the 380MWe plant size was chosen in order to include a metallic fuel PRISM like technology. When compared with the larger reactor size in Option 2 it is possible that economies of scale work against Option 3, leading to higher costs and the larger GDP/jobs created impact.

Figure 12-2 Economic Impact Results

The realization of economic benefits for Ontario as a result of making an investment can depend on several factors. A ranking was devised based on the magnitude of the economic benefits. The economic benefits are in direct proportion to the size of the original investment, the share of investment spent in Ontario, and the allocation of investment for labour expenditure. The greater the magnitude of these factors, the greater the economic benefit. Other factors were also incorporated into deriving the economic benefits. The economic factors considered were incorporated through a multiplier mechanism to arrive at a measurement of economic benefit. For each option, the economic expansion was driven by an investment that required spending on labour in order to provide Ontario with new products. The income generated from an investment would be spent at new and existing businesses, which in turn would create more jobs. The process of economic expansion captures several economic factors channelled through a multiplier mechanism. The multiplier mechanism reflects both behavioural and structural aspects of the economy that is applicable to all options. The differences in economic benefits were driven by two key differentiating economic factors amongst the options: 1) the size of the original investment, and 2) the share of investment spent in Ontario. That being said, Options 1 and 3c generated the greatest economic benefits,

which had approximately the same economic impacts when comparing their order of magnitudes.

13.   CONCLUSION AND FINAL REMARKS

The information presented in this report was intended to provide a high level review of each of the three options (Table 13-1). Only reactors fuelled with plutonium were considered for recycling since the fissile content of uranium in CANDU UNF is very low; similar to tails from an enrichment plant. However, the fissile content of the Pu in CANDU UNF is favourable.

Table 13-1
Facilities by Option

	Reactor	Fuel	Reprocessing	Fabrication
Option 1	Seventeen[43] 818 MWe Refurbished CANDU	MOX with 1.1% Pu	Co-Ex (Demonstrated)	MELOX (Commercialised)
Option 2a	One 1,500 MWe SFR	MOX Fuel with 33% Pu	GANEX (Demonstrated)	MELOX (Commercialised)
Option 2b	Two 1,500 MWe SFRs			
Option 2c	Eight 1,500 MWe SFRs			
Option 3a	Four 380 MWe SFRs	Metallic Fuel with 29% Pu	Pyro-Processing (Novel)	Integrated System (Novel)
Option 3b	Eight 380 MWe SFRs			
Option 3c	Forty 380 MWe SFRs			

13.1   Technology Risk

The different technologies used in this study range from fully commercialised to novel (without demonstration) approaches.

13.1.1   Reprocessing Technology Risk

Canada does not have any experience in commercial reprocessing and would rely heavily on international experts to develop the regulatory framework that would be required. Two of the technologies considered (Co-EX and GANEX) have been demonstrated but not yet commercialised. Commercialisation is expected to be relatively straight forward, since both have evolved from the PUREX process, which has been successfully deployed in France, The United Kingdom and Japan. The other reprocessing technology considered in this study is pyro-processing, which is still under development and therefore has a high degree of risk. If successful, pyro-processing will likely become the preferred reprocessing method for metallic fuels since it can reprocess hotter UNF, therefore the required cooling interval is reduced allowing the UNF to be recycled more quickly.

13.1.2   Fabrication Technology Risk

Currently all commercial power reactors use oxide fuels, with 5% of the world's reactors using mixed oxides from recycling. Therefore, the MELOX fuel fabrication process for Options 1 and 2

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[43] Twelve reactors, eight at the Bruce site and four at the Darlington site were included. Five of these reactors were then again refurbished to extend their operating life.

are well developed internationally. Canada has some experience with oxide fuel fabrication; however additional knowledge and regulations would need to be developed for fuels made from recycled material and fuels with high fissile content. Metal fuels, like the one proposed for Option 3, have been used in military and research applications but not commercial power generation. In order to deploy this technology in Canada safety and licensing issues would need to be addressed.

13.1.3   Reactor Technology Risk

Three reactor technologies were studied: CANDU reactors, MOX-fuelled SFRs and metal fuelled SFRs. The CANDU reactors were designed in Canada and have several decades of operating experience. There is some risk associated with the multiple refurbishments required, and the reactors would have to be relicensed to be able to use MOX fuel; however these risks are assumed to be manageable due to the depth of knowledge and experience with CANDU reactors in Canada.

Proponents advocate for SFRs since they operate with a fast neutron energy spectrum that burns transuranic material (TRUs), such as Pu and MAs, to a greater extent than thermal reactors. The MOX-fuelled SFR has been demonstrated and had operated in several countries with varying degrees of success. If Canada were to deploy this type of reactor outstanding operational and safety concerns would first need to be addressed. The metal fuelled SFR has the most risk of the three options included in this study due to its low technology readiness level. However, if successful it may have an advantage over the MOX-fuelled SFR since metal fuel naturally has a higher heavy metal density which increases the probability of fission within the fuel therefore reducing the amount of fissile material required.

13.2   Criteria Analysis Results

13.2.1   Overall Results

Eight mutually agreed upon evaluation criteria were used to generate a high level assessment of the challenges and benefits associated with each option (Table 13-2).

Table 13-2
Evaluation Criteria

Criteria	Description	Key Measure(s)
Environmental	Impact of the reactor and supporting facilities on the environment: • Radiological discharges • Reprocessing wastes • Implications on marine environments	Volume of UNF and high level waste (HLW) produced

Criteria	Description	Key Measure(s)
Safety and Licensing	Effort required and perceived barriers to licensing the required facilities: • Existing experience (in Canada, internationally) • Known safety concerns • Outstanding technical challenges	Experience and technology maturity
Sustainability	Ability to produce electricity without drawing from Canada's existing uranium reserves.	Electricity produced from recycled material
Non-Proliferation	Barriers in place to impede the diversion or undeclared production of nuclear material. • Accessibility and usefulness for weapons manufacture • Ease of process modifications to increase the purity and quantity of Pu produced	Inherent proliferation resistance of the reprocessing technologies and fuels
Community and Social Considerations	Analysis of the public perception of the risks and benefits, with a focus on tools for measurement and growth of community support.	N/A[44]
Costs	Estimated costs of the capital, operating and decommissioning of all new facilities including. • Reactor costs • Reprocessing & fabrication costs • Storage costs	Total investment measured in $/MWh produced
Waste Disposal	Effort required to properly dispose of all of the waste from the following streams: • UNF in storage at the end of the study period • Waste from reactor operation,reprocessing and fuel fabrication • Reprocessed Uranium	Total cost to dispose of all waste generated
Economic Impact	Analysis of the potential impact on the economy; in direct development and through supply chain growth.	GDP impact and person years of employment

Using the criteria defined in Table 13-2 above, each of the options were analyzed and ranked based on their relative impact (Figure 13-1).

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[44] Given the lack of Canadian specific societal data on the options analyzed in this study, a ranking was not possible. Further study and public surveys are needed.

Impacts	Environmental	Safety & Licensing	Sustainability	Non-Proliferation	Cost	Waste Disposal	Economic Impact
Option 1							
Option 2a							
Option 2b							
Option 2c							
Option 3a							
Option 3b							
Option 3c							

Best Option   Worst Option
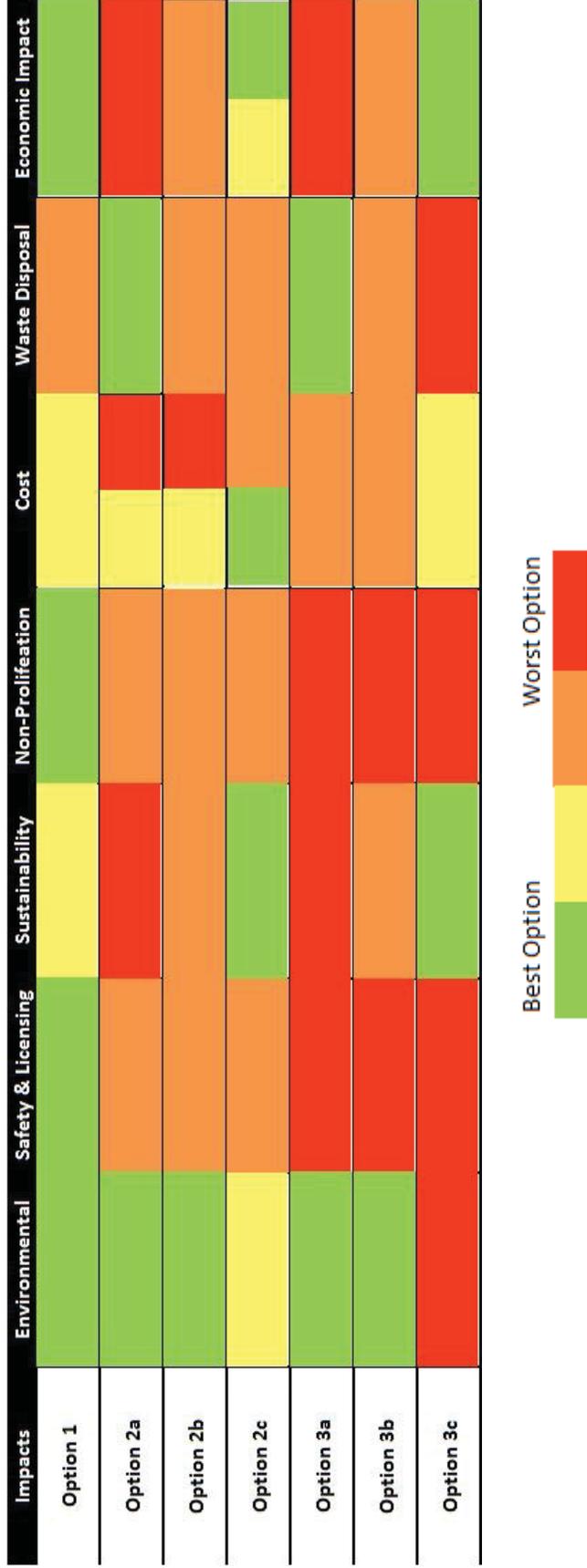
Figure 13-1 Final Rankings

13.2.2   Results of Significant Interest

The key areas of interest identified by the MoE were the impact of recycling on the need for a DGR and the costs.

DGR Findings

Nuclear waste includes long-lived nuclides which can remain dangerous for over 10,000 years; therefore they must be safely isolated in a permanent storage solution. Canada has decided to use a DGR for long-term storage with the option to retrieve the material if desired. In the CANDU-NU system, these nuclides are primarily contained in the UNF from the reactors. In the systems analyzed in this study, these nuclides are dispersed among several waste streams, including: CANDU-NU and advanced reactor UNF, HLW from reprocessing, RU, and ILW from reprocessing and fuel fabrication. The characteristics of the waste produced by each option vary significantly based on total electricity produced, fuel type and mass of UNF reprocessed. These were used to determine the total costs, including conditioning and disposal of all ILW, leftover RU, HLW and UNF in a DGR for each option, which are shown in Table 13-3.

Table 13-3
Total Conditioning and Disposal Costs

			Disposal Costs ($M)			
			UNF and RepHLW[a]		Total	
	ILW	RU	Lower Bound	Upper Bound	Lower Bound	Upper Bound
Reference	54		24,670		24,725	
Option 1	2,080	6,633	25,698		34,411	
Option 2a	262	3,959	22,761	25,847	26,982	30,067
Option 2b	513	7,918	20,856	27,033	29,287	35,464
Option 2c	774	10,098	20,938	25,126	31,809	35,997
Option 3a	301	4,830	22,474	25,747	27,606	30,879
Option 3b	603	9,661	20,272	26,813	30,535	37,077
Option 3c	1,024	11,205	22,131	25,135	34,254	37,258
a: The range provided is based on a low and high estimate for CANDU-NU UNF in SFRs using LWR UNF and weapons PU as proxies, since CANDU-NU UNF in SFRs data was unavailable.						

The lower bound on the disposal costs for every option for recycling CANDU UNF is greater than the reference case of placing all CANDU UNF in the DGR. While all of the SFR options may reduce the cost of the HLW DGR relative to reference system, this reduction in cost is less than the increase in costs due to the disposal of ILW and RU.

Cost & Economic Findings

Undiscounted cost estimates were produced for all the options. These estimates included the cost to construct/refurbish the reactors, to construct the reprocessing and fabrication facilities, to operate all facilities, and to decommission all facilities (Table 13-4). Excluded from these

costs were DGR costs, which were calculated separately, and materials handling/transportation costs of nuclear material since it was deemed inappropriate for inclusion in a high-level feasibility study. The expected impact on GDP and the employment generated were estimated based on the total investment required for each option. It is important to note that all costs in this report are undiscounted. If costs were discounted this would place greater weight on the short-term aspects of a project, favouring projects with a longer duration and lower up front costs. This would have a significant impact on the results presented in this report since investment timelines vary substaintially. However, estimates of GDP and employment growth are traditionally calculated using undiscounted costs; therefore these results would be unaffected.

Table 13-4

Cost and Economic Results

	Electricity Generated (MWh)	Undiscounted Cost ($ M)[b,c]	Unit Cost ($/MWh)	Person Years of Employment (Thousands)	Total GDP Impact ($ M)	DGR Costs ($ M)
Reference[45]	2,980,454,902	463,396	155	594	48,400	24,725
Option 1	670,599,000	463,396	155	11,399	963,753	34,411
Option 2a	670,599,000	107,281 to 145,540	160-217	2,238 to 2,861	188,227 to 241,253	26,982 to 30,067
Option 2b	1,341,198,000	214,532 to 291,049	160-217	4,056 to 5,302	342,213 to 448,266	29,287 to 35,464
Option 2c	4,023,594,000	214,532 to 679,829	106-169	7,556 to 11,685	639,035 to 989,233	31,809 to 35,997
Option 3a	679,540,320	119,344	176	2,445	205,940	27,606 to 30,879
Option 3b	1,359,080,640	237,277	175	4,448	375,721	30,535 to 37,077
Option 3c	5,436,322,560	758,288	139	12,988	1,100,545	34,254 to 37,258

a: Due to limited publicly available data, some information for Options 2 and 3 was extrapolated from LWR data.

b: Reactor capital costs are optimistic and may significantly underestimate the true cost.

c: Due to the long time horizon and low technology readiness of some technologies, all cost estimates have a large uncertainty range estimated to be -50% to +100% of the estimates used in this study.

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[45] Reference option assumes all CANDU UNF from the existing reactors will be placed directly in the DGR, without recycling. Any additional UNF that would be generated from new reactors was excluded from the analysis.

Option 1 was found to be the most economic option to reprocess all of the CANDU-NU UNF using a once through cycle. The SFRs used in Options 2 and 3 are found to have a high unit cost in the 'a' and 'b' sub-options, with a significant reduction occurring in sub-option 'c'. This was primarily due to the multiple recycles system used in Options 2c and 3c which recycles the SFR UNF compared to the once through cycles of Options 2a, 2b, 3a and 3b that only recycle CANDU-NU UNF. This is supported by existing fast reactor literature, which focuses on continuously recycling in order to realise the full potential of the SFR technology. In addition, pyro-processing can reduce the required UNF cooling time, further increasing the number of recycles that can be achieved within a single reactor lifetime. The benefits can be seen by comparing Options 2c and 3c, noting the increased electricity generated and lower unit cost in Option 3c.

All of the options require a substantial investment, which is expected to create a substainial number of jobs in Ontario and make a significant contribution to the province's GDP. In cases where Canadian technology (e.g., CANDU reactors) can be used this impact is larger since more of the investment will stay in Ontario. Conversely, options that rely on technologies developed by others internationally will have reduced GDP and jobs creation since a higher proportion of the components and expertise will have to be imported. For the options that include technologies that have not yet been fully developed, opportunities may exist for Ontario to secure a portion of the supply chain, which could lead to exports if the technology is deployed in other countries.

13.3   Challenges and Benefits

The findings of this report demonstrate that there are sizable challenges to recycling Canada's CANDU used fuel. At current uranium prices, the costs and risks associated with recycling CANDU UNF make recycling unattractive due to its low fissile content. However, reprocessing and reactor technologies are still under development. Therefore, as technologies mature and uranium becomes more expensive, Canada's used fuel inventory could become a valuable source of carbon-free electricity.

Should Ontario consider the deployment of one of the options analyzed in this study, a number of technical challenges would need to be overcome:

1   Each option requires a significant upfront investment to establish the infrastructure needed to recycle fuel (i.e., new or refurbished reactors, and fuel reprocessing and fabrication facilities. Options would also require numerous investments over a long timeframe to maintain/replace this infrastructure as it ages.

2   New waste streams would be created during reprocessing and recycling that do not currently exist in Canada. Waste owners would be required to find a long-term solution for all waste types not covered under the Nuclear Fuel Waste Act.

3   A time horizon of 150 years, ending in 2165, was used for this study. This introduced a number of issues such as; multiple refurbishments of CANDU reactors (Option 1) which

will be technologically challenging and may require additional R&D, recycling systems are evolving and it is difficult to predict how they may progress over the study period

4   SFRs have yet to achieve commercial success. Most of the fast reactors built within the past decade faced significant reliability and safety challenges that have seriously affected performance. More R&D will be required before deployment in Canada.

5   Commercial reprocessing has never been carried out in Canada. Significant knowledge and skills must be developed or introduced to ensure that reprocessing facilities could be safely installed, licensed, regulated, operated and eventually decommissioned. Note: the technologies considered in this report range from unproven to internationally demonstrated designs.

6   Recycling will increase proliferation risk compared to the once-through natural uranium fuel cycle. The Licensee (Operator) and the Canadian Nuclear Safety Commission (CNSC) would have to work closely with the International Atomic Energy Agency (IAEA) to ensure adequate safeguards are put into place.

In addition to the above-mentioned technical challenges, there are potential social and public perception challenges (i.e., social license) which are not included in the scope of this report.

If these challenges are overcome, Ontario could experience benefits related to recycling such as:

1   A reliable source of baseload nuclear electricity, without further depletion of the Canadian uranium reserves.

2   Potential for less long-lived and/or less radiotoxic nuclear waste destined for long-term disposal.

3   Growth in Ontario's nuclear knowledge and supply chain, including potential for international exports of certain components produced by the Canadian supply chain.

4   Opportunities for Canada to participate as a leader/world expert in international deployments of similar recycling systems.

As technologies develop, it is expected that there will be many opportunities to optimize the recycling system, further improving the characteristics of the final nuclear waste material requiring disposal. In addition, further studies may be undertaken on other potential recycling systems which may not require reprocessing and on technology innovations that could manage final nuclear waste material which could result in lower storage costs.

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Appendix A

Molten Salt Reactors

A.1   Molten Salt Reactors

A.1.1   Salts and Molten Salts

A salt is a chemical compound formed by the complete reaction of an acid with a base, replacing the hydrogen of the acid with the anion of the base. For example, common table salt (NaCl) could be formed by the reaction:

$$NaOH + HCl \rightarrow H_2O + NaCl.$$

If the salt formed is solid at standard temperature and pressure, but has an 'elevated' melting point (a few hundred °C), the salt may be called a 'molten salt'. Above the melting point (800 °C for NaCl) a salt dissolves into a homogeneous mixture of the anions and cations (Na^+ and Cl^- for NaCl). In industrial applications, molten salts are of considerable interest for their heat storage capacities, having high heats of fusion and heat capacities which can be greater than that of water. For such an application, the choice of salt would depend on the melting and boiling points and heat capacity required.

A.1.2   Molten Salts as Nuclear Fuel

Another application of molten salts is as a carrier for nuclear fuel in liquid form. There are a number of possible salts for this application, but they are almost invariably fluoride salts. One of the more interesting salts, from a nuclear fuel perspective, goes by the name 'FLiBe', and has the chemical form Li_2BeF_4. This salt dissolves at a melting point of 459°C into $2LiF^+ + BeF_2^{-2}$, and remains a liquid up to its boiling point of 1430 °C. Both the fissile and fissionable actinides in a nuclear fuel cycle can be suspended in the salt as fluorides, producing a mixture which is very chemically stable because the fluorine radicals do not attack the already completely fluoridated actinide compounds. For example, if uranium is to be used as the source of fissile and fertile material, the fuel would be:

$$2LiF^+ + BeF_2^{-2} + UF_4.$$

Such a nuclear fuel would have the following advantages over solid fuels:

1. The fluorine compounds are very tightly bound, making the fuel mixture resistant to radiation damage. Any radiation induced splitting of the anions or cations would be quickly repaired.

2. The low vapour pressure, high heat capacity and large temperature range over which the fluid is liquid are ideal qualities for heat removal.

3. The high temperatures possible in the salt make very efficient electrical generation (50% or higher thermal energy to electrical energy conversion efficiency) possible. Also, after

electricity generation there would likely be considerable heat left over, making MSRs a good choice for consequent production of process heat.

4. Li-7, Be and F all have very low neutron absorption cross sections, so do not interfere with the nuclear reactions.

Potential problems associated with a molten salt fuel could include:

1. The beryllium in the salt is quite chemically toxic. (While other molten salts without Be are possible, and are sometimes chosen for just this reason, the chemical and physical properties are not quite as suitable. One example would be NaF^+ and $ZrF4^-$).

2. The lithium in the salt must be very highly enriched in the isotope Li-7 relative to the other naturally occurring isotope Li-6. This is because Li-6 has an extremely high cross section for the tritium –producing reaction:

$$n+ Li\text{-}6 \rightarrow T + \alpha.$$
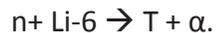
The technology for lithium enrichment is sensitive because Li-6 is a controlled material due to its use in thermonuclear weapons, where the above reaction is key.

Both Li-7 and Be-9 would also produce tritium under the bombardment of high energy neutrons, but at much reduced rates. These rates are large enough to create tritium leakage problems however[46].

3. While the corrosive properties of the salt are relatively benign initially, the salt chemistry must be monitored and adjusted during irradiation to keep the salt in a net reducing[47] state to minimize corrosion. Since the fuel would be highly radioactive, such chemical adjustments would be non-trivial, but the corrosive potential of the salt is considerable otherwise.

A.2   Characteristics of a Reactor with Molten Salt as the Fuel

A.2.1   Primary and Secondary Heat Removal Circuits

In a Molten Salt Reactor (MSR), the functions of fuel and coolant are combined. A primary circuit would pump the molten salt through an active core region, where it is heated directly by fissions occurring in the dissolved nuclear fuel, and then outside the core where the heat would be removed by a secondary circuit. A number of heat removal possibilities for secondary circuits are available, although the high temperatures and violent reaction of molten salts with

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[46] Tritium production rates using FLiBe salt would be similar to those in a CANDU, but the tritium would be produced in a very hot coolant rather than in a cool moderator. Containing the tritium inside the reactor containment is one area of required research.

[47] A reducing state is one in which the constituents of the salt prefer, on net, to donate electrons to the solid structure of the container. This prevents the formation of anion-metal compounds (which can be described as a process in which the metal of the container donates an electron to the anion) on the surface – i.e., corrosion.

water make a secondary gas circuit, or molten salt circuit (without the dissolved nuclear fuel), more likely choices.

A.2.2   Size

The molten salt can sustain a very high fissionable material loading. Coupled with the excellent heat transport characteristics of the salt, the result is that very high power densities (power produced per unit volume) are possible in an MSR and hence, for a given power output, the reactor can be made quite small with consequent economic benefits. In particular, small cores lend themselves to an 'integral' design where the heat exchangers are enclosed in the primary containment. For reactors built off-site, transportation issues are also simplified.

A.2.3   Fuel Type

Fuel performance issues are simplified in an MSR with respect to a solid fuel reactor, as issues with respect to fuel centerline melting, element cracking, or cladding defects no longer exist. The homogenization of the fuel eliminates fuel shuffling issues. Since most fissionable material can be made into fluorides and added to the fuel without significantly changing the physical properties of the fuel (since they are neutral and suspended), an MSR is a good vehicle for reburning the UNF from other reactors, such as CANDU, without chemical or isotopic separation of its components.

A.2.4   Spectrum

An MSR can in principle be a fast-spectrum or thermal-spectrum reactor. The salt-fuel mixture does not provide much moderation, so a thermal spectrum reactor would require the inclusion of moderating regions in the core. Often a fast spectrum reactor type is chosen for an MSR because:

1. The design is simpler, since no moderation is required. The core can be essentially just a tank of molten salt and fuel.

2. The fuel cycle advantages of a traditional fast-spectrum reactor are obtained, namely the ability to fission minor actinides from spent fuel[48] (from other reactors) and the possibility of breeding fertile material into fissile material.

A.2.5   Fuel Enrichment

The advantages of an MSR are mainly in fuel flexibility and safety, and not in neutron economy, so fuel would have to be enriched, especially if:

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[48] Thermal-spectrum reactors can also disposition minor actinides. Although they do not fission in a thermal spectrum, multiple neutron captures create a chain of nuclides containing some highly fissionable ones, where the actinide is likely to be destroyed. The advantage of a fast spectrum is that the destruction of minor actinides can be combined with fissile material breeding (usually U-238 $\rightarrow$ Pu-239 or Th-232 $\rightarrow$ U-233).

1. The reactor is required to burn non-fissile actinides, such as the minor actinides in CANDU UNF.

2. The reactor is fast-spectrum, since the reduced cross section for fast fission is very low.

3. The reactor is quite small, since neutron leakage from the core would be large.

A.2.6   Online Fueling and Reprocessing

The flow of fuel through and out of the core of an MSR lends itself to two online operations: the removal of fission products and the addition of fresh fuel without shutdown. The first is a unique possibility for this reactor type, but the chemistry may be complex and difficult in a high radiation environment. The second is a characteristic shared with pressure tube reactors such as the CANDU, but the low pressure and ability to add small quantities which are quickly homogenized throughout the core make MSR refuelling particularly easy.

A.3   Safety

The MSR has some unique advantages related to the fuel type and lack of a separate coolant:

1. The low vapour pressure of the molten-salt means the reactor can be run unpressurized.

2. There is no bottleneck for heat transfer at the edge of the fuel pins. This prevents two types of accidents on high heat in a fuel element: a) a phase change, such as the production of steam in an LWR next to a fuel element, which blocks further heat transfer and leads to a runaway temperature excursion in the element and b) fuel element centerline melting.

3. A meltable fuel drain plug (called a 'freeze plug'), which disappears at a pre-defined temperature, can be installed at the bottom of the core. Such a drain would allow the molten salt to flow into high surface-to-volume tanks, stopping the nuclear reaction and allowing passive heat removal.

4. The reactor is insensitive to small changes in the fuel, unlike solid fuel reactors whose thermal and radiation damage behaviors may be significantly different as the constituents change, and which therefore requires requalification for each new fuel type.

Small MSRs have the following two safety features common to many SMRs:

5. On shutdown, thermal siphoning in the salt can often be shown to be sufficient to remove decay heat, so that active systems do not have to maintain cooling.

6. The small size of the core lends itself to integral designs, where both the core and secondary electrical production circuit are contained in one containment vessel, reducing or eliminating public exposure to radionuclides in the event of a core malfunction.

A.4 Required Research

A possibility inherent in any liquid-fuel reactor is the ability to remove fission products continuously online. However, because of the varied nature of these fission products, and the very high radiation, the chemistry would be non-trivial and operation would have to be completely automated. Reactor designs requiring this ability would require significant further research. Irrespective of fission product removal, the chemistry of the molten salt would have to be continuously monitored and adjusted to avoid corrosion problems.

If fission products are not to be continuously removed, then they must remain trapped in the molten salt. The chemical properties of the free fluorine ions in a fluoride salt reactor appear to be admirably suited to this purpose, but further research is still required to demonstrate this.

A.5 Experience

A.5.1 The Aircraft Reactor Experiment (ARE)

The ARE project was started in 1946 to design a small, nuclear powered jet engine for the US military. A MSR was chosen as the concept and by 1956 General Electric had produced the X-46, a nuclear reactor coupled to two J-47 jet engines. This reactor had the following specifications:

Table A-1
Characteristics of the ARE – the First Operational MSR in the USA

Characteristic	Value
Molten salt	$NaF\text{-}ZrF_4\text{-}UF_4$
Power	2.5 MW_{th}
Secondary coolant	Sodium
Spectrum	Thermal
Moderator	BeO

Air flowing in to an inlet plenum was directed into the reactor for direct heating by the secondary circuit and out through the turbines of the jet engines. The engine ran for a total of 5000 MWh of operation during its test phase, but was never put in service due to changing military requirements.

A.5.2 The Molten Salt Reactor Experiment (MSRE)

Oak Ridge National Laboratory (ORNL) built a molten salt reactor in 1965 called the Molten Salt Reactor Experiment with the following specifications:

Table A-2
Characteristics of the MSRE – the Second Operational MSR in the USA

Characteristic	Value
Molten salt	LiF-BeF$_2$-ZrF$_4$-UF$_4$
Power	7.4 MW$_{th}$
Secondary coolant	FLiBe
Spectrum	Epithermal
Moderator	Graphite

This reactor was operated for four years at full power between 1965 and 1969. Between 1970 and 1976 the same team designed the Molten Salt Breeder Reactor (MSBR) a fast-spectrum reactor based on the use of thorium as a breeding material. This project was cancelled due to competition for funds with the Liquid Metal Fast Breeder Reactor (a more conventional fast reactor with sodium coolant and solid fuel elements).

A.5.3   Summary of Prior Experience

MSRs were built and operated successfully in the 1950s and 1960s by the U.S., and similar programs in Great Britain and Russia also concluded that there were no insurmountable obstacles to an MSR power program. MSRs were abandoned not for technological reasons, but because of changing requirements, a political shift in favour of conventional fast breeder reactors, and ultimately because of the technological lead of water-cooled reactors which were being built all over the world by the mid 1970s.

A.6   Current Prospects in Canada

A.6.1   The Terrestrial Energy (TE) iMSR

Since MSRs were named one of the six Generation-IV (GIF) reactor technologies by the international GIF consortium in 2002, there has been renewed interest in the technology. GIF technologies are those expected to have: increased safety (particularly by the use of passive safety systems were possible), better economics and sustainability, increased proliferation resistance.

The most interesting prospect for Canada in the short to medium term appears to be the Terrestrial Energy initiative for an integral-core MSR (iMSR). Their concept is approximately that described below:

Table A-3
Characterisic of the iMSR – an MSR Proposed by Terrestrial Energy

Characteristic	Value
Molten salt	FLiBe, or $Na^+F^- + Rb^+F^-$
Fuel	$^{enr}UF_4$, PuF_3 or Th
Power	80, 300 and 600 MW_{th} 32.5 MWe, 141 Mwe
Outlet temperature	700 °C
Electricity Production Efficiency	50%
Conversion Ratio[49]	0.6 – 0.9
Secondary heat removal and power production	Steam, or Gas using a Brayton Cycle, or Molten salt
Tertiary heat storage	'Solar Salt' 60% $Na^+NO_3^-$ 40% $K^+NO_3^-$
Spectrum	Thermal
Moderator	Graphite

The larger size of the reactor means that for isolated, off-grid locations its use would be limited to all but a few with the biggest demand (Whitehorse, Iqaluit, large mining or oilsands extraction operations). The reactor would be more usefully used in multiple units as a replacement for conventional nuclear power plants.

The power coefficient of reactivity is estimated to be negative, but small. This allows quick shutdown of the reactor, but does not slow down the power rise much in accident-transient conditions.

The sodium-rubidium salt would be used if required to eliminate tritium production. It would also have the effect of lowering the conversion ratio (see footnote), though, towards the lower end of the 0.6 – 0.9 range.

The reactor is designed to be a net-burner of fissile material and not to have online processing of the salt to remove fission products[50]. The initial fissile would be in the range of 1.5 to 4% and

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[49] The Conversion Ratio is the rate of production of fissile nuclides divided by the rate of their destruction (including capture without fission). The conversion ratio can vary throughout burnup, but in a continuously refuelled case such as here it can be regarded as essentially a fixed feature of the reactor, and 1/(1-CR) as an energy multiplier. A makeup rate of R atoms/unit time results in a fission rate of R/(1-CR) atoms per unit time. A conversion ratio > 1 is called a Breeding Ratio.

[50] One exception would likely be that Noble gases, in particular Xe-135, may be allowed to come out of the salt. This would improve the fuel performance of the reactor.

would be regularly topped up with the addition of fresh fissile material, probably at an enrichment of 19.9%. The concept calls for the reactor to be run for seven years, after which the entire core would be replaced and the original core would be put into storage on-site. This fairly short period of core operation should eliminate most corrosion issues. The existence of a third molten salt loop allows for power production to be decreased easily for load following. Fairly continuous online refueling would reduce the need for reactivity hold-down devices and therefore would improve the sustainability of the fuel cycle.

At the higher Conversion Ratio of 0.9, the iMSR is projected to use ~1/6th the uranium of an LWR (per unit of power produced). A natural uranium fuelled CANDU has a conversion ratio that is only slightly less, about 0.85, but the CANDU does not have sufficient excess reactivity to achieve the fuel utilization of the iMSR and the MSR U-235 to power ratio is about ¼ that of the CANDU.

Since the fertile part of the fuel is in the reactor for many years, the reactor should score well on ratings for proliferation resistance. If the fuel is plutonium or uranium, plutonium bred into the fuel would be mostly the non-fissile isotopes Pu-242 and Pu-244, significantly denaturing the plutonium content for weapons purposes[51].

A.6.2   Further Research Required

Some research is required into the fission product retention of the molten salts. It is expected that the most problematic fission products for release – isotopes of Cesium and Iodine – would be retained in the fluoride salts but this must be confirmed. Another issue is tritium production in the fuel/coolant. If a high conversion ratio is desired (allowing more power to be produced per plutonium atom available in CANDU UNF) then FLiBe salt would be called for and this produces tritium. Hot tritium mixed in with the fuel has a significant escape probability from the reactor.

A.6.3   Prospects for Reburning CANDU UNF in the TE iMSR

At high conversion ratios, an MSR produces power with considerably less U-235 input than a CANDU, although the fuel costs of an MSR would include enrichment. If run as a plutonium burner instead, using the plutonium from CANDU UNF, the implication is that a given quantity of power could be produced with much smaller plutonium requirements. At the lower conversion ratios the reactor sustainability would be similar[52] to what would be expected in a CANDU.

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[51] The IAEA counts all plutonium as equal for proliferation purposes, unless the primary component is Pu-238, since even a dirty or 'fizzle' nuclear bomb would be devastating. However, in other proliferation assessments the existence of Pu-242 and Pu-244 would be a strong mitigating factor in the desirability of the plutonium in UNF for these purposes.

[52] At a CR of 0.6, the MSR fuel utilization is ¼ of what it is at 0.9 and the fuel utilization is therefore similar to that of a CANDU.

A long residence time implies that long-lived plutonium isotopes and minor actinides in the original fuel loading would be efficiently dispositioned. The presence of fertile material in this original load would result in these isotopes building in to the fuel, probably to a higher value than CANDU. However, since little fertile material is added during top-ups, the total quantity of troublesome minor actinides produced per unit energy in the MSR UNF would be significantly lower than in a solid fuel reactor. Exactly how much lower would require more details to be available on the core design so that an isotopic characterization could be made of the MSR final fuel loading.

Appendix B

Letter from CNSC

**Directorate of Regulatory Improvement and
Major Projects Management**

E-Docs# 4894852 / 2.01

December 17, 2015

Bhaskar Sur
Director, Nuclear Science Division
Canadian Nuclear Laboratories
Chalk River Laboratories
286 Plant Road
Chalk River, Ontario
K0J 1J0

Subject:	**Request to CNSC for assistance with information for CNL Feasibility Study on the Recycling of Used CANDU Fuel**
Reference:	**Letter from Bhaskar Sur to Haidy Tadros (Oct 27, 2015 E-Docs# 4900977)**

Dear Mr. Sur:

Canadian Nuclear Safety Commission (CNSC) staff has reviewed your request for CNSC assistance to provide information for your Feasibility Study on the Recycling of Used CANDU Fuel.

Per the criteria listed in your letter, CNSC as the Canadian nuclear regulator can address the provision of information as follows:

Anticipated challenges to licensing a used fuel recycling facility and a fast reactor in Canada

- FOAK (First of a kind) designs or systems represent greater risks from a safety analysis perspective. In particular, options combining FOAK designs and systems can have more complex and cumulative risks to safety.

- CNSC must ensure that appropriate oversight is developed for each of the progressive stages of licensing and compliance programs through to Operation. While licensing and compliance programs have been developed by the CNSC for new build reactors, they would need to be developed for a used fuel recycling facility. The precise combination and extent of regulatory activities would be documented in project specific licensing and compliance plans once the technology and principal engineering, procurement and construction company has been identified.

- Long lead activities such as obtaining a simulator, performing training and certification of authorized nuclear operators and fuel qualification are also anticipated challenges.

Suitability of the system for construction and operation at an Ontario site

Site suitability can only be assessed during an Environmental Assessment (EA) and the assessment of the application for a licence to prepare a site. Please refer to the following regulatory documents:

- RD-346: Site Evaluation for New Nuclear Power Plants - Canadian Nuclear Safety Commission

- REGDOC-2.5.2, Design of Reactor Facilities: Nuclear Power Plants - Canadian Nuclear Safety Commission

- RD-367: Design of Small Reactor Facilities - Canadian Nuclear Safety Commission

The licensing phases for a fast reactor facility, and most likely for the recycling facility as well, as this is a complex facility and a FOAK in Canada, would be:

- Licence to Prepare Site

- Licence to Construct

- Licence to Operate

Identification of the timeline for deployment of the system in Ontario

CNSC can provide information regarding estimated timelines for licensing reviews only. These timelines are highly dependent upon CNSC receiving full and complete licensing submissions. Licensing timeline estimates do not include time required for applicants to respond to requests for additional required information, external issues related to Environmental Assessments etc. For further information regarding timelines and licensing process please refer to REGDOC 3.5.1 Licensing Process for Class 1 Nuclear Facilities and Uranium Mines and Mills.

Given this context, CNSC would estimate the following timelines and costs for an EA and a Licence to prepare site and construct review, including Commission hearings, for the recycling systems per the table below:

System	Licensing Strategy and Estimated Duration per Licence	Estimated Licensing Cost
Licensing of a used fuel recycling facility (Class 1A Nuclear Facility) *Fuel to be used in existing CANDU fleet* of PHWRs (Pressurized Heavy Water Reactors)***	- Concurrent EA and LTPS - 24 months - LTC -32 months - LTO - 24 months	$115M
Licensing of a used fuel recycling facility (Class 1A Nuclear Facility) AND a Fast Reactor Facility (Class 1A Nuclear Facility).	<u>Recycling Facility Licensing</u> - Concurrent EA and LTPS - 24 months - LTC -32 months - LTO - 24 months <u>Fast Reactor Licensing</u> - Concurrent EA and LTPS - 24 months - LTC -32 months - LTO - 24 months	$230M

*Additional regulatory work would be required by existing plants towards fuel qualification.

**Please note the terminology used in your letter referred to HWRs. CNSC is assuming you are referring to the existing fleet of CANDU PHWR's (Pressurized Heavy Water Reactors).

Assumptions for Estimated Licensing Costs

The estimated costs stated in the table above refer to the complete cost to CNSC for both licensing and compliance programs such as compliance under the LTPS and Construction Licence. Taking into account the time to construct the facility, the time to obtain a licence to operate is estimated to be 9 years.

Lifecycle System Costs

The expected yearly regulatory fees are approximately $9M - $11M per year for either an operating used fuel recycling facility or an operating FOAK (first of a kind) reactor technology. This estimate is based on regulatory oversight program fees for facilities of comparable scale and complexity.

CNSC trusts that this information addresses your requests for information in support of your assessment of potential recycling systems in Canada. Should you have any further questions, please direct them to Dr. Doug Miller, Director, New Major Facility Licensing Division. Dr. Miller can be reached at (613)-943-0089.

Yours sincerely,

H. Tadros
Director General
Directorate of Regulatory Improvement and Major Projects Management

b.c.c.:   D. Newland, B. Howden, G. Frappier, P. Thompson, R .Awad, K. Heppel-Masys, D. Miller,
L. Andrews

Appendix C

GIF-PRPP Assessment Methodology

NNL's inherent PRPP assessment method, referred to as the "U-A methodology", involves calculating a utility function U(x), which represents the potential usefulness of nuclear material to a proliferator. The utility function depends on multiple attributes, represented by the vector x. The utility function is broken down into a value function V(x) that takes account of the mass and isotopic composition of the nuclear material and an access function A(x) that accounts for the difficulty of accessing the nuclear material, either through intrinsic barriers (such as a radiation field) or extrinsic barriers (such as safeguards and security measures).

$$U(\underline{x}) = V(\underline{x}). \, A(\underline{x})$$

Since it is usually the access function that exerts the most control over the utility function, it is helpful to plot U(x) on the y-axis against A(x) on the x-axis, to produce the U-A plot that gives the method its name. The favoured approach for evaluating state sponsored proliferation is to use the categorisation schemes recommended by the GIF-Proliferation Resistance and Physical Protection (GIF-PRPP) [87] methodology to determine the value and access functions.

The Value function uses the nuclear materials type (MT) classification adopted by GIF-PRPP, while the Access function amalgamates the four other GIF-PRPP classifications of Technical Difficulty (TD), Proliferation Cost (PC), Proliferation Time (PT) and Detection Probability (DP).

C.1   GIF-PRPP Methodology for Assessing Proliferation Resistance

The GIF proliferation resistance assessment methodology is based on a set of six metrics (or measures) each of which assigns a nuclear system into one of five categories. The six metrics are:

1. Fissile Material Type (MT): Accounts for the inherent value of the fissile material used in the proliferation pathway. There are five categories: Very Low; Low; Medium; High and Very High, corresponding to High Enriched Uranium (HEU); Weapons-Grade Plutonium (WG-Pu); Reactor-Grade Plutonium (RG-Pu); Deep Burn Plutonium (DB-Pu) and Low Enriched Uranium (LEU). For WG-Pu a nominal value of 94 w/o fissile fraction is assigned, for RG-Pu 70 w/o fissile and for DB-Pu 43 w/o.

2. Technical Difficulty (TD): This is essentially the risk of a proliferation pathway failing from the inherent technical diffculties. There are five categories: Very Low; Low; Medium; High and Very High.

3. Proliferation Cost (PC): This is the cost of implementing a proliferation pathway, arbitrarily set to a fraction of a state's military resource budget. There are five categories: Very Low; Low; Medium; High and Very High, measured as a fraction of the country's total annual defence budget. The five categories equate to 0-5% 5-25%, 25-75%, 75-100% and >100% of the defence budget respectively.

4. Proliferation Time (PT): The time needed to implement the proliferation pathway. There are five categories: Very Low; Low; Medium; High and Very High. Very Low corresponds to 0-3 months; Low 3 months to 1 year; Medium 1-10 years; High 10-30 years and Very High >30 years.

5. Detection Probability (DP): The probability of a clandestine proliferation pathway being discovered by routine safeguards inspections or other surveillance activities. There are five categories: Very Low; Low; Medium; High and Very High

6. Detection Resource Efficiency (DE): This is the level of resource required to implement EURATOM or IAEA safeguards. There are five categories: Very Low; Low; Medium; High and Very High.

These metrics are summarised in Table C-1 and it is seen that the proliferation risk decreases as the metric increases. GIF-PRPP methodology is self-explanatory and easy to implement within the framework of NNL's proliferation resistance assessment methodology.

Table C-1
Description of Metrics

Metric	Scoring					Criteria
MT	HEU	WG-Pu	RG-Pu	DB-Pu	LEU	Grade of nuclear material
TD	Very low	Low	Medium	High	Very High	Risk of proliferation pathway failing
PC	0-5	5-25	25-75	75-100	>100	% national defence budget
PT	0-3 months	3 months-1 year	1-10 years	10-30 years	>30 years	Time needed to implement pathway
DP	Very low	Low	Medium	High	Very High	Probability of being discovered
DE	Very low	Low	Medium	High	Very High	Safeguards resources required to ensure compliance

C.2   GIF-PRPP Value Function

Using the GIF-PRPP model, the Materials Type (MT) classification categories are: HEU, WG-Pu; RG-Pu, DB-Pu and LEU. These types need to be assigned different numerical values and the

approach adopted was the same as that described below for the Access function, setting the value function to:

$$V(P) = q^{P1}$$

Where q is an arbitrary classification parameter set to 1/√10 = 0.31623 and P1 is an exponent that depends on the GIF-PRPP. For the five GIF-PRPP categories, P1 is set to 1, 2, 3, 4 and 5 respectively. Therefore, HEU is assigned the highest value of V and the subsequent categories are assigned progressively lower values. The value 1/√10 chosen is arbitrary and only affects the y-axis scale. Setting q = 1/√10 ensures the numerical scales of the U-A plot are convenient. The arbitrary choice of q is made to facilitate the graphical output and implies that the numerical scales on the plots have no physical meaning and the plots are presented without numerical values for this reason.

C.3   GIF-PRPP Access Function

The same approach is applied to the Access function for four of the five remaining metrics: Technical Difficulty (TD); Proliferation Cost (PC); Proliferation Time (PT) and Detection Probability (DP). The detection resource efficiency (DE) is omitted because it relates only to the resource commitment required from EURATOM or IAEA, so is not relevant here as it is not a discriminating variable.

For each of these four metrics, an Access function is assigned according to:

$$A(TD) = q^{P2}$$
$$A(PC) = q^{P3}$$
$$A(PT) = q^{P4}$$
$$A(DP) = q^{P5}$$

Where the exponents P2 to P5 are assigned by the respective classification categories. The q parameter is the same as that used to define the value function. The overall Access function is then the product of the individual access functions:

$$A = A(TD).A(PC).A(PT).A(DP)$$

C.4   Implementation

All the Value and Access function models described have been implemented in Visual Basic functions embedded in a Microsoft®-Office EXCEL spreadsheet. This allows the calculations to be controlled very closely and facilitates checking, while the EXCEL environment allows easy modification of the inputs and visualisation of the outputs.

Appendix D

Options to Be Analyzed

It should be noted that due to limited publically available information, LWR UNF data was used as a proxy for much of the CANDU UNF data; particularly for the SFR systems. The assumption that the fast reactor plutonium is of LWR origin increases the amount of total plutonium required in a fast reactor, since the fissile content is lower than that of CANDU, and also increases the predicted amount of minor actinides produced in the UNF, since these are mostly produced by neutron absorption on plutonium isotopes. Detailed physics models of the specific fuel cycles using CANDU-NU UNF would be required to remove this assumption.

D.1   Option 1: Recycling in CANDU Reactors

In Option 1 (Figure D-1), CANDU UNF is reprocessed in two Co-extraction (COEX) facilities to be based at both the Bruce and Darlington reactor site.

COEX (Figure 4-1) [5] is an evolution of the existing PUREX process. It co-extracts some U with Pu to form a combined product. This increases the proliferation resistance of the Pu product by increasing the effort required to obtain pure Pu, thus decreasing the attractiveness of the material for use in weapons [5]. The U/Pu stream then undergoes finishing and thermal denitration to produce a homogenous $(U,Pu)O_2$ powder. The bulk of the uranium is processed into a secondary uranium product (a result of the large ratio of U to Pu in UNF). For the purpose of this investigation the required quantity of uranium would be routed to MOX production, the remainder would be converted to U_3O_8 and disposed of in a DGR.

The $(U, Pu)O_2$ powder is used to produce a MOX fuel by the MELOX fabrication process. The ratio of U:Pu in the reprocessed product requires down-blending with additional uranium before MOX production, this occurs at the MOX fabrication facility. MELOX is a fully-developed proprietary process owned by Areva and has been used to produce over 1700tHM of MOX fuel to date [9]. The MOX fuel is transported to the reactor sites and stored prior to its use in the reactors. The refurbished reactors are assumed to be capable of operating with a full MOX core.

Figure D-1 Option 1 Process

D.1.1 Option 1 Assumptions

The following tables detail the assumptions, scenario definition, timeline, waste produced and facilities required for Option 1.

Table D-1
Option 1 Reactor and Fuel Parameters

Parameter	Units	Value
CANDU Thermal Output	MWth	2,737
Reactor Power	MWe	818
CANDU Irradiation Start Date	year	2046
CANDU Reactor Life	Years	30
CANDU Pu proportion	%HM Pu	1.1
CANDU Reactor Core Mass	tHM	98
CANDU Annual Fuel Requirement	tHM/reactor/year	50.964
Reprocessing & Fabrication lead times	years	2

D.1.2 Option 1 Scenario Definition

Table D-2
Option 1 Definition

Parameter	Units	First 12 Reactors	2nd Five Reactors
Dates	Year	2046-2095	2081-2115
CANDU UNF Reprocessed	tHM	101,464	
Electricity Generated	TWh	2,193	786
MOX from CANDU UNF to be Disposed	tHM	35,693	

D.1.2.1 Option 1 Waste Produced

Table D-3
Option 1 Waste Produced

Parameter	Waste type	Units	Value
CANDU UNF for DGR		tHM	170
MOX UNF from CANDUs for DGR		tHM	35,523
HLW Reprocessing Waste		flasks	132
		canisters	3,702[53]
ILW	Reactor Solid	m^3	3,675
	Reactor Decommissioning	m^3	4,636
	Fabrication	m^3	119,002
	Reprocessing	m^3	44,217
	Facilities Decommissioning	m^3	5,850
LLW Reprocessing Waste	Reactor Solid	m^3	174,563
	Reactor Decommissioning	m^3	104,318
	Fabrication	m^3	45,114
	Reprocessing	m^3	96,660
	Facilities Decommissioning	m^3	67,500

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[53] Each canister has a volume of 150L

D.1.2.2   Option 1 Facilities Required

Table D-4
Option 1 Facilities Required

Parameter	Product type	Units	Value
Additional CANDU Refurbishments		Count	17
CANDU UNF Reprocessing Facility		Count	6
		Capacity tHM/yr	700
MOX for CANDUs Fabrication Facility		Count	9
		Capacity tHM/yr	180
Storage facilities	Interim UNF	tHM	35,523
	Vitrified Product	m³	556
	Pu	tHM	390
	U	tHM	66,332
UNF Encapsulation Facility	UNF	kgHM/yr	540

D.2   Option 2: Sodium Cooled Fast Reactor with Oxide Fuel

The Option 2 systems consider the recycle of CANDU UNF as MOX fast reactor fuel for use in Sodium Cooled Fast Reactors (SFRs), a generation IV reactor technology currently under development (originally expected to become deployable by 2030) [10]. The CANDU UNF is to be removed from storage throughout Canada and transported to a reprocessing plant operating an advanced aqueous recycling technology process. The products of reprocessing would then be fabricated into SFR MOX fuel through the use of MELOX based technology.

The fast reactors, reprocessing facilities, fuel fabrication plant and all accompanying facilities required to achieve the scenarios are assumed to be new-builds and located at existing reactor sites in Ontario (Bruce and Darlington). For the purpose of this investigation front-end reprocessing, fuel fabrication, reactor operations and back-end reprocessing (where applicable) are assumed to be located together on a single site to simplify issues related to the transportation of material.

Option 2 was split into three separate options each focussing on a different number of reactors, fuel cycle mode or UNF management strategy. The three options aim to demonstrate the effects and potential benefits of closing the fuel cycle on operation time scales, waste profiles and volumes and utilization of the CANDU UNF inventory. These are discussed in more detail later in this section.

Sodium cooled fast reactors (SFRs) [88], are part of the generation IV reactor initiative centred on deploying actinide management in fast reactor technology by 2030 at the earliest. Being a fast reactor, the SFR operates with a fast neutron energy spectrum which extends uranium reserves and burns transuranic material to a greater extent compared with thermal reactors [88]. As a result of a reduced actinide content (assuming recycling occurs), the radiotoxicity and heat load in wastes would decrease more quickly. The SFR utilises liquid sodium as a primary coolant to transfer fission heat from the fuel to the medium which drives the generator turbines (usually superheated steam). The system must be completely sealed and free of air and moisture due to the high chemical reactivity of sodium in these environments, providing rigorous engineering requirements, but also helping to prevent corrosion. The sodium coolant provides a high power density per volume of coolant and has a temperature of approximately 500°C when exiting the reactor [88].

The manufacture of fast reactor MOX fuel requires the same basic processing steps as most other ceramic oxide fuels such as those used in LWR's. Typically, the oxide powder products from reprocessing operations and storage would be blended and mixed to the correct ratios depending on the fuel specification and fissile feed isotopic qualities. This powder is then pressed into a pellet slightly larger than the final pellet size required and sintered to bind the powder into a ceramic. The ceramic pellet then undergoes inspection and grinding processes to mill the pellet to the exact dimensions required [89]. The MELOX process [90] developed by AREVA is assumed to be able to accommodate the manufacture of fast reactor MOX through modifications to the existing plant design and would be the fabrication process referred to within Option 2.

The MOX required for fast reactors is generally of higher Pu content than thermal reactor fuel; reprocessing plants are designed to produce a MOX powder product of 30wt% Pu or higher. The MOX is then down-blended with U at the fabrication facility to the required levels. The SFR MOX required for the fast reactors discussed in Option 2 requires ≈30wt% Pu product. Due to the very high fissile quantity and the presence of Am and Cm, significant (and expensive) changes to the MOX fabrication facility would be required (e.g., smaller criticality safe vessels (blenders, granulators) and additional shielding and/or remote operations during mechanical fabrication).

D.2.1   Option 2 Assumptions

This scenario assumes a scaled up ASTRID like core is used to irradiate TRU material obtained from reprocessing a proportion of the CANDU UO_2 UNF stockpile. Input parameters required for the SFR model were obtained from a model for an oxide fuelled 380 MW(e) PRISM core due to the lack of detailed mass flows for a fuel cycle involving the ASTRID like SFR. Parameters from the PRISM model have been scaled such that the total net electrical output is 1500 MW(e).

A conversion ratio of 0.5 has been used since a burner configuration was required for this scenario and lower conversion ratios would require significant research and would possibly lead

to an unstable reactor core. Details for the MOX SFR core concept which have been used in scoping calculations are given in Table D-5.

The following tables detail the assumptions, scenario definition, timeline, waste produced and facilities required for Option 2.

Table D-5
Option 2 Parameters

Parameter	Units	Value
SFR Thermal Output	MWth	3950
Reactor Power	MWe	1500
SFR Start Date	year	2050
SFR Operating Life	Years	60
SFR Pu proportion	Wt% Pu/HM	33
Conversion Ratio		0.5
SFR Core Mass	tHM	43.03
SFR TRU feed at startup	tHM	14.24
SFR Annual Fuel Req	tHM/year	7.14
SFR TRU loaded per year	tHM/year	2.37
Reprocessing & Fabrication lead times	years	2

D.2.2 Option 2 Scenario Definition

Table D-6
Option 2 Definition

Parameter	Units	Option 2a	Option 2b	Option 2c
End Date	Year	2110	2110	2150
CANDU UNF Reprocessed	tHM	40,440	80,880	103,000
SFR UNF Reprocessed	tHM	0 (n/a)	0 (n/a)	2,256
Electricity Generated	TWh	671	1,341	4091
SFR UNF to be Disposed	tHM	464	929	645

D.2.2.1 Option 2 Waste Produced

Table D-7
Option 2 Waste Produced

Parameter	Waste Type	Units	Option 2a	Option 2b	Option 2c
CANDU UNF for DGR		tHM	62,560	22,120	0
SFR UNF for DGR		tHM	464	929	645
HLW Reprocessing Waste	CANDU UNF	canister	1,456	2,912	3,708
		flask	52	104	132
	SFRMOX UNF	canister	0	0	2,978
		flask	0	0	107
ILW Reprocessing Waste	Reactor Solid	m^3	300	600	1,800
	Reactor Decommissioning	m^3	273	545	2,182
	Fabrication	m^3	1,554	3,112	9,718
	Reprocessing	m^3	17,839	34,778	45,260
	Facilities Decommissioning	m^3	2,340	4680	7,800
LLW Reprocessing Waste	Reactor Solid	m^3	14,250	28,500	85,500
	Reactor Decommissioning	m^3	6,136	12,273	49,091
	Fabrication	m^3	589	1,180	3,684
	Reprocessing	m^3	38,014	76,027	98,941
	Facilities Decommissioning	m^3	27,000	54,000	90,000

D.2.2.2   Option 2 Facilities Required

Table D-8
Option 2 Facilities Required

Parameter	Material	Units	Option 2a	Option 2b	Option 2c
SFR built		count	1	2	8
CANDU UNF Reprocessing		Count	3	6	8
		Capacity tHM/yr	600	600 each	600
SFR MOX UNF Reprocessing		Count	0	0	2
		Capacity tHM/yr	0	0	50
SFR MOX Fabrication		Count	3	6	6
		Capacity tHM/yr	7	7	7
SFR MOX Fabrication		Count			4
		Capacity tHM/yr			21
Storage facilities	Interim	tHM	464	930	646
	Vitrified Product	m^3	219	438	556
	Pu/TRU	tHM	154	308	1,066
	U	tHM	39,592	79,184	100,976

D.3   Option 3: Sodium Cooled Fast Reactor with Metal Fuel

The Option 3 systems consider the recycle of CANDU UNF as metallic fast reactor fuel for use in small-modular Sodium Cooled Fast Reactors (SFRs), much like GE Hitachi's Power Reactor Innovative Small Module (PRISM) reactor concept [12]. The CANDU UNF is to be removed from storage throughout Canada and transported to a pyro-processing plant, which utilises molten salt and electro-refining technologies to reprocess the reusable material. The products of reprocessing would then be fabricated into metallic fast reactor fuel through an integrated metal ingot casting process incorporated into the pyro-processing plant. Several pyro-processing processes have been developed however the USA and Russia are currently the leaders in this type of reprocessing. The American electro-refining process developed by Argonne National Laboratory is considered in this option due to its compatibility [91] with the PRISM concept.

The fast reactors, reprocessing facilities, fuel fabrication plant and all accompanying facilities required to achieve the scenarios are assumed to be new-builds and located at existing reactor sites in Ontario (Bruce or Darlington). For the purpose of this investigation front-end reprocessing, fuel fabrication, reactor operations and back-end reprocessing (where applicable) are assumed to be located together on a single site to simplify issues related to the transportation of material.

Option 3 was split into three separate options each focussing on a different number of reactors, fuel cycle mode or UNF management strategy. The three options aim to demonstrate the effects and potential benefits of closing the fuel cycle on operation time scales, waste profiles and volumes and utilization of the CANDU UNF inventory. These are discussed later in this section.

The PRISM reactor is a GE Hitachi reactor concept that has evolved from the EBR-II (Argonne National Laboratory), Southwest Experimental Fast Oxide Reactor (SEFOR) (GE) and the Fast Flux Test Facility (FFTF) (US Department of Energy) experimental reactors [12]. The reactor is a pool type metal fuelled SFR with a 380MWe output. The reactor can be grouped into reactor islands comprising of two reactors feeding a single steam generator system. The PRISM relies on passive safety systems typical of generation IV reactor concepts such as an intermediate sodium loop (to provide another barrier between the radioactive primary cooling loop and the steam loop and minimize the potential of a water-sodium chemical reaction). The PRISM reactor concept also possesses a passive cooling system to prevent the core from overheating and melting in an emergency [14].

A PRISM-like SFR has been selected for Option 3 as it is at near-deployment readiness level.

The fabrication of metallic fast reactor fuel occurs in two stages: firstly the metal ingot is cast at the back end of the pyro-processing facility and then the ingot is processed into a final fuel assembly in a separate fabrication facility. The ingot needs to be geometrically accurate and undergoes milling and finishing operations before being encased in cladding. Metallic fast reactor fuel is recommended but not limited to small to medium sized reactors (600MWe and lower) [88].

D.3.1   Option 3 Assumptions

This scenario is similar to Scenario 2 in that a number of SFR units are operated in an open fuel cycle using TRU obtained from reprocessing CANDU UNF. However, instead of an 'ASTRID' like 1,500 MW(e) reactor, 4 smaller PRISM like units each producing 380 MW(e) has been assumed (1520 MW(e) total).

The parameters required to model the PRISM fuel cycle were obtained [92]. As mentioned previously, this report does assume a different fissile feed stream to what is available in these scenarios.

The PRISM units are assumed to have a typical conversion ratio of 0.5 and utilise Pu-U-Zr metallic fuel. According to Hoffman et al [92], lower conversion ratios are possible. However

this is at the expense of significantly lower ^{238}U contents in the fuel leading to potentially undesirable core reactivity coefficients and kinetic behaviour, as well as unfavourable fuel performance characteristics due to a very high fissile fraction. Details for the S-PRISM core concept (which have been used in a model to predict mass flows) are given in Table D-9.

The following tables detail the assumptions, scenario definition, timeline, waste produced and facilities required for Option 3.

Table D-9
Option 3 Parameters

Parameter	Units	Value
SFR Thermal Output	MWth	4,000
Reactor Power (4 units)	MWe	1,520
SFR Start Date	Year	2050
SFR Operating Life	Years	60
SFR Pu proportion	wt% Pu/HM	29%
Conversion Ratio		0.5
Metallic SFR Core Mass	tHM	37.8
Metallic SFR TRU feed at startup	tHM	10.8
Metallic SFR Annual Fuel Req	tHM/year	9.12
Metallic SFR TRU per year	tHM/year	2.68
Reprocessing & Fabrication lead times	years	2

D.3.2   Option 3 Scenario Definition

Table D-10
Option 3 Definition

Parameter	Units	Option 3a	Option 3b	Option 3c
End Date	year	2110	2110	2155
CANDU UNF Reprocessed	tHM	44,515	89,030	103,000
Metallic SFR UNF Reprocessed	tHM	0 (n/a)	0 (n/a)	3,829
Electricity Generated	TWh	680	1,360	5,100
Metallic SFR UNF to be Disposed	tHM	576	1,151	542

D.3.2.1 Option 3 Waste Produced

Table D-11
Option 3 Waste Produced

Parameter	Waste Type	Units	Option 3a	Option 3b	Option 3c
CANDU UNF for DGR		tHM	58,485	13,970	0
SFR UNF for DGR		tHM	576	1151	542
HLW Reprocessing Waste	CANDU UNF	canister	1,603	3,205	3,708
		flask	57	114	953
	Metallic SFR UNF	canister	0	0	22,974
		flask	0	0	821
ILW Reprocessing Waste	Reactor Solid	m^3	1,200	2,400	9,600
	Reactor Decommissioning	m^3	1,091	2,182	10,909
	Fabrication	m^3	1,930	3,856	14,643
	Reprocessing	m^3	19,141	38,283	45,936
	Facilities Decommissioning	m^3	2,340	4680	7020
LLW Reprocessing Waste	Reactor Solid	m^3	57,000	114,000	456,000
	Reactor Decommissioning	m^3	24,545	49,091	245,455
	Fabrication	m^3	732	1,462	5,551
	Reprocessing	m^3	41,844	83,688	100,419
	Facilities Decommissioning	m^3	27,000	54,000	81,000

D.3.2.2 Option 3 Facilities Required

Table D-12
Option 3 Facilities Required

Parameter	Waste type	Units	Option 3a	Option 3b	Option 3c
Metallic SFR built		count	4	8	40
CANDU UNF Reprocessing		Count	3	6	6
		Capacity tHM/yr	660	660	675
Metallic SFR UNF Reprocessing		Count	0	0	2
		Capacity tHM/yr	0	0	80
Metallic SFR Fabrication		Count	3	6	6, 2[54]
		Capacity tHM/yr	9	9	9, 70[55]
Storage facilities	Interim	tHM	576	1152	542
	Vitrification	m^3	240	480	552
	Pu/TRU	tHM	169	338	1538
	U	tHM	43,517	87,034	100,946

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[54] Six facilities fabricate fuel from CANDU UNF and two facilities fabricate fuel from SFR UNF

[55] Facilities to fabricate fuel from CANDU UNF is 9 tHM/yr and from SFR UNF is 70tHM/yr

Appendix E
Decay Power Calculations

E.1   Decay Power of UNF

Estimating the decay power of UNF requires knowledge of its composition, especially the composition of TRUs and FPs. Since there is no publicly available data for the resulting SFR UNF composition when recycling TRUs from CANDU-NU UNF into a SFR, bounding estimates were used in this analysis.

- Lower Bound:
    - The lower bound estimate is based on the relative change in decay power of the TRUs due to irradiation in a SFR. The decay power of TRUs from SFRs fuelled by recycled LWR TRUs and SFRs fuelled by weapons plutonium are provided 50 years after irradiation (shown in Table E-1).
    - The percent changes in decay power were used to estimate the lower bound for the decay power of CANDU-NU TRUs after irradiation in a SFR by multiplying their initial decay power by the percent change for each case for LWR TRUs. This is considered a lower bound due to the decay power of CANDU-NU TRUs being closer to that of weapons plutonium than LWR TRUs with the percent change in decay power for CANDU-NU TRUs likely being closer to that of weapons plutonium.

- Upper Bound:
    - The upper bound for the decay power of CANDU-NU TRUs after irradiation in a SFR is equal to the decay power of LWR TRUs after irradiation in a SFR. This estimate is considered an upper bound due to the decay power of LWR TRUs being well over 1.8 times that of CANDU-NU TRUs, therefore it is expected that the irradiation of CANDU-NU TRUs in a SFR would not increase their decay power by more than the increase in decay power of weapons plutonium shown in Table Table E-1.

Table E-1
The Relative Increase in TRU Decay Power Due to Irradiation in a SFR

TRU Source	Irradiation	Decay Power (W/kgTRU)	Change in Decay Power as a Result of Recycling into an SFR (Fractional Change)
Weapons plutonium [93]	None	2.6	
	1st pass Metal SFR	4.7	$= \dfrac{4.7}{2.6} = 1.81$
LWR [92]	None	34.9	
	1st pass MOX SFR	37.0	1.06
	Eq.[56] MOX SFR	41.2	1.18
	1st pass Metal SFR	34.3	0.98
	Eq. Metal SFR	38.1	1.09

The estimated decay powers for UNF from reactors in each of the SFR options are summarized in Table E-2.

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[56] Equilibrium.

Table E-2
The Decay Power of UNF for Each Option

Waste Type		Ref	Option 1	Option 2		Option 3	
	Recycle Fuel Irradiation	None	1st pass MOX CANDU	1st pass MOX SFR	Eq. MOX SFR	1st pass Metal SFR	Eq. Metal SFR
Fresh Fuel	**TRU (wt. %)**	-	1.1%	33.1%	37.8%	29.4%	33.3%
	TRU Decay Power (W/kgTRU)	-	8.7	8.7	8.7	8.7	8.7
UNF	**TRU (wt. %)**	< 1.0%	0.9%	25.1%	29.1%	23.2%	26.6%
	TRU Decay Power (W/kgTRU)	8.7	17.9	LB: 9.2 UB: 37.0	LB: 10.2 UB: 41.2	LB: 8.5 UB: 34.3	LB: 9.5 UB: 38.1
	TRU Contribution to Decay Power (W/kgUNF)	< 0.1	0.2	LB: 2.3 UB: 9.3	LB: 3.0 UB: 12.0	LB: 2.0 UB: 7.9	LB: 2.5 UB: 10.1
	FP Contribution Decay Power (W/kgUNF)	0.1	0.1	1.2	1.0	1.2	1.0
	Total Decay Power (W/kgUNF)	0.1	0.3	LB: 3.5 UB: 10.5	LB: 4.2 UB: 13.2	LB: 2.9 UB: 8.9	LB: 3.5 UB: 11.1

Since each option results in different volumes of CANDU-NU UNF and UNF from advanced reactors, the total decay power of UNF to be placed in the DGR is the total decay power per unit mass of UNF in Table E-2 multiplied by the mass of the corresponding UNF. Using the volumes of wastes produced documented in Appendix D, the total decay power was calculated and recorded in Table E-3 below. The decay powers shown in Table E-2 and Table E-3 were estimated using data from the VISION fuel cycle simulation [94].

Table E-3
The Total Mass and Decay Power of UNF in Each Option

Option	Recycled Fuel Irradiation	Total Volume of UNF for DGR (kgUNF)		Total Decay Power (MW)	
		CANDU NU UNF	Advanced Reactor UNF	Lower Bound	Upper Bound
	none	103,000	0	18.5	
Option 1	1 pass MOX CANDU	107	35,523	10.3	10.3
Option 2a	1 pass MOX SFR	62,560	464	12.9	16.1
Option 2b	1 pass MOX SFR	22,120	929	7.2	13.7
Option 2c	Eq. MOX SFR	0	645	2.7	7.1
Option 3a	1 pass Metal SFR	58,485	576	12.2	15.7
Option 3b	1 pass Metal SFR	13,970	1,151	5.9	12.8
Option 3c	Eq. Metal SFR	0	542	1.9	5.0

The total decay power of UNF is calculated using the following assumptions:

- Any CANDU UNF that is placed in a DGR has decayed for 30 years prior to emplacement.

- SFR UNF that is placed in a DGR has decayed for 50 years prior to emplacement.

E.2   Decay Power of RepHLW

The largest contributors to the decay power of the RepHLW stream are the FPs and TRUs, the proportion of which depends on the reprocessing technology used. The decay power of the RepHLW streams for each option is shown in Table E-4.

Table E-4
The Mass and Decay Power of RepHLW for Each Option

Option	Total Decay Power of RepHLW for DGR (MW)	
	CANDU-NU[57]	SFR[58]
Reference	-	-
Option 1	9.3	-
Option 2a	3.6	-
Option 2b	7.3	-
Option 2c	9.3	2.6
Option 3a	4.0	
Option 3b	8.0	
Option 3c	9.3	4.6

The decay power of RepHLW from CANDU-NU UNF was calculated by multiplying the number of canisters by the canister thermal loading limit (2.5 kW/canister). The decay power of RepHLW from SFR UNF was calculated by multiplying the HM mass of the GANEX waste stream[59] in Options 2c and 3c by the unit mass decay power of the GANEX waste stream[60] after 50 years of decay, which are shown in Table E-5.

Table E-5
Data for Calculating the Decay Power of RepHLW from SFR UNF

Option	SFR UNF Reprocessed (tHM)	SFR UNF Composition at Reprocessing (wt. %)			Total RepHLW mass (tHM)	RepHLW Decay Power at emplacement (kW/tHM)
		Pu	U	FP		
2c	2256	25.7%	52.4%	18.0%	394	6.7
3c	3829	23.8%	58.8%	14.2%	669	6.9

The decay powers and UNF compositions shown in Table E-3 and Table E-5 were estimated using data from the VISION fuel cycle simulation [69].

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[57] RepHLW from CANDU-NU reprocessing is sent to the DGR immediately after vitrification.

[58] RepHLW from SFR UNF reprocessing is stored for 50 years prior to being sent to the DGR.

[59] 96% of FP, 0.1% of Pu, and 0.1% of U go to the RepHLW stream.

[60] Due to a lack of publicly available data on the pyro-process, the reprocessing efficiency for Option 3c is assumed to be the same as that of the GANEX process. This assumption is reasonable because the pyro-process would be considered technically acceptable if its efficiency rivals that of aqueous methods.

Appendix F

Glossary

Term	Definition
AGR	Advanced Gas-cooled Reactor
Am	Americium
AP	Activation Products
AP1000	An advanced light water reactor designed by Westinghouse.
APM	Adaptive Phased Management
Aqueous reprocessing	Reprocessing processes (e.g. PUREX) which involve dissolving the fuel in aqueous solutions such as nitric acid.
ASTRID	Advanced Sodium Technological Reactor for Industrial Demonstration
Bq	Becquerel, where a Becquerel is a measure of radioactivity
C	Carbon
CAD$	Canadian dollar
CANDU	CANada Deuterium Uranium Reactor, a pressure tube heavy water reactor of canadian design fuelled by natural uranium.
Capacity factor	Ratio of the actual power output of a reactor to its full potential output over a given period of time, i.e. 85% means that out of 100 days the plant can be expected to give 85 days' worth of full-power generation.
CAPEX	Capital Expenditure
CFR	Code of Federal Regulations
CL	Construction Licence
Cm	Curium
CNL	Canadian Nuclear Laboratories
CNSC	Canadian Nuclear Safety Commission
COEX	Co-extraction, A hydrometallurgical reprocessing technique similar to PUREX that produces a Pu+U stream and a U stream.
COL	Construction and Operating Licence
Conversion Ratio	The ratio of new fissile nuclei to fissioned nuclei in a reactor.
DGR	Deep Geological Repository, Underground facility for the long-term storage of nuclear waste.
DOE	U.S. Department of Energy
DU	Depleted Uranium

Term	Definition
EBR-II	Enhanced Boiling water Reactor –II
EFPD	Effective Full Power Days, Term used to detail the duration of fuel irradiation.
Fertile	Capable of producing fissile nuclides via neutron capture.
FFTF	Fast Flux Test Facility
FIAC	First-In-A-Country
Fissile	Capable of sustaining a fission chain reaction.
FOAK	First-Of-A-Kind
FP	Fission Product
FR	Fast Reactor: Nuclear reactors which operate at a high neutron energy spectrum (i.e., they contain no moderator).
FSAR	Final Safety Analysis Report
GANEX	Grouped Actinide Extraction, A hydrometallurgical reprocessing technique similar to PUREX that utilises novel solvents to produce a U product and a mixed Pu-MA product.
GBq	Gigabecquerel equal to 1×10^9 Becquerels
GDA	Generic Design Assessment
GE	General Electric
GENIV	Generation IV, An international initiative to develop 6 new fast reactor design concepts for deployment before 2030
GIF	Generation IV International Forum
GW	GigaWatts
GWd	GigaWatts day
He	Helium
HLW	High Level Waste - UNF when consigned to the DGR or waste products containing the majority of the fission products after reprocessing.
HM	Heavy Metal
I	Iodine
IAEA	International Atomic Energy Agency
IFR	Integrated Fast Reactor
ILW	Intermediate Level Waste - Higher activity waste with a decay heat $<2kW/m^3$.
Iso-breeder	A reactor operating mode where the reactor breeds as much fuel as it consumes.
ITAAC	Inspections, Tests, Analysis and Acceptance
KAERI	Korea Atomic Energy Research Institute
Kr	Krypton

Term	Definition
kW	kilowatt
LKE salt	LiCl-KCl Eutectic salt
LLW	Low Level Waste - Waste with an activity range of 0.4 Bq/g to 10s of kBq/g
LTEP	Long-term energy plan
LWR	Light Water Reactor
m^3	meters cubed
MA	Minor Actinides
MADA	Multi-Attribute Decision Analysis
MAGNOX	Magnesium Non-Oxidising – A UK developed CO_2 cooled, graphite moderated, natural U fuelled reactor technology.
MBq	1×10^6 Bq
MELOX	Areva MOX fabrication plant technology.
Moderator	A material (e.g. heavy water) used to reduce the energy of neutrons released by fission to increase their probability of causing fission reactions.
MOX	Mixed Oxide Fuel, oxide nuclear fuel manufactured from a blend of heavy elements such as plutonium and uranium.
MSR	Molten Salt Reactor, A reactor whose fuel is a molten salt.
MWe	MegaWatt of electricity
MWh	MegaWatt hour
Na	Sodium
NEA	Nuclear Energy Agency
NFWA	Nuclear Fuel Waste Act
NIL	Nuclear Installation Licence
NISL	Nuclear Installation Site Licence
NNL	National Nuclear Laboratory
NNR	National Nuclear Regulator
NRC	U.S. Nuclear Regulatory Commission
NRCan	Natural Resources Canada
NU	Natural Uranium, unenriched uranium with a fissle content of 0.711%
NWMO	Nuclear Waste Management Organisation
O&M	Operations and Maintenance
OECD	Organisation for Economic Cooperation and Development
OPG	Ontario Power Generation
PCM	Plutonium Contaminated Material
PFR	Prototype Fast Reactor

Term	Definition
PHWR	Pressurized Heavy Water Reactor. A Canadian design that utilizes heavy water as the moderator and coolant material.
PRISM	Power Reactor Innovative Small Module
Proliferation	The dissemination of fissile material for non-civil purposes.
PRPP	Proliferation Resistance and Physical Protection
PSAR	Preliminary Safety Analysis Report
Pu	Plutonium
PUREX	Plutonium-Uranium Redox Extraction. A hydrometallurgical nuclear fuel reprocessing process that utilises tributyl phosphate based solvents to separate re-usable U and Pu from UNF.
Pyro-processing	A dry reprocessing technology that utilises high temperatures and electochemistry to dissolve UNF and separate out the reusable components.
Radiotoxicity	The toxicity of radioactive material
RepHLW	The high level waste from reprocessing UNF.
Reprocessing	The separation of reusable components of nuclear fuel from the fission products and cladding material.
RU	Reprocessed Uranium
RWM	Radioactive Waste Management Ltd.
SEFOR	Southwest Experimental Fast Oxide Reactor
SFR	Sodium-cooled Fast Reactor, Nuclear reactors that operate at a high neutron energy spectrum (i.e., they contain no moderator).
SFRMOX	Sodium-cooled fast reactor mixed oxide fuel
Shuffling	Moving nuclear fuel already utilised in the core to another position to maximise fuel efficiency.
SQ	Significant Quantity
T	Metric tonne
TBP	Tri-butyl phosphate
TBq	Terabequerel, equal to 1×10^{12} Bq
Thermal reactor	A nuclear reactor in which neutrons have their energy reduced by a moderator for increased neutron efficiency.
tHM	tonne of heavy metal
TRU	Transuranics, A group of elements with an atomic number greater than 92 (typically refers Pu, Np, Am and Cm).
TWh	TerraWatt Hours
U	Uranium
U.K.	United Kingdom
UK EPR	UK Enhanced Pressurized Reactor

Term	Definition
UK RWI	UK Radioactive Waste Inventory
UNF	Used Nuclear Fuel, Nuclear fuel removed from a reactor post-irradiation.
VHLW	Very High Level Waste - Waste which produces >2.5kW/m^3
Vitrification	A process in which nuclear materials are mixed with glass to immobilise them in a glass matrix.
wt. %	Percent by weight