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Land and Water Use, CO2 Emissions, and Worker Radiological Exposure Factors for the Nuclear Fuel Cycle

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Considerations Relevant to Fuel Cycle Technology Evaluation

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EXECUTIVE SUMMARY

The Department of Energy Office of Nuclear Energy's Fuel Cycle Technologies program is preparing to evaluate several proposed nuclear fuel cycle options to help guide and prioritize Fuel Cycle Technology research and development. Metrics are being developed to assess performance against nine evaluation criteria that will be used to assess relevant impacts resulting from all phases of the fuel cycle. This report focuses on four specific environmental metrics.

- land use
- water use
- CO_2 emissions
- radiological Dose to workers

Impacts associated with the processes in the front-end of the nuclear fuel cycle, mining through enrichment and deconversion of DUF_6 are summarized from FCRD-FCO-2012-000124, Revision 1. Impact estimates are developed within this report for the remaining phases of the nuclear fuel cycle. These phases include fuel fabrication, reactor construction and operations, fuel reprocessing, and storage, transport, and disposal of associated used fuel and radioactive wastes.

Impact estimates for each of the phases of the nuclear fuel cycle are given as impact factors normalized per unit process throughput or output. These impact factors can then be re-scaled against the appropriate mass flows to provide estimates for a wide range of potential fuel cycles. A companion report, FCRD-FCO-2013-000213, applies the impact factors to estimate and provide a comparative evaluation of 40 fuel cycles under consideration relative to these four environmental metrics.

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ACRONYMS

ADS	Accelerator driven system
BWR	Boiling water reactor
DOE-NE	United States Department of Energy Office of Nuclear Energy
DU	Depleted uranium
EIA	United States Energy Information Administration
EIS	Environmental Impact Statement
EPA	United States Environmental Protection Agency
FCRD	Fuel Cycle Research and Development
FEFC	Front-End of the Nuclear Fuel Cycle
FFH	Fission-Fusion Hybrid
GJ	Giga Joules(10^9 Joules)
GW	GigaWatt (10^9 Watts)
GWe	GOgaWatt electric
GWh	GigaWatt hours
GTTC	Greater-than-Class-C radioactive waste
HF	Hydrofluoric acid
HLW	High level radioactive waste
HWR	Heavy water reactor
HTGR	High-temperature gas reactor
ISOM	Inspection, Surveillance, Operations, and Maintenance
J	Joules (energy unit)
kSWU	Kilo-Separative Work Unit (1,000 SWU)
L	Liters
LEU	Low-enriched uranium
LLW	Low level waste
LWR	Light water reactor
MOX	Mixed oxide fuel
mSv	MilliSieverts (10^{-3} Sieverts)
MSR	Molten salt reactor
MTIHM	Metric tonnes of initial heavy metal
MTNU	Metric tonnes of natural uranium
MTTh	Metric tonnes of natural thorium

MW	MegaWatt (10^6 Watts)
MWh	MegaWatt hours
NFC	Nuclear Fuel Cycle
NRC	United States Nuclear Regulatory Commission
PWR	Pressurized water reactor
R	Roentgen
R&D	Research and Development
rad	Radiation absorbed dose (0.01 Gy, 0.01 J/kg)
rem	Roentgen equivalent man (10^{-2} Sieverts)
SFR	Sodium-cooled fast reactor
SNF	Spent Nuclear Fuel
STD	Storage, transport, and disposal
Sv	Sieverts (100 rem, 1 J/kg absorbed dose in tissue)
SWU	Separative Work Unit
TRISO	Tristructural isotropic
U-233	Uranium-233 (also presented as 233U, ^{233}U)
U-235	Uranium-235 (also presented as 235U, ^{235}U)
U-238	Uranium-238 (also presented as 238U, ^{238}U)
U_3O_8	Triuranium octoxide (“yellowcake”)
UF	Used fuel
UF_6	Uranium hexafluoride
UO_2	Uranium dioxide (“UOX”)
Th-232	Thorium-232 (also presented as 232Th, ^{232}Th)
YMP	Yucca Mountain Project
Yr	Year

1. OBJECTIVE AND APPROACH

The Department of Energy Office of Nuclear Energy (DOE-NE) is responsible for developing sustainable nuclear fuel cycles as described in the DOE-NE Research and Development Roadmap. This responsibility is implemented through the DOE-NE Fuel Cycle Research and Development (FCRD) Program with a mission to develop a suite of options that will enable future policy makers and stakeholders to make informed decisions about how best to benefit from nuclear technologies.

In fulfilling this responsibility the FCRD program is developing metrics for assessing overall systems performance of various proposed nuclear fuel cycles. The metrics will assess relevant impacts resulting from all phases of the fuel cycle and will help focus research and development (R&D) efforts and inform allocation of funding to nuclear technologies which will best meet technical objectives as well as other societal needs and concerns.

The objective of this effort is to estimate the impacts of the nuclear fuel cycle (NFC) for the following measures:

Land Use is defined to include land not available for other purposes as a result of the NFC process. This includes both temporary and permanent land occupied by facilities and within the exclusion area (i.e. inside the fence).

Note: Only the permanent land use component accrues. The non-permanent component of the land use reaches steady state where land is being decommissioned and placed back into circulation at the same rate that new land is being taken out of service for new facilities. Thus, inclusion of the temporary land use adds a measure of conservatism in the estimate that is correlated to the ratio of temporary vs. permanent land use.

Water Use is the net water used by the NFC process. This is defined as the total water withdrawals minus any water returned to its source at equal or better purity and within allowable temperature limits.

Note: Water conservation and/or recycling technologies are often commensurate with the local perceived value of this resource. Hence, water usage for a given technology often varies widely.

CO_2 Emissions include the CO_2 emissions associated with all direct and embodied energy consumed by the NFC process. CO_2 is the primary green-house gas and typically accounts for ~84% of U.S. greenhouse gas emissions from human activities[1]. CO_2 emissions are estimated based on both direct

[1] http://www.epa.gov/climatechange/ghgemissions/gases/co2.html

energy use as well as the energy embodied in process materials along with the associated mix of energy carriers.

Occupational Radiological Exposure is a measure of the collective worker dose resulting from the NFC process. It is obtained as the product of the expected average worker dose and the estimated number of workers.

Figure 1-1 is a block flow diagram showing the various phases of the NFC along with the candidate options that may be employed in each phase. Impact estimates, based on existing operational data and supplemented by engineering judgment when needed, are provided for each phase. Impact estimates for the processes in the front-end of the nuclear fuel cycle (FEFC) phase are summarized from [Carlsen 2013]. Estimates for the remaining phases of the NFC are developed herein. Estimated impacts are those associated with steady state process operations and, with the exception of reactor operations, did not include the construction or decommissioning impacts. The CO_2 impacts associated with reactor construction energy and materials were included because of the substantial embodied energy associated with construction materials and also because CO_2 emissions associated with reactor operations are considered negligible. The significance of neglecting construction and decommissioning energy associated with nuclear fuel cycle processes (except reactor construction) may be a subject for future work.

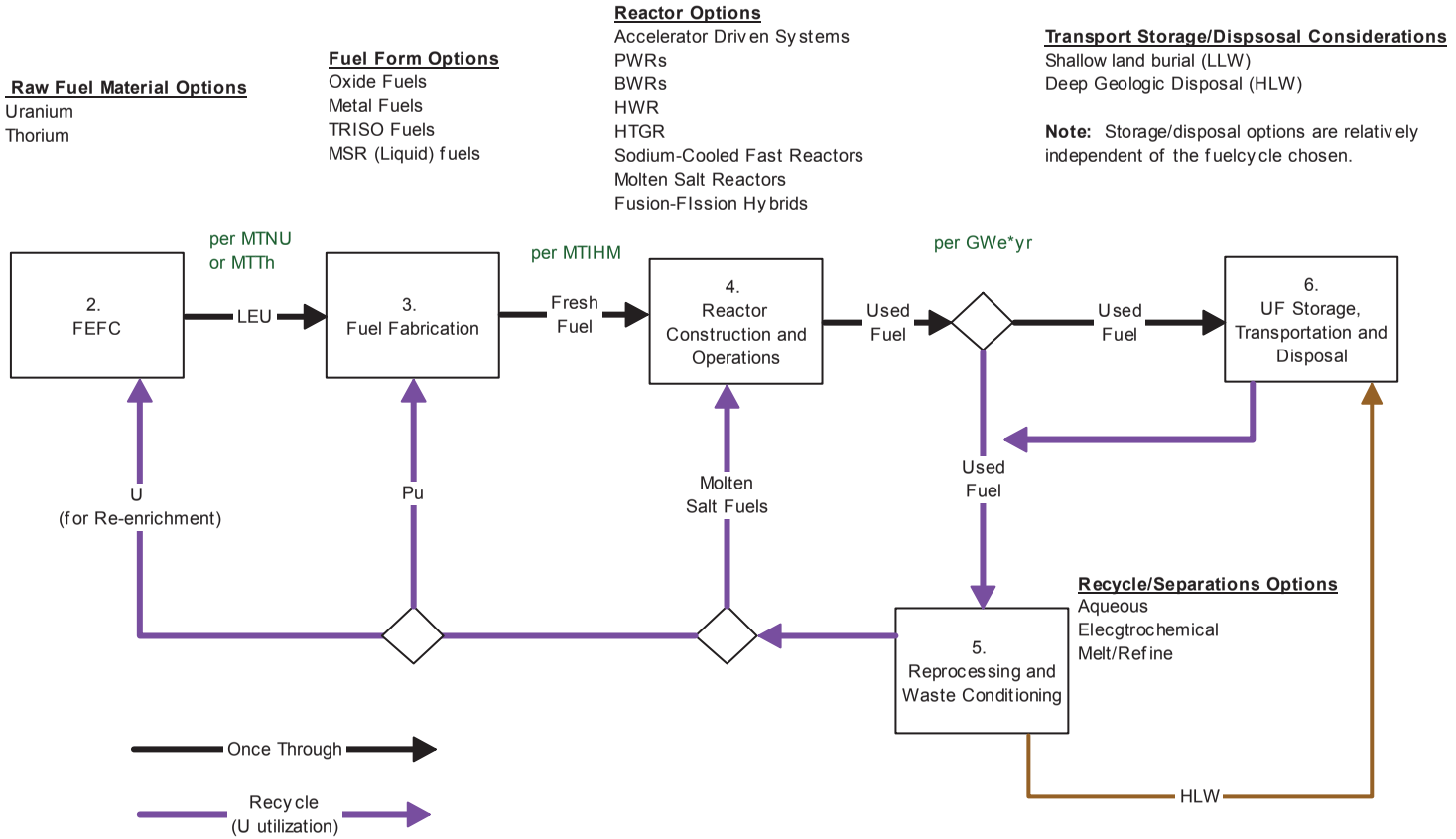
Subject matter experts from within the FCRD Fuel Cycle Technology campaigns responsible for each of the NFC phases were consulted when identifying the options that could be used within each NFC phase. These options are shown in Figure 1-1. Impact estimates are not developed for each of these candidate technologies. Rather, options for each NFC process are grouped into those with similar impacts. Impacts were first estimated based on a reference process based on the current light water reactor (LWR) fuel cycle. Separate impacts were developed for alternate options only if data were available to support development of a separate impact estimate and when the estimate resulted in a significantly different impact with respect to one of the four metrics described above.

As illustrated in Figure 1-2, each NFC phase requires water, material, and energy for process operations. Process operations also produce by-product streams. For this exercise, the by-product and/or waste streams of interest are the CO_2 emissions and radiological wastes that must be managed. Material and energy inputs are used to determine the total (direct and embodied) energy consumed which is used as the basis for estimating CO_2 emissions.

Impacts associated with the main process input are accounted for with upstream process phases. Impacts are normalized per unit process output with the exception of the last phase (i.e. Storage, Transport, and Disposal), which is, by necessity, normalized per unit of input. Using fuel-cycle specific fuel requirements and mass

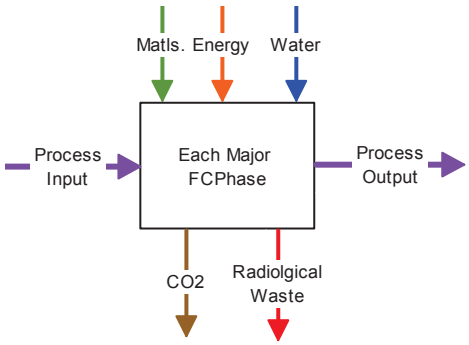
flows through each of these processes, these normalized impact factors can be scaled to estimate impacts from a broad range of potential fuel cycles.

Figure 1-1. Illustration of Range of Candidate Technologies and Options for Phases of the Nuclear Fuel Cycle[2]



[2] The diamonds in the above flowchart do not indicate decision points but simply splits of the material flow process.

Figure1-2. Material and Energy Balance for each NFC Phase



The impact factors are intended to provide a best estimate of the actual impact rather than a conservative or bounding estimate. When developing impact estimates, the following guidelines were used.

- Estimates are based on the presumption of a mature nuclear fuel cycle operating at steady state. In other words, fuel is not consumed from stockpiled reserves, wastes are properly dispositioned, new plants come on line as aging plants retire, etc.

- Estimates are 'forward-looking' in that they are not unduly based on past technologies or practices. Estimates are based, to the extent possible, on contemporary data from operating plants and/or mature designs supported by peer reviewed documentation. However, estimates do not attempt to anticipate technology developments that would cause the impacts to evolve going forward (and in the case of fuel choice, do not attempt to forecast the effects of resource depletion and new discoveries).

- Efforts were made to avoid estimates based on site-specific or other local factors that do not represent the industry as a whole. Examples of this site-specific bias would be things such as unusually high investment in water conservation due to local scarcity, local hydrology or atmospheric conditions that affect doses, etc. When needed, generalized assumptions were made to 'wash out' site-specific factors such that the resulting impact estimates are based on the operation itself rather than by the facility location or the administrative controls applied to the operation. In some cases, it was not possible to identify and compensate for local factors due to limited data.

- Significant effort was not expended pursuing impact estimates for a specific NFC phase if it could be reasoned to be negligible with respect to the impact summed across the full NFC. For example, water use for reactor cooling dominates water use for all other NFC processes with the exception of the FEFC for once-through fuel cycles.

Because the NFC phases for advanced fuel cycles may include technologies for which there is little or no data and/or operational experience, it is expected that there will be areas where there is insufficient data and/or understanding to support a credible estimate. When sufficient data is not available to make a defensible estimate of the impact, surrogate processes are used to represent the impacts and the rationale for selecting the surrogate process is provided.

This report is structured as follows: Section 2 provides a brief summary of the FEFC impacts which this study builds upon. In [Carlsen 2013], representative impacts for each FEFC process (e.g. mining, milling, conversion, etc.) were developed and normalized per unit of mass throughput (e.g. natural uranium, separative work unit [SWU], depleted uranium, etc.). Similarly, this document develops impacts for the remaining phases of the fuel cycle that are normalized based on throughput and can thus be scaled to estimate impacts for many different fuel cycles.

Sections 3 – 6 of this report document the derivation of impacts for the remaining NFC processes: fuel fabrication; reactor operations and construction; and storage, transport, and disposal of spent fuel.

2. FRONT-END OF NUCLEAR FUEL CYCLE

A previous study [Carlsen 2013] developed metrics for evaluating environmental, safety and health, and cost impacts resulting from the front-end of the nuclear fuel cycle Using fuel-cycle-specific parameters such as fuel specifications and product mass flows, these normalized impacts, provided below in Table 2-1, can be converted to impacts per unit electrical energy output from any specified nuclear fuel cycle. Details on impacts and data sources for each part of the FEFC are documented in the previous study and not repeated here.

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Table 2-1. FEFC Impact Estimates

	Uranium Fuels					Thorium
	Mining	**Milling**	**Conversion**	**Enrichment**	**Deconversion**	**Extraction and Refining**
Normalization Unit	**MTNU**		**MTNU**	**SWU**	**MTDU**	**MTTh**
Land Use	2.8E-04		3.3E-06	9.0E-09	9.3E-05	negligible
Permanent (km^2)	3.1E-05		2.6E-08	0	0	negligible
Non-Permanent (km^2)	2.4E-04		3.3E-06	9.0E-09	9.3E-05	negligible
Water Use (ML) net	8.5E-01		6.5E-02	2.9E-05	5.3E-04	1.1E-01
CO2 Emissions (kg)	8.3E+04		2.2E+04	2.8E+01	-3.2E+03*	2.0E+04
Occupational Radiological (person*mSv)	5.2E-01	9.0E-01	8.8E-02	3.1E-05	2.9E-02	3.8E+00

* CO_2 emissions are negative due to the large amount of recovered embodied energy in the HF product stream.

3. FUEL FABRICATION

In addition to traditional LWR oxide fuels, advanced fuel cycles under consideration may employ metallic, tristructural-isotropic (TRISO), molten salt, and potentially other fuel types. Despite the potential for different fuel types and fabrication processes, impact estimates are based on experience and data obtained from fabrication of uranium oxide (UOX) and mixed oxide (MOX) fuels. This is considered appropriate and sufficient for an initial fuel cycle options screening for the following reasons:

- Because fuel fabrication is not a significant contributor to the land and water use across the full NFC, even substantial differences for these metrics, although not expected, would not substantially affect the land and water use impacts for the full NFC.
- A rough estimate, based on construction and operational costs, of energy usage and associated CO_2 emissions from metal fuel fabrication process (see Appendix B) indicates energy use and CO_2 emissions to be approximately three times higher than for fabrication of oxide fuels. Given that this estimate was expected to provide only 'order of magnitude' accuracy, it was concluded that energy and CO_2 emissions for metal fuel fabrication are reasonably represented by those of oxide fuel fabrication.
- Radiological exposures for data for UOX and MOX fuels encompass hands-on, glove-box, and remote fabrication techniques. These techniques will also be used for fabrication of other fuel types.

During fabrication of UOX fuel, enriched uranium hexafluoride (UF_6) is chemically converted into uranium oxide (UO_2) powder, which is then formed into pellets, compacted and sintered, and loaded into fuel rods and assemblies. MOX fuel fabrication follows the same general process, modified to include remote and glove box operations as well as feed purification.

Table 3-1 provides a summary of the impacts associated with the reference fuel cycle technology, UOX fuel fabrication, identified in Section 1. Calculation of these impacts is documented in Sections 3.1 through 3.4, with supplemental information given in Table A-2 and A-3 of Appendix A. Land use data is obtained for the Westinghouse Columbia UOX fuel fabrication facility. The Areva FCFC Romans facility provides the reference for water use and energy consumption, supplemented by information in [Rotty 1975]. Occupational radiological dose is derived from data available on three U.S. UOX fabrication plants currently operating: Areva NP Inc., Westinghouse Columbia, and Global Nuclear Fuel Americas LLC. Impacts for MOX fuel fabrication are also provided in Table 3-1.

Table 3-1. Summary of UOX and MOX fuel fabrication land, water, and CO_2 impacts

	UOX Impacts	MOX Impacts
Normalization Unit	MTIHM	MTIHM
Land Use (km^2)	1.02E-04	1.02E-04
Water Use (ML)[1]	1.41E-01	5.21E-01
CO2 Emissions (kg CO_2)	2.85E+05	4.45E+05
Occupational Radiological Dose (person-mSv)	1.43E+00	1.17E+01
1. Water use estimate reflects potable plus raw water consumption at the reference facility		

3.1 LAND USE

Primary data for the land footprint of the fuel fabrication process are obtained from [Westinghouse 2010] for UOX fabrication and from [Bailly 2009] for MOX fabrication. The Westinghouse Columbia UOX fuel fabrication plant occupies 1,160 acres of land. Given a throughput of 1,150 MTIHM/yr and a facility lifetime of 40 years, land use at the Westinghouse facility is 102 m^2/MTIHM. The Areva MELOX MOX fuel fabrication facility occupies 35 acres of land [Bailly 2009]. Given a throughput of 195 MTIHM/yr and a facility lifetime of 40 years, land use at the MELOX facility is 18 m^2/MTIHM. The discrepancy between the land footprint of the two facilities arises from their locations; the Areva MELOX facility is located on their Marcoule Nuclear Site, currently home to the Phénix prototype fast breeder reactor, while the Westinghouse facility is standalone. The MELOX land use may thus be artificially small as it benefits from infrastructure and land exclusion areas shared with the Phénix facility. Therefore, the larger Westinghouse land use number will be used for both types of facilities.

3.2 WATER USE

Operational water withdrawal for both the Areva FBFC Romans facility and MELOX MOX facility is given in [FBFC 2009] and [MELOX 2010], respectively. Water use at the Romans facility averaged 141,000 L/MTIHM over the 2005-08 period, while water use at the MELOX facility averaged 521,000 L/MTIHM over the 2008-10 period.

3.3 ENERGY USE AND CO_2 EMISSIONS

Reference data for direct energy for UOX fabrication is taken from the Areva FBFC Romans facility. Over the 2005-2008 period, the Romans facility fabricated an average of 524 MTIHM/yr; average direct energy consumption was 212 GJ(e)/MTIHM and 73 GJ(t)/MTIHM [FBFC 2009]. [Rotty 1975] provides an estimate

of 723 GJ(e)/MTIHM and 2,440 GJ(t)/MTIHM for the embodied energy in process materials for the UOX fuel fabrication process.

Production at the reference Areva MELOX MOX fabrication facility averaged 129 MTIHM/yr over the 2008-10 period. The average direct energy consumption was 1,060 GJ(e)/MTIHM and 0.13 GJ(t)/MTIHM [MELOX 2010]. [Rotty 1975] estimates 761 GJ(e)/MTIHM and 2,720 GJ(t)/MTIHM for the embodied energy in process materials for MOX fuel fabrication.

The primary contributor to embodied energy during UOX and MOX fuel fabrication is the zircaloy material input. Zircaloy is employed in fuel cladding due to its transparency to neutrons and corrosion-resistant properties. The age of the data source [Rotty 1975] is recognized. Some environmental impact information related to modern zircaloy production at Areva's Cezus zircaloy plant as well as three others in the zircaloy production chain is available at [Areva 2013]. While energy use is available for the Cezus facility, zircaloy environmental impacts were not able to be isolated as each plant in the chain also produces secondary products not related to nuclear fuel cladding.

Thermal energy is provided through natural gas for both UOX and MOX fuel fabrication. Carbon emissions (see Tables A2 and A3 of Appendix A) are calculated from the energy use information described previously using carbon intensities given in Table A1 of Appendix A.

3.4 OCCUPATIONAL RADIOLOGICAL DOSE

Occupational radiological impact metrics were quantified for fuel types based on the fabrication plant design and operating approach:

- Hands-on: For fuels having small amounts of penetrating radiation and low inhalation radiotoxicity (e.g., low enriched uranium [LEU] fuels in any form)
- Glove box: For fuels having small-to-moderate amounts of penetrating radiation or substantial inhalation radiotoxicity (e.g., fuels containing Pu and/or Th)
- Hot cell: For fuels having high amounts of penetrating radiation (e.g., fuels or targets containing minor actinides or U-233).

Normalized worker exposure estimates for each of these fabrication approaches are given in Table 3-2 and discussed in the text that follows.

Table 3-2. Radiological Impacts to Workers for Fuel Fabrication

Fuel Fabrication	Normalized Impacts (person-mSv/MTIHM)		
	Hands-On	Glove Box	Hot Cell
Technology Basis	*LEU Fuels*	*Pu, Th fuels*	*U-233, MA fuels*
Occupational Radiological Dose	*1.43*	*11.66*	*0.38*

3.4.1 HANDS-ON FUEL FABRICATION

There are two main technological methods to produce LEU oxide fuel – wet and dry. Both methods are analogous to the conversion methods presently in use, as the beginning stages must reverse the original conversion process by changing low-enriched UF_6 to LEU oxides. Operations at LEU-oxide fuel fabrication facilities (wet and dry) that lead to occupational exposure are external exposure from UF_6 cylinders and LEU fuel, and inhalation of UO_2 powder [NRC 1988, 2010].

Currently, there are three U.S. facilities that produce LEU oxide fuel for use in commercial power plants [DOE 2013, ANL 2001]. The three facilities are:

- Areva NP Inc. – Richland, Washington[3]
- Westinghouse Electric Company, LLC – Columbia, South Carolina
- Global Nuclear Fuel Americas LLC – Wilmington, North Carolina

All of the currently operating U.S. facilities use the dry-process. The production capacities of each facility are listed in Table 3-3 [ANL 2011]. The collective occupational doses to workers in these facilities are reported to the Nuclear Regulatory Commission (NRC) annually. The NRC then summarizes the doses and statistics on nuclear fuel cycle facilities in the NUREG-0713 annual series. Collective doses to fuel fabrication plant workers for the last ten years where data is available are given in Table 3-4 [NRC 2001, 2002, 2004, 2005, 2006, 2007, 2008, 2010a, 2011, and 2012]. Total collective radiological impacts to workers are normalized by the mass of fuel produced and are given in in Table 3-5. The average of the 3 facility-specific doses is used for the total collective radiological impact.

Table 3-3. US LEU-Oxide Fuel Fabrication Facilities and Annual Production Capacities

Country	Facility	Location	Capacity (MTIHM/year)	Capacity ($MTUF_6$/year)	Notes
USA	Westinghouse Electric Company LLC - Columbia Fuel Fab Facility	Columbia, South Carolina	1150	1700	Uses Dry Process
USA	Areva NP Inc.	Richland, Washington	700	1035	Uses Dry Process
USA	Global Nuclear Fuel - Americas, LLC	Wilmington, North Carolina	1200	1775	Uses Dry Process
Sources: [ANL 2001, Areva 2013a, NRC 2010, GNF-A 2007]					

Table 3-4. Occupational Radiological Impact Data from LEU-Oxide Fuel Fabrication Facilities from Years 2000-2010

Year	Facility Annual Worker Collective Dose [person-mSv/year]		
	Westinghouse Electric Company	Areva NP, Inc. - Richland	Global Nuclear Fuel - Americas, LLC.
2000	6154.67	1221.37	1126.91
2001	7251.77	1052.24	860.00
2003	2454.92	951.23	572.63
2004	2361.40	855.38	700.70

[3] The Areva site in Richland was previously Framatome ANP; Framatome was formerly Siemens Power Corp.

2005	1912.00	341.80	599.84
2006	2624.57	803.47	589.94
2007	1827.42	728.51	495.66
2008	1587.14	668.84	734.59
2009	1512.54	897.01	480.03
2010	1419.00	999.76	491.68
10-year Average:	2910.54	851.96	665.20
Notes: The 2002 annual report was not available online through the NRC's website			
Sources: [NRC 2001, 2002, 2004, 2005, 2006, 2007, 2008, 2010a, 2011, and 2012]			

Table 3-5. Occupational Radiological Health Impacts for LEU-Oxide Fuel Fabrication facilities

Parameter	Facility-Average Worker Collective Dose [person-mSv/MTIHM]		
	Westinghouse Electric Company	Areva NP, Inc. - Richland	Global Nuclear Fuel - Americas, LLC.
10-yr Average Annual Collective Dose [person-mSv/year] from Table 3-4	2.91E+03	8.52E+02	6.65E+02
Annual Production Capacity [MTIHM/year]	1.15E+03	7.00E+02	1.20E+03
Facility-Specific Normalized Metric [person-mSv/MTIHM]	2.53E+00	1.22E+00	5.54E-01
Average Normalized Collective Dose [person-mSv/LEU-Oxide MTIHM]	**1.43E+00**		

This estimate is likely somewhat conservative relative to what is achievable in future processes. This conclusion is based on the fact that the three sets of plant data differ by a factor of ~10, indicating that much better performance is achievable than is indicated by the average dose. It should be noted however that there is potentially some non-conservatism in that the normalized doses were based on the plant rated capacity and actual production was likely somewhat lower. Nonetheless, based on the considerable data available for hands-on LEU fuel fabrication, confidence in this worker dose estimate is considered high.

3.4.2 GLOVE BOX AND HOT CELL

The value for collective dose to workers from glove-box fuel fabrication was taken from [NEA 2000] and is based on measured doses at AREVA's MELOX plutonium MOX fuel fabrication plant in France.

The value for hot-cell fuel fabrication is assumed to be the same as that for a reprocessing facility and is based on data for the AREVA La Hague reprocessing facility [NEA 2000] (see Section 5.4). This assumes that the hot cell fuel fabrication facility would be designed using the same standards, operating philosophy, and maintenance philosophy as the reprocessing plant – essentially 100% containment of radionuclides, remote operation and maintenance, and sufficient shielding so as to yield very low external dose rates.

The [NEA 2000] study used for these estimates compared the radiological impacts of LWR/LEU oxide once-through and reprocessing fuel cycles. The study was conducted by a multi-national group of technical experts with oversight from the NEA's Committee on

Radiation Protection and Public Health. Table 3-6 provides the worker doses reported in [NEA 2000] and the conversion factors for renormalizing them to units of MTIHM.

Table 3-6. Collective Doses to Workers from Glove Box and Hot Cell Fuel Fabrication

Fuel Fabrication	Glove Box	Hot Cell
Technology Basis	*Pu, Th fuels*	*U-233, MA fuels*
Reported Occupational Radiological Dose (person-mSv/MW_eyr)	*0.43*	*0.014*
Conversion Factor (MW_eyr/MTIHM)*	27.1	27.1
Normalized Occupational Radiological Dose (person-mSv/MTIHM)	***11.66***	***0.38***
* Fuel burnup in [NEA 2000] was 30 GW_td/MT. A thermal efficiency of 0.33 was assumed – leading to a conversion factor of 27.1 MW_eyr/MTIHM		

Confidence levels in this estimate are high for glove box fabrication of MOX fuels, and medium for other fuels using glove box fabrication. Confidence in this estimate is lower for hot cell fabrication because the experience base is very limited.

It should be noted that the hot cell value calculated above is based on the electricity generated from the 30 GW_td/MT SNF fed to the reprocessing plant that was assumed in [NEA 2000]. However, for fabrication the conversion factor should be based on the electricity that will be produced by the fuel that is fabricated, which will vary depending on which representative fuel and reactor are being considered. If this dose represents a significant fraction of the total NFC dose, it would be appropriate to recalculate this impact using the fuel burnup applicable to the specific NFC under consideration.

3.4.3 OTHER CONSIDERATIONS

It is assumed that occupational impacts from fabrication of fuels having similar key radiological characteristics (i.e., penetrating radiation and radiotoxicity) are similar irrespective of differences in enrichment, fuel form (oxide, carbide, metal), or structure (zirconium-based, SS, graphite). For example, fabrication of HTGR LEU fuel would have the same occupational impacts as LWR LEU-oxide fuel. There is no identified, defensible base of experience with fuels other than LWR UOX and MOX on which to base occupational doses for the other fuels.

In some reactor systems (e.g., MSRs and some dry processing) fuel reprocessing and fuel fabrication are integrated into a single hot cell facility. A reasonable assumption is that the value for hot cell worker impacts should be applied only once, i.e., the worker impact should be assumed to be zero for reprocessing or fabrication, but not both.

4. REACTOR CONSTRUCTION AND OPERATIONS

No data is currently available for land, water, and energy use associated with construction and operation of advanced reactor types at a commercial scale. However, the land, water, and energy use impacts based on existing LWRs are considered to be representative for other reactor types for the following reasons:

- Any differences in the actual facility footprint (e.g. accelerator-driven systems may require additional facilities for neutron generation) are likely to be insignificant relative to the stand-off area beyond the facilities that is included in the exclusion area.
- Water use is driven by cooling needs associated with converting heat to electricity. Thus water use for a typical LWR can be scaled using the ratio of thermal efficiencies to account for other reactor types.
- Energy for construction of plant equipment and building structures is expected to be similar to present reactor facilities.

Impacts associated with reactor construction are amortized over the expected lifetime of the plant in order to estimate an equivalent annual impact which can be normalized against the energy produced(i.e. impacts per GW_eyr).

Impacts associated with construction and operation of typical current generation LWR facilities identified in Section 1 are given in Table 4-1. Calculation of the land, water, CO_2 emissions, and radiological impacts is documented in Sections 4.1 through 4.4.

Table 4-1. Reactor construction and operations land, water, and CO_2 impacts

	Reactor Construction Impacts	**Reactor Operations Impacts**
Normalization Unit	GW_eyr	
Land Use (km^2)	See note 1	7.27E-02
Water Use (ML)	See note 2	2.37E+04
CO2 Emissions (kg CO_2)	1.16E+07	See note 3
Occupational Radiological Dose (person-mSv)	Not Applicable	7.30E+02

1. Land use for construction is absorbed into that used during reactor operations.
2. Assumed to be negligible relative to water use during reactor operations.
3. Assumed to be negligible relative to reactor construction.

4.1 LAND USE

Estimating land use for reactor construction and operations presents a unique challenge for two reasons. First, little or no land used for reactor construction and operations is permanent. Although considerable land is committed to the production of nuclear energy, the vast majority of the land is unmodified and merely serves as a buffer zone around the plant. This, along with actual land occupied by the reactor plant and supporting facilities, will be available for other use following reactor decommissioning. Second, the temporary land occupied is not proportional to the energy produced but to the energy production capacity. A $1GW_e$ reactor with a 60-year plant lifetime and a 90% capacity factor will produce 54 GW_eyr of energy. So, the land impact per GW_eyr is $1/54^{th}$ the land that is actually committed to support the associated reactor. Although, defined this way, this metric does not provide a 'snapshot' of the land tied up nuclear plants at any given time, it does provide a land use value for comparison with other

environmental impacts based on impacts per unit energy produced. Note however that, if the same amount of land is used for a reactor over 'n' reactor lifetimes, the land use for reactor operations would be reduced by a factor of 1/n.

The land use impact factor derived below is based on the land committed to a typical reactor plant amortized over the energy produced during its lifetime (km^2/GW_eyr. This is essentially the equivalent of assuming that the land within the exclusion area of a reactor is not recoverable following decommissioning of the reactor. So, in one sense, one might argue that this land use impact factor significantly underestimates (i.e. by a factor of 54) the land that is actually occupied by nuclear power plants at any given time. However, one could also argue that essentially no land is consumed by the process and it therefore significantly overestimates land use. The approach taken is considered to be a reasonable compromise.

Land use for reactor construction and operations is chosen as the exclusion area associated with the reactor facility. For example, Figure 4-1 depicts the exclusion area (red circle) and site boundary (black polygon) of Crystal River Unit 3 [NRC 2012a]. The site boundary may be determined by factors not directly related to reactor needs and may include areas open for public use. For example, Highway 19 runs through the eastern portion of the Crystal River site. The exclusion area is the land withdrawn from public use and is thus chosen as the measure of land used for reactor construction and operations.

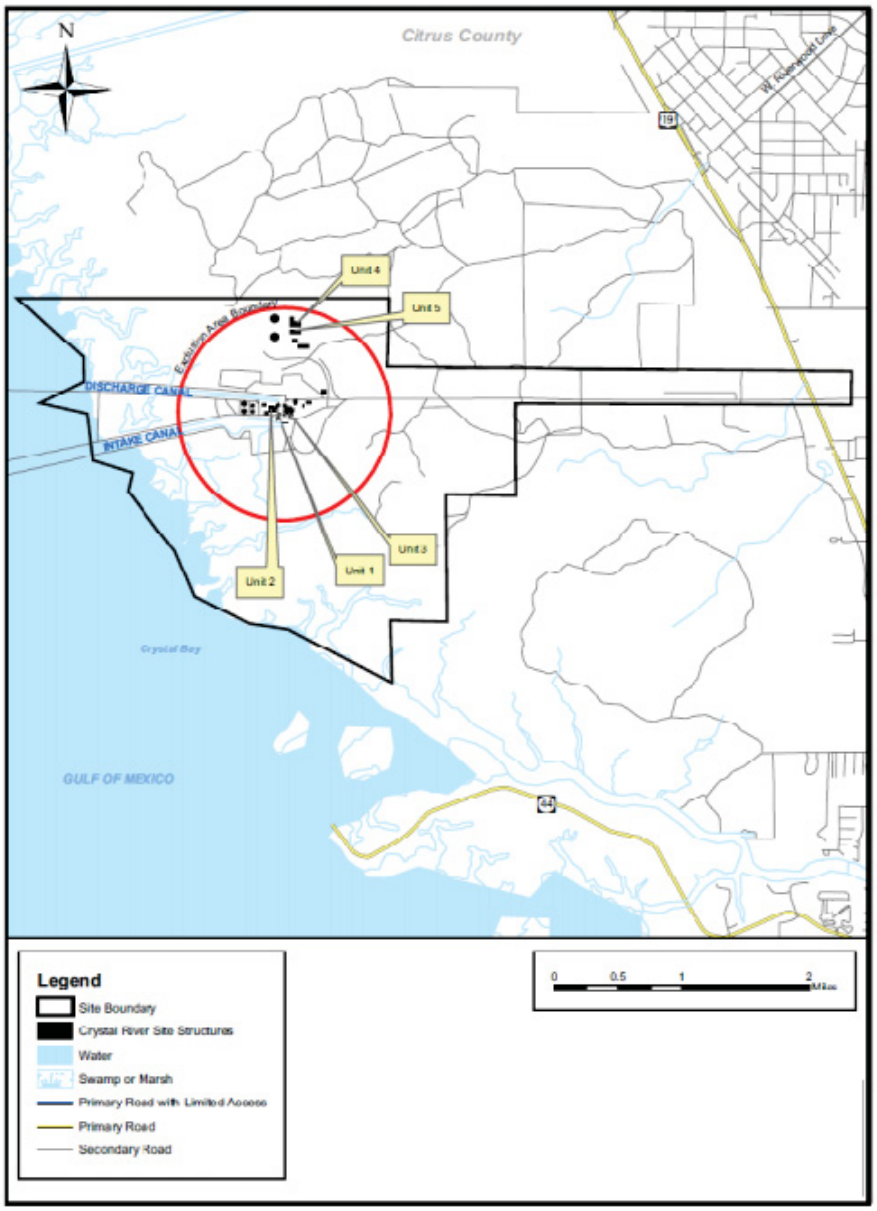

Figure 4-1. Crystal River Unit 3 site boundary and exclusion area. Source [NRC 2012a]

More generally, [NRC 2012b] reviews land use impacts for relicensing currently existing reactors. Within [NRC 2012b], 19 sites explicitly state their exclusion area. The land use for these 19 sites was averaged to estimate the land used for a generic 1 GW_e generating facility as 3.93 km^2. The chosen reference facility, Wolf Creek, which has a generating capacity of 1,165 MW(e), has a circular exclusion area with a radius of 0.75 miles [NRC 2012c]. Assuming a 60 year lifetime and a capacity factor of 0.9, the resulting land use is 0.0727 km^2/GW_eyr.

4.2 WATER USE

[Shropshire 2009] estimates water use for reactors utilizing three cooling technologies. This data is summarized in Table 4-2. Water consumption associated with a cooling tower was selected since it can be applied at any generic site without requiring a large water source (i.e. once-through) or an on-site cooling pond.

Table 4-2. Summary of water withdrawals and consumption for reactor operations. Adapted from [Shropshire 2009].

	Water Withdrawal [ML/GW_eyr]	Water Consumption [ML/GW_eyr]
Once-through	8.32E+05 – 2.01E+06	1.31E+04
Pond cooling	1.66E+04 – 3.68E+04	1.31E+04 to 2.37E+04
Cooling towers	2.63E+04 – 3.68E+04	2.37E+04

4.3 ENERGY USE AND CO_2 EMISSIONS

With the negligible exception of diesel generator operation, energy use during operations is absorbed into net electrical efficiency. Therefore, the energy use and resulting CO_2 emissions associated with reactor operations are those which result from reactor construction.

Energy use in reactor construction is comprised of direct energy consumed in construction operations and energy embodied in building materials and equipment. No source providing direct construction energy use being available, construction energy intensity was assumed to be adequately represented by the construction sector averages reported by the US Energy Information Administration (EIA). The EIA tabulates direct energy coefficients that provide sector energy use per dollar of capital cost expended. To implement this approach, an overnight capital cost of 4,000 \$(2012)/kW_e is obtained from [Dixon 2012]. This cost estimate is combined with the direct energy coefficient data (2.03E-3 GJ per \$(2005)) for construction in the nonmanufacturing sector [EIA 2012].

Embodied energy is estimated from an inventory of the commodity inputs to reactor construction. [Shropshire 2009] reviews a study in which Oak Ridge National Lab (ORNL) estimates for the commodity inputs for construction of a generic 1 GW_e plant. Energy and carbon intensities for these commodities are obtained from [Hammond 2011]. A complete calculation of the energy use and carbon emissions associated with reactor construction is given in Table A4 of Appendix A. This value of 6.29E8 kg CO_2 was amortized over the electrical energy produced over a 60-year lifetime of a plant operating with a 90% capacity factor (1GW_e * 60 yrs *.9 = 54 GW_eyr) to obtain the net energy use and CO_2 impact per unit electrical energy produced (6.29E+008/54=1.16E+07).

Although no data is available for construction of fission-fusion hybrid or accelerator driven reactor systems, it is reasonable to expect that additional CO_2 impacts will be associated with construction of the additional infrastructure and supporting facilities. The additional CO_2 emissions are estimated based on existing LWRs. In the absence of any firm data, a scaling factor of 150% is judged to be a reasonable estimate.

4.4 OCCUPATIONAL RADIOLOGICAL DOSE

Metrics for radiological impacts to workers were quantified for eight candidate reactor technologies shown in Figure 1-1: boiling water reactor (BWR), pressurized-water reactor (PWR), high-temperature gas-cooled reactor (HTGR), a heavy-water moderated reactor (HWR), sodium-cooled fast reactor (SFR), a molten-salt reactor (MSR), an accelerator-driven subcritical reactor (ADS), and a fission-fusion hybrid reactor (FFH). The normalized radiological impacts to workers for each reactor system are given in Table 4-3 and discussed in the following subsections.

Collective occupational dose estimates from reactor decommissioning have been excluded based on NEA conclusions concerning the relatively small contribution of such doses compared to collective doses received during normal operations [NEA 2000][4].

Table 4-3. Radiological Impacts to Workers for Candidate Reactor Systems

Reactor Operations	Normalized Impacts (person-mSv/GW_eyr)							
	PWR	BWR	HWR	SFR	HTGR	MSR	ADS	FFH
Occupational Radiological Dose	730	1570	1830	1200	730	490	2770	2060
Confidence Level	High	High	High	Med	Low	Low	Very low	Very low
Analogous System Used?	No	No	No	No	Yes (PWR)	Yes (Reprocessing)	Yes (SFR + BWR)	Yes (MSR + BWR)

[4] NEA (2000) states that, "Annual collective occupational exposures during decommissioning of all stages of the fuel cycle, including of reprocessing, have been very small, particularly in comparison with worker doses from other stages of the fuel cycle (NEA 1996C, NEA 1999). This is due to the long time period over which decommissioning is conducted, and due to the radiation protection means applied during work activities. Additionally, these doses would be further reduced if normalized with respect to electricity production."

As a cross-check for this NEA conclusion, the Trojan Nuclear Plant was decommissioned while incurring 5910 person-mSv (NUREG-1628). Trojan was a 1.095 GW_e PWR. Projecting this class of reactor into the future with a 90% capacity factor and a 60-year reactor life yields 1.095*60*0.9 = 59.130 GW_eyr, which, when divided into 5910 person-mSv yields 100 person-mSv/GW_eyr which is not negligible but not large compared to reactor doses ranging from 500 to 2800. The impact of decommissioning other facilities will be even smaller, as they may each support several reactors. [NOTE: Putting a reactor in safe storage before decommissioning is estimated to lower worker dose by ~6x, per NUREG-1628].

It should be noted that the normalized collective worker doses in Table 4-3 are (directly or by analogy) based on handling LEU fuel during fuel receipt and refueling. At least the SFR, and possibly all of these reactors, are likely to handle fresh MOX fuel containing reactor-grade Pu that is much more radioactive than LEU and which is expected to take longer to unload from the required secure transports. The associated dose increase has not been calculated because data to support disaggregation of the portion of LEU-fueled reactor dose resulting from fuel receipt and refueling has not been found. As an example of the potential implications, the penetrating radiation dose rates from reactor-grade Pu and Th are about 100 times that from LEU. For the purposes of illustration, if 1% of the presently measured LEU reactor collective dose to workers comes from fresh fuel handling (likely an over-estimate, but used here for arithmetic convenience), then use of Th or Pu fuels would roughly double the total reactor collective dose unless significant additional radiation protection measures are taken.

4.4.1 BWR AND PWR

The NRC has been collecting data on occupational doses and power production at U.S. nuclear power reactors for more than 30 years. The data is analyzed and published in a series of reports with the latest edition at the time of this analysis being [NRC 2012d] containing data through 2010. The information provided includes collective occupational dose and electrical energy production by reactor type (BWR or PWR) for the U.S. fleet. The value adopted in this report is the most recent 3-year average of the collective worker dose for BWRs and PWRs divided by their electricity production taken from Tables 4.1 and 4.2 of [NRC 2012d], respectively. The historical trend of collective dose for workers has been declining as improved technologies and worker radiation protection measures have been brought to bear [Krahn 2013]. As a consequence, the 3-year average from 2008 to 2010 is used because this more recent data better represents future impacts of commercialized systems. The recent data from the NRC reports are given in Tables 4-4 and 4-5 for BWRs and PWRs, respectively, as well as historical data to provide some perspective on the decline in collective dose trends.

Table 4-4. Occupational Dose and Energy Production Data from U.S. BWRs from Years 1994-2010

Year	No. of Individuals with Measurable Dose	Annual Collective Dose (person-mSv)	Average Measurable Dose per Individual (mSv)	Electricity Generated (GW_eyr)	Average Collective Dose per GW_eyr (person-mSv/ GW_eyr)
1994	39171	120980	3.09	22.139	5.46E+03
1995	35686	94710	2.65	24.737	3.83E+03
1996	37792	94660	2.50	24.3222	3.89E+03
1997	34021	76030	2.23	22.8661	3.33E+03
1998	32899	68292.96	2.07	23.7812	2.87E+03
1999	31482	64344.3	2.04	26.9626	2.39E+03
2000	31186	60896.76	1.95	28.4769	2.14E+03
2001	28797	48353.97	1.68	28.7304	1.68E+03
2002	30978	61077.67	1.97	29.46	2.07E+03
2003	30759	56594.34	1.84	29.0944	1.95E+03
2004	33948	54509.82	1.61	29.4248	1.85E+03

2005	33544	59959.75	1.79	29.3868	2.04E+03
2006	34159	49897.61	1.46	30.2384	1.65E+03
2007	37515	53884.16	1.44	30.1893	1.78E+03
2008	34642	45224.13	1.31	31.2483	1.45E+03
2009	36207	52828.69	1.46	30.7627	1.72E+03
2010	37214	48076.56	1.29	31.2746	1.54E+03
				10-year average (2001-2010)	**1.77E+03**
				3-year average (2008-2010)	**1.57E+03**
Sources: [NRC 2012] (Table 4.1: Summary of Information Reported by Commercial Boiling Water Reactors 1994-2010)					

Table 4-5. Occupational Dose and Energy Production Data from U.S. PWRs from Years 1994-2010

Year	No. of Individuals with Measurable Dose	Annual Collective Dose (person-mSv)	Average Measurable Dose per Individual (mSv)	Electricity Generated (GWe-yr)	Average Collective Dose per GW_eyr (person-mSv/ GW_eyr)
1994	44283	95740	2.16	52.3976	1.83E+03
1995	49985	117620	2.35	54.1382	2.17E+03
1996	46852	94170	2.01	55.3378	1.70E+03
1997	50690	95460	1.88	48.9853	1.95E+03
1998	38586	63581	1.65	53.2887	1.19E+03
1999	43938	72313	1.65	56.235	1.29E+03
2000	42922	65620	1.53	57.5299	1.14E+03
2001	38773	62732	1.62	58.8224	1.07E+03
2002	42264	60184	1.42	59.3697	1.01E+03
2003	44054	62961	1.43	57.9206	1.09E+03
2004	35901	49169	1.37	60.3987	8.14E+02
2005	44583	54598	1.22	59.7909	9.13E+02
2006	46106	60314	1.31	59.7513	1.01E+03
2007	42015	47316	1.12	61.9556	7.64E+02
2008	44808	46735	1.04	60.586	7.71E+02
2009	45547	47419	1.04	60.4679	7.84E+02
2010	37796	38237	1.01	60.8594	6.28E+02
				10-year average (2001-2010)	**8.8E+02**
				3-year average (2008-2010)	**7.3E+02**
Sources: [NRC 2012] (Table 4.2: Summary of Information Reported by Commercial Pressurized Water Reactors 1994-2010)					

4.4.2 HEAVY WATER REACTORS

There are multiple HWR designs in the world. For this effort occupational dose data on CANDU reactors in Canada formed the basis for metric quantification. The collective occupational dose for Canadian CANDU reactors was obtained from the NEA Information System on Occupational Exposure (ISOE) 2011[NEA 2012]. Reactor electricity generation capacities were taken from [WNA 2012]. A capacity factor of 0.8 was adopted based on CANDU nuclear station reliability web page from April 2009 [Brown 2009]. The data from these sources for the Canadian CANDU fleet in 2009 are summarized in Table 4-6 leading to an electricity-normalized collective worker dose of 1830 person-mSv/GW_eyr.

CANDU collective worker doses are higher than PWR collective worker doses even though both are two-loop pressurized water reactors. The reasons for this are not clear. However, most worker doses at U.S. LWRs are incurred during refueling/maintenance outages and the situation is similar in CANDUs (see Table 4-

7). The NEA ISOE [NEA 2012] notes that Canadian CANDUs experience an estimated 2-3 planned and forced outages each year as compared to U.S. PWRs which typically shut down for refueling and maintenance once every 18 months. That CANDUs have more outages than PWRs is consistent with the lower capacity factor of 0.8 for the CANDUs compared to over 0.9 in U.S. LWRs. It is possible that doses associated with more frequent maintenance outages at CANDUs accounts for the higher collective dose to workers. Another possibility is that workers received additional doses during on-line refueling.

Table 4-6. Occupational Dose and Energy Production Data for Canadian CANDU HWRs from 2009

CANDU Reactor - Unit	Capacity (MWe)	Year of Exposure	Reactor Collective Dose (person-mSv/yr)	Electrical Energy Produced (GW_eyr/yr)	Reactor Normalized Collective Dose (person-mSv/GW_eyr)
Bruce-A-3	750	2009	1371.5	0.6	2.29E+03
Bruce-A-4	740	2009	1371.5	0.592	2.32E+03
Bruce-B-1	822	2009	1076.75	0.6576	1.64E+03
Bruce-B-2	822	2009	1076.75	0.6576	1.64E+03
Bruce-B-3	822	2009	1076.75	0.6576	1.64E+03
Bruce-B-4	822	2009	1076.75	0.6576	1.64E+03
Darlington-1	881	2009	798.25	0.7048	1.13E+03
Darlington-2	881	2009	798.25	0.7048	1.13E+03
Darlington-3	881	2009	798.25	0.7048	1.13E+03
Darlington-4	881	2009	798.25	0.7048	1.13E+03
Gentilly-2	638	2009	677	0.5104	1.33E+03
Pickering-A-1	515	2009	1220	0.412	2.96E+03
Pickering-A-4	515	2009	1220	0.412	2.96E+03
Pickering-B-5	516	2009	852.5	0.4128	2.07E+03
Pickering-B-6	516	2009	852.5	0.4128	2.07E+03
Pickering-B-7	516	2009	852.5	0.4128	2.07E+03
Pickering-B-8	516	2009	852.5	0.4128	2.07E+03
				Average	**1.83E+03**
Sources: Worker exposure data comes is reported for the year 2009 from [NEA 2012]; Generation capacities of reactors were taken from [WNA 2012] Reactor capacity factor of 0.80 taken from [Brown 2009];					
* After multiplying by capacity factor of 0.8 and converting to GWeyr/yr					

Table 4-7. Occupational Dose Data from Outages and During Electricity Generation from 2009 for Canadian CANDUs (HWRs)

CANDU Reactor - Unit	Dose while Generating Power (person-mSv/yr)	Outage Dose (person-mSv/yr)	Total Collective Occupational Dose (person-mSv/yr)
Bruce-A-3	170.5	1201.0	1371.5
Bruce-A-4	170.5	1201.0	1371.5
Bruce-B-1	142.5	934.3	1076.8
Bruce-B-2	142.5	934.3	1076.8
Bruce-B-3	142.5	934.3	1076.8

Bruce-B-4	142.5	934.3	1076.8
Darlington-1	64.0	734.3	798.3
Darlington-2	64.0	734.3	798.3
Darlington-3	64.0	734.3	798.3
Darlington-4	64.0	734.3	798.3
Gentilly-2	156.0	521.0	677.0
Pickering-A-1	235.0	985.0	1220.0
Pickering-A-4	235.0	985.0	1220.0
Pickering-B-5	143.3	709.0	852.3
Pickering-B-6	143.3	709.0	852.3
Pickering-B-7	143.3	709.0	852.3
Pickering-B-8	143.3	709.0	852.3
Average	**139**	**847**	**986**
Source: [NEA 2012]			
Notes: • Darlington Units 1-4: Outages were extensive due to vacuum building outage that required all units to shutdown. • Bruce-A Units 3 & 4: Two planned outages were performed during 2009 that required the two units to shutdown. • Bruce-B Units 1-4: An unknown number and cause of outages occurred during 2009 • Gentilly Unit 2: A decrease in outage-dose occurred in 2009 vs. 2008 due to less schedule times of maintenance • Pickering-A Units 1 & 4: Reported planned and forced outages occurred in 2009 that resulted in a outages-dose. There was a reduction in routine operations compared to the previous year's operations • Pickering-B Units 5-8: A lesser number of outages were required for the year 2009 vs. 2008 that lead to a lower collective worker dose compared to 2008. Internal doses were a record low from implementing several airborne exposure reduction initiatives (e.g., improved drier performance, decreased tritium curie content in moderator and heat transport of D_2O)			

4.4.3 SODIUM-COOLED FAST REACTORS

Defensible data concerning worker dose at SFRs is limited, primarily because: (a) most SFRs have been prototype, demonstration, or test reactors, where sustained high power operation was not the main objective, and (b) most SFRs had low capacity factors due to reliability issues. An exception is the Russian BN-600 reactor that has been operating relatively reliably (75% - 80% capacity factors) for 20 years. For the 5 years ending in 2010, the average annual collective dose was 540 person-mSv at an average capacity factor of 78% [Vasilyev 2010]; therefore, this data has been used to estimate dose from use of SFRs in an equilibrium fuel cycle. Using a BN-600 electricity generating capacity of 600 MW_e yields the normalized collective worker dose given in Table 4-3 and calculated in Table 4-8 below.

Some SFR designs call for deployment of multiple small modular reactors (SMRs) at a site so as to constitute a virtual large (1000 MW_e-class) reactor. It is likely that the average individual annual dose would be about the same for large reactors and aggregate SMRs. However, there is no basis for concluding that the number of workers required in a large reactor would be the same as the sum of workers for equivalent SMR capacity. To the extent that the number of workers per unit of electrical output differs, so too will the normalized worker dose. Regulations for

SMRs are still evolving and there is no evident way to quantify any differences at present. This same thought is equally applicable to using other reactor technologies such as PWRs and HTGRs for SMRs.

Table 4-8. Occupational Dose and Energy Production Data for the Russian BN-600 from 2005-2010

Parameter Description (Unit)	Parameter Value
Average Annual Worker Collective Dose (person-mSv/yr)	540
Electricity Generating Capacity (MW_e)	600
Capacity Factor (unitless)	0.78
Annual Electrical Energy Production (GW_eyr/yr)	.468
Normalized Worker Collective Dose Metric for SFR system (person-mSv/GW_eyr)	1.2E+03
Source: 5-year average from 2005-2010 (annual breakdown is not available} was adopted from [Vasilyev 2010] describing the Russian BN-600 fast reactor.	

4.4.4 HIGH-TEMPERATURE GAS REACTORS

A helium-cooled, graphite-moderated HTGR using prismatic block fuel was taken as the representative technology for HTGR for occupational dose. Data is limited because experience has mostly involved testing of fuel in non-HTGR reactors, and small prototype or test HTGRs dating to the 1980s. For example, Peach Bottom Unit 1 only generated 15 MWy(e) with graphite fuel and accessible NRC documents containing occupational dose do not go back this far.

The single exception to the non-HTGR and small prototype experience is operation of the Fort St. Vrain reactor (FSV, 342 MWe) between 1974 and 1991. Unfortunately, this reactor had reliability issues and, in its best two years, achieved a capacity factor of only 28% [NRC 1993] which does not provide a defensible basis for estimating worker dose for a future, commercial HTGR–that must be assumed to operate reliably[5].

On balance, an HTGR is a two-loop system like a PWR and, like a PWR, the coolant is not significantly activated (although both coolants would contain trace activation and fission products) so it was assumed that an HTGR would have the same normalized radiological dose to workers as a PWR: 730 person-mSv/MW_eyr. This value is considered more realistic than paper studies that estimated collective worker doses ranging from 0.7 to 2.0 person-mSv/MW_eyr [Su 1980] but the disparity leads to confidence in the result being low.

[5] In 1981 and 1983, the FSV reactor produced 94 GW_eyr out of a possible 330 and worker collective dose was 10 person-mSv leading to an imputed 0.11 person-mSv per MW_eyr. This value is likely low because of the many people doing maintenance.

4.4.5 MOLTEN SALT REACTORS

A representative MSR is taken to be graphite-moderated using thorium/U-233 fuel dissolved in a circulating molten fluoride coolant. It would be a two-loop design similar to PWRs, CANDUs, and SFRs. It would have a fully integrated fuel processing plant to remove fission products, remove and/or feed fissile material, isolate Pa-233 for decay, and feed thorium. There is no reactor operating experience on which to base a collective worker dose estimate. The only MSR to operate was the Molten Salt Reactor Experiment (MSRE) at ORNL, which was a 7.4 MW_t reactor designed to test the reactor concept and materials. The MSRE operated for about 1.5 full-power years during its 5-year life, which is not sufficiently representative of a potential future MSR so as to provide a basis for estimating collective worker doses. Additionally, finding worker dose information for the MSRE has been unsuccessful. An MSR worker collective dose estimate was developed by recognizing that: (a) most worker dose at reactors where the secondary loop is not radioactive results from maintenance performed during maintenance/refueling outages, and (b) both the MSR primary and reprocessing loops will have to be designed as a hot-cell- (or canyon-) type facility with remote maintenance because these loops contain what is essentially spent fuel. Thus, in concept, a MSR is similar to a nuclear fuel reprocessing plant and, thus, the MSR would be expected to have individual worker dose rates similar to a reprocessing plant, i.e., 14 person-mSv/GW_eyr[6]. However, the normalization basis for a reactor is different from that of a reprocessing plant. Thus, in concept, a MSR is similar to a nuclear fuel reprocessing plant in that it would be remotely operated and involves a substantial number of flowing nuclear materials including those in the integral fuel reprocessing plant. As a consequence, we assume the annual individual dose to MSR workers is the same as in a fuel reprocessing plant. Additionally, absent detailed information about the number of exposed workers at a MSR, we assume it to be the same as for the fuel reprocessing plant. This means that the un-normalized collective dose to MSR workers (in person-mSv) is the same as that for fuel reprocessing plant workers. However, a typical aqueous fuel reprocessing plant can support about 35 reactors each generating ~1 GW_eyr annually for the collective dose it imparts to workers whereas a single MSR would be producing ~1 GW_eyr annually. Thus, the electricity-normalized collective dose from a reprocessing plant (14 person-mSv/GW_eyr; see Section 5.4) needs to be multiplied by 35 to yield the electricity-normalized collective dose for the MSR which is 490 person-mSv/GW_eyr.

4.4.6 ACCELERATOR DRIVEN SYSTEMS

This system is composed of a high-energy, high-current proton accelerator and target, which produce spallation neutrons. These neutrons drive a closely coupled subcritical assembly generating fission energy that is converted to electricity by

[6] See Section 5.4; conversion assumes fuel burnup from [NEA 2000] of 30 GW_td/MTIHM and thermal efficiency of 0.33, resulting in 27.1 MW_eyr/MTIHM.

conventional means. In essence, the sub-critical assembly is a stand-alone nuclear reactor. Transport of the accelerator-produced neutrons evenly through the core and efficient production of fissile material that will be used to support other reactors favors use of a fast reactor. On the basis that the reactor portion will effectively share many of the same characteristics as a SFR, we represent this portion of the system with an SFR and adopt the associated radiological dose to workers. The accelerator portion of the system leads to the need to add a contribution from maintaining radioactive components of the accelerator: beam tubes (especially the target), target cooling system, and possibly target material cleanup operations. There is no experience with high-availability, high-current accelerator systems, and target materials vary widely: solid or liquid, various metals such as lead, mercury, and tungsten. For this draft, we assume that the target involves a single loop that is similar a single-loop reactor (BWR). This combination leads to a worker radiological dose of 1200 + 1570 = 2770 person-mSv/GW_eyr and is shown below in Table 4-9.

Table 4-9. Analogous ADS System Components and Associated Worker Dose Estimates

ADS System Component	Analogous Reactor System	Collective Dose for Analogous Reactor System (person-mSv/GW_eyr)
Proton-accelerator with worker doses from high-energy protons	SFR	1200
Metal targets	BWR	1570
	Total:	2.8E+03

4.4.7 FISSION-FUSION HYBRID REACTORS

This concept is sufficiently immature so that it is possible to postulate very different representative technologies. One possibility is a fusion reactor using excess neutrons to produce U-233 (from Th-232), which is then recovered and used to make fuel for fission reactors, and within this various types of fusion and fission reactors are possible. Another possibility is a fusion reactor that produces U-233 in a blanket of Th-232 that is nearly critical and in which most of the power is produced (much like an ADS concept with MSR-like components). For the purpose of this effort we assume the latter. In particular, we assume the FFH is composed of a torus fusion reactor having one or more molten salt fluoride blankets that combine fusion heat removal, a subcritical assembly generating substantial fission power, and tritium production to continue to fuel the fusion reactor.

For the blanket portion of the system, in which most of the power is generated, is similar to a MSR because it transports heat to generate electricity, contains nuclear materials that are dissolved in the subcritical blankets, and produces tritium, all of which are comparable to an MSR. Thus, the MSR worker radiological dose is adopted here. The fusion portion of the system entails the need to maintain highly

activated components of the fusion device in the presence of blanket material after the blankets are assumed to be drained and to maintain tritium storage and feed systems. There is no experience with maintaining such systems and the design philosophy (e.g., remote, semi-remote) that will be used has not been established. Because the primary fusion system components are radioactive and there is no radioactive secondary loop, we assume that worker doses will be similar to that for a single-loop BWR. This combination leads to a worker radiological dose of 490 + 1570 = 2060 person-mSv/GW_eyr and is shown below in Table 4-10.

Table 4-10. Analogous FFH System Components and Associated Worker Dose Estimates

FFH System Component	Analogous Reactor System	Collective Dose for Analogous Reactor System (person-mSv/GW_eyr)
Torus Fusion Reactor Blanket	MSR	490
Primary coolant loop	BWR	1570
	Total:	2.1E+03

5. REPROCESSING AND WASTE CONDITIONING

Used fuel (UF) reprocessing chemically separates discharged nuclear fuel into its constituents. Several separations technologies have been proposed and are often separated into two broad categories of either aqueous or dry processes. Large-scale reprocessing experience is limited to the aqueous PUREX process. A rough estimate, based on construction and operational costs, of energy usage and associated CO_2 emissions from an electro-chemical (dry) process (see Appendix B) indicates energy use and CO_2 emissions to be approximately twice that of PUREX. Given that this estimate was expected to provide only 'order of magnitude' accuracy, it was concluded that energy and CO_2 emissions for reprocessing are reasonably represented by those of the PUREX process.

The reference process described here is the PUREX process, which is the sole large-scale industrially achieved reprocessing technology. The aqueous PUREX process allows for the recovery of uranium and plutonium for recycling, although similar aqueous processes may partition, for example, selected fission products as well or all transuranic elements at once. Table 5-1 summarizes the impacts associated with reprocessing used fuel. Supplemental information is provided in Tables A-6 and A-7 of Appendix A.

Reference data from the Areva La Hague reprocessing facility was used to obtain the impacts given here. Operations at La Hague are dominated by reprocessing, but include other operations that cannot be disaggregated from the data available (March 13th, private correspondence with Paul Murray, Areva) including high level waste vitrification and canistering, intermediate level waste conditioning and

cementaton, and treatment of resins used in water purification (April 24th, personal correspondence with Patricia Paviet-Hartmann). These waste treatment and conditioning operations would be necessary to support any industrial scale reprocessing facility, so it is considered appropriate to report impacts for the entirety of La Hague.

Table 5-1. Summary of reprocessing land, water, and CO_2 impacts

	Impacts
Normalization Unit	MTIHM
Land Use (km^2)	4.41E-05
Water Use (ML)[1]	4.83E-01
CO_2 Emissions (kg CO_2)	5.15E+05
Occupational Radiological Dose (person-mSv)	3.80E-01

1. Water use estimate reflects potable plus raw water consumption at the reference facility.

5.1 LAND USE

The La Hague reprocessing facility sits on a 300 ha site. Given an annual capacity of 1,700 MTIHM processed per year and a 40-year facility lifetime [Bailey 2009], the land use at the La Hague plant is 44 m^2/MTIHM. This land use estimate may be somewhat low due to the fact that production over the facility lifetime will be less than 40 times the 1700 MTHM per year capacity. This is unlikely to be significant since the land use for reprocessing is only a very small fraction of the land impacts associated with the nuclear fuel cycle.

5.2 WATER USE

Operational water withdrawal for the La Hague facility is given in [Areva 2009]. Water use at La Hague averaged 483,000 L/MTIHM over the 2007-09 period.

5.3 ENERGY USE AND CO_2 EMISSIONS

Reference data for energy use in used fuel reprocessing is obtained from the Areva La Hague facility. La Hague processed an average of 938 MTIHM/yr over the 2007-09 period. Direct energy use at La Hague averaged 1,740 GJ(e)/MTIHM and 816 GJ(t)/MTIHM [MELOX 2010], all in the form of natural gas. Average annual chemical consumption at La Hague is also obtained from [MELOX 2010] and is needed to estimate the energy embodied in material inputs to reprocessing and other operations. Inputs outside of chemicals give rise to negligible impacts in this respect. See Tables A5 and A6 in Appendix A for a full calculation of the energy use and carbon emissions associated with used fuel reprocessing.

5.4 OCCUPATIONAL RADIOLOGICAL DOSE

This section addresses radiological dose to workers from SNF reprocessing and, to the extent that it is an integral part of reprocessing (e.g. for MSRs), recycle fuel fabrication. To avoid double-counting for MSRs, MSR fuel fabrication is assigned zero occupational impact since the reprocessing scheme was accounted for in the reactor dose. Any makeup fuel using enriched U or Pu can be made by simply mixing and melting the component chemicals (e.g., UF_4, LiF, BeF) in a chemical lab with hoods. U-233 makeup fuel would be made similarly but inside a hot cell with little additional dose. Values for the normalized collective radiological dose to workers are given in Table 5-2 and discussion of the basis for these values follows.

Table 5-2. Radiological Worker Impacts for Recycling and Reprocessing Operations

Fuel Reprocessing/ Recycling Technology	High Temp/Dry					Aqueous			
	E-Chem	Melt Refining	Halide Slagging	OREOX	In-line MSR (F, Bi extraction)	PUREX	Co-decontamination, UREX-NPEX	THOREX	TRISO (crush, crack, dissolve)
Occupational Radiological Dose (person-mSv/MTIHM)	0.38					0.38			
Confidence Level	Low					High	Med	Med	Low

5.4.1 AQUEOUS TECHNOLOGIES

There is significant experience with commercial reprocessing of LWR fuels using the standard PUREX process and the occupational impacts of reprocessing as a result of the operation of the La Hague reprocessing plants in France and THORP in the U.K. The value for the collective worker radiological dose of 14 person-mSv/GW_eyr was based on experience at La Hague [NEA 2000]. Normalization is based on the 30 GWd/MT burnup assumed in [NEA 2000] which, assuming a thermal efficiency of 33%, leads to a conversion factor of 0.027 GW_eyr/MTIHM and the mass-normalized occupational impact of 0.38 person-mSv/MTIHM shown in Table 5.2.

Currently envisioned co-decontamination and fuel fractionation processes[7] such as the UREX variants include many processes that are essentially identical to PUREX plus additional processes needed to accomplish the fractionation. While any additional processes would be performed in a hot cell or canyon environment, imparting the same low dose rate to individual workers as the PUREX process per

[7] Fractionation processes might be used, for example, to separate minor actinides and lanthanides from the raffinate, minor actinides from the lanthanides, and cesium and strontium from the raffinate.

se, it is likely that some number of additional workers would be needed to conduct the additional processes leading to an increase in collective dose. However, the extent of the increase is unknown because it would depend on the number of additional processes and their design, which has not yet been determined. On balance, the increase is expected to be small in comparison to the scope of the entire reprocessing operation which supports adoption of the same value as for PUREX: 0.38 person-mSv/MTIHM.

There is a limited (and dated) basis for estimating how occupational doses from THOREX processing of thorium-based fuels might differ from PUREX experience [Wymer 1968]. UREX is expected to be a good first approximation for THOREX for thorium-based oxide fuels because a future deployment of THOREX would presumably require additional separations processes functionally similar to what was added to PUREX to yield UREX, and because THOREX and PUREX are basically the same process. However, the additional processes have not yet been conceived and there is presently no basis for differentiating the occupational impacts of THOREX from those of UREX or, as discussed in the previously, from those for PUREX. Thus, we adopt the same value as for PUREX: 0.38 person-mSv/MTIHM.

There is also no data for estimating impacts from reprocessing graphite fuels containing TRISO particles. Separations for HTGR fuel would presumably involve the additional processes and attendant occupational impacts discussed above for a UREX variant or THOREX depending on the nature of the fuel matrix. Further, graphite-based HTGR fuel reprocessing would entail additional occupational impacts because of the need for additional head-end processes to either crush graphite blocks, requiring rotating dust-generating equipment inside cells leading to additional ventilation system complexity, or to remove the fuel-bearing material from the bulk graphite moderator. Either of these steps would be followed by burning some or all of the graphite leading to major expansion of the off-gas system to remove contaminants from the large volume of carbon dioxide. Occupational impacts from handling a substantial volume of stabilized waste containing C-14 would also be increased. However, the extent of the additional worker radiological impacts is presently unknowable because of the lack of experience or even an integrated design for such a facility. Again, we assume the same value as for PUREX: 0.38 person-mSv/MTIHM.

Confidence in the PUREX and UREX co-decontamination values for collective worker radiological dose is high because it is based on experience in an industrial-scale facility for PUREX and the relatively modest differences between PUREX and UREX. Confidence in using the same value for THOREX is medium because, while this process has many conceptual similarities to PUREX, there are many differences in detail—coupled with the potential additions to fractionate UF constituents and no experience with these differences. Confidence in using the same value for graphite-based fuels is low because of the additional uncertainties resulting from dealing with the graphite in addition to the uncertainties THOREX and UREX.

5.4.2 HIGH TEMPERATURE/DRY TECHNOLOGIES

Dry reprocessing technologies have been operated at laboratory and engineering scale at national laboratories in the U.S. and elsewhere to process nuclear materials as a part of ongoing R&D under non-commercial regulations. As a consequence, available information is not adequate to differentiate collective occupational doses among the dry technologies or between dry and aqueous technologies. Because any of these technologies would have to be implemented in hot-cell or canyon-type facilities conceptually similar to those used for PUREX reprocessing and in accordance with the same regulations and standards, the collective occupational dose value based on operating the La Hague aqueous reprocessing plant from [NEA 2000] are also adopted for dry reprocessing: 0.38 person-mSv/MTIHM.

It should be noted that dry reprocessing is typically proposed for deployment at a size adequate to support a single reactor site containing the equivalent of 1 or 2 large reactors, while aqueous reprocessing is typically deployed at a scale so that one plant supports 30-40 large reactors. It is likely that the average individual dose would be about the same for the two facilities because this is driven by regulatory requirements. However, there is no basis for concluding that the number of workers per unit of throughput and, thus, the collective worker radiological dose at a reprocessing plant serving one site scales linearly with that of a large centralized facility. To the extent that the number of dry reprocessing workers required to achieve a given throughput is different than the number of workers for aqueous reprocessing, so too will be the collective worker radiological dose; the lack of a detailed analysis precludes estimating this impact, however.

6. STORAGE, TRANSPORT, AND DISPOSAL

Numerous deep geological repository (DGR) options have been proposed, including disposal in deep boreholes, granite or volcanic rock formations, impermeable clay media, and salt domes. Environmental impacts associated with these disposal strategies would vary widely: surface land use for deep borehole disposal, for instance, may be negligible whereas shallower repositories would remove considerable land from service for other uses. On the other hand, energy (and hence CO_2) impacts of borehole disposal may be significantly larger than shallow disposal due to deep excavation and backfill operations. Consequently, it is not possible to represent generic DGR disposal impacts that are broadly applicable to all disposal concepts by use of a reference facility. On the other hand, estimating the impacts associated with a well-understood DGR can provide valuable guidance as to which impact categories require more intensive study if DGR options are contrasted in future work.

Deep disposal impacts are based on the reference Yucca Mountain disposal option previously examined by the DOE identified in Section 1. This is the best-quantified large repository design presently available. Table 6-1 summarizes the impacts associated with spent fuel disposal in a deep geologic repository. All impacts, except land use, are based on the higher-temperature operating mode described in [DOE 2002] and a repository capacity of 70,000 MTIHM. Impacts associated with shallow-land burial of low and intermediate level wastes are also provided in Table 6-1. The basis for these estimates is given in the remainder of section this section.

Land, water, and CO_2 impacts are based on impacts associated with disposal of DU given in [Schneider 2010]; readers are referred to [Schneider 2010] where more information is sought. Occupational radiological dose accumulated from shallow land burial is based on operation of the U.S. Ecology LLW Disposal Facility.

When evaluating other waste forms and/or repository concepts, each of the impacts shown in Table 6-1 could be scaled with the mass, activity, or decay heat disposed and/or the number of packages emplaced. Further, the scaling relationships depend on the repository design and host medium. To permit scoping estimates, a single scaling metric will be chosen for each impact category.

Land, water, and CO_2 impacts associated with transport of nuclear materials are considered negligible[8]. Occupational radiological dose associated with transportation of nuclear material from extraction through disposal is derived from NEA data. Appendix C discusses differences between transportation of uranium- and thorium-based compounds.

Impacts for the interim storage of used nuclear fuel are based on the Independent Spent Fuel Storage Installation (ISFSI) that was proposed for construction on land belonging to the Skull Valley Band of Goshute Indians in Utah. Two documents supporting the reference facility are used to inform these estimates: the 2001 EIS prepared by the NRC [NRC 2001], and a 2009 EPRI estimate of the cost of constructing and operating a facility based on ISFSI [Kessler 2009]. The ISFSI would be designed and licensed for a capacity of 40,000 MTIHM. Although the EIS was prepared in support of a 20 year operating period, the EIS indicated that the Skull Valley Band expressed interest in an extension of the operational lifetime to 40 years. Therefore, the 40 year lifetime will be taken as the reference value.

[8] Energy consumed in transportation has been shown to represent less than 1 percent of the total energy consumed in the front end of the nuclear fuel cycle [Schneider 2010].

Table 6-1. Summary of disposal land, water, and CO_2 impacts

	Shallow Land Burial Impacts[a]	Geologic Repository Impacts
Normalization Unit	MT waste	MTIHM
Land Use (km2)[a]	9.74E-6	1.50E-03
Water Use (ML)	2.3E-04	1.43E-01[b]
CO2 Emissions (kg CO2)	1.82E+00	see detail below
Excavation & Closure		2.49E+04
Operations		6.32E+04
Waste Packages and Drip Shields		
Fabrication		2.71E+04[c]
Waste Package materials		2.91E+04[d]
Drip Shield materials		1.30E+03
Occupational Radiological Dose (person-mSv)	1.32E-01[e]	1.06E+00[f]

[a] Based on impacts for disposal of DU given in [Schneider 2010]
[b] In [DOE 2002], water use is estimated as total raw water withdrawals. Data on any planned water recycling program is needed to generate an estimate of net water usage.
[c] Includes fabrication of storage, transport, and disposal canisters, waste packages, and drip shields packages because energy use data could not be disaggregated.
[d] Includes materials for storage, transport, and disposal canisters.
[e] Units of dose for shallowland burial are person-mSv/MTIHM. Based on once-through LWR fuel cycle. This value increases by a factor of ~3 if reprocessing and MOX fuel fabrication is included. (see section 6.4.4)
f This value is estimated based on disposal of typical LWR spent fuel. Worker dose impacts from disposal of other high level wastes may be slightly higher (see section 6.4.5)

Table 6-2. Summary of storage land, water, and CO_2 impacts

	Interim Storage Impacts
Normalization Unit	MTIHM
Land Use (km^2)	3.0E-05
Water Use (ML)	negligible[a]
CO2 Emissions (kg CO_2)	see detail below
Concrete Manufacture	3.11E+04
Storage Package Fabrication	See note a
Storage Package Materials	See note a
Occupational Radiological Dose (person-mSv)	1.16E+00

[a] Energy and associated CO2 emissions associated with materials and fabrication of the container used for storage and transport is was included in the disposal packaging estimates that formed the basis for th CO2 estimate for the waste package shown in table 6-1 above.

6.1 LAND USE

6.1.1 INTERIM SURFACE STORAGE

The land footprint of the ISFSI facility would encompass the storage pads themselves and supporting structures within a 40 hectare restricted access area. An isolation perimeter surrounding the restricted area as well as facilities supporting fail access to the site contribute to the total land use reported in the EIS, 120 hectares (1.2 km^2) [NRC 2001]. Land use per MTIHM is thus 3.0E-05 km^2/MTIHM.

6.1.2 DEEP GEOLOGIC REPOSITORY

The Yucca Mountain disposal option consists of a land withdrawal area (Figure 6.1) of 150,000 acres (600 km^2); a smaller controlled area consists of 74,000 acres (300 km^2). However, choosing the control area of the Yucca Mountain facility as a representative land use for geologic disposal is overly conservative[9]. Land at the proposed Yucca Mountain facility was previously federally owned, and transfer of ownership to DOE was straightforward. By doing so, future repository expansions may occur unimpeded.

[Hardin 2012] was used to determine a generic land footprint for geologic disposal. [Hardin 2012] examines potential generic waste emplacement approaches for a 140,000 MTIHM repository. Chapter 4 of [Hardin 2012] provides information on disposal approaches including panel dimensions plus related access, disposal, and service drifts and the number of panels needed to dispose 140,000 MTIHM for 5 representative disposal concepts. The footprint for these concepts ranged from 7 to 45 km^2, with an average of 19.5 km^2. Based on this information, a footprint of 20 km^2 will be used to represent a geology-independent generic repository. In addition to the land footprint, 40 CFR 191 requires a maximum setback distance of 5 km; a 5 km setback from the repository footprint will be assumed for determination of the total controlled area.

There is no established upper limit for repository size. [Hardin 2012] indicates the feasibility of developing a repository of up to 140,000 MTIHM in multiple geologies. [Peterson 2003] estimated a similar capacity of 150,000 MTIHM for spent fuel disposal in Yucca Mountain with minimal perturbations to the then-existing design (loading remains at ~75 MTIHM/acre, drift footprint increased somewhat to 8 km^2). Therefore, a capacity of 140,000 MTIHM will be assumed.

For simplicity, it was assumed that repository footprints were square. Then, each side of the repository measures (10+$\sqrt{20}$) km (5 km setback on either side of the repository, and $\sqrt{20}$ km for one length of the repository) for a total controlled area of 209 km^2. Assuming a repository capacity of 140,000 MTIHM, the total land footprint for geologic disposal is 1.5E-03 km^2/MTIHM.

[9] 40 CFR 191 requires a maximum controlled area of 100 km^2.

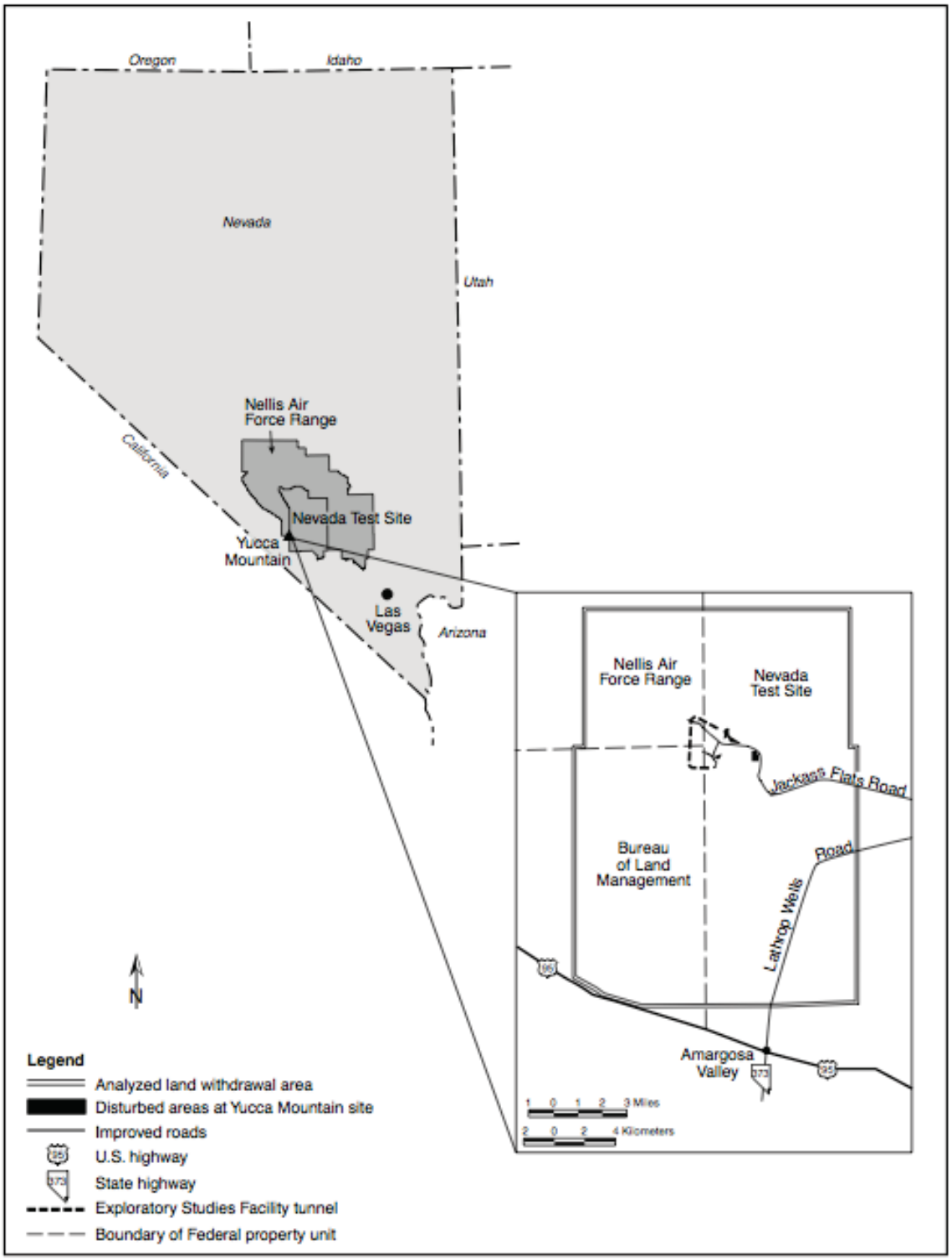

Figure 6.1. Land withdrawal area. Source: [DOE 2002]

6.2 WATER USE

6.2.1 INTERIM SURFACE STORAGE

Water use, as reported in [NRC 2001], is minimal for the passively-cooled ISFSI. It is considered below threshold relative to the water use associated with reactors and other fuel cycle technologies.

6.2.2 DEEP GEOLOGIC REPOSITORY

Estimated water usage for the construction, operation, and closure of the Yucca Mountain spent fuel repository is given in [DOE 2002, Table 4-11] and replicated in Table 6-3. Converting the total water use to units of ML (1.23 ML per acre-foot) gives a total water use of 10,000 ML. Dividing this by the 70,000 MTIHM YMP capacity yields an estimated water use of 1.43E-01 ML/MTIHM. Water use is assumed to scale with the mass of material disposed.

Table 6-3. Water use for repository construction, operation, and closure

Phase	Duration (yr)	Water Demand (acre-feet/yr)	Total Water Demanded during Phase (acre-feet)
Construction	5.00E+00	1.60E+02	8.00E+02
Operation & Monitoring			
Operations Period			
Emplacement and Development	2.20E+01	2.30E+02	5.06E+03
Subsequent emplacement only	2.00E+00	1.80E+02	3.60E+02
Monitoring Period			
Initial decontamination	3.00E+00	2.20E+02	6.60E+02
Subsequent monitoring & caretaking	7.30E+01	6.00E+00	4.38E+02
Closure	1.00E+01	8.10E+01	8.10E+02
		Total Water Use:	**8.13E+03**

6.3 ENERGY USE AND CO_2 EMISSIONS

6.3.1 INTERIM SURFACE STORAGE

Four contributors to energy use may be identified. These are:

1) fabrication of the high density concrete storage pads and overpack,
2) construction of the supporting buildings and container transfer equipment,
3) manufacture of the spent fuel storage, transport, and disposal containers, and
4) direct energy use during the operation phase.

Of these contributors, design and operational data reported in [Kessler 2009] shows that construction of buildings and equipment (#2) as well as direct operational energy use (#4) are negligible (on the order of 1% or less) contributors to total energy use. Therefore, they are not considered further.

Concrete requirements for the storage pads and overpacks are provided in [Kessler 2009]. The reference design calls for 4,000 dual purpose canisters (DPCs) to be emplaced at the facility. Each DPC would require an overpack of 1000 yd^3 in extent plus a storage pad of high density concrete measuring a further 67 yd^3. The waste packages themselves are designed for continued use when the waste is subsequently transferred to a deep geologic repository (DGR). Therefore, the energy and CO_2 intensities for waste package fabrication developed in the DGR section will be used here as well[10].

Table 6-4 summarizes the reference facility design data taken from [Kessler 2009]. Table 2 steps through the energy and CO_2 intensity calculations and provides overall results.

Table 6-4. Interim Spent Fuel Storage Facility data from Ref. [Kessler 2009]

Item	Unit	Value
Capacity	MTIHM	40,000
Operating Lifetime	Yr	40
Concrete required for pads	yd^3	2.68E5
Concrete required for overpacks	yd^3	4.00E6
Total concrete required*	m^3	3.26E6
Total concrete mass**	Kg	7.83E9
Concrete mass per unit of capacity	kg/MTIHM	1.96E5
* at 0.76455 m^3 per yd^3 ** at 2400 kg/m^3 for high density concrete		

Table 6-5. Calculation of CO_2 impact factor for interim surface storage*

Item	Unit	Value	Source
Energy intensity, concrete manufacture	GJ/kg	1.11E-3	[Hammond 2011]
Energy use, concrete manufacture	GJ/MTIHM	2.17E2	calculated
Emission factor, concrete manufacture	kg CO2/GJ	143.2	[Hammond 2011]

[10] Note that if a fuel cycle incorporates both long term surface storage and DGR disposal, the waste package impacts should be deducted from the DGR disposal category in order to avoid double counting them.

CO2 emissions, concrete manufacture	kg CO2/MTIHM	3.11E4	calculated
* Energy and associated CO_2 emissions impacts from materials and fabrication of storage canisters is included in those calculated for the waste package section 6.3.2.3 below.			

6.3.2 DEEP GEOLOGIC REPOSITORY

Major energy use processes associated with constructing, operating, and closing the repository are broken into four categories: (1) initial excavation, (2) operation of ventilation fans for active cooling during repository operations, (3) material inputs and fabrication for the engineered barrier design, and (4) transportation associated with emplacement of waste packages into the final repository. Of these, major energy use processes, transportation associated with emplacement of (and transport to the repository site of) waste packages was considered negligible.

The CO_2 impact metric thus has three significant components. Excavation and closure impacts will scale with the decay heat production if the excavation requirements (e.g., number, length or spacing of tunnels) are tied to a constraint on the heat per unit of tunnel length in the repository. If no such constraint exists, then this impact would likely scale with mass or number of packages to be disposed. The operational active cooling impact, if present for a disposal concept, clearly scales with heat generation. Waste packaging impacts increase with the number of packages, but the waste packing density inside a single package may in turn be coupled to the decay heat of the waste. Therefore, it may be appropriate to modify the geologic disposal impact factors estimated below (kg CO_2/MTIHM) to account for any packaging or emplacement constraints associated with decay heat.

6.3.2.1 EXCAVATION

Excavation energy use is estimated as shown in Table 6-6.

Table 6-6. Energy estimate for repository excavation and backfill

	value	Units	Source
Material extracted for YMP excavation	4.40E+06	m^3	[DOE 2002]
Density of material extracted[1]	3.00E+00	MT/m^3	[Pidwirny 2013]
MT of material extracted	1.32E+07	MT	calculated
Energy required per MT extracted[2]	4.45E-01	GJ/MT	[DOE 2002b]
Excavation energy for YMP	5.87E+06	GJ	calculated
Backfill energy for YMP[3]	5.87E+06	GJ	calculated
YMP capacity	70,000	MTIHM	
Normalized excavation energy and backfill	1.68E+02	GJ/MTIHM	calculated
1. Excavated rock density values from mining operations vary widely; a typical value for a western US coal mine is 2.4 MT/m^3 [EPA 1995]. A density of 3 MT/m^3 is chosen as a			

conservative estimate.
2. Energy intensity of a hypothetical underground room-and-pillar coal mine .
3. Backfilling of drifts with previously excavated material is assumed to be equal to the initial excavation energy.

According to [DOE 2002b], 78 percent of the energy used in the coal mining operation is obtained through electricity, with the remaining energy share obtained through distillate fuels. Using this breakdown and the carbon intensities given in Table A-1, the CO_2 emissions associated with this energy use are estimates as shown in Table 6-7.

Table 6-7. CO2 emissions estimate for repository excavation and backfill

	GJ/MTIHM	kg CO_2/GJ	kg CO_2/MTIHM
Total energy	1.68E+02		
Energy from electricity (78%)	1.31E+02	168	2.20E+04
Energy from distillate fuels (22%)	3.69E+01	79	2.92E+03
		Total CO_2	2.49E+04

6.3.2.2 REPOSITORY OPERATION

The greatest energy expenditure during repository operations is in the ventilation fans used for active cooling of the repository. [DOE 2001] determined 3 fans, operating at an annual cost of $7.3 million total, were required to cool the 70,000 MTIHM in Yucca Mountain. [DOE 2001] assumed an electricity cost of $0.1 per kWh. Under this assumption, energy consumption for active cooling is 376 GJ per MTIHM. [DOE 2002] states that electricity during repository operations will be provided through the Nevada Test Site electric power distribution system; applying the U.S. average carbon intensity (Table A-1) of 168 kg CO_2 per GJ for electricity, carbon emissions due to repository operation are estimated at 63,200 kg CO_2 per MTIHM.

6.3.2.3 WASTE PACKAGE FABRICATION AND MATERIAL INPUTS

Two major material inputs to the repository exist: (1) titanium drip shields and (2) waste packages. These two engineered barriers are intended confine the waste and to protect the waste package from contact with water.

The total estimated cost of fabricating drip shields and waste packages is obtained from [DOE 2008] at $21 billion ($30,000 per MTIHM for the YMP 70,000 MTIHM capacity). Using these figures, the carbon emissions for fabrication of the drip shields and waste packages is calculated as shown in Table 6-8. Because [DOE 2008] and [EIA 2011] combine the fabrication costs of drip shields and waste packages, the energy use for fabricating these cannot be disaggregated.

Table 6-8. CO_2 emissions for fabrication of waste packages and drip shields.

	GJ/$[1]	$/MTIHM	GJ/MTIHM	$kgCO_2/GJ$[2]	kg CO_2/MTIHM
Electricity	3.82E-04	2.92E+05	1.12E+02	168	1.87E+04
Natural Gas	5.27E-04	2.92E+05	1.54E+02	51	7.85E+03
Distillate Fuels	1.99E-06	2.92E+05	5.81E-01	79	4.59E+01
LPG	3.98E-06	2.92E+05	1.16E+00	51	7.90E+01
Coal	1.69E-05	2.92E+05	4.94E+00	89	4.39E+02
				Total	**2.71E+04**

1. From Table A-7 of Appendix A
2. From Table A-1 of Appendix A

[DOE 2002a] estimates a total of 3,087 MT of titanium is used to fabricate drip shields for the repository. Material inputs for waste packages are determined from [DOE 2002a] and [Armijo 2006]; see Tables A10 through A14 for data input for determining material inputs for waste packages and Figure A1 for a depiction of a typical waste package. Energy (GJ per kg material) and carbon (kg CO_2 per kg material) intensity coefficients are obtained from [Hammond 2011] and applied to obtain the final energy and carbon intensities given in Table 6-1.

6.4 OCCUPATIONAL RADIOLOGICAL DOSE

This section quantifies normalized collective radiological dose to workers from SNF and HLW storage, transportation, and disposal (STD) operations.

The STD operations deal with "packages" of nuclear material and the dose rate from a package is regulated. Thus, if there are more packages, there is correspondingly an increase in STD handling operations and collective dose to workers. Lower burnup fuels, for example, may require more packages per MW_eyr or per MTIHM. Additionally, even for a given burnup, limits on package sizes may require the use of more packages as might be the case for bulkier SNF such as that from HTGRs. However, radioactivity or decay heat (i.e. higher burnup and/or less cooling time) can also decrease the capacity per package to meet regulatory limits, which would result in additional packages. These factors should be taken into account if additional precision is needed in the STD dose estimate.

Estimating these doses was complicated by several other factors. For example, dose data for at-reactor storage facilities are not typically segregated from the dose associated with reactor operations and thus is likely already counted in the dose associated with reactor operations (Section 4.4). Further, one must know how the duration of the storage period in order to estimate the associated dose. Similarly, doses associated with transportation and storage operations are directly related to the number of handling and transport operations, the distance traveled, and the

population density along the route. Lastly, it is not known what, if any repackaging or other preparations may be needed prior to disposal.

Despite potentially large variability resulting from the factors discussed in the previous paragraph, values for normalized collective doses to workers were developed and are presented in Table 6-9 which is followed by discussion of the bases for the values. The remainder of this subsection provides additional qualitative insights regarding potential differences between these values and collective dose values for thorium-based fuel cycles.

Table 6-9. Radiological Worker Impacts for ST&D Operations for Repository Wastes[1]

Process	Storage[2]		Transportation	Disposal	
Technology	Wet	Dry		Near-Surface	Deep
Once-through LWR cycle (person-mSv/MTIHM)	3.8E+00	1.2E+00	5.9E-01	1.32E-01	1.06E+00
LWR fuel reprocessing cycle	NA	NA	5.9E-01	1.7E-01	1.27
LWR MOX fabrication cycle	NA	NA	5.9E-01	9.7E-02	3.07

1. Dose data given per unit electricity produced was converted to dose per MTIHM using 45.2 MWeyr/MTIHM, based a typical LWR fuel cycle (i.e. 50 GWtd/MTIHM and 33% thermal efficiency)
2. The storage doses are based on a 10-year period for wet storage and an 88-year period for dry storage (see sections 6.4.1 and 6.4.2)

There is relatively high uncertainty in the estimates shown above. For wet storage, it is unclear how much of the dose is already counted in reactor operations. For transportation, the data in [NEA 2000] is based on European data and there is considerable scatter. Additionally, transportation distances in Europe are much less than in the U.S. and adjusting for this is likely to increase these values. There also appears to be inconsistent accounting for transportation impacts across the various phases of the fuel cycle. For disposal, Greater-than-Class-C (GTCC) wastes and DU are not included.

6.4.1 WET INTERIM STORAGE

Obtaining worker doses for wet interim storage is complicated by the fact that individual or collective dose data for at-reactor storage facilities is not typically reported separately from the total dose associated with reactor operations (Section 4.4). This complication reduced the available database to that from standalone wet storage facilities of which there are few. For this report, collective doses to workers for the CLAB facility in Sweden were used because it is such an 'away-from-reactor' wet storage facility. A range of normalized occupational doses are provided (50 to 140 person-mSv/GWeyr) for years between 1986 to 1996, along with the normalized mass throughput of 25 MT/GWeyr [NEA 2000]. The average of the two extremes of the electricity-normalized collective doses was used for this report (95 person-mSv/GWeyr). Using these two sets of data, the mass-normalized radiological

metric is calculated in Table 6-10. Confidence that the value is representative of the actual impacts is medium because of the limited amount of data available.

Table 6-10. Occupational Dose and Production Data for a Wet Interim Storage Facility from 1986-1996

Parameter Description (Unit)	Parameter Value
Collective Worker Dose Normalized by Energy Production (person-mSv/GW_eyr)	95
Mass Throughput for 1 unit of GW_eyr Produced (MT SNF/ GW_eyr)	25
Radiological Occupational Metric (person-mSv/MTIHM)	3.8

6.4.2 DRY INTERIM STORAGE

A study of the impacts of moving SNF from wet pools to dry storage after various cooling times[11] provides collective doses to workers for representative PWRs and BWRs from dry storage including loading, annual maintenance and inspection, and construction during ISFSI operation [EPRI 2010]. Using the EPRI results to quantify a radiologic metric for worker impacts faces two major complications. The first complication is that the worker dose is composed of three components: cask loading and unloading, construction of new storage pads or vaults adjacent to existing ISFSIs, and ongoing inspection, surveillance, operations, and maintenance (ISOM). Based on industry experience, EPRI assumes that loading, unloading, and construction are one-time events for each cask. Based on the EPRI's assessment of industry experience, the collective dose from cask loading and unloading[12] is 8 person-mSv/cask and the dose from construction is 1.7 person-mSv/cask. In the EPRI baseline scenario a total of 10,822 casks contain 136,600 MTIHM of SNF or an average of 12.6 MTIHM/cask. However, the collective worker dose from ISOM activities is assumed to be incurred annually at a rate of 16.2 person-mSv/cask-yr. The relatively large worker dose from ISOM and it's being time-dependent means that it dominates the collective dose to workers from dry storage, and that the value of a mass-normalized metric increases as the assumed storage time increases. The second complication is that EPRI's baseline scenario assumes that the inventory of SNF in dry storage increases approximately linearly from 1400 casks to 10,822 casks between 2011 and 2050, and remains constant thereafter until 2099. As a

[11] It should be noted that, under the assumption of steady state operations being used in DOE/NE's fuel cycle option screening, there should be no need for dry interim storage because SNF and HLW would presumably be going to disposal sites at the same rate as spent fuel is received into storage. In a "steady-state" nuclear economy, there would be no driving force, except perhaps to allow further cooling, for utilities to incur the extra cost, worker dose, and liabilities to buy, load, and maintain dry casks as opposed to routinely shipping older from directly from the pool to the repository.

[12] EPRI did not account for cask unloading so it was assumed that worker dose from unloading was equal to the dose from loading.

consequence, the amount of SNF subject to ISOM activities (i.e., the normalization basis) and the worker dose therefrom is not constant until 2050. There are also other complications such as variations in SNF burnups and cask capacities but the magnitude of the variation is small compared to the impacts of collective doses to workers having multiple components, one of which is time dependent, and the time- and scenario-dependence of the SNF inventory in dry storage.

A simplified approach to obtaining a mass-normalized value for collective radiological impacts to workers from dry storage involves dividing the cumulative worker dose from the EPRI baseline scenario over 88 years (158,000 person-mSv) by the total amount of SNF in dry storage (136,600 MTIHM) to yield 1.16 person-mSv/MTIHM. This value is unique to the scenario analyzed and would change depending on the duration of the scenario, and assumptions concerning the rate at which the inventory changes. This result is summarized in Table 6-11.

Confidence that the value in Table 6-11 is representative of the expected radiological impact to workers is medium because the value is scenario-specific, but based on industry analysis and exposure data.

Table 6-11. Occupational Dose and Production Data for Dry SNF Interim Storage Based on Baseline Scenario in [EPRI 2010]

Parameter Description (Unit)	Parameter Value
Storage time assumed in scenario(years)	88
Cumulative Collective dose in scenario (person-mSv)	158,000
Steady-state SNF mass storage in Dry Interim Facility (MTIHM)	136,600
Radiological Occupational Metric (person-mSv/MTIHM)	1.16

6.4.3 TRANSPORT

Nuclear materials are transported between operations from uranium and thorium recovery through waste disposal. Noticeable doses (on the order of 20 mrem/yr) are received by drivers – who can be exposed for long times – and workers loading and unloading the nuclear materials because of their proximity to the packages. Additionally, the doses from transportation in the front end of the fuel cycle are not negligible [WNTI 2006] because of the relatively large amount of nuclear material involved and because these materials are not transported with the amount of radiation shielding used in the backend of the fuel cycle. The types of material transported may include natural uranium, natural thorium, uranium hexafluoride, and LLW in the front end of the fuel cycle; fresh nuclear fuel and LLW in the middle of the fuel cycle; and SNF, HLW, LLW, and recycled nuclear material in the backend of the fuel cycle.

Data on nuclear material transportation is sparse, and the data sets that exist are not complete, contain ranges of values, and at times combine occupational and

public dose. Data from [NEA 2000] was used as a basis for the collective dose to workers from transportation because it attempts to cover the whole once-through fuel cycle although the data are not complete, and there appear to be errors in some tables (e.g., UK dose "All" value should be 3.6, not 36 or the values comprising it do not add up), and transportation of uranium ore and yellowcake in the front end of the fuel cycle, and SNF or HLW to the repository was not included. The consensus range given in Table 16 of [NEA 2000], which is for a once-through fuel cycle was used: 0.005 – 0.022 person-mSv/MWe-yr. Taking the average yields 0.013 person-mSv/MW_eyr. Applying a conversion factor of 45.2 MW_eyr/MTIHM based on a typical LWR fuel cycle (50GW_d/MTIHM and 33% thermal efficiency) yields 5.9E-01 person-mSv/MTIHM(as shown in Table 6-12).

Table 6-12. Occupational Dose and Production Data of Fuel Cycle Material Transportation from 1986-1996

Parameter Description (Unit)	Parameter Value
Collective Worker Dose Normalized by Energy Production (person-mSv/MW_eyr)	1.3E-02
Electricity per unit fuel mass (MW_eyr/MTIHM)	45.2
Radiological Occupational Metric (person-mSv/MTIHM)	5.9E-01

Transportation impacts for fuel cycles involving SNF reprocessing and MOX fuel fabrication were assumed to be the same as for the once-through cycle because the limited data base for transportation is not sufficient to support differentiation. There is relatively high uncertainty in the estimates shown above. For transportation, the data in [NEA 2000] is based on European data, and there is considerable scatter. Additionally, transportation distances in Europe are shorter than in the U.S. and adjusting for these distances is likely to increase these values. There also appears to be inconsistent accounting for transportation impacts across the various phases of the fuel cycle, e.g., for disposal, GTCC wastes and DU are not included.

6.4.4 NEAR-SURFACE/SHALLOW LAND DISPOSAL

Near-surface burial is assumed to be used for Class A, B, and C low level wastes (LLW). Collective radiological doses to workers values are based on the 2004 EIS of the Richland LLW Disposal Facility owned and operated by U.S. Ecology [Washington State 2004]. Operational capacity in units of volume of LLW disposed per year for the Richland facility was obtained from the NRC [NRC 2013]. During 2005-2008, the average annual LLW volumes[13] disposed were 1234 m^3/year (850,

[13] Sealed sources used for industrial and medical purposes are disposed at LLW facilities, but only "comprise of less than 1% by volume and activity of all LLRW" [LANL 2009]. Due to the low percent of volume and activity of sealed sources, and the fact that sealed sources are used in many industrial applications outside of nuclear power generations, the worker

704, 2738, and 645 m^3/year). The volume-normalized collective occupational dose at the LLW facility estimated below in Table 6-13.

Table 6-13. Radiological Worker Impacts for LLW Shallow Land Burial Operations at U.S. Ecology

Number of workers	28
Average worker dose (mSv)	0.96
Collective Dose (person-mSv)	26.9
Volume of LLW Disposed of (m^3)	1234
LLW Radiological Worker Metric (person-mSv/m^3 LLW disposed)	0.022

Converting the volume-normalized impact to a mass-normalized impact from specific fuel cycle facilities involves a multi-step process. First, the volume-normalized value of collective worker dose from LLW disposal in Table 6-13 was used to calculate a volume-normalized value of collective worker dose from disposal of LLW from LWRs. The LWR volume-normalized value was then converted to a mass-normalized value. Then, the normalized volume of waste from front-end and back-end fuel cycle operations from a variety of sources was combined with representative assumptions and the LLW worker metric in Table 6-10 to yield mass-normalized values for collective worker dose from disposal of LLW from these facilities. These steps are elaborated below.

First, a breakdown of types of dry and wet waste and the production of LLW by BWRs and PWRs was obtained from [Saling and Fentiman 2002] and is shown in Table 6-14.

Table 6-14. Annual LLW Volumes Produced by LWRs

LLW Type	1978-1981		1982-1985		1985-1986	
	PWR	BWR	PWR	BWR	PWR	BWR
Average LLW Volume (m^3 per year)						
Dry waste						
Compacted	136	405	180	296	122	222
Noncompacted	156	228	105	228	59	139
Filters	1	1	7	3	6	1
Subtotal	299	634	292	527	187	362
Wet Waste						
Resins	30	59	42	62	31	68
Sludges	0	157	7	170	11	123
Concentrates	113	130	35	50	23	48
Oils	--	--	8	25	8	31
Miscellaneous	--	--	3	1	4	6
Subtotal	143	347	96	309	78	276

doses attributed to disposing of this type of material is excluded from the quantified metric presented here.

Total Average Annual LLW Volume, m^3/yr	442	981	388	835	265	639
Source: (Saling and Fentiman 2002) Table 1.2 LLW from Nuclear Energy Plants (pg 8); Originally taken from Radioactive Waste						

Next, data in Table 6-13 are combined with net electricity generation from reactors in the respective time periods, obtained from [EIA 2012a], and with average discharge burn-up of reactors in the respective time periods, obtained from [EIA 2004]. The calculated electricity-normalized LLW volume is calculated to be 0.82 m^3/MTIHM as shown in Table 6-15.

Table 6-15. Electricity-Normalized LLW Volumes Generated by LWRs

Parameter	1978-1981		1982-1985		1985-1986	
	PWR	BWR	PWR	BWR	PWR	BWR
LLW Produced (m^3/yr) [from Table 6-14]	442	981	388	835	265	639
Electricity Production GW_ed/yr)[a]	10994		13415		16619	
Electricity-Normalized LLW Production from LWRs [m^3 LLW/ GW_ed]	0.129		0.091		0.054	
Average Burn-up from LWRs [GW_ed/MTIHM][b]	8.54		9.41		9.32	
Initial fuel content normalized LLW production from LWRs [m^3 LLW/MTIHM]	1.102		0.856		0.503	
Average initial fuel content normalized LLW production from LWRs [m^3 LLW/MTIHM]	0.82					

[a] Derived from [EIA 2012a]
[b] Derived from [EIA 2012a]. Assumes 33% thermal-to-electrical efficiency. Weighted based on initial uranium content for PWRs and BWRs.

For non-reactor fuel cycle operations, the volumes of LLW produced by uranium enrichment and uranium fuel fabrication from Saling and Fentiman (2002)[14]. These results are summarized in Table 6-16 which also includes the electricity-normalized volume of LLW from LWRs taken from Table 6-15. These values were converted to mass-normalized collective dose to workers from LLW disposal as described in Table 6-12.

[Carlsen 2013] gives a production of 7.57 MTDU per MTIHM for uranium enrichment at 4.2% U-235 product enrichment, 0.25% U-235 tails assay, and 0.711% U-235 feedstock enrichment. Collective doses to workers from disposition of depleted uranium from enrichment assume that the uranium is de-converted to uranium oxide which is disposed by near-surface burial. The bulk density of the uranium oxide is assume to be 3 g/cc which leads to a DU oxide volume of 2.52 m^3 DU per MTIHM and, when multiplied by 0.022 person-mSv/m^3 LLW a collective worker dose of 0.056 person-mSv/MTIHM.

[14] The waste disposal doses from mining/milling are not included because these doses are included in the worker doses we already have for the FEFC.

The sources and methods described in Sect. 6.4.5 were used to estimate that an LWR SNF fuel reprocessing plant would produce 37.8 LLW packages per MTIHM of SNF processed by the plant. Each LLW package is a 55-gallon (200 L) drum leading to a volume of 7.56 m^3 per MTIHM which, when multiplied by 0.022 person-mSv/m^3 LLW disposed of yields 0.17 person-mSv/MTIHM. Similarly, using the sources and methods described in Sect. 6.4.5, the radiological impact to workers from LLW from LWR MOX fuel fabrication that is disposed of in the near-surface is estimated to be 0.097 person-mSv/MTIHM as MOX.

Table 6-16. Mass-Normalized Radiological Worker Impacts for Near-Surface LLW Disposal from Individual Fuel Cycle Operations[1]

Operation	Value	Native Unit	Value	Normalized Unit
Enrichment:	3.11E-05	[m^3 / kg SWU][2]	4.3E-03	[person-mSv/ MTIHM]
Uranium Fuel Fabrication:	2.5	[m^3 / MTIHM] of 2-3% enriched LEU[3]	5.4E-02	[person-mSv/ MTIHM]
LWR Reactor Operation:	0.82	[m^3/ MTIHM]	1.8E-02	[person-mSv/MTIHM]
Depleted uranium disposal	2.52	[m^3 / MTIHM]	5.6E-02	[person-mSv/MTIHM]
Once-through fuel cycle total			1.32E-01	[person-mSv/MTIHM]
LWR Fuel Reprocessing	7.56	[m^3/MTIHM of SNF]	1.7E-01	[person-mSv/MTIHM]
LWR MOX Fuel Fabrication	0.44	[m^3/MTIHM as MOX]	9.7E-02	[person-mSv/MTIHM]

1. The normalized radiological impact incurred from a unit volume of LLW disposed is 0.022 person-mSv/ m^3 LLW. The value of 0.022 was multiplied by the values in the second column to produce the values found in 4th column.
2. To calculate the enrichment impacts: 1 MT SWU = 1 kiloSWU = 1000 kg SWU = 1.60E-01 MTIHM at LEU U235 wt% as 4.2%
3. Volume of LLW generated from fuel fabrication is expected to relatively be independent of LEU product enrichment.

6.4.5 DEEP GEOLOGICAL DISPOSAL

Deep geologic disposal metrics for direct disposal of PWR SNF were quantified based on [DOE 2002] and constitute estimates because a SNF/HLW repository has not yet operated. The metric value is based on the Supplemental EIS for Yucca Mountain, Table D-12 which gives a collective worker dose of 7,400,000 person-mrem (74,000 person-mSv) for the nominal 70,000 MTIHM that would have been disposed of in YM, or 1.06 person-mSv/MTIHM (as shown in Table 6-17).

Table 6-17. Radiological Worker Impacts for Deep Geological Disposal (SNF)

Parameter (unit)	Value
Collective Dose (person-mSv)	74,000
Mass Disposed in the Disposal Facility (MT SNF)	70,000
SNF Disposal Rad Worker Metric (person-mSv/MT SNF disposed)	1.06E

Reprocessing Waste Disposal Rad Worker Metric (person-mSv/MT SNF reprocessed	1.27
MOX Fuel Fabrication Waste Disposal Rad Worker Metric (person-mSv/MT MOX fuel fabricated	3.07

Deep geologic disposal metrics for fuel cycle options involving reprocessing of PWR SNF and fabrication of PWR MOX fuel were quantified based on information provided in recent estimates of the amount of various process and secondary wastes produced by a 800 MTIHM/yr reprocessing plant coupled with a MOX fabrication facility producing about 80 MT/yr of MOX fuel [Foare 2013, Phillips 2013]. It was assumed that the radiological impact to repository workers is proportional to the number of waste packages that have to be disposed. Plans for operating Yucca Mountain called for the SNF to be disposed of in a large sealed transportation and disposal (TAD) package that would hold 21 PWR fuel assemblies which means that 0.10 TAD is required per metric ton of heavy metal in the PWR SNF.

To provide a comparable value for reprocessing and MOX fuel fabrication wastes that require repository disposal, i.e., vitrified HLW, metal wastes (e.g., cladding), I-129 waste, and greater-than-Class C (transuranic) secondary wastes (deep geological disposal is one of the options under consideration by DOE for this class of wastes), it was assumed that the HLW and metal wastes would be in 2 ft diameter by 15 ft long cylindrical containers of which five fit into a TAD, and the remaining wastes would be contained in 55-gallon drums of which 25 fit into a TAD. The radiological impact to workers is obtained by dividing the fraction of a TAD required for reprocessing or MOX fabrication wastes requiring repository disposal by the fraction of a TAD required for disposal of SNF and multiplying the worker impact for disposal of SNF (1.06 person-mSv/MTIHM) in Table 6-14 by the result. Based on estimates from the sources cited in the previous paragraph, reprocessing wastes requiring repository disposal would require 0.12 TAD per MTIHM of fuel reprocessed which leads to a worker impact of 1.27 person-mSv per MTIHM of SNF reprocessed. PWR MOX fuel fabrication wastes would require 0.29 TAD per MTIHM as MOX leading to a worker impact of 3.07 person-mSv/MT of MOX produced.

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APPENDIX A – ENERGY AND CO2 CALCULATIONS

Table A1. Carbon intensities used in this document (Adapted from [EIA 2010])

Carrier	Factor	Unit
Gasoline	68	kg CO_2/GJ(t)
Distillate Fuel	79	kg CO_2/GJ(t)
Coal - Industrial Coking	89	kg CO_2/GJ(t)
Natural Gas	51	kg CO_2/GJ(t)
Coal - fired electricity (US avg)	272	kg CO_2/GJ(e)
Natural Gas fired elec. (US avg)	114	kg CO_2/GJ(e)
Electricity - US Grid Average	168	kg CO_2/GJ(e)
Electricity - Canada Grid Average	59	kg CO_2/GJ(e)
Electricity - Australia Grid Avg	248	kg CO_2/GJ(e)
Electricity - World Average	153	kg CO_2/GJ(e)

Carbon intensities given in Table A1 are used to estimate carbon emissions resulting from the consumption of direct energy carriers (e.g., through electricity consumption, or coal burning). Embodied energy and carbon intensities for material inputs are obtained from [Hammond 2011].

Table A2. CO_2 emissions calculations for UOX fuel fabrication

	Energy Use (GJ/MTIHM)		CO_2 Intensity (kg CO_2/GJ)	Overall Carrier CO_2 Emissions (kg CO_2/MTIHM)
	Electricity	Natural Gas		
Direct Energy Inputs[1]				
Electricity	2.12E+02		1.68E+02	3.56E+04
Natural Gas		7.31E+01	5.10E+01	3.73E+03
Embodied Energy Inputs[2]				
Material inputs - electricity	7.23E+02		1.68E+02	1.21E+05
Material inputs - natural gas		2.44E+03	5.10E+01	1.24E+05
Total CO_2 Emissions (kg CO_2/MTIHM)				**2.85E+05**

1. From (FBFC 2009) for the Romans facility over the 2005-2008 period. Average production of 524 MTIHM/yr.
2. From (Rotty 1975)

Table A3. CO_2 emissions calculations for MOX fuel fabrication

	Energy Use [GJ/MTIHM]		CO_2 Intensity (kg CO_2/GJ)	Overall Carrier CO_2 Emissions (kg CO_2/MTIHM)
	Electricity	Natural Gas		
Direct Energy Inputs[1]				
Electricity	1.06E+03		1.68E+02	1.78E+05
Natural Gas		1.32E-01	5.10E+01	6.74E+00
Embodied Energy Inputs[2]				
Material inputs - electricity	7.61E+02		1.68E+02	1.28E+05
Material inputs - natural gas		2.72E+03	5.10E+01	1.39E+05
Total CO2 Emissions (kg CO2/MTIHM)				**4.45E+05**

1. From (MELOX 2010) averages over 2008 to 2010 period, with average procuction of 129 MTIHM/yr
2. From (Rotty 1975)

Table A4. Energy use and CO_2 calculation for reactor construction

Commodity	Value	Units	Source	Energy Intensity Conversion Factor	Units	Source	Energy Intensity (GJ/GW_e)	Carbon Intensity Conversion Factor	Units	Source	Carbon Intensity ($kg\ CO2/GW_e$)
Direct Energy Inputs											
Electricity	3.52E+09	$(2005)/$GW_e$	[Dixon 2012]	1.20E-04	GJ/$ (2005)	[EIA 2012]	4.21E+05	1.68E+02	kg CO2/GJ	[EIA 2010]	7.08E+07
Natural Gas	3.52E+09	$(2005)/ GW_e	[Dixon 2012]	3.91E-04	GJ/$ (2005)	[EIA 2012]	1.38E+06	5.10E+01	kg CO2/GJ	[EIA 2010]	7.02E+07
Petroleum	3.52E+09	$(2005)/ GW_e	[Dixon 2012]	1.52E-03	GJ/$ (2005)	[EIA 2012]	5.34E+06	6.80E+01	kg CO2/GJ	[EIA 2010]	3.63E+08
						Total, Direct	**7.13E+06**			**Total, Direct**	**5.04E+08**
Material Inputs											
Aluminum	1.80E+04	kg/GW_e	[Shropshire 2009]	1.55E-01	GJ/kg	[Hammond 2011]	2.79E+03	8.24E+00	kg CO2/kg	[Hammond 2011]	1.48E+05
Brass	1.00E+04	kg/ GW_e	[Shropshire 2009]	4.40E-02	GJ/kg	[Hammond 2011]	4.40E+02	2.46E+00	kg CO2/kg	[Hammond 2011]	2.46E+04
Bronze	2.50E+04	kg/ GW_e	[Shropshire 2009]	6.90E-02	GJ/kg	[Hammond 2011]	1.73E+03	3.73E+00	kg CO2/kg	[Hammond 2011]	9.33E+04
Carbon steel	3.27E+07	kg/ GW_e	[Shropshire 2009]	3.54E-02	GJ/kg	[Hammond 2011]	1.16E+06	2.71E+00	kg CO2/kg	[Hammond 2011]	8.86E+07
Concrete	1.80E+08	kg/ GW_e	[Shropshire 2009]	1.11E-03	GJ/kg	[Hammond 2011]	2.00E+05	1.59E-01	kg CO2/kg	[Hammond 2011]	2.86E+07
Copper	6.94E+05	kg/ GW_e	[Shropshire 2009]	4.20E-02	GJ/kg	[Hammond 2011]	2.91E+04	2.60E+00	kg CO2/kg	[Hammond 2011]	1.80E+06
Galvanized iron	1.26E+06	kg/ GW_e	[Shropshire 2009]	2.50E-02	GJ/kg	[Hammond 2011]	3.14E+04	1.91E+00	kg CO2/kg	[Hammond 2011]	2.40E+06
Inconel	1.24E+05	kg/ GW_e	[Shropshire 2009]	1.70E-01	GJ/kg	[Hammond 2011]	2.10E+04	1.04E+01	kg CO2/kg	[Hammond 2011]	1.29E+06
Insulation	9.22E+05	kg/ GW_e	[Shropshire 2009]	4.50E-02	GJ/kg	[Hammond 2011]	4.15E+04	1.86E+00	kg CO2/kg	[Hammond 2011]	1.71E+06
Lead	4.60E+04	kg/ GW_e	[Shropshire 2009]	2.52E-02	GJ/kg	[Hammond 2011]	1.16E+03	1.57E+00	kg CO2/kg	[Hammond 2011]	7.22E+04
Nickel	1.00E+03	kg/ GW_e	[Shropshire 2009]	1.64E-01	GJ/kg	[Hammond 2011]	1.64E+02	3.00E+00	kg CO2/kg	[Hammond 2011]	3.00E+03
Paint	7.94E+04	kg/ GW_e	[Shropshire 2009]	7.00E-02	GJ/kg	[Hammond 2011]	5.56E+03	2.42E+00	kg CO2/kg	[Hammond 2011]	1.92E+05
						Total, Embodied	**1.49E+06**			**Total, Embodied**	**1.25E+08**
					Total (GJ/GW_e)		8.63E+06		**Total ($kg\ CO2/GW_e$)**		6.29E+08

Table A5. Energy use calculation for used fuel reprocessing

Commodity	Value	Units (per MTIHM)	Energy Intensity Conversion Factor	Units	Energy Intensity (GJ/MTIHM)
Direct Energy Inputs					
Electricity	4.83E+02	MWh	3.60E+00	GJ/MWh	1.74E+03
Natural Gas	2.24E+07	L	3.64E-05	GJ/L	8.15E+02
				Total, Direct	2.55E+03
Material Inputs					
Nitric Acid (HNO3)	3.80E+00	MT	6.19E-01	GJ/MT	2.35E+00
24% Formaldehyde (CH2O)	3.07E+00	MT	9.57E+00	GJ/MT	2.94E+01
Sodium Nitrite (NaNO2)	1.03E+00	MT	1.76E+03	GJ/MT	1.81E+03
Sodium Carbonate (Na2CO3)	5.18E-02	MT	3.76E+00	GJ/MT	1.95E-01
Tributyl Phosphate (TBP)	2.64E-02	MT	1.19E+02	GJ/MT	3.14E+00
35% Hydrazine (N2H4)	1.08E-01	MT	7.66E+01	GJ/MT	8.31E+00
Ammonia (NH3)	1.57E-03	MT	2.83E+01	GJ/MT	4.44E-02
				Total, Embodied	1.86E+03
				Final Energy Intensity	4.41E+03

Direct energy inputs and chemical inventory obtained from [Areva 2009].

Table A6. Carbon emission calculation for used fuel reprocessing

	Direct Energy Inputs		Material Inputs (GJ/MTIHM)									
Energy Carrier	Electricity (GJ/ MTIHM)	Natural Gas (GJ/ MTIHM)	HNO_3	CH_2O	$NaNO_2$	Na_2CO_3	TBP	N_2H_4	NH_3	Overall Carrier Energy Intensity (GJ/MTIHM)	CO_2 Intensity (kg CO_2/GJ)	Overall Carrier CO_2 Emissions (kg CO_2/MTIHM)
Electricity	1.74E+03	0.00E+00	2.65E-02	1.16E+01	7.16E+02	7.70E-02	1.52E-01	3.28E+00	3.36E-03	2.47E+03	1.68E+02	4.15E+05
Natural Gas	0.00E+00	8.15E+02	1.79E+00	1.74E+01	1.07E+03	1.16E-01	1.27E+00	4.93E+00	3.75E-02	1.92E+03	5.10E+01	9.77E+04
Coal	0.00E+00	0.00E+00	2.30E-01	1.24E-01	7.68E+00	8.26E-04	0.00E+00	3.52E-02	1.52E-03	8.07E+00	8.90E+01	7.18E+02
Distillate Fuel	0.00E+00	0.00E+00	2.30E-01	1.60E-01	9.87E+00	1.06E-03	0.00E+00	4.52E-02	1.52E-03	1.03E+01	7.90E+01	8.14E+02
Gasoline	0.00E+00	0.00E+00	6.97E-02	7.11E-02	4.39E+00	4.72E-04	1.72E+00	2.01E-02	4.51E-04	6.26E+00	6.80E+01	4.26E+02
									Total CO2 Emissions (kg CO2/MT UF)			**5.15E+05**

Table A7. Energy intensity coefficients for the Fabricated Metals industry (adapted from [EIA 2012])

	Energy Carriers									
	Electricity	Units	Natural Gas	Units	Distillate	Units	LPG	Units	Coal	Units
Fabricated Metals	4.05E-04	GJ/$(2005)	5.59E-04	GJ/$(2005)	2.11E-06	GJ/$(2005)	4.22E-06	GJ/$(2005)	1.79E-05	GJ/$(2005)
Fabricated Metals	3.82E-04	GJ/$(2007)	5.27E-04	GJ/$(2007)	1.99E-06	GJ/$(2007)	3.98E-06	GJ/$(2007)	1.69E-05	GJ/$(2007)

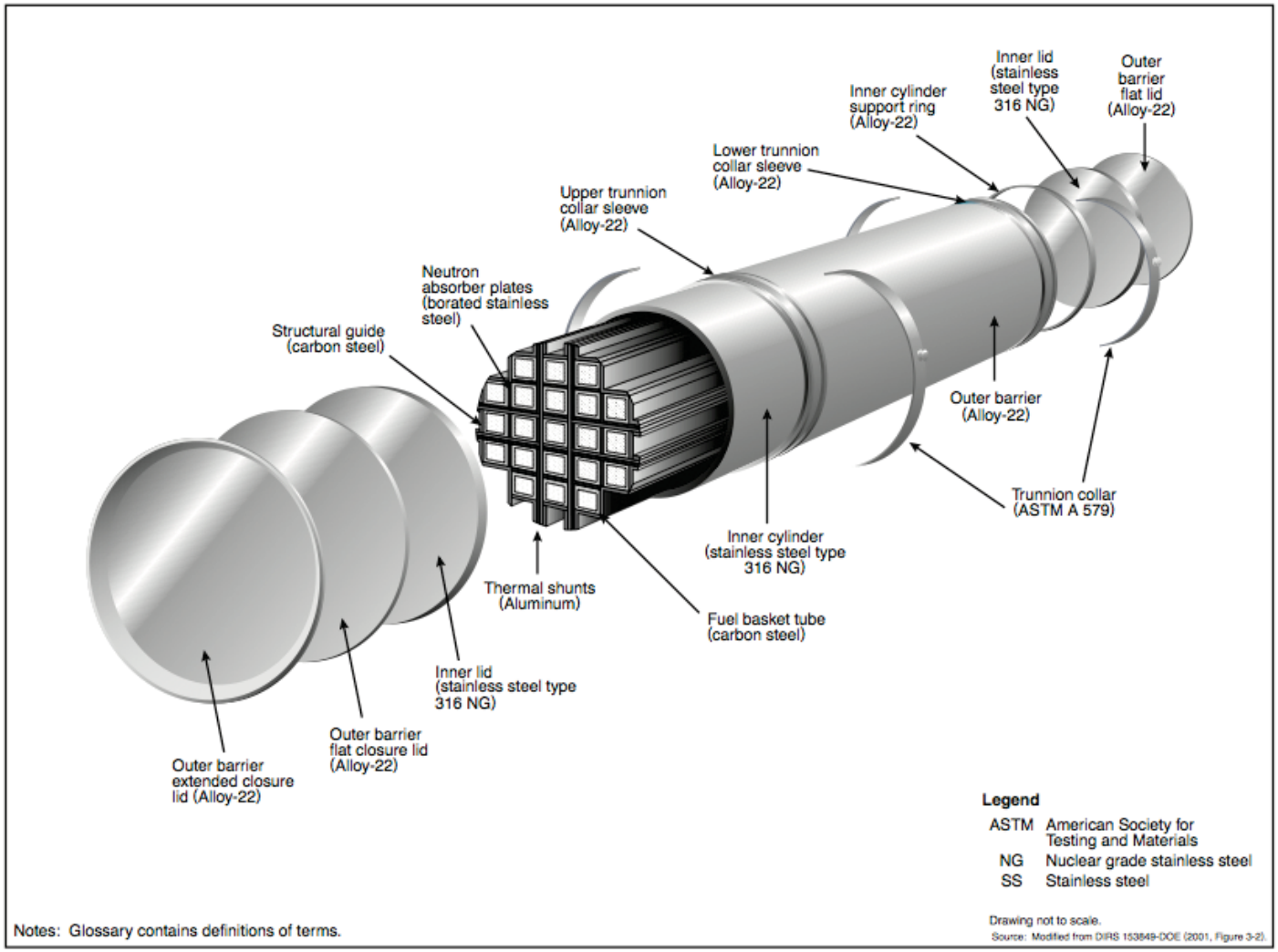
Table A8. Waste package and drip shield fabrication lifetime repository costs (in Millions of $(2007)) (adapted from Tables 2-3 through 2-5 in [DOE 2008])

Repository Phase	Costs $(2007)
Engineering, Procurement, and Construction Costs	240
Operations	12,580
Monitoring	7,630
Closure	0
Total	20,450

Table A9. Energy use and CO_2 emissions associated with waste package and drip shield fabrication (for final value appearing in Table 6-1)

Costs $(2007)	Energy Carrier	Energy Intensity [GJ/$(2007)]	Energy Use [GJ]	CO_2 Intensity [kg CO_2/GJ]	CO_2 Emitted [kg CO_2]
2.05E+10	Electricity	3.82E-04	7.81E+06	1.68E+02	1.31E+09
	Natural Gas	5.27E-04	1.08E+07	5.10E+01	5.49E+08
	Distillate Fuel	1.99E-06	4.07E+04	7.90E+01	3.21E+06
	LPG	3.98E-06	8.13E+04	6.80E+01	5.53E+06
	Coal	1.69E-05	3.46E+05	8.90E+01	3.08E+07
			Total CO_2 Emitted [kg CO_2]		1.90E+09
			Repository Capacity [MTIHM]		7.00E+04
			CO_2 Intensity [kg CO_2/MTIHM]		2.71E+04

Figure A1. Schematic of typical waste package design (Source: [DOE 2002])



Notes: Glossary contains definitions of terms.

Table A10. Data input for determination of materials for waste package fabrication (Sources: [DOE 2002a] and [Armijo 2006])

Component	L [cm] x W [cm] x T [cm]	Material	Number per Waste Package
Outer Cylinder	508 x 254 x 2	Alloy 22[a]	2
Inner Cylinder	498 x 254 x 5	Stainless Steel Type 316NG[b]	2
Outer Lid	380 x 198 x 2	Alloy 22[a]	1
Inner Lid	360 x 180 x 5	Stainless Steel Type 316NG[b]	1
Support Ring	484 x 10 x 2	Alloy 22[a]	1

[a] Density of Alloy 22 is taken as 8.72 g/cm^3; obtained by assuming composition as 22.5% Cr, 14.5% Mo, and 63% Ni. Energy and carbon intensities obtained using same composition from [Hammond 2011].
[b] Density of Stainless Steel Type 316NG as 7.82 g/cm^3; obtained by assuming composition as 18% Cr, 3% Mo, and 79% Fe. Energy and carbon intensities obtained for generic stainless steel from [Hammond 2011].

Average weight of a typical waste package is obtained from data in Table A11-A12. Weight of Alloy 22 and Stainless Steel Type 316NG is obtained from data in Table A6. Remainder of waste package weight is attributed to carbon steel, which is used to fabricate internal components of the waste packages.

Table A11. Breakdown of Waste Packages for 70,000 MTIHM (Source: Table 3-3 of [DOE 2002a])

Waste Package Design	Approximate Percentage of Waste Packages by Waste Package Design	Approximate Percentage of MTHM by Waste Package Design
21-PWR Absorber Plate	38%	55%
21-PWR Control Rod	1%	1%
12-PWR Long	2%	2%
44-BWR	25%	32%
24-BWR	1%	<1%
5-DHLW/DOE SNF Short[a]	14%	3%
5-DHLW/DOE SNF Long[a]	15%	4%
2-MCO/2-DHLW Long	1%	<1%
Naval SNF Short	2%	<1%
Naval SNF Long	1%	<1%

Table A12. Physical dimensions of commercial waste package designs (Source: Table 3-7 [DOE 2002a])

No.	Waste Package Design	Outer Diameter mm (in.)	Outer Length mm (in.)	Mass of Empty WP kg (lb)	Mass of Loaded WP kg (lb)
1	21-PWR Absorber Plate	1,644 (64.7)	5,165 (203.3)	26,000 (57,300)	42,300 (93,300)
2	21-PWR Control Rod	1,644 (64.7)	5,165 (203.3)	26,000 (57,300)	42,300 (93,300)
3	12-PWR Long	1,330 (52.4)	5,651 (222.5)	19,500 (43,000)	30,100 (66,400)
4	44-BWR	1,674 (65.9)	5,165 (203.3)	28,000 (61,700)	42,500 (93,700)
5	24-BWR	1,318 (51.9)	5,105 (201)	19,400 (42,800)	27,300 (60,200)

NOTES: Control rods do not add any mass to the package because they displace the mass of nonfuel components (e.g., existing control rods, in-core detectors) included in the fuel assembly mass. WP = waste package. Source: CRWMS M&O 2000au; BSC 2001n.

Table A13. Calculation of average weight of typical waste package (data sources: Table A11-12)

Waste Package Design	Mass of Empty WP [kg]	Percentage of WPs by WP design	Fractional Mass of WP [kg]
21-PWR Absorber Plate	2.60E+04	38%	9.88E+03
21-PWR Control Rod	2.60E+04	1%	2.60E+02
12-PWR Long	1.95E+04	2%	3.90E+02
44-BWR	2.80E+04	25%	7.00E+03
24-BWR	1.94E+04	1%	1.94E+02
		67%	**1.75E+03**
		Total Mass[a] [kg]	**2.65E+04**

[a.] Obtained by dividing total fractional mass by total percentage of waste packages considered. Excludes DOE SNF and Naval SNF waste packages, as these are meant for disposal of HLW, not commercial reactor fuel.

From Table A3, Alloy 22 and Stainless Steel account for 1.83E+04 kg of a typical waste package (total weight is 2.65E+04). The remainder of the waste package weight (8.18E+03 kg) is attributed to carbon steel, which is used to fabricate the internal components of the waste packages.

Table A14. Carbon intensity calculation of material inputs for waste package fabrication and drip shields

Material	**Weight [kg]**	**Carbon Intensity [kg CO_2/kg]**	**CO_2 Emitted [kg CO_2]**
Drip Shields			
Titanium	3.09E+06	2.94E+01	**9.08E+07**
	Repository Capacity [MTIHM]		**7.00E+04**
	CO_2 Intensity [kg CO_2/MTIHM]		**1.30E+03**
Waste Packages			
Alloy 22	5.84E+03	1.34E+01[a]	7.84E+04
Stainless Steel Type 316N	1.24E+04	6.15E+00[b]	7.65E+04
Carbon Steel	8.18E+03	2.71E+00	2.22E+04
	Total CO_2 Emitted per WP [kg CO_2/WP]		**1.77E+05**
	Total number of WPs[c]		**1.15E+04**
	Repository Capacity [MTIHM]		**7.00E+04**
	CO_2 Intensity [kg CO_2/MTIHM]		**2.91E+04**

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APPENDIX B – CO_2 FOR METAL FUEL FABRICATION AND ELECTROCHEMICAL REPROCESSING

Estimate of Energy and CO_2 Intensities for Metal Fuel Fabrication and Electro-Chemical Reprocessing

Metal Fuel Fabrication and Electrochemical Reprocessing: Energy Use and CO_2 Impact

Energy, material and commodity use data for these technologies is not available at a sufficient level of detail to permit the bottom-up energy and CO_2 intensity estimation approach applied elsewhere to be used. Therefore, a top-down method suitable for obtaining order-of-magnitude estimates is applied.

The top-down approach begins with capital and operating cost estimates taken from [Kim 2013]. It then applies industry-average energy intensities, in units of GJ of energy consumed per dollar of capital or operating expenditures, to these costs. To estimate the energy use associated with construction, the capital expenditures were multiplied by the EIA industry-average direct energy use intensities for industrial building construction [EIA 2012]. Energy embodied in construction materials is also significant; [UNCHS 1991] provides average scaling factors for building construction in units of Joules of energy embodied in materials per Joule of direct energy consumed.

Direct operational electricity use expenditures are provided in the source. These are divided by the average US cost of industrial electricity [EIA 2013] to obtain the direct electricity use. [Kim 2013] also provides the operational expenses associated with materials consumption. Material consumption contributes to energy use in the form of the energy embodied in the preparation/manufacture of the materials. EIA, EPA, NETL and others all publish data on the average energy use per dollar of product value for basic industries such as metals, chemicals, and concrete/building materials production. But a detailed breakdown of the types and quantities of materials used in the fabrication and reprocessing facilities is not available, so a further assumption is required. This assumption is that the energy embodied in materials consumed by both technologies is represented by the energy use of the chemicals basic industry sector. Energy intensity coefficients, in GJ of energy consumed per dollar of output of chemical products, are obtained from [Worrel 2000]. Finally, energy use is converted to CO_2 intensity using the same approach applied elsewhere in this report.

Table B-1. Metal Fuel and Electrochemical Reprocessing Plant Data from [Kim 2013]

Item	Unit	Value for Metal Fuel Fabrication Plant	Value for E-chem Reprocessing Plant
Plant Capacity	MTIHM/yr	38.6	38.6
Operating Lifetime	Yr	60	60
Overnight Capital Cost	$	5.51E8	3.68E8
Annual Operating Cost	$/yr	8.94E7	5.96E7

of which: materials	$/yr	5.54E7	3.70E7
of which: electricity	$/yr	6.26E6	4.17E6

Table B-2. Energy intensity data

Item	Unit	Value	Source
Electrical energy intensity, industrial facility construction	MJ/$ of capital cost	0.12	[2]
Thermal (assumed natural gas) energy intensity, industrial facility construction	MJ/$ of capital cost	1.91	[2]
Units of embodied energy used per unit of direct energy, construction sector	Joules/Joule	3	[3]
Cost of electricity, industrial sector	$/kWh	0.069	[4]
Energy use intensity, chemical production basic industrial sector	MJ/$ of product value	16.2	[5]
CO2 intensity, chemical production basic industrial sector	kg CO_2/MJ	54.9	[5]

Table B-3. Calculations and results

Item	Unit	Value for Metal Fuel Fabrication Plant	Value for E-chem Reprocessing Plant
Direct electrical energy use, construction	MJ	6.62E7	4.41E7
Direct thermal energy use, construction	MJ	1.05E9	7.02E8
Direct plus embodied electrical energy use, construction	MJ	2.65E8	1.76E8
Direct plus embodied thermal energy use, construction	MJ	4.21E9	2.81E9
Direct electrical energy use, operations	MJ/yr	2.52E7	1.68E7
Embodied energy use (materials consumption), operations	MJ/yr	8.99E8	5.99E8
Electrical energy use per unit of throughput	MJ/MTIHM	7.67E5	5.11E5
Thermal energy use per unit of throughput	MJ/MTIHM	1.82E6	1.21E6

Embodied energy use (materials consumption) per unit of throughput	MJ/MTIHM	2.32E7	1.56E7
CO2 emissions, electrical energy use	Kg CO_2/MTIHM	1.29E5	8.59E4
CO2 emissions, thermal energy use	Kg CO_2/MTIHM	9.27E4	6.18E4
CO2 emissions, energy embodied in operational materials use	Kg CO_2/MTIHM	1.28E6	8.25E5
Total CO2 emissions intensity	**Kg CO_2/MTIHM**	**1.50E6**	**1.00E6**

The comparable CO_2 intensity for mixed oxide fuel fabrication is 4.45E5 kg CO_2/MTIHM and for aqueous reprocessing 5.15E5 kg CO_2/MTIHM. Therefore, on a per MTIHM basis, the CO_2 emissions impact for metal fuel fabrication is approximately 3 times larger than that of MOX fuel fabrication. The electrochemical reprocessing CO_2 emissions impact is approximately 2 times higher than that of aqueous reprocessing.

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APPENDIX C –THORIUM VS. URANIUM STORAGE, TRANSPORTATION, AND DISPOSAL

Introduction

Documented experience with waste from thorium reactors has been limited; however, commercial power production from thorium reactors occurred in the 1960's through 1980's in the U.S. (e.g., Elk River, Peach Bottom, Indian Point, Fort St. Vrain, Shippingport), and a number of test reactors have operated on a smaller scale. At the Shippingport LWBR, spent fuel was shipped in 39 modified M-130 cylinders with an approximate volume of 14,000 L each, while the reactor produced about 0.243 $GW_e yr$ over the course of its lifetime [INEEL 2002]. The Ft. St. Vrain HTGR discharged 2,208 fuel blocks each having a volume of about 89 L over the course of 12-years of operation. Though Ft. St. Vrain was rated for 842 MW*thermal, it typically operated well below this over its 12 year lifespan [ORNL 1992].

Assertions have been made by thorium proponents of a potential volume reduction of high level waste and spent nuclear fuel of about half for Th-U PWRs. Independent reviews have expressed that there is indeed some waste volume reduction associated with thorium, but probably not 50%. The ten-year storage requirement has been roughly estimated to be about 4.6 core volumes for a uranium PWR and 4.19 core volumes for a uranium-thorium PWR, about a 10% reduction [Kasten 1998]. The impacts of the storage-transportation-disposal (STD) portion of the fuel cycle are likely to be dependent on the number of waste packages handled which depends on the volume of waste generated because the capacity of a package is limited. Thus, a smaller volume would imply lesser number of person-hours of exposure to handle the additional packages and vice-versa.

Storage

The duration of interim storage is not expected to differ significantly between thorium and uranium cycles. Thorium-uranium PWRs may achieve some reduction in interim storage volume due to slightly lower decay heat, corresponding to small savings in interim storage operational cost. Initial decay heat of Th-U fuel has been estimated to be about 72% of uranium fuel [Kasten 1998]; however, other factors often dictage how long spent nuclear fuel remains in storage, therefore the impact this would have on occupational dose is unclear. The change in radiological impact on the overall fuel cycle would likely be negligible with regard to the rest of a thorium-uranium fuel cycle, given the dominance of reactor contributions, but it might represent a significant impact if considering STD in isolation.

An issue with the interim storage of used thorium fuel concerns Pa-233, a relatively long-lived actinide ($t1/2$ ~27 days) which ultimately decays to U-233. Sufficient storage time, at least a year, must be allowed before reprocessing to maximize the U-233 recovered from the SNF [IAEA 2005].

Transportation

Radiological impacts from a single shipment of thorium fuel or waste are not expected to differ from the impacts from transporting uranium fuels or wastes because the

transportation casks must be designed to meet the same regulatory requirements. However, as discussed earlier, the radiological impacts should be proportional to the number of waste shipments (assuming the distance traveled is the same); thus, the radiological impacts should vary linearly with the generated waste volume.

Disposal

If the comparison is LEU SNF disposal vs. Th/U-233 SNF disposal then the activities are comparable for a long time. For a real system where reactors operating on LEU or Pu are needed to provide the U-233 feed for a Th/U-233 reactor, generalizations regarding toxicity are difficult. Thorium also provides a less mobile, more stable waste form in environments where oxidizing groundwater is present. In light-water reactors, radiotoxicities are projected to be lower for thorium fuel cycles, although the degree of reduction varies over time [Kazimi 1999]. Calculations project at least a 30-fold decrease in the normalized radiotoxicity of Th-U MOX FBRs or MSRs wastes when compared to a comparable FBR using U-Pu MOX after ten years [Nuttin 2004] but the radiotoxicity at 10 years would not seem relevant given that a repository will not yet be closed. This degree of reduction is expected to become less significant over time, as U-233 becomes the dominant source of radiotoxicity.

Limitations and Conclusion

Data concerning thorium waste form stability is limited for irradiated samples; behavior is mostly projected from non-irradiated behavior. As such, the effects of irradiation on long-term stability are not certain. The chemical and thermodynamic behavior of thorium's hydroxide and carbonate forms also requires more research [Kazimi 1999]. The correlation between radiotoxicity and dose ranges from fair for oxidizing repository conditions to poor for reducing repository conditions, so the usefulness of this parameter is marginal at best. Strictly comparing the STD impacts of uranium fuel cycles to those of Th or Th-U cycles, differences resulting from radiotoxicity and volume generation characteristics may be non-negligible.

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