Evaluating Environmental, Health and Safety Impacts from Two Nuclear Fuel Cycles: A Comparative Analysis of Once-Through Uranium Use and Plutonium Recycle in Light Water Reactors

By

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To Mom and Dad, infinitely supportive

and

To Duncan, my best friend

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NOMENCLATURE

Abbreviation/Symbol	Description
ACGIH	(U.S.) American Conference of Governmental Industrial Hygienists
ACNW	(U.S.) Advisory Committee on Nuclear Waste
AFCI	Advanced Fuel Cycle Initiative
ALARA	as-low-as-reasonably-achievable
ANL	(U.S.) Argonne National Laboratory
AR-ISFSI	at-reactor independent spent fuel storage installations
ASTM	(U.S.) American Society for Testing and Materials
ATSDR	(U.S.) Agency for Toxic Substances and Disease Registry
BLEU	blended low-enriched uranium
BRC	(U.S.) Blue Ribbon Commission's on America's Nuclear Future
BWR	boiling water reactor
CEDE	committed effective dose equivalents
CNSC	Canadian Nuclear Safety Commission
DOE	(U.S.) Department of Energy
DOE-EM	(U.S.) Department of Energy - Office of Environmental Management
DPC	dual purpose cask (type of dry cask)
DSC	dry storage cask
DU	depleted uranium
DUF_6	depleted uranium hexafluoride
DU-Oxides	depleted uranium-oxides
EC	European Commission
EDE	effective dose equivalents
EH&S	environmental, health & safety
EIA	(U.S.) Energy Information Administration
EIS	environmental impact statement
ENU	enriched natural uranium
EPA	(U.S.) Environmental Protection Agency
EPRI	Electric Power Research Institute
ERA	Energy Resources of Australia Ltd.
ERU	enriched reprocessed uranium
FA	fuel assemblies
FEFC	front-end of the (nuclear) fuel cycle
FWMS	federal waste management system
GAO	(U.S.) Government Accountability Office
GEMSO	Generic Environmental Statement of the Use of Recycle Plutonium in Mixed Oxide Fuel in Light Water Reactors
GNEP	Global Nuclear Energy Partnership
GTCC	greater-than-class C (type of radioactive waste)
GWdthermal	gigaWatt-thermal day (unit of thermal energy)
GWe	gigaWatt-electric

GWe-yr	gigaWatt-electric year (unit of electrical energy)
HEU	highly-enriched uranium
HF	hydrogen fluoride (also hydrofluoric acid)
HIC	high-integrity container
HLW	high-level (radioactive) waste
HSM	horizontal storage module
IAEA	International Atomic Energy Agency
ICRP	International Commission on Radiological Protection
IFNEC	International Framework for Nuclear Energy Cooperation
IIFP	International Isotopes Fluorine Products, Inc.
INEEL	(U.S.) Idaho National Engineering and Environmental Laboratory (outdated name of INL)
INL	(U.S.) Idaho National Laboratory (current name of this facility)
ISFSI	independent spent fuel storage installations
ISL	in-situ leach (uranium-recovery technology; see ISR)
ISR	in-situ recovery (uranium-recovery technology; see ISL)
KY	(U.S. state of) Kentucky
LCF	latent cancer fatality
LES	Louisiana Enrichment Services
LEU	low-enriched uranium
LLW	low-level (radioactive) waste
LNT	linear no-threshold
LWR	light water reactor
m^3	cubic meters
MIT	Massachusetts Institute of Technology
MLLW	mixed low-level (radioactive) waste
MOC	modified-open (nuclear fuel) cycle
MOX	mixed-oxide (type of reactor fuel using Pu and U)
MPC	multi-purpose cask (type of dry cask)
mrem	milli-Roentgen equivalent man (unit of EDE)
MRS	monitored retrievable storage (type of UNF storage facility, see ISFSI)
MRSRC	(U.S.) Monitored Retrievable Storage Review Commission
MSHA	(U.S.) Mine Safety and Health Administration
mSv	milliSievert (unit of EDE)
MT	metric ton(s)
MTBLEU	metric tons blended-low enriched uranium
MTDU	metric tons depleted urnaium
MTENU	metric tons enriched natural uranium
MTERU	metric tons enriched reprocessed uranium
MTHLW	metric tons high-level waste
MTHM	metric tons heavy metal
MTMOX	metric tons mixed-oxide
MTNU	metric tons natural uranium
MTRepU	metric tons reprocessed uranium
MTUNF	metric tons used nuclear fuel
MWe	megaWatt-electric

MWe-yr	megaWatt-electric year (unit of electrical energy)
NAS	(U.S.) National Academy of Sciences
NEA	Nuclear Energy Agency
NEI	(U.S.) Nuclear Energy Institute
NFC	nuclear fuel cycle
NFS	Nuclear Fuel Services
NIST	(U.S.) National Institute of Standards and Technology
NM	(U.S. state of) New Mexico
NNSA	(U.S.) Nuclear National Security Administration
NPP	nuclear power plant
NRC	(U.S.) Nuclear Regulatory Commission
NRG	(Netherlands) Nuclear Research and Consultancy Group
NU	natural uranium
NWTRB	(U.S.) Nuclear Waste Technical Review Board
OH	(U.S. state of) Ohio
OP	open pit (uranium) mining
ORNL	(U.S.) Oak Ridge National Laboratory
OSHA	(U.S.) Occupational Safety and Health Administration
OTC	once-through (nuclear fuel) cycle
PDF	probability distribution function
person-mSv	person-milliSievert
Pu	plutonium
Pu-238	plutonium-238
Pu-239	plutonium-239
Pu-240	plutonium-240
Pu-241	plutonium-241
PUREX	plutonium-uranium reduction and extraction (reprocessing technology)
PWR	pressurized water reactor
rad	radiation absorbed dose (unit of radiation exposure)
RD&D	research, development, and demonstration
Re-ENU	re-enriched natural uranium
rem	Roentgen equivalent man (unit of EDE)
RepU	reprocessed uranium
SCDHEC	South Carolina Department of Health and Environmental Control
SEU	slightly-enriched uranium
SFP	spent fuel pool
SLB	shallow-land burial disposal (type of disposal for some radioactive waste)
SOP	storage overpack
SPC	single purpose cask (type of dry cask)
SRNL	(U.S.) Savannah River National Laboratory
Sv	Sievert (unit of EDE)
SWU	separative work units
TAD	transportation-aging-disposal (type of dry cask)
TEDE	total effective dose equivalents
TENORM	technologically enhanced naturally occurring radioactive material
TOP	transport overpack

TRU	transuranic (radioactively contaminated waste)
U	uranium
U.S.	United States (of America)
U-232	uranium-232
U-233	uranium-233
U-235	uranium-235
U-236	uranium-236
U-238	uranium-238
U3O8	(one form of) uranium oxide
UC-C	universal canister (for holding vitrified HLW)
UC-V	universal canister (for holding compacted metal waste considered GTCC)
UF6	uranium hexafluoride
UG	underground (uranium) mining
UNF	used nuclear fuel
UNSCEAR	United Nations Scientific Committee on the Effects of Atomic Radiation
UO2	(one form of) uranium oxide
UOX	uranium-oxide (type of reactor fuel)
U-recovery	uranium-recovery
VEPCO	Virginia Electric Power Company
VSC	vertical storage casks
VU	Vanderbilt University
WASDH	Washington State Department of Health
WEC	Westinghouse Electric Company
WH	Westinghouse
WNA	World Nuclear Association
WSRC	Washington Savannah River Company
yr	year

CHAPTER 1

INTRODUCTION

1.1. Overview

In the face of a large number of retiring nuclear power plants with limited resources to support an aging infrastructure, the nuclear industry is assessing its own means of prioritizing investments into future options to maintain the competiveness of nuclear energy [*Mourogov et al.*, 2002; *Jewell*, 2011]. This includes evaluating several advanced technologies that are most promising in terms of multiple performance metrics and evaluation criteria to assess the viability of a successful research, development, and demonstration (RD&D) program [*Stamford & Azapagic*, 2011]. Establishing an effective RD&D program for nuclear fuel cycle (NFC) technologies will benefit from transparent and consistent assessment methods to evaluate and document the advantages and disadvantages of NFC options.

The determination of the most viable options is often uprooted by scrutiny of other issues that are important to a diverse set of stakeholders of nuclear energy [*Pierpoint*, 2011]. Past arguments for various fuel cycle options have been placed into terms of financial costs [*EPRI*, 2009b; *Shropshire et al.*, 2009], but with rising concerns of potential impacts to human health and the environment, cost has become only one of many metrics that are considered necessary for informing decision-making. Limited studies evaluating relative performance in terms of safety, protection of human health, and mitigation of environmental consequences are often are omitted

from comparative studies because of the rationale that actions to meet regulations must be fulfilled in order for the system to be licensed and continue operations [*Piet et al.*, 2006a]. Positing that each promising advanced NFC option will perform in the limits of the regulations diminishes the ability to differentiate and prioritize options.

Not only are studies limited to the bases of comparison by available performance metrics, but also by the hypothetical scenarios that represent the advanced NFC as implemented in the future. The way in which most comparative nuclear energy options studies are performed to understand potential impacts lie on two sides of the spectrum with respect to the level of detail: from site-specific to a generic total fuel cycle treatment [*IAEA*, 1996; *UNSCEAR*, 2008], both within simplifying boundaries of a steady-state scenario of the evaluated NFC. Performed more infrequently are dynamic studies that estimate impacts when transitioning between NFCs [*NRG*, 2010; *Feng et al.*, 2012]; the infrequent performance of dynamic/transition analyses is most likely caused by the good understanding of the end goal, but uncertainty concerning the journey to get there [*EPRI*, 2012]. Transient periods of time introduce complexities when implementing an advanced fuel cycle and potentially unique considerations for RD&D needs that are not as transparent when an advanced NFC option is fully deployed and operating at steady-state [*NEA*, 2012a].

Examining potential transitions between NFC options is important because an "evolutionary and progressive pathway is likely to be more realistic than a revolutionary pathway that attempts to simultaneously solve all real or perceived fuel cycle issues with advanced technologies [*EPRI*, 2010a]." As the modified-open nuclear fuel cycle (MOC) is itself an advanced NFC, it can also

be viewed that the MOC is a logical step to be considered in moving the U.S. towards an NFC incorporating a fast reactor technology and/or a more comprehensive recycling scheme. The MOC consists of the once-through nuclear fuel cycle (OTC) as currently implemented in the United States and then adding a hypothetical transition from the OTC to recycling plutonium (Pu) and reprocessed uranium (RepU). The MOC is situated in the direct path to many advanced fuel cycle options that are postulated and warrants further investigation of the incremental risks or benefits that may be incurred during this initial transition.

1.2. Structure of Dissertation

The work presented in this dissertation represents a systems-level approach to investigate potential net impacts with respect to human health and the environment associated with transitioning to the MOC for the U.S. In Chapter 2, an updated systems-level conceptual model of the OTC is presented to more accurately portray the OTC as currently implemented in the U.S. The conceptual model is the basis for estimating the worker collective doses at each operational stage, and the first demonstration of a quantitative comparative radiological impact assessment from expected normal operations is presented. In the course of evaluating worker collective dose associated with modern OTC practices, it was found that the relative contributions from the two grouped operations (front-end operations for preparing reactor fuel and reactor operations) were substantially different from historical data and conventional wisdom. As a bookend to Chapter 2, a summary is provided that describes the nature of the differences and factors that led to these differences. Detailed information of the work as part of the published journal article based off of this corollary work is included as an Appendix (C).

In Chapter 3, the study of worker collective doses from the phased introduction of reprocessing in the MOC scenario, and is presented similarly to the results in Chapter 2. MOC performance was also estimated by evaluating the radioactive waste generated that can be disposed and managed through known disposal practices in shallow-land burial. Relative to the OTC, MOC performance with respect to worker collective dose was not discernibly different; while the volume of radioactive waste generated decreased. It was found that although the sheer volume of radioactive waste avoided is large, the waste disposition pathway is known for the majority of this waste. The radioactive waste that requires disposal at a licensed off-site facility is examined in closer detail. The verification process for this study's comparative impacts of worker collective doses elucidates the dependence of net radiological impacts to workers to fuel-type use. This verification exercise then leads to concluding remarks that fuel-use proportions employed at the end of the hypothetical advanced NFC scenario within the reactor fleet can determine what level of analysis may be required to estimate the net impacts that may be incurred from an advanced NFC.

In Chapter 4, a study of potential worker collective doses incurred from carrying out the strategy to manage and dispose of used nuclear fuel outlined by the U.S. Department of Energy (DOE) as part of a comprehensive federal waste management system (FWMS) is discussed. It was estimated that the worker collective dose from repository operations leads to the large part of the radiological impact of the new FWMS. The contribution to worker collective dose was compared to that of the contemporary OTC quantitative model presented in Chapter 2. The additional worker collective dose contributed by FWMS activities is small and when the contributions from each grouped operation of the OTC are renormalized, the FWMS ranges annually from 4-8%.

Finally, Chapter 5 offers ideas for future work and provides a summary of the findings of this dissertation.

CHAPTER 2

AN UPDATED OTC MODEL AND ESTIMATED WORKER COLLECTIVE DOSES

2.1. Introduction

One of the earliest studies that recognize the importance of estimating potential impacts related to commercial nuclear power, and within each stage of the NFC, is the "classic" study that focuses on the environmental, health, and safety (EH&S) impacts of the OTC: the WASH-1248 report [*AEC,* 1974]. EH&S impacts are not just evaluated according to particular reactor sites, WASH-1248 extends the bounds of the study to the comprehensive set of operations that constitute the OTC (and the MOC). A NFC is composed of a sequence of facilities producing, processing, or using nuclear materials to accomplish some purpose – in this case commercial nuclear power[1]. The facilities are typically connected by transportation of nuclear materials because the facilities are usually not collocated. The facilities constitute distinct operations that are referred to throughout this thesis. Within WASH-1248 study, eight facilities constitute the OTC: (1) mining, (2) milling, (3) conversion, (4) enrichment, (5) fuel fabrication, (6) fuel irradiation in LWRs, (7) wet interim storage in spent fuel pools, (8) disposal of used nuclear fuel (UNF).

[1] While NFCs can have various purposes (e.g., power production, medical isotope or nuclear weapons production), the only purpose considered herein is for the use of nuclear material for production of electricity by nuclear reactors (commercial nuclear power).

Since the publication of WASH-1248 the OTC in the U.S. has evolved technologically while simultaneously including ad-hoc adjustments that have been made in response to U.S. governmental programmatic changes. Only a few studies have included considerations of the new methods for fuel supply and the waste disposal issues that have evolved as a result of a series of recent events and policy decisions, the new processes include: (1) re-enrichment of depleted uranium hexafluoride (DUF_6) tails to produce re-enriched natural uranium (Re-ENU) [*GAO,* 2011; *WNN,* 2013] (2) deconversion of DUF_6 to a stable oxide form (depleted uranium-oxides) [*DOE,* 2008b; *ANL,* 2012b], (3) downblending of weapons-grade uranium to form another source of reactor fuel, blended low-enriched uranium (BLEU) [*NNSA,* 2007; *DOE,* 2008c] and (4) dry interim storage of UNF [*VEPCO,* 2002].

Neglecting new operations that have become important in the U.S. is problematic due to the fact that the OTC is serving as the reference case to measure relative future impacts related to future advanced NFCs. By nature of studying impacts of the entire nuclear fuel cycle, excluding these four operations would be inconsistent with the objectives of these comparative studies. In order to accurately portray estimated impacts for the OTC, a contemporary model has been developed and a brief description of each of the four new operations is provided in the following section.

One type of performance metric that is given much attention is the radiological impacts by nuclear power production [*Frischknecht et al.,* 2000; *UNSCEAR,* 2008]. Radiological impacts from normal operations of NFC facilities can be estimated in several ways for comparing NFC options [*IAEA,* 1996]. A review of the types of performance metrics that intend to capture the radiological impact incurred from normal operations of the OTC is presented; how worker

collective dose came to be the metric used here to represent the fuel cycle safety strategic criteria is also discussed.

Finally, worker collective doses are estimated for a simplified energy demand scenario for the U.S. light water reactor (LWR) fleet for a window of 50 years in the future using a new model that includes the four new processes and traditional OTC operations that support nuclear energy production. Because this chapter's study of OTC worker collective doses can be divided as two overall sections of (1) the material flow analysis (2) quantifying performance metrics, the methodology and results are discussed topically according to this division.

2.1.1. Objectives of this Chapter

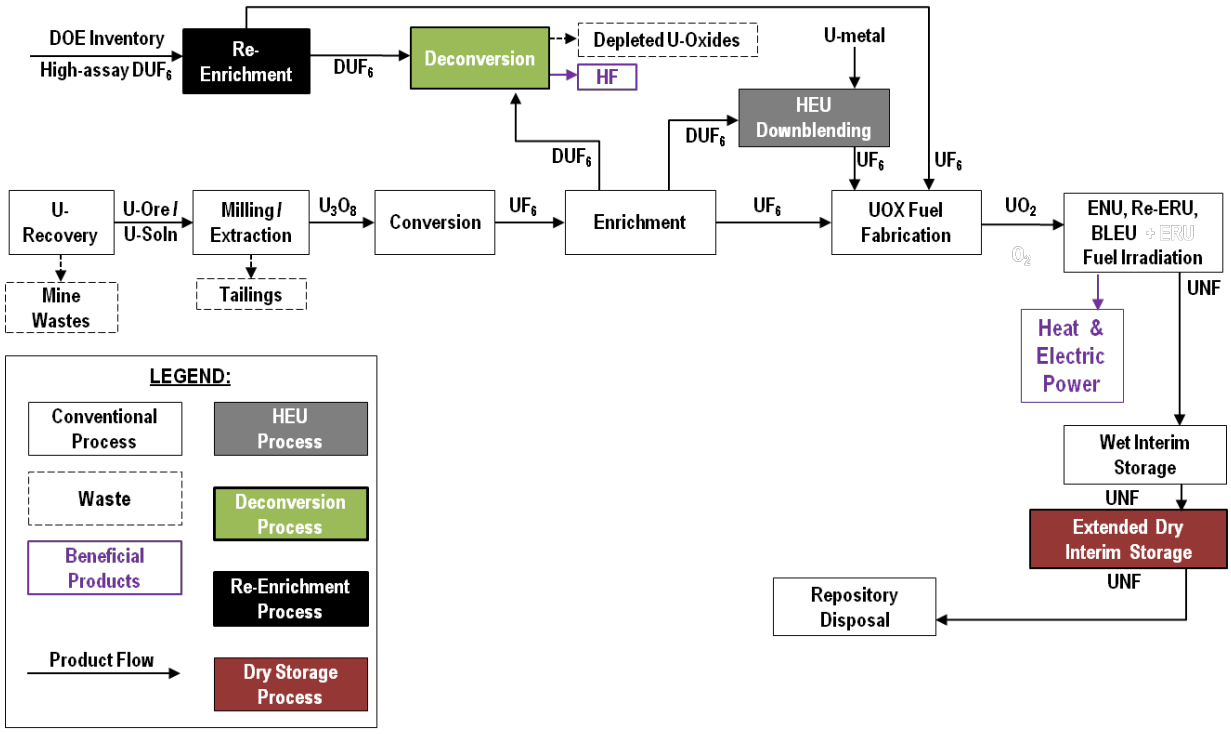
There are three overall objectives of this chapter:

- Determine if adequate data are available to comprehensively evaluate EH&S impacts of NFC options.
- Update the model the OTC EH&S impacts of a plausible simplified energy demand scenario for the next 50 years given new processes that have recently become important in the U.S.
- Model the OTC impacts to serve as a more accurate baseline for judging the performance of an advanced NFC that incorporates limited reprocessing of used nuclear fuel. This advanced NFC is referred to as the "modified-open nuclear fuel cycle" or "MOC" throughout this work.

2.1.2. Changes to the OTC and a Qualitative Systems Model

The objective of developing a revised conceptual qualitative OTC systems model was to serve as a basis for performing comparative radiological risk assessments of advanced nuclear fuel cycles leading to a quantitative evaluation of multiple NFC options. Developing the conceptual OTC model was divided into three tasks: (1) review and analysis of pertinent information; (2) delineate various contributors relating to different kinds of risks addressed for each fuel cycle phase; (3) develop the conceptual and qualitative system models for fuel cycles to be analyzed, as illustrated by the completion of a detailed qualitative model of the once-through fuel cycle (the reference case).

Pertinent OTC facilities, nuclear material mass flow modeling, and identifying EH&S impacts (informed by the risk-assessment methodology) were the bases for the qualitative OTC scenario and conceptual model for EH&S risks. Detailed descriptions of the NFC phases important in the U.S. are described in detail in EPRI (2012a), Smith et al. (2012a) and Smith et al. (2012b). Two important outcomes were: (1) an updated, detailed description of the U.S. OTC that reveals far greater complexity than is represented in other conceptual models, in terms of facilities and activities, shown graphically in Figure 2.1; and (2) identification and adoption of a suitable risk assessment template (that is discussed later). New operations are shown as highlighted boxes that demonstrably complicate the U.S. traditionally-defined system model that are shown in white boxes of Figure 2.1.



Notes: BLEU = blended low-enriched uranium (from downblending HEU and DUF_6), ENU = enriched natural uranium (processed through conventional front-end stages), HEU = highly-enriched uranium, HF = hydrofluoric acid, UF_6 = uranium hexafluoride, UO_2 = uranium oxide, Re-ENU = re-enriched natural uranium. Even though the "Repository Disposal" operations are shown here, the impacts are not estimated for this operation; this will be the focus of Chapter 4.

Figure 2.1 | Updated Baseline: The OTC Conceptual Systems Model

As noted above, the OTC has changed in response to U.S. governmental programmatic changes. Historically, the first change to the traditionally-defined U.S. OTC was the introduction of on-site dry cask interim storage of UNF in 1986 when the first dry storage installation at the Surry power plant in Virginia was licensed by NRC [*VEPCO*, 2002; *NRC*, 2012h]. Nuclear power plants (NPPs) determined that additional required capacity to store UNF was needed when federal milestones had not been met for developing and constructing a repository [*GAO*, 2012].

Downblending activities were initiated by DOE's "Megatons-to-Megawatts" program, enacted in 1993 by the signing of treaties between Russia and the U.S., enabling the downblending of 500

metric tons of Russian highly-enriched uranium (HEU, ~80-95 wt% U-235 fuel) with depleted uranium into blended-low-enriched uranium (BLEU) for U.S. commercial nuclear power plant (NPP) use. From the late 1990s to 2007, the U.S. has contributed to the BLEU inventory by downblending 100 MT HEU of the originally declared 175 MT HEU usable surplus [*NNSA,* 2007]. The economics and potential impacts to the uranium fuel market are published periodically by the U.S. DOE; this source meets 10% of the 100-reactor fleet fuel needs and DOE estimates few economic impacts are to be expected. This assessment has lead to continued commitments of HEU to the production of BLEU [*DOE,* 2008c].

DOE currently has an inventory of 700,000 metric tons[2] of DUF_6 tails from past defense and commercial activities [*GAO,* 2011; *ANL,* 2012a]. Re-enrichment of high-assay DUF_6 tails was originally a programmatic decision by DOE to continue Paducah's gaseous diffusion plant operations while maintaining an active part of the NFC infrastructure; meanwhile discussions transpired with other enrichment companies to operate the Kentucky site [*GAO,* 2011; *WNN,* 2013]. Beginning in May 2012 to May 2013, the contribution to the supply of LEU fuel by re-enrichment substituted the need for low-enriched uranium (LEU) fuel that had been sourced from mined uranium (either through conventional mining or in-situ leach/in-situ recovery (ISL/ISR) [*WNA,* 2012b, 2012c]).

During evaluations of ES&H impacts of disposition pathways for depleted uranium hexafluoride tails (DUF_6), it was concluded that a more stable form of depleted material as an oxide form was the preferred option for growing inventories of DUF_6 [*DOE,* 2008b; *ANL,* 2012b]. The NRC has

[2] This inventory (as of calendar year 2012) is comprised of two categories: (1) 600,000 tons of low-assay material, and (2) 100,000 MT DUF_6 (68,000 MTU) that contains "high-assays" of U-235 [*WNA,* 2012b, 2012c].

recently recognized that dealing with DUF_6 by deconversion to depleted uranium oxides is a reality by listing the deconversion phase as part of the fuel cycle and depicting it in public information [*NRC,* 2013b]. In 2010, the Portsmouth, Ohio deconversion facility began deconverting low-assay DUF_6 tails to depleted uranium oxides [*DOE,* 2009b; *NRC,* 2011b]. There is also an operational deconversion facility operational in Paducah, Kentucky. Recently, the International Isotopes Fluorine Products, Inc. (IIFP) deconversion plant in New Mexico was licensed by the NRC for construction and operation in 2012 [*IIFP,* 2009; *NRC,* 2013a].

Along with the extended fuel cycle of four additional processes, advances in drilling technology pioneered by the gas and oil industry have benefitted uranium recovery with the use of in-situ leach/in-situ recovery (ISL/ISR) technology and now smaller, less concentrated deposits of uranium are commercially viable [*Mudd,* 2001; *WNA,* 2012a].

Higher enrichment capacity and efficiencies are now capable through the use of centrifuge technology and even higher capacities will be available through laser isotope separation technology [*NRC,* 2012a; *WNA,* 2012c]. Both technologies have potential to impact the uranium fuel cycle supply with re-enrichment processes of DUF_6 past the May 2013 mark when Paducah will end re-enrichment operations.

2.1.3. Radiological Impact Performance Metrics under Fuel Cycle Safety

The topic of human health impacts potentially incurred from radiological sources terms is a wide and complex subject matter. Human health impacts from radioactive source terms are

extensively studied in, but not limited to, the nuclear, health physics, medical, and epidemiological fields. The regulatory-accepted predictive models of effects from ionizing radiation are based on deterministic high dose cases that are then linearly extrapolated, the model known as the linear no-threshold (LNT) model [*Brenner & Sachs,* 2006; *NAS,* 2006a].

Two schools of thought have emerged in the stochastic low-dose radiation area of study where potential radiological impacts are reported and predicted with concepts of dose versus risk. Dose is reported as total effective dose equivalents (TEDE) which is a biologically- and radiation-type-weighted absorbed energy from ionizing radiation specific to the human body [*McCollough & Schueler,* 2000]. TEDE is the sum of internal dose (committed EDE, CEDE) via ingestion or inhalation and external dose (referred to as "exposure" of radiation attenuated in the air medium) [*ICRP,* 2010b; *Knoll,* 2010]. Alternatively, risk can be presented as a probability, as in one in a million ($1x10^{-6}$), that represents the likelihood of incurring a specific end-state impact, such as induction of cancer, latent death, and other internal organ-specific effects that are targeted due to radionuclide form, geometry and chemistry [*NAS,* 2006a].

Risk is a useful metric under circumstances where individual chemicals and radionuclides are of concern and are singled out for site remediation [*EPA,*1989]; risk may also be reduced by eliminating pathways between the source and receptor [*Brown,* 2008]. A direct linear relationship can describe risk to dose [*EPA,* 1989]; thus it is possible to normalize risk by mass flow and energy production in a NFC comparative study; Rashad and Hammad (2000) and have applied mortality metrics to a comparative study of different energy sources and Machiels (1992) has looked at the normalized mortality and morbidity values relating to individual conventional

OTC phases. However, the definitive tie to a specific end-state limits modeling of all other potential end-states unless an extensive list of multiple types of end-state impacts is included. Producing many risk variant forms (that are not additive [*NAS,* 2006a]) would not be appropriate for a high-level decision framework unless some sort of valuation judgment based system is used to empirically combine or eliminate risk types (e.g., valuing the risk of death more than cancer, etc.).

Radiological human-health metrics typically attempt to capture potential impacts to two receptor groups – workers and members of the public. The hazard sources for the two receptor groups are the same, but the impacts are different as environmental transport processes dictate which radionuclides are carried to receptors and at what concentrations. From there, the physical form of radionuclides determines the exposure pathways into the human body and the chemical form controls the biological half-life (retention time in the body) and which organs uptake and utilize the radioactive chemical compound [*Eidson,* 1994].

Workers are commonly exposed to external radiation at all NFC operations while only a limited number of NFC phases provide opportunities of internal dose via the inhalation pathway and minimal events that lead to ingestion [*Stoetzel et al.,* 1981, 1982; *NAS,* 2011]. Inhalation of uranium products is most likely to occur in the front end stages when contact handled methods are used) [*Stoetzel et al.,* 1981, 1982]. When occupational radiological metrics are quantified, they are done in the context of TEDE, including CEDE when applicable [*IAEA,* 1996; *NEA,* 2000; *UNSCEAR,* 2008]. Specifically, worker collective TEDE dose is used and normalized by mass throughput of the facilities or energy produced by the reactor fleet. Individual worker dose

does not correlate directly with production due to dose management and radiation protection practices (e.g., workers can rotate through stations that provide higher doses; effectively increasing the collective dose but maintaining small average individual doses [*Stoetzel et al.*, 1981, 1982]. Understanding that this practice is a method to apply as-low-as-reasonably-achievable (ALARA) principles to an individual worker and not withstanding that it can be assumed that production does not change when the another shift of workers is present [*DOE*, 2009a], the collective dose does scale linearly with production for annual mass throughputs for NFC stages [*IAEA*, 1996; *NEA*, 2000; *UNSCEAR*, 2008] over short time periods. This is true even though the long-term trend has been towards lower individual TEDE, even with increasing nuclear power production.

In this same line of reasoning, all metrics described were developed under equilibrium assumptions that normal operations and required routine maintenance are included in a combined reported collective worker dose data set. This metric is based on real events and measured doses and is valuable for verifying actual impacts that are occurring in NFC facilities.

In order to avoid reliance on site-specific operational practices, it is preferred to relate potential radiological dose impacts to mass flow and isotopic content of the major material flows, regardless of the type of human receptor (i.e., worker or member of the public). Material streams of nuclear material, paired with relatively good performance of isotopic decay and burnup codes, has made the use of the radiotoxicity indicator a fairly common method to measure potential human radiological impacts. In the available literature, radiotoxicity metrics are divided into CEDE and external exposure and are not specific to one type of receptor group and more focused

on typical human characteristics based on International Commission on Radiological Protection (ICRP) models and internal dosimetry conversion factors [*Eastham et al.,* 2012; *Menzel & Harrison*, 2012].

Public radiological impacts are defined by the environmental transport mechanisms that result in likely exposure pathways. Exposure pathways are numerous and can range from ingestion of water and food sources to doses received during recreational water events [*EPA*, 1983a; *Menzel & Harrison,* 2012]. The process of modeling the public human health risk assessment becomes more and more site specific and is less susceptible to adapting to a general public-impacts radiological metric due to the required pathway analysis. A pathway analysis begins with a characterization of the source, such as a chemical, then a characterization of changes to the chemical and/or dilution as it moves through the environment, then a characterization of the mode of exposure (inhalation, ingestion, dermal) to the receptor, and finally a characterization of harm to the receptor. Each of these characterization steps may also involve several stages. While it would be ideal to carry out such detailed evaluations for every potential exposure or emission, adequate data and resources are not normally available to do so in a broad comparative study such as this one.

2.1.4. Worker Collective Dose Performance Metrics under Fuel Cycle Safety

The additive property of TEDE, and inherent descriptive nature of energy deposition by ionizing radiation, can be more easily melded to a comparative metric that is normalized to production of nuclear material streams (or electrical energy production) and compared to strict units of risk

(probability of a particular end-state). Conjointly, the method of reporting radiological impacts by the nuclear industry (NRC[3], IAEA, and DOE) is by dose, with units of [milliSievert, mSv] that establishes a literature basis for developing and quantifying risk metrics for the OTC model and by international operational experience for the MOC scenario.

When occupational radiological metrics are quantified, worker collective TEDE dose is used and normalized by mass throughput of the facilities or energy produced by the reactor fleet. Individual worker dose does not correlate directly with production due to dose management and radiation protection practices (as described in Section 2.1.3.) with current application of ALARA principles. Withstanding that it can be assumed that production does not change when another shift of workers is present, the collective dose does scale linearly with production for annual mass throughputs for NFC stages over short time periods; thus collective occupational dose with units of [person-mSv per normative throughput basis] will be used.

All metrics described were developed under assumptions that normal operations and required routine maintenance are included in a combined reported collective worker dose data set. This metric is based on real events and measured doses and is valuable for verifying actual impacts that are occurring in NFC facilities; but this metric is not without some limitations as discussed in Section 2.1.3.

At this time, calculating metrics for public and ecological impacts (collective or individual) relating to radioactive source terms are not within the scope of this study largely due to three

[3] NRC uses the units of "Roentgen equivalent man (rem)" instead of the SI units of Sievert (Sv). The units of rem and Sv can be converted by the following conversion equation: 1 rem = 0.01 Sv ; 100 Sv = 1 rem.

reasons (as expressed in Section 2.1.3): (1) the lack of a technical basis for choosing a representative site for all NFC process facilities to draw important parameters; (2) the exposed public population is not as closely related to the type(s) of NFC operation as the number of workers, and (3) worker exposure occurs contemporaneously with the NFC operation whereas important contributors to public exposure (e.g., long-lived effluents and waste disposal) can occur over long times.

2.2. Methods

A deterministic modeling approach was used for estimating OTC related worker collective doses are estimated using three high-level steps: (1) develop a hypothetical nuclear energy demand scenario, that is used to calculate material flows and analyze resultant waste at each NFC facility per year (Section 2.2.2); (2) calculate the normalized worker collective dose metrics which is the quotient of measured annual worker collective dose quantities and reported annual throughput of nuclear material (or for reactors, electrical energy produced, Section 2.2.2); and (3) scale normalized metrics by the material flow values found in step 1 for the simulation time frame. The methodology and results are discussed topically according to this division. The methodology section also provides background on the rationale for choosing modeling parameters for the OTC.

2.2.1. Modeling Platform and Calculational Tool

This initial demonstration of a comparative impact assessment of two NFCs was performed as a deterministic evaluation within a spreadsheet modeling platform (using Microsoft Excel). This means that deterministic data, or point values, were used to calculate other necessary parameters for normalized performance metrics, and scaled results of worker collective doses and volumes of radioactive wastes. These estimated parameters and results were then also presented in a deterministic manner. The deterministic values for input values were found by taking the average of input value options that were collected through literature reviews. The set of input values could have also been used as the training data points for calculating a probability distribution function (PDF) that could serve as the basis for a probabilistic evaluation (also referred to as a Monte Carlo Simulation). A probabilistic evaluation takes into account the probability profile of each input value by picking randomly an input value constrained by the assigned PDF. A probabilistic simulation runs the deterministic model several hundreds to thousands of times, each time picking a random point value from each input PDF. The PDF of the outcome results, in this case the worker collective doses, volumes of radioactive wastes, and the comparison between the OTC and MOC can then be further evaluated with additional statistical methods.

Because a probabilistic evaluation was not performed as part of this work, a high-level discussion of the uncertainty of input values is provided if multiple values were available from literature (often times, only one deterministic value was found in the literature). This discussion has been structured in the methods subsections such that the deterministic input value used is stated and then the available range of input values is provided in parentheses. The potential

impacts to the overall deterministic results by choosing higher or lower deterministic input values are difficult to assess without a formal sensitivity analysis. When wide ranges of input values are present, the potential to have a differing conclusion than the ones presented increase as the uncertainty increases with each input parameter and the number of input parameters with associated uncertainty bands (referred to as uncertainty propagation).

2.2.2. OTC Material Flow Analysis

Quantitative analysis of the worker collective dose was performed with essentially five modeling steps, focusing on: mass flows of uranium fuel supply, mass flows of DUF_6 management and UNF dry interim storage activities at reactor sites:

1. Establishing a simplified nuclear energy demand scenario for the 50-year simulation time period (2013 to 2063) by assuming a 1% growth per year.
2. Distributing the annual fuel requirements for the LWR fleet distinct supply sources (the extent of their use is described in the following sections):
    a. **BLEU:** Blended-low enriched uranium fuel, a product from downblending highly-enriched uranium (HEU) and DU;
    b. **Re-ENU:** Re-enriched natural uranium, higher-assay DUF_6 tails that are passed through the enrichment stage again to produce low-enriched fuel; and
    c. **ENU:** Enriched natural uranium, fuel sourced from conventional means of uranium mining, milling, conversion, enrichment, and fuel fabrication.

3. Apply deconversion and tails waste management parameters to re-enrichment and NU enrichment phases, while incorporating depletion of the DOE low-assay and high-assay DUF_6 stocks
4. Calculate the back-end interim independent spent fuel storage installations (ISFSIs) expansion and maintenance at existing reactor sites and scale risk metrics to number of dry casks produced each year
5. Scale the mass-normalized and energy-normalized performance metrics to find the total worker collective doses associated with the OTC

2.2.2.1. Energy Demand Growth and Reactor Annual Fueling Requirements

The U.S. electrical energy demand has increased around 1% per year since the year 2008, with ranges of 0.9 to 1.5% [*EIA*, 2012]. With continual NRC-approved upratings of LWR operating capacities, nuclear power plants have contributed 20% of the total U.S. electricity generated for over two decades [*NRC,* 2011a]; these rates are unchanged in this analysis. To account for the 1% growth each year LWRs are added while maintaining the same pressurized water reactor (PWR) to boiling waste reactor (BWR) technology ratio as the current LWR fleet.

Total fuel requirements for PWRs and BWRs each year are calculated with standard burnup, capacity factors, thermal efficiencies, cycle length and number of fuel batches. Several operating parameters were assumed to be the same for both PWRs and BWRs: capacity factor (90%), thermal efficiency (34%), electricity generation capacity per reactor (1200 MWe [range: 1000 to 1500 MWe]), and cycle length for each fuel batch (18 months [range: 12 to 24 months]). The

BWR and PWR parameters assumed to be dissimilar are enriched natural uranium (ENU) enrichment (4.35 ; 4.5 wt% U-235 [range: 4.25 to 5 wt%]), number of fuel batches (4 ; 3) and burnup levels (50 ; 55 GWdthermal/metric ton heavy metal [MTHM] with a range of 50 to 60 GWdthermal/MTHM). Annual refueling requirements[4] of a BWR and PWR are calculated to be 23.2 MTHM/yr and 21.1 MTHM/yr, respectively. The total mass of the cores were taken from a representative PWR and BWR[5].

2.2.2.2. Front-End Operations

Front-end operations are considered the consecutive fuel preparation stages before fuel is loaded into reactors. Specifically for this analysis, the stages preparing ENU, BLEU, and Re-ENU fuel are considered the front-end (as this definition is important when considering later on use of fuel that recycles Pu and RepU). The extent of U.S. and international operations are reflected as best as possible with current information available and a brief description is provided here.

From 1995 to 2012, the average and median percentage of domestically-supplied uranium for U.S. NPPs has been 8% (with a range of 2 to 12%). The U.S. Energy Information Administration (EIA) anticipates that 10% of the 2013 uranium NPP requirements of 50 million pounds of U_3O_8 will be met by U.S. uranium recovery operations [*EIA,* 2013]. Therefore, 10% will be used for 2013 to 2063 for U.S. self-supplied natural uranium (NU). Divisions between U-recovery

[4] If different input values (capacity factor, thermal efficiency, electricity generation capacity, cycle length, initial enrichment, number of batches, and burnup levels) were used within the available stated ranges, the mass of the annual fueling requirements would change and this change would affect all the results of worker collective doses and radioactive waste volumes because the normalization basis is by nuclear material mass processed each year. For reactor operations, the normalization basis is the electrical energy produced.

[5] A single reactor unit at the Sequoyah plant (PWR) and a single unit of Brown's Ferry (BWR) were used as the representative cases for sizing reactor units that provide baseload electricity. Both reactor sites are owned and operated by the Tennessee Valley Authority (TVA) [*Roddy et al.,* 1986, *Smith & Bevard,* 2013].

technologies were taken from the "Redbook" [*NEA*, 2011] for the U.S. and then globally and shown in Table 2.1. It was assumed that when NU was source from other countries other than the U.S., the percent of each technology shown in Table 2.1 was used.

Table 2.1 | Share by Mass Percent of Natural Uranium Recovery Technologies

Region	OP % in the Region (% Globally)	UG % in the Region (% Globally)	ISR/ISL % in the Region (% Globally)	% Region (% Globally)
U.S.	0% (0%)	25% (2.5%)	75% (7.5%)	100% (10%)
International	17.3% (15.6%)	37.5% (33.8%)	45.2% (40.7%)	100% (90%)
Global	(15.6%)	(36.3%)	(48.2%)	100% (100%)

Notes: All % describe U recovered for commercial use, all wt% U-235 are at natural concentrations of 0.71wt% U-235. OP = open pit uranium mining; UG = underground uranium mining; ISR/ISL = in-situ recovery/in-situ leach. "International" are the countries that provide NU other than the US. "Global" considers the amount of NU from both the U.S. and international partners.

The only conversion facility located within the U.S. is the Metropolis, Illinois conversion plant that is operated by Honeywell Specialty Chemicals. Within an IAEA (2005) report, it is stated that in the early 2000's Metropolis converted material for a total of 10 countries: 1) United States, 2) Belgium, 3) France, 4) Germany, 5) Japan, 6) Republic of Korea, 7) Slovenia, 8) Spain, 9) Sweden, and 10) Switzerland. Metropolis has a reported annual conversion capacity of 17,000 MTU until 2020 when there will be an increase in annual capacity to a total of 23,000 MTU/yr [*WNA*, 2014]. It is unknown what amount of material is processed by Metropolis for each country. For purposes of modeling efforts, it is assumed that Metropolis converts all U.S. material (if there is any the excess demand past 23,000 MTU/yr, it is assumed to be processed internationally).

The EIA lists the annual enrichment services purchased each year from U.S. located enrichment facilities and international-located services for the years of 1994 to 2012 [*EIA*, 2013]. The percent of the domestically provided enrichment services has been calculated for each of the 19

years. The average of the 19-year time frame is 32% of the total separative work units (SWU) services were provided (with a range of 9 to 82%); however, this information was not available during the time when modeling was being performed. The then assumed amount of domestically provided enrichment services was that U.S. facilities fulfilled the enrichment needs first and if any excess capacity was required, it was met by international partners with an assumed sufficient level of capacity. Within the EIA (2013) reference, the percent of domestically-supplied enrichment services is lower than the 32% average for years 2001 to 2011. This is due to lessened demand of enrichment services from downblended HEU and the partial dedication of Paducah's gaseous diffusion plant capacity to re-enrichment activities. In 2012, the Louisiana Enrichment Services (LES) company opened the centrifuge enrichment plant in New Mexico and the percent of domestically-supplied enrichment services increased for that year. The current LES annual capacity is 3,000 MT SWU/yr and is expected to increase to 5,700 MT SWU/yr in the year 2018. The U.S. domestic enrichment capacity is expected to increase due to plans made public by USEC to begin operations at the Piketon, Ohio centrifuge plant in the 2015 time period with an annual capacity of 3,800 MT SWU/year [*WNA*, 2014].

UO_2 fuel fabrication capacity in the U.S. exceeds the required annual amount of around 2,200 MTHM/yr with three facilities open and operating: (1) Areva NP facility in Richland, Washington [capacity: 700 MTHM/yr], (2) Global Nuclear Fuel facility in Wilmington, North Carolina [capacity: 1,200 MTHM/yr], (3) Westinghouse Electric facility in Columbia, South Carolina [capacity: 1,150 MTHM/yr]. The total U.S. capacity each year to produce UO2 fuel is 3,050 MTHM/yr [*WNA*, 2013c]. According to IAEA (2005), at that time, there was one

international supplier[6] of LEU UO_2 fuel to the U.S. and that was Westinghouse Sweden. In the early 2000's, Westinghouse Sweden fabricated fuel to a total of eight countries: 1) Sweden, 2) Spain, 3) Germany, 4) France, 5) Finland, 6) Belgium, 7) Switzerland, and 8) United States. Westinghouse Sweden produces approximately 400 MTU within UO_2 fuel for BWRs and PWRs per year. In the conversion of UF_6 into UO_2 powder, the capacity as well as the plant license is limited to 600 MTU in UO_2 fuel [*Westinghouse Sweden,* 2014]. Another source is in agreement that the WH Sweden plant's maximum capacity is 600 MTU [*WNA,* 2014]. It is unknown what amount of material is processed by Westinghouse Sweden for each country. Furthermore, the capacity is much lower than other facilities that utilize the same technology for producing LWR fuel. Therefore, it is assumed that fabricating U.S. fuel is the priority of U.S. fabrication facilities and only if fuel demand exceeds U.S. capacity, the Westinghouse Sweden facility is relied upon for the remaining small portion of annual fuel demands.

DOE's "Megatons-to-Megawatts" program was established in 1993 following agreement between Russia and the U.S. on nuclear weapons reductions to produce BLEU as fuel in U.S. commercial nuclear reactors. From the late 1990s to 2007, the U.S. has contributed to the BLEU inventory by downblending 100 MT HEU of the originally declared 175 MT HEU usable surplus [*NNSA,* 2007]. The HEU inventory at the beginning of the modeling year 2013 is approximately 525 MT HEU (675 MT HEU total of U.S. commitment plus Russian commitments less 150 MT of HEU already downblended for use by NPPs since 1993). The fresh fuel made from BLEU entering the market has historically been limited to 10% each year (recent dialogue has evaluating increasing this to 20%) [*NNSA,* 2007; *WNA,* 2011]. It was assumed in this study that

[6] There is likely to be more international nuclear fuel suppliers presently, but a comprehensive list is not readily available.

10% of the commercial fuel demand was met through the use of BLEU (HEU downblended and mixed with DUF_6 tails assumed to have a U-235 assay of 0.20 wt%). This use of low-assay DUF_6 tails as a diluent reduces backlog of material that must be processed in the deconversion facilities. Although the most recent US-Russian agreement for blend-down of HEU has lapsed, talks between Russia and the U.S. continue [*McGraw Hill*, 2013]; a number of inventories remain and this modeling effort assumes continued BLEU use through the end of the modeling time frame. It is assumed here that additional commitments of HEU will allow a 10% contribution of reactor fuel through to the end of the simulation since the initial 525 MTU is modeled to be exhausted in the year 2058.

2.2.2.3. Back-End Operations

The back-end of the OTC modeled includes deconversion and managing UNF by storing at reactor-sites. Worker collective dose resultant from managing UNF off-site and then to disposal is considered in Chapter 4.

The deconversion of low-assay DUF_6 tails to depleted uranium oxides began in 2010 at the Portsmouth, Ohio facility. There is also a new deconversion facility operational in Paducah, Kentucky and the International Isotopes Fluorine Products, Inc. (IIFP) plant in New Mexico was licensed by the NRC for construction and operation of a deconversion plant in 2012. Within the model, the only deconversion plant dedicated to deconverting future commercial tails is the IIFP plant in New Mexico with an annual capacity of 3,400 MTU/yr [*IIFP,* 2009; *NRC,* 2012]. It was assumed that the IIFP plant begins operations in 2013 and deconverts the annual production of

DUF_6 from its commercial enrichment process, while the OH and KY plants' purpose are to deconvert the existing DOE stockpile of 600,000 $MTUF_6$ (420,000 MTU) low-assay DUF_6 tails [*WNA,* 2012c]. Any commercial DUF_6 low-assay enrichment tails previous to 2013 is included in the DOE historical stockpile will be deconverted by the Paducah and Portsmouth facilities. Within the model, Paducah and Portsmouth facilities are only allowed to deconvert non-DOE material when excess capacity become available; that is, after the DOE stockpile has been worked off.

Occupational exposure due to spent fuel pool (SFP) loading, maintenance, and re-racking is included in the doses of operating reactors and reported to the NRC as part of normal reactor operations [*NRC,* 2011b, 2012c]. The occupational collective dose was assumed in this study to continue in a manner consistent with present practices.

The first major addition or change to the traditionally-defined U.S. NFC could be considered the introduction of on-site dry cask interim storage in 1986 when the first dry storage installation at the Surry NPP in Virginia was licensed by NRC [*VEPCO,* 2002; *NRC*, 2012h]. From 1986 to 2010, 54 independent spent fuel storage installations (ISFSIs) were constructed and brought into operation. EPRI (2010b) estimates that by 2030, 73 ISFSIs will exist in the U.S. It is assumed that UNF is added to the existing on-site ISFSIs and requires continued expansion of the facilities.

The mass capacity of dry casks was estimated by EPRI (2010b) for a typical existing and future LWRs for UNF that has cooled in pools for 10 years at around 11.25 MT UNF/ dry cask.

However, the holding capacity has increased for the newer dry casks [*BRC,* 2011, 2012b]. The mass capacity of a single dry cask was averaged for heat and volume constraints of specific vendors where a single dry cask can hold around 13.05 MTHM [*BRC,* 2011, 2012b]. The new reactors that are constructed during the simulation time frame are not uniquely modeled; if dry interim storage is required for new reactors, the fact that the newest reactors under construction are being added at sites that already have operating reactors (Watts Bar, Sumner, Vogtle), it is assumed that such dry cask storage will be added as new casks resulting in expansion of capacity by adding additional concrete pads (extra exposure is incurred during construction of the additional ISFSI pads).

2.2.2.4. OTC Material Flow Analysis Methods Summary

Below is a bulleted list of the modeling parameters that were used for the OTC (as described above):

- Reactor Fleet and Bringing Reactors Online:
    - A new reactor is assumed to begin operations when the energy demand is 500 MWe-yr higher than the existing reactor capacity at that given year.
    - The reactor fleet is 2/3 PWRs and 1/3 BWR throughout the simulation.
    - When reactors are retired and brought online, both the defueling stage and initial fueling occurs at the same frequency and duration of regular annual fueling (that occurs in batches) to avoid fluctuations with discharged UNF and loading to and from the reactor.

- Priority of fuel used in reactors is: 10% of annual total fueling requirements will be met by BLEU; all the Re-ENU fuel produced that year is loaded and used by reactors; the rest of fuel requirements for any given year are met by the use of ENU.
- Inventories of nuclear materials at beginning of year 2013:
    - Natural Uranium (NU) = unlimited
    - Used nuclear fuel (UNF) = 70,000 MT
        - Wet storage = 18,000 MT UNF (BWR); 33,000 MT UNF (PWR)
        - Dry storage = 6,000 MT UNF (BWR); 13,000 MT UNF (PWR)
    - Highly-enriched uranium (HEU) = 525 MT HM
        - HEU mass proportion used in BLEU is 5%; DU mass proportion = 95%
    - Higher-Assay DUF_6 DOE Inventory = 73,500 MTU in DUF_6
    - Lower-Assay DUF_6 DOE Inventory = 362,474 MTU in DUF_6
    - DU-oxides of DOE Inventory already deconverted = 51,526 MTU in DU-Oxides
- U-235 wt% in each nuclear material stream:
    - UNF = 93.7 (does not change with aging over 50 year modeling)
    - Highly-enriched uranium (HEU) = Assumed at 90 wt%; DU as diluent at 0.20 wt%
    - Higher-Assay DUF_6 DOE Inventory = Average assay of 0.30 wt%
    - Lower-Assay DUF_6 DOE Inventory = Average assay of 0.25 wt%
    - DU-oxides of DOE Inventory already deconverted = Average assay of 0.25 wt% U-235
- Nuclear Fuel Cycle Facility Operations and Characteristics

- NU supply country = 10% US (10% = 2.5% UG; 7.5% ISL); 90% International (90% = 33.8% OP + 15.6 UG + 40.7 ISL)
- Conversion and ENU enrichment: Use of U.S. facilities are priority for conversion and ENU enrichment until demand exceeds US capacity and assumption that international partners have sufficient capacity
- Re-enrichment: Annual enrichment capacity for re-enrichment is limited to half of Paducah's gaseous diffusion plant for year 2013 (4000 MT SWU/yr). For 2014, the centrifuge enrichment plant in New Mexico is assumed to take over with half of the capacity (1,500 MT SWU/yr)
- ENU and Re-ENU Fuel Fabrication: Use of U.S. facilities are priority for fuel fabrication until demand exceeds US capacity and assumption that international partners have sufficient capacity
- BLEU Fuel Fabrication: 100% of BLEU fuel is manufactured at the Areva Richland Fuel Fabrication plant, any remaining capacity of Areva Richland plant is used for ENU or Re-ENU
- Deconversion: The DOE faculties at Paducah and Portsmouth are only allowed to deconvert commercial DUF6 material if the DOE inventory of lower-assay DUF_6 material is worked down to levels lower than the annual capacities of both plants.
- Dry interim storage:
    - UNF already in dry storage casks (DSC) = DSC capacity = 11.25 MT UNF/DSC
    - UOX UNF discharged from either a PWR or BWR during simulation to DSC = 13.05 MT UNF/DSC

- At-Reactor ISFSIs: Number of sites in 2013 = 64; Increases incrementally to 73 sites total modeled by year 2029

2.2.3. Quantifying Worker Collective Dose Performance Metrics

Quantitative data from the literature survey that enables development of OTC worker collective dose performance metrics to describe fuel cycle safety performance in terms of radiological impact has been identified. Available data for individual components of the radiological impact assessment model have been quantified empirically or presented as empirical distributions of dose measurements by four generally categorized entities: (i) individual facilities that self-report occupational exposure levels as mandated by regulations (e.g., [*Denison Mines,* 2007b]); (ii) regulatory agencies that collect, assimilate, and provide statistics of routine exposure data and violations of exposure limits (values and nature of violation) such as the NRC, Mine Safety and Health Administration (MSHA), and Occupational Safety and Health Administration (OSHA) [*NRC,* 2012b; *MSHA,* 2012a]; (iii) research/academic institutions that produce data at various levels (e.g., universities, Agency for Toxic Substances and Disease Registry [ATSDR]) [*Cavallo,* 1997; *Porstendörfer,* 2001]; (iv) agencies that set exposure guideline levels and that act as the bridging entity between regulations and the research institutions (e.g., International Commission on Radiological Protection [ICRP] and the American Conference of Governmental Industrial Hygienists [ACGIH]) [*ICRP*, 2012; *Menzel & Harrison*, 2012]. These agencies are accepted as authoritative sources on the current body of knowledge. Data sets have been used for this work that represent measured levels worker collective doses based upon industry experience with emphasis on collecting as recent data as possible that is within the public domain. The

importance of this nuance is described in detail within Chapter 2 (and Appendix C), but the crux is that impacts have been reduced each decade due to many factors; near future estimated impacts should reflect performance metrics of actual industry performance.

The normalization parameters were determined by throughput of the individual facilities, as the mass flow, with a predetermined percent of nuclear material processed (e.g., natural uranium [NU] concentration vs. LEU vs. UNF, etc.); normalization units and bases were chosen, and such that an implicit assumption was utilized that linear relationships adequately described the estimation of future impacts to operational throughput. The equation to calculate the worker collective dose metric is shown below (Equation 2.1) where M_{WCD} is the worker collective dose metric, D is annual individual dose per monitored worker, W is the number of monitored workers at the facility, m is the mass flow rate in units if metric tons of heavy metal per year (for the reactor operations, m is electrical energy produced in a given year).

$$M_{WCD} = \frac{D * W}{m}, \tag{2.1}$$

The data are provided in terms of the occupational radiological risk metric (collective dose) normalized by the mass of uranium in each facility's product stream, as shown in Table 2.2 and Table 2.4). Worker collective doses associated with reactor operation are normalized by electrical energy output (MWe-yr) and dry interim storage operations are normalized by dry cask or by ISFSI (Table 2.5). Linear relationships with the normalization bases were assumed for the risk metrics with the following data sets: (1) individual worker radiation doses for a given year, (2) production data for uranium processed for the same given year, and (3) receptor population

exposed for that same given year (number of workers). The sources of data used to quantify the worker collective dose metric (Table 2.3) and values for each metric are provided (Table 2.4 and Table 2.5).

This is important to note that data sets have been used for this work that represent actual measured levels of worker collective doses and are based upon real events and industry experience with emphasis on collecting as recent data as possible that is within the public domain. The emphasis was also placed on using most recent measurements due to observed trends that doses are generally decreasing worldwide from many factors but mostly because of the ALARA (As-low-as-reasonably-achievable) operational philosophy.

A bulleted list of normalized worker collective dose metric modeling assumptions for Chapter 2 is below:

- The preferred and most common source of data used for modeling parameters and building normalized impact metrics were taken from actual measured doses of operating facilities worldwide. The emphasis was also placed on using most recent measurements due to observed trends that doses are generally decreasing worldwide from many factors but mostly because of the ALARA operational philosophy (ALARA stands for As-low-as-reasonably-achievable).
- Worker collective doses as part of handling and disposing of UG mining, OP mining, and milling are included in the normal operations of these mines and mills each year. Data

separating the onsite activities of waste and operations that directly work to make the product are not readily available.

- No worker collective doses are calculated for disposing of depleted uranium-oxides (which is not presently occurring). Worker collective doses for storage and maintenance of depleted uranium-oxides are assumed to be included in the normal operations of deconversion.
- Worker collective doses for maintenance, refueling outages, and wet interim storage are included in the worker collective doses normalized metrics for normal reactor operations. Data were not readily available for those individual activities at reactor sites and were included within reported values to the NRC [*NRC,* 2011e, 2012b].

Table 2.2 | OTC Worker Collective Dose Metric Parameter Data Sets (Front-End, Reactors)

OTC Operation Technology	**Metadata**		**Parameter Data Sets**			**Row Number of Refs. Listed Table 2.3**
	Country (Site)	**Data Measure-ment Years**	**Average Individual Worker Dose (*D*) [mSv/yr]**	**Receptor Population (*W*) [persons]**	**Annual Production (m) [Normative basis unit /yr] (see footnote 25)**	
Mining						
Open Pit	US (representative site)	1979	1.01	115	122	1
	Australia (Ranger)	2007-2011	1.09	287	3812	2
	Namibia (Rössing)	2009-2011	1.40	1610	2432	3
Open Pit Mining Average			*1.17*	*671*	*2122*	--
Underground	Australia (Olympic)	2008-2009	2.12	1992	3570	4
	Canada (McArthur)	2001-2007	1.16	611	6900	5
Underground Mining Average			*1.64*	*1302*	*5235*	--
ISR/ISL						
Alkaline Solution	US (Smith-Ranch)	2002-2010	6.20	146	540	6
	US (Crowe Butte)	1997-2010	1.20	69	308	7
	Australia (Beverley)	2008	0.28	148	636	8
ISR/ISL Average			*2.56*	*121*	*495*	--
Milling						
Acid Leach	US (White Mesa)	1999	1.70	80	1400	9
	US (Cotter Corp.)	2005	1.68	167	281	10
Acid Leach Milling Average			*1.69*	*124*	*841*	--
Conversion						
Dry	US (Metropolis)	2009	3.30	450	12600	11
Wet	Canada (Cameco Port Hope)	2010-2011	0.33	299	8000	12
	France (Areva Comhurex)	2008-2010	0.72	97	12500	13
Wet Conversion Average			*0.53*	*198*	*10250*	--
Enrichment & Re-enrichment						
Diffusion	US (Paducah, KY)	1989, 2012	0.36	227	480	14
Centrifuge	US (USEC, OH)	2006	0.29	600	1022	15
	US (LES, NM)	2005	0.20	210	438	16
Centrifuge Enrichment and Re-enrichment Average			*0.25*	*405*	*730*	--
Laser (US)	US (GE-Hitachi NC)	2012	1.50	37	150	17
Deconversion						
DUF_6 to DU-Oxides (US)	US (IIFP, NM)	2009	0.75	139	2300	18
Downblending						
HEU with Low-Assay DUF_6 Tails	US (NFS)	2010, 2012	0.27	130	92	19
	US (NFS, BWXT, Y-12,SRS)	1996, 2007	0.90	125	92	20
Downblending Average			*0.59*	*128*	*92*	--
Fuel Fabrication						
UO_2 Oxide	US (NFS)	2001, 2009	0.27	726	1200	21
	US (Westinghouse Electric)	2001, 2009	2.18	650	1150	22
Fuel Fabrication Average			*1.23*	*688*	*1175*	--
Fuel Irradiation (Reactor Operation)						
BWR	US (all 35 US reactors)	2010	0.70	69174	31275	23
PWR	US (all 69 US reactors)	2010	0.35	110474	60859	24

Notes: The blue highlighted rows are averages from several sites and are carried over to Table 2.4.

Table 2.3 | Worker Collective Dose Metric Parameter Data References (Front-End, Reactors)

Footnote	References Used in Table 2.2 *Notes*
1	[*Miller & Scott*, 1981; *EPA*, 1983a] *Worker population is the number of workers exposed onsite*
2	[*ERA*, 2005, 2006, 2007, 2008, 2009, 2010, 2011; *WNA*, 2013a)] *Worker population is the number of designated workers. Designated workers are those employees and contractors who have the potential to exceed 5 mSv per year from occupation exposure to radiation. Designated workers + non-designated workers = total monitored workers on site. The number of non-designated workers was not provided in the above-listed references.*
3	[*Rössing*, 2011] *Worker population was the total number of employees*
4	[*BHP*, 2009; *WNA*, 2013a] *Worker population was the total monitored workers (designated + non-designated workers)*
5	[*CNSC*, 2009; *WNA*, 2013b] *Worker population was the total monitored workers.*
6	[*NRC*, 2009, 2010, 2011c,. 2011d]
7	[*Crowe Butte*, 2007]
8	[*Nilsson & Randheim*, 2008]
9	[*Denison Mines*, 2007a, 2007b, 2010]
10	[*SENES*, 2009; *EIA*, 2011]
11	[*EPA*, 1989; *NRC*, 2011e]
12	[*Cameco*, 2011, 2012]
13	[*Areva*, 2012a, 2012b]
14	[*EPA*, 1989; *WNA*, 2012c]
15	[*USEC*, 2006; *WNA*, 2012c]
16	[*LES*, 2005; *WNA*, 2012c]
17	[*NRC*, 2012a]
18	[*IIFP*, 2009; *NRC*, 2012c, 2013a]
19	[*NFS*, 2009, 2012; *NRC*, 2012b; *Power*, 2012]
20	[*DOE*, 1996, 2009; *NNSA*, 2007; *WSRC*, 2008]
21	[*ANL*, 2001; *NRC*, 2012b]
22	[*ANL*, 2001; *NRC*, 2012b]
23	[*NRC*, 2012b] *Worker population is the total number of monitored workers.*
24	[*NRC*, 2012b] *Worker population is the total number of monitored workers.*
25	Normative basis unit is listed in Table 2.4

Table 2.4 | OTC Worker Collective Dose Metrics (Front-End, Reactors)

Worker Collective Dose Performance Metric Set			Parameter Data Sets Used		
Once-Through NFC Operation Technology (Country of Exposure Data)	**Normalized Average Occupational Collective TEDE (M_{WCD}) [person-mSv/ Normative Unit]**	**Normative Basis: Units (*m*)**	**Average Individual TEDE (*D*) [mSv/year]**	**Receptor Population (*W*) [Workers]**	**Annual Production Data as the Denominator (*m*) [Normative basis unit / yr]**
Mining					
Open Pit (Australia and Namibia)	3.69E-01	MTNU	1.17	671	2122
Underground (Australia and Canada)	4.08E-01	MTNU	1.64	1302	5235
ISR/ISL					
Alkaline Solution (US and Australia)	6.26E-01	MTNU	2.56	121	495
Milling					
Acid Leach (US)	2.48E-01	MTNU	1.69	124	841
Conversion					
Dry (US)	1.18E-01	MTNU	3.30	450	12600
Wet (Canada and France)	1.01E-02	MTNU	0.53	198	10250
Enrichment & Re-enrichment					
Diffusion (US)	1.70E-01	MTLEU	0.36	227	480
Centrifuge (US)	1.36E-01	MTLEU	0.25	405	730
Laser (US)	3.70E-01	MTLEU	1.50	37	150
Downblending					
HEU metal with Low-Assay DUF_6 Tails (US)	8.09E-01	MTBLEU	0.59	128	92
Deconversion					
DUF_6 to U Oxides (US)	4.53E-02	MTDU	0.75	139	2300
Fuel Fabrication					
UO_2 Oxide (US)	7.17E-01	MTIHM	1.23	688	1175
Fuel Irradiation (Reactor Operation)					
BWR (All US 35 BWRs in 2010)	1.54E+00	MWe-yr	0.70	69174	31275
PWR (All US 69 PWRs in 2010)	6.28E-01	MWe-yr	0.35	110474	60859

Worker collective dose performance metrics associated with dry interim storage activities were previously estimated and reported in EPRI (2010b) and are adopted for this work. The metrics below (Table 2.5) describe a generic LWR UNF setting for current burnup rates and cooling times in spent fuel pools (around 10 years) and have been validated by industry experience recently within Weck (2013) and DOE estimations [*DOE*, 2008d, 2008e]. Four activities are pertinent to dry interim storage at a reactor site: (1) loading of UNF assemblies from the spent fuel pool into casks[7], transport from reactor building to the ISFSI located outside of the reactor

[7] Wet storage of UNF within reactors' spent fuel pools is where the majority of UNF is currently stored [*Wagner et al.*, 2012; *Peterson & Wagner*, 2014]. Wet storage operations are a common activity for reactor operations and mostly only comprise of UNF remote-handling and transferring spent fuel assemblies from the reactor to spent fuel pools. Because unloading fuel from the reactor is part of normal reactor operations, worker doses related to wet fuel

building, and final loading of the casks into a concrete vault; (2) expansion of the existing ISFSI; (3) operations and maintenance of the ISFSI with loaded UNF; and (4) inspections of concrete housing units and security surveillance.

Table 2.5 | OTC Worker Collective Dose Metrics (Dry Interim Storage)

Loading operations for a dry storage system [person-mSv/cask]	Construction of one additional dry cask at an existing ISFSI [person-mSv/ cask]	ISFSI operations and maintenance [person-mSv/ISFSI site/year]	ISFSI inspections and security surveillance [person-mSv/ISFSI site/year]
4	1.7	15	1.2

Performance metrics, when multiplied by the annual amount of activity in each fuel cycle operation constitute a measure of annual worker collective dose for each operation that is utilized to fulfill reactor fueling needs for that year. The summation of collective doses at each operation is a measure of radiological impact from the OTC assumed as the baseline performance.

2.3. Results and Discussion

In the following, a summary of the results from modeling the worker collective doses of the detailed, revised OTC within a simplified energy growth scenario is presented as two parts. The first is the material flows of fuel and waste (DUF_6 and UNF as part of interim dry storage); the second part of the results is presented as the estimated worker collective dose modeled as the scaled worker collective dose performance metrics to the material flows modeled in the first section. The summation of worker collective doses from the conventional front-end operations,

pool UNF management are not reported separately from the other reactor management roles, and therefore not widely available through NRC's annual report on occupational exposures throughout the U.S. [*NRC*, 2012b].

additional operations and technologies described above for the modified front end of the fuel cycle, and interim dry storage constitute the OTC and serves as the baseline for comparing advanced fuel cycle options.

2.3.1. Material Flow Analysis

Three material flow analyses were performed for the OTC scenario. First was the annual reactor fuel required and then loaded by different types: available from conventional means, re-enrichment and downblending activities. The second material flow analysis was the DUF_6 management and deconversion activities that result from enrichment, re-enrichment, and deconversion activities (with consideration of the DOE inventories of previously accumulated low-assay DUF_6 tails). Dry interim UNF storage transfers and maintenance is the final material flow analysis that is tracked.

2.3.1.1. Annual LWR Fuel Loadings

The electrical energy demand for the U.S. LWR fleet is modeled to increase at a constant rate of 1% per year. The amount of energy that each reactor can produce depends on historic availability coupled with capacity upgrades. Annual fuel requirements do not change until new reactors are required to be brought online. The increased annual fueling requirements are shown as the black line in Figure 2.2, and the breakdown in contributions from BLEU, Re-ENU, ENU are shown. Because there were no constraints placed on the raw uranium resources available and the modeling assumption set that 10% of the fresh fuel used must come from the excess uranium

weapons stock, the most influential factor affecting fuel supply was the decision to re-enrich the high-assay tails from DOE's historical stock starting in the year 2013 at the Paducah gaseous diffusion plant. After 2014, the stair-step function is caused by the assumption that the Louisiana Energy Service (LES) centrifuge plant in NM would assume the responsibility of re-enrichment until the high-assay stock was exhausted. In 2013, the contribution of re-enriched natural uranium (Re-ENU) is around 19% (434 MT Re-ENU fuel) from running the Paducah gaseous diffusion plant and then drops to 5% (119 MT Re-ENU fuel) due to lower enrichment (SWU) capacity from the NM centrifuge plant. In 2022, the DOE high-assay DUF_6 inventory is depleted and re-enrichment activities cease.

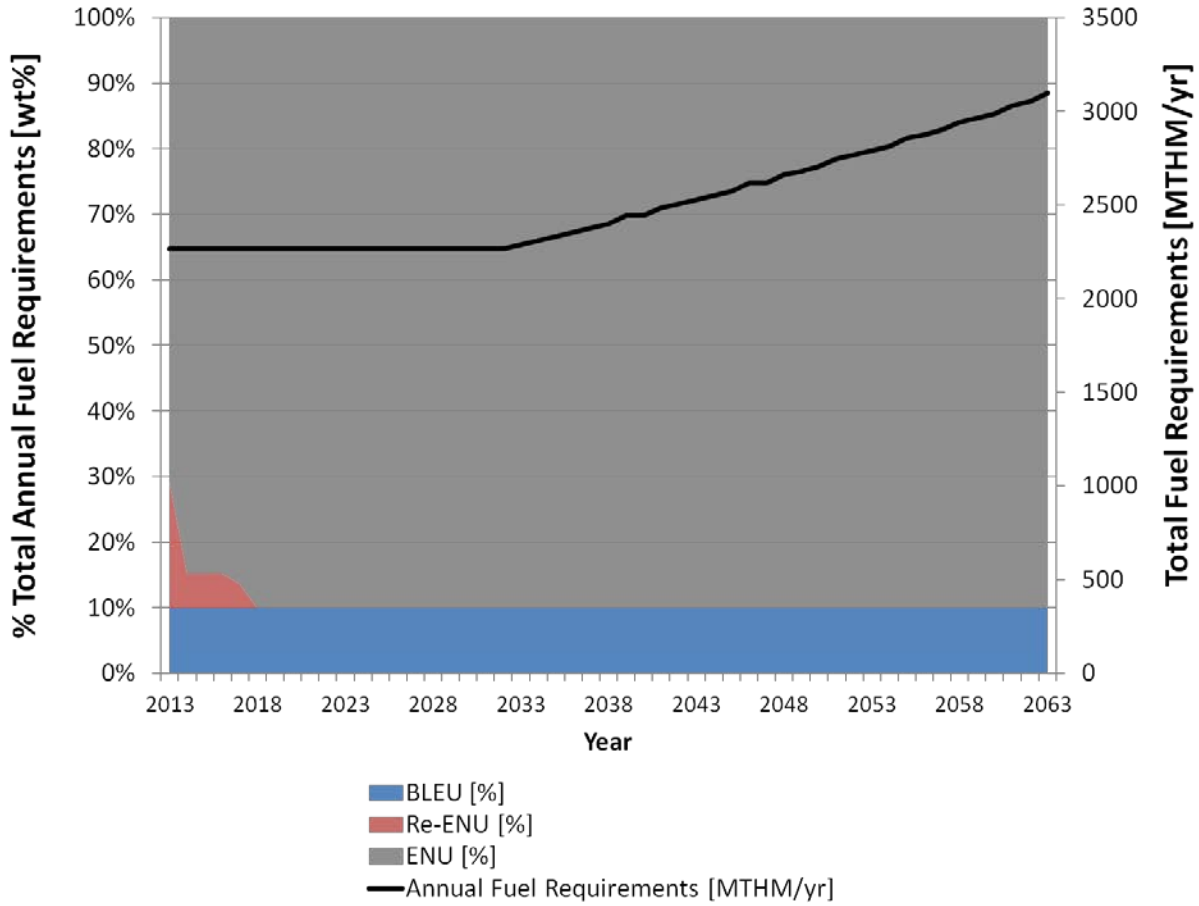


Figure 2.2 | OTC Percent of LWR Loaded Fuel Type and LWR Fuel Loaded Each Year

2.3.1.2. DUF_6 Management and Deconversion

DUF_6 is produced from the enrichment of ENU, and the re-enrichment of high-assay DUF_6 tails. The amount of DU produced per 1 MTIHM of each type of fuel was calculated with standard conservation of mass and value functions related to SWU estimations [*MIT,* 2001; *EPRI,* 2010b]:

- ENU: (PWR @ 4.5wt%): 6.88 MT DU; (BWR @ 4.35 wt%): 6.57 MT DU
- Re-ENU: (PWR @ 4.5 wt%): 12.78 MT DU; (BWR @ 4.35 wt%): 12.26 MT DU

DUF_6 tails generated domestically in the model can be used in the production of BLEU (5 wt% HEU, 95 wt% DU), as shown in Figure 2.3 as a dashed orange line. DUF_6 tails not used in BLEU are deconverted and stored until final disposal.

The re-enrichment operation begins at the same time as the commercial deconversion at the IIFP facility and creates a large stream of Re-ENU tails (purple solid line in Figure 2.3) and follows the stair-step trend of the contribution of Re-ENU to fuel production (red-shaded area in Figure 2.2). Re-enrichment activities cause a backlog of tails (blue solid line in Figure 2.3) that must be stored until excess deconversion capacity from any of the three facilities becomes available. The accumulation of backlogged re-enrichment continues until 2029 and DOE historical and adopted inventories are drawn down sufficiently by 2030 (blue line in Figure 2.4). After 2030, all three facilities can drawdown the accumulated backlog of commercial tails and deconvert the annual production of tails for each year until 2043 when the deconversion capacity is exceeded by U.S. production of DUF_6 tails (blue solid line in Figure 2.3). ENU tails are produced when the

Paducah plant is running (green solid line in Figure 2.3), but then drops due to much of the NM centrifuge plant capacity is being used for re-enrichment activities in 2013. The U.S. enrichment capacity increases due to opening the Ohio centrifuge plant in 2015 and the level of domestic ENU tails produced also increases until the maximum capacity of enrichment services dedicated to U.S. stocks are reached at year 2018. International enrichment services increase rapidly in response to the lack of U.S. capacity in 2014 (pink solid line in Figure 2.3), then stay constant as the fuel requirements for reactors stays constant until additional reactors are constructed in 2032 and the international community has sufficient capacity to enrich the remaining growing need for ENU fuel.

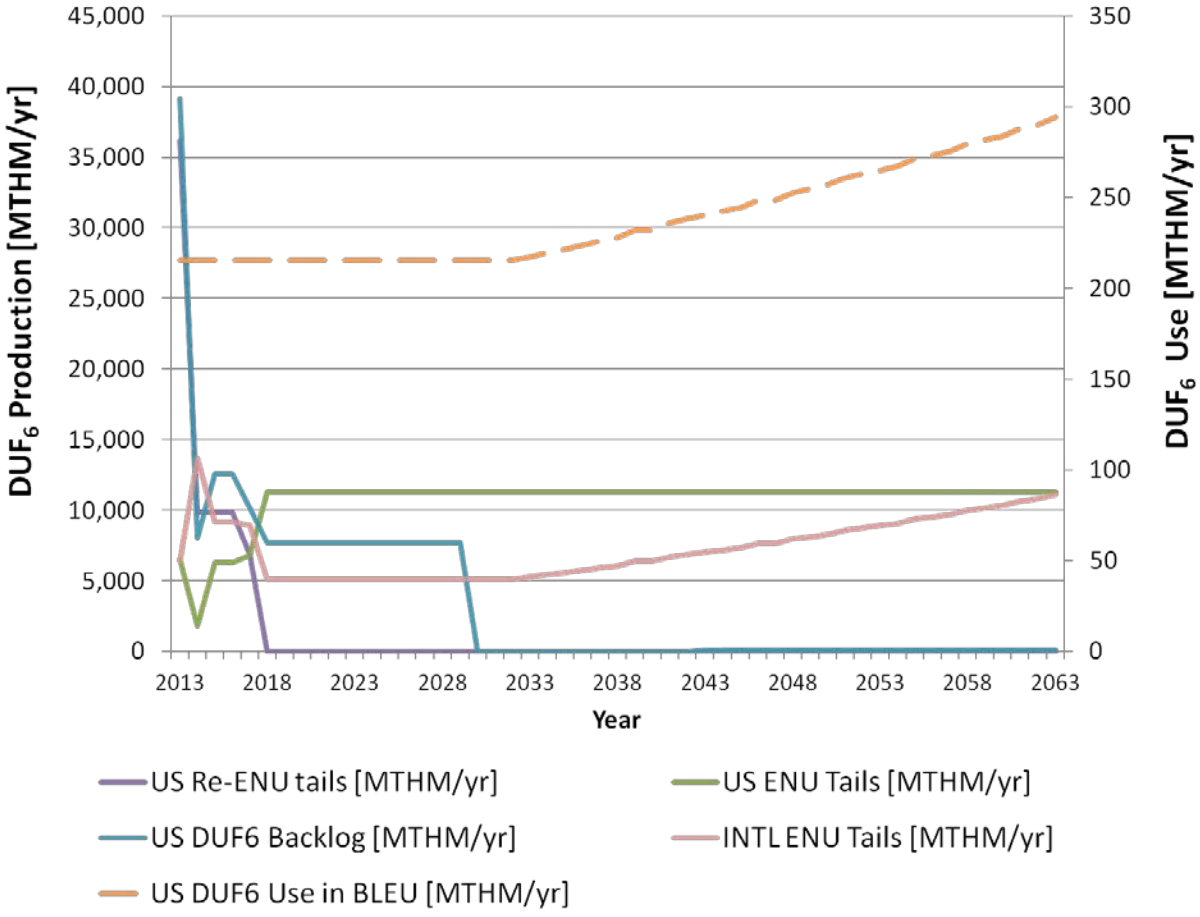


Figure 2.3 | OTC DUF_6 Annual Production and Use

Low-assay tails that are produced both domestically and internationally are assumed to be stored until deconversion capacity is available and then on-site storage and maintenance of depleted uranium-oxides is performed at deconversion facilities. The commercial DUF_6 tails are deconverted in the U.S. by the IIFP plant with a capacity of 3,400 MTHM as DU/yr (blue line in Figure 2.4) until the Paducah and Portsmouth deconversion facilities become available. These two facilities do not become available until the DOE inventory of low-assay tails is worked down in 2029 (red line in Figure 2.4). In 2030, the amount of annual commercial depleted uranium-oxides increases to the capacity of all three U.S. deconversion plants at around 24,645 MTHM as DU/yr and remains at that level for around 13 years (2030-2042) due to the backlog of DUF_6 created from re-enrichment and the cascading effect of additional backlog accumulated from normal enrichment requirements.

The increase in international deconversion efforts is resultant from when insufficient U.S. capacity was available to self-supply enrichment services in 2014, and assumes that international consensus matches current U.S. practice with regard to waste form stabilization. The amount of depleted uranium-oxides produced outside of the U.S. reflects the assumption that international deconversion activities would match the rate of DUF_6 production from enrichment facilities (green line in Figure 2.4). At 2033 when increasing amounts of fuel are required due to newly constructed reactors, the enrichment needs must be met by international enrichment partners and thus, the deconversion level of activity increases correspondingly. The sum of international and domestic depleted uranium-oxides produced each year (purple line in Figure 2.4) reaches a

maximum of over 31,000 MTHM as DU/yr in 2041 while deconverting the last of the DUF_6 re-enrichment tails backlog is cleared.

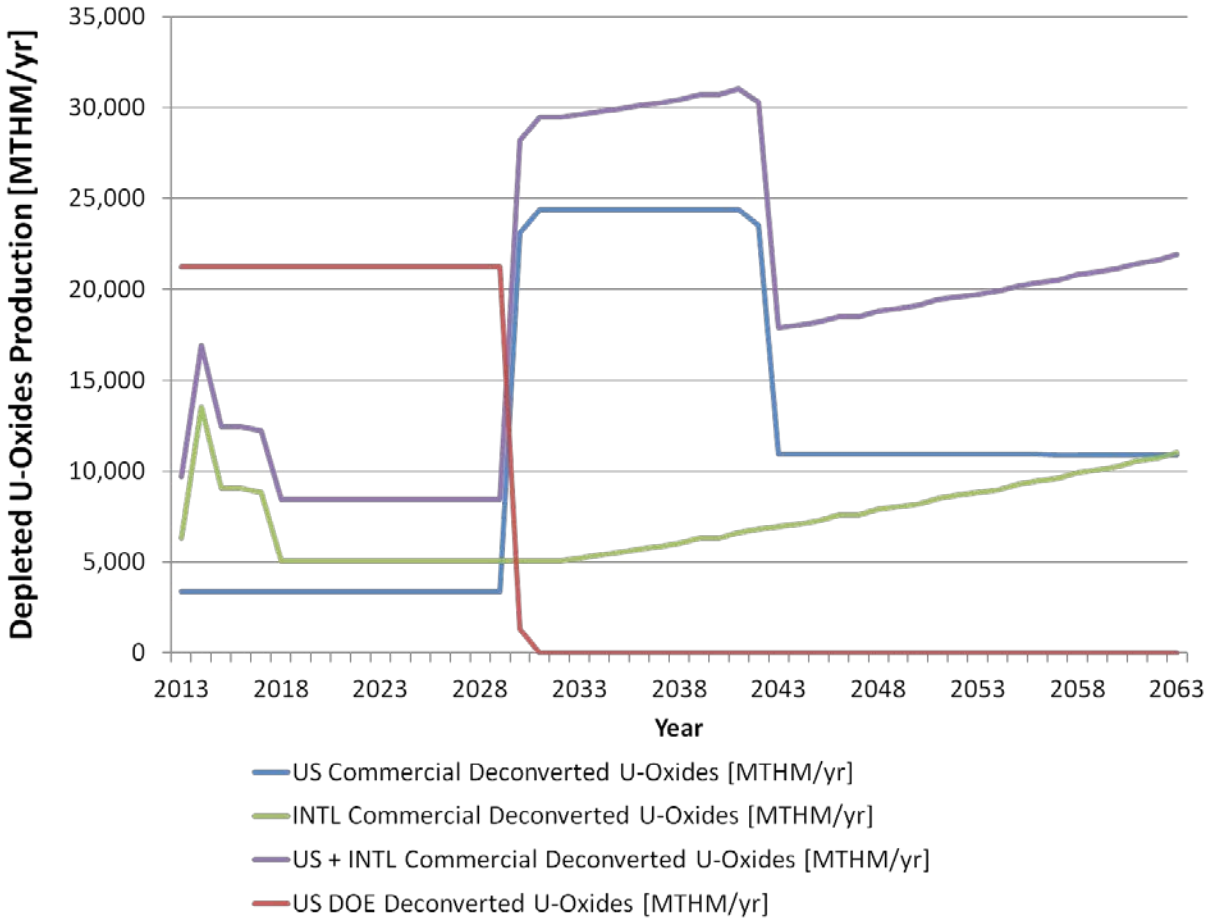


Figure 2.4 | OTC Depleted Uranium-Oxides Annual Production

2.3.1.3. Dry Interim Storage

An estimated 73 independent spent fuel storage installations (ISFSI) will exist in the U.S. by 2020. Continued expansion of the ISFSIs is modeled with each dry storage cask (DSC) holding UOX fuel storing 11.25 MTHM UOX UNF [*EPRI*, 2010b]. The annual additions to the DSC inventory increases each year according to the amount of fuel that is used by the reactor fleet

(blue solid line in Figure 2.5). In 2012 the total inventory of DSCs was around 2000 DSCs loaded units in the U.S. [*EPRI,* 2013a] and was used as the starting inventory of the OTC model. The cumulative plot of the annual additions added to the original 2,000 DSCs in the U.S is shown as the dashed orange line in Figure 2.5. The slight change in the rate of inventory growth reflects the gradual increased rates of the annual additions. By the end of 2063, it is estimated that the total DSC inventory will be near 10,600 DSCs stored at reactor sites around the U.S.

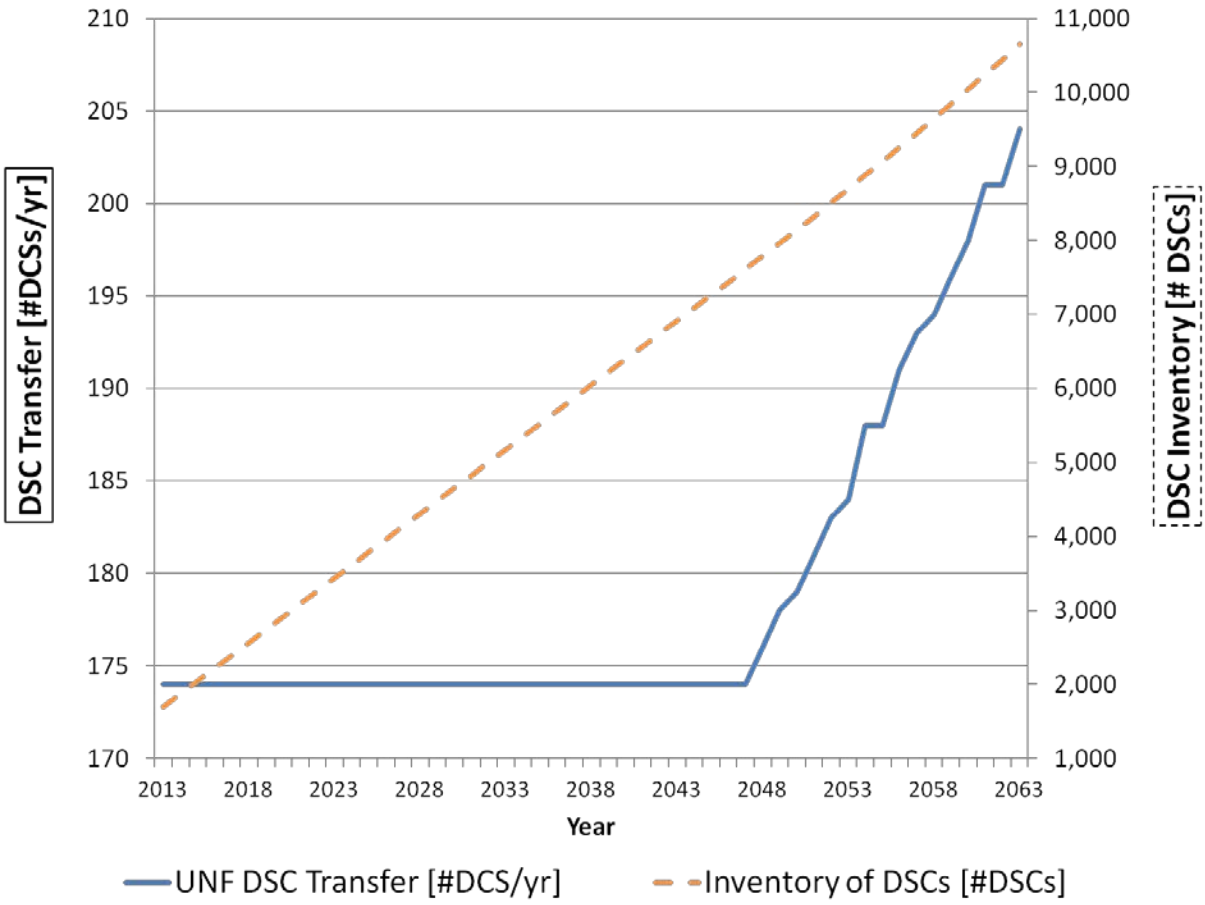


Figure 2.5 | OTC Dry Interim Storage DSC Additions and Inventory

2.3.2. OTC Worker Collective Doses

The estimated total worker collective doses each year for the OTC are shown in Figure 2.6 for the modeling years of 2013 to 2063. Worker collective doses range from 110,000 person-mSv/yr to 179,000 person-mSv/yr (110 – 179 person-Sv/yr) and increase for the most part in a steadily increasing rate, similar to that of the assumed growth of the rate of nuclear energy demand.

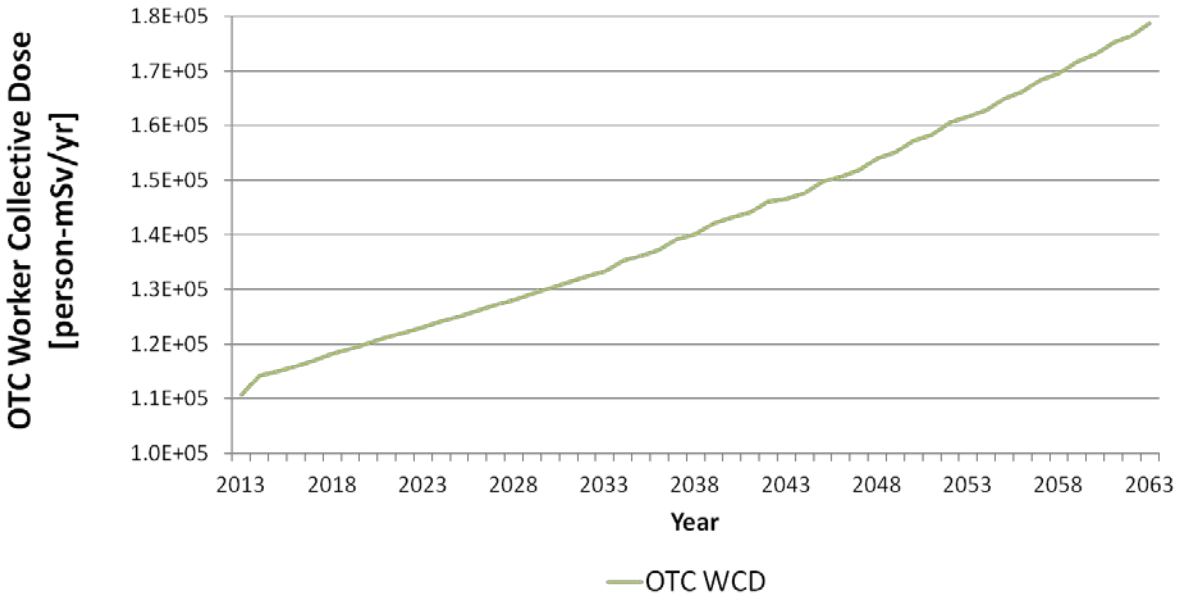


Figure 2.6 | OTC Worker Collective Doses

The worker collective doses from each OTC operation are graphically displayed in Figure 2.7. As measured by worker collective dose (person-mSv/yr), the highest overall dose was for the operating reactors (PWRs and BWRs). It is interesting to note that even though BWRs were 1/3 of the total reactor fleet, they contributed about the same proportion of annual dose as PWRs. This result is due to the fact that in the U.S., BWRs typically have more monitored workers per plant (~1000 workers/plant) than PWRs (~600 workers/plant) receiving a measurable dose. This operating difference between the two reactor types has been documented and trends are long-established, and with higher annual individual worker doses for BWRs than PWRs. Considering

the assumed constant share of the two reactor technologies as part of the reactor fleet, the worker collective doses for the two reactor fleets become comparable [*NRC*, 2012b].

The next highest worker collective dose was associated with fuel fabrication from ENU, Re-ENU, and BLEU material within the U.S. shown in Figure 2.7. The high collective occupational dose for U.S. fuel fabrication is attributed to the high relative throughput of mass that is required for fuel feeding into the reactors each year and not a reflection of a high individual worker dose (estimated as 2.5 mSv/yr, compared to U.S. NRC limits of 5 rem/yr [50 mSv/yr]).

In Figure 2.7, the worker collective doses incurred from U.S. and international OTC activities to support a U.S. reactor fleet are shown for a select number of years. The amount of nuclear material processed by front-end operations outside of U.S. is small, as U.S. fuel fabrication facilities have sufficient capacity. The same is true for international conversion operations due to the assumed large dedicated capacity of the Metropolis conversion facility.

The next highest contributors to OTC worker collective dose were associated with natural uranium recovery (through OP and UG mining, and ISR/ISL technologies), followed by collective occupational doses from milling, conversion, and U.S. LLW disposal. The concentration of U-235 found in nature is around 0.71 wt% U-235, compared to that of the concentration of required to sustain nuclear fission in a modern LWR is around 4-5 wt% U-235. This disparity translates to a high volume of raw uranium ore that must be mined, recovered, and processed. When multiplied by a small worker collective dose metric, this results in a large worker collective dose from both U.S. and international operations. A high volume of LLW

disposal is required due to LLW produced in the U.S. from all operations other than mining and milling; the worker collective dose from LLW disposal increases accordingly to the increased reactor fuel and front-end operations (Figure 2.7).

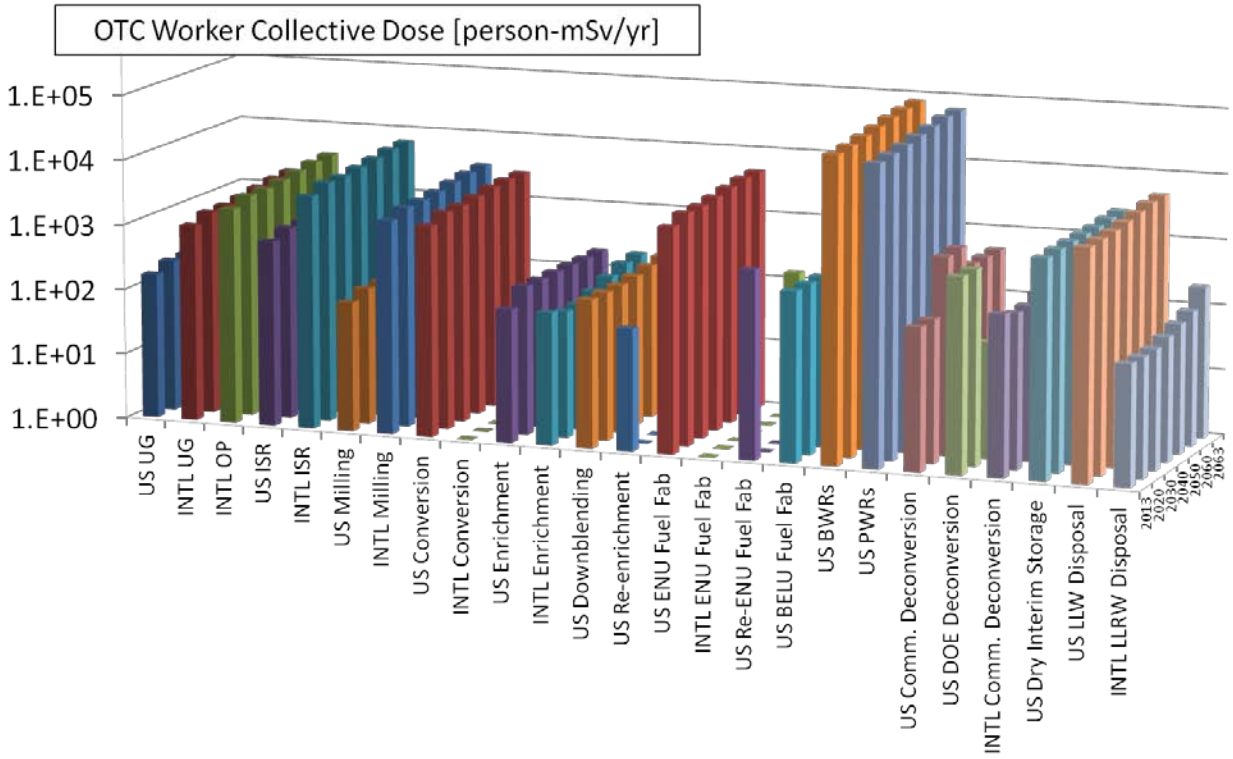


Figure 2.7 | Worker Collective Dose from OTC Operations

Aggregating the fine granularity of detail shown in Figure 2.7 to a number of grouped operations eases interpretation of the contributors to the OTC worker collective dose. In Figure 2.8 the U.S. and international activities are combined by operation and the results are displayed as a relative percent of the total OTC worker collective dose for a select number of years.

The worker collective dose from the front-end of the OTC which includes re-enrichment and downblending is approximately 14.3%. Reactor operations are around 80% of the modeled OTC

worker collective dose for the seven selected years shown in Figure 2.8. When worker collective doses from LLW disposal, management of UNF at ISFSIs, and deconversion operations are considered, the relative contribution is around 5-6.5 % of the total OTC worker collective dose (noting that modeling of impacts incurred from disposal of UNF in a deep geologic repository is deferred to Chapter 4).

Even with dynamic behavior of fuel cycle activities, including large contributions of Re-ENU fuel, there are minimal changes of the contributions to OTC worker collective dose from U-recovery and back-end operations in the selected years shown in Figure 2.8. While the reactor operation contribution to OTC worker collective stays almost constant, the contribution of U-recovery and back-end operations change complementarily; but the changes are not directly related. The cause of a higher worker collective dose for back-end operations at the first few decades is that deconversion of the large DOE inventory of DUF_6 low-assay tails and then the backlog of DUF_6 tails from re-enrichment are then deconverted.

These values appear to go against the generally held belief that the front end of the OTC is responsible for most of the occupational radiological risk [*UNSCEAR,* 1993] and this result is furthered evaluated in Chapter 3.

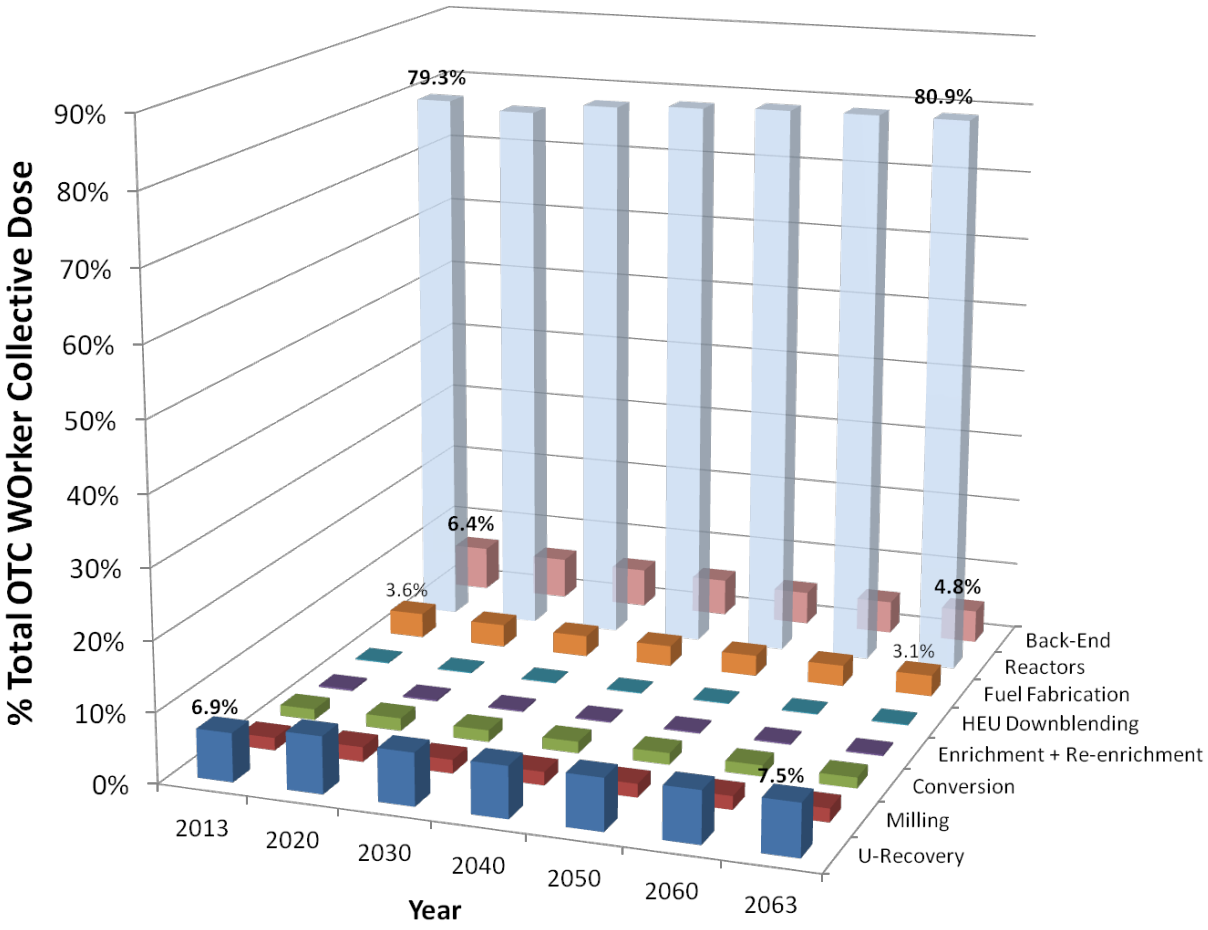


Figure 2.8 | Percent of Worker Collective Dose from Grouped OTC Operations

2.4. Corollary Verification Study: A Summary

As part of the quality control and verification process for determining OTC worker collective doses, the process of evaluating the most recent UNSCEAR data was performed [*UNSCEAR*, 2008], comparing occupational doses and numbers of workers employed in these stages of the nuclear fuel cycle to the results of the initial model runs. From this analysis, it was observed that the relative contribution of each NFC operation to radiological dose to workers was substantially different from widely used historical data and conventional wisdom. A manuscript in the form of

a journal article was published in February of 2014 [*Krahn et al.*, 2014] describing this evaluation. In the manuscript and detailed discussion found in Appendix C, the causes of the differences were explored by describing the NFC and the nature of radiological impacts to workers, the results from the new NFC assessment tool, historical data on radiological impacts to workers, and trends in radiological worker impacts over time. A brief summary is provided here of Krahn et al., (2014) and the discussion in Appendix C that covers the highlights of Chapter 2 on calculating worker collective doses from the OTC, the historical trends of contributors of OTC operations to overall NFC worker collective doses, and the factors that have led to the change in occupational radiological risk contributors of the OTC.

2.4.1. Recent Estimates of OTC Worker Collective Doses

Previous modeling of the OTC was performed to estimate the total worker collective doses at each operation to support an increasing electricity demand in the U.S. from 2013 to 2063 and results were presented in Chapter 2. A finding of importance here is that over the 50-year modeling time frame, the contribution by the reactors to overall OTC worker collective dose was much greater than of other operations (as can be seen in Figure 2.8). Additionally, contributions of each operation did not vary greatly over time.

The UNSCEAR (1993) report was used as the verification basis for understanding global trends of worker collective doses through time. Verification against the UNSCEAR (1993, 2008) reports required a simple method to aggregate worker collective doses from major operations that are traditionally used world-wide to prepare fuel [*UNSCEAR*, 1993, 2008]. This meant that

contributions to OTC worker collective dose from re-enrichment, downblending and deconversion, that were not part of the groupings of operations and the contributing operations, were reorganized into the conventional phases of: (1) uranium-recovery[8], (2) milling, (3) conversion and enrichment[9] , (4) fuel fabrication[10], and (5) reactor operations. The operations that were re-grouped according to the five operations considered for verification against world-wide UNSCEAR data.

2.4.2. Historical Data on Worker Collective Dose from Traditionally-Defined OTC Operations

Results from historical surveys of worker occupational dose from NFC operations are available. However, such surveys are infrequently done because they are costly and time-consuming. The seminal work on this subject is documented in the "UNSCEAR 1993 Report to the General Assembly," which, among other things, documents a global survey of information related to worker doses for NFC operations covering the years 1975–1989 [*UNSCEAR*, 1993]. From the 1970s to late 1980s, the majority of the worker collective doses were from the front-end of the OTC (most notably the uranium recovery operations). The UNSCEAR (1993) report accounts for the historical ''conventional wisdom'' that impacts from the front end of the NFC are dominant. However, estimates from the recent VU work are substantially different from the historical data and do not support conventional wisdom, which led to the central question: Could these disparate results be reconciled? The answer to this question was and is "yes," and was

[8] Uranium recovery includes underground mining, open pit mining, and in-situ leach recovery.
[9] [*UNSCEAR*, 2008] combines the enrichment and conversion operations when evaluating worker collective doses
[10] The modeled worker collective doses from fabricating BLEU, and Re-ENU fuel were excluded for this companion exercise

addressed by focusing on the two terms used to calculate worker collective dose: the average dose per worker and the number of workers exposed to this average dose.

2.4.3. Historical Data on Worker Individual Dose from Traditionally-Defined OTC Operations

There are hints that the average annual worker individual doses for both uranium recovery and reactor workers had started to decline in the late 1980s in UNSCEAR (1993). However, in a more recent version of the UNSCEAR survey that includes data through 2002 [*UNSCEAR*, 2008], the trend of declining average annual worker individual dose of a uranium recovery and reactor worker is very evident. This trend is driven by regulatory enforcement that has become more stringent over the years and more vigorous application of the as-low-as-reasonably-achievable (ALARA) radiation protection philosophy. However, the magnitude of the decrease is about the same for both operations and could not account for the substantial change in the relative contribution of these operations to annual worker collective dose from the NFC.

2.4.4. Historical Data on Number of Monitored Workers from Traditionally-Defined OTC Operations

As with the individual worker dose, to the extent that the number of exposed uranium recovery workers has changed over time relative to the number of exposed reactor workers, the contribution of these two operations to total worker collective dose will vary. Data on the number of uranium recovery and power reactor operations workers revealed substantially different trends. The number of uranium recovery workers has declined by 20-fold while at the

same time the number of reactor operations workers increased during the 1990s due primarily to the global growth of nuclear power and has remained essentially constant in the 2000s.

Data concerning the global workforce at power reactors reveal a trend of high growth from an average of 290,000 workers in the 1980s to a peak of 530,000 in the early 1990s and then declining slightly to 437,000 in the early 2000s. To a large extent the reactor workforce trends reflect the global growth of nuclear power the reactor workforce trend in recent years verified by the U.S. reactor workforce trends is essentially unchanging.

Data concerning the number of global uranium recovery workers reveals a much different trend than of reactor operations; the number of global uranium recovery workers has declined precipitously since the 1980s. Specifically, the average number of uranium mining workers in the 1980s was 270,000 but was only 12,000 in the 2000–2002 period, which represents a reduction of ~96% (~26x). The small workforce shown in the early 2000s is the result of a combination of factors, some temporary, some enduring, and some unlikely to reoccur. These factors are related to the combination of changing production technologies (i.e., by transitioning a substantial fraction of uranium production from underground mining to in situ leach and by-product recovery technologies), continuous improvements in worker productivity for all technologies, and the use of remote technologies for some very productive mines containing high-grade ores.

2.4.5. Conclusions and Larger Implications

From the overall perspective of producing nuclear energy, the result that is most important is that the annual collective dose to workers incurred in doing so has declined substantially even though the production of nuclear energy has increased.

The relative contributions of annual collective dose from uranium recovery and reactor operations to total annual collective dose from the once-through NFC developed in this analysis are clearly different from what they were prior to the1990s and compare favorably to recent predictions of the VU-EPRI model.

Analysis of recent information concerning annual worker collective doses from the NFC reverses the relative importance of uranium recovery, and the front end of the NFC, as compared to reactor operations to the point that worker impacts as measured by annual collective dose are dominated by nuclear reactors. The paper summarized here was intended to capture the interest of organizations that are comparatively evaluating NFC options due to the potential implications of the results describing the relative attractiveness of once-through vs. reprocessing NFC options (e.g., Wigeland et al. [2014]) with respect to a figure of merit based on estimated occupational radiological impacts.

CHAPTER 3

COMPARING WORKER COLLECTIVE DOSE AND RADIOACTIVE WASTE OF THE MOC TO THE OTC

3.1. Introduction and Background

Examining potential transitions between NFC options is important because an "evolutionary and progressive pathway is likely to be more realistic than a revolutionary pathway that attempts to simultaneously solve all real or perceived fuel cycle issues with advanced technologies [*EPRI,* 2010a]." Even though the modified-open cycle (MOC) is situated in the direct path to several advanced fuel cycles, few studies have investigated the incremental risks or benefits that could be incurred during this initial transition to MOC. Analyses tend to jump directly to an NFC incorporating a fast reactor technology and/or a more comprehensive recycling scheme.

The MOC has been interpreted in two main ways, one with the use of recycled plutonium (Pu) one time in LWRs and then with the use of Pu and reprocessed uranium (RepU) one time in LWRs. The modeling of a dynamic scenario moving from the OTC to the MOC is represented here by incorporating five additional processes compared to the OTC (Figure 3.1): (1) reprocessing UNF to produce Pu and RepU; (2) mixed-oxide (MOX) fuel fabrication; (3) conversion of RepU to the hexafluoride form; (4) enrichment of RepU to produce enriched RepU (ERU); (5) ERU fuel fabrication. The stated benefits of a MOC vary, but revolve around the reduction of Pu inventory and requiring less natural uranium including:

- Providing means of utilizing a “valuable” asset (Pu) that otherwise would be discarded [*Collins et al.*, 2010]
- Reducing the repository space required for UNF [*Wigeland et al.*, 2010)]

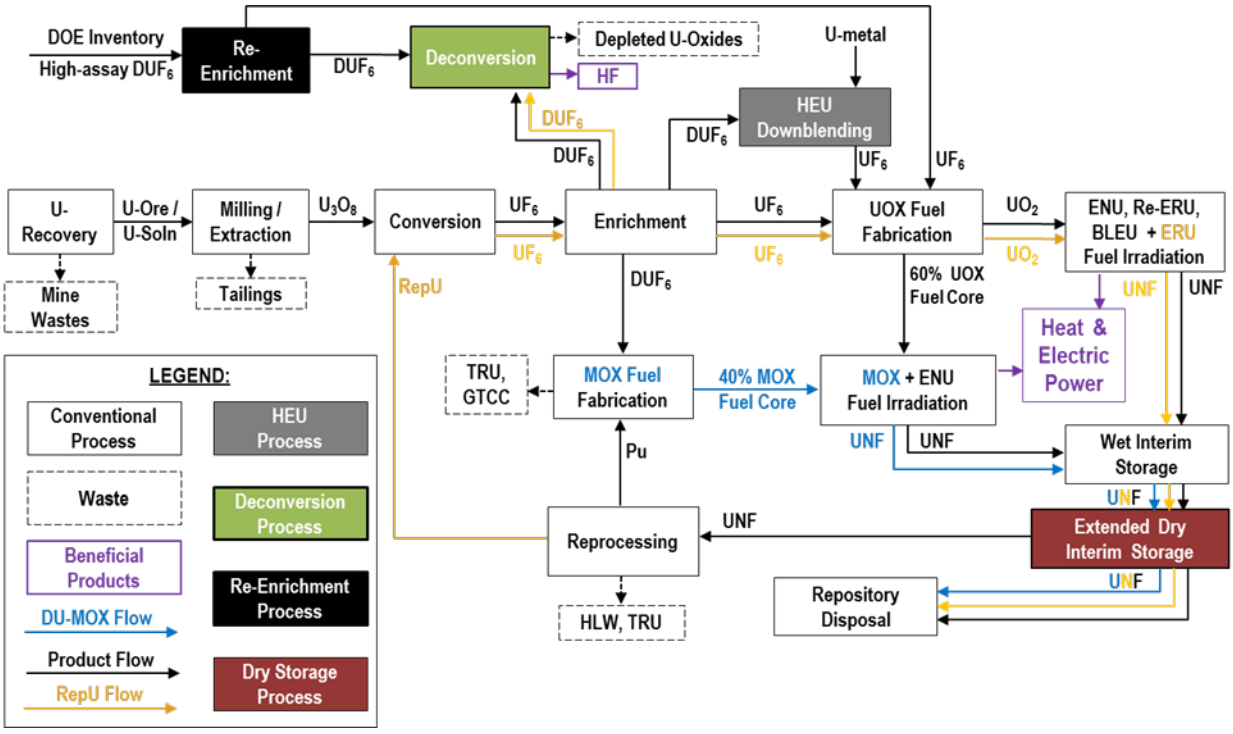


Figure 3.1 | The MOC Conceptual Systems Model

There are a few notable examples of past studies that have looked at possible transitions to the MOC. The related ES&H impacts are described in the following subsection. In addition to the evaluation of worker collective dose in Chapter 2, the volume of radioactive waste generated is now considered. A review of what types of radioactive waste performance metrics have been considered previously is presented and the rationale for the chosen performance metric for this analysis is discussed.

Finally, worker collective doses and volumes of radioactive waste are estimated for a simplified energy demand scenario for the U.S. LWR fleet for a window of 50 years in the future, one that transitions from the OTC (still including the four new processes and traditional OTC operations) to a MOC that utilizes RepU and Pu to help support nuclear energy production. Because this chapter's study of MOC worker collective doses and volume of radioactive waste can be divided as three overall sections of (1) the material flow analysis (2) quantifying performance metrics, and (3) comparison of the two scaled performance metrics of the MOC to the OTC, the methodology and results are discussed topically according to these three divisions.

3.1.1. Objectives of this Chapter

There are four overall objectives of this chapter:

- Determine if adequate data are available to comprehensively evaluate EH&S impacts of NFC options
- Model the U.S. MOC near-future EH&S impacts through investigating impacts of a plausible simplified energy demand scenario for the next 50 years given new processes that have recently become important in the U.S. and operations likely to begin to support the hypothetical MOC presented in this work.
- Compare and contrast EH&S impacts from the transition to the MOC from the OTC
- Identify EH&S risk metrics that are amenable and adequate discriminators of different fuel cycle options (of the small sample presented here)

3.1.2. Past Comparative Studies of the MOC and OTC

One of the earliest studies that recognized the importance of estimating potential EH&S impacts of future NFCs is the "Generic Environmental Statement of the Use of Recycle Plutonium in Mixed Oxide Fuel in Light Water Reactors," (popularly known as "GESMO") compared to that of the traditional operations of the OTC [*NRC*, 1976]. In particular, GESMO evaluated the potential scale of reprocessing Pu and U from once-irradiated, low-enriched natural uranium fuel for reuse as fuel in LWRs within an exponentially increasing energy demand scenario. GESMO estimated by the year 2000 that over 500 GWe installed capacity in the U.S. would have been required to keep with then-current increasing need for electricity and interest in nuclear power (compared to that of ~100 GWe of nuclear energy was installed in the early 2000s). GESMO's objective was to understand potential issues in maintaining an adequate supply of fuel for such an extensive growth of nuclear energy. Because fuel supply was of concern, comparative results of net reductions in the feed stock from uranium mines and following front-end operations were tracked through the 26-year modeling period from 1974 to 2000. A net reduction of 13% of ENU fuel would be achieved if both Pu and RepU were recycled and irradiated in half of the LWRs (~250 reactors) by the end of year 2000. Integrating the net reduction of fuel over the 26 years resulted with a net 20% decrease, if the same increase in energy output had been provided by ENU fuel rather than MOX and ERU. Comparative results shown for waste production included UNF, high-level waste (HLW), transuranic waste (TRU), and low-level waste (LLW) for only reactor and reprocessing operations; this prohibited showing potential net decreases in LLW and other types of front-end waste streams (e.g., mill tailings).

The next effort to understand net differences in likely EH&S impacts was under the multi-national program, the Global Nuclear Energy Partnership (GNEP)[11] that began in 2006 [*DOE,* 2008a; *WNA,* 2012e]. The programmatic EIS that was performed included six programmatic alternatives, DOE's preferred alternative was stated as closing the fuel cycle [*DOE,* 2008a]. Closing the fuel cycle was a means to support GNEP's mission objectives focused around global security (minimizing proliferation)[12] [*NNSA,* 2008]. All alternatives proposed assessed the impacts of the NFC to support a 200 GWe reactor fleet that reflected a growth in U.S. annual electricity demand of 1.3% ended the model in the range of calendar years 2060-2070 (equal to a 50 year implementation period of the new fuel cycle). One of the alternative sub-options was a mono-recycle of Pu in MOX for use in LWRs (MOC). The MOC includes modeling from the current OTC fuel cycle to a complete transition to the MOC alternative. Impacts of waste production (HLW, UNF, GTCC, and LLW) were provided and radiological impacts to workers with units of latent cancer fatalities (LCFs) were integrated over the 50-year modeling time frame for the OTC and the MOC for comparison.

The information to support the underlying calculations of resultant impacts were not provided to discern which operations of the MOC to the OTC are responsible for this increase; however, the information that has been provided was evaluated. MOC using LWRs supported with MOX resulted with cumulative increase in worker LCFs of 4% and a LLW volume increase in a range from no increase to 4x the amount of the OTC. LCF's are linearly scaled from worker collective

[11] GNEP was initiated within the U.S. within the program, "Advanced Fuel Cycle Initiative," (AFCI). AFCI was terminated in 2009 and the remaining countries in GNEP adopted a new name in June 2010 called, "International Framework for Nuclear Energy Cooperation," (IFNEC). The five countries originally part of GNEP were the U.S., Russia, China, France, and Japan. Now IFNEC includes 31 nations [*WNA,* 2012e]

[12] Carrying out GNEP would have likely included support for commercialization of a fast reactor technology and comprehensive recycling scheme that avoided a pure plutonium stream; therefore initial work by NRC was prompted to assess the impacts of alternative recycling technologies [*NRC,* 2008, 2012d]

dose [*NAS*, 2006a] and this study relies on the UNSCEAR (1993) report for worker radiological data (It has been shown in Appendix C and within Krahn et al. (2014) that worker collective doses associated with older data sets from the same UNSCEAR (1993) report equate the dose from uranium recovery to reactor operations at a 1:1 ratio and modern radiation protection practices have changed that ratio to about 1:10).

More modest efforts of modeling potential transitions to future NFCs, looking at limited mono-recycle of Pu and U in LWRs, have been performed by the U.S. Nuclear Waste Technical Review Board (NWTRB). The expansion of nuclear energy past the 104-reactor fleet was limited by the number of reactors that had placed license applications to the U.S. NRC at the time of the study (~25 new reactors) and the modeling of the extended LWR fleet began in 2010 and ran to 2130 [*NWTRB,* 2011, 2012]. The extent of use of Pu in MOX and RepU in enriched RepU (ERU) fuel was limited to reprocessing annual capacity of 1,500 MT UNF per year. The focus of the study was to evaluate complex interactions of the NFC operations when considering the introduction of MOC operations and what scale of waste generation, would result in order to plan waste management efforts for both LLW and repository waste. Net impacts of the MOC to the OTC as estimated by the NWTRB (2011) study resulted with a net decrease of LLW (~15%) and the number of dry UNF casks required by 62% and the number of waste packages sent to the repository (HLW and UNF) was reduced overall 30% compared to the OTC. The amount of natural uranium required to be processed through traditional front-end operations was reduced around 14%.

To avoid the many issues of very different dynamic scenarios described previously, many studies have based the comparison on a simplified steady-state, reactor-to-reactor basis that evaluates an ideal NFC for a given year's performance. Four studies provide sufficient data to make a quantitative comparison of radioactive waste impacts (Table 3.1) and radiological impacts to workers (Table 3.2); these studies only include the use of Pu in MOX and do not consider the further use of RepU.

The net impacts of UNF waste between the MOX reactor-based NFC and ENU reactor-based NFC can be reduced from 25% to 100% (Table 3.1) according to the references cited below. LLW generation spans from a reduction when going to recycle Pu in MOX to an increase of 429% [*DOE*, 2008a]. Decommissioning of 200 reactors contributes to the large difference (and large uncertainty bounds) seen in LLW disposal requirements listed in [*DOE*, 2008a].

Table 3.1 | Comparative Waste Impact Summary of MOC and OTC Steady-State Studies

Case Description (Annual Basis)	[*DOE*, 2008a]	[*NEA*, 2000]	[*Park et al.*, 2011]
Reactors on MOX: Reactors of ENU	1:1	1:1	1:7.25
Natural U Feed	0.84	~0.80	0.88
LEU	0.76	~0.80	0.88
Enrichment Requirements	1.05	~0.80	0.88
UNF to Repository	0.00	0.75	0.07
LLW to Disposal [m^3/yr]	1.0 – 4.29	~0.80	1.22

Notes: It is unclear whether [*DOE*, 2008] included depleted uranium-oxides within the LLW values. [*Park et al.*, 2011] included GTCC, TRU, etc. in the LLW values due to the international system based on the IAEA's waste classification scheme of low-to-intermediate-waste.

Radiological impacts to workers incurred from a MOX reactor-based NFC to that of an ENU reactor-based NFC were estimated to slightly increase from 2% to 9% (Table 3.2). The values of the increase in radiological impacts to workers from the MOX reactor-based NFC for DOE

(2008a) were based upon outdated radiological values from the late 1970s (discussed previously); however, they still fell between the estimates of an NEA (2000) study that used radiological impact from the late 1990s. NEA (2000) reports that overall increase in worker radiological impact is due to increased conversion and enrichment and uses a report from the European Commission (1995) that relies on occupational data from 1990-1992 studies of French facilities [*EC*, 1995].

Table 3.2 | Comparative Radiological Worker Impact Summary of MOC and OTC Steady-State Studies

Case Description	[*DOE*, 2008a]	[*NEA*, 2000]
Reactors on MOX: Reactors of ENU	1:1	1:1
Normal Operation Worker Radiological Impacts	--	~0.80
Mining and Milling	--	~0.80
Conversion and Enrichment	--	13.43
Fuel Fabrication	--	1.00
Reprocessing and Vitrification	--	--
Transportation	--	1.00
Total	1.07	1.02-1.09

Notes: [*NEA*, 2000] relies on mostly industry data from the late 1990s to support calculating the values shown here, but uses [*EC*, 1995] as the basis for conversion and enrichment that represents the years of 1990-1992 for French facilities and uses [*UNSCEAR*, 1993] data from the mid-1980s as values for transportation operations. [*DOE*, 2008a] also uses [*UNSCEAR*, 1993] data for uranium-recovery operations.

The most recent undertaking of estimating steady-state ES&H impacts of future NFC options, was conducted by multiple institutions led by DOE-NE from 2011 to 2014 [*Wigeland et al.*, 2014]. Individual NFCs were grouped, "option groups," and were established to be representative of major functions of a NFC; particular technologies (e.g., reactor designs, enrichment methods, etc.) were not assigned to the NFCs as discussed in the study's charter [*DOE*, 2011a]. This study was done to provide information to support long-term RD&D programmatic goals. Transitions to new NFCs and assumptions about electricity growth

scenarios were not developed as part of the comparative basis for NFC alternatives; instead each option was assumed to be fully deployed at steady-state.

At the time of the writing of this chapter, the final report on the comparative results had not been published and an interim report that was described at a high-level was used to understand initial results, to date [*Wigeland et al.,* 2014]. The high-level results are reported as orders of magnitude in improvement (10:1 OTC impact amount to new NFC impact amount) between the best ranking advanced options to the OTC. Overall, the performance metrics that were most useful in discriminating between fuel cycle option groups (i.e., the net results of the impacts were substantially different) were the waste management and resource utilization metrics. Fuel cycle safety and environmental impacts were moderately useful and showed modest changes in benefits when moving to a future NFC option when running at steady-state. The waste management criteria had a change in an order of magnitude and the resource utilization changed by two orders of magnitude with the most promising NFC option group.

Of the dynamic studies reviewed and presented here, the overall theme is that when only one element is recycled (Pu or RepU) from UNF the net reductions of radioactive waste for a new NFC are not as extensive [*NWTRB,* 2011] and the amount of waste can even be greater than the OTC (as the case for GNEP's PEIS when only using Pu). This signifies that there is insufficient fissile material being recycled (restricted and defined by the assumed reprocessing capacity) that can be used to replace ENU and offset front-end impacts from the additional recycling operations. If both are elements are recycled, and thus replace a greater amount of ENU fuel, then LLW waste avoidance matches closely to the amount of ENU fuel replaced (around 10-

15%). For either case where one or both elements are recycled, the savings of UNF are extensive. The steady-state comparisons of using MOX in an LWR compared to that of ENU, results are consistently showing that worker radiological impacts and LLW waste production increase while the UNF destined for repository disposal decrease dramatically. It is interesting that this result is consistent, even when the use of outdated performance metric data sets and poorly defined study bounds have established.

3.1.3. Radioactive Waste Volumes

Radioactive waste management in the U.S. has historically been considered source-based, meaning that the fuel cycle stage of origin defines waste categorization rather than purely the characteristics of its content [*Croff,* 2006]. The source-based categorization scheme can reflect the inherent level radiological hazard by the required depth beneath the land surface at which the waste is disposed. Moreover, within major source-based categories, divisions of material class have been established such as low-level waste classes (A, B, C, Greater-than-class-C [GTCC]) that are defined by radiological-hazard level [*ACNW,* 2007] and recent events provide evidence that hazard-based approach for dealing with waste is becoming more prominent[13].

There are two established and overall disposition pathways for the multiple U.S. waste streams: (1) shallow-land burial disposal (also referred to in literature as near-surface disposal) and (2) deep land disposal that waste is intended to stay within a repository for durations of time on a

[13] A recent modification to this approach to dealing with waste has emphasized characterizing the content, by enabling risk-informed reclassification schemes. An example of such reclassification is the Waste-Incident-to-Reprocessing (WIR) category that allows HLW to be determined LLW after appropriate processing and disposal [DOE (2012)].

geological time scale (deep geological disposal). The majority of existing U.S. radioactive waste classifications can be divided into either shallow-land burial or deep geological disposal options. The wastes that have known and/or codified disposition pathways for shallow-land burial disposal are: (1) mining waste, (2) byproduct waste (ISL waste, mill tailings), and (3) LLW, MLLW. The wastes that have known and/or codified disposition pathways for geological repository disposal are: (1) UNF, (2) HLW, and (3) TRU.

The disposition pathways for the six radioactive waste streams listed above are known; however there are two radioactive waste categories that have yet to be assigned a definitive disposition pathway: depleted uranium-oxides and GTCC [*DOE,* 2011c; *NRC,* 2012c]. Previously, the NRC allowed licensed LLW disposal facilities to approve or deny disposal of depleted uranium-oxides dependent on the their waste acceptance criteria and limits of activity concentration, volume, and waste form; however, the inventory of depleted oxides is very large and the disposition pathway for the large inventory of depleted uranium-oxides that will result from the continued deconversion operations is currently under review by the NRC [*NRC,* 2012c].

Although the DOE Record-of-Decision for a preferred disposal option for DOE-GTCC has not been released, the option of disposing GTCC at the operating DOE-owned deep geological repository Waste Isolation Pilot Plant located at Carlsbad, New Mexico was included in the draft EIS [*DOE,* 2011c]. DOE's "GTCC-like" waste is still classified under LLW, but can have radiological hazards similar to that of HLW and UNF [*Croff,* 2006]. Commercial GTCC has its own designation under LLW and results mostly from activated metal materials from operating reactors and decommissioning reactors. Commercial GTCC will increase if MOC operations are

employed, specifically when reprocessing occurs due to the requirements of handling contaminated metal from fuel assembly structural material, treatment and technological wastes [*NRC,* 2012d; *Foare et al.,* 2013].

When comparing radioactive waste generation from various NFC options, the radioactive waste performance metrics are kept as major waste classifications (e.g., LLW, HLW, UNF) and priority of elucidating the net benefits of a potential NFC option are most often expressed in terms of geological repository wastes. A widely used waste management metric is the amount (mass or volume) of UNF and HLW produced by the reactor and supporting reprocessing infrastructure [*Piet et al.,* 2006b; *DOE,* 2011c; *Wilson,* 2011]. The draw to this metric is that it directly relates to reactor performance; thus, allowing impacts to be projected for reactor systems with little to no operational experience [*Hill et al.,* 2011] and to reactors with longstanding commercially safe operations. Reactors and resultant waste is a dominant driver in some evaluations of advanced NFCs.

Radioactive waste performance metrics have been less frequently evaluated outside of reactor and reprocessing operations. Using waste metrics that are generated at the remainder of NFC facilities can provide an enhanced ability to discriminate NFC options [*DOE,* 1997]. LLW results from MOC and OTC stages are available, thus tracking LLW when performing NFC comparative assessments is able to capture finer levels of detail when comparing individual operations. Because natural uranium and front-end operations are unlikely to become obsolete with many of the popular advanced NFCs (e.g., even fully closed) due to a constant need to replace inefficiency losses, the wastes specific to the front-end operations remain useful waste

performance metrics to track. These waste streams include: production of waste rock and overburden at conventional uranium mining sites [*EPA*, 1983a], mill tailings at the milling phase [*Hartley et al.*, 1984; *DOE*, 1997], and depleted uranium-oxides at the deconversion stage [*Spradley & Camper*, 2009]. These are typically not included in NFC evaluations but inventories of waste and data to build metrics are readily available; also, the amounts of these streams run to the hundreds of thousands of metric tons.

A tenet of conducting comparative NFC options studies is to include the comprehensive scope of the impacts from constituent operations. This has led the author to believe that waste management material flows and radioactive waste volume impacts should include waste classifications that are usually ignored in other studies (e.g., mine wastes, depleted uranium-oxides) are thus, the two general categories will be used in this study: (1) waste that have the potential to be disposed through shallow land burial disposal methods, and (2) waste that have the potential to be disposed within a geological repository. Shallow land burial eligible wastes are the following:

- Mine overburden
- Byproduct waste (ISL waste and mill tailings)
- LLW and MLLW
- Depleted uranium-oxides from deconversion (based on hazard and past practices)

Geological repository-eligible wastes that are considered and tracked as part of the waste material flow analysis are UNF, HLW, TRU, and GTCC (based on the draft DOE-EM EIS).

3.2. Methods

Similar in structure to the Methods Section in Chapter 2 (Section 2.2), this chapter evaluates the MOC worker collective doses and volume of radioactive waste. The discussion is divided into three overall sections of (1) the material flow analysis and the rationale for selected modeling parameters from a literature review (Section 3.2.2), (2) quantifying performance metrics (Sections 3.2.3 and 3.2.4), (3) scaling performance metrics to the material flow analyses found in the first section and then comparing the two scaled performance metrics of the MOC to the OTC (Section 3.2.5) for the simulation time frame.

3.2.1. Modeling Platform and Calculational Tool

Similar to Chapter 2, the comparative impact assessment presented in this Chapter 3 was performed as a deterministic evaluation within a spreadsheet modeling platform (using Microsoft Excel). The deterministic values for input values were found by taking the average of input value options that were collected through literature reviews. This discussion has been structured in the methods subsections such that the deterministic input value used is stated and then the available range of input values is provided in parentheses. A high-level discussion of the uncertainty of input values is provided if multiple values were available from literature (often times, only one deterministic value was found in the literature).

3.2.2. MOC Material Flow Analysis

Quantitative analysis of the collective occupational radiological risk was performed with essentially five modeling steps focusing on mass flows of uranium fuel supply, mass flows of DUF_6 management and UNF dry interim storage activities at reactor sites:

1. Establishing a simplified nuclear energy demand scenario for the 50-year simulation time period (2013 to 2063) by assuming a 1% growth per year.
2. Distributing the annual fuel requirements for the LWR fleet distinct supply sources (the extent of their use is described in the following Sections):
    a. **BLEU:** Blended-low enriched uranium fuel, a product from downblending HEU and DU
    b. **Re-ENU:** Re-enriched natural uranium, higher-assay DUF_6 tails that are passed through the enrichment stage again to produce low-enriched fuel
    c. **ENU:** Enriched natural uranium, fuel sourced from conventional means of uranium mining, milling, conversion, enrichment, and fuel fabrication.
    d. **MOX:** Mixed-oxide of Pu separated from UNF at the reprocessing MOC operation mixed with DU resultant from the enrichment phase
    e. **ERU:** Enriched reprocessed uranium that is separated from UNF at the reprocessing MOC phase and passes through the enrichment stage again to produced low-enriched fuel made from RepU
3. Apply deconversion and tails waste management parameters to re-enrichment and NU enrichment phases, while incorporating depletion of the DOE low-assay and high-assay DUF_6 stocks

4. Calculate the back-end ISFSIs expansion, maintenance, and required withdrawals for reprocessing at existing reactor sites
5. Scale the mass-normalized, energy-normalized, and dry cask-normalized performance metrics to find total collective dose for workers and radioactive waste volumes[14] associated with the MOC

3.2.2.1. Energy Demand Growth and Reactor Annual Fueling Requirements

The same 1% growth in nuclear energy demand as assumed for the OTC is used for the MOC analysis. The PWR and BWR operating parameters remain the same as well with the MOC as the OTC and the ratio of PWR:BWR reactor technologies remains 2:1. The same ranges of input values that were presented in Chapter 2 are also applicable to the reactor operating input values for LWRs in this chapter.

The three fuel types from the OTC that are utilized in the MOC are BLEU, Re-ENU, and ENU. MOX and ERU are two additional nuclear fuel types that are introduced with the MOC. In this analysis, MOX and ERU are only used in PWRs due to the more extensive experience with MOX in PWRs and the French PWR-based experience with ERU [*Bernard,* 2007; *Areva,* 2010b)]. MOX and ERU fuel performance characteristics are assumed to act similarly to ENU fuel for PWRs based upon a number of studies and reports of operational experience (MOX: [*Watteau et al.,* 2001; *INEEL,* 2003; *Provost & Debes,* 2006]; ERU: *IAEA,* 2007a, 2009; *ORNL,* 2009; *EPRI,* 2010c]); thus, the same PWR operating parameters listed above are assumed for

[14] Radioactive waste volumes for the OTC are calculated in order to perform a comparative analysis of the MOC to the OTC.

MOX and ERU. PWRs are initially loaded with MOX fuel at 40 wt%[15] and 60 wt% ENU, if the entire 40 wt% is available for a full core replacement from current MOX inventories. PWRs are not converted to MOX unless there is sufficient inventory to support refueling of existing and future MOX-converted reactors. The same constraints of ERU inventory availability as MOX were applied to PWRs when switching to the 100 wt% core of ERU fuel.

The extent of fuel type use is divided into three categories: (1) ten (10) percent of LWRs fuel requirements (BLEU) ; (2) as much as NFC facilities can produce annually and support existing reactors for Re-ENU, MOX, and ERU; (3) any remaining fuel requirements is ENU. In this analysis, priority of use corresponds to the category number.

3.2.2.2. Front-End Operations

The front-end operations that are used to produce ENU are adopted from the OTC modeling parameters and details are provided in Section 2.2.2.2.

3.2.2.3. Reprocessing Operations

The MOC worldwide is evaluating moving towards a PUREX[16]-derivative process that addresses emerging global non-proliferation policy by keeping a combined U-Pu stream throughout the reprocessing and MOX fuel fabrication process [*NRC*, 2012d]. The most

[15] The percent of the core loaded with MOX ranges from 33% to 100%. Using a percent core loaded with MOX value other than 40% would result with differing number of PWRs that could be loaded with MOX as the amount of MOX produced remained constrained to the assumed set amount of reprocessing capacity.
[16] PUREX = plutonium-uranium reduction and extraction (reprocessing technology)

advanced plant is the Rokkasho Reprocessing Plant in Japan that recombines RepU and Pu streams prior to fuel fabrication [*IAEA,* 2003; *WNA,* 2013d]. France is developing the co-extraction (CO-EX) process and has plans to build a first-of-a-kind facility in China [*Areva,* 2011c]. Consistent with other recent analyses, capacity is modeled after the Rokkasho facility of 800 MT UNF/yr [*IAEA,* 2008; *Areva,* 2011b, 2011d; *Murray et al.,* 2012)]. Data availability for quantifying worker collective dose and radioactive waste volume metrics has dictated the use of aqueous PUREX technology (or a close variant) as the reprocessing technology likely to be used for this MOC study. The annual capacity is set at 800 MT UNF/yr and the U-Pu recovery is set as 99.8%, based on literature values ranging from 98 to 99.9% [*NEA,* 1989; *Areva,* 2010a; *DOE,* 2010; *NRC,* 2012d]. A reprocessing plant is assumed to begin operations at full capacity 15 years from the start of the simulation time frame, with the ability to add another 800 MT UNF/yr capacity every decade thereafter until 2048 (3 total); this set of assumptions is consistent with the recent NEA (2013a) analysis and is used within this study.

3.2.2.4. UNF Inventory Available for Recycling

Optimized time between discharging UNF and reprocessing has been analyzed to be around 3-5 years based on buildup fission product decay, minimizing loss of fissile Pu-241, along with buildup of Am-241 that affects ease of reprocessing [*EPRI,* 2006a, 2009a; *DOE,* 2010; *Arnold et al.,* 2012]. However, this time span of 3-5 years does not closely match operations in the U.S. The average spent fuel pool time for an assembly is 10 years [*EPRI,* 2008a, 2010b], considering the lag time for handling until reprocessing can occur adds another 2 to 5 years [*NEA,* 1989;

IAEA, 2005b]. Thus, the likely time period between reactor discharge and reprocessing is 15 years[17].

The timing of when the reprocessing facility opens (2028, 15 years after simulation start time, 2013) also matches the age of UNF to be reprocessed, meaning that the UNF available for reprocessing is discharged beginning in 2013 with the annual inventory around 2000 MT UNF each year. In this analysis, a one-time recycling scheme of Pu and U is adopted such that the only UNF eligible for reprocessing is that of ENU, Re-ENU and BLEU. Irradiated MOX and ERU are not considered for reprocessing and are stored for disposal. The inventory of UNF in the U.S. has been extensively documented and categorized [*DOE,* 2011b, 2012a; *Wagner et al.,* 2012] and the majority of fuel existing has isotopic content that is a product of lower burnups and initial U-235 enrichment and less extensive burnup levels. Updated UNF isotopics were calculated for 15-year cooled fuel based off of aging 10-year cooled fuel, resulting with total U and Pu in 15-year UNF is around 93.7% and 1.34%, of the total UNF mass [*EPRI,* 2010a; *Collins et al.,* 2010]. The inventory of UNF and the ages of UNF were provided in Wagner et al. (2012) up to the year end of 2010 containing a total of nearly 67,800 MTHM UNF. Assuming that around 2000 MTHM UNF is discharged per year by the reactor fleet, the inventory was tracked and aged by adding an additional ~4,000 MTHM UNF in order to update the inventory for the modeling start year of 2013.

[17] If a shorter time was used for cooling of UNF before reprocessing, theoretically there would be a greater amount of fissile Pu-241. A great amount of Pu-241 would result with a larger amount of MOX that could be produced each year from the same amount of UNF reprocessed and therefore an increased number of PWRs that could be loaded with MOX. However, the data associated with normalized worker collective dose for handling UNF is for older UNF that has been cooled for 10 or more years. The modeling uncertainty would increase due to lack of data on handling UNF that has cooled for less than 10 years.

3.2.2.5. MOX Fuel Fabrication

There are operating MOX fuel fabrication facilities in six countries, mostly producing MOX from separated Pu and depleted uranium (DU) [*WNA,* 2013e]. Future plans are mixed, with France and Japan planning to use RepU with Pu and the U.S.'s MOX Fuel Fabrication Facility (MFFF) using DU as a diluent [*NRC,* 2005]. Because 15 year old UNF is used in this analysis, MOX compositions that are modeled herein will incorporate 10 wt% Pu and 90 wt% DU (around 0.25 wt% U-235) to achieve the assumed 55 GWd/MT burnup [*IAEA,* 2003; *Porsch et al.,* 2005; *Trellue,* 2006; *Inagakit et al.,* 2009]. MOX fuel fabrication facilities are assumed to be co-located with the reprocessing facility within this study that is consistent with other studies and is used herein [*Areva,* 2011c]. The rate of MOX fuel production is assumed to consume the plutonium output of the reprocessing plant (typically 8 MT UNF is reprocessed to produce one MTHM MOX [*Areva,* 2011c]); however, to account for processing 15 year-old fuel, the ratio of 8.22:1 is used. This means that when the reprocessing plant is operating at full capacity, 97 MTHM MOX/yr can be produced [*WNA,* 2013e, 2013c].

3.2.2.6. Recycled Uranium Processing

Recycling uranium recovered in reprocessing has been done by enriching the RepU stream by conventional enrichment technology [*IAEA,* 2005b; *WNA,* 2013f]. Other means of using RepU can include blending higher enrichment unirradiated or recycled uranium, or a combination of both enriching and blending [*Del Cul et al.,* 2009; *IAEA,* 2009a; *WNA,* 2013f]. Our model for RepU utilization assumes it is accomplished through conversion and enrichment. ERU fuel

fabrication involves three steps: (1) conversion, (2) enrichment, and (3) oxide fuel fabrication. The composition of RepU is different than the composition of natural uranium (NU) resulting from the presence of (a) small amounts of U-232 (requires additional radiation shielding), and (b) U-236 which is a neutron absorber and, subsequently requires higher U-235 enrichment levels (the "U-236 penalty," see below) in fuel [*Del Cul et al.*, 2009; *WNA*, 2009]. Separate conversion and enrichment lines are typically used (due to U-232), consistent with present French practice [*IAEA*, 2009a; *WNA*, 2013f].

For this paper, it is assumed that the currently employed conversion technology is used for converting RepU to $RepUF_6$ with an efficiency of 99.9% and the annual capacity is modeled to match the RepU production from the assumed reprocessing plant capacity. The operations for ERU fuel preparation begins in 2028, but with fuel qualification and material handling time lags, it is assumed that there is a 5-year delay from start of reprocessing operations to the use of ERU in PWRs [*IAEA*, 2007a]. Estimations of the duration to qualify a fuel have been adopted from IAEA (2007a).

Enrichment of RepU to increase the fissile U-235 content must account for an increased parasitic U-236 content [*IAEA*, 2007a, 2009a, *Del Cul et al.*, 2009]. We assume that U-236 levels will not exceed the ASTM[18] limits (as listed in Del Cul et al. [2009]) and RepU-235 is enriched to higher levels to compensate for the U-236 penalty with an addition of 0.5 wt% U-235 leading to PWR ERU fuel being enriched to 5% U-235. Separative work units (SWU) requirements for enriching RepU from 0.97 wt% to 5 wt% U-235 require 6.76 MT SWU per 1 MTHM ERU, using standard enrichment value functions [*MIT*, 2001; *EPRI*, 2010b]. By the time RepU enrichment is needed

[18] ASTM = (U.S.) American Society for Testing and Materials

the LES NM centrifuge plant and the USEC American Centrifuge facility should both be operating; thus sufficient overall capacity should be available to support dedicated RepU enrichment.

The ERU fuel fabrication facility in France is located at the Romans UOX fuel fabrication and it has an annual capacity for ERU fuel fabrication of 150 MTHM ERU/yr [*Areva,* 2011c; *WNA,* 2013f]. The same facility capacity is adopted for this model with an efficiency of 99.9% of processing heavy metals. It is assumed that ERU centrifuge plant capacity will match the ERU fuel fabrication capacity (120, 240, 360 MTHM ERU/yr) building new fuel fabrication capacity as required.

3.2.2.7. DUF_6 Management and Deconversion

DUF_6 is produced from the enrichment and re-enrichment of ENU, Re-ENU, and ERU. The amount of DU produced per 1 MTHM of each type of fuel was calculated with standard conservation of mass and value functions related to SWU estimations [*MIT,* 2001; *EPRI,* 2010b]:

- ENU: (PWR @ 4.5wt%): 6.88 MT DU; (BWR @ 4.35 wt%): 6.57 MT DU
- Re-ENU: (PWR @ 4.5 wt%): 12.78 MT DU; (BWR @ 4.35 wt%): 12.26 MT DU
- ERU: (PWR @ 5 wt%): 6.77 MT DU (not used in BWRs for this model)

DUF_6 tails generated domestically in the model can be used in the production of MOX (10 wt% Pu, 90% DU) and BLEU (5 wt% HEU, 95 wt% DU), or if not used in fuel, are deconverted and stored until final disposal. Tails that are produced internationally are assumed to be deconverted to a more stable oxide form and disposed as low-level radioactive waste [*WNA,* 2013g].

Within the model, we assume that the present Paducah and Portsmouth facilities are only allowed to deconvert non-DOE material when excess capacity becomes available; Paducah is assumed to stop deconversion after the backlog of non-DOE material because no operating enrichment or re-enrichment will remain (DUF_6 storage and maintenance will continue throughout the model for depleted oxides at Paducah). Deconversion activities after the point of completion of deconverting the DOE re-enrichment backlog will be divided equally between the sites that have operating centrifuge enrichment plants in the model.

3.2.2.8. Dry Interim Storage

Dry cask interim storage began in 1986 [*VEPCO,* 2002; *NRC*, 2012h] and an estimated 73 independent spent fuel storage installations (ISFSI) will exist in the U.S. by 2020 [*EPRI,* 2010b]. Continued expansion of the ISFSIs is modeled with each DSC holding UOX fuel storing 13.05 MTHM UOX UNF [*BRC,* 2012a]. It is assumed that ERU UNF can be handled similarly to other UOX fuel [*IAEA*, 2007a, 2009a]. Decay heat levels from irradiated MOX are higher than those of UOX UNF in the short-term, during dry interim storage [*Arndt et al.*, 2003; *IAEA*, 2007]; thus, a lower heavy metal capacity of MOX DSCs is assumed in modeling. AREVA's experience with storing irradiated MOX is extensive and thus, the MOX DSC capacity is set at the present limits of their Transnuclear TN 24^E design that can hold up to 17 PWR MOX fuel assemblies [*IAEA,* 1999; *Areva,* 2010b]. A standard 17x17 PWR UOX fuel assembly contains approximately 0.460 MTU [*Roddy et al.,* 1986] and it is assumed that the difference in bulk density of MOX and

UOX is negligible. Therefore, the DSC capacity of holding solely MOX fuel assemblies was estimated to be 7.82 MTHM/DSC.

3.2.2.9. Reprocessing Waste

Handling of reprocessing waste is logistically modeled after the La Hague plant that co-locates the HLW vitrification process and the compaction of metals from fuel assembly hulls and other structural material [*Areva,* 2011d; *Foare et al.,* 2013]. HLW contains 0.2 wt% of the U and Pu (reprocessing efficiency was set at 99.8 wt% of major actinides). Vitrification technology at La Hague now incorporates cold crucible induction melter and has waste loadings ranging from 18.5 to 25 wt% vitrified HLW[19] [*Areva,* 2011c; *Murray et al.,* 2012; *Foare et al.,* 2013]. The average of these values was used and a waste loading of 22% is considered in this model. Process wastes containing fission products are placed in universal canisters for vitrified waste (UC-V) which can hold approximately 1.43 MT HLW/container, with the specified glass loading limits [*Foare et al.* 2013]. Compacted metal waste is considered GTCC and is discussed in Appendix B.

3.2.2.10. MOC Material Flow Analysis Methods Summary

Below is a bulleted list of the modeling parameters that were used for the OMOC (as described above). The assumptions that were carried forward from Chapter 2 are denoted with an asterisk (*):

[19] If a higher mass loading rate of HLW within the vitrified HLW was used, the number of vitrified glass logs would be lower. The opposite would be true if a lower mass loading rate of HLW within the glass logs were used in the calculations.

- *Reactor Fleet and Bringing Reactors Online:
    - *A new reactor is assumed to begin operations when the energy demand is 500 MWe-yr higher than the existing reactor capacity at that given year.
    - *The reactor fleet is 2/3 PWRs and 1/3 BWR throughout the simulation.
    - *When reactors are retired and brought online, both the defueling stage and initial fueling occurs at the same frequency and duration of regular annual fueling (that occurs in batches) to prevent large prevent large spikes and fluctuations with discharged UNF and loading to and from the reactor.
- Priority of fuel used in reactors is:
    - *10% of annual total fueling requirements will be met by BLEU;
    - *All the Re-ENU fuel produced that year is loaded and used by reactors;
    - All the MOX and ERU that can be loaded given that sufficient inventory from an assumed level or reprocessing is available to perform a full core loading for ERU (40% PWR core for MOX; 100% PWR core for ERU) while maintaining the ability to meet annual refueling needs of MOX and ERU.
    - *The rest of fuel requirements for any given year is met by the use of ENU.
- Inventories of nuclear materials at beginning of year 2013:
    - *Natural Uranium (NU) = unlimited
    - *Used nuclear fuel (UNF) = 70,000 MT
        - *Wet storage = 18,000 MT UNF (BWR); 33,000 MT UNF (PWR)
        - *Dry storage = 6,000 MT UNF (BWR); 13,000 MT UNF (PWR)
        - *UNF already in dry storage casks (DSC) = DSC capacity = 11.25 MT UNF/DSC

        - *UOX UNF discharged from either a PWR or BWR during simulation to DSC = 13.05 MT UNF/DSC
        - MOX UNF discharged from a PWR during simulation to DSC = 7.82 MT UNF/ DSC
    - *Highly-enriched uranium (HEU) = 525 MT HM
        - *HEU mass proportion used in BLEU is 5%; DU mass proportion = 95%
    - * Higher-Assay DUF6 DOE Inventory = 73,500 MTU in DUF6
    - *Lower-Assay DUF6 DOE Inventory = 362,474 MTU in DUF6
    - *DU-oxides of DOE Inventory already deconverted = 51,526 MTU in DU-Oxides
    - Global Pu Inventory in UNF = 1,300 MT Pu
        - Global production from LEU UNF = 70 MT Pu/yr
        - Pu Inventory already separated from UNF = 370 MT
            - Pu used in MOX and irradiated from already separated Pu Inventory = 130 MT Pu)
- U-235 wt% in each nuclear material stream:
    - *Used nuclear fuel (UNF) of UOX fuel not previously irradiated= 93.7 (does not change with aging over 50 year modeling)
    - *Highly-enriched uranium (HEU) = Assumed at 90 wt%; DU as diluent at 0.20 wt%
    - *Higher-Assay DUF6 DOE Inventory = Average assay of 0.30
    - *Lower-Assay DUF6 DOE Inventory = Average assay of 0.25 wt%
    - *DU-oxides of DOE Inventory already deconverted = Average assay of 0.25 wt% U-235

- Reprocessed Uranium (RepU) = Average assay of 0.25 wt% U-235
- Enriched Rep U (ERU) = 5 wt% U-235

- Nuclear Fuel Cycle Facility Operations and Characteristics
  - *NU supply country = 10% US (10% = 2.5% UG; 7.5% ISL); 90% International (90% = 33.8% OP + 15.6 UG + 40.7 ISL)
  - *Conversion: Use of U.S. facilities are priority for conversion until demand exceeds US capacity and assumption that international partners have sufficient capacity
  - RepU Enrichment to make ERU: Priority to use enrichment facilities in the U.S. is placed on ERU production before ENU production
  - *ENU enrichment: Use of U.S. facilities are priority for ENU enrichment until demand exceeds US capacity and assumption that international partners have sufficient capacity
  - *Re-ENU Re-enrichment: Annual enrichment capacity for re-enrichment is limited to half of Paducah's gaseous diffusion plant for year 2013 (4000 MT SWU/yr). For 2014, the centrifuge enrichment plant in New Mexico is assumed to take over with half of the capacity (1,500 MT SWU/yr)
  - *ENU and Re-ENU Fuel Fabrication: Use of U.S. facilities are priority for fuel fabrication until demand exceeds US capacity and assumption that international partners have sufficient capacity
  - *BLEU Fuel Fabrication: 100% of BLEU fuel is manufactured at the Areva Richland Fuel Fabrication plant, any remaining capacity of Areva Richland plant is used for ENU or Re-ENU

- MOX and ERU fuel fabrication: New facilities are brought online specifically for MOX and ERU fuel fabrication. The annual capacity of the MOX facility matches the output rate of reprocessing. The annual capacity of the ERU fuel fabrication matches the output rate of the reprocessing and then enrichment step of ERU preparation.
- *Deconversion: The DOE faculties at Paducah and Portsmouth are only allowed to deconvert commercial DUF6 material if the DOE inventory of lower-assay DUF6 material is worked down to levels lower than the annual capacities of both plants.
- *At-Reactor ISFSIs: Number of sites in 2013 = 64; Increases incrementally to 73 sites total modeled by year 2029

3.2.3. Quantifying Worker Collective Dose Performance Metrics

Metrics in this paper are developed with the assumption that operations at fuel cycle facilities produce a linearly related impact, in this case volume of radioactive waste and worker collective dose. All operational phases are normalized by mass flow of major fissile content, except for the reactor operation which is normalized by electrical energy production; thus scaling of EH&S risk estimates can be done for each major NFC facility.

This is important to note that data sets have been used for this work that represent actual measured levels of worker collective doses and are based upon real events and industry experience with emphasis on collecting as recent data as possible that is within the public

domain. The emphasis was also placed on using most recent measurements due to observed trends that doses are generally decreasing worldwide from many factors but mostly because of the ALARA (As-low-as-reasonably-achievable) operational philosophy.

Required data sets for worker collective dose were presented previously in Section 2.2.3 for OTC operations and the resultant metrics are repeated here in this section. Some of the worker collective dose metrics parameter data sets were available in a straight forward fashion for the MOC operations; some were not and the data sets required additional explanation and calculations are explained in Appendix A. A summary of some of the major assumptions for quantifying worker collective dose metrics for the MOC operations is presented below. The calculated worker collective dose performance metrics are presented here for operations related to the MOC and are presented in Table 3.3.

Workers receive relatively small occupational doses each year during routine NFC facility operations. Individual doses for monitored workers are low compared to the annual limits in 10 CFR 20 of 50 mSv (5 rem) and, over long time periods do not scale linearly to operational throughput of nuclear material because of ALARA operational procedures [*NRC,* 2012e; *Krahn et al.,* 2014]. However, over short time periods, the impacts can be assumed to scale linearly.

Worker collective doses are estimated for all operations listed in Table 3.3. The bulleted list of normalized worker collective dose metric modeling assumptions for Chapter 3 is below, (the assumptions that were carried forward from Chapter 2 are denoted with an asterisk [*]):

- *The preferred and most common source of data used for modeling parameters and building normalized impact metrics were taken from actual measured doses of operating facilities worldwide. The emphasis was also placed on using most recent measurements due to observed trends that doses are generally decreasing worldwide from many factors but mostly because of the ALARA operational philosophy (ALARA stands for As-low-as-reasonably-achievable).
- *Worker collective doses as part of handling and disposing of UG mining, OP mining, and milling are included in the normal operations of these mines and mills each year. Data separating the onsite activities of waste and operations that directly work to make the product are not readily available.
- *No worker collective doses are calculated for disposing of depleted uranium-oxides (which is not presently occurring). Worker collective doses for storage and maintenance of depleted uranium-oxides are assumed to be included in the normal operations of deconversion.
- *Worker collective doses for maintenance, refueling outages, and wet interim storage are included in the worker collective doses normalized metrics for normal reactor operations. Data were not readily available for those individual activities at reactor sites and were included within reported values to the NRC [*NRC*, 2011e, 2012b].
- Dry interim storage impacts for UOX DSCs [*EPRI*, 2008a, 2010b] are assumed to be similar for DSCs storing MOX. Handling and storage of GTCC, TRU, and HLW at the reprocessing and TRU at the MOX fuel fabrication plant are assumed to be part of the normal operations and so reported.

- Worker collective doses from waste handling during the reprocessing operation are included in the normal operations worker collective dose normalized metric. Waste handling includes handling HLW, vitrifying HLW, storing HLW, storing GTCC and TRU material that are not modeled to be moved to a repository in this work.
- Worker collective doses for RepU enrichment are calculated based on a ratio of the ERU and UOX fuel fabrication and scaled to the available centrifuge enrichment values for ENU (see Appendix A, Section A.4)

Table 3.3 | MOC Worker Collective Dose Performance Metrics

NFC Operation	Worker Collective Dose	
Technology	Value	Units
Mining		
Open Pit	3.69E-01	person-mSv/MTNU
Underground	4.08E-01	person-mSv/MTNU
ISR/ISL		
Alkaline Solution	6.26E-01	person-mSv/MTNU
Milling		
Acid Leach	2.48E-01	person-mSv/MTNU
Conversion		
Dry	1.18E-01	person-mSv/MTNU
Wet	1.01E-02	person-mSv/MTNU
Deconversion		
DUF_6 to DU Oxides	4.53E-02	person-mSv/MTDU
Enrichment & Re-enrichment		
Diffusion	1.70E-01	person-mSv/MTLEU
Centrifuge	1.36E-01	person-mSv/MTLEU
Downblending		
HEU metal with Low-Assay DUF_6 Tails	8.09E-01	person-mSv/MTBLEU
Fuel Fabrication		
ENU (Hands-On Technology)	1.75E+00	person-mSv/MTENU
BLEU (Hands-On Technology)	1.75E+00	person-mSv/MTBLEU
Fuel Irradiation (Reactor Operation)		
BWR	1.54E+00	person-mSv/MWe-yr
PWR	6.28E-01	person-mSv/MWe-yr
Interim Storage [Dry on-site (assumed no LLW produced)]		
Loading operations	4.00E+00	person-mSv/DSC
Expansion Construction	1.70E+00	person-mSv/DSC
ISFSI operations and maintenance	1.50E+01	person-mSv/ISFSI site/year
ISFSI inspections and security surveillance	1.20E+00	person-mSv/ISFSI site/year
Reprocessing		
Aqueous - PUREX	5.21E-01	person-mSv/MTUNF
Conversion for RepU		
UN to UF_6	5.59E-03	person-mSv/MTRepU
Enrichment of RepU		
Centrifuge	1.36E-01	person-mSv/MTERU
Fuel Fabrication		
MOX Fuel Fabrication (Glove-box Technology)	1.65E+01	person-mSv/MTMOX
ERU Fuel Fabrication (Hands-on Technology)	1.40E-01	person-mSv/MTERU
Shallow Land Disposal		
Shallow-Land Burial (LLW, MLLW)	2.20E-02	person-mSv/m^3 LLW, MLLW

3.2.4. Quantifying Radioactive Waste Volume Performance Metrics

The minimum number of required data sets for volumetric radioactive waste were the annual volumes of waste destined for shallow-land burial [m^3 /yr] and the annual throughput of nuclear material [MTHM/yr] (electrical energy for reactors was used). The performance metric is the quotient of the annual volume of radioactive waste and the annual throughput of product at the nuclear facility [m^3/yr]/[MTHM/yr].

Required data sets to develop radioactive waste volume performance metrics were not discussed in Chapter 2 and require descriptions of sources and assumptions. Similar to the worker collective dose metric data sets, straight forward information was prevalent and additional discussion with calculations are provided in Appendix B. The calculated radioactive waste volume performance metrics are presented in see Table 3.4 and a summary of the major assumptions for quantifying the radioactive waste volume performance metrics for the MOC operations is presented below.

Radioactive waste volume metrics were quantified for types of waste that can be disposed by shallow-land burial (SLB) since final disposition pathways are known and operations go on currently.

- The normalized metric used is the volume of radioactive waste that is eligible for shallow-land burial disposal. This includes the waste classifications of:

- Technologically enhanced naturally occurring radioactive material (TENORM) radioactive waste from open pit and underground uranium mines [*EPA*, 2008] (this is referred to as “mining overburden” and “mining waste” in the text),
- Byproduct radioactive waste from in-situ leach uranium recovery operations and milling operations. [*EPA*, 1983b; *NRC*, 2012g]. Mill tailings are classified as byproduct material under the Uranium Mill Tailings Radiation Control Act of 1978 (this is referred to as “in-situ leach waste” and “mill tailings” in the text),
- Low-level radioactive waste (LLW) from all NFC operations except for operations listed above that produce TENORM and Byproduct radioactive wastes. LLW is referred to only as “LLW” in the text,
- Mixed low-level radioactive waste (MLLW) from a number of NFC operations, but are modeled here as conversion (dry technology only), enrichment (diffusion and centrifuge technologies), and MOX fuel fabrication (glove-box technology). MLLW is referred to only as “MLLW” in the text, and
- Depleted uranium-oxides from deconversion operations (although the final disposition pathway is currently under review by the NRC). This waste classification is referred to as “depleted uranium-oxides” or “depleted U-oxides” in the text.

- If waste volumes are described in any other term other than “radioactive waste volume” or “radioactive waste eligible for shallow-land burial disposal,” then they are called by their specific waste classification name that is described above (e.g., if text refers to “LLW and byproduct waste”

- It was assumed that gaseous diffusion technology produces MLLW but data were not available to calculate MLLW for gaseous diffusion technology. Gas diffusion MLLW volumes were extrapolated to the ratio of MLLW to LLW that gas centrifuge produces.

Wastes that are anticipated to require disposal in a deep geological repository are GTCC, TRU, UNF, and HLW. These four wastes streams are tracked as part of the waste management material flow analysis portion of this work and impacts from storage and handling are considered, but not disposal (geological repository impacts will be modeled in Chapter 4).

The majority of radioactive waste volume performance metrics that were considered related to reprocessing was adopted from Foare et al. (2013). The underlying assumptions for reprocessing rely on La Hague operational data, except for the treatment of I-129, which the Rokkasho Reprocessing Plant (Japan) technology for trapping off-gases were used.

Table 3.4 | MOC Radioactive Waste Volume Performance Metrics

NFC Operation	Radioactive Waste Volume	
Technology	Value	Units
Mining		
Open Pit	6.06E+03	m^3 overburden/MTNU
Underground	5.48E+02	m^3 overburden /MTNU
ISR/ISL		
Alkaline Solution	1.90E-01	m^3 byproduct /MTNU
Milling		
Acid Leach	1.01E+03	m^3 byproduct (mill tailings)/MTNU
Conversion		
Dry	3.71E-02	m^3 LLW/MTNU
	4.76E-04	m^3 MLLW/MTNU
Wet	1.03E-02	m^3 LLW/MTNU
Deconversion		
DUF_6 to DU Oxides	1.51E-01	m^3 LLW/ MTDU
	5.65E-01	m^3 DUO_2/ MTDU
Enrichment & Re-enrichment		
Diffusion	1.20E-01	m^3 LLW/MTLEU
	6.91E-06	m^3 MLLW/ MTLEU
Centrifuge	1.36E+00	m^3 LLW/MTLEU
	7.84E-05	m^3 MLLW/MTLEU
Downblending		
HEU metal with Low-Assay DUF_6 Tails	3.07E-01	m^3 LLW /MTBLEU
Fuel Fabrication		
ENU (Hands-On Technology)	4.91E+01	m^3 LLW/MTENU
BLEU (Hands-On Technology)	4.91E+01	m^3 LLW/MTBLEU
Fuel Irradiation (Reactor Operation)		
BWR	6.27E-01	m^3 LLW/ MWe-yr
PWR	2.88E-01	m^3 LLW/ MWe-yr
Reprocessing		
Aqueous - PUREX	6.2E-02	MTHLW/MTUNF
	2.8E-01	MTVitrified HLW/MTUNF
	8.4E-01	m^3 Vitrified HLW/MTUNF
	1.81E-01	m^3 TRU / MTUNF
	5.66E-01	m^3 GTCC / MTUNF
	4.65E+00	m^3 LLW/ MTUNF
Conversion for RepU		
UN to UF_6	1.03E-02	m3 LLW / MTRepU
Enrichment of RepU		
Centrifuge	1.36E+00	m^3 LLW / MTERU
Fuel Fabrication		
MOX Fuel Fabrication (Glove-box Technology)	4.43E+00	m^3 TRU/ MTMOX
	1.46E+01	m^3 LLW / MTMOX
	1.20E-01	m^3 MLLW / MTMOX
ERU Fuel Fabrication (Hands-on Technology)	4.91E+01	m^3 LLW / MTERU

3.2.5. Comparing Impacts from MOC to the OTC

A number of methods to compare outputs of multiple options are available. Options studies can express the each of the results as a nominal value or relative value. Of studies looking at NFC options, most commonly the nominal results are tabulated or graphically shown [*DOE,* 2008a]. Placing the options in a relative context to a chosen baseline, the OTC in this case, is a format that allows quicker understanding and interpreting of relative performance of options [*Eastham et al.*, 2012]. A ratio of multiple options to the OTC can be expressed at a time [*Dixon & Wigeland,* 2008], or a pair-wise comparison can be performed and calculated simply by the quotient shown below in Equation 4.1. The ratio of steady-state impacts is denoted as R_i, MOC_i and the OTC_i are the impacts associated with the MOC and the OTC, respectively.

$$R_i = \frac{MOC_i}{OTC_i}, \tag{4.1}$$

Steady-state performance of multiple NFC options can be compared by the above equation. NFC options studies evaluating the transitions between NFC end-states can also use the above equation but the ratio can be used for each discretized time step evaluated or Equation 4.1 can be altered to integrate the impacts over the modeling time frame, as represented in Equation 4.2. *R* denotes the integrated (or cumulative) impacts, *n* is the total number of discretized time steps evaluated, *i* is the individual time step, MOC_i and OTC_i are impacts from each time step.

$$R = \frac{\sum_{i=1}^{n} MOC_i}{\sum_{i=1}^{n} OTC_i}, \tag{4.2}$$

3.3. Results and Discussion

In the following, a summary of the results from modeling the worker collective doses and radioactive waste volumes eligible for shallow-land disposal of the transition from the OTC to the MOC within a simplified energy growth scenario is presented as three parts. The first is the material flows of fuel and waste; the second part of the results is presented as the estimated two performance metrics scaled to the material flows modeled in the first section. Lastly, comparative performances of the MOC to that of the OTC are presented in two forms, relative performance each year of the 50-year modeling scenario and then the integrated impacts from two NFCs are then compared.

3.3.1. Material Flow Analysis

This section includes in the material flow analysis of nuclear material that concerns the fueling requirements of the LWR fleet. First, results are presented on the UNF reprocessed and sent to MOX and ERU fuel fabrication. Second, the total amount of nuclear fuel and the types of UOX and MOX fuel are considered. Next, the extent of use of MOX and ERU in PWRs is presented. Finally, the amount of Pu burned associated with MOX use is evaluated.

3.3.1.1. Reprocessing RepU and Pu

Reprocessing capacity is modeled as a step function and the amount of RepU and separated Pu, MOX and ERU fabricated follows the same increases each ten years 2028 to 2048 (Figure 3.2).

The reprocessed Pu calculated for the 3 decades were (9.8, 19.6, 29.4 MT Pu/yr; red line in Figure 3.2), annual MOX production modeled as (97.0, 194.1, 291.1 MT MOX/yr; green line in Figure 3.2), and separated RepU modeled as (745.8, 1491.7, and 2237.5 MT RepU/yr; purple line in Figure 3.2).

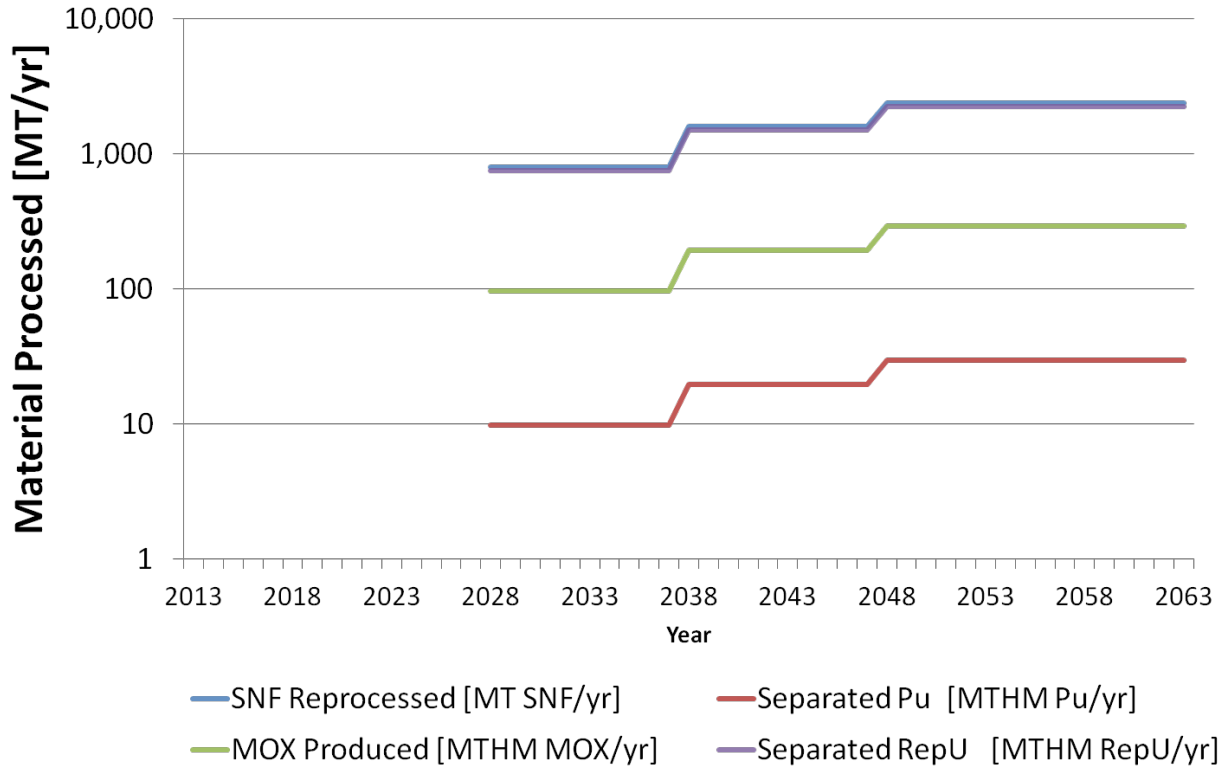


Figure 3.2 | Annual UNF Reprocessing, Pu, RepU and MOX Produced

3.3.1.2. Extent of MOX and ERU Use in PWRs

The extent of MOX use in PWRs is constrained by the annual production of MOX (blue line in Figure 3.3) that is directly linked to the installed reprocessing capacity; hence, MOX production matches the increasing stepped trend of annual reprocessing capacity in decadal intervals. The

annual use of MOX varies (green line in Figure 3.3) around the fuel fabrication amounts as inventory becomes available to support annual fueling requirements of already converted PWRs and then 40% core replacement of MOX to initially convert PWRs. The cumulative number of PWRs that are converted to MOX use is shown as purple circles and corresponds to the right Y-axis (Figure 3.3) In year 2059, MOX use reaches steady state and is used in 34 PWRs (total 101 PWRs projected to be in the U.S. reactor fleet) using ~290 MT MOX/yr and ~430 MT ENU/yr (the 40% MOX and 60% ENU core composition). Effectively, each reprocessing plant can support annual MOX refueling of ~11 PWRs. At the end of the simulation in year 2063, a little less than 25% of the 142-LWR fleet is supported by the MOX/ENU blend.

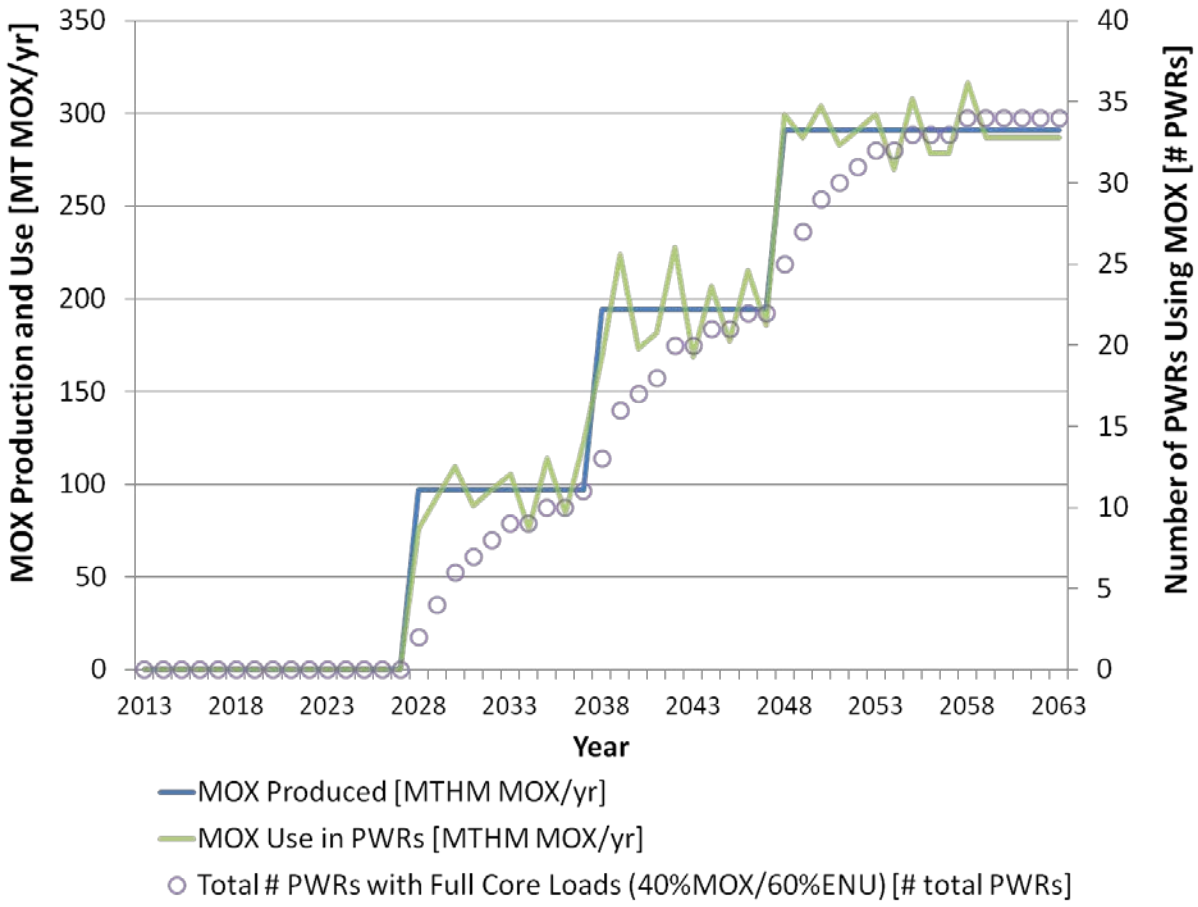


Figure 3.3 | MOX Production and Use in PWRs

Similar to the MOX scenario, the extent of ERU use in PWRs is constrained by installed reprocessing capacity and ERU production matches the increasing stepped trend of annual reprocessing capacity in decadal intervals (Figure 3.4). RepU is produced directly from the reprocessing stage (blue line in Figure 3.4) and then must be enriched. The feed to product ratio of RepU to ERU is 5.7, accounting for minor inefficiencies in enrichment and that reduction in material to be further processed is shown (yellow line in Figure 3.4). Effectively, each reprocessing plant can support annual ERU refueling of ~7 PWRs.

The annual use of ERU varies around the fuel fabrication amounts as inventory becomes available to support annual fueling requirements of already converted PWRs and then the 100% core replacement of ERU to initially convert PWRs. ERU use does not reach steady-state within the model timeframe, but by the end of the simulation in year 2063, ERU is used in 22 PWRs using ~465 MT ERU/yr which is around 16% of the 142-LWR fleet is supported by ERU fuel.

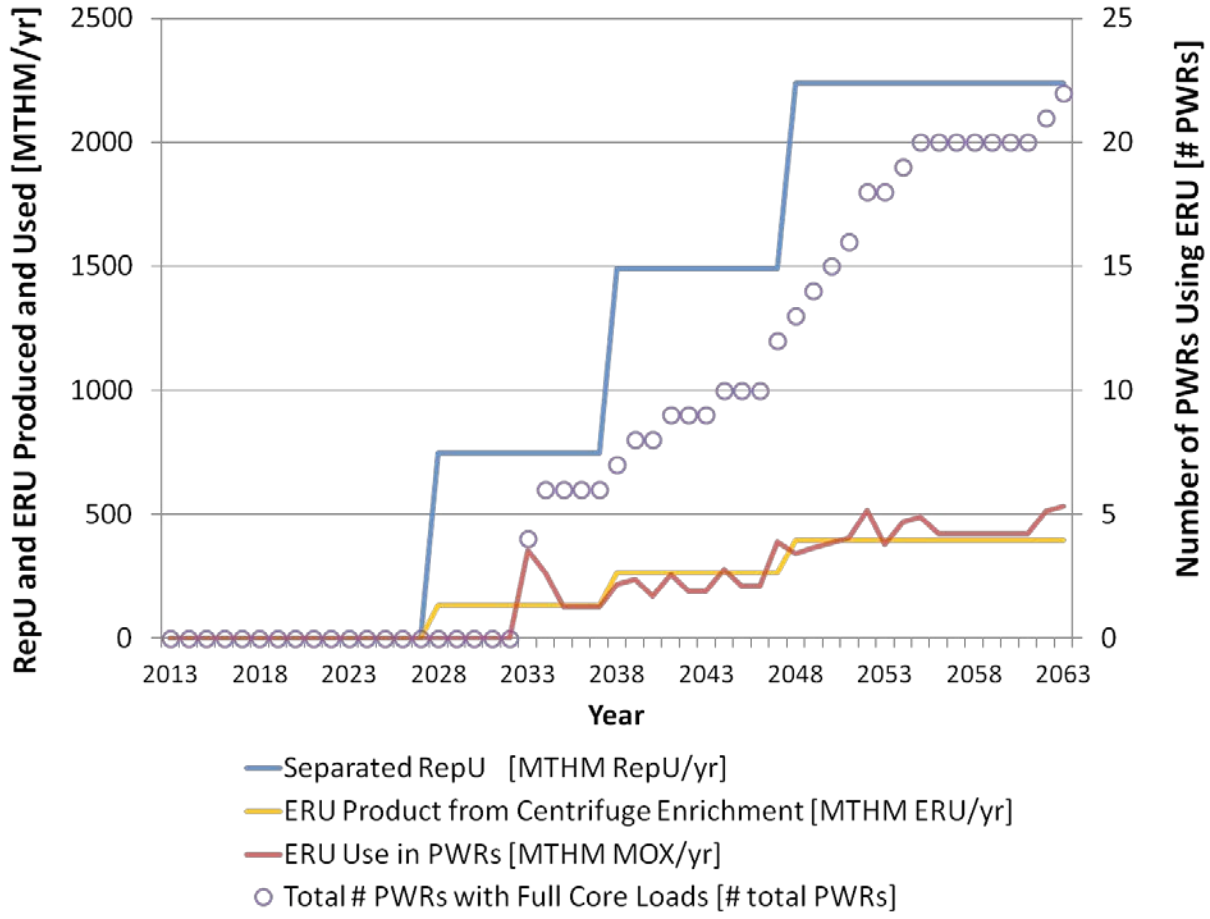


Figure 3.4 | RepU and ERU Production, ERU Use in PWRs

3.3.1.3. Annual LWR Fuel Loadings

The breakdown of the fuel types loaded into the reactor fleet each year is shown in Figure 3.5, represented by shaded areas and corresponding to the left Y-axis. The annual fuel requirements is shown as the black solid line as the baseline if the reactor fleet were fueled regularly with ENU. The purple solid line is the amount of fuel loaded into the reactor fleet due and deviates from the baseline due to the assumption of when converting PWRs to ERU and MOX use, there is a full core replacement.

BLEU use stays consistent at 10%, as discussed in Chapter 2. Re-ENU contributes 19% when Paducah is re-enriching higher-assay DUF6 and lowers to 5% when the New Mexico centrifuge plant is assumed to take responsibility of the remaining higher-assay DUF_6 inventory (consistent with modeling in Chapter 2).

In 2028, MOX use begins, shown as the green band in Figure 3.5. The percent of MOX as loaded fuel begins at 3% and peaks in year 2058 at 11%. In year 2033, ERU use begins, shown in the orange band in Figure 3.5. There is an initial spike due to the assumption that the inventory of ERU production would be initially stored for five years during fuel qualification. The five years of stored inventory allows a greater number of PWRs to be converted initially with full core replacements and then similar spikes in the orange band appear when new PWRs are converted to ERU.

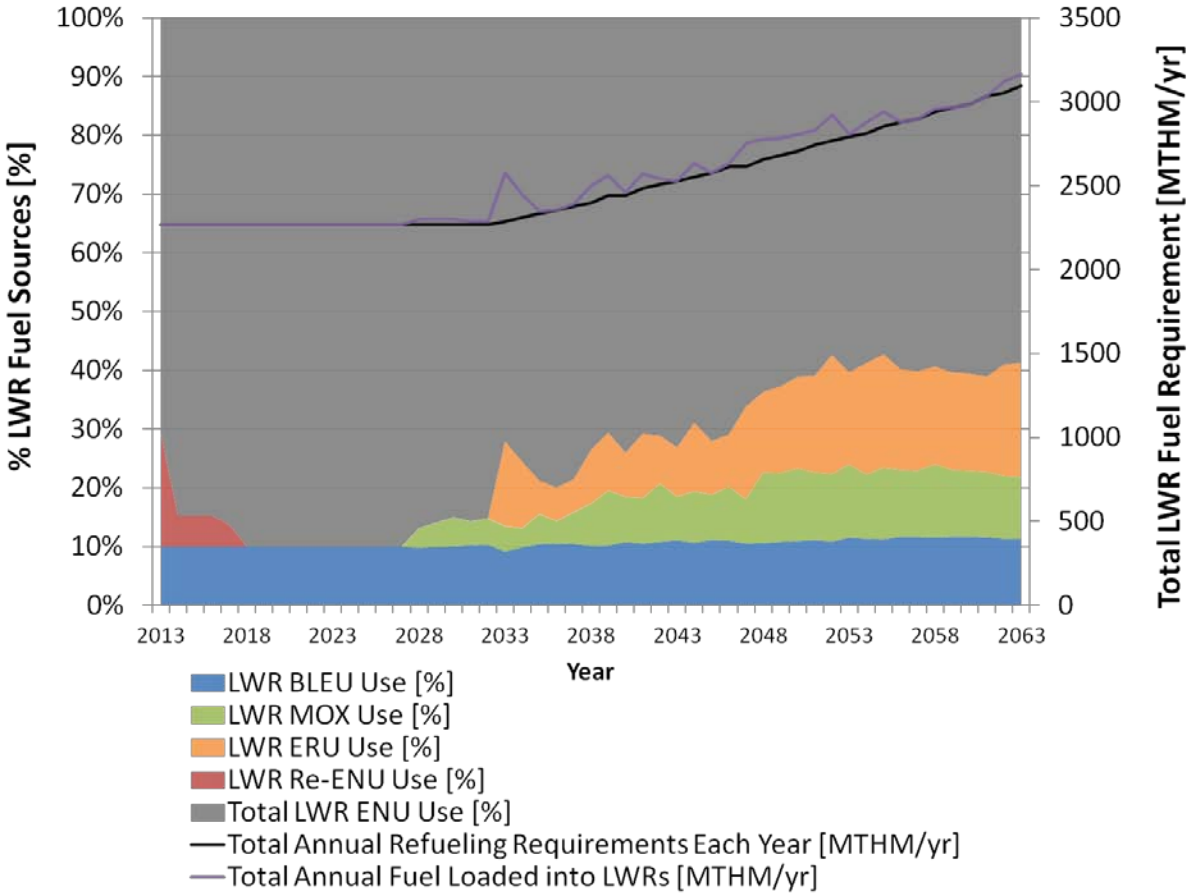


Figure 3.5 | MOC Percent of LWR Loaded Fuel Type and LWR Fuel Loaded Each Year

3.3.1.4. Pu Reduction

Approximately 7 MTHM Pu/yr is eliminated through burning of MOX (Figure 3.18, turquoise line) by the end of the simulation, 2063, due a 28% reduction of Pu in initial MOX compared to irradiated MOX [*EPRI*, 2006a]. The cumulative amount of Pu reduced from the Pu inventory is about 95 MT Pu that represents about 0.5% reduction compared to that of the inventory of UNF if no MOX irradiation had been performed. The remaining plutonium in after irradiation of MOX is shown (Figure 3.18, green line) reaching about 19 MT Pu/yr.

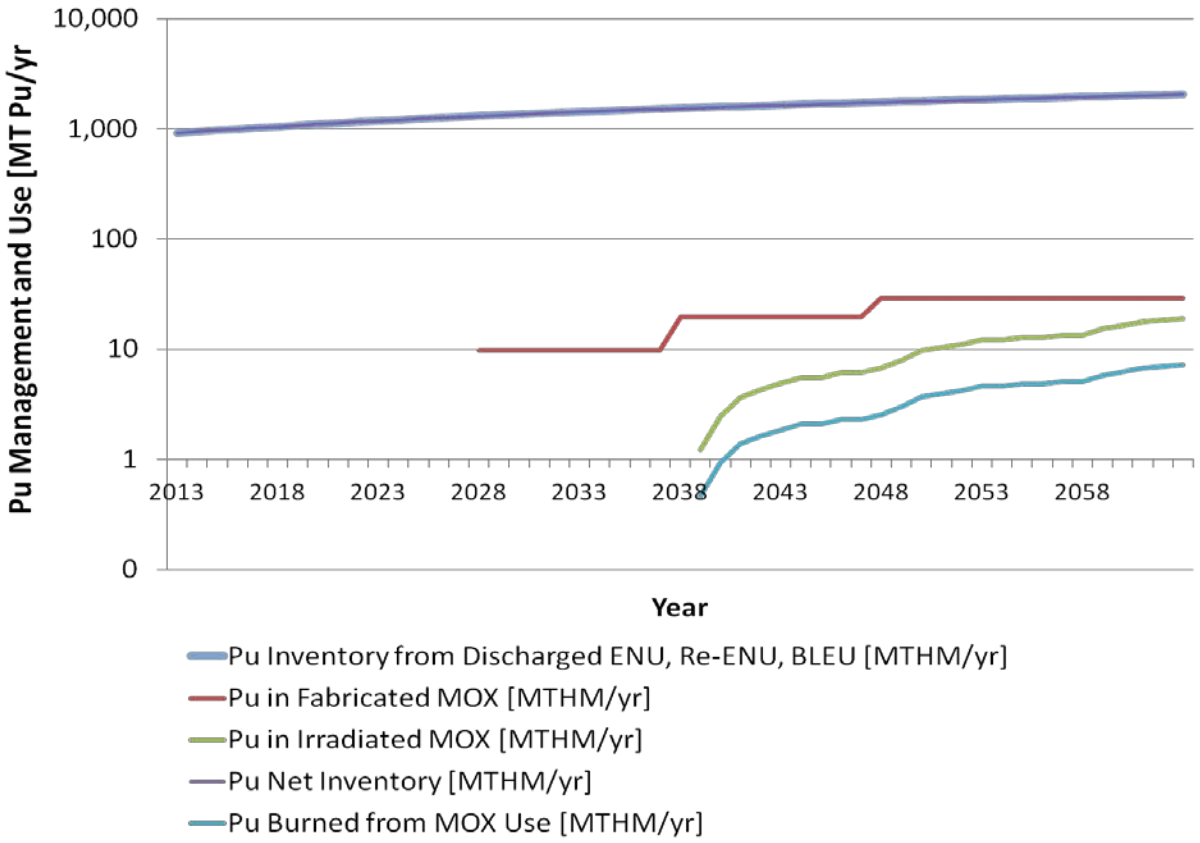


Figure 3.6 | Pu Inventories and Reductions from MOX Irradiation

3.3.2. MOC Waste Management Material Flow Analysis

Waste management material flows and radioactive waste volume impact results are presented here within two categories: (1) waste that have the potential to be disposed through shallow land burial disposal methods, and (2) waste that have the potential to be disposed within a geological repository. Shallow land burial eligible wastes are the following:

- Mine overburden
- Byproduct waste (ISL waste and mill tailings)
- LLW and MLLW
- Depleted uranium-oxides

Geological repository eligible wastes are the following (and are defined in Appendix B, Section B.18):

- ENU, BLEU, Re-ENU, ERU UNF (mass of UNF and number of dry storage casks) [Section 3.3.2.3]
- MOX UNF (mass and number of dry storage casks) [Section 3.3.2.3]
- HLW (mass, mass of vitrified HLW, number of Vitrified containers) [Section 3.3.2.4]
- TRU (volume of TRU) [Section 3.3.2.4]
- GTCC (volume of GTCC) [Section 3.3.2.4]

The results from scaling the radioactive waste volumes are presented first because the LLW, MLLW, and UNF values from operations were used to calculate the worker collective doses for back-end operations in the following Section (Section 3.3.3).

3.3.2.1. MOC Shallow-Land Burial-Eligible Radioactive Waste Generation

The total volume of radioactive wastes eligible for shallow-land burial are shown in Figure 3.7 for each year of the MOC from 2013 to 2063. The volumes range from 43 million m^3 to 54.7 million m^3 of wastes considered. The lower end of that range is observed when Re-ENU and BLEU replace ~30% and then ~15% of the required ENU fuel, then immediately after these two alternative sources of fuel are discontinued in the year 2018, the amount of shallow-land burial eligible wastes increase to the upper bound of 54.7 million m^3 each year until 2028 when the introduction of MOX fuel enters the MOC system and decreases until 2033. A number of smaller

spikes are observed between 2033 and 2054 when LWRs are added to the MOC faster than MOX and ERU can support and thus, must use ENU as the source of fuel; the same is true for towards the end past year 2054 when the reprocessing capacity is fully deployed and the production and use of MOX and ERU remain constant while ENU use continues to grow to meet the energy demand scenario.

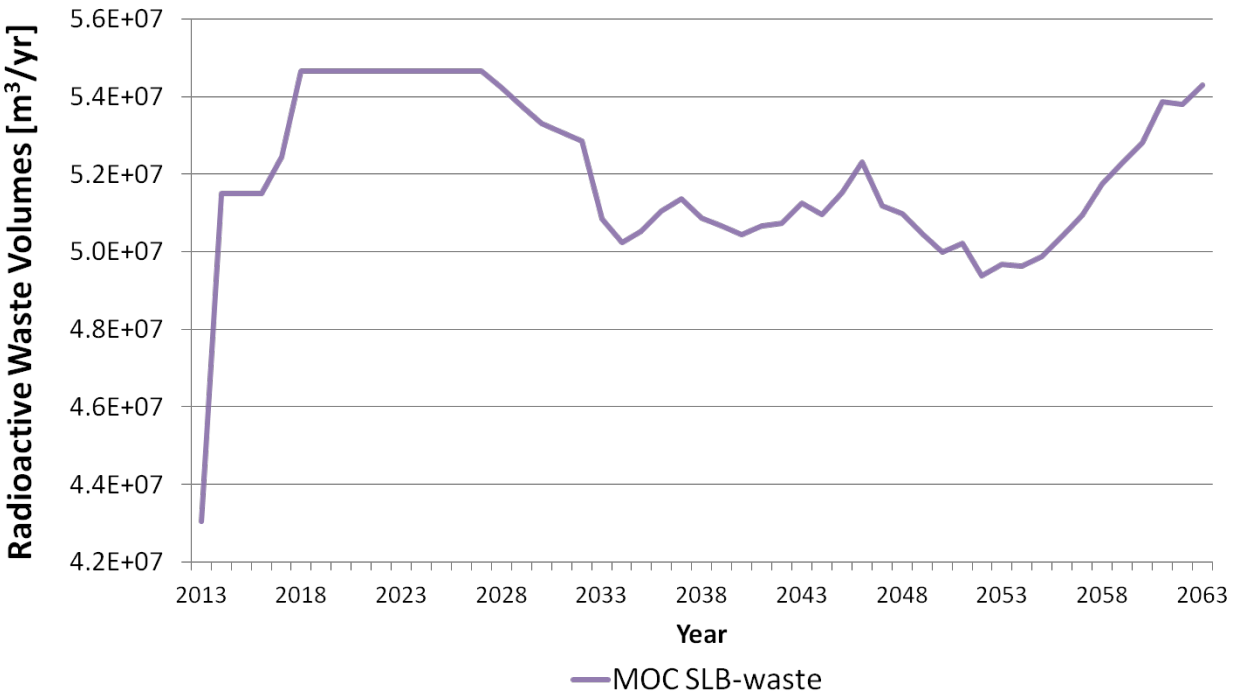


Figure 3.7 | MOC Radioactive Waste Volumes

LLW is produced at most MOC operations; it was assumed that negligible LLW is generated during disposal of LLW and during dry interim storage activities. Wastes from mining, milling and ISL are classified distinctly from LLW and are included separately in this analysis. The total volume of LLW for each year of the MOC is shown in Figure 3.8 and the range from 158,000 m^3/yr to almost 200,000 m^3/yr. The lower bound occurs at the start of the simulation when ENU

fuel and energy demand is at the lowest point during the modeling time frame. There are three peaks of total annual LLW produced during the years of 2028, 2038, and 2048 and correspond to each expansion of recycling operations. The peaks of LLW production last for a duration of about 5-years due to two factors: (1) PWRs are not converted to either MOX or ERU until sufficient inventory is available and (2) the assumption that PWRs cores undergo full replacement (100% for ERU; 40% for MOX) when PWRs are converted to MOX and ERU fuel. Each time the recycling capacity expands and the initial PWRs are converted, it takes time for the MOX and ERU inventories to accumulate to the point of being able to convert additional PWRs while simultaneously supporting the annual refueling needs of previously converted PWRs.

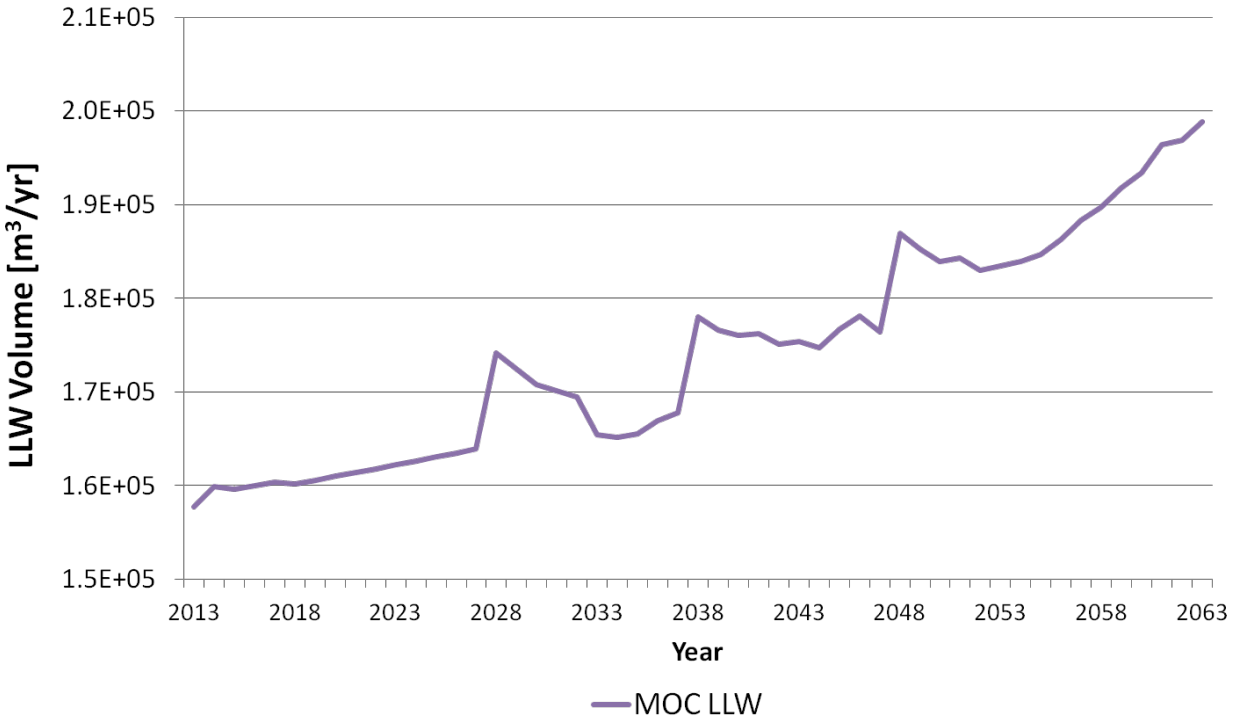


Figure 3.8 | MOC LLW Volumes

The nominal values of radioactive waste volumes from the front-end of the fuel cycle (including downblending and re-enrichment), middle of the fuel cycle (considered as reactors and recycling), and the back-end of the fuel cycle (deconversion) are plotted as a percent of the total waste volumes each year in Figure 3.9. The Y-axis has been truncated in Figure 3.9 to emphasize that 99.6% of the MOC wastes are mining and milling (~80% + ~20%, respectively) waste volumes, indicated by the dotted black line superimposed on the plot. The vast majority of the waste being directly associated with natural uranium production was from mining and milling waste that is dealt with and disposed on site and not at licensed disposal facilities [*EPA*, 1983a].

The remaining volume of radioactive waste that is described above the black dashed line shown in Figure 3.9, that is 0.4% of the waste is re-plotted into Figure 3.10 to more easily discern between various operations contributions to LLW and depleted uranium-oxides and what must be disposed at an off-site licensed disposal facility. The front-end dominates for LLW generation ranging at around 60% at near the beginning and lowering to 45% towards the end. Also, towards the end, recycling and reactor operations account for nearly 40% of the LLW produced and the remaining percent is related to deconversion operations. From both Figure 3.9 and Figure 3.10, it is apparent that the contribution of front-end impacts is lessening as less ENU fuel is required.

ISL byproduct wastes are not plotted in either 4.8 or 4.9 because annual waste volumes (~1,500 m^3/yr) are ~0.003% of the total radioactive waste volumes each year and 1% compared to total LLW values. MLLW produced each year is even smaller than ISL waste (~35 m^3/yr) and is 0.00002% of the total radioactive waste volumes and 0.01% of the LLW values.

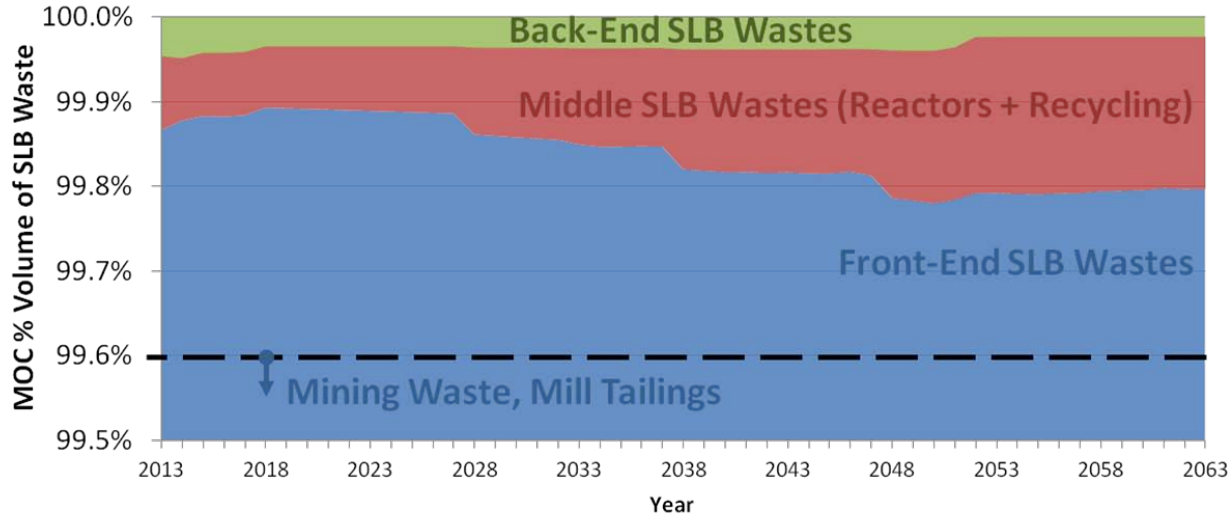


Figure 3.9 | Radioactive Waste Volumes Generated from MOC Operations

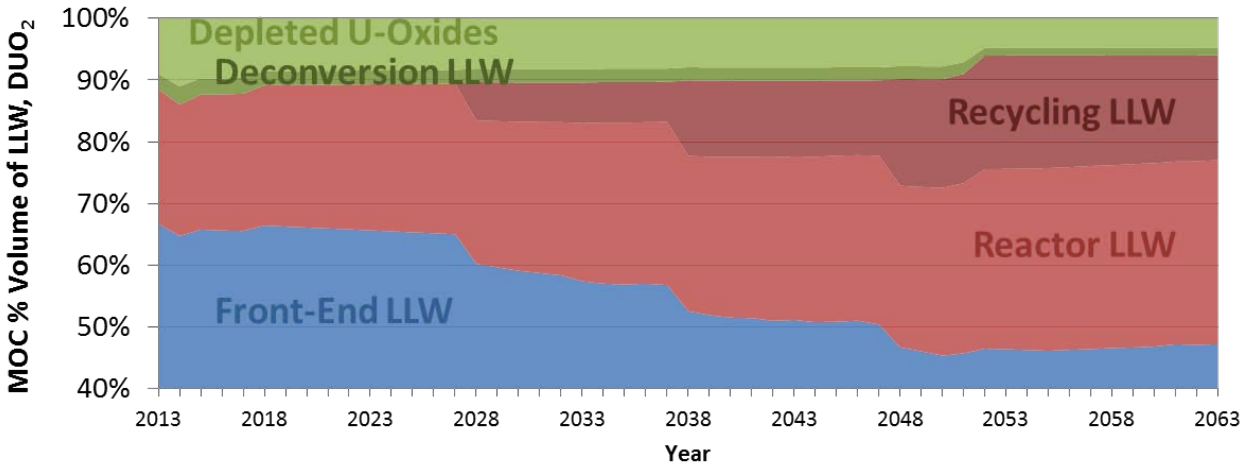


Figure 3.10 | Percent of Radioactive Waste Volumes from Grouped MOC Operations

3.3.2.2. MOC DUF_6 Management and Deconversion

DUF_6 waste management is more complex because of the many uses and pathways that DUF_6 is routed. Commercial and DOE-owned enrichment tails for the base case and MOC are the same for the modeling time frame of 2013 to 2027 (Figure 2.3) and the use of DU as part of BLEU

within the MOC scenario is the same as the OTC for the entire modeling time (orange-dashed line in Figure 2.3 and Figure 3.11). When the recycling operations begin in year 2028, the differences between the MOC and OTC DUF_6 waste management is due to the ERU tails production and MOX use of DU (red-dashed line, Figure 3.12).

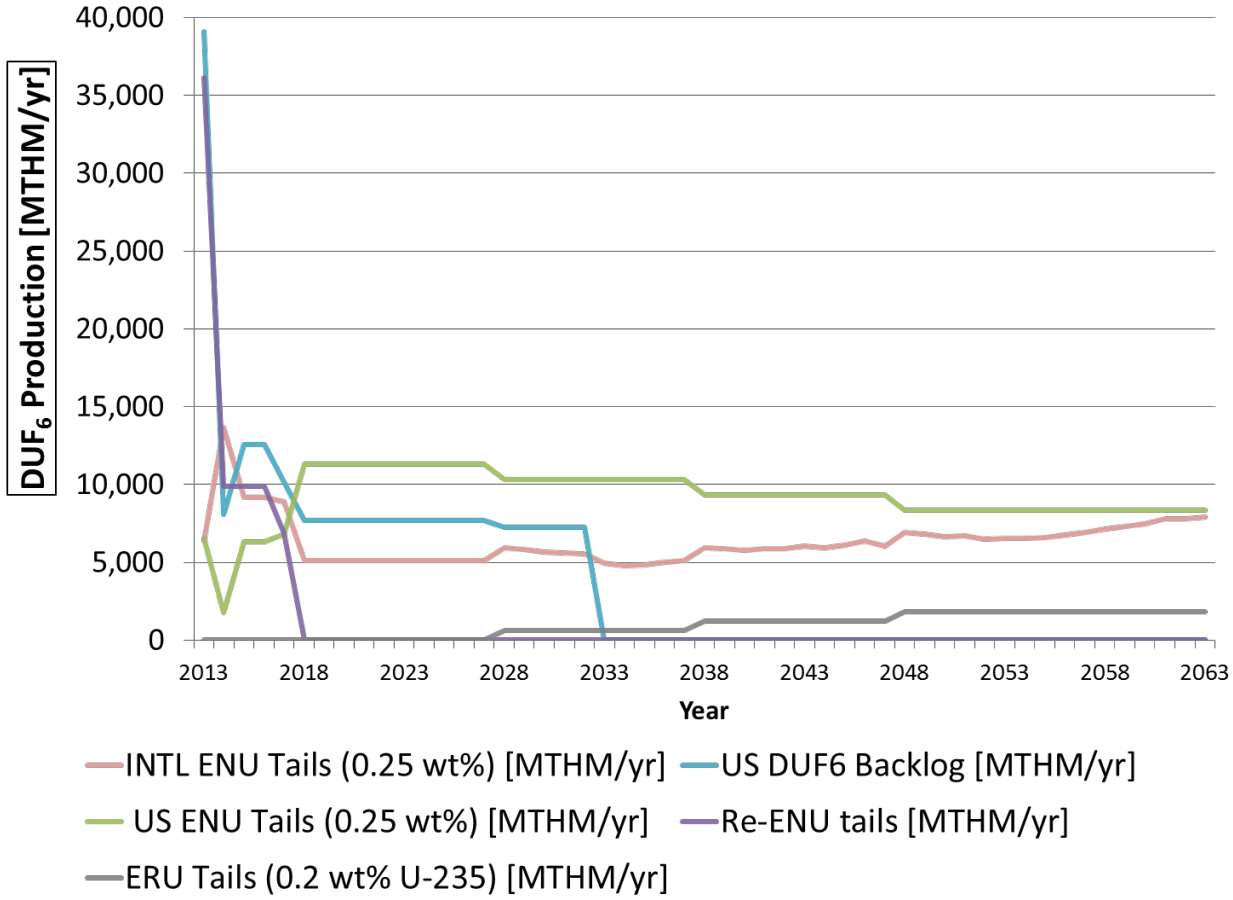


Figure 3.11 | MOC DUF_6 Annual Production

U.S. ENU tails production (green line, Figure 3.11) within the OTC scenario continues to be constant at around 12,000 MTHM of DU from 2028 to 2063, but with the MOC, the U.S. ENU tails production is shown to have a decreasing trend that is shown as a stair-stepped function two reasons: (1) lessened ENU required to be processed and enriched, (2) increased use of DUF_6 in

MOX because MOX is composed of 10% Pu and 90% DU (red-dashed line, Figure 3.12). The use of DU in MOX increases from 87, 175, to 262 MT DU.

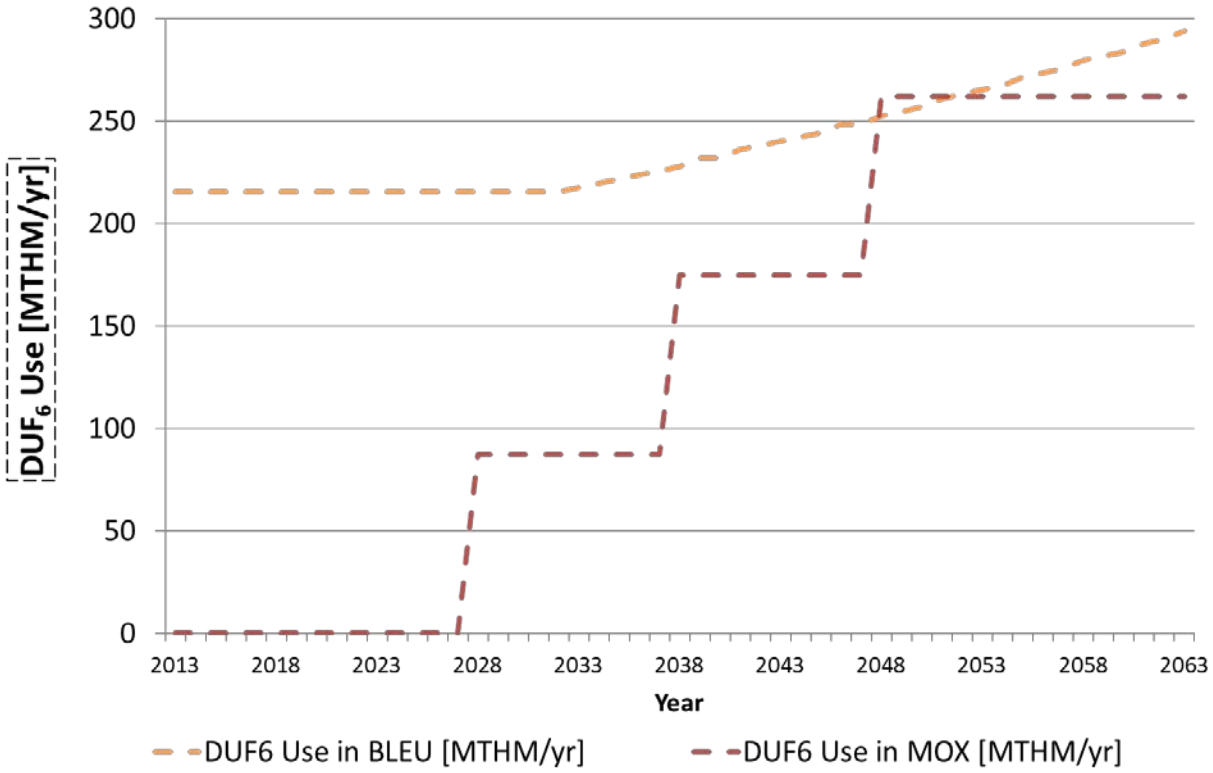


Figure 3.12 | MOC DUF_6 Annual Use

ENU tails produced by enrichment services outside of the U.S. (pink line, Figure 3.11) is caused by the increasing need for ENU but limited U.S. enrichment annual capacity. The use of international enrichment services continues to grow, especially at the end of the simulation (around 2054 to 2063) when reactors are coming online must use ENU because no additional inventories of MOX and ERU are available without building a fourth reprocessing plant.

The amount of ERU tails also follows the shape of the increases in reprocessing capacity in decadal intervals (gray line, Figure 3.11). A longer duration is observed for the MOC compared

to that of the OTC for backlogged DUF_6 awaiting deconversion. This is due to the increases in ERU tails at the initial five years of recycling operations and before excess capacity of Paducah and Portsmouth deconversion facilities become available in 2033 to support the one commercial IIFP deconversion facility. After 2033, the backlogged inventory of DUF_6 is eliminated.

The depleted uranium-oxides produced from 2013 to 2028 of the MOC scenario (Figure 3.13) are the same as the OTC modeling (Figure 2.4, Section 2.3.1.2). There are two main differences in the two scenarios. The first is that the a longer amount of time is required for all three deconversion plants (IIFP, Paducah and Portsmouth) for the MOC must operate (Figure 3.13, blue line) compared to the OTC is due to the additional ERU tails that would have originally remaining in UNF, but now are being rerouted from geological disposal to shallow-land burial disposal. Both depleted uranium-oxides and UNF have unresolved finalized disposition pathways; however, they are modeled herein as SLB and geological disposal, respectively.

The second difference is the amount of depleted uranium-oxides produced internationally reflects the same trend in the increases of enrichment and ENU fuel needs because of limited U.S. enrichment capacity beyond the two centrifuge plants (LES facility in New Mexico and USEC facility in Ohio) instead of following a gradually increasing need for fuel that was set in the OTC (as shown in Figure 2.4, Section 2.3.1.2).

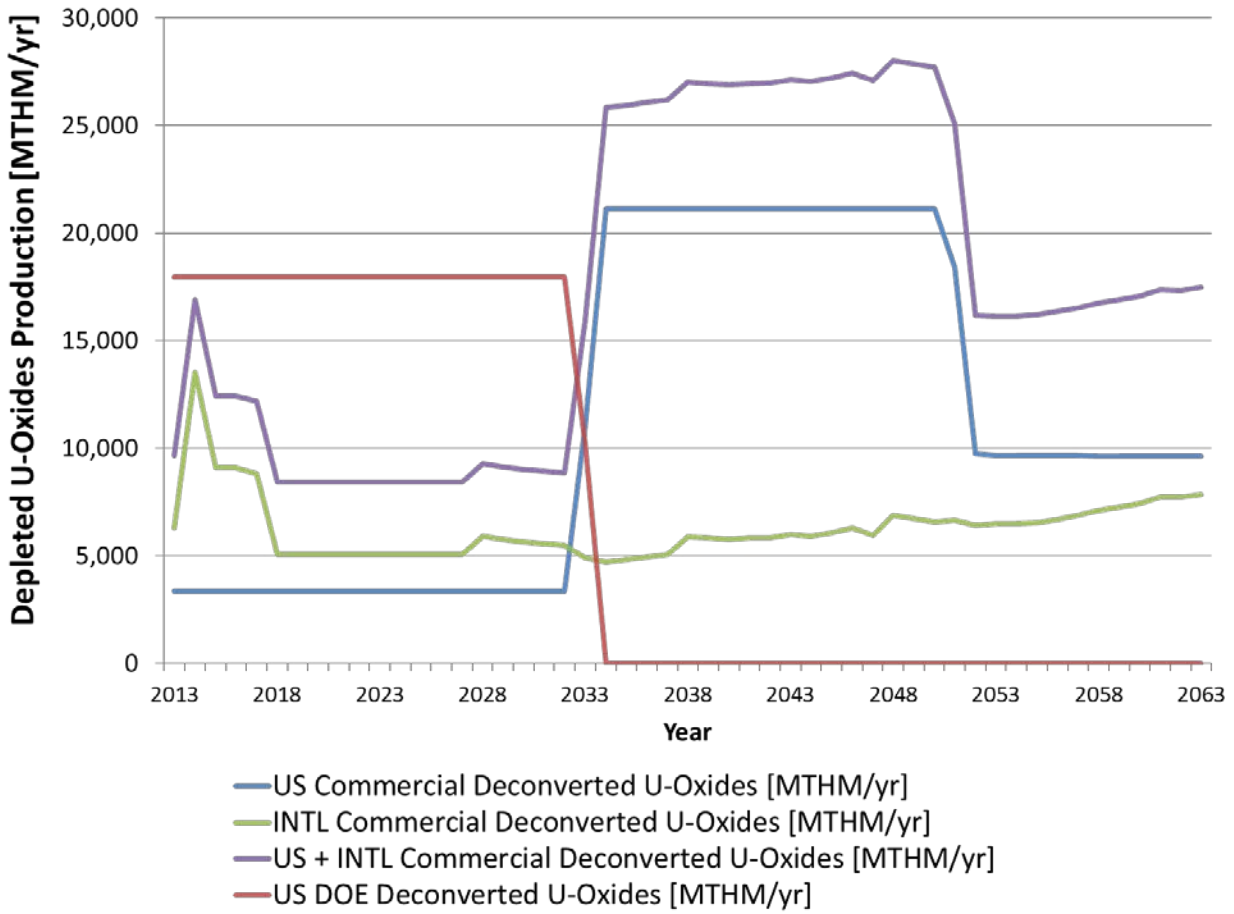


Figure 3.13 | MOC Depleted Uranium-Oxides Annual Production

3.3.2.3. MOC Dry Interim Storage

UNF discharged from UOX and MOX reactors is shown in Figure 3.14 – staying constant at first according to constant fuel requirements. The spikes observed in the number of DSCs holding UOX fuel (ENU, ERU, Re-ENU, BLEU, blue line in Figure 3.14) reflects the increases of fuel discharged from the reactor for full core replacements of ERU and 40% of the core for MOX when PWRs are converted from 2038 to 2058. From 2059, to 2063, the increases of UOX DSCs are from the increased ENU requirements.

The number of ENU DSCs taken from ISFSIs for reprocessing and further recycling is shown (green line, Figure 3.14) and ranges from 62 rising to 184 per year. The net number of DSCs being added to ISFSIs each year ranges from 174 declining to 10 DSCs (Figure 3.14, purple line) and is equal to the number of MOX DSCs plus number of UOX DSCs minus the number of UOX DSCs that have been taken from the ISFSI for reprocessing. This trend shows that on the opening year of the third reprocessing plant in 2048 that the number of DSCs required to support recycling operations almost exceeds the annual production of UNF DSCs—thus it could be postulated that the backlog of DSCs awaiting disposal (the total UNF inventory) could have changed inflection and the inventory would decrease as a result, if another reprocessing plant were brought online in 2058 (however, this has yet to be modeled).

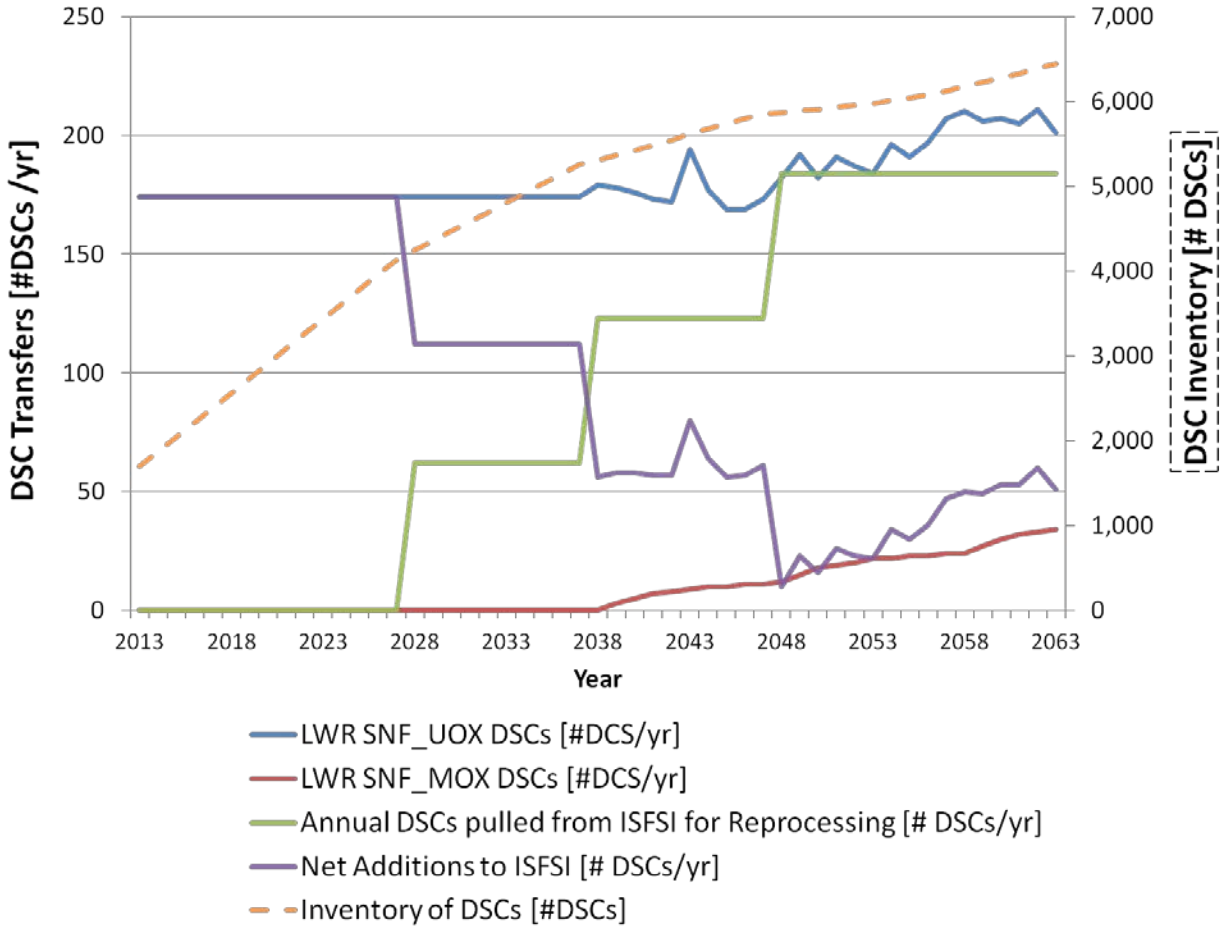


Figure 3.14 | MOC Dry Interim Storage DSC Additions and Inventory

3.3.2.4. MOC Reprocessing and Recycling Waste

Reprocessing HLW is generated at the rates that matched the stepped increase in reprocessing capacity yielding 40, 80, 120 MTHLW/yr (Figure 3.15, blue line), corresponding to vitrified HLW masses of 227, 453, 680 MT /yr (Figure 3.15, green line). HLW was loaded to 22 wt% of the total vitrified HLW and the corresponding number of containers (UC-Vs) to store vitrified glass (560, 1120, 1680 UC-Vs/yr, purple circles in Figure 3.15).

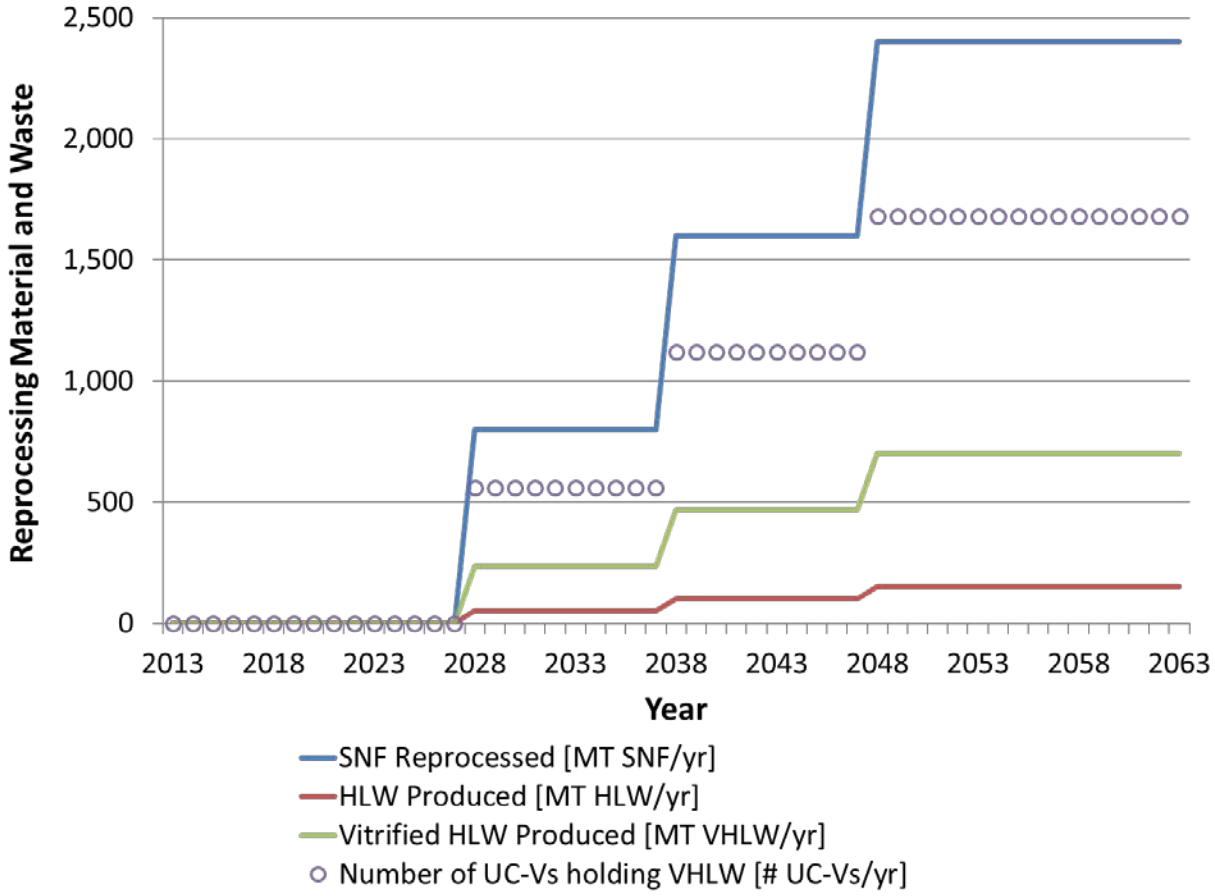


Figure 3.15 | MOC Reprocessing HLW Production

Transuranic contaminated material (TRU) and GTCC waste were calculated and results matched the increased-step function of the reprocessing and recycling facilities' capacity. TRU reprocessing waste for the three time frames was (145, 289, and 434 m^3 TRU/yr) and MOX fuel fabrication TRU waste was (431, 863, and 1294 m^3 TRU/yr); GTCC waste resultant from reprocessing was (453, 906, and 1359 m^3 GTCC/yr) for the same modeling time frames.

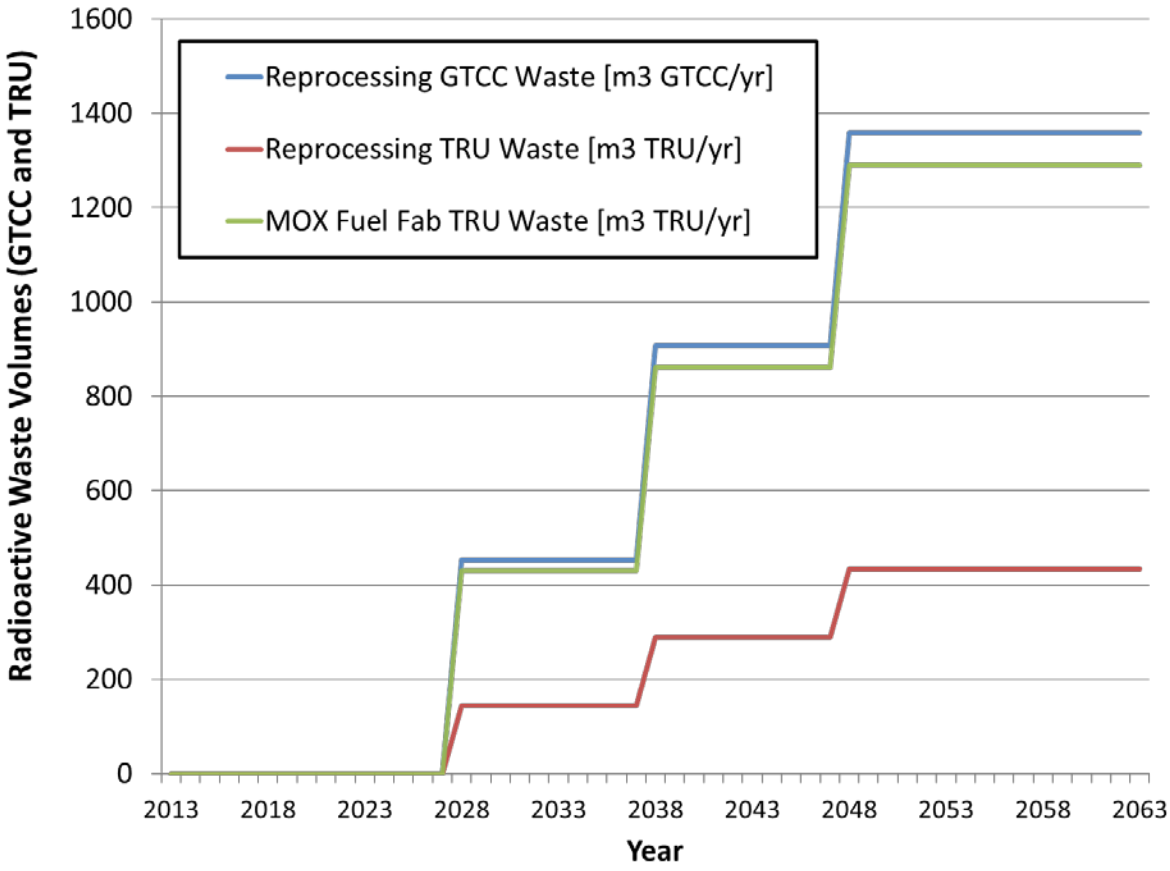


Figure 3.16 | MOC Recycling TRU and GTCC Production

3.3.3. MOC Worker Collective Doses

The estimated total worker collective doses each year for the MOC are shown in Figure 3.17 from 2013 to 2063. Worker collective doses range from 110,000 person-mSv/yr to 177,000 person-mSv/yr (110 – 177 person-Sv/yr) and increase for the most part in a steadily increasing rate, similar to that of the assumed growth of the rate of nuclear energy demand.

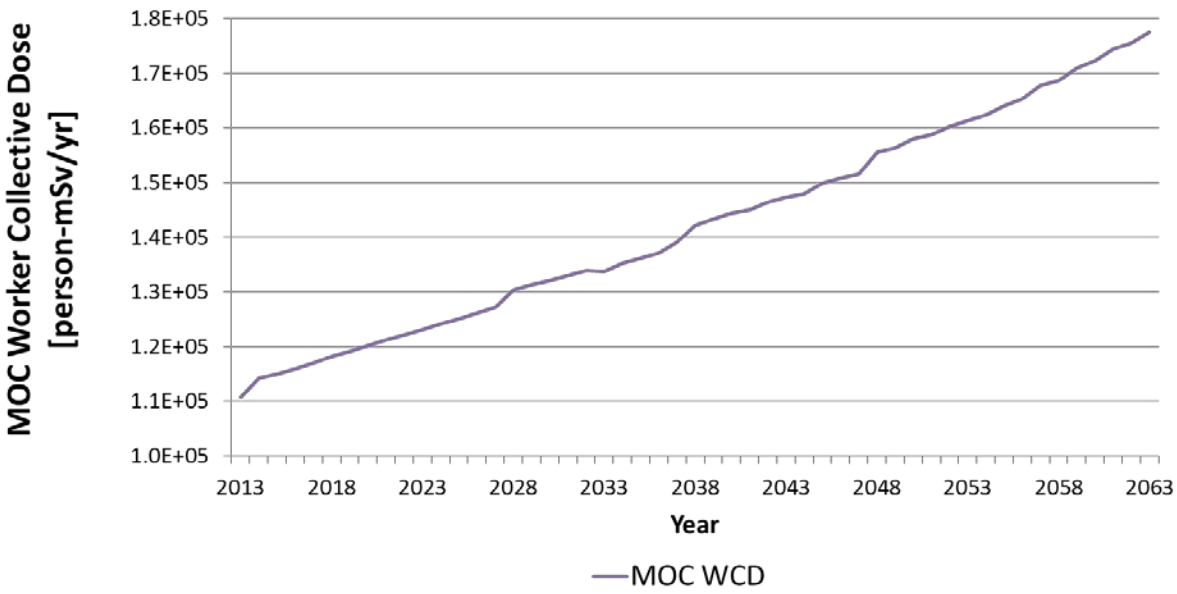


Figure 3.17 | MOC Worker Collective Doses

The worker collective doses from each OTC operation are graphically displayed in Figure 3.18. As measured by worker collective dose (person-mSv/yr), the highest overall dose was for the operating reactors (PWRs and BWRs). This is similar to the OTC results, even when operations from recycling of Pu and RepU are considered.

Many operations of the MOC could be considered to contribute the third highest worker collective dose each year for the select years shown in Figure 3.18 and are on the order of magnitude of in the 1000s person-mSv/yr. This includes international UG mining, OP mining, ISL operations, and milling; U.S. operations were dry conversion, MOX fuel fabrication, and dry interim storage activities.

The basis for the worker collective dose normalized metric for maintenance, surveillance, and security of ISFSIs is the number of ISFSI sites. The data sets to allow calculating this metric with

the basis of metric tons of UNF stored on site were not readily available at the time of the calculation, but a preliminary analysis has been performed and presented in Appendix A (Section A.12). This preliminary analysis was not incorporated in to the larger study because the parameter data sets to understand potential reductions in worker collective doses from decreased UNF stored at ISFSIs is required. There is doubt associated with this preliminary estimation of the normalized metric using a UNF-mass normalization basis to reflect the actual reduction in worker collective dose from withdrawals required for reprocessing. This is because this preliminary analysis results indicate that the reduction of worker collective dose from withdrawals of UNF is higher than the current estimates of all of dry interim storage activities and it is not possible to result with a negative worker collective dose.

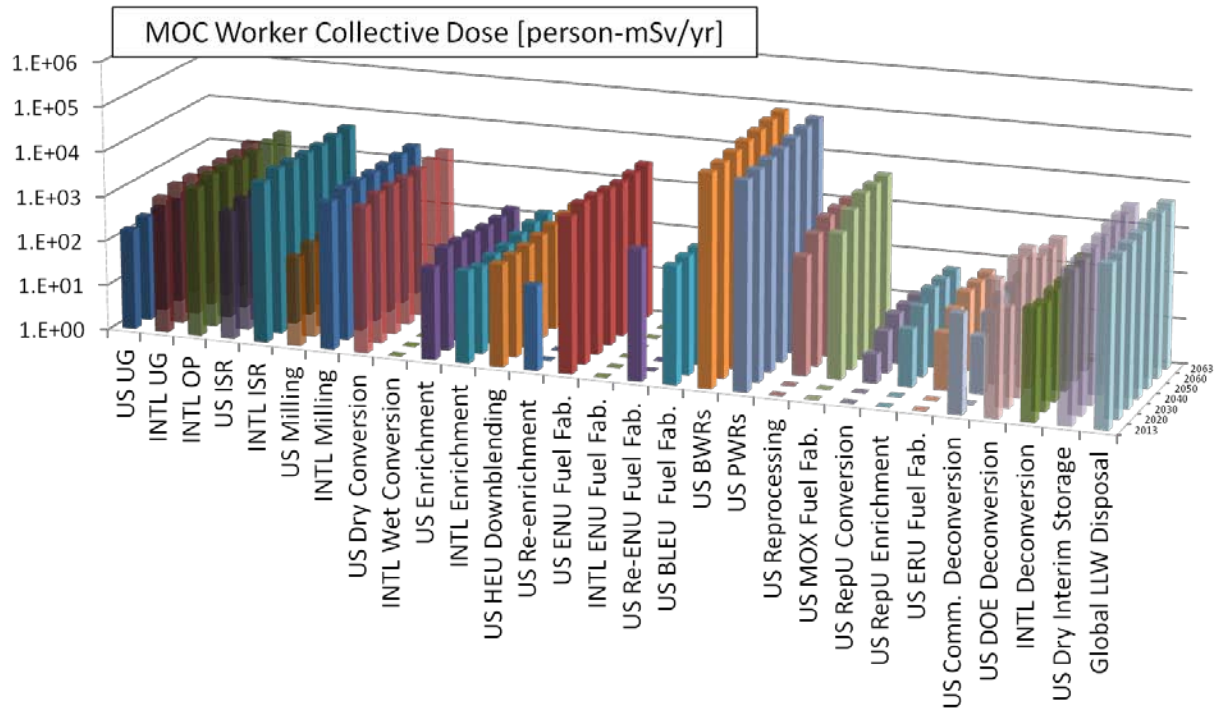


Figure 3.18 | Worker Collective Dose from MOC Operations

Aggregating the fine granularity of detail shown in Figure 3.18 to a number of grouped operations eases interpretation of the contributors to the OTC worker collective dose. In Figure 3.19 the U.S. and international activities are combined by operation and the results are displayed as a relative percent of the total OTC worker collective dose for a select number of years.

Similar to the OTC case (as shown in Figure 2.7), the reactor operations contribute ~80% of the total MOC worker collective doses. When recycling operations are included, the renormalized percent of contributions from the front-end of the MOC (that includes re-enrichment and downblending) is 10.5% at the year 2063. Worker collective doses from back-end operations of LLW disposal, management of UNF ISFSIs, and deconversion operations constitute around 4.5% of the total MOC worker collective dose.

Even with dynamic behavior of fuel cycle activities, the contribution of MOC worker collective doses of reactor operation stays almost constant. The contribution from the front-end and back-end operations shift in a manner to such that recycling becomes a small, but increasing part of the contributors to total MOC worker collective dose. This is the balance between worker collective doses decreasing in response to lessened ENU related front-end operations and waste production, small increases from deconversion of ERU tails and handling of UNF DSCs to use as feedstock for reprocessing.

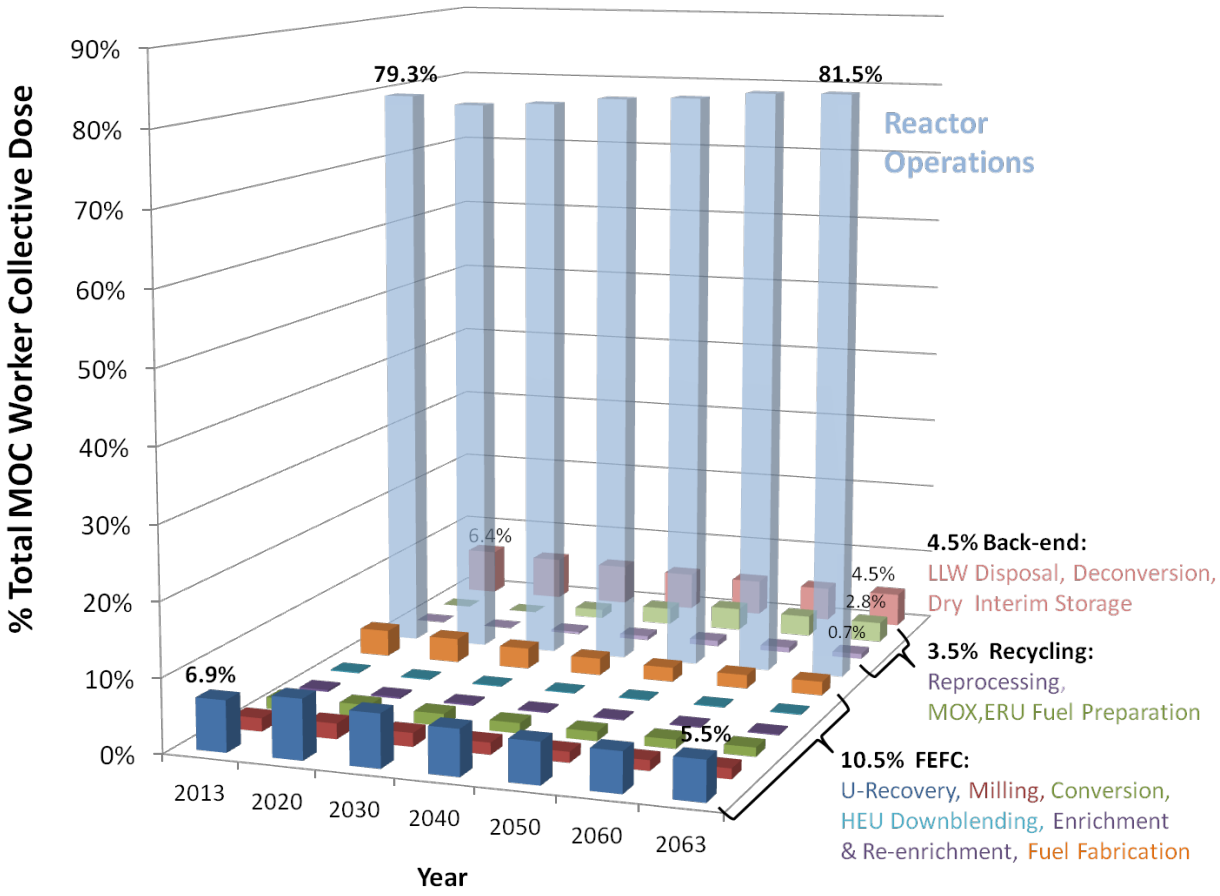


Figure 3.19 | Percent of Worker Collective Dose from Grouped MOC Operations

3.3.4. Comparing MOC to OTC Worker Collective Doses and Radioactive Waste Volumes

The expected differences in radioactive waste volumes and worker collective doses between the OTC and MOC involve competing factors. The MOC requires 10-15% less natural uranium than the OTC, so front-end operations contribute less to the MOC total worker collective dose. However, the MOC involves the addition of reprocessing, MOX and ERU fuel fabrication, RepU conversion and enrichment, and more DSC units are required due to the characteristics of spent MOX fuel. It is of interest to calculate the net results of "savings" from reduced activity in the

front-end operations that may be matched or exceeded by the dose "costs" associated with recycling RepU and Pu as part of the MOC.

Methodology for comparing impact results is simply the ratio, or quotient, of impacts from the MOC to the OTC; the MOC is the numerator and the OTC is the baseline and is therefore, the denominator in this ratio (as discussed in Section 3.2.5, Equation 4.1). The ratio is applied for each year of the OTC and MOC results and referred to as "annual ratios" in the following sections. The inherent nature of a dynamic study produces results that vary throughout the simulation and therefore another method to gauge overall comparative performance is helpful. Integrating the impacts of each NFC and then calculating the quotient of the MOC integrated impacts to OTC integrated impacts is used for comparing overall NFC performance (as discussed in Section 3.2.5, Equation 4.2). This is referred to as the "integrated ratio" when evaluating integrated worker collective doses and radioactive waste volumes in the following sections.

Both annual and integrated ratios will be presented for first worker collective doses and then radioactive waste volumes. Annual ratios of UNF DSCs transferred to ISFSIs are examined as this will aid in verification since many studies do not evaluate using the same metrics used in this work. Insights into translating radioactive waste volume ratios back to nominal values will be explored to understand the magnitude of the avoidance, or increased, burden of transitioning to the MOC. Next, concluding remarks of overall performance of the MOC compared to that of the OTC are also provided. Finally, the performance of the MOC relative to the OTC is reviewed against previous studies discussed in the beginning of this chapter.

3.3.5. Comparing MOC to OTC Worker Collective Doses

Annual and integrated ratios describe the extent of the differences estimated for MOC and OTC worker collective doses in the following sections.

3.3.5.1. Comparing MOC to OTC Worker Collective Doses on an Annual Basis

There is not a noticeable difference in the worker collective dose incurred each year between the MOC and OTC, as shown in Figure 3.20 (top graph) with MOC represented as the purple line and OTC as the green line. The steadily increasing trend upwards is indicative of the 1% energy demand increase over the entire 50 year time frame. There are somewhat notable sections where MOC becomes slightly higher than the OTC worker impacts and that is due to increased recycling activities beginning in 2028 and continuing to 2048.

The differences in the total worker collective dose amounts are more easily identifiable from the lower diagram in Figure 3.20 showing the ratio of MOC over OTC impacts. There are peaks at the beginning years of opening the recycling facilities but this represents only a 1% to 1.7% increase overall at these specific points. After these peaks, the ratio tends to return to just slightly below 1.0 that is representative of less front-end processes required due to increased MOX and ERU use.

Similar to the reasons explained for MOC LLW production (Section 3.3.2.1, Figure 3.8), three peaks of total annual ratios are observed with starting years of 2028, 2038, and 2048 and

correspond to each expansion of recycling operations. The peaks of LLW production last for a duration of about 5-years to fully complete converting PWRs. Each time the recycling capacity expands and the initial PWRs are converted, it takes time for the MOX and ERU inventories to accumulate to the point of being able to convert additional PWRs while concurrently supporting the annual refueling needs of previously converted PWRs.

Simultaneous to the inventory balance of ERU and MOX, the increasing demand of nuclear energy continues to grow at the exponential rate of 1% per year and the rate of PWR conversions to MOX and ERU are less than that of the overall rate of adding reactors that are supported by ENU. The driving trend for the annual ratio to return to 1.0 (Figure 3.20, bottom graph) is that the higher rate of additions of ENU-supported PWRs exceeds the MOX- and ERU-supported PWRs and the MOC moves closer to looking like an OTC with a reactor fleet supported by ENU. This is also the same reason why the slopes of worker collective dose increase a faster rate for when recycling facilities are brought online and then return to the same slope of the OTC when the 5-year period to use the MOX and ERU in PWRs.

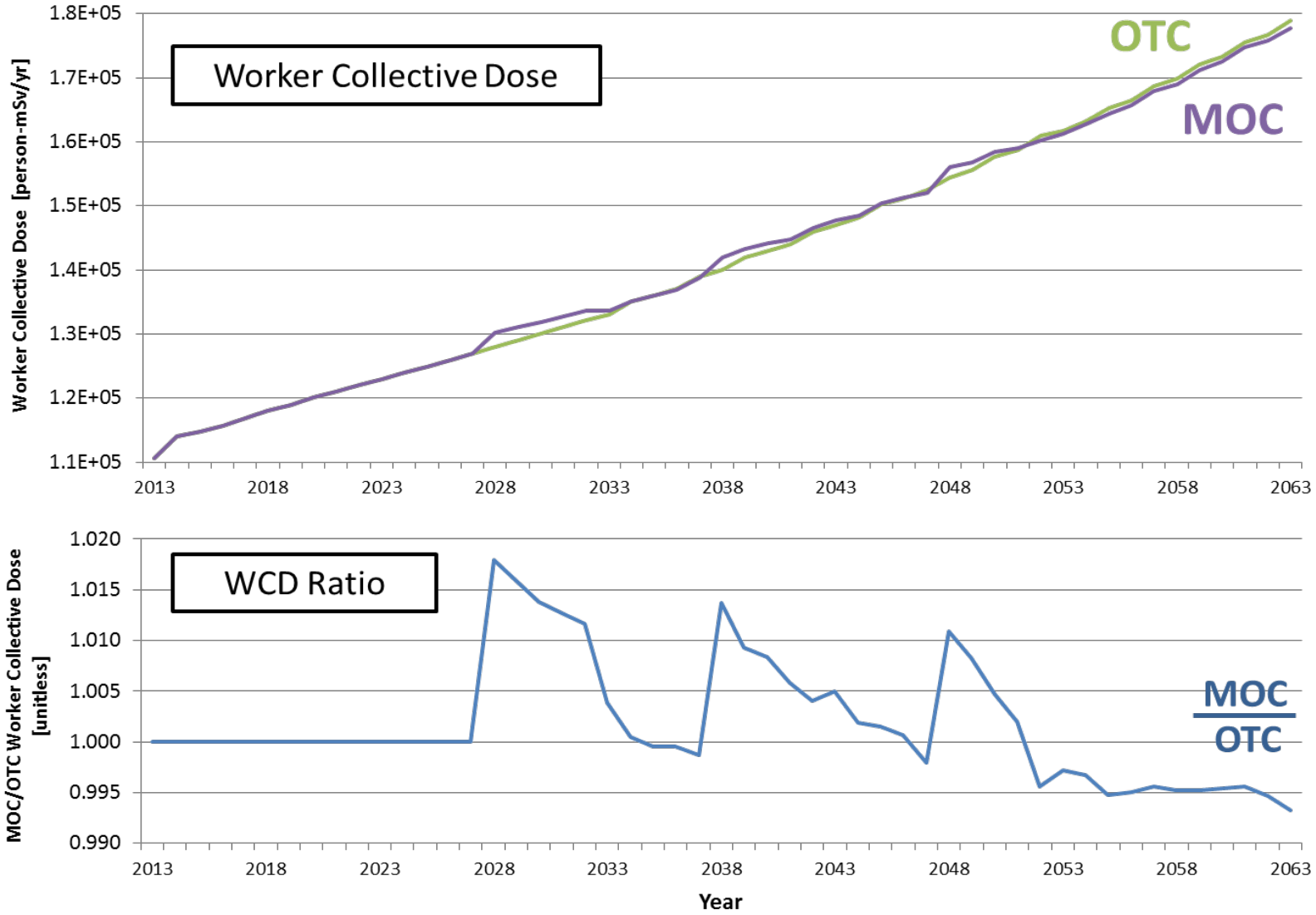


Figure 3.20 | Comparing Worker Collective Doses of the MOC and OTC

3.3.5.2. Comparing Integrated MOC to Integrated OTC Worker Collective Doses

Integrated worker collective doses over the 50-year simulation for the MOC and OTC are listed by operation and grouped operations (shown in Table 3.5) and then are used to calculate integrated ratios. The distance away from 1.0 the integrated ratio, when multiplied by 100, is the percent difference of the integrated worker collective doses of the MOC to the OTC. The integrated ratio for the entire NFC was 1.002, equal to 0.2% increase expected of the MOC to OTC.

The largest difference of integrated ratios observed was with fuel fabrication operations with a 40% difference that represented a decrease of expected worker collective doses. The use of ERU as a replacement of ENU combined with lower worker collective dose metrics for ERU than ENU fuel fabrication is cause for this reduction (ENU: 1.7 person-mSv/MT, MOX: 16.5 person-mSv/MT, and ERU: 0.14 person-mSv/MT). All three performance metrics were based on measured and recorded dose measurements and were taken from reported years of the early to mid-2000s.

A 14% increase in the integrated worker collective dose ratio for dry interim storage operations is a result of additional handling of UNF DSCs in order to send to reprocessing and the greater number of DSCs required for storing MOX fuel assemblies compared to UOX-based fuel assemblies (7.82 MT UNF of MOX, 11.25 MT UNF of UOX-based fuels).

Reactor operations, MOX fuel fabrication, RepU conversion, RepU enrichment, and ERU fuel fabrication are considered as the “middle of the fuel cycle” in this analysis and are aggregated in order to consider the worker collective doses estimated from recycling operations within the annual and integrated ratios. The middle of fuel cycle integrated ratio is around 3% higher for the MOC to that of the OTC. Due to the assumption that reactor worker collective doses on a per MWe basis were the same, independent of the fuel options used in this model, the 3% increase is attributed to the additional operations to recycle RepU and Pu. The reactor associated worker collective dose is ~80% of total estimated MOC worker impacts each year so the anticipated and observed increases of recycle are relatively low.

HEU downblending had an integrated ratio of 1.0 because it was assumed that downblending and BLEU use were the same in both MOC and OTC scenarios. Reactor operations integrated and annual ratios were also unchanged from the MOC to OTC scenarios because of the assumption described above.

Table 3.5 | Comparative Integrated Worker Collective Doses

Worker Collective Dose [person-mSv]	MOC	OTC	Difference (OTC-MOC)	Integrated Ratio (MOC/OTC)	Range of Annual (MOC/OTC) Ratios LB	UB
TOTAL FRONT-END	9.07E+05	1.06E+06	1.55E+05	0.85	0.70	1.00
U-Recovery	4.73E+05	5.47E+05	7.42E+04	0.86	0.72	1.00
Milling	1.22E+05	1.41E+05	1.92E+04	0.86	0.72	1.00
Conversion	1.06E+05	1.21E+05	1.53E+04	0.87	0.73	1.00
Enrichment & Re-enrichment	1.36E+04	1.57E+04	2.08E+03	0.87	0.73	1.00
HEU Downblending	1.05E+04	1.05E+04	0.00E+00	1.00	1.00	1.00
Fuel Fabrication	1.82E+05	2.26E+05	4.44E+04	0.80	0.61	1.00
TOTAL MIDDLE	5.96E+06	5.80E+06	-1.61E+05	1.03	1.00	1.05
BWRs, PWRs	5.80E+06	5.80E+06	0.00E+00	1.00	1.00	1.00
Reprocessing	3.25E+04	0	-3.25E+04	--	--	--
MOX Fuel Fab., RepU Conv., RepU Enrich., ERU Fuel Fab.	1.28E+05	0	-1.28E+05	--	--	--
TOTAL BACK-END	3.91E+05	3.86E+05	-5.56E+03	1.01	0.99	1.06
Deconversion	5.69E+04	6.04E+04	3.51E+03	0.94	0.79	1.03
Dry Interim Storage	1.37E+05	1.20E+05	-1.73E+04	1.14	1.00	1.32
LLW + MLLW	1.97E+05	2.06E+05	8.25E+03	0.96	0.89	1.06
TOTAL NFC	7.26E+06	7.25E+06	-1.10E+04	**1.002**	0.99	1.02

3.3.5.3. Worker Collective Doses: Annual and Integrated Ratios

The range of annual MOC/OTC ratios are listed in Table 3.5 and are found by the maximum ratio and minimum ratio throughout the model years; this upper bound (UB) and lower bound (LB) is graphically represented in Figure 3.21 as a gray bar that spans the range of annual ratios. The integrated MOC/OTC ratios are shown as the yellow circles that are superimposed on the annual ratio bands that were also listed in Table 3.5.

The visual range of the worker collective doses of the MOC to the OTC of each operation (and groups of operations) is shown and reflects the variability of net impacts during the transition to the MOC. Most bands of worker collective dose annual ratios stay on either side of the 1.0 line (for most operations, the 1.0 bound represents the annual ratios towards the beginning of the simulation when recycling operations had yet to be deployed). The wider the annual ratio band, or the difference in the upper and lower bound, the importance of using an integrated ratio is

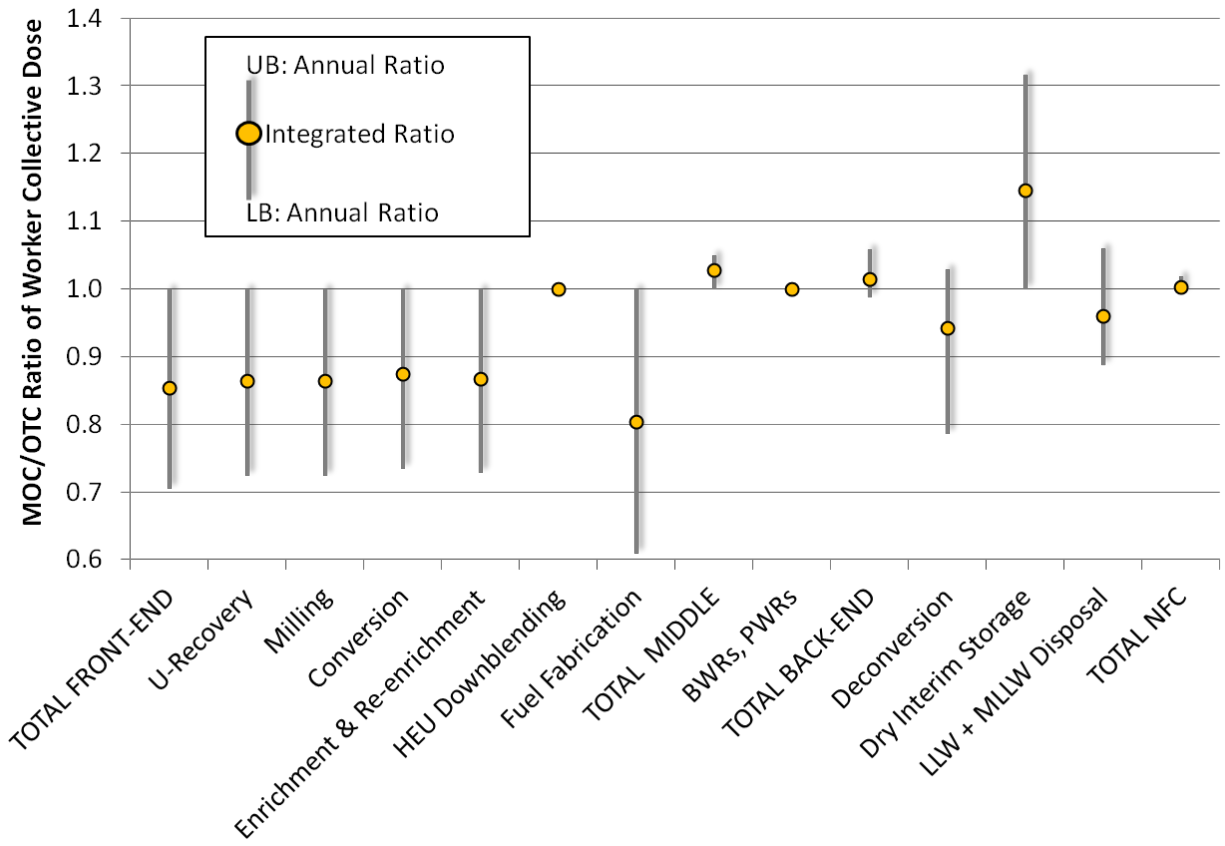
reiterated as the net benefit or disadvantage of MOC worker collective dose can fluctuate between individually modeled years.

Fuel fabrication annual ratios band had the widest range spanning from 0% decrease to 40% decrease (1.0 to 0.6) and dry interim storage annual ratios band was the second largest but from 1.0 to 1.3 that represents a 0% increase to 30% increase.

As expected, HEU downblending and reactor operations had no variability in the annual ratios of worker collective dose because it was assumed that the same amount of BLEU and electrical energy were to be produced and worker collective doses were independent of fuel types.

Additional support for the use of an integrated ratio for comparing impacts of dynamic impact studies is when the integrated ratio does not align in the middle of the band of annual ratios; these instances occur also when the worker collective dose annual ratio bands extend in both directions of above and below 1.0: (1) deconversion, (2) LLW, MLLW disposal, and (3) the grouped operation of total back-end operations.

As anticipated from earlier discussions, the integrated worker collective dose ratio for the entire MOC and OTC are nearly one (1.002). Although the widths of the individual operations' annual ratio bands may indicate wide variability, the annual ratio band is small but spans across both sides of 1.0 (0.99 to 1.02). The large proportion of reactor operations to worker collective dose can account for the low variability of the overall MOC to OTC annual ratios.



Notes: Fuel fabrication as part of the front-end grouped operation excludes MOX and ERU fuel fabrication. MOX and ERU fuel fabrication are included in the "TOTAL MIDDLE" category with reprocessing and reactor operations. LB = lower bound; UB = upper bound.

Figure 3.21 | Worker Collective Dose Integrated and Annual Ratios

3.3.6. Comparing MOC to OTC Radioactive Waste Volumes

Annual and integrated ratios describe the extent of the differences estimated for MOC and OTC radioactive waste volumes are discussed in the following sections. Consideration of the total radioactive volumes that are eligible for shallow-land disposal, LLW, and UNF are considered separately.

3.3.6.1. Comparing MOC to OTC Radioactive Waste Volumes on an Annual Basis

The difference in MOC and OTC radioactive waste volumes begins the same year that reprocessing and other recycling operations begin in 2028 (top graph, Figure 3.22) and the difference is described as waste avoidance and the cumulative waste avoided is shown in the middle graph. The radioactive waste volume avoidance grows each year as the increase of ERU and MOX fuel replace the use of ENU in reactors. The end of the simulation of years 2054 to 2063, the annual waste avoidance lessens as the use of ENU increases as reactors come online faster than the recycling activities to produce MOX and ERU can support. By the end of the simulation, the accumulated waste avoided is over 414 million m^3 of radioactive waste.

The annual ratios of radioactive waste volumes are all below1.0 (bottom graph, Figure 3.22) from 2028 to 2063. Annual ratios reach to nearly 0.72, representing a decrease of 28% from the MOC generated radioactive waste to the same time of the OTC scenario.

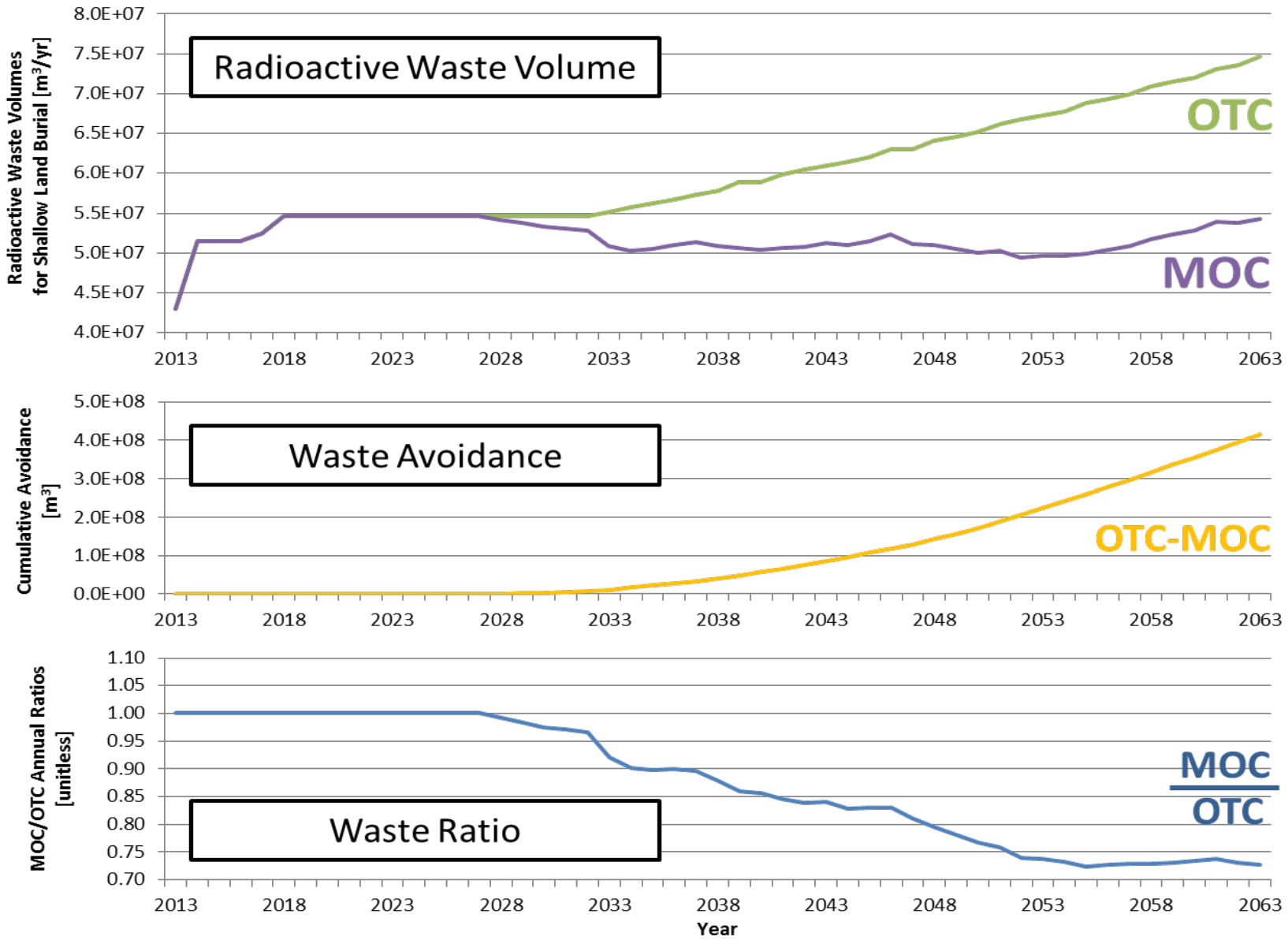


Figure 3.22 | Comparing Radioactive Waste Volumes of the MOC and OTC

LLW volumes produced annually and the cumulative LLW avoidance are shown in Figure 3.23 and do follow the same trend of the total radioactive waste volumes as shown in Figure 3.22. The first five years during recycling operations, beginning in 2028, the LLW generated is greater for the MOC than of the OTC and means that the annual ratios are greater than 1.0 in the bottom graph in Figure 3.23. From 2028 to 2033, a net increase results due to that ERU use has not begun and the use of MOX is not sufficient to lower the LLW volumes to levels lower than front-end resultant wastes to that of LLW volumes from the recycling operations.

Identifiable peaks in LLW volumes generated from the MOC are shown in the annual volumes (top graph, shown in Figure 3.22) annual ratios (bottom graph). The reasons for these peaks and their durations were described previously in Sections 3.3.2.1 and 3.3.5.

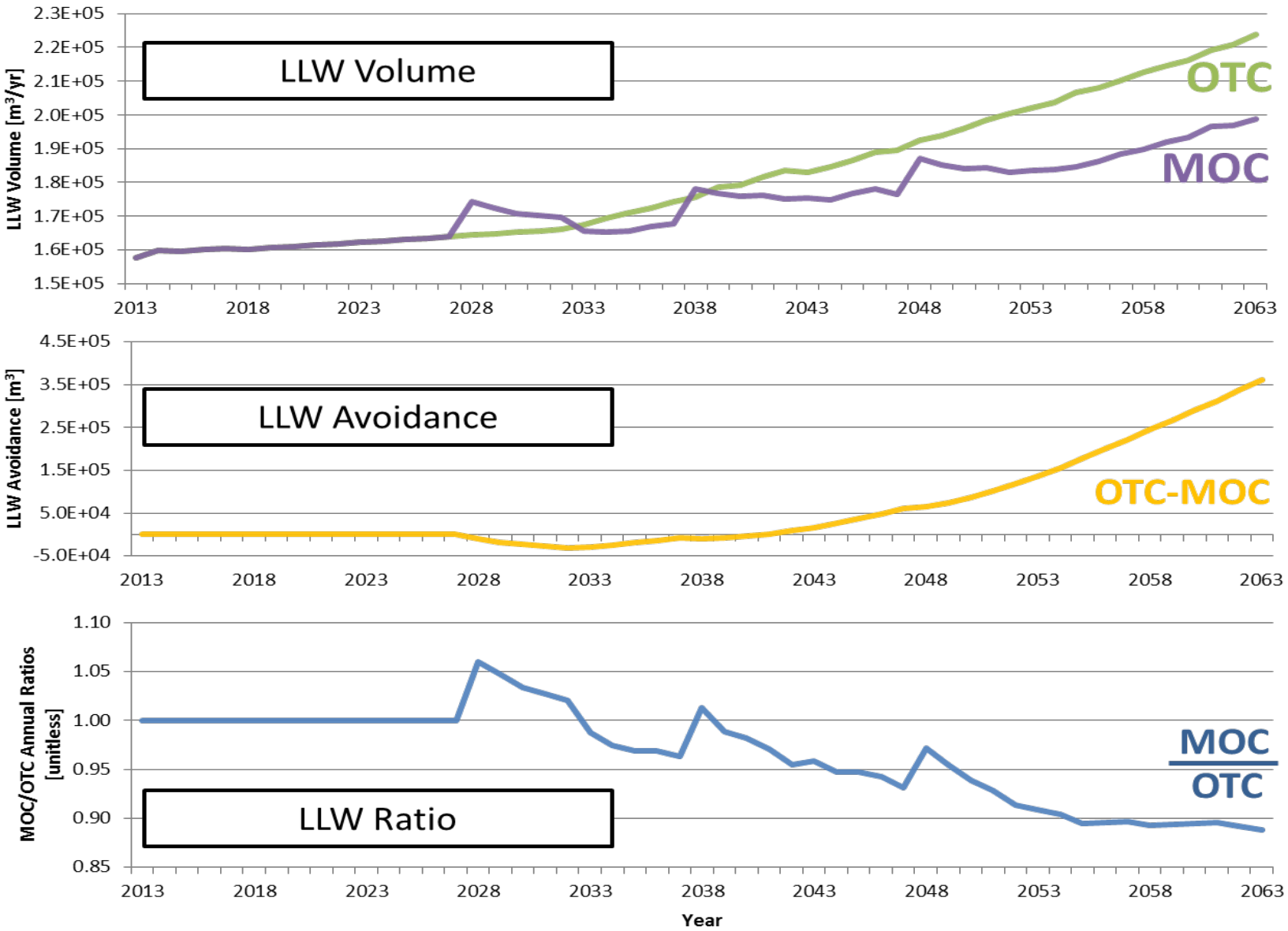


Figure 3.23 | Comparing LLW Volumes of the MOC and OTC

The posited benefits of recycling RepU and Pu in literature tend to focused on the avoidance of UNF destined for the repository [*NRC,* 1976; *DOE,* 2008a; *NWTRB,* 2011; *Park et al.,* 2011] with estimates up to 60 to 100% relative to the OTC. The findings of this study are in agreement to past studies, if and only if, discussions are limited to UNF, with reprocessing wastes that also require disposal in a geological repository being excluded (e.g., HLW, GTCC, and TRU). The amount of discharged UNF from the spent fuel pools is evaluated as the number of DSCs loaded and placed into long-term at-reactor storage as an annual ratio (blue line, Figure 3.24) The annual ratios of DSCs added to ISFSIs remain above 1.0 because increased fuel discharges occur from converting PWRs to ERU and MOX and that DSCs holding MOX are limited to less UNF because of higher short-term decay heat loadings of MOX.

The net additions of DSCs are plotted for annual ratios of UNF are plotted (purple line, Figure 3.24 and consider the withdrawals from ISFSIs for reprocessing. Net DSC additions of the MOC compared to the OTC is represented as a well-defined stepped function until 2038 when the discharged fuel from PWR conversions now has cooled for 10 years in wet spent fuel pools. Small incremental additions of 40% core discharges related to MOX are observed at the beginning from 2038-2043. Then in 2043, the initially 4 converted PWRs to ERU that occurred in 2033 due to fuel qualification purposes all now have fuel that has cooled for the 10 years and not this curve is really the shift of 10 years from the actual fuel loadings in Figure 3.5 (Section 3.3.1.3).

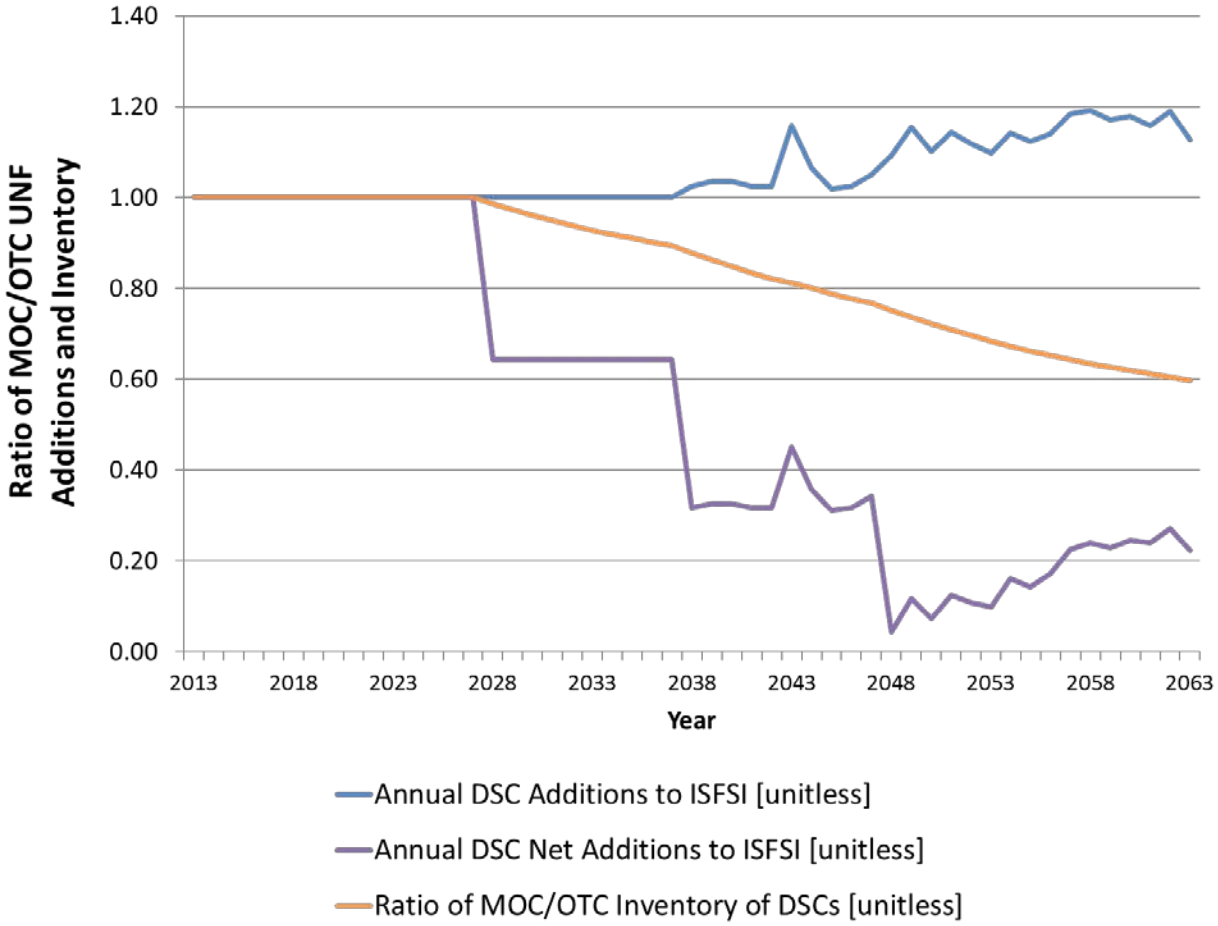


Figure 3.24 | Comparing UNF Additions and Inventories of the MOC and OTC

3.3.6.2. Comparing Integrated MOC to Integrated OTC Radioactive Waste Volumes

Integrated worker collective doses over the 50-year simulation for the MOC and OTC are listed by operation and grouped operations (shown in Table 3.6) and then are used to calculate integrated ratios. The greatest reduction in integrated ratios of waste was associated with the mining and milling wastes of around 14%. The largest increase in waste volumes was associated with the middle of the fuel cycle for LLW generation of 37%; though it should be noted that the total LLW generated from all MOC operations actually results in a 4% decrease and represents over 362,000 m^3 of LLW is avoided during the 50-year modeling time.

Table 3.6 | Comparative Integrated Radioactive Waste Volumes

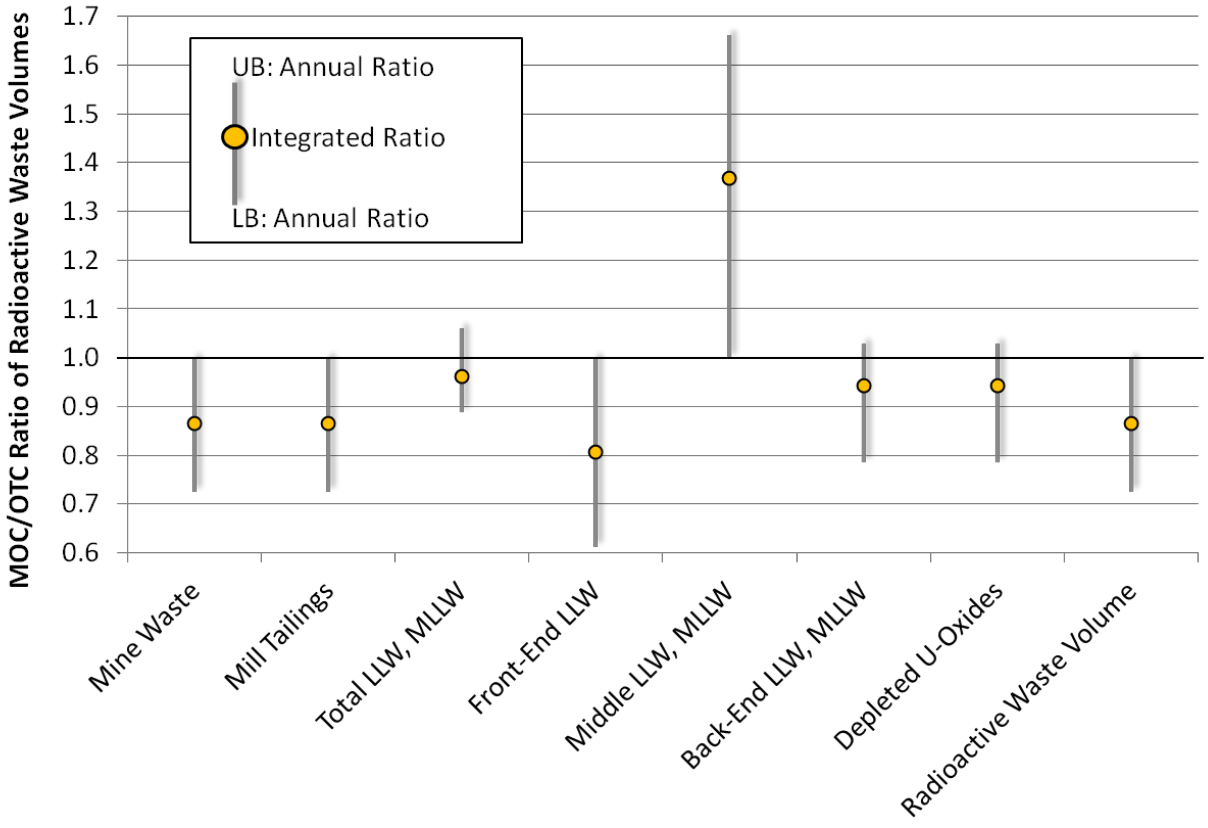
Radioactive Waste Volumes [m^3]	MOC	OTC	Difference (OTC-MOC)	Integrated Ratio (MOC/OTC)	Range of Annual (MOC/OTC) Ratios LB	UB
Mine Waste	2.14E+09	2.48E+09	3.36E+08	0.86	0.72	1.00
Mill Tailings	4.97E+08	5.75E+08	7.80E+07	0.86	0.72	1.00
LLW and MLLW	8.89E+06	9.25E+06	3.62E+05	0.96	0.89	1.06
Front-end	5.28E+06	6.55E+06	1.27E+06	0.81	0.61	1.00
Middle	3.42E+06	2.50E+06	-9.21E+05	1.37	1.00	1.66
Back-end	1.90E+05	2.01E+05	1.17E+04	0.94	0.79	1.03
Depleted Oxides	7.10E+05	7.53E+05	4.37E+04	0.94	0.79	1.03
Total Radioactive Waste Volume for Shallow Land Burial	2.65E+09	3.06E+09	4.14E+08	0.86	0.72	1.00

The range of annual MOC/OTC ratios are listed in Table 3.6 and are found by the maximum ratio and minimum ratio throughout the model years; this upper bound (UB) and lower bound (LB) is graphically represented in Figure 3.25 as a gray bar that spans the range of annual ratios. The integrated MOC/OTC ratios are shown as the yellow circles that are superimposed on the annual ratio bands that were also listed in Table 3.6.

The width of annual ratios for the operations of mining waste, mill tailings, LLW and MLLW disposal are the same as found in Figure 3.21 because the worker collective doses are directly proportional to operational throughput of mining, milling, and the amount of LLW produced.

The widest annual ratios band is observed for the LLW and MLLW for the middle of the fuel cycle (reactors and recycling operations), but the influence of the net production of LLW from all operations is at least smaller than combined contributions from other LLW-producing operations because a resultant overall 4% reduction in integrated LLW volumes occurs.

The large band of MOC and OTC annual ratios indicates that it is important to use the integrated ratio for radioactive waste volumes as a leading indicator of the MOC performance to that of the OTC. Singling out a particular year during the modeling time would be difficult, other than just using the end year of 2063 as the basis, namely because the dynamic nature of the ratios and the demand for ENU grows exponentially causing many fluctuations in annual ratios while the contribution of MOX and ERU are directly dependent on the stepped function of reprocessing capacity.



Notes: LB = lower bound; UB = upper bound

Figure 3.25 | Radioactive Waste Volumes Integrated and Annual Ratios

3.4. Verification of MOC Worker Collective Doses

Relative radiological impacts to workers (worker collective dose and LCFs) estimated by others have resulted with a net increase of around 2-9% (see Table 3.2, Section 3.1.2) [*NEA*, 2000; *DOE*, 2008a]. The integrated worker collective dose ratio for this study estimates an increase of 0.2% of cumulative worker collective doses incurred during the 50-year model of the MOC to that of the OTC (see Section 3.3.5, Table 3.6, and Figure 3.25). Be that as it may, the worker collective dose annual ratios at the end of the simulation shown in Figure 3.20 (Section 3.3.5) are less than 1.0 and represent a near 0.5-0.7% decrease will be expected from transitioning to the MOC relative to that of the OTC. This is important as the annual ratio towards the end of the simulation is the basis for verification against the NEA (2000) and DOE (2008a) studies.

Although both sets of values of (+) 2-9% and (–) 0.7% could draw conclusions that there is almost difference of the MOC to OTC in terms of the worker collective dose metric, but the disparity of the results must be evaluated between the present study and the literature. The factors that contribute to these differences and a method to verify the present MOC study are presented within the following sections. Finally, an analysis to further investigate the potential net worker collective doses is performed and provides a potential model to elucidate the possible net outcomes with respect to varying fuel-use scenarios.

3.4.1. Distinct Analyses: Model Behavior, Reactor Portfolios, Fuel Recycle Extent and Use

After further investigation of the models, objectives, and assumptions employed in the NEA (2000) and DOE (2008a) studies, it can be fair to claim that the present comparative study results, as presented in this work cannot be directly verified against NEA (2000) and DOE (2008a) when evaluating relative worker collective doses of the MOC to OTC. Three factors make the case that the current MOC study is sufficiently different from the two references:

(1) Simulation Approaches: Steady-state vs. Dynamic

(2) Recycled Materials: Pu-only vs. RepU + Pu

(3) Fuel Type Use in Reactor Mix at MOC End-State: [1 : 1] vs. [1 : 1.5 : 3.9]

The first and most obvious difference is that the approach to simulating the overarching forcing function of energy demand is constant in the NEA (2000) study. Within dynamic studies, the overall benefit or additional risk is influenced by the extent of recycling employed, energy growth assumptions, and the end-state reactor technology mix. The steady-state scenarios eliminate these constraints, but cannot capture the complexities and interactions of individual NFC operations that can point to unexpected and unforeseen issues that contribute or detract from the total MOC worker collective doses (or for any other performance metric).

The second factor is that both ERU and MOX are used in the current MOC study and only MOX is considered in NEA (2000) and DOE (2008a). At steady-state the extent of replaced ENU for MOX-only cases is only 40% whereas the 100% of the cores are replaced with ERU and the use of ENU is eliminated. The NWTRB (2011, 2012) study demonstrated that the use of both MOX

and ERU within the MOC will result with a greater amount of ENU savings than of when one of the fuel types are used; resultant net impacts of waste generation will be enhanced accordingly to the amount of ENU avoided (as discussed in Section 3.1.2).

Furthering that concept of ENU savings by the incorporation of both ERU and MOX into the MOC model is valuable to the verification process because the current MOC study uses a combination of fuel types incorporated as a function of reprocessing capacity (not setting the reactors and assuming unlimited reprocessing capacity to meet demand). NEA (2000) and DOE (2008a) assume that the comparison of worker collective dose is against a MOC with 100% MOX at the MOC end-state and the current MOC study ends with a reactor fleet using 24% MOX, 15.5% ERU and 60.5% ENU. The extent of use of MOX, ERU, and ENU in varying combinations and the requirement of ENU reduction is demonstrated with a simplified example that only includes 3 PWRs and running at steady-state.

The combination of recycling both elements, Pu and RepU, broken down by each type of combination of ENU, MOX and ERU-loaded PWRs and the reduction of ENU use is shown below (Table 3.7). Within Table 3.7 is an exercise to show the reduction of ENU when using a combination of the three fuel types in various ratios. The possible nine mutually exclusive combinations of the use of ENU, ERU, and MOX are shown for a 3-PWR fleet, all running at steady-state. The ratios of ENU fuel use for each PWR according to the designation of fuel type are shown under the column headings “PWR 1, PWR 2, and PWR 3.” A ratio for a PWR running on MOX lists an ENU fuel use of 0.6 (since 60% of the core is ENU and the remaining 40% is MOX), a PWR on ENU is 1.0 and a PWR on ERU is 0.0. Not surprisingly, as the number of

PWRs loaded with MOX increases, the reduction of ENU also increases; when the compounding effect of ERU additions and then eventually the 3-PWR fleet is loaded with ERU only, then the savings of ENU reaches 100%.

Table 3.7 | ENU Use Reductions of Combinations of ERU and MOX Use in 3-PWR Fleet

Scenarios Mix of # PWRs Loaded ENU: MOX: ERU	PWR 1	PWR 2	PWR 3	Ratio: Scenario ENU Use to 3-PWR ENU Use	% Reduction of ENU Use
3 : 0 : 0	1.0	1.0	1.0	1.00	0 %
2 : 1 : 0	1.0	1.0	0.6	0.87	13 %
1 : 2 : 0	1.0	0.6	0.6	0.73	27 %
0 : 3 : 0	0.6	0.6	0.6	0.60	40 %
1 : 1 : 1	1.0	0.6	0.0	0.53	47 %
0 : 2 : 1	0.6	0.6	0.0	0.40	60 %
1 : 0 : 2	1.0	0.0	0.0	0.33	67 %
0 : 1 : 2	0.6	0.0	0.0	0.20	80 %
0 : 0 : 3	0.0	0.0	0.0	0.00	100 %
0.7 : 0.47 : 1.83	--	--	--	0.25	75 %

The extent of use of MOX, ERU, and ENU in varying combinations and the requirement of ENU reduction is demonstrated with a simplified example that only includes 3 PWRs and running at steady-state. The purpose of this exercise in Table 3.7 is that the DOE (2008a) and NEA (2000) study assumes that the reactor fleet will completely be supported by MOX (as shown in the row highlighted in gray in Table 3.7). A reactor fleet supported by all MOX uses 60% of the amount of ENU compared to the OTC and the reductions related to worker collective dose is influenced by the reduction of front-end processes and additions of reprocessing and MOX fuel fabrication. Within this MOC study, the ratio of reactors that use ENU, ERU, and MOX do not split evenly nor result in integers (see yellow-highlighted box in Table 3.7). Clearly, the reductions in ENU use for the NEA (2000) and DOE (2008a) study do not match the ENU savings used here as the end-state for the MOC model of 75% ENU savings to that of the OTC.

In summary, the only recent analyses [*NEA*, 2000; *DOE*, 2008a] that compare MOC to OTC worker radiological impacts are similar to each other because of the sole use of MOX as a recycled fuel product in the modeled MOC along with the assumption of unlimited reprocessing capacity to produce Pu for MOX. Additionally, the use of MOX in the MOC in the two studies is utilized in all LWRs in the reactor fleet modeled and then the worker radiological doses are then compared to the OTC when the MOX-supported LWR fleet is then at steady-state. This is unlike the present study where both ERU and MOX are incorporated into the MOC model at a rate that is supported by realistic rates of reprocessing and construction resulting with a net ENU savings of 75%; this is dissimilar to the NEA (2000) and DOE (2008a) studies in which a savings of 40% would be observed.

3.4.2. Renormalizing MOC Worker Collective Doses for Verification

To compare and contrast the present study results to NEA (2000) and DOE (2008a), the worker collective dose performance metrics must be scaled by a steady state scenario that includes use of ENU, ERU, and MOX. Worker collective dose metrics are listed in Table 2.4 and Table 2.5 (Section 2.2.3) and Table 3.3 (Section 3.2.3) and are applied to a steady-state scenario where each reactor is supported by its own set of NFC facilities. In other words, the worker collective doses associated with the three unique NFCs to support the sole reprocessing and use of each fuel (as listed as the column heading “Fuel Use in Isolated System,” in Table 3.8). With that, there is no “sharing” of worker collective doses for withdrawing fuel from ISFSIs and

reprocessing to make RepU and Pu are accounted separately[20] (the worker collective doses form separating sufficient UNF to make ERU is a separate calculation from worker collective doses from separating sufficient UNF to produce MOX).

When the worker collective dose performance metrics are applied to the steady-state scenarios[21] shown in Table 3.8, the NEA (2000) and DOE (2008a) results are verified that a reactor fleet completely fueled with MOX will result with a higher worker collective dose (1065.0 person-mSv/GWe) than if fueled completely with ENU or ERU. ERU-fueled PWRs and the supporting NFC resulted with the lowest steady-state (858.1 person-mSv/GWe). A single-loaded PWR with ENU was estimated to produce 977.5 person-mSv/GWe.

When evaluating the steady-state reactor fleet, ERU fuel fabrication worker collective doses are less than ENU and MOX. This is interesting as ERU contains traces of fission products and other radionuclides from previous irradiation cycles that could present a higher worker collective dose than the other two types of fuels. The fuel fabrication metrics of MOX, ERU, and ENU were taken from recent industry experience (see Appendix A), but the most notable reason why ERU fuel fabrication worker collective doses would be lower is because the physical radiation protection that is implemented for each type of fuel. ERU is produced at the Romans (France) facility and the ERU line is separated and more radiation protection barriers are provided to protect workers against external radiation [*IAEA*, 2007a] compared to MOX and ENU. ENU fuel fabrication is done so by hand without radiation protection while MOX fuel fabrication is

[20] It should be noted that the sole use of RepU as part of ERU in an "isolated system" that does not co-produce Pu as part of MOX not likely feasible from a policy perspective because burning Pu was, and still is a major driver for MOX and MOC development.

[21] It should be noted that the three NFCs shown in Table 3.8 does not exactly match the MOC evaluated because of the exclusion of BLEU use that used throughout the modeling time frame.

performed within a glove-box type facility where inhalation of Pu is the main radiological concern so little external radiation protection is provided with glove-box equipment.

The ratio of MOX use and ENU use from the values in Table 3.8 is 1.09, or a 9% increase of the MOX use compared to that of the OTC. This is consistent with the NEA (2000) and DOE (2008a) results that estimated an increase of between 2-9%. It is interesting to note that the decrease in worker collective doses associated with an NFC using ERU-only in the PWR will result with a 12% reduction if using the values from Table 3.8.

Table 3.8 | Worker Collective Dose Performance Metrics Renormalized for Steady-State Comparisons of ENU, MOX, ERU Use in PWRs

Nuclear Material Mass	Fuel Use in Isolated/Unique Systems [MT/yr]		
	MOX	ERU	ENU
Annual Refueling Requirements			
ENU	12.7	0.0	21.1
MOX	8.4	0.0	0.0
Pu	0.8	0.0	0.0
DU	7.6	0.0	0.0
ERU	0.0	21.1	0.0
Feed (including inefficiencies)			
NU	132.2	0.0	220.3
DU	7.59	0.0	0.0
Pu	0.86	0.0	0.0
RepU	0.0	123.0	0.0
UNF	69.6	119.9	0.0
NFC Operations	Worker Collective Dose [person-mSv/GWe-yr]		
U-recovery (global mix)	65.0	0.0	108.4
Milling	29.6	0.0	49.4
Conversion	13.9	0.0	23.2
Enrichment	1.7	0.0	4.7
Fuel Fabrication	22.2	0.0	59.7
Reactor Operations	678.6	678.6	678.6
Reprocessing	36.2	62.4	0.0
RepU Conversion	0.0	3.8	0.0
RepU Enrichment	0.0	16.5	0.0
ERU Fuel Fabrication	0.0	3.0	0.0
MOX Fuel Fabrication	140.8	0.0	0.0
Subtotal [person-mSv/GWe-yr]	**988.0**	**764.3**	**923.9**
Dry Interim Storage	37.3	53.3	10.7
Add	12.6	10.7	10.7
Withdrawal	24.7	42.6	0.0
Deconversion	4.9	4.6	8.8
LLW Disposal	34.7	35.9	34.1
Total [person-mSv/GWe-yr]	**1065.0**	**858.1**	**977.5**

In the previous section (Section 3.4.1) Table 3.7 was part of an exercise to show the reduction of ENU when using a combination of the three fuel types in various ratios. The possible nine mutually exclusive combinations of the use of ENU, ERU, and MOX are shown for a 3-PWR fleet, all running at steady-state; Table 3.7 also presented the fuel mix that was employed in the current MOC study. Following that same vein of logic, the steady-state values of estimated worker collective doses for each reactor system was weighted to the fuel mix that was incorporated by year 2063 of the current MOC model (23.9% MOX; 15.5% ERU; 60.6% ENU)[21] and is shown in Table 3.9.

To avoid double counting worker collective doses for this exercise, the higher of the worker collective doses responsible from MOX or ERU was used for UNF withdrawals from ISFSIs and reprocessing. The rationale was that for the MOC modeled, the ISFSI withdrawal and reprocessing amount would be the required and the other smaller of the MOX or ERU would be made from the remainder of material. If worker collective doses were scaled from the amount of UNF required to produce MOX and then a unique stream of UNF for sourcing ERU material, effectively this would result with double counting as the UNF stream contains nuclear material for both fuel products.

The weighted average of the reactor fleet with the 23.9% MOX; 15.5% ERU; 60.6% ENU fueled PWRs resulted in a total of 964.5 person-mSv/GWe which represents the modeled MOC in this study; this is 13 person-mSv/GWe less than the total worker collective dose for ENU-only and MOX-only loaded PWRs shown in Table 3.8, respectively. This is a decrease of 1.3% of the mix that is employed in the current MOC study to the ENU-loaded reactors and 10.5% lower than the

MOX-loaded reactor case. The current MOC study is now verified; when using the mix between ENU, ERU, and MOX, the results will not be the same as when comparing to a NFC strictly using MOX.

Table 3.9 | Worker Collective Dose Performance Metrics Renormalized for Steady-State Comparisons of ENU, MOX, ERU Use in PWRs

Nuclear Material Mass	MOC Fuel Use [MT/yr]		
	MOX (23.9%)	ERU (15.5%)	ENU (60.6%)
Annual Refueling Requirements			
ENU	3.0	0.0	12.8
MOX	2.0	0.0	0.0
Pu	0.2	0.0	0.0
DU	1.8	0.0	0.0
ERU	0.0	3.3	0.0
Feed (including inefficiencies)			
NU	31.7	0.0	133.4
DU	1.8	0.0	0.0
Pu	0.2	0.0	0.0
RepU	0.0	19.1	0.0
UNF	47.3		0.0
NFC Operations	Worker Collective Dose [person-mSv/GWe-yr]		
U-recovery (global mix)	15.6	0.0	65.6
Milling	7.1	0.0	29.9
Conversion	3.3	0.0	14.1
Enrichment	0.4	0.0	2.8
Fuel Fabrication	5.3	0.0	36.2
Reactor Operations	162.5	105.1	411.0
Reprocessing	24.6		0.0
RepU Conversion	0.0	0.6	0.0
RepU Enrichment	0.0	2.6	0.0
ERU Fuel Fab	0.0	0.5	0.0
MOX Fuel Fab	33.7	0.0	0.0
Subtotal [person-mSv/GWe-yr]	**252.5**	**108.7**	**559.5**
Dry Interim Storage	--	--	--
Add	12.6	10.7	10.7
Withdrawal	16.8		0.0
Deconversion	1.2	0.7	5.3
LLW Disposal	8.3	5.6	20.6
Subtotal [person-mSv/GWe-yr]	**281.8**	**116.7**	**566.0**
Total [person-mSv/GWe-yr]	**964.5**		

It is obvious that the answer will be a reactor set that utilizes all ERU fuel since the worker collective dose per GWe basis was the lowest for this fuel type and the highest net impact will be a scenario that only uses MOX within the advanced NFC (as seen in Table 3.8). Nonetheless, the net reductions of worker collective dose caused from changing the mix of fuels employed (as demonstrated in Table 3.9) demonstrates that benefits can vary depending on the mix of PWRs loaded with different fuel types – all within the same reactor fleet running at steady-state. Correspondingly, the net differences of worker collective doses are reduced by lessened front-end operations, but increased doses from recycling operations and waste handling; the net result does not necessarily correlate linearly to the amount of ENU avoided. This begs further questions relating to what other combinations are possible that will also provide a lower net worker collective dose impact compared to the OTC.

An evaluation has been performed of all of the possible combinations of use for MOX, ERU, and ENU and resultant net ratios of worker collective doses as shown in Figure 3.26 and Figure 3.27. The weighted average method demonstrated in Table 3.9 was applied to all combinations in discretized intervals of 0.05 for each of the three axes in the 3-d surface plot (Figure 3.26) and then a plan-view of that 3-d plot is made to show all combinations possible (Figure 3.27) and the net worker collective dose impact that would result compared to the OTC. The fuel-use combination are plotted on the x- and y-coordinates while the z-coordinate shows the net worker collective dose impact (and designated with coordinated color mapping). The y-coordinate variable is the ratio of MOX and ERU fuel to the total fuel amount required when both the MOC and OTC are operating at steady-state. The ratio ranges from 0.0 to 1.0, representing the lower bound of no MOX and ERU (essentially the OTC using only ENU) to the use of only ERU and

MOX as a possible scenario for the MOC. The total amount of fuel is the sum of ENU, MOX, and ERU fuel used in the MOC steady-state scenario. The x-coordinate variable is the ratio of ERU to MOX use in the steady-state scenario and also ranges from 0.0 to 1.0; where 0.0 represents no use of ERU and all use of MOX and 1.0 denotes an MOC case that uses only ERU fuel within LWRs.

When the ratios of worker collective dose of the hypothetical MOC scenario, as defined by the combination of the x- and y-variable to the OTC, are plotted in a 3-d manner, the possible combinations create a surface plot. This plot allows one to see the places of maximum, minimum, etc., between the options and net worker collective dose impacts as a function of the fuel-use combinations of ERU, ENU, and MOX. The OTC in this case is represented by any point along the x=0.0 coordinate because y is invariant due to the specification of no use of ERU or MOX.

Figure 3.26 is the graphical equivalent of demonstrating that the NEA (2000) and DOE (2008a) studies are substantially different studies compared to this current MOC study and that it would be expected that the ratio of worker radiological impacts would not match. The assumption that only MOX will be used within the end-state of NEA (2000) and DOE (2008a) is represented as a green arrow and shows that the increase in worker collective dose expected is around 9% compared to the OTC; the yellow arrow represents the current study and shows that the fuel mix combination[22] of (x,y) = (0.39, 0.39) results with a 0.4% reduction of worker collective doses. As expected and discussed previously, the lowest ratio of the worker collective doses is shown at

[22] Where the x-coordinate is the ratio of [ERU/(ERU+MOX)] = 15.5 % /(15.5+ 23.9%) = 0.39;
and the y-coordinate is the ratio of [(MOX+ERU)/Total] = (23.9% + 15.5%) / 100% = 0.39;

coordinates (1.0, 1.0) where a hypothetical MOC scenario only using ERU fuel is shown with a reduction of worker collective dose impacts of 12%. The annual ratio of 1% reduction of worker collective doses agrees with the results produced for the annual ratio of the simulation year of 2063 when it was estimated that 0.5% reduction would be expected.

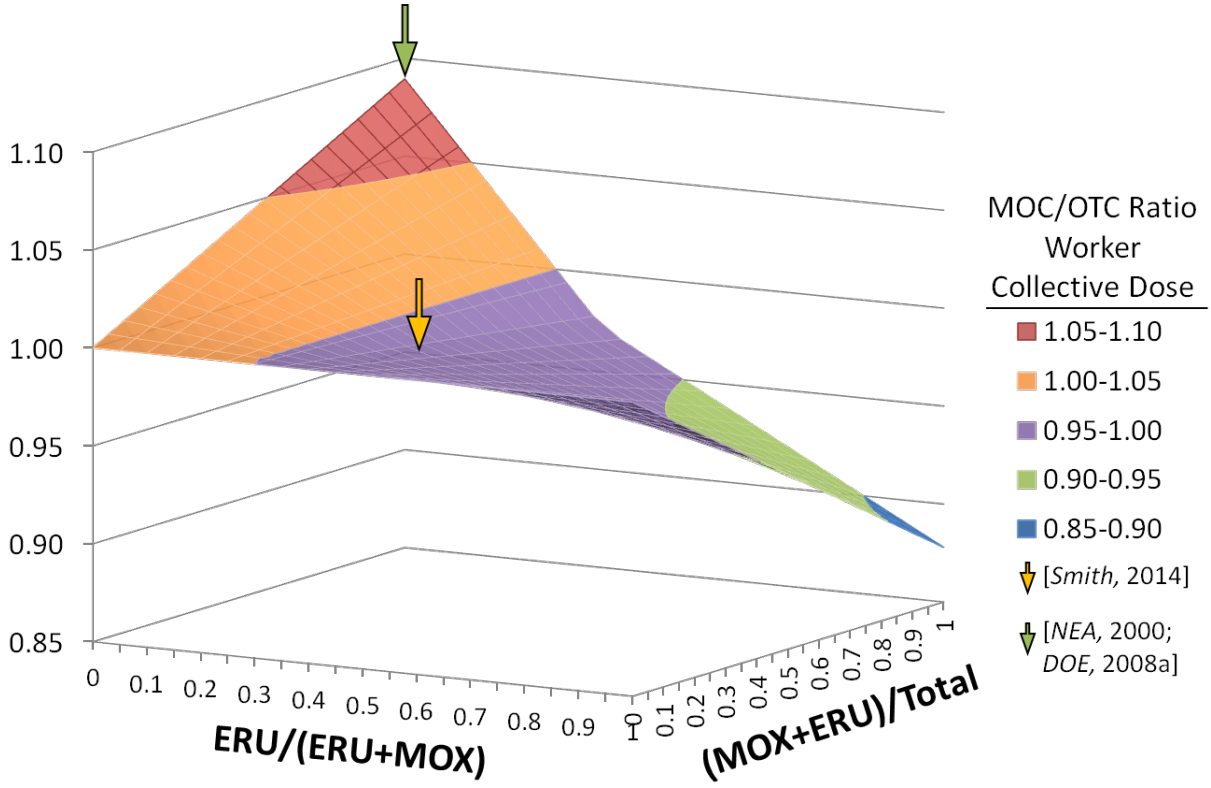


Figure 3.26 | Steady-State MOC/OTC Worker Collective Dose Ratios Mapped on Potential Fuel-Use Combinations Surface Plot

Figure 3.27 represents the same information shown in Figure 3.26, but as a 2-d plot such that all worker collective dose ratios can be seen in one image. The current MOC study sample space is shown as a yellow star and the NEA (2000) /DOE (2008a) location is shown on the 2-d plot as a green star. The utility of Figure 3.27 is such that it can be used to evaluate any other possible

combinations of steady-state MOC scenarios that use ERU, MOX, and ENU. For combinations falling between the major gridlines for x- and y-coordinates, linear interpolations can be used for calculating the ratio of MOC and OTC worker collective doses.

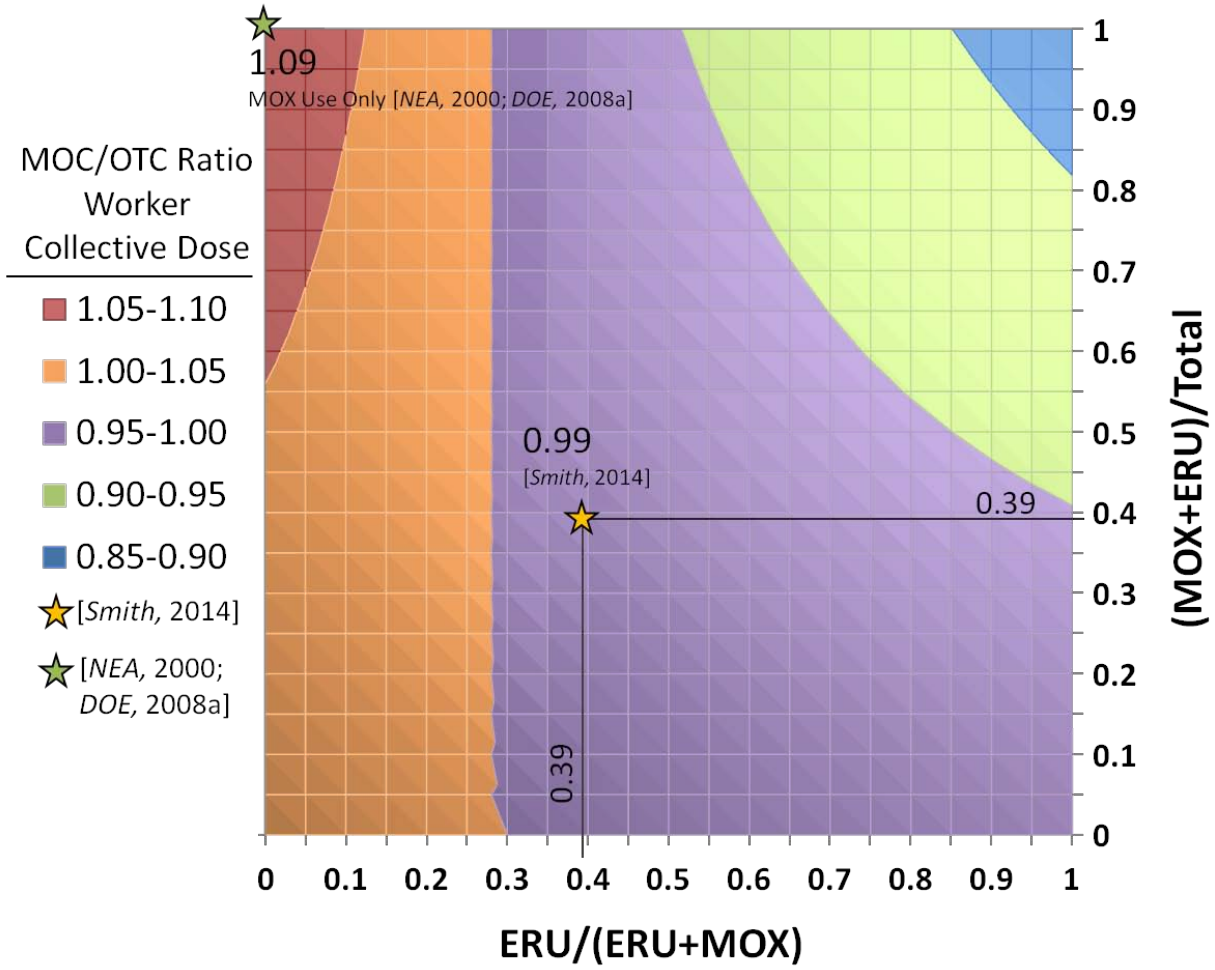


Figure 3.27 | Steady-State MOC/OTC Worker Collective Dose Ratios Mapped on Potential Fuel-Use Combinations Plan-View of Surface Plot

3.5. Verification of MOC Radioactive Waste Volumes

Looking to past studies of the estimated net impacts for waste production, the study presented here aligns closely with the net avoidance of UNF (as discussed above) and the radioactive waste volumes that are eligible for disposing via shallow-land burial. Using values from the NWTRB study (2011), which most closely matched the present study's modeling boundaries, the avoidance of radioactive waste volumes was estimated to be nearly 15% when ENU use is replaced with a combination of both ERU and MOX in the 130 LWR fleet modeled in 2063. The integrated avoidance of radioactive waste volumes predicted for this study was 14%.

DOE (2008a) LLW volumes were estimated for reactor and reprocessing operations[23] only and that around 63,600 m^3 LLW will be produced from reprocessing each year to support a 200 GWe LWR fleet using only MOX fuel – that rounds to nearly 320 m^3 LLW/GWe. The values estimated here are similar to 310 m^3 LLW/ GWe for sufficient reprocessing capacity to support 1 GWe PWR (as described in Appendix B, Section B.13).

However, the estimates of LLW volumes performance metrics for reactor operations are not consistent with the present analysis. The DOE (2008a) reports that approximately 21 m^3 to 79 m^3 of LLW is produced in a year for a typical 1 GWe LWR. The source of information for the LLW volume performance metrics was from the Nuclear Energy Institute [*NEI,* 2007] but the website is no longer accessible. When checking against more recent references, the World Nuclear Association [*WNA,* 2013i] states that a typical 1 GWe LWR anywhere in the world will produce

[23] GTCC LLW were estimated for decommissioning of reactors within DOE (2008a), but is not included in this discussion since only LLW volumes produced from routine operations were considered for this present analysis.

around 200-350 m^3 of LLW and the values that were used in producing the LLW volume performance metric (as described in Appendix B, Section B.13) are consistent with this value for PWRs (288 m^3 LLW/GWe PWR) but reports a higher value for BWRs (627 m^3 LLW/ GWe BWR). The higher LLW volumes associated with BWRs are due to additional maintenance for more extensive piping systems that carry radioactively contaminated primary coolant water and additional reactor control chemistries of borated solutions are required due to the engineered coolant system [*Saling & Fentiman*, 2002; *EPRI*, 2007].

Because there is a difference in the LLW volume performance metric with DOE (2008a) values that cannot be traced to the original data sources for further clarification, the DOE (2008a) integrated and annual ratios for LLW will not be used as a verification source. However, results will be presented for cases that only use MOX and it is postulated that if values in DOE (2008a) were updated, the net impacts would be similar to the ones estimated here.

3.5.1. Renormalizing MOC LLW Volumes

LLW volumes resultant from the current MOC study were in alignment to those predicted by NWTRB (2011, 2012); but even though they could not be verified against DOE (2008a), it would still be of interest to evaluate in a similar manner the expected net LLW volumes associated with mixed fuel uses in hypothetical MOC scenarios as was done in Section 3.4.2. The first step in which to do so is to calculate the three unique 1-PWR NFCs using MOX, ENU, or ERU and use the LLW volume performance metrics to calculate the single-PWR NFC LLW volumes for each. The results are shown in Table 3.10.

When the LLW volume performance metrics are applied to the steady-state scenarios, shown in Table 3.10 a reactor fleet completely fueled with MOX will result with the lowest LLW volumes (1419 m^3 LLW/GWe) on a PWR than either a PWR using ERU (2102 m^3 LLW/GWe) or a PWR using ENU (2069 m^3 LLW/GWe).

The ratio of ERU use and ENU use from the values in Table 3.10 is 1.02, or a 2% increase of the ERU use compared to that of the OTC. It is interesting to note that the decrease in LLW volume generation associated with an NFC using MOX in the PWR will result with a 31% reduction if using the values from Table 3.10.

Table 3.10 | LLW Volume Performance Metrics Renormalized for Steady-State Comparisons of ENU, MOX, ERU Use in PWRs

Nuclear Material Mass	Fuel Use in Isolated/Unique Systems [MT/yr]		
	MOX	ERU	ENU
Annual Refueling Requirements			
ENU	12.7	0.0	21.1
MOX	8.4	0.0	0.0
Pu	0.8	0.0	0.0
DU	7.6	0.0	0.0
ERU	0.0	21.1	0.0
Feed (including inefficiencies)			
NU	132.2	0.0	220.3
DU	7.59	0.0	0.0
Pu	0.86	0.0	0.0
RepU	0.0	123.0	0.0
UNF	69.6	119.9	0.0
NFC Operations	LLW Volumes [m^3 LLW/GWe-yr]		
U-recovery[a] (global mix)	--	--	--
Milling[a]	--	--	--
Conversion	4.4	0.0	7.3
Enrichment	17.4	0.0	47.0
Fuel Fabrication	621.2	0.0	1673.9
Reactor Operations	311.5	311.5	311.5
Reprocessing	323.1	557.1	--
RepU Conversion	0.0	7.0	--
RepU Enrichment	0.0	165.2	--
ERU Fuel Fabrication	0.0	1046.1	--
MOX Fuel Fabrication	124.8	0.0	--
Subtotal [m^3 LLW/GWe-yr]	**1402.3**	**2086.9**	**2039.6**
Dry Interim Storage[b]	--	--	--
Add	--	--	--
Withdrawal	--	--	--
Deconversion	16.5	15.2	29.4
LLW Disposal[b]	--	--	--
Total [m^3 LLW/GWe-yr]	**1418.8**	**2102.1**	**2069.0**

Notes: [a] Uranium recovery and milling wastes are considered TENORM (technologically enhanced naturally-occurring radioactive material) and byproduct waste ("mill tailings"). [b] It is assumed that dry interim UNF storage operations and LLW disposal facility operations produce negligible LLW volumes.

The steady-state values of estimated LLW volumes for each reactor system was weighted to the fuel mix that was incorporated by year 2063 of the current MOC model (23.9% MOX; 15.5% ERU; 60.6% ENU) and is shown in Table 3.11

The weighted average of the reactor fleet with the 23.9% MOX; 15.5% ERU; 60.6% ENU fueled PWRs resulted with a total of 1823 m^3 LLW/GWe which represents the modeled MOC in this study; this is 246 m^3 LLW /GWe less than the total LLW volume for ENU-only and 404 m^3 LLW /GWe more for MOX-only loaded PWRs shown in Table 3.11 respectively. The specific combination of ERU, ENU, and MOX results with a decrease of 12% compared to that of ENU-loaded reactors and 14% lower than the ERU-loaded reactor case, but will result with a 28% higher LLW volume amount on a per GWe basis at steady-state for MOX-only loaded PWRs.

Table 3.11 | LLW Volume Performance Metrics Renormalized for a Combination of ENU, MOX, ERU Use in LWR Fleet

Nuclear Material Mass	MOC Fuel Use [MT/yr]		
	MOX (23.9%)	ERU (15.5%)	ENU (60.6%)
Annual Refueling Requirements			
ENU	3.0	0.0	12.8
MOX	2.0	0.0	0.0
Pu	0.2	0.0	0.0
DU	1.8	0.0	0.0
ERU	0.0	3.3	0.0
Feed (including inefficiencies)			
NU	31.7	0.0	133.4
DU	1.8	0.0	0.0
Pu	0.2	0.0	0.0
RepU	0.0	19.1	0.0
UNF	47.3		0.0
NFC Operations	LLW Volumes [m^3 LLW/GWe-yr]		
U-recovery[a] (global mix)	--	--	--
Milling[a]	--	--	--
Conversion	1.0	--	4.4
Enrichment	4.2	--	28.4
Fuel Fabrication	148.7	--	1013.8
Reactor Operations	74.6	48.3	188.6
Reprocessing	86.3		--
RepU Conversion	0.0	1.1	--
RepU Enrichment	0.0	25.6	--
ERU Fuel Fab	0.0	162.1	--
MOX Fuel Fab	29.9	0.0	--
Subtotal [m^3 LLW/GWe-yr]	**344.7**	**237.0**	**1235.3**
Dry Interim Storage[b]	0.0	0.0	0.0
Add	0.0	0.0	0.0
Withdrawal	0.0		0.0
Deconversion	3.9	2.4	17.8
LLW Disposal[b]	0.0	0.0	0.0
Subtotal [m^3 LLW/GWe-yr]	**348.7**	**239.4**	**1235.3**
Total [m^3 LLW/GWe-yr]	**1823.3**		

Notes: [a] Uranium recovery and milling wastes are considered TENORM (technologically enhanced naturally-occurring radioactive material) and byproduct waste ("mill tailings"). [b] It is assumed that dry interim UNF storage operations and LLW disposal facility operations produce negligible LLW volumes.

An evaluation similar to the worker collective dose section (Section 3.4.2) has been performed of all of the possible combinations of use for MOX, ERU, and ENU and resultant net ratios of LLW volume generation as shown in Figure 3.28 and Figure 3.29. The weighted average method demonstrated in Table 3.11 was applied to all combinations in discretized intervals of 0.05 for each of the three axes in the 3-d surface plot (Figure 3.28) and then a plan-view of that 3-d plot is made to show all combinations possible (Figure 3.29) and the net LLW volume generation that would result compared to the OTC. The fuel-use combination are plotted on the x- and y-coordinates while the z-coordinate shows the net LLW volume generation (and designated with coordinated color mapping). The y-coordinate variable is the ratio of MOX and ERU fuel to the total fuel amount required at a certain point in time when both the MOC and OTC are operating at steady-state. The ratio ranges from 0.0 to 1.0, representing the lower bound of no MOX and ERU (essentially the OTC using only ENU) to the use of only ERU and MOX as a possible scenario for the MOC. The total amount of fuel is the sum of ENU, MOX, and ERU fuel used in the MOC steady-state scenario. The x-coordinate variable is the ratio of ERU to MOX use in the steady-state scenario and also ranges from 0.0 to 1.0; where 0.0 represents no use of ERU and all use of MOX and 1.0 denotes an MOC case that uses only ERU fuel within LWRs.

When the ratios of LLW volumes of the hypothetical MOC scenario as defined by the combination of the x- and y-variable to the OTC are plotted in a 3-d manner, the possible combinations creates a surface plot that allows one to see the places of maximum, minimum, and all between of the options net LLW volume generation as a function of the fuel-use combinations of ERU, ENU, and MOX. The OTC in this case is represented by any point along the x=0.0 coordinate because y is invariant due to the specification of no use of ERU or MOX.

Figure 3.28 is the graphical equivalent of demonstrating studies that only consider MOX in PWRs are substantially different studies against the current MOC study and that it would be expected that the ratio of LLW volume generation would not match. The assumption that only MOX will be used within the end-state of a comparative study is represented as a green arrow and shows that the increase in LLW volumes expected is around 31% compared to the OTC; the yellow arrow represents the current study fuel-use combination and shows that the fuel mix combination of (x,y) = (0.39, 0.39) results with a 11% reduction of LLW volume generation. The annual ratio of 11% reduction of LLW volume generated agrees with the results produced for the annual ratio of the simulation year of 2063 when it was estimated that 12 % reduction would be expected. Also expected and discussed previously, the lowest ratio of the LLW volumes is shown at coordinates (0.0, 1.0) where a hypothetical MOC scenario only using MOX fuel is shown with a reduction of LLW volume generation of 31%.

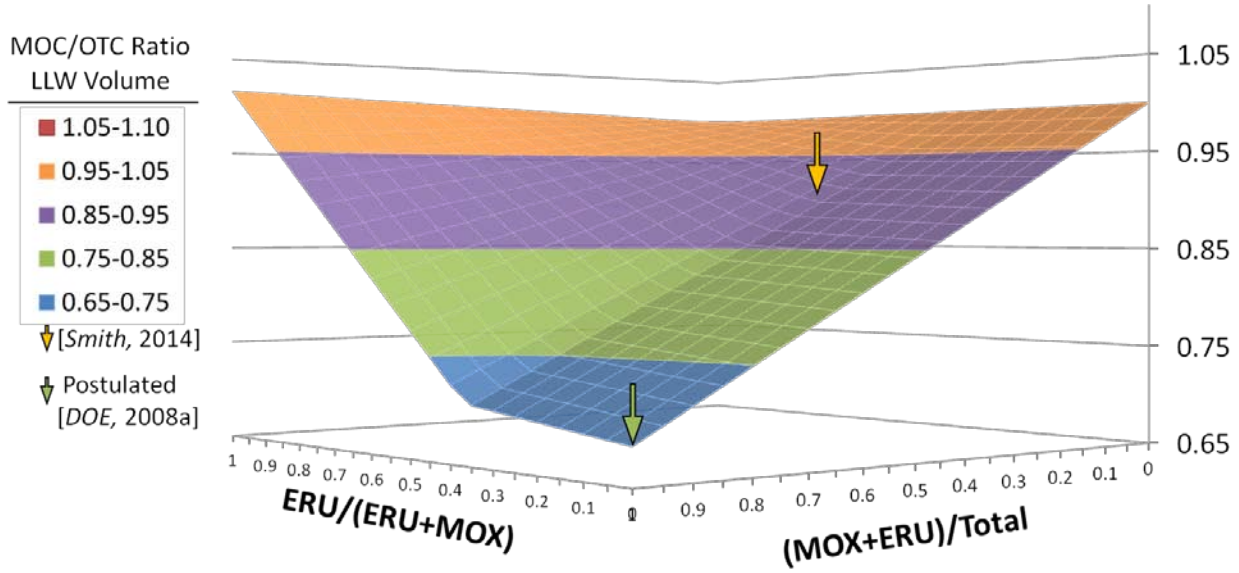


Figure 3.28 | Steady-State MOC/OTC LLW Volume Ratios Mapped on Potential Fuel-Use Combinations Surface Plot

Figure 3.29 represents the identical data in Figure 3.28, but as a 2-d plot such that all LLW volume ratios can be seen in one image. The current MOC study sample space is shown as a yellow star. The MOX-only scenario is shown on the 2-d plot as a green star; it is postulated that this would be result for the DOE (2008a) steady-state comparative study if updated values were used. The utility of Figure 3.27 is such that it can be used to evaluate net LLW volume generations any other possible combinations of steady-state MOC scenarios that use ERU, MOX, and ENU by linearly interpolating between major gridlines.

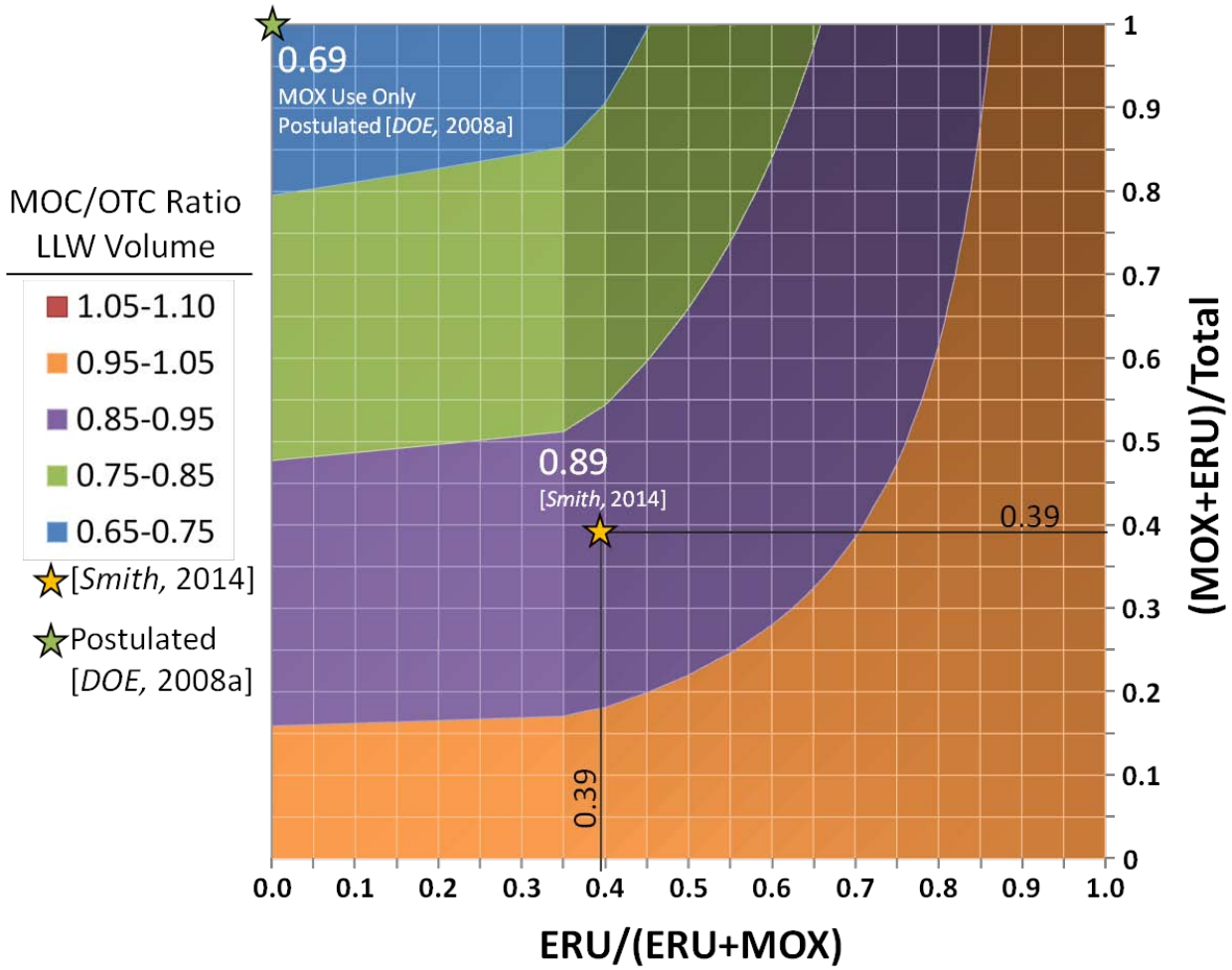


Figure 3.29 | Steady-State MOC/OTC LLW Volumes Ratios Mapped on Potential Fuel-Use Combinations Plan-View of Surface Plot

3.6. Integrated Ratios of Dynamic Studies and Annual Ratios of Steady-State Studies

Worker radiological impacts are estimated as latent cancer fatalities (LCF) in DOE (2008a) were evaluated for both an integrated and two annual ratios that represented the start-year and end-year for the dynamic scenario. The start-year and end-year represented two NFCs, one with a reactor fleet only using ENU (OTC) and one reactor fleet only using MOX (MOC). The differences and ratios are shown in Table 3.12.

From the approach in DOE (2008a) (study representing the transition of the OTC to a possible MOC using only MOX) the MOC/OTC integrated ratio and annual ratio at the end-states indicate a higher net impact with respect to worker radiological impacts when transitioning to the MOC over a 50-year modeling time frame. The magnitude of the integrated ratio of worker radiological impacts is higher than of when only evaluating end-state to end-state when ENU and MOX are fully implemented in the start-year and then end-year, respectively. This suggests that when net impacts are to be estimated for a scenario employing only one type of fuel, a steady-state comparative analysis may be a good surrogate for quickly assessing performance of the MOC to the OTC but may not include uncertainty of the magnitude of the relative performance.

Table 3.12 | Integrated Dynamic Ratio to Steady-State Ratio of Radiological Worker Impacts [*DOE,* 2008a]

Simulation Approach: Performance Metric	OTC	MOC	Difference (OTC-MOC)	Ratio (MOC/OTC)
Steady-State (End-State to End-State Ratio)				
Worker Rad. Impacts [LCF]	13	14	-1	1.08
Dynamic (Integrated Ratio)				
Worker Rad. Impacts [LCF]	455	575	-120	1.26

When considering a potential future MOC scenario transition that utilizes a combination of ERU, ENU, and MOX, there is not a clear trend like shown above in Table 3.12 that shows a consistent pattern in the integrated ratios and end-state ratios (shown in Table 3.13). This is the case for worker collective doses where a net increase is observed for the integrated ratio, but the end-state ratios describe a net decrease will occur for the dynamic MOC case evaluated here. The use of a steady-state end-state ratio will not be a good surrogate to understand the net impacts during the transition or what the changes in magnitude of the net impact.

Table 3.13 | Integrated Ratio to Steady-State End-State of Radiological Worker Impacts and Waste Shipments [*Smith,* 2014]

Simulation Approach : Performance Metric	OTC	MOC	Difference (OTC-MOC)	Ratio (MOC/OTC)
Steady-State End-State to End-State Ratio for 3 PWR fleet (from Table 3.9 and Table 3.11)				
Worker Rad. Impacts [person-mSv/yr]	9.775E+02	9.645E+02	1.297E+01	0.987
Dynamic End-State to End-State Ratio for 142 LWR fleet (94 PWR, 48 BWR)				
Worker Rad. Impacts [person-mSv/2063 year]	1.787E+05	1.775E+06	1.205E+03	0.993
Dynamic Integrated Ratio (2013-2063)				
Worker Rad. Impacts	7.248E+06	7.259E+06	-1.104E+04	1.002

It is posited that the combination of fuel used in a hypothetical advanced NFC could indicate whether a dynamic analysis was required or if a simpler, steady-state analysis would be sufficient to capture the results if one were to see a net benefit or increased impact between NFCs When an advanced NFC using only one type of fuel was compared to a baseline NFC also only using one type of fuel, an initial steady-state comparative analysis would be sufficient to understand if an improvement or decline in performance was expected; however, the magnitude of net impacts when integrating over the transition would not be captured and a dynamic analysis

should be used. Additionally, when a heterogeneous fuel-use portfolio was the end-state within the simulation, the net impacts could not be determined as easily by a steady-state comparative analysis and would require an in-depth dynamic analysis to understand the integrated impact ratio of the two NFCs under evaluation.

3.7. Conclusions

Within this Chapter 3, a comparative analysis of the worker collective doses and radioactive waste volumes of the MOC to that of the OTC was performed. The MOC as modeled in this study utilizes both Pu and RepU a single-time more within PWRs from previously irradiated UNF. Large reductions in waste generation were observed that is consistent with other systems studies comparing the MOC and the OTC generated wastes from front-end operations and UNF destined for geological repository waste. It is interesting to note that the reduction translates to a tremendous volume of waste avoided that lie on both sides of the spectrum of inherent radiological hazard: mining and milling waste that is relatively low hazard compared to other fuel cycle wastes and the disposition pathway for these wastes are known, and UNF which the disposal path is clear but the timing of when such a facility will begin accepting fuel is unknown.

It is apparent that under this set of assumptions for transitioning and adopting the MOC does not present a deviation from estimated OTC worker collective doses in the near future. However, larger trends have been explored and it can be concluded that the most influential factor to estimating comparative impacts is the end-states of the scenarios, whether a steady-state approach or dynamic approach is used. The end-state of the scenario means, for example, that the

number of reactors supported on each fuel type (ERU, ENU, MOX) within the reactor fleet. As the options of fuel types will likely become diversified and reactor fleets will be a combination of technologies, this should be important to note when evaluating net worker radiological impacts.

Through the verification process of recent studies of worker collective doses, a map of potential combinations of fuel types used in steady-state scenarios and the corresponding resultant worker collective dose ratios was created (with a similar one for LLW volumes generated). With the small sample size of recent studies and the current MOC study, it was postulated that the end-state of the simulation that described the proportion of fuel used in the reactor fleet could indicate whether a dynamic analysis was required or if a simpler, steady-state analysis would be sufficient to capture the results if one were to see a net benefit or increased impact between NFCs However, when a heterogeneous fuel-use portfolio was the end-state within the simulation, the net impacts could not be determined as easily and would require an in-depth dynamic analysis to understand the integrated impact ratio of the two NFCs under evaluation.

CHAPTER 4

WORKER COLLECTIVE DOSES OF A FEDERAL WASTE MANAGEMENT SYSTEM FOR COMMERCIAL USED NUCLEAR FUEL

4.1. Introduction

In 2013, the U.S. Department of Energy (DOE) published a report[24] on their strategy to manage and eventually dispose of commercial used nuclear fuel (UNF) and high-level waste (HLW) [*DOE*, 2013], in response to the Blue Ribbon Commission's on America's Nuclear Future (BRC) series of reports in released in 2012 [*BRC*, 2012a, 2012b]. Recommendations by the BRC proposed that a "phased, adaptive, and consent-based approach" should be used during siting and implementation of the waste management system and supporting infrastructure. Within the 2013 *Strategy Report*, DOE outlined a program including a set of two types of facilities that would be part of the proposed federal waste management system (FWMS): (1) centralized storage facilities (i.e., independent spent fuel storage installations [ISFSI]) and (2) a geological repository. The major milestones of the FWMS were described at a high-level and were stated by DOE (2013) and used within this study as the following:

1. Start operations at the consolidated "Pilot" ISFSI by 2021 and accepting stranded fuel at non-operating reactors across the country

[24] [*DOE*, 2013] U.S. Department of Energy (2013). Strategy for the Management and Disposal of Used Nuclear Fuel and High-Level Radioactive Waste. (available at: http://energy.gov/downloads/strategy-management-and-disposal-used-nuclear-fuel-and-high-level-radioactive-waste).

2. Start operations at “Larger” ISFSI by 2025 with broad potential to implement waste handling and preparation for disposal
3. Start operations at the repository site by 2048 and emplacing fuel for final disposition

Since the start of planning how to deal with UNF, there have been many that ask the question of the costs of a comprehensive waste management and disposal system [*Wood et al.,* 1989; *TRW,* 1992; *EPRI,* 2009a; *GAO,* 2009; *Brinton & Kazimi,* 2013; *NEA,* 2013b]. This question has now been asked regarding the individual facilities of the FWMS to be required as part of the DOE’s *Strategy Report* including the use of, at minimum, two interim facilities and one repository. Most recently, a MIT study estimates the monetary costs of the three waste management facilities required to handle such a large inventory of UNF [*Brinton & Kazimi,* 2013].

However important, monetary costs are not the only “costs” that can be incurred by the many stakeholders involved with this process. Other ways to assess performance can be in the form of estimating the potential degradation of the local environment. Additionally, human health impacts should always be considered for any engineered facility and system. The recent MIT study has noted that other types of performance metrics should be evaluated and provided an example environmental protection metric of water usage as a possibility [*Brinton & Kazimi,* 2013]. Regarding human health and safety, worker dose is a readily available and widely used metric [*NEA,* 2000; *EPRI,* 2008a; *Weck,* 2013]. The BRC stated that individual doses to workers must be minimized during the implementation of the waste management system; due to the potential for higher dose rates are associated with handling UNF and HLW when compared to other nuclear material streams handled in other fuel cycle facilities [*BRC,* 2012b].

As part of the regulatory siting process, the facility must undergo rigorous technical review to ensure the safety and health of potential receptors that must bear the costs of the proposed facilities. After passing the licensure hurdle, it is assumed that during operations, annual individual worker doses will not exceed regulatory limits of 50 mSv per year (5 rem per year). Qualitatively at this stage, there is assurance that a waste management system will not operate if regulatory standards are not met; yet there remains a need to understand and estimate what the potential occupational radiological impacts could be due to operations associated with these new facilities.

The work presented in this chapter is intended to understand the potential occupational radiation impacts resultant from the decision to meet the three DOE milestones, as part of a systems approach, within a larger effort to quantify occupational radiation impacts of the entire U.S nuclear fuel cycle (NFC), as described later. The performance metric used for human health is the potential worker collective dose incurred from implementing the comprehensive FWMS. This is assuming that the ALARA (as low as reasonably achievable) operational philosophy and standards promulgated by DOE and/or NRC, are adhered to at each new facility as part of the waste management system concerning individual doses to workers. The benefit of using the collective dose for workers compared to the individual dose is that the collective dose is scalable to the amount of material and the extent of the operations part of the waste management system [*Weck,* 2013; *Krahn et al.,* 2014]. Thus, the worker collective dose is a function of the material flows of the waste management system and the number of facilities required for implementing all the functions that are part of meeting the three DOE waste management milestones.

Estimating the potential worker collective doses from a proposed FWMS depends on what a future comprehensive FWMS looks like and how it functions from a high-level perspective. In order to do so, first, a literature survey of past work focusing on estimated radiological impacts to workers from hypothetical FWMSs is presented. A qualitative model is then described and depicted graphically to understand how the components of the new FWMS could work together to meet the three stated milestones as described by the DOE (2013) *Strategy Report*. Next, literature is reviewed to aide in indentifying necessary parameters for modeling the new comprehensive FWMS for both the material flows of UNF and the worker collective dose metrics. Results are presented as scaled worker collective doses each year from implementing the proposed FWMS. Finally, concluding remarks are made of the potential performance of the FWMS compared to previous estimated worker collective doses of the once-through NFC (OTC) that includes up to at-reactor dry interim storage.

4.1.1. Objectives of this Chapter

There are two overall objectives of this chapter:

- Extend the OTC model to include geological disposal of UNF and estimated worker collective doses.

- Include worker collective doses associated with operations of consolidated, centralized storage facilities, as outlined by a recent report[25] published by the U.S. Department of Energy (DOE) in 2013 [*DOE,* 2013].

4.1.2. Background and Past Studies

When a potential need for alternative actions to store and manage UNF became apparent in the mid-1980s (with associated amendments to the national Nuclear Waste Policy Act of 1987), the U.S. Congress created the Monitored Retrievable Storage (MRS) Review Commission [*MRSRC,* 1989]. The MRS Review Commission was tasked to document the technical considerations of an interim, consolidated facility with input from multiple stakeholders, and then to finally report whether a MRS facility would be required. Many reports were published from this Congressional mandate, but in 1989 the first comprehensive study was published to estimate worker radiological impacts associated with a federal waste management system that included a monitored, consolidated location for storing UNF [*MRSRC,* 1989]. Concluding remarks by the MRS Review Commission were that the MRS facility could be constructed and implemented by 2000[26] and could be done so in a safe manner from a technical perspective. The expected differences in worker collective doses of a waste management system with and without an MRS could be small or result with benefits to worker safety (see Table 4.1). The MRS Review Commission also recognized that an MRS facility start-date that was not tied to a repository

[25] [*DOE,* 2013] U.S. Department of Energy (2013). Strategy for the Management and Disposal of Used Nuclear Fuel and High-Level Radioactive Waste. (available at: http://energy.gov/downloads/strategy-management-and-disposal-used-nuclear-fuel-and-high-level-radioactive-waste).

[26] The Nuclear Waste Policy Act (NWPA) of 1982 and the Amendment Acts of 1987 included an opening date of 1998 for the Yucca Mountain site for the only repository option. The MSR Review Commission (1989) study rounds up to the year 2000 for the opening date of the repository.

start-date could begin accepting waste sooner rather than later and could potentially reduce the overall worker collective doses compared to the base case without an MRS. The summary of worker collective doses is provided for three scenarios that vary by the year the repository begins to accept waste; worker collective doses reported are integrated over the 50 year period from year 1995 to 2045. The detailed data and methods underpinning these values in Table 4.1 are not publicly available. From the little information that is available, it is interesting to note that including a MRS reduces the FWMS total worker collective doses ranging from 2% to 42% (2023 repository start date to accept waste).

Table 4.1 | Worker Collective Doses of UNF Management Strategies [*MRSRC*, 1989]

Case	Worker Collective Doses [person-mSv] (Ratio to No MRS Case)[b]		
	Repository Start Date		
	2003	2013	2023
No MRS	166,000 (1.0)	261,000 (1.0)	315,000 (1.0)
Linked MRS[a]	166,000 (1.0)	223,000 (0.85)	310,000 (0.98)
Unlinked MRS	166,000 (1.0)	170,000 (0.65)	182,000 (0.58)

Notes: [a] The "linked MRS" represents a federal waste management system that includes a MRS that is tied to the performance of the geological repository and the MRS must begin accepting waste three years prior to the repository acceptance date. [b] The ratio of worker collective dose is to the reference case of the No MRS case shown in the first row.

There have been several follow-on studies that analyze variant scenarios that were posed in the MRS Review Commission (1989) study:

- UNF Acceptance Scenarios for Shutdown Reactors [*Wood et al.,* 1989]
- MRS Conceptual Design Studies [*TRW,* 1992]
- Analysis of Radiation Doses from Operation of Postulated UNF Transportation Systems Analysis of a System with a MRS Facility [*Smith et al.,* 1992]
- Multi-Purpose Canister System Evaluation A Systems Engineering Approach [*DOE,* 1994]

Wood et al., (1989) and TRW (1992) do not address potential radiological worker impacts but recognize many logistical issues that would arise when implementing a FWMS; these reports are discussed in more detail in the following section to inform the quantitative modeling phase. The publication of the MRS Review Commission (1989) report led to two other comprehensive systems-level studies for a FWMS evaluating worker collective doses. The report by Smith et al., (1992) is the most in depth analysis of routine (non-accident) worker collective doses and includes the base case of a FWMS with a repository and no MRS and several alternatives with a MRS that vary the parameters listed below:

- Location of MRS (east or west of the Mississippi River – i.e., Oak Ridge National Laboratory and Idaho National Laboratory)
- Transportation modes (legal-weight truck only, non-dedicated train, dedicated train)
- Fuel form (intact fuel or consolidated fuel[27])

Smith et al., (1992) reports that the worker collective doses for any of the alternative FWMS scenarios that includes a MRS is higher than a FWMS without an MRS (1,340 to 2,370 person-mSv/yr, [15-27% increase] depending on the scenario as shown in Table 4.2). Factors that increase the worker collective doses for a FWMS with a MRS are associated with increased handling of casks. Additionally, Smith et al. (1992) reports that total worker collective doses associated with the eastern MRS alternatives are lower than western MRS alternatives, due to less in-transit miles are required because most reactors are located in the eastern U.S. For the

[27] Intact fuel is UNF that remains in fuel assembly hardware. Consolidated UNF refers to used nuclear fuel assemblies mechanically separated: used fuel pellets are shipped within one container and Zircaloy structural material is compacted and shipped within a different container than the fuel pellets. The U.S. conducted consolidation demonstrations in the 1980s that achieved volume reductions of 50%, but in-pool cutting of fuel assemblies proved to be complicated and all companies involved in the demonstrations chose not to pursue full scale UNF consolidation [*EPRI,* 2010d]. Due to this, consolidated UNF is not considered in this work.

alternatives including shipped consolidated fuel is lower compared to intact UNF (~5-8% reduction).

The most applicable scenarios listed within Table 4.2 are the "intact fuel" cases (see footnote 27), as the commercial industry has not decided to implement fuel compaction. Total worker collective doses for a FWMS with an MRS were ~31 to 33% higher than no-MRS case; this is unlike the MRS Review Commission report (1989) estimates of lower worker collective doses (see Table 4.1).

Table 4.2 | Worker Collective Doses of UNF Management Strategies [*Smith et al.*, 1992]

FWMS + Repository + Case listed below	Annual Worker Collective Doses [person-mSv/yr] (Ratio to No MRS Case)[a]		
	Receptor Population		
	Facility Workers	Transport Workers	Facility + Transport Workers
No MRS	8,758 (1.0)	2,229 (1.0)	10,987 (1.0)
Eastern MRS Consolidated Fuel	10,100 – 10,393 (1.15-1.19)	1,470 (0.66)	11,570 – 11,863 (1.05-1.08)
Western MRS Consolidated Fuel	10,256 – 10,577 (1.17-1.21)	2,691 (1.21)	12,947 – 13,248 (1.18-1.21)
Eastern MRS Intact Fuel	10,868 (1.22)	1,490 (0.69)	12,358 (1.12)
Western MRS Intact Fuel	11,125 (1.27)	2,704 (1.21)	13,829 (1.26)

Notes: [a] The ratio of annual worker collective dose is to the reference case of no MRS shown in the first row.

In 1994, DOE published a report of impacts associated with multi-purpose casks (MPC) designed for freight-rail and legal-weight truck shipments [*DOE,* 1994]. The alternatives discussed were placed in the context of a MRS facility that is included and excluded from the proposed FWMS. The FWMS scenario evaluated in detail included the concept of dry UNF transfer in hot cells into vertical cask storage (shown below, Table 4.3). The worker collective doses are integrated over the specified 41 year period from year 2000 to 2041.

Table 4.3 | Worker Collective Doses of UNF Management Strategies [*DOE,* 1994]

Facilities	Worker Collective Doses [person-mSv]	
	FWMS Scenario[a]	
	No MRS	With MRS
Reactor Sites	282,700	256,600
Monitored Retrievable Storage	--	107,000
Cask Maintenance Facility[a]	600	600
Geological Repository	225,300	205,600
FWMS Total	508,600 (1.0)	569,800 (1.12)

Notes: [a] The worker collective doses are summed over the "life of the system" from operating years of 2000 to 2041. [b] The scenario that includes a MRS assumes that the cask maintenance facility is assumed to be co-located to with the MRS but is a separate building away from the main MRS area. A cask maintenance facility with no MRS is located at one of the two repositories assumed to be operating.

Consistent with the conclusions of Smith et al. (1992), DOE (1994) estimated an increase in total worker collective dose will be associated with a FWMS that includes an MRS. The percent increase of 12% for DOE (1994) is lower than that of Smith et al. (1992) with a range of 31-33% increase depending on the location of the MRS.

The three studies described above present a wide array of what could be expected from a FWMS that includes an MRS. Estimates of worker collective doses comparative ratios lay on both sides of the 1.0 marker which demonstrates the need for this evaluation using updated performance metrics and a FWMS that matches current objectives and milestones.

4.1.3. In-Transit Worker Collective Doses and Other Transportation Considerations

Transportation risk assessment for a FWMS transporting UNF from reactor sites to the FWMS facilities has been a topic of a number of studies [*Smith et al.,* 1992; *DOE,* 2002b; 2008d, 2008g;

EPRI, 2006b]. Smith et al. (1992) was previously discussed and as shown in Table 4.2, the transportation associated worker collective dose is around 12 to 21% of the total worker collective doses for any of the alternative FWMS scenarios that includes a MRS. Worker collective doses for the U.S. OTC at the time of when the study was conducted are not known, thus, it is difficult to assess whether transport of UNF was mainly responsible. However, deduction can be used to estimate that transportation-associated worker collective doses were then, and still are, a small portion of overall OTC worker collective doses from conclusions found in Krahn et al., (2014); it was found that the majority of the OTC worker collective doses are from the front-end and the reactor operations. This deduction can be verified by a recent NEA (2000) study reporting that transportation contributes around 0.5 to 0.7% of the total OTC worker collective dose.

In 1996, the U.S. Nuclear Waste Technical Review Board (NWTRB) made a public statement that in response to the multiple published reports on transportation risk [*NRC,* 1987; *MRSRC,* 1989; *Brentlinger et al.,* 1989; *Smith et al.,* 1992]:

> "Numerous analyses have been performed in recent years concerning transportation risks associated with shipping spent fuel. Although any analysis of transportation radiological risks is extremely sensitive to the assumptions made (e.g., routing, the amount of material shipped by rail versus by truck, the number of people at stops along the route), the results of these analyses all show very low levels of risk under both normal and accident conditions. The safety record has been very good and corroborates the low risks estimated analytically. In fact, during the decades that spent fuel has been shipped, *no* accident has caused a radioactive release." [*NWTRB,* 1996]

The same message has been repeated in a National Academies (2006b) report and that "no fundamental technical barriers to the safe transport of spent fuel and high-level radioactive waste in the United States" could be identified. During the four decades of transporting over 3,000

shipments in the U.S., there has never been a release of radioactive contents and zero injuries due to radioactive contents over the 1.7 million miles transported [*McCullum,* 2012].

To press this argument further that routine worker collective doses from transporting UNF is a small portion of the entire OTC worker collective doses expected, a recent publication by Marincel et al. (2007) explains that previous worker collective doses related to transportation were grossly overestimated, due to assumptions of dose rates for nearby casks. The source terms for standard transportation risk assessments (including the EIS for Yucca Mountain) are assumed to be equal to the regulatory limit of 0.1 mSv/hour (10 mrem/hr) at a distance of 2 meters away (non-air attenuated). Marincel et al. (2007) argues that when actual measurements are taken and updated dosimetry models are used, the actual dose rates are $1/10^{th}$ to $1/1{,}000^{th}$ of the estimated dose rates when using 0.1 mSv/hr as the assumed dose rate.

It has been deduced that worker collective doses incurred by the transportation process are negligible to the FWMS and thus, also to the overall OTC. Calculating transportation associated worker collective doses also necessitates that a future location of all facilities of the FWMS are known. There is a lack of a technical basis for doing so, thus worker collective doses incurred in-transit are not included as part of the quantitative model of this initial study.

4.2. Methods

Environmental, health and safety (EH&S) risks were discussed as important performance indicators in previous chapters and two NFCs have been evaluated within a dynamic modeling

scenario and then contrasted using estimated worker collective doses and radioactive waste volumes.

For both the OTC and the modified open NFC (MOC), the order for developing the quantitative EH&S impact models has been to first develop a qualitative systems model of the NFC of interest. Second, a quantitative model is established for the NFC within a simplistic energy demand scenario but includes complex interactions of how LWR fuel requirements are met to support U.S. electricity generation. The EH&S impacts are then linearly scaled from the material flow balances produced by the quantitative model. The work described in the subsequent sections follows these same methodological steps by first presenting the qualitative systems model and then rationale for parameter considerations of a UNF material flow analysis for back-end FWMS operations for a 50-year modeling scenario from years 2013 to 2063. Worker collective dose performance metric data sets are also described and the set of performance metric applicable for a FWMS is presented.

4.2.1. Modeling Platform and Calculational Tool

Consistent with Chapter 2 and Chapter 3, the comparative impact assessment presented in this Chapter 4 was performed as a deterministic evaluation within a spreadsheet modeling platform (using Microsoft Excel). The deterministic values for input values were found by taking the average of input value options that were collected through literature reviews. This discussion has been structured in the methods subsections such that the deterministic input value used is stated and then the available range of input values is provided in parentheses. A high-level discussion

of the uncertainty of input values is provided if multiple values were available from literature (often times, only one deterministic value was found in the literature).

It is important to note that the uncertainty associated with many of the required modeling parameters is much higher than previous chapters due to the nature of this modeling exercise of a set of posited facilities that has not operated, nor have any available documentation that provides evidence that a number of facilities will be included in the FWMS (such as the case with a proposed number of facilities with the Larger ISFSI). The input values choices are minimal as most values only have one reference that includes applicable parameters that describe how a FWMS could operate. These past studies contain much uncertainty to the ability to predict future operations of the FWMS as these plans are associated with a FWMS that failed to be implemented in the late 1980s.

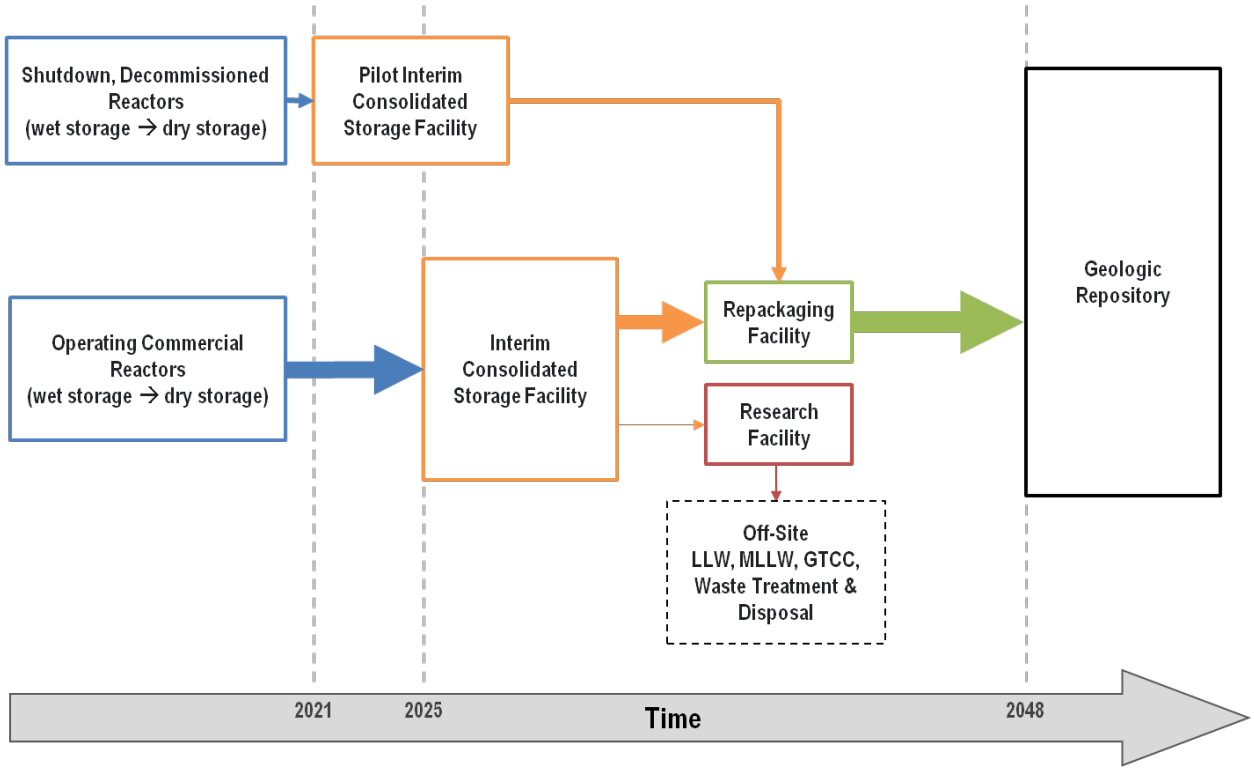
4.2.2. FWMS Qualitative Systems Model

To minimize future legal encumbrances on the DOE, it was stated by the BRC that the phased FWMS approach should begin operations first of the interim storage for the large inventory of UNF, with preference for centralized locations; soon thereafter, a repository would open for UNF disposal. Beyond that initial direction, the decisions regarding finer details of the phased approach was to be determined by DOE; the details are now explored as in a model the worker collective doses of a potential FWMS based on the DOE 2013 *Strategy Report*. The first step is formulating a conceptual, or qualitative, model of how, when, and where UNF will be handled at

the posited three back-end facilities (plus consideration for additional operations suggested by the BRC in conjunction with the three facilities).

As specified in the DOE (2013) *Strategy Report*, there are three types of facilities with distinct roles that will be part of the new FWMS (as shown in Figure 4.1):

- Pilot ISFSI: Enable turning over potentially valuable land at decommissioned reactors by accepting stranded fuel, whereby eliminating high overhead costs for ISFSI maintenance, surveillance, and high-security measures at these sites.
- Larger ISFSI, (herein "ISFSI"): Reduce monetary penalties and minimize litigious interactions between utilities and DOE by accepting UNF stored at reactors in dry storage.
- Geological Repository: Emplace UNF for permanent disposal



Notes: Blue boxes and arrows represents UNF initially moved from reactor sites, orange boxes represent ISFSIs and arrows represent UNF moving from ISFSIs, green boxes represent co-located activities to the ISFSI, and the green arrows represent. Widths of arrows shown indicate relative amounts of fuel

Figure 4.1 | Conceptual Systems Model of the Federal Waste Management System

The BRC explains that utilities were forced to deal with shortages of on-site UNF storage through the past three decades and that an ad hoc method of adding storage was used. This has resulted with over 1,700 dry storage casks stored as of 2012 that constitute more than 20 unique dry storage system designs within 32 states [*BRC*, 2012b; *EPRI*, 2013a]. Solely looking at the stranded UNF inventory at the nine shutdown reactors, DOE will have to deal with "18 different canister designs, eight storage overpack designs and eight different transport casks systems [*McGraw Hill*, 2013a]."

In 2005, DOE announced that the transportation-aging-disposal (TAD) canister design was the exclusive waste package design for Yucca Mountain [*EPRI*, 2008b; *McCullum*, 2008] and

worked towards a TAD system performance specification document that was initially released in 2002 and a revision in 2008 [*DOE,* 2008h]. DOE recognized that industry use of dual-purpose canisters (DPCs) was widespread and stated that a limited number of DPCs would be accepted at the Yucca Mountain site where head-end processes before emplacement would require repackaging UNF fuel assemblies into TAD canisters[28]. This would be performed in a hot cell facility (dry repackaging) and would dispose the DPCs as LLW [*EPRI,* 2008b; *McCullum,* 2008]. However, no hot cell of this magnitude has been constructed within the U.S. [*McCullum,* 2008] and within the international community, the "majority of spent fuel cask operations involve loading or unloading of spent fuel assemblies under water to move from one place to another for storage" [*IAEA,* 2007b]. Spent fuel pools are not technically challenging compared to hot cell facilities, but pools do require a high level of maintenance [*BRC,* 2012b]. Repackaging is assumed to be performed in a pool that is scaled to handle a much higher throughput than that currently on going at a single reactor site (which this assumption is adopted according the BRC (2012b) and McCullum (2008) reports).

From the standpoint of increasing efficiency at the repository, the BRC recommended that repackaging be part of the back-end operations of the consolidated ISFSI. If repackaging were to occur during the front-end of the repository operations before emplacement (as was the plan announced by DOE [*EPRI,* 2008b]), the BRC concluded that a foreseen bottleneck would be caused by the repackaging activities. One part of the rationale for repackaging activities at the back-end of the large ISFSI operations and before sending UNF to the repository is that the

[28] In response, EPRI (2008b, 2008c) analyzed the thermal loadings and repository performance of 100% of waste packages be DPCs emplaced within Yucca Mountain and a mix of 2100 DPC/5010 TAD canisters. The reported differences in repository performance with respect to thermal loading were expected to be minimal compared to that of 100% TAD emplacement.

ISFSI will have sufficient inventory to blend "hot" and "cold" (older and younger) UNF in order to create a stream of uniform thermal load of packages sent for disposal. This would simplify the heat requirements of the repository and potentially increase the rate of emplacement [*BRC,* 2012b]. Repackaging was and is expected to result with higher worker collective doses simply because an increased exposure time to handle UNF is required [*EPRI,* 2008b; *Weck,* 2013]; additional low-level waste (LLW) would be produced from using DPCs only once, and additional costs would be added to the total costs of the FWMS [*BRC,* 2012b; *McCullum,* 2012]. To avoid the unnecessary production of LLW from used DPCs, it will be assumed that the DPCs will be sent back to the owners (utilities) and then will be reused. The same is also true for DPC and TAD transport overpacks where they will be sent back to the original owner with the intent for reuse. It also must not be omitted that the Pilot ISFSI, as planned in the DOE (2013) *Strategy Report*, does not repackage UNF in DPCs and will require transport to the Larger ISFSI that has the capability of repackaging in order to be disposed at the repository.

There has also been discussion of a research facility that would examine UNF of various burnups, reactor conditions, initial U-235 enrichments, and ages of UNF [*BRC,* 2012b]. Post-irradiation examination would most likely occur in hot cells with remote-handling technologies. The location of this type of facility would not as likely be constrained as the repackaging facility, but would most likely benefit from being located at the ISFSI if repackaging activities are ongoing (that is, while casks are being opened and readied for the repository). The amount of UNF that will be used for research can be assumed to be small compared to other UNF mass flows to the various FWMS facilities. The types of waste resulting from these activities are assumed to be similar to the example hot cell research facility (Oak Ridge National Laboratory).

The two types are Greater-than-Class C (GTCC) low-level waste (LLW) from fuel assembly structural material and mixed LLW (MLLW); the volumes of GTCC and MLLW waste that will result from research will be small [*DOE,* 2008b]. GTTC and MLLW are sent off site for handling, treatment, and disposal. The resultant worker collective doses from waste-related operations for the research facility are assumed to be negligible, compared to the other incurred doses.

As planning discussions are occurring to prepare for the new FWMS [*NWTRB,* 2013], it is becoming more of a reality that repackaging of UNF will have to be performed. Input from the BRC (2012b) and research needs on long-term storage degradation mechanisms make the case that the two facilities should be located at the Larger ISFSI and will be incorporated into the qualitative and quantitative systems model.

4.2.3. FWMS Quantitative Systems Model

Worker collective doses will be estimated for the new waste management system using three high-level steps: (1) develop a hypothetical scenario used to calculate a material flow analysis based on the three DOE milestones and the objectives of each of the three facilities; (2) calculate the normalized worker collective dose metrics, which is the quotient of measured annual worker collective dose quantities and reported annual throughput or extent of handling of nuclear material; and (3) scale normalized metrics by the material flow values found in step 1 for the appropriate years within the simulation time frame, 2013 through 2063, which is consistent with previous Vanderbilt and EPRI models.

Using the three DOE milestones that describe objectives of each of the three facilities as guides to inform the UNF material flow analyses, the following high-level parameters will need to be finalized for modeling purposes (and are still under consideration):

1. Facilities' capacities to handle UNF based on DOE strategy objectives
2. Facilities' acceptance rates for UNF on an annual basis
3. Inventories of UNF at operating and shutdown facilities

Presently, these three parameters are still under consideration and a literature review is presented in the following sections as a discussion of the options for modeling the material flow between reactor sites, on-site ISFSIs, consolidated ISFSI facilities, and the repository.

4.2.3.1. Capacities of FWMS Facilities

In 2021, 7 years from the present, a pilot ISFSI is to begin accepting stranded fuel from shutdown, decommissioned reactors across the U.S. The term "pilot" indicates that only one facility will exist and the estimated capacity of this facility has been discussed by DOE at around 10,000 MT of SNF [*McGraw Hill*, 2013a]. EPRI has estimated that a 6-year schedule would be required for a generic consolidated ISFSI facility for siting, design, licensing and construction [*EPRI,* 2009a]. The assumption that the Pilot ISFSI could be licensed and constructed in a reasonable amount of time (~6 years) such that the Pilot ISFSI operations can begin in 2021 are adopted from the EPRI (2009a) report.

The option still exists to size the pilot capacity closer to the anticipated amount of stranded UNF at shutdown reactors by 2021. At the time of the BRC publication release in 2012, there was an estimated 2,812 MT of stranded fuel at nine decommissioned reactor sites (excluding DOE-owned inventories at Fort St. Vrain in Colorado, shown in Table 4.4) [*BRC,* 2012b]. In 2012, the amount of UNF at non-decommissioned shutdown reactors is 770 MT UNF [*McCullum,* 2012].

Since the BRC report release, market conditions and other maintenance issues have led utilities to shut down four reactors at three power plant sites: Crystal River (Florida), Kewaunee (Wisconsin), and San Onofre (California) [*McGraw Hill,* 2013a; *NRC,* 2013d]. The decommissioning plan (SAFSTOR option[29]) for Crystal River was submitted to the NRC for review in 2013; SAFSTOR decommissioning can extend to 40 to 60 years post reactor-shutdown and UNF will remain in the existing on-site fuel pool until a new ISFSI is constructed on the Crystal River site [*Duke Energy,* 2014]. Because the BRC defined "stranded" fuel as UNF that remains at completely decommissioned reactors, the amount of UNF stored on-site will be considered stranded within 50 years after decommissioning ends at the three sites [*BRC,* 2012b; *NRC,* 2014a]. In total, there is approximately 2,805 MT of UNF at the three planned shutdown sites [*McGraw Hill,* 2013a]. For purposes of informing the quantitative model and without further information, the total of *potentially* stranded fuel is around 6,387 MT of UNF that will likely need to be considered as the capacity of the Pilot ISFSI beginning in 2021 (which estimate

[29] Three decommissioning options are allowed by NRC: (1) DECON: Immediate decontamination and dismantlement, (2) Safe storage (SAFSTOR), also called delayed decontamination: Placing the facility into a safe storage configuration, requiring limited staffing to monitor plant conditions, until the eventual dismantling and decontamination activities occur, usually in 40 to 60 years, (3) Entombment (ENTOMB): Radioactive contaminants are permanently encased on site in structurally sound material, such as concrete, and appropriately maintained and monitored until the radioactivity decays to a level permitting restricted release of the property. To date, no NRC-licensed facilities have requested the ENTOMB option. [*Duke Energy,* 2014; *NRC,* 2014]

inventories were adopted from BRC (2012a), McCullum (2012), and McGraw Hill (2013a) and used here).

Table 4.4 | UNF at Shutdown Reactors (Decommissioned and in SAFSTOR)

Shutdown Reactor (D = Decom. Complete; I = Decom Incomplete)	Reactor Type	Location State	Shutdown Date	Storage Type	UNF Onsite [MT UNF]	Number of Dry Storage Casks[a, b]
Big Rock Point (D)	BWR	MI	1997	Dry	58	7
Haddam Neck (D)	PWR	CT	1996	Dry	412	40
Maine Yankee (D)	PWR	ME	1996	Dry	542	60
Rancho Seco (D)	PWR	CA	1989	Dry	228	21
Trojan (D)	PWR	OR	1992	Dry	359	34
Yankee Rowe (D)	PWR	MA	1991	Dry	127	15
Humboldt Bay (D)	BWR	CA	1976	Dry	29	5
Zion 1 & 2[b] (D)	PWR	IL	1996-1997	Pool	1019	61
LaCrosse [b] (D)	BWR	WI	1987	Pool	38	5
Subtotal					*2812*	*248*
Indian Point 1 (I)	PWR	NY	1974	Dry	33	3
Dresden 1 (I)	BWR	IL	1978	Dry	69	11
Millstone 1[b] (I)	BWR	CT	1988	Pool	522	47
San Onofre 1 (I)	PWR	CA	1992	Dry	146	13
Crystal River 3[b,c] (I)	PWR	FL	2013	Dry and Pool	2805	215
San Onofre 2 & 3[b,c] (I)	PWR	CA	2013			
Kewaunee[b,c] (I)	PWR	WI	2013			
Subtotal					*3575*	*289*
TOTAL					**6387**	**537**

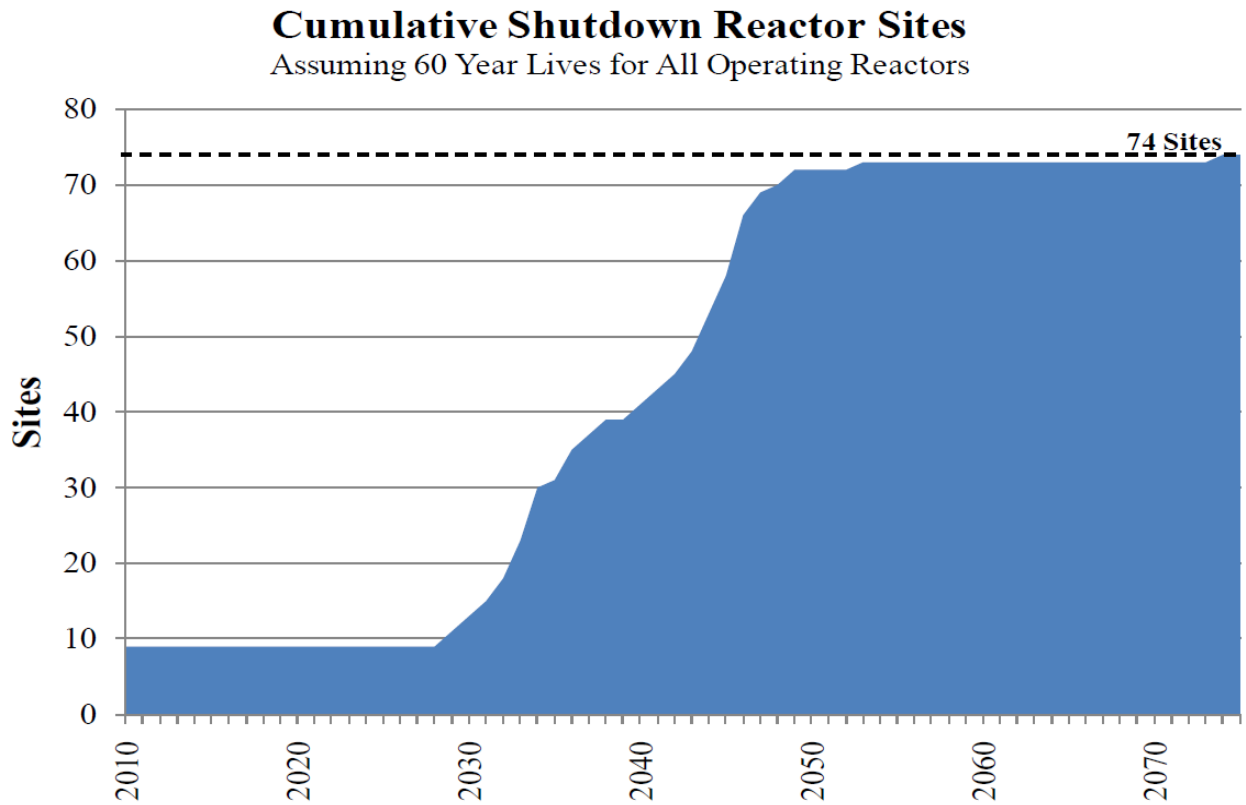
Notes: [a] Number of DSCs required at decommissioned reactors once UNF is moved from pool to dry storage are estimated by BRC (2012b). [b] At non-decommissioned reactors, UNF mass capacity of dry storage casks is set as 11.25 MT UNF/DSC when capacity was lower in the past than current day where the current average is 13.05 MT UNF per DPC (see Table 4.5). [c] The amount of UNF was not reported for individual reactors shut down in 2013, but was reported as a sum in McGraw Hill (2013a). It was also assumed that half of UNF was stored in pool storage and the other half in dry storage. **Sources:** [*BRC,* 2012b; *McCullum,* 2012; *McGraw Hill,* 2013a; *NRC,* 2014a].

The amount of potentially stranded fuel by the time the ISFSI opens in 2021 is around two-thirds of what the stated capacity will be (10,000 MT UNF); which presents an issue of inefficiency if no other reactors shut down between now and 2021. The trend in expected closures of reactors, and thus increases in the potentially stranded fuel is shown in Figure 4.2 [*BRC,* 2012b] and no reactors are expected to shut down by 2021. Wood et al., (1989) had already reached the same

conclusion by looking at two scenarios of transferring UNF at reactor sites to the MRS[30] by stating that taking shutdown reactor fuel first does not fully utilize the UNF acceptance capacity and at-reactor storage is higher for this option than an UNF-age hierarchal system. It should be also noted that experts on rail transport of hazardous materials have recommended that the U.S. FWMS ship oldest fuel first to reduce transport risks [*Halstead,* 2010]. This study assumes that the remaining one-third of the Pilot ISFSI is filled with UNF from operating reactor sites in order to fully utilize the available Pilot ISFSI storage capacity. Because stranded fuel is given higher priority when shipping to the Pilot ISFSI, that same principle is carried forward and it is assumed that the fuel from the Pilot ISFSI will be the first to be sent to the repackaging facility (collocated at the Larger ISFSI) and then sent to the repository for disposal. The rates of shipments are described later on in Section 4.2.3.4.

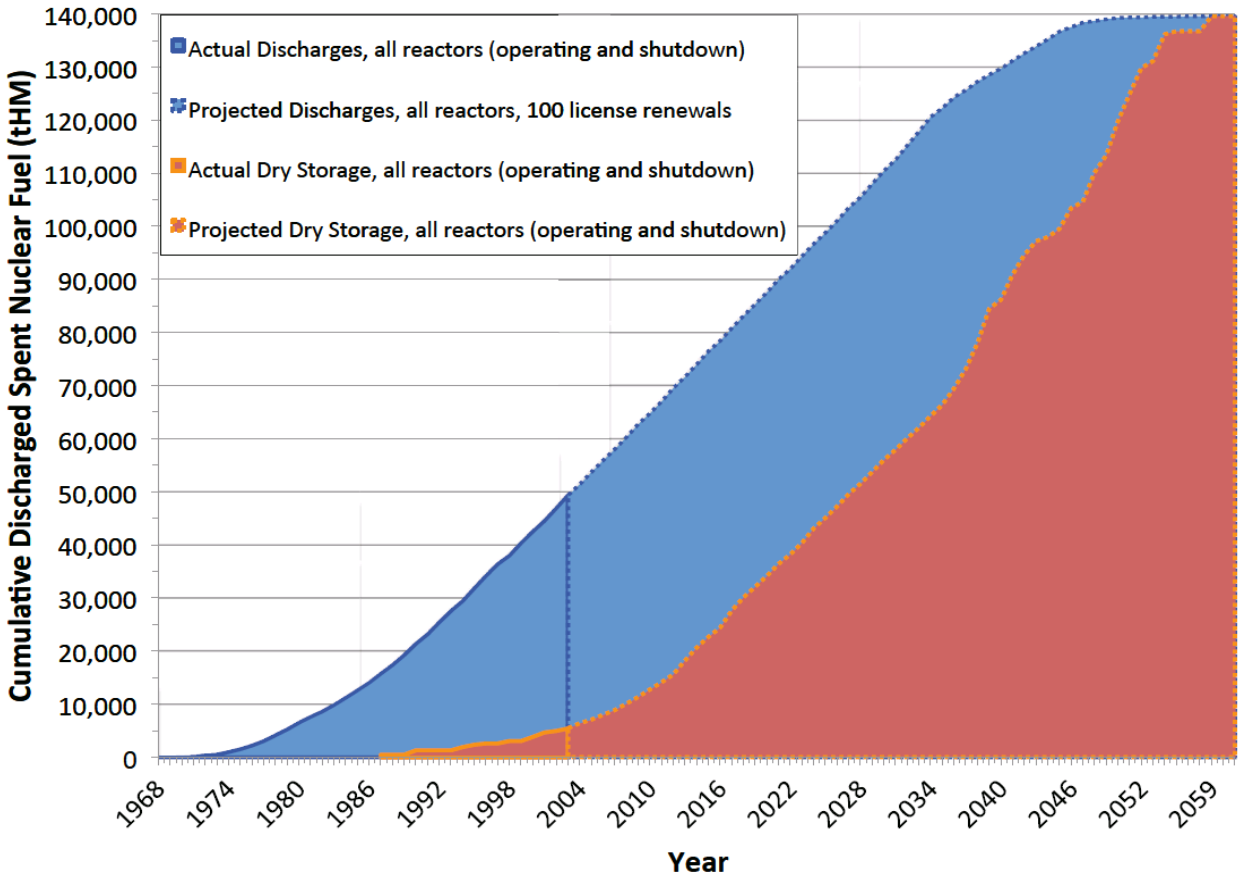
The cumulative number of reactors that will be expected to reach the 60-year licensed operational lifetime by 2030 and through 2050 are shown Figure 4.2 . It is estimated that a sharp increase in the number of reactors reaching the end of the operating life; this is during the midst of the operations of the Larger ISFSI. The cumulative number of dry storage systems required from discharged fuel and shutdown reactors will reach nearly 140,000 MT UNF(Figure 4.3) equating to ~10,000 DSCs loaded [*BRC,* 2012b; *Peterson & Wagner,* 2014]

[30] The MRS term is used in older literature when referring to efforts in the late 1980s to mid 1990s to create a FWMS with a centralized storage facility. ISFSI is the same concept as proposed by DOE (2013) Strategy Report and ISFSI will be used herein as designated by “Pilot ISFSI” and “ISFSI” for the larger facility.



Source: [*BRC*, 2012b]

Figure 4.2 | Projected Growth in Cumulative Shutdown Reactor Sites



Notes: Assuming 60-year operating durations for all reactors **Source:** [*Peterson & Wagner*, 2014]

Figure 4.3 | Historical and Projected UNF Discharges to Dry and Wet Storage

A "Larger" ISFSI is reported as having a minimum capacity of 20,000 MT UNF and would begin accepting fuel in 2025, four years after the Pilot ISFSI begins operations and 13 years before the opening of the geological repository [*BRC*, 2012a, 2012b; *McGraw Hill*, 2013a]. An annual acceptance rate for the Larger ISFSI was proposed as 3,000 MT UNF per year. If the constant 3,000 MT UNF/yr rate is put into place, the 20,000 MT UNF capacity will be fully utilized in less than 7 years; filling the remaining capacity from the Pilot ISFSI extends this to around 8 years (5 years short of the repository start date). Even if the BRC suggestion is considered where a ramp-up acceptance rate, on an annual basis, would provide flexibility for the system of facilities, the Larger and Pilot ISFSI capacity would be filled before the repository is

ready. There is likely a link between the design minimum as 20,000 MT UNF as the starting point and the DOE statements that suggest expansion of the original design capacity.

The repository capacity options in this study are limited to Yucca Mountain and the Waste Isolation Pilot Plant (WIPP). A capacity of 70,000 MT of heavy metal[31] was set as the limit of Yucca Mountain [*BRC,* 2012a]. By year end of 2013, there was approximately 70,000 MT UNF alone [*Peterson & Wagner,* 2014]. The number of packages of UNF is dependent on the individual waste package design capacity. The stated capacity considering the multiple types of UNF, defense UNF, and HLW containers destined for Yucca Mountain totals 10,676 [*Rechard & Voegele,* 2014]. However, the number of TAD (with disposal overpacks) that can be emplaced within the 70,000 MT UNF stated capacity equals 8,235 TADs using the TAD holding capacity of 13.05 MT UNF; the capacity of Yucca Mountain modeled herein will be set as 8,235 to balance arithmetically the flows of the quantitative model.

The potential limit of UNF associated with WIPP is difficult to assess because waste disposed at WIPP differs from Yucca Mountain. WIPP accepts defense-related transuranic contaminated material (TRU waste) and uses volumes and number of standardized canisters depending on the handling method (contact-handled or remote-handled). Presently, the emplaced TRU waste at WIPP is nearly 88,000 m^3 (11,000 shipments) and has been approved to expand and add rooms to the existing deep geological repository [*DOE,* 2011e; *NMED*, 2013]. Anticipated volumes from DOE sites to be shipped to WIPP are around an additional 80,000 m^3 for decommissioning defense related sites [*DOE,* 2011e].

[31] 90% of the capacity at Yucca Mountain was intended for UNF, 3.3% was for defense UNF, and 6.7 was for HLW [*DOE,* 2002a]

An important aspect of WIPP that could be applied to the initially limited capacity of Yucca Mountain is that once WIPP began operations in 1999 to present, relatively no major incidents[32] have occurred and approval for expansion was accepted by the local community. The expansion of a repository past the 70,000 MT UNF capacity is a possibility once imbued confidence is established with the local community.

There is little to no literature available to determine the annual capacity of repackaging activities and the numbers of UNF fuel assemblies that can be examined at the proposed co-located repackaging and research facilities at the larger ISFSI. A research facility will contain a number of hot cells that houses post-irradiation examination (PIE) equipment; required PIE equipment depends on whether destructive or nondestructive examination is performed [*Morris*, 2006]. It is likely that destructive testing will be performed with intent of carrying out radiochemical analyses due to recent reports of research needs [*BRC*, 2012b; *Hansen et al.*, 2012]. It is assumed that a research facility will operate at a pace similar to facilities at Oak Ridge National Laboratory and the annual level of performed research with correspond to known worker collective doses at the Irradiated Fuel Examination Laboratory [*DOE*, 2008a, *ORNL*, 2010]. Annual repackaging capacities will be assumed to match the required rate for emplacement into the repository. The extent of repackaging will be discussed in Section 4.2.3.4.

[32] Although, just recently two unrelated incidents have occurred at WIPP, the first was a subsurface truck fire that hauls salt cuttings away from the mined rooms on February 5, 2014 [*DOE*, 2014a]. The second incident on February 14, 2014 was the detection of contaminated air within an exhaust vent of americium-241 and plutonium-239/240 [*DOE*, 2014b]. Emplacement of waste packs has been suspended until further notice [*DOE*, 2014a].

4.2.3.2. Containers for Holding UNF

There are several terms that are used in literature and industry to describe containers for dry-storing UNF[33] [*NAS*, 2005]:

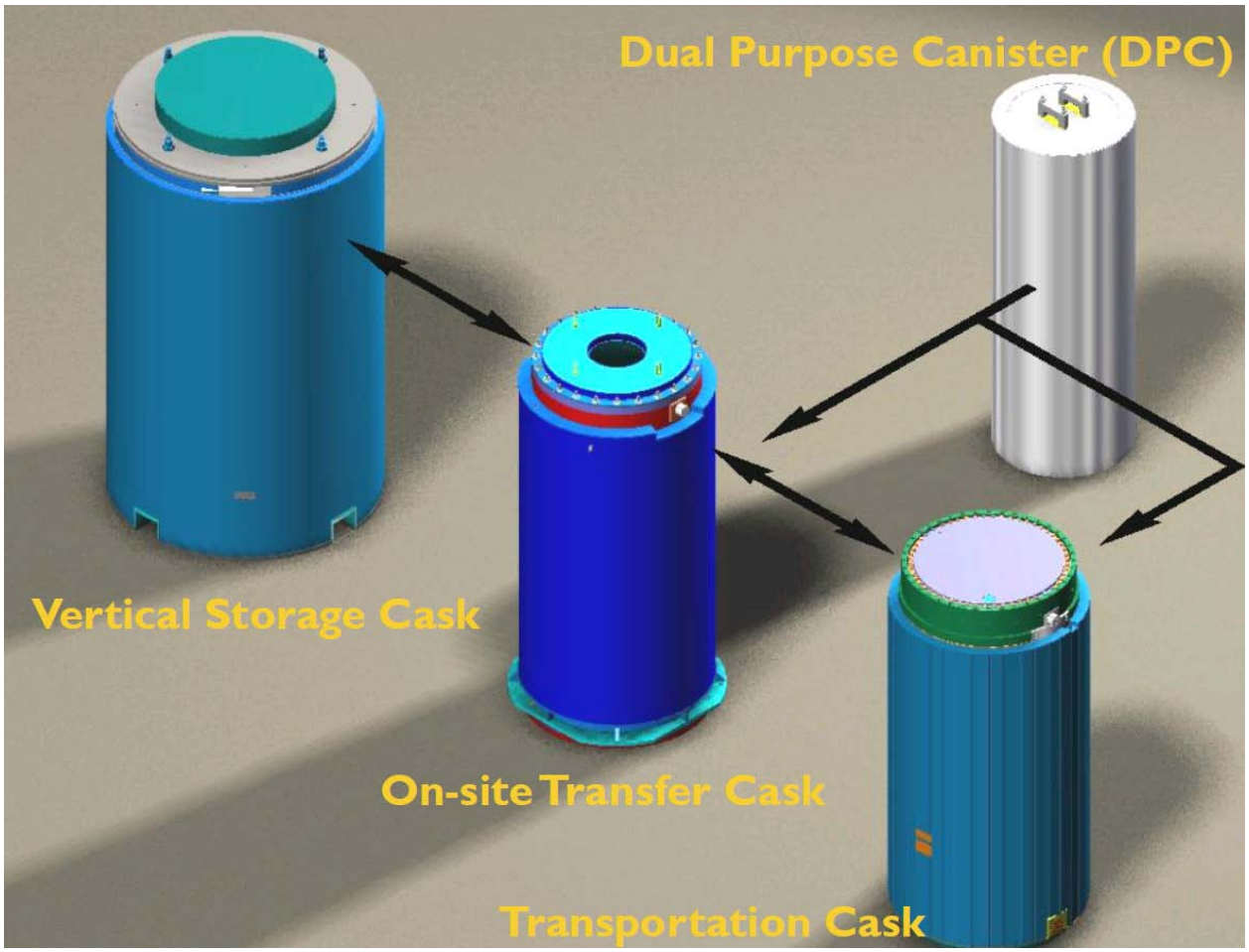
- Single-, dual-, and multi-purpose casks
- Bare-fuel and canister-based casks (also referred to as "thick-walled" and "thin-walled", respectively)

When a cask is referred to as single-, dual-, or multi-purpose, the intent and application of the cask's use is described. Single-purpose casks are for storage purposes only. Dual-purpose casks (DPC) are intended for storage and transportation uses. The MPC design mentioned previously within Section 4.1.2 within previous studies were designed for storage, transportation, and disposal with a repository. There is no currently accepted MPC design because the disposal waste package has not been finalized by DOE and no MPCs are in use [*NAS*, 2005].

The differentiation of bare-fuel casks and canister-based casks is the design of embedded and nesting shielding and the method that UNF is placed into these containers. Bare-fuel casks have an integrated internal basket permanently attached to a concrete shell (hence, the name "thick-walled); when UNF is loaded into a bare-fuel cask, the entire bare-fuel cask is placed into the spent fuel pool and UNF is loaded underwater. Canister-based casks are designed by using an internal stainless steel metal canister (here the name "thin-walled" casks misnomer applies here) that have designated "overpacks" that are made of concrete and the canister and the overpack are

[33] The term dry storage cask (DSC) is the general term for either a bare-fuel casks or a canister-based cask that is placed in a storage overpack.

separable; loading UNF for canister-based casks requires only the metal canister being submerged into the spent fuel pool and then loaded into the on-site transfer cask. In Figure 4.4 it is shown that the DPCs can be directly transferred from the on-site transfer casks into a transport or vertical storage casks [*McCullum*, 2012]. Another important distinction of bare-fuel casks and canister-based casks are that bare-fuel casks can only be stored on-site in a vertical geometry where canister-based casks can be placed in a vertical storage casks (VSC) or a horizontal storage module (HSM) [*NAS*, 2005].



Source: [*McCullum*, 2012]

Figure 4.4 | Dual-Purpose Canister and Overpack Options

As mentioned previously, DOE had included a particular waste package design in the Yucca Mountain license-application that was a multi-purpose canister-design called, "transportation-aging-disposal (TAD)." The DOE TAD design could hold up to 21 PWR fuel assemblies or 44 BWR fuel assemblies (see Table 4.5), but the exterior was made of stainless steel and not concrete [*Rechard & Voegele,* 2014].

While the DPCs capacity has in the past been similar to the DOE-proposed TAD capacities, the current and near future capacities have increased and now have limits of up to 32 PWR fuel assemblies or 68 BWR fuel assemblies [*BRC,* 2011, 2012b].

Table 4.5 | UNF Dry Storage Casks Holding Capacities

Dry Storage Cask Design	Fuel Assembly Capacity PWR (BWR)	MT UNF Capacity PWR (BWR)
MPC (truck)[a]	2 (5)	0.924 (0.930)
MPC (freight rail)	14 (36)	6.47 (6.78)
TAD	21 (44)	9.5 (7.9) Average = 8.7
DPC (current average)	32 (68)	14.4 (11.7) Average = 13.05
DPC (max currently licensed)[b]	37 (87)	15.7 (16.7)
Single-Purpose Bare-Fuel Casks[c]	-- (68)	-- (12.2)

Notes: [a] There were three MPC designs proposed in [*Smith et al.,* 1992] for general truck, single general rail car and a dedicated train. The truck and single rail car are shown here. [b] The DPC maximum values were shown in the draft 2011 version of the 2012 final report [BRC, 2011, 2012b] [c] Bare fuel casks are less common than DPCs and capacities for BWR fuel assemblies was found for the Transnuclear TN-68 model [*EPRI,* 2010d].

Transportation overpacks are considered as part of the TAD- and DPC-types of canister-systems but are only for legal-weight trucks [*NAS,* 2005; *DOE,* 2008h]. The capacity of the transport casks would be the same as the UNF-loaded canister which are listed in Table 4.5. Rail-approved overpacks for the U.S. are not available (as discussed in the following section), even though most

studies consider a rail option [*MRSRC,* 1989; *Smith et al.,* 1992; *DOE,* 1994, 2008a, 2008d, 2008e]. Rail cars that are considered in these studies have capacities that allow up to 11.35 MT UNF [*Smith et al.,* 1992] and is placed in a transport pack with a gross weight of 100 MT with a maximum of 131.5 MT [*Weck,* 2013]. Areva (2011c) states that UNF shipments by rail in Europe are limited to only 6 metric tons of UNF and 110 metric ton casks.

Designing a new waste package to match a representative geological medium is the newest approach that DOE has begun to evaluate [*Clayton et al.,* 2011], and the possibilities are listed in Table 4.6. The likelihood of design changes are high as was documented by Rechard and Voegele (2014) noting that there were seven designs for Yucca Mountain over the 26 years until the license application submission (1982 to 2008). Considering there have been no leads concerning sites willing to host the repository, the use of the TAD and the capacity of Yucca Mountain will be modeled herein as the repository waste package.

Table 4.6 | DOE Waste Packages for Representative Repository Designs

Dry Storage Cask Design	PWR Fuel Assembly Capacity per Waste Package	MT UNF Repository Capacity	Repository Capacity [# of waste packages]
Tuff (Yucca Mountain)	21	70,000	10,676[b]
Salt (WIPP)	10	140,000	32,154
Granite	10	140,000	32,154
Clay[a]	4	5,800	13,500
Deep Borehole[a]	4	696	400

Notes: [a] 0.435 MT UNF in a PWR fuel assembly was used to calculate the mass capacity of the repository knowing the number of waste packages to be disposed in at a single repository. [b] The stated capacity considering the multiple types of UNF, defense UNF, and HLW containers destined for Yucca Mountain totals 10,676 – however, the number of TAD (with disposal overpacks) that can be emplaced within the 70,000 MT UNF stated capacity equals 8,235 TADs; the capacity of Yucca Mountain modeled herein will be set as 8,235 to balance arithmetically the flows of the quantitative model. **Sources:** [*Clayton et al.,* 2011; *Rechard & Voegele,* 2014]

4.2.3.3. Transport Casks Considerations

Although in-transit worker collective doses are not considered in this work, worker collective doses differ when placing overpacks designed for rail and legal-weight trucks. Therefore, the mode of transportation that likely could be used is presented to inform what type of transport overpack is required.

Past considerations of transporting UNF throughout the FWMS have included rail, truck, and a mix of the two modes [*Smith et al.,* 1992; *DOE,* 2008a, 2008g] However, the rail option does not currently exist and the BRC (2012b) states that critical elements of the rail infrastructure related to safety measures would take a decade to develop. Based on technical considerations, it is reasonable to believe that rail car qualifications that would carry UNF could be completed within 10 years because defense-UNF is transported by rail. The Naval Nuclear Propulsion Program sends used naval cores from ships to the Naval Reactors Facility located at Idaho National Laboratory via rail [*BRC,* 2012b]. In addition, there are international guidelines based on international industry experience [*IAEA,* 2007b].

As with many other aspects of the FWMS, the technical barriers are only one part of the managing UNF. There are more barriers for extensive use of rail shipments to transport UNF because of recent legislation set forth by the U.S. Department of Homeland Security and Transportation. The BRC (2012b) and Halstead (2010) state that "rail routes to Yucca Mountain would require chain of custody and control procedures through 30 High Threat Urban Areas in 25 states, 20 other major cities, and require coordination among 18 railroads" [*Halstead,* 2010;

BRC, 2012b]. With considerations that rail seems more and more unlikely for moving extensive inventories of UNF, at least within the foreseeable future, the use of transport overpacks designed for legal-weight trucks to carry UNF will be assumed for the entire model.

4.2.3.4. Inventories and Transfer Rates

The repository waste inventory[34] of commercial spent nuclear fuel (UNF) is "distributed, diverse, and changing with time" [*Peterson & Wagner,* 2014]. By the end of 2013, the inventory of UNF was approximately 70,000 MT located at 75 sites in 33 states stored as varying amounts within wet fuel pools and dry interim storage at the reactor sites [*Peterson & Wagner,* 2014]. The 70,000 MT UNF consists of approximately 104,000 PWR fuel assemblies and 138,000 BWR fuel assemblies. There is an estimated 20 different dry storage system designs between the 75 sites [*BRC,* 2011] and a reported 238 dry storage canisters and bare-fuel casks that are licensed for storage-only purposes and will need to be repackaged at the reactor site to a DPC canister [*McCullum,* 2012]. It is assumed that utilities will not use single-purpose casks (SPCs) within the model employed here and 238 SPCs are all that need to be repackaged at the reactor sites as described in McCullum (2012) such that the use of SPCs were declining in use as DPCs become more prevalent. The inventory of UNF stored at shutdown sites was adopted from previous studies and considered in this analysis to be 6387 MT UNF (536 DSCs) and the breakdown of

[34] The work presented here excludes defense UNF and HLW. The defense waste that must be disposed in a repository is also vast and is spread around the country. Rechard and Voegele (2014) states that "at the end of 2010, ~2458 MTHM Defense UNF had been generated with (a) 2130 MTHM stored at Hanford in ~400 multi-canister overpacks; (b) 282 MTHM in mostly dry storage at INL, which includes 25 MTHM (of eventually 65 MTHM) of Naval UNF; (c) 29 MTHM of Defense-UNF in wet storage at Savannah River Site (SRS); (d) 15 MTHM of graphite blocks in vault storage at the dismantled Fort St. Vrain reactor in Colorad, and (e) 2 MTHM stored elsewhere. About 3175 canisters of HLW borosilicate glass had been poured with 275 canisters stored at West Valley, New York and the remainder stored at SRS. SRS was projected to produce another 6300 canisters, Hanford ~9700 canisters and INL between 3590 and 5090 canisters (pg 65)." [*Rechard & Voegele,* 2014]

these values were presented in Table 4.4 in Section 4.2.3.1. It was assumed that stranded fuel in wet fuel pools were prepared for transfer to the pilot ISFSI in 2015 by transferring fuel to DPCs at a rate of 50 DPCs packaged each year in order to calculate the associated worker collective doses.

The majority of UNF in dry storage are in DPCs [*McCullum,* 2012; *EPRI,* 2013a]. Of those DPCs, in the past the trend has been shifting away from pure metal casks (instead of stainless steel canisters or baskets within concrete) and the majority right now is being purchased as HSM (horizontal storage modules) due to economics [*NAS,* 2005]. The future additions of UNF to the 70,000 MT UNF inventory in 2013 will be taken from the results of the OTC scenario presented in Chapter 2 that assumes growth of nuclear energy production of 1%,

Waste acceptance and emplacement rates discussed in the license application for Yucca Mountain are stated as 3,000 MT UNF/yr [*DOE,* 2002; *Rechard & Voegele,* 2014]. However, the BRC (2012b) states that each element of the FWMS "must be flexible enough ... to 'ramp up' when necessary" which will be beneficial when facilities are first opened and initial learning can take place. Ramp up rates that were posited for the FWMS in the 1990s were associated with acceptance rates for ISFSIs [*TRW,* 1992; *DOE,* 1994] and emplacement rates for the repository [*DOE,* 1994] are shown in Table 4.7. There are two options for use of the ISFSI ramp up rates when modeling the FWMS worker collective doses; because there are two ISFSIs that will be included (Pilot and Larger ISFSI) the slower ramp up rate will be assumed to be used with the Pilot ISFSI from TRW (1992) data and the faster ramp up rate reported by DOE (1994) will be used for the Larger ISFSI. With lack of any other references available for repository waste

emplacement ramp up rates, the data and ramp-up modeling assumptions from DOE (1994) will be used for this analysis.

Table 4.7 | Proposed Ramp Up Rates FWMS Facilities

Year	MRS/ Pilot ISFSI Acceptance Rate [*TRW*, 1992] [MT UNF/yr] (DPC/yr)[a]	MRS/ Larger ISFSI Acceptance Rate [*DOE*, 1994] [MT UNF/yr] (DPC/yr)[a]	Repository Emplacement Rate [*DOE*, 1994] [MT UNF/yr] (TAD/yr)[b]
1	400 (31)	900 (69)	300 (35)
2	600 (46)	1400 (108)	600 (69)
3	900 (69)	2000 (154)	1200 (138)
4	3000 (230)	2600 (200)	2000 (230)
5	3000 (230)	3000 (230)	3000 (345)
...	**...**	**...**	**...**
n	3000 (230)	3000 (230)	3000 (345)

Notes: [a] The DPC holding capacity of 13.05 MT UNF is assumed. [b] The TAD holding capacity is assumed to be 8.7 MT UNF

4.2.3.5. FWMS Material Flow Analysis Methods Summary

The following is a list of modeling assumptions used in the FWMS material flow analysis that was described in detail above:

- Locations the Pilot ISFSI, Larger ISFSI and Repository are different to avoid concerns of ISFSIs becoming the *de facto* repository, The locations are not determined, nor assumed
- Capacities of FWMS facilities are fixed and limited:
    - Pilot ISFSI (10,000 MT UNF)
    - Larger ISFSI (20,000 MT UNF)
    - Repository (70,000 MTUNF)
- UNF Inventories:

- Stranded fuel considered: ~6,400 MT UNF
- SPCs at reactor sites that must be repackaged to DPCs: ~2,800 MT UNF (238 SPCs)
- Current UNF inventory across the country: ~70,000 MT UNF

- Ramp up rates are used for each ISFSI and repository, equilibrating at 3,000 MT UNF/yr
- Wet pool repackaging rates match emplacement rates each year
- DPCs are used for storage and transport (13.1 MT UNF), TADs are used for disposal (8.7 MT UNF)

4.2.4. Worker Collective Dose Performance Metrics

A literature review has been performed thus far of the identified tasks and activities at each facility that would likely result with measurable individual dose (this is keeping in mind that annual individual doses will be under regulatory limits[35]), and then tracking of worker collective dose could be estimated. The associated worker collective doses of the listed tasks were found through a literature review and data sets to build metrics are based on industry experience [*EPRI,* 2010b, 2012b; *Weck,* 2013]. Even though some of the activities and facilities have not been constructed (e.g., a centralized consolidated ISFSI) and measured doses have not been recorded, there are similar and analogous activities to the current day practices such that worker collective doses can be reasonably approximated. The following table (Table 4.8) presents the worker collective dose performance metrics of applicable foreseen high-level actions required to

[35] Regulatory dose rate limits for the non-accident scenarios of storing UNF at an ISFSI at the controlled area boundary cannot exceed 0.25 mSv/yr (25 mrem/yr). Regulatory dose rate limits for transporting UNF is set at 0.1 mSv/hour at a distance of 2 meters away (10 mrem/hour at 2 meters away) and 2 mSv/hour on contact (200 mrem/hour on contact). An individual worker dose limits are set at 50 mSv/yr (5 rem/yr) [*NRC,* 1998, 2012e; *McCullum,* 2012].

implement the FWMS facilities with additional explanation and calculations are provided in Appendix A. The worker collective dose performance metrics that were already presented from Table 2.5 within Section 2.2.3 are not repeated here.

It is repeated here that the worker collective dose metrics that were used to estimate FWMS worker collective doses were based on actual measured levels of worker collective doses and are based upon real events and industry experience with emphasis on collecting as recent data as possible that is within the public domain. The emphasis was also placed on using most recent measurements due to observed trends that doses are generally decreasing worldwide from many factors but mostly because of the ALARA (As-low-as-reasonably-achievable) operational philosophy. A bulleted list of the modeling assumptions is provided below:

- Even though some of the activities and facilities have not been constructed (e.g., a centralized consolidated ISFSI) and measured doses have not been recorded, there are similar and analogous activities to the current day practices such that worker collective doses can be reasonably approximated.
- Doses from the research facility are modeled after ORNL facilities that perform non-destructive and destructive testing of UNF assemblies
- Doses from handling small streams of waste produced from FWMS facilities are considered negligible
- Worker collective doses associated with transportation were not included because the locations of the facilities are unknown
- In-transit doses are not considered, however the dose metrics for loading onto a truck and rail differ – it is assumed that UNF is loaded onto trucks

Table 4.8 | Worker Collective Dose Performance Metrics (FWMS)

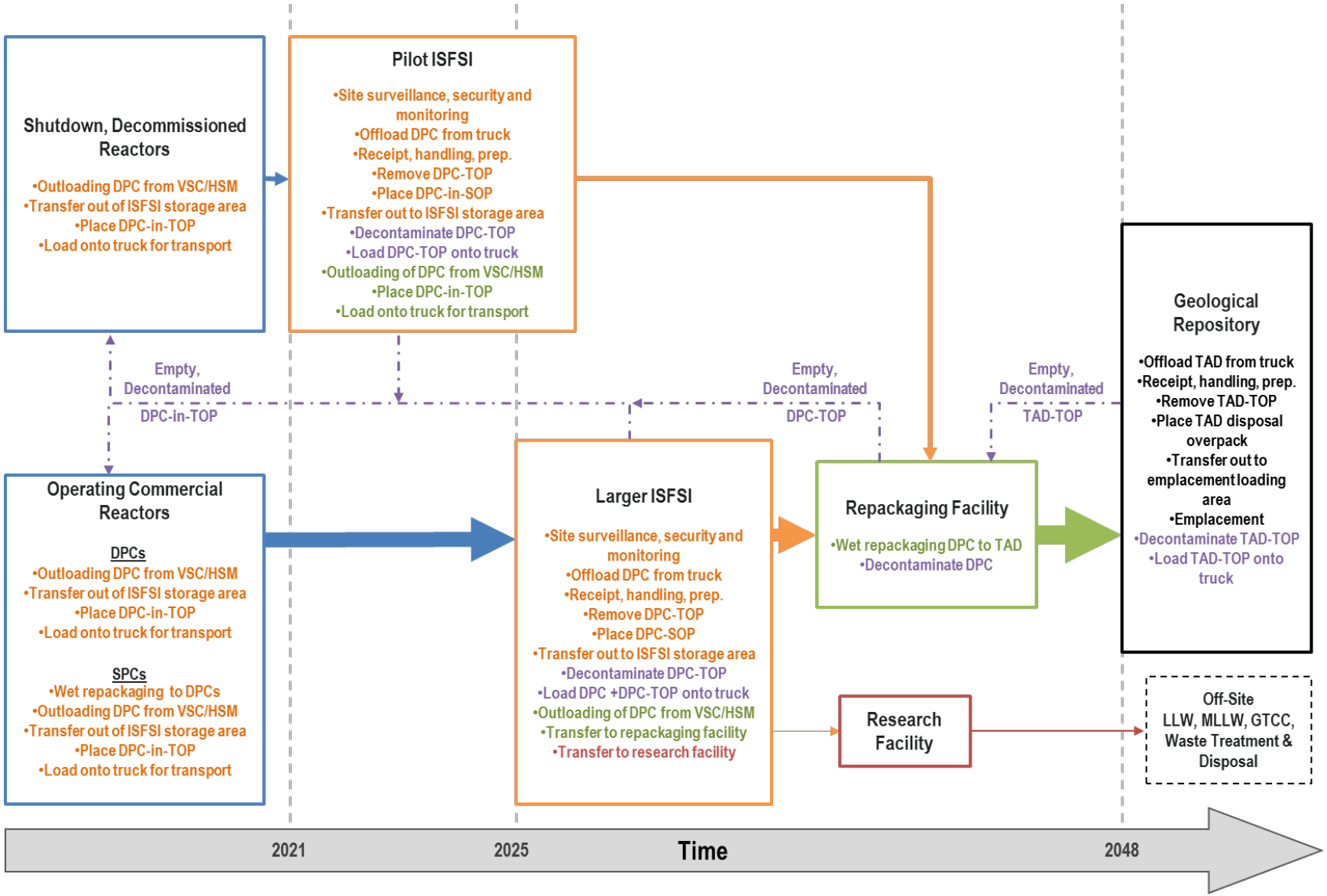
FWMS Facility	Value	Unit
Operation Step		
At Reactor		
Loading of UNF from spent fuel pool storage to DPC	4	person-mSv/DPC loaded of UNF in spent fuel pool storage
DPC Prep for Transport (outloading from storage, loading transport overpack and truck loading)	3.33	person-mSv/DPC
Wet Repackaging of SPC	2.5	person-mSv/SPC
SPC Decontamination and on-site storage	0.27	person-mSv/SPC decontaminated
Pilot ISFSI		
Front-end	7.53	person-mSv/DPC
Prep for transport to larger ISFSI for repackaging	3.33	person-mSv/DPC
DPC-TOP Decontamination and prep for sending back to reactors	0.27	person-mSv/DPC-TOP decontaminated
Annual site maintenance, security and surveillance	16.2	person-mSv/ISFSI site/yr
Larger ISFSI		
Front-end	7.53	person-mSv/DPC
DPC-TOP Decontamination and prep for sending back to reactors	0.27	person-mSv/DPC-TOP decontaminated
Annual site maintenance, security and surveillance	16.2	person-mSv/ISFSI site/yr
TAD Prep for Transport to Repository	2.6	person-mSv/TAD
Repackaging		
Wet Repackaging of DPC	2.5	person-mSv/DPC
DPC Decontamination and prep for sending back to reactors	0.27	person-mSv/DPC decontaminated
Research Facility		
Destructive testing and research	2.5	person-mSv/research facility operations/yr
Repository		
Front-end	7.53	person-mSv/TAD
Emplacement	9.22	person-mSv/TAD
TAD-TOP Decontamination and prep for sending back to Repackaging Facility	0.27	person-mSv/TAD-TOP decontaminated

Notes: SPC = single-purpose casks; TOP = transport overpack.

4.2.5. A More Detailed Qualitative Systems Model with Worker Collective Dose Considerations

The depth of the description required for understanding FWMS operations worker collective dose performance metrics is more extensive that what was described previously in Sections 4.2.2 and 4.2.3 describing the qualitative and quantitative model. The qualitative systems-model has been updated to reflect the increased level of detail of actions and interactions of the FWMS components and are shown in Figure 4.5.

The individual steps outlined in the worker collective dose performance metrics sections are now listed and color-coded to the FWMS component. The colors of the text shown within the FWMS facilities denote what stage the actions are intended for which FWMS component. An example is shown in the shutdown reactor FWMS facility text is shown in orange, representing that the actions that produce additional worker collective dose that was not previously accounted for are all done with the intent for moving UNF to the pilot ISFSI (shown in orange). In Figure 4.5, the interactions of decontaminating and reusing the overpacks for transport are shown between the FWMS facilities (shown as purple dashed arrows).



Notes: HSM = horizontal storage module; SOP = storage overpack; SPC = single-purpose cask; TOP = transport overpack; VSC = vertical storage cask

Figure 4.5 | Detailed Conceptual Systems Model of the Federal Waste Management System

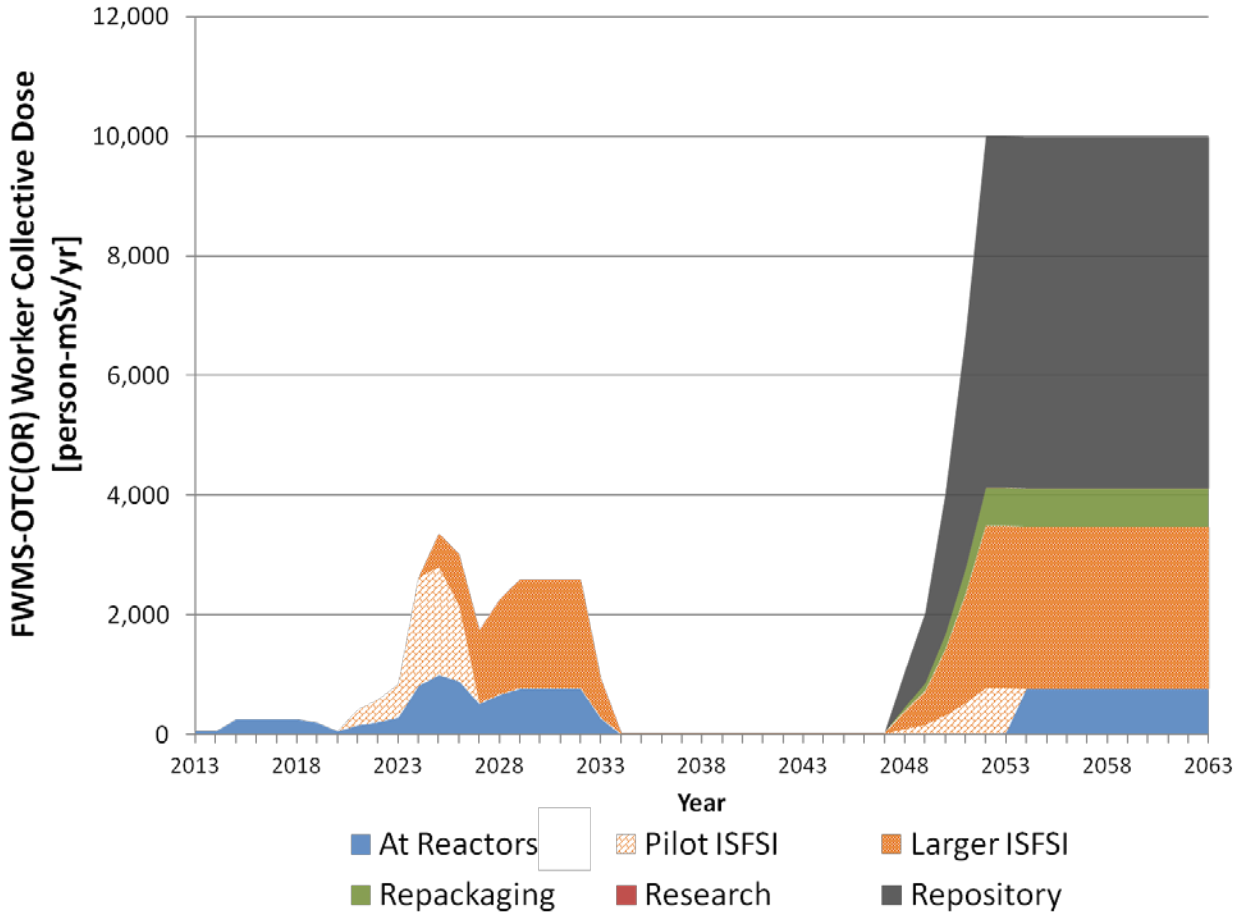
4.3. Results and Discussion

Worker collective doses from implementing the new FWMS were estimated based on the three DOE (2013) *Strategy Report* milestones to open a Pilot ISFSI by 2021, Larger ISFSI by 2025, and start emplacing UNF into the repository by 2048. The results were divided into two overall sections, the annual expected worker collective doses of the FWMS and the verification of results based on previous studies. Annual worker collective doses were presented that includes the additional actions that must be performed to implement the FWMS past the dry interim storage at operating reactors. Annual worker collective doses are also presented as all activities past the point of spent-fuel pool storage and then placed in context of the entire OTC that was discussed in Chapter 2. The previous studies have compared FWMS worker collective doses associated with and without a MRS plus a geological repository; thus an additional analysis calculating worker collective doses was performed that excludes the ISFSIs (pilot and larger).

4.3.1. FWMS Worker Collective Doses

The additional activities required to implement the new FWMS were considered as any handling of UNF besides what is currently being done at reactors sites, that is, past on-site dry interim storage at operating reactors (this is denoted in Figure 4.6 as “FWMS-OTC(OR)”). Worker collective doses associated with each FWMS component is shown for each year from 2013 to 2063 that was not already accounted for previously in Chapter 2 for at-reactor dry interim storage.

The worker collective doses are incurred from the FWMS each year are shown in Figure 4.6. Towards the beginning of the simulation, worker collective doses are resultant from transferring UNF assemblies from single-purpose casks (SPCs) into DPCs throughout 2013 to 2024 (~50 person-mSv/yr). The blue band in Figure 4.6 also includes worker collective doses from loading stranded UNF assemblies still in wet fuel pool storage into DPCs and then preparing for shipment to the Pilot ISFSI in years 2021 to 2024 (blue band, Figure 4.6). The Pilot ISFSI begins accepting stranded UNF in year 2021 (speckled-patterned orange band, Figure 4.6) and then other UNF from operating reactors with the remaining capacity. The Pilot ISFSI acceptance rates and ramps up to accepting 230 DPCs during 2024 and 2025 before slowing to 160 DPCs in 2026 due to reaching full capacity of 766 DPCs (~10,000 MT UNF).



Notes: Excludes worker collective doses from the OTC previously calculated from dry interim storage calculations at operating reactors

Figure 4.6 | FWMS-OTC(OR) Worker Collective Doses

The Larger ISFSI begins accepting UNF from operating reactors beginning in year 2025 and ramps up to 230 DPCs per year by 2029 and continues at this rate until 2032. In 2033, this is the last year that the larger ISFSI can accept UNF (160 DPCs) until the repository opens in 2048, without expansion. The worker collective doses resulting with this temporary activity is shown in Figure 4.6 (solid orange band). Worker collective doses are almost negligible from research operations, surveillance, monitoring, and security of the pilot and larger ISFSIs (~35 person-mSv/yr) during 2034 to 2047 during the time when maximum capacity has been reached for both facilities.

In 2048, the worker collective doses increases due to many activities occurring at once to prepare to move UNF throughout the FWMS. The first action is to send the stranded fuel that was originally placed at the Pilot ISFSI to the repackaging facility that is collocated at the Larger ISFSI which is shown as the small amount of worker collective dose from 2048 to 2053 (Figure 4.6, speckled-patterned orange band). Repackaging UNF from DPCs to TADs is limited to the repository emplacement ramp up rate and is the same as the rate of movement of originally stranded fuel to the repackaging facility and moves directly to the repository. Repackaging doses are shown in Figure 4.6 as the solid-green band.

Once the repackaging of originally-stranded fuel is performed, then the other UNF from operating reactors stored at the Pilot and Larger ISFSI are eligible for repackaging and sent to the repository (Figure 4.6). The need for the Pilot ISFSI is eliminated when all of the UNF is moved to the repackaging facility collocated at the Larger ISFSI by 2053. The worker collective doses from surveillance, monitoring, and security also drop to zero each year when there is no UNF at the Pilot ISFSI. With the current set of modeling parameters, it does not make sense to continue operations at the Pilot ISFSI because the repository steady-state emplacement rate is the same as the Larger ISFSI steady-state acceptance rate of 3,000 MT UNF/yr.

Once the larger ISFSI UNF inventory decreases (because the repository opened), the movement of UNF from reactors to the larger ISFSI begins (blue solid band, Figure 4.6). The annual worker collective dose incurred from repackaging is 575 person-mSv/yr at steady-state when repackaging 230 DPCs into 345 TADs (because TADs have smaller capacity than DPCs).

The majority of the annual worker collective dose is from receipt and emplacement of UNF at the repository (solid gray band, Figure 4.6). The ramp-up rate of emplacing TADs is reflected by the increasing worker collective doses at the repository. At year 2052, the rate of emplacement reaches steady-state and the worker collective doses each year is 5,870 person-mSv (~2,690 person-mSv/yr receipt, handling of TAD truck shipments; ~3,180 person-mSv/yr emplacement). The receipt and handling of waste packages at the repository is more than what is observed at the pilot and larger ISFSIs mainly due to the fact that more TADs are required to keep with the pace of emplacing 3,000 MT UNF at steady state. The DPCs received at the pilot and larger ISFSI are only limited to 230 DPCs/yr at steady state but the number of TADs is around 345 TAD/yr associated with the 3,000 MT UNF/yr emplacement rate.

When aggregating the worker collective doses associated with dry interim storage previously accounted for in Chapter 2, the result is shown in Figure 4.7, and now shows the complete picture of back-end UNF activities (denoted as "FWMS"). This is the scenario that most past studies described as a FWMS with a MRS, but in this case with two ISFSIs. The additional worker collective doses from operating reactors (blue solid band) has now increased and has essentially shifted the curves from Figure 4.6 around 2,000 person-mSv/yr. Towards the middle of the model in 2033, the amount of UNF discharged from the OTC each year increases due to the increased number of reactors to meet the 1% growth of nuclear energy demand.

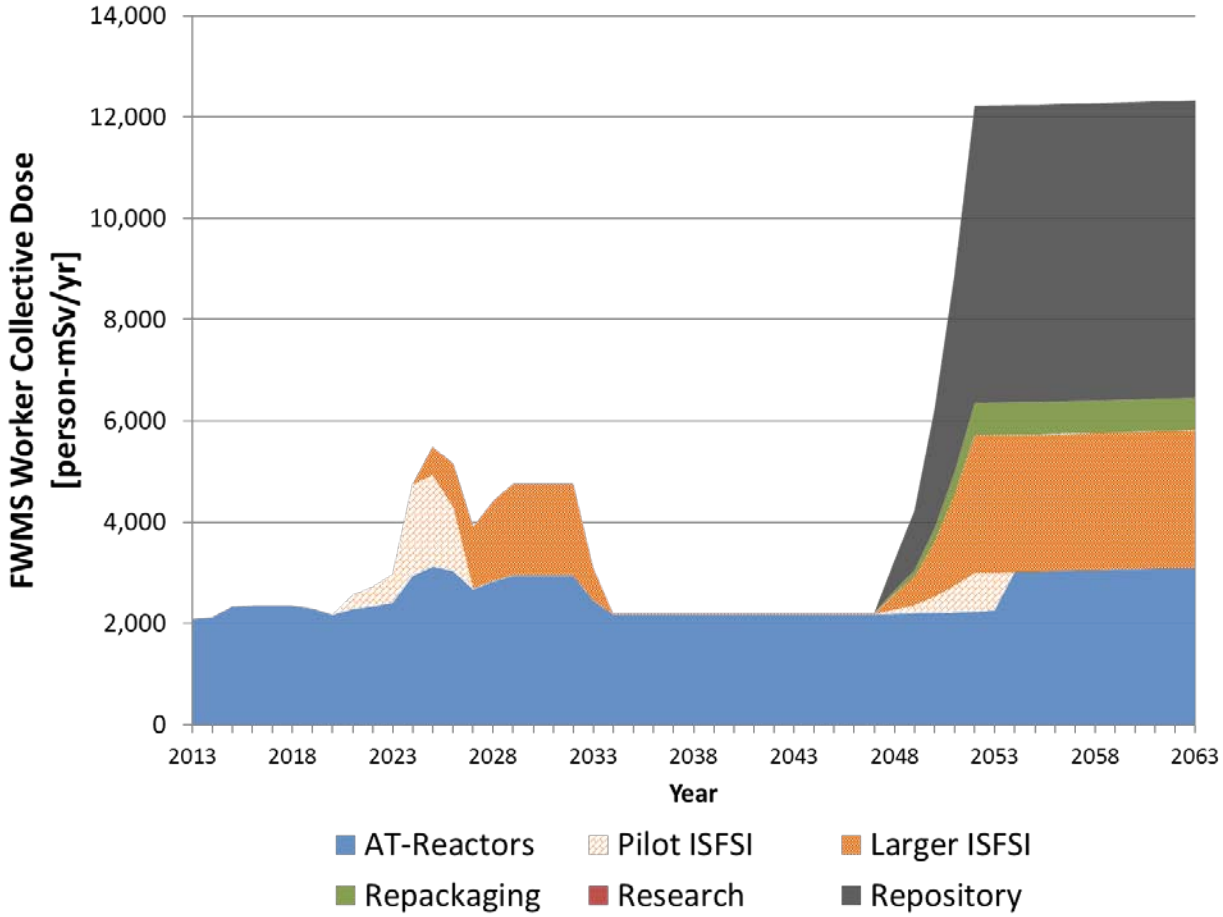


Figure 4.7 | FWMS Worker Collective Doses

FWMS+OTC activities are shown as a percentage of each type of FWMS component in Figure 4.8. The majority of the time, the worker collective doses from at-reactor dry interim storage activities dominates the total back-end total worker collective doses. Until the repository opens in 2048, the contribution of worker collective doses from each FWMS component changes substantially from year to year due to multiple modeling constraints.

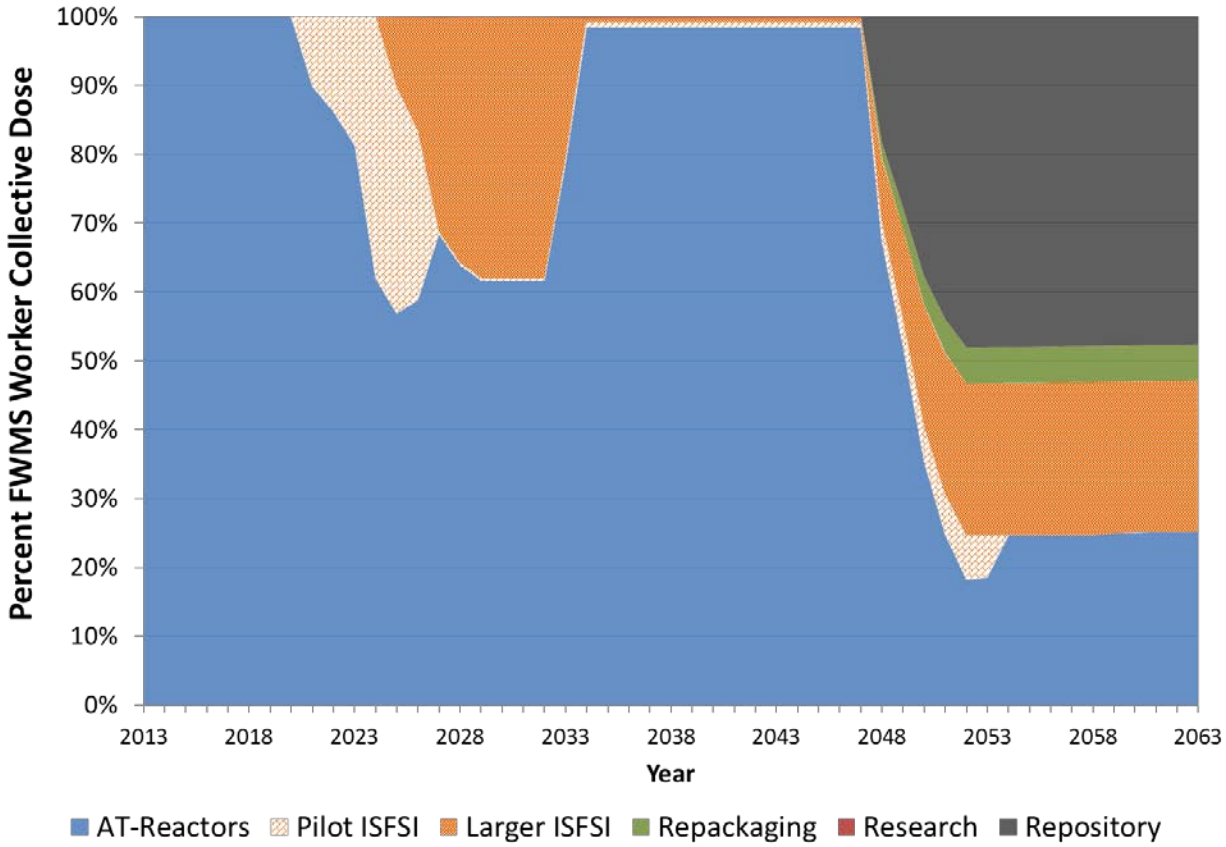


Figure 4.8 | Percent FWMS Worker Collective Doses

4.3.2. FWMS Contribution to Worker Collective Dose of Current OTC Operations

To place into context the additional worker collective dose from FWMS operations, the results from Chapter 2 for the OTC are now shown with the FWMS modeled in Figure 4.9 (denoted as "OTC+FWMS"). Worker collective doses for each year modeled are shown by the following grouped operations:

- Front-end (preparing enriched natural uranium, blended-low enriched uranium, and low re-enriched uranium fuel)
- Reactor operations (plus wet interim storage)
- Deconversion of depleted uranium hexafluoride tails

- Disposal of LLW and MLLW
- FWMS operations (dry interim storage at operating and shutdown reactors, pilot ISFSI, larger ISFSI, repackaging, research facility, repository)

It should be noted that worker collective dose is added due to additional handling of UNF. Because increased handling is required during the ISFSIs operations compared to currently ongoing operations of the OTC, the FWMS increases the total worker collective associated with the OTC. The estimated exposure rates are low and should not present a challenge to maintaining worker doses within regulatory limits and ALARA.

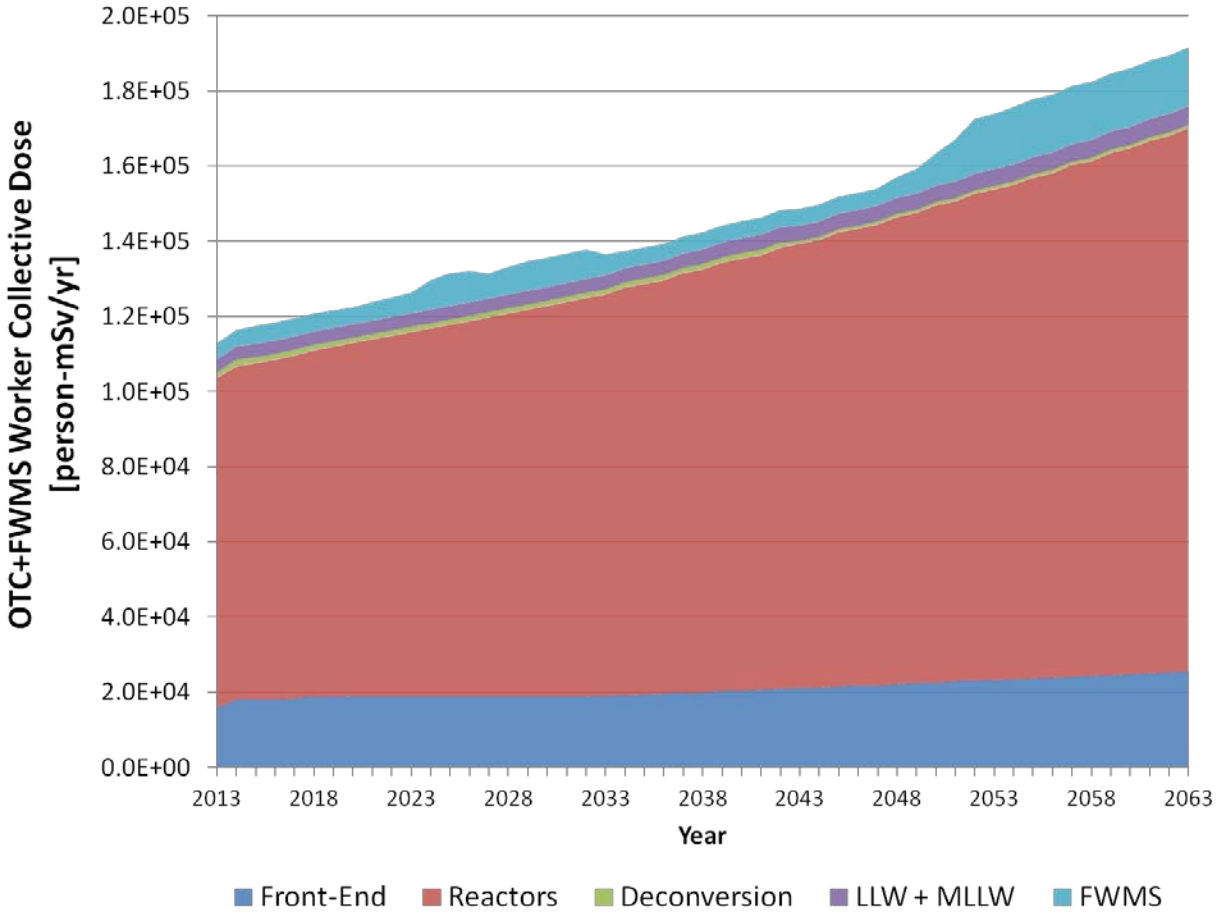


Figure 4.9 | OTC+FWMS Worker Collective Doses

The total OTC+FWMS worker collective doses range from 112,700 to 191,570 person-mSv/yr (Figure 4.9) and generally increases at the slow exponential curve of increasing nuclear energy demand. The addition of the FWMS (turquoise band, Figure 4.9) results with a higher overall worker collective dose towards the end of the model beginning in 2048. This reflects the flurry of activity when the repository opens and movement of UNF from the pilot ISFSI flows through to the repackaging facility and then directly to the repository.

As expected from findings of Krahn et al. (2014), the reactors and the front-end of the OTC constitute the majority of the total worker collective doses each year when looking at the OTC+FWMS (Figure 4.10). Note in Figure 4.10 of the truncated Y-axis showing that OTC reactors and front-end operations are 88-93% of the annual worker collective doses. During the same time, the back-end operations for handling UNF, LLW+MLLW, and deconversion are only a small part of the overall worker collective dose.

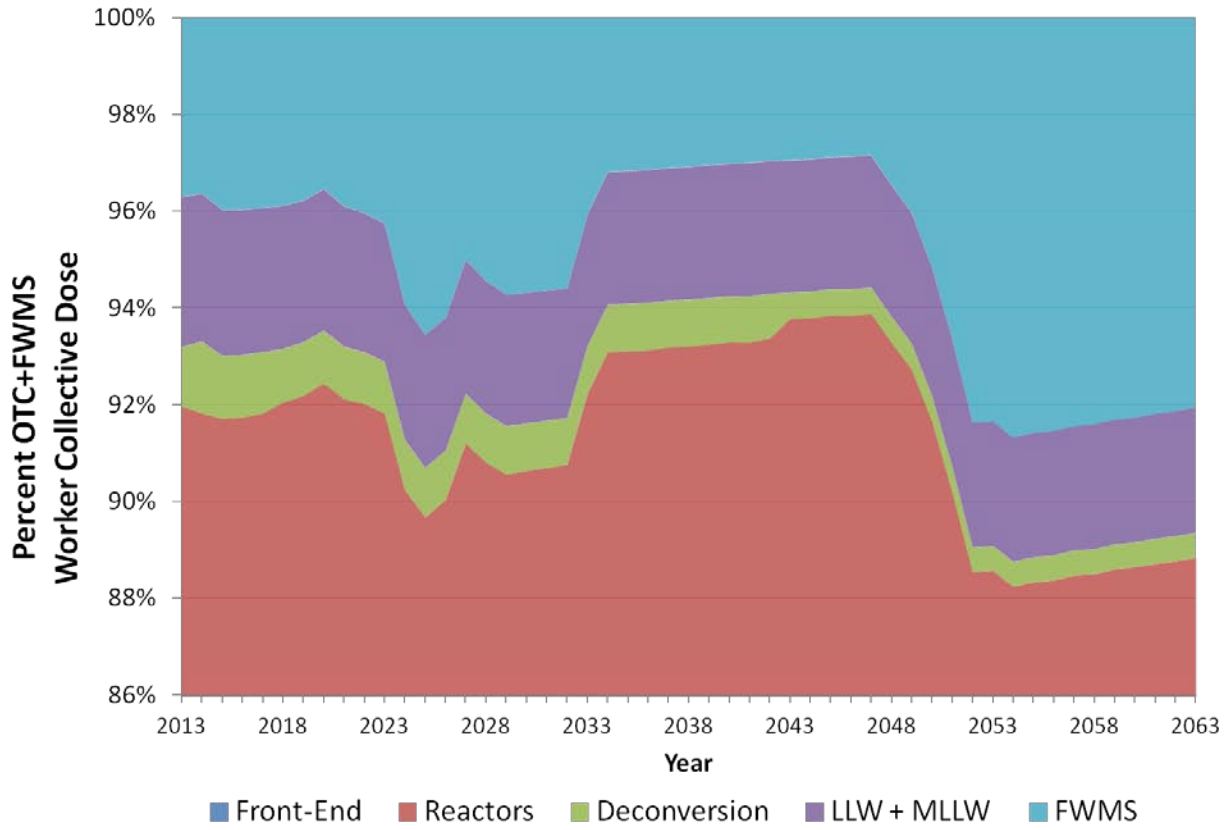


Figure 4.10 | Percent OTC+FWMS Worker Collective Doses

4.4. Conclusions

From modeling the FWMS that meets the stated three milestones in the DOE (2013) *Strategy Report*, the following conclusions can be made:

- The use of ISFSIs within a FWMS increases the total worker collective dose. The more ISFSIs used, the more worker collective dose is added due to additional handling of UNF. The increase is not necessarily linear.
- The worker collective doses from the FWMS modeled results with only a small overall contribution to the OTC (9% increase compared to current OTC operations). When the contribution of the FWMS is considered to total worker collective doses summed from

the OTC with FWMS operations, the overall contribution from the FWMS is about 8% when the emplacement of UNF reaches steady-state at the repository (3,000 MT UNF/yr).

- During the time period of transferring UNF from reactor sites to Pilot ISFSI and the Larger ISFSI, these operations increase worker collective dose approximately 1-3%.

CHAPTER 5

SUMMARY

5.1. Future Work

The work presented in this dissertation will hopefully serve as a launching pad for future studies. The expansion of this study is already underway at Vanderbilt where the transition from the MOC to a closed-fuel cycle system that breeds Pu within a fast-spectrum sodium-cooled fast reactor (SFR) would occur after a full implementation of MOX utilization in thermal-spectrum LWRs, thereby, setting up an evolutionary path. This path involves first understanding the OTC, moving to the MOC, and evolving towards a closed-fuel cycle. NFCs that utilize MOX fuel in a fast-neutron spectrum reactor is posited to be an energy system that fulfills some potential programmatic objectives such as fuel sustainability and minimizing uranium that must be mined from the earth.

Further refinements of the existing study could prove beneficial for understanding scenarios that are preferred by the commercial nuclear industry (e.g., instead of UNF withdrawn from at-reactor ISFSIs, consider withdrawing UNF from the spent fuel pool. If the time that the UNF is cooled in the spent fuel pool stays constant and carried forward from this analysis of 5 years, this effectively makes note that the UNF will be cooled equal to or less than 5 years before being withdrawn for reprocessing) or potential new sources for nuclear material as diluent for HEU to

make BLEU (e.g., extent of use of depleted uranium (DU), natural uranium (NU), or slightly enriched uranium (SEU) for use as the diluent).

Future work also includes investigating the availability of additional performance metric data to potentially expand the current set of performance metrics to safety issues particular to non-radiological, chemical aspects that are often prevalent at NFC facilities. The leading indicator for potential human health impacts related to on-site chemical use will be estimated as the reported chemical inventories required at each facility type. This metric will be a starting place for a safety-related metric and was chosen for keeping the metrics at a systems analysis level.

5.2. Conclusions

While this research focuses on a specific scenario, the MOC implemented in the U.S., the MOC on a whole represents a logical step while transitioning towards other advanced fuel cycle options that can seem esoteric and too far-reaching from the OTC starting point. Yet, if decision-makers and a multitude of stakeholders cannot be convinced that the MOC (or any other variant) can be carried out in a safe and environmentally responsible manner, the case for fast-reactor technology and more comprehensive recycling schemes will be even more difficult to make. Work presented in this dissertation supports the fact that worldwide industry experience with MOC operations can be done safely with potential benefits with regard to radioactive waste management and nearly identical performance of likely worker radiological impacts within the foreseeable future.

In Chapter 2, it was concluded that the OTC as implemented in the U.S. is not as simple as depicted in textbooks and other fuel cycle evaluations and must be updated in comparative NFC impact studies to serve as an accurate baseline to just performance of future potential NFCs. The OTC has changed in response to U.S. governmental programmatic changes and now includes operations of deconversion, re-enrichment, downblending, and extended dry interim storage.

Radiological impacts associated with the OTC were modeled by estimating worker collective dose within Chapter 2; however, these recent estimates are substantially different from historical data and do not support conventional wisdom that says that impacts from the front end of the OTC are dominant. Further analysis in Chapter 2 (with expanded discussion in Appendix C) concluded that the reversal of major contributors to the OTC from uranium recovery operations to reactor operations. This result is primarily driven by a substantial reduction in the global number of uranium recovery workers since 1980 despite increased uranium demand; albeit both reactor and mining operations have reduced annual individual worker doses to current levels that are 60-70% of individual doses expected in the 1980s.

Chapter 3 presents an analysis of the worker collective doses and radioactive waste volumes of the MOC to that of the OTC. The MOC as modeled in this study utilizes both Pu and RepU a single-time, as MOX and ERU fuel loaded in PWRs. Through the verification process of recent studies, there was agreement in modeling results that radioactive waste volumes correlated strongly to the amount of ENU avoided; but the verification evaluation was more complex during the comparison of worker collective doses. Two models of potential combinations and resultant MOC/OTC ratios of worker collective dose ratios and LLW volume generation were

created for the steady-state scenario. With the small sample size of recent studies and the current MOC study, it was postulated that the end-state of the simulation that described the proportion of fuel used in the reactor fleet could indicate whether a dynamic analysis was required or if a simpler, steady-state analysis would be sufficient to capture the results if one were to see a net benefit or increased impact between NFCs However, when a heterogeneous fuel-use portfolio was the end-state within the simulation, the net impacts could not be determined as easily and would require an in-depth dynamic analysis to understand the integrated impact ratio of the two NFCs under evaluation.

In Chapter 4, the impact of implementing the proposed federal waste management system (FWMS) was evaluated in terms of the worker collective dose for the OTC scenario as modeled in Chapter 2. Contributions by FWMS activities change the contributions of radiological worker impacts by increasing an additional 3-9% to that of the currently employed OTC without a disposal option. When renormalized, the FWMS modeled contributes around 4-8% of the total OTC worker collective dose annually, with higher percentages related to repository operations.

Research on potential benefits, increased burdens to human health and the environment, and other performance metrics must be considered when evaluating future nuclear energy production and the associated fuel cycle infrastructure that supports fuel production and waste management. Periodic assessments of modeling transitions to advanced NFCs are not sufficient when the implementation course is changing thanks to rapidly advancing technologies and changing policy preferences.

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APPENDIX A

WORKER COLLECTIVE DOSE PERFORMANCE METRIC DEVELOPMENT

The contents of this Appendix are the additional descriptions, references, and calculations that were used to develop some of the worker collective dose metrics and/or the three data sets (described in Chapter 4, Section 3.2.3, and listed in Table 3.3).

The worker collective dose metric calculations require three sets of data for MOC operations listed in Chapter 4.

- Number of monitored workers at the NFC facility during a specified year
- Average annual individual worker dose received for the same specified year
- Amount of nuclear material processed at the NFC facility for the same specified year

A.1. Reprocessing Worker Collective Dose Performance Metric

An NEA (2000) report was used for worker collective dose values describing reprocessing operating conditions during the years 1996 to 1997 (Table A.1). The worker collective dose metrics data are assumed to reflect all operations included inside of the La Hague reprocessing plant. Foare et al. (2013) lists the operations that are included in La Hague:

- Aqueous separations of Pu, U, fission products, metals using hot cell technology
- Shearing, compaction of structural material from the UNF fuel assemblies

- Vitrification of fission products and some transuranic contaminated process waste
- Capturing and treating effluents (liquid and gaseous)
- Resins management
- Vitrified HLW in UC-V handling and storage

Table A.1 | Reprocessing Worker Collective Dose Metric Parameter Data Sets

Country (Site)	Years of Operation	Annual Average Individual Worker Dose [mSv/yr]	Number of Monitored Workers	Annual Production Data [MTHN UNF Reprocessed/yr]
France (La Hague)	1996-1997	0.84	1045	1676

A.2. MOX Fuel Fabrication Worker Collective Dose Performance Metric

MOX fuel fabrication is performed with a glove-box to reduce internal doses caused from inhalation of Pu if handled by hand. An average of the worker collective dose metric data sets of the U.S. MOX fuel fabrication facility and France's MELOX facility operating experience were used to calculate the metric used here (Table A.2). Data from the MOX fuel fabrication facility was taken from the environmental impact statement [*NRC*, 2005] and the MELOX data were taken from NEA (2000) report.

Table A.2 | MOX Fuel Fabrication Worker Collective Dose Metric Parameter Data Sets

Country (Site)	Years of Operation	Reported Worker Collective [person-mSv/yr]	Annual Production Data [MTHM MOX /yr]
France (MELOX)	2000	2273	195
US (MFFF)	2005	2600	100

A.3. RepU Conversion Worker Collective Dose Performance Metric

The worker collective doses for France's Comhurex I conversion facility where conversion of reprocessed uranium occurs was obtained from [*Areva*, 2012a, 2012b] for the operational years of 2008 to 2010 (Table A.3).

Table A.3 | RepU Conversion Worker Collective Dose Metric Parameter Data Sets

Country (Site)	Year of Operation	Annual Average Individual Worker Dose [mSv/yr]	Number of Monitored Workers	Annual Production Data [MTHM RepU/yr]
France (Comhurex I - Pierrelatte at Tricastin)	2008-2010	0.72	97	12500

A.4. RepU Enrichment Worker Collective Dose Performance Metric

France separates RepU at the La Hague plant but then sends the RepU to Russia for enrichment [*IAEA*, 2007a]. Data on the worker doses at the Russian centrifuge facility is not available and so the worker doses are calculated based on a ratio of the ERU and UOX fuel fabrication and scaled to the available centrifuge enrichment values for ENU with the following equation (Equation A.1)

Where $M_{ERU\text{-}E}$ is the worker collective dose metric for the ERU centrifuge enrichment operation, $M_{ERU\text{-}F}$ is the worker collective dose metric for the ERU fuel fabrication operation, $M_{ENU\text{-}E}$ is the

worker collective dose metric for the ENU centrifuge enrichment operation, and $M_{ENU\text{-}F}$ is the worker collective dose metric for the ENU fuel fabrication operation.

The three variables used to calculate the RepU Enrichment metric are listed in Table A.4.

Table A.4 | RepU Enrichment Worker Collective Dose Metric Parameter Data Sets

Operation [Country] (Site)	Year of Operation	Annual Average Individual Worker Dose [mSv/yr]	Number of Monitored Workers	Annual Production Data [MTHM Enriched Uranium Product/yr]	Worker Collective Dose Metric
ENU Centrifuge Enrichment [US] (Average, LES and USEC)	2005-2006	0.25	405	730	1.36E-01
ERU Fuel Fabrication [France] (Romans Fuel Fabrication Facility)	2000	0.15	140	150	1.40E-01
ENU Fuel Fabrication [US] (Areva NP, Global Nuclear Fuel, Westinghouse Electric)	2000-02, 2004-2010	2.48	718	1017	1.75E+00
ERU Centrifuge Worker Collective Dose					1.09E-02

A.5. ERU Fuel Fabrication Worker Collective Dose Performance Metric

Worker collective dose information for ERU fuel fabrication was taken from the NEA (2000) report and annual production capacity information was taken from [*WNA*, 2013f] describing France's Romans UOX based fuel fabrication facility where ERU fuel fabrication takes place (Table A.5).

Table A.5 | ERU Fuel Fabrication Worker Collective Dose Metric Parameter Data Sets

Country (Site)	Year of Operation	Annual Average Individual Worker Dose [mSv/yr]	Number of Monitored Workers	Annual Production Data [MTHM ERU/yr]
France (Romans Fuel Fabrication Facility)	2000	0.15	140	150

A.6. LLW, MLLW Disposal Worker Collective Dose Performance Metric

Worker collective dose metrics were calculated for the disposal of LLW and MLLW via shallow-land burial on a volumetric basis according to reported values from the U.S. Ecology and Barnwell low-level waste disposal facilities as shown in Table A.6 [*WASDH,* 2004; *SCDHEC,* 2007].

Table A.6 | LLW, MLLW Disposal Worker Collective Dose Metric Parameter Data Sets

Country (Site)	Years of Operation	Reported Worker Collective Dose [person-mSv/yr]	Annual Production Data [m^3 LLW disposed/yr]
US (Average of U.S. Ecology , Barnwell)	1994-2002; 2005-2008	26.8	1234

A.7. FWMS: Additional At-Reactor Activities for Preparing UNF

Worker collective doses performance metrics associated with dry interim storage activities were previously quantified in an EPRI 2008 report [*EPRI,* 2008a] of UNF disposition options, and were used for estimations in previous chapters (Chapter 2, Section 2.2.3, and Table 2.5). The worker collective dose estimations for the OTC were already included in Chapter 2 and the were reported for a generic LWR USNF setting for current burnup rates and typical cooling times in spent fuel pools of 10 years and moved to on-site dry storage [*EPRI,* 2008a].

Additional operations at reactor sites that are not extensively performed currently that will be required for the FWMS are the following [*Weck*, 2013]: (1) Outloading of DPC from storage unit, either a vertical storage cask or from the horizontal storage module and, (2) Loading DPCs into a transport overpack and loading onto a truck for transport. Worker collective dose performance metrics associated with DPC outloading and preparation for transport are listed in Table A.7.

Table A.7 | Worker Collective Dose Performance Metrics (At-Reactor DPC Preparation for Transport)

Reference	Outloading DSC from VSC or HSM + Transport out of ISFSI storage area[c] [person-mSv/cask]	Loading DPCs into Transport Overpacks[d] and Truck Loading [person-mSv/cask]
[*Weck*, 2013][a]	0.73	2.2314
[*DOE*, 1987]	--	2.2314
[*Rawl et al.*, 2004][b]	--	2.35
		3.77
Average	**0.73**	**2.6**

Notes: [a] Weck (2013) reports the values from DOE (1987) of 0.4 person-mSv/cask for outloading a DSC from a VSC or HSM. [b] Rawl et al., (2004) lists two values of 2.35 person-mSv/cask for U.S. operations and 3.77 person-mSv/casks for French operations. Worker collective doses for France were higher because there are two areas that the DSC must be monitored; the redundancy increases the worker collective dose for this step. [c] The sum of placing the DPC with storage overpack on crane and then movement out to storage area was calculated from Weck (2013) as 0.062 + 0.269, respectively [d] It was estimated that loading DPCs into various overpacks are 0.4 person-mSv/cask and the remainder of the worker collective doses are from loading activities onto the truck.

If the single-purpose cask is used for dry storage, the UNF will have to be repackaged into a transport-approved DPC. There are only a limited number of these single-purpose casks; McCullum (2012) reported that in 2010 there were 238 of these single-purpose casks. It is assumed that utilities will not use single-purpose casks within the model employed here and 238 single-purpose casks are all that need to be repackaged at the reactor sites. Industry experience of at-reactor repackaging single-purpose casks within the spent fuel pool has been recorded as

around 2.5 person-mSv/cask [*Weck,* 2013]. Scaling the worker collective dose performance metric to the assumed fixed 238 single-purpose casks that must be repackaged into DPCs equates to 717.5 person-mSv. The DPCs then must be placed into a transport overpack and follow the same process as listed above in Table A.7 as 2.6 person-mSv/cask worker collective dose for preparing transport.

A.8. FWMS: Pilot and Larger ISFSI Operations

The Larger ISFSI is anticipated to be a larger scaled version of the Pilot ISFSI with minor design changes. Information on the likely small doses expected from the Pilot ISFSI will be applicable to the Larger ISFSI [*TRW,* 1992; *Weck,* 2013]. Front-end activities related to the Pilot ISFSI will be similar to those at the at-reactor smaller ISFSIs. Instead of transporting the UNF within the dry storage cask from the reactor building, there will be a receipt, inspection, and unloading area of DPCs with transport overpacks that will be unloaded and prepared for movement into the storage area (as listed in Table A.8). Next, the transport overpack must be replaced with a storage overpack and transported from the unloading area to the main storage area of the ISFSI. Finally, placing the cask vertically on a concrete pad or horizontally in a concrete vault will result with small, but measureable individual doses [*Weck,* 2013].

Because the DPC transport overpack is only required for a short amount of time and is re-sealable, it is assumed that the DPC transport overpack can be reused by the reactor sites. The steps of reusing transport overpacks are shown in Table A.8 and include (1) decontaminating the empty overpack and (2) loading onto a returning truck to the reactor sites.

Table A.8 | Worker Collective Dose Performance Metrics per Cask (Front-end of Pilot ISFSI and Larger ISFSI)

Operation Step	Worker Collective Dose Performance Metric [person-mSv/dry storage unit]	Reference
DPC Off-loading truck[a]	3.3	[*Smith et al.*, 1992; *NRC*, 2004]
DPC Receipt, handling, preparation[b]	3.1	[*Weck*, 2013]
Remove transport overpack on DPC[c]	0.4	[*Weck*, 2013]
Place storage overpack on DPC	0.4	[*Weck*, 2013]
Transport out to ISFSI storage area[d]	0.33	[*Weck*, 2013]
Subtotal	*7.53*	--
Decontaminate DPC transport overpack[e]	0.27	[*DOE*, 1987; *NRC*, 1992; *Weck*, 2013]
Move and load empty DPC transport overpack, decontaminated DPC overpack onto truck[f]	0.0	--
Subtotal	*0.27*	--
Total	**7.8**	--

Notes: [a] Average of two references for truck unloading: Smith et al. (1992) reports 2.75 person-mSv/cask and NRC (2004) reports 3.8 person-mSv/casks. [b] Weck (2013) provides a range from 2.2 to 3.93 person-mSv/cask. The average of these values were used. [c] Removing a transport overpack was assumed to be the same as placing a storage overpack on a DPC [d] The sum of placing the DPC with storage overpack on crane and then movement out to storage area was calculated from Weck (2013) as 0.062 + 0.269, respectively. [e]Average of 0.2336 and 0.3 person-mSv/empty casks Weck (2013) takes the values from DOE 1987 (0.236). [f] Assumed to be negligible for an empty, decontaminated transport overpack.

Again, it is expected that operations and maintenance of the ISFSI with loaded UNF, inspections of concrete housing units and security surveillance will be required, and loading of UNF into transport casks will have some associated small, but measureable doses. It is anticipated that worker collective doses can be reasonably approximated from the at-reactor dry storage activities (as was shown in Table 2.5 in Section 2.2.3) and was approximately 16.2 15 person-mSv/ISFSI each year (15 person-mSv/ISFSI site for operations and maintenance; 1.2 person-mSv/ISFSI site for security and surveillance).

When the repository is operating and ready to receive the UNF stored at the Pilot ISFSI, a detour to the repackaging facility at the Larger ISFSI is required since the smaller-capacity TAD canisters (with the disposal overpack) are the approved waste disposal form. At the pilot ISFSI, DPCs must be prepared for transport using the same process listed in Table A.7: (1) Outloading of DPC from storage unit, (2) Loading DPCs into a transport overpack and loading onto a truck for transport. The resultant worker collective dose performance metric is 3.33 person-mSv/DPC when readying for transport to the repackaging facility at the Larger ISFSI.

A.9. FWMS: Repackaging at the Larger ISFSI

Repackaging of fuel assemblies from DPCs into the repository-approved TAD canisters will be an important and large operation within the FWMS. UNF at the Larger ISFSI, in addition to the UNF from the Pilot ISFSI, will be channeled through the repackaging facility. As discussed in Section 4.2.3.1, repackaging will be performed in a pool and the throughput of repackaging operations must match the emplacement rates at the repository. Industry experience of at-reactor repackaging single-purpose casks within the spent fuel pool has been recorded as around 2.5 person-mSv per repackaged single-purpose cask [*Weck*, 2013]. It will be assumed that repackaging of DPCs to TAD canisters will result with the same worker collective dose performance metric as repackaging a single-purpose cask (2.5 person-mSv/repackaged DPC shown in Table A.9).

The reuse of DPCs is also assumed to be performed in this FWMS to prevent generation of high volumes of LLW. The front-end operation of the larger ISFSI has already accounted for

decontaminating the DPC transport overpack and it is expected to result with similar worker collective doses for decontaminating the DPC (as shown in Table A.9). With the empty, decontaminated DPC and DPC transport overpack, the two can both be transported in the same truck shipment back to the reactor site.

Table A.9 | Worker Collective Dose Performance Metrics per Cask (Repackaging at the Larger ISFSI)

Operation Step	Worker Collective Dose Performance Metric [person-mSv/dry storage unit]	Reference
Wet Repackaging	2.5	[*Weck*, 2013]
Subtotal	*2.5*	--
Decontaminate DPC	0.27	[*DOE*, 1987; *NRC*, 1992; *Weck*, 2013]
Move and load empty DPC, place DPC in DPC transport overpack, load decontaminated DPC+transport overpack onto truck[a]	0.0	--
Subtotal	*0.27*	--
Total	**2.77**	**--**

Notes: [a] Assumed to be negligible for an empty, decontaminated DPC with transport overpack.

The back end of the larger ISFSI involves preparing the TAD for transport to the repository. It will be assumed that the placement of the TAD transport overpack and truck loading will result with the same worker collective doses of 2.6 person-mSv/TAD as shown in Table A.7 similar to DPC preparation for transport.

A.10. FWMS: Collocated Research Facility at the Larger ISFSI

The tasks and capabilities of the collocated research facility at the larger ISFSI are unknown at this time; however, the BRC has indicated at a high-level of research needs of UNF that vary in

age, burnups, initial U-235 enrichments [*BRC*, 2012b]. There are multiple national laboratories that have the capability to do such experiments but the most readily available information on the worker collective dose performance metrics are listed in DOE (2008a) that describe the capabilities of Oak Ridge National Laboratory. There are two buildings that carry out research on fuel: (1) the Irradiated Fuel Examination Laboratory (IFEL) and the Radiochemical Engineering Development Center (REDC). IFEL performs destructive testing of UNF, is used for the receipt, segmentation, and testing of UNF and includes equipment used for voloxidation experiments and fission gas capture. UNF is typically processed through IFEL before being transferred to REDC for chemical processing" [*DOE*, 2008a]. The use of worker collective dose information for both the IFEL and REDC will be used to estimate performance metrics. The data on amount of fuel that can be tested and the duration of time for a fuel assembly is not readily available; thus the worker collective dose performance metric will be based on the combined sites on an annual basis of operations. DOE (2008a) reports that IFEL operation[36] results with 2 person-mSv per year of worker collective dose and REDC with 0.5 person-mSv per year. The worker collective dose performance metric of 2.5 person-mSv/year for the research facility collocated at the larger ISFSI will be used.

A.11. FWMS: Repository Operations

The repository will include components for handling UNF similar to that of the front-end operations for the pilot and larger ISFSI (as shown in Table A.10). Emplacing fuel worker dose

[36] There are an estimated 10 full-time employees that are monitored at IFEL, back calculating results with an average annual worker individual dose of 0.2 mSv/year. This is below the regulatory occupational limit of 50 mSv/year. The number of full time employees at REDC is not provided in DOE (2008a). UNSCEAR 2008 reports that the global annual average from 2000 to 2002 for individuals researching nuclear materials received 0.4 mSv/yr. This is in good alignment of doses reported in DOE (2008a) and the values from DOE (2008a) will be used.

estimates are taken from the reported calculations from the Yucca Mountain supplemental environmental impact statement [*DOE,* 2002a]. The total worker collective dose from emplacing 70,000 MT UNF into Yucca Mountain was estimated as 74,000 person-mSv throughout the operational life of the repository. This equates to a worker collective dose performance metric of 1.06 person-mSv/MT UNF emplaced. The mass capacities of TADs within disposal packs varied between BWR and PWR fuel assemblies (7.9 MT BWR UNF/TAD and 9.5 MT PWR UNF/TAD). The average mass capacities of the TAD is taken as 8.7 MT UNF/TAD and multiplied by the 1.06 person-mSv/MT UNF emplaced calculated above. The resulting worker collective dose performance metric is the value shown in Table A.10 of 9.22 person-mSv/TAD emplaced in the repository.

Table A.10 | Worker Collective Dose Performance Metrics per Cask (Repository)

Operation Step	Worker Collective Dose Performance Metric [person-mSv/dry storage unit]	Reference
Off-loading truck	3.3	Table A.8
Receipt, handling, preparation	3.1	Table A.8
Remove transport overpack on TAD	0.4	Table A.8
Place disposal overpack on TAD	0.4	Table A.8
Transport out to emplacement loading area	0.33	Table A.8
Emplacement	9.22	[*DOE*, 2002a]
Subtotal	*16.75*	--
Decontaminate TAD transport overpack[e]	0.27	Table A.8
Move and load empty TAD transport overpack, decontaminated TAD overpack onto truck	0.0	--
Subtotal	*0.27*	--
Total	**17.02**	--

A.12. Preliminary Evaluation of Worker Collective Dose Reductions from UNF Removal from At-Reactor ISFSIs

The basis for the worker collective dose normalized metric for maintenance, surveillance, and security of ISFSIs is the number of ISFSI sites. This normalization basis does not allow further consideration of the amount of UNF stored there (other than zero). It is likely that the resultant worker collective dose from maintenance, surveillance, and security would b a function of the number of UNF dry storage casks stored on site; however the data sets to allow calculating this metric with the basis of metric tons of UNF stored on site were not readily available at the time of the calculation, but a preliminary analysis has been performed and presented here.

This preliminary evaluation would assist in understanding this relationship of number of casks and the normalized worker collective dose for ISFSI maintenance, surveillance, and security. This preliminary evaluation also assists in understanding the potential reduction of worker collective dose associated with the decreased number of casks holding UNF due to the assumption employed here that the withdrawals of UNF for reprocessing occur at the ISFSI.

From EPRI (2008a), the collective worker dose from ISFSI maintenance, surveillance, and security activities is assumed to be incurred annually at a rate of 16.2 person-mSv/ISFSI site-yr (see Table 2.5 in Section 2.2.3). Within EPRI (2008a), the normalized metric is time-dependent because EPRI (2008a) baseline scenario assumes that the inventory of UNF 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 consequence, the amount of UNF subject to

maintenance, surveillance, and security activities at one site (i.e., the normalization basis) and the worker dose there from is not constant until 2050

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 MT UNF) to yield 1.16 person-mSv/MT UNF. 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 A.10.

There is doubt associated with this preliminary estimation of the normalized metric using a UNF-mass normalization basis to reflect the actual reduction in worker collective dose from withdrawals required for reprocessing. This is because when this developed metric is scaled to the reprocessing capacity at the end of the simulation (2400 MT UNF/yr) this results with 2,784 person-mSv/yr reduction. However, the worker collective dose previously calculated is 1,180 person-mSv/yr for maintenance, surveillance, and security for the 73 sites at 2063, and the total worker collective dose of the dry interim storage activities is 2,350 person-mSv/yr. The reduction of worker collective dose from withdrawals of UNF is higher than the current estimates and it is not possible to result with a negative worker collective dose; therefore this preliminary analysis was not incorporated in to the larger study because the parameter data sets to understand potential reductions in worker collective doses from decreased UNF stored at ISFSIs is required.

Table A.11 | Worker Collective Dose and Production Data for Dry UNF Interim Storage Based on Baseline Scenario in EPRI (2008a)

Parameter Description [Unit]	Parameter Value
Storage time assumed in scenario [years]	88
Cumulative worker collective dose in EPRI (2008a) scenario [person-mSv]	158,000
Steady-state UNF mass storage at ISFSI [MT UNF/yr]	136,600
Worker collective dose normalized metric [person-mSv/MT UNF]	1.16

APPENDIX B

RADIOACTIVE WASTE VOLUME PERFORMANCE METRIC DEVELOPMENT

Because radioactive waste volumes were not produced in Chapter 2 for OTC operations, the contents of this Appendix are descriptions, references, and calculations that were used to develop all of the OTC and MOC radioactive waste volume metrics (described in Chapter 4, Section 3.2.4, and listed in Table 3.4).

The radioactive waste volume metric calculations require three sets of data for OTC and MOC operations listed in Chapter 4.

- Average annual radioactive waste volumes produced for the same specified year
- Amount of nuclear material processed at the NFC facility for the same specified year

Radioactive waste volumes describe the volumes of these wastes that are eligible for shallow-land burial:

- Technologically enhanced naturally occurring radioactive material (TENORM) radioactive waste from open pit and underground uranium mines,
- Byproduct radioactive waste from in-situ leach uranium recovery operations and milling operations,
- Low-level radioactive waste (LLW) from all NFC operations except for operations listed above that produce TENORM and Byproduct radioactive wastes.

- Mixed low-level radioactive waste (MLLW) from a number of NFC operations, but are modeled here as conversion (dry technology only), enrichment (diffusion and centrifuge technologies), and MOX fuel fabrication (glove-box technology).
- Depleted uranium-oxides from deconversion operations (although the final disposition pathway is currently under review by the NRC).

Geological repository eligible wastes are the following [and discussed in the following Sections]:

- ENU, BLEU, Re-ENU, ERU UNF (mass of UNF and number of dry storage casks) [Section 3.3.2.3]
- MOX UNF (mass and number of dry storage casks) [Section 3.3.2.3]
- HLW (mass, mass of vitrified HLW, number of Vitrified containers) [Section 3.3.2.4]
- TRU (volume of TRU) [Section 3.3.2.4]
- GTCC (volume of GTCC) [Section 3.3.2.4]

B.1. Underground Uranium Mining Radioactive Waste Volume Performance Metric

The underground uranium mine data available for understanding the amount of mine waste (waste rock) compared to the amount of uranium mass found at natural concentrations of (0.71 wt% U-235) was taken from an EPA (1983a) report for a typical underground uranium mine located in New Mexico. The UG mine produces around 18,000 MT of uranium ore with an ore grade of 0.15%. This translates to around 27 MT of U_3O_8 is contained in this ore and 23 MT of natural uranium is produced by one UG mine each year. The stripping ratio is the quotient of mass of mine waste to the amount of uranium ore produced. EPA (1983a) states that UG mine

stripping ratio is half of that of an open pit (OP) uranium mine. The stripping ratio for the OP mine is around 2.9 (see next section, Section B.2.), thus it was assumed that modern day stripping UG ratios were around 1.4 and was used here. The amount of mine waste produced each year is 25,200 MT per year (18,000 MT * 1.4). The normalized amount of mine waste to the annual production of MT of uranium is 1096 MT mine waste per 1 MTHM natural uranium. The density of uranium ore, overburden and mine waste is given in EPA (1983a) and is estimated around 2 MT/m^3 which results with the radioactive waste volume metric of 548 m^3 UG mine waste/MTNU.

B.2. Open Pit Uranium Mining Radioactive Waste Volume Performance Metric

The Rössing open pit uranium mine data from a Rössing Uranium Limited company report for the production years of 2007 to 2011 [*Rössing*, 2011]. The average annual volume of overburden was calculated and normalized to the annual average MTNU produced that resulted with 6059 m^3 of overburden produced per 1 MTNU for an open pit mine. The stripping ratio of 2.9 was found by finding the quotient of the annual average overburden mass and the annual average ore produced (Table B.1).

Table B.1 | OP U-Mining Radioactive Waste Volume Metric Parameter Data Sets

Year	Number of Employees	U produced [MTNU/yr]	U_3O_8 produced [MT U_3O_8 /yr]	Ore Processed [MT Ore / yr]	Overburden [MT Overburden/ yr]	Overburden [m^3 Overburden/yr]
2007	1155	2589.1	3046	12613000	21396000	10698000
2008	1307	3491.8	4108	12858000	33899000	16949500
2009	1415	3527.5	4150	12633000	38755000	19377500
2010	1592	3083.8	3628	11598000	41955000	20977500
2011	1637	1825.8	2148	10729000	39913000	19956500
Average		2903.6	3416	12086200	35183600	17591800

B.3. Milling Radioactive Waste Volume Performance Metric

The volume of mill tailings was calculated first by taking the difference of uranium ore mass produced from a uranium mine and the amount of U_3O_8 mass in the ore. The average of uranium ore mass and amount of U_3O_8 mass produced on an annual basis from an UG and OP mine (from the Sections B.1. and B.2., respectively) were averaged and then the volume of the mill tailings were assumed to be 2 MT/m^3 [*EPA*, 1983a].

B.4. In-Situ Leach Radioactive Waste Volume Performance Metric

Radioactive waste from in-situ leach (ISL) is classified as 'byproduct waste' under the Uranium Mill Tailings Radiation Control Act (UMTRCA) of 1978, as amended within 40 CFR 192 [EPA, 2012]

Byproduct waste from ISL operations consists of waste from the yellowcake recovery process, water treatment, and waste from the use of evaporation ponds to reduce water treatment volumes. Data on radioactive waste production were taken from two U.S. sites at Crowe Butte, Nebraska and Nichols Ranch, Wyoming and the average of the two sites are listed here (Table B.2) [*NRC*, 2010, 2011c]. The annual production capacity for each site producing U_3O_8 and the mass of uranium (MTHM NU) was found in [*WNA*, 2013h].

Table B.2 | In-Situ Leach Radioactive Waste Volume Metric Parameter Data Sets

Country (Site)	Years of Operation	Volume of ISL waste [m^3 / yr]	Annual Production Data [MTNU/yr]
US (Average, Crowe Butte, NE and Nichols Ranch, WY)	2009, 2011	55.24	290.25

B.5. Conversion (Wet Technology) Radioactive Waste Volume Performance Metric

Waste production values for low activity waste that is produced during the Port Hope refinery and conversion facility employing wet conversion technology was found to be the following (Table B.3) [*Canada*, 2012]:

Table B.3 | Wet Conversion Radioactive Waste Volume Metric Parameter Data Sets

Country (Site)	Years of Operation	Volume of radioactive waste [m^3 / yr]	Annual Production Data [MTNU/yr]
Canada (Cameco, Port Hope Refinery and Conversion Facility)	2010	90	8750

B.6. Conversion (Dry Technology) Radioactive Waste Volume Performance Metric

Dry conversion technology is only used in the U.S. at the Honeywell conversion facility in Metropolis, Illinois. The amount of LLW produced, MLLW produced, and amount of material converted at the plant in 2006 were reported (Table B.4) [*NRC*, 2006]:

Table B.4 | Dry Conversion Radioactive Waste Volume Metric Parameter Data Sets

Waste Type	Country (Site)	Year of Operation	Volume of radioactive waste [m^3/yr]	Annual Production Data [MTHM NU/yr]
LLW	US (Honeywell Metropolis)	2006	468	12600
MLLW	US (Honeywell Metropolis)	2004	6	12600

B.7. Enrichment (Gaseous Diffusion) Radioactive Waste Volume Performance Metric

LLW produced from enriching uranium was provided in Saling & Fentiman (2002) for the operational years of 1983 to 1995 but MLLW were not provided. It was assumed that gaseous diffusion technology produces MLLW and was extrapolated to the ratio of MLLW to LLW that gas centrifuge produces. The ratio of MLLW volume and LLW is 5.75E-05 and was calculated in the next section (Section B.8.). Saling & Fentiman (2002) reported that 0.12 m^3 of LLW was produced from producing one MT of low-enriched uranium, thus 6.91E-06 m^3 (0.12 m^3 LLW * 5.75E-05 MLLW/LLW) of MLLW was calculated and used as the amount of MLLW per MTHM low-enriched uranium product.

B.8. Enrichment (Gas Centrifuge) Radioactive Waste Volume Performance Metric

The Louisiana Enrichment Services centrifuge facility radioactive waste volume data were taken from the environmental impact statement (Table B.5) [*LES*, 2005]. The ratio of MLLW to LLW was found to be 5.75E-05 (0.034 m^3 MLLW to 597 m^3 LLW).

Table B.5 | Centrifuge Enrichment Radioactive Waste Volume Metric Parameter Data Sets

Waste Type	Country (Site)	Year of Operation	Volume of radioactive waste [m^3/yr]	Annual Production Data [MTHM LEU/yr]
LLW	US (LES NM)	2005	597	438
MLLW	US (LES NM)	2005	0.034	438

B.9. Downblending Radioactive Waste Volume Performance Metric

Downblending of highly enriched uranium with depleted uranium hexafluoride tails occurs at the Nuclear Fuel Services (NFS) facility and data from the operational year 1996 has been used to estimate the LLW and MLLW waste volume metrics (Table B.6) [*DOE,* 1996].

Table B.6 | Downblending Radioactive Waste Volume Metric Parameter Data Sets

Waste Type	Country (Site)	Year of Operations	Volume of radioactive waste [m^3/yr]	Annual Production Data [MTHM LEU/yr]
LLW	US (NFS)	Early 1990s	46.00	150
MLLW	US (NFS)	Early 1990s	0.00	150

B.10. Fuel Fabrication (Non-Recycled) Radioactive Waste Volume Performance Metric

LLW volumes resulting from fuel fabrication of low-enriched natural uranium fuel (that includes ENU, BLEU and Re-ENU in this study) were taken from Saling & Fentiman (2002) and the Westinghouse Electric Company (2004) report and shown below (Table B.7) [*Saling & Fentiman,* 2002; *WEC,* 2004]. Saling & Fentiman (2002) reported normalized values and represented an average U.S. value from the operational years of 1983 to 1995. Westinghouse (2004) values were averaged over the seven years of data. The average of the two reports was used for the radioactive waste volume metric for hands-on fuel fabrication technology for ENU, BLEU and Re-ENU.

Table B.7 | Fuel Fabrication Radioactive Waste Volume Metric Parameter Data Sets

Country (Site)	Year of Operations	Volume of radioactive waste [m^3/yr]	Annual Production Data [MTHM LEU/yr]	Radioactive Waste Volume Metric [m^3 /MTHM LEU]
US	1983-1995	--	--	8.85E+01
US (Westinghouse Electric, Columbia SC)	1996-2000; 2002-03	11104	1150	9.66E+00
UOX Fuel Fabrication Average				4.91E+01

B.11. PWR Radioactive Waste Volume Performance Metric

PWR LLW generation on a per reactor unit basis was presented in both Saling & Fentiman (2002) and EPRI (2007) [*Saling & Fentiman*, 2002; *EPRI*, 2007]. An average of the reactor groups reported in Saling & Fentiman (2002) was taken from the years 1978 to 1986 with a range of reactors from 28 to 56 PWRs. LLW generation was tracked by EPRI (2007) for four years from 2003 to 2006 for 41 PWR reactor units and was used within a NRC (2012e) report for estimating potential LLW future production from ion exchange resins [*EPRI*, 2007; *NRC*, 2012e]. An average of the total LLW from the two references was used for PWR normalized LLW volume metrics (Table B.8).

Table B.8 | PWR Radioactive Waste Volume Metric Parameter Data Sets

Waste Type	Country	Year of Operations	Volume of radioactive waste [m^3/yr]	Annual Production Data [MWe-yr/yr]	Waste Volume Metric	Waste Volume Metric Unit
Dry LLW	US	1978-1986	259.3	1000	2.59E-01	m3 non-GTCC LLW/ MWe-yr
Wet LLW	US	1978-1986	105.7	1000	1.06E-01	m3 non-GTCC LLW/ MWe-yr
Total non-GTCC LLW	US	1978-1986	365.0	1000	3.65E-01	m3 non-GTCC LLW/ MWe-yr
Class A	US	2003-2006	208.1	1000	2.08E-01	m3 LLW-A / MWe-yr
Class B	US	2003-2006	3.1	1000	3.10E-03	m3 LLW-B/ MWe-yr
Class C	US	2003-2006	0.6	1000	6.00E-04	m3 LLW-C/ MWe-yr
Total non-GTCC LLW	US	2003-2006	211.8	1000	2.12E-01	m3 LLW/ MWe-yr
PWR Average			--	--	2.88E-01	m3 LLW/ MWe-yr

B.12. BWR Radioactive Waste Volume Performance Metric

Similar to the PWR LLW metric, BWR LLW generation on a per reactor unit basis was presented in both Saling & Fentiman (2002) and EPRI (2007) [*Saling & Fentiman*, 2002; *EPRI*, 2007]. An average of the reactor groups reported Saling & Fentiman (2002)was taken from the years 1978 to 1986 with a range of reactors from 18 to 27 BWRs. LLW generation was tracked by EPRI (2007) for four years from 2003 to 2006 for 24 BWR reactor units. An average of the total LLW from the two references was used for PWR normalized LLW volume metrics (Table B.9).

Table B.9 | BWR Radioactive Waste Volume Metric Parameter Data Sets

Waste Type	Country	Year of Operations	Volume of radioactive waste [m^3/yr]	Annual Production Data [MWe-yr/yr]	Waste Volume Metric	Waste Volume Metric Unit
Dry LLW	US	1978-1986	507.7	1000	5.08E-01	m^3 non-GTCC LLW/ MWe-yr
Wet LLW	US	1978-1986	310.7	1000	3.11E-01	m^3 non-GTCC LLW/ MWe-yr
Total non-GTCC LLW	US	1978-1986	818.3	1000	8.18E-01	m^3 non-GTCC LLW/ MWe-yr
Class A	US	2003-2006	431.9	1000	4.32E-01	m^3 LLW-A / MWe-yr
Class B	US	2003-2006	4.0	1000	4.00E-03	m^3 LLW-B/ MWe-yr
Class C	US	2003-2006	0.4	1000	4.00E-04	m^3 LLW-C/ MWe-yr
Total non-GTCC LLW	US	2003-2006	436.3	1000	4.36E-01	m^3 LLW/ MWe-yr
BWR Average			--	--	6.27E-01	m^3 LLW/ MWe-yr

B.13. Reprocessing Radioactive Waste Volume Performance Metric

Reprocessing technology and resultant waste volume values were mostly based upon values published by AREVA for HLW, GTCC, LLW and TRU waste categories [*Foare et al.*, 2013]. There were other values for volumes of GTCC and TRU within the NRC's environmental topical

report [*NRC,* 2012d]; the average between AREVA's values and the NRC's values has been calculated and used for GTCC and TRU waste (Table B.10). The volume of GTCC compacted metal waste per metric ton of UNF processed is given in NRC (2012d) but is not as clearly reported in Foare et al. (2013). An estimated 0.65 UC-C (universal canister containing compacted metal, Table 23) is provided along with the mass loading limit of a UC-C at 3.6 MT. It was assumed that the compaction of structural material created a solid mass of Zircaloy where the density of Zircaloy of 6.53 MT/m^3 could be used to estimate the volume of GTCC compacted metal waste [*NIST,* 1989; *Foare et al.,* 2013].

Table B.10 | Reprocessing GTCC Compacted Metal Radioactive Waste Volume Metric Parameter Data Sets

Type of GTCC waste	Country (Site)	Year of Estimation	Waste Quantity (per MT UNF processed)		Reference
Compacted Metal Mass	France (La Hague)	2013	5.51E-01	m^3 compacted metal/ MT UNF	[*Foare et al.,* 2013]
Compacted Metal Mass	France (La Hague)	2012	1.00E-01	m^3 compacted metal/ MT UNF	[*NRC,* 2012d]
Compacted Metal in Canisters	France (La Hague)	2013	6.50E-01	# UC-C/MT UNF	[*Foare et al.,* 2013]
Compacted Metals for GTCC Waste Average			3.26E-01	m^3 compacted metal/ MT UNF	--

Technological waste (TW) that was considered GTCC and was not compacted metal waste is placed in a high-integrity container (HIC) concrete cylinder. HICs can vary in size but is assumed to have a payload volume of 4.8 m^3 with reported dimensions of 6-ft in diameter and 6-ft high [*SCDHEC,* 2007; *NRC,* 2013c]. With the reported number of HIC's required for reprocessing 1 MT SNF at La Hague [*Foare et al.,* 2013], 0.05 HIC's are required for disposing GTCC technological waste, or 0.24 m^3 GTCC/MT UNF.

TRU contaminated materials are considered secondary waste streams according to Foare et al. (2013). As with GTCC, both the NRC and AREVA provide estimations of the amount of TRU

waste produced per MT UNF reprocessed (Table B.11) [*Foare et al.*, 2013; *NRC*, 2012d]. Technological waste considered TRU differs from technological waste considered as GTCC by the constituents of the contamination found on piping, metal tools, etc.

Table B.11 | Reprocessing TRU Radioactive Waste Volume Metric Parameter Data Sets

TRU Waste Form	Country (Site)	Year of Estimation	Waste Quantity (per MTHM UNF processed)		Reference
TW in 55-gallon drums	France (La Hague)	2013	2.91E-01	m^3 TRU	[*Foare et al.*, 2013]
Solid TRU Waste	France (La Hague)	2012	7.00E-02	m^3 TRU	[*NRC*, 2012d]
TW (tools, piping, etc.)	France (La Hague)	2012	1.00E-02	m^3 TRU	[*NRC*, 2012d]
Alpha Waste	France (La Hague)	2012	6.00E-02	m^3 TRU	[*NRC*, 2012d]
Technological Waste + Alpha-emitting wastes for TRU Waste			1.81E-01	m^3 TRU	--

Foare et al. (2013) reports the volume of vitrified HLW similarly to GTCC compacted metal waste where 0.7 UC-V (universal canisters holding vitrified HLW) results from reprocessing 1 MT UNF. The internal volume for 1 UC-V holding vitrified HLW is reported as 1.2 m^3 and the mass of glass encapsulating fission products and transuranic material can be calculated from the weight loading of radionuclides. Foare et al. (2013) provides the value of 25.0 wt% and Murray et al. (2012) reports a value of 18.5 wt%. The average of the two AREVA values is used for finding the mass of the vitrified HLW with a waste loading of 22 wt%. Murray et al. (2009) reports the masses of a filled UC-V as 500 kg and that 400 kg is the vitrified glass containing HLW. The HLW metrics used for estimating likely total HLW masses and volumes to be produced for the MOC study are listed below (Table B.12).

Table B.12 | Reprocessing Vitrified HLW Radioactive Waste Volume Metric Parameter Data Sets

HLW Parameters	Country (Site)	Waste Quantity (per MT UNF processed)	
HLW	France (La Hague)	6.2E-02	MT HLW/ MT UNF
Waste Loading in Glass (wt % HLW)	France (La Hague)	22	Wt% HLW / Vitrified HLW
Mass of Vitrified HLW	France (La Hague)	2.8E-01	MT Vitrified HLW/ MT UNF
Universal Canister for Vitrified Waste (UC-V)	France (La Hague)	0.7	UC-V / MT UNF
Volume of Vitrified HLW	France (La Hague)	8.4E-01	m^3 Vitrified HLW/MT UNF

There were four references that published LLW values related to reprocessing UNF and were averaged and used within the MOC model (Tble B.13) [*Saling & Fentiman,* 2002; *Areva*, 2011a; *NRC,* 2012d, *Foare et al.*, 2013]. Foare et al. (2013) provided the highest level of detail regarding waste types and radionuclides that are cemented and disposed as LLW.

Table B.13 | Reprocessing non-GTCC LLW Radioactive Waste Volume Metric Parameter Data Sets

Waste Type	Main Contents	Form	Quantity (m^3 per MT UNF Processed)	Category/ Class	Reference
LLW (Non-GTCC)			8.42E+00	Non-GTCC LLW	[*Foare et al.*, 2013]
Process	Kr-85	Gas under Pressure	5.0E-03	A	
	C-14 Waste	Cement	7.5E-02	A	
	Tritiated Wastes	Cement	5.0E+00	B	
	Salt-Bearing Wastes	Cement	1.4E+00	A	
Secondary	Spent Resins	Cement	7.5E-03	A	
	Solvent Residue	Cement	1.35E-02	A	
	Nitrate Effluents	Cement	1.2E+00	A	
	Technological Waste	Cement	6.4E-01	A	
	Technological Waste	Cement	8.0E-02	C	
LLW (Non-GTCC)			1.20E+00	Non-GTCC LLW	[*Areva*, 2011a]
LLW (Non-GTCC)			7.56E+00	Non-GTCC LLW	[*Saling & Fentiman*, 2002]
LLW (Non-GTCC)			1.40E+00	Non-GTCC LLW	[*NRC*, 2012d]
Reprocessing Total Non-GTCC LLW Average			4.65E+00	Non-GTCC LLW	--

B.14. MOX Fuel Fabrication Radioactive Waste Volume Performance Metric

Two data sets were available to estimate the LLW volume produced per MTHM MOX fuel fabricated (Table B.14) [*Saling & Fentiman*, 2002; *NRC*, 2005]. The environmental impact statement of the U.S. MOX fuel fabrication plant listed the solid and liquid contributions to the total volume of estimated LLW each year. The average of the two data points were used for LLW volume produced on a per MTHM MOX production basis.

Table B.14 | MOX Fuel Fabrication LLW Radioactive Waste Volume Metric Parameter Data Sets

LLW Waste Type	Country (Site)	Year of Operation	Volume of radioactive waste [m^3/yr]	Annual Production Data [MTHM MOX/yr]	Waste Volume Metric	Waste Volume Metric Unit	Reference
Total non-GTCC LLW	US	1983-1995	--	--	4.40E-01	m^3 LLW/ MTHM MOX	[*Saling & Fentiman*, 2002]
Total non-GTCC LLW	US (MFFF)	2005	--	--	2.88E+01	m^3 LLW/ MTHM MOX	[*NRC*, 2005]
Liquid LLW	US (MFFF)	2005	2.28E+03	100	2.28E+01	m^3 liquid LLW/ MTHM MOX	
Solid LLW	US (MFFF)	2005	6.05E+02	100	6.05E+00	m^3 solid LLW/ MTHM MOX	
MOX Fuel Fabrication Total Non-GTCC LLW					1.46E+01	m^3 LLW / MTHM MOX	--

MLLW and TRU waste annual volumes were provided in the MOX fuel fabrication facility environmental impact statement [*NRC*, 2005] and included in this analysis (Table B.15).

Table B.15 | MOX Fuel Fabrication TRU, MLLW Radioactive Waste Volume Metric Parameter Data Sets

Waste Type	Country (Site)	Year of Operation	Volume of radioactive waste [m^3/yr]	Annual Production Data [MTHM MOX/yr]	Waste Volume Metric	Waste Volume Metric Unit	Reference
TRU	US (MFFF)	2005	4.43E+02	100	4.43E+00	m^3 TRU/ MTHM MOX	[*NRC*, 2005]
MLLW	US (MFFF)	2005	1.20E+01	100	1.20E-01	m^3 MLLW / MTHM MOX	[*NRC*, 2005]

B.15. RepU Conversion Radioactive Waste Volume Performance Metric

The uranium conversion Comhurex facility in France utilizes the wet conversion technology but data on waste volume generation were not available. It was assumed that the wet conversion technology used in Canada would produce similar LLW volumes as the French facility. LLW is not a recognized waste category anywhere but the U.S.; waste produced from the conversion facility would be classified as low-level radioactive waste to intermediate-level radioactive waste and is assumed to be of the same hazard level and metrics were based off of the LLRW values reported by [*Canada*, 2012]. The amount of uranium product that can be converted annually was taken from WNA (2013g) (Table B.16).

Table B.16 | RepU Conversion Radioactive Waste Volume Metric Parameter Data Sets

Country (Site)	Year of Operation	Volume of radioactive waste [m^3/yr]	Annual Production Data [MTHM /yr]	Waste Volume Metric	Waste Volume Metric Unit
Canada (Cameco, Port Hope Refinery and Conversion Facility)	2010	90	8750	1.03E-02	m^3 LLRW & ILRW / MTHM

B.16. RepU Enrichment Radioactive Waste Volume Performance Metric

LLW volumes produced from gas centrifuge enrichment have been estimated in Section B.8 for the front-end of the fuel cycle and assumed here that the enrichment of RepU will produce similar LLW volumes. The LES centrifuge plant in New Mexico was used as the representative technology and the environmental impact statement was used as the reference resulting with 1.36 m^3 of LLW produced per 1 MT of low-enriched uranium product [*LES,* 2005].

B.16. ERU Fuel Fabrication (Hands-on) Radioactive Waste Volume Performance Metric

The representative fuel fabrication technology for enriched RepU was sampled after the Romans plant in France in which an IAEA (2009a) describes as a two-step process. First, a conversion step must be performed to change the uranium hexafluoride to the oxide form and second, the fuel undergoes UOX fuel fabrication with pelletizing and sintering to a ceramic fuel material. Additional scrap material and residues are removed from the first conversion step where the fuel stream moves on to the second step [*IAEA,* 2009a]. The data from the Romans plant on waste production were not available and analogous processes from other nuclear fuel cycle facilities were used. The first step involves removing excess scrap material from the conversion process and the available data on waste volumes generated for such a specific step is only described in the dry conversion technique at the Honeywell conversion plant [*NRC,* 2006]. The second step of pelletizing and sintering was assumed to be the same as the UOX fuel that does not include reprocessed uranium. The values from the UOX fuel fabrication were used and added to the amount of waste produced from the first step (as shown in Table B.17).

Table B.17 | ERU Fuel Fabrication Radioactive Waste Volume Metric Parameter Data Sets

Process	Country (Site)	Year of Operation	Volume of radioactive waste [m^3/yr]	Annual Production Data [MTHM /yr]	Waste Volume Metric	Waste Volume Metric Unit
Scraps compaction - put in same canisters as UC-C from reprocessing	US (Honeywell Metropolis)	2006	213	12600	1.69E-02	m^3 LLW / MTERU
Final residues - cemented and disposed	US (Generic); US (Westinghouse Electric, SC)	--	--	--	4.91E+01	m^3 LLW/MTERU
ERU Fuel Fabrication Total Non-GTCC LLW					4.91E+01	m^3 LLW / MTERU

B.17. Deconversion Radioactive Waste Volume Performance Metric

Two types of modeled radioactive waste volumes were depleted uranium oxides and LLW and the sample deconversion facility was modeled after the IIFP facility in New Mexico. The environmental impact statement from 2012 was used for both types of radioactive waste volumes [*NRC*, 2012c]. It was estimated that 1300 m^3 of depleted uranium oxide would be produced per year and 2,300 MT of depleted uranium oxide would be deconverted; the resulting volume of depleted uranium oxide is 5.65E-01 m^3 DUO_2/MTHM DU (The inverse of this metric does not represent the bulk density of depleted uranium oxide because of the non-compacted storage form). LLW and the types of materials considered as LLW were detailed in NRC (2012c) and listed below (Table B.18).

Table B.18 | Deconversion Radioactive Waste Volume Metric Parameter Data Sets

Type of LLW	Country (Site)	Year of Operation	Volume of radioactive waste [m^3/yr]	Annual Production Data [MTHM /yr]	Waste Volume Metric	Waste Volume Metric Unit
LLW - Activated alumina	US (IIFP)	2011	9.3	2300	4.03E-03	m^3 LLW/MTDU
LLW - Air ventilation filters	US (IIFP)	2011	0.2	2300	1.01E-04	m^3 LLW/MTDU
LLW - Carbon	US (IIFP)	2011	85.8	2300	3.73E-02	m^3 LLW/MTDU
LLW - DUF_4 clinkers	US (IIFP)	2011	23.3	2300	1.01E-02	m^3 LLW/MTDU
LLW - Coke	US (IIFP)	2011	30.9	2300	1.34E-02	m^3 LLW/MTDU
LLW - Crushed drums	US (IIFP)	2011	6.3	2300	2.76E-03	m^3 LLW/MTDU
LLW - Dust collector bags	US (IIFP)	2011	5.6	2300	2.43E-03	m^3 LLW/MTDU
LLW - Ion exchange resin	US (IIFP)	2011	4.6	2300	2.01E-03	m^3 LLW/MTDU
LLW - Radioactive waste trash	US (IIFP)	2011	140.7	2300	6.12E-02	m^3 LLW/MTDU
LLW - Scrap metal	US (IIFP)	2011	18.5	2300	8.06E-03	m^3 LLW/MTDU
LLW - Sintered metal tubes	US (IIFP)	2011	4.6	2300	2.01E-03	m^3 LLW/MTDU
LLW - Sodium fluoride	US (IIFP)	2011	9.3	2300	4.03E-03	m^3 LLW/MTDU
LLW - Spent blasting sand	US (IIFP)	2011	0.5	2300	2.01E-04	m^3 LLW/MTDU
LLW - Wood trash (pallets)	US (IIFP)	2011	7.7	2300	3.36E-03	m^3 LLW/MTDU
Deconversion Total Non-GTCC LLW			347.4	2300	1.51E-01	m^3 LLW/MTDU

B.18.Definitions of Repository Eligible Wastes

Regulatory definitions of each of the repository-eligible wastes are provided. Some classifications of waste have multiple definitions dependent if these wastes are produced within the commercial space or within DOE (both definitions are provided).

High-Level Waste (HLW) under 10CFR63.2 Definitions of Disposal of High-Level Radioactive Wastes in a Geologic Repository at Yucca Mountain, Nevada [*NRC*, 2014b] (this applies to both commercial and DOE HLW):

> "(1) The highly radioactive material resulting from the reprocessing of spent nuclear fuel, including liquid waste produced directly in reprocessing and any solid material derived from such liquid waste that contains fission products in sufficient concentrations; (2) Irradiated reactor fuel; and (3) Other highly radioactive material that the Commission, consistent with existing law, determines by rule requires permanent isolation."

Transuranic (Contaminated material) as defined by DOE (TRU waste is a waste category that applies to wastes owned or generated by DOE) [*DOE,* 2011c, 2011e]:

> "Transuranic (TRU) waste is radioactive waste containing more than 100 nanocuries (nCi) of alpha-emitting transuranic radionuclides with half-lives greater than 20 years per gram of waste, except for (1) high-level radioactive waste; (2) waste that the Secretary of Energy has determined, with the concurrence of the Administrator of the U.S. Environmental Protection Agency, does not need the degree of isolation required by the 40 CFR Part 191 disposal regulations; or (3) waste that the NRC has approved for disposal on a case-by-case basis in accordance with 10 CFR Part 61."

"Greater-than-Class C (GTCC)" type of Low-Level Waste is produced commercially (the NRC definition is provided) and "GTCC-Like Waste" is produced by DOE [*DOE,* 2011c]:

The NRC classification system for the four classes of LLW (A, B, C, and GTCC) is established in 10 CFR 61.55 and is based on the concentrations of specific short- and long-lived radionuclides given in two tables. Classes A, B, and C LLW are generally acceptable for disposal in near-surface land disposal facilities. GTCC LLW is LLW "that is not generally acceptable for near-surface disposal" as specified in 10 CFR 61.55(a)(2)(iv). As stated in 10 CFR 61.7(b)(5), there may be some instances where waste with radionuclide concentrations greater than permitted for Class C would be acceptable for near-surface disposal with special processing or design.

The definition and classification of GTCC-like waste produced by DOE is still being reviewed and the draft EIS from 2011 has no stated record-of-decision (ROD) of which disposition pathway DOE prefers [*DOE,* 2011c]. The DOE (2011c) EIS discusses the inventories and types of materials that are being evaluated under the classification of GTCC-like waste as the

following (About three-fourths of the waste by volume is GTCC LLRW; GTCC-like waste accounts for the remainder. It should be noted that DOE considers much of the GTCC-like waste meets the DOE definition of TRU waste):

The wastes being addressed in this EIS are divided into three distinct types. These three waste types and their estimated total volumes and radioactivities are as follows:

- Activated metals: 2,000 m^3 (71,000 ft^3) and 160 MCi
- Sealed sources: 2,900 m^3 (100,000 ft^3) and 2.0 MCi
- Other Waste: 6,700 m^3 (240,000 ft^3) and 1.3 MCi

APPENDIX C

SHIFTS IN CONTRIBUTORS TO WORKER RADIOLOGICAL IMPACTS IN THE OTC

Within Chapter 2, a summary of the auxiliary study to verify OTC worker collective doses was provided. This auxiliary study is now published [*Krahn et al.*, 2014] and the discussion is extended within this appendix.

C.1. Background

The focus of this portion of the study is to understand the relative radiological impacts of nuclear fuel cycle (NFC) operations on the workers that perform the operations and how they have evolved. The impact of radiation on an individual is measured by the total effective dose equivalent (just ‘‘dose’’ hereinafter) received in some amount of time, such as milliSievert (mSv) per year (mSv/yr). This individual dose takes into account external and internal radiation exposures weighted by the biological effects of various types of radiation (gamma ray, alpha particle, etc.) and the susceptibility of various organs to adverse impacts such as cancer. The average annual individual worker dose rate (mSv/yr) for a particular NFC operation, such as uranium mining or reactor operation, can be determined by summing the individual annual doses for all workers in the operation and dividing by the number of workers. However, this average is not a meaningful measure of the radiological impact of various NFC operations because the number of exposed workers in each operation can vary substantially. To compensate for this, the accepted measure of the radiological impact of NFC operations on workers is the annual

collective (or population) occupational dose, which is the average annual dose to the workers in a particular NFC operation multiplied by the number of workers in that operation with typical units being person-mSv/yr[37].

Comparative analysis of estimated annual collective dose to workers among NFCs, and NFC operations, from different sources is complicated by the fact that the collective dose depends on, among other things, the number of workers, which typically depends on the size or throughput of the cycle or facility. For NFCs, this complication is addressed by normalizing the collective dose to the throughput [e.g., metric ton (MT) of uranium (MTU)] of the cycle or the electrical energy [GW(electric)-yr] produced by the cycle (or portion thereof).

With the concept of collective doses having been established as the measure of radiological impact to workers in this chapter, the NFC and its component operations that provide the basis for discussion of worker collective dose estimates and data are next described for this remaining section. This is followed by a brief review of the highlights from previously performed assessments (found in Chapter 2), a summary of historical global collective worker dose data, leading to a comparison of the recent assessment results to historical data and identification of important differences.

[37] The same collective dose could also be calculated by simply summing the average annual individual doses for the workers in a NFC operation. However, the two-part representation here provides a more transparent framework for subsequent discussions.

C.1.1. Recent Estimates of OTC Worker Collective Doses

Previous modeling of the OTC was performed to estimate the total worker collective doses at each operation to support an increasing electricity demand in the U.S. from 2013to 2063 and results were presented in Chapter 2. A finding of importance here is that over the 50-year modeling time frame, the contribution by the reactors to overall OTC worker collective dose was much greater than of other operations. Additionally, contributions of each operation did not vary over time (as shown in Figure 2.8).

As part of the quality control and verification process, we evaluated available data published by the United Nation's Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) data, especially for uranium-recovery[38] and comparing worker collective doses and numbers of workers employed in these stages of the nuclear fuel cycle to the results of the initial model runs. Verification against the UNSCEAR (1993, 2008) reports required a simple method to aggregate worker collective doses from major operations that are traditionally used world-wide to prepare fuel [*UNSCEAR,* 1993, 2008]. This meant that contributions to OTC worker collective dose from re-enrichment, downblending and deconversion, that were not part of the groupings of operations and the contributing operations, were reorganized into the conventional phases of: (1) uranium-recovery, (2) milling, (3) conversion and enrichment[39] , (4) fuel fabrication[40], and (5) reactor operations. The operations that were shown in Figure 2.7 are now highlighted by re-grouped operations and labeled in Figure C.1 according to the five operations considered for verification

[38] Uranium recovery includes underground mining, open pit mining, and in-situ leach recovery.
[39] [*UNSCEAR,* 2008] combines the enrichment and conversion operations when evaluating worker collective doses
[40] The modeled worker collective doses from fabricating BLEU, and Re-ENU fuel were excluded for this companion exercise

against world-wide UNSCEAR data. The resultant worker collective doses from regrouping are shown in Figure C.2. The use of worker collective doses from the modeling year 2018 will be used for this verification process (as seen later) because the contribution of Re-ENU has ceased in that year (fuel sources return to 90% of the fuel needs are met by ENU prepared through the traditionally-defined operations).

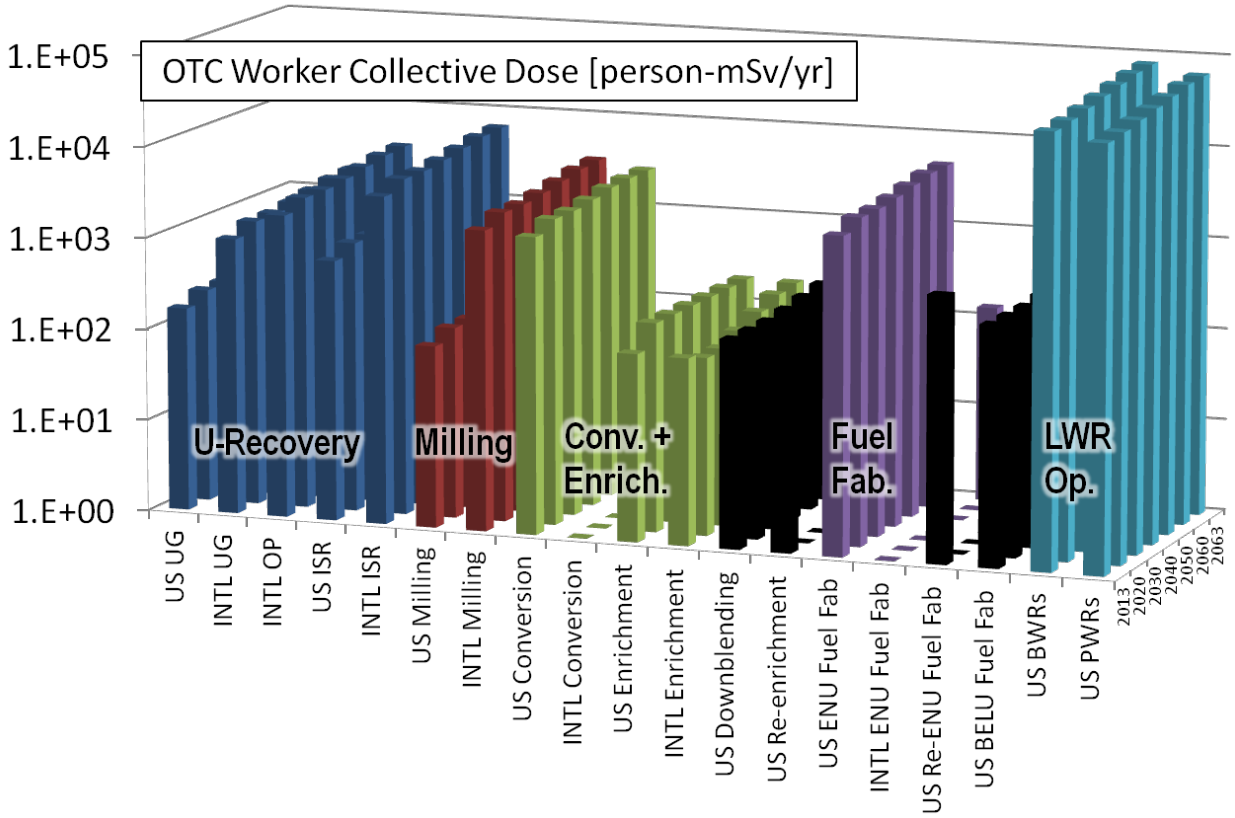


Figure C.1 | Regrouping OTC Operations to Conventional Phases for Verification

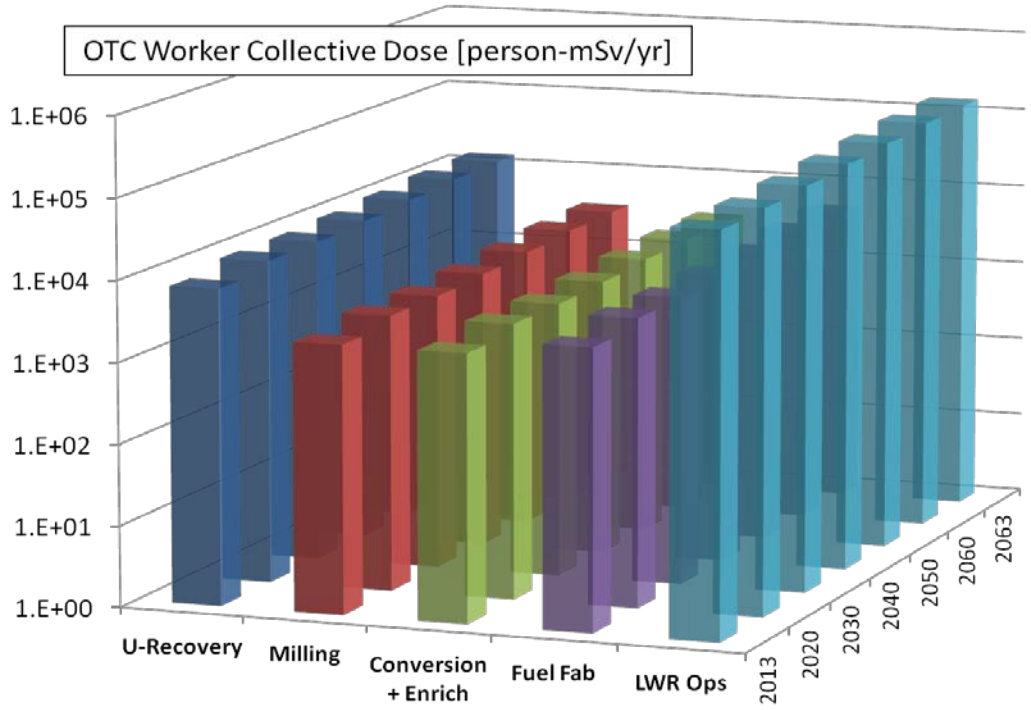


Figure C.2 | Worker Collective Doses of Regrouped Global OTC Operations

C.1.2. Historical Data on Worker Collective Dose from Traditionally-Defined OTC Operations

Results from historical surveys of worker occupational dose from NFC operations are available. However, such surveys are infrequently done because they are costly and time-consuming; this is due to the large amount of data that is acquired and the need to ensure consistent nomenclature, data collection, and assumptions. The seminal work on this subject is documented in the "UNSCEAR 1993 Report to the General Assembly," which, among other things, documents a global survey of information related to worker doses for NFC operations covering the years 1975–1989 that are summarized in Table C.1 and Table C.2, for three 5-yr time periods [*UNSCEAR*, 1993]. For each time period the following are reported (on a global basis): average worker individual dose for each operation, number of workers involved in each operation,

worker collective dose to workers for each operation, and contribution of each operation to the total collective dose to workers. Two percentages are given for each operation in each time period: one that includes all operations and one that includes all operations except reprocessing and research (which is essentially the OTC).

Table C.1 | Global Worker Collective Dose Parameter Data Sets [*UNSCEAR*, 1993]

NFC Operation	1975-1979		1980-1984		1985-1989	
	Average Individual Dose [mSv/yr]	Workforce [number of monitored workers]	Average Individual Dose [mSv/yr]	Workforce [number of monitored workers]	Average Individual Dose [mSv/yr]	Workforce [number of monitored workers]
Mining	5.5	2.40E+05	5.1	3.10E+05	4.4	2.60E+05
Milling	10	1.20E+04	5.1	2.30E+04	6.3	1.80E+04
Enrichment	0.5	1.10E+04	0.2	4.00E+03	0.1	5.00E+03
Fuel Fabrication	1.8	2.00E+04	1.0	2.10E+04	1.0	2.80E+04
Reactor Operations	4.1	1.50E+05	3.6	2.90E+05	1.4	4.30E+05
Reprocessing	7.1	7.80E+04	4.9	9.00E+03	1.5	1.70E+04
Research	1.4	1.20E+05	1.1	1.30E+05	0.8	1.30E+05
Subtotal (excludes reprocessing and research)	21.9	4.33E+05	15	6.48E+05	10.3	7.41E+05
Total	30.4	5.60E+05	21	8.00E+05	12.6	8.88E+05

Table C.2 | Global Worker Collective Dose and Percent Contribution [*UNSCEAR*, 1993]

NFC Operation	1975-1979		1980-1984		1985-1989	
	Annual Worker Collective Dose [person-mSv/yr]	Percentage Total Worker Collective Dose[a] [%]	Annual Worker Collective Dose [person-mSv/yr]	Percentage Total Worker Collective Dose[a] [%]	Annual Worker Collective Dose [person-mSv/yr]	Percentage Total Worker Collective Dose[a] [%]
Mining	1.30E+06	56.5 (63.0)	1.60E+06	53.3 (58.4)	1.10E+06	44.0 (47.0)
Milling	1.24E+05	5.4 (6.0)	1.17E+05	3.9 (4.3)	1.16E+05	4.6 (5.0)
Enrichment	5.00E+03	0.2 (0.2)	1.00E+03	0.0 (0.0)	4.00E+02	0.0 (0.0)
Fuel Fabrication	3.60E+04	1.6 (1.7)	2.10E+04	0.7 (0.8)	2.20E+04	0.9 (0.9)
Reactor Operations	6.00E+05	26.1 (29.1)	1.00E+06	33.3 (36.5)	1.10E+06	44.0 (47.0)
Reprocessing	5.30E+04	2.3 (0.0)	4.60E+04	1.5 (0.0)	3.60E+04	1.4 (0.0)
Research	1.70E+05	7.4 (0.0)	1.50E+05	5.0 (0.0)	1.00E+05	4.0 (0.0)
Subtotal (excludes reprocessing and research)	2.07E+06	89.8 (100)	2.74E+06	91.3 (100)	2.34E+06	93.5 (100)
Total	2.30E+06	100	3.00E+06	100	2.50E+06	100

Notes: [a] Percentages listed in parentheses are percent of the operations: mining, milling, enrichment, fuel fabrication, and reactor operations only.

C.1.3. Analysis of Historical Data and OTC Modeled Worker Collective Doses

Analysis of the information in Figure C.2 and Table C.2 [*UNSCEAR,* 1993], for comparative purposes, can be simplified with the recognition that on a system-wide basis, worker collective dose from uranium recovery and reactor operations accounts for ~92-94% of the OTC, which means that other three OTC operations are negligible. With this assumption, and using an average of the three time periods in the UNSCEAR (1993) data, the detailed information concerning contributions to total worker collective dose can be distilled into the information given in Table C.3.

Table C.3 | Regrouped Estimated Worker Collective Doses Compared to Historical Data

OTC Operation	VU Model Estimations 2018 Data	[*UNSCEAR,* 1993] Report[a] 1975-1989 Data
Uranium Recovery	8.8 %	56.1 %
Reactor Operations	83.5 %	37.5 %
Total	92.3%	93.6 %

Notes: [a] The average of the three time periods from Table C.2 is presented here.

The contribution percentages based on UNSCEAR data are consistent with other data seen [*Machiels*, 1992] in and estimates performed during the 1970s and 1980s which draw on a number of other sources such as the ‘‘Final Generic Environmental Statement on the Use of Recycle Plutonium in Mixed Oxide Fuel in Light Water Cooled Reactors (GESMO)’’ [*NRC,* 1976] and ‘‘Health Effects Attributable to Coal and Nuclear Fuel Cycle Alternatives’’ (NUREG-0332) [*NRC*, 1977]. These results account for the historical ‘‘conventional wisdom’’ that impacts from the front end of the NFC are dominant. However, estimates from the recent VU work are substantially different from the historical data and do not support conventional wisdom, which leads to the central question: Can these disparate results be reconciled? The answer to this

question is “yes,” and will be addressed in the following sections by focusing on the two terms used to calculate worker collective dose: the average dose per worker (Section C.2) and the number of workers exposed to this average dose (Section C.3).

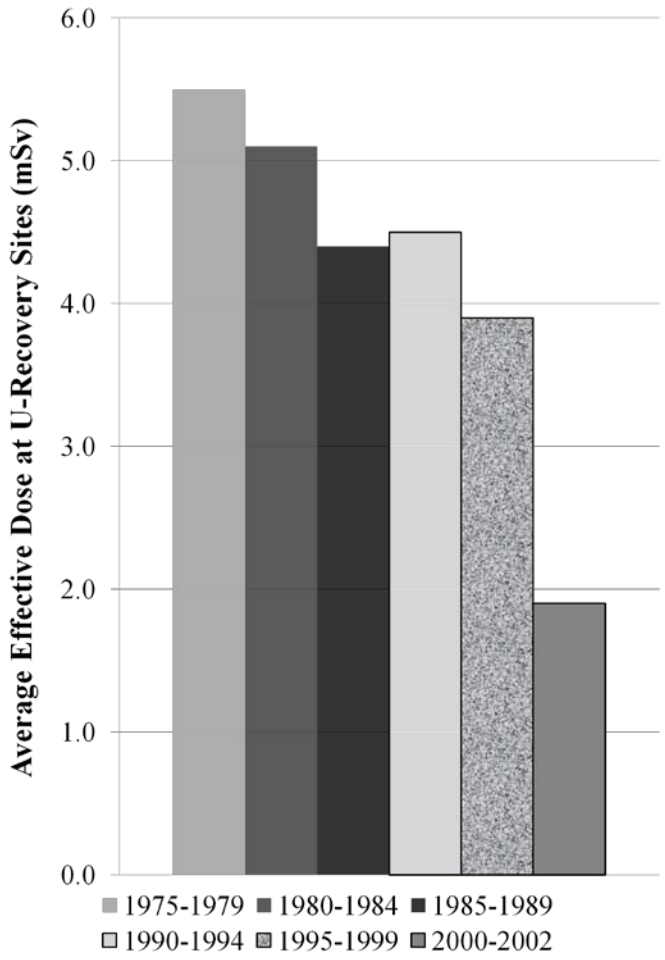
C.2. Worker Individual Dose Trends at U-Recovery and Reactor Operations

The total annual worker collective dose for a fuel cycle operation is directly proportional to the average annual individual dose to a worker. As a consequence, to the extent that the average annual worker individual dose from uranium recovery has changed over time relative to the average annual worker individual dose from reactor operations, the contribution of these two operations to total worker collective dose will vary. These trends will be examined in the remainder of this section.

C.2.1. Uranium Recovery

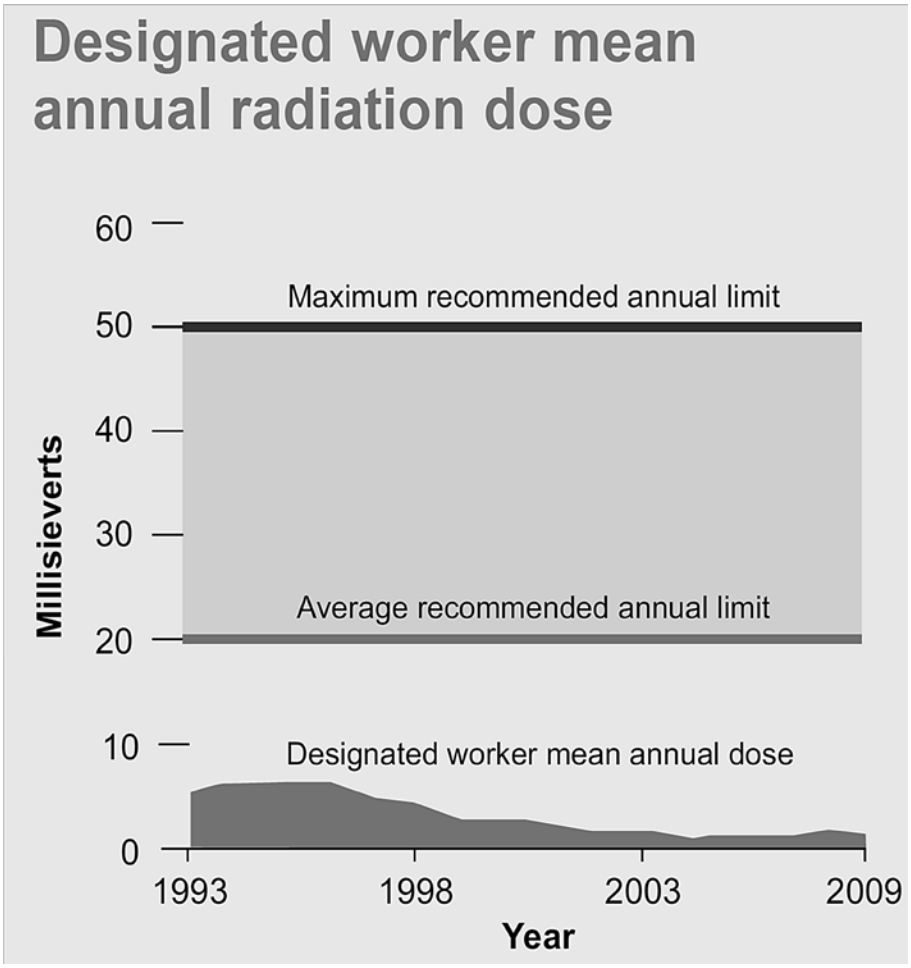
There are hints that the average annual worker individual doses for uranium recovery workers had started to decline somewhat in the late 1980s, as can be seen in Table C.1. However, in a more recent version of the UNSCEAR survey that includes data through 2002 [*UNSCEAR,* 2008], the trend of declining average annual worker individual dose of a uranium recovery worker is very evident (see Figure C.3). In fact, the average annual worker individual dose decreased by 62% from 5 mSv/yr in [*UNSCEAR,* 1993] to 1.9 mSv/yr in [*UNSCEAR,* 2008]. This trend is verified by more recent measurements at specific sites such as the Ranger uranium mine in Australia, as shown in Figure C.4 [*Guilfoyle & Martin,* 2010]. This trend is driven by

regulatory enforcement that has become more stringent over the years and more vigorous application of the as-low-as-reasonably-achievable (ALARA) radiation protection philosophy.



Notes: Adapted from [*UNSCEAR*, 2008]

Figure C.3 | Trend in Average Individual Worker Dose in Global U-Recovery Operations

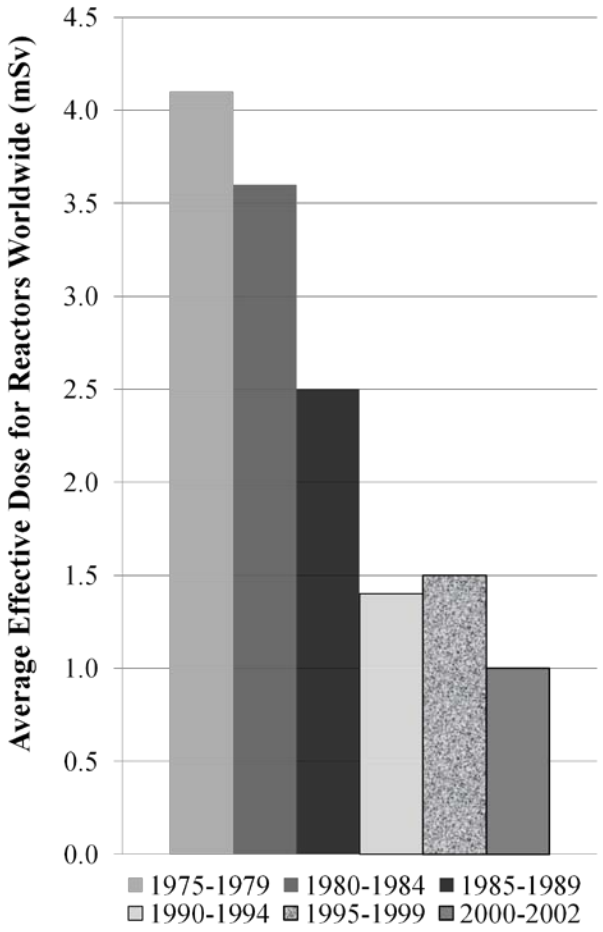


Source: [*Guilfoyle & Martin*, 2010]

Figure C.4 | Average Annual Worker Individual Dose at the Ranger Uranium Mine

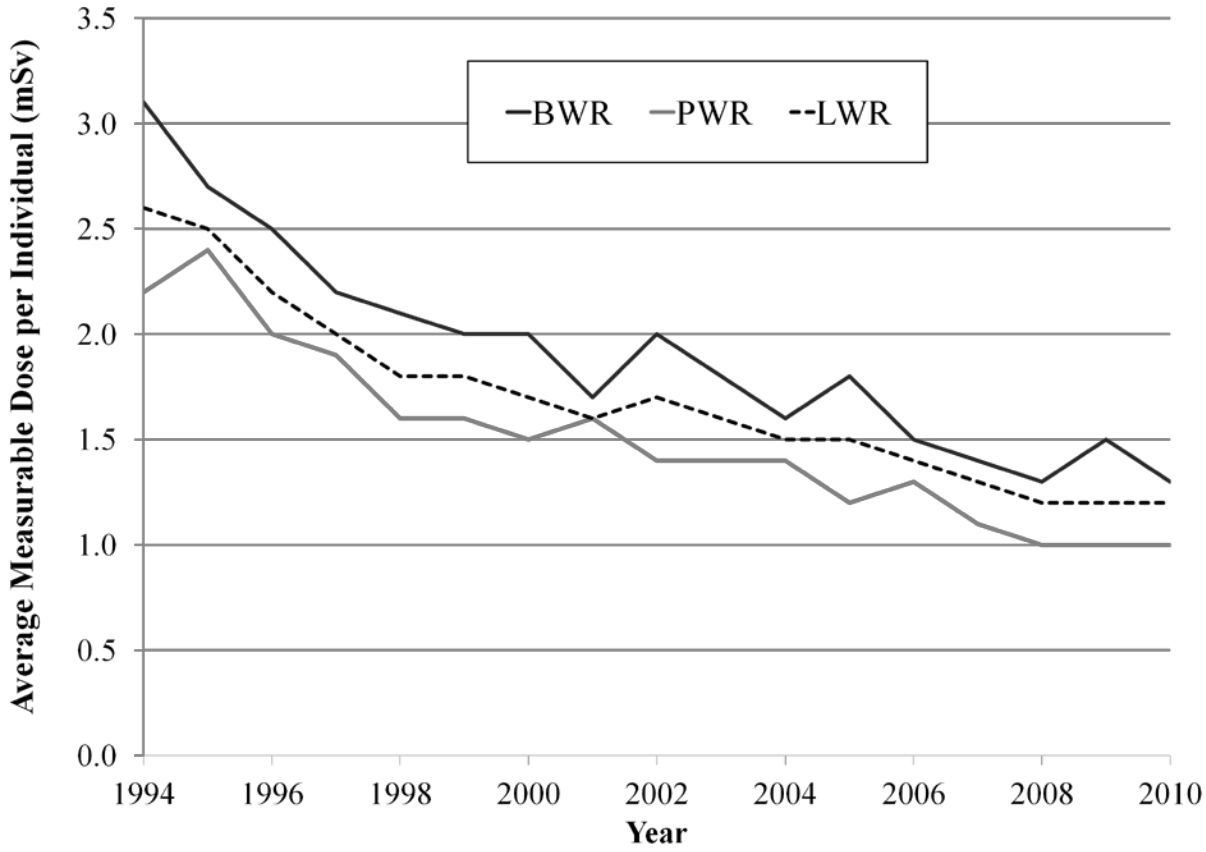
C.2.2. Reactor Operations

As with uranium mining, Table C.2 also hints at a trend of declining average annual dose for reactor operations. In fact, this has continued to be the case (Figure C.5), with the average annual dose decreasing from an average of 3.4 mSv/yr in the 1980s to 1 mSv/yr (Table C.1) in the most recent time period in [*UNSCEAR,* 2008], which is a reduction of 71%. This trend is verified by more recent average reactor worker annual dose data from the U.S. reactor fleet shown in Figure C.6, including BWRs, PWRs, and consideration of both as LWRs [*NRC,* 2012b]. This decline is driven by ALARA considerations and occurs through better plant design to reduce radiation sources and releases, and better work planning to reduce exposures during activities such as maintenance.



Notes: Adapted from [*UNSCEAR*, 2008]

Figure C.5 | Trend in Average Individual Worker Dose in Global Reactor Operations



Notes: Adapted from [*NRC,* 2012b]
Figure C.6 | Average Annual Worker Individual Dose at U.S. Reactors

C.2.3. Summary

Average annual doses to workers in both uranium recovery and reactor operations have decreased substantially since the 1980s. However, the magnitude of the decrease is about the same for both operations and cannot account for the substantial change in the relative contribution of these operations to annual worker collective dose from the NFC. In fact, based only on average annual worker dose (i.e., with no change in the exposed population), the relative contribution of uranium mining to annual collective dose would be increased slightly because the

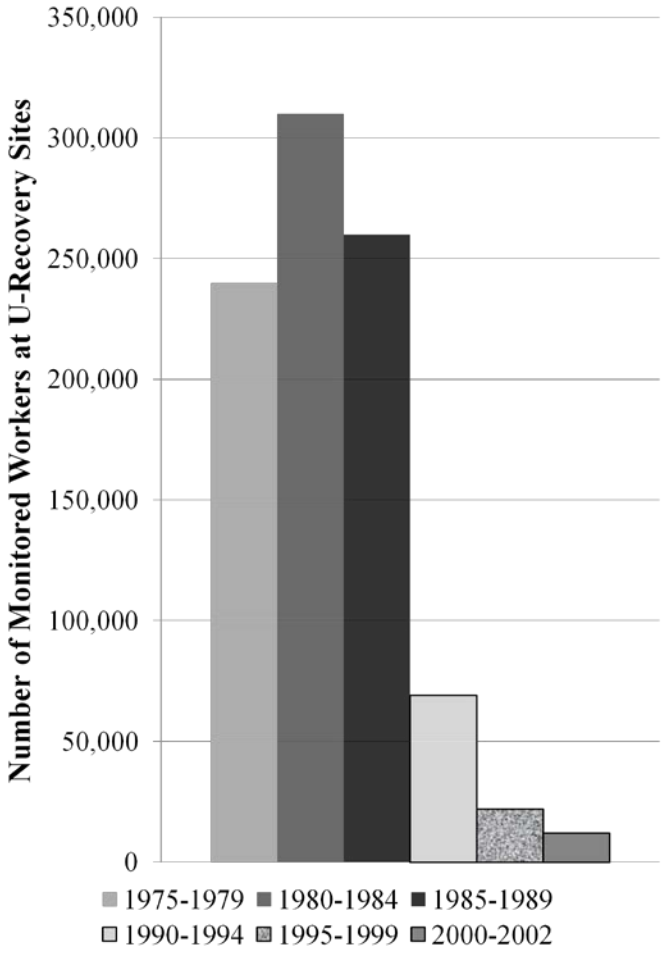
average annual dose to reactor workers decreases by about 4-fold whereas the average annual dose to uranium recovery workers decreases by a factor of 2.6.

C.3. Trends in Number of Monitored Workers at U-Recovery and Reactor Operations

The total annual worker collective dose from a NFC operation is also directly proportional to the number of workers exposed to the average annual worker individual dose. As with the individual worker dose, to the extent that the number of exposed uranium recovery workers has changed over time relative to the number of exposed reactor workers, the contribution of these two operations to total worker collective dose will vary. These trends will be examined in the remainder of this section.

C.3.1. Uranium Recovery

As shown in Figure C.7 [*UNSCEAR*, 2008], the number of global uranium recovery workers has declined precipitously since the 1980s. Specifically, the average number of uranium mining workers in the 1980s was 270,000 but was only 12,000 in the 2000–2002 period, which represents a reduction of ~96% (~26x). The small workforce shown in the early 2000s is the result of a combination of factors, some temporary, some enduring, and some unlikely to reoccur. These factors are discussed in the following sections.



Notes: Adapted from [*UNSCEAR*, 2008]

Figure C.7 | Trend in Global Number of Monitored Workers at U-Recovery Sites

C.3.1.1. Supply-Demand Disequilibrium

Uranium is produced and sold as a commodity in an open market. Such markets are often not "perfect," especially on a short-to-midterm basis, in economic terms[41]. Market imperfection can

[41] Market perfection defined in economic terms, occurs when a number of conditions are met, collectively referred to as "perfect competition." The main five conditions are: (1) rivals, or sellers, act independently, (2) a sufficient number of sellers must exist to inhibit monopolization, (3) sellers have perfect knowledge of market opportunities, (4) freedom from social restraints (e.g., governments) must be present and no externalities exist that prevent equal access to entering the market, resources, and knowledge, and (5) sufficient time must elapse for the flow of goods to move in intended directions [*Stigler*, 1957; *Debreu*, 1972].

result in disequilibrium, i.e., decoupling supply and demand, at times producing surpluses and uranium has been no exception [*Macusani,* 2011].

From the inception of using nuclear technology in 1945 until about 1980, uranium production substantially exceeded uranium consumption because a number of countries were stockpiling uranium for future defense applications[42], which distorted the subsequent uranium market [*NEA,* 2006]. This situation changed in the 1980s when defense programs began to release their surplus uranium stockpiles into the commercial market [*NEA,* 2006]. Thus, while commercial demand was being met, the demand for mining uranium plummeted to less than half of the peak demand in about 1990. In about 1990, the trend again reversed to a degree because the surplus had been used. Growth in production of uranium from mines then began to roughly parallel the growth in commercial nuclear energy but at a level greatly below demand. The cause of the difference between demand and new uranium produced from mines is discussed in the next section.

C.3.1.2. Influence of Secondary Uranium Sources

Even during the downturn in traditional uranium mining during the 1990s, uranium began to be produced in large quantities from what can be termed ‘‘secondary sources.’’ Our study distinguishes between two general types of secondary sources: by-product and recycle. For the purposes of this study, the following definitions are provided:

[42] The amount of uranium used for defense purposes is not readily available

1. “By-product” uranium is produced by activities such as being recovered as part of milling/refining other ores (e.g., gold, copper, phosphate), leaching uranium from uranium tailings piles (i.e., heap leaching), and mine recovery water.
2. “Recycle” uranium is that obtained from activities such as downblending HEU from dismantled nuclear weapons, recycle of uranium recovered from reprocessing, and re-enrichment of high-assay DUF_6 tails.

By-product uranium has been produced in large quantities because its cost tends to be low as a result of the cost of ore mining and beneficiation being partially or wholly allocated to other mining products. By-product uranium has provided from 8% to 14% of uranium production for the last two decades [*OECD,* 2003, 2007, 2011; *NEA,* 2006] and the expectation is that this will continue into the foreseeable future.

The contribution of recycle uranium to the uranium supply is a somewhat newer phenomenon than byproduct uranium recovery, starting in about 1991 [*NEA,* 2006]. The importance of recycle uranium results from a confluence of events: international policy decisions to reduce HEU stockpiles, gas centrifuge technology that can economically re-enrich tails produced from the original enrichment process, and reprocessing of spent nuclear fuel yielding a uranium product having about the same U-235 concentration as natural uranium, but which does not require mining or milling. Recycled uranium accounted for ~35%, 26% (estimate), and 20% (estimate) of the total uranium supply (total = mined + byproduct + recycled) in 2004, 2013, and 2028 [*Resource Investor,* 2006]. The absolute amount of recycle uranium put into the supply chain is estimated to remain about constant in absolute terms for the foreseeable future, but it constitutes

a smaller part of increased amounts of uranium from mines into the future. The extent to which recycled uranium will continue to contribute to the uranium supply is uncertain because there are complex noneconomic, international, and domestic policy factors involved (see additional discussion in [*Cornell*, 2005]).

C.3.1.3. Combined Effects of Uranium Supply-Demand Disequilibrium and Secondary Sources

The combined effects of releasing the pre-1980 uranium surpluses from defense stockpiles, the increased production of by-product uranium, and the use of recycled uranium since the 1990s has decreased the production of uranium from mines; it has gone from providing over 2.5 times the amount of uranium required to fuel reactors to only about 0.5 times the annual fuel loadings [43] [*NEA*, 2006].

At the beginning of Section C.3.1., it was noted that the number of uranium recovery workers declined by 26-fold from its peak in about 1980–2002. Within the NEA (2006), it was reported that the decline in production of newly mined uranium accounts for about 20 percent of the 26-fold leaving about another 5-fold to be accounted for. The remaining 5-fold reduction in workers involved in uranium mining will be accounted for in the next section.

[43] NEA (2006) also considers substituting plutonium-uranium mixed oxide fuel for ~2% of low-enriched uranium (LEU) oxide fuel.

C.3.1.4. Worker Productivity

The number of global uranium recovery workers required should be roughly proportional to the amount of uranium the mines must produce, all other things being equal. However, all other things are not equal: substantial amounts of uranium have been produced for five decades – much of it outside full open market influences – and the technology used to produce it – as well as other mined commodities – has not been stagnant during this interval. At the beginning of the nuclear age, labor intensive methods (with accompanying high radiation annual doses) were used to mine varying grades of ore (e.g., pick-and shovel, hand-sorting of ores) to provide uranium for defense purposes [*NEA,* 2006; *Power,* 2008a] with the dominant source being underground mines [*OECD,* 2003, 2007, 2011; *NEA,* 2006; *Power,* 2008a]. As time passed, mining technology, including that related to uranium, evolved along the following lines to improve substantially worker productivity:

1. *Improved exploration techniques:* Early exploration focused mostly on surface outcrops of desirable ores, and then the uranium recovery workers followed the higher-grade ores wherever they led, mostly underground. Now, the ore body is extensively mapped, and efficient approaches to extracting the ore using any technology is planned beforehand [*Uranium Producers,* 2008]
2. *Use of mechanized equipment:* From the pick-and shovel, uranium ore extraction has progressed to smaller scale excavators, drills, and haulers for underground mining, and massive excavating and hauling equipment for open pit mines [*Power,* 2008a]. Based on experience in the coal industry [*EIA,* 2014], the miner productivity for open pit mines can

be estimated as approximately 3.3x that for underground mines (as well as offering more operational flexibility and safety) [*NAS,* 2002], and this efficiency accounted for a trend away from underground mines [*OECD,* 2003, 2007, 2011; *NEA,* 2006] to open pit mining technology. In addition, worker productivity from both technologies has continued to improve in absolute terms, as can be seen in Table C.4. For example, uranium worker productivity improvement in Niger, which has both open pit and underground mines have increased 1.5x over the past 20 years [*Capus et al.,* 2004].

Table C.4 | Labor Productivity in Uranium Mining and Milling

Type of Mining/ Milling	Year	Labor Productivity [kg / worker–yr]	Productivity Compared to 1970-1980
Conventional	1970-1980 (average)	7,716	100 %
Conventional	1993	8,818	114 %
In Situ Leach, Wyoming	2006	32,628	423 %
In Situ Leach, DOE estimate	1993	44,092	571 %
In Situ Leach, NRC estimate	2007	52,911	686 %
In Situ Leach, Hydro Resources Inc. estimate	1997	29,321	380 %

Source: [*Power,* 2008a]

3. *In situ leach technology:* The use of in situ technology for uranium recovery has increased from several percent of worldwide uranium production to ~40% at present [*OECD,* 2003, 2007, 2011; *NEA,* 2006]. This technology has 4 to 7 times better worker productivity than older conventional (open pit and underground) technology [*Resource Investor,* 2006; *Power,* 2008b].
4. *By-product technology:* Uranium recovery as a byproduct of other mineral extraction and milling operations should have relative worker productivity at least as good as in situ leach as compared to conventional technologies because it avoids both mining and part of the milling operation.

5. *Remote ore mining:* Relatively recently, some uranium ores have been recovered using remote technologies. The concept is similar to in situ leach except that the ore is pulverized in the subsurface and brought to the surface as slurry instead of the return fluid being an aqueous uranium stream. This technology is being used on very-high-grade ores such as those at McClean Lake in Canada [*NEA*, 2006; *Maschinen*, 2011] where radiation hazards from conventional subsurface approaches would be substantial. Such technology allowed Canada to account for 30% of the world's uranium production with only ~2% of the world's uranium workers [*NEA*, 2006].

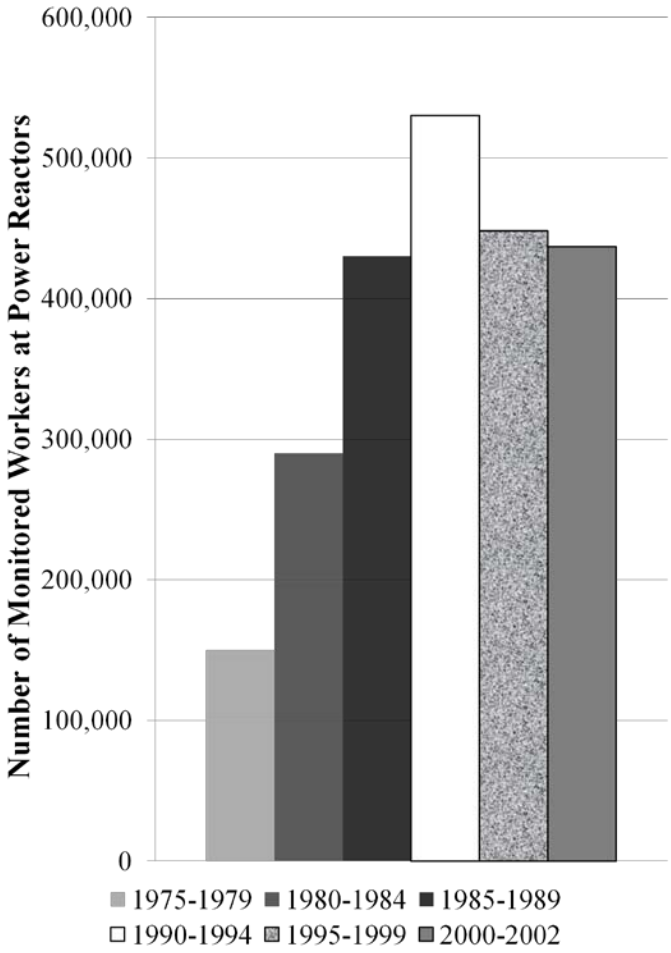
The views of one expert on the economics of uranium production offer the following perspectives concerning worker productivity: "Each technological innovation has allowed both the processing of lower grade ores and the reduction of labor requirements per unit of production. That is, labor productivity in uranium mining has steadily and dramatically increased . . . Between 1980 and the early 1990s the output per employee in uranium mining tripled . . . primarily due to the introduction of in situ leach techniques . . . Over the next several decades new technological developments will reduce the number of workers needed for each million pounds of uranium produced even further [*Power*, 2008a]."

In summary, the remaining 5-fold reduction in the number of uranium production workers can be accounted for by a combination of changing production technologies (i.e., by transitioning a substantial fraction of uranium production from underground mining to in situ leach and by-product recovery technologies), continuous improvements in worker productivity for all

technologies, and the use of remote technologies for some very productive mines containing high-grade ores. There is no evident reason to believe that the trend will change.

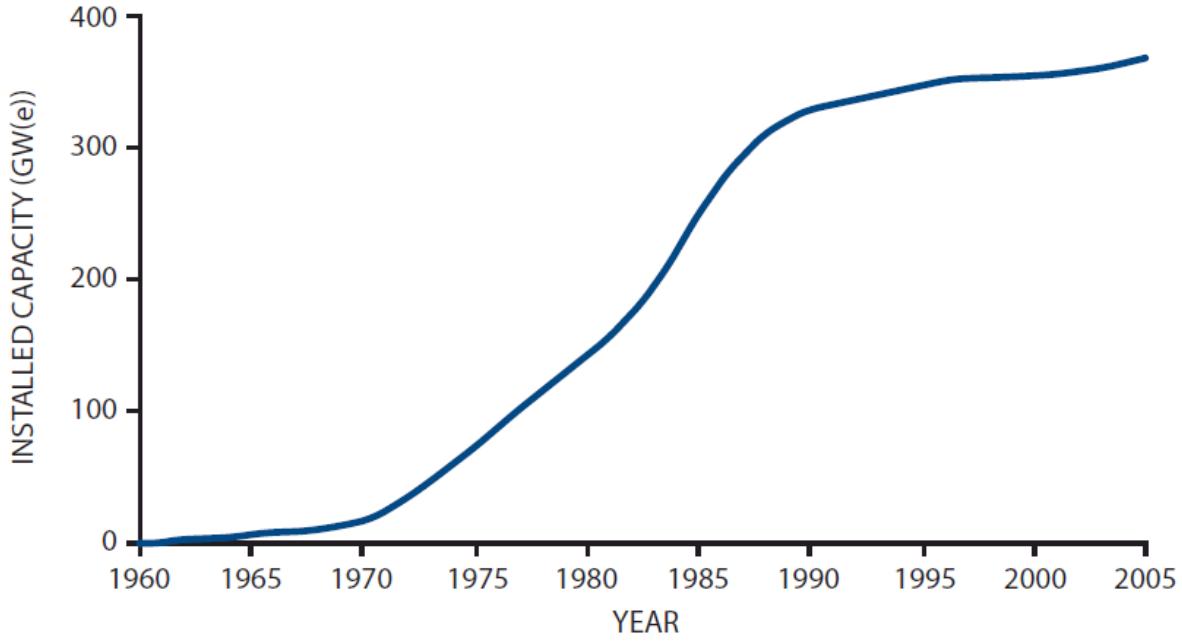
C.3.2. Reactor Operations

Data concerning the global workforce at power reactors reveal a trend very different from that of the uranium recovery workforce. As shown in Figure C.8, growth from an average of 290,000 workers in the 1980s to a peak of 530,000 in the early 1990s and then declining slightly to 437,000 in the early 2000s. To a large extent the reactor workforce trends seen in Figure C.8 reflect the global growth of nuclear power, as shown in Figure C.9 and Figure C.10 [*UNSCEAR,* 2008; *WNA,* 2012d]. The reactor workforce trend in recent years verified by the U.S. reactor workforce trends shown in Figure C.11 [*NRC,* 2012b] is essentially unchanging.



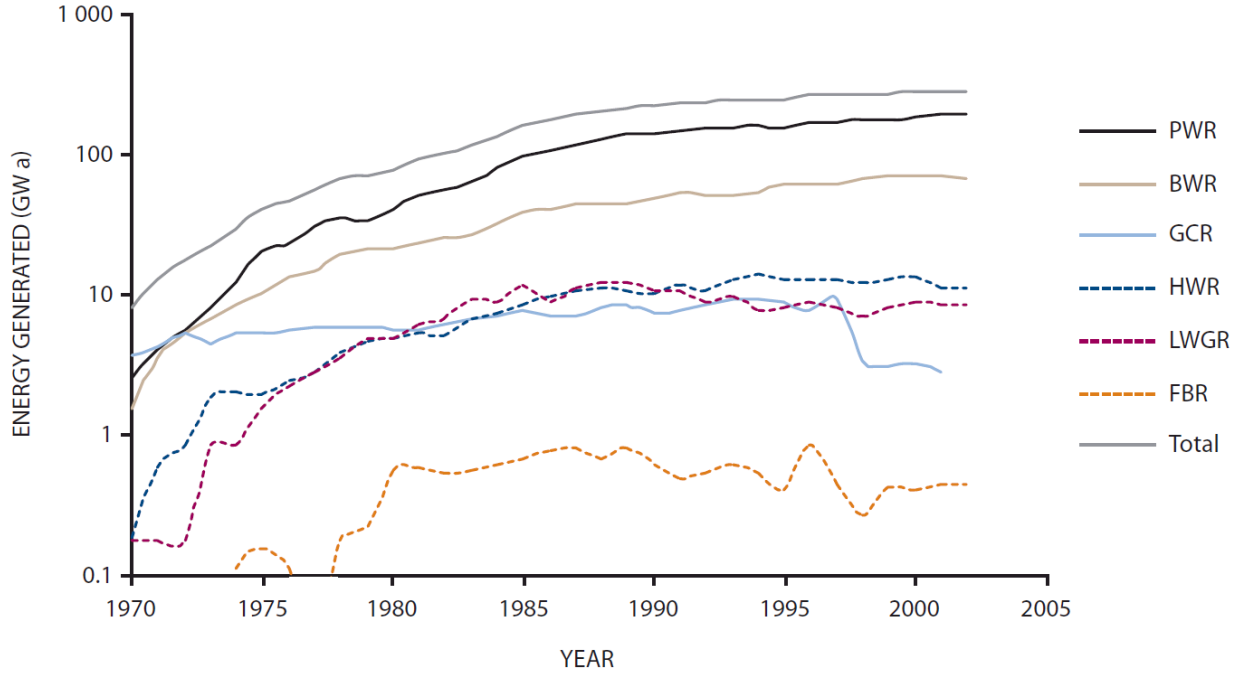
Notes: Adapted from [*UNSCEAR*, 2008]

Figure C.8 | Trend in Global Number of Monitored Workers at Reactors



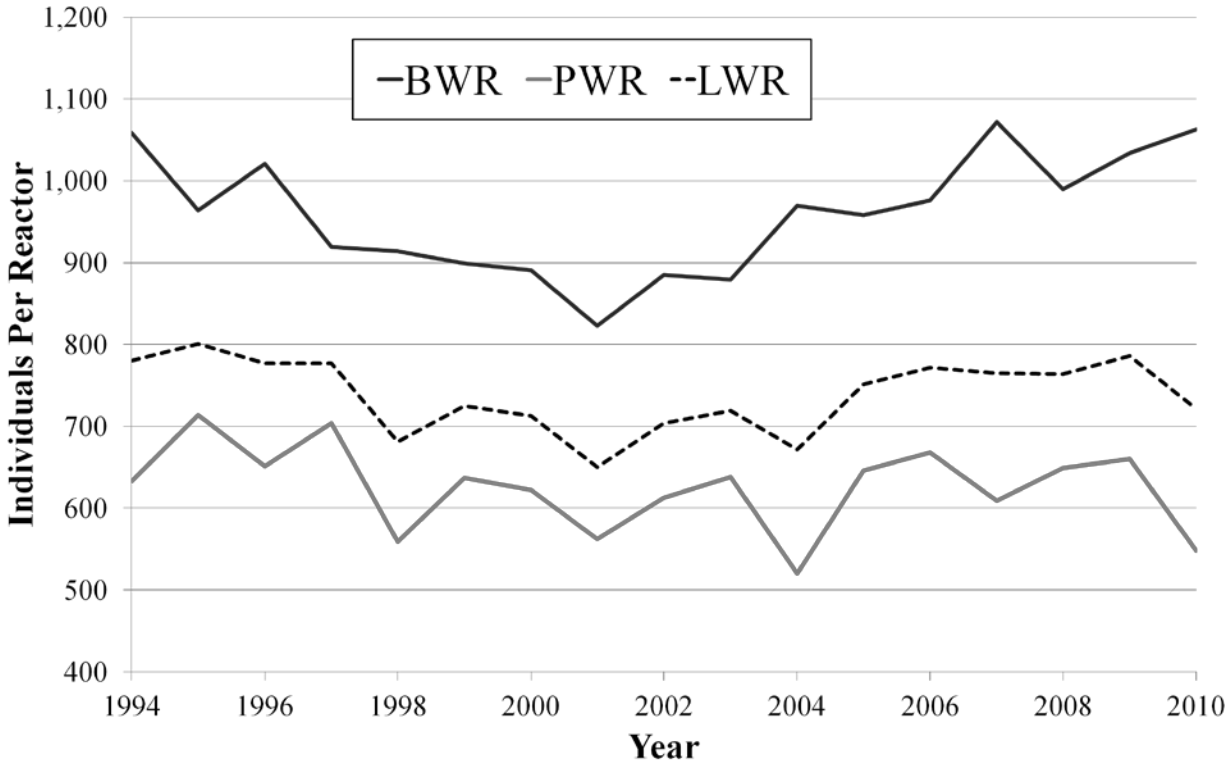
Source: [*UNSCEAR*, 2008]

Figure C.9 | Global Nuclear Energy Installed Capacity



Notes: Most of the electrical energy (90%) from 1995 to current day is produced from PWRs and BWRs. GCR = gas cooled and graphite-moderated reactor; HWR = heavy water-cooled and heavy water-moderated reactor; LWGR = light water-cooled graphite-moderated reactors; FBR = fast breeder reactor. **Source:** [*UNSCEAR*, 2008]

Figure C.10 | Global Nuclear Electrical Energy Production



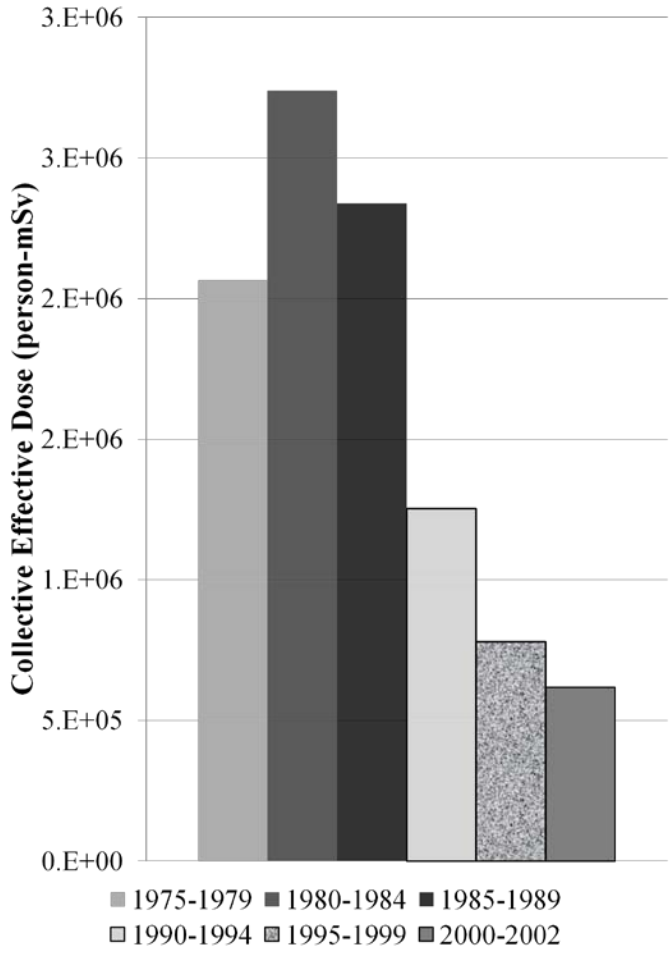
Notes: Adapted from [*NRC*, 2012b]
Figure C.11 | Average Workforce at U.S. Reactors

C.3.3. Summary

Data on the number of uranium recovery and power reactor operations workers reveal substantially different trends. The number of uranium recovery workers has declined by 20-fold while at the same time the number of reactor operations workers increased during the 1990s due primarily to the global growth of nuclear power and has remained essentially constant in the 2000s [*UNSCEAR*, 1993]. The 20-fold decline in the global uranium recovery workforce is partially an artifact of downblending of HEU and a supply-demand oscillation in the uranium market. Smoothing out the oscillation would still yield a uranium recovery workforce decline of about 10-fold (see below).

C.4. Contributors to Annual Worker Collective Dose to Workers from NFC Operations

Based on the information in Sections C.2 and C.3, it is clear that there have been substantial changes in some of the factors that dominate occupational annual collective dose from NFC operations since the 1980s. In particular, average annual worker dose from uranium recovery and reactor operations, and the uranium recovery workforce have all declined substantially while the number of reactor workers peaked and has then declined slightly. Taken together, this has resulted in a dramatic reduction in the total annual collective dose to workers, as shown in Figure C.12 [*UNSCEAR,* 2008]. In and of itself, declining annual collective dose to workers in the face of increasing production of nuclear power is a noteworthy and positive trend.



Notes: Adapted from [*UNSCEAR*, 2008]; Excludes worker collective doses from reprocessing and research operations

Figure C.12 | Trend in Global NFC Worker Collective Dose

The extent to which changes in the relative contribution of annual worker collective dose from uranium recovery and reactor operations can be estimated by using information developed in previous sections (C.2 and C.3). For uranium recovery the average annual worker individual dose has declined from 5 to 1.9 mSv/yr, and the exposed population has declined from 270,000 workers to 24,000 workers[44], which leads to the recent uranium recovery annual worker collective dose being 3.4% of that in the 1980s. For reactor operations, the average annual

[44] Data for the recent number of uranium recovery workers in [*UNSCEAR*, 2008] have been doubled to account for the effects of oscillations resulting from uranium supply-demand disequilibrium as discussed in Section C.3.1.

worker individual dose declined from 3.4 to 1.0 mSv/yr, but the exposed population increased from 290,000 to 437 000, leading to the recent annual collective dose from reactor operations being 44.3% of that in the 1980s. Normalizing these two percentages to the ~92% of the total annual collective dose of the OTC NFC that they comprise[45] leads to an estimate of uranium recovery now accounting for 6.2% of annual worker collective dose and reactor operations accounting for 85.8% of annual worker collective dose, shown in Table C.5.

Table C.5 | Comparison of Historical to Recent Major Contributors to Total Annual Worker Collective Dose

OTC Operation	VU Model Estimations 2018 Data	This Analysis	[*UNSCEAR*, 1993] Report[a] 1975-1989 Data
Uranium Recovery	8.8 %	6.2 %	56.1 %
Reactor Operations	83.5 %	85.8 %	37.5 %
Total	92.3%	92.0 %	93.6 %

Notes: [a] The average of the three time periods from Table C.2 is presented here.

The relative contributions of annual collective dose from uranium recovery and reactor operations to total annual collective dose from the once-through NFC developed in this analysis are clearly different from what they were prior to the1990s and compare favorably to recent predictions of the VU-EPRI model.

C.5. Potential Implications

From the overall perspective of producing nuclear energy, the result that is most important is that the annual collective dose to workers incurred in doing so has declined substantially even though the production of nuclear energy has increased. However, there is another perspective: that of

[45] Collective dose to workers from NFC operations other than uranium recovery and reactors also declined in the intervening years so that it constituted ~8% of the total in 2002 [*UNSCEAR*, 2008].

needing to evaluate which NFC concepts and technologies should be researched, developed, demonstrated, and deployed. For example, the DOE's Office of Nuclear Energy (DOE-NE) has a program under way to evaluate and screen nuclear energy system concepts [*DOE,* 2011a] to inform its decisions on which NFC concepts should be the focus of its RD&D efforts[46]. This is to be followed by evaluation and screening of technologies that could be used to implement the preferred NFC concepts to inform DOE-NE's decisions at this level. In addition, as described at the beginning of this paper, EPRI has a similar effort under way [*EPRI,* 2011, 2012d, 2013b]. For applications such as these, the relative contribution of various NFC operations to worker impacts, such as annual collective dose is relevant. As an example, consider a conventional once-through LWR using LEU fuel and a fast reactor using fuel composed of recycled Pu recovered from reprocessing and depleted uranium. Under the "conventional wisdom," where the impacts of uranium recovery dominated, the use of fast reactors and recycle drastically reduced uranium recovery requirements and, thus, collective annual doses from the front end of the NFC, which is favorable for the above fast reactor scenario even considering the additional annual collective worker doses from reprocessing (e.g., see Machiels (1992)). However, using the new results (discussed in Smith et al. [2013a] and this section) where the impacts of uranium recovery are relatively smaller, using fast reactors and reprocessing to reduce uranium recovery annual dose further does not offer as much of an advantage. Such a paradigm shift would tend to increase the preference for fast reactor fuel cycle, all other things being equal.

[46] The trend analyzed in this section was initially identified by the author during the ES&H evaluation associated with a concurrent effort DOE-NE effort to evaluate nuclear fuel cycle options to inform R&D planning [*Wigeland et al.,* 2014].

C.6. Conclusions

Analysis of recent information concerning annual worker collective doses from the NFC reverses the relative importance of uranium recovery, and the front end of the NFC, as compared to reactor operations to the point that worker impacts as measured by annual collective dose are dominated by nuclear reactors. This result is primarily driven by a substantial reduction in the global number of uranium recovery workers since 1980, despite increased uranium demand, a trend that does not seem likely to be reversed in the foreseeable future.