Comparison of Radioactive Waste Volumes from Single Used Nuclear Fuel Recycling and the Once-Through Nuclear Fuel Cycle

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Evaluation of competing advanced nuclear technologies is useful to inform decisions concerning the prioritization of finite resources available for research, development and demonstration. Advanced nuclear technologies and nuclear fuel cycle (NFC) options offer improvements that are only fully realized when deployed and running at steady-state: however, such NFCs will not be implemented instantaneously. A potential transition from the current U.S. once-through NFC (OTC) to a modified-open NFC (MOC) involving a single recycle of plutonium and use of enriched reprocessed uranium has been modeled for a simplified energy demand scenario. The most limiting modeling constraint is the rate at which reprocessing capacity is brought into operation, which dictates the extent to which fuel can be recycled. This approach is different from many previous dynamic modeling efforts because the reality that reprocessing capacity will not be unlimited is addressed explicitly. Volumes of radioactive waste are estimated and tracked for each operation of the OTC and MOC for waste that can be disposed near the surface of the earth and waste that is assumed to require disposal within a geological repository. The volumes of these categories of waste generated by the MOC were 14% and 25% smaller than from the OTC, respectively, over the 50-year simulation.

I. PROJECT BACKGROUND & MOTIVATION

The Electric Power Research Institute (EPRI) is building tools and capacity for assessing NFC options based on a platform of software, simplified relationships, and decision-making and evaluation guidelines [1]. This paper describes the results to date on the development of EPRI's comparative risk assessment tool and its application for evaluating the environmental, health and safety (EH&S) related consequences of NFC choices. Alignment of NFC options with safety criteria and metrics, including EH&S effects, represents an important input for technology assessments and decision-making. For this phase of tool development and demonstration, radioactive waste volumes serve as a proxy for environmental/waste management impacts. Waste volumes were estimated separately for wastes that can be disposed of on or near the earth's surface and wastes that require deep geologic disposal to better approximate the risk posed by the wastes. While volume is an imperfect measure of risk, its use is driven by operational relevance and availability of supporting data. Other metrics are considered in several previous EPRI reports [1, 2].

The objective of this study is to compare the radioactive waste volumes estimated to be generated from the MOC to those generated by the OTC on an annual basis, and then integrated across a 50-year modeling time frame (2013-2063). This time period makes it possible to evaluate the magnitude and timing of substantial net differences in radioactive waste production that could occur during the realistic implementation schedule modeled in this study.

II. METHODOLOGY

Radioactive waste volumes for the OTC and MOC were estimated in three phases: (1) develop the conceptual systems model (as shown in Fig. 1); (2) develop normalized radioactive waste production rates (e.g., m^3 waste per metric tons (MT) facility throughput or GWe-yr generated) that are based on data from recent industry experience; and (3) calculate waste production rates (normalized production multiplied by throughput or production). Waste production rates are used to compare performance of the MOC to the OTC; waste volume is one attribute (among many, including radionuclide content) that are important to waste storage and disposal operations and decision making.

II.A. MOC Conceptual Model

Previously, the conceptual model of the OTC was updated in an EPRI-sponsored Vanderbilt study to include four new fuel cycle operations [2]. The MOC (Fig. 1) adds operations that enable a single-time reuse of the plutonium (Pu) and reprocessed uranium (RepU) as mixed-oxide (MOX) and enriched RepU (ERU), respectively. The MOC includes reprocessing, MOX fuel fabrication, RepU conversion and enrichment, and ERU fuel fabrication. The feed of used nuclear fuel (UNF) to reprocessing is assumed to be withdrawn from the dry interim storage located at operating reactor sites.

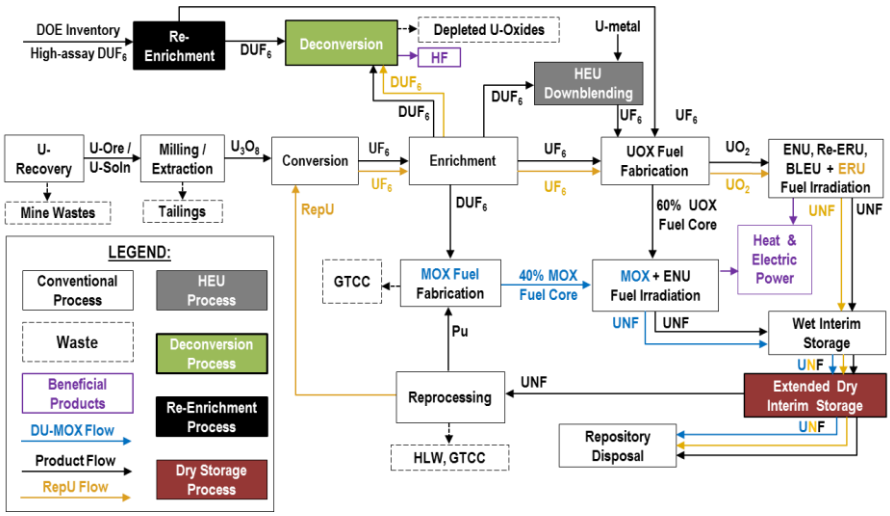

Figure 1. The MOC Conceptual Systems Model

II.B. MOC Nuclear Fuel Material Flow Analysis

Quantitative analysis of fuel cycle mass flows was performed with five modeling steps focusing on: nuclear fuel supply, depleted uranium (DU) management and UNF dry interim storage activities at reactor sites, as follows:

1. Establish a simplified nuclear energy demand scenario for the 50-year simulation time period (2013 to 2063) by assuming a 1% growth of nuclear energy demand per year;
2. Distribute the annual fuel requirements for the light water reactor (LWR) fleet among the following distinct supply sources:
   a. **BLEU:** Blended low enrichment uranium fuel, a product from blending highly-enriched uranium (HEU) and DU;
   b. **Re-ENU:** Re-enriched natural uranium, obtained from higher assay DU enrichment tails that are passed through the enrichment stage again to produce low enrichment fuel;
   c. **ENU:** Enriched natural uranium, obtained by conventional means involving uranium mining, milling, conversion, enrichment, and fuel fabrication;
   d. **MOX:** Mixed-oxide obtained by mixing Pu separated from UNF in the reprocessing MOC operation with DU resultant from the enrichment phase; and
   e. **ERU:** Enriched RepU that is separated from UNF in the reprocessing MOC phase and passes through the enrichment stage again to produce low enrichment fuel made from RepU
3. Model deconversion and waste management of the streams from enriching high-assay tails, natural uranium, and RepU that contain low concentrations of U-235, while accounting for depletion of the DOE low-assay (for MOX fuel) and high-assay (for re-enrichment) DU stocks;
4. Calculate the back-end independent spent fuel storage installations (ISFSIs) expansion, maintenance, and required withdrawals for reprocessing at existing reactor sites; and
5. Scale normalized performance metrics to calculate radioactive waste volumes associated with the MOC to compare with volumes from the OTC in [2]).

The extent of fuel type use was assumed to be: (1) 10% of LWRs fuel requirements (BLEU); (2) as much Re-ENU, MOX, and ERU fuel as NFC facilities can produce annually; and (3) any remaining fuel requirements are met with ENU. In this analysis, MOX and ERU are only used in PWRs based on the more extensive experience with MOX in PWRs and the French PWR-based experience with ERU; also, the rate at which reprocessing was introduced did not require introduction of MOX into BWRs [3, 4]. The calculation is dynamic, in that reactors are only converted to recycled fuel as reprocessing plants produce sufficient feed for the associated MOX and ERU fuel fabrication plants to provide a full core reload for ERU and for 40% of a full core for MOX; Furthermore, reactors are converted to MOX or ERU use only when fresh fuel feedstock backlogs are sufficient to maintain the reactors throughout their life cycle. Thus, the incremental implementation of reprocessing, coupled with the need to demonstrate sufficient fuel availability, is intended to provide a more realistic scenario for a potential transition to recycle of UNF.

The present-day U.S. reactor fleet of ~100 LWR was modeled as 2/3 PWRs and 1/3 BWRs. Many operating

parameters were assumed to be the same for both: capacity factor (88%), thermal efficiency (33.7%), electricity generation capacity per reactor (1000 MWe/year), and cycle length for each fuel batch (18 months). The BWR and PWR parameters assumed to be different are listed in Table 1. The eligible fuel types for PWRs and BWRs are listed (as the distinction of MOX and ERU use is explained later). Table 2 lists the enrichments and mass contents of the major element of interest and the U-235 initial enrichment is listed. PWRs that use MOX are loaded with MOX and ENU (described later) and the ENU used is the PWR-type ENU fuel.

Table 1: BWR and PWR Input Parameters

Reactor Type	Eligible Fuel Type	Number of Fuel Batches	Burnup ($GWd_{thermal}$ /MTHM)
PWR	ENU (PWR) , BLEU, Re-ENU, MOX, ERU	3	55
BWR	ENU (BWR), BLEU, Re-ENU	4	50

Table 2: Fuel Input Parameters

Fuel Type	U-235 Enrichment (wt%)	Total U Content (wt%)	Pu Content (wt%)
ENU (PWR)	4.5	100	0
ENU (BWR)	4.35	100	0
BLEU	4.5	100	0
Re-ENU	4.5	100	0
MOX	0.25	90	10
ERU	5.0	100	0

II.C. Reprocessing and Other Recycling Operations

The newest reprocessing facility is the Rokkasho Reprocessing Plant in Japan that combines relatively pure RepU and Pu streams for co-denitration prior to MOX fuel fabrication. Consistent with other recent analyses [5, 6], capacity is modeled after the Rokkasho facility of 800 MT UNF/yr. The annual capacity is set at 800 MT UNF/yr and the U-Pu recovery is set as 99.8% consistent with literature values[7, 8]. 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 a recent Nuclear Energy Agency analysis [9].

Only fuel discharged from the start of the scenario (2013) onward is assumed to be available for reprocessing. This produces a fifteen year cooling time for UNF, meaning that the timing of when the first reprocessing facility opens (2028) matches the age of the first UNF to be reprocessed. In this analysis, a one-time recycling scheme of Pu and U is adopted (i.e., used MOX and ERU fuel are not recycled).

Operating MOX fuel fabrication facilities most frequently produce MOX from separated Pu and DU [10]. Based on the 15-year-old UNF used in this analysis, MOX compositions that are modeled herein contain 10 wt% Pu and 90 wt% DU (assumed to be 0.25 wt% U-235) to achieve the assumed 55 GWd/MT burnup [11, 12]. MOX fuel fabrication facilities are assumed to be co-located with the reprocessing facility [13]. 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 [13]. This is the same strategy for the already-built Rokkasho reprocessing facility and its adjacent MOX fuel fabrication facility that is presently under construction [10]. To account for decay losses of Pu-241 when processing 15 year-old fuel, a UNF/MOX 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.

Recycled uranium recovered in reprocessing is reused by enriching the RepU stream by conventional enrichment technology [14, 15]. ERU fuel fabrication involves three steps: (1) conversion, (2) enrichment, and (3) oxide fuel fabrication. The composition of RepU is different from the composition of natural uranium (NU) as a result of the presence of (a) small amounts of U-232, which requires additional radiation shielding; and (b) U-236, which is a neutron absorber and, subsequently necessitates higher U-235 enrichment levels of 5 wt% U-235 (the "U-236 penalty") [16, 17]. Separate conversion and enrichment lines are typically used (due to U-232), consistent with present French practice [15, 18].

It is assumed that currently employed conversion technology is used for converting RepU to $RepUF_6$ with an efficiency of 99.9% and that the annual conversion capacity matches the RepU production from the assumed reprocessing plant capacity. It is further assumed that ERU fuel preparation begins in 2028; however, with fuel qualification and material handling time lags, there is postulated a 5-year delay from the start of reprocessing operations to the use of ERU in PWRs [19].

The ERU fuel fabrication facility in France (Romans) has an annual capacity for ERU fuel fabrication of 150 MTHM ERU/yr [13, 15]. The same facility capacity is adopted for this model with a material recovery of 99.9%. It is assumed that ERU enrichment capacity will match the ERU fuel fabrication capacity (120, 240, 360 MTHM ERU/yr), adding new fuel fabrication capacity as required.

II.D. Waste Management Material Flow Analysis

Waste management material flow analysis tracks DU as DU-hexafluoride (DUF_6) along with its management and uses, the number of dry storage casks (DSCs) of UNF discharged, mass of high level waste (HLW), and the mass of vitrified HLW and GTCC waste from recycling

processes. Several specific aspects of waste modeling are discussed below.

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 the well-known separative work unit (SWU) calculational methodology [20]. DU 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), but if not used in fuel, it is deconverted and stored until final disposal. DU produced internationally is assumed to be deconverted to a more stable oxide form and disposed as low-level radioactive waste [21]. It was assumed that the present Paducah and Portsmouth facilities are allowed to deconvert non-DOE material when excess capacity becomes available.

Dry cask interim storage began in 1986 [22] and expansion continues today collocated at reactors. Each DSC that holds UOX fuel stores 13.05 MT UOX UNF. In this model, UNF is stored in wet interim storage for five years in a spent fuel pool following reactor discharge. Then, UNF moves to dry storage at reactor sites and loaded in DSCs. UNF from ENU, Re-ENU, and BLEU are assumed to be identical isotopically and eligible to feed reprocessing. The 15-year cooled UNF eligible for reprocessing has cooled in the spent fuel pool for five years and then cooled within a DSC for 10 years. The DSC external volume was calculated based on the outer dimensions of a circular cylinder: 190-inches in height; 70-inches in diameter.

It is assumed that ERU UNF can be handled similarly to other UOX fuel [18]. However, decay heat levels from irradiated MOX are higher than those of UOX UNF during the required time period when dry interim storage is used [19]; thus, a lower heavy metal capacity of MOX DSCs is assumed due to the set decay heat limits of licensed DSCs. Areva's experience with casks that are used for dry storage and transport of irradiated MOX is used as the assumed basis for the MOX DCS, which is 17 PWR MOX fuel assemblies [4, 23]. DSC capacity for solely MOX fuel assemblies was estimated to be 7.82 MTHM/DSC.

Reprocessing waste handling is modeled after La Hague which co-locates HLW vitrification and structural metal hull compaction [24, 25]. HLW contains 0.2 wt% of the U and Pu. Vitrification technology at La Hague incorporates a cold crucible induction melter and the average waste loading is 22%. 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 specified glass loading limits [24] and equates to around 0.7 UC-Vs holding vitrified HLW per 1 MT UNF reprocessed. Compacted metal waste is considered GTCC LLW.

II.E. Normalized Performance Metrics

Metrics in this paper are developed with the assumption that the impacts of operations at fuel cycle facilities, in this case volume of radioactive waste, scale linearly with mass flows. All operational phases are normalized by mass flow of heavy metal content, except for the reactor operation, which is normalized by electrical energy production; thus, scaling waste volumes can be done for each major NFC facility.

Table 3: MOC Normalized 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	1.21E-01	m^3 byproduct /MTNU
Milling		
Acid Leach	1.01E+03	m^3 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 DU	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
Dry Interim Storage		
UOX (ENU, Re-ENU, BLEU, ERU)	9.18E-01	m^3 UNF in DSCs/MTUNF (UOX)
MOX	1.57E+00	m^3 UNF in DSCs/MTUNF (MOX)
Reprocessing		
Aqueous – PUREX	6.16E-02	MTHLW/MTUNF
	2.83E-01	MTVitrified HLW/MTUNF
	1.56E-01	m^3 Vitrified HLW/MTUNF
	7.00E-01	Vitrified HLW within a Universal Canister (UC-V)/ MTUNF
	1.90E-01	m^3 GTCC / MTUNF
	8.76E+00	m^3 LLW/ MTUNF
Conversion for RepU		
UN to UF_6	1.03E-02	m^3 LLW / MTRepU
Enrichment of RepU		
Centrifuge	1.36E+00	m^3 LLW / MTERU
Fuel Fabrication		
MOX Fuel Fabrication (Glove-box Technology)	1.68E-01	m^3 GTCC/ MTMOX
	6.21E-01	m^3 LLW / MTMOX
	1.20E-01	m^3 MLLW / MTMOX
ERU Fuel Fabrication (Shielded Handling Technology)	4.91E+01	m^3 LLW / MTERU

The radioactive waste volume metric is quantified by multiplying the normalized waste volume for each facility by the throughput of that facility. The data required to calculate the normalized radioactive waste volume are 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]. The external volumes of the waste packages are used as the calculational basis when waste packages are required (as shown in Table 3).

It is important to note that data sets which have been used for this work represent actual measured volumes of waste and are based upon real events and industry experience, with emphasis on collecting as recent data as possible [26, 27]. The majority of radioactive waste volume performance metrics that were considered related to reprocessing was adopted from recent studies done by Areva and Energy Solutions [26, 27]. Values of waste volumes provided in these two reports vary and engineering judgment has been used to determine the most appropriate values that are to be used in this study. The remaining fuel cycle normalized metrics are not as disparate and details are listed in [1].

II.F. Comparing Impacts: MOC to OTC

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. A side-by-side comparison can be performed and calculated simply by the quotient shown below. The ratio (Equation 1) of steady-state impacts is denoted as R_i, while MOC_i and the OTC_i are the impacts associated with the MOC and the OTC, respectively. The subscript i denotes an individual year of the simulation beginning in 2013 and ending at the year 2063. The R_i represents the radioactive waste volume impact for that individual year.

$$\mathrm{R_i} = \frac{\mathrm{MOC_i}}{\mathrm{OTC_i}} \tag{1}$$

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 needs to be expressed at each discretized time step or the equation shown below (Equation 2) can be altered to integrate the impacts over the modeling time frame. R denotes the integrated (or cumulative) impacts, n is the total number of discretized time steps evaluated from years 2013 to 2063 (there are 51 years when the year 2013 is included in the total time of simulation), i is the individual time step, MOC_i and OTC_i are impacts from each time step.

$$\mathrm{R} = \frac{\sum_{1=\mathrm{i}}^{\mathrm{n=51\ years}} \mathrm{MOC_i}}{\sum_{\mathrm{i=1}}^{\mathrm{n=51\ years}} \mathrm{OTC_i}} \tag{2}$$

III. RESULTS & DISCUSSION

Results from modeling the radioactive waste volumes eligible for shallow-land disposal and waste that is assumed to require deep geological repository disposal, during the transition from the OTC to the MOC, are presented in three parts. The first part includes the material flows of fuel and waste for each year of the simulation. The second part includes results of the analysis of the MOC in terms of radioactive waste volumes, when using the material flows calculated in the first part. Lastly, comparative performances of the MOC to that of the OTC are presented in two forms: (1) relative performance, on a yearly basis, during the 50-year modeling scenario; and (2) the impacts from two NFCs integrated over 50 years.

III.A. MOC Nuclear Fuel Material Flow Analysis

Reprocessing capacity is modeled as a step function addition of 800 MTHM per year and the amount of RepU and separated Pu, MOX and ERU fabricated follows in proportional step increases every ten years from 2028 to 2048: separated Pu was calculated as (10, 20, 30 MT Pu/yr), annual MOX production was modeled as (97, 195, 292 MT MOX/yr), and separated RepU was modeled as (750, 1500, 2250 MT RepU/yr).

The extent of MOX use in PWRs is based on the annual production of MOX (Fig. 2) 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 around the fuel fabrication amounts as inventory becomes available to support annual fueling requirements of already converted PWRs and the 40% core replacement of MOX to initially convert PWRs. The cumulative number of PWRs that are converted to MOX use is shown using purple circles and corresponds to the right-hand Y-axis. 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.

Similar to the MOX scenario, the extent of ERU use in PWRs is based on installed reprocessing capacity; ERU production matches the increasing decadal intervals of reprocessing capacity (Fig. 3). RepU from reprocessing must be enriched. The feed-to-product ratio of RepU to ERU is 5.7 and is calculated by established enrichment

SWU formulae. This means that when inefficiencies are accounted for, the material produced is reduced at a rate of nearly 6 MT of RepU feed and 1 MT is remaining that is ERU. Effectively, each reprocessing plant can support annual ERU refueling of ~7 PWRs.

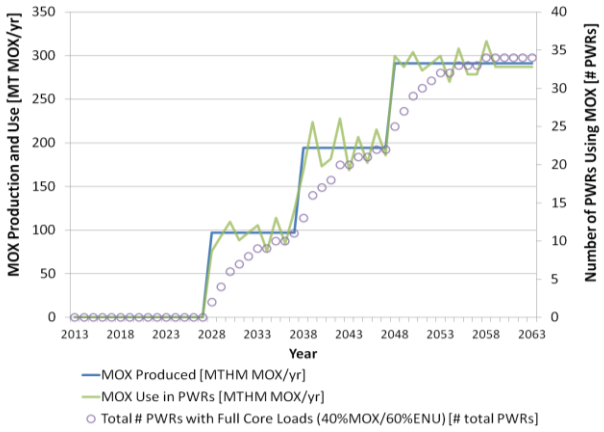

Figure 2. MOX Production and Use in PWRs

The annual use of ERU is influenced by available inventory to support annual fueling requirements of already-converted PWRs and then the assumed 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.

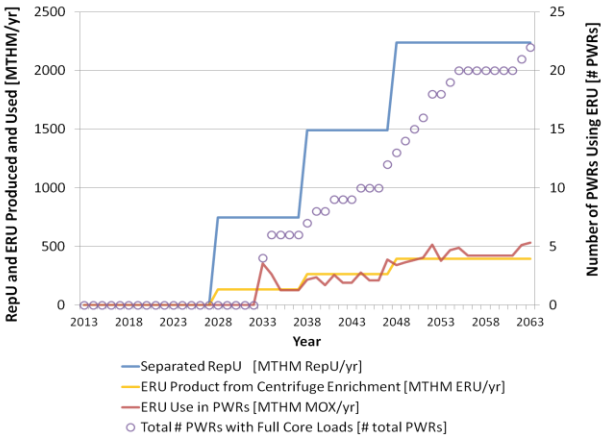

Figure 3. RepU and ERU Production and Use in PWRs

The breakdown of the fuel types loaded into the reactor fleet each year is shown in Fig. 4. The baseline annual fuel requirement if the reactor fleet were fueled regularly with ENU is shown as the black solid line. The purple solid line represents the amount of fuel loaded into the reactor fleet due and deviates slightly from the baseline due to the assumption of full core replacement when converting PWRs to ERU and MOX use. BLEU use stays consistent at 10%, Re-ENU contributes 19% when Paducah was re-enriching higher-assay DUF_6 and lowers to 5% when the NM centrifuge plant was assumed to take responsibility of the remaining higher-assay DUF_6 inventory (as discussed in our prior modeling of the OTC) [2]. In 2028, MOX use begins, shown as the green band in Fig. 4. 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 Fig. 4. 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. Smaller spikes in the orange band appear when new PWRs are converted to ERU.

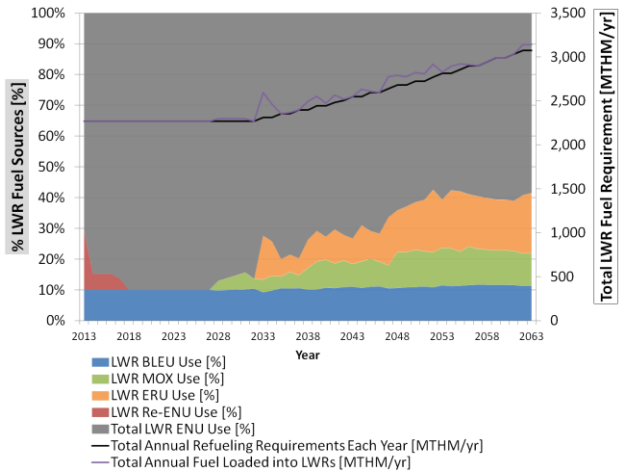

Figure 4. MOC Percent of LWR Loaded Fuel Type and LWR Fuel Loaded and Demanded Each Year

III.B. Shallow-Land Burial-Eligible Radioactive Waste Volume Comparative Results of the MOC to OTC

III.B.1. Annual Basis Comparisons of the MOC to OTC

The total volume of radioactive wastes eligible for shallow-land burial is shown in Fig. 5 for each year of the MOC from 2013 to 2063 (purple line, Fig. 5). The volumes range from 27 million m^3 to 34 million m^3 of wastes considered each year. The lower end of the range is observed when Re-ENU and BLEU replace ~30% and then ~15% of the required ENU fuel, respectively; then after these two alternative sources of fuel are discontinued in the year 2018, the amount of shallow-land burial eligible wastes increases to the upper bound of 34 million m^3 each year, until 2028 when the introduction of MOX fuel occurs and waste volumes decrease. A number of smaller spikes are observed between 2033 and 2054 during periods when LWRs are added to the MOC faster than MOX and ERU can support (due to continued increase in electricity demand) and thus, these reactors must use ENU as the source of fuel. The same is true 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. The annual ratio of MOC to OTC shallow land burial waste volumes are all below 1.0 (bottom graph, Fig. 5) from 2028 to 2063. The annual ratio reaches a minimum of about 0.70, representing a decrease of almost 30% from the shallow land burial-eligible wastes calculated for the OTC scenario. However large this waste avoidance is, it is important to realize that more than 99% of both OTC and MOC shallow land burial-eligible wastes are from mining and milling. By the end of the simulation, the accumulated waste avoided is more 410 million m^3 of waste.

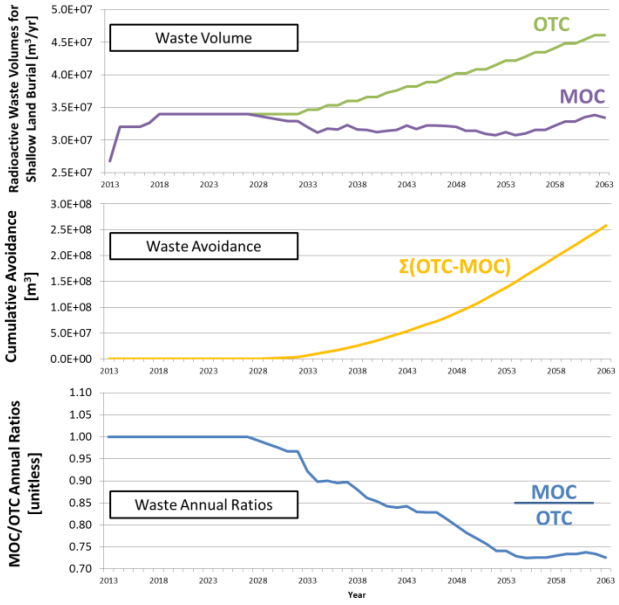

Figure 5. Comparing Radioactive Waste Volumes for Shallow Land Burial of the MOC and OTC

The vast majority of lower-hazard waste is associated with NU production, results from mining and milling, and is dealt with and disposed on site [28]. The remaining volume of radioactive waste, that is, 0.4% of the waste volume, is composed of LLW and DU-oxides that must be disposed at a licensed disposal facility. The front-end of the fuel cycles dominates for LLW generation ranging at around 60% at the beginning and decreasing to 45% towards the end of this study. Also, towards the end the study, recycling and reactor operations account for nearly 40% of the LLW produced (the remaining LLW is related to deconversion operations). It is apparent that the contribution of front-end impacts is reduced as less ENU fuel is required. MLLW produced each year is small (~43 m^3/yr) and is 0.0002% of the total radioactive waste volumes and 0.01% of the LLW values.

The difference between MOC and OTC radioactive waste volumes commences when reprocessing and other recycling operations begin in 2028 (top graph, Fig. 5) and the difference is described herein as "waste avoidance"; the cumulative waste avoided is shown in the middle graph of Fig. 5. Radioactive waste volume avoidance grows as ERU and MOX fuel replace the use of ENU in reactors. At the end of the simulation (years 2054 to 2063), the annual waste avoidance is reduced as ENU use rises because reactors come online faster than the recycling activities produce MOX and ERU.

LLW is produced throughout the MOC and OTC fuel cycles. Wastes from mining, milling and (including in-situ leach, ISL) are classified separately, above. The total volume of LLW for each year of the MOC is shown in Fig. 6 and ranges from 158,000 m^3/yr to 203,000 m^3/yr. The lower bound occurs at the start of the simulation when ENU fuel and energy demand are at their lowest points during the simulation.

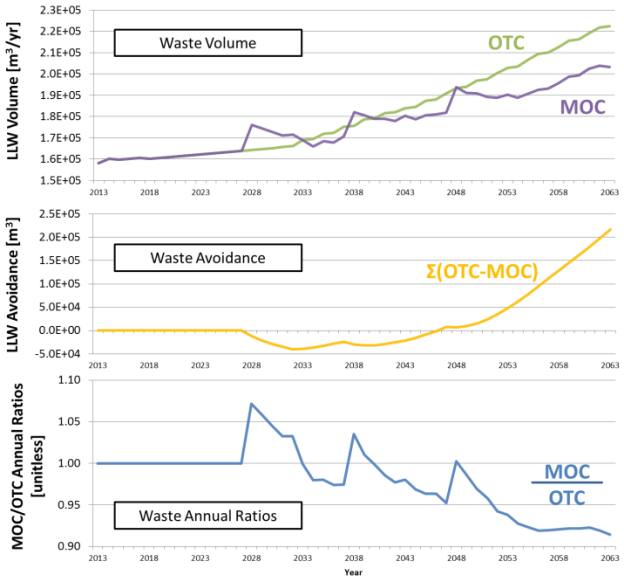

Figure 6. Comparing LLW Volumes for Shallow Land Burial of the MOC and OTC

There are three peaks of total annual LLW produced during the years of 2028, 2038, and 2048 which 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 converted to either MOX or ERU only when 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 (i.e., MOX and ERU fuel are not incrementally introduced to cores). 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. During this time LLW is being

produced for both annual ENU refueling and accumulation of the MOX and ERU inventories for full core replacement, then as MOX and ERU replace ENU cores the LLW production drops until the next reprocessing plant is built.

III.B.2. Integrated Impact Comparison of the MOC to OTC

Annual radioactive waste volumes eligible for shallow-land burial are integrated over the 50-year simulation for the MOC and OTC are listed by operation and grouped operations (see Table 4) and then are used to calculate integrated ratios. The greatest reduction of integrated waste impacts was associated with the front-end: LLW from the front-end is reduced around 20% and mining and milling wastes of around 15%. A relative increase of about 40% in waste volumes was associated with the middle of the fuel cycle for LLW generation; though it should be noted that the total LLW generated from all MOC operations actually results in a small (2%) overall decrease; this represents more than 217,000 m^3 of LLW that is avoided during the 50-year modeling time for non-mining related LLW.

Table 4. Comparative Integrated Radioactive Waste Volumes: Shallow-Land Burial-Eligible

Radioactive Waste Volumes [m^3]	MOC	OTC	Difference (OTC-MOC)	Integrated Ratio (MOC / OTC)	Range of Annual Ratios (MOC/OTC) LB	UB
Mine Waste	1.14E+09	1.32E+09	1.80E+08	0.86	0.72	1.00
Mill Tailings	4.97E+08	5.76E+08	7.83E+07	0.86	0.72	1.00
LLW and MLLW	9.04E+06	9.26E+06	2.17E+05	0.98	0.91	1.07
Front-end	5.28E+06	6.56E+06	1.28E+06	0.80	0.61	1.00
Middle*	3.57E+06	2.50E+06	-1.08E+06	1.43	1.00	1.77
Back-end	1.90E+05	2.02E+05	1.18E+04	0.94	0.75	1.03
DU-Oxides	7.10E+05	7.54E+05	4.41E+04	0.94	0.75	1.03
TOTAL NFC	1.65E+09	1.91E+09	2.59E+08	0.86	0.73	1.00

Notes: LB = lower bound; UB = upper bound; * Middle of the fuel cycle includes reactor and recycling operations

III.C. Deep Geological Repository-Eligible Radioactive Waste Volume Comparative Results of the MOC to OTC

III.C.1. Annual Basis Comparisons of the MOC to OTC

There are several considerations when accounting for the volumes of waste that are assumed to require geological repository disposal. The four categories of waste volumes that have been considered are UNF from enriched-uranium fuels, MOX UNF, vitrified HLW, and GTCC waste; each are briefly discussed below.

The volumes of UNF (UOX, MOX) are considered separately due to higher volumetric requirements in dry interim storage on a mass unit basis of MOX UNF, due to higher decay heat generation of MOX compared to UOX. Reprocessing and MOX fuel fabrication wastes are generated in a stepped-interval fashion corresponding to the reprocessing capacities that were assumed to increase in decadal intervals (see Fig. 7).

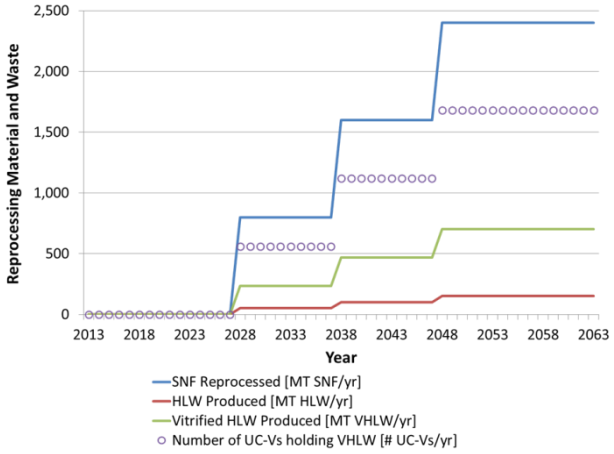

Figure 7. MOC Reprocessing HLW Production

The spikes observed in the number of DSCs holding UOX fuel (ENU, ERU, Re-ENU, BLEU) 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 (see Fig. 8). From 2059, to 2063, the increases of UOX DSCs are from the increased ENU requirements. The UNF inventory (non-reprocessed BWR and PWR assemblies, irradiated MOX PWR assemblies) in metric tons is around 133,900 MT UNF by the year of 2063.

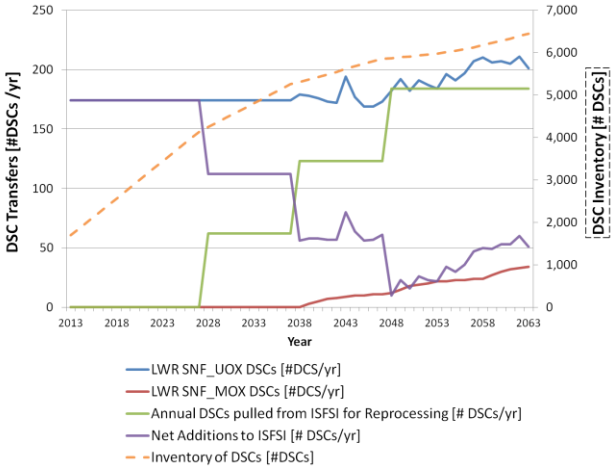

Figure 8. MOC Dry Interim Storage DSC Additions, Withdrawal, and Inventory

Total volume of waste produced by the MOC that is assumed to require the protection provided by geological disposal are consistent with the OTC case through 2028,

since reprocessing has not yet been introduced (purple line, top graph in Fig. 9).

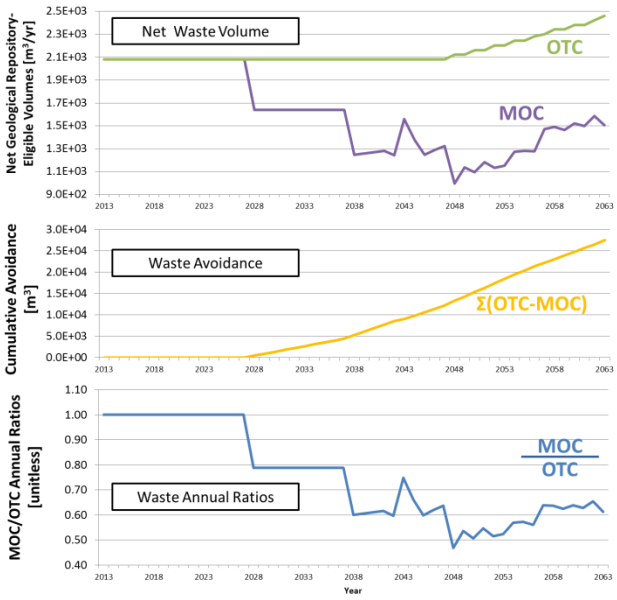

Figure 9. Comparing Geological Repository-Eligible Volumes of the MOC and OTC

When the withdrawal of UOX UNF is accounted for each year to feed reprocessing, the net UNF volume produced each year is much lower than the gross volume coming out of wet pool storage each year. UNF gross volume of both UOX and MOX sums to around 2000 m^3/yr from 2013 to 2028 (because 2028 is when the model initiated reprocessing) and then increases to 2,420 m^3/yr by 2063 for the OTC. After reprocessing withdrawals of UOX UNF are subtracted from the gross UNF volume, the net UNF volume is reduced to its lowest point at 130 m^3/yr at year 2048 when the third reprocessing facility begins operations. After 2048, the net volume of UNF increases to approximately 640 m^3/yr by the end of the modeling timeframe, due to continued addition of reactor capacity to meet electricity demand. The resulting trend shows that on the opening year of the third reprocessing plant (2048) that the number of DSCs (and volume of UOX UNF) emptied to support recycling operations almost exceeds the annual number of UNF DSCs filled — thus it can be postulated that the backlog of DSCs awaiting disposal (the total UNF inventory) would decrease if another reprocessing plant were brought online in 2058 (however, this has yet to be modeled).

The first discharged MOX fuel assemblies that have completely cycled through the PWR (5 year residence time) are then sent to the spent fuel pool for 5 years to cool. By the year 2038, the MOX fuel assemblies are 10 years old from first being placed into a PWR in year 2028. Also by year 2038, a slight increase of volume of net UNF volumes is observed since this initial MOX stream is now being placed into dry storage. This slight increase is due to higher DSC volume requirements on a unit mass basis for MOX due to higher decay heat (only 17 PWR-MOX assemblies are loaded into a single DSC compared to 32 PWR-UOX assemblies that can now be placed in the average DSC). Then, in year 2043, the increased net UNF volumes spike, representing the arrival of ERU from a delay of 5 years after MOX loadings resultant from the assumption that a 5-year time qualification delay of ERU fuel would allow fuel to be loaded by 2033.

Once other streams of high-activity radioactive waste from reprocessing and MOX fuel fabrication are included, there is a ~880 m^3 increase by the time that the third reprocessing and third MOX fuel fabrication plant begins operations. The resultant net volume of waste potentially requiring disposal in a geological repository from the MOC is estimated to be around 1,520 m^3/yr (640 m^3 of net UNF volume + 880 m^3 from recycling) at year 2063.

The net volumes of geological repository-eligible waste are plotted from the OTC and MOC in Fig. 7 (top graph). The volumes associated with the MOC are shown to be lower than the OTC. Cumulative waste volume continues to decrease (middle graph, Fig. 7) throughout the simulation resulting with around 27,400 m^3 less material requiring assumed to require disposal in a geological repository.

III.C.2. Integrated Impacts Comparison of the MOC to OTC

The volumes of waste assumed to require disposal in a geological repository that result from the MOC and OTC were integrated over the 50-year simulation and then were used to calculate integrated ratios listed by fuel cycle operation (and grouped operations, shown in Table 5). The "net" volumes of MOC geological waste were considered as most accurately representing the volumes that are ultimately to be disposed of in the repository because these net volume account for the UOX UNF withdrawn and used as feed for reprocessing. The greatest reduction in integrated ratios of waste was associated with the withdrawal of UNF for reprocessing feed during the MOC, including a 45% reduction of UNF volume in dry storage. However, in the OTC, the waste needing to be disposed at a geological repository only includes UNF. The MOC produces more types of waste assumed to required disposal in a geological repository than the OTC (i.e., vitrified HLW and GTCC streams). The integrated ratio of the MOC to the OTC for total geological repository-eligible waste estimates that the MOC produces 25% less geological repository waste volume than the OTC during the 2013 to 2063 window. This reduction would more closely approach 40% if the timeframe were extended further with continuing to open

a fourth reprocessing plant and the percentage of reactors supported by recycled fuel were increased.

Table 5. Comparative Integrated Radioactive Waste Volumes: Geological Repository-Eligible

Geological Repository-Repository Waste Volumes [m^3]	MOC	OTC	Differ-ence (OTC-MOC)	Inte-grated Ratio (MOC / OTC)	Range of Annual Ratios (MOC/OTC) LB	UB
Dry Interim Storage (Net)	5.90E+04	1.09E+05	5.03E+04	0.54	0.12	1.00
Reprocessing	2.16E+04	--	-2.16E+04	--	--	--
Vitrified HLW	9.70E+03	--	-9.70E+03	--	--	--
GTCC	1.19E+04	--	-1.19E+04	--	--	--
MOX Fuel Fab. GTCC	1.27E+03	--	-1.27E+03	--	--	--
TOTAL RECYCLING	2.28E+04	--	-2.28E+04	--	--	--
TOTAL NFC	8.18E+04	1.09E+05	2.74E+04	0.75	0.47	1.00

Notes: LB = lower bound; UB = upper bound

IV. SUMMARY AND CONCLUSIONS

Under the stated capacity assumptions and realistic, construction-limited transition to recycling, substantial reductions in lower-hazard (shallow-land burial destined) and higher-hazard (geological repository-eligible) waste volumes are estimated. For shallow-land burial eligible wastes, mining- and milling-related waste volumes are 14% lower for the MOC than the OTC over the 50-year model period and a 10% net savings in annual fuel cycle LLW volumes is observed by year 2063. For geological repository-eligible wastes, including used fuel and HLW, volumes are 25% lower for the MOC over the transition period modeled and approach a 40% decrease relative to the OTC at longer times as the system approaches steady-state. It was concluded that radioactive waste volumes, both lower-hazard and higher-hazard, can be relevant discriminators for NFC performance for the MOC and OTC (given both annual and integrated impacts are evaluated under the present set of model assumptions). However, results for waste volume estimates are particularly sensitive to the waste generation assumptions used; therefore, any conclusions drawn should be considered in the context of the uncertainties in waste generation rates. Future analysis efforts refining estimates of radioactive waste volumes as a metric appear warranted, both to compare potential fuel cycles and to resolve some of the uncertainties in normalized waste metric estimates.

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