Waste Estimates for a Future Recycling Plant in the US Based Upon AREVA Operating Experience – 13206

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ABSTRACT

Estimates of process and secondary wastes produced by a recycling plant built in the U.S., which is composed of a used nuclear fuel (UNF) reprocessing facility and a mixed oxide (MOX) fuel fabrication facility, are performed as part of a U.S. Department of Energy (DOE) sponsored study [1]. In this study, a set of common inputs, assumptions, and constraints were identified to allow for comparison of these wastes between different industrial teams. AREVA produced a model of a reprocessing facility, an associated fuel fabrication facility, and waste treatment facilities to develop the results for this study. These facilities were divided into a number of discrete functional areas for which inlet and outlet flow streams were clearly identified to allow for an accurate determination of the radionuclide balance throughout the facility and the waste streams. AREVA relied primarily on its decades of experience and feedback from its La Hague (reprocessing) and MELOX (MOX fuel fabrication) commercial operating facilities in France to support this assessment. However, to perform these estimates for a U.S. facility with different regulatory requirements and to take advantage of some technological advancements, such as in the potential treatment of off-gases, some deviations from this experience were necessary. A summary of AREVA's approach and results for the recycling of 800 metric tonnes of initial heavy metal (MTIHM) of LWR UNF per year into MOX fuel under the assumptions and constraints identified for this DOE study are presented.

INTRODUCTION

In the U.S., numerous studies have been performed that predict the volume of waste produced from a U.S. recycling plant for UNF. However, to date, the results of these studies have often produced wildly varying differences in waste volumes and classifications. These differences are mainly attributed to differences in the characteristics of the selected UNF to be recycled (e.g., age and burn-up), the potential inaccurate estimates of process efficiencies, the differences in the deployed industrial technology/flow sheets, and the differences in the assumed gaseous and liquid effluent releases.

To better understand and address the issue associated with differences in waste estimates produced from a U.S. recycling plant, the DOE initiated a study with industry to more precisely determine waste

generation from a UNF recycling plant located in the U.S. under a common set of high-level assumptions and constraints. This study is developing detailed estimates for waste volumes, waste forms, and waste characteristics for input fuel, specified through government furnished information (GFI), with two varying cooling times: 5 and 50 years. Current results are provided in this paper and are potentially subject to revision as task work proceeds.

To perform this study, AREVA has drawn upon its decades of experience as a leading designer, builder, and operator of back-end fuel cycle facilities. By developing the model of a U.S. recycling plant based on this experience, AREVA estimated the types and quantities of the waste streams produced by such a facility. In addition, these identified and characterized waste streams were preliminarily classified according to U.S. regulations (namely, HLW and Class A, B, C and GTCC LLW) and any outlet effluent streams were verified to meet U.S. regulatory limits. The exact waste form classification will depend upon the regulations in effect at the time the waste is ultimately created.

ASSUMPTIONS AND MODEL

Assumptions and Constraints

To allow for the comparison of the results between different industrial teams, a common set of inputs, assumptions, and constraints for the recycling activities have been specified. DOE provided through GFI the composition of two reference LWR uranium oxide (UOX) fuels to be recycled which were based on a burn-up of 50 gigaWatt-days per metric ton uranium (GWd/MTU) in a PWR with a cooling time of 5 and 50 years. The DOE also identified the capacity of the recycling plant to be 800 MTIHM/yr. Additional key assumptions included:

- Discharge of excess liquid effluent is acceptable provided the discharge meets the U.S. drinking water standards.
- Secondary wastes are classified per 10 CFR 61 and are packaged in 4 cubic meter steel boxes, except for GTCC which is packaged in containers similar to those approved for WIPP. Intentional blending of waste to change the classification is prohibited.
- Process wastes containing fission products are to be packaged in DOE standard 0.6 meter (2 foot) diameter by 4.6 meter (15 foot) tall canisters with an internal volume of 1.2 m^3. Heat generating waste cannot exceed 14,000 watts per canister.
- Metal process wastes are to be packaged in DOE identified 0.6 meter (2 foot) diameter by 3.0 meter (10 foot) tall canisters with a waste weight limit of 3,600 kg of waste.
- Off-gases of C-14, Kr-85, and I-129 are treated to ensure U.S. regulatory limits are met and wastes created from this treatment are treated to ensure disposal requirements are satisfied.
- U and (U, Pu)/MOX products comply with American Society for Testing and Materials (ASTM)

standards.

AREVA utilized its expertise on backend facilities to adapt the design of and results from existing industrial units to comply with these assumptions and constraints [2] and in some cases, recommended modification of assumptions to allow for use of its expertise in waste optimization (e.g., vitrified waste packaging).

Facility Model

As part of the study and to allow comparison with other industrial partners utilizing different technologies/flow sheets, a model of the recycling facility, associated fuel fabrication facility, and waste treatment facilities was developed. These facilities were divided into a number of discrete functional areas (FAs) for which inlet and outlet flow streams were clearly identified to allow for the determination of the radionuclide balance throughout the facility and the waste streams. Figure 1 provides an overview of the FAs for AREVA's proposed recycling plant, where the FAs are identified by the larger yellow boxes (e.g., "Receipt and Storage"). The blue squares in Figure 1 represent the main processing steps of the recycling plant. The purple squares represent the support functions required to effectively operate the recycling plant and these functions can be used to support the operation of multiple reprocessing and fuel fabrication facilities. Note that only LWR MOX fuel fabrication was considered for this part of the study (i.e., no activities associated with recycled U were captured in this study).

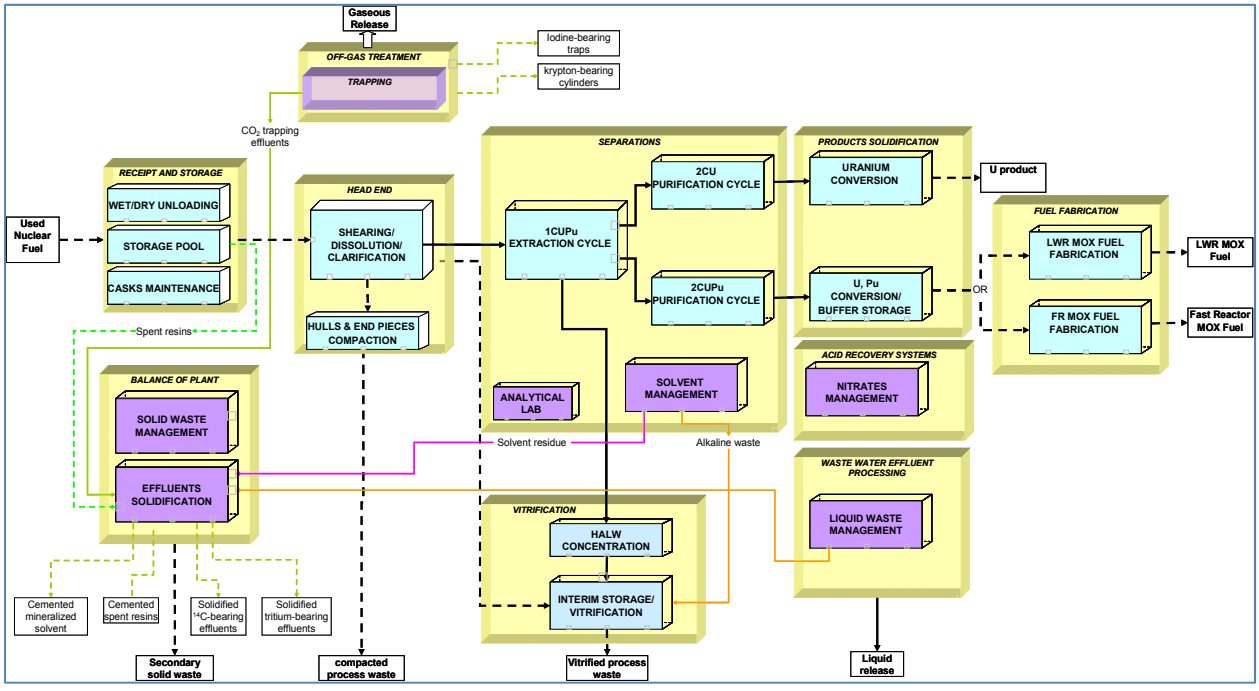

FIGURE 1: Functional Diagram of the Recycling Plant

In addition to the FA overview, each FA was broken down into more detail that included a description of the inputs to and outputs from each FA and included connections not detailed in Figure 1. Figure 2 provides an example of one of these detailed assessments for the "Head End" FA.

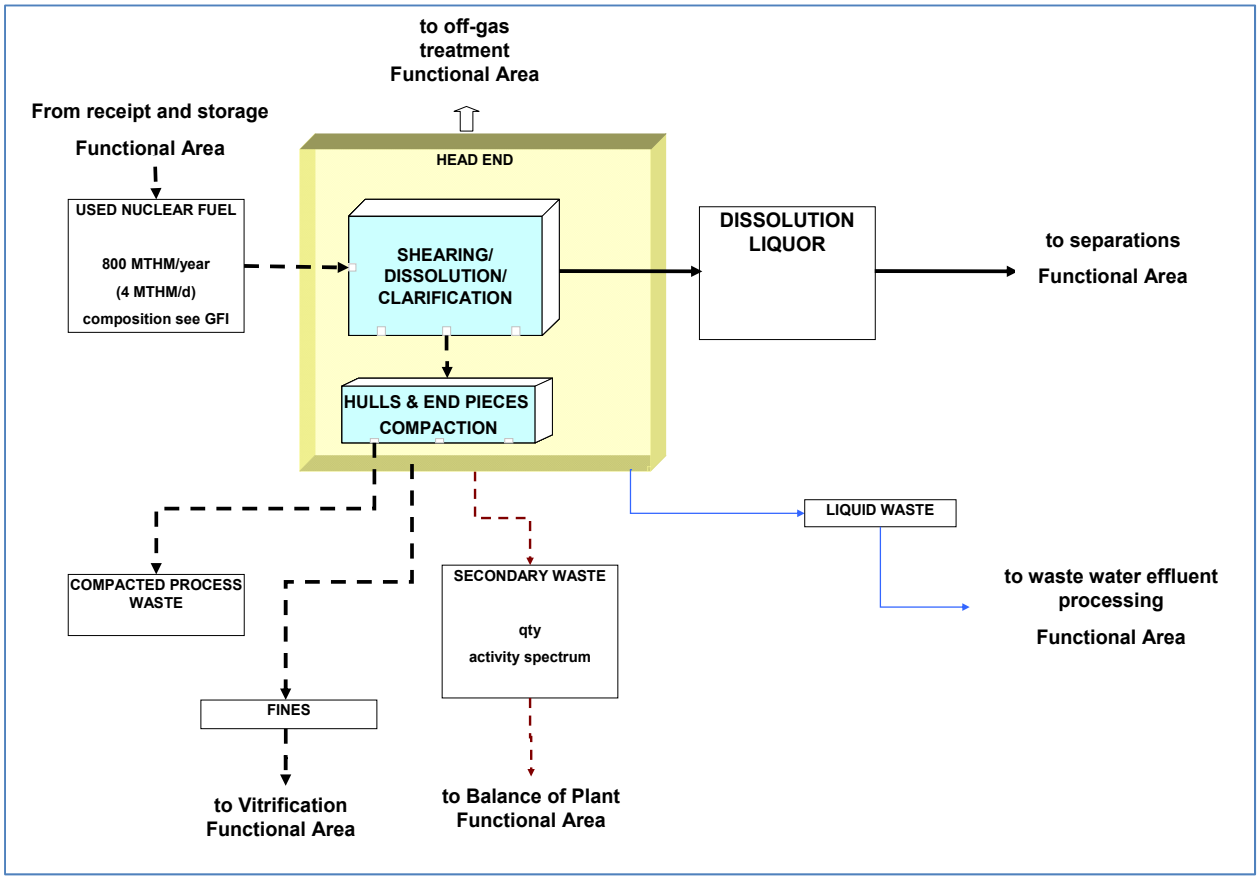

FIGURE 2: Inlet and Outlet Streams for the Head End Functional Area

PROCESS PARAMETERS AND PROCESS WASTE ESTIMATE

Once the FAs were defined, the critical parameters that established the inlet and outlet flows for each FA were identified. These critical parameters were relationed to AREVA's existing facilities for default values and variations to these default values were identified, as necessary, to meet the assumptions and constraints of this study. Flowsheets were then produced using this information to establish waste streams produced from each of the FAs. The key parameters and waste estimates for the wastes associated directly with the processing of the LWR UNF and the production of the MOX fuel (i.e., process wastes) are presented below.

Vitrified Process Wastes

The vitrified process waste contains nearly all the activity entering the plant, except for the U, Pu, gaseous fission products, and activation fission products (FP) included in the metal process wastes (e.g., fuel cladding). The cold crucible induction melter (CCIM) is the selected process from the La Hague plant used to produce the vitrified process wastes. The process solution sent to the CCIM is prepared in the high active liquid waste (HALW) processing unit within the Vitrification FA. The HALW unit optimizes the mixing of concentrated fission products, alkaline waste, and undissolved solids such that the heat limit constraints of the final glass form (14 kW/canister) are satisfied while minimizing the volume of waste produced. The final product from the CCIM is a robust canistered glass waste form packaged in a Universal Canister for vitrified waste (UC-V) produced at a rate of approximately 0.7 UC-V per MTIHM recycled [6].

The specified fission product (vitrified) waste package in this study (0.6 m diameter x 4.6 m tall U.S. canister) and its associated maximum heat load (< 14 kW per canister) results in a maximum incorporation rate of FP and actinide oxides for the 5-year cooled UNF of approximately 12%. In contrast, the present typical R7/T7 glass produced at La Hague and placed into UC-V's incorporates approximately up to 18.5% and AREVA is currently considering the possibility of increasing this value up to 19.6% for the higher burn-up fuels in order to minimize, to the extent possible, the total waste volume produced to 0.7 UC-V per MTIHM recycled. Such high values are possible because of the design of the volume and diameter of the AREVA reference universal canisters (UC-V is 0.4 m in diameter and has a capacity of about 150 L), which allows for efficient cooling during storage and transportation even with the higher activity and heat load associated with high burn-up fuels. AREVA therefore recommends adoption of the UC-V, especially for the recycling of the shorter cooled, higher burn-up UNF.

For 50 year cooled UNF, the heat limitation is far less restrictive, however the waste volume associated with the recycling of this UNF will be about the same or potentially larger than for the 5 year cooled UNF due to the in-growth of Am and oxides in this UNF. To preserve a low crystalline yield after cooling, a maximum loading of approximately 28 weight percent is applied for radioactive oxides.

Metal Process Wastes

Metal process wastes are composed of the structural parts of the fuel elements (hulls and end-pieces) which are super-compacted. Experience at the La Hague plant provides the reference super compaction data utilized in this study. Waste volume reduction at this unit is by a factor of 4 to 5. The reference package used to store and ultimately dispose of this compressed waste is the universal canister for compressed wastes (UC-C) which has the same dimensions as the UC-V container. AREVA's experience finds that approximately 0.65 UC-C per MTIHM recycled would be produced from this process unit for

this compactable waste stream.

AREVA did not use the specified waste package in this study for this waste stream (0.6 m diameter x 3.0 m tall canister) because the use of this canister was outside AREVA's industrial expertise and would have required confirmation that the super compaction process could be re-designed and qualified to reach an equivalent compaction factor with this larger container. Based on AREVA's experience at the reference unit, the use of the larger canister would have required redevelopment of the compaction line which would include: modification of the process feeding hulls and end-pieces to the press; development of a larger press and mechanical handling, transfer, and welding devices; and design of specific nuclear measurement posts. AREVA therefore recommends the adoption of the UC-C which provides the added benefit of using the same handling equipment and similar transportation casks as the UC-V and hence eliminating potentially redundant equipment and reducing safety hazards associated with the potential mixing of handling equipment.

Off-Gas Trapping Waste

I-129 is mainly released during the dissolution process and can be trapped in the off-gas treatment. AREVA's experience in this area in France is limited and hence experience from the Rokkasho Reprocessing Plant is applied where about 98% of the iodine is trapped from the off-gas on a solid media (silver alumina) [9]. About 1% is expected to be released to the atmosphere and the remaining fraction captured in the waste streams to be solidified.

The trapping and final conditioning of iodine waste in the silver alumina is the subject of ongoing research and development actions in several countries. None has reached industrial deployment yet, but the following three processes [9] were considered in this study for the estimation of the final solid waste produced from the capture of I-129:

- Hot Isostatic Press (HIP)
- $BiPbO_2$ low temperature vitrification
- High integrity confinement (HIC) cementation

Table I provides the waste estimates and key parameters associated with each of these processes. Due to the long half-life of I-129, the quantity of solid wastes produced from these processes during the recycling of 5 and 50 year cooled UNF does not differ and is classified GTCC.

TABLE I: Iodine-Bearing Solid Wastes

Process	I-129 Bearing Solid Waste (m^3/MTIHM)	Iodine Incorporation Rate (weight %)	Specific Activity Ci/m^3 (TBq/m^3)
HIP	6.25×10^{-4}	14%	75.3 (2.78)
Low Temperature Vitrification	2.50×10^{-3}	2%	18.8 (0.7)
HIC Cementation	1.08×10^{-2}	1.85%	4.4 (0.16)

Kr-85 is currently released from existing commercial plants as the radiological impact is acceptably low and the trapping of noble gas is very difficult, expensive, and potentially hazardous. In this study however, Kr capture was necessary for the 5 year cooled UNF to meet specific U.S. regulatory limits (40 CFR 190 [8]). Therefore, the existing trapping technology consisting of liquefying off-gases and separating the elements by rectification at -152°F and 0.67 MPa was applied without further consideration of cost or safety [7]. To prevent the clogging of the off-gas line, a pre-treatment of the off-gas is necessary and includes removal of iodine, NO_x, CO_2, H_2O, and Xe. The captured Kr, which includes the non-radioactive species, is to be stored in 50L cylinders loaded with 3,700 TBq of Kr-85.

Approximately 0.1 cylinder per MTIHM recycled are expected to be produced from the recycling of the 5 year cooled UNF. An increase in the cooling time of the UNF to 50 years negates the need for trapping Kr-85 and thus dramatically reduces the cost of the off-gas treatment unit and the potential safety risks thereof. The captured Kr in the cylinders would be decay stored at the plant and not disposed of, but would be classified as Class A LLW.

C-14 released during the dissolution process is absorbed by alkaline scrubbing solution which produces a salt-bearing waste that is subsequently solidified. About 0.075 m^3 of solid waste containing C-14 is produced per MTIHM recycled and will be classified as Class A LLW. Note that the trapping technique for C-14 will also trap large amounts of $^{12}CO_2$ present in the air flow, thus resulting in large volumes of waste material with a very low C-14 to C-12 ratio.

Tritium and Liquid Waste Management

To limit tritium migration around the process and thus, simplify design and operation in the low (or non) tritiated portions, the tritium will be segregated, to the extent possible, and maintained in the portion of the facility designated as “tritiated” [4]. This is the design approach AREVA has proposed for its next generation recycling plant (e.g., COEX™ [3]). Tritiated nitric acid is recovered by rectification in an evaporator and totally reused in the tritiated portions of the facility. Intensive recycling of high tritiated

recovered water is effective but leads to several drawbacks (e.g., increased tritium content and accumulation of impurities). Therefore a compromised optimum between the volume of excess tritiated waste water and the drawbacks associated with intensive recycling has to be found [5]. The considered value derived from the selected compromise is 2,500 L/MTIHM recycled. Excess tritiated water is purified in a series of evaporators to limit the total activity of the final waste. Table II includes the expected global decontamination factors (DF) for radionuclides other than tritium. The final effluent from the tritiated portion of the facility (i.e., excess tritiated water) is solidified by grouting and results in a Class B LLW. Approximately 5 m^3 of solidified tritium effluents are produced per MTIHM recycled.

TABLE II: Decontamination Factors Used to Calculate the Activity of High Tritiated Excess Water

Functional Area	Vitrification	Acid Recovery		Waste Water Effluent	Total
	HALW	Purification	Rectification		
Purification Factor	> 10^7	> 10^4	~2,000	> 10^4	
Concentration Factor	~20	~20	~10	~20	
Decontamination Factor	> 5 x 10^5	> 5 x 10^2	> 2 x 10^2	> 5 x 10^2	> 2.5 x 10^{13}

Low tritiated water and nitric acid are purified in a series of evaporators to limit the total activity of the final liquid effluents from the low tritiated portion of the plant. Table III includes the expected global DFs for radionuclides other than tritium. Table III also includes the global ^3H DF of 1CUPu (describing tritium distribution between high and low tritiated portions of the plant). This value is used to estimate the tritium content in the low tritiated effluents.

Most of the low tritiated acid is recycled back into the process to limit the effluent volume. The excess portion is grouted (see solid wastes). Low tritiated water may be recycled. Excess low tritiated water from the "low tritiated" portion of the facility may be released without further treatment as they are processed through a sufficient number of evaporators to satisfy the release criteria.

Liquid waste treatment also includes an evaporator dedicated to the treatment of atypical and salt-bearing liquid wastes which include alkaline liquid waste from the Solvent Regeneration units, analytical lab liquid wastes, wastes produced by decontamination of failed equipment, etc. The distillates from this evaporator are treated in the Low Tritiated Waste Water Treatment unit and the concentrates are solidified to Class A LLW. Approximately 1.4 m^3 of solidified salt-bearing wastes are produced per MTIHM recycled.

TABLE III: Decontamination Factors Used to Calculate the Activity of Low Tritiated Excess Water

Functional Area	Separation	Acid Recovery			Waste Water Effluent Processing	Total
		Purification 1	Purification 2	Rectification		
Purification Factor		> 1 x 10^4	> 1 x 10^4	~2,000	> 1 x 10^4	
Concentration Factor		~20	~20	~10	~20	
DF for FP (identified to Cs)	> 1 x 10^7	> 5 x 10^2	> 5 x 10^2	> 2 x 10^2	> 5 x 10^2	> 2.5 x 10^{17}
DF for Ru/Rh	> 2 x 10^4					> 5 x 10^{14}
DF for ^{3}H	> 200 x 100	~ 1 (Purification Factor)				> 200 x 100

SECONDARY WASTE PARAMETERS AND ESTIMATE

Secondary wastes produced from the recycling plant fall into one of two categories:

- Technological solid wastes (e.g., worn components produced during operations, maintenance wastes from radiological zones within the facilities, and tools used during the maintenance operations)
- Solidified effluents produced during operation of the facilities.

The approach to the treatment of secondary wastes at the recycling plant is shown in Figure 3.

Spent resins are linked to the operation of the UNF storage pool and wet unloading facility of the recycling plant. Their quantity and characteristics (e.g., chemical species and radionuclides) are dependent on the pool capacity and status (e.g., leaker) and characteristics (e.g., corrosion products) of the UNF. Experience from the La Hague plant provided the annual quantity and the activity spectrum for resins at the proposed recycling plant. These resins are solidified in a dedicated cementation unit and results in the production of approximately 7.5 x 10^{-3} m^3 of Class A LLW per MTIHM recycled.

Solvent residue is the ultimate waste from the management of the solvent stream used in the separations units. The solvent is normally regenerated in specific units through contact with washing solutions. Excess spent solvent loaded with impurities and degradation products is treated by distillation in the Organic Wastes Treatment unit and then recycled. The residue from this distillation process is a stream of very low volume but with highly concentrated residual activity. Table IV provides the DFs that establish the activity of the solvent residue. Application of the La Hague treatment process to this stream

(mineralization followed by cementation of the ashes) results in 1.35 x 10^{-2} m^3 of Class A LLW per MTIHM recycled.

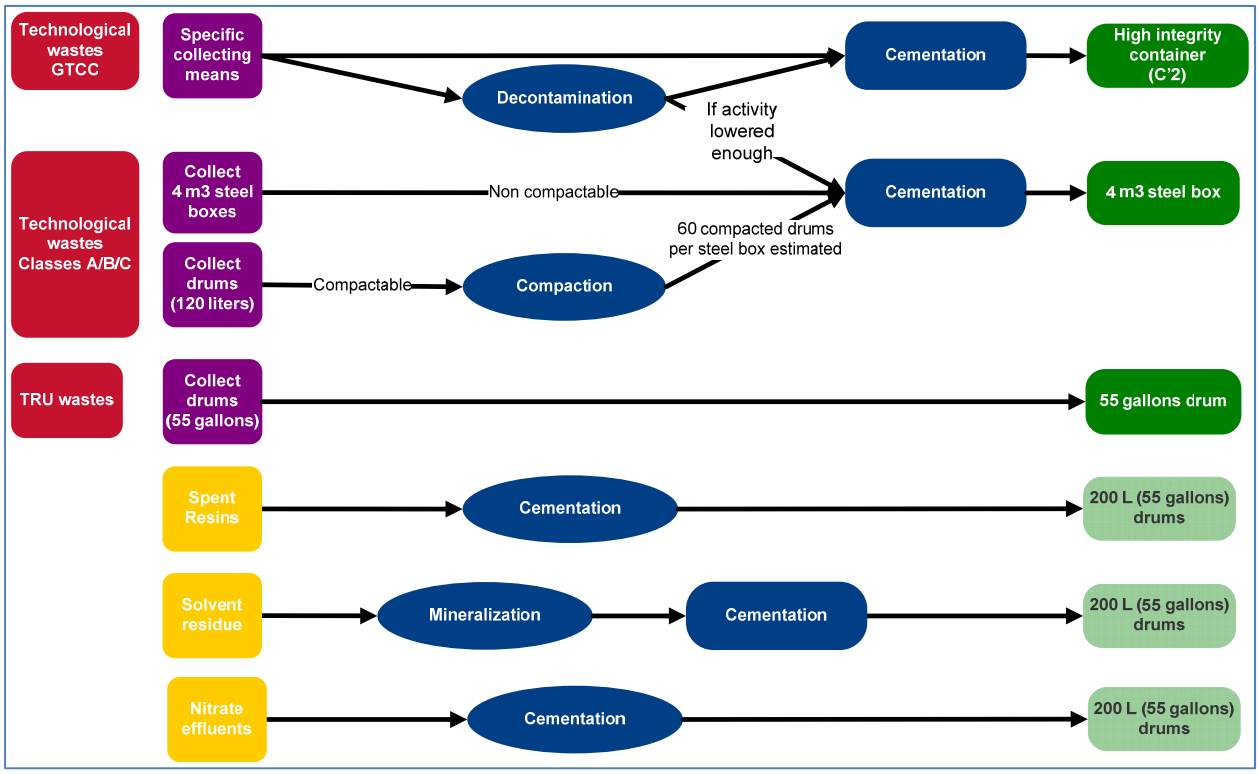

FIGURE 3: Management of Secondary Wastes

Nitrate effluents are a specific liquid waste from the low tritiated area of the acid recovery system. Most of the low tritiated acid is recycled back into the process so that the volume of the excess nitrates from the plant is reduced to an as low as reasonably achievable quantity. These excess nitrates are solidified and amount to approximately 1.2 m^3 of Class A LLW per MTIHM recycled. The residual activity is established using the DFs presented in Table V.

TABLE IV: Decontamination Factors Used to Calculate the Activity of the Solvent Residue

Functional Area	Separation				Total
DF for:	FP Scrubbing	U Pu Separation	U Stripping	Solvent Regeneration of 1 CUPu	
FP (identified to Cs)	> 1 x 10^7	~ 3		~1,000	> 3 x 10^{10}
Ru/Rh	> 2 x 10^4			~100	> 6 x 10^6
U	-	-	~1,000	~1,000	> 1 x 10^6
Pu	-	~1 x 10^6	~2	~100	> 2 x 10^8
Np	= 1/80%	-	~1,000	~1,000	> 1.25 x 10^6
^{3}H	200 x 100	~1,000 (assumed conservative)			> 2 x 10^7
^{14}C	~1,000 (assumed conservative)				> 1,000

TABLE V: Decontamination Factors Used to Calculate the Activity of the Excess Nitrates

Functional Area	Separation	Acid Recovery			Total
DF for:		Purification 1	Purification 2	Excess Nitrates Drawing Off	
FP (identified to Cs)	> 1 x 10^7	> 5 x 10^2	> 5 x 10^2	= 1/12%	> 2.1 x 10^{13}
Ru/Rh	> 2 x 10^4				> 4.2 x 10^{10}
^{3}H	> 200 x 100	~1			> 1.7 x 10^5

Technological Wastes

The main purpose of the solid waste management unit is to treat all the solid technological wastes produced within the recycling plant into final waste forms that are compatible with potential disposal pathways. At the reference plants operated by AREVA (La Hague and MELOX), the solid waste management unit has considerably and continuously evolved based on the experience gained from: maintenance and repair activities, the efforts for waste minimization at the source, and the selective sorting and safe preconditioning of the waste according to the nature of the waste and the area of its generation. A centralized solid waste treatment facility is devoted to controlling treatment activities including: providing additional decontamination and/or compaction (if required) and ultimate conditioning.

To estimate the technological waste produced at an integrated recycling facility, AREVA utilized its operational data from both the La Hague and MELOX plants to establish quantities and activities of each type of waste. However, waste treatment facilities can be utilized by multiple plants and facilities and this is the case for the waste treatment facilities at La Hague which treat wastes from the entire site (which includes multiple reprocessing lines and D&D activities located throughout the site). Hence, a detailed assessment of the technological wastes treated at the La Hague site had to be performed. Data from the years 2008 through 2010 were utilized as the basis for the estimate and the following activities had to be performed to establish a best estimate:

- Wastes from out-of-scope facilities and activities (e.g., D&D wastes) were discounted.
- Wastes not applicable to the scope of this study (e.g., retro-fitting of vitrification systems) were discounted.
- Wastes volumes had to be adjusted from the La Hague yearly throughputs to the 800 MTIHM/yr throughput proposed for this study.
- Including the benefits of applying only the new technologies that reduce waste volumes (e.g., some La Hague technological wastes were associated with operation of the hot induction melters which produce larger quantities of technological wastes than the CCIM, so the database was corrected for this difference).

For technological wastes associated with the fuel fabrication facility, the MELOX data from the years 2007 and 2008 (latest available) was utilized. However this data also had to be evaluated to correct for:

- The use of PuO_2 cans for transfers between La Hague and MELOX (a site with these activities co-located, as proposed for this study, will not include the wastes associated with these cans).
- The difference in fuel fabrication throughputs (the MELOX throughput is greater than the throughput of the proposed fuel fabrication facility and hence, must be scaled accordingly).
- No need for a primary blending step at the proposed fuel fabrication facility since this study is based on a powder containing only 35% Pu, unlike MELOX which requires a primary blending step with UO_2 (hence the fraction of wastes produced with this step can be discounted).

In addition to these corrections, this evaluation of the technological wastes was also complicated for the following three reasons:

- The functional areas identified in this study do not exactly align with those associated with the reference facilities. To adjust for this, the wastes remaining from the above identified and corrected databases were mapped to the best fitting functional area. Ultimately, the wastes are optimally combined in the solid waste management unit and hence, their mapping to specific functional areas is not that important.
- The waste packages and categorization of the wastes at the reference plants correspond to definitions specific to the French waste management agency (ANDRA) and its disposal sites requirements and these definitions do not match those in the U.S. (10 CFR 61). Thus, average characteristics of each type of French package have been determined (e.g., size, weight, waste contents, and radionuclides content) to tentatively establish the classification of the waste according to the U.S. regulations (after the next correction is made).
- The UNF recycled at La Hague to produce the characteristics of these technological wastes had different characteristics than the UNF assumed recycled in this study. Thus, the recorded

activities for fission products from La Hague were scaled to the processed UNF assumed in this study by ratio-ing the spectrum associated with this study's UNF against the original spectrum associated with the inventory of UNF processed at La Hague (note that an average value was established for the UNF processed at La Hague over the database timeframe). The resulting radiological characteristics (TBq/m^3 or Bq/g) for each radionuclide were used to perform the classification of the wastes.

Table VI presents the final results for the technological wastes after the above modifications were performed on the data collected from the operating facilities. Note that the very low active waste, which is subject to incineration in France, has been included in the Class A LLW volume. In addition, technological waste from the most contaminated portions of the facilities has been conditioned and placed into high integrity fiber-concrete containers (HICs). Finally, the GTCC-TRU waste has been placed in 55 gallon drums without application of any thermal treatment or super-compaction and the majority of it assigned to this category due to the relatively low alpha limit for GTCC waste (3.7 KBq/g).

TABLE VI: Estimated Quantities and Classifications of Technological Wastes from Recycling Plant

Number of packages (per MTIHM)	Type	Classification
0.16	Steel Boxes (4 m^3)	LLW CLASS A
0.02	Steel Boxes (4 m^3)	LLW CLASS C
0.05	High Integrity Containers	GTCC
1.4	Drums (55 gallons)	GTCC-TRU

CONCLUSIONS

A study sponsored by the DOE has been performed by AREVA to estimate the process and secondary wastes produced from an 800 MTIHM/yr recycling plant built in the U.S. utilizing two types of UNF defined by DOE and some DOE defined assumptions and constraints. AREVA based its waste estimates on its current proposed recycling technology (e.g., COEX™ process [3]), operational data from its backend facilities in France (La Hague and MELOX), supplemental data from the Rokkasho Reprocessing plant in Japan, and from recent advances in waste treatment technology. A summary of the process and secondary wastes are provided in Table VII for the 5 year cooled UNF defined by DOE.

For the 50 year cooled UNF defined by DOE, the waste quantities are fairly similar with the following exceptions:

- There may be a fractional increase in the quantity of UC-V's produced from the fission products as a result of an increase in Am and oxides in the older UNF.

- Due to the decay of Kr-85, there will be no requirement for its capture and hence, no cylinders for this waste type.
- Due to the decay of tritium, there will be a significant decrease in the volume of tritiated wastes.
- Due to the decay of the short lived radionuclides, there will be a small decrease in the quantity of technological wastes.

In summary, a study was performed that provides detailed waste estimates of projected volumes, masses, radionuclide concentrations, and other characteristics of both process and secondary wastes produced for the recycling of two different types of UNF into MOX fuel. By providing a common set of inputs, assumptions, and constraints, DOE can now compare wastes estimates between different industrial teams.

TABLE VII: Summary of Process and Secondary Wastes Produced from a Recycling Plant Processing 5 Year Cooled UNF

Waste Type	Main Contents	Form	Quantity (per MTIHM)	Category/Class
Process	Fission Products	Vitrified Glass	0.7 UC-V	HLW
	Metal Process Wastes	Super-Compacted	0.65 UC-C	GTCC
	Iodine 129 Waste	Synthetic Rock	6.25×10^{-4} m^3	GTCC
	Krypton 85	Gas under Pressure	0.1 cylinder	A
	Carbon 14 Waste	Cement	0.075 m^3	A
	Tritiated Wastes	Cement	5 m^3	B
	Salt-Bearing Wastes	Cement	1.4 m^3	A
Secondary	Spent Resins	Cement	7.5×10^{-3} m^3	A
	Solvent Residue	Cement	1.35×10^{-2} m^3	A
	Nitrate Effluents	Cement	1.2 m^3	A
	Technological Waste	Cement	0.16 steel box	A
	Technological Waste	Cement	0.02 steel box	C
	Technological Waste	Cement	0.05 HIC	GTCC
	Technological Waste	Cement	1.4 drum	GTCC-TRU

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