

COMPARISON OF THE PROPOSED AND ALTERNATIVE MOX FUEL FABRICATION FACILITY PLUTONIUM-POLISHING PROCESSES

INTRODUCTION

The U.S. Nuclear Regulatory Commission (NRC) is currently reviewing a license application from the consortium of Duke Cogema Stone & Webster (DCS) to build a mixed oxide (MOX) fuel fabrication facility (MFFF) at the Department of Energy's (DOE's) Savannah River Site (SRS). To aid in the NRC's technical and environmental review for acceptability to license, this study has been tasked with obtaining and reviewing the available information and literature regarding the aqueous (or wet) and dry plutonium polishing processes for a comparison of the efficiencies, waste generation, environmental effects, cost, throughput, and other issues. Open sources were used for the technical information presented in this report, which included aqueous process information from documents provided by DCS to the NRC and dry process information from publically available documents.

As presented in the MFFF Construction Authorization Request (CAR) and Environmental Report (ER), the aqueous plutonium-polishing technology using solvent extraction is the preferred process. In addition, another aqueous process using an ion exchanger for the removal of impurities was presented as an alternative in DOE documents. DOE also funded the research and development of a dry plutonium-polishing process principally through the Los Alamos National Laboratory (LANL). Each process will be reviewed and documented for a comparison of plutonium-polishing processes.

The following sections provide a summary of each process from its reference source material, a comparison between the waste streams of each alternative, and the conclusions that can be drawn from information currently available and, if necessary, any lack of information that may affect the conclusions.

PROCESS DESCRIPTIONS

The principal assumption of this study is that the feed material in the plutonium polishing process is plutonium oxide from the disassembly and conversion of the plutonium pits from DOE's surplus plutonium disposition program. This plutonium oxide contains various impurities including gallium, americium, uranium, aluminum, and fluorides. The plutonium isotopic composition is as follows (DCS, 2001a):

- Pu-236 < 1 ng/g Pu (1 ppb), at the origin of pit.
- Pu-238 < 0.05%.
- 90% < Pu-239 < 95%.
- 5% < Pu-240 < 9%.
- Pu-241 < 1% during lifetime of plant.
- Pu-242 < 0.1%.

Table 1 lists the feed chemical impurities. These impurities must be removed to satisfy the feed material requirements of the MFFF. In particular, the gallium (as much as 12,500 $\mu\text{g/g}$ Pu (12,500 ppm) in the weapons plutonium) must be removed due to the corrosive nature of this element when in contact with zirconium, stainless steel, and other metals (Philip, 1997 and DCS, 2001b).

**Table 1 Chemical Impurities of Plutonium Oxide into the Plutonium Polishing Process
(Reprinted from Table 11.3-27 of DCS, 2001a)**

Chemical Component	Maximum Content ($\mu\text{g/g Pu}$ [ppm])	Maximum Exceptional Content ($\mu\text{g/g Pu}$ [ppm])	Chemical Component	Maximum Content ($\mu\text{g/g Pu}$ [ppm])	Maximum Exceptional Content ($\mu\text{g/g Pu}$ [ppm])
Ag	NA	10,000	N	400	400
Al	150	10,000	Na	300	10,000
B	100	1,000	Nb	100	10,000
Be	100	10,000	Ni	200	10,000
Bi	100	1,000	P	200	1,000
C	500	1,500	Pb	200	1,000
Ca	500	10,000	S	250	1,000
Cd	10	1,000	Si	200	200
Cl	(+F<250)	500	Sm	2	1,000
Co	100	10,000	Sn	100	10,000
Cr	100	250	Ti	100	10,000
Cu	100	500	Th	100	100
Dy	1	1,000	V	300	10,000
Eu	1	1,000	W	200	10,000
F	(+Cl<250)	350	Zn	100	1,000
Fe	500	1,000	Zr	50	1,000
Ga	12,000	12,500	Boron Equivalent	Not Applicable	
Gd	3	1,000	Total Impurities	18,800	
In	20	1,000			
K	150	10,000			
Li	400	10,000			
Mg	500	10,000			
Mn	100	10,000			
Mo	100	1,000			

The CAR presents the following concentrations for the principal impurities after polishing (DCS, 2001a):

- Gallium $\leq 0.1 \mu\text{g/g Pu}$ (0.1 ppm).
- Uranium $\leq 100 \mu\text{g/g Pu}$ (100 ppm).
- Americium-241/(Americium-241 + Plutonium Total) $< 0.01\%$.
- Silver (Ag) $\leq 10 \mu\text{g/g Pu}$ (10 ppm).

The Fuel Qualification Plan (FQP) outlines the step-by-step process for implementing the qualification of MOX fuel using surplus weapons-grade plutonium (DCS, 2001b). Appendix D of the FQP presents the summary of the MOX fuel pellet specifications including impurity concentration levels. There are two different impurity criteria in the MOX fuel pellet specifications. The first is a maximum summation impurity level of $1,500 \mu\text{g/g Pu}$ (1,500 ppm) for twenty-one chemical elements from Table 1 and shown in Table 2. The other impurity criteria is based on individual limits for selected chemical elements as shown in Table 3. Based on these two impurity criteria and the maximum possible impurity level of the plutonium oxide entering the plutonium polishing process, Tables 2 and 3 show the necessary decontamination factors¹ that must be obtained to yield the impurity criteria in Appendix D of the FQP. Therefore, the plutonium polishing process selected for use with MFFF must have the capability to satisfy a decontamination factor of 10^5 as given from Table 3.

Aqueous Processes

Several DCS or DOE documents present two aqueous processes as the basis of the aqueous plutonium-polishing process. The first is the mature plutonium purification process known as plutonium-uranium extraction, or PUREX.² This process uses solvent extraction and is the selected plutonium-polishing process for the MFFF (ORNL, 1998 and DCS, 2001a). The second process applies ion exchange technology to separate the dissolved plutonium from the impurities (ORNL, 1998).

Solvent extraction is a continuous process suitable for a wide range of flow rates and concentrations. Ion exchange also can be used for similar separation applications, but it is more routinely applied in a semi-batch mode for materials at lower concentrations. In addition, ion exchange methods can be more selective than solvent extraction techniques.

Both solvent extraction and ion exchange processes have been successfully applied to many metal recovery and nuclear applications. Both processes create an aqueous acidic waste solution with the various separated impurities such as gallium, americium, aluminum, fluorine, and other materials. Either of the aqueous processes have decontamination factors of 10^4 to 10^5 . The actual achievable decontamination factors would have to be measured or demonstrated for the particular application or equipment process stream (ORNL, 1998).

¹Decontamination Factor: The dimensionless ratio of the concentration of an impurity that is in the original, input stream to the polishing process to the concentration of this impurity in the final, output stream from the polishing process.

²The PUREX process was developed in the early 1950's to reprocess irradiated reactor fuels from various DOE production and single-pass nuclear reactors for the recovery of uranium, plutonium, and neptunium (DOE, 1997 and BHI, 2001). The PUREX process was applied at DOE facilities built at Hanford, Idaho National Engineering & Environmental Laboratory, and Savannah River sites (DOE, 1997).

Table 2 MOX Pellet Impurity Chemical Grouping and Necessary Decontamination Factor

Chemical Component	Maximum Exceptional Content ¹ ($\mu\text{g/g Pu}$ [ppm])	Chemical Component	Maximum Exceptional Content ¹ ($\mu\text{g/g Pu}$ [ppm])
Al	10,000	Gd	1,000
B	1,000	Mg	10,000
C	1,500	Mo	1,000
Ca	10,000	Ni	10,000
Cd	1,000	Si	200
Cl	500	Sm	1,000
Cr	250	Ti	10,000
Dy	1,000	Th	100
Eu	1,000	W	10,000
F	350	Zn	1,000
Fe	1,000		
Total impurity level of all of the above chemicals			71,900
MOX Pellet Impurity Specification ²			1,500
Necessary Decontamination Factor			48

¹ Table 11.3-27 of DCS, 2001a

² Appendix D of DCS, 2001b

Table 3 Individual MOX Pellet Impurity Limits and Necessary Decontamination Factors

Chemical Component	Maximum Exceptional Content ¹ ($\mu\text{g/g Pu}$ [ppm])	MOX Pellet Specification ² ($\mu\text{g/g Pu}$ [ppm])	Necessary Decontamination Factor
Al	10,000	250	40
C	1,500	100	15
Cl	500	25	20
F	350	15	23.3
Fe	1,000	500	2
Ga	12,500	<0.12	$\sim 10^5$
N	400	75	5.3
Si	200	250	0.8
Th	100	10	10

¹ Table 11.3-27 of DCS, 2001a

² Appendix D of DCS, 2001b

Solvent Extraction

Generic process descriptions for solvent extraction are provided in the documents related to the Draft Surplus Plutonium Disposition Environmental Impact Statement (ORNL, 1998 and DOE, 1999). Site-specific process descriptions for solvent extraction are given in the DCS documents (Section 3.2 of DCS, 2000 and Section 11.3 of DCS, 2001a) and are the principal source of information on solvent extraction used in this report.

The solvent-extraction process proposed for the MFFF is a refinement of the aqueous polishing process used as a part of Cogema's MELOX plant and the La Hague Plutonium Finishing Facilities in France (DCS, 2000). Therefore, Cogema has proven that the aqueous polishing process produces plutonium oxide that meets their MOX feed material specification. It also appears that the proposed aqueous process improves upon the PUREX process by recovering and recycling acid and solvent to reduce the liquid waste streams. For the MFFF, the proposed aqueous process is highly automated and continuous. Also, the impurities in the feed material into the proposed aqueous process differs significantly from the PUREX process. PUREX, as applied by DOE, refined irradiated fuel so its waste stream had highly radioactive fission products, while the radioactive waste stream from MFFF is predominately alpha emitters of americium and uranium. This difference in the radioactive content of the removed impurities will also make the liquid wastes from the MFFF easier to manage and minimize their environmental impact in comparison to the high-level wastes produced from the SRS's PUREX process.

Solvent extraction plutonium polishing requires several discrete steps as presented in the MFFF documents (DCS, 2000 and DCS, 2001a). These steps are shown in Figure 1 and described below:

1. Dissolution: Plutonium oxide is electrochemically dissolved with silver (Ag^{2+}) in nitric acid forming a plutonium nitrate solution with the impurities.
2. Purification: An organic solvent (tributyl phosphate) in an aliphatic diluent (dodecane) extracts the plutonium nitrate and uranium from the nitrate solution. Nitrate impurities (i.e., americium, gallium, and silver) remain in the aqueous phase. The solution containing the uranium and plutonium nitrate is washed with nitric acid to ensure good decontamination (i.e., removal of impurities). The plutonium is then reduced to the trivalent state, Pu(III), by the introduction of hydroxylamine nitrate. The plutonium is removed from the solvent using a solution of nitric acid, hydrazine, and hydroxylamine nitrate. The plutonium valence is then oxidized back to Pu(IV) by driving nitrous fumes (NO_x) through the plutonium solution before being sent to the precipitation step. The waste streams are processed to separate the impurities, strip out the uranium from the solvent, and recover-recycle the acid and solvent.
3. Precipitation: The plutonium is precipitated with excess oxalic acid and collected on a filter as plutonium oxalate.
4. Oxidation-Homogenization: The moist oxalate is dried and calcined to PuO_2 . The homogenization unit mixes the calcined PuO_2 and feeds into the canning unit where the PuO_2 powder is packaged into reusable stainless steel cans for use in the MOX fuel-fabrication process.

It is important to note that because uranium impurities are contained in the plutonium oxide (predominately uranium-235 from the alpha decay of plutonium-239), the uranium is also extracted with

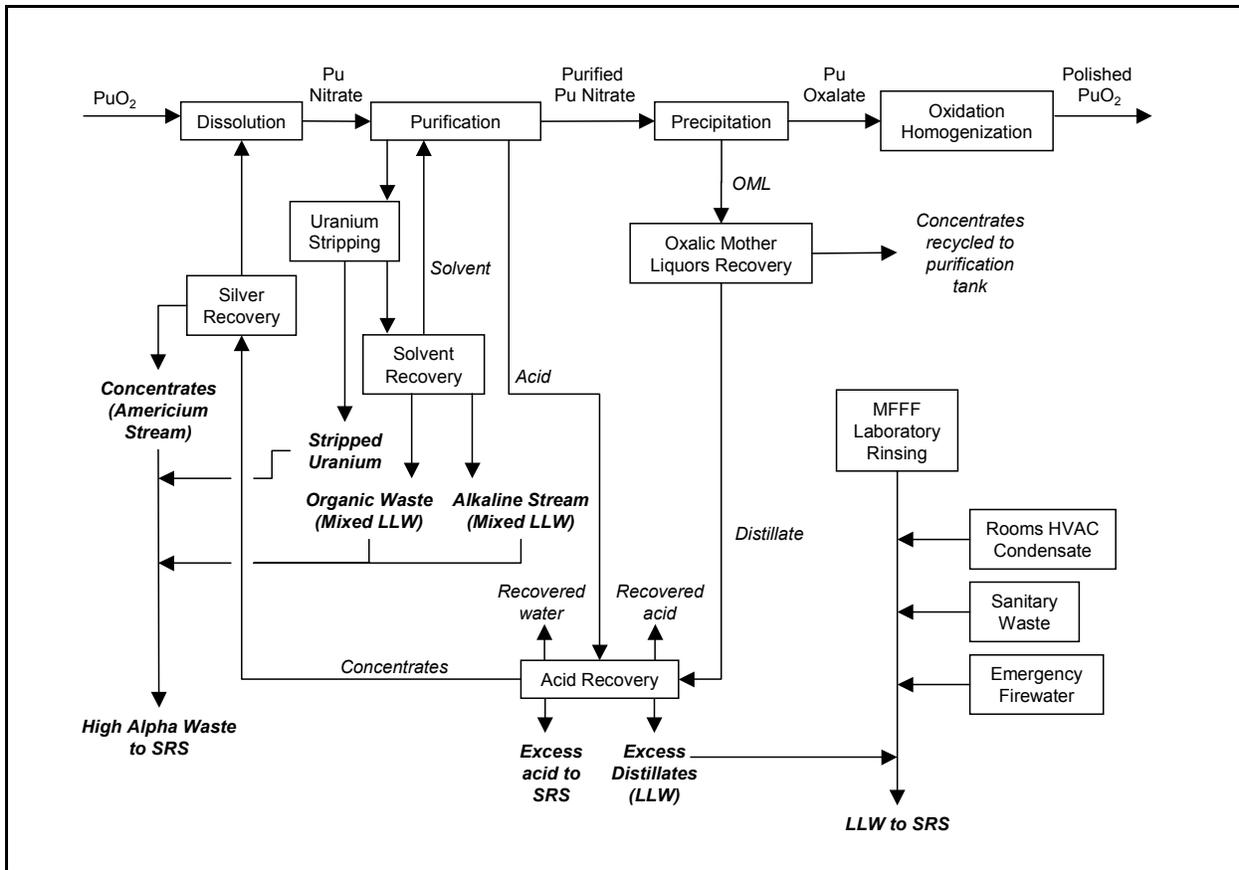


Figure 1 Aqueous Plutonium-Polishing Flowchart (based on Figure 11.3-1 of DCS, 2001a)

the plutonium nitrate from the nitrate solution into the organic solvent. Once the plutonium is removed, the organic solvent is routed to the uranium-stripping process where the uranium is removed with dilute nitric acid. Depleted uranium is added to reduce the uranium-235 concentration to below 30 percent isotopic concentration, thus reducing the potential for criticality. After the uranium-stripping process, the uranium stream is further diluted with additional depleted uranium to approximately 1 percent of uranium-235, mixed into the high alpha waste stream and sent to SRS (EA RAI Response to Question No. 8 of DCS, 2001c and CAR RAI Response to Question No. 135 of DCS, 2001d).

The intent of the proposed solvent-extraction process is to maintain the plutonium losses and the liquid waste generation as low as technically and economically possible by applying several recovery and recycling streams. Acid is recovered and reused from the raffinate stream, the NO_x -Air stripping, and the oxalic mother liquors distillates. The organic solvent, once the plutonium and uranium are both removed, is routed to solvent recovery mixer-settlers before being recycled back to the process. Silver is recovered from the concentrates remaining after acid recovery by a batch process using electrolytic separation and then is recycled back into plutonium oxide dissolution. The resulting wastes streams for the aqueous process are presented in Table 4.

Table 4 Plutonium-Polishing Waste Streams

Aqueous Solvent Extraction [Ion Exchange]			Dry	
Waste Stream	Annual Quantity ^{1, 2, & 3}	Chemical Concentration, Flow Rate, or Annual Quantity ^{1, 2, & 3}	Waste Stream	Annual Quantity ⁴
Impurities: Americium, Gallium, etc.	31,600 L ²	Am-241: <24.5 kg (Max. 0.7% by weight of Pu in the feed material) Pu: <150 g Ga: <42 kg (Max. 1% by weight of Pu in the feed material) H ions: 3 N Nitrate salts: 200 kg	Gallium (Ga)	35 kg for annual processing of 3.5 metric tons of Pu (Max. 1% by weight of Pu in the feed material). Note 5.
Excess Acid	5,000 L ²	Am: <14 mg H ions: 13.6 N	N/A	---
Stripped Uranium	134,000 L ²	Pu: <16 g U: <2150 kg (~1% U-235)	N/A	---
Distillate (or Acid Recovery Condensate)	320,000 L ²	Pu: <4E-03 mg Am-241: <0.8 mg Activity 1E+08 Bq	N/A	---
Solvent Regeneration Alkaline Wash	9,400 L ²	Pu: <13 g U: <13 g Na: <115 kg	N/A	---
Excess Solvent Residues [Spent Resin]	10,600 L ¹ [163 kg ³]	Solvent: 30% tributyl phosphate in branched-dodecane Pu: <17 mg [Reilex HPQ resin]	N/A	---
Rinsing Water (MFFF Lab. Rinsing, HVAC Condensate, Sanitary Washing)	500,000 L ²	Alpha activity: <5 Bq ω /L	Rinsing Water	Unknown
Off-gases	Various	Scrubbed NO _x : 300 m ³ /hr Effluents to the Acid Recover Unit: 50 L/hr	Off-gases	Unknown
Canning Decontamination	Unknown	Unknown	Canning Decontamination	40 L of sludge

1 - Source: Table 3-3 of DCS, 2000

2 - Source: DCS, 2001d

3 - Source: ORNL, 1998

4 - Source: Schwartz, 1998

5 - Source (Schwartz, 1998) gives 70 kg Ga for processing 3.5 metric tons of weapons grade Pu but this conflicts with the 1% by weight of the weapons grade Pu that is Ga (i.e., 3.5 metric tons of weapons grade Pu yields 35 kg of Ga).

Ion Exchange

The basic difference between aqueous plutonium polishing with solvent extraction or with ion exchange is in the second step; namely, separating the plutonium from the impurities in the aqueous solution occurs in an anion exchange resin column (ORNL, 1998, Schwartz, 1998). The ion exchange process requires a valence adjustment to the plutonium feed solution using nitric oxide gas at 60 to 80 °C before it enters the resin column. This adjustment is to ensure the plutonium is in the tetravalent state, Pu(IV), and to remove nitrite ions by air-sparging to prevent damage to the resin.

Multiple ion exchange columns of Reillex HPQ resin (ORNL, 1998) are loaded with feed solution to bind the plutonium with the resin. Each column would probably have the capacity to process 1 kg of plutonium per batch. The resin is washed with nitric acid (HNO₃) to remove the impurities and then undergoes elution of the plutonium with dilute nitric acid containing hydroxylamine nitrate to reduce the plutonium to the trivalent state, Pu(III). The plutonium washed from the resin columns is processed through the remaining two steps (Precipitation and Oxidation-Homogenization) previously described in Solvent Extraction section.

Differences Between Solvent Extraction and Ion Exchange

The principal difference between solvent extraction and ion exchange would be the replacement of an annual consumption of 2,800 liters of tributyl phosphate (from Table 3-2 of DCS, 2000) with 163 kg of ion exchange resin (Table 4.2 of ORNL, 1998). Differences in operation resources or in the waste generated have not been found to be significant as shown in the generic plutonium-polishing facility study (ORNL, 1998). Section 3.5 of ORNL, 1998 states that an annual process rate of 3.3 metric tons of plutonium³ would produce less than 5 kg (~0.004 m³/year) of solvent degradation products (dibutyl and mono butyl phosphate) or 163 kg (0.2 m³/year) of spent resin for solid waste treatment if either is used for the entire year.

Determining which process is superior depends on the process capability needed. Either technology can be made to work, but each is best applied in different scenarios: whether it is a continuous process to meet large capacity needs or multiple batch runs for lower capacity needs. For large capacity, the continuous solvent extraction process has lower manpower requirements through automation with lower associated exposure, waste, and potential for human error. Clearly, the use of the ion exchange is best for small lots or intermittent process operations.

Dry Process

Due to the well-known volume of liquid and corrosive waste generated from the aqueous form of plutonium polishing, DOE funded research and development for a dry plutonium polishing process for the purification of weapons-grade plutonium. This dry process is a subset of the Advanced Recovery and Integrated Extraction System (ARIES) developed at the LANL (Schwartz, 1998). ARIES is intended as the process stream for the Pit Disassembly and Conversion Facility (PDCF) (DOE, 2001).

ARIES (see flowchart of Figure 2) would accept the plutonium pits for processing and conversion to plutonium oxide to become the feed material into the MOX Fuel Fabrication process. Once the

³The ER states on page 1-2 that the annual process rate of the MFFF would be 3.5 metric tons (DCS, 2000).

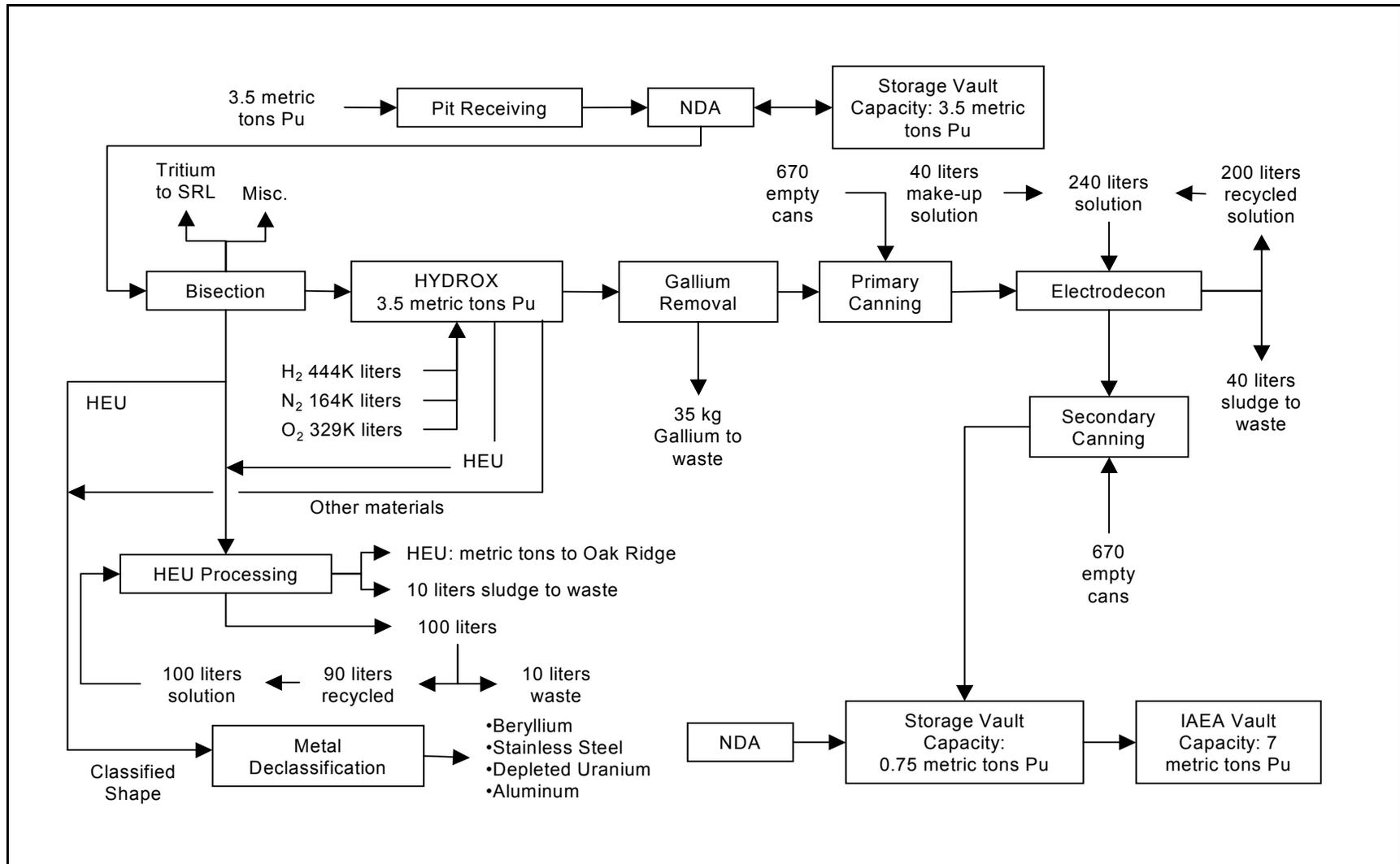


Figure 2 Pit Disassembly and Conversion Flowchart with Dry Plutonium Polishing with Yearly Throughput (Schwartz, 1998)

plutonium pits are bisected and processed to remove external equipment and non-plutonium components, the plutonium-bearing components are placed into the hydride/oxidation (HYDOX) unit and converted into plutonium oxide. The plutonium oxide then undergoes purification by thermally induced gallium removal (TIGR). During TIGR, the plutonium oxide material is heated in an atmosphere of argon with 6 percent hydrogen to a temperature of about 1100 °C. This high temperature ensures a relatively rapid process of the gallium oxide vaporizing as gallium suboxide and being pulled off by negative pressure for condensation, collection, and disposal. The processed plutonium oxide would then undergo packaging, decontamination, and placement into storage or transportation to the MFFF.

The dry process is a batch operation producing a gallium, an off-gas waste stream, and some sludge from canning decontamination by electrodecon (Table 4). There are several technical issues of the TIGR segment of ARIES that must be resolved before full-scale use. The primary technical issue is that only gallium is removed leaving the other impurities in the plutonium oxide (such as americium, aluminum, and uranium). From TIGR experiments, about 25 µg/g Pu (25 ppm) remains in the plutonium oxide after 24 hours exposure at 1200 °C (Kolman, 2000). Other sources describe that as much as 200 µg/g Pu (200 ppm) could remain in the plutonium oxide after TIGR (Schwartz, 1998). Such output levels from TIGR would not meet the MOX feed material requirement of less than 0.12 µg/g Pu (0.12 ppm or 120 ppb) (DCS, 2001b and DCS, 2001c). Thus, the plutonium oxide from the dry process would not be acceptable for use in the MOX fuel fabrication process without additional purification by an aqueous polishing process. There are also known technical issues to resolve with the collection of the vaporized gallium suboxide in a manner that prevents the deposition of corrosive gallium onto furnace and vent surfaces. Again, collector experiments have been performed (Philip, 1998) but have not been demonstrated under full-scale production conditions.

Because of the various technical issues, DOE did not pursue further development of TIGR and ended all funding after 1998 (James, 2001). Thus, while most aspects of the TIGR purification process have been demonstrated in a laboratory setting, the process has not been fully demonstrated nor characterized under full-scale production conditions (Schwartz, 1998).

COMPARISON OF PROPOSED AND ALTERNATIVE PROCESS WASTE STREAMS

As shown in Figure 1 and Table 4, there are eight waste streams from the aqueous process. Based on the information provided in the ER and the responses to the NRC's request for additional information (DCS, 2000 and DCS 2001c), the total annual waste volume would be approximately 1.01 million liters (267,000 gallons). From Figure 2 and Table 4, the principal waste stream for the dry process would be no greater than 1 percent of the mass of the weapons-grade plutonium processed⁴. However, because the ARIES process was not fully developed and tested in a production-scale facility, any other waste streams associated with full operations and quality control of the dry process (such as sampling rising water, equipment maintenance, and cleaning) cannot be determined.

It is also clear that the plutonium oxide produced by the dry process, by itself, would not meet the technical requirements for the MOX feed material. Thus, an additional aqueous process would still be

⁴While Figure 2 shows for the dry process a sludge waste stream from electro-decontamination of the plutonium oxide cans of about 40 liters, a comparable value from the MFFF documents is not available. However, due to the significant technical and developmental differences between the two processes, this lack of information does not change the final result.

necessary after the dry process to further purify the plutonium oxide to the MOX feed material requirements of less than 0.12 $\mu\text{g/g}$ Pu (0.12 ppm or 120 ppb). As discussed previously and in Section 11.2.6 of the CAR, the level of gallium after the aqueous process would be less than 0.1 $\mu\text{g/g}$ Pu (0.1 ppm) in the MOX feed material (DCS, 2001a) and would meet the MOX feed material requirements. For these reasons, a direct comparison between the aqueous and dry polishing processes waste streams cannot be made.

There are additional factors that also should be considered when evaluating the aqueous process waste streams. The radioactive waste stream from the plutonium-polishing process for the MFFF results in predominately americium and uranium along with non-radioactive materials such as gallium, aluminum, etc. As described in Section 10.1.4.1.1 of the CAR, the high-alpha-activity stream is made up of the silver recovery and alkaline waste streams (DCS, 2001a). The silver recovery waste stream contains the removed impurities (gallium, americium); trace amounts of silver, plutonium, and uranium; and excess acid. The solvent regeneration and the stripped uranium waste streams are pumped into the high-alpha-activity stream, and all are sent to SRS high-level radioactive waste (HLW) tanks (EA RAI Response to Question No. 8 of DCS, 2001c).

The information available in the CAR indicates that many of the aqueous process waste streams have a dilute nature (see Table 4). The dilute streams could be volume-reduced very effectively if needed (e.g., by ion exchange or evaporation). However, DCS elected via their contract to have the SRS treat these wastes via storage in the HLW tanks with eventual vitrification by the Defense Waste Processing Facility (DWPF).

CONCLUSIONS

This report presents a summary of the technical issues and an initial comparison of the possible waste streams for two plutonium-polishing processes. Based on the information in the documents currently available, it is clear that the solvent extraction aqueous process is the best technology for the MFFF. Because of this clear technological advantage and the lack of information available for the dry process, issues such as the relative hazards or environmental safety and health impacts were not evaluated. In addition, a true direct comparison of the waste streams between the aqueous and dry processes cannot be made since the dry process was not nor will be fully developed and evaluated.

The aqueous plutonium polishing with solvent extraction is proposed for the MFFF due to the apparent advantages of being a mature technology and the capacity for removal of a significant quantity of all impurities found in weapons-grade plutonium. The solvent-extraction process also lends itself better to a continuous automated process stream with the other segments of the complete plutonium-polishing process. The ion exchanger process could also remove the impurities; however it does not add any value over solvent extraction in reducing waste stream volumes nor in process compatibility.

While the dry process may appear to be advantageous for a reduced waste stream, it can only remove gallium and to a level that apparently would not meet the MOX feed material technical specification. This leaves the remaining gallium and other impurities still to be removed by a follow-on purification process to meet the MOX fuel feed material technical requirements. There is an additional technical disadvantage with the dry process in the collecting the gallium suboxide vapor waste stream and there could be operational and maintenance problems with scaling up to a production facility.

It must be noted that additional references concerning the dry plutonium process were requested but are not currently available to include in this report. However, it is anticipated that they would only provide additional details for the technical or production difficulties and potentially for the material and quantity in the dry process's waste streams. Therefore, the additional reference material should not affect the conclusions of this report.

REFERENCES

Bechtel Hanford, Inc. (BHI). "Plutonium-Uranium Extraction Plant (PUREX) Hazards Assessment," BHI-03158, Rev. 1, July 18, 2001.

Duke Cogema Stone & Webster (DCS). *Mixed Oxide Fuel Fabrication Facility Environmental Report*. December 19, 2000.

---, 2001a. *Mixed Oxide Fuel Fabrication Facility Construction Authorization Request*. February 28, 2001.

---, 2001b. *Fuel Qualification Plan*, DCS-FQ-1999-001, Rev. 2. April 2, 2001.

---, 2001c. *ER RAI Responses*. July 12, 2001.

---, 2001d. *Mixed Oxide Fuel Fabrication Facility Construction Authorization Request Responses to NRC Request for Additional Information*. August 31, 2001.

James, C. of Los Alamos National Laboratory. Personal communications with D. Palmrose of ATL International, Inc., on July 5, 2001.

Kolman, D.G., et al. "Thermally Induced Gallium Removal from Plutonium Dioxide for MOX Fuel Production," *Journal of Nuclear Materials*, Vol. 282, pages 245-254. July 17, 2000.

Oak Ridge National Laboratory (ORNL). "Technical Report for Generic Site Add-On Facility for Plutonium Polishing." ORNL/TM-13662. June 1998.

Philip, C.V., W.W. Pitt, and C. Beard. "Investigations in Gallium Removal." Amarillo National Resource Center for Plutonium, ANRCP-1997-3. November 5, 1997.

Philip, C.V., R.G. Anthony, and S. Chokkaram. "Collector for Recovering Gallium from Weapons Plutonium." Amarillo National Resource Center for Plutonium, ANRCP-1998-14. September 1998.

Schwartz, K.L. "Process Modeling of Plutonium Conversion and MOX Fabrication for Plutonium Disposition." Amarillo National Resource Center for Plutonium, ANRCP-1998-17. October 1998.

U.S. Department of Energy (DOE). Office of Environmental Management. *Linking Legacies: Connecting the Cold War Nuclear Weapons Production Processes to Their Environmental Consequences*, Appendix B: Eight Major Processes, DOE/EM-0319, p. 175. January 1997.

---. Office of Fissile Materials Disposition. *Draft Surplus Plutonium Disposition Environmental*

Impact Statement, Appendix N: Plutonium Polishing, DOE/EIS-0283, Volume I - Part A. November 1999.

--- . “National Nuclear Security Administration - Pit Disassembly and Conversion.” 2001.
<http://www/doe-md.com/pu_pdcf.htm> (16 July 2001).