APPENDIX C
DESCRIPTION OF PROCESSING TECHNOLOGIES

C.1 INTRODUCTION

This appendix presents detailed descriptions of the technologies evaluated in the Environmental Impact Statement (EIS) for the processing of certain plutonium residues and scrub alloy stored at the Rocky Flats Environmental Technology Site (Rocky Flats) so that they are brought into compliance with safeguards termination limits for ultimate disposition. The chronological development of the safeguards termination limits as a part of overall safeguards protection is presented in a series of memos and letters. The most relevant of these are included in Attachment 1 to this appendix. This appendix also describes the screening process and approach used to select and evaluate the most suitable processing technologies for these materials in the proposed action. Processing technologies discussed include those that do not remove plutonium from the material (e.g., immobilization technologies) and those that separate plutonium from the material (e.g., acid dissolution technologies). No Action Alternative processing technologies that were analyzed in the *Environmental Assessment, Finding of No Significant Impact, and Response to Comments-Solid Residue Treatment, Repackaging, and Storage* (DOE 1996d), or Solid Residue Environmental Assessment, are also presented in this appendix.

C.2 SCREENING AND EVALUATION APPROACH

The U.S. Department of Energy (DOE) used a screening process to identify a reasonable set of technologies for detailed evaluation in this EIS. In selecting these technologies, a number of factors were considered, including the following:

- Direct applicability of the technology to the particular material type
- Maturity and timing of the technology so that processing could be accomplished in the 1998-2004 time frame within reasonable cost
- Potential impact of processing technology implementation to ongoing mission activities at the site
- Experience of the DOE site in employing the technology and availability of facilities and equipment
- Minimization of the number of process steps to reduce worker exposures
- Amount of secondary wastes generated and appropriate secondary waste disposition methods.

The initial screening process began with the assessment of a wide range of potential processing technologies that were identified in the following DOE studies and during the public scoping process.

❒ **Trade Studies (DOE 1995, 1996a, 1996b, 1996e, 1996f, and 1997a)—**DOE conducted a series of trade studies to identify the best possible technologies for stabilizing plutonium residues to an end state suitable for disposition. The trade studies were developed by the DOE Nuclear Material Stabilization Task Group, which comprised representatives from the DOE sites that store plutonium residues or have capabilities in treating the residues, as well as DOE Headquarters and other interested individuals. The trade studies resulted in a technical assessment of various approaches and a bounding of the range of alternative stabilization approaches for further consideration.

- ❐ **Environmental Assessment—**DOE prepared the Solid Residue Environmental Assessment (DOE 1996d) to address the environmental impacts associated with stabilizing the entire 106,600 kilogram (kg) (235,000 pound [lb]) inventory of Rocky Flats plutonium residues to allow for safe storage pending final disposition. This environmental assessment addressed stabilization technologies that would provide for safe storage. It did not address technologies for the further processing needed to comply with safeguards termination limits required for ultimate disposition because the environmental assessment was prepared before the safeguards termination limits were developed and implemented. The "no action" alternatives in this EIS are based on the technologies identified in the Solid Residue Environmental Assessment.

- ❐ **The Rocky Flats Residue Rebaselining Study—**Rocky Flats prepared a study entitled *Residue Rebaselining for Combustibles, Fluorides, Ash, and Miscellaneous Residues* (DOE 1997b) to identify the most viable options for removal of residues from the site. This study resulted in proposed paths and alternative technologies for preparing the residues for final disposition (i.e., to meet or exceed the safeguards termination limits).

- ❐ **Public Scoping Comments—**Public scoping comments were received by DOE during the November 1996 to January 1997 time frame and were considered during the screening process. Many of the comments included preferences for certain technology paths and locations for stabilization.

After the initial screening process, DOE Headquarters requested the candidate processing sites (Rocky Flats, the Savannah River Site, Los Alamos National Laboratory, and Lawrence Livermore National Laboratory) to assess the technologies identified in the initial screening process from a site-specific perspective, considering the screening and evaluation factors described previously. Each site provided input on which technologies could be implemented at their site, taking into account their respective capabilities, facilities, and equipment. Working sessions were held between DOE candidate site and Headquarters representatives to review the benefits and constraints of processing technologies at each site and to reach consensus on the sites and technologies that should to be evaluated in this EIS.

The technologies described herein were determined to be technologically mature enough to be considered as viable options for stabilization of the various residue materials at Rocky Flats.

A further discussion of the screening process is given in Sections 2.3 and 2.9 of this EIS.

C.3 PROCESSING TECHNOLOGIES FOR ROCKY FLATS PLUTONIUM RESIDUES AND SCRUB ALLOY

The following sections give detailed descriptions of the candidate processing technologies for each of the material categories discussed in Appendix B of this EIS. The proposed technologies are presented by material category in **Table C–1**. Each material has a No Action processing technology–Alternative 1 (Section C.4), a processing technology without plutonium separation–Alternative 2 (Section C.5), and a processing technology with plutonium separation –Alternative 3 (Section C.6). In addition, DOE has combined certain elements of Alternatives 1 and 2, with the application of a variance to safeguards termination limits, for the residues to form an Alternative 4.

Table C–1 Candidate Process Technologies by Material Category

Material Category	*No Action Stabilization*	*Processing without Plutonium Separation*	*Processing with Plutonium Separation*	*Combination of Processing Technologies*
Ash Residues				
Incinerator Ash	Calcination and cementation	Calcination and vitrification Blend down Cold Ceramification	Purex with ash fusion preprocessing Mediated electrochemical oxidation with preprocessing	Calcination/Cementation Repackaging
Sand, Slag, and Crucible	Calcination and cementation	Vitrification Blend Down	Purex with preprocessing	Calcination/Cementation Repackaging
Graphite Fines	Calcination and cementation	Vitrification Blend Down Cold Ceramification	Mediated electrochemical oxidation with preprocessing	Calcination/Cementation Repackaging
Inorganic	Calcination and cementation	Vitrification Blend Down Cold Ceramification		Calcination/Cementation Repackaging
Salts Residues				
Electrorefining Salts	Pyro-oxidation	Pyro-oxidation and blend down	Pyro-oxidation and salt distillation Pyro-oxidation and water leach with plutonium oxide recovery Salt Scrub with Purex processing of newly created alloy	Repackaging
Molten Salt Extraction	Pyro-oxidation	Pyro-oxidation and blend down	Pyro-oxidation and salt distillation Pyro-oxidation and water leach with plutonium oxide recovery Salt scrub with Purex processing of newly created scrub alloy	Repackaging
Direct Oxide Reduction	Pyro-oxidation	Pyro-oxidation and blend down	Water leach with plutonium oxide recovery Acid dissolution with plutonium oxide recovery Salt scrub with Purex processing of newly created alloy	Repackaging
Combustible Residues				
Aqueous-contaminated	Neutralization and drying	Sonic wash Catalytic chemical oxidation	Mediated electrochemical oxidation	Neutralize/Dry
Organic-contaminated	Thermal desorption and steam passivation	Blend down		Thermal Desorption/Steam Passivation
Dry	Repackaging			Repackaging

Material Category	*No Action Stabilization*	*Processing without Plutonium Separation*	*Processing with Plutonium Separation*	*Combination of Processing Technologies*
Plutonium Fluoride Residues	Acid dissolution with plutonium oxide recovery	Blend down	Acid dissolution with plutonium oxide recovery Purex with plutonium metal or oxide recovery	Does not apply
Filter Media Residues	Neutralization	Vitrification (HEPA filters only) Blend down (HEPA filters only) Sonic wash	Mediated electro-chemical oxidation	Neutralize/Dry (IDC 338 only) Repackaging (All Other Filter Media)
Sludge Residues	Filtration and drying	Vitrification Blend down	Acid dissolution with plutonium oxide recovery (except IDCs 089, 099, and 332)	Filter/Dry (Except IDCs 089, 099, and 332) Repackage (IDCs 089, 099, and 332)
Glass Residues	Neutralization and drying	Vitrification Blend down Sonic wash	Mediated electrochemical oxidation	Neutralize/Dry
Graphite Residues	Repackaging	Cementation Vitrification Blend down	Mediated electrochemical oxidation	Repackaging
Inorganic (Metal and Other) Residues	Repackaging	Vitrification Blend down	Mediated electrochemical oxidation	Repackaging
Scrub Alloy	Repackaging	Vitrification	Purex with plutonium metal or oxide recovery	Does not apply

HEPA = high-efficiency particulate air

The technology descriptions consist of a summary of the technology process; flow chart diagrams; and a description of each process step. The proposed technologies are as follows:

❒ No Action Processing Technologies

- Calcination and cementation of ash residues
- Pyro-oxidation of pyrochemical salts
- Neutralization and drying of aqueous-contaminated combustibles
- Thermal desorption and steam passivation of organic-contaminated combustibles
- Repackaging of dry combustibles
- Acid dissolution and plutonium oxide recovery of plutonium fluorides
- Neutralization of filter media
- Filtration and drying of sludge residues
- Neutralization and drying of glass residues
- Repackaging of graphite residues, inorganic residues, and scrub alloy.

❒ Technologies without Plutonium Separation

- Immobilization (vitrification)
- Immobilization (cementation)
- Blend down

- Pyro-oxidation and blend down of pyrochemical salts
- Sonic wash
- Catalytic chemical oxidation of combustibles
- Cold ceramification.

❒ **Processing Technologies with Plutonium Separation**

- Purex process with plutonium metal or oxide recovery
- Mediated electrochemical oxidation
- Salt distillation
- Water leach with plutonium oxide recovery
- Salt scrub with Purex processing of newly created scrub alloy
- Acid dissolution with plutonium oxide recovery.

❒ **Combination of Processing Technologies**

(In addition to these processes, materials may also be blended with low plutonium concentration materials or inert materials to achieve a 10 percent plutonium concentration and a variance to safeguards termination limits would be applied.)

- Calcination/Cementation
- Repackaging
- Pyro-oxidation
- Neutralization/Drying
- Thermal Desorption and Steam Passivation
- Filtration and Drying

For each technology, it would be necessary to perform a nondestructive assay after the packaging process to ensure compliance with interim safe storage criteria, Waste Isolation Pilot Plant (WIPP)/Waste Acceptance Criteria, and TRUPACT II shipping requirements. The WIPP/Waste Acceptance Criteria are summarized in Table 2–5 of this EIS. One of the criteria limits the amount of packaged fissile gram equivalents to 200 per drum. The assay would allow for maximizing the amount of container loadings, which in turn would minimize the number of drums destined for interim site storage and disposal. The assay would be performed using either neutron multiplicity counters in concert with gamma-ray isotopic spectrometers or by using segmented gamma scanners.

For shipment to WIPP, there are criteria that must be followed in using the TRUPACT II shipping container. Based on these criteria, the residues, where necessary, would be packaged according to the maximum allowable plutonium—83.5 grams (g) (2.9 ounces [oz])—per individual packing container. Since there are two containers per drum, this would ensure that the 200-g limit per drum would not be exceeded. For other residues, where the quantity of plutonium per package is too low to be of concern regarding packaging, the weight of each individual package becomes a concern for handling within a glovebox. An estimated 9.1 kg (25 lb) would be used as a basis for this type of packaging. These values are used throughout this appendix, where appropriate.

C.4 DETAILED PROCESS DESCRIPTIONS FOR NO ACTION PROCESSING TECHNOLOGIES

C.4.1 Calcination and Cementation of Ash Residues

The proposed cement-based immobilization process is an adaptation of a cement-based waste immobilization process that has been used within the DOE complex and the commercial nuclear industry. This process was approved by the U.S. Environmental Protection Agency (EPA) as a best demonstrated available technology for use in waste stabilization. At Rocky Flats, cement-based waste immobilization processes have been operated successfully for several years and have produced thousands of cubic yards of solidified waste. The process has been used for the solidification of low-level waste (saltcrete) in Building 374 and for the solidification of transuranic waste in Building 774 (bottlebox process). The cement-based ash residue solidification process would take place in either Building 707 or Building 371, and it would be an in-container solidification process.

Calcination of powdered or granular materials in muffle furnaces is considered to be a proven technology. Capabilities necessary to satisfy all alternatives are currently being installed at Rocky Flats as part of the ongoing stabilization programs, and should be operational within several months of issuance of the EIS. Cementation of materials necessary to immobilize fines and to form an acceptable solid is considered to be a proven technology, although optimization studies are routinely performed to improve specific characteristics. Rocky Flats would have to install or remodel gloveboxes to provide additional area for the curing step, so approximately one year would be required after the issuance of the Record of Decision before the cementation capability would be fully operational. The specific location of the cementation processing is uncertain, although the process would be consistent with either Building 371 or Building 707.

The cement-based immobilization process is shown in **Figure C–1**. The process steps are drum unloading and bag-in, feed preparation for calcination, calcination, feed preparation for cementation, in-line nondestructive assay, cement mixing station, curing and bag-out, and final drum packaging and storage.

Cement-based immobilization would blend cement and water with the prepared ash residues. The advantage of cement-based immobilization technology is its proven performance. Well-established protocols, when followed, ensure an acceptable final product. Elements included within these protocols include waste characterization, both physical and chemical, treatability formula development, bench scale testing, pilot scale studies, and detailed project planning for full-scale operations.

A cement-based immobilization process has several disadvantages associated with it. The mixing of the cement and water components produces heat during the curing process. Any active metals remaining in the residue stream after calcination may react with water to produce hydrogen gas. Also, during mixing, curing, and after final packaging, the potential for hydrogen generation exists due to both radioalysis and hydrolysis of the water of hydration by the radiological and reactive metal components, respectively.

Two waste streams would be generated in addition to the cemented residue product stream. The first waste stream would be a solid transuranic waste stream consisting of size-reduced steel containers, plastic containers and plastic bags. The second waste stream would be a gaseous effluent stream consisting primarily of water vapor, nitrogen, hydrogen, carbon dioxide and particulates. Any tramp material removed from the waste before size-reduction either would be combined with the solid transuranic waste stream or would be placed into a cemented waste container before curing.

❒ Detailed Process Description

Drums with capacity of 208-liter (L) (55-gallon [gal]) would be transferred from storage into a contamination control enclosure. The contamination control enclosure is designed to control airflow in the event of a bag failure within a drum. The drums would be opened and the integrity of the packaging would be checked. If the packaging has not been compromised, the containers would be transferred into the glovebox. The containers would be removed from the drums and bagged into the glovebox. If the integrity of the packaging has been compromised, the package would be overpacked with a new plastic bag before transfer to the glovebox.

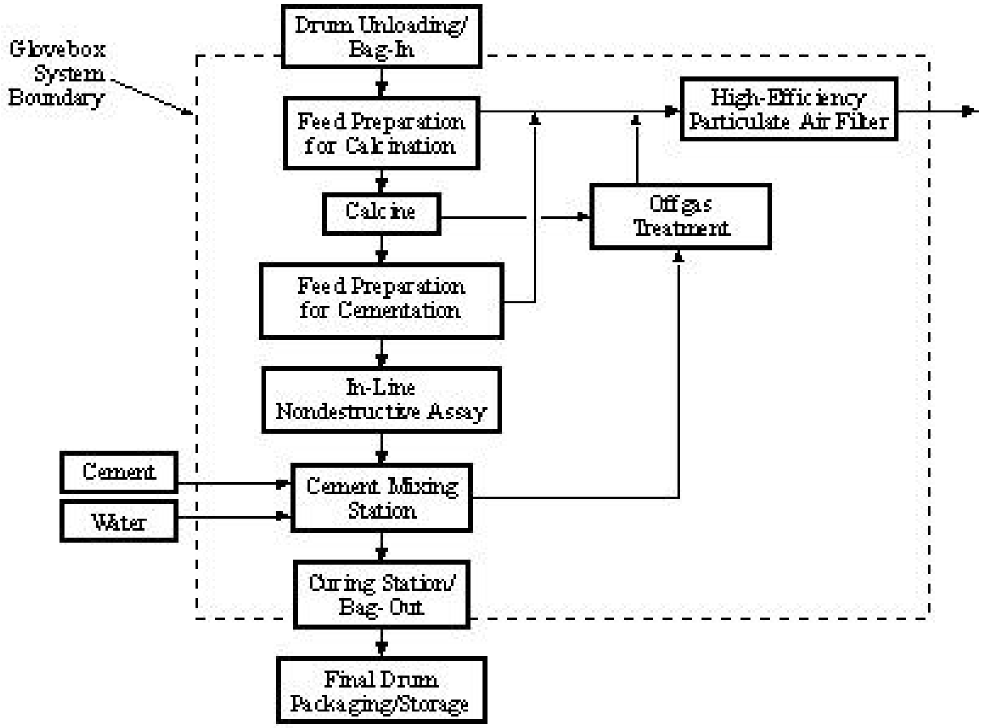

Figure C–1 Cement-Based Immobilization Process for Ash Residues

After bag-in, the Item Description Codes (IDCs) of the residue containers would be verified and the original residue containers would be transferred to a residue sorting and loading station, which would provide local dust control and would contain a 1/8" sieve that would be used to separate all oversized residue and tramp material (e.g., nuts, bolts). The sieved residue fines would be transferred into burn boats. Tramp material would be separated and transferred for transuranic waste size reduction and packaging or placed into cemented waste containers before curing. Oversized residue would be crushed and fed back to the loading station for sieving. Each burn boat would be filled to contain approximately 83.5 g (2.9 oz) of plutonium. After the filling step, the burn boats would be transferred to a furnace for calcination.

Calcination is required to high-fire the incinerator ash residue which would remove the reactive characteristics in the residue stream. Each batch would be calcined at 900 degrees Celsius (°C) [1,650 degrees Fahrenheit (°F)] for 4 hours, which would oxidize carbon and organic to carbon dioxide and eliminate water, thereby increasing the bulk density of the ash residue. After cooling, the residue would be transferred for feed preparation for cementation.

The burn boats containing the calcined residue will be transferred to a residue sorting and loading station. As described previously, the residue would be sieved and the residue fines would be loaded into metal containers. As required, oversized residue will be crushed and loaded into the containers. Each container would be filled to contain approximately 83.5 g (2.9 oz) of plutonium. After the filling step, the containers would be transferred to an in-line nondestructive assay station.

Following nondestructive assay, the container would be moved to the mixing station. Then, measured quantities of water and cement would be blended into the residue containers. The material would be mixed until all of the water has been absorbed by the cement and the mixture thickens. Because of the potential for heat generation, provisions for actively cooling the container during and after mixing may be required for certain residue IDCs. During mixing, there would be a potential for vapor generation produced by an exothermic reaction associated with the hydration of the cement and through hydrogen gas generation produced from radioalysis and hydrolysis. Therefore, provisions would be incorporated as necessary for the collection and extraction of these vapors in both the mixing station and curing station.

The container would be moved from the mixing station into a set of curing gloveboxes and set aside for a 24-hour curing period. After curing has been completed, the cans would be bagged out of the glovebox. Assayed, cemented residue containers would be transferred for final drum packaging. The container would be loaded into pipe components already staged in the drums. The drums would be placed in interim storage until a final disposition decision is made.

C.4.2 Pyro-Oxidation of Pyrochemical Salts

Pyro-oxidation technology converts reactive metals in salt residues to nonreactive oxides. The resulting products would be stored as stabilized plutonium salts at Rocky Flats. This technology would be used on all three types of pyrochemical salt residues, including electrorefining salts, molten salt extraction salts, and direct oxide reduction salts. The pyro-oxidation process would be conducted inside gloveboxes located in Module A of Building 707 or in Building 371.

Pyro-oxidation of salts in stationary furnaces is considered to be a proven technology, although specific process variables are being evaluated in an attempt to make the pyro-oxidation process more compatible with a pyro-distillation follow-on processing step. Pyro-oxidation of reactive salts is part of the Rocky Flats response to Defense Nuclear Facilities Safety Board Recommendation 94-1 to stabilize potentially higher-risk or reactive materials. Rocky Flats has the capability to support the ongoing stabilization programs, and operations are pending. While not a technology risk for the pyro-oxidation process, the salts, once pyro-oxidized, cannot be subsequently salt scrubbed, which is the only current process to allow plutonium separation using the Purex process. The on-going stabilization program trades the technical and programmatic risk of not using a proven Savannah River Site residue disposition approach (Purex) against the reduction of an immediate safety risk. The pyro-oxidation process is, however, a prerequisite step for both salt distillation and aqueous distillation.

The pyro-oxidation process for pyrochemical salts is shown in **Figure C–2**. The salt residues would be sorted and batched in preparation for pyro-oxidation. The residues would be pyro-oxidized to convert reactive metals to oxides. After pyro-oxidation, the oxidized plutonium salts would be packaged for storage. The packaged material would be removed from the glovebox, nondestructively assayed for accountability purposes, and transferred to plutonium storage.

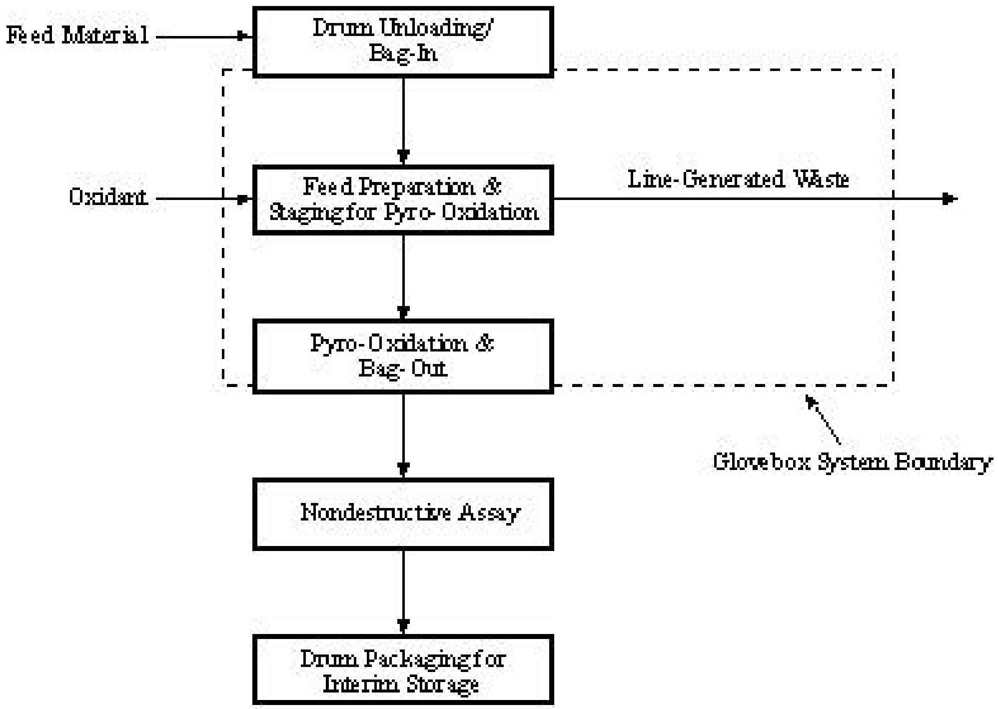

Figure C–2 Pyro-Oxidation Process for Pyrochemical Salts

❐ Detailed Process Description

Drums would be manually transferred into a contamination control enclosure and unpacked. This step is to contain any contamination which could result from an individual package containment damaged by radiolysis, or physical damage to the package during storage. Any unnecessary packaging materials would be removed to limit the amount of packaging introduced into the salt feed preparation glovebox.

The salt feed would be introduced into the glovebox, one stream at a time, and the IDC verified. The individual packages would be opened and loaded into a crucible in preparation for pyro-oxidation. Sodium carbonate or another oxidant would also be added to the crucible at this time. Combustible packaging materials from the individual packages would be bagged out of the glovebox and managed as transuranic waste. Other materials would be bagged out and managed appropriately.

Once the crucible is loaded with salt feed, it would be placed in a glovebox furnace and heated to approximately 800°C (1,470°F) with sodium carbonate or another oxidant as a reagent for 2 to 3 hours, stirring continuously (8-hour cycle time). The product would be a stabilized plutonium salt matrix. Pyro-oxidation can be applied both to a sodium chloride/potassium chloride matrix and to a calcium chloride matrix. This process converts reactive metals (such as calcium and sodium) to oxides. When the furnace has cooled to below 100°C (212°F), the crucible would be removed from the furnace. During the heating, stirring, and cooling phases, argon would flow through the furnace. During the last part of the heating phase, argon would be replaced by a mixture of air and argon.

Once the crucible is removed from the furnace, it would be allowed to cool. Because metal crucibles are used, the pyro-oxidized salt would remain in the crucibles to be sealed and bagged out directly in nominal 2.5–kg (5.5 lb) bulk (net) batches. The material would then be bagged from the glovebox and placed into containers for plutonium storage.

Nondestructive assay would be performed to ensure requirement limits are met and to obtain data to ensure that required accountability procedures are followed. Nondestructive assay methods would be selected to ensure that the best accountability data are obtained. Assayed product packages containing the plutonium-bearing salt matrix would be transported to appropriate plutonium storage areas.

C.4.3 Neutralization and Drying of Aqueous-Contaminated Combustibles

Aqueous-contaminated combustible residues include acidic liquids and generally do not have free liquid present. They were generated from an aqueous process and some degree of moisture will be present. The neutralization and drying process for aqueous-contaminated combustible residues removes the nitric acid from the organic matrix, eliminating a possibly unstable condition. The residue consists of materials, such as cloth, paper, rags, coveralls, rubber, wood, and other miscellaneous materials, some of which is above the safeguards termination limit for combustibles. The neutralization and drying process is not intended to remove the plutonium from the residue. As a result, this process would preclude ultimate shipment to WIPP unless the residue is subjected to further processing. This process would be conducted in Room 3701 of Building 371.

The neutralize-dry process, consisting of washing materials in alkaline solutions, allowing them to drain or partially dry, and mixing the resulting solids with water-absorbing materials, is considered to be a proven technology. The capability for Rocky Flats is being installed to support the disposition of below-safeguards termination limit materials, and should be available several months after the issuance of the EIS. Activities are underway to optimize the process and reduce the quantity of water-absorbing materials required for meeting disposal requirements.

The neutralization and drying process for aqueous-contaminated combustible residues is shown in **Figure C–3**. The process steps are drum unloading and bag-in, feed preparation, the neutralization and decant/filtration, oven drying, and packaging and bag-out. Nondestructive assay would be performed and the drums would be packaged for interim site storage.

❒ Detailed Process Description

Drums with the capacity of 208-L (55-gal) would be transferred from storage into a contamination control enclosure and unpacked. The contamination control enclosure is designed to control airflow in the event of a bag failure within a drum. The drums would be opened and the integrity of the packaging would be checked. If the packaging has not been compromised, the package would be transferred into the glovebox.

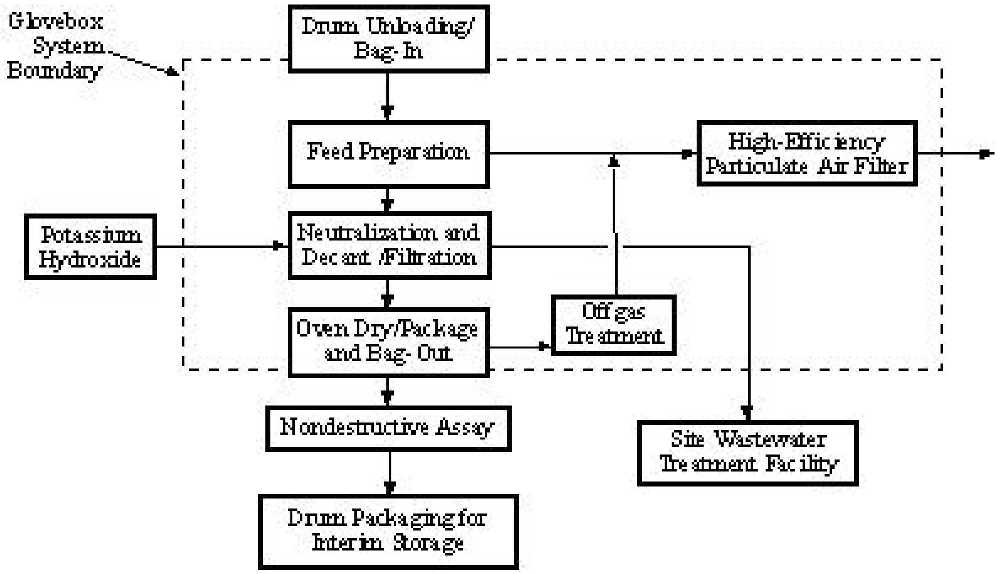

Figure C–3 Neutralization and Drying Process for Aqueous-Contaminated Combustible Residues

Any unnecessary packaging materials, would be removed to limit the amount of packaging introduced into the glovebox. If the integrity of the primary packaging has been compromised, the package would be overpacked with a new plastic bag before transfer to the glovebox.

Following bag-in, the IDCs would be verified and the plastic bags would be unpacked and the residue sorted. Each bag would be opened to remove any tramp metal or other unwanted materials. This material would be bagged out of the glovebox and managed appropriately. Following the sorting, the residue feed material would be shredded and batched to 5-kg (11-lb) batches for neutralization.

Neutralization is intended to remove the nitrate contamination from the combustible waste and to neutralize any residual nitric acid contained within the residue. The 5-kg (11-lb) batches of combustible would be washed with 50 L (13.2-gal) of water containing 10 percent excess potassium hydroxide. After 2 hours, the acid would be neutralized, forming potassium nitrate and water. None of the plutonium would be removed from the residue during the neutralization process. The combustible solids would be separated from the nitrate solution and processed through decanting and filtration. The combustible solids would contain approximately 20 percent solution by weight with a proportionate quantity of nitrates. These solids would be transferred to a drying pan. The neutralization solution would be sent, at intervals, to Building 374 for evaporation using the site wastewater treatment process.

The combustible residue that was neutralized and transferred to a drying pan would then be placed into a drying oven. The residue would be dried under a vacuum at 80°C (176°F) for 2 hours. Offgas from drying will be treated before high-efficiency particulate air filtration. After cooling, the combustible residue would be weighed, and the quantity of plutonium estimated as the waste would be transferred to 8.2-L (2.2-gal) containers. Each container would contain approximately 44.6 g (1.6 oz) of plutonium with the container loading based on an approximate bulk density of the solids of 0.3 kg/L (2.5 lb/gal). These containers would be bagged out of the glovebox and packaged into convenience cans. None of the plutonium would be removed in the neutralization; therefore, the plutonium remaining in the combustibles would be above the safeguards termination limit and would preclude shipment and disposal of this population at WIPP.

Nondestructive assay would be performed, and then the assayed and repackaged residue containers would be transported for drum packaging. These 208-L (55-gal) drums cannot be shipped to WIPP because the percentage of plutonium in the waste exceeds the safeguards termination limit. The drums would remain in the interim site storage until subjected to an appropriate stabilization process that would reduce the plutonium content below the safeguards termination limit.

C.4.4 Thermal Desorption and Steam Passivation of Organic-Contaminated Combustibles

Thermal desorption and steam passivation removes residual organic contaminants from organic-contaminated combustible residues and converts plutonium fines in the residue to plutonium oxide. Organic-contaminated combustible residues consist of materials, including wet and dry combustibles and leaded rubber gloves, some of which are above the safeguards termination limit for combustibles. The thermal desorption and steam passivation process and the repackaging of this material would satisfy the requirements for safe interim site storage. This process would be conducted in Room 3701 of Building 371.

Thermal desorption/steam passivation to remove volatile organics and oxidize plutonium fines is considered to be a proven technology; however, the processing times are currently under investigation as are final process parameters. The capability for Rocky Flats is being installed to support the disposition of below-safeguards termination limit materials, and should be available several months after the issuance of the EIS.

The thermal desorption and steam passivation process for organic-contaminated combustible residues is shown in **Figure C–4**. The process steps include drum unloading/bag-in and feed preparation, followed by thermal desorption and steam passivation. Absorbent is added and the material repackaged and bag-out. After nondestructive assay is performed, the final drum packaging and storage would take place.

❒ Detailed Process Description

Drums with the capacity of 208-L (55-gal) would be transferred from storage into a contamination control enclosure and unpacked. The contamination control enclosure is designed to control airflow in the event of a bag failure within a drum. The drums would be opened and the integrity of the packaging would be checked. If the packaging has not been compromised, the package would be transferred into the glovebox. Any unnecessary packaging materials would be removed to limit the amount of packaging introduced into the glovebox. If the integrity of the primary packaging has been compromised, the package would be overpacked with a new plastic bag before transfer to the glovebox.

Following bag-in, the IDCs would be verified and the plastic bags would be unpacked and the residue sorted. Each bag would be opened to remove any tramp metal or other unwanted materials. This material would be bagged out of the glovebox and managed appropriately. Following the sorting, the residue feed material would be shredded and batched to 1 kg (2.2 lb) for thermal desorption/steam passivation.

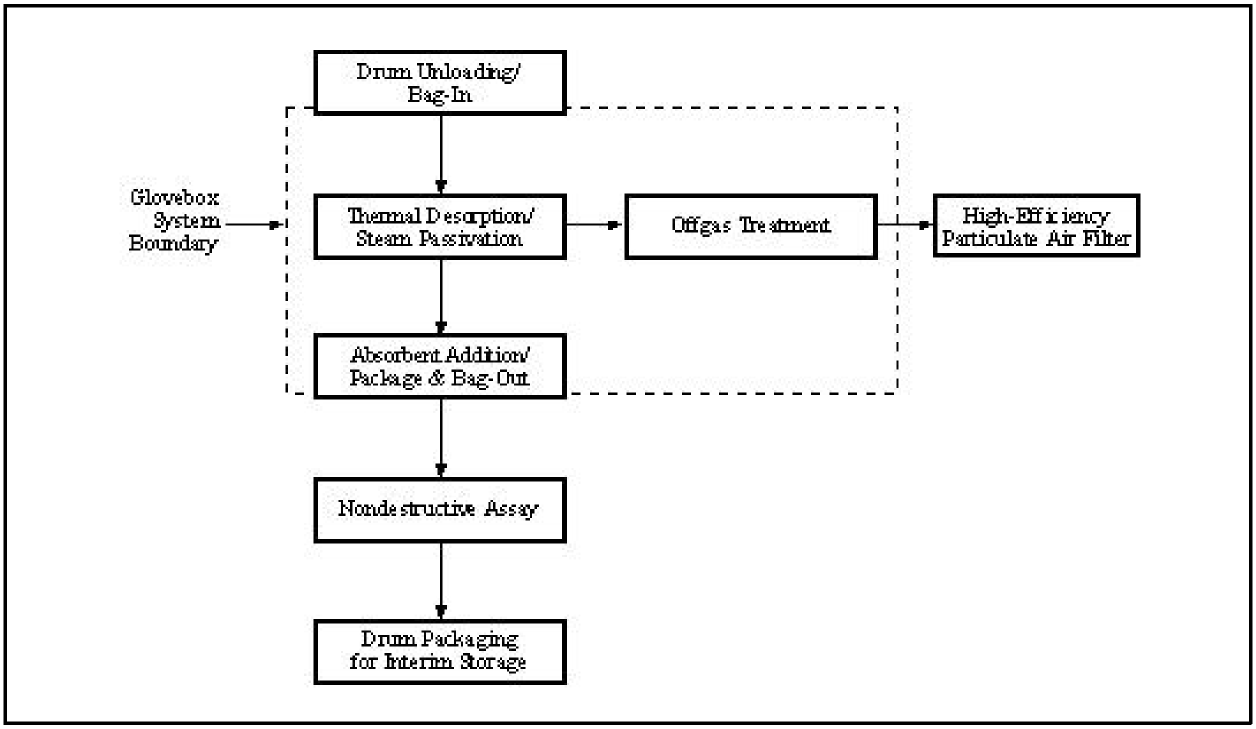

Figure C–4 Thermal Desorption/Steam Passivation Process for Organic-Contaminated Combustible Residues

Thermal desorption is intended to remove the organic solvent contaminants from the combustible residue. The 1-kg (2.2-lb) batches of combustible residue would be heated to 80°C (176°F) for 2 hours under reduced pressure to volatilize the organic solvent contaminants. The offgases would be collected on granulated activated charcoal. Then, low temperature steam would be injected for 1 hour to oxidize any plutonium fines present in the residue.

The processed combustible residue would be allowed to cool to room temperature and approximately 1 kg (2.2 lb) of dry absorbent would be added to dry the wet matrix. The residue would then be batched to approximately 4 kg (8.8 lb) and placed into an 8.2-L (2.2-gal) container. The 4-kg (8.8-lb) batch is based on the volume of shredded combustible waste and absorbent that may reasonably fit into an 8.2-L (2.2-gal) container. The can would be sealed, taped, and bag-out of the glovebox and placed into a 30.5-centimeter (cm) (12-inch [in]) convenience can. Each can would contain approximately 37.8 g (1.3 oz) of plutonium.

Nondestructive assay would be performed and the assayed and repackaged residue containers would be transported for final drum packaging. The containers would be transferred for final drum packaging and then placed in interim storage until a final disposition decision is made.

C.4.5 Repackaging of Dry Combustibles

Repackaging of dry combustibles would be performed to achieve the criteria for safe interim site storage. Dry combustible residue consists of such materials as paper, rags, cloth, plastic, wood, surgical gloves, tape, paper coveralls, booties, personal protective equipment waste, full-face masks, v-belts, polyvinyl chloride, polyethylene, polypropylene, supplied-air suits, and gaskets, some of which are above the safeguards termination limit for combustibles. After repackaging, the combustible residues above the safeguards termination limit would remain above the limit. This would preclude ultimate shipment of this material to WIPP unless it is subjected to further processing. Preparation of direct repackage residues would be conducted within glovebox lines in Modules D, E and F of Building 707.

Repackaging to package and assay appropriate residues is considered to be a proven technology. The capability for Rocky Flats is being installed to support the disposition of below-safeguards termination limit materials, and should be available several months after the issuance of the EIS.

The direct repackage process is shown in **Figure C–5**. The process steps include drum unloading and bag-in, feed preparation and repackaging, and bag-out. Nondestructive assay would be performed and the drums would be packaged for interim site storage.

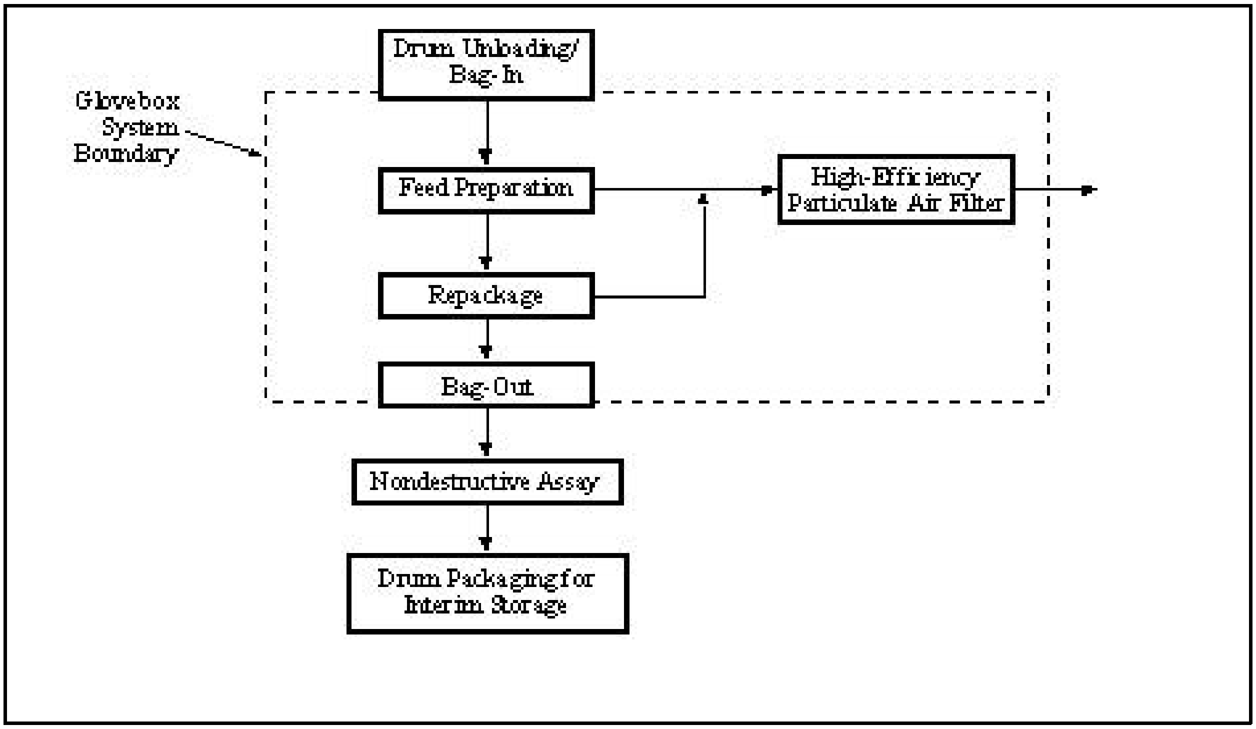

Figure C–5 Direct Repackaging Process for Dry Combustibles

❒ Detailed Process Description

Drums with the capacity of 208-L (55-gal) would be transferred from storage into a contamination control enclosure in Module D of Building 707. The contamination control enclosure is designed to control airflow in the event of a bag failure within a drum. The drums would be opened and the integrity of the packaging would be checked. If the packaging has not been compromised, the package would be transferred into the glovebox in Module E of Building 707. Other packaging materials would be removed from the drum and bagged into the glovebox. If the integrity of the packaging has been compromised, the package would be overpacked with a new plastic bag before transfer to the glovebox.

Following bag-in, the plastic bags would be unpacked and the residue sorted. The residue would then be repackaged into metal containers. If the material requires size-reduction and/or compaction to minimize the volume of the repackaged residue, the sorted residue would be transferred to a size-reduction station. The residue would be shredded and repackaged into metal containers. If required, the repackaged material would be compacted within the metal containers to gain additional volume reduction, which would reduce the number of drums requiring shipment to WIPP. Each repackaged container would be filled to approximately 83.5 g (2.9 oz) of plutonium. After the container filling step, the sealed container would be bagged out and transferred to nondestructive assay.

Nondestructive assay would be performed and the assayed and repackaged residue containers would be transported into Module F of Building 707 for drum packaging. Two containers would be loaded into a pipe component which would be staged inside of a 208-L (55-gal) drum. These drums cannot be shipped to WIPP because the percentage of plutonium in the waste exceeds the safeguards termination limit. The drums would remain in interim site storage until subjected to an appropriate stabilization process that would reduce the plutonium content below the safeguards termination limit.

C.4.6 Acid Dissolution and Plutonium Oxide Recovery of Plutonium Fluorides

Acid dissolution of plutonium fluorides would involve dissolution of the fluorides, followed by precipitation and filtration of plutonium oxalate, and calcination to plutonium oxide for storage. The filtrate from the oxalate precipitation would be treated with magnesium hydroxide to precipitate the plutonium remaining in the solution. That precipitate would then be filtered, calcined, repackaged, and placed in interim site storage until a final disposition decision is made. The dissolution process would be conducted inside gloveboxes located in Room 3701 of Building 371.

The acid dissolution/plutonium oxide recovery process is considered to be a proven technology. The process to be used for the limited quantities of materials identified in these categories would be consistent with equipment and activities that can be performed in the neutralize-dry process area. Thus, the capability for Rocky Flats is currently being installed to support the disposition of below-safeguards termination limit materials, and should be available several months after the issuance of the EIS. However, the use of this equipment for Acid Dissolution would generally be preceded by the neutralize-dry processing of the combustible residues required by the Defense Nuclear Facilities Safety Board Recommendation 94-1 stabilization program, and may not be able to start until 4 years after issuance of the Record of Decision.

The plutonium fluoride acid dissolution process is shown in **Figure C–6**. The feed materials would be unpacked and batched for acid dissolution. The dissolved fluorides would be sent through precipitation to form plutonium oxalate precipitate in slurry form, which would then be filtered to separate the effluent solution from the precipitate. The oxalate would be calcined, nondestructively assayed, calcined again for long-term storage, again nondestructively assayed, and then packaged for storage. Magnesium hydroxide would be mixed into the oxalate precipitation effluent to precipitate the remaining plutonium, and the effluent filtered to form magnesium hydroxide and effluent. The magnesium hydroxide would be calcined and packaged. The packaged magnesium hydroxide product would be removed from the glovebox and nondestructively assayed for accountability purposes, packaged in the final transport/storage container, and placed in interim storage. The last filtration effluent would be sent for evaporation at the Rocky Flats wastewater treatment facility.

Most of the fluoride residues are located in the Building 371 storage vault, and would be transferred from the vault into the glovebox system by a remote handling system through an input/output station. Other fluoride packages would be manually transferred and bagged into the feed preparation glovebox.

❒ Detailed Process Description

The residue feed would be introduced into the glovebox, and the IDC will be verified. The materials would then be removed from the containers and batched to a maximum of 200 g (7 oz) of plutonium in preparation for nitric acid dissolution. Combustible packaging materials from the individual containers would be

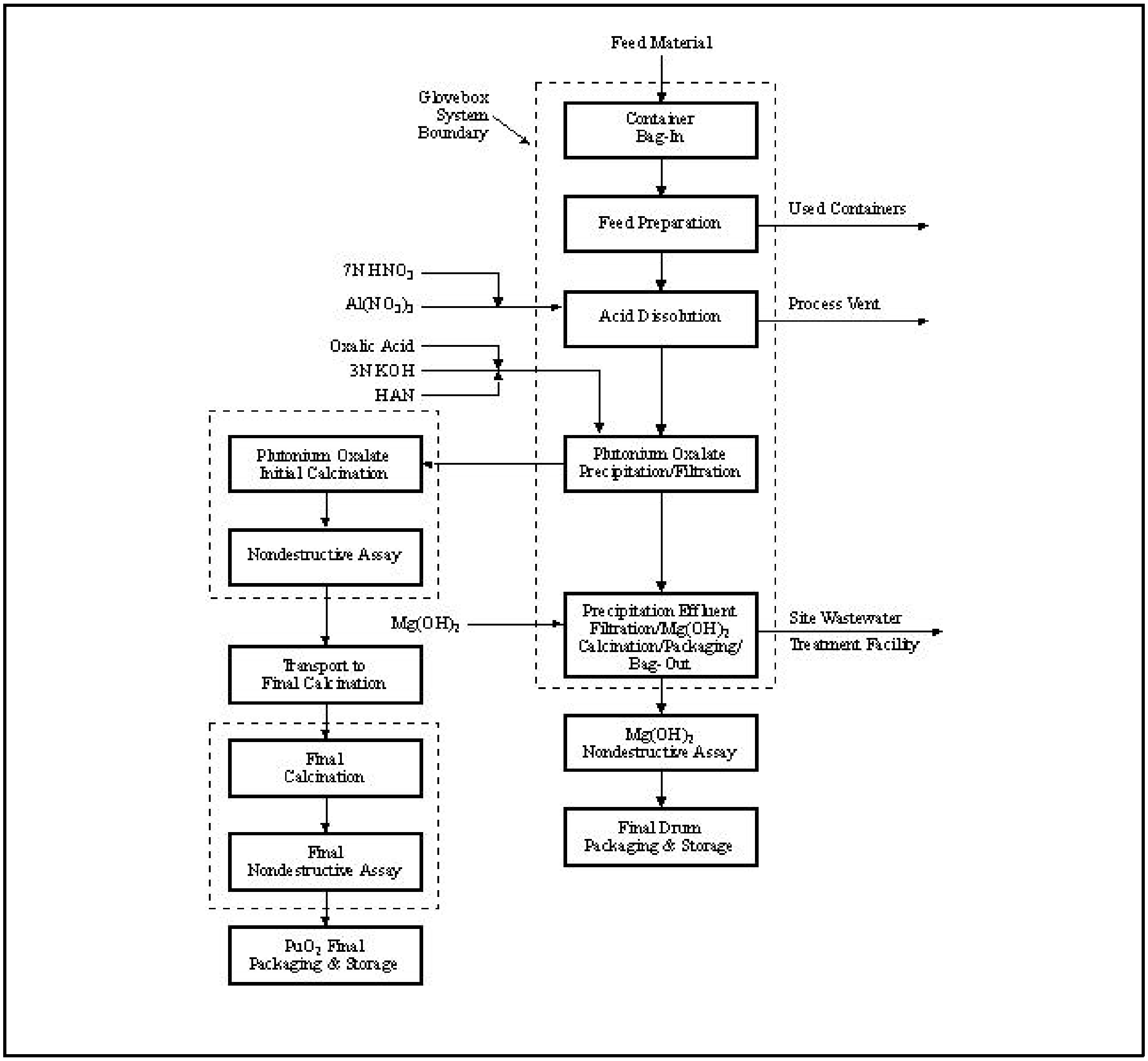

Figure C–6 Acid Dissolution Process for Plutonium Fluorides

bagged out of the glovebox and sent to a combustible handling process. Other unwanted materials would be bagged out of the glovebox and managed appropriately.

The contents of the residue cans would be transferred to one of two heated stirrers. The operator would add 7N nitric acid (HNO_3) and 60 percent aluminum nitrate ($Al(NO_3)_3$) solution to each dissolver before stirring. $Al(NO_3)_3$ would be added to complex residue ions during dissolution. The slurry would be heated to approximately 80°C (176°F) and stirred until dissolution is achieved. Vented fumes would be cooled in a condenser, and then piped to the process vent system. The batch would be filtered to remove any undissolved solids and then split into two equal amounts and transferred to the adjacent heated stirrers for precipitation.

For plutonium oxalate precipitation, 3N potassium hydroxide (KOH) would be added to each can to adjust the normality to 0.75N nitric acid. Hydroxylamine nitrate (HAN) would then be added as a 1.9M solution to adjust the plutonium valence to +3. After these adjustments have been made, solid oxalic acid would be added to form plutonium oxalate precipitate. The solution would be heated to approximately 80°C (176°F) and stirred to form a slurry.

The slurry from the two stirrer assemblies would be poured onto an R-4 filter. Filtration of plutonium oxalate would be achieved by pulling a vacuum through the filter and drawing effluent liquids into a filtrate tank. The plutonium oxalate precipitate would be scooped into a filter boat in preparation for calcining.

The plutonium oxalate would require calcining at 450°C (840°F) to convert the oxalate into the oxide form. In this process, the filter boat would be placed on a pneumatic lift, placed into the calcination furnace, and the precipitate would be heated to 450°C (840°F). Glovebox air would be drawn down through the precipitate at a rate of approximately 0.10 cubic meter (m^3) (3.5 cubic feet [ft^3]) per minute during the heating cycle. After a cooling cycle, the calcined oxide would be transferred from the filter boat back into a can, batched to 1,000 g (2.2 lb), sealed, and sent to calorimetry.

The plutonium oxide can would be assayed for plutonium content based on its rate of thermal generation using calorimeters and gamma-ray isotopic spectrometer equipment. This activity is required to maintain accountability within the acid dissolution material balance area. After assay, the containers would be ready for final calcination. The cans containing the plutonium oxide would be placed into appropriate outer containers and transferred to the Building 371 loading dock. The containers would then be transported to the Building 707 loading dock by intra-site truck transportation, and moved to appropriate vault storage pending final calcination.

The plutonium oxide cans would be transferred from the Building 707 storage vault to Module J and bagged into the plutonium stabilization and packaging system. The plutonium oxide would be removed from the cans, placed into furnaces, and calcined at 1,000°C (1,830°F) for 8 hours. The material, now suitable for long-term storage or transportation, would be weighed, characterized, and placed into a 3013 inner container. This container would then be removed from the glovebox by the bagless transfer process and sent to calorimetry. The plutonium oxide package would be assayed for plutonium content based on its rate of thermal generation using calorimeters and gamma-ray isotopic spectrometer equipment. After assay, the containers would be placed into vault storage, pending a final disposition decision.

Magnesium hydroxide, $Mg(OH)_2$, 30 percent by weight, would be added to the effluent liquid in the filtrate tank from the precipitation filtration step, and the tank would be mixed by sparging. The liquid and precipitate would then be drained onto an R-4 filter. Filtration would be achieved by pulling a vacuum through the R-4 filter and drawing effluent liquids into the transfer tank. The magnesium hydroxide precipitate would then be scooped into a filter boat in preparation for calcining. The magnesium hydroxide would be calcined at 450°C (840°F). In this process, the filter boat would be placed on a pneumatic lift, placed into the calcination furnace, and the precipitate would be heated to 450°C (840°F). Glovebox air would be drawn down through the precipitate at a rate of approximately 0.10 m^3 (3.5 ft^3) per minute during the heating cycle. After a cooling cycle, the calcined hydroxide would be transferred from the filter boat back into a can, batched to 9.1 kg (20 lb), sealed, and bagged out into convenience cans.

Nondestructive assay of the magnesium hydroxide would be performed to ensure requirements limit are met and to obtain data to ensure that required accountability procedures are followed. Nondestructive assay methods would be selected to ensure that the best accountability data are obtained. Assayed product packages would be selected for final packaging to minimize the number of shipping containers and placed in interim storage pending a final disposition decision. Selected packages would be loaded into an inner container and sealed before placing the container into the final outer shipping container.

C.4.7 Neutralization and Drying of Filter Media

The neutralization and drying process for filter media residues would treat the nitric acid contaminant on the residue to eliminate the potential flammable hazard. The neutralization and drying process is not intended to remove the plutonium from the residue. As a result, this would preclude ultimate shipment to WIPP unless the residue is subjected to further stabilization processing. This process would be conducted in Room 3710 of Building 371.

The neutralize-dry process, consisting of washing materials in alkaline solutions, allowing them to drain or partially dry, and mixing the resulting solids with water-absorbing materials, is considered to be a proven technology. The capability for Rocky Flats is being installed to support the disposition of below-safeguards termination limit materials, and should be available several months after the issuance of the EIS. Activities are underway to optimize the process and reduce the quantity of water-absorbing materials required for meeting disposal requirements.

The neutralization and drying process for filter media residues is shown in **Figure C–7**. The process steps are drum unloading and bag-in, feed preparation, the neutralization and decant/filtration, oven drying, packaging, and bag-out. Nondestructive assay would be performed, and the drums would be packaged for interim site storage.

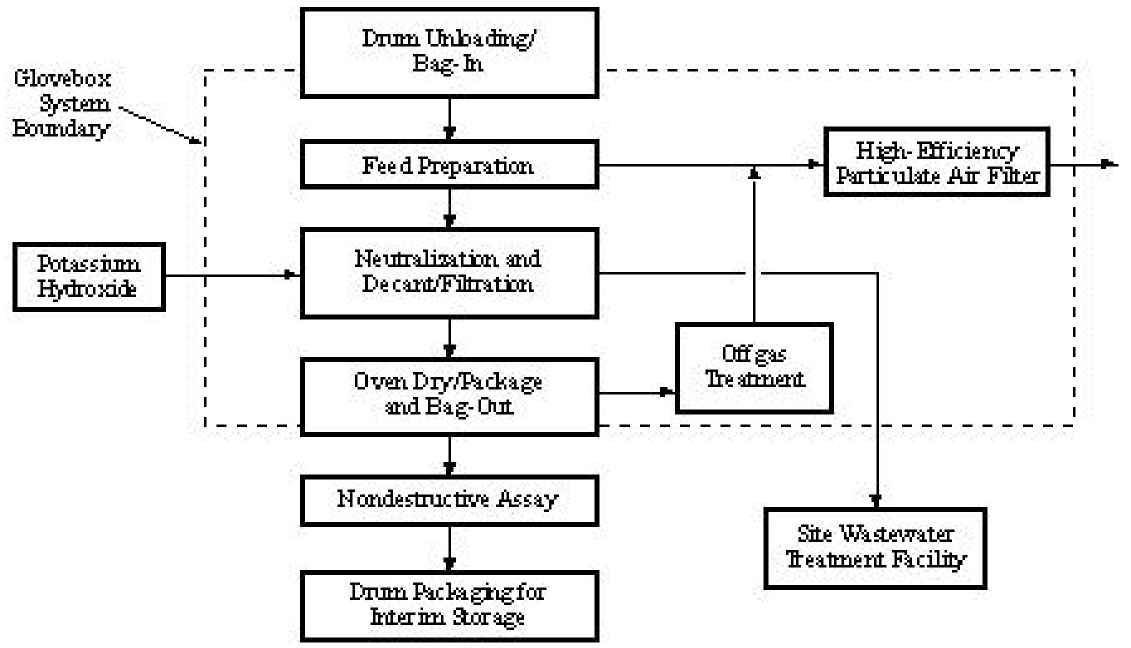

Figure C–7 Neutralization and Drying Process for Filter Media

❒ Detailed Process Description

Drums with the capacity of 208-L (55-gal) would be transferred from storage into a contamination control enclosure and unpacked. The contamination control enclosure is designed to control airflow in the event of a bag failure within a drum. The drums would be opened and the integrity of the packaging would be checked. If the packaging has not been compromised, the package would be transferred into the glovebox. Any unnecessary packaging materials, would be removed to limit the amount of packaging introduced into the glovebox. If the integrity of the primary packaging has been compromised, the package would be overpacked with a new plastic bag before transfer to the glovebox.

Following bag-in, the IDCs would be verified and the plastic bags would be unpacked and the residue sorted. Each bag would be opened to remove any tramp metal or other unwanted materials. High-efficiency particulate air filter frames and stiffeners would be separated from the filter media. These materials would be bagged out of the glovebox and managed appropriately. Following the sorting, the residue feed material would be shredded and batched to 5-kg (11-lb) batches for neutralization.

Neutralization is intended to remove the nitrate contamination from the combustible waste and to neutralize any residual nitric acid contained within the residue. The 5-kg (11-lb) batches of combustible residues would be washed with 50 L (13.2 -gal) of the water containing 10 percent excess potassium hydroxide. After 2 hours, the acid would be neutralized forming potassium nitrate and water. None of the plutonium would be removed from the residue during the neutralization process. The combustible solids would be separated from the nitrate and plutonium containing solution by decanting and filtration. The combustible solids would contain approximately 20 percent solution by weight with a proportionate quantity of nitrates and transferred to a drying pan. At intervals, as required during the process, the neutralization solution would be sent to Building 374 for evaporation using the site wastewater treatment process.

The filter media residue that had been neutralized and transferred to the drying pan would then be placed into a drying oven. The residue would be dried under a vacuum at 80°C (176°F) for 2 hours. Offgas from drying would be treated before high-efficiency particulate air filtration. After cooling, the residues would be weighed, and the quantity of plutonium estimated as the waste would be transferred to 8.2-L (2.2-gal) containers. Each container would hold approximately 83.5 g (2.9 oz) of plutonium. The containers would be bagged out of the glovebox and packaged into convenience cans for transfer to nondestructive assay.

Nondestructive assay would be performed and the assayed and packaged residue containers would be transported for drum packaging. These drums cannot be shipped to WIPP because the percentage of plutonium in the waste exceeds the safeguards termination limit. The drums would reside in interim site storage until subjected to an appropriate stabilization process that would reduce the plutonium content below the safeguards termination limit.

C.4.8 Filtration and Drying of Sludge Residues

The filtration and drying process for sludge residues filters off any excess liquid and drys the remaining material by mixing it with an absorbent. After drying and repackaging, the sludge residues would be placed in interim storage pending a final disposition decision. This process would be conducted in Room 3701 of Building 371.

The filter-dry process, consisting of allowing wet materials to drain and partially dry, and mixing the resulting solids with water-absorbing materials, is considered to be a proven technology. The capability for Rocky Flats is being installed to support the disposition of below-safeguards termination limit materials, and should be available several months after the issuance of the EIS. Activities are underway to optimize the process and reduce the quantity of water-absorbing materials required for meeting disposal requirements.

The filtration and drying process for sludge residues is shown in **Figure C–8**. The process steps are drum unloading and bag-in, feed preparation and decant/filtration, absorbent addition, and bag-out. Nondestructive assay would be performed, followed by drum packaging for interim site storage.

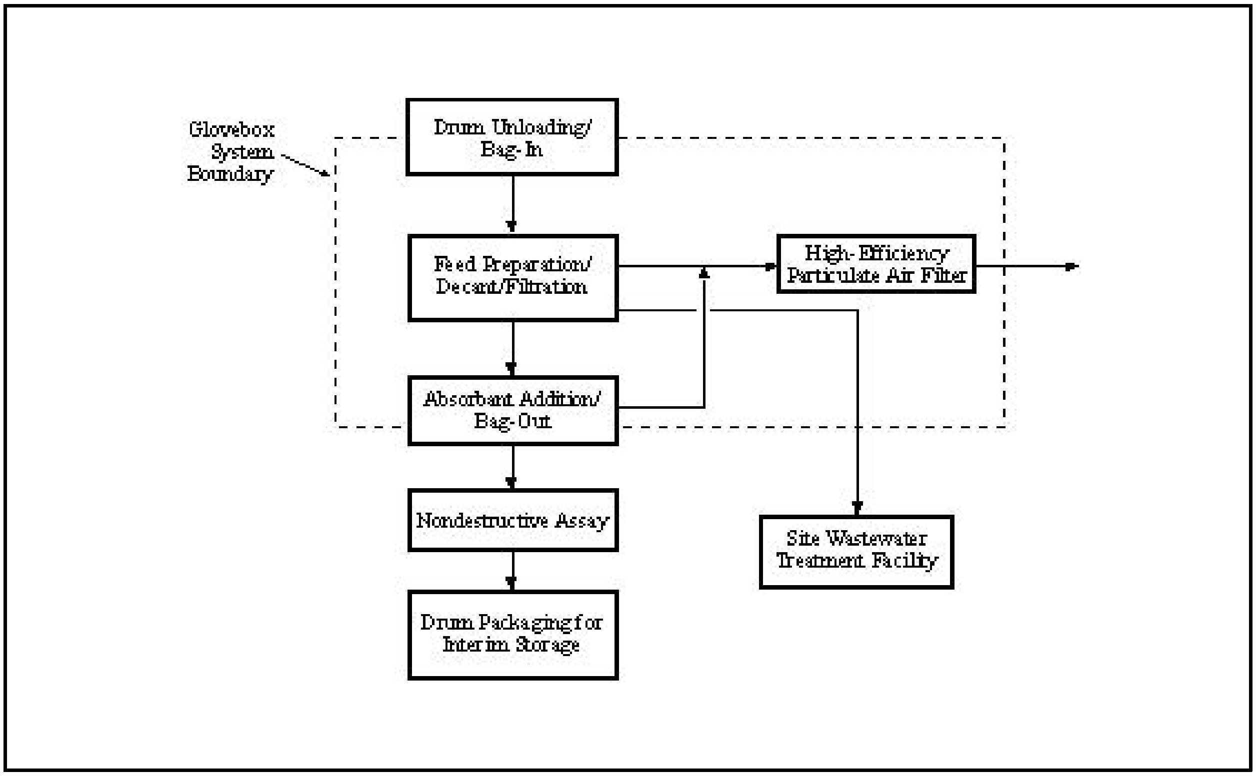

Figure C–8 Filtration and Drying Process for Sludge Residues

❒ Detailed Process Description

Drums with the capacity of 208-L (55-gal) would be transferred from storage into a contamination control enclosure and unpacked. The contamination control enclosure is designed to control airflow in the event of a bag failure within a drum. The drums would be opened and the integrity of the packaging checked. If the packaging has not been compromised, the package would be transferred into the glovebox. Any unnecessary packaging materials, would be removed to limit the amount of packaging introduced into the glovebox. If the integrity of the primary packaging has been compromised, the package would be overpacked with a new plastic bag before transfer to the glovebox.

Following bag-in, IDCs would be verified, containers would be unpacked, and the residues would be sorted. Any unwanted materials found in the sludge, such as plastics or metals, would be bagged out of the glovebox and managed appropriately. As required, free liquids would be decanted and vacuum filtered to collect any suspended solids. At intervals, as required during the process, the decanted and filtered liquids would be sent to the site wastewater treatment process in Building 374 for evaporation. After decanting, the sludge would be removed from the container and the resulting packaging materials would be bagged out of the glovebox and managed appropriately. The sorted residue material would be weighed into 8.2-L (2.2 -gal) containers. The amount of residue added to the container would be based on the total weight of the container, after absorbent addition, being 9.09 kg (20.0 lb) or less to meet physical handling constraints. Dry absorbent would be blended with the wet sludge residue for absorbent addition at a ratio of 4 parts absorbent to 1 part sludge, by weight. After blending, the containers would be sealed with a lid and bagged out of the glovebox. The containers would be placed in convenience cans, sealed and taped, and sent to nondestructive assay.

The assayed and packaged residue containers would be transported for drum packaging, which would include first placing the containers into pipe components. The drums would be transferred to interim storage until a final disposition decision is made.

C.4.9 Neutralization and Drying of Glass Residues

The neutralization and drying process for glass residues would treat the nitric acid contaminant on the residue. This process may remove up to 99 percent of the plutonium from the residue based on results from washing Raschig rings. This process would be conducted in Room 3701 of Building 371.

The neutralize-dry process, consisting of washing materials in alkaline solutions, allowing them to drain or partially dry, and mixing the resulting solids with water-absorbing materials, is considered to be a proven technology. The capability for Rocky Flats is being installed to support the disposition of below-safeguards termination limit materials, and should be available several months after the issuance of the EIS. Activities are underway to optimize the process and reduce the quantity of water-absorbing materials required for meeting disposal requirements.

The neutralization and drying process for glass residues is shown in **Figure C–9**. The process steps are drum unloading and bag-in, feed preparation, neutralization and decant/filtration, and oven drying, packaging, and bag-out. Nondestructive assay would be performed, followed by drum packaging for interim site storage.

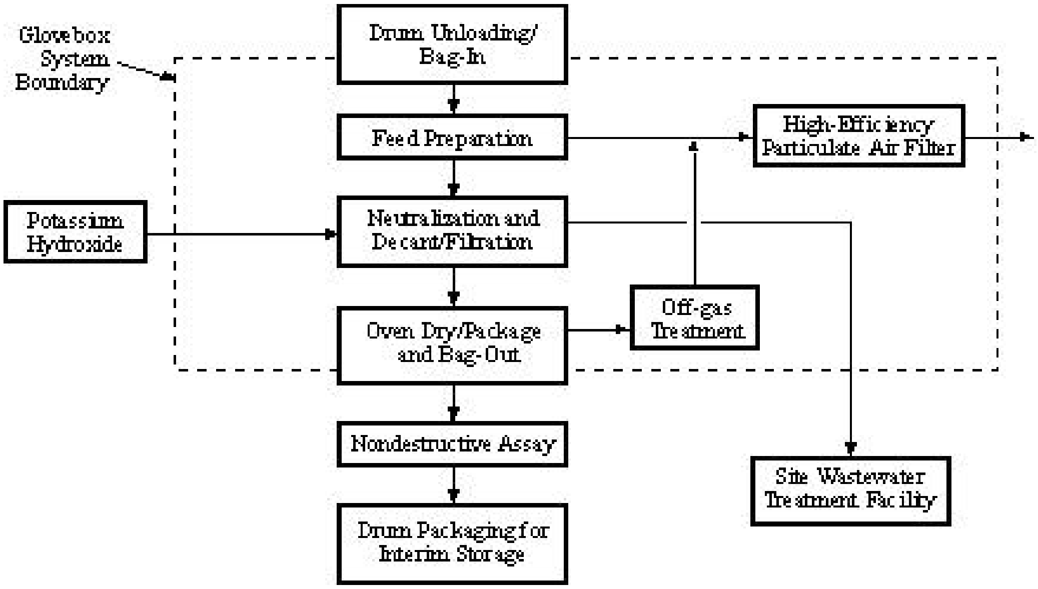

Figure C–9 Neutralization and Drying Process for Glass Residues

❐ Detailed Process Description

Drums with the capacity of 208-L (55-gal) would be transferred from storage into a contamination control enclosure and unpacked. The contamination control enclosure is designed to control airflow in the event of a bag failure within a drum. The drums would be opened and the integrity of the packaging checked. If the packaging has not been compromised, the package would be transferred into the glovebox. Any unnecessary packaging materials would be removed to limit the amount of packaging introduced into the glovebox. If the integrity of the primary packaging has been compromised, the package would be overpacked with a new plastic bag before transfer to the glovebox.

Following bag-in, the IDCs would be verified, and the containers would be unpacked and the residue sorted. Any unwanted materials would be bagged out of the glovebox and managed appropriately. As required, the sorted residue material would be size-reduced by crushing and batched to 5-kg (11-lb) batches for neutralization.

Neutralization is intended to remove the nitrate contamination from the glass waste and to neutralize any residual nitric acid on the residue. The 5-kg (11-lb) batches of residue would be washed with 50 L (13 -gal) of water containing 10 percent excess potassium hydroxide. After 2 hours, the acid would be neutralized forming potassium nitrate and water. The neutralization process should remove approximately 99 percent of the plutonium from the residue. The residue solids would be separated from the nitrate and plutonium-containing solution by decanting and filtration. The residue, after neutralization, would contain approximately 0.5 percent neutralization solution by weight with a proportionate quantity of nitrates and transferred to a drying pan. At intervals, as required during the process, the neutralization solution with the removed plutonium would be sent to the site wastewater treatment process in Building 374 for evaporation.

The glass residue, neutralized and transferred to the drying pan, would then be placed into a drying oven. The residue would be dried under a vacuum at 80°C (176°F) for 2 hours. Off-gas from drying would be treated before high-efficiency particulate air filtration. After cooling, the residue would be weighed and the quantity of plutonium estimated as the waste is transferred to plastic bags. These bags would be bagged out of the glovebox and packaged in 8.2-L (2.2-gal) containers to approximately 42.9 g (1.5 lb) of plutonium per container based on a maximum container weight of 9.09 kg (20.0 lb) because of physical handling constraints. After being removed from the glovebox, the containers would be packaged into convenience cans.

Nondestructive assay would be performed, and the assayed and packaged residue containers would be transported for drum packaging, which includes first placing the containers into pipe components. The drums would be transferred to interim storage until a final disposition decision is made.

C.4.10 Repackaging of Graphite Residues, Inorganic Residues, and Scrub Alloy

Repackaging of graphite and inorganic residues and scrub alloy would be performed to achieve the criteria for safe interim site storage. For the graphite and inorganic residues, after repackaging, the residues would remain above the safeguards termination limits, which would preclude ultimate shipment to WIPP unless the material is subjected to further stabilization. Preparation of direct repackage residues for all three materials would be conducted within glovebox lines in Modules D, E and F of Building 707.

Repackaging to package and assay appropriate residues is considered to be a proven technology. The capability for Rocky Flats is being installed to support the disposition of below-safeguards termination limit materials, and should be available several months after the issuance of the EIS.

The direct repackage process is shown in **Figure C–10**. The process steps are drum unloading and bag-in, feed preparation (for graphite and inorganic), container examination and verification (for scrub alloy), repackaging, and bag-out. Nondestructive assay would be performed, followed by drum packaging for interim site storage.

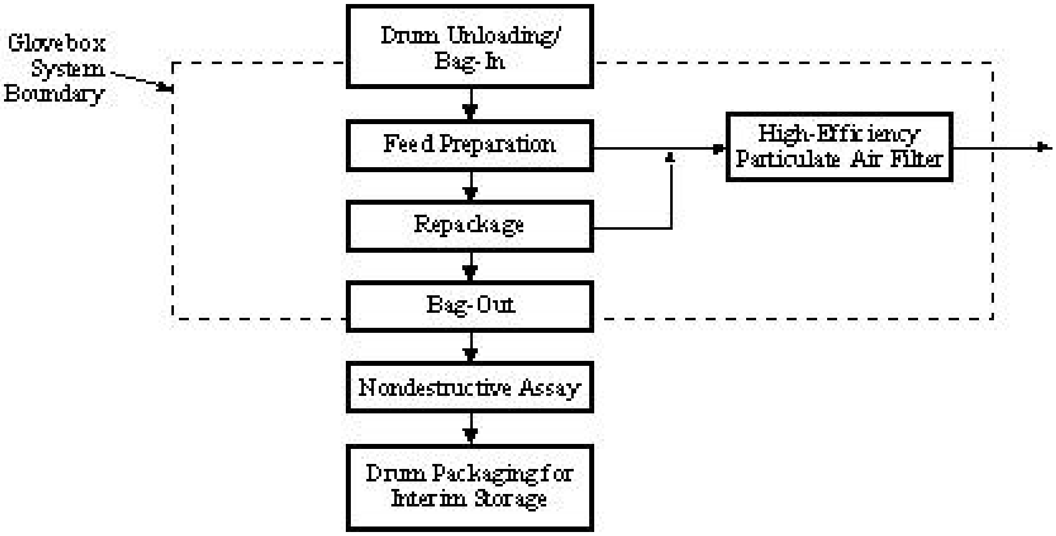

Figure C–10 Direct Repackaging Process for Graphite and Inorganic Residues

❒ Detailed Process Description

Drums with the capacity of 208-L (55-gal) would be transferred from storage into a contamination control enclosure in Module D of Building 707. The contamination control enclosure is designed to control airflow in the event of a bag failure within a drum. The drums would be opened and the integrity of the packaging would be checked. If the packaging has not been compromised, the containers would be transferred into the glovebox in Module E of Building 707. The containers, including outer packaging materials, would be removed from the drum and bagged into the glovebox. If the integrity of the packaging has been compromised, the package would be overpacked with a new plastic bag before transfer to the glovebox.

For the graphite and inorganic residues, following bag-in, the plastic bags would be unpacked and the residue sorted. The residue would then be repackaged into metal containers. If the material requires size-reduction and/or compaction to minimize the volume of the repackaged residue, the sorted residue would be transferred to a size-reduction station, after which the residue would be repackaged into metal containers. If required, the repackaged material would be compacted within the metal containers to gain additional volume reduction. Each repackaged container would be filled to approximately 83.5 g (2.9 oz) of plutonium. After the container filling step, the sealed container would be bagged out and transferred to nondestructive assay. The assayed and repackaged residue containers would then be transported into Module F of Building 707 for drum packaging. The containers would be loaded into 208-L (55-gal) drums. These drums cannot be shipped to WIPP because the percentage of plutonium in the waste exceeds the safeguards termination limit. The drums would reside in interim site storage until subjected to an appropriate stabilization process that would reduce the plutonium content below the safeguards termination limit for graphite and inorganic residues.

For the scrub alloy, following bag-in, the containers housing the scrub alloy buttons would be unpacked. The scrub alloy would then be repackaged into metal containers meeting the safe storage standard. After the scrub alloy button is repackaged, the sealed container would be bagged out and transferred to nondestructive assay. The assayed and repackaged residue containers would then be transported to Building 371 for safe interim site storage. These containers cannot be placed into pipe components, drummed, and shipped to WIPP because the percentage of plutonium exceeds storage limits. The containers would reside in interim site storage until subjected to an appropriate stabilization process.

C.5 DETAILED PROCESS DESCRIPTIONS FOR PROCESSING TECHNOLOGIES WITHOUT PLUTONIUM SEPARATION

C.5.1 Immobilization (Vitrification)

For ash, high-efficiency particulate air filter media, sludge, glass, graphite, and inorganic residues, the proposed vitrification immobilization process would use a furnace vitrification technology similar in concept to calcination. This process has been proposed for interim processing to allow safe interim storage at Rocky Flats until shipment to WIPP is approved. The process would be conducted in gloveboxes located in Module D, E, and F of Building 707 using muffle furnaces to heat the residue material to approximately 700 to 1,300°C (1,300 to 2,400°F) for 4 hours. The end product would consist of a solidified monolith contained inside a 20-cm (8-in) diameter by 25.4-cm (10-in) high metal can.

Calcination of powdered or granular materials in muffle furnaces is considered to be a proven technology. Capabilities necessary to satisfy all alternatives are being installed at Rocky Flats as part of the ongoing stabilization programs, and should be operational within several months after issuance of the EIS. The vitrification process is also considered to be a proven technology for most residue types to which it may be applied. A technical development program is underway for the vitrification of ash residues. The muffle furnace capability for Rocky Flats is being installed to support the disposition of below-safeguards termination limit materials, and should be available several months after the issuance of the EIS. Activities are underway to optimize the process and reduce the steps necessary to achieve an acceptable waste form.

The vitrification process for residues is shown in **Figure C–11**. The process steps are drum unloading, feed preparation, vitrification, and bag-out. Nondestructive assay would be performed, followed by final drum packaging and storage.

The proposed vitrification for scrub alloy requires a two-step heating process. First, the scrub alloy would be converted to an oxide by calcining at 600°C (1,100°F) and 1,000°C (1,800°F), respectively. Then, calcined scrub alloy would be blended with frit and vitrified using the furnace vitrification process. This entire process would be conducted in gloveboxes located in Modules D, E, and F of Building 707, similar to the residues. The calcining and vitrification steps would use identical muffle furnaces.

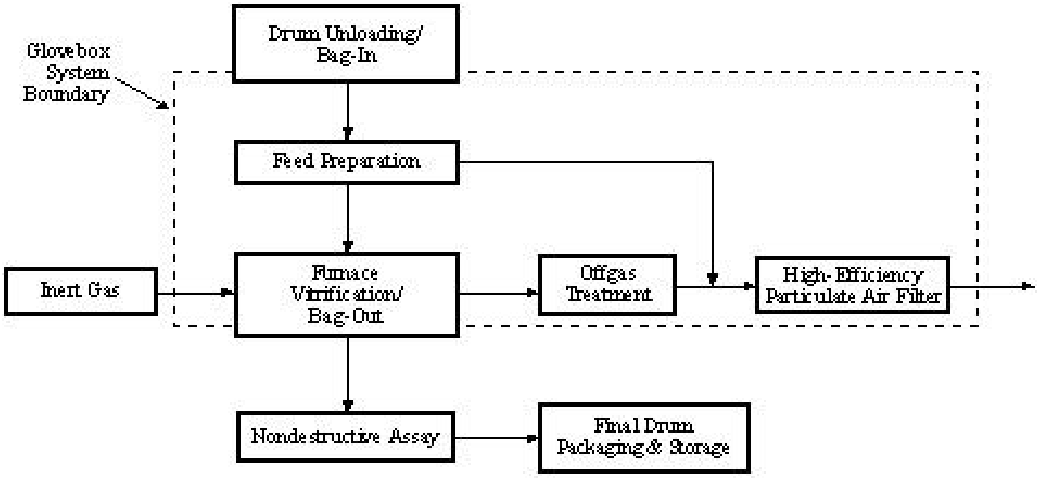

Figure C–11 Furnace Vitrification Process for Residues

Since the calcination of powdered or granular materials in muffle furnaces is considered to be a proven technology and plutonium metals and other alloys have been routinely burned in the past, calcination of scrub alloy is considered to be a low-risk technology although not specifically proven in this context. Capabilities necessary to satisfy all alternatives are being installed at Rocky Flats as part of the ongoing stabilization programs, and should be operational within several months of issuance of the EIS. The vitrification process is considered a proven technology for most residue types for which it may be applied. The muffle furnace capability for Rocky Flats is being installed to support the disposition of below-safeguards termination limit materials, and should be available several months after the issuance of the EIS. The disposition of scrub alloy through a calcination and vitrification process was not envisioned as a disposal approach at the time of the WIPP EIS and therefore was not included in the WIPP Baseline Inventory Report. In the event that this technology would be implemented, the resulting transuranic waste, although of satisfactory composition and form, might be subject to delays in disposal due to the necessity of revising regulatory documentation. Since this material has historically been considered "War Reserve" material, its final disposition to WIPP has not been programmatically evaluated. As such, DOE does not consider the calcination and vitrification of scrub alloy at Rocky Flats to be a preferred processing technology.

The process to vitrify scrub alloy residues is shown in **Figure C–12**. The process steps would be container bag-in, feed preparation for calcination, calcination, feed preparation for vitrification, and vitrification and bag-out. Nondestructive assay would also be performed, followed by final drum packaging and storage.

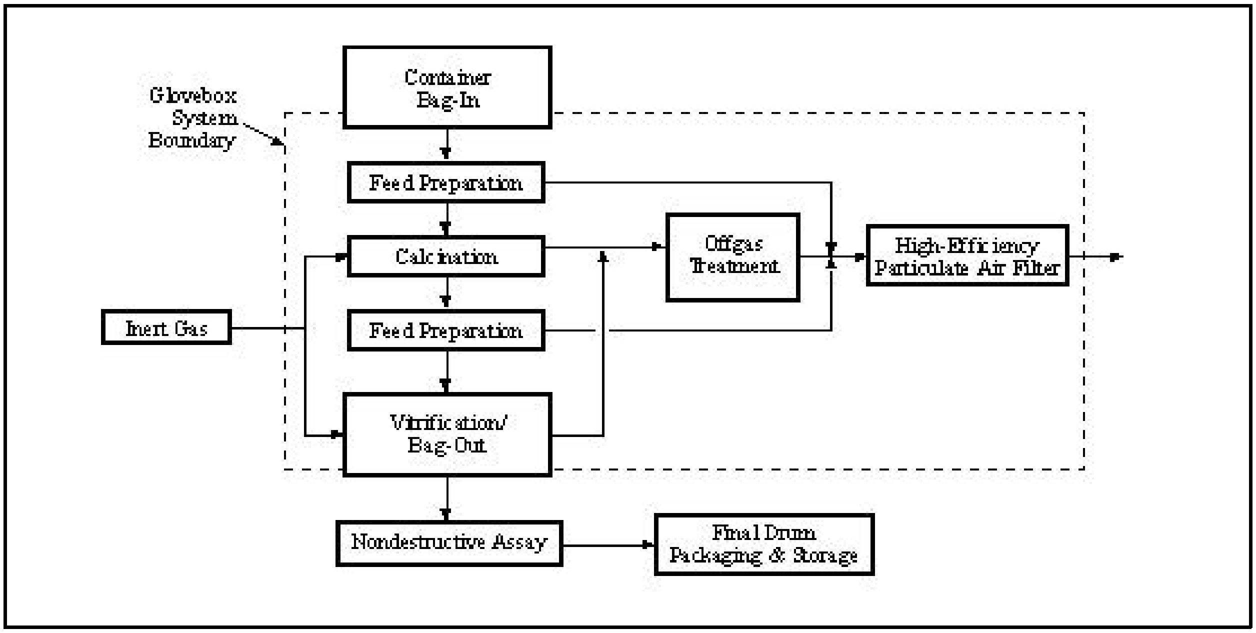

Figure C–12 Furnace Vitrification Process for Scrub Alloy

Furnace vitrification involves the addition of siliceous material called "frit" to the residues or scrub alloy followed by heating at 700 to 1,300°C (1,300 to 2,400°F) to produce a glass matrix. Two waste streams may be generated in addition to the vitrified product stream. The first waste stream would be a solid transuranic waste stream consisting of size-reduced stainless steel cans, plastic containers, plastic bags and containers. The second waste stream would be a gaseous effluent stream consisting primarily of one or more of the following, depending on the residue or scrub alloy type: nitrogen, oxygen, trace acid gases, carbon dioxide, nitrous oxide, water, and/or particulates. The off-gas stream would be configured to cool the effluents and remove acids and particulates before discharge into the glovebox exhaust high-efficiency particulate air filter system.

❒ Detailed Process Description for Residues

Drums with the capacity of 208-L (55-gal) would be transferred from storage into a contamination control enclosure in Module D of Building 707. The contamination control enclosure would be designed to control airflow in the event of a bag failure within a drum. The drums would be opened and the integrity of the packaging would be checked. If the packaging has not been compromised, the containers would be transferred into the glovebox in Module E of Building 707. The containers, including outer bags, clamshells, and other packaging materials, would be removed from the drum and bagged into the glovebox. If the integrity of the packaging has been compromised, the package would be overpacked with a new plastic bag before transfer to the glovebox.

For the residues, after bag-in, the IDCs of residue containers/packages would be verified and the original residue packages would be transferred to a residue sorting and loading station or to a crushing station, either of which would provide local dust control. For most residue types, the sorting and loading station would contain a sieve to sort and to separate out oversized residue and tramp material (e.g., nuts, bolts). Tramp material and large pieces of residue would be collected on the sieve. The sieved residue fines would be collected in the new containers. The tramp material will be separated and transferred for transuranic waste size-reduction and packaging. Oversized residue pieces, would be sent through a size-reduction process and either mechanically crushed or shredded. Once size-reduced, the residues would be fed back to the loading station. High-efficiency particulate air filter frames would be separated out and shredded for size-reduction and sorted glass residues would be crushed, then both will be loaded into new metal containers for the vitrification process. Sorted sludge residue materials would be loaded directly into a new metal container for the vitrification process.

The residues would be batched with an average of 83.5 g (2.9 oz) of plutonium per container. For all of the residues, following batching, a blending step would be required wherein the materials would be blended and diluted with low melting temperature frit. Each material stream to be immobilized by the vitrification process would be analyzed to determine the appropriate proportions of material and frit to meet the WIPP/Waste Acceptance Criteria requirements. The material containers would then be ready for vitrification.

After a container is charged and blended, it would be positioned into the heating chamber of the muffle furnace. The furnace would be energized and there would be a gradual ramp-up in temperature within the chamber. The temperature range for the vitrification process would be between 700 and 1,300°C (1,300 and 2,400°F). The actual vitrification temperature would be determined for each specific type of residue before vitrification.

Engineering investigations are underway to identify the most effective method to extract and capture the off-gases generated by the heating process. Various constituents may be generated during vitrification depending on the residue type, including one or more of the following: water vapor, carbon dioxide, nitrous oxide, trace quantities of acid gases, organic, and/or particulates. This description assumes the use of a dry scrubber using potassium carbonate for off-gas treatment. A concern with residual organic contaminants in the residue feed stream subsequently volatilizing during the heating process necessitated an investigation into the incorporation of a design modification which would continuously purge the heating chamber with inert gas during the processing of the residues.

The heating process would be approximately 4 hours in duration. The container would be allowed to cool to 100°C (212°F) before removal from the furnace. After final cooling, the container would be sealed with a lid and bagged out of the glovebox for nondestructive assay.

Post-vitrification nondestructive assay would be performed on all material containers to determine the amount of fissile material present. Following nondestructive assay, residue containers that meet the WIPP/Waste Acceptance Criteria fissile material limits would be transferred into Module F of Building 707 for final drum packaging. Two containers would be loaded into a pipe component staged inside a 208-L (55-gal) drum. The sealed drums would be placed into interim site storage awaiting shipment to WIPP.

❒ Detailed Process Description for Scrub Alloy

Stainless steel containers of scrub alloy would be transferred from storage and bagged directly into the glovebox. After bag-in, the alloy buttons would be unpacked and placed in a burn boat in a muffle furnace and calcined at approximately 600°C (1,100°F) for 2 hours to convert the scrub alloy to an oxide. After cooling, the powdery oxide would be transferred to another muffle furnace and calcined at 1,000°C (1,800°F) for 2 hours. After being allowed to cool, the boats would be transferred to the loading station. At this point in the process, oversized scrub alloy pieces would be sent back through the calcination process and transferred again to the loading station.

The scrub alloy would be batched to average 18.1 g (0.64 oz) of plutonium per container. Following batching, a blending step would be required wherein the materials would be blended and diluted with low-melting-temperature frit. The material stream to be immobilized by the vitrification process would be analyzed to determine the appropriate proportions of material and frit to meet the WIPP/Waste Acceptance Criteria requirements. The material containers would then be ready for vitrification.

After a container is charged and blended, it would be positioned into the heating chamber of the muffle furnace. The furnace would be energized and there would be a gradual ramp-up in temperature within the chamber. The temperature range for the vitrification process would be between 700 and 1,300°C (1,300 and 2,400°F). The actual vitrification temperature would be determined for each specific type of material before vitrification.

Engineering investigations are underway to identify the most effective method to extract and capture the off-gases generated by the heating process. This description assumes the use of a dry scrubber using potassium carbonate for off-gas treatment. A concern with residual organic contaminants in the feed stream subsequently volatilizing during the heating process necessitated an investigation into the incorporation of a design modification that would continuously purge the heating chamber with inert gas during the processing of the scrub alloy.

The heating process would be approximately 4 hours in duration. The container would be allowed to cool to 100°C (212°F) before removal from the furnace. After final cooling, the container would be sealed with a lid and bagged out of the glovebox for nondestructive assay.

Post vitrification nondestructive assay would be performed on all material containers to determine the amount of fissile material present. Following nondestructive assay, containers that meet the WIPP/Waste Acceptance Criteria fissile material limits would be transferred into Module F of Building 707 for final drum packaging. Two containers would be loaded into a pipe component staged inside a 208-L (55-gal) drum. The sealed drums would be placed into interim site storage awaiting shipment to WIPP.

C.5.2 Immobilization (Cementation) of Graphite Residues

The proposed cement-based immobilization process is an adaptation of a Portland cement-based waste immobilization process that has been used within DOE and the commercial nuclear industry. This process was approved by EPA as a best demonstrated available technology for use in waste stabilization. At Rocky Flats, cement-based waste immobilization processes have been operated successfully for several years and have produced thousands of cubic yards of solidified waste. The process has been used for the solidification of low-level waste (saltcrete) in Building 374 and for the solidification of transuranic waste in Building 774 (bottlebox process). The graphite residue cement solidification process would be located in either Building 707 or Building 371 and would involve the cementation of graphite molds, scarfed graphite molds, coarse graphite, and coarse firebrick as feed materials.

Cementation of materials necessary to immobilize fines and to form an acceptable solid is considered to be a proven technology, although optimization studies are routinely performed to improve specific characteristics. Rocky Flats would have to install or remodel gloveboxes to provide additional area for the curing step, so approximately one year would be required after the issuance of the Record of Decision before the cementation capability would be fully operational. The specific location of the cementation processing is uncertain, although the process would be consistent with either Building 371 or Building 707.

The cement-based immobilization process is shown in **Figure C–13**. The process steps are drum unloading and bag-in, feed preparation for calcination, calcination, feed preparation for cementation, in-line nondestructive assay, process mixing, curing and bag-out, and final drum packaging and storage.

Cement-based immobilization blends cement and water with the prepared graphite residues. The advantage of cement-based immobilization technology is its proven performance. When well-established protocols are followed, a WIPP acceptable final product would be ensured. Elements included within these protocols include waste characterization, both physical and chemical; treatability formula development; bench scale testing; pilot scale studies; and detailed project planning for full-scale operations.

There are several disadvantages associated with a cement-based immobilization process. First, unrecognized variability in the waste feed stream can compromise the acceptability of the final product in meeting the WIPP/Waste Acceptance Criteria. Second, the mixing of the cement and water components produces heat during the curing process, and the active metals in the waste stream react with water to produce hydrogen gas. Third, during mixing, curing, and after final packaging, there is a potential for hydrogen generation due to both radiolysis and hydrolysis of the water of hydration by the radiological and reactive metal components, respectively. This allows for less transuranic material to be transported per shipment to WIPP.

In addition to the cemented-residue product stream, there would be two waste streams generated. The first waste stream would be a solid transuranic waste stream consisting of size-reduced steel containers, plastic containers and plastic bags. The second waste stream would be a gaseous effluent stream consisting primarily of nitrogen, water vapor, carbon dioxide, hydrogen, and particulates. Any tramp material removed from the waste would be either combined with the solid transuranic waste stream or placed into a cemented waste container before curing.

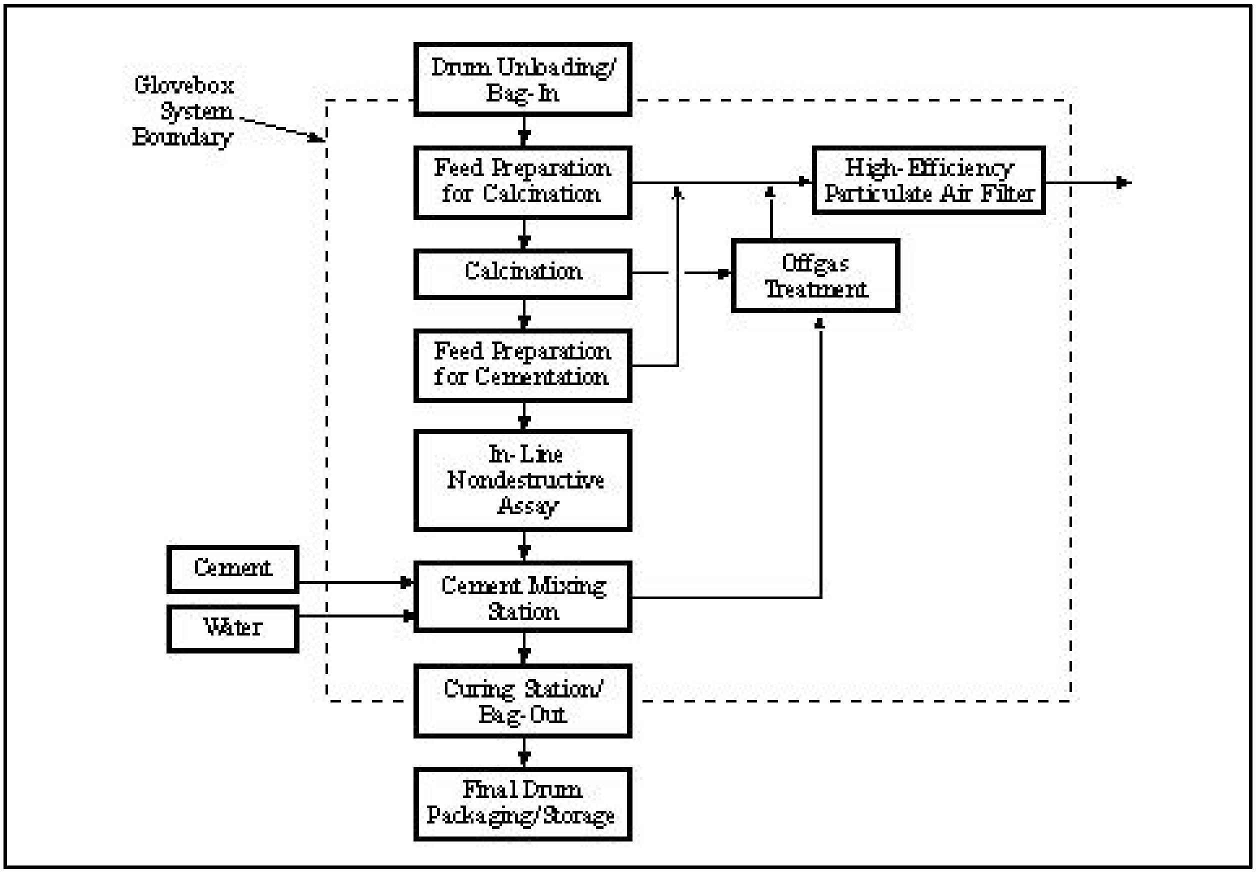

Figure C–13 Cement-Based Immobilization Process for Graphite Residues

❐ Detailed Process Description

Drums with the capacity of 208-L (55-gal) would be transferred from storage into a contamination control enclosure. The contamination control enclosure is designed to control airflow in the event of a bag failure within a drum. The drums would be opened and the integrity of the packaging would be checked. If the packaging has not been compromised, the containers would be transferred into the glovebox. The containers, including outer bags, clamshells and other packaging materials, would be removed from the drum and bagged into the glovebox. If the integrity of the packaging has been compromised, the package would be overpacked with a new plastic bag before transfer to the glovebox. There would be approximately five containers in each drum.

After bag-in, the IDCs of the residue containers would be verified and the original residue containers would be transferred either to a residue sorting and loading station or, for the large sized residue such as graphite molds, directly to a crusher. Both the sorting and loading station and crusher would provide local dust control. The sorting and loading station would contain a 0.32-cm (1/8-in) sieve that would be used to separate oversized residue and tramp material (e.g., nuts, bolts). The sieved residue fines would be transferred into burn boats. Tramp material would be separated and transferred for transuranic waste size reduction and packaging or placed into the cemented waste before curing. Oversized residues would be crushed and fed back to the loading station. Each container would be filled to contain approximately 83.5 g (2.9 oz) of plutonium. After the filling step, the burn boats would be transferred to the muffle furnace for calcination.

Calcination is required to high-fire the residue, which would remove the reactive characteristics in the residue stream. Each batch would be calcined at 900°C (1,650°F) for 4 hours, which would oxidize carbon and organic to carbon dioxide and eliminate water, thereby increasing the bulk density of the residue. After cooling, the residue would be transferred for feed preparation for cementation.

The burn boats containing the calcined residue would be transferred to a residue sorting and loading station. As described previously, the residue will be sieved and the residue fines would be loaded into metal containers. As required, oversized residues would be crushed and loaded into the containers. Each container would be filled to contain approximately 83.5 g (2.9 oz) of plutonium. After the container filling step, the containers would be transferred to an in-line nondestructive assay station.

Nondestructive assay would be performed, after which, the container would be moved to the mixing station. Then, measured quantities of water and cement would be manually blended into the residue containers. The material would be mixed until all of the water has been absorbed by the cement and the mixture thickens. Because of the potential for heat generation, provisions for actively cooling the container during and after mixing may be required for certain residue IDCs. During mixing, there is a potential for vapor generation produced by an exothermic reaction associated with the hydration of the cement and through hydrogen gas generation produced from radiolysis and hydrolysis. Therefore, provisions would be incorporated as necessary for the collection and extraction of these vapors in both the mixing station and curing station. The container would then be removed from the mixing station into a set of curing gloveboxes and set aside for a 24-hour curing period. After curing has been completed, the cans would be bagged out of the glovebox.

Assayed, cemented residue containers that meet the WIPP/Waste Acceptance Criteria would be transferred for final drum packaging. Two containers would be loaded into a pipe component already staged inside of a 208-L (55-gal) drum. The sealed drums would be placed into interim site storage awaiting shipment to WIPP.

C.5.3 Blend Down

Blend down technology would involve mixing residues with other materials to reduce plutonium concentrations below safeguards termination limits. The blending process would be conducted inside a glovebox located in Module E of Building 707 for all residues except salts or in Room 3701 of Building 371. Uranium oxide and other nonradioactive materials, such as magnesium oxide sand, have been proposed as the blending diluent. Most uranium oxide currently at Rocky Flats has been classified as Resource Conservation and Recovery Act hazardous waste and a decision on its use must be made. Uranium oxide imported from another site, such as Savannah River Site, may be required.

Blending of granular or powdered residue materials with inert or lower-assay powdered residues, and the subsequent packaging and assaying, is considered to be a proven technology. Capabilities necessary to satisfy all alternatives are being installed at Rocky Flats as part of the ongoing stabilization programs, and should be operational within several months of issuance of the EIS. The specific location of the blending process is uncertain, although the process would be consistent with either Building 371 or Building 707.

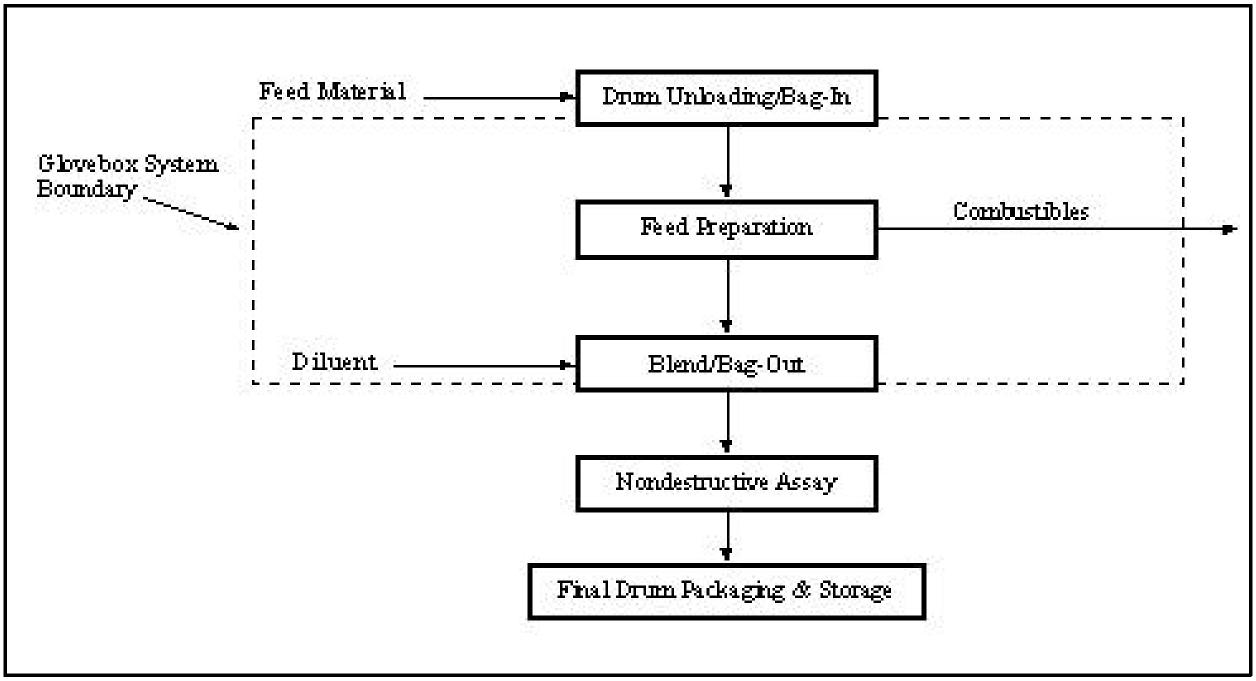

Figure C–14 Blend Down Process for All Residues Except Salts

The blend down process for all residues is shown in **Figure C–14**. For most of the residues, the feed materials would be sorted and size-reduced to enhance uniform mixing with the diluent. Calcination would be required for ash residues (except for graphite fines) in order to convert reactive metals to unreactive oxides and to meet the high-fired qualification. The feed materials would then be blended with the diluent. For plutonium fluorides, the feed materials would be unpacked and then blended with the diluent without size reduction. The packaged residue product for each residue would be removed from the glovebox, nondestructively assayed for accountability purposes, packaged in the final transport/storage container, and placed in interim storage.

❒ Detailed Process Description

Drums and containers would be manually transferred from storage into a contamination control enclosure and examined for damage. The contamination control enclosure is designed to control airflow in the event of damage or a bag failure within a drum or container. The drum or container would be opened and the integrity of the packaging would be checked. If the packaging has not been compromised, the drum/container would be transferred to the glovebox. Any unnecessary packaging materials would be removed to limit the amount of packaging introduced into the feed preparation glovebox. If the integrity of the packaging has been compromised, the packaging would be overpacked with a new plastic bag before transfer to the glovebox. All individual drums/containers would be bagged into the feed preparation glovebox.

For most residues, the feed would be introduced into the glovebox, the IDC verified, and each package/bag would be opened and sorted/sieved to remove any tramp metal and other unwanted materials. Following the sorting/sieving, combustible and filter media residue feed materials would be shredded, and ash, sludge, glass, graphite, and inorganic residue feed materials would be crushed and sieved to produce a particle size for uniform mixing with the blending diluent. The crushed feed would again be sieved with a finer mesh screen and any large chunks will be returned to the crusher for reprocessing. Both the shredded and crushed materials would then be batched so that each new container will average 83.5 g (2.9 oz) of plutonium, except sand, slag, and crucible, which would be batched to 18.1 g (0.64 oz) of plutonium because of the high ratio of diluent to residue matrix required. Calcination would be required to high-fire the incinerator ash residues and firebrick fines. It may also be required for certain feeds, such as sand, slag, and crucible, with reactive characteristics. Further study must be completed to ensure that dilution of the feed will negate reactivity characteristics in the feed streams. Each batch would be calcined at 900°C (1,650°F), which would oxidize carbon and organic materials to carbon dioxide and eliminate water, and increase the bulk density of the ash residues. Crushing may be required after calcination. After crushing, the batches would be available for blending.

For plutonium fluorides, the materials would be removed directly from containers and would be batched so each new container would average 18.1 g (0.64 oz) of plutonium, due to its high ratio of diluent to residue matrix. It would be necessary to batch the fluorides with 18.1 g (0.64 oz) plutonium per package to maintain the final package weight at less than 9 kg (20 lb) to allow for physical handling within the glovebox. The batching used for each residue would allow for maximum packaging flexibility during the final packaging step after nondestructive assay has been completed and accountability data has been analyzed. Combustible packaging materials from the individual containers would be bagged out of the glovebox and sent to a combustible handling process. Any metals or other unwanted materials would be bagged out of the glovebox and managed appropriately.

Blending may be done manually or mechanically using a blender. In either case, blending would be a hands-on operation, whether the addition of the diluent to the batched feed and subsequent mixing would be accomplished in small batches manually or whether loading and unloading steps must be accomplished for use of a mechanical blender. If Rocky Flats uranium oxide is used as the diluent, it would be calcined and sieved in another location to convert it to a uniformly sized powdery oxide form. If uranium oxide is imported from the Savannah River Site, this step would not be necessary. Additional or different stabilization processings may be needed if salt, magnesium oxide sand, or other blending material is used as a diluent instead of, or in addition to, uranium oxide. This would ensure that the diluent material, when added to the crushed and sieved feed materials, would blend uniformly. The blended material would then be bagged from the glovebox and placed in a convenience container for safe handling.

Nondestructive assay would be performed to ensure requirement limits are met and to obtain data to ensure that required accountability procedures are followed. Nondestructive assay methods would be selected to ensure that the best accountability data are obtained. Assayed product packages would be selected for final packaging to minimize the number of shipping containers required to be shipped to WIPP. Selected packages would be loaded into an inner container and sealed before placing of the container into the final outer shipping container.

C.5.4 Pyro-Oxidation and Blend Down of Pyrochemical Salts

The pyro-oxidation and blend down process would remove reactive metals from the salts and mixes them with a matrix to reduce the plutonium concentration to below the safeguards termination limit for pyrochemical salts. This technology can be used on electrorefining salts, molten salt extraction salts, and direct oxide reduction salts. The pyro-oxidation and blending processes would be conducted inside gloveboxes located in Module E of Building 707 or in Room 3701 of Building 371. Uranium oxide and other nonradioactive materials, such as magnesium oxide sand, have been proposed as the blending diluent. Most uranium oxide currently at Rocky Flats has been classified as Resource Conservation and Recovery Act hazardous waste and a decision on its use must be made. Uranium oxide imported from another site, such as the Savannah River Site, may be required.

Pyro-oxidation of salts in stationary furnaces is considered to be a proven technology, although specific process variables are being evaluated in an attempt to make the pyro-oxidation process more compatible with a pyro-distillation follow-on processing step. Pyro-oxidation of reactive salts is part of the Rocky Flats response to Defense Nuclear Facilities Safety Board Recommendation 94-1 to stabilize potentially higher-risk or reactive materials. Rocky Flats has the capability to support the ongoing stabilization programs, and operations are pending. While not a technology risk for the pyro-oxidation process, the salts, once pyro-oxidized, cannot be subsequently salt scrubbed, which is the only current process to allow plutonium separation using the Purex process. The on-going stabilization program trades the technical and programmatic risk of not using a proven Savannah River Site residue disposition approach (Purex) against the reduction of an immediate safety risk. The pyro-oxidation process is, however, a prerequisite step for both salt distillation and aqueous distillation. Blending of granular or powdered residue materials with inert or lower-assay powdered residues, and the subsequent packaging and assaying, is considered to be a proven technology. Capabilities necessary to satisfy all alternatives are being installed at Rocky Flats as part of the ongoing stabilization programs, and should be operational within several months of issuance of the EIS. The specific location of the blending process is uncertain, although the process would be consistent with either Building 371 or Building 707.

The pyro-oxidation and blending process steps for pyrochemical salt residues are shown in **Figure C–15**. The salt residues would be sorted and batched in preparation for pyro-oxidation. The salts would be pyro-oxidized to convert reactive metals to oxides. After pyro-oxidation, the oxidized salts and plutonium oxide would be size-reduced. They would then be blended with the diluent. The packaged product would be removed from the

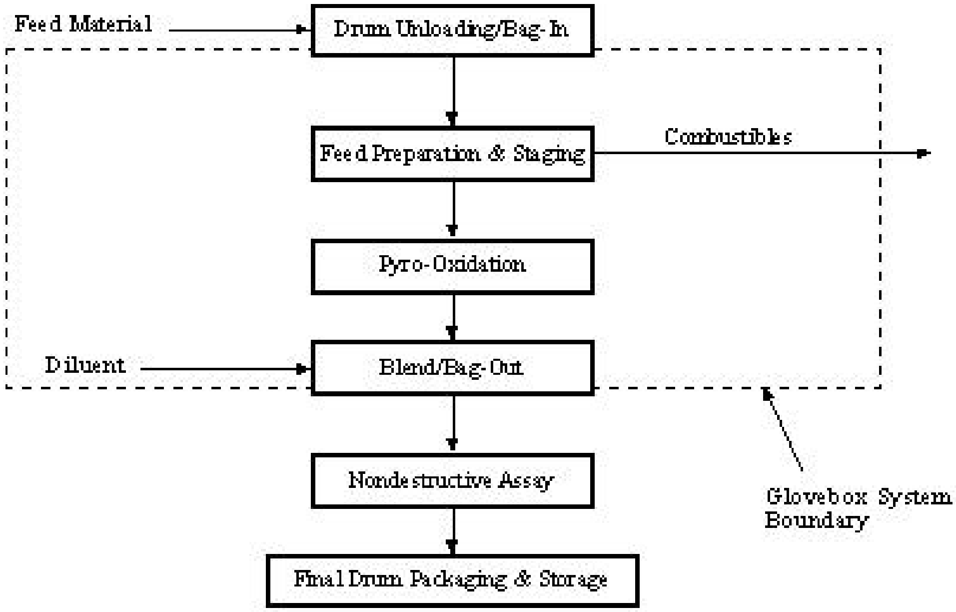

Figure C–15 Pyro-Oxidation and Blend Down Process for Pyrochemical Salt Residues

glovebox and nondestructively assayed for accountability purposes, packaged in the final transport/storage container, and placed in interim storage.

❒ Detailed Process Description

As required, drums would be manually transferred from storage into a contamination control enclosure. The contamination control enclosure is designed to control airflow in the event of a bag failure within a drum. The drum would be opened and the integrity of the packaging will be checked. If the packaging has not been compromised, the containers would be transferred to the glovebox. Any unnecessary packaging materials would be removed to limit the amount of packaging introduced into the salt feed preparation glovebox. If the integrity of the packaging has been compromised, the packaging would be overpacked with a new plastic bag before transfer to the glovebox.

The salts would be introduced into the glovebox, one package at a time, and the IDC verified. The individual packages would be opened and loaded into a magnesium oxide crucible in preparation for pyro-oxidation. Combustible packaging materials from the individual packages would be bagged out of the glovebox and sent to a combustible handling process. Other materials would be bagged out and managed appropriately.

Once the crucible is loaded with salt feed, it would be placed in a glovebox furnace and heated to approximately 800°C (1,500°F) with an oxidant, such as sodium carbonate, as a reagent for 2 hours, stirring continuously. The product would be a lower plutonium-bearing salt matrix on top and plutonium oxide bound in a salt matrix at the bottom of the crucible. Pyro-oxidation could be applied to both sodium chloride-potassium chloride and calcium chloride matrices. This process converts reactive metals (calcium and sodium) to oxides. Stirring is discontinued during the cooling phase. When the furnace has cooled to below 100°C (212°F), the crucible would be removed from the furnace. During the heating, stirring, and cooling phases, argon would flow through the furnace. During the last part of the stirring phase, argon would be replaced by a mixture of air and argon.

Once the crucible is removed from the furnace, it would be allowed to completely cool before breakout. The salt matrix and plutonium oxide would then be removed from the crucible. The crucible would be discarded and treated as inorganic ash residue. At this point, the salt matrix and plutonium oxide would be screened and sent through a crusher in order to achieve a uniform size for blending with diluent, and placed in containers in preparation for blending. After the materials are size-reduced, they would be batched to 18.1 g (0.64 oz) or less of plutonium due to the high ratio of diluent to residue matrix required. This would allow for maximum packaging flexibility during the final packaging step after nondestructive assay has been completed and accountability data has been analyzed.

Blending may be done manually or mechanically using a blender. In either case, blending would be a hands-on operation, whether the addition of the diluent to the batched feed and subsequent mixing is accomplished in small batches manually or whether loading and unloading steps must be accomplished for use of a mechanical blender. If Rocky Flats uranium oxide is to be used as the diluent, it would be calcined and sieved in another location to convert it to a uniformly-sized powdery oxide form. If uranium oxide is imported from the Savannah River Site, this step would not be necessary. Additional or different stabilization processings may be needed if salt, magnesium oxide sand, or other blending material is used as a diluent instead of, or in addition to, uranium oxide. This would ensure that the diluent material, when added to the crushed and sieved feed materials, would blend uniformly. The blended material would then be bagged from the glovebox and placed in a convenience container for safe handling.

Nondestructive assay would be performed to ensure requirement limits are met and to obtain data to ensure that required accountability procedures are followed. Nondestructive assay methods would be selected to ensure that the best accountability data are obtained. Assayed product packages would be selected for final packaging to minimize the number of shipping containers required to be shipped to WIPP. Selected packages would be loaded into an inner container and sealed before placing the container into the final outer shipping container.

C.5.5 Sonic Wash

Sonic washing for combustible, filter media, glass, and inorganic residues removes the organic and nitrate contaminants from the residue waste to eliminate the potential flammable hazard and allow for its disposal at WIPP. Along with nitrate removal, the sonic wash process may remove up to 90 percent of the plutonium from the residue waste (up to 99 percent for glass residues). At this removal efficiency, the waste would meet safeguards termination limits for disposal at WIPP. The plutonium and nitrate removed from the residues would be vitrified to meet the safeguards termination limit for vitrified waste disposal at WIPP. The sonic washing process would be conducted inside a glovebox located in Room 3701 of Building 371.

Sonic washing of materials, using sound waves to enhance the partition of a residue into a below-safeguards termination limit (washed) component and a concentrated component which would then be vitrified to meet safeguards termination limit, has been demonstrated with residue-type materials on a bench scale. Due to the significant effort required to demonstrate a consistent process and develop the procedures and analysis necessary for routine operation, the estimated time required to deploy this operation would be 2 years after the issuance of the Record of Decision.

The sonic washing process for residues is shown in **Figure C–16**. The process steps include drum unloading and package bag-in, feed preparation, thermal desorption and steam passivation (for combustible and filter media residues), sonic washing and decant/filtration, evaporation and water recycle, plutonium vitrification and package bag-out, and oven drying and package bag-out. Nondestructive assay would be performed, followed by final drum packaging and storage.

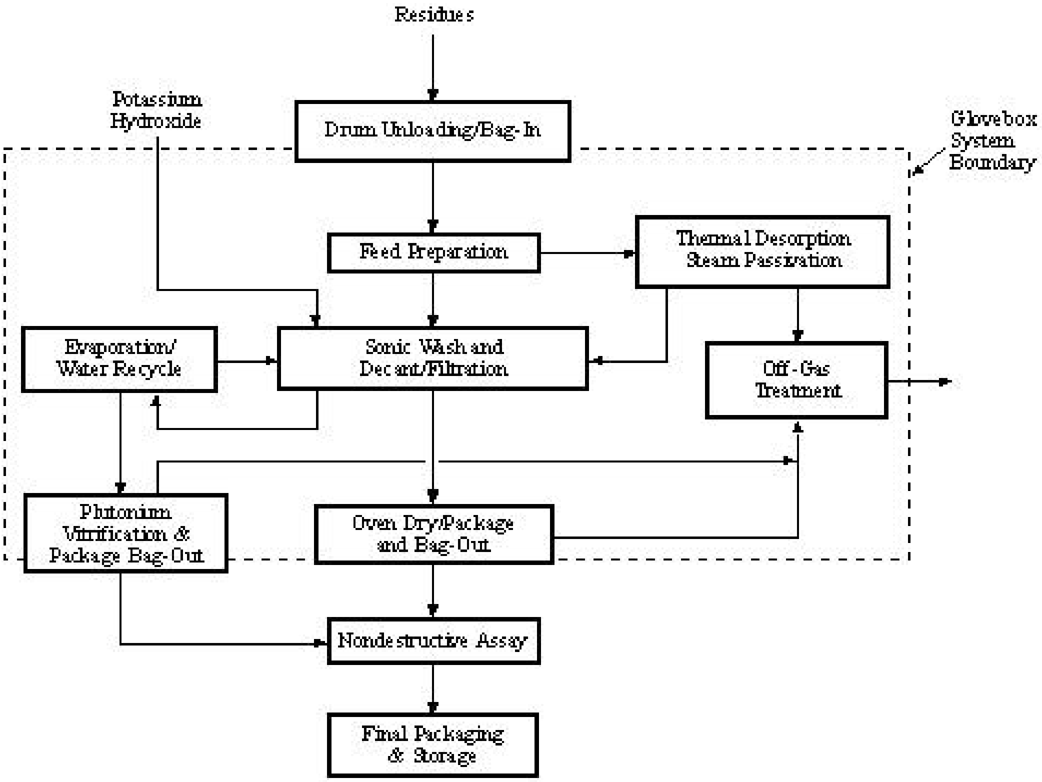

Figure C–16 Sonic Wash Process

❒ Detailed Process Description

Drums would be manually transferred into a contamination control enclosure and unpacked. The contamination enclosure is designed to control airflow in the event of a bag failure within a drum. The drums would be opened and the integrity of the packaging will be checked. If the packaging has not been compromised, the containers would be transferred into the glovebox. Any unnecessary packaging materials would be removed to limit the amount of packaging introduced into the feed preparation glovebox. If the integrity of the packaging has been compromised, the package would be overpacked with a new plastic bag before transfer to the glovebox. Each of the individual bags would be bagged into the feed preparation glovebox.

The residue feed would be introduced into the glovebox and the IDC verified. Each bag would be opened to remove any tramp metal and other unwanted materials, and sorted to separate the organic contaminated residues from the nonorganic contaminated residues. Following the sorting, the feed materials would be shredded and batched to 5 kg (11 lb) of residue in preparation for low temperature thermal desorption and sonic washing. Any metals and other unwanted materials would be bagged out of the glovebox and managed appropriately.

The organic contaminated residues would be fed in 5 kg (11 lb) batches to a low temperature thermal desorption unit which operates under a vacuum at 80°C (176°F). During the 6-hour process, the organic contamination would be volatilized from the residues. The organic containing off-gas would be treated by silent discharge plasma destruction before being vented through a high-efficiency particulate air filter. The resulting residues would then be sonic washed.

Sonic washing would be intended to remove both the nitrate and plutonium contamination from the residue waste to meet acceptable waste storage criteria at WIPP. The 5-kg (11-lb) batches of residue from feed preparation would be sonic washed with 50 L (13-gal) of aqueous solution containing 10 percent excess potassium hydroxide to neutralize any residual nitric acid contained within the waste. After 2 hours of sonic washing, the acid would be neutralized forming potassium nitrate and water, and approximately 90 percent of the plutonium would be removed from the residue waste (and 99 percent of the plutonium from glass residues). The residue solids would be separated from the nitrate and plutonium-containing solution by decanting and filtration. The solids would be transferred to a drying pan. The solution from sonic washing would contain approximately 90 percent of the plutonium, as solids, and more than 97 percent of the nitrates, which are dissolved. This solution would flow to the evaporation and recycle step.

The nitrate and plutonium bearing solution would be evaporated in a forced circulation evaporator to produce water which may be recycled to the sonic washing step, and dried nitrate and plutonium solids. While the evaporator type has not yet been selected, it would evaporate approximately 50 L (13 -gal) of water within a 3-hour period, and would probably operate under a vacuum at a temperature below 100°C (212°F). This would require a heat load of approximately 40,000 Btu (11.7 kilowatts) per hour, and the capability of evaporating the liquid while preventing the collection of the plutonium on the heat transfer surface. After evaporation, the solids would be weighed into batches containing an average of 83.5 g (2.9 oz) plutonium and placed into 8.2-L (2.2-gal) containers in preparation for vitrification.

Following batching, a blending step would be required where the residue material would be blended and diluted with low-melting-temperature frit. Each residue stream to be immobilized by the vitrification process would be analyzed to determine the appropriate proportions of residue and frit to meet the WIPP/Waste Acceptance Criteria requirements. The residue containers would then be staged for vitrification. Once the residue container is charged and blended, it would be positioned into the heating chamber of the muffle furnace. The furnace would be energized and there will be a gradual ramp-up in temperature within the chamber. The target temperatures for vitrification would be between 700 and 1,300°C (1,300 and 2,400°F). The actual temperature would be determined for each specific type of residue before vitrification.

Engineering investigations are underway to identify the most effective method to extract and capture the off-gases generated by the heating process. Water vapor, carbon dioxide, nitrous oxide, trace quantities of acid gases, organic, and particulates may be generated during vitrification. This description assumes the use of a dry scrubber using potassium carbonate for off-gas treatment. A concern with residual organic contaminants in the residue feed stream subsequently volatilizing during the heating process necessitated an investigation into the incorporation of a design modification which would continuously purge the heating chamber with inert gas during the processing of the residues.

The heating process would be approximately 4 hours in duration. The container would be allowed to cool to 100°C (212°F) before removal from the furnace. After final cooling, the container would be sealed with a lid and placed into a convenience can before being bagged out of the glovebox for nondestructive assay. The sonic wash technology would produce 8.2-L (2.2-gal) containers of vitrified plutonium/nitrate waste.

The sonic-washed residue waste would be transferred from sonic washing to the drying oven in drying pans. The waste would be dried under a vacuum at 80°C (176°F) for 2 hours, producing a dry waste (containing approximately 1 percent water). Off-gas from drying would be treated before high-efficiency particulate air filtration to capture or destroy any volatilized contaminants. After cooling, the waste would be weighed, and the quantity of plutonium estimated, as the waste is transferred to 8.2-L (2.2-gal) containers. These containers would be bagged out of the glovebox and packaged into convenience cans. Based on 90 percent of the plutonium being removed in the sonic wash, the plutonium remaining in the waste would be below the safeguards termination limit required for shipment and disposal at WIPP.

Nondestructive assay would be performed to ensure requirement limits are met and to obtain data to ensure that required accountability procedures are followed. Nondestructive assay methods would be selected to ensure that the best accountability data are obtained. Within this single step, 8.2-L (2.2-gal) waste containers from both plutonium vitrification with bagout and residue drying with bagout steps would be analyzed. Assayed product packages would be selected for final packaging from both the vitrified plutonium/nitrate waste and the dried residue waste to minimize the number of shipping containers required to be shipped to WIPP. Selected packages would be loaded into an inner container and sealed before placing the container into the final outer shipping container. The sonic wash technology would produce drums containing pipe components loaded with nitrate-washed combustibles and vitrified plutonium waste.

C.5.6 Catalytic Chemical Oxidation of Combustible Residues

The catalytic chemical oxidation process is a relatively new dissolution process that has been incorporated into a standard aqueous separation process for the processing of plutonium-containing residues. The catalytic chemical oxidation/aqueous process would be used to remove all of the plutonium from the residue matrix material, creating a concentrated plutonium oxide stream and converting the residual material into carbon dioxide and water. Catalytic chemical oxidation processing can be used on combustible residues, including wet and dry combustibles, plastic, and leaded gloves. The catalytic chemical oxidation process would principally be conducted inside gloveboxes located in Room 3701 of Building 371. The catalytic chemical oxidation process for combustible residues is shown in **Figure C–17**.

Catalytic chemical destruction of combustibles at elevated temperatures and pressures, while demonstrated in a commercial environment, is completely unproven as a production process in the size and service required, and for residue material applications. Due to the significant effort required to demonstrate a consistent process and to develop the procedures and analysis necessary for routine operation, the estimated time to deployment of this operation would be 4 years after the issuance of the Record of Decision.

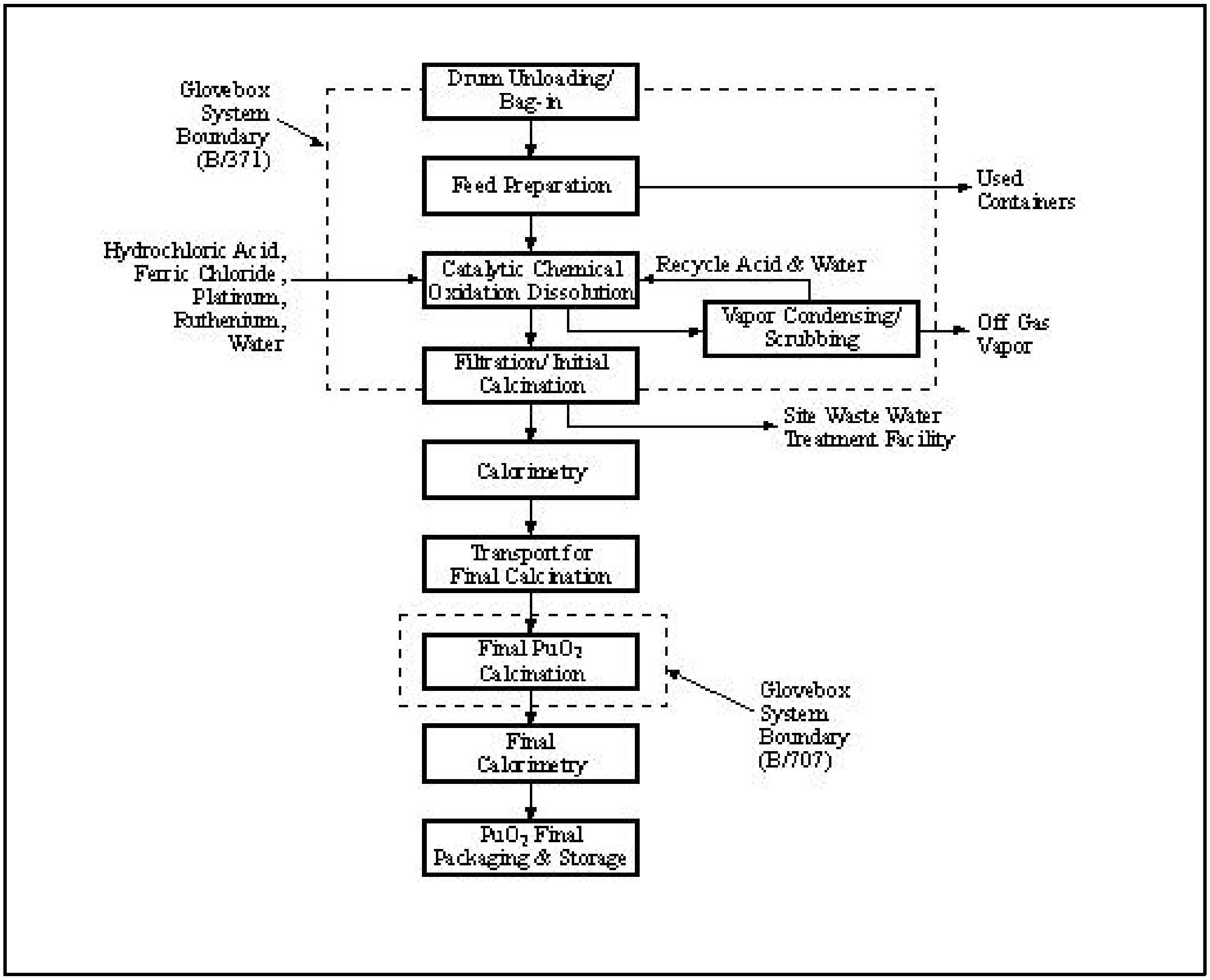

Figure C–17 Catalytic Chemical Oxidation Process for Combustible Residues

The catalytic chemical oxidation process is a dissolution process to separate plutonium from a residue using a catalyst to enhance the oxidation of liquid or solid organic materials, and dissolve metallic components of the residues. Catalytic chemical oxidation would utilize a hydrochloric acid solution at elevated temperatures and pressures that would maintain the solution below its boiling point. As the material is oxidized, the catalyst would be regenerated using injected oxygen. Once the plutonium species are dissolved and all of the combustible material is destroyed, the solution would be neutralized and the dissolved solids, including plutonium, would be precipitated as oxides. The resulting solids would be separated and treated for storage and shipment, and the liquids treated in the site wastewater treatment facility.

❒ Detailed Process Description

Drums would be manually transferred into a contamination control enclosure and unpacked. This step would contain any contamination which could result from an individual package containment damaged by radiolysis or other physical damage to the package during storage. Any unnecessary packaging materials would be removed to limit the amount of packaging introduced into the feed preparation glovebox.

After bag-in, the IDCs of the combustible residue containers would be verified. The original residue containers would be transferred to a residue sorting and loading station which would provide local dust control and contain a sorting station to separate all oversized residue and tramp material. As the dissolver would only convert combustible residues, all other material would be sorted out and removed from the glovebox with the tramp material. The combustible residues would be shredded and weighed into dissolver feed containers. Each dissolver feed container would hold a 100-g (3.5-oz) charge of bulk shredded residue, each charge containing an average of 1.87 g (0.066 oz) of plutonium. The dissolver feed containers would be transferred to the dissolver glovebox as required.

The catalytic chemical oxidation dissolution step would consist of a 7.6-L (2-gal) catalytic chemical oxidation dissolver, a condenser for off-gas treatment, and piping and tankage to support the equipment. The process would be operated on a batch basis. First, the dissolver tank would be filled with 6M ferric chloride and 1M hydrochloric acid solution. Platinum and ruthenium would also be added from 0.001M solutions of each. The solution would be heated to 175 to 200°C (350 to 390°F) at a pressure of 60 to 110 pounds per square inch gauge (410 to 760 kilopascal gauge) to maintain the solution below boiling. The prepared residue would be fed into the heated solution at one 100-g (3.5-oz) residue charge per hour, and would be agitated to maintain the solution in contact with the solid particles. As each dissolution charge would take 2 hours of active dissolution time, an additional 100-g (3.5-oz) residue charge would be fed to the dissolver every hour until the dissolver contains 100 g (3.5 oz) of plutonium. Each 100-g (3.5-oz) plutonium batch would require approximately 54 100-g (3.5-oz) residue charges. Once the dissolver would contain 100 g (3.5 oz) of plutonium, no further charges would be added to the dissolver. Heat and oxygen would continue to be applied to the unit for an additional hour to vaporize all of the acid and neutralize the solution. As the acid is removed from the solution, the dissolved metals would precipitate as oxides. The solution would be cooled and the slurry pumped from the dissolver to a filter holding tank. All of the carbonaceous materials in the residue feed would be oxidized to carbon dioxide and water.

The slurry would be drained from the filter holding tank into stainless steel filter boats with a sintered metal filter. The liquids would be sucked through the filter, leaving the solid metal precipitate material within the filter boat. The filtrate would be collected and sent to the site wastewater treatment facility. These filter solids would include the plutonium, the iron precipitated from the ferric chloride reagent, platinum and ruthenium catalysts, and any other trace metals dissolved and precipitated during the catalytic chemical oxidation reaction. The filter boats would be placed in a calciner and heated to approximately 400°C (750°F) for an hour to convert the plutonium and metal precipitates to plutonium and metal oxides and carbon dioxide. The plutonium oxide would then be consolidated into slip-lid cans, weighed, and bagged out of the glovebox. The package would be loaded into a convenience can, as necessary, before being nondestructively assayed and transported to Building 707 for final calcination.

The plutonium oxide package would be assayed for plutonium content based on its rate of thermal generation using calorimeters and gamma-ray isotopic spectrometer equipment. After assay, the containers would be ready for storage. This activity is required to maintain accountability within the catalytic chemical oxidation material balance area. The cans containing the plutonium oxide would be placed in appropriate outer containers and transferred to the Building 371 loading dock. The containers would then be transported to the Building 707 loading dock by intra-site truck transportation, and moved to appropriate vault storage pending final calcination.

The plutonium oxide cans would be transferred from the Building 707 storage vault to Module J and bagged into the plutonium stabilization and packaging system. The plutonium oxide would be removed from the cans and placed in furnaces, and calcined at 1,000°C (1,800°F) for 4 hours. The material, now suitable for long-term storage or transportation, would be weighed, characterized, and placed in a 3013 inner container. This container would then be removed from the glovebox via the bagless transfer process and sent to calorimetry. The plutonium oxide package would be assayed for plutonium content based on its rate of thermal generation using calorimeters and gamma-ray isotopic spectrometer equipment. After assay, the containers would be placed in vault storage pending DOE decisions on eventual disposition of the plutonium.

The vapors produced during the catalytic chemical oxidation process would be condensed to recover the majority of the acid and water volatilized during the reaction. The condensed acid and water would be returned to a catalytic chemical oxidation feed tank where the amount of acid contained would be determined so the appropriate quantity of recycle and fresh acid could be added to the subsequent catalytic chemical oxidation batch dissolution. The vapor from the condenser would be scrubbed to further reduce the quantity of acid and water contained in the discharge vapor.

C.5.7 Cold Ceramification for Ash Residues

The Cold Ceramification Process would stabilize the waste stream by converting chemical wastes and contaminated materials into chemically bonded phosphate ceramics. The waste stream would be mixed with reagents such as magnesium oxide and monopotassium phosphate to produce low temperature chemical reactions that would yield a ceramic material in which the hazardous constituents would be both physically encapsulated and chemically bonded. The ceramics produced would be dense, highly leach resistant, impermeable and very strong. The equipment required for cold ceramification is similar to the equipment used in current cement stabilization processes.

The cold ceramification process would blend magnesium oxide powder and monopotassium phosphate powder with the prepared ash residue, then would mix this blend with water to produce a low temperature chemical reaction. This chemical reaction would yield a dense, impermeable and highly leach resistant ceramic material that would encapsulate the contaminants. Due to the densification that occurs during the process and the high waste loadings, the final waste volume would typically be less than that of conventional treatment processes. Additional advantages of cold ceramification technology are its insensitivity to pH, lack of impact on the binding process from salts in high concentrations, and negligible hydrogen release from the final waste form.

In addition to the ceramic encapsulated residue product stream, there would be two waste streams generated. The first waste stream would be a solid transuranic waste stream consisting of size-reduced steel containers, plastic containers, and plastic bags. The second waste stream would be a gaseous effluent stream consisting primarily of water vapor, nitrogen, hydrogen, carbon dioxide, and particulates. Any tramp material removed from the waste prior to size reduction would be combined with the solid transuranic waste stream.

The cold ceramification process for Ash Residue would be located in Building 371. It is an in-container treatment process that encapsulates all Ash Residues to meet the safeguards termination limit for disposal at WIPP.

The cold ceramification process is shown in **Figure C-18**. The process steps would include drum unloading and bag-in, feed preparation, in-line nondestructive assay, ceramic mixing station, curing and bag-out, and final drum packaging and storage.

❒ Detailed Process Description

Upon demand, 208-L (55-gal) drums would be transferred from storage into a contamination control enclosure. The contamination control enclosure is designed to control airflow in the event of a bag failure within a drum. The drums would be opened and the integrity of the packaging would be checked. If the packaging has not been compromised, the containers would be transferred into the glovebox. The containers, including outer bags, clamshells, and other packaging materials, would be removed from the drum and bagged into the glovebox. If the integrity of the packaging has been compromised, the package would be overpacked with a new plastic bag, prior to transfer to the glovebox.

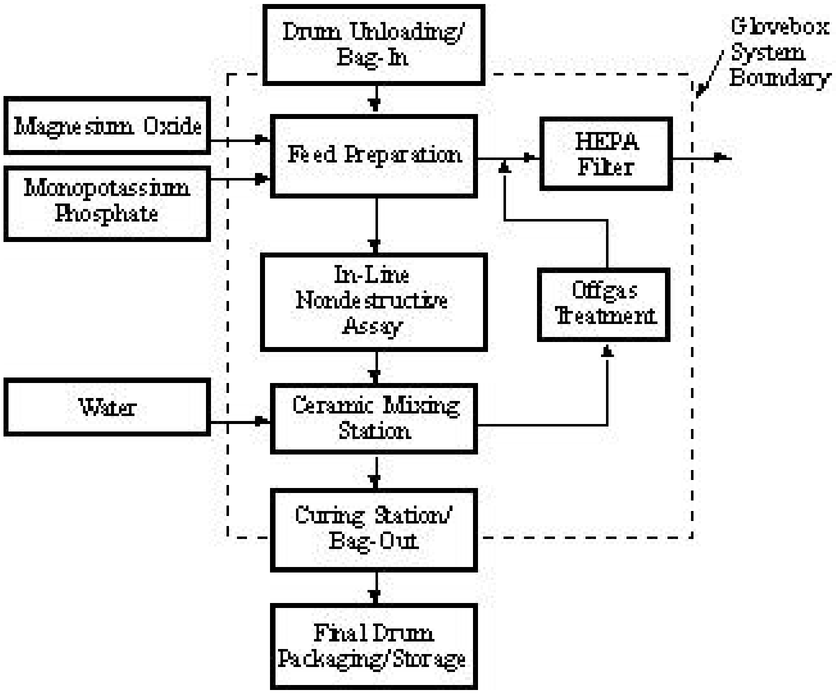

Figure C–18 Cold Ceramification Process for Ash Residues

After bag-in, the IDCs of the ash residue containers would be verified. Then, the original residue containers would be transferred to a residue sorting and loading station. This loading station would provide local dust control and would contain a 0.318 cm (0.125 in) sieve that would be used to separate all oversized residue and tramp material (nuts, bolts, etc.). The sieved residue fines would be transferred into 4-L (1.06-gal) metal containers. Tramp material would be separated and transferred for transuranic waste size reduction and packaging. Oversized residue would be crushed and fed back to the loading station for sieving. Each container would be filled to contain approximately 83.5 g (2.9 oz) of plutonium. Then, the magnesium oxide and monopotassium phosphate would be blended into the container with the residue. After the blending step, the containers would be transferred to an in-line nondestructive assay station.

Following the nondestructive assay, the container would be moved to the mixing station. Then, measured quantities of water would be blended into the residue containers. The material would be mixed until the mixture thickens and appears homogeneous. Because of the potential for heat generation, provisions for actively cooling the container during and after mixing might required for certain residue IDCs. During mixing, there would be a potential for some vapor generation produced by the chemical reaction of the reagents. Therefore, provisions would be incorporated as necessary for the collection and extraction of these vapors in both the mixing station and curing station. The container would be moved from the mixing station into a set of curing gloveboxes and set aside for a 24-hour curing period. After curing has been completed, the cans would be bagged out of the glovebox.

Assayed containers of stabilized residue that meet the WIPP waste acceptance criteria would be transferred for final drum packaging. Two containers would be loaded into a pipe component already staged inside of a 208-L (55-gal) drum.

C.6 DETAILED PROCESS DESCRIPTIONS FOR PROCESSING TECHNOLOGIES WITH PLUTONIUM SEPARATION

There are several categories of residues (identified by the Item Description Code) selected for offsite shipment from Rocky Flats and processing elsewhere that presently have Resource Conservation and Recovery Act hazardous waste numbers associated with them. The residues are in the process of being recharacterized and having those hazardous waste numbers verified or removed. If the hazardous waste numbers are validated, any site receiving and processing the residues must comply with the Resource Conservation and Recovery Act hazardous waste regulations which the site's State (South Carolina, New Mexico, or Colorado) requires for the storage, treatment, and/or disposal of hazardous waste.

C.6.1 Purex Process with Plutonium Metal or Oxide Recovery at the Savannah River Site with Preprocessing at Rocky Flats

Purex stabilization processing of residues has been proposed at the Savannah River Site. This processing technology would require residue preprocessing at Rocky Flats.

The Purex processing of sand, slag, and crucible, and plutonium fluoride residues, and scrub alloy at the Savannah River Site is considered to be a proven technology, as is any preprocessing, packaging, and transportation which must occur to allow shipping of the materials. The capability for preprocessing and packaging at Rocky Flats is being installed to support the disposition of below-safeguards termination limit materials, and should be available several months after the issuance of the EIS.

After Rocky Flats preprocessing, the packaged residues would be shipped to the Savannah River Site and processed. Sufficient numbers of 6M shipping containers are available to ship the currently-stored scrub alloy. The Type B shipping containers required for shipping of powdered materials have been recently certified; these containers have been ordered by DOE, with expected delivery in September 1998. Safe secure trailers are available as required. The Purex process is considered to be a proven technology and an on-going operation, and the processing "canyon" will be available for scheduled windows of processing consistent with its other on-going missions. The technical and programmatic risks associated with residue shipping and processing at the Purex facility are considered minimal, with the exception of Fluoride "Heels," where this previously-extracted material may not be compatible with canyon operations because of their difficult dissolution characteristics.

❒ Preprocessing at Rocky Flats

Preprocessing options include various technologies depending on the residue type: ash fusion for incinerator ash residues; packaging of ash; grinding and packaging of sand, slag, and crucible; and repackaging of plutonium fluoride residues and scrub alloy. The preprocessing activities would be drum unloading/bag-in; feed preparation/bag-out; calcination and fusion with sodium peroxide (for incinerator ash); repackaging (for plutonium fluorides); nondestructive assay of cans for accountability purposes; loading cans into shipping containers; and moving them to interim storage with shipment to the Savannah River Site. Calcination of the incinerator ash would be required to convert reactive metals to unreactive oxides before mixing with sodium peroxide. For plutonium fluorides and scrub alloy, special precautions would need to be taken to minimize operator exposure due to high radiation fields. All glovebox operations for ash residues would be performed in Building 707, while all glovebox operations for plutonium fluoride residues and scrub alloy would be performed in Building 371. The packaging process for shipment to Savannah River Site is shown in **Figure C–19**.

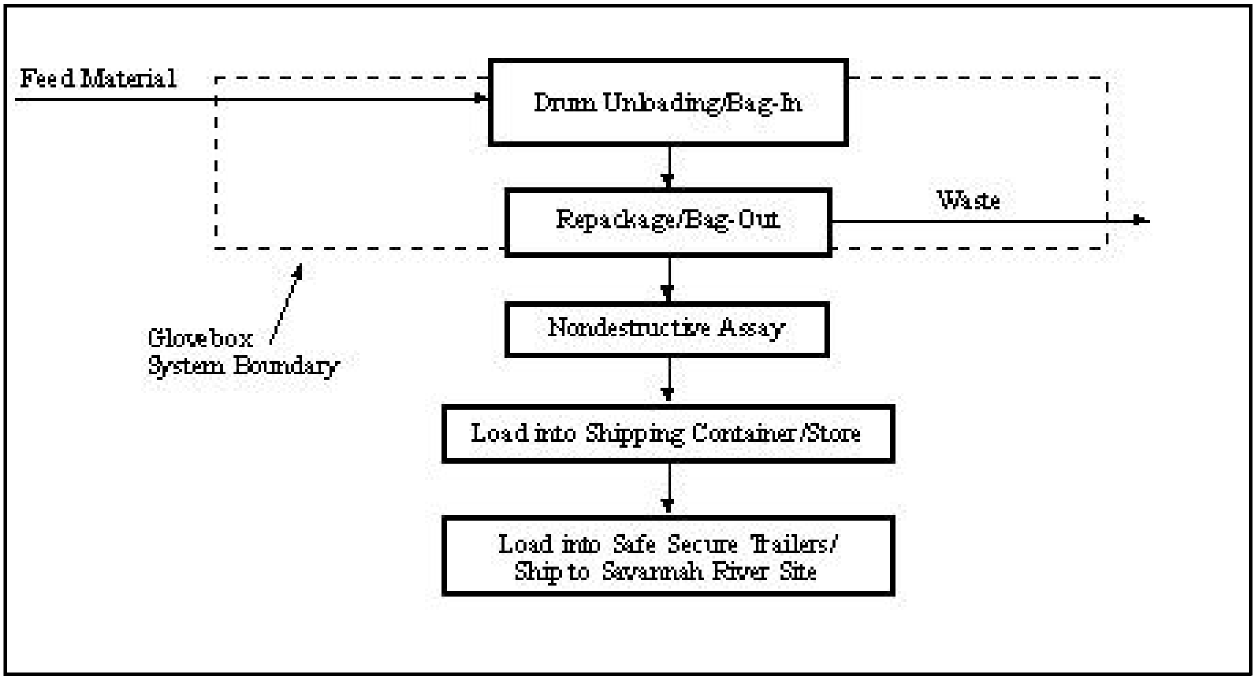

Figure C–19 Packaging Process at Rocky Flats for Shipment to Savannah River Site

Detailed Process Description

For the ash residues, drums would be manually transferred from storage into a contamination control enclosure. The contamination control enclosure is designed to control airflow in the event of a bag failure within a drum. The drum would be opened and the integrity of the packaging would be checked. If the packaging has not been compromised, the containers would be transferred to the glovebox. Any unnecessary packaging materials would be removed to limit the amount of packaging introduced into the feed preparation glovebox. If the integrity of the packaging has been compromised, the packaging would be overpacked with a new plastic bag before transfer to the glovebox.

- *Ash Fusion Preprocessing for Incinerator Ash Residues*—Incinerator ash would be introduced into the glovebox and the IDC verified. The individual packages would be opened and if necessary sieved to remove any tramp metal and other unwanted materials. Following the sieving, the incinerator ash would be crushed, as necessary, to produce a particle size which would facilitate calcination and subsequent fusion. The treated incinerator ash would again be sieved, if necessary, with a finer mesh screen and any large chunks returned to the crusher for reprocessing. The material would then be batched for calcining at nominally 2 kg (4.4 lb) bulk per batch. Combustible packaging materials from the individual packages would be bagged out of the glovebox and sent to a combustible handling process. Other materials would be bagged out and treated appropriately. Each batch would be calcined at 900°C (1,650°F), which would oxidize carbon and organic to carbon dioxide and eliminate water, and increase the bulk density of the residues. Crushing may be required after calcination. After crushing, the batches would be available for fusion. The calcined batch, at approximately 1.5 kg (3.3 lb) bulk weight, would be mixed with about 400 g (14 oz) of crushed sodium peroxide reagent materials and placed in a 2-L (0.53-gal) mild steel can.

The mixture would be heated in a furnace to 450°C (840°F) for 2 hours and allowed to cool (4-hour cycle time). The dissolvable cans would be sealed and bagged out of the glovebox using dissolvable nylon bags. They would then be placed in larger "tall cans," also of mild steel.

- *Packaging Preprocessing for Sand, Slag, and Crucible Ash Residues*—After bag-in, the IDCs of the residue containers would be verified. As the containers are emptied, they would be transferred to a sorting and loading station. At the loading station, the residue would be removed from the containers and loaded into the crusher. The empty residue containers would be removed from the glovebox as solid waste. From the loading station, the residue would be processed through a crusher for size-reduction. The crushed material would then be screened through a mesh screen, packaged into a dissolvable mild steel can, crimp sealed, and weighed. Each can would contain approximately 2 kg (4.4 lb) bulk material based on estimated weight and volume limitations. If necessary, the contents would be sampled for plutonium assay. The cans would then be bagged out of the glovebox using special dissolvable nylon bags and sent to nondestructive assay. Coarse materials from screening would be re-crushed. The crusher and screening stations would provide local dust control.

- *Preprocessing for Plutonium Fluoride Residues*—Plutonium fluorides are currently stored in containers within an in-line vault in Building 371 and in Building 777. Because of the alpha-neutron reaction between plutonium alpha particles and fluorine nuclei, the unshielded radiation exposure of operators routinely handling this material may approach administrative limits. The principal radiation is neutrons, thus hydrogenous shielding (water walls) is necessary for operator protection. Cans of fluoride would be transferred from the storage area (an in-line vault) to a glovebox in Building 371 containing suitable neutron shielding (such as 5.1- to 10.2-cm [2- to 4-in] water walls). The materials would be transferred into a dissolvable mild steel tared container, crimp sealed, and weighed. The high assay fluorides may be sampled for plutonium analysis. The dissolvable container would be bagged out of the glovebox line. Special bags (nylon) that are readily dissolvable in the Savannah River Site dissolver would be used. The empty containers would be bagged out of the glovebox line, assayed with nondestructive assay, and disposed of as waste.

- *Preprocessing for Scrub Alloy* —Scrub alloy contains a high americium content, therefore, special precautions must be taken to minimize personnel radiation exposure. The alloy currently in stainless-steel containers needs to be repackaged. They would be removed from storage vaults (located in several buildings other than Building 371), transferred to Building 371, and bagged into a glovebox line. Outer packaging materials would be bagged out and managed appropriately. Once in the glovebox line, the scrub alloy would be removed from the stainless steel can and placed into a dissolvable mild-steel 1-L (0.26-gal) can that would be crimp-sealed, weighed, and bagged out of the line using dissolvable nylon bags. Two 1-L (0.26-gal) cans would be placed into a tall mild steel can. The original stainless steel cans and other packaging would be removed from the glovebox and disposed of as waste. The scrub alloy already packaged in dissolvable containers would be statistically sampled and inspected to verify integrity of the package. Some of the alloy is already stored in Type 6M containers ready to be shipped, and some is currently in dissolvable containers stored in building vaults ready to load into shipping containers. For materials already packaged, the dissolvable container is aluminum. Inspection would require opening the outer container or the shipping container and inspecting the condition of the inner container. If deterioration of the inner container is found, then the entire package would be bagged into the glovebox line and repackaged. Repackaging would use mild steel as the dissolvable container. Dissolvable containers in shipping containers which are not inspected would be transferred directly to interim storage or to Safe Secure Transport or other DOE-approved transport, as appropriate loading. Containers that need to be loaded into shipping containers but do not require nondestructive assay would be transferred directly to shipping container loading. Packages ready for shipment would not be re-assayed using nondestructive assay.

After preprocessing of each residue and scrub alloy, nondestructive assay would be performed to confirm the amount of plutonium being shipped to Savannah River Site. After nondestructive assay, the packages would be loaded into Type 9975 (Type 6M for scrub alloy) shipping containers, and transferred to the shipping facility. There would be two cans (one can for scrub alloy) placed into each shipping container. The shipping containers would be cleaned and surveyed for contamination before transfer to interim storage within the process building or to the Building 707 shipping facility. The loaded Type 9975 (Type 6M for scrub alloy) shipping containers would be picked up at the process building and transferred to the shipping facility, where they would be loaded into a safe secure trailer or other DOE-approved transport, as appropriate and transported to the Savannah River Site. The distance from Rocky Flats to Savannah River is approximately 2,620 km (1,625 mi).

❐ Purex Processing at the Savannah River Site

The Purex process at Savannah River Site following the preprocessing at Rocky Flats is shown in **Figure C–20**. The preprocessed residues or scrub alloy would be dissolved in a Savannah River Site Canyon facility with plutonium being separated from the residue using solvent extraction technology. The plutonium would be converted to metal or oxide, prepackaged into cans, and transferred to either the FB-Line or 235-F vault until the Savannah River Actinide Packaging and Storage Facility vault is complete, where packaging would be completed (container to meet DOE-STD-3013-96 [DOE 1996c]), and stored until decisions are made on fissile material disposition. Any plutonium separated would be disposed of using an immobilization process. This process is currently in operation and no changes to the process are required to process residues and scrub alloy.

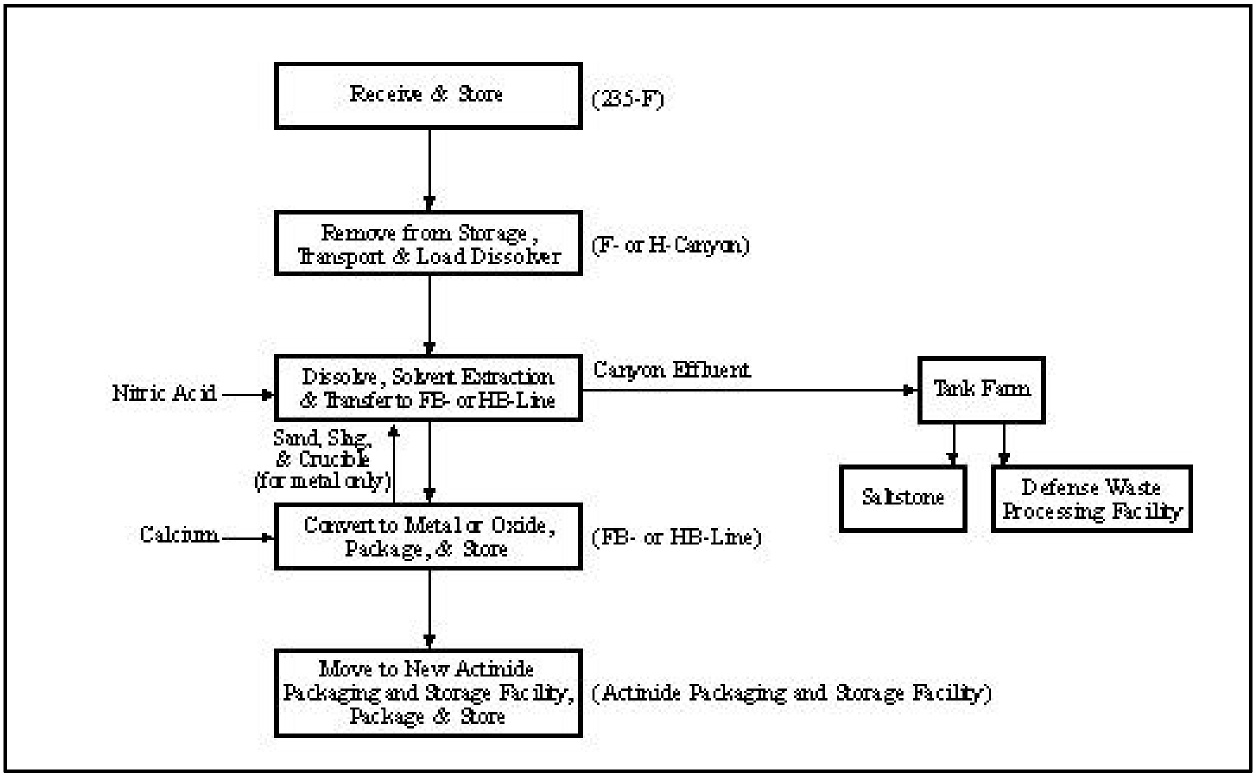

Figure C–20 Purex Process

Detailed Process Description

The shipping containers received from Rocky Flats would be unloaded, confirmatory measurements would be made, and the containers would be placed in a vault-like room in the 235-F facility. Shipping containers would be removed from storage and transported to the F- or H-Canyon crane maintenance area where the shipping containers would be opened and the cans loaded into a dissolver tube. The dissolver tube would then be loaded into a dissolver by remote control. Heated nitric acid in the tank would dissolve the residue or scrub alloy, resulting in a solution containing many constituents (dependent on material type). The solution would be purified by removing the impurities in an aqueous stream. The waste liquid would be transferred from the Savannah River F- or H-Canyon to the Savannah River high-level waste system. The plutonium product solution would be transferred to canyon hold tanks for later transfer to the finishing line.

The FB-Line process would include concentration of plutonium by cation exchange, precipitation of plutonium as a trifluoride, recovery of the trifluoride by filtration, drying of trifluoride in an oxygen atmosphere, and reduction with calcium metal to form plutonium metal buttons. The sand, slag, and crucible generated from button reduction will be dissolved in F-Canyon. The HB-Line process would include concentration of plutonium through anion exchange, precipitation of plutonium as plutonium oxalate, recovery of the oxalate by filtration, drying and calcining the oxalate, converting it to plutonium oxide. The metal buttons and oxide would be prepackaged into cans and placed in an F area vault for temporary storage. The cans would then be removed from the F area vault, placed into shipping containers, and transported to either the FB-Line or 235-F vault until the Actinide Packaging and Storage Facility is complete. At the vault the cans would be removed from the shipping containers, packaged into an outer 3013 container, and placed into the vault for long-term storage pending disposition in accordance with decisions reached under the *Surplus Plutonium Disposition Environmental Impact Statement.* Any plutonium separated would be disposed of using an immobilization process.

C.6.2 Mediated Electrochemical Oxidation

The mediated electrochemical oxidation process is a relatively new dissolution process which has been incorporated into a standard aqueous separation process for the stabilization of plutonium-containing residues. The mediated electrochemical oxidation/aqueous process would be used to remove the majority of the plutonium from the residue matrix material, creating a concentrated plutonium oxide stream and leaving the residual material suitable for disposal at WIPP. It is a dissolution process to separate plutonium from a residue using a highly oxidizing metal cation generated in an acid solution using an electrochemical cell. These metal cations would migrate from the anode to the residue surface, and oxidize any reactive substance present on exposed surfaces. The mediated electrochemical oxidation process would be used to dissolve less reactive plutonium materials from residues, along with some of the residue matrix. Depending on the substrate material, the mediated electrochemical oxidation process would oxidize some materials into carbon dioxide and water. Once dissolved, the plutonium species would be removed from the other dissolved solids by precipitation as a plutonium oxalate solid. All separated and residual solids would be treated for storage and shipment, and the liquids solidified as transuranic or low-level waste.

Mediated electrochemical oxidation has been proposed at Rocky Flats (for combustible, filter media, glass, graphite, and inorganic residues) and at Savannah River with preprocessing at Rocky Flats (for incinerator ash and graphite and firebrick fines, and graphite and inorganic residues). Though similar, enough details differ to warrant two discussions.

C.6.2.1 Mediated Electrochemical Oxidation at Rocky Flats

The mediated electrochemical oxidation process would principally be conducted inside gloveboxes located in Room 3701 of Building 371. The mediated electrochemical oxidation process for residues is shown in **Figure C–21**.

The mediated electrochemical oxidation process at Rocky Flats, consisting of dissolving the plutonium and oxidizing “combustible” constituents contained in various residues, filtering the solution, and precipitating and calcining a plutonium oxalate, is considered to be a well-demonstrated technology with radioactive materials, although not yet used in production operations in DOE facilities. The process would be required to be installed in areas of Building 371 adjacent to the neutralize-dry process to take advantage of the liquid treatment facilities. The requirements for using this area for other residue activities (e.g., neutralize dry, cementation) would impact the installation, testing, and operational schedule of new process equipment. Therefore,

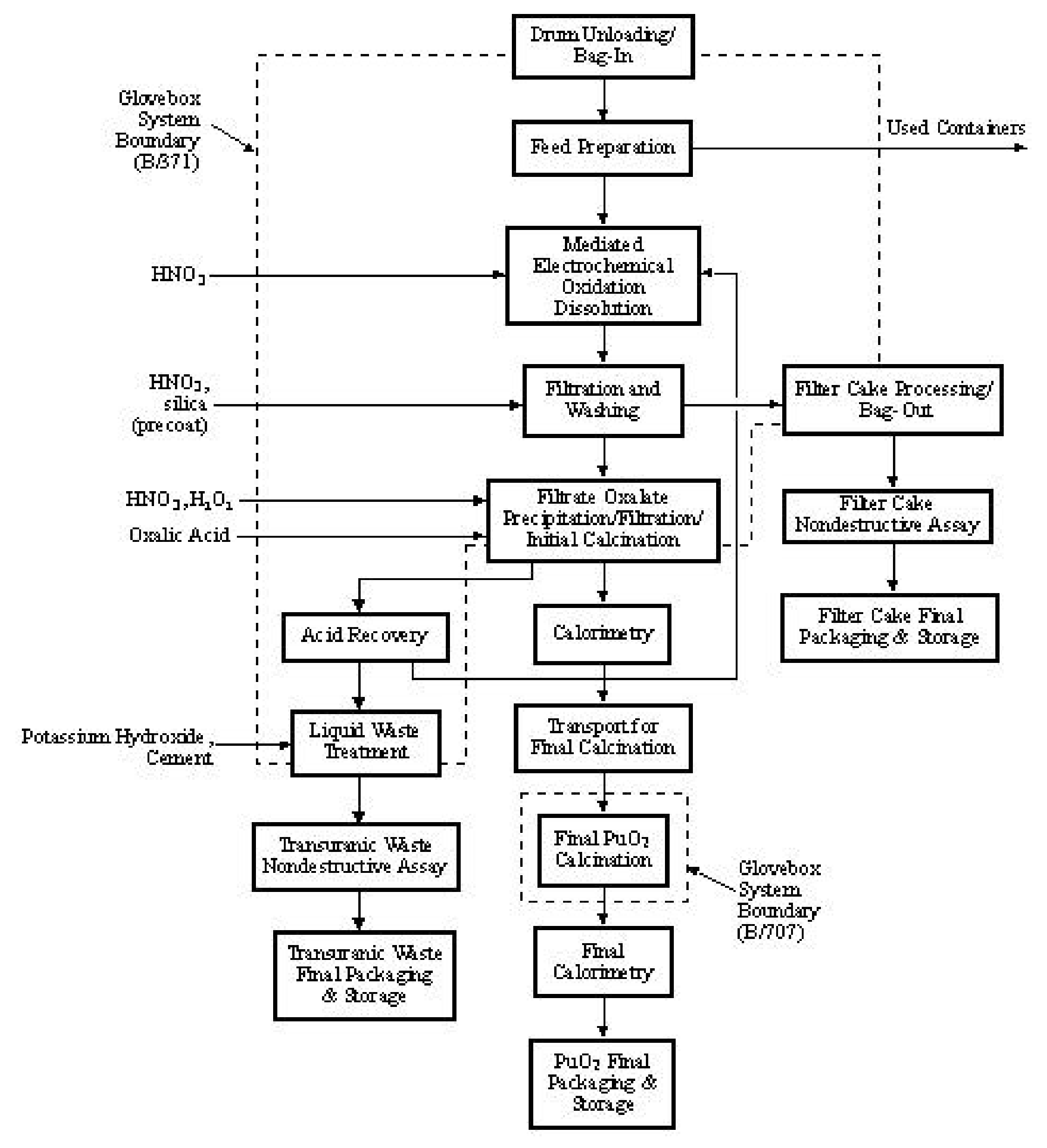

Figure C–21 Mediated Electrochemical Oxidation Process at Rocky Flats

operations of the mediated electrochemical oxidation process may not be able to start until a minimum of 4 years after issuance of the Record of Decision.

❒ Detailed Process Description

Drums would be manually transferred into a contamination control enclosure and unpacked. This would contain any contamination which could result from an individual package containment damaged by radiolysis or other physical damage to the package during storage. Any unnecessary packaging materials would be removed to limit the amount of packaging introduced into the feed preparation glovebox.

After bag-in, the IDCs of the residue containers would be verified and the original residue containers would be transferred to a residue sorting and loading station which would provide local dust control and a sorting station to separate all oversized residue and tramp material. The residues would either be shredded and batched (combustibles) or just batched (filter media, glass, and inorganic residues) into a dissolver feed container.

Graphite residues would be crushed and sieved over a 0.32-cm (1/8-in) sieve, with oversized pieces returned to the crusher, and the material passing the sieve would be batched into a dissolver feed container. Each dissolver feed container may be batched to hold either 5 kg (11 lb) of bulk residue or 200 g (7 oz) or less of plutonium. A verification step would take place to ensure that each transfer container would contain less than 200 g (7 oz) of plutonium using a gram estimator. The dissolver feed container would be transferred to the dissolver glovebox as required.

The mediated electrochemical oxidation dissolution step would consist of a 40-L (10.6-gal) mediated electrochemical oxidation dissolver, an electrolysis cell where divalent silver ions would be generated, a catholyte regeneration system, a condenser for off-gas treatment, and piping and tankage to support the equipment. The process would be operated on a batch basis.

First, the dissolver tank/anolyte compartment would be filled with concentrated nitric acid and monovalent silver ions. The electrolysis cell would be started and solution will be recirculated between the cell and the tank. The pre-batched residue would be fed into the solution, which would be agitated to maintain the solution in contact with the solid particles. The solution temperature would be maintained at between 80 and 90°C (176 and 194°F). Each dissolution batch would take two hours of active dissolution time, with another 2 hours of tank draining/filtering time and recharging and feed time. Most carbonaceous materials in the residue feed would be oxidized to carbon dioxide and water.

A filtration and washing step would filter the dissolver slurry through two vacuum drum filters with 30.5-cm (12-in) diameter drums covered with a precoat of porous silica material. The precoat would be sucked onto the rotating, fabric-covered drum in an initial step to create a filter cake, and then the dissolver slurry would be fed into the pan in which the drum would be rotating. As vacuum is applied, some of the liquid from the slurry would be sucked through the precoat while the slurry solids adhere to the precoat surface. The solids remaining on the cylindrical precoat surface would cause the liquid flow through the precoat to diminish and nearly stop in a given area. The rotation of the drum would bring these blinded areas out of the pan, where they would be spray-washed with nitric acid to displace some of the entrained solution. Before the blinded/washed area would rotate back into contact with the slurry in the pan, the outer layer of solids and precoat would be cut off to expose fresh precoat surface for filtration. The mixture of the residual solids and precoat cut from the filter, and liquids which would be entrained with it, would be collected in pans as the waste from the dissolution process. Liquids collected in a vacuum receiving tank would be transferred as feed for the oxalate precipitation process.

The pans of filter cake solids (residual solids and precoat) would be placed in an oven and heated to 150°C (300°F) for approximately 1 hour, and then placed in a 20-L (5.3-gal) can. When a can is filled, it would be taped and bagged out of the glovebox. After the solids would be dried, packaged, and removed from the glovebox line, nondestructive assay would be performed on the cans. Containers of assayed solids that meet the WIPP/Waste Acceptance Criteria fissile material limits would be transported as necessary for final drum packaging. Approximately two containers would be loaded into a pipe component staged inside a 208-L (55-gal) drum. The sealed drums would be placed into interim site storage awaiting shipment to WIPP.

The plutonium-rich solution recovered from filtration would be transferred to the precipitation feed tanks, where it would be prepared for precipitation. The batched material would be placed in glass agitated precipitator columns, and oxalic acid would be added. After a digestion period to allow for the formation and growth of plutonium oxalate crystals, the slurry would be drained into stainless steel filter boats with a sintered metal filter. The liquids would be sucked through the filter, leaving the solid plutonium oxalate material within the filter boat. This filter boat would be placed in a calciner and heated to approximately 400°C (750°F) for an hour to decompose the plutonium oxalate to plutonium oxide and carbon dioxide. The plutonium oxide would then be consolidated into slip-lid cans, weighed, and bagged out of the glovebox. The package would be loaded into a convenience can as necessary prior to being nondestructively assayed and transported to Building 707 for final calcination.

The plutonium oxide package would be assayed for plutonium content based on its rate of thermal generation using calorimeters and gamma-ray isotopic spectrometer equipment. After assay, the containers would be ready for storage. This activity is required to maintain accountability within the mediated electrochemical oxidation material balance area. The cans containing the plutonium oxide would be placed in appropriate outer containers and transferred to the Building 371 loading dock. The containers would then be transported to the Building 707 loading dock by intra-site truck transportation, and moved to appropriate vault storage pending final calcination.

The plutonium oxide cans would be transferred from the Building 707 storage vault to Module J and bagged into the plutonium stabilization and packaging system. The plutonium oxide would be removed from the cans, placed in furnaces, and calcined at 1,000°C (1,800°F) for 4 hours. The material, now suitable for long-term storage or transportation, would be weighed, characterized, and placed in a 3013 inner container. This container would then be removed from the glovebox via the bagless transfer process and sent to calorimetry. The plutonium oxide package would be assayed for plutonium content based on its rate of thermal generation using calorimeters and gamma-ray isotopic spectrometer equipment. After assay, the containers would be placed in vault storage pending disposition in accordance with decisions reached under the *Surplus Plutonium Disposition Environmental Impact Statement*. Any plutonium separated would be disposed of using an immobilization process.

The spent solution would be transferred to a batch evaporator where approximately 82 percent of the water and acid in the solution would be evaporated and condensed to be recycled through the dissolution and filtration wash steps as recycled acid. The unevaporated acid solution, containing the remaining dissolved solids, would be transferred to liquid waste treatment. The use of this acid recovery step would reduce the amount of low-level waste generated by about 80 percent. The waste acid stream from the recycle evaporator would be combined with potassium hydroxide in a cooled neutralization tank to produce a solution pH between 6.0 and 9.0. The neutralized solution and cement would be mixed together in a 208-L (55-gal) drum at a water to cement ratio of 0.2 to 0.4, and a waste loading of 15 percent to 25 percent. After the solidified transuranic waste solids are cured and removed from the glovebox line, nondestructive assay would be performed on the drums. Containers of assayed solids that meet the WIPP/Waste Acceptance Criteria fissile material limits would be transported as necessary for final drum packaging and the sealed drums would be placed into interim site storage awaiting shipment to WIPP.

C.6.2.2 Mediated Electrochemical Oxidation at Savannah River Site with Preprocessing at Rocky Flats

The stabilization of residues with the mediated electrochemical oxidation process at the Savannah River Site would require preprocessing at Rocky Flats, which would include crushing the residues as necessary, calcining (for incinerator ash and graphite and fire brick fines), and repackaging the residue materials in preparation for shipment. The cans would be bagged out of the glovebox using dissolvable nylon bags and nondestructively assayed for accountability purposes, packaged in the final transport/storage container, and stored in interim storage or sent directly to a safe secure trailer or other DOE-approved transport, as appropriate for shipment to the Savannah River Site.

The calcining and packaging process for incinerator ash and graphite and firebrick fines would be conducted inside a glovebox located in Module E of Building 707, while Mediated Electrochemical Oxidation glovebox operations would be performed in Building 371. Calcination of the incinerator ash and graphite and firebrick fines would be required in order to high-fire the material to prevent off-gassing during shipment. The packaging process for shipment to Savannah River Site is shown in **Figure C–22**. The mediated electrochemical oxidation process at Savannah River is considered to be a proven technology. The process would be required to be installed in the New Special Recovery facility. Operations of the mediated electrochemical oxidation process may not be able to start until 2 years after issuance of the Record of Decision.

❒ Preprocessing at Rocky Flats

Detailed Process Description

Drums would be manually transferred into a contamination control enclosure and unpacked. This would contain any contamination which could result from any individual package containment failure or damage by radiolysis or physical damage to the package during storage. The drum would be opened and the integrity of the packaging would be checked.

If the packaging has not been compromised, the containers would be transferred to the glovebox. Any unnecessary packaging materials would be removed to limit the amount of packaging introduced into the residue preparation glovebox. If the integrity of the packaging has been compromised, the packaging would be overpacked with a new plastic bag prior to transfer to the glovebox. All individual containers would be bagged into the preparation glovebox.

The residue material would be introduced into the glovebox, the IDC verified, and each package would be opened and sorted/sieved to remove any tramp metal and other unwanted materials. Following the sorting/sieving, the residue materials would be crushed, if necessary. For organic and graphite residues, this would provide small enough material to fit inside a 1-L (0.26-gal) dissolvable can.

The materials would then be batched into the dissolvable cans so that the cans would contain an average bulk amount of 2 kg (4.4 lb). For the materials requiring calcining, the residues would again be sieved, if necessary, with a finer mesh screen and any large chunks returned to the crusher for reprocessing. The materials would then be batched for calcining so that, after calcination, the shipping cans would contain an average bulk amount of 2 kg (4.4 lb). The repackaging of all residues is bulk weight dependent, not plutonium weight dependent.

The cans would be bagged out of the glovebox using dissolvable nylon bags. Combustible packaging materials from the individual containers would be bagged out of the glovebox and sent to a combustible handling process. Other unwanted materials would be bagged out of the glovebox and managed appropriately.

For the incinerator ash and graphite and firebrick fines, each batch would be calcined at 900°C (1,650°F), which would oxidize carbon and organic to carbon dioxide and eliminate water, and increase the bulk density of the residues. Crushing may be required after calcination. After crushing, the batches would be available for final packaging.

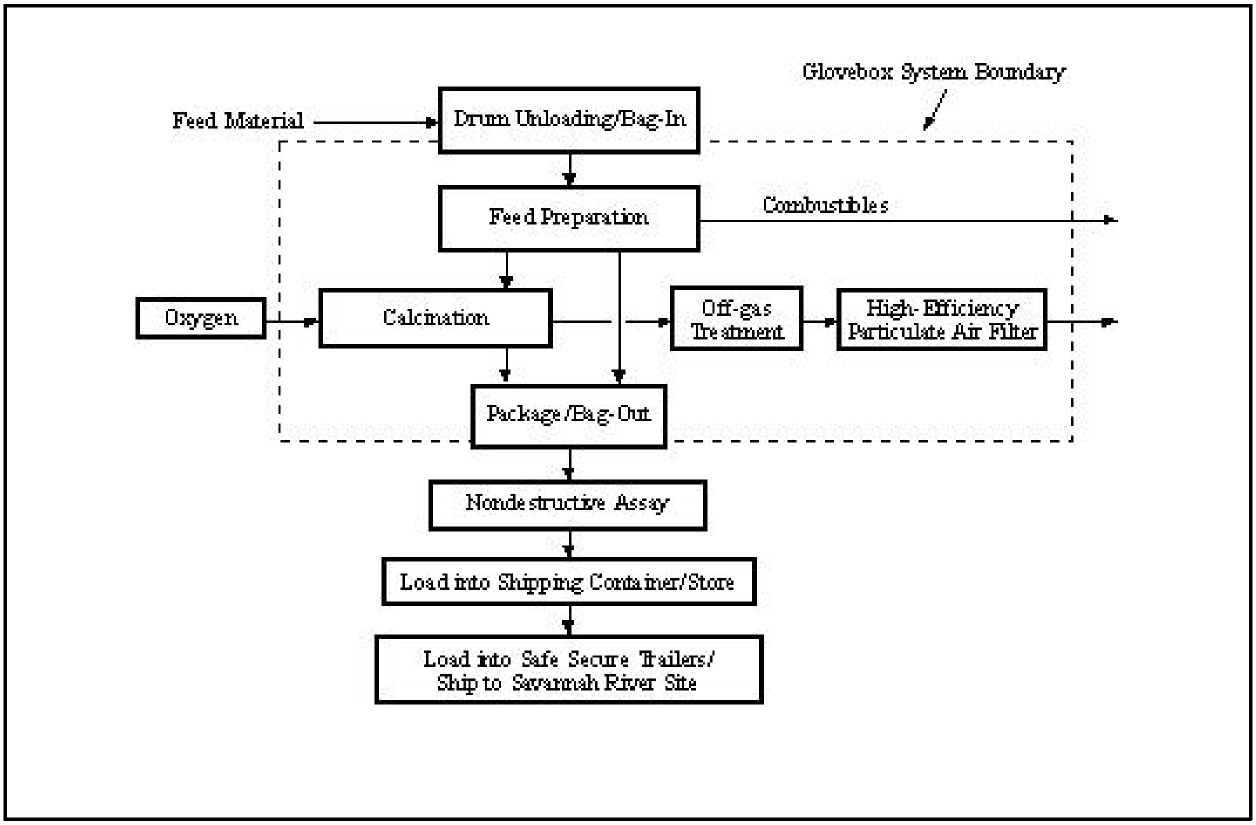

Figure C–22 Packaging Process at Rocky Flats for Shipment to Savannah River Site

Nondestructive assay would be performed using a segmented gamma scanner to confirm the amount of plutonium being shipped to the Savannah River Site. After nondestructive assay, the packages would be loaded into Type 9975 shipping containers and transferred to the shipping facility. There would be two cans placed into each shipping container. The shipping containers would be cleaned and surveyed for contamination before transfer to interim storage within the Building 371 or to the shipping facility. The loaded Type 9975 containers would be picked up at the process building and transferred to the shipping facility, where they would be loaded into a safe secure trailer or other DOE-approved transport, asappropriate and transported to the Savannah River Site. The distance from Rocky Flats to the Savannah River is approximately 2,620 km (1,625 mi).

❒ Mediated Electrochemical Oxidation at the Savannah River Site

The mediated electrochemical oxidation process at the Savannah River Site following preprocessing at Rocky Flats is shown in **Figure C–23**. The plutonium within the preprocessed residues would be leached and/or dissolved in the New Special Recovery or HB-Line facility using two newly installed dissolvers that use the silver II ion to dissolve the normally intractable plutonium in the residue. Once the plutonium is in solution, the residue would be separated/filtered out and discarded as transuranic waste.

The plutonium would be converted to metal or oxide which would be prepackaged into cans and placed in the FB- or HB-Line. It would be transferred to the Savannah River Actinide Packaging and Storage Facility or another Savannah River vault where packaging would be completed (outer container to meet DOE-STD-3013-96 [DOE 1996c]) and it would be stored until decisions are made on fissile material disposition. The Plutonium Storage Facility and New Special Recovery facility are not currently in operation and would require two silver dissolvers to be installed and the facilities started up; the HB-Line is operating and would require modification to existing, or installation of new silver dissolvers. All other facilities are currently in operation and no changes to the process would be required.

Detailed Process Description

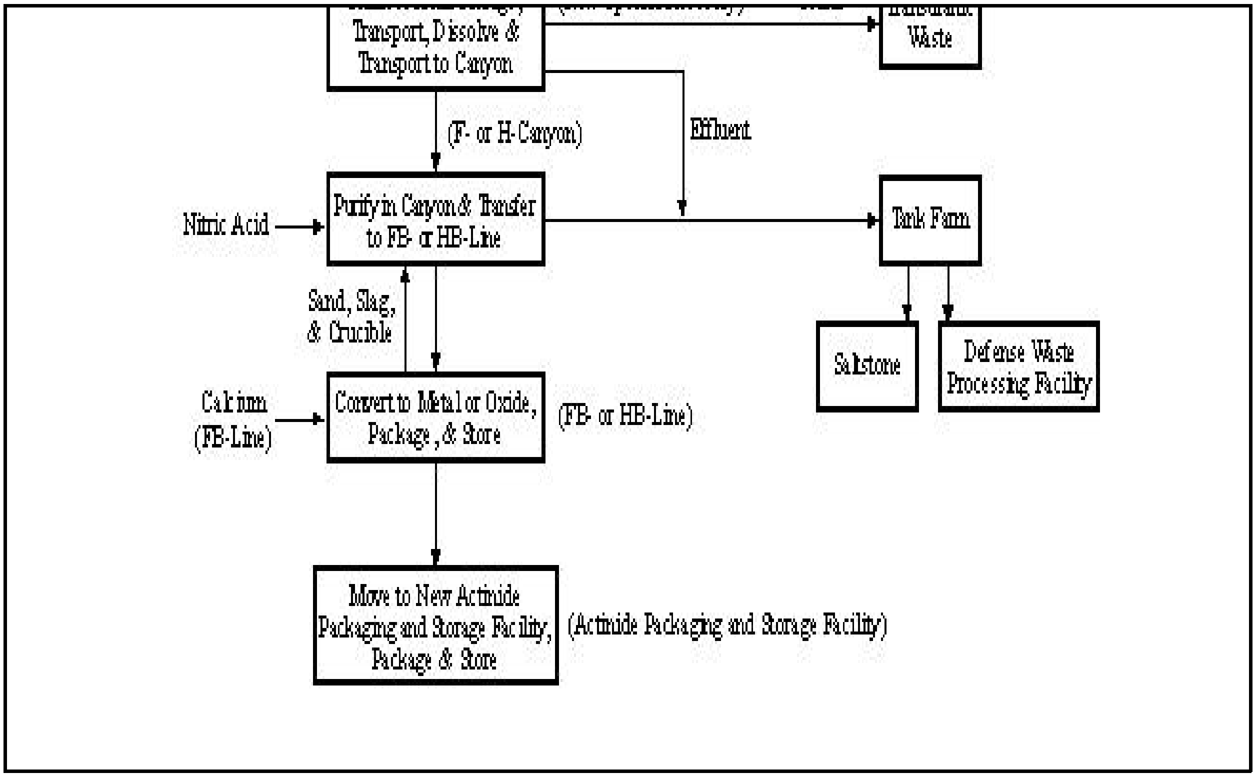

Figure C–23 Mediated Electrochemical Oxidation Process at Savannah River

The shipping containers received from Rocky Flats would be unloaded, confirmatory measurements made, and the containers placed in the Plutonium Storage Facility or other Savannah River Site vault (such as 235-F). One batch of shipping containers would be removed from storage and moved via a conveyor to New Special Recovery or transferred to the HB-Line where each shipping container would be opened up and the two cans removed. The cans would be opened and the contents processed through a leach/dissolve (or wash/filter/dissolve) and filter cycle in a silver dissolver. Any nondissolved material would be discarded as transuranic waste. The silver would be continually reused and the filtered plutonium solution would be transferred to F- or H-Canyon.

Waste liquid containing impurities and residual plutonium would be transferred from the Savannah River Site canyon to the Savannah River Site high-level waste system. The residual plutonium would be vitrified as borosilicate glass in the Savannah River Site Defense Waste Processing Facility. Savannah River Site high-level waste glass is scheduled for disposal in a deep monitored geological repository beginning in 2015. Decontaminated aqueous solutions containing the residue and associated spent processing reagents (the bulk of the secondary waste) would be transferred to the Savannah River Site Z-Area Saltstone Treatment and Disposal Facility. The resultant nonhazardous stabilized waste form would be disposed of in engineered vaults at the Savannah River Site low-level radioactive industrial landfill.

Within the canyon facilities, nitric acid would be added to the primary plutonium solution to dissolve the remaining solids (for incinerator ash and fines) and purified (for all proposed residues) by removing the impurities in an aqueous stream. The plutonium product solution would be transferred to F- or H-Canyon hold tanks for later transfer to FB- or HB-Line.

The FB-Line process would include concentration of plutonium by cation exchange, precipitation of plutonium as a trifluoride, recovery of the trifluoride by filtration, drying of trifluoride in an oxygen atmosphere, and reduction with calcium metal to form plutonium metal buttons. The sand, slag, and crucible generated from button reduction would be dissolved in F-Canyon. The HB-Line process would include concentration of plutonium through anion exchange, precipitation of plutonium as plutonium oxalate, recovery of the oxalate by filtration, drying and calcining the oxalate, converting it to plutonium oxide. The metal buttons and oxide would be prepackaged into cans which would be placed in an F area vault for temporary storage. The cans would then be removed from the F area vault, placed into shipping containers, and transported to the Actinide Packaging and Storage Facility. At the Actinide Packaging and Storage Facility, the cans would be removed from the shipping containers, packaged into an outer 3013 container, and placed into the vault for long-term storage pending disposition in accordance with decisions reached under the *Surplus Plutonium Disposition Environmental Impact Statement.* Any plutonium separated would be disposed of using an immobilization process.

C.6.3 Salt Distillation

Salt distillation technology would require pyro-oxidation of the sodium/potassium chloride pyrochemical salts to convert reactive metals to oxides prior to salt distillation. This technology could be used on electrorefining salts and molten salt extraction salts. Salt distillation has been proposed at Rocky Flats and at Los Alamos National Laboratory with preprocessing at Rocky Flats. Though similar, enough details differ to warrant two discussions.

Salt distillation, consisting of the separation of the higher-vapor pressure alkali halide salts from the transuranic oxides, is considered to be a technology which has been well demonstrated on a pilot scale with actual residue materials, although optimization studies are ongoing and final designs of the production equipment would be required. Operations of the salt distillation process may not be able to start until 2 years after the issuance of the Record of Decision. The capability for salt distillation at Los Alamos National Laboratory is already installed and operational at Los Alamos National Laboratory on a pilot scale. Additional capabilities could be installed if necessary, however this capability would not be available for between 2-4 years after issuance of the Record of Decision.

An additional uncertainty involved in the salt distillation process is the disposition of the resultant transuranic oxide materials resulting from the processing of the molten salt extraction salts. These materials contain elevated concentrations of americium by comparison to other plutonium oxide materials, resulting in elevated gamma radiation levels which would have to be addressed in handling. Estimates of radiation levels from these oxides packaged in normal containers which meet DOE-STD-3013-96 indicate that the materials may not be suitable for storage at the new vault being constructed at the Savannah River Site, although special shielding approaches are being evaluated. In the event that shielding is an unacceptable alternative, these materials may have to be processed in another manner or stored separately prior to final disposition.

C.6.3.1 Salt Distillation at Rocky Flats

The vacuum distillation process would reduce the plutonium concentration below the safeguards termination limit for pyrochemical salts. The resulting products would be a lean transuranic salt waste to be shipped to WIPP and plutonium oxide to be stored at Rocky Flats. Vacuum distillation has not been shown to be effective on calcium chloride (direct oxide reduction) salts. The entire distillation process would be conducted inside gloveboxes located in Module A and Module B of Building 707 or in Building 371.

The salt distillation process for pyrochemical salts is shown in **Figure C–24**. Electrorefining and molten salt extraction salts would be sorted and batched in preparation for pyro-oxidation. After pyro-oxidation, the salts would be vacuum distilled to separate the plutonium oxide from the salts. The packaged salts would be removed from the glovebox and nondestructively assayed for accountability purposes. The salts would be packaged in the final transport/storage container, and moved into interim storage, pending disposal at WIPP. The plutonium oxide would be transferred to the bagless transfer system for final calcination, removed from the glovebox line, nondestructively assayed for accountability purposes, and then transferred to plutonium storage.

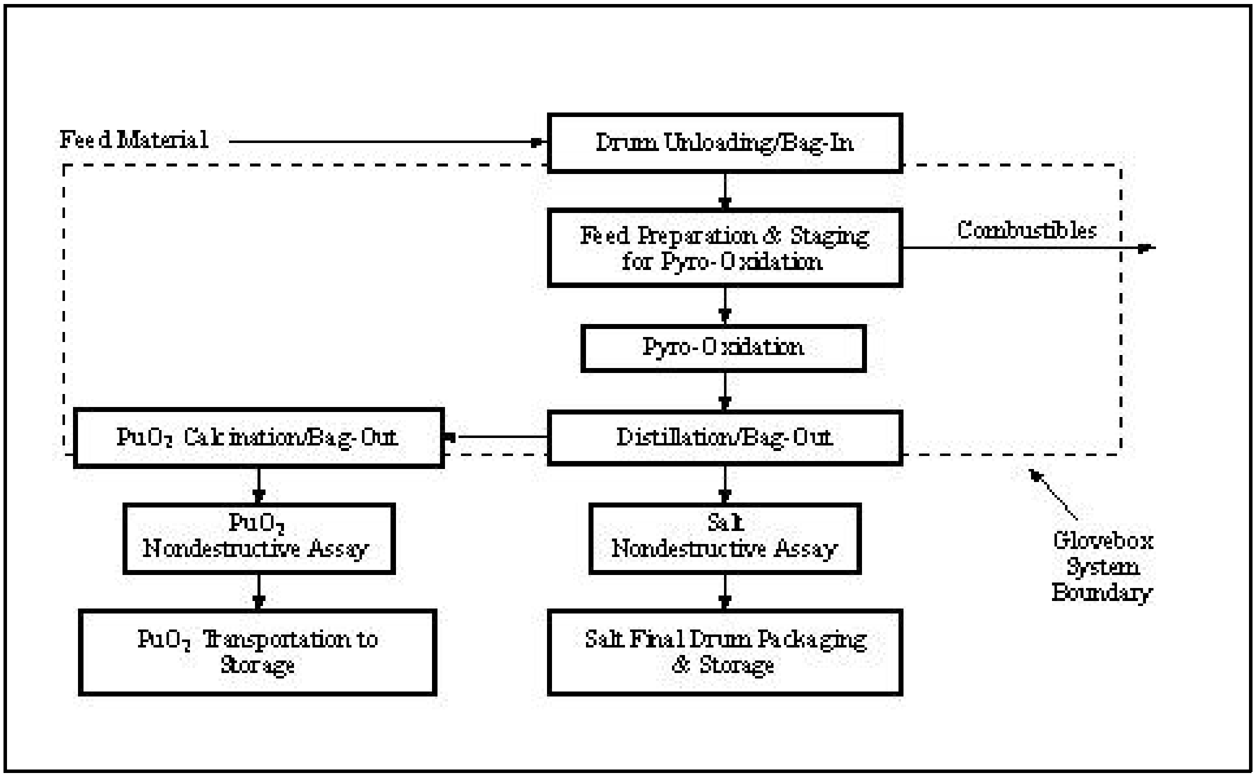

Figure C–24 Salt Distillation Process at Rocky Flats

❐ Detailed Process Description

Drums would be manually transferred into a contamination control enclosure and unpacked. This step is to contain any contamination which could result from an individual package containment damaged by radiolysis, or physical damage to the package during storage. Any unnecessary packaging materials would be removed to limit the amount of packaging introduced into the feed preparation glovebox. All of these individual containers would be bagged into the feed preparation glovebox.

The feed materials would be introduced into the glovebox, one stream at a time, and the IDC verified. The individual packages would be opened and loaded into a crucible in preparation for pyro-oxidation. Sodium carbonate or another oxidant would also be added to the crucible at this time. Combustible packaging materials from the individual packages would be bagged out of the glovebox and managed as transuranic waste. Other materials would be bagged out and managed appropriately.

Once the crucible is loaded with salt feed, it would be placed in a glovebox furnace and heated to approximately 800°C (1,470°F) with sodium carbonate or another oxidant as a reagent for 2 to 3 hours (8-hour cycle time), stirring continuously. The product would be a stabilized plutonium salt matrix. This process would convert reactive metals (such as sodium, calcium, or potassium) to oxides. When the furnace has cooled to below 100°C (212°F), the crucible would be removed from the furnace. During the heating, stirring, and cooling phases, argon would flow through the furnace. During the last part of the heating phase, argon would be replaced by a mixture of air and argon. Once the crucible has been removed from the furnace, it would be allowed to completely cool. The material would be placed into containers and transferred to the distillation glovebox via the chain conveyer system.

Each batch of pyro-oxidized salts would be placed into a vacuum distillation unit and distilled under vacuum for several hours (12-hour cycle time). The distillation process would remove the salts in gaseous form. The salt gases would be condensed to form a lean transuranic salt waste, leaving behind plutonium oxide. At this point, the salts are assumed to contain only parts per million amounts of plutonium. The salts would be batched into containers, bagged out of the glovebox, and packaged for nondestructive assay. The plutonium oxide would be placed into interim storage or directly transferred to the calcination glovebox for the required final plutonium oxide calcination, if necessary.

Nondestructive assay would be performed to ensure requirement limits are met and to obtain data to ensure that required accountability procedures are followed. Nondestructive assay methods would be selected to ensure that the best accountability data is obtained. Assayed product packages would be selected for final packaging to minimize the number of shipping containers required to be shipped to WIPP. Selected packages would be loaded into an inner container and sealed prior to placing the container into the final outer shipping container.

Plutonium oxide from the distillation process step would be removed from the cans and placed in furnaces, and calcined at 1,000°C (1,800°F) for 4 hours, if necessary to meet 3013 criteria. The material, now suitable for storage or transportation, would be weighed, characterized, and placed in a container. This container would then be removed from the glovebox using the bagless transfer process and sent to nondestructive assay. The plutonium oxide would be assayed for plutonium content based on its rate of thermal generation using calorimeters and Gamma-Ray Isotopic Spectrometer equipment. After assay, the containers would be ready for storage. Assayed product packages containing the plutonium oxide would be transported to appropriate plutonium storage areas pending disposition in accordance with decisions reached under the *Surplus Plutonium Disposition Environmental Impact Statement.* Any plutonium separated would be disposed of using an immobilization process.

C.6.3.2 Salt Distillation at Los Alamos National Laboratory with Preprocessing at Rocky Flats

- **Preprocessing at Rocky Flats**

A pyro-oxidation process, if necessary, would be conducted inside gloveboxes located in Module A of Building 707 or in Building 371 at Rocky Flats and the resulting products would be shipped to Los Alamos National Laboratory by safe secure trailer or other DOE-approved transport, as appropriate for final processing. The pyro-oxidation preprocessing process for pyrochemical salts and subsequent shipment to Los Alamos National Laboratory are shown in **Figure C–25**. The residue materials would be sorted and batched in preparation for pyro-oxidation. The salts would be pyro-oxidized to convert reactive metals to oxides. After pyro-oxidation, the oxidized plutonium salts would be packaged for storage and shipment. The packaged material would be removed from the glovebox, nondestructively assayed for accountability purposes, and then packaged in the final transport/storage container and stored, if necessary, until it could be shipped to Los Alamos National Laboratory.

Detailed Preprocessing Description

Drums would be manually transferred into a contamination control enclosure and unpacked. This is to contain any contamination which could result from an individual package containment damaged by radiolysis, or physical damage to the package during storage. Any unnecessary packaging materials would be removed to limit the amount of packaging introduced into the feed preparation glovebox. All of these individual containers would be bagged into the feed preparation glovebox.

The salt feed would be introduced into the glovebox, one stream at a time, and the IDC verified. The individual packages would be opened and loaded into a crucible in preparation for pyro-oxidation. Sodium carbonate or another oxidant would also be added to the crucible at this time. Combustible packaging materials from the individual packages would be bagged out of the glovebox and managed as transuranic waste. Other materials would be bagged out and managed appropriately.

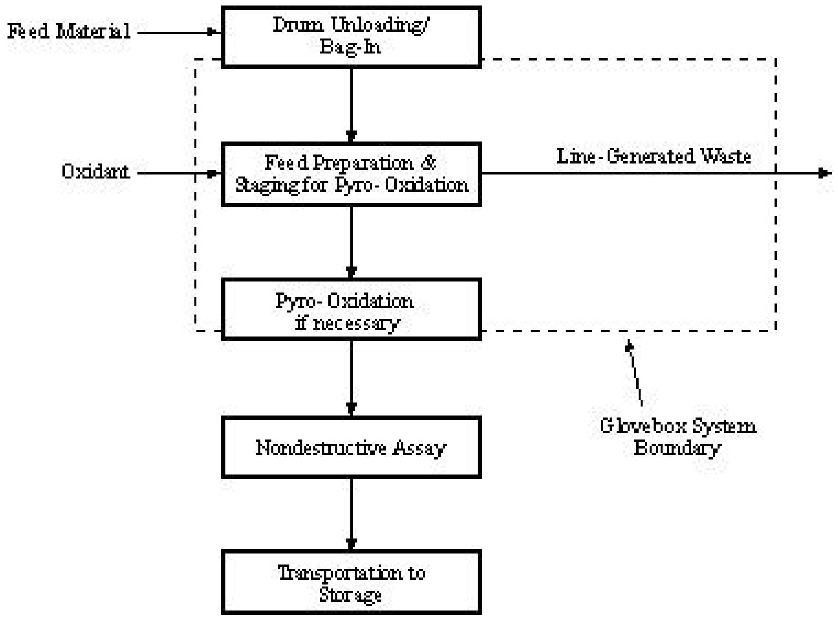

Figure C–25 Pyro-Oxidation Preprocessing Process

Once the crucible is loaded with salt feed, it would be placed in a glovebox furnace and heated to approximately 800°C (1,470°F) with sodium carbonate or another oxidant as a reagent for 2 to 3 hours (8-hour cycle time), stirring continuously. The product would be a stabilized plutonium salt matrix. This process would convert reactive metals, such as calcium, sodium, and potassium, to oxides. When the furnace has cooled to below 100°C (212°F), the crucible would be removed from the furnace. During the heating, stirring, and cooling phases, argon would flow through the furnace. During the last part of the heating phase, argon would be replaced by a mixture of air and argon.

Once the crucible is removed from the furnace, it would be allowed to completely cool before breakout. The salt matrix would then be removed from the crucible and the crucible discarded. The salt matrix would be placed into stainless steel containers in nominal 3.5-kg (7.7-lb) bulk (net) batches. The containers would be crimp sealed, weighed, and bagged out of the glovebox line. After bagout, two smaller cans would be placed in one tall can and sealed and nondestructive assay would be performed.

Calorimetry may be used for determining a heat signature for a shipping package. Packages would be loaded into 9975 Type shipping containers and transferred to interim vault storage or the shipping dock. There would be one can in each shipping container. The shipping containers would be cleaned and surveyed for contamination before transfer to either interim vault storage or the shipping dock. All transfers within the process building would be made by forklift. The loaded 9975 Type containers would be picked up at the process building or interim vault storage and transferred to the shipping facility, where they would be loaded into a safe secure trailer or other DOE-approved transport, as appropriate by forklift and transported to Los Alamos National Laboratory. The distance from Rocky Flats to Los Alamos is approximately 730 km (450 mi).

❐ Salt Distillation at Los Alamos National Laboratory

The salt distillation process at Los Alamos National Laboratory would separate sodium chloride/potassium chloride pyrochemical salt residues into a slightly contaminated (<100 parts per million [ppm] plutonium) salt fraction, suitable for disposal as transuranic waste, and a chloride-free, plutonium oxide powder suitable for long-term storage. The separation is based on the large difference in vapor pressure between alkali metal chlorides and actinide oxides at elevated temperatures. Calcium chloride has a much lower vapor pressure than the alkali metal chlorides and cannot be processed by present distillation equipment. The distillation of the plutonium salts would be carried out at Technical Area 55, Building PF-4, Room 420. The salt distillation process is shown in **Figure C–26**.

Detailed Process Description

The pretreated feed residues would be directly loaded into the Los Alamos National Laboratory salt distillation apparatus. The distillation unit has been designed to handle 3-kg (6.6-lb) salt batches. The distillation unit would be sealed and a vacuum applied. The furnaces on the evaporator side would be heated to 950°C (1,740°F) so that the salt would begin to evaporate and solidify on cool condenser surfaces. The condensing salt would raise the temperature of the condenser. Once all the salt has distilled from the evaporator side, the temperature of the condenser would begin to fall, signaling completion of the evaporation step. Typically, this occurs at 4 to 5 hours after heat is first applied. At this point, the unit would be backfilled with argon to atmospheric pressure and the condenser to 850°C (1,560°F) to melt the salt into a receiving mold. This would provide a convenient salt monolith for disposal and would typically require one hour. All power would be shut off and the unit allowed to cool to room temperature overnight.

If shorter times are required, active cooling could be used to speed the cycle up. It is unlikely a full cycle could be completed in an 8-hour shift; twelve hours is usually required for a complete load/unload cycle.

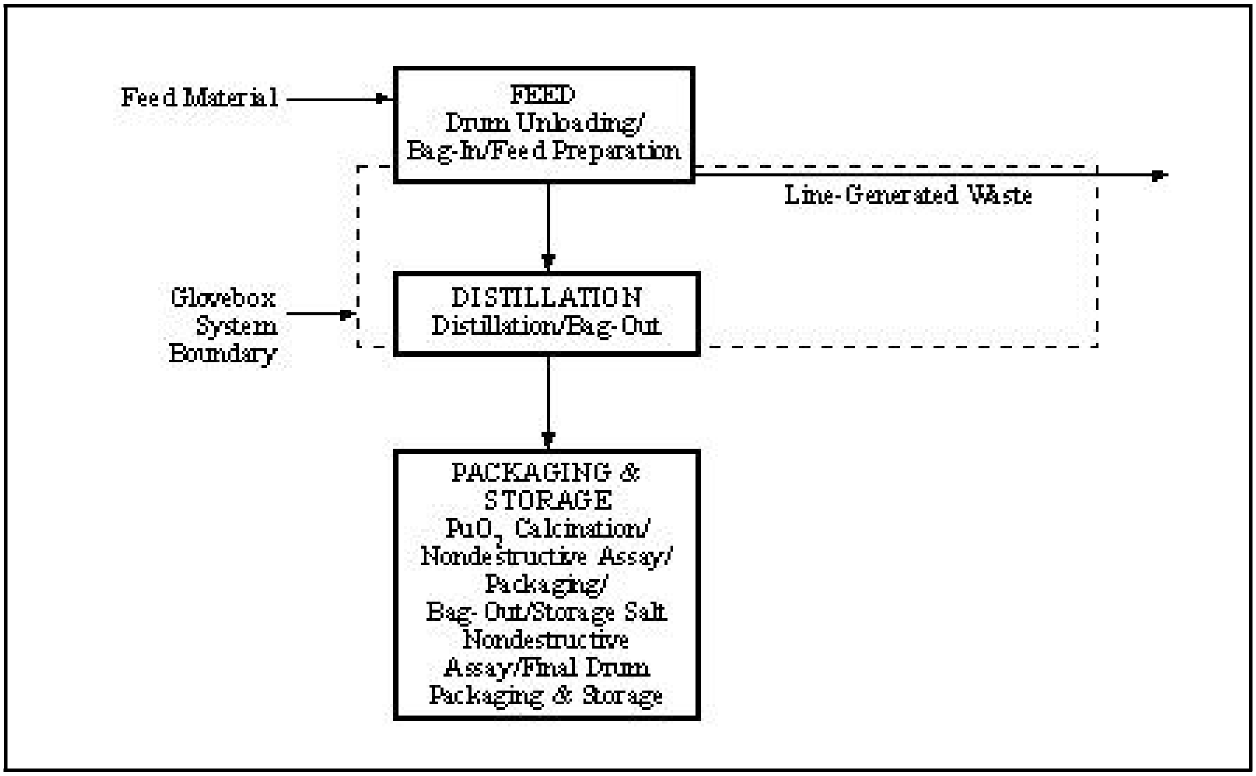

Figure C–26 Salt Distillation Process at Los Alamos

The waste salt would be packaged and removed from the glovebox line for nondestructive assay by neutron counting, and then loaded into a 208-L (55-gal) drum destined for WIPP. Because of the small amount of plutonium in the salt (<100 ppm), the drum could be filled to a volume capacity without exceeding the radionuclide or safeguards termination limit on waste for WIPP.

The oxide distillation heel would meet the criteria of DOE-STD-3013-96 (DOE 1996c) without further processing. However, several distillation runs would be required before the 4-kg (8.8-lb) batch size for packaging is accumulated. The oxide would begin to absorb atmospheric moisture once removed from the distillation unit. Unless rigorously dry conditions are maintained for in-line oxide storage, calcination would be required before final packaging. The oxide would be stored in TA-55 pending disposition in accordance with decisions reached under the *Surplus Plutonium Disposition Environmental Impact Statement.* Any plutonium separated would be disposed of using an immobilization process.

C.6.4 Water Leach with Plutonium Oxide Recovery

The water leach technology for pyrochemical salts would include the pyro-oxidation of calcium chloride pyrochemical salts to oxidize any reactive metals, followed by selective aqueous dissolution of the soluble portion of the salt. The insoluble plutonium-containing material would remain undissolved, and would be filtered and calcined to plutonium oxide for storage. The filtrate would be evaporated to dryness. Water leach has been proposed at Rocky Flats and at Los Alamos National Laboratory with preprocessing at Rocky Flats. Though the proposals are similar, enough details differ to warrant two discussions.

The water leach process at Rocky Flats is considered to be a proven technology. The process would be required to be installed in areas of Building 371 adjacent to the neutralize-dry process to take advantage of the liquid treatment facilities. The requirements for using this area for other residue activities (e.g., neutralize-dry, cementation) would impact the installation, testing, and operational schedule of new process equipment. Therefore, operations of the water leach process may not be able to start until a minimum of 4 years after issuance of the Record of Decision. The capability for water leach at Los Alamos National Laboratory is already installed and operational on a limited scale. Additional capabilities are available using a similar aqueous dissolution process. If any additional capabilities were necessary, they could be installed, however this capability would not be available for between 1–2 years after issuance of the Record of Decision.

Although principally considered for direct oxide reduction salts, if the water leach process were to be used to process molten salt extraction salts (or those calcium chloride salts used for a molten salt extraction-type process) there is an additional uncertainty involving the disposal of the resulting transuranic oxide materials. These materials contain elevated concentrations of americium by comparison to other plutonium oxide materials, resulting in elevated gamma radiation levels which must be addressed in handling. Estimates of radiation levels from these oxides packaged in normal containers which meet DOE-STD-3013-96 indicate that the materials may not be suitable for storage at the new vault being constructed at the Savannah River Site, although special shielding approaches are being evaluated. In the event that shielding is an unacceptable alternative, these materials may have to be processed in another manner or stored separately prior to final disposition. Although these materials have been identified as being difficult to handle due to their higher than normal radiation levels, they are only one of a number of similar materials which must be accommodated for storage at the Savannah River Site.

C.6.4.1 Water Leach with Plutonium Oxide Recovery at Rocky Flats

The entire water leach process would be conducted inside gloveboxes located in Rooms 3305 and 3701 of Building 371, except for the final calcination step, which would be done in Module J of Building 707. The resulting products would be a lean transuranic salt waste to be shipped to WIPP and plutonium oxide to be stored at Rocky Flats.

The water leach process at Rocky Flats is shown in **Figure C–27**. The feed materials would be sorted and batched in preparation for pyro-oxidation. The salts would be pyro-oxidized to convert reactive metals to oxides. After pyro-oxidation, the salts would go through aqueous dissolution to dissolve the salts and soluble oxides. The solution would be filtered to separate the plutonium and americium oxides from the salt solution. The plutonium and americium oxides would be dried, nondestructively assayed for accountability purposes, calcined, and sent to storage pending DOE decisions on eventual disposition of the plutonium. The salt solution would be evaporated and the resulting salts would be dried, cast, packaged, and nondestructively assayed for accountability purposes. The salts would be packaged in the final transport/storage container, and moved into interim storage, pending disposal at WIPP.

During the heating, stirring, and cooling phases, argon would flow through the furnace. During the last part of the heating phase, argon would be replaced by a mixture of air and argon. Once the crucible is removed from the furnace, it would be allowed to completely cool before breakout. The salt matrix would then be removed from the crucible and crushed to be more amenable to dissolution; the crucible would be discarded. The salt matrix would be packaged suitable for dissolution, bagged out, and transferred to Building 371 for the water dissolution step.

After bag-in, the salt would be treated using the water dissolution process on a batch basis. Water dissolution would consist of placing the pyro-oxidized salts into a vessel approximately 15 L (4.0 -gal) in volume, adding approximately two parts slightly acidified (1.7N HCl) water to one part total residue, and stirring for approximately 4 hours until the salts have dissolved. Approximately 90 percent of the water needed would be recycled from the filtrate evaporation step. Thus, a small quantity of 12.4N HCl and makeup water would be added to the recycle water in order to achieve the desired normality. After the salts have dissolved, the resulting solution would be treated in the filtration step.

❒ Detailed Process Description

Drums would be manually transferred from storage into a contamination control enclosure. The contamination control enclosure is designed to control airflow in the event of a bag failure within a drum. The drum would be opened and the integrity of the packaging would be checked. If the packaging has not been compromised, the containers would be transferred to the glovebox. Any unnecessary packaging materials would be removed to limit the amount of packaging introduced into the feed preparation glovebox. If the integrity of the packaging has been compromised, the packaging would be overpacked with a new plastic bag prior to transfer to the glovebox. All of these individual containers would be bagged into the salt feed preparation glovebox. The salt materials would be introduced into the glovebox and the IDC verified. The individual packages would each be opened, batched to a 200 g (7 oz) maximum of plutonium, and then loaded into a magnesium oxide crucible in preparation for pyro-oxidation. An oxidant, such as sodium carbonate, would also be added to the crucible at this time. Combustible packaging materials from the individual packages would be bagged out of the glovebox and sent to a combustible-handling process. Other materials would be bagged out and managed appropriately.

Once the crucible has been loaded with salt feed, it would be placed in a glovebox furnace and heated to approximately 800°C (1,470°F) with an oxidant such as sodium carbonate as a reagent for 2 to 3 hours, stirring continuously. The product would be a stabilized plutonium salt matrix. This process would convert reactive metals (such as calcium) to oxides. When the furnace has cooled to below 100°C (212°F), the crucible would be removed from the furnace.

After bag-in, the salt would be treated in the water dissolution process on a batch basis. Water dissolution consists of placing the pyro-oxidized salts into a vessel approximately 15 L in volume, adding approximately two parts slightly acidized water to one part total residue, and stirring for approximately 4 hours until the salts have dissolved. Approximately 90 percent of the water needed would be recycled

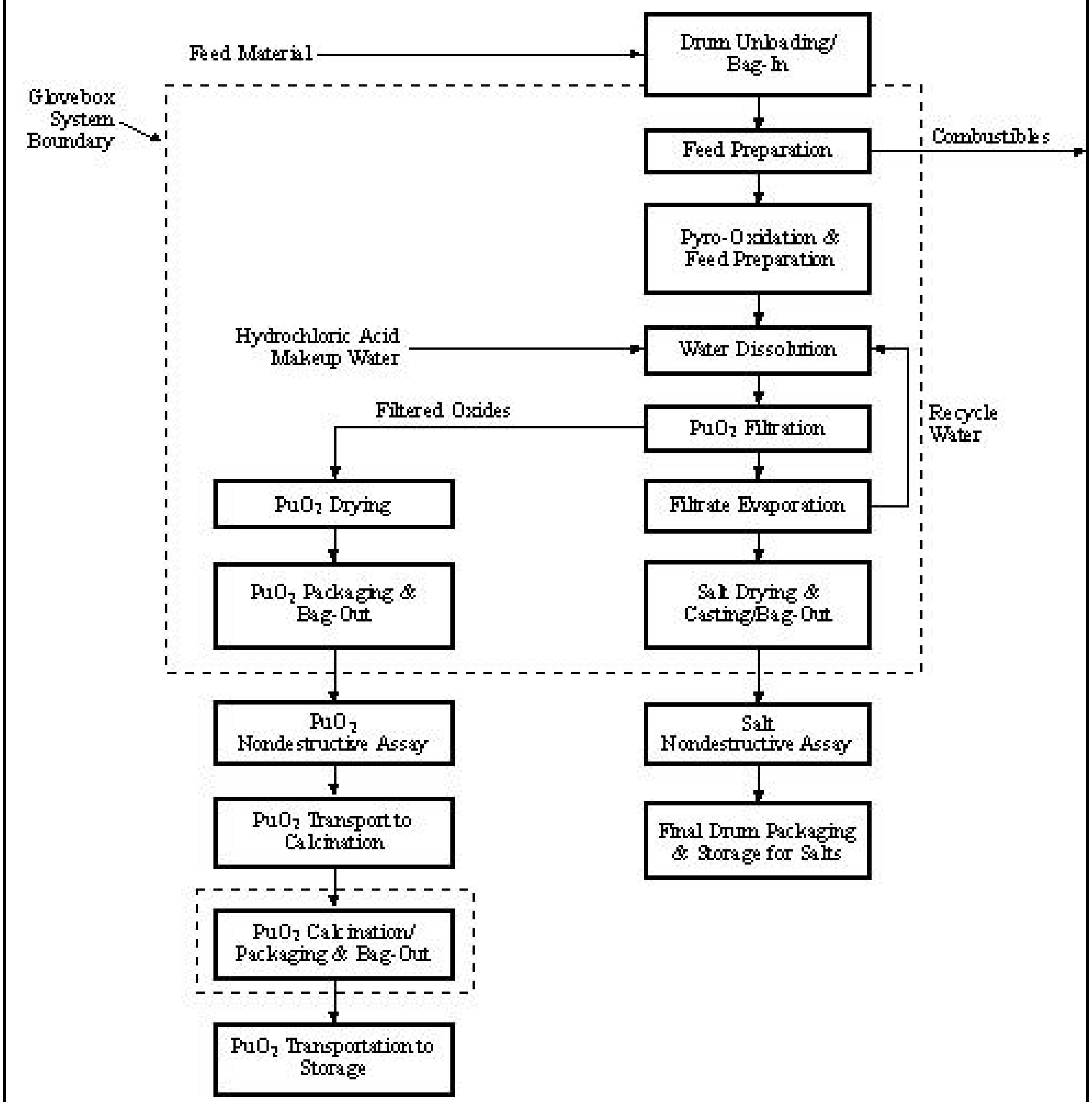

Figure C–27 Water Leach Process at Rocky Flats

from the filtrate evaporation step. Thus, a small quantity of high normality hydrochloric acid and makeup water would be added to the recycle water in order to achieve the desired normality. After the salts have dissolved, the resulting solution would be treated in the filtration step.

The salt solution from water dissolution would be decanted and the resultant wet solids vacuum filtered to remove the plutonium oxide and americium oxide solids from the salt solution. The filtered oxides would go to plutonium oxide drying and the lean salt solution would go to evaporation. The wet plutonium and americium oxides would contain about 20 percent water after filtration. They would be placed into a small furnace and dried for about 4 hours at about 400°C (750°F). The resultant material is assumed to be only plutonium and americium oxides.

After drying, the oxides would be batched to 1 kg (2.2 lb) of plutonium batches. The batches would be placed into slip-lid cans and bagged out into convenience cans for nondestructive assay. The plutonium and americium oxides would be assayed for plutonium content based on their rates of thermal generation using calorimeters and gamma-ray isotopic spectrometer equipment. After assay, the containers would be ready for final calcination. The oxides would be transferred by truck from Building 371 to Building 707 to be calcined in Module J.

Plutonium oxide from the nondestructive assay step would be removed from the cans, batched into 3-kg (6.6-lb) batches, placed into furnaces, and calcined at 1,000°C (1,800°F) for 4 hours. The material, now suitable for storage or transportation, would be weighed, characterized, and placed in a 3013 container. This container would then be bagged out and sent to storage. Product packages containing the calcined plutonium oxide would be transported to appropriate plutonium storage areas pending disposition in accordance with decisions reached under the *Surplus Plutonium Disposition Environmental Impact Statement.* Any plutonium separated would be disposed of using an immobilization process.

Two batches of filtered salt solution would be combined, placed into an evaporator unit and evaporated to a damp solid. The process would evaporate the water and cause the nonvolatile salts to remain in the product solids. The distillate water would be condensed and recycled back to the dissolution step, and the salts would be sent to the drying ovens. The batch of damp solids from evaporation would be placed into drying ovens and dried for about 4 hours at about 200°C (390°F) .

After drying, the salts would be placed into salt casting furnaces, heated to approximately 500°C (930°F) (melted) for about 1 hour, and cooled into solid form. The salts would be removed from the furnaces after cooling and packaged for bag-out. After bag-out, the salts would be sent to nondestructive assay.

Nondestructive assay would be performed to ensure requirement limits are met and to obtain data to ensure that required accountability procedures are followed. Nondestructive assay methods would be selected to ensure that the best accountability data is obtained. Assayed product packages would be selected for final packaging to minimize the number of shipping containers required to be shipped to WIPP. Selected packages would be loaded into an inner container and sealed prior to placing of the container into the final outer shipping container.

C.6.4.2 Water Leach with Plutonium Oxide Recovery at Los Alamos National Laboratory with Preprocessing at Rocky Flats

❒ Preprocessing at Rocky Flats

The stabilization of salts with the water leach process at Los Alamos National Laboratory would require preprocessing at Rocky Flats which would include pyro-oxidation, if necessary, to convert reactive metals to oxides. The pyro-oxidation process would be conducted inside gloveboxes located in Module A of Building 707. The feed materials would be sorted and batched in preparation for pyro-oxidation. The salts would be pyro-oxidized to convert reactive metals to oxides. After pyro-oxidation, the oxidized plutonium salts would be packaged for storage and shipment. The packaged material would be removed from the glovebox, nondestructively assayed for accountability purposes. It would then be packaged in the final transport/storage container and stored, if necessary, until it could be shipped by safe secure trailer or other DOE-approved transport, as appropriate to Los Alamos National Laboratory. The pyro-oxidation preprocessing process is shown in Figure C–25.

Detailed Process Description

Drums would be manually transferred into a contamination control enclosure and unpacked. This step is to contain any contamination which could result from any individual package containment which was damaged by radiolysis or physical damage. Any unnecessary packaging materials would be removed during this step to limit the amount of packaging introduced into the feed preparation glovebox. All of these individual containers would be bagged into the feed preparation glovebox.

The salt feed would be introduced into the glovebox, the IDC verified, and the individual packages would be opened and loaded into a magnesium oxide crucible in preparation for pyro-oxidation. An oxidant such as sodium carbonate would also be added to the crucible at this time. Combustible packaging materials from the individual packages would be bagged out of the glovebox and sent to a combustibles handling process. Other materials would be bagged out and managed appropriately.

Once the crucible is loaded with salt feed, it would be placed in a glovebox furnace and heated to approximately 800°C (1,470°F) with an oxidant such as sodium carbonate as a reagent for 2 to 3 hours (8-hour cycle time), stirring continuously. The product would be a stabilized plutonium salt matrix. This process would convert reactive metals (such as, sodium, calcium, or potassium) to oxides. When the furnace has cooled to below 100°C (212°F), the crucible would be removed from the furnace. During the heating, stirring, and cooling phases, argon would flow through the furnace. During the last part of the heating phase, argon would be replaced by a mixture of air and argon. Once the crucible is removed from the furnace, it would be allowed to completely cool before breakout. The salt matrix would then be removed from the crucible and the crucible discarded. The salt matrix would be placed into stainless steel containers in nominal 3.5-kg (7.7-lb) bulk (net) batches. The salt matrix batches would be weighed and placed in steel containers. The containers would be crimp sealed, weighed, and bagged out of the glovebox line. After bag-out, two smaller cans would be placed in one tall can and sealed, and nondestructive assay would be performed. Calorimetry may be used for determining a heat signature for a shipping package.

Packages would be loaded into 9975 Type shipping containers with one tall can per shipping container, the shipping containers would be cleaned and surveyed for contamination, and then they would be transferred to interim vault storage or the shipping dock. The loaded 9975 Type containers would be picked up at the process building or interim vault storage and transferred to the shipping facility, where they would be loaded into a safe secure trailer or other DOE-approved transport, as appropriate. The pretreated salt shipments would be transported to Los Alamos National Laboratory via safe secure trailer or other DOE-approved transport, as appropriate. The distance from Rocky Flats to Los Alamos National Laboratory is approximately 730 km (450 mi.).

❐ Water Leach with Plutonium Oxide Recovery at Los Alamos National Laboratory

Pyrochemical salts would be received from Rocky Flats for final processing. These salts would be dissolved in a water leaching process. The resulting products would be lean calcium chloride salt for shipment to WIPP, and plutonium oxide to be stored at Los Alamos National Laboratory in TA-55. The water leach of the plutonium salts would be carried out at Technical Area 55, Building PF-4, Room 420. The water leach process is shown schematically in **Figure C–28**. The process steps would include shipment by safe, secure trailer or otheer DOE-approved transport, as appropriate, unloading and receiving at Los Alamos National Laboratory, shipping container unloading and nondestructive assay, aqueous leaching and filtration, calcination of plutonium oxide, casting of calcium chloride salt, and bag-out. Nondestructive assay for plutonium oxide and salts would be performed, followed by final drum packaging and storage for salts, and transfer of the plutonium oxide to storage.

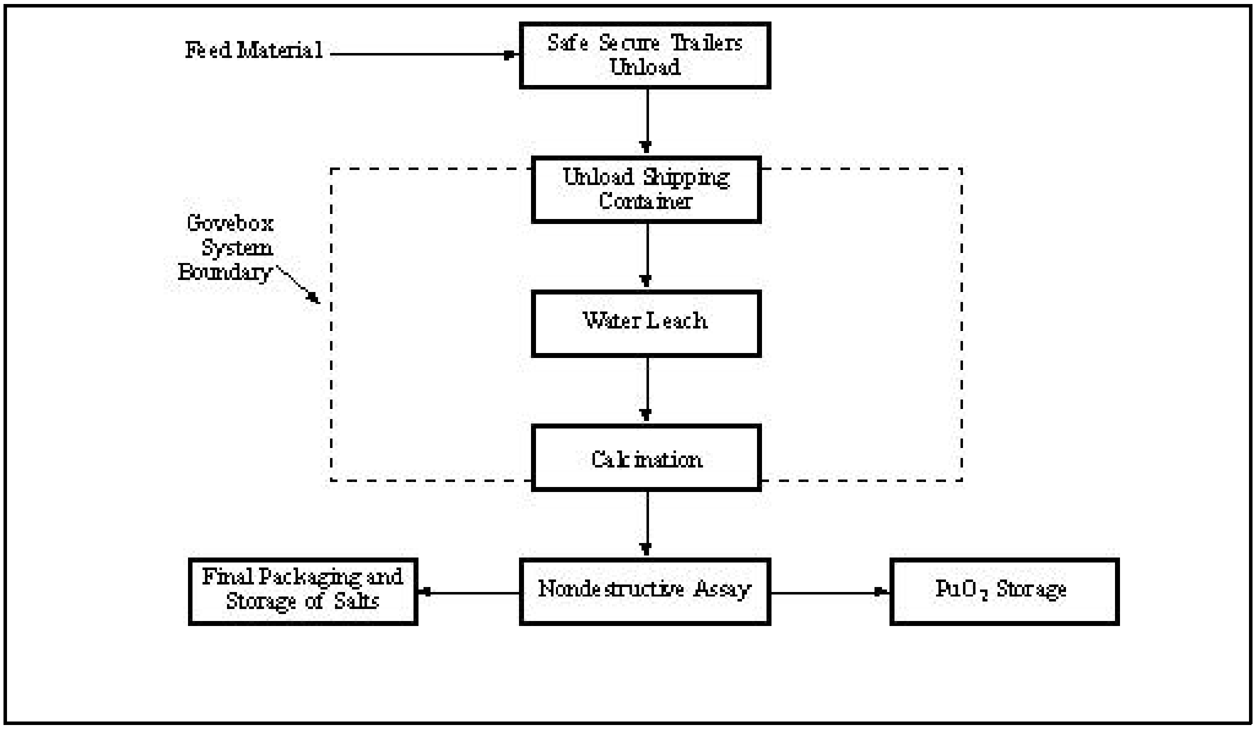

Figure C–28 Water Leach Process at Los Alamos

Detailed Process Description

The leach equipment would be sized to process a complete salt batch at one time. The salt batch would be placed in a leaching vessel sized to dissolve all the calcium chloride salt in the residue. During the leaching operation, the solution would become slightly alkaline from dissolution of excess sodium carbonate and the slight solubility of calcium oxide. Aqueous hydrochloric acid would be added to convert calcium oxide and sodium carbonate into the respective chlorides. This would be necessary to maintain the plutonium concentration in the filter cake above 50 percent. The pH of the solution would be monitored and would always be maintained above 7. The plutonium and americium oxides would remain insoluble. A 3-kg

(6.6-lb) batch of salt can be dissolved in 3 hours. During this time, an operator would monitor the operation for parameters such as temperature, mixing rate, leaching time, and dissolver condition. Once leaching has been completed, the slurry containing the plutonium and americium insolubles would be ready for filtration.

The slurry would be transferred to the filtration system, where the solids containing plutonium and americium oxides would be removed. The solids would be washed with water to remove salt contaminants. The clarified solution would be collected in a geometrically favorable tank and sampled for alkalinity and plutonium/americium concentration. The volume would be estimated at 6 L (1.6 -gal) per 3-kg (6.6-lb) batch. The solution would then be evaporated to dryness and the solid salt transferred to a furnace and heated to 850°C (1,560°F) for melt consolidation. The final plutonium concentration in the salt would be expected to be about 100 ppm. The insolubles collected on the filter would be removed and transferred to the calcination workstation. The wet cake from filtration would be placed in a crucible and calcined at 950°C (1,740°F) to remove water and other volatiles. The calcined product would contain less than 50 percent plutonium and would be stabilized. The stabilized product would be weighed and sampled for Pu and Am analysis, Loss on Ignition and transferred to a packaging workstation. The calcined product would be packaged in accordance with DOE-STD-3013-96 (DOE 1996c) and stored in TA-55 pending disposition in accordance with decisions reached under the *Surplus Plutonium Disposition Environmental Impact Statement*. Any plutonium separated would be disposed of using an immobilization process. Where batches contain small quantities of plutonium, multiple batches may be combined for storage after calcination.

C.6.5 Salt Scrub with Purex Processing of Newly Created Scrub Alloy

❒ Preprocessing at Rocky Flats

The salt scrub process for pyrochemical salts would reduce the plutonium level of the salts below the safeguards termination limit for pyrochemical salts and produces a high plutonium yield scrub alloy that would be shipped to Savannah River Site for further processing. The resulting low plutonium-bearing pyrochemical salts would be a lean transuranic waste to be shipped to WIPP. The salt scrub process can be used on electrorefining salts, molten salt extraction salts, and direct oxidation reduction salts. The salt scrub process would be conducted inside gloveboxes located in Modules A and B of Building 707.

The salt scrub process, consisting of the reduction and capture of plutonium and americium from chloride salts into a metal “button” in a pyrochemical process at Rocky Flats, and the subsequent shipment of the button to Savannah River for processing in the Purex process is considered to be a proven process for clean, recently-packaged salt residues. Technical uncertainties exist for this process as applied to less pure salts and/or salts which have absorbed moisture during storage. Development work would be required prior to or in parallel to the operations to address these uncertainties, with the result possibly being a population of salts not amenable to this technique. Since the scrub alloy process could be performed in the stationary furnaces that have been installed at Rocky Flats as part of the No Action Alternative, currently-installed capability exists to support the this process, although the scrub alloy processing would have to be coordinated with the current pyro-oxidation commitments. The salt scrubbed by this process may not all meet the safeguards termination limits and could need some subsequent processing prior to disposition.

The salt scrub process for pyrochemical salts and subsequent shipment of resultant scrub alloy to Savannah River Site are shown in **Figure C–29**. Because of differences in salt composition, each of the salt types would be processed separately; however, the process steps are the same for each. The feed materials would be sorted and batched in preparation for salt scrub. The salts would be scrubbed to remove as much plutonium as possible. This description assumes all salts would be scrubbed as a bounding condition;

certain lots of material may be unsuitable for this process due to age, condition, or low plutonium content, and would require alternative processing.

After salt scrub, the salts would be re-batched for pyro-oxidation. The salts would be pyro-oxidized to convert any reactive metals to oxides. After pyro-oxidation, the salts would be removed from the glovebox and nondestructively assayed for accountability purposes, packaged in the final transport/storage container, and placed in interim storage. The scrub alloy would also be removed from the glovebox, nondestructively assayed for accountability purposes, and packaged in the final transport/storage container and stored, if necessary, until it can be shipped to Savannah River Site by safe secure trailer or other DOE-approved transport, as appropriate.

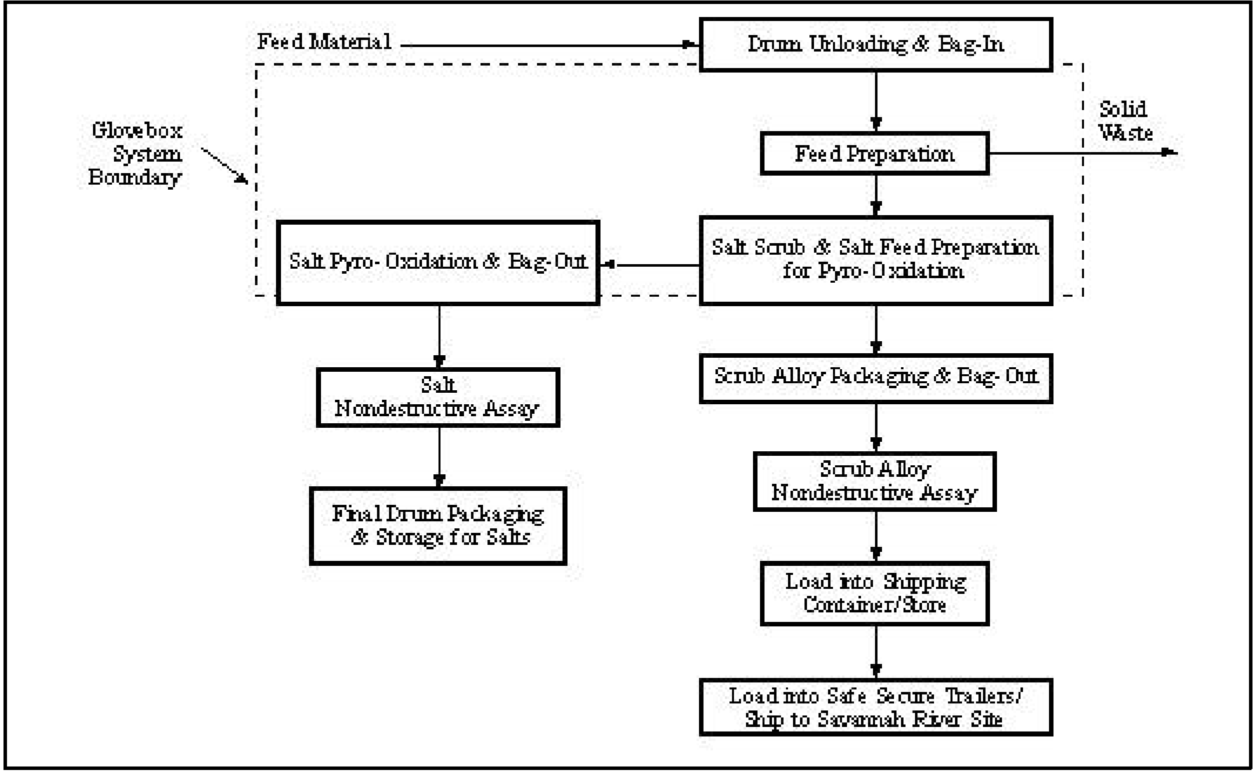

Figure C–29 Salt Scrub Process and Subsequent Shipment of Resultant Scrub Alloy to Savannah River Site

Detailed Process Description

As required, drums would be manually transferred into a contamination control enclosure and unpacked. This step would be to contain any contamination on the outside of the inner package which could result from radiolysis or physical damage to the package during storage. Any unnecessary packaging materials would be removed to limit the amount of packaging introduced into the feed preparation glovebox. All of these individual containers, after examination and/or repackaging in the contamination control enclosure, would be bagged into the feed preparation glovebox.

The feed materials would be introduced into the glovebox, one stream at a time, and the IDC verified. The salt would be removed from the original container, weighed, and batched with the appropriate amount of metal reductants and matrix (e.g., gallium and calcium metal). The quantities of gallium and calcium used would be dependent on the plutonium and americium content. The maximum batch size will be 2.5 kg (5.5 lb) of bulk residue, which produces approximately 200 g (7 oz) plutonium metal. Combustible packaging materials from the individual packages would be bagged out of the glovebox and sent to a combustibles handling process. Other materials would be bagged out and managed appropriately.

Although, in the salt scrub process, an aluminum/magnesium alloy has been used in the past and may be used in specific cases, the newer gallium/calcium alloy system would lower the overall radiation levels, since alpha-neutron reactions would be minimized. The charge, containing the salt/metal mixture, would be placed into a furnace and heated at 800°C (1,470°F) for 2 hours (8-hour cycle time). During heating, the molten salt/metal mixture would be mechanically stirred. The furnace would then be allowed to cool, and the crucible would be removed from the furnace and allowed to completely cool before breakout. The scrub alloy button and the plutonium-depleted salts would be removed from the crucible and the crucible would be discarded. The salts would be either stored in-line or batched into magnesium oxide or other crucibles and sent to pyro-oxidation by chain conveyer.

Once the crucible has been loaded with salt feed, it would be placed in a glovebox furnace and heated to approximately 800°C (1,470°F) with sodium carbonate as a reagent for 2 to 3 hours, stirring continuously (8-hour cycle time). The product would be a stabilized plutonium salt matrix. Pyro-oxidation could be applied to both sodium chloride-potassium chloride and calcium chloride matrices. This process would convert reactive metals (i.e., calcium, sodium, and potassium) to oxides. When the furnace has cooled to below 100°C (212°F), the crucible would be removed from the furnace. During the heating, stirring, and cooling phases, argon would flow through the furnace. During the last part of the heating phase, argon would be replaced by a mixture of air and argon.

Once the crucible has been removed from the furnace, it would be allowed to completely cool before breakout. The salt matrix would then be removed from the crucible and the crucible would be discarded. The material would be batched to 9.1 kg (20.0 lb) of total residue (based on an estimated maximum weight to be handled in a glovebox), placed into a container, bagged from the glovebox, and placed in a convenience container for safe handling. If metal crucibles are used, the pyro-oxidized salt would remain in the crucibles and be sealed and bagged out directly in nominal 2.5-kg (5.5-lb) bulk (net) batches.

Nondestructive assay would be performed to ensure requirement limits are met and to obtain data to ensure that required accountability procedures are followed. Nondestructive assay methods would be selected to ensure that the best accountability data are obtained. Assayed product packages would be selected for final packaging to minimize the number of shipping containers required to be shipped to WIPP. Selected packages would be loaded into an inner container and sealed before placing of the container into the final outer shipping container.

The scrub alloy buttons would be weighed and placed in a dissolvable (mild steel) container. The containers would be crimp-sealed, weighed, and bagged out of the glovebox line, using special dissolvable, nylon bags. After bag-out, two smaller cans would be placed in one tall dissolvable can and sealed. Although aluminum containers have been used in the past, mild steel cans would be used on all future shipments.

Nondestructive assay would be performed and calorimetry may be used for determining a heat signature for a shipping package. Packages would be loaded into Type 6M shipping containers and transferred to interim vault storage or the shipping dock. There would be one can in each shipping container. The shipping containers would be cleaned and surveyed for contamination before transfer to either interim vault storage or the shipping dock. The loaded Type 6M containers would be picked up at the process building or interim vault storage and transferred to the shipping facility, where they would be loaded into a safe secure trailer by forklift. Safe, secure trailer transported shipments to Savannah River Site would be required for the newly-created scrub alloy. The distance from Rocky Flats to Savannah River Site is approximately 2,620 km (1,625 mi.).

❐ **Purex Processing at Savannah River Site of Newly Created Scrub Alloy**

The scrub alloy would be dissolved in the Savannah River Site F- or H-Canyon. The plutonium would be separated from americium and aluminum using the solvent extraction technology. The plutonium would be converted to metal or oxide prepackaged into cans that are placed in the FB- or HB-Line. That metal or oxide would be transferred to Savannah River Site's FB-Line or 235-F vault until the Actinide Packaging and Storage Facility vault is complete, packaging completed (outer container) to meet DOE-STD-3013-96 (DOE 1996c) and stored until decisions are made on fissile material disposition. This process is currently in operation and no changes to the process are required due to salt scrub alloy. The salt scrub alloy Purex processing to metal or oxide at Savannah River Site is shown in **Figure C–30**.

Detailed Process Description

The shipping containers received from Rocky Flats would be unloaded, confirmatory measurements made and placed in a vault-like room in 235-F. Twelve shipping containers at a time would be removed from storage and transported to the F- or H-Canyon crane maintenance area where the shipping containers would be opened up and the cans loaded into a dissolver tube. The dissolver tube would then be loaded into a dissolver by remote control. Twelve cans make up one dissolving batch.

Heated nitric acid in the tank dissolves the salt scrub alloy, resulting in a solution containing americium, chloride, aluminum, magnesium, and plutonium. The plutonium would be recovered and purified by solvent extraction; the impurities remain in the aqueous stream. The waste liquid containing americium, aluminum, and residual plutonium would be transferred from the Savannah River Site canyon facility to the Savannah River Site high-level waste system. The plutonium product solution would be transferred to canyon hold tanks for later transfer to FB- or HB-Line.

The FB-Line process would include concentration of plutonium by cation exchange, precipitation of plutonium as a trifluoride, recovery of the trifluoride by filtration, drying of trifluoride in an oxygen atmosphere, and reduction with calcium metal to form plutonium metal buttons. The sand, slag, and crucible generated from button reduction would be dissolved in F-Canyon. The HB-Line process would include concentration of plutonium through anion exchange, precipitation of plutonium as plutonium oxalate, recovery of the oxalate by filtration, drying and calcining the oxalate, converting it to plutonium oxide. The buttons and oxide would be prepackaged into cans which would be placed in an F area vault for temporary storage. The cans would then be removed from the F area vault, placed into shipping containers, and transported to the Actinide Packaging and Storage Facility. At the Actinide Packaging and Storage Facility, the cans would be removed from the shipping containers, packaged into an outer 3013 container, and placed into the vault for long-term storage pending disposition in accordance with decisions reached under the *Surplus Plutonium Disposition Environmental Impact Statement*. Any plutonium separated would be disposed of using an immobilization process.

C.6.6 Acid Dissolution with Plutonium Oxide Recovery of Fluoride and Sludge Residues

The acid dissolution of either fluoride or sludge residues would involve dissolution of the residues, followed by precipitation and filtration of plutonium oxalate, and calcination to plutonium oxide for storage pending a final disposition decision. The filtrate from the oxalate precipitation would be treated with magnesium hydroxide to precipitate the plutonium remaining in the solution. That precipitate would then be filtered, calcined, repackaged, and placed in interim site storage before being shipped to WIPP. The dissolution process would be conducted inside gloveboxes located in Room 3701 of Building 371.

The acid dissolution/plutonium oxide recovery process consisting of dissolving the plutonium contained in fluorides or sludges, filtering the solution, and precipitating and calcining a plutonium oxalate, is considered to be a proven technology. The process to be used for the limited quantities of materials identified in these categories would be consistent with equipment and activities that can be performed in the neutralize-dry process area. Thus, the capability for Rocky Flats is currently being installed to support the disposition of below-safeguards termination limit materials, and should be available several months after the issuance of the EIS. However, the use of this equipment for acid dissolution would generally be preceded by the neutralize-dry processing of the combustible residues required by the Defense Nuclear Facilities Safety Board Recommendation 94-1 stabilization program, and may not be able to start until 4 years after issuance of the Record of Decision.

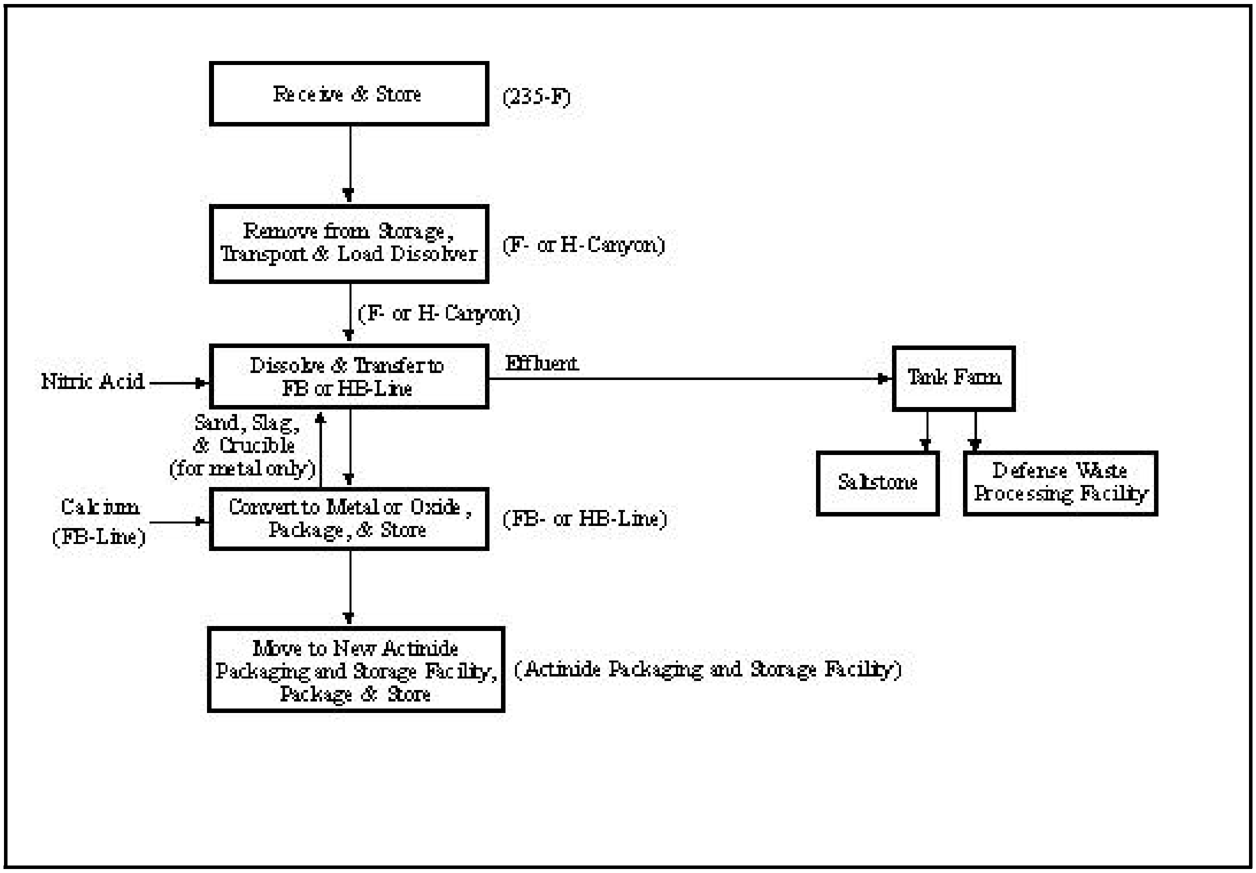

Figure C–30 Salt Scrub Alloy Purex Process at Savannah River Site

The plutonium residue acid dissolution process is shown in **Figure C–31**. The feed materials would be unpacked and batched for acid dissolution. The dissolved residues would be sent through precipitation to form plutonium oxalate precipitate in slurry form, which would then be filtered to separate the effluent solution from the precipitate. The oxalate would be calcined, nondestructively assayed, calcined again for long-term storage, again nondestructively assayed, and then packaged for storage. Magnesium hydroxide would be mixed into the oxalate precipitation effluent to precipitate the remaining plutonium, and the effluent filtered to form magnesium hydroxide and effluent. The magnesium hydroxide would be calcined and packaged. The packaged magnesium hydroxide product would be removed from the glovebox and nondestructively assayed for accountability purposes, packaged in the final transport/storage container, and placed in interim storage. The last filtration effluent would be sent for evaporation at the Rocky Flats wastewater treatment facility.

❒ Detailed Process Description

The residue feed would be introduced into the glovebox, and the IDC would be verified. The materials would then be removed from the containers and batched to a maximum of 200 g (7 oz) of plutonium in preparation for nitric acid dissolution. Combustible packaging materials from the individual containers would be bagged out of the glovebox and sent to a combustible handling process. Other unwanted materials would be bagged out of the glovebox and managed appropriately.

The contents of the residue cans would be transferred to one of two heated stirrers. The operator would add 7N nitric acid (HNO_3) and 60 percent aluminum nitrate ($Al(NO_3)_3$) solution to each dissolver before stirring. $Al(NO_3)_3$ would be added to complex residue ions during dissolution. The slurry would be heated

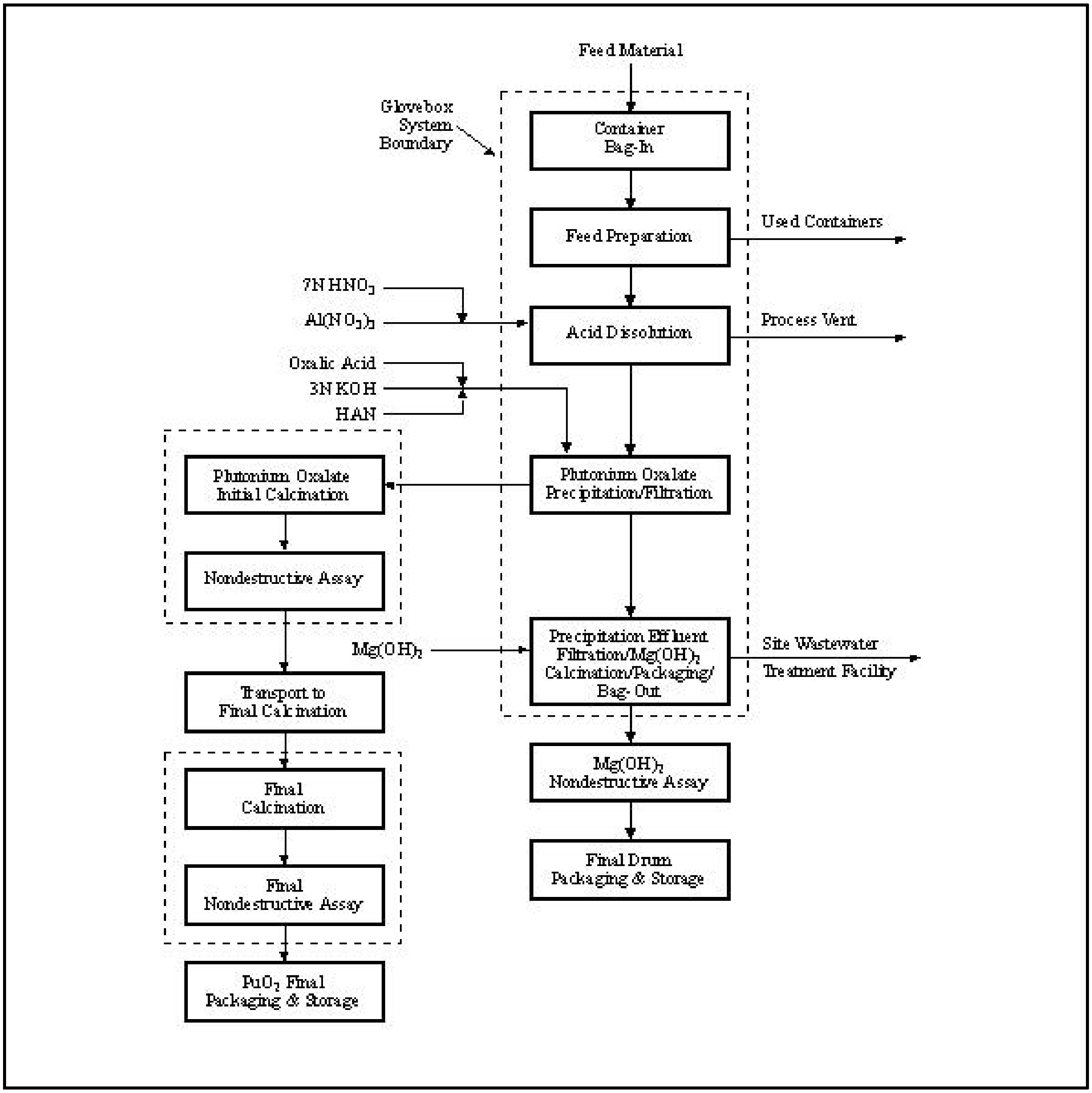

Figure C–31 Acid Dissolution Process for Fluoride and Sludge Residues

to approximately 80°C (176°F) and stirred until dissolution is achieved. Vented fumes would be cooledin a condenser, and then piped to the process vent system. The batch would be filtered to remove any undissolved solids and then split into two equal amounts and transferred to the adjacent heated stirrers for precipitation.

For plutonium oxalate precipitation, 3N potassium hydroxide (KOH) would be added to each can to adjust the normality to 0.75N nitric acid. Hydroxylamine nitrate (HAN) would then be added as a 1.9M solution to adjust the plutonium valence to +3. After these adjustments have been made, solid oxalic acid would be added to form plutonium oxalate precipitate. The solution would be heated to approximately 80°C (176°F) and stirred to form a slurry.

The slurry from the two stirrer assemblies would be poured onto an R-4 filter. Filtration of plutonium oxalate would be achieved by pulling a vacuum through the filter and drawing effluent liquids into a filtrate tank. The plutonium oxalate precipitate would be scooped into a filter boat in preparation for calcining.

The plutonium oxalate would require calcining at 450°C (840°F) to convert the oxalate into the oxide form. In this process, the filter boat would be placed on a pneumatic lift, placed into the calcination furnace, and the precipitate would be heated to 450°C (840°F). Glovebox air would be drawn down through the precipitate at a rate of approximately 0.10 m^3 (3.5 ft^3) per minute during the heating cycle. After a cooling cycle, the calcined oxide would be transferred from the filter boat back into a can, batched to 1,000 g (2.2 lb), sealed, and sent to calorimetry.

The plutonium oxide can would be assayed for plutonium content based on its rate of thermal generation using calorimeters and gamma-ray isotopic spectrometer equipment. This activity would be required to maintain accountability within the acid dissolution material balance area. After assay, the containers would be ready for final calcination. The cans containing the plutonium oxide would be placed into appropriate outer containers and transferred to the Building 371 loading dock. The containers would then be transported to the Building 707 loading dock by intra-site truck transportation, and moved to appropriate vault storage pending final calcination.

The plutonium oxide cans would be transferred from the Building 707 storage vault to Module J and bagged into the plutonium stabilization and packaging system. The plutonium oxide would be removed from the cans, placed into furnaces, and calcined at 1,000°C (1,830°F) for 8 hours. The material, now suitable for long-term storage or transportation, would be weighed, characterized, and placed into a 3013 inner container. This container would then be removed from the glovebox by the bagless transfer process and sent to calorimetry. The plutonium oxide package would be assayed for plutonium content based on its rate of thermal generation using calorimeters and gamma-ray isotopic spectrometer equipment. After assay, the containers would be placed into vault storage, pending a final disposition decision. Any plutonium separated would be disposed of using an immobilization process.

Magnesium hydroxide would be added to the effluent liquid in the filtrate tank from the precipitation filtration step, and the tank would be mixed by sparging. The liquid and precipitate would then be drained onto an R-4 filter. Filtration would be achieved by pulling a vacuum through the R-4 filter and drawing effluent liquids into the transfer tank. The magnesium hydroxide precipitate would then be scooped into a filter boat in preparation for calcining. The magnesium hydroxide would be calcined at 450°C (840°F). In this process, the filter boat would be placed on a pneumatic lift, placed into the calcination furnace, and the precipitate would be heated to 450°C (840°F). Glovebox air would be drawn down through the precipitate at a rate of approximately 0.10 m^3 (3.5 ft^3) per minute during the heating cycle. After a cooling cycle, the calcined hydroxide would be transferred from the filter boat back into a can, batched to 9.1 kg (20 lb), sealed, and bagged out into convenience cans.

Nondestructive assay of the magnesium hydroxide would be performed to ensure requirements limit are met and to obtain data to ensure that required accountability procedures are followed. Nondestructive assay methods would be selected to ensure that the best accountability data are obtained. Assayed product packages would be selected for final packaging to minimize the number of shipping containers and placed in interim storage before being shipped to WIPP. Selected packages would be loaded into an inner container and sealed before placing the container into the final outer shipping container.

C.6.7 Acid Dissolution with Plutonium Oxide Recovery at Los Alamos National Laboratory with Preprocessing at Rocky Flats for Direct Oxide Reduction Salts

Acid dissolution processing of direct oxide reduction salts has been proposed at Los Alamos National Laboratory. This technology option requires salt preprocessing at Rocky Flats.

- ❒ **Preprocessing at Rocky Flats**

The preprocessing of these salts at Rocky Flats is the same as given in Section C.6.4.2 for preprocessing prior to water leach.

- ❒ **Acid Dissolution with Plutonium Oxide Recovery at Los Alamos National Laboratory**

The acid dissolution of direct oxide reduction salts at Los Alamos National Laboratory would involve dissolution of the salts, followed by solvent extraction to separate the plutonium from the salts, oxalate precipitation, and calcination to convert the plutonium compound into plutonium oxide, and hydroxide precipitation and calcination to convert the lean residues to filter cake. The hydroxide filtrate would be processed in the Liquid Waste Treatment Facility. The resulting products would be a lean transuranic salt waste to be shipped to WIPP and plutonium oxide to be stored in TA-55 pending a final disposition decision. The entire acid dissolution process would be conducted inside gloveboxes located in the Los Alamos Plutonium Facility in Technical Area 55. This process is considered to be a proven technology.

The acid dissolution process is shown in **Figure C-32**.

The feed materials would be unpacked and sorted in preparation for acid dissolution to dissolve the salts. After acid dissolution, the plutonium-bearing solution would go through solvent extraction, generating plutonium in the four valence state, which would then be converted prior to the precipitation step to a valence of three, making plutonium (III) oxalate. The plutonium (III) oxalate would then be converted to plutonium oxide by calcination. The plutonium oxide would be nondestructively assayed for accountability purposes, re-calcined, and stored pending a final disposition decision. The waste solutions from solvent extraction and oxalate precipitation would go through hydroxide precipitation and calcination. Wastewater from this would then be sent to the Liquid Waste Treatment Facility and the magnesium hydroxide would be packaged and nondestructively assayed for accountability purposes. The magnesium hydroxide would be packaged in the final transport/storage container and moved into interim storage pending disposal at WIPP.

Detailed Process Description

As required, shipping containers would be manually transferred from storage into a materials management room. The materials management room is designed to control airflow in the event of an inner container failure. The shipping container would be opened and the integrity of the packaging would be checked. If the packaging has not been compromised, the inner containers would be transferred to the glovebox. If the integrity of the inner container has been compromised, it would be overpacked with a plastic bag, prior to transfer to the glovebox.

The primary feed for this process would be direct oxide reduction salts. Once the cans have been bagged into the glovebox, the IDCs would be verified. First, the tall cans would be opened, and then the individual small cans, each containing about 2,200 grams of salt, would be opened, and the contents placed into a dissolver.

The salt would be treated in the acid dissolution process on a batch basis. Acid dissolution would consist of mixing equal weights of water and salts, turning on both vacuum and argon sparging for the dissolver, and adding 12M hydrochloric acid in 200-ml increments to the process. Hydrochloric acid would be added in the amount of milliliters equal to four times the weight of the salts. After 30 to 45 minutes, the solution would be filtered and the dissolver would be washed out with an equal amount of water as was added previously. The wash water would be sent to the Liquid Waste Treatment Facility. The plutonium-bearing product solution would be sent to the solvent extraction feed tank.

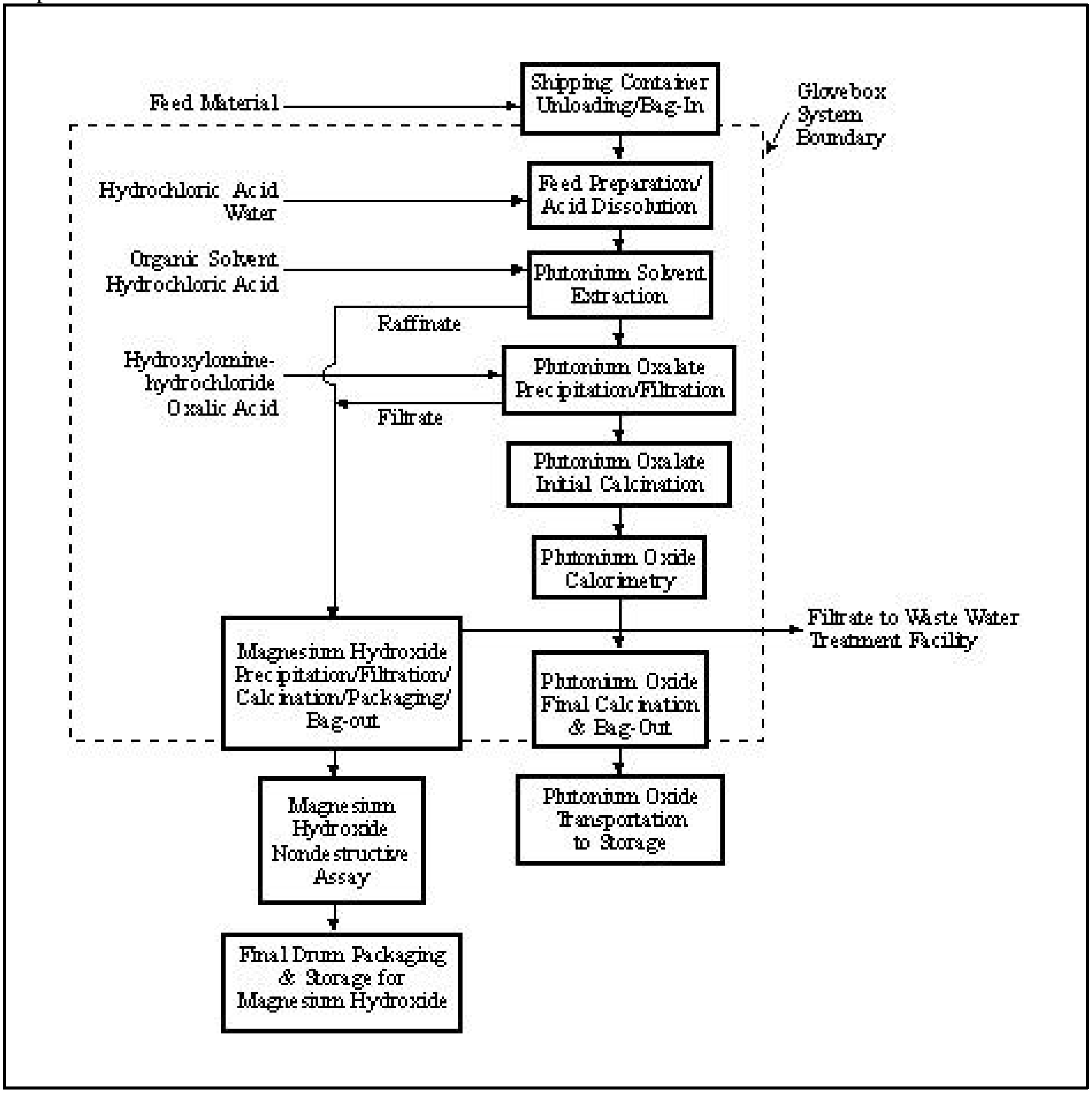

Figure C–32 Acid Dissolution Process for Direct Oxide Reduction Salts at Los Alamos National Laboratory

An oxidizer, in the form of sodium chlorite and water, would be added to the acid dissolution product solution, now called the feed solution, in the four-valence tank, while continuously mixing. Based upon the amount of chloride present, concentrated hydrochloric acid would then be added while continuously mixing the tank, to adjust the molarity to between 6 and 8M. After adjusting the molarity, the organic phase (composed of 70% dodecane and octanol and 30% tributylphosphate) flow would be turned on and then the feed solution from acid dissolution would be fed into the annular centrifugal contactors. The plutonium and americium would move into the organic phase, and the resulting lean acid phase would exit the contactors and be sent to the raffinate catch boat. The strip solution composed of 0.2M hydrochloric acid would then be fed into the contactors, and the plutonium would move from the organic phase into the dilute acid phase. The resulting product solution would exit the contactors and be sent to the product catch boat. Vacuum would be used to pull the raffinate and product solutions to their respective tanks.

The plutonium-rich solution coming from solvent extraction would be transferred to the precipitation feed tanks, where it would be converted to the three-valence state using hydroxylamine-hydrochloride. The batched material would be placed in glass agitated-precipitator columns, and oxalic acid would be added. After mixing for at least 30 minutes, the supernatant would be decanted and filtered into a holding tank. The plutonium oxalate would be drained into a filter boat. The tank would be washed with 0.1M oxalic acid and drained through the filter. The oxalate would dry on the filter and then be transferred to a platinum-lined furnace can. The filtrate would be sampled and sent to hydroxide precipitation.

The furnace can containing the plutonium oxalate would be placed into a calciner and heated to approximately 400°C (750°F) for an hour to decompose the plutonium oxalate to plutonium oxide and carbon dioxide. The plutonium oxide would then be consolidated into slip-lid cans, weighed, and transferred to calorimetry for nondestructive assay. The plutonium oxide would be assayed for plutonium content based on the rate of thermal generation using calorimeters and gamma-ray isotopic spectrometer equipment. After assay, the containers would be ready for final calcination.

Plutonium oxide from the nondestructive assay step would be removed from the cans, batched, placed into furnaces, and calcined at 1,000°C (1,830°F) for 4 hours. The material, now suitable for storage or transportation, would be weighted, characterized, and placed into a 3013 container. The material would be bagged out and stored at TA-55 pending a final disposition decision. Any plutonium separated would be disposed of using an immobilization process.

The raffinate from solvent extraction and the filtrate from oxalate precipitation would be collected in separate holding tanks in preparation for plutonium removal by precipitation. Magnesium hydroxide (30% by weight) and raffinate or filtrate would be mixed in the precipitation tanks by sparging. The liquid and precipitate would then be drained onto an R-4 filter. Filtration would be achieved by pulling a vacuum through the R-4 filter and drawing effluent liquids into the transfer tank. The magnesium hydroxide precipitate would then be scooped into a filter boat in preparation for calcining. The magnesium hydroxide would be calcined at 450°C (840°F). After a cooling cycle, the calcined hydroxide would be transferred from the filter boat back into a can, batched to 9.1 kg (20 lb), sealed, and bagged out into convenience cans and nondestructively assayed. The remaining filtrate would be sent to the Liquid Waste Treatment Facility in Technical Area 55. Assayed product packages would be selected for final packaging to minimize the number of shipping containers required to be shipped to WIPP. Selected packages would be loaded into a pipe component and then the piped component would be loaded into the final outer shipping container. One pipe component would be placed into each 208-L (55-gal) drum for shipment to WIPP.

C.7 DETAILED PROCESS DESCRIPTIONS FOR COMBINATION OF PROCESSING TECHNOLOGIES

The process descriptions for residues that have combined, including blending and repackaging (combination) technologies, are essentially the same as the process descriptions given for No Action and for Processing without Plutonium Separation (Sections C.4 and C.5, respectively), except for the application of a variance to safeguards termination limits. Explanations for each of the combination processing technologies are provided in this section.

The variances to safeguards termination limits include all residues that have a plutonium content of less than or equal to 10 percent. Residues above 10 percent plutonium would be combined with below 10 percent plutonium residues to maintain the 10 percent limit on plutonium content; however, if this type of blending is insufficient to reach 10 percent plutonium, the above 10 percent plutonium residues may be hand-blended with enough virgin material to reach the 10 percent limit on plutonium content.

C.7.1 Combination Process for the Calcination/Cementation of Ash Residues

The residues in the EIS are divided into categories based upon similar applicable process technologies. The residues in the ash category have received a variance to safeguards termination limit, and are addressed in this process description.

The combination process for the calcination/cementation of ash residues calcines, sand, slag and crucible, and inorganic ash residues. Then, along with the graphite fines, all the ash residue would be blended to no more than 10 percent plutonium and cemented, if necessary. The material would then be packaged for interim site storage and ultimate shipment to WIPP. This process would be conducted in Building 707.

The process steps would include drum unloading and bag-in, feed preparation for calcination and calcination, feed preparation (including blending) for cementation, in-line nondestructive assay, mixing of ash with cement, cement curing and bag-out, and final drum packaging and storage.

The description of the combination process for the calcination/cementation of ash residues is similar to the description of the calcination and cementation of ash residues process in the No Action Alternative, except that the graphite fines are not calcined, blending to 10 percent plutonium occurs, and cementation would only be performed if necessary.

C.7.2 Combination Process for the Repackaging of Ash Residues

The residues in the EIS would be divided into categories based upon similar applicable process technologies. The residues in the ash category have received a variance to the safeguards termination limit. In addition, stabilization may not be necessary, which would allow direct repack into pipe components and into drums for shipment to WIPP. The ash residues receiving a variance to the safeguards termination limit and not needing stabilization are addressed in this process description.

Under the combination concept for the repackaging of ash residues, incinerator ash; graphite fines; sand, slag, and crucible; and inorganic ash residues would be blended to no more than 10 percent plutonium and repacked into pipe components for shipment to WIPP. The process would be conducted within a glovebox located in Module A, Building 707.

The combination process for the repackaging of ash residues is shown in **Figure C-33**. The process steps would include drum unloading and bag-in, feed preparation (including blending), repackaging, and bag-out.

The packaged material would be nondestructively assayed for accountability purposes, and packaged into pipe components and then into drums for shipment to WIPP.

❐ Detailed Process Description

Drums with the capacity of 208-L (55-gal) would be manually transferred from storage into a contamination control enclosure and unpacked. The contamination control enclosure is designed to control airflow in the event of a bag failure within a drum. The drums would be opened and the integrity of the packaging would be checked. If the packaging has not been compromised, the containers would be transferred into the glovebox. The containers, including outer packaging materials, would be removed from the drum and bagged into the glovebox. If the integrity of the packaging has been compromised, the package would be overpacked with a new plastic bag prior to transfer to the glovebox.

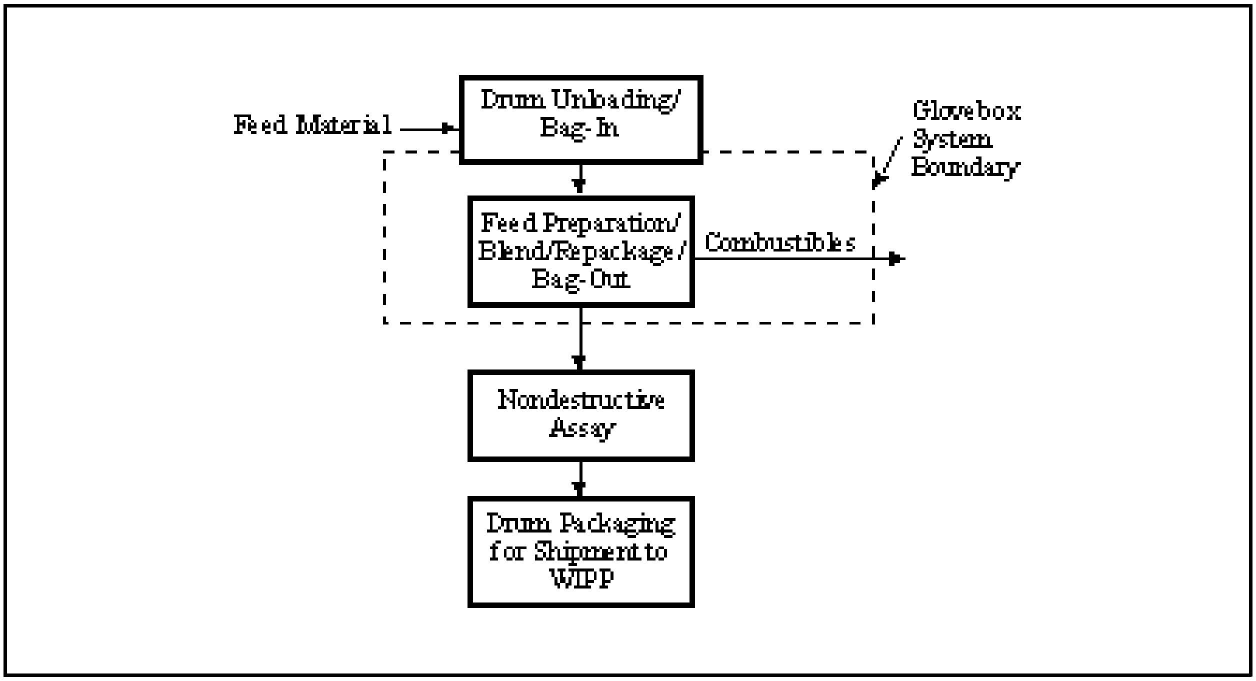

Figure C–33 Combination Process for the Repackaging of Ash Residues

The feed material would be introduced into the glovebox and the IDC verified. The individual packages would be opened, sorted, blended to no more than 10 percent plutonium, and placed into 8.2-L (2.2-gal) containers. The containers would then be bagged from the glovebox. Combustible packaging materials from the individual packages would be bagged out of the glovebox and sent to a combustible handling process. Other materials would be bagged out and managed appropriately.

Nondestructive assay would be performed and the assayed and repackaged residue containers would be loaded into a pipe component, one container each, which would be staged inside of a 208-L (55-gal) drum. These drums would be shipped to WIPP immediately.

C.7.3 Combination Process for the Pyro-Oxidation of Molten Salt Extraction/Electrorefining Salts

The residues in the EIS would be divided into categories based upon similar applicable process technologies. The residues in the molten salt extraction/electrorefining salt category have received a variance to the safeguards termination limit. The molten salt extraction/electrorefining salts that have received this variance are addressed in this process description.

The combination process for the pyro-oxidation of molten salt extraction/electrorefining salts pyro-oxidizes electrorefining salts and molten salt extraction salts to convert reactive metals to oxides. The resulting products would be blended to below 10 percent plutonium and would be packaged for interim site storage and ultimate shipment to WIPP. The pyro-oxidation process would be conducted inside gloveboxes located in Module A, Building 707.

The process steps would include drum unloading and bag-in, feed preparation, blending, and staging for pyro-oxidation, pyro-oxidation and bag-out, nondestructive assay, and drum packaging for interim site storage.

The description of the combination process for the pyro-oxidation of molten salt extraction/electrorefining salts would be similar to the process description for the pyro-oxidation of pyrochemical salts, except that the salts would be loaded into pipe components for ultimate shipment to WIPP. (See Section C.4.2.)

C.7.4 Combination Process for the Repackaging of Molten Salt Extraction/Electrorefining Salts

The residues in the EIS are divided into categories based upon similar applicable process technologies. The residues in the molten salt extraction/electrorefining salt category have received a variance to the safeguards termination limit. In addition, stabilization may not be necessary, thereby allowing direct repack into pipe components and into drums for shipment to WIPP. The molten salt extraction/electrorefining salts receiving a variance to the safeguards termination limit and not needing stabilization are addressed in this process description.

Under the combination concept for the repackaging of molten salt extraction/electrorefining salts, electrorefining salts and molten salt extraction salts would be blended to no more than 10 percent plutonium and repacked into pipe components for shipment to WIPP. The repack process would be conducted inside gloveboxes located in Module A, Building 707.

The combination process for the repackaging of molten salt extraction/electrorefining salts is shown in **Figure C-34**. The process steps would include drum unloading and bag-in, feed preparation (including blending), repackaging, and bag-out. The packaged material would be nondestructively assayed for accountability purposes, and packaged into pipe components and then into drums for shipment to WIPP.

❒ Detailed Process Description

Drums with the capacity of 208-L (55-gal) would be manually transferred from storage into a contamination control enclosure and unpacked. The contamination control enclosure is designed to control airflow in the event of a bag failure within a drum. The drums would be opened and the integrity of the packaging would be checked. If the packaging has not been compromised, the containers would be transferred into the glovebox. The containers, including outer packaging materials, would be removed from the drum and bagged into the glovebox. If the integrity of the packaging has been compromised, the package would be overpacked with a new plastic bag prior to transfer to the glovebox.

The two primary feeds for this process would be: 1) electrorefining salts, and 2) molten salt extraction salts. These materials would be introduced into the glovebox, one stream at a time, and the IDC verified. The individual packages would be opened, sorted, blended to no more than 10 percent plutonium, and placed into 8.2-L (2.2-gal) containers. The containers would then be bagged from the glovebox.

Combustible packaging materials from the individual packages would be bagged out of the glovebox and sent to a combustible handling process. Other materials would be bagged out and managed appropriately.

Nondestructive assay would be performed and the assayed and repackaged salt containers would be loaded into a pipe component, one container each, which would be staged inside of a 208-L (55-gal) drum. These drums would be shipped to WIPP immediately.

C.7.5 Combination Process for the Pyro-Oxidation of Direct Oxide Reduction Salts

The residues in the EIS are divided into categories based upon similar applicable process technologies. The residues in the direct oxide reduction salt category have received a variance to the safeguards termination limit. The direct oxide reduction salts that have received this variance are addressed in this process description.

The combination process for the pyro-oxidation of direct oxide reduction salts pyro-oxidizes direct oxide reduction salts to convert reactive metals to oxides. The resulting products would be blended to below 10 percent plutonium and would be packaged for safe interim site storage and ultimate shipment to WIPP. The pyro-oxidation process would be conducted inside gloveboxes located in Module A, Building 707.

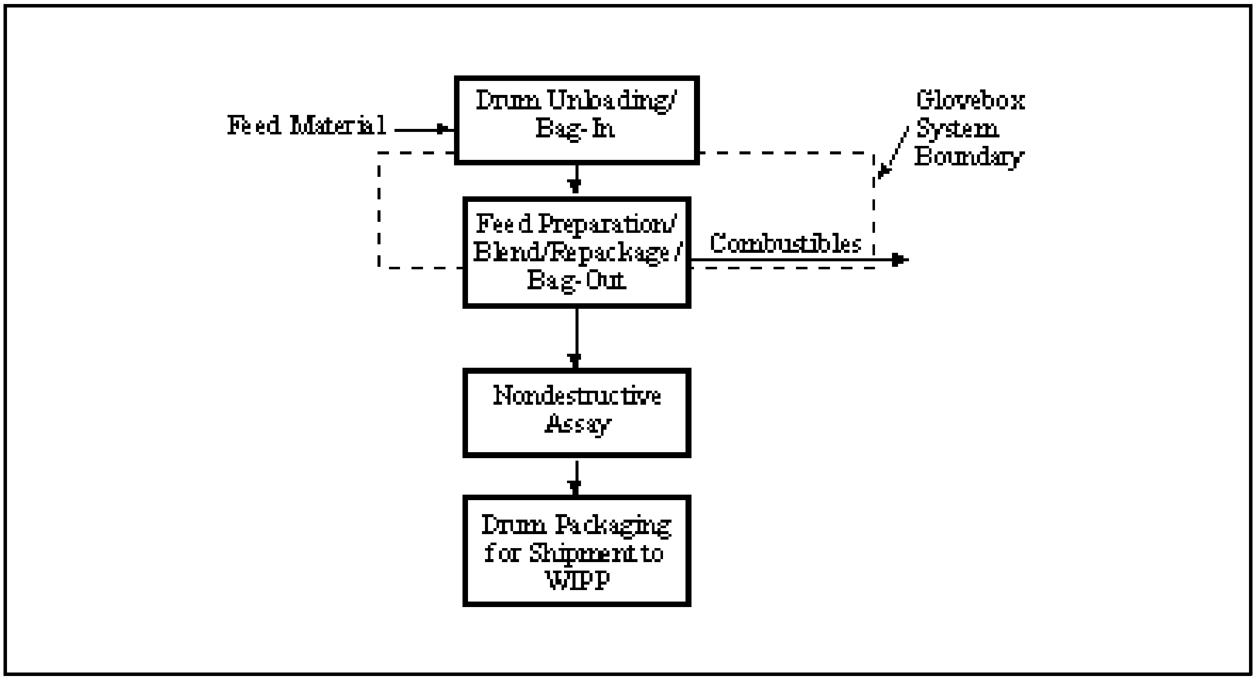

Figure C–34 Combination Process for the Repackaging of Molten Salt Extraction/Electrorefining Salts

The process steps would include drum unloading and bag-in, feed preparation, blending, and staging for pyro-oxidation, pyro-oxidation and bag-out, nondestructive assay, and drum packaging for interim site storage.

The description of the combination process for the pyro-oxidation of direct oxide reduction salts is similar to the pyro-oxidation of pyrochemical salts process description, except that the salts would be loaded into pipe components for ultimate shipment to WIPP. (See Section C.4.2.)

C.7.6 Combination Process for the Repackaging of Direct Oxide Reduction Salts

The residues in the EIS are divided into categories based upon similar applicable process technologies. The residues in the direct oxide reduction salt category have received a variance to the safeguards termination limit. In addition, stabilization may not be necessary, thereby allowing direct repack into pipe components and into drums for shipment to WIPP. The direct oxide reduction salts receiving a variance to the safeguards termination limit and not needing stabilization are addressed in this process description.

Under the combination concept for the repackaging of direct oxide reduction salts, direct oxide reduction salts would be blended to no more than 10 percent plutonium and repacked into pipe components for shipment to WIPP. The repack process will be conducted inside gloveboxes location in Module A, Building 707.

The combination process for the repackaging of direct oxide reduction salts is shown in **Figure C-35**. The process steps would include drum unloading and bag-in, feed preparation, repackaging (including blending), and bag-out. The packaged material will be nondestructively assayed for accountability purposes, and packaged into pipe components and then into drums for shipment to WIPP.

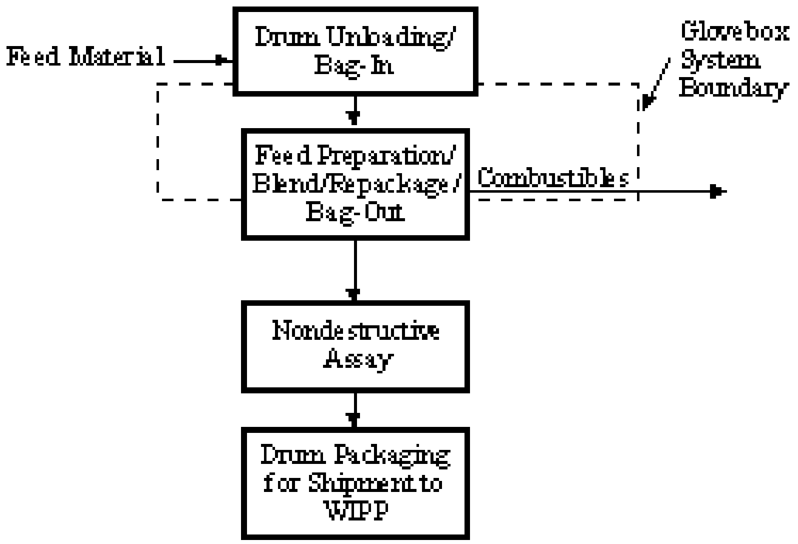

Figure C–35 Combination Process for the Repackaging of Direct Oxide Reduction Salts

❐ Detailed Process Description

Drums with the capacity of 208-L (55-gal) would be manually transferred from storage into a contamination control enclosure and unpacked. The contamination control enclosure is designed to control airflow in the event of a bag failure within a drum. The drums would be opened and the integrity of the packaging would be checked. If the packaging has not been compromised, the containers would be transferred into the glovebox. The containers, including outer packaging materials, would be removed from the drum and bagged into the glovebox.

If the integrity of the packaging has been compromised, the package would be overpacked with a new plastic bag prior to transfer to the glovebox.

The primary feed for this process would be direct oxide reduction salts. These materials would be introduced into the glovebox and the IDC verified. The individual packages would be opened, sorted, blended to no more than 10 percent plutonium, and placed into 8.2-L (2.2-gal) containers. The containers would then be bagged from the glovebox. Combustible packaging materials from the individual packages would be bagged out of the glovebox and sent to a combustible handling process. Other materials would be bagged out and managed appropriately.

Nondestructive assay would be performed and the assayed and repackaged salt containers would be loaded into a pipe component, one container each, which would be staged inside of a 208-L (55-gal) drum. These drums would be shipped to WIPP immediately.

C.7.7 Combination Process for the Neutralization and Drying of Aqueous-Contaminated Combustibles

The materials in the aqueous-contaminated combustible residues category have received a variance to the safeguards termination limit. The processing of these residues is addressed in this technology description.

The combination process for the neutralization and drying of aqueous-contaminated combustible residues would remove nitric acid from the organic matrix, eliminating a possibly unstable condition. The residues-consist of materials, such as cloth, paper, rags, coveralls, rubber, wood, and other miscellaneous materials, some of which are above the safeguards termination limit for combustibles. The application of a variance to the safeguards termination limit would allow shipment of this material to WIPP for disposal as transuranic waste. This process would be conducted in Room 3701 of Building 371.

The process steps would include drum unloading and bag-in, feed preparation, neutralization and decant/filtration, oven drying, packaging and bag-out, nondestructive assay, and drum transfer to interim site storage.

The description of the combination process for the neutralization and drying of aqueous-contaminated combustibles is identical to that given in Section C.4.3, Neutralization and Drying of Aqueous-Contaminated Combustibles.

C.7.8 Combination Process for the Thermal Desorption and Steam Passivation of Organic-Contaminated Combustibles

The materials in the organic-contaminated combustible residues category have received a variance to the safeguards termination limit. The processing of these residues is addressed in this technology description.

The combination process for the thermal desorption and steam passivation of organic-contaminated combustible residues would remove volatile organic contaminants from the residues and converts any plutonium fines present to plutonium oxide. The residues consist of materials such as wet and dry combustibles and leaded rubber gloves, some of which are above the safeguards termination limit for combustibles. The application of a variance to the safeguards termination limit would allow shipment of this material to WIPP for disposal as transuranic waste. This process would be conducted in Room 3701 of Building 371.

The process steps would include drum unloading and bag-in, feed preparation, followed by thermal desorption and steam passivation, addition of absorbent, packaging and bag-out, nondestructive assay, and drum transfer to interim site storage.

The description of the combination process for the thermal desorption and steam passivation of organic-contaminated combustibles is identical to that given in Section C.4.4, Thermal Desorption and Steam Passivation of Organic-Contaminated Combustibles.

C.7.9 Combination Process for the Repackaging of Dry Combustibles

The materials in the dry combustible residues category have received a variance to the safeguards termination limit. The processing of these residues is addressed in this technology description.

Under the combination concept for the repackaging of dry combustibles, repackaging of dry combustibles would be performed to achieve the criteria for safe interim site storage. Dry combustible residues consist of such materials as paper, rags, cloth, plastic, wood, surgical gloves, tape, paper, coveralls, booties, personal protective equipment waste, full-face masks, v-belts, polyvinyl chloride, polyethylene, polypropylene, supplied-air suits, and gaskets. After repackaging, the combustible residues above the safeguards termination limit will remain above the limit. The application of a variance to the safeguards termination limit would allow shipment of this material to WIPP for disposal as transuranic waste. Preparation of direct repackage residues would be conducted within glovebox lines in Modules D, E and F of Building 707.

The process steps include drum unloading and bag-in, feed preparation, repackaging and bag-out, nondestructive assay, and drum transfer to interim site storage.

The description of the combination process for the repackaging of dry combustibles is identical to that given in Section C.4.5, Repackaging of Dry Combustibles.

C.7.10 Combination Process for the Neutralization and Drying of Filter Media

The residues in the EIS are divided into categories based upon similar applicable process technologies. The residues in the filter media category have received a variance to the safeguards termination limit. The filter media residues with this variance that require neutralization are addressed in this technology description.

The filter media residues with a variance to the safeguards termination limit that require neutralization include Ful Flo filters and certain high-efficiency particulate air filters. The combination process for the neutralization and drying of filter media treats the nitric acid contaminant on the residue to eliminate the potential flammable hazard. After drying and repackaging, these residues would remain above the safeguards termination limit, which would preclude shipment to WIPP without the variance. This process would be conducted in Room 3701, Building 371.

The process steps would include drum unloading and bag-in, feed preparation, naturalization, decanting, and filtration, oven drying, packaging and bag-out, nondestructive assay, and drum transfer to interim site storage.

The description of the combination process for the neutralization and drying of filter media is identical to that given in Section C.4.7, Neutralization and Drying of Filter Media Residues.

C.7.11 Combination Process for the Repackaging of Filter Media

The residues in the EIS are divided into categories based upon similar applicable process technologies. The residues in the filter media category have received a variance to the safeguards termination limit. The filter media residues with these variances that do not require acid neutralization are addressed in this technology description.

The filter media residues with a variance to the safeguards termination limit that do not require acid neutralization include all filter media except for Ful Flo filters and IDC 338 high-efficiency particulate air filters. Under the combination concept for the repackaging of filter media, the filter media would be hand-blended to no more than 10 percent plutonium and repacked into 208-liter (55-gallon) drums for shipment to WIPP. The repack process would be conducted inside gloveboxes located in Room 3701, Building 371.

The combination process for the repackaging of filter media is shown in **Figure C-36.** The process steps would include drum unloading and bag in, feed preparation, repackaging (including blending), and bag-out. The packaged material would be placed into 208-liter (55-gallon) drums, non-destructively assayed for accountability purposes, and then would be ready for shipment to WIPP.

C.7.12 Combination Process for the Filtration and Drying of Sludge Residues

The residues in the EIS are divided into categories based upon similar applicable process technologies. The residues in the sludge category have received a variance to the safeguards termination limit. The sludge residues with this variance are addressed in this process description.

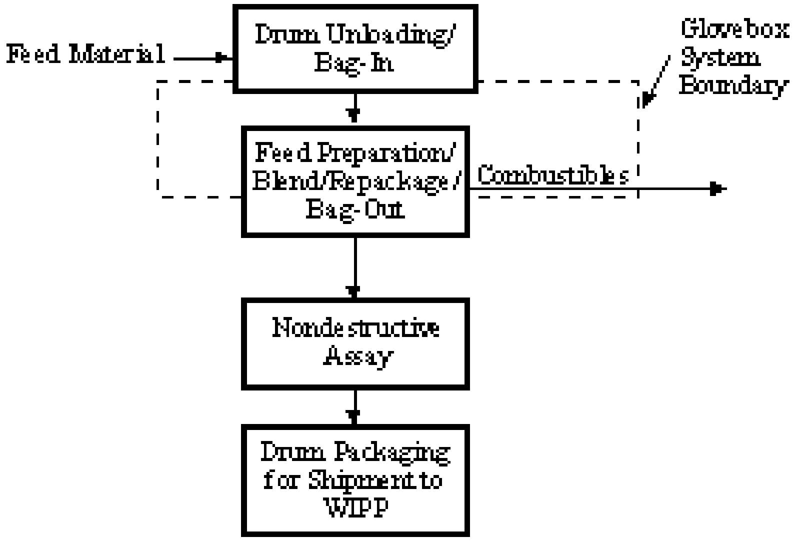

Figure C–36 Combination Process for the Repackaging of Filter Media

In the combination process for the filtration and drying of sludge residues, sludges would be filtered, if necessary, to remove excess liquid, and then dried by mixing the remaining material with an absorbent. After drying and repackaging, the sludges would remain above the safeguards termination limit, which would preclude ultimate shipment to WIPP without the safeguard termination limit variance. This process would be conducted in Room 3701, Building 371.

The process steps would include drum unloading and bag-in, feed preparation, decanting, and filtration, absorbent addition and bag-out, nondestructive assay, and drum packaging for interim site storage.

The description of the combination process for the filtration and drying of sludge residues is identical to that given in Section C.4.8, Filtration and Drying of Sludge Residues.

C.7.13 Combination Process for the Repackaging of Sludge Residues

The residues in the EIS are divided into categories based upon similar applicable process technologies. The residues in the sludge category have received a variance to the safeguards termination limit. The sludge residues with this variance that do not require filtration and drying are addressed in this technology description.

The sludge residues with a variance to the safeguards termination limit that do not require filtration and drying include grease oxide, grease fluoride, and oily sludge. Under the combination concept for the repackaging of sludge residues, the sludges would be hand-blended to no more than 10 percent plutonium and repacked into 208-liter (55-gallon) drums for shipment to WIPP. The repack process would be conducted inside gloveboxes located in Room 3701, Building 371.

The combination process for the repackaging of sludge residues is shown in **Figure C-37.** The process steps would include drum unloading and bag-in, feed preparation, repackaging (including blending), and bag-out. The packaged material would be placed into 55-gallon drums, non-destructively assayed for accountability purposes, and then would be ready for shipment to WIPP.

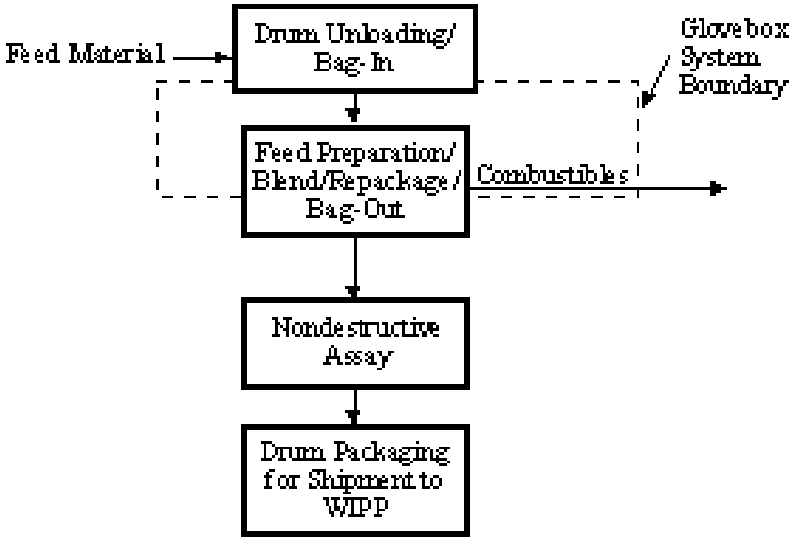

Figure C–37 Combination Process for the Repackaging of Sludge Residues

C.7.14 Combination Process for the Neutralization and Drying of Glass Residues

The materials in the glass residues category have received a variance to the safeguards termination limit. The processing of these residues is addressed in this technology description.

The combination process for the neutralization and drying of glass residues would remove the nitric acid contaminant on the residues eliminating a possibly unstable condition. The process would consist of washing the materials in an alkaline solution, allowing them to drain or partially dry, and mixing the resulting solids with water-absorbing materials. After processing, the glass residues may remain above the safeguards termination limit. The application of a variance to the safeguards termination limit would allow the shipment of this material to WIPP for disposal as transuranic waste. The process would be conducted in Room 3701 of Building 371.

The process steps would include drum unloading and bag-in, feed preparation, neutralization and decant/filtration, oven drying, blending, if necessary, packaging and bag-out, nondestructive assay, and drum transfer to interim site storage.

The description of the combination process for the neutralization and drying of glass residues is identical to that given in Section C.4.9, Neutralization and Drying of Glass Residues.

C.7.15 Combination Process for the Repackaging of Graphite Residues and Inorganic Residues

The materials in the graphite residue and inorganic residue categories have received a variance to the safeguards termination limit. The processing of these residues is addressed in this technology description.

Under the combination concept for the repackaging of graphite and inorganic residues, the graphite and inorganic residues would be repackaged to achieve the criteria for safe interim site storage. After repackaging, the residues may remain above the safeguards termination limit. The application of a variance to the safeguards termination limit would allow the shipment of this material to WIPP for disposal as transuranic waste. The process would be conducted within glovebox lines in Modules D, E and F of Building 707.

The process steps would include drum unloading and bag-in, feed preparation, blending, if necessary, repackaging and bag-out, nondestructive assay, and drum transfer to interim site storage.

The description of the combination process for the repackaging of graphite residues and inorganic residues is identical to that given in Section C.4.10, Repackaging of Graphite Residues, Inorganic Residues, and Scrub Alloy.

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