

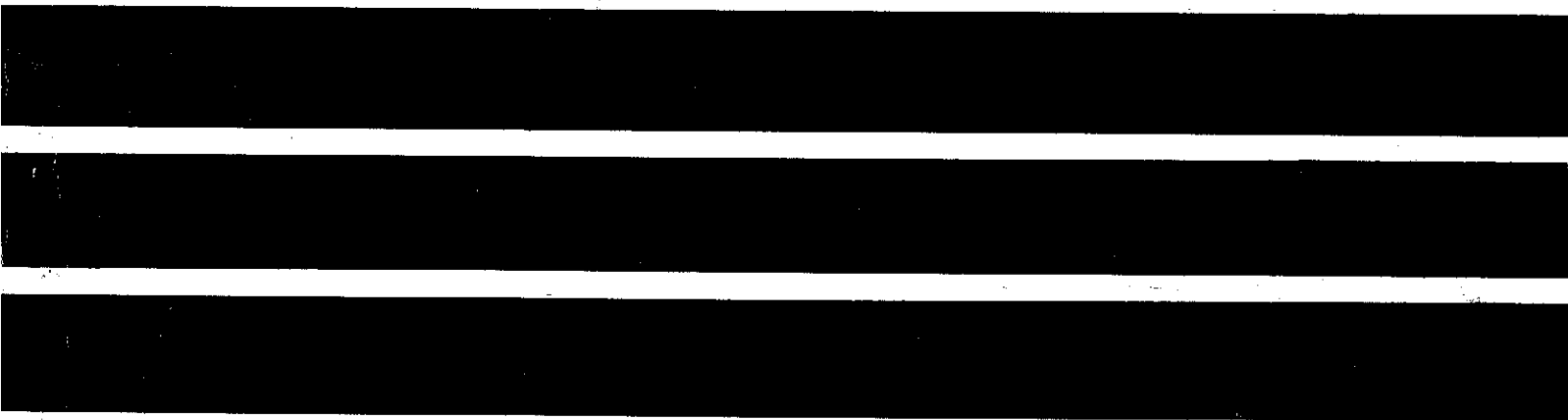
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# NUCLEAR MATERIALS AND FUEL CYCLE SERVICES SOURCES, INVENTORIES AND STOCKPILES

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heavy hydrogen or deuterium is increased by successively raising and lowering the temperature difference between water and a gaseous hydrogen compound ( $H_2S$ ). All naturally occurring hydrogen compounds contain some deuterium which can be extracted. The process is based on the fact that deuterium migrates to the water stream at low temperature and to the hydrogen sulphide gas at high temperature. By suitable arrangement of the flow in separating towers, deuterium can be extracted from a feed of ordinary water. In each tower the water flows down through a series of perforated plates, while the hydrogen sulphide bubbles up through the traps. This arrangement promotes efficient mixing.

In this manner, the hydrogen gas is enriched in deuterium and, leaving the top of the hot towers, passes into a cold tower where deuterium migrates to the water feed. A portion of the  $H_2S$  gas stream enriched in deuterium is extracted and passes to the next stage. This process is then repeated in the second stage and in a further stage. Enriched water from the third stage then passes to a finishing section where it is distilled to a reactor product that is 99.75% pure  $D_2O$ .

## 12. Reprocessing

The earliest known reprocessing occurred at the Hanford Laboratories in the early-to-mid 1940's. The process utilized bismuth phosphates and lanthanum fluorides to precipitate plutonium from the low burn-up, metallic fuel in a batch process (since the sole objective was to obtain plutonium for weapons, residual uranium was discharged with the fission product wastes).

Apparently, the process worked well, achieving acceptably high efficiencies for fission product removal and plutonium extraction. However, the waste volume was high.

Other than the very fact that it worked, the primary lasting technological achievements of the Hanford facility were the concepts of remote operation and maintenance.

The next technological steps took place in both the US and the UK in the late 1940's and early 1950's. In both countries, the objectives were to develop a process which could operate continuously and which would recover uranium as well as plutonium.

At Hanford, the basic solvent extraction process was refined and accepted. In this process, countercurrent flows of aqueous and organic solutions are made to move through some sort of mixing chamber (column, bowl, tank, etc.). The organic solvent strips both uranium and plutonium from the aqueous feed solution, leaving the fission products behind. Later, by adjusting valence, the plutonium can be made insoluble, thus separating the uranium and plutonium.

Hanford called this the Redox process and used Hexone (Methyl-isobutyl Getone) as the organic solvent. While the process indeed achieved the objectives of continuous operation and extraction of both uranium and plutonium, Hexone was expensive and flammable and the process generated very large quantities of waste.

The UK, constructing a separations plant at Windscale, developed a similar process but utilized a different solvent - Butex. Its major advantage over the US solvent, Hexone, was a significant reduction in waste volume.

Finally, during construction of the Savannah River Plant near Aiken, South Carolina in the early 1950's, the Purex process was developed. This process was tributylphosphate (TBP) dissolved in a kerosene-like solvent. TBP has

advantages of being chemically stable in nitric acid, relatively cheap, generating low waste volumes, and superior separations capability.

The basic Purex process is now generally accepted as the fundamental operating process for all reprocessing plants (with the notable exception of the abortive GE Morris effort). Different head ends (the means of dissolving the fuel and chemically preparing it for entry to the Purex process) and different final conversions (the means of converting the recovered uranium and plutonium to the desired chemical/physical form) are used depending upon the type of fuel to be reprocessed and the end-use of the recovered products.

Virtually all development work from then on has focused upon the methods used to bring the aqueous and organic liquid streams together and then separate them. The apparatus to do this is called a contactor. This is, of course, the heart of the solvent-extraction process and the key to both the process efficiency and waste volume.

Simple vertical columns were used as the first and most obvious contactor. These extraction columns were packed with various metal or ceramic shapes to create a very long flow path. The heavy aqueous solution was introduced at the top of the column and allowed to flow downward under the influence of gravity. The lighter organic solution, introduced near the bottom of the column, was displaced by the aqueous solution and forced to flow upwards. Thus, a countercurrent flow could be established and because of the metal or ceramic shapes, intimate aqueous/organic contact occurred. However, since flow rates were low and aqueous/organic contact was gentle, the columns had to be very tall to achieve reasonable efficiency. (The process building at Windscale is said to be 20 stories tall.)

The first step in the evolution of more efficient contactors involved pulsed columns. Pulsed columns utilize multiple perforated plates and, by applying alternating positive/negative pressure "pulses", force the liquids to pass back and forth through the perforations. This imparts a vigorous mixing action of the two streams. Thus, even though the two liquid streams still moved only by gravity, the extraction efficiency was significantly improved and column height could be reduced. This type of pulsed extraction column was used at the Hanford, Idaho National Engineering Laboratory (for naval propulsion and research reactor fuel reprocessing), and Nuclear Fuel Services' plants.

The next evolution in contactors involved a device called a "mixer-settler". This was a horizontal device (i.e. - required much lower buildings) of multiple stages. In each stage, the aqueous and organic streams are first drawn together and vigorously "mixed" by an agitator. Then the mixed solutions are driven by the agitator into a long, horizontal "settling" chamber. In this chamber, gravity again takes effect and the lighter organic solvent rises to the top while the heavier aqueous solution settles to the bottom. Both solutions are then separately drawn off from the chamber and introduced to further mix/settle stages.

This contactor must be considered a major step in that the mixing action was very strong and - for the first time - the two streams were mechanically propelled through the stages. Since the device was horizontal it was very amenable to either remote maintenance (as in its first use at Savannah River) or to a design where the mixer motors were physically remote from the mixers (as at later Windscale facilities). Unfortunately, the design inherently involves large volumes of mixed solutions and thus, major inventories of dissolved uranium and plutonium. This in turn causes con-

siderable chemical and radiolytic solvent degradation - involving both solvent makeup expense and somewhat increased waste volumes. This problem was the driving force to further improve contactor performance.

The centrifugal contactor, developed at Savannah River, was the next improvement. In this device, the mixed solutions are forced to flow from the motor driven agitator to a small centrifugal separator bowl mounted on the same shaft. In this bowl (which effectively replaces the long horizontal settler chamber), the aqueous and organic solutions are separated by centrifugal force.

In this device, for the first time, mechanical force was applied to all three primary solvent extraction sub-processes - stream movement, stream mixing, and stream separation. The results are impressive, including much lower in-process inventory (~25% of a mixer-settler), high separations efficiency, and low solvent degradation (and therefore, lower waste volume). Because of these advantages, Savannah River replaced their existing mixer-settler contactors with centrifugal settlers.

A derivative of the centrifugal contactor is the multi-stage centrifugal contactor developed by Saint-Gobain Techniques Nouvelles. This contactor, called Robatel, incorporates the equivalent of eight separate centrifugal contactors on a single motor driven shaft. This device is used in the first extraction stage at the AGNS Barnwell facility. Presumably, it may also be used in the planned new French plants and in any plants exported by the French.

Fuel reprocessing technology has been well established over the last twenty years in several countries, particularly for low burnup metallic fuels. Large scale experience has not yet been obtained in handling high-burnup oxide fuels.



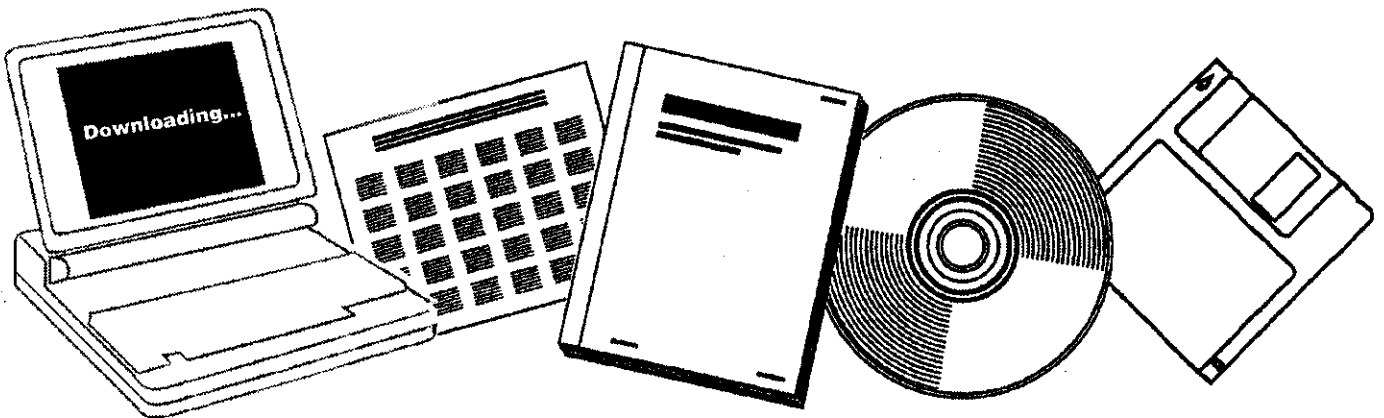
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# NUCLEAR PROLIFERATION AND CIVILIAN NUCLEAR POWER: REPORT OF THE NONPROLIFERATION ALTERNATIVE SYSTEMS ASSESSMENT PROGRAM. VOLUME II. PROLIFERATION RESISTANCE

DEPARTMENT OF ENERGY  
WASHINGTON, DC

DEC 1979



U.S. Department of Commerce  
**National Technical Information Service**



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3.3.2 Reprocessing

Basic Separation Processes

The PUREX process is the most widely known and used method of separating plutonium. It represents the culmination of U.S. efforts after World War II to develop the most efficient way of obtaining the purest possible plutonium for its military program. In fact, reprocessing of uranium-based nuclear-reactor fuel was first utilized about 35 years ago to separate the plutonium for one of the first U.S. nuclear weapons. Since only a minute part of the material processed was plutonium, stringent requirements were placed on the chemistry utilized. It was necessary to purify the plutonium by removing even traces of light elements and fission products so that the product material was usable in a military weapon and had a minimum amount of radioactivity. Under the pressures of World War II, a reprocessing plant (the first of a kind) was designed, built, and successfully operated in less than 18 months.

One of the process criteria for the weapons program emphasized obtaining very pure material so that radiation levels would be low. Early methods did not recover the uranium, minimize the high-level waste volume, or minimize the release of gaseous fission products. Nuclear-weapons stockpiling programs provided the incentive to develop better and more efficient techniques for removing plutonium from the irradiated fuel and to reduce the materials required for the total process. These developments resulted in the PUREX process.

With the prospects of a large civilian nuclear-power program, the incentive to minimize the life-cycle costs of the power has been an additional driving force for maximizing the economic utilization of fuel-cycle materials. This incentive has spurred the continued development of reprocessing technology as a potential contributor to this goal.

Although the nuclear-power fuel to be reprocessed is different--burnup ten times or more greater, oxide instead of metal is used, and zircalloy instead of aluminum clad is used--the PUREX process was readily adapted to commercial purposes. It appeared to minimize the financial or technical risk of expanding to a commercial scale and also promised the lowest cost. Accordingly, the commercial plants that have been designed for use today have been based on this process.

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PUREX-based technology has spread to other nations although the U.S. has not permitted the sale of reprocessing facilities. Numerous training programs have been used to export the technology, however.

The PUREX process uses solvent extraction for a very high degree of separation of uranium and plutonium from fission products and, as implemented, produces pure separate streams of plutonium and uranium in the form of nitrates. The process generally involves cutting fuel elements into small pieces (1 to 3 inches long) prior to dissolution in nitric acid. The fuel is dissolved out of the clad. The spent-fuel cladding and end fittings, or hulls, are then compacted and treated for disposal. The hulls are radioactive and contaminated with uranium, plutonium, and fission products. Vault storage is utilized for the plutonium oxide until it is transferred to fuel fabrication.

The reprocessing of thorium-based nuclear fuels is a somewhat less developed technology than is PUREX, although hundreds of tons of thorium have been processed in government facilities using the Thorex process. The principles are well understood because the Thorex separation process is similar to PUREX. The recovery of plutonium from denatured uranium-thorium fuels would introduce new uranium-plutonium separation techniques, but the commercial development required could probably be based on existing techniques. There are several differences between reprocessing thorium-uranium-plutonium fuel and reprocessing uranium-based fuel.

- Thorium is less soluble in nitric acid than is uranium.
- The U-233 produced may contain significant quantities of U-232, the decay products of which emit gamma-ray radiation. The conversion equipment and fuel-fabrication processes require greater shielding and may require remote operation.
- The separation of thorium fuels is more difficult to control; the formation of solids with solvent degradation products is more likely.
- The throughput for a given facility is less for the Thorex than for the PUREX process (larger equipment is required for the same uranium fuel throughput).

The oxide-conversion step for the U-233 product might be a modified coprecipitation process with remote fabrication required for U-233 fuels.

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The proliferation vulnerabilities of reprocessing of thorium-uranium (U-233)-plutonium fuels are not significantly different from uranium-plutonium fuels; that is, highly purified streams of weapons-usable materials are available. While U-233 may be isotopically denatured, plutonium would still potentially be available. An option not to separate the plutonium, that is, leaving the plutonium unseparated in the fission-product waste stream, would shift safeguards concerns from spent fuel to nuclear waste. Modification of the reprocessing facility to separate plutonium would not be formidable, however.

### Alternative Separation Processes

Many different processing schemes have been demonstrated at the laboratory level (see Table 3.3-1) and have been described in the open literature; some are easier and more economical to commercialize than others. The basic steps in reprocessing involve dissolution of the fuel elements and separation of uranium and plutonium. The fission products may be separated first (PUREX, REDOX) or uranium may be separated (fluoride volatility, ion exchange) or the plutonium may be separated (bismuth phosphate).

There are several processes, some aqueous, some nonaqueous, that can be used to perform separations: solvent extraction, ion exchange, volatility-absorption, and precipitation. In all cases, the process chemistry is controlled by factors which include mass flow, temperature, and solvent/acid concentrations although aqueous processes apparently show a higher degree of flexibility than nonaqueous ones. All of the processes which have been investigated on a laboratory scale, with the possible exception of the pyro-processes discussed below, are capable of separating fission products (decontamination) so that there is less than one part in  $10^5$  of fission products remaining with the product. Also, the chemical separation factors for uranium and plutonium are relatively high; that is, in most cases separation of uranium compounds from plutonium compounds is easy to perform, with less than one part in a hundred remaining in the product. Some experienced reprocessing engineers believe that processes that only partially separate uranium or fission products from plutonium can be readily modified to obtain separated plutonium by staging, by material recycling, or by modifying the process control variables.

The certainty with which this analysis can be supported varies, of course, with the current state of knowledge regarding specific processes, and it should be noted that none of the nonaqueous processes have been developed to a plant-design stage. Because substantial technological development is required, it would appear that these processes could be considered for application only to alternate fast-breeder fuel cycles using metal instead of oxide

Table 3.3-1. Reprocessing Schemes for Oxide Fuels

PROCESS	BASIC STEPS	STATUS	COMMENTS
BISMUTH PHOSPHATE	Head end, dissolve in nitric acid, precipitate with bismuth phosphate	First process used in weapons program	Uranium goes with fission products
REDOX	Head end, dissolve with nitric acid, separate fission products, solvent extraction, partitioning with hexone	Second process used in weapons program	Produces purer material, large volume of waste
PUREX	Head end, dissolve with nitric acid, separation of fission products by solvent extraction, partitioning of U and Pu by using tributyl phosphate	Third process used in weapons program. Has also been operated commercially	Requires cooling of fuel to prevent solvent degradation
THOREX	Similar to PUREX except used to reprocess thorium fuels	Has been operated commercially	Requires higher-volume process equipment
FLUORIDE VOLATILITY	Convert U to $UF_6$ by acid and fluorine. Pu goes with fission products	Demonstrated in lab	Very corrosive process, must keep moisture out of system.
ION EXCHANGE	Head end, dissolve with nitric acid, cation or anion exchange	Demonstrated in lab	Radiation damage of resins requires longer cooling times, 5% uranium loss
PYRO-PROCESSING	Head end, convert to metal, melt, salt extraction, solidify, melt, cast	Demonstrated in lab	Produces metals, instead of oxides

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fuels. Some of these processes, however, like pyrometallurgy, have potentially interesting features for proliferation resistance and have received preliminary analysis. Most, but not all, of the process steps have been demonstrated on either a laboratory or a pilot-plant scale. In the zinc distillation process, the normal product would be a uranium and plutonium alloy contaminated with fission products yielding estimated radiation levels of about five hundred rems per hour at one meter. Both process and equipment modifications would be required to yield a plutonium stream which would still be contaminated to a level of about ten rems per hour at one meter. If these changes were planned for and components were pretested covertly, it is estimated that several months would be required to effect them after the facility had been seized. It would be quicker to recover the plutonium in a dedicated facility which would take a year or more to prepare covertly.

Alternative Reprocessing Schemes

Alternative reprocessing schemes fall into three classes:

- Those in which plutonium compounds are never separated from uranium.
- Those in which there is a radiation barrier with the uranium and plutonium compounds (i.e., the fission products are not totally separated) or in which a radioactive spikant is added.
- Those in which the facility design is engineered to reduce access to plutonium or to inhibit process modification and enhance safeguards.

Dilution with uranium may be effected either by coprocessing or co-conversion. Radioactive contamination may be effected by either spiking or partial decontamination (alternatively, a radiation barrier could be introduced by preirradiating the fuel). Other technical measures include passive engineered features to reduce accessibility, active denial features, and perhaps integral separation and fabrication facilities. To obtain pure plutonium would require changes in some cases to the process itself, in some cases to the process equipment, in some cases to the facility, and in some cases to all three. The activities required in cases where changes in the chemical process are involved can be characterized by the times and efforts needed to prepare for the changes and to carry them out. These times and efforts are summarized in Table 3.3-2.

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Table 3.3-2. Characteristics of Activities Required for Plutonium Production from Different Separation and Reprocessing Schemes\*

Technology	Time/Resources to Plan and Develop Changes for Pure Pu	Time/Effort to Install Changes	Time to Produce Pure Pu	Comments
Normal PUREX	None	None	None	Standard process
Co-conversion	None	None	None	Separated Pu is present in the process
Coprocessing	Weeks/low	Days/minor	Days	
Partial decontamination	Weeks/low	Days/minor	Days	
Spiking	Little or none	Little or none	Little or none	
Pyrometallurgy (zinc distillation)	Months/substantial	Months/major	Weeks	Process requires substantial technology development

\*All schemes except the last listed in the table are based on the aqueous PUREX process and could be used to produce variations of the oxide fuels of the reference recycle and breeder fuel cycles. The pyrometallurgy process would be used for metallic fuels, depends on technology to be developed, and would probably be considered for application only to alternative fast breeders.

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The overall proliferation resistance of a number of such alternatives in terms of the removal of material, the misuse of a reprocessing or refabrication facility, and the conversion in dedicated facilities required to obtain plutonium metal is discussed in Section 2.2.3. The results are summarized in Table 2.2-1 of that section.

### 3.3.3 Plutonium Conversion and Mixed-Oxide Fuel Fabrication

After separation of fission products in the reference PUREX process, the conversion of liquid uranium and plutonium nitrates to oxides must be performed before the fabrication of fuel elements. There are several processes which can be utilized for this conversion (see Table 3.3-3). The alternative reprocessing schemes just described generate different products and therefore require different conversion processes. Since some of these conversion processes may at the same time provide some separation of spikants, fission products, or uranium from the plutonium, special care must be used to match the conversion process to the reprocessing product stream so that the desired proliferation resistance, if any, is retained. These relationships are indicated in the table. Moreover, precisely because some conversion processes can separate spikants, fission products, and uranium from plutonium, the proliferation implications of this process step are similar to those of reprocessing. The uranium and plutonium oxides are appropriately blended during the conversion process or afterwards to the desired ratio, that is, about 5%  $\text{PuO}_2$  for recycle in LWR's and 15 to 25%  $\text{PuO}_2$  for fast-breeder designs. In the reference case, the blending is achieved by the mechanical mixing of the oxide powders after conversion. In coprocessing, the appropriate mix is achieved before conversion.

The blended materials are then pressed to form a pellet, and fired (sintered) at about  $1200^\circ\text{C}$ . The pellets are mechanically assembled into fuel rods. To recover fissile material from the MOX fuel, the pellets must be removed from the clad and dissolved in acid. Some processes have been developed to improve the dissolution characteristics of MOX. The Coprecal process was developed so that fuel-fabrication plants could prepare MOX that was readily dissolvable (as scrap or as spent fuel) in nitric acid for recycling of fissile fuel.

### 3.3.4 Alternative Forms of Plutonium

The previous discussion has emphasized that plutonium in different forms occurs at different points in the fuel cycle. Table 3.3-4 provides a qualitative scale, from G to A in order of increasing sensitivity, for judging the

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Table 3.3-3. Plutonium Conversion Processes (Nitrate Solution to Oxide)

Type	Advantages	Disadvantages
Peroxide Precipitation	No reductant needed, very high decontamination; $H_2O_2$ only reagent	Potential explosions; long digestion times; low-density product; moderate losses (need recycling); not useful for coprocessing or prespiked material
Pu(IV) Oxalate Precipitation	High-stability solids and solutions, high decontamination; good in batch and continuous processes	Careful control of process required; numerous reagents; high losses (needs recycling); not useful for coprocessed or prespiked material
Pu(III) Oxalate Precipitation	Rapid settling, easily filtered, noncritical conditions, low losses, good impurity separation	Not so much experience as with Pu(IV); not useful for coprocessed or prespiked material
Direct Denitration	No added reagents; simple equipment; useful for coprocessed or prespiked material	Unproven technology; mechanical problems; no decontamination (an advantage for prespiked material)
Sol-Gel Process	Adaptable to coprocessing; high-density beads produced directly (remote handling of spiked material may be feasible in Sol-Gel equipment)	Complex process; elaborate equipment; needs testing; requires good control; no decontamination (an advantage for prespiked material). Density of sphere-pak fuel lower than pellet fuel.
Coprecal Process (ammonia coprecipitation and fluidized-bed calcination process)	Good for coprocessed material. May be adaptable to spiked material. Developed to improve dissolvability of MOX fuels. Cannot be used if $PuO_2 > 40\%$ .	No large-scale proof of feasibility yet; no decontamination (an advantage for prespiked material)



Table 3.3-4. Alternative Forms of Plutonium at Different Points in the Reference Recycle System

Level of Protection	Material	Location in fuel cycle (including transportation between facilities)	Processing Necessary	Approx. mass needed for 10 kg of Pu (arbitrary)	Other difficulties
G	Fuel assembly during irradiation	Within reactor core	Shut down of reactor, removal and cooling, the mechanical and chemical separation followed by conversion	1000 kg (2 PWR sub-assemblies)	Intensely radioactive
F	Discharged irradiated fuel subassembly	Reactor storage ponds, interim storage, reprocessing plant, long-term storage	Mechanical and chemical separation followed by conversion	1000 kg (2 PWR sub-assemblies)	Intense radioactivity falling with time after discharge
E	Fuel sub-assembly (prior to irradiation)	Recycle fuel-fabrication plant, reactor site	Mechanical and chemical separation followed by conversion	140 kg (1 PWR sub-assembly) 50 kg (1 LMFBR sub-assembly)	Toxicity Radioactivity*
D	Mixed Oxide (Pu+U)O <sub>2</sub>	Recycle fuel-fabrication plant (possibly reprocessing plant)	Dissolution and separation followed by reduction to metal	140 kg (PWR fuel) 50 kg (LMFBR fuel)	Toxicity Radioactivity*
C	Nitrate Pu(NO <sub>3</sub> ) <sub>4</sub>	Reprocessing plant	Conversion to oxide followed by reduction	17 kg	Toxicity Radioactivity*
B	Oxide PuO <sub>2</sub>	Reprocessing plant, plutonium storage site, recycle fuel-fabrication plant	Reduction to metal and fabrication	12 kg	Toxicity Radioactivity*
A	Pu metal	Not in reference recycle system**	Fabrication only	10 kg	Toxicity Radioactivity*

\*Depends on the plutonium isotopic composition

\*\*Can be present in other civilian nuclear activities, like, for example, critical facilities

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relative sensitivities according to the different forms of plutonium. The table gives the alternative forms in which plutonium exists at different points in the reference recycle system, which includes PUREX reprocessing. The forms are listed in decreasing order of inherent protection, that is, in decreasing order of the amount of processing required to obtain weapons-usable material. The key proliferation activities required to obtain weapons-usable material from fuel-cycle material includes preparation, acquisition of material, and processing of material. The elaboration of this table in terms of the resources and times required and of considerations relevant to detectability are taken up next in Section 3.3.5. Table 3.3-4 also notes the approximate amounts needed of each form of plutonium to yield 10 kg of separated plutonium and related handling difficulties.

The effects of the alternative processes on the form of plutonium available in the fuel cycle influences the entries in the table. For example, the effect of co-conversion is to eliminate pure  $\text{PuO}_2$  (Level B) everywhere in the cycle. The level of protection in the reprocessing plant would be further increased by coprocessing which would eliminate Level C. Pre-irradiation would increase the protection of fuel assemblies from Level E to Level F.

### 3.3.5 Dedicated Processing Facilities

Dedicated processing facilities are out-of-system facilities that are part of a military nuclear fuel-cycle system. This section describes typical chemical facilities that would be capable of processing spent fuel, fresh fuel, or intermediate fuel-cycle materials into metallic weapons-usable material.

Dedicated processing facilities would differ significantly from commercial fuel-cycle facilities in that economic, environmental, and long-term operating considerations would not dominate the design. One major design objective would be to remain undetected, especially during facility construction and prestartup testing. The operating scale of such a facility would be similar to chemical pilot-plant equipment. The most extensive facilities would be required for processing spent fuel; the complexity is reduced successively for spiked fuel materials, for fresh fuel, for MOX powder, and for  $\text{PuO}_2$ . The product of the dedicated facilities is plutonium metal; other facilities needed to convert the buttons into weapon shapes or to prepare nonnuclear weapon components are not discussed here, but such facilities would, of course, also be built and sited for maximum secrecy.

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Dedicated Facility Design Guidelines

Guidelines permitting comparisons of dedicated processing facilities are required for the estimates of the resources required, the times required, and detectability to be as consistent as possible:

- Plant capacity should be sized to produce about 10 kg of plutonium as soon as reasonably possible once material was removed from the fuel cycle; 100 kg should be obtained expeditiously.
- Plant capacity need not recover more than about 100 kg of plutonium in one year for covert scenarios.
- Plant should be sited to minimize detection during construction; possible options are within military reservations, within other chemical plants, or within an aircraft hanger or warehouse.
- Plant facilities should emphasize a high probability of success and simple, reliable operation.
- No special environmental regulations need be met; however, personnel exposure rates of up to 50 rem/yr can be assumed.
- Product recovery rates can be much lower than in industrial facilities (e.g., 85%), and liquid and solid wastes can be stored on-site.

Typical features of dedicated facilities for converting fuel-cycle materials into weapons-usable material are shown in Table 3.3-5. Specific details would depend on the material form in the fuel cycle, the chemical training available to the personnel directing the program, and the conditions which exist in the nation of concern. Thus, there are major uncertainties in the estimates of resources and times required for dedicated facilities. These uncertainties stem from such considerations as (a) a choice of different chemical techniques (e.g., some might select ion exchange to separate materials while others might select solvent extraction), (b) variations in feed material composition (e.g., plutonium in spent fuel varies with exposure and reactor design), (c) the interdependence of cost and time, and (d) specific conditions in a nation. In addition, specific difficulties in operating dedicated facilities affect these estimates. Some specific difficulties are

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Table 3.3-5. Features of Dedicated Processing Facilities  
for Processing Spent Fuel

Feature	Purpose and Effort
1. Siting of plant as an adjunct to an existing chemical plant or laboratory.	Construction is less obvious than at a new site. Services (utilities, laboratories) are not required.
2. Process Selection Examples	
a. Transport of spent fuel would be accomplished in a fuel-assembly truck cask built in the country.	Smaller capacity cask is more maneuverable and less detectable.
b. Size reduction of spent fuel rods would be performed with abrasive saw in water pools.	Less obvious than a specialized rod-shearing device.
c. Ion exchange would be used to recover plutonium rather than solvent extraction system if feasible.	Ion exchange systems more generally used in chemical operations. Tributyl phosphate solvent would not be required.
d. Liquid wastes from operation would be neutralized and stored in underground vault or steel tanks.	Acid waste requires special stainless steel; waste solidification is not required.
e. Mirrors and TV cameras used for remote viewing of operations.	Manipulators and shielding windows might be traceable.
f. Precipitation processes could be considered for Pu separation if personnel were inexperienced in ion exchange systems.	Pu oxalate, Pu fluoride or Bi-phosphate system separates Pu from uranium, but more fission-product removal requires multiple cycles, more wastes.
3. Batch processes with high yield but provisions for problems made by including multiple lines and spares.	Minimum dependence on skill and experience of personnel.
4. Few process samples would be taken to reduce product losses, confirm proper chemical additions but not to establish close accountability.	Not concerned about achieving high yield in reactions or in recovering all the material but avoid major plant delays (criticality accident, fire, etc.).
5. Use hands-on operation, shadow shielding, and tongs for radiation levels up to tens of rem/hr.	Reduce problems and facilities requirements associated with remote operations, sophisticated equipment.

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given in Table 3.3-6. Most importantly, there exists little or no comparable experience to serve as a basis for making these estimates. They are the collective informed judgment of experienced professionals in the chemical reprocessing field.

The estimates of resources and time associated with dedicated facilities for misuse of fuel-cycle materials are summarized in Table 3.3-7. The estimates are reported as ranges rather than single values but cannot accurately reflect the range of the uncertainties just cited. The analysis was made by taking a consistent approach for reliability and competence of design while allowing for typical minor problems in new facilities. The lower end of the range does not represent a minimum time or cost, nor does the upper end allow for incompetence. Specifically, a range of 1 to 3 weeks is estimated to produce 10 kg of plutonium from spent fuel. If LWR fuel remains in the reactor for its designed lifetime, then 1 year after it is discharged, processing it was estimated to require about 12 days to recover 10 kg of plutonium metal; with minimum delays, the time could be 8 days. If the design were poor, if equipment were shoddy or makeshift, or if the operational personnel made poor decisions, many weeks could elapse. Similar factors affect development time (design, procurement, facility construction, and cold testing). Specific factors associated with the industrial base present in the nation might affect vessel fabrication or instrumentation and piping deliveries (either longer or shorter than in the U. S.). However, modifying existing facilities could shorten the construction effort (but might extend operating time). System checkout time would depend on the program management and the results in checkout of individual equipment subsystems. The time to recover 100 kg of plutonium would be dependent on the operating performance of the equipment; about 50% availability is considered likely for a system built when great speed is urgent. Estimates of the time to build and prepare to operate facilities to process spent fuel range from 12 to 24 months (versus a wider range of 4 to 30 months reported by the General Accounting Office). For dedicated facilities, much of this range may be associated with different assumptions on quality of equipment, operating lifetime, and personnel risk, used by different analysts. In general, there is little agreement among United States engineers on the variation in these estimates when applied to other countries and to engineers with a potential wide variation in experience and training. The times might be a little shorter if several experienced staff were directing the effort; however, the times could be significantly longer (perhaps by a factor of two or three) if the level of technology and the experience of the staff were minimal.

Individual estimates differ for spent fuel, for spiked fuel materials, and for unspiked fuel materials. However, whether all the differences provide a meaningful discrimination between fuel cycles is unclear. For example, the

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Table 3.3-6. Difficulties in Operation of Dedicated Facilities  
for Spent Fuel

PROCESS STEP	DIFFICULTY	CONSIDERATION TO OVERCOME DIFFICULTY
1. Move spent fuel from storage site, receive and unload at dedicated facility.	Mechanical equipment problems; damage to fuel in transit	Spent-fuel handling in reactor basins is routine work. Cask handling is routine. Special casks may be less restrictive. Practice using cold fuel
2. Cut fuel (abrasive sawing).	Equipment failure; poor visibility in basin	Spare equipment. Provide filter and water circulation in basin. Practice using simulated fuel
3. Transfer cut fuel to dissolver.	Mechanical problems, contamination	Practice using simulated fuel
4. Dissolve fuel.	Acid too hot or too concentrated; solution spills	Accepted contamination. Provide sump; pump to waste. Enlarge dissolver
5. Filter or centrifuge dissolver solution.	Filter clogs or centrifuge fails	Replace, backwash, or by-pass equipment. Solids no problem in solvent extraction; solids can eventually plug ion exchange columns
6. Feed to ion exchange or solvent extraction.	Poor control of hydraulics, poor separation	Adequate instrumentation proven in cold runs
7. Precipitate $PuF_3$ , collect, wash, dry.	Excess losses if chemistry not correct	Process is published
8. Reduce $PuF_3$ to Pu, clean up, pickle.	Impure product	Accept or recycle
9. General problems typical for many facilities.	Equipment leaks or spills	Accepted in design
	Corrosion	Minimize by design
	Criticality	Gross mass controls; some equipment design
	Plutonium solids form at low acid (<0.1M)	Monitor acidity

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Table 3.3-7. Summary of Information on Dedicated Processing Facilities

	Millions of Dollars Capital Cost	Operating Manpower	Development Time (months)	Cold-Test Time* (weeks)	Time for 10 kg Pu (weeks)	Time for 100 kg Pu (weeks)
Spent Fuel (cooled)**	12-24	100	12-24	4-8	1-3	25-35
PuO <sub>2</sub>	1/2-1	20	6-9	1-3	1/2-1	9-12
PuO <sub>2</sub> - UO <sub>2</sub> (cold) MOX	1-2	20	8-12	2-4	1/2-1	10-20
Fresh-Fuel (cold MOX) Assemblies	1-2	30	8-12	2-4	1/2-1	10-20
PuO <sub>2</sub> - UO <sub>2</sub> (hot) MOX***	5-10	50	10-15	3-6	1/2-1	15-30****
Fresh-Fuel (hot MOX) Assemblies***	6-12	80	10-15	3-6	1/2-1	15-30****

\*Time assumes training of operating personnel using cold materials during construction phase.

\*\*"Cooled" means radioactive at levels one year after discharge.

\*\*\*Hot means partially decontaminated (or pre-irradiated, in the case of fresh fuel assemblies).

\*\*\*\*Time is that allowed for remote maintenance not required for cold facilities.

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cost estimates vary up to a factor of ten or more, but all costs are sufficiently low to be within the resource limits of nations with nuclear-power plants. Small operating staffs are needed for such facilities. The processing time to recover materials for a few weapons is a matter of weeks in any case.

The clearest difference is between spent fuel and plutonium once it has been separated and converted to oxide. Thereafter, the incremental differences are less substantial, and their significance is less clear especially when considered in the light of the uncertainties to which these estimates are subject.