**Nuclear Reprocessing**

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Nuclear reprocessing technology was developed to chemically separate and recover fissionable plutonium from irradiated nuclear fuel. Reprocessing serves multiple purposes, whose relative importance has changed over time. Originally reprocessing was used solely to extract plutonium for producing nuclear weapons. With the commercialization of nuclear power, the reprocessed plutonium was recycled back into MOX nuclear fuel for thermal reactors. The reprocessed uranium, which constitutes the bulk of the spent fuel material, can in principle also be re-used as fuel, but that is only economic when uranium prices are high. Finally, the breeder reactor can employ not only the recycled plutonium and uranium in spent fuel, but all the actinides, closing the nuclear fuel cycle and potentially multiplying the energy extracted from natural uranium by more than 60 times.

Nuclear reprocessing reduces the volume of high-level waste, but by itself does not reduce radioactivity or heat generation and therefore does not eliminate the need for a geological waste repository. Reprocessing has been politically controversial because of the potential to contribute to nuclear proliferation, the potential vulnerability to nuclear terrorism, the political challenges of repository siting (a problem that applies equally to direct disposal of spent fuel), and because of its high cost compared to the once-through fuel cycle. The Obama administration stepped back from President Bush's plans for commercial-scale reprocessing and reverted to a program focused on reprocessing-related scientific research.

**Separated components and disposition**

The potentially useful components dealt with in nuclear reprocessing comprise specific actinides (plutonium, uranium, and some minor actinides). The lighter elements components include fission products, activation products, and cladding.

|  |  |
| --- | --- |
| **material** | **disposition** |
| plutonium, minor actinides, reprocessed uranium | fission in fast, fusion, or subcritical reactor |
| reprocessed uranium, cladding, filters | less stringent storage as intermediate-level waste |
| long-lived fission and activation products | nuclear transmutation or geological repository |
| medium-lived fission products 137Cs and 90Sr | medium-term storage as high-level waste |
| useful radionuclides and noble metals | industrial and medical uses |

**History**

The first large-scale nuclear reactors were built during World War II. These reactors were designed for the production of plutonium for use in nuclear weapons. The only reprocessing required, therefore, was the extraction of the plutonium (free of fission-product contamination) from the spent natural uranium fuel. In 1943, several methods were proposed for separating the relatively small quantity of plutonium from the uranium and fission products. The first method selected, a precipitation process called the Bismuth Phosphate process, was developed and tested at the Oak Ridge National Laboratory (ORNL) in the 1943–1945 period to produce quantities of plutonium for evaluation and use in weapons programs. ORNL produced the first macroscopic quantities (grams) of separated plutonium with these processes.

The Bismuth Phosphate process was first operated on a large scale at the Hanford Site, in the latter part of 1944. It was successful for plutonium separation in the emergency situation existing then, but it had a significant weakness: the inability to recover uranium.

The first successful solvent extraction process for the recovery of pure uranium and plutonium was developed at ORNL in 1949. The PUREX process is the current method of extraction. Separation plants were also constructed at Savannah River Site and a smaller plant at West Valley, New York which closed by 1972 because of its inability to meet new regulatory requirements.

Reprocessing of civilian fuel has long been employed in Europe, at the COGEMA La Hague site in France, the Sellafield site in the United Kingdom, the Mayak Chemical Combine in Russia, and at sites such as the Tokai plant in Japan, the Tarapur plant in India, and briefly at the West Valley Reprocessing Plant in the United States.

In October 1976, fear of nuclear weapons proliferation (especially after India demonstrated nuclear weapons capabilities using reprocessing technology) led President Gerald Ford to issue a Presidential directive to indefinitely suspend the commercial reprocessing and recycling of plutonium in the U.S. On April 7, 1977, President Jimmy Carter banned the reprocessing of commercial reactor spent nuclear fuel. The key issue driving this policy was the serious threat of nuclear weapons proliferation by diversion of plutonium from the civilian fuel cycle, and to encourage other nations to follow the USA lead. After that, only countries that already had large investments in reprocessing infrastructure continued to reprocess spent nuclear fuel. President Reagan lifted the ban in 1981, but did not provide the substantial subsidy that would have been necessary to start up commercial reprocessing.

In March 1999, the U.S. Department of Energy (DOE) reversed its own policy and signed a contract with a consortium of Duke Energy, COGEMA, and Stone & Webster (DCS) to design and operate a Mixed Oxide (MOX) fuel fabrication facility. Site preparation at the Savannah River Site (South Carolina) began in October 2005.

**Separation technologies**

**Water and organic solvents**

**PUREX**, the current standard method, is an acronym standing for ***P****lutonium and* ***U****ranium* ***R****ecovery by* ***EX****traction*. The PUREX process is a liquid-liquid extraction method used to reprocess spent nuclear fuel, in order to extract uranium and plutonium, independent of each other, from the fission products. This is the most developed and widely used process in the industry at present. When used on fuel from commercial power reactors the plutonium extracted typically contains too much Pu-240 to be useful in a nuclear weapon. However, reactors that are capable of refueling frequently can be used to produce weapon-grade plutonium, which can later be recovered using PUREX. Because of this, PUREX chemicals are monitored.

**Modifications of PUREX**

**UREX**

The PUREX process can be modified to make a **UREX** (**UR**anium **EX**traction) process which could be used to save space inside high level nuclear waste disposal sites, such as the Yucca Mountain nuclear waste repository, by removing the uranium which makes up the vast majority of the mass and volume of used fuel and recycling it as reprocessed uranium.

The UREX process is a PUREX process which has been modified to prevent the plutonium from being extracted. This can be done by adding a plutonium reductant before the first metal extraction step. In the UREX process, ~99.9% of the uranium and >95% of technetium are separated from each other and the other fission products and actinides. The key is the addition of acetohydroxamic acid (AHA) to the extraction and scrub sections of the process. The addition of AHA greatly diminishes the extractability of plutonium and neptunium, providing greater proliferation resistance than with the plutonium extraction stage of the PUREX process.

**TRUEX**

Adding a second extraction agent, octyl(phenyl)-N, N-dibutyl carbamoyl-methyl phosphine oxide(CMPO) in combination with tributyl-phosphate, (TBP), the PUREX process can be turned into the **TRUEX** (**TR**ans**U**ranic **EX**traction) process. TRUEX was invented in the USA by Argonne National Laboratory and is designed to remove the transuranic metals (Am/Cm) from waste. The idea is that by lowering the alpha activity of the waste, the majority of the waste can then be disposed of with greater ease. In common with PUREX this process operates by a solvation mechanism.

**DIAMEX**

As an alternative to TRUEX, an extraction process using a malondiamide has been devised. The DIAMEX (**DIAM**ide**EX**traction) process has the advantage of avoiding the formation of organic waste which contains elements other than carbon, hydrogen, nitrogen, and oxygen. Such an organic waste can be burned without the formation of acidic gases which could contribute to acid rain. The DIAMEX process is being worked on in Europe by the French CEA. The process is sufficiently mature that an industrial plant could be constructed with the existing knowledge of the process. In common with PUREX this process operates by a solvation mechanism.

**SANEX**

**S**elective **A**cti**N**ide **EX**traction. As part of the management of minor actinides it has been proposed that the lanthanides and trivalent minor actinides should be removed from the PUREX raffinate by a process such as DIAMEX or TRUEX. In order to allow the actinides such as americium to be either reused in industrial sources or used as fuel, the lanthanides must be removed. The lanthanides have large neutron cross sections and hence they would poison a neutron driven nuclear reaction. To date the extraction system for the SANEX process has not been defined, but currently several different research groups are working towards a process. For instance the French CEA is working on a bis-triazinyl pyridine (BTP) based process. Other systems such as the dithio-phosphinic acids are being worked on by some other workers.

**UNEX**

The ***UN****iversal* **EX**traction process was developed in Russia and the Czech Republic; it is designed to completely remove the most troublesome radioisotopes (Sr, Cs and minor actinides) from the raffinate remaining after the extraction of uranium and plutonium from used nuclear fuel. The chemistry is based upon the interaction of cesium and strontium with polyethylene glycol) and a cobalt carborane anion (known as chlorinated cobalt dicarbollide). The actinides are extracted by CMPO, and the diluent is a polar aromatic such as nitrobenzene. Other dilents such as *meta*-nitrobenzotrifluoride and phenyl trifluoromethyl sulfone have been suggested as well.

**Electrochemical methods**

An exotic method using electrochemistry and ion exchange in ammonium carbonate has been reported.

**Obsolete methods**

**Bismuth phosphate**

The bismuth phosphate process is an obsolete process that adds significant unnecessary material to the final radioactive waste. The bismuth phosphate process has been replaced by solvent extraction processes. The bismuth phosphate process was designed to extract plutonium from aluminum-clad nuclear fuel rods, containing uranium. The fuel was decladded by boiling it in caustic soda. After decladding, the uranium metal was dissolved in nitric acid.

The plutonium at this point is in the +4 oxidation state. It was then precipitated out of the solution by the addition of bismuth nitrate and phosphoric acid to form the bismuth phosphate. The plutonium was coprecipitated with this. The supernatant liquid (containing many of the fission products) was separated from the solid. The precipitate was then dissolved in nitric acid before the addition of an oxidant such as potassium permanganate which converted the plutonium to PuO22+ (Pu VI), then a dichromate salt was added to maintain the plutonium in the +6 oxidation state.

The bismuth phosphate was next re-precipitated leaving the plutonium in solution. Then an iron (II) salt such as ferrous sulfate was added, and the plutonium re-precipitated again using a bismuth phosphate carrier precipitate. Then lanthanum salts and fluoride were added to create solid lanthanum fluoride which acted as a carrier for the plutonium. This was converted to the oxide by the action of an alkali. The lanthanum plutonium oxide was next collected and extracted with nitric acid to form plutonium nitrate.

**Hexone or redox**

This is a liquid-liquid extraction process which uses methyl isobutyl ketone as the extractant. The extraction is by a *solvation* mechanism. This process has the disadvantage of requiring the use of a salting-out reagent (aluminum nitrate) to increase the nitrate concentration in the aqueous phase to obtain a reasonable distribution ratio (D value). Also, hexone is degraded by concentrated nitric acid. This process has been replaced by the PUREX process.

Pu4+ + 4 NO3− + 2S → [Pu(NO3)4S2]

**[edit] Butex, β,β'-dibutyoxydiethyl ether**

A process based on a solvation extraction process using the triether extractant named above. This process has the disadvantage of requiring the use of a salting-out reagent (aluminum nitrate) to increase the nitrate concentration in the aqueous phase to obtain a reasonable distribution ratio. This process was used at Windscale many years ago. This process has been replaced by PUREX.

**Pyroprocessing**

**Pyroprocessing** is a generic term for high-temperature methods. Solvents are molten salts (e.g. LiCl+KCl or LiF+CaF2) and molten metals (e.g. cadmium, bismuth, magnesium) rather than water and organic compounds. Electrorefining, distillation, and solvent-solvent extraction are common steps.

These processes are not currently in significant use worldwide, but they have been researched and developed at Argonne National Laboratory and elsewhere.

**Advantages**

* The principles behind them are well understood, and no significant technical barriers exist to their adoption.
* Readily applied to high-burnup spent fuel and requires little cooling time, since the operating temperatures are high already.
* Does not use solvents containing hydrogen and carbon, which are neutron moderators creating risk of criticality accidents and can absorb the fission product tritium and the activation product carbon-14 in dilute solutions that cannot be separated later.
	+ Alternatively, voloxidation can remove 99% of the tritium from used fuel and recover it in the form of a strong solution suitable for use as a supply of tritium.
* More compact than aqueous methods, allowing on-site reprocessing at the reactor site, which avoids transportation of spent fuel and its security issues, instead storing a much smaller volume of fission products on site as high-level waste until decommissioning. For example, the Integral Fast Reactor and Molten Salt Reactor fuel cycles are based on on-site pyroprocessing.
* It can separate many or even all actinides at once and produce highly radioactive fuel which is harder to manipulate for theft or making nuclear weapons. (However, the difficulty has been questioned.) In contrast the PUREX process was designed to separate plutonium only for weapons, and it also leaves the minor actinides (americium and curium) behind, producing waste with more long-lived radioactivity.
* Most of the radioactivity in roughly 102 to 105 years after the use of the nuclear fuel is produced by the actinides, since there are no fission products with half-lives in this range. These actinides can fuel fast reactors, so extracting and reusing (fissioning) them reduces the long-term radioactivity of the wastes.

**Disadvantages**

* Reprocessing as a whole is not currently (2005) in favor, and places that do reprocess already have PUREX plants constructed. Consequently, there is little demand for new pyrometalurgical systems, although there could be if the Generation IV reactor programs become reality.
* The used salt from pyroprocessing is less suitable for conversion into glass than the waste materials produced by the PUREX process.
* If the goal is to reduce the longevity of spent nuclear fuel in burner reactors, then better recovery rates of the minor actinides need to be achieved.

**Electrolysis**

**PYRO-A and -B for IFR**

These processes were developed by Argonne National Laboratory and used in the Integral Fast Reactor project.

**PYRO-A** is a means of separating actinides (elements within the actinide family, generally heavier than U-235) from non-actinides. The spent fuel is placed in an anode basket which is immersed in a molten salt electrolyte. An electrical current is applied, causing the uranium metal (or sometimes oxide, depending on the spent fuel) to plate out on a solid metal cathode while the other actinides (and the rare earths) can be absorbed into a liquid cadmium cathode. Many of the fission products (such as cesium, zirconium and strontium) remain in the salt. As alternatives to the molten cadmium electrode it is possible to use a molten bismuth cathode, or a solid aluminum cathode.

As an alternative to electrowinning, the wanted metal can be isolated by using a molten alloy of an electropositive metal and a less reactive metal.

Since the majority of the long term radioactivity, and volume, of spent fuel comes from actinides, removing the actinides produces waste that is more compact, and not nearly as dangerous over the long term. The radioactivity of this waste will then drop to the level of various naturally occurring minerals and ores within a few hundred, rather than thousands of, years.

The mixed actinides produced by pyrometallic processing can be used again as nuclear fuel, as they are virtually all either fissile, or fertile, though many of these materials would require a fast breeder reactor in order to be burned efficiently. In a thermal neutron spectrum, the concentrations of several heavy actinides (curium-242 and plutonium-240) can become quite high, creating fuel that is substantially different from the usual uranium or mixed uranium-plutonium oxides (MOX) that most current reactors were designed to use.

Another pyrochemical process, the PYRO-B process, has been developed for the processing and recycling of fuel from a transmuter reactor ( a fast breeder reactor designed to convert transuranic nuclear waste into fission products ). A typical transmuter fuel is free from uranium and contains recovered transuranics in an inert matrix such as metallic zirconium. In the PYRO-B processing of such fuel, an electrorefining step is used to separate the residual transuranic elements from the fission products and recycle the transuranics to the reactor for fissioning. Newly-generated technetium and iodine are extracted for incorporation into transmutation targets, and the other fission products are sent to waste.

**Voloxidation**

Voloxidation (for *volumetric oxidation*) involves heating oxide fuel with oxygen, sometimes with alternating oxidation and reduction, or alternating oxidation by ozone to uranium trioxide with decomposition by heating back to triuranium octoxide. A major purpose is to capture tritium as tritiated water vapor before further processing where it would be difficult to retain the tritium. Other volatile elements leave the fuel and must be recovered, especially iodine, technetium, and carbon-14. Voloxidation also breaks up the fuel or increases its surface area to enhance penetration of reagents in following reprocessing steps.

**Volatilization in isolation**

Simply heating spent oxide fuel in an inert atmosphere or vacuum at a temperature between 700°C and 1000°C as a first reprocessing step can remove several volatile elements, including cesium whose isotope cesium-137 emits about half of the heat produced by the spent fuel over the following 100 years of cooling (however, most of the other half is from strontium-90 which remains). The estimated overall mass balance for 20,000 grams of processed fuel with 2,000 grams of cladding is:

|  |  |  |  |  |  |
| --- | --- | --- | --- | --- | --- |
|  | **Input** | **Residue** | **Zeolitefilter** | **Carbonfilter** | **Particlefilters** |
| Palladium | 28 | 14 | 14 |  |  |
| Tellurium | 10 | 5 | 5 |  |  |
| Molybdenum | 70 |  | 70 |  |  |
| Cesium | 46 |  | 46 |  |  |
| Rubidium | 8 |  | 8 |  |  |
| Silver | 2 |  | 2 |  |  |
| Iodine | 4 |  |  | 4 |  |
| Cladding | 2000 | 2000 |  |  |  |
| Uranium | 19218 | 19218 |  |  | ? |
| Others | 614 | 614 |  |  | ? |
| Total | 22000 | 21851 | 145 | 4 | 0 |

Tritium is not mentioned in this paper.

**Fluoride volatility**

Main article: Fluoride volatility



Blue elements have volatile fluorides or are already volatile; green elements do not but have volatile chlorides; red elements have neither, but the elements themselves or their oxides are volatile at very high temperatures. Yields at 100,1,2,3 years after fission, not considering later neutron capture, fraction of 100% not 200%. Beta decay Kr-85→Rb, Sr-90→Zr, Ru-106→Pd, Sb-125→Te, Cs-137→Ba, Ce-144→Nd, Sm-151→Eu, Eu-155→Gd visible.

In the fluoride volatility process, fluorine is reacted with the fuel. Fluorine is so much more reactive than even oxygen that small particles of ground oxide fuel will burst into flame when dropped into a chamber full of fluorine. This is known as flame fluorination; the heat produced helps the reaction proceed. Most of the uranium, which makes up the bulk of the fuel, is converted to uranium hexafluoride, the form of uranium used in uranium enrichment, which has a very low boiling point. Technetium, the main long-lived fission product, is also efficiently converted to its volatile hexafluoride. A few other elements also form similarly volatile hexafluorides, pentafluorides, or heptafluorides. The volatile fluorides can be separated from excess fluorine by condensation, then separated from each other by fractional distillation or selective reduction. Uranium hexafluoride and technetium hexafluoride have very similar boiling points and vapor pressures, which makes complete separation more difficult.

Many of the fission products volatilized are the same ones volatilized in non-fluorinated, higher-temperature volatilization, such as iodine, tellurium and molybdenum; notable differences are that technetium is volatilized, but cesium is not.

Some transuranium elements such as plutonium, neptunium and americium can form volatile fluorides, but these compounds are not stable when the fluorine partial pressure is decreased. Most of the plutonium and some of the uranium will initially remain in ash which drops to the bottom of the flame fluorinator. The plutonium-uranium ratio in the ash may even approximate the composition needed for fast neutron reactor fuel. Further fluorination of the ash can remove all the uranium, neptunium, and plutonium as volatile fluorides; however, some other minor actinides may not form volatile fluorides and instead remain with the alkaline fission products. Some noble metals may not form fluorides at all, but remain in metallic form; however ruthenium hexafluoride is relatively stable and volatile.

Distillation of the residue at higher temperatures can separate lower-boiling transition metal fluorides and alkali metal (Cs, Rb) fluorides from higher-boiling lanthanide and alkaline earth metal (Sr, Ba) and yttrium fluorides. The temperatures involved are much higher, but can be lowered somewhat by distilling in a vacuum. If a carrier salt like lithium fluoride or sodium fluoride is being used as a solvent, high-temperature distillation is a way to separate the carrier salt for reuse.

Molten salt reactor designs carry out fluoride volatility reprocessing continuously or at frequent intervals. The goal is to return actinides to the molten fuel mixture for eventual fission, while removing fission products that are neutron poisons, or that can be more securely stored outside the reactor core while awaiting eventual transfer to permanent storage.

**Chloride volatility and solubility**

Many of the elements that form volatile high-valence fluorides will also form volatile high-valence chlorides. Chlorination and distillation is another possible method for separation. The sequence of separation may differ usefully from the sequence for fluorides; for example, zirconium tetrachloride and tin tetrachloride have relatively low boiling points of 331°C and 114.1°C. Chlorination has even been proposed as a method for removing zirconium fuel cladding, instead of mechanical decladding.

Chlorides are likely to be easier than fluorides to later convert back to other compounds, such as oxides.

Chlorides remaining after volatilization may also be separated by solubility in water. Chlorides of alkaline elements like americium, curium, lanthanides, strontium, cesium are more soluble than those of uranium, neptunium, plutonium, and zirconium.

**Economics**

The relative economics of reprocessing-waste disposal and interim storage-direct disposal has been the focus of much debate over the past ten years. Studies have modeled the total fuel cycle costs of a reprocessing-recycling system based on one-time recycling of plutonium in existing thermal reactors (as opposed to the proposed breeder reactor cycle) and compare this to the total costs of an open fuel cycle with direct disposal. The range of results produced by these studies is very wide, but all are agreed that under current (2005) economic conditions the reprocessing-recycle option is the more costly.

If reprocessing is undertaken only to reduce the radioactivity level of spent fuel it should be taken into account that spent nuclear fuel becomes less radioactive over time. After 40 years its radioactivity drops by 99.9%, though it still takes over a thousand years for the level of radioactivity to approach that of natural uranium. However the level of transuranic elements, including plutonium-239, remains high for over 100,000 years, so if not reused as nuclear fuel, then those elements need secure disposal because of nuclear proliferation reasons as well as radiation hazard.

* Recycled Nuclear Fuel Cost Calculator designed by the WISE Uranium Project

On 25 October 2011 a commission of the Japanese Atomic Energy Commission revealed during a meeting calculations about the costs of recycling nuclear fuel for power generation. These costs could be twice the costs of direct geological disposal of spent fuel: the cost of extracting plutonium and handling spent fuel was estimated at 1.98 to 2.14 yen per kilowatt-hour of electricity generated. Discarding the spent fuel as waste would cost only 1 to 1.35 yen per kilowatt-hour.

In July 2004 Japanese newspapers reported that the Japanese Government had estimated the costs of disposing radioactive waste, contradicting claims four months earlier that no such estimates had been made. The cost of non-reprocessing options was estimated to be between a quarter and a third ($5.5-7.9 billion) of the cost of reprocessing ($24.7 billion). At the end of the year 2011 it became clear that Masaya Yasui, who had been director of the Nuclear Power Policy Planning Division in 2004, had instructed his subordinate in April 2004 to conceal the data. The fact that the data were deliberately concealed obliged the ministry to re-investigate the case and to reconsider whether to punish the officials involved.

**List of sites**

|  |  |  |  |  |  |
| --- | --- | --- | --- | --- | --- |
| **Country** | **Reprocessing site** | **Fuel type** | **Procedure** | **Reprocessingcapacity tU/yr** | **Commissioningor operating period** |
| Belgium | Mol | LWR, MTR (Material test reactor) |  | 80 | 1966–1974 |
| China | intermediate pilot plant |  |  | 60–100 | 1968-early 1970s |
| China | Plant 404 |  |  | 50 | 2004 |
| Germany | Karlsruhe, WAK | LWR |  | 35 | 1971–1990 |
| France | Marcoule, UP 1 | Military |  | 1,200 | 1958-1997 |
| France | Marcoule, CEA APM | FBR | PUREX DIAMEX SANEX | 6 | 1988–present  |
| France | La Hague, UP 2 | LWR | PUREX | 900 | 1967–1974 |
| France | La Hague, UP 2–400 | LWR | PUREX | 400 | 1976–1990 |
| France | La Hague, UP 2–800 | LWR | PUREX | 800 | 1990 |
| France | La Hague, UP 3 | LWR | PUREX | 800 | 1990 |
| UK | Windscale | Magnox |  | 1,000 | 1956–1962 |
| UK | Sellafield, B205 | Magnox | PUREX | 1,500 | 1964 |
| UK | Dounreay | FBR |  | 8 | 1980 |
| UK | THORP | LWR | PUREX | 1,200 | 1990 |
| Italy | Rotondella | Thorium |  | 5 | 1968 (shutdown) |
| India | Trombay | Military | PUREX | 60 | 1965 |
| India | Tarapur | PHWR |  | 100 | 1982 |
| India | Kalpakkam | PHWR and FBTR |  | 100 | 1998 |
| India | Tarapur | PHWR |  | 100 | 2011 |
| Japan | Tokaimura | LWR |  | 210 | 1977 |
| Japan | Rokkasho | LWR |  | 800 | 2005 |
| Pakistan | New Labs, Rawalpindi | Military/Plutonium/Thorium |  | 80 | 1982–Present |
| Pakistan | Khushab Nuclear Complex, Atomic City of Pakistan | HWR/Military/Tritium |  | 22 kg  | 1986–Present |
| Russia | Mayak Plant B | Military |  | 400 | 1948-196?  |
| Russia | Mayak Plant BB, RT-1 | LWR | PUREX + Np separation | 400 | 1978 |
| Russia | Zheleznogorsk (Krasnoyarsk-26), RT-2 | VVER |  | 1,500 | under construction |
| USA, WA | Hanford Site | Military | bismuth phospate, REDOX, PUREX |  | 1944–present |
| USA, SC | Savannah River Site | Military/LWR/HWR/Tritium | PUREX, REDOX, THOREX, Np separation | 5000 | 1952–2002 |
| USA, NY | West Valley | LWR | PUREX | 300 | 1966–1972 |