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Nuclear Proliferation Technology Trends Analysis

M.D. Zentner
G.L. Coles
R.J. Talbert

September 2005

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NUCLEAR PROLIFERATION TECHNOLOGY TRENDS ANALYSIS

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TABLE OF CONTENTS

TABLE OF FIGURES

TABLE OF TABLES

1.0 INTRODUCTION

One approach to providing a basis for predicting and evaluating future proliferation events is to understand past proliferation events, that is, the different paths that have actually been taken to acquire or attempt to acquire special nuclear material. In order to provide this information this report, describing previous material acquisition activities (obtained from open source material) has been prepared.

This report describes how, based on an evaluation of historical trends in nuclear technology development, conclusions can be reached concerning:

1. The length of time it takes to acquire a technology
2. The length of time it takes for production of special nuclear material to begin
3. The type of approaches taken for acquiring the technology.

In addition to examining time constants, the report is intended to provide information that could be used to support the use of the different non-proliferation analysis methodologies. Accordingly, each section includes:

- Technology description
  - Technology origin
  - Basic theory
  - Important components/materials
- Technology development
  - Technological difficulties involved in use
  - Changes/improvements in technology
- Countries that have used/attempted to use the technology
- Technology Information
  - Acquisition approaches
  - Time constants for technology development
- Required Concurrent Technologies

2.0 TECHNOLOGY TRENDS ANALYSIS APPROACH

In this study, past approaches taken towards acquiring special nuclear material have been categorized as “enrichment technology based” (primarily oriented to acquisition of highly enriched uranium) and “reactor technology based” (primarily oriented towards the acquisition of plutonium). Many different technologies have been investigated and proposed for developing these materials. The technologies to be evaluated in this report are those that have received significant funding and development and have actually produced special nuclear material. These approaches are:

1. Enrichment technology
   - Gaseous Diffusion
   - Gaseous Centrifuge

- Electromagnetic Isotope Separation (EMIS)
- Chemical and Ion Exchange Enrichment
- Aerodynamic Isotope Separation
- Laser enrichment

2. Reactor based technology
    - Graphite moderated
    - Heavy water moderated
    - Research
    - Reprocessing

Some other enrichment technologies that have received some attention and evaluation include thermal diffusion, mass diffusion, and plasma separation. Of these, thermal diffusion was used in the early days of the Manhattan Project by the Unites States.; but due to cost and excessive power requirements, its use was discontinued. Similarly mass diffusion was investigated, but it too had excessive cost and power requirements and so was never developed on an industrial scale. The physics of plasma separation are still being studied, but nothing beyond laboratory experiments has been done. Consequently, these technologies are not evaluated in this study. Conventional commercial light water moderated reactors are also not evaluated, as they have not to date been used in proliferation programs.

A technology can be acquired in three ways, indigenous development, purchase, or covert acquisition. The following discussion summarizes the technical development of each of the approaches described above in the countries that have used or attempted to use it and evaluates the differences in development based on the acquisition method.

3.0 ENRICHMENT TECHNOLOGY

This section describes the various technologies that have been used in the production of enriched uranium. Commercial uranium enrichment programs are included in this section, as they have the potential to be modified for use in proliferation programs.

3.1 GASEOUS DIFFUSION ENRICHMENT TECHNOLOGY HISTORY

3.1.1 Technology Description

3.1.1.1 Origin

The first use of gaseous diffusion for isotopic separation was in 1920 to separate isotopes of neon. The first large scale gaseous diffusion plant to enrich U^{235} (using technology developed by the British in the early 1940's) was constructed during World War II. Called the K-25 plant, it was built at Oak Ridge Tennessee by the Manhattan Project and went into operation in 1945[1]. It was primarily used to create Highly Enriched Uranium (HEU).

3.1.1.2 Basic theory

The gaseous diffusion process makes use of the phenomenon of *molecular effusion* to effect separation. In a vessel containing a mixture of two gasses, molecules of the gas of lower weight have higher speeds and strike the walls of the vessel more frequently than the molecules of the gas with higher molecular weight. If the walls of the vessel (i.e. barrier) have holes just large enough to allow passage of molecules without permitting the flow of the gas in a continuous fluid, then more of the lighter molecules will flow through the wall, relative to concentration, than the heavier molecules. The flow of individual molecules through minute holes is known as *molecular effusion*.

The relative frequency with which molecules of different species enter a small hole is inversely proportional to the square root of their molecular weight. For example, for a mixture of $^{235}UF_6$ and $^{238}UF_6$ the separation factor (α) is 1.00429. Because this value is so small, the process must be repeated many times in a number of stages, to obtain a useful degree of separation. Such a series of stages is called a *cascade* of stages. The number of stages required to enrich uranium to 90% HEU is estimated to be 3,500 to 4,000.

Uranium hexafluoride (UF_6) is used in the process because it is the only known chemical compound suitable for the purpose. It is a solid compound at room temperatures but can be maintained as a gas under controlled temperature and pressure conditions.[2] Figure 1 displays the stage arrangement and various components in a U.S. gaseous diffusion plant.[3]

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[1] Benedict, Pigford, and Levi, "Nuclear Chemical Engineering", McGraw-Hill Book Company, 1981, Second Edition

[2] Final Environmental Impact Statement, Portsmouth Gaseous Diffusion Site, ERDA 1555, May 1977

[3] Technology and the Proliferation of Nuclear Weapons, Richard Kokoski, Sipri, Oxford University Press

3.1.1.3 Important components/materials

The main components of a gaseous-diffusion stage are:

1. A large cylindrical vessel called a diffuser or converter that contains the barrier (or diffuser),
2. A compressor used to compress the gas to the pressures needed for flow through the barrier,
3. An electric motor used to drive the compressor,
4. A heat exchanger to remove the heat of compression,
5. Piping and valves for stage and inter-stage connections and process control.

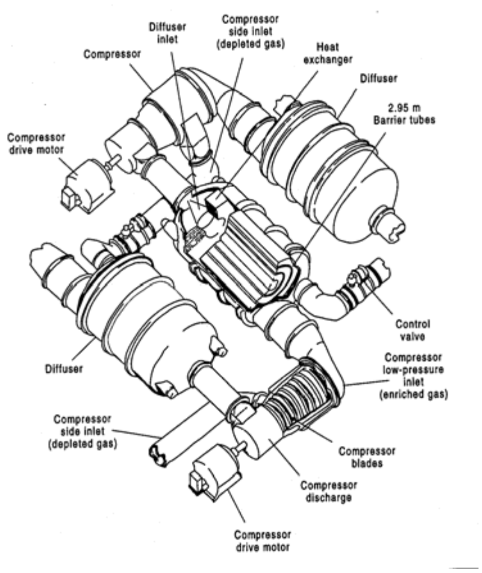

Figure 1 Stage Arrangement and components in a U.S gaseous diffusion plant

Table 1[4] identifies specific technologies used in gaseous diffusion enrichment facilities, describes the technology level needed for successful use, and lists critical materials and unique equipment used.

3.1.2 Technology Development

3.1.2.1 Technological difficulties in use

There are a number of technical difficulties with gaseous diffusion technology. These included the difficulties of making and maintaining a suitable barrier, large energy

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[4] Department of Defense, Militarily Critical Technologies List, Part II: Weapons of Mass Destructions Technologies, Section 5 – Nuclear Weapons Technology, February 1998.

Technology	Sufficient Technology Level	Critical Materials	Unique Test, Production, and Inspection Equipment	Unique Software and Parameters
Barrier materials	Thin, porous filters with small pore size (100 to 1,000 A), thickness of $\leq$5 mm, diameter $\leq$25 mm, sufficient mechanical strength, stable, chemically inert to UF_6	UF_6-corrosion resistant metallic, polymer or ceramic materials. Compounds and powders including nickel or alloys containing $\geq$ 60% nickel, aluminum oxide, fully fluorinated hydrocarbon polymers, etching acid such as HNO_3.	Scanning or transmission microscope, x-ray diffraction system, and other test equipment for measuring the following barrier properties: mechanical strength corrosion resistance, porosity and permeability.	Barrier performance models
Diffuser housings	Hermetically sealed cylindrical vessels >20-cm diameter. And >70-cm length (or comparable rectangular vessel) having inlet and outlet connections all >5-cm diameter, designed for operation at high vacuum, designed for horizontal or vertical installation	Nickel plated steel, aluminum, or nickel alloys containing $\geq$60% nickel; special UF_6 compatible gaskets for bolted flanges	None identified	None identified
Gas blowers and compressors	Axial, centrifugal, or positive displacement compressors/blowers with suction capacity $\geq$ $1m^3$/min of UF_6 environment. Pressure ration between 2:1 and 6:1	Nickel or high nickel alloy casing or plating on casing; rotor blades and impellers of same material or Al alloys.	UF_6 test loop and instrumentation to determine compressor performance characteristics	Compressor design and performance models and blade design codes for heavy gases.
Rotary shaft seals	Vacuum seals with seal feed and seal exhaust connections. Seals designed for a buffer gas in-leakage of <1,000 cm^3 /min. Adaptable to wide range of gas pressures and pressure disturbances, ease of maintenance, and UF_6 corrosion resistance.	Materials resistant to UF_6 corrosion.	Instrumentation to measure seal feed and exhaust pressures and flows to check seal performance.	Seal design and performance models for heavy gases.
Heat Exchangers	Heat exchangers made of, or lined with UF_6 – corrosion resistant materials, and intended for a leakage pressure change rate <10 N/m^2 (0.0015psi) per hour under a pressure difference of $100kN/m^2$ (15 psi).	UF_6 corrosion resistant materials	Test loop to determine heat transfer coefficients and pressure drop.	Heat transfer codes for compact heat transfer surfaces and heavy gases.
Feed systems	Process systems including feed autoclaves for passing UF_6 to the gaseous diffusion cascades and capable of operating at pressures $\leq$ 300 kN/M^2 (45psi). Cylinders and autoclaves ~3-m long and 1.8-m in diameter, and UF_6 corrosion resistant	UF_6 corrosion resistant materials	UF_6 mass spectrometers/ion sources. Autoclaves. UF_6 compatible flow, mass, pressure, and temperature instrumentations	None identified
Product and tails withdrawal systems	Compression liquefaction or desublimation (cold traps) systems for withdrawal. Cylindrical equipment is ~ 1 m in diameter. When insulated, and 2-3 m long. For HEU: diameter. 12.5 cm may include Boron alloys to preclude criticality.	Nickel, high-nickel alloys, aluminum, or copper	UF_6 mass spectrometers/ion sources. UF_6 compatible flow, mass pressure, and temperature instrumentation.	Compressor design codes and heat transfer design codes applicable to UF_6

Table 1 **Technologies and components used in gaseous diffusion enrichment facilities**

Technology	Sufficient Technology Level	Critical Materials	Unique Test, Production, and Inspection Equipment	Unique Software and Parameters
Header piping systems	Arrays of pipes $\geq$ 5 cm in diameter. Made of or lined with UF_6 resistant materials, normally of the double header system type, fabricated to very high vacuum and cleanliness standards, for handling UF_6 within the gaseous diffusion cascades	Materials resistant to UF_6 including stainless steel, aluminum, aluminum alloys, nickel, or alloys containing $\geq$ 60% nickel.	None identified	None identified
Vacuum systems	Large vacuum manifolds, vacuum headers, and vacuum suction pumps having a suction capacity of $5m^3$/min or more. UF_6 corrosion-resistant positive displacement vacuum pumps that may have special working fluids.	Aluminum, nickel or alloys containing $\geq$ 60% nickel. Hydrocarbon or fluorocarbon vacuum pump oils.	None identified	None identified
Shut-off and control valves	Manually or automatically operated, 5 mm or greater in nominal size, made of UF_6 –resistant materials.	UF_6 resistant materials. Bellows seals instead of packing glands because a bellow seal is the more effective technology.	None identified	None identified
Product storage and sampling cylinders	Cylinders designed for operation up to 30 atmospheres, with appropriate diameter and length to avoid criticality with HEU	Valves and connectors resistant to corrosion form UF_6	None identified	None identified

Table 1 **Technologies and components used in gaseous diffusion enrichment facilities (cont)**

Technology	Technical Issues
Barrier Materials	Fabrication of barrier. Maintain fine pore size, high permeability, and structural integrity over long periods of operation. Control non-separative flow mechanisms.
Diffuser Housings	Procurement of large quantities required, sealing and welding technologies, aerodynamic efficiency, minimum leakage and corrosion.
Gas Blowers and Compressors	Procurement of large quantities required, blade design, nozzle design, lubrication system for bearings, minimum leakage and corrosion.
Rotary Shaft Seals	Procurement of large quantities required, minimize in-leakage and out-leakage, long-term running reliability
Heat Exchangers	Minimize leakage and corrosion, cooling tower design
Feed Systems	Maintain material balance: reveal cascade leakage, consumption on surfaces or material freeze-outs
Product and Tail Withdrawal systems	Maintain material balance: reveal cascade leakage, consumption on surfaces or material freeze-outs. Criticality concerns with HEU.
Vacuum Systems	Minimize leakage. Containment and cleanliness.
Shutoff and Control Systems	Procurement of large quantities required, minimize leakage and corrosion, provide proper pressure drop to move UF6 inventory and minimize stage efficiency losses, isolation of stages for maintenance
Product Storage and Sampling Cylinders	Maintain operational integrity with minimum leakage and corrosion. Criticality concerns with HEU.

consumption, the requirement for procuring large quantities of specialized stage equipment, large in-process inventory requirements, and long process times. Table 2[4] lists important technologies used in gaseous diffusion enrichment and associated technical issues.

Table 2 <u>Technology and technical issues involved with gaseous diffusion enrichment</u>

3.1.2.2 <u>Changes/improvements in technology</u>

Once a gaseous diffusion plant is designed, constructed, and put into successful operation, evolutionary changes tend to occur at widely-spaced intervals. The first facility that showed significant changes was the French “Eurodif” gaseous diffusion facility, which was an improvement on the American and British design. The components were arranged more compactly, greatly reducing the length of interconnecting piping and consequent required floor area and building space. Additional improvements included development of new diffusion barrier technologies.

Beginning in 1977, the U.S. diffusion plants underwent a major revision. New diffusion barriers were installed allowing operation at higher gas pressure and throughput. Gas compressors were improved via upgrades in blades, flow paths, stators, and nozzles. These modifications reportedly improved productivity by a factor of twenty-three[5].

3.1.3 Countries that have used/attempted to use technology

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[5] Topics in Applied Physics: Uranium Enrichment, S. Villant, Editor, Springer-Verlag, 1979

Table 3 lists the fourteen countries that have expressed interest in acquiring gaseous diffusion enrichment technology. Of these, five had a successful indigenous program, and one (China) purchased the technology.

Argentina[2]	Italy[2]
Belgium[1]	Japan[1]
China[3]	The Netherlands[2]
France[3]	Soviet Union[3]
Germany[2]	Spain[1]
Iran[1]	United Kingdom[3]
Iraq[1]	United States[3]

[1]Research and development, [2]Pilot plant, [3]Industrial facility

Table 3 <u>Countries interested in gaseous diffusion enrichment technology</u>

Table 4 lists the six countries that have had successful gaseous diffusion programs, and the time frames involved in the completion of the initial phase of these programs for each country. All of these facilities had significant problems in initial operation. So, the dates shown are of the first reported throughput and not necessarily the first successful production of HEU. Most of theses facilities (except in Argentina, Section 3.1.3.6) were initially intended to produce weapons material rather than for commercial production, were national priorities, and were funded as such.

3.1.3.1 <u>United States</u>

The United State had constructed three gaseous diffusion enrichment facilities by 1954. Originally intended to supply highly-enriched uranium for weapons programs, these facilities operated for over forty years.

Construction of the first U.S. facility began in 1943, with small quantities of enriched uranium available in 1945. The plant, however, did not produce significant quantities of HEU until about 1947.

Two later plants, Paducah (began construction 1950, first product 1952) and Portsmouth (began construction 1952, first product 1956), began production of low-enriched uranium for use in commercial nuclear fuel in the mid 1960's[6]. During most of their operating life, these facilities operated at over a 90% plant capacity factor[6]. By 1976, these gaseous diffusion plants had provided more than 90% of the enriched uranium for the western world's nuclear power plants.

As with all initial or early gaseous diffusion facilities, there were significant initial problems to be solved before operations could proceed. Currently, only one U.S. gaseous diffusion plant remains in operation.

Time from initial construction to first operation ranged from two to four years; for the first facility, time from construction start to HEU was four years.

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[6] United States Enrichment Corporation (USEC) Facilities fact sheet.

Country	Program initiation	Facility operational*	Years to operation
United States	1942	1945	3
Soviet Union	1946	1951	5
United Kingdom	1950	1954	4
China	1958	1964	6
France	1960	1964	4
Argentina	1979	1986	7

*Time to first enrichment output

Table 4 Successful gaseous diffusion enrichment programs

3.1.3.2 Soviet Union

The Soviet Union began construction of its first enrichment plant, known as D-1 at Sverdlovsk-44, as a part of its weapons program in 1946[7]. Construction was complete in 1948; but, because of a number of technical problems (primarily corrosion and leakage), it did not produce HEU until 1951. This first facility was shut down in 1955 and was reportedly dismantled, with components sent to China for use in the Chinese enrichment program.

The Soviet Union had an aggressive program for enhancing and improving diffusion technology. The second plant at the Sverdlovsk-44 facility, D-3, was built between 1950-1951 using equipment similar to that in the D-1 plant and implemented improvements identified during the efforts to solve the initial problems at D-1. Using information gained from these two facilities, two more plants were constructed (D-4 and D-5) that used new technological advances in diffusion stage technology. Throughput and energy efficiency were greatly improved. The largest of the final diffusion stages had a separation capacity of 850 swu/yr. This can be compared to the production of the smallest stages of the first generation of 0.13 swu/year[8]

By the end of 1953, approximately fifteen thousand diffusion stages were operating at Sverdlovsk-44. Three additional facilities were constructed between 1949 and 1964 at Krasnoyarsk-45, Angarsk, and Tomsk-7.

Primarily because of its enormous energy requirements and the availability of the more efficient, gas-centrifuge machines, the Soviet Union halted gaseous diffusion enrichment in 1991.

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[7] Stalin & the Bomb, The Soviet Union and Atomic Energy, 1939-1956, David Holloway, Yale University Press, 1994

[8] The History of the Soviet Atomic Industry, Arkadii Kruglov, Taylor and Francis Press, 2002

As with the United States, time from initial construction to first operation took a minimum of two years but because of process difficulties with the initial equipment, it took an additional three years for production of HEU.

The total time from construction start to HEU production at the D-1 facility was five years.

3.1.3.3 United Kingdom

The U.K. began construction of a gaseous diffusion plant as a part of its weapons program in 1950. The plant, completed by early 1952, did not produce LEU until 1953 and HEU until 1954. The facility's capacity was tripled in 1959. In 1969, production was switched to LEU. In the 1970's, the facility was used to "re-enrich" depleted uranium until 1980, when the process became uneconomical due to falling "fresh" uranium prices[9]. The facility was decommissioned in 1982.

Similarly to the United States and the Soviet Union, the first facility took two years to construct, and then took another two years to produce HEU.

The time from construction start to HEU production was four years.

3.1.3.4 China

Instead of developing an indigenous gaseous diffusion technology, China purchased surplus equipment from a Soviet designed facility (Section 3.1.3.2). Construction began in 1958. In 1960, with the plant nearly completed, political issues caused Soviet technicians to leave China and Chinese technicians began modification of the original Soviet plans to simplify the design and shorten construction time.

Construction was essentially complete in 1961, but because the Chinese had to develop expertise in operations and maintenance the facility did not produce 90% HEU until 1964[10].

Plant construction took three years, and the time to produce HEU once construction was complete was three years.

3.1.3.5 France

Construction of the first French gaseous-diffusion facility for military purposes began in 1960. The plant was constructed in phases, with the final phase being complete in 1967. As with other such facilities, problems occurred in construction and operation, but the initial production of LEU occurred in 1964.

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[9] "That Secret Factory" Dismantled, Financial Times (London), 11/3/82

[10] China Builds the Bomb, John Wilson Lewis and Xue Litai, Stanford University Press, 1988

Construction began on a second facility for commercial purposes in 1974, which met its planned completion date of 1979. The facility has a capacity of 10 million swu/yr, and requires 2400 megawatts to operate. Four nuclear reactors were constructed to provide power.

The second facility underwent major upgrades in the 1980's. It can now operate in a load-following mode from full capacity to 20%, adapting to peaks and dips in power availability in order to take advantage of cheap electricity[11].

The first French facility took seven years to be fully complete, but only four to begin initial operation. The second took five years.

3.1.3.6 Argentina

In 1983, Argentina revealed that it had been secretly constructing a gaseous-diffusion enrichment facility. The facility was reported to be intended to produce 20% HEU. Construction on the facility had begun in 1979, and it was originally planned to be in operation by 1985. Continued delays occurred, caused by both economic and technical issues[12], but the plant began to produce in 1987. However, by 1990, production delays occurred, and in 1994 the plant was shut down for repairs. The facility started up and operated briefly but was closed in 1997.

In 2000, Argentina announced development of a "revolutionary" new gaseous diffusion process known as the Sigma process. The technology is said to overcome the main drawback of the old military-based diffusion technology: the need to build huge plants to achieve economy because of the low efficiency of compressors at low-flow stages. A pilot-scale facility based on the Sigma technology has been built to demonstrate the components and some engineering innovations.[13]

It took about seven years after plant construction began for the first enriched uranium to be produced, but the faculty never successfully operated.

3.1.4 Technological Information Acquisition/Development

3.1.4.1 Acquisition approaches

Of the six countries that successfully developed gaseous diffusion technology, five had indigenous programs and one purchased the information to develop the technology. One of the indigenous programs (Argentina) was originally surreptious, with apparently most equipment developed internally. When Argentina revealed its program, it was considered a “startling and dismaying failure of intelligence gathering”.[14]

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[11] Cogema Tries Again on Centrifuge Development, this time with Japan, Mark Hibbs, Ann MacLachlan, Nuclear Fuel: Vol. 23, No. 22; Pg 1, 11/2/1998

[12] Financial Problems for Argentina's Nuclear Fuel Reprocessing Programme, Noticias Argentinas

[13] Argentina seeking partners to develop 'revolutionary' diffusion technology, Ann MacLachlan, Nuclear Fuel, Vol. 25, No. 8; Pg. 5, 4/17/2000

[14] Argentine Enrichment Pronouncement Characterized As Startling, Richard Kessler and Michael Knapik, Nucleonics Week, Vol. 24, No. 47; Pg. 1, 11/24/1983

3.1.4.2 Time constant for technology development

Table 4 shows that the average time for completion of a gaseous diffusion facility is about five years. This includes two to three years to complete construction and another two to three years for successful operation to begin.

3.1.5 Required Concurrent Technologies

Materials needed to develop gaseous diffusion enrichment processing facilities can be grouped into several categories. Governmental and international agreements control the acquisition of many of these materials[15, 4].

3.1.5.1 Fluorine-related equipment

- Fluorine specific equipment such as fluorine generators.
- Process related equipment such as
    - Fluorine compatible compressors,
    - Corrosion resistant valves,
    - Corrosion resistant piping,

3.1.5.2 Diffusion related equipment

- Diffuser housings
- Diffusion barriers

3.1.5.3 Auxiliary equipment

- Compressors,
- Gas coolers,
- Motors,
- Rotary shaft seals,
- Heat exchangers,
- Water supplies,
- Power supplies,
- Uranium Feed systems,
- Uranium Withdrawal systems

In addition to UF_6 production facilities, other necessary technologies include a large electric power distribution system, cooling towers to dissipate waste heat, a steam plant, a diffusion barrier production plant, and a plant to produce dry air and nitrogen[4].

3.2 CENTRIFUGE ENRICHMENT TECHNOLOGY HISTORY

3.2.1 Technology Description

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[15] LA-13131-M, A Handbook for the Nuclear Suppliers Group Dual-Use Annex, April 1996

3.2.2.1 Origin

The first suggestion for separating isotopes by use of a centrifuge occurred in 1919. The method was developed at the University of Virginia. In 1938, the concept was proven through the separation of isotopes of chlorine. In 1941, 1.2 grams of 4% enriched uranium was produced on prototype machines.

German scientists in the Soviet Union, in 1945, developed modern centrifuge technology. The first Soviet pilot centrifuge enrichment plant was completed in 1953, the first demonstration plant in 1957, and the first full-scale facility was completely operational by 1964.

After the success of the Soviet Union's program, the German scientists were allowed to return to Germany, and in 1960 a seminal paper was presented that revealed the new technological approach taken by the Soviet Union's centrifuge program. At this point, a number of countries, especially the United Kingdom, The Netherlands, and Germany began their own centrifuge programs.

3.2.1.2 Basic theory

In the gas centrifuge uranium-enrichment process, gaseous UF_6 is fed into a cylindrical rotor that spins at high speed inside an evacuated casing. Because the rotor spins so rapidly, centrifugal force results in the gas occupying a thin layer next to the rotor wall, with the gas moving at approximately the speed of the wall. Centrifugal forces cause the heavier $^{238}UF_6$ molecules to tend to move closer to the wall than the lighter $^{235}UF_6$ molecules, thus partially separating the uranium isotopes. This separation is increased by a relatively slow axial countercurrent flow of gas within the centrifuge that concentrates the relatively lighter enriched gas at the top of the centrifuge and the relatively heavier depleted gas at the other. This flow can also be driven mechanically by scoops and baffles or thermally by heating the bottom end cap.

The separating capacity of a single centrifuge increases with the length of the rotor and the rotor wall speed. Consequently, centrifuges containing long, high-speed rotors are the goal of centrifuge development programs.

3.2.1.3 Important components/materials

The main subsystems of the centrifuge (Figure 2)

- Rotor and end caps
- Top and bottom bearing/suspension system
- Electric motor and power supply (frequency changer)
- Center post, scoops and baffles
- Vacuum system
- Outer casing

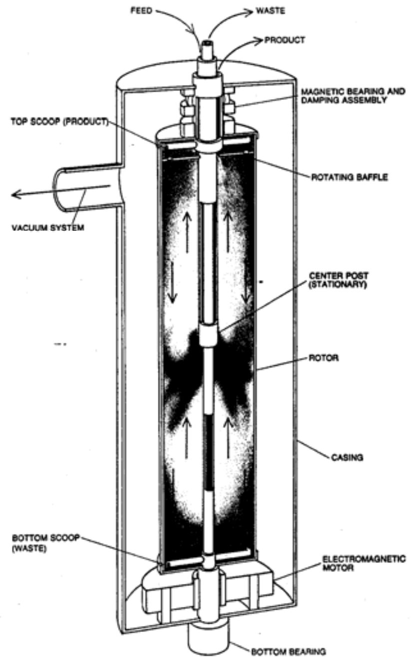

Figure 2 Sub-critical gas centrifuge components[16]

Because of the corrosive nature of UF$_6$, all components that come in direct contact with it must be fabricated from or lined with corrosion-resistant materials.

The power supply is a key component of the gas centrifuge process. The power supply (frequency converter) must accept alternating current input at the 50 or 60 Hz line frequency of the electric power grid and provide an ac output at a much higher frequency (typically 600 Hz or more). The high frequency output from the frequency converter is fed to the high-speed gas centrifuge drive motors (the speed of an ac motor is proportional to the frequency of the supplied current). The centrifuge power supplies must operate at high efficiency, provide low harmonic distortion, and ensure precise control of the output frequency.[16]

A variety of materials have been used for centrifuge construction. The three most common (in order of increasing tensile strength) are high tensile strength aluminum alloys, maraging steel, and carbon fiber/resin. Specific design issues are discussed in Section 3.2.2.1.

Table 5 identifies specific technologies used in centrifuge enrichment, describes the technology level needed for successful use, and lists critical materials and unique equipment.

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[16] Scientific American, Vol., No. 2, August 1978, Pg 29

3.2.2 Technology Development

3.2.2.1 Technological difficulties in use

As will be seen in Section 3.2.3, a number of countries have tried, with varying success, to develop centrifuge based uranium enrichment. The following section discusses some difficulties in developing this technology

The rotational speed and the length of the rotor determine the separative power of a gas centrifuge. The separative power increases rapidly with rotor speed and is proportional to its length.

The peripheral speed of the rotor is limited by the ratio of strength to density of the material with which it is made. Aluminum alloys are capable of maximum peripheral speeds to 425 m/s, maraging steel has an approximate maximum speed of 525 m/s, and carbon fiber can reach 700 m/s.[17]

The limits to rotor length include:

1. Critical speeds

Every spinning object encounters *critical speeds* as it accelerates and the greater its length, the more quickly it reaches them. When it reaches the first critical speed, a cylindrical rotor bends outward like a banana. At the second critical speed, it bends like an S. As more and more critical speeds are reached, the rotor bends into more and more curves. Designers of earlier centrifuges considered the first critical speed an insurmountable barrier, because the rotor wobbled and crashed when it reached this speed. They built machines with short rotors that could attain high velocities without running into any critical speeds. These are called "subcritical" centrifuges. Later improvements in centrifuge technology provided means for passing the first critical speed. These improvements included dividing longer centrifuges into several sections that allowed centrifuges to “bend”. Centrifuges that can exceed at least the first critical speed are called “supercritical” centrifuges[18].

2. Manufacturing processes

Because of the very high rotational speeds of centrifuge rotors, quality control is vital for successful operations. Minor variations in the straightness of the rotor or wall thickness will result in early catastrophic rotor failure.

3. Durability of bottom bearing

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[17] Uranium Enrichment and Nuclear Weapon Proliferation, Krass, Boskma, Elzen, and Smit; Stockholm International Peace Research Institute, 1983

[18] In the beginning was uranium, Dan Charles, New Scientist, Vol. 136 ; No. 1844 , 10/24/92

The bottom bearing must support the weight of the rotor, efficiently supply a “frictionless’ surface area, and have an extended lifetime. With a design lifetime of 10 years, a high rotational speed (1200 rpm or more), and the small bearing size, it is an important and critical component.

Table 6 lists important components used in centrifuge enrichment, and associated technical issues.

3.2.2.2 Changes/improvements in technology

Seven countries have successfully developed and utilized centrifuge technology over the last fifty years. Two primary approaches have been used, a great many relatively low enrichment *sub-critical* centrifuges or a lesser number of super-critical centrifuges that pass at least the first centrifuge critical speed. A variety of improvements and evolutionary changes have occurred in materials used, electronic speed control, bearing design, and internal improvements.

1. Critical speeds

Two basic approaches are taken to overcome the critical speed issue. The Soviets, the first to use centrifuge technology, selected the sub-critical approach. They used aluminum alloys and built machines about 50 cm tall and 50 mm in diameter. The original throughput was small, approximately 1 swu/yr or less. Accordingly, it took many thousands of centrifuges to meet production requirements.

The second approach is to build ‘super-critical’ machines. Two approaches (sometimes combined) are taken here. In the first approach, flexible joints, called bellows, are built into the rotor, so it could bend freely with the dynamic forces and adopt new shapes as it accelerated through each critical speed. In the second approach, new materials, primarily carbon fiber, are used. These have a much higher tensile strength (almost twice that of aluminum alloys) and can spin at much higher speeds.

2. Manufacturing processes

Modern computer numerically-controlled machine tools in specially constructed "clean rooms” must be used. This ensures that the large quantity of high-precision parts needed to produce components for high-speed centrifuges are of sufficient quality to ensure long-term successful operation.

Technology	Sufficient Technology Level	Critical Materials	Unique Test, Production, and Inspection Equipment	Unique Software and Parameters
Rotating Component: (Complete Rotor Assemblies)	Thin-walled cylinders (>30 cm in length) or interconnected thin-walled cylinders up to 15 m in length made from high strength-to-density ratio material.	High strength-to- density ratio (HSD) materials: maraging steel, high-strength aluminum alloys, filamentary materials suitable for use in composite structures.	Equipment to manufacture, assemble, and balance complete rotor assembly.	Rotor dynamics/stress analysis software
Rotating Component: Rotor Tubes	Thin-walled cylinders wall thickness $\leq$ 12 mm, diameter: 75 to 400 mm thick, made from high I strength-to-density material, length-to-diarneter ratio typically >2	HSD materials: maraging steel, high-strength aluminum alloys, filamentary materials suitable for use in composite structures.	Equipment to manufacture and balance rotor tubes; spin-forming and flow-forming machines, filament winding machines. Spin-testing equipment.	Rotor dynamics/stress analysis software
Rotating Component Rings or Bellows	Cylinder of wall thickness ~3 mm, diameter 75 to 400 mm, made of high strength-to-density ratio material, and having a convolute. Used to provide local support to rotor tube or to join rotor tubes.	HSD materials: maraging steel, high-strength aluminum alloys, filamentary materials suitable for use in composite structures.	Equipment to manufacture and balance rings and bellows. Spin-testing equipment.	Rotor dynamics/stress analysis software
Rotating Component Baffles	Disc-shaped high strength- to-density ratio components, 60 to 500 mm in diameter, designed to be mounted in rotor tubes to isolate take-off chamber of rotor tube and/or to assist UF6 gas circulation in main separation chamber.	HSD materials: maraging steel, high-strength aluminum alloys, filamentary materials suitable for use in composite structures.	Equipment to manufacture and balance baffles. Spin-testing equipment.	Rotor dynamics/stress analysis software
Rotating Component top caps/bottom caps	Disc-shaped or cup-shaped HSD components, 75 to 400 mm in diameter, designed to fit the ends of rotor tubes, contain the UF6, within the rotor, and support the upper bearing elements or to carry rotating elements of motor	HSD materials: maraging steel, high-strength aluminum alloys, filamentary materials suitable for use in composite structures.	Equipment to manufacture and balance end caps. Spin-testing equipment.	Rotor dynamics/stress analysis software
Static Component: Magnetic Suspension Bearings (includes ring magnets)	Homogeneous ring-shaped annular magnet suspended within UF6 resistant housing, deviation of the magnetic axes from the geometrical axes limited to very small tolerances	Ring magnet, samarium-cobalt Alnico	Precision balancing and magnetic properties measuring equipment.	None identified
Static Component: Bearings, Dampers (for lower end of rotor tube)	Bearing comprised of pivot cup assembly mounted on a damper. Pivot is normally hardened steel shaft polished into a hemisphere. Cup has a hemispherical indentation in one surface. Shaft may. have hydrodynamic bearing	Hardened or maraging steel, stainless steel, aluminum having high-quality machined surface	None identified	None identified

Table 5 **Technologies and components used in centrifuge enrichment facilities**

Technology	Sufficient Technology Level	Critical Materials	Unique Test, Production, and Inspection Equipment	Unique Software and Parameters
Static Component: Molecular Pumps	Cylinders having internally helical grooves and internally machined bores. Grooves are typically rectangular in cross section.	Steel, stainless steel, aluminum	Precision manufacturing and mensuration equipment.	None identified
Static Component: Motor Stators	Ring-shaped stators having multiphase windings on low- loss laminated iron core for synchronous operation of AC hysteresis motors in vacuum. Power range is 50 to 1,000 VA for frequencies 600 to 2,000 Hz.	Low-loss iron core	Precision manufacturing of laminated structure, coil winding and mounting.	Motor design software for unusual motor geometries and high frequency operation.
Static Component: Scoops	Tubes up to 12 mm (0.5 in) internal diameter for extraction of UF 8 from within the rotor tube by Pitot tube action and capable of being fixed to the central gas extraction system	UF_6 resistant materials	None identified	CFD codes for heavy gases in strong rotation with shocks.
Feed Systems/Product and Tails Withdrawal Systems	Feed autoclaves that pass UF6 to centrifuge cascades, desublimers that remove UF6 from the cascades, product and tails stations for trapping UF6 into containers.	UF_6 resistant materials used in piping	Mass spectrometers/ion sources. Autoclaves. UF6 compatible flow, mass, pressure, and temperature instrumentation.	Heat transfer codes applicable to UF6 desublimers.
Machine Header Piping System	Piping network normally of the “triple” header system with each centrifuge connect to each of the headers. Line connections at the centrifuge may be individually flanged or combined in a single flange	UF_6 resistant materials used in piping	Fabrication techniques applicable to very high vacuum and cleanliness standards.	None identified
Frequency changers (also called converters or inverters)	Multiphase output capable of providing an output of >40 W, operating in the range of 600 to 2,000 Hz, high stability with frequency control <0.1 %, harmonic distortion <10%, high efficiency, large MTBF, ability to drive on or more centrifuges	None identified	None identified	None identified

Table 5 Technologies and components used in centrifuge enrichment facilities (cont.)

Technology	Technical Issues
Rotating Component: Complete Rotor Assemblies	Rotor dynamics, critical frequencies, proper balancing and damping, continuous operation
Rotating Component: Rotor Tubes	Material properties, balancing resistance to corrosion attack, continuous operation, uniformity of manufacture
Rotating Component: Rings or Bellows	Material properties, balancing resistance to corrosion attack, continuous operation, uniformity of manufacture
Rotating Component: Baffles	Material properties, balancing resistance to corrosion attack, continuous operation, uniformity of manufacture
Rotating Component: Top Caps/Bottom Caps	Material properties, balancing resistance to corrosion attack, continuous operation, uniformity of manufacture
Static Component: Magnetic Suspension Bearings (includes ring magnets)	Homogeneity of magnet material, deviation of magnetic axes
Static Component: Bearings, Dampers (for lower end of rotor tube)	Probe damping to control rotor vibration and restrain later movement
Static Component: Molecular Pumps	Maintain low pressure in casing
Static Component: Motor Stators	Provide low-loss, high speed, high frequency, synchronous and uninterrupted service.
Static Component: Scoops	Aerodynamics and materials
Feed Systems/Product and Tails Withdrawal System	Maintain material balance. Criticality concerns with HEU.
Machine Header Piping System	Minimize leakage and corrosion, sealing, and welding technologies
Frequency Changers (also called converters or inverters)	Trouble-free operation for extended periods of operation, no maintenance requirements

Table 6 Technology and technical issues involved with centrifuge enrichment

3. Bearing development

The bottom bearing is a difficult part of the centrifuge to perfect. The rotor of a centrifuge is attached to a thin metal pin that widens to a ball at its end, a few millimeters wide. Adequate lubrication systems must be provided to prevent wear and ensure smooth operation.

Design of the top bearings took another turn. Ring magnets are used to hold the top of the rotor steady without any physical contact. This magnetic bearing creates a small opening at the top of the centrifuge that allows an entry and exit point for the UF_6 gas.

3.2.3 Countries that have used/attempted to use technology

Table 7 lists the eighteen countries that have expressed some interest in acquiring centrifuge enrichment technology. Of these, seven had a successful indigenous program, five attempted to develop it (one successfully) using a covert acquirement program, and four investigated the approach but for one reason or another halted development. The following discussion summarizes each countries program.

Australia[6]
Brazil[1,2]
China[2]
France[6]
Germany[1,4]
India[1]
Iran[3]
Iraq[3]
Japan[1]
Lybia[3]
Pakistan[3]
North Korea[6]
South Africa[5]
Soviet Union[1]
Syria[3,6]
The Netherlands[1,4]
United Kingdom[1,4]
United States[5,6]

[1]Successful indigenous development
[2]Purchase technology
[3]Covert acquisition
[4]Unrenco Participant
[5]Development halted
[6]Recent or past interest

Table 7 Countries interested in Centrifuge technology

Table 8 shows the estimated timeframes for the various successful indigenous centrifuge enrichment programs. It should be noted that the dates are approximate, because reports from which the information was extracted were not always consistent. However, these dates are "best estimates" and are accurate within a year or two.

Country	Program Initiation	First Demonstration Facility* Operational	Time to First Demonstration Facility (yrs)	First Full Scale** Facility	Time to first full scale facility
Soviet Union	1945	1953	8	1957	12
Urenco	1960	1971	11	1973	13
Japan	1971	1979	8	1982	10
Pakistan	1974	1980	6	1984	10
Brazil	1980	1990	10	2002	22
India	1975	1985	10	1992	17
Iran	1987	2004 (?)	17	?	?

*Linked cascades of 100 or more centrifuges producing enriched uranium
**Facility producing 5,000 or more swu/yr

Table 8 Successful indigenous centrifuge enrichment programs

3.2.3.1 Soviet Union

The first successful centrifuge enrichment program was created in the Soviet Union. The program was initiated in 1945, and a successful pilot plant was in operation by 1953. Although the Soviets had plans and prototypes for super-critical centrifuges, the decision was made to use sub-critical centrifuge technology at the beginning of the program, and this remains the primary technology choice today.

The original purpose of the Soviet program was to develop the capability to produce highly enriched uranium for their weapons program, because their gaseous diffusion plants did not function well.

The first production centrifuges were of aluminum sub-critical design, about fifty cm in height, and had an estimated original throughput of less than one swu/yr. The first pilot plant was constructed in 1953, the first demonstration plant in 1957, and the first full scale plant in 1959. However, stable and productive output of centrifuge cascades did not occur until Soviet engineers mastered the dynamics of ultra-high-speed rotors and creation of fluorine resistant materials[19]. Four facilities were built over the last fifty years and continue to operate today.

Seven generations of aluminum centrifuges were developed and put into operation, with the last generation having a throughput of eight to ten swu/yr. An eighth generation carbon fiber centrifuge is currently under development that has an expected throughput of ten to fifteen swu/yr[20].

As new generations of centrifuges were developed and installed, the Soviets stored the replaced units. In 1989, they were reported to have an excess of centrifuge machines (several warehouses full) and were willing to sell them to a suitable company[21]. Russia agreed in 1993 to build a centrifuge enrichment plant in China[22] reportedly using these machines.

Analysis of centrifuge development shows that the Soviets (now Russians) averaged about six years between centrifuge generations.

3.2.3.2 Urenco

In 1964, the first commercial company intended to develop a reliable and economic centrifuge for uranium enrichment on an industrial scale was formed in Germany. In March 1970, Germany, The Netherlands and the U.K. signed the Treaty of Almelo, which was the basis for collaboration between the three countries for the development and industrial exploitation of centrifuge technology to enrich uranium.

For a variety of reasons, the three countries decided to consolidate national programs that, while using concepts developed in the Soviet Union, were based on different choices of material, diameter, length (sub- to super-critical), and operating speed.

In 1971, the three countries had pilot plants under construction, one in the U.K. and two (Dutch and German) in The Netherlands. These pilot plants were built in two phases; the first with designs from the national programs, the second with modifications to these designs incorporating improvements identified by comparing technical variations

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[19] Soviet Official Says Centrifuge Process Allows TSE Flexibility for SWU Sales, Mark Hibbs, Nuclear Fuel, Vol. 14, no. 20, pg 6, 10/2/1989

[20] Minatom says its centrifuge plants are competitive with those of URENCO, Mark Hibbs, Nuclear Fuel, Vol. 17, No. 22; Pg.3, 10/26/1992

[21] Soviet Centrifuge Capacity said to free more low-cost SWU for Western markets, Michael Knapik, Nuclear Fuel: Vol. 14, no. 13; pg. 1, 6/26/1989

[22] Nuclear Fuel, Vol. 22, No 20: pg 3, Mark Hibbs, 10/6/1997

between the three programs. Initial production from these pilot plants began in 1972 and full production was reached by early 1976[23].

From these programs, two were selected to construct initial facilities. One, using subcritical aluminum centrifuges of Dutch design was constructed in the United Kingdom; the other subcritical machine of German design using maraging steel was constructed in The Netherlands.

Urenco machines have been used exclusively for commercial nuclear fuel production. However, in 1982, the United Kingdom began construction of a facility to be used to enrich to 20% HEU for use in military programs. This facility (Capenhurst A-3) took five years to complete and put into production. It produced 20% HEU for seven years, when its military production role ended and commercial production began.

Urenco, which has the worlds most sophisticated and successful centrifuge enrichment program, has developed six distinct generations of centrifuges examples of which are listed below:

- Original models
  - 1977 CNOR: aluminum subcritical centrifuges (Dutch model); 0.5-1.0 swu/yr. SNOR: aluminum supercritical centrifuges (Dutch model); 1-2 swu/yr,
  - 1973 G-1 maraging steel subcritical centrifuge (German model), 1-2 swu/yr
- G-2/G-3 model
  - Developed in the Mid-70's were the first operational supercritical centrifuges using maraging steel, the G-2 was essentially two G-1 rotors connected by bellows, and was capable of up to 5 swu/yr (the G-3 model was three G-1 rotors connected by bellow and was never used commercially)
- TC-11 model
  - 1987 carbon fiber rotors, output 50% greater (5-8 swu/yr), capital costs ⅓ of G-2 model
- TC-12 model
  - 1991 50% more output than TC-11, power costs ¼ of original (80 Kwh/swu vs. 20 Kwh/swu)
- TC-21
  - 1999 double the capacity of the TC-12, output 30 times of original centrifuges, ten times longer in length, and rotates twice as fast.

Urenco reportedly spends about eight years developing each generation. Urenco machines are designed for a "no maintenance policy" [24] and have an excellent failure rate

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[23] "Status Report On Urenco's Progress and Plans", J.V.L. Parry, Recent Developments in Uranium Enrichment, American Institute of Chemical Engineers Symposium Series, 221, Volume 78, 1982.

[24] Centrifuge technology: The Future for Enrichment, Pat Upson, World Nuclear Association Annual Symposium 2001.

history, with only a few per thousand failing over twenty years of continuous operation[25]. The design output has been better than expected, 80% above that predicted for a 10 year lifetime.

3.2.3.3 Japan

Japan had expressed an interest in centrifuge enrichment as early as 1959, but did not begin a national centrifuge enrichment development program until 1976. Japan's program was aggressive, with announced plans to build a centrifuge demonstration plant with 500,000 SWU/yr plant on line by 1984, a 1 Million SWU/yr plant by 1988, and a 4 million SWU/yr capacity by 1995[26].

Japan's original design was a subcritical maraging steel centrifuge. As the program evolved, three new models were introduced during an eight year period from 1977 to 1985.

Although the original construction dates were primarily met, centrifuge failure rates were reported to be much higher than expected. It was determined that one significant, life-limiting cause was a defective end cap design. This allowed corrosive UF_6 to plate out on the end caps, resulting in stress corrosion cracking and early machine failure.

Analysis showed that it was possible to redesign the end caps, but developing and qualifying the design, bench testing machines, licensing, and production would take nine years. It was determined that a "second generation," more powerful supercritical machine using advanced materials could be developed and put into commercial operation in ten years[27].

Japan has had a centrifuge enrichment program for about 25 years. Significant design issues have arisen as their facility ages. Failure rates for their first commercial production line were about 1% failure in 5 years. The current advertised time to develop and put a new centrifuge design into operation is ten years.

3.2.3.4 Brazil

Brazil's centrifuge enrichment program was a project of the Brazilian navy. In 1975, Brazil signed agreements with West Germany to transfer nuclear technology. By 1980, two pilot plants had been constructed, probably using West German technology. The pilot plant machines were based on early German sub-critical designs using Brazilian maraging steel. They reportedly had a throughput of about 1-2 swu/yr[28]. By 1990, a demonstration plant of about one thousand centrifuges of a new, subcritical design using carbon fiber with a throughput of 3-5 swu/yr was in operation and reportedly able to enrich to 20%[29]. In 2000, Brazil announced plans to enrich uranium on a commercial

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[25] Urenco Building up SWU capacity to handle growing market share, Mark Hibbs, Nuclear Fuel, Vol. 22, No. 10; Pg.1, 5/19/1997

[26] More power for uranium enrichment, Chemical Week, pg 33, 7/21/1978

[27] JNFL opts for advanced centrifuge after examining end cap corrosion, Mark Hibbs, Nuclear Fuel, Vol. 25, No.26; Pg. 4, 12/25/2000

[28] Brazil to Build Centrifuge plant at Resende Fuel Processing Complex, Mark Hibbs, Nuclear Fuel, Vol. 25, No. 14, Pg. 1, 7/10/2000

[29] Nuclear Fuel, Vol. 15, No 15: pg 4, Richard Kesler, 7/23/1990

scale, using the centrifuge designs developed by the Navy. The commercial enrichment plant is based on a third generation super critical centrifuge with a carbon fiber rotor. Throughput is expected to be 5-10 swu/yr.

It took the Brazilians eighteen months to construct the initial phase of the facility, and it began commercial operation in 2002. They plan to be able to produce 20,000 swu/yr in the initial phase, reaching full capacity in 2007.

Brazil had three distinct centrifuge generations, each separated by about a ten-year development program. The generations went from sub-critical maraging steel rotors to sub-critical carbon fiber rotors to super-critical carbon fiber rotors. The throughput approximately doubled for each generation, going from 1-2 swu/yr to 3-5 swu/yr to 5-10 swu/yr.

3.2.3.5 India

India began research into centrifuge enrichment in 1975 and, by 1985, had a 100-centrifuge cascade that operated successfully. India's centrifuge design used a subcritical maraging steel rotor that had a likely throughput of less than 3 swu/yr. After two years of pilot plant operation, they began construction of a second facility and had it in operation by 1992. It consisted of several hundred operating centrifuges of domestically produced maraging steel. The purpose of this plant was to develop capability to enrich fuel for India's reactors, as the French were ceasing to provide fuel for them[30].

Many of the Indian centrifuges machines prematurely crashed or otherwise proved defective. Due to technical limitations encountered, it was decided in 1997 to build and install rotor assembles of an improved design at its pilot centrifuge plant[31].

Indications are the Indian program had only one design over a 25 year period, which proved ineffective. The new designs were apparently not a new generation, but rather are improvements.

3.2.3.6 Pakistan

The Pakistani program (besides the Soviet Union the only successful program to date specifically intended to develop weapons grade HEU) did not use intrinsically-developed technology.

Pakistan began a centrifuge enrichment program in 1973. In 1975 Pakistan acquired plans for early Urenco centrifuges, using both aluminum and maraging steel, and built cascades of both types[32]. As it had a limited technological capability, Pakistan purchased centrifuge-related equipment from companies in Germany, The Netherlands, the United States, France, and China. In this process, an international clandestine

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[30] Second Indian Enrichment Facility Using Centrifuges is Operational, Mark Hibbs, Nucleonics Week, Vol. 33, No.13; Pg 9, 3/26/1992

[31] India to equip centrifuge plant with improved rotor assemblies, Mark Hibbs, Nuclear Fuel, vol.22 No. 24; Pg 7, 12/1/1997

[32] Pakistan Builds Second Plant to Enrich Uranium, Simon Henderson, Financial Times (London), 12/11/1997

network for manufacturing and acquiring centrifuge and enrichment related components was developed[33] (see Section 3.2.4.1.3).

Using the equipment and technology acquired through purchases, Pakistan developed aluminum and maraging steel centrifuges. The aluminum centrifuges (called P1 or Pak-1) were based on the Urenco CNOR/SNOR centrifuge models, the maraging steel centrifuges (called P2 or Pak-2) were based on the Urenco G-2 design.

In 1976, Pakistan began construction of facilities for both a pilot and full-scale plant. By 1979, a 54-stage cascade was nearing completion[34]. A decision had been made to concentrate on P2 style centrifuges but manufacturing difficulties with maraging steel bellows used in the P2 resulted in having to build 14,000 centrifuges to get 1,000 that functioned.

By 1984, it was reported that despite a difficulty in developing proper centrifuge cascade operation, Pakistan had uranium enriched to 3.4% U-235[35]. A demonstration facility was in operation, with a production rate of about 5,000 swu/yr[36]. It was reported, by 1993, that about 14,000 centrifuges had been installed in Pakistan[37]. However, the reliability of these centrifuges was low[38]. Crashes required replacement of 1,000-2,000 units per year[39]. As new units are constructed and installed, improvements are made, but the basic models remain the same.

Pakistan has had a centrifuge-enrichment program for over 25 years. It appears that while improvements are made as new units are constructed, no major design change has occurred. Quality control appeared to have been low in the early stages of the program, with many early units failing, and existing centrifuges being replaced at the rate of one out of fourteen per year.

Buying components and assembling them in country allowed Pakistan to develop an enrichment capability in nine years. However, the Pakistani program exhibited significant quality problems in its early years.

3.2.3.7 China

After a period of in-country development and research, China made the decision to purchase centrifuge enrichment technology from Russia. In 1993, an agreement was signed between the two countries. In 1996, the 200,000-swu/yr facility went into operation. The plant uses older sub-critical Russian centrifuges featuring aluminum rotor tube assemblies.

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[33] *Thriving Nuke Trade Revealed*, Rohan Sullivan, The Commercial Appeal (Memphis, TN), February 21, 2004

[34] Long Road to Chagai, Shahid-Ur-Rehman, Print Wise Publication, 9/1/1999

[35] Banned Book Charts Pakistan bomb Course, Shahid-Ur-Rehman Khan, Nucleonics Week: Vol. 30, No.33; Pg. 3, 8/17/1989

[36] World Nuclear Industry Handbook – 2001, Business and Industry, Wilmington Publishing Limited, 2001

[37] China said aiding Kahuta project; U.S. credits, NPT status on line, Mark Hibbs, Nucleonics Week, Vol. 37, No 6; Pg 1, 2/8/1996

[38] *India and Pakistan Fail To Include New SWU Plants on Exchanged Lists*, Mark Hibbs, Nuclear Fuel, Vol. 17, No. 7; Pg. 6 March 30, 1992

[39] Nuclear History in India, Pakistan, AP Online, 5/28/1998

China and Russia are collaborating on design and engineering for advanced centrifuges. China has made substantial progress in indigenous development of centrifuges using composite carbon fiber[40].

This is a good example of a minimum time to develop an enrichment facility. It essentially took three years to get the facility into operation, including constructing the buildings and installing the centrifuges. This particular facility was co-located with an older gaseous diffusion plant, so UF_6 handling facilities already existed there and did not need to be developed.

3.2.3.8 United States

The United States had an early centrifuge research program, first enriching uranium at the University of Virginia in 1941. Approximately 1.2g of 4% enriched uranium was produced in this initial attempt. This first centrifuge, known as a "Beams"-type centrifuge after its inventor, had a length of 3.35 meters.

For a variety of reasons, development of centrifuge enrichment in the United States stopped until 1958, when information about work in the Soviet Union became available. In 1965, a demonstration centrifuge (SET I) was built, followed by SET II in 1967, and SET III in 1969. While a pilot plant was built in 1978 to demonstrate the technology, the centrifuge program was canceled in 1985 in favor of concentrating on laser enrichment technology (AVLIS).

The United States had a different centrifuge design philosophy than Urenco. Urenco designed for many small machines with very low failure rates, with no maintenance, while the United States went for fewer, far bigger machines and planned to repair them. It was reported that individual machine throughput was 300 swu/yr[41]. United States centrifuges were 50 cm in diameter, 15 meters long[42]. Compare this to the 5 cm diameter, 50 to 100 cm long machines being developed in other countries at the same time. High failure rates and what was thought to be a more promising technology halted U.S. centrifuges[43] development.

3.2.3.9 Iraq

Like Pakistan, Iraq planned to develop centrifuge enrichment technology through surreptious acquisition. It is reported[44] that Iraq attempted to develop centrifuge technology based on Urenco G1/G2 technology. This program was to be made possible through acquisition of equipment, plans, and models from France, Germany, Switzerland, Liechtenstein,[45] and Pakistan[46].

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[40] Nuclear Fuel, Vol. 24, No 10: pg 11, Mark Hibbs, 5/17/1999

[41] Fuel Issues, Plans for an Enriched Future, Dennis Spurgeon, Nuclear Engineering International, pg 44, 3/31/2002

[42] In the beginning was uranium….., Dan Charles, New Scientist; Vol.136; No 1844; pg 30, 10/24/1992

[43] Uranium Enrichment: why the U.S. is turning to lasers, David Fishlock, Financial Times (London) Section I; Technology; pg 6, 9/16/1985

[44] Iraqi Evidence Points to Unknown Enrichment Mentor, Mark Hibbs, Nucleonics Week, Vol.32, No 34; pg 7, 8/22/1991

[45] The trail of secrets that gave Saddam deadly power, The Sunday Times, 12/16/1990

[46] Special Report: How Saddam got the bomb, Mednews - Middle East Defense News; Proliferation; Vol. 5, No. 1, 10/14/1991

Iraq began its centrifuge enrichment program in parallel with efforts in electromagnetic isotope separation and chemical isotope separation (sections 3.3.3.3 and 3.4.3.3). By 1984, it had set up a centrifuge production facility. Equipment included spinning lathes, precision lathes, milling machines, and other equipment that had been acquired from European manufactorers.

After the Gulf War, inspectors discovered a number of components related to G-1/2 centrifuges, and even several G-2 type centrifuge prototypes believed to be of Pakistani origin. One hundred tons of maraging steel centrifuge-preforms (of Belgium origin), enough to make 5000 centrifuges, were discovered[47] by inspectors and then destroyed.

Iraq's program was primarily concentrated on first and second generation Urenco G-1/G-2 centrifuge technology. Interestingly, a rotor tube obtained by the IAEA indicates that Iraq may have tried to build G-1 rotor tubes using carbon fiber material instead of maraging steel. G-1 maraging steel type rotors are about 50 cm long; because of their stronger tensile strength, carbon fiber rotors can be longer, thus more efficient[48].

There is no indication that Iraq was successful in using centrifuge enrichment to produce any significant quantities of HEU. However, with a robust program underway and sufficient component inventory (even if they had the same quality problems as Pakistan with 13 of 14 failures of completed centrifuges, see section 3.2.3.6), they would have had enough for several cascades.

3.2.3.10 Iran

Iran had a strong nuclear program prior to 1979, but at that point most activities ceased. The program was restarted in 1985, beginning with plans to develop a uranium enrichment program using centrifuges. In 1987, Pakistan began a joint collaboration with Iran to assist in this effort. Pakistan provided plans and components in several installments from the late 1980s to the mid-1990s, allowing Iran to overcome several major technological hurdles in developing its enrichment capabilities[49,50].

Iran's enrichment program consisted of three phases. During the first phase (1985-1997), efforts were concentrated on achieving an operating centrifuge. To do this the Iranians acquired components from abroad using the Pakistani network and through other foreign intermediaries, or directly by the Iranians themselves. During this period, Iran acquired high-strength aluminum, maraging steel, electron beam welders, balancing machines, vacuum pumps, computer-numerically controlled machine tools, and flow-forming machines for both aluminum and maraging steel. Many of these items were obtained in Europe, especially from Germany and Switzerland[51].

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[47] Centrifuge-grade Maraging Steel Shipped from Germany is Missing, Mark Hibbs, Nuclear Fuel, Vol. 19, No 21: pg 8, 10/24/1995

[48] Experts believe Iraqi Program Focused on Urenco's G-1 centrifuge, Mark Hibbs, Nuclear Fuel: Vol. 16, No. 16; Pg. 6, 8/5/1991

[49] *Nuclear Program in Iran Tied To Pakistan, Complex Network Acquired Technology and Blueprints*, Joby Warrick, Washington Post, Sunday, December 21, 2003

[50] *How Pakistani's network offered the whole kit; Nuclear proliferator/Scientist and black marketer*, William J. Broad, David E. Sanger and Raymond Bonner, The International Herald Tribune, February 13, 2004

[51] *Fuel Cycle; Tracking The Technology*, Jack Boureston, Nuclear Engineering International, September 30,2004

In the second phase (1997-2002), efforts focused on centrifuge construction, assembly, and mechanical testing. The third phase (since 2002), has involved research, assembly, installation, and completion of pilot centrifuge cascades.

Iran explored the use of both P1 and P2 centrifuge technology from Pakistan (see section 3.2.3.6). In the mid-1990s, Pakistan supplied about 500 P1 centrifuges to Iran. These were probably scrapped machines that Pakistan had retired from its main centrifuge program[52]. By 2003, Iran reported it had 920 P1 centrifuges of which some were indigenously manufactured or assembled. Iran also received blueprints and information from Pakistan for constructing P2 machines, but lacked the capability to manufacture maraging steel rotors, and attempted to make them from carbon composites. This attempt reportedly failed and the decision was made in 2003 to abandon the P2 approach and scrap the P2 equipment.[53]

Although Iran has reportedly not yet actually enriched any significant quantities of uranium, it has apparently continued to work to advance its indigenous aluminum centrifuge technology. It has been reported that Iran may have developed an advanced supercritical gas centrifuge, with a potential of producing as much as 10 SWU/yr[54]. An aluminum centrifuge with a throughput of 6 SWU/yr or 7 SWU/yr would be of supercritical design, connecting two or more segments with a bellows, allowing it to survive the first critical speed. An aluminum machine with a throughput of 14 SWU/yr would have three or more segments. This higher throughput would reflect a very advanced aluminum machine design and a production capability nearly three times that of first-generation supercritical centrifuges such as the G2 or P2 design[55].

Iran's centrifuge development program took over twenty years to produce what is apparently an indigenous, advanced design for a supercritical aluminum centrifuge based on early Urenco or Pakistani designs.

3.2.3.11 Libya

In 1995 Libya began to acquire the capability to perform gas centrifuge uranium enrichment. In 1997, after failing to develop centrifuge technology indigenously, Libya decided to purchase the technology surreptiously and contacted the international underground procurement network based in Pakistan that had already supplied Iran (Section 3.2.3.10) and North Korea (Section 3.2.3.12) with centrifuge components and designs.

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[52] *Smuggling Of Weapons Of Mass Destruction,* David Albright, Capitol Hill Hearing Testimony, Senate Governmental Affairs Committee, June 23, 2004

[53] *The spread of nuclear know-how*, Peter Grier, Christian Science Monitor, March 2, 2004

[54] *Iran Has Developed An Advanced Supercritical Aluminum Centrifuge,* Nucleonics Week Special Report, Volume 44.Special, March 7. 2003.

[55] *Estimates of Natanz Centrifuge Power by **Iran,** IAEA Differed by Factor of Two*, Mark Hibbs, Nuclear Fuel, Vol. 28, No. 10; Pg. 3, May 12, 2003

For what was reportedly about $100 million[56], Libya contracted with the network to develop a turn-key gas centrifuge facility. The network planned to supply approximately ten thousand centrifuges, piping to connect them together, detailed project designs for the centrifuge plant, electrical and electronic equipment, uranium feed and withdrawal equipment, an initial 20 tonnes of uranium hexafluoride, and equipment and technology to allow Libya to make more centrifuges indigenously, and on- going technical assistance to help Libya overcome any obstacles in assembling and operating the centrifuges in the plant[57].

The network sold the Pakistani "P1" and "P2" centrifuges (Section 3.2.3.6). The P1 centrifuges are similar to the early Dutch-designed aluminum CNOR/SNOR centrifuges. The P2 design is based on the more advanced German designed maraging steel G-2 centrifuge.

In 1997 the Libyan government received twenty assembled P1 centrifuges and components for 200 more. These were reportedly scrapped machines (one of which had been in use as late as 1987) that Pakistan had retired from its main centrifuge program, and that members of the network had apparently been able to remove them in secret and sell them. IAEA inspectors have found contamination from highly enriched uranium as well as low enriched uranium on gas centrifuge equipment in Libya, and this was almost certainly from the used and probably contaminated equipment acquired from Pakistan[58].

Nine of these were assembled into a cascade in 2000. The first successful test was in October 2000, and three cascades had been assembled by April 2002. However, the cascades were reportedly never used for enrichment but only for testing and training purposes, and were then disassembled and moved for security reasons, and have remained packed in boxes ever since.

In addition to the P1s, Libya acquired two P2 model demonstration centrifuges. One of was not suitable for enrichment since it did not have the final surface coating necessary to prevent corrosion by uranium hexafluoride gas[59].

In addition to the acquisition of a complete centrifuge facility, Libya planned to construct a sophisticated manufacturing center, code-named Project Workshop 1001, to make centrifuge components. The original plan called for this center to be able to make additional centrifuges after the network delivered the first ten thousand, either to replace broken ones or add to the total number of centrifuges. However, if the network would potentially have had difficulty supplying a component, this center may have been intended to manufacture it.

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[56] AP Investigation: *Head of Pakistan's nuclear ring made repeated visits to uranium-rich Africa*, Edward Harris, Ellen Knickmeyer, The Associated Press, April 17, 2004

[57] *Testimony of David Albright before the Senate Governmental Affairs Committee*, June 23, 2004, Federal Document Clearing House Congressional Testimony

[58] Michael Adler, Agence France Presse – English, May 29, 2004 Saturday

[59] *Libya Politics: Nuclear Weapons Program Diverse, Says IAEA*, Mark Huband Financial Times, February 23, 2004

In all, the network helped Libya purchase more than a hundred machine tools for its facility. Most of the machine tools, furnaces, and other equipment for the center came from Europe, particularly from or through Spain and Italy. The equipment was not under international controls, but was still apparently suitable for use in a centrifuge manufacturing program, particularly because the network also supplied detailed manufacturing information for almost all the parts.

In October 2003, a ship containing centrifuge parts manufactured by the network and bound for Libya was boarded and the parts were seized. In December, 2003, Libya announced that it would abandon its nuclear plans[60].

If Libya had continued with its nuclear ambitions and the network had not been exposed, some have estimated that with the assistance of the network, Libya could have succeeded in about four to five more years in assembling its centrifuge plant and operating it to produce significant amounts of HEU systems.[61] This means that the project would have been completed in 2007 or 2008. Since the project was initiated in 1997, an evaluation of this estimate shows that the total elapsed time for the project to be completed could have been as short as about ten years.

3.2.3.12 North Korea

North Korea may have imitated discussions with the Pakistani network about acquiring centrifuge enrichment technology as early as the 1980s[62]. However, the effort to acquire the technology accelerated in 1994, when North Korea began an intensive centrifuge enrichment technology development program in violation of the 1994 Agreed Framework, in which North Korea had pledged to freeze its indigenous nuclear program[63].

Although details of North Korea’s program are not clear, there are suggestions that it is similar to the uranium enrichment programs in Iran (Section 3.2.3.10) and Libya (Section 3.2.3.11) in that they are all based on original and virtually identical Urenco centrifuge design information. It is suspected that North Korea's program was based on the G-1 Urenco design[64].

It has been reported that the Pakistani network shipped centrifuge designs, a small number of assembled centrifuges, depleted UF_6 gas, and a 'shopping list' of equipment needed to produce 'thousands' more, to North Korea in the late 1990s.

In 2002, a consignment of aluminum tubing, shipped from Germany and headed for North Korea, was confiscated. It is likely that this was intended for a pilot uranium enrichment cascade containing between about 100 and 200 gas centrifuges. The basis for this consignment may have been a centrifuge cascade design identical or closely

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[60] *Treachery: How America's Friends and Foes Are Secretly Arming Our Enemies, Libyan sincerity on arms in doubt*, Bill Gertz, The Washington Times, September 9, 2004

[61] *Smuggling Of Weapons of Mass Destruction,* Testimony of David Albright before the Senate Governmental Affairs Committee, June 23, 2004, Federal Document Clearing House Congressional Testimony

[62] *Fuel Cycle; Tracking the Technology* ,Jack Bourseton Nuclear Engineering International, September 30, 2004.

[63] *Urenco report said to concur aluminum for DPRK fits centrifuge,* Mark Hibbs, Nuclear Fuel, Vol. 28, No. 14; Pg. 20, July 7, 2003

[64] *Netherlands probing suspected **centrifuge**-related diversions*, Mark Hibbs, Nucleonics Week, Vol. 45, No. 4; Pg. 16, January 22, 2004

resembling one used in Iran that called for a cascade of 164 centrifuges. This may indicate a similarity between the North Korean and Iranian program. [65]

The aluminum tubing apparently matched the dimensions of casings for a rotor assembly similar to Urenco G-2 centrifuges. Other reports indicated that enough aluminum tubing had been ordered to make casings for about 3,500 G-2 centrifuges, which are estimated to have a throughput of about 5 SWU/machine/yr.

The DPRK initiated its program in 1994. The current status is unclear, but significant quantities of centrifuge components were being ordered as late as 2002, an elapsed time of about eight years.

3.2.4 Technological Information Acquisition/Development

There are three approaches that can be taken to acquire and develop a technology. A country can develop the technology independently, team in development with another country, purchase the technology openly, or acquire it covertly. A review of the descriptions of the centrifuge enrichment programs will provide examples of all of these approaches.

3.2.4.1 Acquisition approaches

The above summaries of national centrifuge programs allow some conclusions to be drawn. This section summarizes the experience of the different acquisition approaches: in-country development, purchase, and overt/covert acquisition.

3.2.4.1.1 In-country development

Five countries have had indigenous centrifuge development programs. The Soviet Union and the Urenco countries were the most successful. The other three countries (Brazil, India, and Japan) based their original programs on early Urenco designs. Time frames for initial development are relatively constant (Table 8): eight to eleven years for a demonstration facility, and four to eight years later to have a production facility in operation (India and Brazil seventeen years and twenty-two years, respectively).

Successful programs have continual improvement programs in place and a period of between six and eight years between generations. Brazil's program had about ten years between generations.

Less successful programs, such as India and Japan, spend more of their effort on trying to make their centrifuges operate successfully, spending much less time on new component development.

3.2.4.1.2 Technology Purchase

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[65] *Aluminum tubing North Korea sought believed meant for pilot cascade*, Mark Hibbs, Nuclear Fuel, Vol. 28, No. 21; Pg. 7, October 13, 2003

Although a number have countries have expressed an interest, only China has openly purchased centrifuge enrichment technology. It took three years from the time the agreement was signed until the facility (200,000 swu/yr) went into operation. Several factors were involved in this rapid development: 1) The Russian (Soviet) centrifuges were reportedly early models taken out of service that were immediately available and were known to be functional, 2) the Russians were very familiar with the technology and had proven ways of setting up the cascades and other parts of the facility, 3) the facility was constructed on a site where other enrichment technology existed, so proven UF_6 infrastructure was available as were technicians trained in handling it.

Brazil's decision to construct a commercial facility could in some ways be considered a purchase, because the technology was developed by the Brazilian navy and was sold to a commercial entity. It took Brazil about 18 months to put their facility (20,000 swu/yr) into initial operation. This is about 1/10th the size of the Russian/Chinese enrichment facility. As with the Chinese facility, the Brazilian commercial plant was located on a site where enrichment technology (an aerodynamic enrichment facility, see section 3.5) existed, so the infrastructure and trained personnel were available.

From this, it can be inferred the minimum time to put a centrifuge enrichment facility into operation in a location with existing UF_6 technology infrastructure is between eighteen months and three years.

3.2.4.1.3 Overt/covert acquisition

At least five countries, Pakistan, Iraq, Iran, North Korea, and Libya, have acquired or attempted to acquire technology through covert means. The most successful of the five, Pakistan, may have been responsible for providing the means for the other four countries to develop their programs, especially Iran, North Korea, and Libya.

Pakistan is a classic example of a country attempting to acquire technology through covert means. In order to bypass existing international controls, Pakistan was forced to develop an illicit international procurement network. The key technology holders and several of its leaders were in Pakistan. But many other leaders were spread throughout the world and located in Europe, Dubai in the United Arab Emirates, Turkey, South Africa, and Malaysia. The network also depended on a variety of unwitting manufacturing companies and suppliers on many continents. Buying components and assembling them in-country allowed Pakistan to develop an enrichment capability in nine years.

In the mid 1980's, when the Pakistani program was mature, the process changed, and some key members of the network decided to make the technology available to whoever would buy it[66,67]. While operational and producing HEU, the early Pakistani facilities suffered from severe quality problems. It may be inferred that this is because the

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[66] *Smuggling of Weapons of Mass Destruction*, David Albright, Capitol Hill Hearing Testimony, June 23, 2004

[67] *How Pakistani's network offered the whole kit; Nuclear proliferator/Scientist and black marketer*, William J. Broad, David E. Sanger and Raymond Bonner, The International Herald Tribune, February 13, 2004

technology was not initially developed in-country, and therefore the technological and manufacturing base was insufficient to develop and expand a high quality program until a number of years of experience were gained.

Iraq attempted a similar program. Despite international controls, it pursued an aggressive component acquisition program, which was halted during the Gulf War, with much of the acquired components destroyed.

The experience of Iran, North Korea, and Libya shows that it was possible for an international network such as that developed by Pakistan to provide significant support to any country that wished to develop its own centrifuge enrichment capability.

Although Iran has officially not yet enriched any uranium, it appears that that it is nearly ready. Its program has existed for about twenty years.

It took Pakistan about nine years to reach the point of enrichment. Accordingly, based on Pakistan's experience it is expected that a successful overt/covert acquisition time scale could be around nine years.

3.2.4.2 Time constant for technology development

Based on the above discussions the following conclusions can be reached:

- Minimum time to develop indigenous centrifuge technology – 8 years
- Expected time to develop technology surreptitiously using stolen technology - 9 years
- Time to purchase and install technology in significant quantities in new facility - 18 to 36 months

3.2.5 Required Concurrent Technologies

Required concurrent technologies can be separated into three types. The first type includes the technologies involved with centrifuge construction. This is important and specific for centrifuges, since so many individual components are required. The second type of concurrent technologies is electric power related, as stable electric power is essential to successful operation. The third is fluorine-related technology.

3.2.5.1 Centrifuge construction

Important centrifuge construction technologies include:

- Numerically-controlled machine tools
- Clean-room technology
- Metallurgical techniques, including material etching
- Magnet technology

3.2.5.2 Electric power control

Centrifuges rotate at a very high, precise speed; therefore, control of the frequency of rotation is very important. Important elements include:
`

- Frequency converters
- SCADA systems
- Stable off-site power supplies
- Very stable electric power grid

3.2.5.1 Fluorine related equipment

As with gaseous diffusion facilities, fluorine related equipment is necessary. These include:

- Feed autoclaves used for passing UF_6 to the centrifuge cascades
- Desublimers (or cold traps) used to remove UF_6 from the cascades
- Product and tails stations used for passing UF_6 into containers

3.3 ELECTROMAGENTIC ISOTOPE SEPARTION TECHNOLOGY HISTORY

3.3.1 Technology Description

3.3.1.1 Origin

The process of electromagnetic isotope separation (EMIS) was developed in the United States as a part of the Manhattan project. Starting in 1941 and using already existing cyclotrons to demonstrate the technology, the U.S. decided to build bigger machines called “calutrons” to produce enriched uranium. Calutrons produced the first HEU, using slightly enriched uranium from other processes as feed. When the first gaseous diffusion plant began operating effectively, the use of calutrons for enriching uranium ceased. They are, however, still being used for other isotopic separation tasks.

3.3.1.2 Basic theory

The EMIS approach is based on the physical principal that ions of the same energy but of different masses describe different trajectories in a magnetic field. In particular, the trajectory of ^{238}U will have a larger diameter than that of ^{235}U. The different diameters allow for separation and collection of the material in receivers or “collection pockets”. EMIS is a batch process that produces weapons grade HEU from natural uranium in only two steps. However, hundreds to thousands of units would be required to produce large quantities of HEU due to the process’s relatively low product collection rate and the long cycle time required to recover material between runs.

In the uranium EMIS process, uranium ions are generated within an evacuated enclosure (called a “tank”) located in a strong magnetic field. For the EMIS ion source, solid uranium tetrachloride (UCl_4) is electrically heated to produce UCl_4 vapor. The UCl_4 molecules are bombarded with electrons, producing U^+ ions. The ions are accelerated to high speed by an electric potential and follow a circular trajectory in the plane perpendicular to the magnetic field. In U.S. EMIS separators, the ion beam traverses an 180^o arc before the ions pass through slit apertures at the collector. These are called *180^o machines*. A major problem with the EMIS process is that less than half of the UCl_4 feed is typically converted into the desired to U^+ ions and less than half of the U^+ ions are actually collected. Recovery of unused material deposited on the interior surfaces of the tanks is a laborious, time consuming process that reduces the effective output of an EMIS facility and requires a large material recycle operation[68].

It is important to note that EMIS is used to separate many different kinds of isotopes and has been used on a laboratory scale to separate ^{239}Pu from reactor grade spent fuel. See Figure 3 for EMIS system components and configuration.[69]

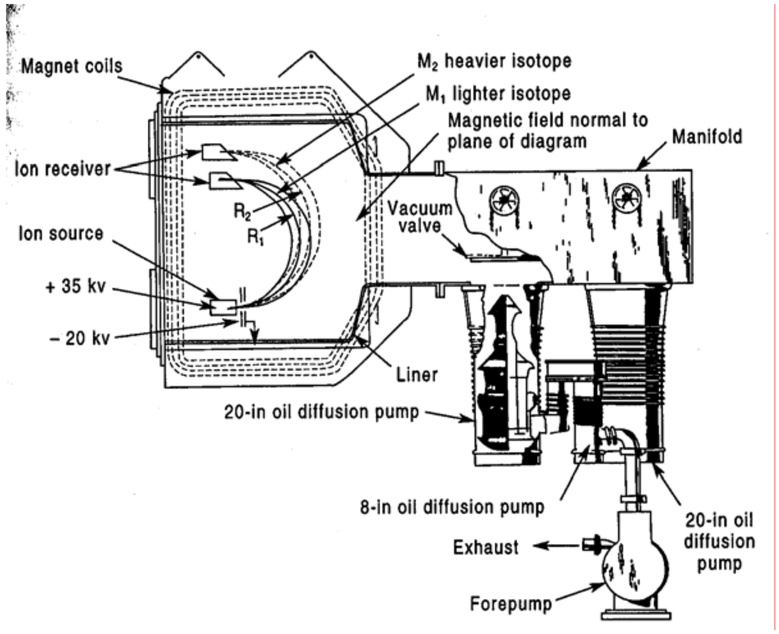

Figure 3 Electromagnetic Isotope Separation System Configuration

3.3.1.3 Important components/materials

Following is a list of important components and materials needed to support development of EMIS facilities. Governmental and international agreements control the acquisition of many of these materials.

- Ion sources
- Ion collectors

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[68] Department of Defense, Militarily Critical Technologies List, Part II: Weapons of Mass Destructions Technologies, Section 5 – Nuclear Weapons Technology, February 1998.
[69] Technology and the Proliferation of Nuclear Weapons, Richard Kokoski, Sipri, Oxford University Press

- Vacuum housings
- Magnet pole pieces
- High-voltage power supplies
- DC-magnet power supplies
- Vacuum pumps
- Uranium-recovery equipment
- Uranium tetrachloride (UCl_4) processing equipment

Table 9 identifies specific technologies used in EMIS enrichment facilities, describes the technology level needed for successful use, lists critical materials, and unique equipment used in that technology.

Technology	Sufficient Technology Level	Critical Materials	Unique Test, Production, and Inspection Equipment	Unique Software and Parameters
Ion Source	Single or multiple uranium ion sources consisting of a vapor source, ionizer, and beam accelerator. Capable of providing a total ion beam current of ≥ 100 mA	Uranium chloride, graphite, stainless steel, copper, tantalum, tungsten	None identified	Validated ion source models including 3-dimensional solution of Poisson's equation for multiple species and taking into account the effect of the accelerating structure.
Ion Collectors	Collector plates of two or more slits and pockets for collection of enriched and depleted uranium ion beams, minimize sputtering	Graphite, stainless steel, copper	None identified	Validated ion beam dynamics software and algorithms that optimize isotope separation design from ion source through vacuum and into collector.
Vacuum Housings	Vacuum vessels large enough to contain two or more sets of injectors and collectors with appropriate beam current geometry. Two or more provide the scaling required for reasonable electromagnetic separation.	Nonmagnetic materials (e.g., stainless steel)	None identified	None identified
Magnet Pole Pieces	Diameter >2 meters, able to maintain a time-invariant magnetic field within a separator, ability to transfer magnetic field between adjoining separators.	Low resistance wire, magnet iron	Precision field measurement and adjustment. Precision shaping of pole tips, precisely controlled windings.	Validated 3-dimensional singly- (predominant) and multiply-charged high current ion beam dynamics codes and algorithms
High voltage DC Power Supplies	Capable of continuous operation, output voltage ≥20,000 V, output current ≥1 A, voltage regulation <0.01% over 8-hour interval	None identified	None identified	None identified
DC Magnet Power Supplies	Capable of continuously producing a voltage ≥ 100 V, current ≥500 A, and current or voltage regulation <0.01% over 8-hour interval.	None identified	None identified	None identified
Vacuum Pumps	Input throat size ≥ 38 cm, pumping speed ≥ 15,000 liters/sec, vacuum <10^{-4} Torr (1.33 x 10^{-4} nbar), oil-diffusion pump systems of sufficient capacity to provide minimum downtime when removing collectors.	Pumping fluid, such as a hydrocarbon oil	Fast-acting shutoff valves to protect vacuum system and minimize downtime	None identified
Uranium Recovery	Extract enriched uranium in small batches without going critical, efficient chemical processes to extract enriched uranium from graphite collector	Cadmium (neutron poison) used to prevent criticality. Must be removed at end of process	Mass spectrometers	None identified

Table 9 Technologies and components used in EMIS facilities

3.3.2 Technology Development

3.3.2.1 Technological difficulties in use

A number of technical difficulties exist in using of the EMIS process for producing significant quantities of HEU:

1) There is a very low efficiency in feed utilization, due to the tendency of ionized uranium vapor to deposit on all available surfaces inside the vacuum chamber. Only ten to fifteen percent of the feed material is actually ionized, meaning that the cut (the ratio of product flow to feed flow) is below .01 for low enriched feed.
2) Control of the ion beam current is important for various physical reasons. As the collection rate is directly proportional to the ion beam strength, the amount of material deposited is limited.
3) To maximize the amount of material arriving at the collectors, the beams must be focused by adjusting the magnetic fields.

Table 10 lists important technologies and related technical issues.

Technology	Technical Issues
Ion Source	Obtain high U+ beam currents from source, control expansion of beam, properly focus ion beam on collector slits, heater life, insulator breakdown, damage to source components due to high energy ions
Ion Collectors	Retain and measure collected uranium, retain shape over wide temperature range, resist sputtering, conduct heat, permit recovery of deposited uranium.
Vacuum Housings	Leakage rate; open and close with minimum downtime
Magnet Pole Pieces	Maintain low magnetic field ripple
High-voltage Power Supplies	Maintain stable voltage
DC Magnet Power Supplies	Maintain stable current
Vacuum Pumps	Maintain high vacuum in large evacuated region
Uranium Recovery	Substantial chemical processing facility required, labor intensive

Table 10 **Technology and technical issues involved with EMIS**

3.3.2.2 Changes/improvements in technology

In 1946, Swedish scientists developed an improved calutron design. By varying magnetic field focus and having the ion beam go through a 255^0 arc, the separation power could be increased by a factor of 1.5, and higher intensity ion currents could be used. This type of calutron, called a 255^0 machine, has a significantly higher production rate than the 180^0 machines[70].

Efforts to increase the performance of magnetic spectrometers have led to significant improvements in magnet design, which can increase the ability to focus the ion beams and increase production efficiency accordingly.

3.3.3 Countries that have used/attempted to use technology

A number of countries have acquired EMIS technology. Table 11 lists those with significant programs, but only three have actively tried to use EMIS to acquire HEU. The others acquired single or small numbers of units and used them for other purposes.

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[70] Iraq's calutrons: Electromagnetic isotope separation, beam technology, and nuclear weapon proliferation, Andre Gsponer and Jean-Pierre Hurni, ISRI-95-03, 10/19/95

China	Japan
France	Soviet Union
India	Sweden
Iran	United Kingdom
Iraq	United States
Israel	

Table 11 Countries with EMIS technology

The three countries with active programs to produce HEU using EMIS were the United States, the Soviet Union, and Iraq. As the other eight countries did not use EMIS for uranium enrichment, their programs are not evaluated.

3.3.3.1 United States

The U.S. began working on enrichment using EMIS in 1941. The first large quantity of highly enriched uranium was produced in 1944, after three years of work on the process.

In the U.S. EMIS program, production of weapon-grade uranium took place in two enrichment stages, referred to as the α and β stages. The first (α) stage used natural or slightly-enriched uranium as feed and enriched it to 12-20 % ^{235}U. The second (β) stage used the product of the (α) stage as feed and further enriched it to weapons grade HEU. Both stages used 180^0 machines. To allow more efficient use of magnets and floor space, the individual stages were arranged in continuous oval or rectangular arrays (called "race-tracks" or simply "tracks") using separator tanks alternated with electromagnetic units.

Over 1100 calutrons were used in the U. S. program, 864 α machines, and 288 β machines. It took three years of intense effort and a billion dollars to get material for one weapon[71], and the use of calutrons for uranium enrichment ceased after the gaseous diffusion facility became operational. Some calutrons were modified to produce enriched stable and other radioactive isotopes.

3.3.3.2 Soviet Union

The Soviet Union built its first cyclotron in 1932. In 1943, researchers used it to make ^{239}Pu. In 1946, a program similar to that of the United States using 180^o machines was initiated to develop the capability to produce HEU. It was never successful. In 1949, using 40% HEU from the Soviet gaseous diffusion plant, 0.4kg of HEU was produced "with great difficulty".[72] After the Soviet gaseous diffusion facility became operational in 1950, the EMIS uranium-enrichment program was de-emphasized.

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[71] Uranium enrichment and nuclear weapon proliferation, Alan S. Krass, et. al., sipri, 1983

[72] Stalin & the Bomb, The Soviet Union and Atomic Energy, 1939-1956, David Holloway, ale University Press, 1994

EMIS technology continued to be used for other purposes such as in the United States, and designs continued to evolve. In 1957, a design for a 255^0 machine was announced that could be used for enrichment of heavy isotopes, such as uranium or plutonium.

Although advanced designs were developed later, the active Soviet enrichment program from 1947-1949 never produced significant amounts of material.

3.3.3.3 Iraq

As early as 1979, Iraqi engineers were attempting to collect information about state of the art magnet design for use in an EMIS facility. In 1982, construction began on the first of two planned facilities with each having seventy 255^0 type α stage machines and twenty β stage machines. At the same time, construction began on plants to produce UCl_4 feed for the enrichment facility.

By 1990, construction was complete on the calutrons and on feed stock production facilities. However, technical difficulties limited production, including:

- Severe corrosion problems and furnace and chiller difficulties at the UCl_4 production facility,[73] resulting in limited feedstock availability.
- Difficulties in acquiring good ion-sources, which limited the calutrons' production efficiency.

Iraq's calutrons and other supporting facilities were destroyed before production problems could be solved.

Analysis of available information shows that it took eight years, from the decision to construction start in 1982, to begin preliminary operation in 1990.

3.3.4 Technological Information Acquisition/Development

3.3.4.1 Acquisition approaches

EMIS technology was first developed in the United States. After WWII, information about the technology was unclassified and widely disseminated. Many isotope separation facilities using EMIS technology have been constructed. The technology to construct such facilities is therefore easily available and can be obtained with little trouble. Accordingly, all the programs could be considered as an in-country development acquisition approach.

3.3.4.2 Time constant for technology development

It is difficult to determine a time constant for a successful program. The United States program (three years) could be considered to represent a minimum time, while the Iraqi

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[73] Special Report: How Saddam got the bomb, Mednews - Middle East Defense News; Proliferation; Vol. 5, No. 1, 10/14/1991

program (eight years to initial production) may be a more appropriate example of what would be done today.

3.3.5 Required Concurrent Technologies

Technologies required for EMIS programs include:

- Large electromagnets
- High voltage power equipment
- High current ion sources
- Vacuum/molecular diffusion pumps
- UCl_4 processing equipment
- Uranium processing equipment

3.4 CHEMICAL AND ION EXCHANGE ENRICHMENT TECHNOLOGY HISTORY

3.4.1 Technology Description

3.4.1.1 Origin

Chemical-enrichment programs began in France in 1968 when French scientists discovered a chemical process for enriching uranium and began developmental research[74]. In 1972, Japan began work on a different process based on ion exchange. A primary reason for developing this type of technology is that it is difficult and time consuming to enrich to high level. Therefore, it can be considered a proliferation-resistance approach to enrichment.

3.4.1.2 Basic theory

Chemical-exchange isotope separation requires segregation of two forms of an element in separate but contacting streams. Since many contacts are required to achieve the desired separation, the contacting process must be fast and achieve as much separation as possible. For heavy elements such as uranium, achieving a suitable separation factor involves contact between two valence (oxidation state) forms such as hexavalent [U^{6+} as in uranyl chloride (UO_2Cl_2)] and the quadrivalent [U^{4+} as in uranium tetrachloride (UCl_4)]. The ^{235}U isotope exhibits a slight preference for the higher valence, for example the hexavalent over the quadrivalent in the Japanese Asahi process or the quadrivalent over the trivalent (U^{3+}) in the French solvent-extraction process[75].

3.4.1.2.1 Chemex

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[74] Nuclear Energy and Nuclear Weapons Proliferation: Chapter 2. Practical suggestions for the improvement of proliferation resistance within the enriched Uranium fuel cycle, J.H. Coates and B. Barre, Stockholm International Peace Research Institute, 1979

[75] Department of Defense, Militarily Critical Technologies List, Part II: Weapons of Mass Destructions Technologies, Section 5 – Nuclear Weapons Technology, February 1998.

Chemex, the chemical exchange process developed by the French, uses the exchange reaction that takes place between two valence states (U^{3+} and U^{4+}) of uranium ions in aqueous solution. Isotopic enrichment results from the tendency of ^{238}U to concentrate in the U^{3+} compound while ^{235}U concentrates in the U^{4+} compound. It is therefore possible to obtain enriched uranium by removing the U^{4+} compounds with an organic solvent that is immiscible with the aqueous phase (concentrated hydrochloric acid). Several possible extractants are available; however tributyl phosphate (TBP) is typically used. TBP is diluted with an aromatic solvent, and this organic phase moves countercurrent to the aqueous phase through a series of pulsed columns.

In the pulse column, the heavier aqueous phase is fed into the top of the column and the lighter organic phase is fed into the bottom of the column. A rapid reciprocating motion (*pulsing*) is applied to the contents of the column, providing efficient and intimate contact of the two phases.

After passing through the column, the enriched and depleted uranium streams must be chemically treated to be re-circulated through the column again (refluxed) or sent to another column for additional enrichment. This requires complicated refluxing equipment at both ends of the column.

3.4.1.2.2 Ion-exchange

The ion-exchange process developed by the Japanese (the Ashai Chemical Exchange Process, ACEP) uses the chemical isotope effect between two valences (U^{4+} and U^{6+}) of uranium. In this process, the organic phase is replaced by a proprietary ion-exchange resin. The aqueous phase flows through the stationary resin held in a column and the net effect of all the chemical reactions is a “band” of uranium that moves through the ion-exchange column. The exchange between the unadsorbed uranium flowing through the band and that adsorbed on the resin enhances the isotopic separation. In this continuous separation system, ^{235}U and ^{238}U tend to accumulate respectively at the entrance and exit ends of the adsorption band. In this process, it is economical to regenerate many of the chemicals by reaction with oxygen and hydrogen in separate equipment.

The adsorbent in the columns is a spherical bead of porous anion-exchange resin with both a very high separation efficiency and exchange rate. The development and manufacture of the appropriate adsorbent beads has required years of experimental research and development.

3.4.1.3 Important components/materials

Components and materials necessary to develop a chemical or ion-exchange enrichment plant include:

- Liquid-liquid exchange columns
- Liquid-liquid centrifugal contactors
- Electrochemical reduction systems and reduction cells

- Feed preparation systems
- Uranium oxidation systems
- Ion exchange columns
- Ion exchange reflux systems

An important key to making these processes work was to reduce chemical reaction times through the use of catalysts, in particular resins which lowered previous reaction times by a factor of 1000[76].

Table 12 identifies specific technologies used in chemical and ion exchange enrichment facilities, describes the technology level needed for successful use, lists critical materials and unique equipment used, and identifies export control references in place for that technology.

3.4.2 Technology Development

3.4.2.1 Technological difficulties in use

Chemical-exchange reactions involving different isotopes tend to have extremely small separation factors especially for heavy elements like uranium. This means that the extraction or exchanges must consist of many stages usually measured in the thousands. If the chemical reaction rates are relatively fast and if good mixing can be achieved in each stage, the stages can be relatively short. Even in the best of circumstances, however, the columns will be large, which implies large inventories and very long equilibrium times.

As the chemicals must be recycled to make either process (Chemex or Ion-exchange) economical, both processes involve substantial chemical reflux processes. These are energy-consuming and technically demanding, since they must be accomplished without significant remixing of the uranium isotopes. Criticality is also a problem as the uranium is in liquid solution. However, this can be controlled by the addition of strong neutron-absorbing substances to the solutions.

Table 13 lists important technologies and related technical issues for the chemical and/or ion enrichment process.

3.4.2.2 Changes/improvements in technology

Although chemical and ion exchange technology is used extensively for lighter elements and has been studied by several countries, it has not been shown to be more cost effective for uranium enrichment than other approaches, in particular modern centrifuge technology. Although successful enrichment has taken place (see section 3.4.3)

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[76] Uranium enrichment and nuclear weapon proliferation, Alan S. Krass, et. al., sipri, 1983

Technology	Sufficient Technology Level	Critical Materials	Unique Test, Production, and Inspection Equipment	Unique Software and Parameters
Liquid-liquid exchange columns	Ability to produce pipes of various diameters and lengths that are internally coated with material resistant to UCl and have mechanical power input systems to provide mixing of two immiscible liquids with residence times of ≤ 30 seconds.	Corrosion resistant pipes and their internals made of or protected by suitable plastic materials (such as fluorocarbon polymers) or glass	Mechanical power systems. Sieve plates, reciprocating plates. Or internal turbine mixers	None identified
Liquid-liquid centrifugal contactors	Capability to build and operate centrifuge systems that disperse and separate two immiscible liquids with stage residence times of ≤ 30 seconds and are corrosion resistant to concentrated UCl.	None identified	Contractors made of or are lined with suitable plastic materials (such as fluorocarbon polymers) or with glass	None identified
Electrochemical reduction systems and reduction cells	Skills in the design, production, and operation of reduction cells that are corrosion resistant to concentrated UCl and prevent the re-oxidation of U3+ to U4+	Parts in contact with process stream: suitable materials (glass, fluorocarbon polymers, polyphenyl sulfate, polyether sulfone, and resin-impregnated graphite) to avoid contamination of aqueous stream with certain metal ions. Electrodes (graphite).	Potentiometers	Precise control of uranium valence
Feed preparation systems	Ability to prepare high-purity aqueous solutions of uranium chloride. Concentration of certain metal ions such as chromium, iron, vanadium molybdenum, and other bivalent or higher multivalent cations must be more than a few parts per million.	Parts in contact with final feed solutions: suitable materials (glass, fluorocarbon polymers, poly-phenyl sulfate, poly-ether sulfone, and resin-impregnated graphite) to avoid contamination of the aqueous stream with certain metal ions.	Analytical equipment to monitor purity of solutions	None identified
Uranium oxidation systems	Knowledgeable in the operation of systems for the oxidation of U^{3+} to U^{4+}. Familiarity with the handling of chlorine and oxygen gases and distillation of UCl solutions.	For portions of system processing high-purity U3+ streams: suitable materials (glass, fluorocarbon polymers, polyphenyl sulfate, polyether sulfone, and resin-impregnated graphite) to avoid contamination	Potentiometers	Accurate control of uranium valence
Ion exchange columns	Ability to design, construct, and operate cylindrical columns > 1 m in diameter made of or protected by materials resistant to concentrated UCI and are capable of operating at a temperature of 100^{o} C to 200^{o} C and pressures >0.7Mpa (102psi)	Fast–reacting ion exchange resins or absorbents	Provide characteristics of glass substrate and resin	Physical and chemical characteristics of resin

Table 12 <u>Technologies and components used in chemical and isotope exchange enrichment facilities</u>

Technology	Technical issues
Liquid-liquid Exchange Columns	Judicious handling of columns to prevent breaching of interior coating or lining. The instability of U^{3+} in aqueous solution demands expertise in uranium-solution chemistry.
Liquid-liquid Centrifugal Contactors	Protection of corrosion-resistant lining is paramount. The instability of U3+ in aqueous solution demands expertise in uranium-solution chemistry.
Electrochemical Reduction Systems and Reduction Cells	Must prevent re-oxidation of uranium
Feed Preparation Systems	Product must be of very high-purity with little metallic contamination.
Uranium Oxidation Systems	Chlorine gas is highly toxic and must be handled with extreme care. Pure oxygen gas may bring about rapid combustion and fire.
Ion Exchange Columns	The preparation of the resin/absorbent is the key and has proven very difficult.
Ion Exchange Reflux Systems	The appropriate metals to use in the regeneration system have not been well identified.

Table 13 Technology and Technical Issues involved with chemical and isotope exchange enrichment facilities

reductions in energy use, improved catalysts, and improved regeneration techniques must occur before the technology can compete with other enrichment processes.

3.4.3 Countries that have used/attempted to use technology

Although a number of countries (e.g., the United States, Brazil) have expressed an interest in chemical and isotope exchange technologies, only France and Japan established research and development programs to determine if the process was technically viable and competitive. As a part of its nuclear weapons program, Iraq entered into discussions with Japan and France in an attempt to purchase or surreptiously acquire the technology. Table 14 shows the time frames involved in program development.

Country	Program Initiation	First Lab Scale Pilot Plant Operational	Time to First Lab Scale Pilot Plant	First Significant Enrichment Quantities	Time to first Significant Enrichment	Program stops
France	1968	1974	6	1977	16	1988
Japan	1972	1979	7	1984	12	1992

Table 14 Chemical isotope enrichment programs

3.4.3.1 France

France began its chemical enrichment developmental research program in 1968 after French scientists discovered a new chemical process for enriching uranium[77]. By 1974,

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[77] Nuclear Energy and Nuclear Weapons Proliferation, J.H. Coates and B. Barre, Stockholm International Peace Research Institute, 1979

two laboratory scale enrichment plants were operating[78]. The assumed advantage of this process was that it is difficult and time consuming to enrich to high levels, so this could be considered a "non proliferation approach".

In 1984, a demonstration plant was constructed and put into operation. The facility was modular to allow easy scale-up. It was a pre-industrial process pilot plant that used large-diameter, low-height pulse columns twenty meters high and 380 mm in diameter and was to be capable of producing 1000 swu/yr. A hypothetical commercial plant of this type would have 20 columns per cascade arranged in modules of two vertical cascades, with columns 25-30 m. high and 1,200-1,600 mm in diameter. Each cascade would be capable of producing 250,000 swu/yr and an industrial module capacity would be 500,000 swu/yr. The process would be economic at "several million swu/yr." This facility ran for one year, completed its test runs, and was shut down[78,79].

In 1987, another test facility was constructed and operated. But, in 1988, the French decided to terminate the chemical-enrichment program to concentrate on laser enrichment despite positive results.[80]

A production, chemical-enrichment facility was never constructed. It took six years for the French to build a lab-scale, pilot plant and sixteen years to build a pilot scale demonstration facility.

3.4.3.2 Japan

In 1972, Japan began research on chemical enrichment and, in 1979, completed a small bench chemical-enrichment facility. The facility consisted of a module of five columns. By 1982, it had enriched uranium to 2.2%. It took about four months after input of uranium solution for the enrichment process to reach this level[81].

In 1984, a larger facility, using improved adsorbent and larger reaction columns, obtained 3.2% enriched uranium. Construction of a larger demonstration facility was planned but delayed. In 1992, Japanese officials decided to slow development on chemical enrichment even though the development had achieved visible success. Work on chemical enrichment was halted.

A production, ion-exchange, enrichment facility was never constructed. It took seven years to construct a lab scale pilot plant and twelve years to construct a pilot scale demonstration enrichment facility.

3.4.3.3 Iraq

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[78] CEA Readies Chemical Enrichment Pilot, But Expansion Hinges on Foreign Demand, Ann MacLachlan, Nuclear Fuel: Vol. 9, No.3; Pg. 5, 1/30/84

[79] French find Chemical enrichment is a pleasant economic surprise, Ann MacLachlan, Nuclear Fuel; Vol. 12, No. 2; Pg. 1, 1/26/1987

[80] Laser enrichment up, breeder R&D down in CEA's new budget, Ann MacLachlan, Nucleonics Week Vol. 28, No. 5; Pg. 5, 1/29/87

[81] Asahi Enriches to 3.2% with Chemical Process, Nuclear Fuel, Vol. 9, No. 10; Pg 8, 5/7/84

As a part of its nuclear weapons acquisition program, Iraq began investigating both the Japanese and French chemical and ion exchange enrichment processes. In 1991, Iraq admitted that they had constructed facilities based on both French and Japanese chemical enrichment processes[82]. It was revealed that they had been pursuing laboratory-scale work similar to both the French and Japanese methods, had obtained encouraging results, and were planning on building a pilot plant to prove the concept.

They were studying the possibility of producing 6-8% enriched uranium and using the product as feed for their EMIS program[83].

The Iraqi work never got beyond the laboratory stage.

3.4.4 Technological Information Acquisition/Development

3.4.4.1 Acquisition approaches

Of the three possible approaches for acquisition, Japan and France attempted in-country development, while Iraq attempted to purchase or divert information about the processes.

3.4.4.2 Time constant for technology development

Both France and Japan concurrently developed different approaches to chemical enrichment. It took France six years and Japan seven to produce laboratory-scale demonstration facilities, going from original concept to demonstration of that concept.

3.4.5 Required Concurrent Technologies

Chemical enrichment programs are based on standard chemical engineering technology (except for proprietary catalysts and ion-exchange resins), so such a level of expertise would be needed to develop this process. Particular skills demanded include expertise in uranium solution technology, preparation of resins and adsorbents, and handling of chlorine gas.

3.5 AERODYNAMIC ISOTOPE SEPARATION TECHNOLOGY HISTORY

3.5.1 Technology Description

3.5.1.1 Origin

German scientists first reported research in aerodynamic isotope separation in 1955. Two processes for the use of this technology were developed during the 1960's, one in Germany (the jet-nozzle approach) and one in South Africa (the advanced vortex tube approach). In 1975, Brazil and Germany agreed to jointly develop and commercialize the

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[82] Iraq's Nuclear Secrets, Ze'ev Schiff, The Jerusalem Report, 8/15/91

[83] A proliferation primer: Nuclear Proliferation, David Albright, Bulletin of the Atomic Scientists; Vol. 49; No. 5; Pg 14, 6/1/1993

jet nozzle approach. South Africa used the advanced vortex-tube approach to produce weapons grade HEU and LEU for commercial reactor fuel.

3.5.1.2 Basic theory

A key breakthrough in the development of aerodynamic isotope separation was the discovery that the separation effect of the processes can be substantially increased by diluting the UF_6 with a light auxiliary gas, either hydrogen or helium. Addition of these light gases increases the UF_6 flow velocity significantly for a given pressure ratio.

3.5.1.2.1 Jet Nozzle

In this process, a jet of gas consisting of roughly 96% hydrogen and 4% UF_6 is allowed to expand through a narrow slit. The gas moves at supersonic speeds (comparable to that found in centrifuges) parallel to a semicircular wall of very small radius of curvature (see Figure 4). If the speed of the gas is 400 m/s and the radius of curvature is 0.1mm, then the centrifugal acceleration achieved is 1.6×10^9 m/s, or 160 million times that of gravity. The centrifugal forces on the molecules cause the streamlines of the heavier $^{238}UF_6$ molecules to move closer to the curved wall than the $^{235}UF_6$ molecules. At the other side, where the gas has changed direction by 180^o, a sharp 'skimmer' separates the flow into an inner 'light' fraction and a 'heavy' fraction.

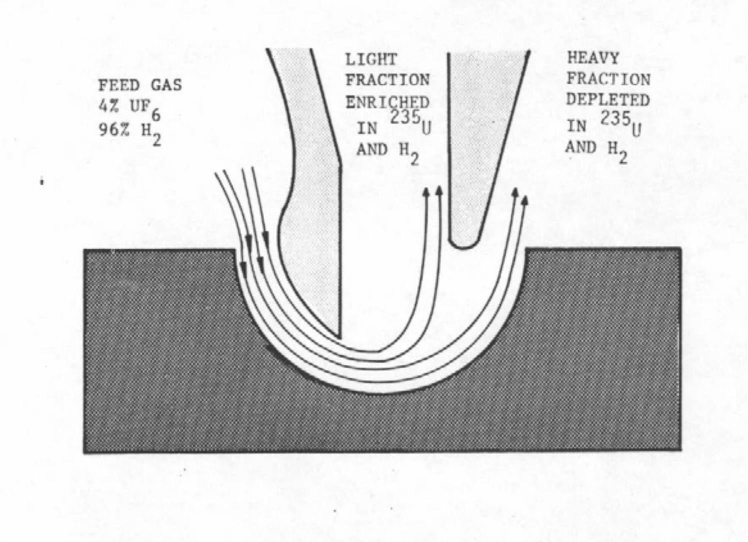

Figure 4 Separation nozzle

The position of the skimmer is arranged so that one-quarter of the total UF_6 content is extracted in the light fraction. This is the 'cut' (the ratio of product flow to feed flow). The other three-quarters of the total UF_6 in the heavy fraction is depleted in $^{235}UF_6$ and forms the tails from the separating element[84].

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[84] Uranium Enrichment and Nuclear Weapon Proliferation, Krass, Boskma, Elzen, and Smit; Stockholm International Peace Research Institute, 1983

3.5.1.2.2 Advanced Vortex Tube

In this process, also called the "stationary-walled centrifuge" approach, a mixture of 1-2% of UF_6 and 98-99% hydrogen is compressed and enters a vortex tube tangentially (see Figure 5). This tangential injection of gas results in a spiral or vortex motion within the tube, and two gas streams are withdrawn at the opposite end of the vortex tube. The spiral-swirling flow decays downstream of the feed inlet due to friction at the tube wall. Consequently, the inside diameter of the tube is tapered to reduce the decay in the swirling flow velocity. This process is characterized by a separation with a very small cut, about 1/20 (compare with ¼ cut for the jet nozzle approach).

Due to the combination of the very small cut of the vortex tube stages and extremely difficult piping requirements necessary based on traditional piping stage methods, the South Africans developed a cascade design technique called a *Helikon*. In essence, the Helikon technique permits 20 separate stages to be combined into one large module, and all twenty stages share a common pair of axial-flow compressors. The axial-flow compressors must successfully transmit parallel streams of different isotopic compositions without significant mixing for this method to succeed.[85]

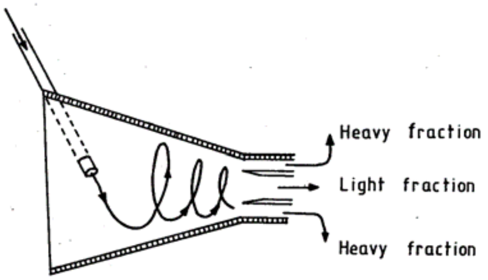

Figure 5 Vortex tube

3.5.1.3 Important components/materials

Following is a list of important components and materials needed to support development of aerodynamic, isotope–separation, processing facilities. Governmental and international agreements control the acquisition of many of these materials.

- Separator elements: nozzles, jets and vortex tubes
- UF_6 carrier-gas separation equipment
- Separation element housings
- UF_6-hydrogen (or helium) gas compressors, gas blowers, and rotary shaft seals
- Heat Exchangers
- Feed Systems/Product and Tail Withdrawal Systems

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[85] Department of Defense, Militarily Critical Technologies List, Part II: Weapons of Mass Destructions Technologies, Section 5 – Nuclear Weapons Technology, February 1998.

- Process piping systems and header systems
- Vacuum Systems and Pumps

Table 15 identifies specific technologies used in aerodynamic isotope separation enrichment facilities, describes the technology level needed for successful use, lists critical materials and unique equipment, and identifies export control references in place for that technology.

Technology	Sufficient Technology Level	Critical Materials	Unique Test, Production, and Inspection Equipment	Unique Software and Parameters
Separator elements: nozzles, jets, and vortex tubes	Nozzle slit-shaped, curved channels with a radius of curvature less than 1 mm, knife-edge to separate the gas flow. Vortex tubes: cylindrical or tapered, 0.5cm to 4-cm diameter, length to diameter ratio of $\leq$ 20:1, one or more tangential inlets	UF$_6$ resistant materials	Test facility to measure isotopic separation performance, pressure drops, etc.	CFD software for nozzle design and performance
UF$_6$/carrier gas separation systems	Designed to reduce UF$_6$ content in carrier gas to $\leq$ 1 ppm. Use of cryogenic heat exchangers and cryo-separators, cryogenic refrigeration units, separation nozzle or vortex tube units, or UF$_6$ cold traps.	UF$_6$ resistant materials	None identified	None identified
Separation element housings	Cylindrical vessels >30cm in diameter and 90 cm in length or rectangular vessels of comparable dimensions. Made of protected by UF$_6$ –resistant materials.	UF$_6$ resistant materials	None identified	None identified
UF$_6$ –hydrogen (or helium) gas compressors, gas blowers, and rotary shaft seals	Axial, centrifugal, or positive displacement compressors or gas blowers, suction volume capacity of $\geq$ 2 M^3 /min typical pressure ratio between 1.2:1 and 6:1. Seals with feed and exhaust connections, provide a reliable seal against out-leakage or in-leakage.	UF$_6$ resistant materials	UF$_6$ –hydrogen test loop and instrumentation to determine compressor performance characteristics. Instrumentation to measure seal feed and exhaust pressures and flows to check seal performance	Compressor and seal design and performance models. Blade design codes.
Heat Exchangers	Provide adequate gas cooling, made or protected by materials resistant to UF$_6$	UF$_6$ resistant materials	Test loop to determine heat transfer coefficients and pressure drop	Heat transfer codes for compact heat transfer surfaces.
Shut-off control and bellows-sealed valves	Manually or automatically operated, 40 to 1,500 mm in diameter, made of or protected by UF$_6$ resistant materials	UF$_6$ resistant materials; bellows seals rather than packing glands	None identified	None identified
Feed systems/product and tail withdrawal systems	Feed autoclaves to pass UF$_6$ to the enrichment process; desublimers (cold traps or solidification or liquefaction stations for removal of UF$_6$ into containers	UF$_6$ resistant materials	Mass spectrometers/ion sources. Autoclaves. Flow, mass, pressure, and temperature instrumentation	None identified
Process piping systems and header systems	Piping network normally of the “double” header design with each stage or group of stages connected to each header	UF$_6$ resistant materials	None identified	None identified
Vacuum systems and pumps	Vacuum systems having a suction capacity of $\geq$ 5m^3/min with vacuum manifolds, headers and pumps designed for service in corrosive atmosphere. In this context, the materials being treated may contain strong acids or fluorine that reacts with materials in pumps and headers. Pumps may have fluorocarbon seals and special working fluids.	UF$_6$ resistant materials. Hydrocarbon or fluorocarbon vacuum pump oils.	None identified	None identified

Table 15 Technologies and components used in aerodynamic isotope-separation facilities,

3.5.2 Technology Development

3.5.2.1 Technological difficulties in use

There are a number of technological difficulties involved in the use of aerodynamic isotope separation approach. Only two countries (Brazil and South Africa) have attempted to use it for commercial purposes but found it not competitive with other processes. It can be made to work, as South Africa proved when this process formed the basis for their weapons program.

A list of potential difficulties includes

- Heat removal - the high proportion of carrier gas (hydrogen or helium) required in relation to UF_6 results in high specific-energy consumption and substantial requirements for removal of waste heat.
- Component production – due to economic considerations, process designers select separation nozzles with physical dimensions as small as manufacturing technology will allow. The curved wall of the nozzle may have a radius of curvature as small as 10 μm (0.0004 inch). Production of these tiny nozzles, by such processes as stacking photo-etched metal foils, is technically demanding.
- Energy consumption - a primary reason for the South African phase-out of their process was because of the high energy cost of the process.[86]
- Hydrogen handling - UF_6 reacts strongly with hydrogen at elevated temperatures, so care must be taken to keep this reaction from occurring

3.5.3 Countries that have used/attempted to use technology

The concept of aerodynamic isotope separation was developed in Germany in the 1950's. Only Brazil, with the help of Germany, and South Africa continued the development of this approach.

3.5.3.1 Germany

Scientists in Germany published papers beginning in the 1950's describing successful experiments using isotope separation. In 1960, patents were issued addressing process basics, and serious development work began. In 1967, an experimental laboratory-scale jet-nozzle facility was put into operation.[87] In 1976, Germany and Brazil signed an agreement to jointly develop jet-nozzle technology. In 1976, a demonstration pilot plant was completed that successfully demonstrated the process.[88] In 1980, the facility was moved to Brazil where the development work was being performed. By 1989, all research and development in Germany had ended.[89]

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[86] Centrifuges or Lasers may replace South Africa's present SWU plant, Mark Hibbs, Nuclear Fuel: Vol 16, No. 1; Pg 4, 1/7/1991
[87] Jet Nozzling, The Economist: "The world: international report:", 10/11/1975
[88] More power for uranium enrichment, Chemical Week, pg 33, 7/21/1978
[89] Karlsruhe Deemphasizing Traditional Nuclear Endeavors, Mark Hibbs, Nucleonics Week: Vol. 28, No. 11; Pg. 10, 3/12/1997

Table 16 provides a list of technologies and related technical issues.

Technology	Technical Issues
Separator elements: nozzles, jets, and vortex tubes	Precision in fabricating very small nozzles, sophisticated machine shop
UF_6 carrier-gas separation equipment	Large building ventilation system, H_2 generating site, explosive mixture concerns
Separation element housings	Sealing and welding technologies, aerodynamic efficiency, minimum leakage and corrosion.
UF_6-hydrogen (or helium) gas compressors, gas blowers, and rotary shaft seals	Aerodynamics, rotor dynamics, lubrication, blade/vane stress and vibration, minimize leakage, corrosion, failure rates
Heat Exchangers	Substantial waste heat, cooling tower design
Shut off, control, and bellows-sealed valves	Minimize leakage and corrosion
Feed Systems/Product and Tail Withdrawal Systems	Maintain material balance. Criticality concerns with HEU
Process piping systems and header systems	Minimize leakage and corrosion, sealing and welding technologies
Vacuum Systems and Pumps	Minimize leakage. Containment and cleanliness.

Table 16 Technology and Technical Issues involved with aerodynamic isotope separation

It took seven years to get a laboratory-scale pilot plant in operation and another nine for a pilot plant to be constructed to prove the concept.

3.5.3.2 Brazil

After the agreement with Germany was signed in 1976, Brazil made plans to build a 5 million swu/yr ‘demonstration’ enrichment facility. The German pilot plant was moved to Brazil in 1980, where it was used to test functional parameters of the process, including dynamic stability of the gases used and stress and fatigue limitations of equipment.[90]

Over the next few years a number of problems with plant equipment (especially compressors) and design caused significant delays in the plant completion schedule. The facility was completed in 1985, two years behind schedule.[91] It produced LEU enriched to 0.85% during testing in 1988, and went into operation in 1990 with one complete cascade. It was capable of enrichment only to about 0.8%.[92]

In 1994, Brazil closed its facility because of high production cost and a history of technical problems.

To summarize, Brazil did not build a pilot plant, but rather purchased one from Germany. Its demonstration facility took nine years to construct, and after two years of testing was able to enrich to only 0.8%. After four years of operation, the plant was closed.

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[90] Critics see Brazil as taken for a ride on its "Last Train" to SWU technology, Charles Thurston, 4/26/82

[91] Budget cuts delay testing and expansion of Brazil's Jet-nozzle enrichment plant, Paul Lyons,

[92] Brazil will develop domestic fuel cycle if its nuclear power capacity warrants, Mark Hibbs, Nuclear Fuel: Vol. 14, No. 19; Pg. 6, 9/18/89

3.5.3.3 South Africa

South Africa began work on aerodynamic isotope separation in 1960. By 1967, a laboratory-scale facility had demonstrated the feasibility of the vortex tube approach. In 1969 a pilot-scale facility was built, with production capability of 10-20,000 swu/yr. While construction of the pilot facility was complete in 1974, it took four years before significant quantities of HEU were produced.

In 1978, the facility was able to produce 45% enriched HEU for research reactor fuel and some HEU for the South African weapons program. However, technical problems with facility design and component materials resulted in the facility shutdown for rework in 1979.[93] The facility was taken apart and reconstructed. Improvements included a change of component materials (e.g., nickel-plated steel components replaced aluminum ones).[94] In 1981, the facility resumed operation with improved throughput and continued to operate until 1990.

A “semi-commercial” facility intended to produce LEU only and with a design throughput of 300,000 swu/yr was completed in 1986 and went on line in 1988. The facility was shut down in 1995 as it was not competitive on the world market due to high-energy costs.[95] This was not unexpected because when the facility was originally planned, it was acknowledged that it eventually would need to produce 3 million swu/yr to be competitive in the world market.[96]

To summarize South Africa’s experience from the time it began work on aerodynamic isotope separation, it took seven years to construct a laboratory-scale demonstration facility, eighteen years to produce HEU in a pilot-scale facility, and twenty eight years for a commercial scale plant to produce LEU.

3.5.4 Technological Information Acquisition/Development

3.5.4.1 Acquisition approaches

Table 17 summarizes the aerodynamic isotope separations program timelines. The German and Brazilian timelines are combined, as the Brazilian program was essentially a continuation of the German one. As such, both of these programs can be considered to be indigenous programs.

The timeframes for both programs are remarkably similar, although the South African program could be considered more successful than the German/Brazilian one in that it actually produced significant quantities of HEU. In both cases, economics was the prime cause for program cancellation.

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[93] South Africa and the affordable bomb, David Albright, Bulletin of the Atomic Scientists, 7/1/1994

[94] South Africa's Secret Nuclear Program: From a PNE to a Deterrent, Mark Hibbs, Nuclear Fuel: Vol. 18, No. 10; Pg. 3, 5/10/1993

[95] Centrifuges or Lasers may replace South Africa's present SWU plant, Mark Hibbs, Nuclear Fuel: Vol. 16, No. 1; Pg 4, 1/7/1991

[96] Inside Valindaba: South African Enrichment Plant Steadily Taking Shape, Rob Laufer, Nucleonics Week: Vol. 23, No.14; Pg. 1, 4/8/1982

Country	Program Initiation	First Laboratory Scale Pilot Plant	Years To First Laboratory Scale Pilot Plant	First Demonstration Plant Operation	Years To First Demonstration Plant	First Production Facility	Years to First Production Facility
Germany/ Brazil	1960	1967	7	1976	16	1990	30
South Africa	1960	1967	7	1978	18	1988	28

Table 17 Aerodynamic Isotope Separation Program

3.5.4.2 Time constant for technology development

The time constants for the two programs are similar, seven years for a pilot plant, eighteen for a demo plant. The South African facility was much larger and produced significant quantities of HEU, as a part of their weapons program.

3.5.5 Required Concurrent Technologies

Among concurrent technologies needed to support aerodynamic isotopic separation are; cryogenics, advanced compressors, micro photo-etching (to produce nozzles), hydrogen handling.

3.6 LASER ENRICHMENT TECHNOLOGY HISTORY

3.6.1 Technology Description

3.6.1.1 Origin

Laser enrichment technology originated in the United States at the Los Alamos Laboratory in 1971. In that year, measurable quantities of enriched uranium were first produced on a laboratory scale using prototype laser enrichment technologies.

3.6.1.2 Basic theory

The basic theory behind laser-enrichment technology involves the use of specific a wavelength of light to excite the Uranium-235 atoms in a source material. This allows desired atoms to be separated from the source material and provides the possibility for enrichment to be performed.

Four laser-enrichment methods have been pursued. They are 1) Atomic Vapor Laser Isotope Separation (AVLIS), 2) Molecular Laser Isotope Separation (MLIS), 3) Chemical Reaction Isotope Selective Laser Activation (CRISLA), and 4) Separation of Isotopes by Laser Excitation (SILEX).

3.6.1.2.1 Atomic Vapor Laser Isotope Separation

Lasers used in AVLIS are tuned so that only ^{235}U atoms are excited, creating positively-charged ions. The ^{235}U atoms are deflected by an electrostatic field and collect on a product collector. Other atoms remain neutral and pass through the collector.[97]

The process consists of a laser system and a separation system. The laser system is a pumped laser system comprised of one laser used to optically pump a separate dye laser (dye master oscillator laser) that produces the light used in the separation process.

Principal advantages of the AVLIS process include a high separation factor, low energy consumption (approximately the same as the centrifuge process), and a small volume of generated waste. However, no AVLIS process has been deployed because process implementation is difficult and expensive. The process requires sophisticated hardware constructed of specialized materials that must be capable of reliable operation for extended periods of time in a harsh environment.

This process is referred to as SILVA in France.

3.6.1.2.2 Molecular Laser Isotope Separation

During the MLIS process, UF_6 is energized using an infrared laser system, exciting the $^{235}UF_6$. Photons from a second laser system (infrared or ultraviolet) preferentially dissociate the excited $^{235}UF_6$ to form $^{235}UF_5$ and free fluorine atoms. The $^{235}UF_5$ precipitates and can be filtered from the gas stream.

Principal advantages of this process are low power consumption and use of UF_6 as the process gas. There are, however, many complexities with the MLIS systems, and most countries have terminated their programs.

3.6.1.2.3 Chemical Reaction Isotope Selective Laser Activation

In the CRISLA process, a carbon monoxide (CO) laser is used to illuminate an intra-cavity cell filled with gaseous UF_6 and a co-reactant.[98] The laser frequency is tuned to a special selected value at which UF_6 molecules containing ^{235}U are preferentially excited. This is possible because of the isotope shift between the ^{235}U and ^{238}U absorption bands. In the process, the CO laser at a selected laser frequency illuminates a mixture of natural UF_6 and a proprietary reagent called *"RX"*. At that point, the reaction rate of excited $^{235}UF_6$ molecules with RX is enhanced by more than a thousand times over the reaction rate of unexcited $^{238}UF_6$ molecules with the RX. The reaction product is therefore enriched. Since the enriched product is chemically and physically different from UF_6, it can be separated by standard chemical engineering techniques.

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[97] *Militarily Critical Technologies List*, Part II, Office of the Under Secretary of Defense for Acquisition and Technology, Washington D.C., February 1998

[98] Nuclear Engineering International, Fuel Review: Market Trends; Enrichment: In The Wake Of The USEC Privatization, PG. 16, Wilmington Publishing Limited, September 30, 1998

The laser serves only as an activator. The energy to separate ^{235}U from ^{238}U is mostly chemical. No expensive laser power is required to dissociate or ionize uranium-bearing molecules or atoms as proposed in other laser-enrichment schemes.

CRISLA has also been used to separate deuterium from hydrogen.

3.6.1.2.4 Separation of Isotopes by Laser Excitation

This technology has the potential advantages of being simpler and less expensive than AVLIS. It is a UF_6-based technology and not a uranium metal technology and therefore would not involve a change from the existing UF_6 fuel cycle. SILEX is suitable for separating isotopes of chlorine, molybdenum, and uranium. SILEX has the potential to be efficient enough to use depleted uranium to produce the same assays as natural uranium. This is a third-generation, laser-enrichment technology that has the potential for low capital cost in addition to low operating costs

The Australian company, Silex Systems Limited, began development of the process in 1982. The technology was proven to work on a laboratory scale in 1994. In 1996, a license agreement was signed with the United States Enrichment Corporation (USEC). Under the agreement USEC will fund all of Silex's research and development costs as long as certain milestones are met. All development work is being conducted in Australia.

The SILEX development program with USEC involves three stages. The Pilot Module Program was successfully completed in January 2000, triggering a U.S. $5 million milestone bonus payment to Silex. The Pilot Engineering Study was scheduled for completion in 2001 and the Pilot Plant Program in 2003.

The SILEX technology has applications other than uranium enrichment that are also being pursued, including both silicon and carbon enrichment.

3.6.2 Technology Development

As described above, several processes have been tried but none have been successfully deployed.

3.6.3 Countries that have used/attempted to use technology

The countries that have attempted the development of these techniques are shown in Table 18.

Country	**Technology**	**Dates**	**Reason for stop**
Australia	SILEX	1982 - present	--
United Kingdom	MLIS	1983 - 1994	Urenco announces stop of work, funds go elsewhere
Canada	CRISLA	1990-1993	Stops funding
France	SILVA	1984-1996	France starts looking at MLIS
France /South Africa	MLIS	1996- 1997	France withdraws from development
Japan	AVLIS	1982 - 2001	Japan's Ministry of Economy cancels program
United States	AVLIS	1984-1999	Funding stopped
United States/Australia	SILEX	1997 - present	--

Table 18 **Laser Technology Acquisition/Development**

Other countries have been identified that have attempted development programs for laser-enrichment technologies but no formal program has been defined or results shown. These include but are not limited to:

- Brazil
- China
- Germany
- India
- Iran
- Israel

3.6.4 Technological Information Acquisition/Development

3.6.4.1 Expert Knowledge

Laser enrichment technology is still in the research/development stage. Therefore, the most valuable resource that can be acquired is the knowledge of the experts in the field. For example, in 1997, when South Africa stopped development on their laser enrichment program, their laser experts immigrated to Australia.

Other items that are necessary for development that may be pursued for acquisition include:

3.6.4.2 Material

A difficult acquisition for this process may be the uranium feed material. Laser enrichment usually requires UF_6 feed. Agreements between countries can result in transfer of feed material for research use. For example, in 2000, the NRC approved export of 33.5 kilograms of uranium enriched to a maximum of 9.9 % ^{235}U to Australian for Silex-process evaluation.

3.6.4.3 Equipment

Although lasers are used in many countries for a variety of purposes, the laser enrichment process requires the use of specific wavelengths. This means that acquisition of equipment aids in accelerating the research/development efforts for interested countries.

For example, in 1978 American scientists built and delivered four lasers to an Iranian research center. In 1998, India tried to obtain modern nuclear-use lasers; and, in 1998, Pakistan attempted to procure AVLIS equipment in Europe.

3.6.4.4 Technology

Although laser enrichment technology has not yet been successful, there are countries that are willing to pursue the sale of information that they have developed on different processes. For example, in 2000, it was detected that Russia may be exploring the sale of laser enrichment technology to Iran.

3.6.5 Required Concurrent Technologies

Since a successful laser-enrichment technology has yet to be developed, it is not certain what additional technologies will be required to sustain a laser-enrichment process. However, as described above, the process does require correct feed material. Most processes tried, including the SILEX process currently under development, require a source of UF_6. Successful deployment of the current laser-enrichment schemes will require that this feed material be present.

4.0 REACTOR BASED TECHNOLOGY

This section describes the various technologies that have been used in the production of weapons grade plutonium. Plutonium does not exist in nature, it is a man-made element. For example, when a ^{238}U nucleus absorbs a neutron in a nuclear reactor, it becomes ^{239}U, which then undergoes two nuclear reactions. First it emits an electron and decays to Neptunium (^{239}Np) and then emits another electron and decays to ^{239}Pu, the desired isotope. As long as it stays in the operating reactor, there is a chance that the ^{239}Pu nucleus will absorb another neutron and become ^{240}Pu. ^{240}Pu is an undesirable isotope, because it is a neutron emitter and makes it difficult to construct a nuclear weapon. Plutonium that contains more than 93% ^{239}Pu is termed "weapons grade", if it contains more than 7% ^{240}Pu it is called "reactor grade".

In a nuclear reactor, neutrons are principally emitted from reactions that occur when a neutron is absorbed by the nucleus of an atom such as ^{235}U. The nucleus splits, or *fissions*, and emits more high energy or "fast' neutrons. In most nuclear reactors, these fast neutrons must be slowed down by collisions or interactions with other nuclei to effectively continue the chain reaction. These "slowing down" materials are called moderators, and one of the most effective moderators is the hydrogen in ordinary or "light" water. However, there is a chance that instead of slowing the neutron down, the hydrogen will absorb it resulting in an overall loss of neutrons. *Heavy water*, consisting of deuterium and oxygen, is another good moderator. It is less likely to absorb neutrons during the slowing down process. Consequently, reactors using heavy water as a moderator can be fueled with natural uranium (containing 0.7% ^{235}U) while reactors using light water must have fuel that is enriched in ^{235}U, usually to at least 3 to 4%. *Graphite*, also a good moderator, has a smaller chance than light water of absorbing neutrons during the slowing down process. Reactors with graphite as a moderator can also use natural uranium as fuel.

Reactors considered in this section are separated into three classes: 1) graphite moderated production reactors, 2) heavy-water moderated production reactors, and 3) research reactors. Both graphite and heavy water moderated reactors have several different configurations that will be described further. Research reactors come in many different configurations, but only those that could be used for plutonium production will be described here. For historical reasons and for ease of narration (excluding certain specific facilities), reactors of less than 100 MWt will be considered research reactors, reactors equal to or greater than 100 MWt are considered production reactors.

Conventional commercial light water moderated reactors are not evaluated in this section, as they have not to date been used in proliferation programs.

No matter what reactor type is pursued, reprocessing of the spent fuel to extract the plutonium is required. Trends in reprocessing are covered in a separation section.

4.1 GRAPHITE MODERATED REACTOR TECHNOLOGY HISTORY

4.1.1 Technology Description

4.1.1.1 Origin

The construction of the first graphite-moderated reactors for the purpose of plutonium production occurred in the United States in the early 1940s. Two versions of graphite-moderated reactors have been developed. The first version is a light-water-cooled, graphite-moderated design (LWGR). The second version is gas-cooled and graphite-moderated (GGR).

The design used in the U.S., the former Soviet Union, and China production programs was the LWGR. The design used in the British and French weapons programs and subsequently the North Korean weapons program was the GGR. The British and French reactors were operated as dual-purpose electricity and production facilities in order to spread costs. They typically utilized batch production for limited weapons runs.

Two other countries, Lithuania and Ukraine, have operated Soviet-designed commercial graphite moderated reactors, but only for the generation of electrical energy. The following is a discussion of the development of graphite-moderated reactor technology in the six countries that have used them for plutonium production.

4.1.1.2 Basic Theory

When graphite is used as a moderator, natural uranium can be used as fuel. This is an advantage because it eliminates the need for enrichment facilities. The usual design concept for graphite-moderated reactors consisted of a configuration of graphite blocks surrounding fuel inserted in tubes or channels. The fuel is cooled either by water or gas passing through or around the tubes containing the fuel. Although graphite is less likely to absorb neutrons than water, it is also less effective at slowing them down. Consequently for a given quantity of fuel there must be a larger volume of graphite than there would be water for a light-water-moderated reactor. Therefore, graphite-moderated reactors are generally larger than light-water-moderated reactors.

The longer a given amount of fuel is irradiated in a reactor, the greater the buildup of ^{240}Pu. Consequently, production reactors are designed and operated to have the fuel stay in the reactor for a short period of time compared to a commercial power reactor; nominally a month as opposed to a year or more. Some reactor designs allow refueling while operating at power, others require shutdown with rapid refueling.

4.1.1.3 Important components/materials

Most of the components/materials needed to construct a graphite-moderated reactor can be commercially obtained. Graphite of very high purity is required to produce neutrons in the correct energy range where weapons grade plutonium can be effectively produced.

4.1.2 Technology Development

4.1.2.1 Technological difficulties in use

Graphite-moderated reactor design is a mature and well-understood technology. Enrichment facilities are not required, as natural uranium is used. Only uranium, fuel manufacturing technologies, and reprocessing facilities are necessary.

Commercial civilian designs use graphite-moderated reactors for producing electricity. A major operating difference between commercial civil reactors used to generate electricity and plutonium production reactors is that fuel is discharged from the production reactors more frequently. A significant consequence of this lower exposure time is that the discharged fuel is less radioactive and would not have to be left for as long to cool before processing. [99]

4.1.2.2 Changes/improvements in technology

The concept and design of the graphite-moderated reactor is relatively simple. From a proliferating country's point of view, a reactor with a design similar to one of the original GGR U.S. or British designs (e.g. CP-1 or X-10 see Figure 6[100]) would be probably be sufficient to meet their needs.

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[99] Plutonium and Highly Enriched Uranium 1996, World Inventories, Capabilities and Policies, David Albright, Frans Berkhout, and William Walker, Stockholm International Peace Research Institute (SIPRI), Oxford University Press, Inc, New York 1997.

[100] Penn State University Press

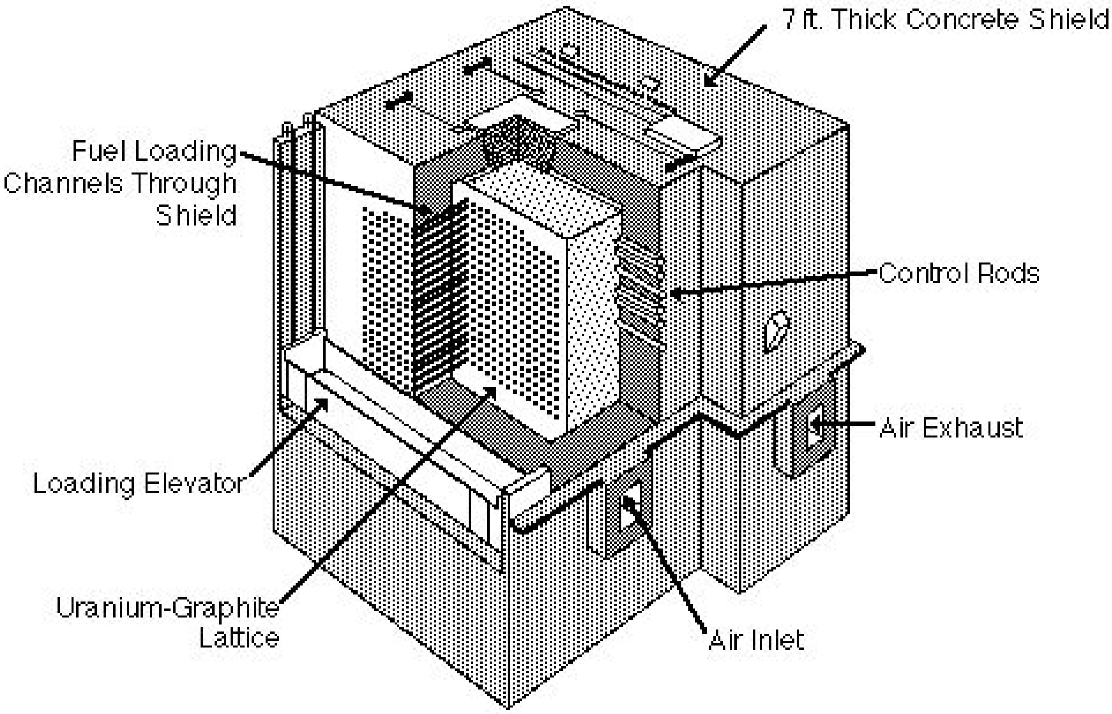

Figure 6 **X-10 Reactor**

4.1.3 Countries that have used/attempted to use the technology

Six counties (Table 19) have used graphite moderated reactor technology to produce weapons grade plutonium:

United States	France
Soviet Union	China
United Kingdom	North Korea

Table 19 **Countries with graphite-moderated production reactors**

The graphite-moderated reactor programs in each of the six countries having plutonium production programs are described below. Because many reactors have been constructed, individual construction times for each reactor will not be described. Instead average time constants for development and construction will be developed and described in section 4.1.5.

4.1.3.1 United States

In December 1942, the world's first self-sustaining nuclear chain reaction was demonstrated on a squash court beneath Stagg Field at the University of Chicago campus by a graphite reactor (CP-1). Before large reactors could be built, a pilot plant was constructed to prove the feasibility of scaling up from laboratory experiments. The X-10 pilot scale reactor was constructed at the Oak Ridge site in Tennessee and went critical in

October of 1943. Both the CP-1 and X-10 were small reactors (<100 MWt) and took only a few months to assemble.

Between 1945 and 1963, nine light water cooled graphite moderated production reactors were constructed at the Hanford Reservation in Washington State. Six of these reactors (B, C, D, DR, F and H Reactors) initially operated at 25 MW, but eventually reached 2500 MWt. The remaining three (KE, KW, and N Reactors) operated at 4400 MWt.

The U.S. LWGRs original design used aluminum-clad, natural uranium fuel with horizontal pressure tubes. They were operated in a "batch mode", i.e. shut down approximately every thirty days to discharge and reload fuel. The N Reactor, which began operation in 1963, was an improved design. Instead of aluminum pressure tubes and fuel cladding, it used zirconium clad fuel in zirconium pressure tubes. It was a "dual purpose" reactor, operated to produce both plutonium and electricity.

Between 1965 and 1971, the eight original graphite production reactors put into service in the 1940s and 1950s were closed down. By the end of 1988, the N Reactor was shut down and all plutonium production had ceased at Hanford.

4.1.3.2 United Kingdom

The initial plutonium production site in the U.K. was at Sellafield (formally known as Windscale), in northwest England. The two Windscale "piles" (an archaic term used because the reactor cores in these early facilities consisted of piles of graphite blocks) used aluminum clad metallic natural uranium fuel and were air cooled. Weapons grade plutonium was produced at the two Windscale piles between 1951 and 1957, when these two plants were closed after a graphite fire occurred in the core of Windscale unit 1.

After 1957, production came from four reactors at the Calder Hall site and four reactors at the Chapelcross site. These dual purpose reactors were used both for plutonium production and electrical generation. The initial design ratings of these reactors were 180 MWt and 42 MWe. These capacities were upgraded in the 1960s to 240 MWt (54 MWe) and down-graded in the 1970s to 48 MWe (Chapelcross) and 50 MWe (Calder Hall).

The next generation of British GGRs, beginning with the Calder Hall and Chaplecross facilities, were called "Magnox" reactors. These were named after the magnesium alloy (an alloy of magnesium oxide and aluminum) used to clad the natural uranium metal used as the fuel. They used pressurized CO_2 as the coolant. In early designs, the core was contained within a steel pressure vessel surrounded by a steel and concrete biological shield more than one meter thick, with the boilers located outside the shield. Later designs had a steel-lined, pre-stressed concrete pressure vessel that also acted as the biological shield, with the boilers contained inside. Twenty six Magnox reactors were built, and, as of 2003, sixteen were still operating[101].

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[101] "World List of Nuclear Power Plants", Nuclear News, March 2003.

The successor to the Magnox reactors was the advanced gas-cooled reactor (AGR). These reactors use enriched uranium fuel with stainless steel cladding and pressurized CO_2 as the coolant. The advanced gas-cooled reactors operate at a higher temperature than the Magnox design. The reactor is encased in a steel-lined, pre-stressed concrete pressure vessel several meters thick that acts as the biological shield, with the boilers inside. The CO_2 coolant conveys heat from the reactor to the boilers that, in turn, heat water in an isolated steam circuit used to turn the turbines, just as in coal, oil, or gas-fired stations. Fifteen AGRs were constructed, and as of 2003, fourteen were still operating.

Other sources of weapons grade plutonium were the first discharges of fuel from civilian reactors. Until 1969, there was no clear distinction between civil and military plutonium in the United Kingdom, rather the distinction was between weapon- and reactor-grade plutonium.

4.1.3.3 France

France produced weapons-grade plutonium in both military and civilian reactors. France's first graphite reactor was a GGR called G1 that reached criticality in 1956. Initially, its rated power level was 38 MWt but it was upgraded to 42 MWt by 1962. The G2 and G3 reactors, which reached full-power operation in 1959, were also GGRs originally rated at 200 MWt. During the 1960s, the power was gradually increased to 260 MWt. The G1 reactor was shutdown in 1968, the G2 in 1980, and the G3 in 1984.

The French reactors were similar to the British GGRs, but used a magnesium zirconium alloy for the fuel cladding instead of the magnesium oxide and aluminum (Magnox) used by the British. There are few differences between the two materials.

France has acknowledged that it also used its civilian GGRs to make plutonium for its military programs. Reactors in this category include Chinon 1, Chinon 2 and 3, St Laurent-1 and 2, and Bugey 1 with power levels ranging from 300 MWt to 2000MWt. France had a total of nine GGRs, now all shut down.

4.1.3.4 Soviet Union

Although knowledge of the Soviet Union's plutonium production system has grown in recent years, less is known about it than other countries. Site numbers and types of production reactors are known but power ratings and production history remain undisclosed. It is known that although production started at about the same time as in the U.S., it developed more slowly.

There were three production sites in the Soviet Union: 1) Chelyabinsk on the eastern side of the Ural Mountains, 2) Tomsk in southern Siberia, and 3) Krasnoyarsk in eastern Siberia. Five LWGRs were constructed at Chelyabinsk in the late 1940s and 1950s and were shutdown between 1987 and 1990. Five LWGRs were built at Tomsk, three of which have been shut down. Three LWGRs were built at Krasnoyarsk in a secret underground facility, one of which remains in operation. The design rating for these reactors is 2500 MW[102].

Soviet production reactor designs used aluminum-clad, natural uranium fuel slugs; the reactors were of vertical pressure tube design capable of on-line refueling.

Today in the former Soviet Union countries (Lithuania, Russia and Ukraine), there are twenty civilian LWGRs in operation. For many of these facilities, design life dates extend past 2020.

4.1.3.5 China

The Soviet Union began helping China construct a LWGR in 1960, but then withdrew its technical assistance in the same year. However, enough technology had been transferred to enable construction to continue. The first production reactor, a LWGR (250 MWt), began operating in 1966. This reactor, at the Juiquan Atomic Energy Complex, experienced frequent technical difficulties and interruptions because of political turmoil in China during the Cultural Revolutions and was shut down for repair between 1973 and 1975. In the mid-1980s, it was re-engineered so that it could operate as a dual-purpose electricity and plutonium producer.[103]

Another production site was developed at Guanyuan in Sichuan Province where a larger LWGR (500 MWt) was constructed. There is no evidence to suggest that any other units besides the Juiquan and Guanyuan units have existed. The two LWGRs were shut down in 1991.

4.1.3.6 North Korea

Korea initiated a highly-secret nuclear program. It escaped detection by outside international inspectors until a U.S. intelligence report revealed in the early 1980s that North Korea was constructing a small nuclear reactor at Yongbyon. The reactor was a GGR similar to an early British Magnox design with a power level of 20-30 MWt.[104]

In 1989, it was discovered that North Korea was building bigger reactors copied after French Magnox designs; one at Yongbyon rated at about 200 MWt and one at Taechon rated at about 800 MWt. Still, North Korea refused, at first, to enact the International Atomic Energy Agency (IAEA) safeguards on all its nuclear activities but eventually entered into a safeguards agreement with IAEA in 1992. Despite this, North Korea disallowed access to the suspected construction sites of two other reactors and had many other confrontations with the IAEA.

At the end of 1995, North Korea signed the “Agreed Framework” with the U.S., Japan, and South Korea, agreeing to shutdown the small reactor and halt construction on the two

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[102] The History of the Soviet Atomic Industry, Arkadii Kruglov, Taylor and Francis Press, 2002

[103] China Builds the Bomb, Stanford University Press, John Lewis and Xue Litai, Stanford California, 1988.

[104] Solving the North Korean Nuclear Puzzle, David Albright and Kevin O’Neil, Institute for Science and International Security (ISIS) 2000.

larger ones. Late in 2002, however, North Korea announced that the Agreed Framework was “dead” and began preparations to restart the reactor.

4.1.4 Technological Information Acquisition/Development

4.1.4.1 Acquisition approaches

The design construction of graphite-moderated reactors for the purpose of plutonium production was developed indigenously in the U.S., the Soviet Union, the United Kingdom and France. Each country developed a different graphite-moderated reactor design. Technology for the reactors in China was transferred from the Soviet Union before they withdrew assistance in 1960. The first small North Korean reactor was modeled after the British Magnox and the larger reactors after the France GGRs. General technical support was provided to North Korea by China.

4.1.4.2 Time constants for technology development

Following is a determination of time constants for the development and construction of a graphite-moderated reactor. Construction time appears to differ according to 1) whether it was a simple early generation reactor or a later, more evolved one, 2) the country of origin, and 3) whether it was a civil- or government-sponsored program.

Table 20 provides the type and number of dual purpose and production reactors put into operation in each country. Figure 7 shows this graphically. Table 21 provides summary construction information for all known graphite-moderated reactors in the world (except research reactors rated less than 100 MWt). For each country, the average construction time, the number of reactors, and the fastest and slowest construction times are given for each decade starting in the 1940s up until the present. Table 22 is a distillation of the information in Table 21 for general reactor regimes. The values in Table 22 could be viewed as time constants for the construction of graphite-moderated reactors.

Country	Reactor Type	Number
Britain	Dual purpose production and civilian GGRs	43
China	Production LWGRs	2
France	Dual purpose production and civilian GGRs	9
North Korea	Production GGRs	1
Lithuania	Civil LWGRs	2
Ukraine	Civil LWGRs	4
USSR	Production and civil LWGRs	43
U.S.	Production LWGRs	9

GGR – Gas cooled graphite moderated
LWGR – Light water cooled graphite moderated

Table 20 Countries with Graphite moderated reactors and number put into operation

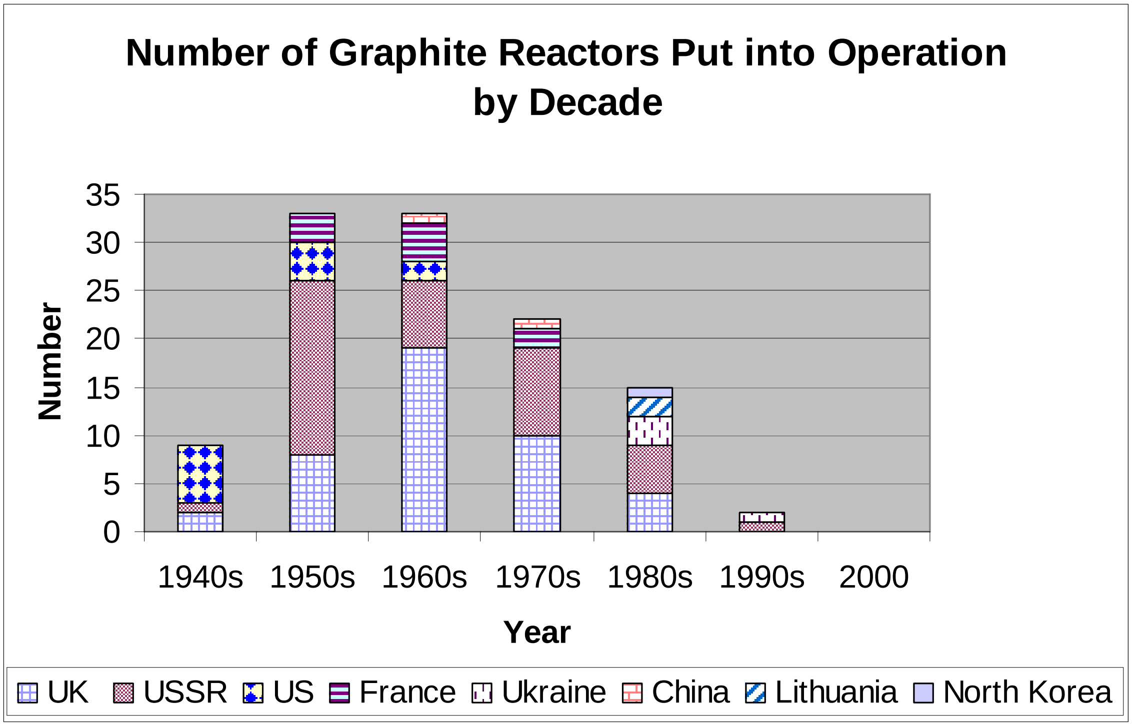

Figure 7 **Graphite reactors put into operation by decade**

Country		**1940s**	**1950s**	**1960s**	**1970s**	**1980s**	**1990s**
Britain (GGR)	Ave. time No. Units Range	3 yrs 2 units 3-4 yrs	4 yrs 8 units 3-6 yrs	6 yrs 19 units 5-8 yrs	13 yrs 10 units 9-19 yrs	9 yrs 4 units 8-9 yrs	
China (LWGR)	Ave. time No. units Range			6 yrs 1 unit 6 yrs	8 yrs 1 unit 8 yrs		
France (GGR)	Ave. time No. units Range		4 yrs 3 units 4 yrs	6 yrs 6 units 5-7 yrs			
Korea (GGR)	Ave. time No. units Range					6 yrs 1 unit	9 yrs[1] 2 unit 8-10 yrs
Lithuania (LWGR)	Ave. time No. units Range				6 yrs 1 unit 6 yrs	9 yrs 1 unit 6 yrs	
Ukraine (LWGR)	Ave. time No. units Range				5 yrs 3 units 4-5 yrs	4 yrs 1 unit 4 yrs	
Russia (LWGR)	Ave. time No. units Range	1 yr 3 units 1 yrs	6 yrs 22 units[2] 4-9 yrs	6 yrs 5 units[3] 5-6 yrs	5 yrs 9 units 3-7 yrs	6 yrs 4 units 4-9 yrs	
U.S. (LWGR)	Ave. time No. units Range	1 yr 8[4] units 1-3 yrs	2 yrs 3 units 1-3 yrs	4 yrs 1 units 4 yr			

1. Construction on Yongbyon 50Mwe and Taechon 200 MWe reactors were frozen in 1994. However, it was estimated that construction could have been completed in two years. So the construction times provided are based on that projection.
2. Average Time Assessment based on only 7 of 22 reactors. Construction times for production reactors not well documented.
3. Time Assessment based on only 2 of 5 reactors production reactors.
4. Includes Chicago piles and X-10 reactor

Table 21 Construction Time for Graphite Moderated Reactors

Stage	Period Primarily Operated in	Description	Construction Time (Years)
Stage 1	1940s	Developmental prototypes in the U.S. and Russia	1-2
Stage 2	1950s	First production reactors in the 1940s, 1950s and early 1960s in Britain, France, Russia, the United States and China (lagged by a few years)	2-6
Stage 3	1960-80s	Civil reactors in Britain and France, as well as Russia and former Republics of the USSR (Ukraine and Lithuania)	5-14
Stage 4	1990s	Production reactors in North Korea	6-9[1]

[1] Construction on Yongbyon 50Mwe and Taechon 200 MWe reactors, frozen in 1994, was never completed. Estimates are that construction could have been completed in two years. Construction times provided are based on that projection.

Table 22 Construction Time Constants for Graphite-Moderated Reactors for Different Regimes

4.1.5 Required Concurrent Technologies

Technologies required to support plutonium production include fuel manufacturing, production of high quality graphite, reactor control technology, heat removal systems, and fuel reprocessing. For GGRs, additional requirements include the capability to use high pressure gas technologies.

4.2 HEAVY WATER MODERATED REACTOR TECHNOLOGY HISTORY

4.2.1 Technology Description

4.2.1.1 Origin

The first heavy water moderated reactor, CP-3, was built in 1944 in the U.S. at Argonne National Laboratory as part of its weapons grade material production program. The second heavy water moderated reactor, ZEEP, was built in 1945 at the Nuclear Research Center at Chalk River in Canada; it was the predecessor to the commercial CANDU (CANadian Deuterium Uranium) design.

The weapon production programs in France, the Soviet Union and the United States constructed heavy-water-cooled, heavy-water-moderated reactors (HWR). These reactors were used to produce tritium as well as plutonium. This same design was also used by Israel and India to provide weapons grade plutonium and possibly tritium.

Canada pioneered the transition from research reactors to commercial power reactors. India also has a large number of heavy-water-moderated reactors. They are also currently operated to produce electricity in Argentina, Japan, Pakistan, Romania, and South Korea.

4.2.1.2 Basic theory

As with graphite, when heavy water is used as a moderator, natural uranium can be used as fuel eliminating the need for enrichment facilities. Because heavy water absorbs relatively less neutrons (also known as having a "high-neutron economy") than graphite, uranium requirements are lower, and plutonium production rates are higher[105] .

Although heavy water reactors (HWRs) can use natural uranium as fuel, they can also use enriched uranium. Their high-neutron economy allows them to use spent fuel from other light water reactors (LWRs) and even other fuels such as thorium.

Several possible HWR design configurations have been investigated:

- Pressure tube heavy water cooled
- Pressure tube light water cooled
- Pressure tube gas cooled
- Pressure tube organic cooled
- Pressure vessel

The most common commercial HWR design (used in reactors of Canadian and Indian design) is the pressure tube heavy-water cooled and moderated design (Figure 8). The fuel is contained in horizontal *pressure tubes* through which heavy water flows at high temperature and pressure. Surrounding the pressure tubes is low-temperature, low-pressure, heavy water contained in a tank called the "calandria." Reactors with this design can be refueled on line while the reactor is at full power, allowing for short fuel residence in the reactor, thus minimizing the buildup of ^{240}Pu.

4.2.1.3 Important components/materials

Heavy water, the key material needed for the operation of heavy-water-moderated reactors, is not easily produced and is contained under international controls. Heavy water moderated reactors have the advantage of easy adaptability to various fuel types (i.e., natural uranium, slightly enriched U^{235}, plutonium fuel, and a thorium/uranium 233 fuel cycle).

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[105] "Heavy Water Reactors: Status and Projected Development", Technical Reports Series Number 407, IAEA, 2002.

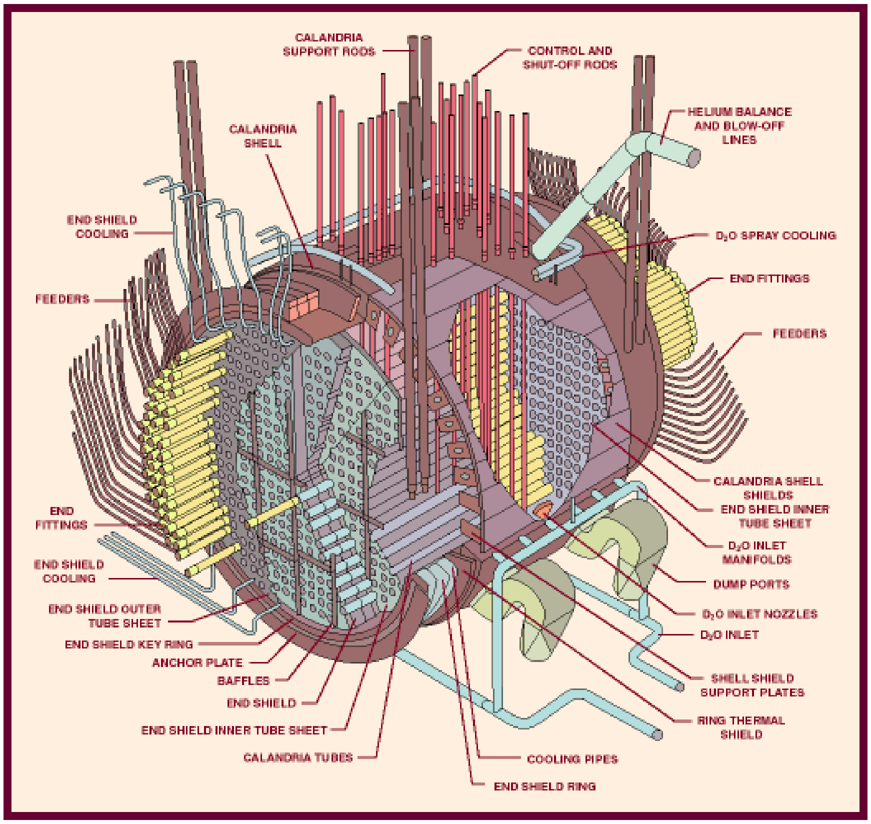

Figure 8 **CANDU Reactor Schematic**

4.2.2 Technology Development

As described in section 4.2.1.2, five distinct types of heavy water reactors have been developed. The most widely used is the CANDU type, with forty built in Argentina, Canada, India, the Republic of Korea, and Pakistan, and two under construction in China.

4.2.3 Countries that have used/attempted to use the technology

Table 23 lists the four counties that have used heavy water moderated reactor technology to produce weapons grade plutonium and tritium. Israel and Pakistan (see section 4.3.3.7) are not included in this list because their reactors were of less than 100 MWth and were originally designated as “research reactors”.

United States	India
Soviet Union	France

Table 23 **Countries with heavy water moderated production reactors**

Several other countries operate commercial heavy-water-moderated reactors for the generation of electrical energy. The following is a discussion of the development of heavy water moderated reactor technology in the five countries that have used them in the production of plutonium.

4.2.3.1 United States

The first heavy-water-moderated reactor, CP-3, was built at and started operation at Argonne in 1944 as part of its weapons grade material production program. It was primarily a research reactor. Another heavy water research reactor, CP-5, started operation at Argonne in 1954 and was primarily used to irradiate samples. The CP-3 was shutdown in 1954 and CP-5 was shutdown in 1979.

Five heavy water moderated production reactors were put into operation between 1953 and 1955 at the Savannah River site in South Carolina. The R Reactor operated at levels up 2260 MWt, the L Reactor up to 2700 MWt, the P Reactor up to 2680 MWt, the K Reactor up to 2710 MWt, and the C Reactor up to 2915 MWt. These reactors produced both plutonium and tritium.

By the end of 1992, all production had ceased at these facilities.

4.2.3.2 France

France produced weapons-grade plutonium in both military and civilian reactors. The twin Celestin heavy-water reactors at Marcoule were initially rated at 190 MWt. The first started in 1967 and the second in 1968. They were originally intended to produce tritium for nuclear weapons but were later operated to produce military plutonium and civil isotopes.

Conversion to plutonium production started in the 1970s; and, by the 1980s, the facilities were producing primarily plutonium although some tritium continued to be produced. Plutonium production was reported to end in 1991, but some tritium is still produced.

4.2.3.3 Soviet Union

Soviet research on heavy-water reactors began in 1947. An experimental reactor was constructed beginning in 1947 and completed in 1949. It was a “pool” type reactor; heavy-water moderated and cooled producing about 2.5 MWth. Two similar reactors with a power level of 10 MWth were constructed in China and Yugoslavia in 1959.

A “commercial” pressurized HWR called OK-180 was constructed beginning in 1949 and was completed and place in operation in 1952. It operated until 1965, producing plutonium, ^{233}U, and tritium. Work on a second reactor, OK-190, began in 1953. It began operation in December 1955 and operated until November 1965. After an extensive renovation, it was renamed OK-190m, restarted in April 1966, and continued operation until 1986.[106]

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[106] The History of the Soviet Atomic Industry, Arkadii Kruglov, Taylor & Francis, 2002.

A replacement reactor for OK-180, Ruslan (1000 MWt), was originally built as a heavy water reactor and was converted to a light water reactor in 1979. A replacement for OK-190m, Ludmila LF-2 (1000 MWt), was put into operation in 1987[105].

4.2.3.4 India

India indigenously designed and built the 100 MWt Dhruva facility based on the CIRUS research reactor design (section 4.3.3.3). Dhruva commenced construction in 1972 with a planned completion date of 1980. The reactor went critical in 1985. Because of severe vibrations in the reactor core, Dhruva was shutdown soon after starting. In late 1986, it began operating at 25 MWt but did not solve its vibration problems until 1988 when it achieved the 100 MWt level[107].

It required 16 years, for a nation that was fairly mature in nuclear technology, to produce an indigenous reactor designed for the purpose of producing weapons material.

India also has twelve operating Pressurized Heavy Water Reactor (PHWR) type power reactors with the capability to make weapons grade plutonium.

4.2.4 Technological Information Acquisition/Development

4.2.4.1 Acquisition approaches

The design of heavy-water-moderated reactors for the purpose of plutonium production was developed indigenously in the U.S., the Soviet Union, and France. Canada pioneered the development of the heavy-water design for commercial use. India developed its own heavy water reactor, Dhruva, in 1985 after importing heavy-water technology from Canada.

4.2.4.2 Time constant for technology development

The following is a determination of time constants for the development and construction of heavy-water-moderated reactors. Construction time differs according to whether it was a simple early generation reactor or a later more evolved one. It also appears to be affected by location (country) and program sponsorship (civil or government). Table 24 provides the type and number of reactors put into operation in each country. Figure 9 shows this graphically by time frame. Table 25 provides summary construction information for known heavy-water-moderated reactors in the world. For each country the average construction time, the number of reactors constructed, and the fastest and slowest construction times are given for each decade from the 1940s to the present. Table 26 lists construction time constants for heavy-water-moderated reactors for different regimes.

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[107] India's Nuclear Bomb: The Impact on Global Proliferation, George Perkovich, University of California Press, 1999

Country	Reactor Type	Number
Argentina	Civil CANDU reactor and PHWR	2
Canada	Civil CANDU reactors	23
China	Civil CANDU reactors	2
France	Production PHWRs	2
Germany	Small gas cooled heavy water moderated reactor	1
India	Civil CANDU reactors and production PHWRs	16
Israel	Production PHWR	1
Japan	Civil light water cooled heavy water moderated reactor	1
Pakistan	Civil PHWR and dual civil/production CANDU reactor	2
Romania	Civil PHWR	1
South Korea	Civil CANDU reactors	3
USSR	Production PHWRs	2
U.S.	Production PHWRs	5

PHWR - Pressurized Heavy Water Reactor

CANDU – Canadian designed heavy-water-moderated and cooled reactors

Table 24 **Countries with Heavy Water Moderated Reactors**

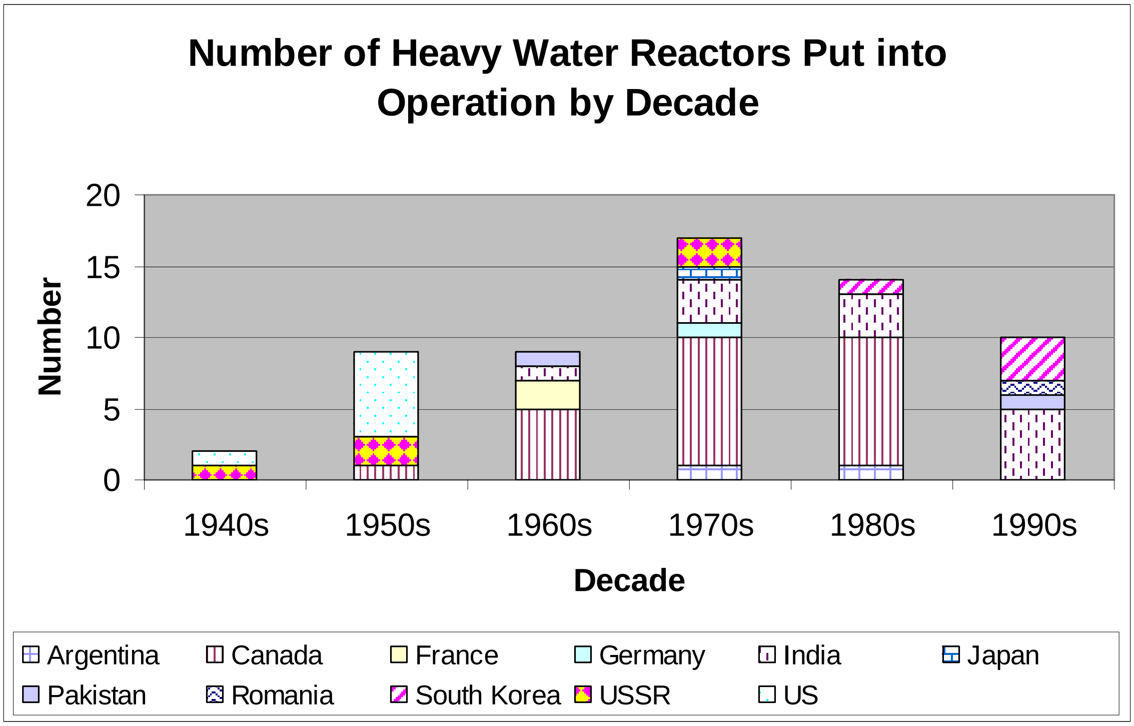

Figure 9 **Heavy water reactors put into operation by decade**

	Decade that construction primarily occurred in						
Country		**1940s**	**1950s**	**1960s**	**1970s**	**1980s**	**1990s**	**2000**
Argentina	Average time Total units Range				6 yrs 1 unit 6 yrs	9 yrs 1 unit 9 yrs		
Canada	Average time Total units Range			5 yrs 5 units 5-7 yrs	6 yrs 7 unit 6-8 yrs	8 yrs 10 units 6-10 yrs	8 yrs 2 units 7-10 yrs	
France	Average time Total units Range			6 yrs 2 units 5-6 yrs				
Germany	Average time Total units Range			7 yrs 1 unit 7 yrs				
India	Average time Total units Range			1 yrs 1 unit 1 yrs	12 yrs 3 units 12-13 yrs	11 yrs 8 units 8-15 yrs	11 yrs 2 units 11 yrs	
Japan	Average time Total units Range				8 yrs 1 unit 8 yrs			
Pakistan	Average time Total units Range			5 yrs 1 unit 5 yrs			6 yrs[1] 1 unit 6 yrs	
Romania	Average time Total units Range					16 yrs 1 unit 16 yrs		
South Korea	Average time Total units Range				5 yrs 1 unit 5 yrs		6 yrs 3 units 6 yrs	
Soviet Union	Average time Total units Range	1 yr 1 unit 1 yr	3 yr 2 units 2-3 yr		? yr[2] 2 units			
U.S.	Average time Total units Range	1 yr 1 unit 1 yr	3 yr 5 units 2-4 yr					

1. Reactor discovered by Russian intelligence 5 years before it began operation. Six year construction is best estimate.
2. Construction time for Russian (Soviet Union) production reactors is not well documented.

Table 25 <u>Construction Times for Heavy Water Reactors</u>

Stage	Period Primarily Operated in	Description	Construction Time (Years)
Stage 1	1940s	The first developmental prototypes in the U.S. and Russia	1
Stage 2	1950s	Early production reactors in Russia and the United States.	2-4
Stage 3	1960s	Production reactors in France, India, Israel, and Pakistan, as well as civil reactors in Canada and Germany	5-6[1]
Stage 4	1970-2000	A group of civil reactors, many of the CANDU design	6-11

1. With the exception of Cernavoda 1 in Romania that took 16 years because of political work stoppage

Table 26 **Construction Time Constants for Heavy Water Moderated Reactors for Different Regimes**

4.2.5 Required Concurrent Technologies

Technologies required to support plutonium production include fuel manufacturing, heavy water production, reactor control technology, heat removal systems, and fuel reprocessing.

4.3 RESEARCH REACTOR TECHNOLOGY HISTORY

4.3.1 Technology Description

4.3.1.1 Origin

The world's first research reactor was the Chicago pile (CP-1). In this reactor, the first self-sustaining nuclear chain reaction was produced. This reactor, as well as subsequent early plutonium production reactors, had a very simple design.

4.3.1.2 Basic theory

Basic "production" reactor theory is discussed in sections 4.1.1.2 and 4.2.1.2. The two most productive reactor designs for the production of weapons material are heavy-water-moderated and graphite-moderated, gas (or light-water) cooled, research reactors. These reactors can operate on natural uranium, negating the need for enrichment technology. From these reactors, weapons material is accumulated by separating the plutonium from the uranium and the various fission products.

Plutonium can be produced either from reprocessing the uranium reactor fuel or by the irradiation of uranium targets, as the Iraqis may have planned at the Osiraq reactor (Section 4.3.3.5). Producing weapons material by irradiating "targets" is technically possible, but the production rate will be considerably lower than if the fuel itself is the target material.

4.3.1.3 Important components/materials

The most important materials for reactors intended to produce weapons grade material are the fuel and the moderator of choice. If either heavy water or graphite is selected as the moderator then natural uranium can be used as the fuel; and, there is no need for enrichment facilities, only for fuel production capability. Accordingly, the acquisition of the moderator is the critical action; acquisition of natural uranium and the ability to manufacture suitable fuel are the next critical actions.

4.3.2 Technology Development

4.3.2.1 Countries that have used/attempted to use technology

Several hundred research and test reactors have been constructed and operated throughout the world for a broad spectrum of purposes. Only a few have been used for producing weapons material. The countries that are believed to have used research reactors in some way for the production of weapons material are listed in table 27. Norway, Canada, and Switzerland have built and operated heavy-water-moderated, natural-uranium-fueled research reactors. It is doubtful that they were ever planned for the purpose of weapons production.

Algeria	Pakistan
Argentina	Romania
India	South Africa
Iran	Sweden
Iraq	Taiwan
Israel	Yugoslavia

Table 27 **Countries believed to have used or planned to use research reactors to produce plutonium**

4.3.2.2 Technological difficulties in use

There are few *technical* difficulties to overcome in acquiring the reactors. The principle difficulties are political. The theory of reactor design and control is well known. Obtaining heavy water or highly purified graphite and the manufacturing components, fuel, and parts are the challenges.

4.3.2.3 Changes/improvements in technology

For reactors used to produce weapons material, few technological advances have been made, needed, or desired. A low-tech approach works adequately and requires less time to design, construct, and assemble. Basically, a large CP-1 or X-10 (see Figure 6) style pile that can be cooled and refueled could produce weapons grade Pu.

4.3.3 Countries that have use/attempted to use research reactors

Following are highlights of activities of those countries that are believed to have successfully used or attempted to use research reactors for weapons material production:

4.3.3.1 Algeria

In 1985, Algeria purchased a 0.5 MWth research reactor from Argentina[108]. It became operable in 1992[109]. In 1991, China provided a 15 MWth heavy water research reactor at Es Salam (Ain Oussera site). The reactor commenced operation in December 1993[110]. Construction took seven years.

4.3.3.2 Argentina

Argentina was perhaps the most successful country to develop an indigenous reactor technology based on a United States design. In 1958, the United States provided detailed plans for the RA-1 reactor at Constituyentes and aided the Argentines in its construction. By 1967, the Argentines had built three additional research reactors based on the U.S. design.[111]

4.3.3.3 India

The Cirus research reactor (a 40MWth heavy water facility) was purchased from Canada in 1955 and commenced operation[112] in 1960. It was removed from service in 1997. This reactor was a known weapons material producer.

4.3.3.4 Iran

In 1990, China agreed to build a 27 MWth heavy water moderated plutonium production reactor in Isfahan in Iran. U.S. satellite photographs taken in September 1991 documented major construction work at the site.[113,114] However, as of 2003 it appears to have not been completed.

4.3.3.5 Iraq

In 1976, a 40 MWth material test reactor was purchased from France to be built at the Iraqi nuclear research site Tuwaitah. The reactor was light water cooled and moderated and used aluminum clad HEU fuel. It was based on the French “Osiris” reactor, a tank type reactor with twenty one positions for experiments to be placed for irradiation[115]. The reactor was named “Osiraq" by the French, a combination of Osiris and Iraq. There are indications that the Iraqi’s planned to irradiate natural uranium “targets” in the

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[108] Nucleonics Week, October 1986

[109] Nuclear News, April, 1992

[110] “Plutonium and Highly enriched Uranium” Albright et al, 1996

[111] http://www.ceip.org/programs/npp/nppargn.htm

[112] “India's Nuclear Bomb, the Impact on Global Proliferation,” George Perkovich, 1999, page 27

[113] The New Republic, April 24, 1995

[114] Washington Times, May 8, 1995

[115] Directory of Nuclear Reactors, Vol. VIII Research, Test and Experimental Reactors, International Atomic Energy Agency, Vienna, 1970

experimental locations to make plutonium. The reactor was destroyed by Israel in 1981 before it could be fueled.

4.3.3.6 Israel

In 1957, the Dimona reactor project began with assistance from France.[116] The French provided a heavy water moderated 24MWth reactor (based on the French E-3 research reactor), but the facilities were upgraded to 70 MWth before project completion in 1963. The French also built a Pu separation facility for Israel at the Dimona site. Around 1970, Dimona was reportedly operating at well over 100 MWth, for the sole purpose of Pu production. In 1980, it is reported that tritium was produced at Dimona (from lithium targets).

4.3.3.7 Pakistan

Pakistan built a small, heavy-water research reactor, KANUPP, which began operation in 1971. This reactor did not appear to be used for weapons production, but probably provided technology and training. In 1993, it was reported that a heavy-water-moderated reactor of about 40 MWth was being built at Khushab, most likely for the purpose of Pu production. Ostensibly, it was of indigenous design and probably based on the KANUPP design. The facility is not subject to the IAEA non-proliferation agreement. The facility was completed in 1996, but couldn't begin operation at that time due to a lack of heavy water. Pakistan received significant help from China in design and construction of this facility[117].

In 1998, Khushab commenced operation, after producing heavy water using equipment from another source, probably China. It is estimated that the power of the reactor is between 50 and 70 MWth.

4.3.3.8 Romania

During the early 1980's[118], Romania used a 14 MWth TRIGA reactor to produce a small amount of weapons grade plutonium. The weapons program ended in 1989 after the overthrow of the government and the reactor was shutdown. It was restarted in 1991 with LEU fuel.

4.3.3.9 South Africa

South Africa had two research reactors. The first, SAFARI-I, a 20 MWth research reactor, was constructed with the help of the United States. It uses HEU, and is under international controls. It was commissioned in 1965 and is still operating. In 1967, an indigenously designed facility called SAFARI- II or SAFARI-Zero went into operation.

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[116] "Israel and the Bomb" Avner Cohen, Columbia University Press, Oct, 2002.

[117] Pakistan's Chashma Nuclear Power Plant, Zia Mian, A.H. Nayyar, Princeton University Center for Energy and Environmental Studies Report No. 321, December 1999

[118] *Romania Separated Tiny Amount Of Plutonium In Secret In 1985*, Ann Maclachlan, Nucleonics Week, Vol. 33, No. 26; Pg. 16, June 25, 1992

It was heavy-water moderated critical facility and used 2% enriched fuel supplied by the U.S. In 1969, the facility was shutdown and abandoned as it was drawing too many resources from the uranium enrichment program[119].

4.3.3.10 Sweden

Sweden had a clandestine, military sponsored plutonium production research program that began after World War II and continued until 1972. Sweden built its first experimental reactor, named R1, in an underground facility located in Stockholm. R1 was loaded with three tons of metallic uranium from France and five tons of heavy water purchased from Norway. It was operated as a zero power reactor. The Agesta power reactor (also known as R3), which produced energy (65 MWth) for civilian consumption and nuclear material for military research purposes, was intended to be the first source of plutonium for weapons research. It was a heavy water reactor, also built underground. Construction began in 1954, and was completed in 1958. The reactor operated for sixteen years. It produced plutonium and generated district heat as a by-product, which provided a public rationale for the plant. The Swedish plutonium production program was halted in 1972, and the R3 was taken out of service in 1974[120].

4.3.3.11 Taiwan

In 1969, Canada supplied a 40 MWth Natural U fuel, heavy water moderated reactor to Taiwan, called the "TRR" (Taiwan Research Reactor). The design was identical to India's CIRUS reactor. The facility achieved criticality in 1973, but was shutdown in 1988 under intense U.S. political pressure[121]. In 1991, the fuel (with exception of 118 spent rods) was shipped to the U.S. for reprocessing. Taiwan planned to convert the reactor to light-water moderated LEU fuel for beam port experiments and isotope production.[122]

4.3.3.12 Yugoslavia

In 1956, construction started on the "RA" test reactor facility, which was completed and achieved criticality in 1959. The reactor, of Soviet design, was heavy-water moderated and cooled, presumably fueled with natural U[123]. It is known that the facility was modified to use HEU fuel in 1976. The reactor was permanently shutdown in 1984.

4.3.4 Technological Information Acquisition/Development

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[119] The Nuclear Axis, Zdenek Cervenka and Barbara Rogers, NYT Books, 1978

[120] Atomic Bombast: Nuclear Weapon Decision-Making in Sweden1946-72, Paul M. Cole, The Washington Quarterly, ATOMS FOR PEACE AND WAR; Vol. 20, No. 2; Pg. 233, 1997 Spring ,The Center for Strategic and International Studies and the Massachusetts Institute of Technology

[121] U.S. Pleased Taiwan Is Shutting Down Reactor Producing 'Good Quality Pu', Michael Knapik, Donald Shapiro, Gamini Seneviratne, Nuclear Fuel, Vol. 13, No. 7; Pg. 12, April 4, 1988

[122] Construction Of Research Reactor To Be Suspended For Three Years, Maubo Chang, August 21, 2001, Central News Agency – Taiwan,

[123] Directory of Nuclear Reactors, Vol. VIII Research, Test and Experimental Reactors, International Atomic Energy Agency, Vienna, 1970

4.3.4.1 Acquisition Approaches

The transfer of technology has been the primary way for third-world nations to acquire research reactors and Pu separation technology. Early on, many Western nations provided nuclear technology for peaceful purposes. The U.S. provided technology to Argentina, Canada provided the technology to India, and France transferred the technology to Israel and Iraq. Until very recently, China was willing to help any nation friendly to them to develop nuclear technology.

Nations lacking a strong industrial base will probably not be capable of using reactors to produce weapons material unless they receive significant help. Pakistan attempted to obtain heavy water for at least eight years before they were successful in obtaining enough to start the KANUPP facility.[124] Most recently, both Algeria and Pakistan had significant help from China in developing their reactors and reprocessing facilities. Iran had significant help from China but has so far failed in its attempt to produce weapons material from a reactor.[125]

It should be noted that South Africa, Taiwan, and Sweden could all have been successful in producing significant quantities of weapons grade plutonium had they chosen to continue down the reactor path.

4.3.4.2 Time constant for technology development

Third world nations that have been successful in producing weapons material from research reactors have generally had significant help. A turnkey operation in the 1960s and 1970s could break ground and complete a research reactor capable of producing weapons material in typically 4 or 5 years.

Indigenous designs have been strongly predicated on turnkey designs supplied to countries earlier in their nuclear infancy. Frequently, technical problems arise tending to cause delays in achieving success. Typically, more than a decade would be required for a country to indigenously clone a foreign design. The technology is simple and the underlying theory is also simple. Obtaining all the materials is the challenge. The technology and materials could, at one time, be purchased and installed in typically a three to four year period. Canada and France, the two principal producers of the desirable type of research reactors (Heavy Water Moderated), no longer export the technology.

Yongbyon in North Korea is an exception. This design appears to be an original indigenous design but modeled after the British gas-cooled, graphite-moderated design (Calder Hall). The small 20-30 MWt reactor was constructed in about 7 or 8 years. This graphite-moderated, natural-uranium-fueled and light water-cooled design is simple. It is believed that weapons grade Pu was separated and processed from the reactor fuel about two years after initial startup. It is unclear if North Korea had help.

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[124] "Nuclear History In India, Pakistan," New York Times, May 28, 1998

[125] The New Republic, April 24, 1995

A simple graphite-moderated, light-water-cooled reactor could be constructed solely for the purpose of the production of weapons material in a short period of time, if proper materials could be obtained.

4.3.5 Required Concurrent Technologies

In parallel to the development of a weapons program is the construction of substantial hot cells in which to perform the chemical separation (see section 4.4 Reprocessing Technology) of the plutonium from the uranium fuel rods (or U-233 from thorium target rods). Hot cells must be robust, typically requiring greater than 40 inches of concrete for adequate worker shielding. All non-automated functions must be performed using remote manipulators. Indigenous construction of remote manipulators is a challenging task.

4.4 REPROCESSING TECHNOLOGY HISTORY

4.4.1 Technology Description

4.4.1.1 Origin

Nuclear fuel reprocessing technology began to be developed in the United States after plutonium was first created in 1942. While the basic chemistry of uranium was understood, the complications caused by the radioactive byproducts of the nuclear reactions producing plutonium were not. In addition, it was expected that the concentration of plutonium produced would be on the order of a few hundred grams of plutonium per ton of irradiated reactor fuel. Consequently, the recovery process had to be very efficient to separate and recover the small quantity of plutonium from the uranium and radioactive fission products in the fuel.

A number of approaches to extract plutonium and uranium from spent reactor fuel have been explored over the years. The approaches to reprocessing can be generally divided into two categories; those based on aqueous chemistry and those based on non-aqueous materials such as molten salts and molten metals[126]. For a variety of reasons discussed below, aqueous-based technologies have been the most widely used to date.

4.4.1.1.1 Aqueous Repossessing Technology Origins

The first large-scale reprocessing technology, called the “Bismuth Phosphate” process, was developed beginning in 1942, demonstrated at a pilot plant in Oak Ridge, and put into operation at Hanford in 1944. Although it was very effective at recovering plutonium, it had several drawbacks. It recovered only the plutonium, leaving the valuable remaining uranium in the process waste. It was an inefficient “batch” process, requiring large quantities of process chemicals and generating large volumes of waste.

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[126] Nuclear Wastes: Technologies for Separation and Transmutation, National Academy of Sciences, 1996

Despite its processing inefficiencies, it was able to extract 95% of the plutonium in the fuel. The Bismuth Phosphate process operated at Hanford until 1956.

Because of these drawbacks, efforts continued to develop improved reprocessing technology. Solvent-based processes were explored, as these were already in use for recovery of uranium during the fuel manufacturing process. A process called Redox (for *oxidation-reduction* separations chemistry) was developed by 1948 at Argonne National Laboratory, tested in a pilot plant from 1948 to 1949 at Oak Ridge, and put into service in a full scale facility at Hanford in 1951.

The primary Redox process chemicals were *hexone* (methyl isobutyl ketone) and aluminum nitrate. The process was continuous rather than batch and was capable of efficiently extracting both plutonium and uranium, leaving behind the fission products and other wastes. The disadvantages of the Redox process were the volatility and flammability of the hexone solvent, and the large volume of non-volatile reagents added to the radioactive wastes. The Redox facility at Hanford operated until 1966.

During this period other processes were developed. The Soviet Union developed a batch process using acetates and nitric acid that was used from 1948 to 1976. Two other early processes, *Trigly* and *Butex*, were developed by the British and Canadians at the Chalk River Laboratory in Canada. The Trigly process was similar to Redox, but used triglycol dichloride, aluminum nitrate, and nitric acid. Hexone was used in a final purification step. It was a batch rather than a continuous process.

The Redox and Trigly processes added large quantities of nitrate salts to the waste streams, increasing the overall waste volume by a factor of ten. The Butex process avoided this by using dibutyl carbitol and nitric acid, and then evaporating and reusing the nitric acid. The Butex process was used at the Windscale facility in the United Kingdom until the 1970s.

The most widely-used reprocessing technology in the world is the *Purex* process. It was developed at the Knolls Atomic Power Laboratory in 1950, demonstrated in a pilot plant at Oak Ridge in 1952, and put into initial production at the Savannah River Site in 1954. The Purex process replaced the Redox process at Hanford in 1956 and has since been used in countries such as the Soviet Union, India, Germany, France, and Japan[127]

The Purex process uses a mixture of tributyl phosphate (TBP) and a hydrocarbon diluent to extract uranium and plutonium from an aqueous solution containing nitric acid. The Purex process has four advantages over the Redox process it replaced: 1) Waste volumes can be greatly reduced as the nitric acid is removed via evaporation, 2) The solvent, TBP, is less volatile and less flammable than hexone, 3) TBP is more stable against attack by nitric acid, and 4) operating costs are lower[128]

4.4.1.1.2 Nonaqueous Repossessing Technology Origins

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[127] Nuclear Chemical Engineering; Benedict, Pigford, Levi, 1981.

[128] Nuclear Chemical Engineering; Benedict, Pigford, Levi, 1981).

Three types of nonaqueous reprocessing have received most of the development attention: 1) pyrometallurgical processes, involving high-temperature processing of metallic fuels, 2) pyrochemical processes, involving high-temperature processing of oxide or carbide fuel, and 3) fluoride volatility processes, in which elements in the fuel are converted to fluorides and then separated by fractional distillation.

Possible advantages over aqueous processing include: 1) no radiation damage to chemicals involved in the process, so fuel could be processed quickly after removal from the reactor, 2) fewer chemical steps involved, 3) more compact processing equipment and smaller shielded volumes, and 4) larger batches possible, as critical masses are larger without moderators.

As work on the nonaqueous process has continued, the following disadvantages have been found that discourage widespread use: 1) separations are not complete in nonaqueous processes, excluding the fluoride volatility process; that is, the product is not completely decontaminated and refabrication must be done remotely, and 2) operation at high temperatures with corrosive or reactive agents requires special, costly construction materials and makes maintenance difficult.[129]

Development of nonaqueous processes has been generally limited to engineering studies and some pilot plants. Pyrometallurgical processes were studied at Brookhaven, Argonne, and Oakridge; and, a facility was built at EBR-II to test one such process, electrochemical separations (as part of the Integral Fast Reactor demonstration). It is based on the selective *electrorefining* of uranium, plutonium, and heavier actinides from a molten solvent. The basis for this process is that each metallic chemical element has a unique electrical potential where it can be dissolved in some medium, such as molten chloride salt, and a unique potential where it can be plated out. Thus, this process is driven electrically rather than chemically. From a weapons production aspect, one problem with this approach is that the produced plutonium contains rare earth contaminants.[130]

Engineering studies of the pyrochemical process have been performed at both Argonne National Laboratory and Pacific Northwest National Laboratory, and engineering studies were performed at EBR-II and Hanford in the 1960s. The fluoride volatility process was studied at Brookhaven, Argonne, Oak Ridge, and laboratories in France. Engineering studies were performed on a variety of fuel types, including fuel from the molten-salt reactor experiment at Oak Ridge. It was found that the fluoride volatility process worked best on fuel that contained little plutonium, i.e. fuel that already had the plutonium removed[131].

Although ongoing work continues in development of nonaqueous processes, no evidence exists that nonaqueous reprocessing techniques have been used to date to extract

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[129] Nuclear Chemical Engineering; Benedict, Pigford, Levi, 1981.

[130] Nuclear Wastes: Technology for Separations and Transmutations, National Academy Press, Washington D.C., 1996.

[131] Nuclear Chemical Engineering; Benedict, Pigford, Levi, 1981.

significant quantities of weapons grade plutonium. However, pyrochemical processes are extensively used in the production of UF_6.

4.4.1.2 Basic theory

The extraction of plutonium and uranium from spent fuel using aqueous chemical technology is based on the chemical differences between the multiple oxidation states of plutonium and uranium. After the fuel cladding has been removed, by mechanical means (e.g., cutting the cladding off) or chemical means (e.g., dissolving the cladding in a nitric acid solution), the fuel is subjected to a series of chemical steps designed to extract the plutonium.

It is important to note there are significant differences in the radioisotopic content of fuel used in power reactors (and consequently residing in the reactor for extended periods of time) and fuel used in production reactors. The dominate radioisotopes in production reactor fuel are cesium (^{137}Cs), strontium (^{90}Sr), technetium (^{99}Tc), and plutonium (^{239}Pu). Additional radioisotopes present in power reactor fuel include americium, curium, neptunium, and other radioisotopes in lesser quantities. If an attempt is made to use power reactor fuel to produce weapons grade material, additional processing steps and shielding must be included to remove these radioisotopes.

Two representative aqueous reprocessing approaches will be described in some detail to provide a basic understanding of the chemistry involved. The two are Bismuth Phosphate process, the first developed, and the Purex process, the most widely used.

4.4.1.2.1 Bismuth Phosphate Process

The keys to this process were the quantitative, selective co-precipitation of Pu^{4+} from an acid solution by a bismuth phosphate carrier and the ability of Pu^{6+} to remain in solution in the presence of the bismuth phosphate carrier.

The process is as follows. The irradiated, clad (aluminum is a typical cladding material used for plutonium production) fuel rods are removed from the reactor to a pool of water where they must remain for a number of days to allow the short-lived, high-activity fission products to decay. The (aluminum) cladding is removed from the rods with a concentrated sodium hydroxide solution. The contents of the fuel rods are dissolved in nitric acid. After the addition of sulfuric acid to keep the uranium and other fission products in solution, Pu^{4+} is co-precipitated with bismuth phosphate. The precipitate containing the plutonium is then dissolved in nitric acid and the Pu^{4+} is oxidized to Pu^{6+} with sodium dichromate. This time the plutonium remains in solution as Pu^{6+} while any remaining uranium and fission products are precipitated with additional bismuth phosphate. The Pu^{6+} is then reduced to Pu^{4+} and the cycle repeated.

At this point, the carrier is changed to lanthanum fluoride, $LaF_{2,}$ and a similar oxidation-reduction cycle is performed to achieve further purification and concentration of the plutonium. The concentration of plutonium is high enough so that no carrier is required for the final purification step, the precipitation of plutonium peroxide from a basic solution. The peroxide is typically converted to a plutonium-nitrate paste that is used to produce Pu metal for weapons purposes.

The overall recovery of plutonium by this process is typically greater than 95% and the plutonium will contain less than 0.1 part per million (PPM) impurities. The process generates large amounts of chemical and radioactive wastes, approximately 10,000 gallons per metric ton of uranium processed.

4.4.1.2.2 Purex Process

This method relies on solvent extraction to perform the separations. It is capable of recovering plutonium in a continuous extraction using organic solvents. This process also generates less waste that the bismuth-phosphate process.

After dissolution in nitric acid, the plutonium and uranium nitrates transfer into the organic phase while the fission products are removed in the aqueous phase. The plutonium is then separated from the uranium in a second solvent extraction process. The detailed process follows.

The fuel cladding is typically removed through a combination of mechanical and chemical processes. After the fuel is dissolved in nitric acid, the resulting solution is combined with a solution of tri-butyl phosphate (TBP) in refined kerosene. The TBP solution preferentially extracts uranium and plutonium nitrates, leaving fission products and other nitrates in the aqueous phase. Then, chemical conditions are adjusted so that the plutonium and uranium are re-extracted into a fresh aqueous phase.

Normally, two solvent extraction cycles are used for the separation; the first removes the fission products from the uranium and plutonium, while the second provides further decontamination. Uranium and plutonium are separated from one another in a similar second extraction operation.

Solvent extraction usually takes place in a pulse column, a several-inch diameter metal tube resistant to nitric acid and used to mix together the two immiscible phases (organic phase containing TBP and an aqueous phase containing U, Pu, and the fission products). The mixing is accomplished by forcing one of the phases through the other via a series of pulses with a repetition rate of 30 to 120 cycles/minute and amplitudes of 0.5 to 2.0 inches. The metal tube contains a series of perforated plates that disperses the two immiscible liquids.

While the plutonium and uranium from the solvent extraction are almost chemically pure, additional decontamination or separation from one another, from fission products, and from other impurities may be required. Large plants use additional solvent extraction cycles to provide this service. Small plants may use an ion exchange process for the final purification step known as “polishing”.

To prepare the plutonium in the metal state, plutonium may be precipitated as PuF_3 from aqueous nitrate solution by reducing its charge from +4 to +3 with ascorbic acid and adding hydrofluoric acid (HF). The resulting solid is separated by filtration and dried. Reprocessed uranium is converted to the oxide and then stored or to the hexafluoride and then re-enriched. Plutonium (and uranium) metal may be produced by the reaction of an active metal (calcium or magnesium) with a fluoride salt at elevated temperature in a sealed metal vessel (called a "bomb"). The metal product is freed from the slag, washed in concentrated HNO_3 to remove residue, washed with water, dried, and then re-melted in a high temperature arc furnace.

4.4.1.3 Important components/materials

Generally, chemicals and industrial equipment used in chemical processing facilities and radiation shielding are needed for separation. Robust hot cells and radiation protection technology is crucial. Table 28 lists important technologies and related critical materials and components.

A typical process layout for all separation chemistry performed on radioactive materials is to have the product moving in one direction down a chemical process line, with the waste and residue moving in the opposite direction. This approach serves two purposes. First, it guards against the product becoming contaminated with waste or residue, as the waste from the process diminishes as the product progresses. Second, it allows for reduced requirements for hot cell robustness, as the product progresses down the chemical process line. With hot cell separation, the first steps require robust hot cells, while the last few steps can be performed in shielded gloveboxes.

In the past, most countries with weapons programs performed reprocessing in large-scale operations, as their intentions were to develop many weapons. In the future, it is expected that smaller countries may take a scaled-down approach to Pu separation, especially if surreptitious production is the objective.

In this scenario, a hot cell approach could be used. The hot cells would need to be quite robust, typically requiring greater than 40 inches of concrete for adequate worker shielding. All non-automated functions would be performed using remote manipulators. Indigenously constructing remote manipulators is a challenging task.

It should be noted that an entire Purex-separation process line was constructed at Hanford in a single hot cell (approximately 10' by 12') as a proof-of-principle prototype for larger facilities.

4.4.2 Technology Development

4.4.2.1 Technological difficulties in use

Chemical separation processes are mature. The chemistry involved is straight-forward and well understood. Other than the typical chemical processing hazards (chemical explosion, fire, etc.), the major issue that must be dealt with is radiation protection and accidental criticality as the plutonium is concentrated.

Note, however, India (section 4.4.3.4) did have significant problems in making the Tarapur reprocessing facility run successfully. From 1979 through 1990, the facility underwent many modifications before achieving successful operation.[132]

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[132] India’s Nuclear Bomb, George Perkovich, University of California Press, 1999

Technology	Sufficient Technology Level	Critical Materials	Unique Test, Production And Inspection Equipment	Unique Software And Parameters
Heavy industrial construction	Ability to fabricate a facility, which will protect workers and the environment from radioactivity and hazardous materials (note: some countries may have different criteria that the United States in this regard).	High-density concrete	Radiation monitoring (applies to all processes) Fuel storage pool Cranes, Hot cells, Remote manipulators, High-density radiation shielding windows, Radiation-hardened TV cameras Air filtration, Evaporators	Shielding software Criticality software Radiation generation/ Depletion software
Fuel storage and movement	Sufficient storage pool capacity and depth. Ability to move radioactive material.	None identified	Remotely operated cranes Specially designed shipping casks Criticality Control	None identified
Fuel disassembly (breaching)	Capability to separate cladding from fissile material mechanically or chemically.	None identified	Cut-off wheel Shear dissolver (for Al cladding) Laser	None identified
Fuel dissolution	Ability to handle highly corrosive liquids containing radioactivity. Adequate knowledge of uranium, plutonium, and fission product chemistry.	Nitric acid (HNO_3) Hydrogen fluoride (HF) HNO_3 resistant tanks of a specific configuration to prevent a nuclear excursion	Analytical chemistry facility for fission products, U and Pu	None identified
Fissile element separation (solvent extraction)	Familiar with liquid-liquid extraction systems. Understand distribution of uranium, plutonium, and fission products between two immiscible liquids.	None identified	Mixer/settlers Pulse columns Centrifugal contractors	Distribution coefficients for many elements. Aqueous solubility for many substances.
U and Pu product purification	Cognizant of liquid-liquid extraction systems Familiar with ion exchange resin systems	Tri-butyl phosphate (TBP) Refined kerosene Ion exchange resins	Mixer/settlers Pulse columns Centrifugal contractors Chemical holding or storage vessels	Distribution coefficients for many elements Aqueous solubility for many substances
Metal preparation (Pu exclusively)	Ability to handle plutonium in glove boxes	HF Reducing agents (high-purity Ca or Mg) CaF_2 or MgF_2 (used as liner for reduction bomb) Iodine (serves as catalyst in reduction)	Drying Furnace; Fluoride resistant Furnace capable of reaching 600°F Sealed reaction tube Temperature control/measurement High temperature furnace (arc)	None identified
Waste treatment/recycle	Ability to recycle valuable components (TBP, HNO_3) Ability to process streams containing high levels of radioactivity and hazardous materials	Resistant to HNO_3 (stainless steel, titanium alloys)	Chemical storage tanks	None identified

Table 28 <u>Technologies used in reprocessing facilities</u>

Based on India's experience, it is apparent that nations undertaking indigenous development of reprocessing technology may face significant challenges. Early successes by Pakistan, Israel, and North Korea were accompanied by significant help from nations that had already developed the technology. Table 29 outlines technologies and technical issues involved in reprocessing.

Technology	Technical Issues	Alternative Technologies
Heavy industrial construction	Ability to construct a thick walled, relatively sealed structure with adequate shielding.	May not be needed if nation unconcerned about its workers or the environment and reprocessing is to be a short-term endeavor.
Fuel storage and movement	Adequate depth of storage pool to shield spent fuel. Sufficient storage capacity for fuel. Cranes of sufficient capacity to handle shipping casks.	Use reactor storage pool if close proximity to reprocessing facility. Possible storage (dry) in specially designed casks.
Fuel disassembly (breaching)	Capability to remove as much, extraneous material from fuel element as possible. Knowledgeable in the construction and use of one of the breaching tools.	None identified
Fuel dissolution	Ability to prevent a nuclear excursion	Several non-aqueous processes have been developed but most are complicated (pyrometallurgical, pyrochemical, and fluoride volatility)
Fissile element separation (solvent extraction)	Ability to prevent a nuclear excursion. Aqueous solution from separation process contains extremely hazardous radioactive materials.	Use one of the non-aqueous processes. Replace solvent extraction with ion exchange process.
U and Pu product purification	Ability to obtain a pure product. Availability of ion exchange resins and sufficient knowledge of their use.	Use one of the precipitation processes (peroxide, oxalate)
Metal preparation (Pu exclusively)	Capability to handle molten Pu metal.	Electrolytic process (requires molten salts – 1,300°F). Reduction of other halides
Waste treatment/recycle	High level radioactive waste must be handled with extreme care.	Discharge all aqueous waste solutions to the environment. Minimal recycling (expensive but may be used for limited production).

Table 29 Technical issues with reprocessing operations

4.4.2.2 Changes/improvements in technology

Aqueous reprocessing is a mature technology, with Purex being the dominant process. The majority of current work on technology development is in non-aqueous reprocessing technologies.

A potential advance in reprocessing technology involves a combination of a molten salt reactor (MSR) or a slurry reactor and an on-line reprocessing system. This process, being developed in the Czech Republic, uses molten fluorides for the MSR reprocessing. Currently, it is at a laboratory scale only. An MSR can be very effective in producing medical radioisotopes. It is doubtful that this initiative in the Czech Republic is for the purpose of weapons material production. However, the technology could be used very effectively for that purpose, as a very high quality Pu with a very low Pu^{240} content could be extracted from this type of system.

4.4.3 Countries that have used/attempted to use the technology

Nations that have reprocessed or attempted to construct facilities for reprocessing nuclear fuel are listed in Table 30.

Algeria*	Israel*
Argentina*	Italy*
Belgium*	Japan****
Brazil*	North Korea*
China**	Pakistan*
Czech Republic*	Soviet Union***
France***,****	Taiwan*
Germany*	United States***
India**	United Kingdom***,****
Iraq*	Yugoslavia*

*Small facilities, i.e. hot cells, pilot plants, technology demonstration facility
**Medium scale reprocessing facilities.
***Large-scale plants for the separation of weapons grade Pu
****Plants for reprocessing commercial reactor fuel

Table 30 **Countries with reprocessing facilities**

4.4.3.1 Algeria

In 1992, construction began on a hot-cell facility provided by China near the Es Salem reactor at Ain Oussera. If a simplified Purex process were to be installed and used in the facility, it was thought that it could separate weapons-grade plutonium. By 1997, Algeria had promised the IAEA that it would operate the hot-cell facility under safeguards, would allow IAEA environmental sampling, and would not separate plutonium from spent fuel. Of additional interest is a larger facility nearby. Algeria has not declared it as a nuclear facility, but some Western officials believe it may be intended as a large-scale reprocessing facility.[133]

4.4.3.2 Argentina

In 1968, a laboratory-scale reprocessing facility was constructed to extract Pu from irradiated fuel. The facility was closed in 1973, however, after reportedly extracting less than 1 kg of Pu.[134] In 1978, the United States cut off supplies of nuclear material and technology to Argentina for failure to accept IAEA safeguards on all its nuclear facilities. Construction on another separations facility began in 1978 with work stopped in 1990 due to political pressure and economic constraints. In 2000, Argentina was selected to supply material to Australia's replacement research reactor and to reprocess the fuel from

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[133] Tracking Nuclear Proliferation 1998, Rodney W. Jones, Mark G. McDonough, et. al., Carnegie Endowment for International Peace, 1998.

[134] Argentina on Threshold of Nuclear Reprocessing**;** Argentina Plans Own Reprocessing**,** Milton R. Benjamin, The Washington Post, October 16, 1978

this reactor. In spite of this, Argentina has yet to successfully reprocess any significant quantity of fuel.

4.4.3.3 Belgium

A demonstration Purex reprocessing facility was built at Mol, Belgium by Eurochemic, a joint venture of member countries of the Organization for Economic Cooperation and Development (OECD) Nuclear Energy Agency (NEA). The facility was capable of reprocessing approximately 0.35 MT/day. Construction on the facility began in 1960, and was complete by 1966. It operated from 1966 to 1974. From 1966 to 1975, it reprocessed approximately 180 tons of natural and low-enriched uranium fuels and 30 tons of high-enriched uranium fuels.[135,136]

At the Mol facility, Eurochemic developed advanced chemical process technologies for use in plutonium separation. In particular, the consortium developed a capability to remove cladding from irradiated-uranium fuel and bind the medium-level liquid reprocessing wastes (MLW) to bitumen (a substance similar to asphalt). It was shown that this process was particularly advantageous for reprocessing magnesium alloy fuel (such as used in British and French GGRs), as it generated on average about half of the waste volume produced by chemical decladding of zircaloy-clad fuels. As this work was not classified, information about it was publicly available by 1978.

This decladding and waste treatment process was copied by North Korea (section 4.4.3.14) in the plutonium reprocessing facility under construction at Yongbyon.

4.4.3.4 Brazil

For thirty years, Brazil had a secret program to develop reprocessing capability. Beginning in 1960, the Brazilian army was responsible for research into Pu separation activities but had limited success. A laboratory-scale facility was operated for several years, extracting gram quantities of plutonium. The facility was reported closed in 1989, and no further work has been done. Consequently, Brazil has yet to succeed in the development of separations technology.[137]

4.4.3.5 China

Ground was broken in 1960 at the Juiquan complex for a co-located reactor, chemical separation plant, and plutonium processing plant. The reprocessing plant was completed in 1970.[138] It was based on early USSR technology, using the co-precipitation of sodium

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[135] Nuclear Chemical Engineering, Benedict, Pigford & Levi, McGraw Hill, Inc. (1981) pages 472

[136] North Korea Obtained Reprocessing Technology Aired by Eurochemic, Mark Hibbs, Nuclear Fuel, Vol. 19: pg 6, February 28, 1994

[137] Plutonium and Highly Enriched Uranium 1996 World Inventories, Capabilities and Policies, Albright, Berkhout and Walker, sipri, Oxford University Press, 1997

[138] "China Builds the Bomb" John Wilson Lewis & Xue Litai, Stanford University Press, 1988

uranyl acetate. Two facilities were eventually built and operated until 1984, when they were shut down due to process difficulties[139].

A replacement plant using the Purex process was built beginning in the mid-1960s and completed in 1974 in Guanyuan, Sichuan. It operated until the early 1990s.

At Lanzhou, Gansu, a construction of a multipurpose, commercial fuel, reprocessing plant commenced in 1997. Cold commissioning was expected in 2002. A larger plant (800 Metric Tons/yr) is to be constructed and be operational by 2020.[140].

4.4.3.6 Czech Republic

At the Nuclear Research Institute Řezplc in 2002, construction of a co-located molten salt reactor and reprocessing facility was completed. The reprocessing technology uses molten fluorides. The facility is at laboratory scale only and was not built for the purpose of weapons production[141].

4.4.3.7 France

The main facility for the production of plutonium for military purposes in France is the complex located at Marcoule, in the commune of Bagnols-sur-Ceze in the Gard. Founded in 1952, Marcoule was equipped with France's first plutonium production reactor, the natural uranium fueled, graphite moderated, gas-cooled G1 reactor, and its first plutonium separation plant based on Purex technology, known as UP1. Plutonium production for weapons use ceased in 1992. Marcoule/UP1 was capable of reprocessing 1-2 tons/day of metal fuel from G1 (and later G2 and G3)[142]. Three different reprocessing plants have been built at La Hague (1976, 1990, and 1994) for reprocessing commercial light water reactor fuel. Two additional plants were built at Marcoule for the reprocessing of fast reactor fuel and non-LWR fuel.

4.4.3.8 Germany

Pilot scale reprocessing facilities, built and operated at Karlsruhe between 1971 and 1991, were intended to advance development of Purex technology. Improved technologies used included: remote-controlled mixer-settlers, including titanium mixer-settlers for electrochemical extraction of U and Pu; a remote-controlled dissolver system allowing post-separation analysis; separations technology for all product streams; chemical treatment technology for waste solutions; new solvent regeneration systems; improved metering systems; and automatic data collection and data processing technology. These advances made it possible to reduce the number of process chemical

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[139] China Said To Be Preparing For Decommissioning Defense Plants, Mark Hibbs, Nuclear Fuel, Vol. 24, No. 10; Pg. 11, May 17, 1999.

[140] Chinese Pu Lab To Operate In 2002, But Interim Storage Now Foreseen, Mark Hibbs, Nuclear Fuel, Vol. 25, No. 22; Pg. 9, October 30, 2000

[141] Nuclear Research Institute Řež plc, Presentation at GEDEON-PRACTIS meeting Chateau de Caderache 2002 (http://www.gedeon.prd.fr/ATELIERS/AT19_20_06_2002)

[142] "The Nuclear Fuel Cycle from Ore to Wastes," P.D. Wilson, Oxford University Press, 1996

steps and maximize potential production of the Purex method"[143]. The facilities were shut down in 1991.

4.4.3.9 Italy

From 1961 through 1975, Italy worked with the concept of reprocessing thorium fuels. A small plant was built in Taranto in 1966, and designs for a large plant were developed in 1975[144]. The larger plant was never built.

4.4.3.10 India

India began designing and acquiring equipment for the Trombay plutonium reprocessing facility in 1958 with significant training and technical help from the United States. In 1961, construction began on the Trombay facility and in 1964 hot operations began, using the Purex process. The facility, known as Phoenix, produced the plutonium for India's first nuclear device. The facility was shutdown for repair between 1970 and 1972 and was decommissioned in 1973. In 1982, Trombay returned to service and is still operating.

A second reprocessing facility at Tarapur commenced operation in 1979. This facility was intended to be capable of reprocessing 100-150 MT/yr of Candu fuel using the Purex process, but serious operating problems prevented operation until 1990 when it became fully operational[145, 146].

Construction of a third facility began in 1996. Originally designed to reprocess 1000 MT of fuel/yr, it was de-rated to 100 MT/yr. This facility is sufficient to handle all of India's reactor fuel. In 1998, the facility was completed, cold and hot testing was performed, and the facility was placed into operation.

India is currently working on a variety of other fuel reprocessing methodologies. In 2000, construction began on a Fast Reactor Fuel Reprocessing Plant (FRFRP). The facility uses a Purex flow sheet with centrifugal contactors. Also in 2000, testing on an advanced pyro-electrochemical process began at the Indira Gandhi Centre for Atomic Research. Techniques are being developed at the Centre for the reprocessing of thorium fuels irradiated in the Advanced Heavy Water Reactor (AHWR). Finally, work has been commissioned on a Lead Mini Cell for Fast Breeder Test Reactor (FBTR) carbide fuel reprocessing technology using a Purex flow sheet with an electrolytic dissolver and centrifugal contactors.[147]

4.4.3.11 Iraq

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[143] "Future Uncertain For German Pilot Reprocessing Program", Mark Hibbs, Nuclear Fuel, Vol. 15, No. 1; Pg. 6, January 8, 1990

[144] "A Guide to Nuclear Power Technology," Rahn, Adamantiades, Kenton, & Braun, EPRI, John Wiley & Sons, New York, 1984

[145] India's Nuclear Bomb, George Perkovich, University of California Press, 2001.

[146] http://www.dae.gov.in/milestone.htm (official Indian government website)

[147] Processing of Nuclear Wastes, UIC Nuclear Issues Briefing Paper # 72, December 2001

An Italian-supplied, laboratory-scale, fuel reprocessing facility became operational in 1976. This facility was used to learn how to separate plutonium from irradiated uranium. In 1979, an Italian company provided a pilot plutonium-separation facility. Iraq also discussed a production-scale facility to be supplied by Italy. $200M was invested in the plant.[148] By 1982, the pilot plant was operational and, between 1982 and 1988, had separated 5.26 g of Plutonium [149]. In 1991, the reprocessing plant and the hot laboratories used for irradiated fuel reprocessing research and development were destroyed during the Gulf War.[150]

4.4.3.12 Israel

As early as 1950, Israeli scientists were involved in the construction of the first French production reactor and reprocessing plant. In 1957, the Dimona project began with French help, constructing a Purex based reprocessing facility. Te project was completed in 1963. The entire Dimona complex (reactor and reprocessing facilities) was completed in less than 6 years.[151]

4.4.3.13 Japan

Several reprocessing facilities have been built in Japan. In 1975, a plant was commissioned at Tokai Mura for reprocessing MONJU fast reactor fuel. The Tokai Reprocessing Plant (TRP) possesses the capacity of reprocessing approximately 0.7 kg/day using the solvent extraction process with mixer settler contactors.

In 1988, a facility was commissioned at Rokkasho for reprocessing commercial LWR fuel. The facility has the capacity to reprocess approximately 90 tonnes of LWR per year using the Purex process. The facility is still in operation.

Japan has also been advancing the reprocessing technology. Advanced Purex & Truex processes had been reported to be at the Hot Laboratory System Level Semiworks stage (step immediately prior to commercialization) as of 1998. The Truex process uses solvent extraction to separate the various transuranics from uranium and plutonium.[152]

In 2001, a larger facility was under construction at Rokkashamura for reprocessing Monju fast reactor fuel. When completed, the facility will be capable of reprocessing approximately 10 kg of fast reactor fuel per day.

4.4.3.14 North Korea

In 1987, a reprocessing facility under construction was detected. Known as the "Radiochemical Laboratory", it underwent test runs in 1990. It is believed that weapons grade Pu was separated and processed at this facility. Construction was observed on a

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[148] http://nuketesting.enviroweb.org/hew/Iraq/IraqAtoZ.html

[149] http://www.thebulletin.org/issues/1991/o91/o91reports.html

[150] http://www.iaea.or.at/worldatom/Programmes/ActionTeam/nwp2.html#product

[151] "Triple Cross - Israel, the Atomic Bomb & the Man Who Spilled the Secrets" by Louis Toscano, Carol Publishing Group, 1990

[152] http://www.nea.fr/html/trw/docs/mol98/session2/SIIpaper1.pdf

second, larger facility that reportedly incorporates technology obtained from western sources, included technology developed in Belgium.[136,153] Construction was halted in the early 1990s under international agreements.

4.4.3.15 Pakistan

In the late 1960s, Pakistan contracted with Belgian countries to design pilot plutonium separation facilities using Purex technology. The plans for the plant were completed in 1971, and the plant was constructed by 1973. The facility was capable of extracting approximately 15 kg Pu/y. Its capability was expanded in the mid 1990s.[154]

4.4.3.16 Soviet Union

In February 1948, the Mayak/B Plant produced the first Soviet plutonium. The process was a co-precipitation process using sodium uranyl acetate and plutonyl acetate precipitated from fuel dissolved in nitric acid. It only required 15 months to go from groundbreaking to initial plutonium production.[155,102] In the mid-70s, B Plant was modified to use the Purex process and designated as RT-1. RT-1 has a throughput of 400 tons of heavy metal per year[156].

Purex processing was again used in 1956 for the Siberian Chemical Combine (SCC)/Object 15 plant.

A major accident occurred at the Mayak/Kyshtym facility on September 29, 1957. The accident involved an acetate waste explosion. Approximately 740 PBq (20 MCi) were released in a plume over 1 km high, contaminating the Chelyabinsk, Sverdlovsk and Tyumen oblasts.

An underground chemical separation plant at Krasnoyarsk (Zheleznogorsk) was completed in 1964, and is believed to have used the Purex process. A second plant, the Mining & Chemical Combine Zheleznogorsk/RT-2 (at Krasnoyarsk) was also to have used the Purex process. Construction has been suspended on RT-2 since 1989.

The chemical separation plant at Tomsk (Seversk) was converted to the Purex process in 1983.

4.4.3.17 Taiwan

In 1965 a program was initiated to procure and operate a reprocessing research laboratory and plutonium separation plant. A facility was constructed consisting of hot cells and

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[153] "Solving the North Korean Nuclear Puzzle," D. Albright & K. O'Neill, The Institute for Science & International Security, Washington, D.C. 2000

[154] "The Long Road to Chagai," by Shahid-ur-Rehman

[155] "Making the Russian Bomb," Thomas Cochrane, Robert Norris, & Oleg Bukharin, Natural Resources Defense Council. Inc. Westview Press, Boulder, Colorado, 1995, Pages 79, 140 & 153

[156] 2001 World Nuclear Industry Handbook, 2001 Nuclear Engineering International

related reprocessing equipment for research. The program was halted in 1988 for political reasons.[157]

4.4.3.18 United Kingdom

The United Kingdom began reprocessing fuel from the Windscale reactors for the purpose of producing weapons grade plutonium in 1950. The reprocessing was performed at Stellafield using the co-precipitation process. In 1964, a plant was built at Stellafield for the reprocessing of commercial non-LWR Fuel. The reprocessing capacity was 1500 tonnes per year using the Purex process. This facility has processed over 35,000 tons of fuel to date. In 1994, a commercial LWR fuel reprocessing plant was built at Stellafield that was capable of reprocessing 1200 tonnes per year using the Purex process.[158]

In 1960, a plant was built at Dounreay for reprocessing fast reactor oxide fuel with a capacity of approximately 25 kilograms per day. The plant was reconstructed in 1980 and can now reprocess 30 kilograms per day.[159]

4.4.3.19 United States

Oak Ridge operated pilot-scale nuclear reactors and built a demonstration plant called X-10 for reprocessing nuclear fuel to recover plutonium and uranium for nuclear weapons. In 1943, 1.54 milligrams of plutonium were extracted at X-10. Operations ceased there in early 1945. When plutonium separation ceased at X-10, the Graphite Reactor and separations plant had produced a total of 326.4 grams of plutonium by the bismuth-phosphate process.[160]

The Hanford T Plant was a bismuth-phosphate batch precipitation process. It operated from 1944 until 1952. Construction began in June 1943 and was completed in September 1944 (15 months from ground breaking to first Pu production). The first batch of irradiated fuel rods from the B-Reactor was processed on December 26-27, 1944. The T Plant produced the plutonium for the Trinity test and the Nagasaki weapon.[161]

The Hanford/ B Plant followed the T plant and used a similar process. A bismuth-phosphate batch precipitation process was employed, repeatedly dissolving and centrifuging the plutonium-bearing solution. The B Plant operated from 1945 until 1952 and was superseded by the Hanford Redox plant.[162]

ORNL continued to be the lead facility in developing the demonstration processes. The Redox process was developed between 1946 and 1948. The process was used on low

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[157] http://www.nti.org/e_research/e1_taiwan_1.html

[158] "Processing of Nuclear Wastes" UIC Nuclear Issues Briefing Paper # 72, December 2001 (http://www.uic.com.au/nip72.htm)

[159] International Nuclear Societies Council's Committee Report "A Vision For The Second Fifty Years Of Nuclear Energy" 2001 (http://www2s.biglobe.ne.jp/~INSC/)

[160] http://www.em.doe.gov/oranlyt/ornl.html

[161] http://www.hanford.gov/history/mr0452/mr0452.htm

[162] "221-B (B Plant or B Canyon) Historical Facts," Westinghouse Hanford Company (http://www.hanford.gov/history/misc/bplant.htm)

burn-up fuel only for the recovery of enriched uranium. ORNL also constructed the demonstration facility for the Purex process development. From 1950 through 1953, 7 kg of Plutonium & 7.5 MT of uranium were processed by the Purex process at the demonstration facility at ORNL. ORNL also built the Thorex demonstration for the reprocessing of thoria fuel for the recovery of U^{233}. From 1954 through 1958, approximately 500 kg of U^{233} was recovered at the ORNL facility.[163]

The Hanford/Redox was the first solvent-extraction process plant. The solvent employed was hexone (methyl isobutyl ketone) used with packed column contactors. Hexone is very volatile and flammable and has a poor decontamination factor. Construction commenced in 1949, with plant closure in 1967.

The Idaho Chemical Plant made modifications to the Redox process. The updated process was used for Materials Test Reactor and spent naval reactor fuel reprocessing and involved the use of pulsed and packed columns. In 1992, reprocessing at the INEL Chemical plant was discontinued.

The Purex process was used for separation of weapons grade Pu from 1956 until reprocessing was suspended in the United States. The Hanford Purex plant began construction in 1953 and was operational in 1956. It was inactivated in 1974, resumed operation 1983, and again shutdown in 1988.

Savannah River had reprocessing plants using the Purex process, F Canyon (began operation in 1954) and H Canyon (began operation in 1955). F Canyon could reprocess 9 metric tons of uranium fuel per day. Mixer-settlers and centrifugal contactors were added in 1966. Operation was suspended in 1992 and resumed in 1995 for stabilization runs, with shutdown projected in 2002.

Argonne National Laboratory West developed a pyrometallurgical process for reprocessing the EBR-II fast reactor spent fuel. Argonne operated the process facility from 1963 through 1994.

At West Valley, NY, a large Purex plant was constructed and operated from 1966 through 1972. West Valley was capable of processing 300 tons of fuel per year. Escalating regulations required plant modifications and the plant was deemed uneconomical and shut down.

A 300 ton per year plant was built at Morris, Illinois, incorporating new technology that was proven on a pilot-scale, but failed to work in production scale. The foundation of the process was solvent extraction with pulse column contactors.

The construction of a 1500 ton-per-year plant at Barnwell, South Carolina, was aborted due to government policy change as one facet of the U.S. non-proliferation policy.

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[163] "Fuel Reprocessing at ORNL," J.W. Roddy, ORNL, 2000

4.4.3.20 Yugoslavia

In 1966, a lab-scale reprocessing facility capable of processing 1.5 tons/yr using hot cells and the Purex process was put into operation. Its operating history is unclear.

4.4.4 Technological Information Acquisition/Development

4.4.4.1 Acquisition approaches

4.4.4.1.1 In-country development

Of greatest interest are the successful developments in India, Taiwan, Israel, North Korea, and Pakistan. India and Taiwan appear to be the only nations that were successful in indigenous development of reprocessing with little outside help. India prepared its own design and went from design to operation in 6 years. The process was a Purex process. This was accomplished between 1958 and 1964, when concerns regarding India's development of nuclear weapons were not high, and chemical processing equipment could be purchased from the U.S. and Europe.

India did, however, receive training for their engineers and technicians in the U.S.. Clearly, the training received in the U.S. aided significantly in the design effort.

India's second effort at Tarapur, designed and constructed without outside help, had many difficulties and required 11 years of troubleshooting and modification after the initial operation of the facility to make it run successfully. However, the lessons learned at Tarapur allowed success in constructing and commencing operation of a large facility at Kalpakkam in approximately 3 years.

Taiwan developed and ran a covert weapons program from 1964 through 1988. Little is known regarding how the processing laboratory was developed. It was successful in separating approximately 10 kg of Plutonium[157] .

4.4.4.1.2 Overt/covert Purchase

In the early years (1950s and 1960s), chemical processing equipment could be purchased from the U.S. and Europe rather easily. At that time, the reprocessing of fuel was generally believed to be for the purpose of developing an economic nuclear fuel cycle. Since the mid-1970s, the purchase of or the attempt to purchase chemical equipment that could be used for plutonium separation has been restricted by international agreements.

4.4.4.1.3 Covert Acquisition

Transfer of technology has been the primary way third world nations have developed plutonium separation technology. Early on, many Western nations provided nuclear technology to third world nations under the guise of the use of the technology for peaceful purposes. America provided technology to Argentina and training to India.

France assisted Israel in the Middle East and transferred important technology. Belgium built Pakistan's Pinstech reprocessing facility. Russia supported North Korea's Yanggang-Do reprocessing facility and provided the technology and equipment. Italy sold technology to Iraq and Germany to Brazil. Until very recently, China has been willing to help any nation friendly to them to develop nuclear arms. In the future, it will be more difficult for a nation to develop a weapons program based on plutonium produced from research reactors and separated in hot cells.

4.4.4.2 Time constant for technology development

4.4.4.2.1 Minimum time

The time required for success varies widely and is strongly dependent on either help from nations who have already developed the technology or the nuclear and industrial maturity of the nation. The measure of success also plays heavily into the time factor. Brazil attempted several times with virtually no success. Argentina developed a laboratory process in 1968, but produced less than a kilogram of Pu^{239} by 1973.

India developed the Trombay facility in six years, but clearly had technological help from the United States. Their next attempt at Tarapur required nearly seventeen years for successful operation. However with the Tarapur experience, India successfully built and commenced operation at Kalpakkam in less than three years. Israel, with significant help from France, constructed the Dimona facility, reactor plus reprocessing, in less than six years.

Yugoslavia developed a laboratory scale PUREX process in ten years, starting in 1956 and completing in 1966. It must be taken into account that this was during a period when equipment was readily available from the West.

Belgium built a laboratory-scale facility in Pakistan between 1970 and 1973, but had already fully developed the technology and equipment. The three years was basically the construction period.

Six years would probably be a minimum time for indigenous development of a small to medium reprocessing facility if equipment could be purchased. In the current atmosphere of non-proliferation, ten years would be very optimistic for a country that has yet to make an attempt at reprocessing.

4.4.4.2.2 Country Average

The average time to develop a reprocessing technology is approximately ten years. Several nations made attempts to develop this technology and were unsuccessful. Currently, a totally indigenous process would probably require much more than a decade. Outside help, from countries possessing the technology and equipment, can reduce the time considerably.

4.4.5 Required Concurrent Technologies

Reprocessing technology is based on standard chemical operations. Literature on the chemistry and equipment required is widely available. Technologies required to support development of a facility include 1) fuel element equipment, 2) fuel dissolution, 3) fissile element separation, 4) uranium or plutonium product purification, 5) metal preparation, and 6) waste treatment/recycle facilities.

5.0 INSIGHTS AND CONCLUSIONS

Ten technologies having the potential to be used in proliferation programs have been examined. Table 31 summarizes program information derived in this analysis.

Technology	Number Of Countries Interested in Technology	Number Of Countries With Successful Production Programs*	Average Time To Pilot Plant**	Average Time To Production***
Gaseous diffusion enrichment	6	5	-	6 years
Centrifuge enrichment	18	7	8 years	14 years
Electromagnetic isotope separation	11	1	2 years	3 years
Chemical isotope separation	3	-	6 years	11 years
Aerodynamic isotope separation	3	1	7 years	18 years
Laser enrichment	14	-	-	-
Graphite-moderated production reactors	6	6	1 year	2-11 years
Heavy-water-moderated reactors	12	5	1 year	2-6 years
Research reactors	14	3	-	4-5 years
Reprocessing	19	13	6 years	10 years

*More than gram quantities of material produced
**Technological capability demonstrated
***Significant quantities of material produced

Table 31 Program summary

The three approaches with the shortest time frames shown in Table 31, graphite and heavy water reactors and electromagnetic isotope separation, have short time frames because they were the products of very expensive and personnel-intensive wartime programs to develop the techniques. Other approaches to obtaining special nuclear material take about six years to demonstrate the technology and another six to eleven years to actually produce significant quantities of material.

It is reasonable to assume that these time frames can be used as starting points when non-proliferation models are built, even when different technologies are considered. These timeframes also indicate it is important that models constructed include aspects that evaluate means for detection of covert programs

5.1 Enrichment Technology Summary

Enrichment technology continues to evolve. There is a need for new enrichment facilities; by 2005, 90% of existing enrichment capacity will be more than 15 years old, with 70% more than 25 years old. It appears that the current trend in enrichment technology development is to concentrate on centrifuge enrichment for commercial production. Other proposed commercial enrichment technologies such as laser enrichment and chemical enrichment, while showing some technical promise, have been considered non-competitive with centrifuge technology. Thus, government-funded research has essentially stopped. Gaseous diffusion enrichment facilities are aging and except for the French Eurodif plant are scheduled to be closed.

Two enrichment technologies successfully used in weapons programs, electro-magnetic and aerodynamic isotope separation, are very expensive, difficult to operate, and are unlikely to be used commercially. They would only be used in other weapons programs.

Accordingly, the timeframes identified above can be considered representative of current development efforts.

5.2 Plutonium Production Technology Summary

Plutonium production technology can be considered mature. It is doubtful that great advances will be made in either graphite- or heavy-water-moderated reactor technology. The same can be said for research reactors. The technology is well known and understood. Reprocessing technology is in a similar state. Although research is on going in advanced reprocessing techniques (i.e., non-aqueous technologies such as pyrochemical processing), conventional reprocessing approaches such as Purex and bismuth-phosphate reprocessing are well known.

As with enrichment techniques, the timeframes identified above can be considered representative of current development efforts.

Appendix A – Glossary

Word	Definition/Description
Advanced gas-cooled reactor (AGR)	A graphite-moderated, CO_2-cooled thermal reactor with slightly enriched uranium as a fuel.
Aerodynamic enrichment	A process of uranium enrichment based on the centrifugal effects of a fast-moving uranium hexafluoride (UF_6) gas in very small curved-wall chambers.
Alpha particle	A charged particle emitted from the nucleus of an atom, having a mass and charge equal in magnitude to a helium nucleus.
Atom	A particle of matter indivisible by chemical means-the fundamental building block of the chemical elements.
Atomic number	The place occupied by an element in the Periodic Table of Elements. It is determined by the number of protons in the nucleus of an atom.
Beta particle	An electron or positron emitted from a nucleus during radioactive decay.
Burn-up	A measure of reactor fuel consumption. It is expressed as the amount of energy produced per unit weight of fuel in the reactor.
Calutron	(From Ca lifornia U niversity Cyclo tron). A calutron is an electromagnetic uranium enrichment machine. Calutrons were used early in the Manhattan Project to produce highly enriched uranium (HEU) for the Hiroshima bomb and also developed in the Iraqi bomb program. Alpha machines are the first stage, producing low-enriched uranium (LEU) from natural uranium; beta machines are the second stage, producing HEU from the output of the alpha machines.
CANDU	A reactor of Canadian design, which uses natural uranium as fuel and heavy water as moderator and coolant.
Cascade	A connected series of enrichment machines, material from one being passed to another for further enrichment,
Centrifuge	A rotating vessel used to enrich uranium. The UF_6 gas molecules containing heavier isotopes of uranium concentrate at the walls of the rotating centrifuge and are drawn off.
Centrifuge isotope separation	An enrichment process in which lighter isotopes are separated from heavier ones by means of ultra high speed centrifuges.
Chain reaction	A reaction that stimulates its own repetition. In a fission chain reaction, a fissionable nucleus absorbs a neutron and undergoes fission, releasing additional neutrons. These in turn can be absorbed by other fissile nuclei, releasing still more neutrons. A fission chain reaction is self-sustaining when the number of neutrons released in a given time equals or exceeds the number of neutrons lost by absorption in non-fissile material or by escape from the system.
Chemical enrichment	A method of uranium isotope separation that depends on the slight tendency of ^{235}U and ^{238}U to concentrate in different molecules when uranium compounds are continuously brought into contact. Catalysts are used to speed up the chemical exchange.
Cladding	The material (aluminum, stainless steel, magnesium alloy, or zirconium alloy) in which the fuel elements in a reactor are sheathed.
Coolant	A substance circulated through a nuclear reactor to remove or transfer heat. Common coolants are light or heavy water, carbon dioxide and liquid sodium.
Core	The central portion of a nuclear reactor containing the fuel elements and usually the moderator, but not the reflector.
Cut	The process ratio of the amount of material enriched to the amount of feed material.
Depleted uranium	Uranium with a smaller percentage of ^{235}U than the 0.7 per cent found in natural uranium. It is a by-product of the uranium enrichment process, during which ^{238}U is culled from one batch of uranium, thereby depleting it, and added to another batch to increase its concentration of ^{235}U.
Electromagnetic isotope separation (EMIS)	A process of uranium enrichment in which UCl_4 ions are passed through a magnetic field. Since they have different masses, ^{235}U and ^{238}U atoms will pass through different trajectories, and can be collected at different locations (see calutron).
Enriched uranium	Uranium in which the percentage of ^{235}U has been increased beyond that found in natural uranium.
Enrichment	A process by which the relative abundances of the isotopes of a given elements are altered, thus producing a form of the element enriched in one particular isotope.

Word	**Definition/Description**
Feed material	Material introduced into a facility at the start of the process, such as uranium hexafluoride (UF_6) in an enrichment plant
Fertile material	Material composed of atoms which readily absorb neutrons to produce fissionable materials. One such element is ^{238}U, which becomes ^{239}Pu after it absorbs a neutron. Fertile material alone cannot sustain a chain reaction.
Fissile material	A material fissionable by neutrons of all energies, especially thermal neutrons: for example, ^{235}U and ^{239}Pu.
Fission	The process by which a neutron strikes a nucleus and splits it into fragments or 'fission products'. During the process of nuclear fission, several neutrons are emitted at high speed and radiation is released. Fissions can occur spontaneously but usually are caused by absorption of neutrons.
Fissionable material	Material whose nuclei can be induced to fission by a neutron.
Fuel	Fissile material used or usable to produce energy in a reactor. Also applied to a mixture, such as natural uranium, in which only part of the atoms are fissile, if the mixture can be made to sustain a chain reaction.
Fuel cycle	The series of steps involved in preparation and disposal of fuel for nuclear reactors. It includes mining, refining the ore, fabrication of fuel elements, their use in a reactor, chemical processing to recover the fissile material remaining in the spent fuel, re-enrichment of the fuel material, and refabrication into new fuel elements
Fuel element	A rod, tube, plate, or other mechanical shape or form into which nuclear fuel is fabricated for use in a reactor.
Fuel reprocessing	The chemical processing of spent reactor fuel to recover the unused fissile material.
Gamma radiation	High-energy electromagnetic radiation emitted from nuclei as a result of nuclear reactions and decay.
Gas-centrifuge process	See Centrifuge isotope separation
Gas-Cooled reactor	A nuclear reactor employing a gas (usually carbon dioxide, CO_2) as a coolant, rather than water or liquid metal.
Gaseous diffusion	A method of isotopic separation based on the fact that gas atoms or molecules with different masses will diffuse through a porous barrier (or membrane) at different rates. The method is used to separate ^{235}U from ^{238}U.
Gas-graphite reactor	A nuclear reactor in which a gas is the coolant and graphite is the moderator.
Graphite	A form of pure carbon, used as a moderator in nuclear reactors.
Half-life	The time in which half of the atoms in a given amount of a specific radioactive substance disintegrate.
Heavy water	Water in which the ordinary hydrogen is replaced by deuterium.
Heavy water moderated reactor	A reactor that uses heavy water as its moderator. Heavy water is an excellent moderator that permits the use of natural uranium as a fuel.
Highly enriched uranium (HEU)	Uranium in which the percentage of ^{235}U nuclei has been increased from the natural level of 0.7 per cent to some level greater than 20 per cent, usually around 90 per cent.
High temperature gas-cooled reactor (HTGR)	A graphite-moderated, helium-cooled reactor using highly enriched uranium as fuel.
Hot cells	Lead-shielded rooms with remote handling equipment for examining and processing radioactive materials. In particular, hot cells are used for examining spent reactor fuel.
Irradiation	Exposure to a radioactive source; usually in the case of materials being placed in an operating nuclear reactor.
Isotopes	Nuclides of the same chemical element but different atomic weight, that is with the same number of protons but different numbers of neutrons.
Jet nozzle enrichment	Process of uranium enrichment based on pressure diffusion in a gaseous mixture of uranium hexafluoride and an additional light gas flowing at high speed through a nozzle along curved walls. (see aerodynamic enrichment)
Laser enrichment	An isotope separation technique, in which ^{235}Uatoms are selectively excited or ionized by lasers.
Light water-cooled, graphite-moderated reactor	A reactor cooled by light water and moderated with graphite. The fuel and coolant are contained in pressure tubes that pass through the graphite moderator.
Magnox reactor	An early version of the AGR; using natural uranium as fuel and magnesium oxide alloys as the fuel cladding.
Maraging steel	Special hardened steel used in the fabrication of centrifuge rotors and rocket engines.

Word	Definition/Description
Mass number	The number of protons and neutrons in the atomic nucleus. Elements may occur in forms (isotopes) displaying a range of mass numbers i.e., ^{231}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu and ^{242}Pu.
Megawatt electric (MWe)	The amount of power, in megawatts, generated by a reactor in the form of electricity.
Megawatt thermal (MWth)	The amount of power, in megawatts, generated by a reactor in the form of heat
Megawatt-day per kilogram (MWd/kg)	A unit used for expressing the burn-up of fuel in a reactor: specifically, the number of megawatt-days of heat output per kilogram of fuel in the reactor.
Moderator	A material, such as ordinary water, heavy water, or graphite used in a reactor to slow down fast neutrons to thermal energies.
Natural uranium	Uranium as found in nature, containing 0.7 per cent of ^{235}U, 99.3 per cent of ^{238}U, and a trace of ^{234}U.
Neutron	An uncharged elementary particle with a mass slightly greater than that of the proton, and found in the nucleus of every atom heavier than hydrogen.
Non-weapon grade material	A material containing fissile nuclides but at a concentration so low as to make it unsuitable for nuclear weapons.
Nuclear energy	The energy liberated by a nuclear reaction (fission or fusion) or by radioactive decay.
Nuclear reactor	A device in which a fission chain reaction can be initiated, maintained, and controlled. Its essential component is a core with fissile fuel. It usually has a moderator, a reflector, shielding, coolant, and control mechanisms.
Nuclide	Species of atom characterized by the number of protons and the number of neutrons in its nucleus.
Production reactor	A reactor designed primarily for large-scale production of ^{239}Pu by neutron irradiation of ^{238}U
Plutonium (Pu)	A radioactive, man-made, metallic element with atomic number 94. Its most important isotope is fissile ^{239}Pu, produced by neutron irradiation of ^{238}U. It is used for reactor fuel and in weapons.
Radioactive decay	The gradual decrease in radioactivity of a radioactive substance due to nuclear disintegration, and its transformation into a different element. Also called radioactive disintegration.
Radioactivity	The spontaneous decay or disintegration of an unstable atomic nucleus.
Radioisotope	Radionuclide-any nuclide which undergoes radioactive decay
Research reactor	A low power reactor primarily designed to supply neutrons for experimental purposes. It may also be used for training, materials testing and production of radioisotopes.
Separative work unit (SWU)	A measure of the work required to separate uranium isotopes in the enrichment process. It is used to describe the capacity of an enrichment plant.
Tail assay	The percentage of ^{235}U left in the depleted uranium after passing through the enrichment plant.
Tails/Tailings	1. The uranium ore left after the extraction of the uranium in the milling plant. 2 The depleted uranium remaining after the enrichment process.
Thermal neutron	A neutron in thermal equilibrium with its surrounding medium. Thermal neutrons are those that have been slowed down by a moderator to an average speed of about 2200 meters per second (at room temperature) from the much higher initial speeds they had when expelled by fission.
Thorium (Th)	A naturally radioactive element with atomic number 90. The isotope thorium-232 can be transmuted to fissile ^{233}U by neutron irradiation.
Urenco	The Uranium Enrichment Company, created in 1970 by the signing of the Treaty of Almelo by the Federal Republic of Germany, the United Kingdom, and The Netherlands.
Weapon-grade material	Nuclear material of the type most suitable for nuclear weapons, i.e., uranium enriched to over 90 per cent ^{235}U or plutonium that is primarily ^{239}Pu.
Yellowcake	A uranium compound consisting mainly of U_3O_8.