Appendix V: TECHNICAL DESCRIPTION OF FUEL CYCLE FACILITIES AND EVALUATION OF DIVERSION POTENTIAL

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APPENDIX V

TECHNICAL DESCRIPTION OF FUEL CYCLE FACILITIES

AND DIVERSION POTENTIAL

1. INTRODUCTION

Considerable concern has been expressed over the possibility that nations could extract from their nuclear power systems the fissile material essential for nuclear weapons. There is, in fact, no intrinsic reason why they could not do so, although no nation which has nuclear weapons has effected them by these means. It is the intent of this section to examine the existing and potential reactors and their associated fuel cycles. With this background, the possibility of diversion from each system can be understood and compared. In the past, proliferation potential has not been considered as a parameter in the design of nuclear power systems. If diversion is increasingly perceived as a problem, however, it may be found desirable to favor   those systems which are least vulnerable.

2.1 INTRODUCTION

Nuclear energy is derived from the conversion of mass to energy by splitting or joining nuclei. A fission reaction occurs when a heavy nucleus is struck by a neutron and shatters into two or more intermediate weight nuclei and additional neutrons with slightly less total mass than the original nucleus. This mass defect is converted to energy - in the form of radiation and particle motion. The only naturally occurring nucleus that readily fissions (i.e. is fissile) when struck by a neutron is ^{235}U (the isotope of uranium containing 92 protons and 143 neutrons for an atomic mass of 235). All others usually either deflect or absorb neutrons. There are other notable fissile isotopes, all manmade. These are ^{233}U and all isotopes of plutonium.

A chain reaction occurs when neutrons emitted from fissioning nuclei cause other nuclei to fission. This can happen only under certain conditions. **There** must be sufficient fissile material present, arranged in an appropriate geometry. A moderator may have to be present to slow the high energy (fast) neutrons emerging from the fissioning nuclei so that they may be more readily captured by other fissile nuclei. There cannot be too many other nuclei present which absorb neutrons. To produce useful power some means of control to keep the chain reaction at a constant rate must be included **and the heat generated must be removed by a coolant.**

Uranium is naturally found as a mixture of two isotopes: the fissile ^{235}U **(0.71%) and** ^{238}U **(99.29%). Natural uranium can be made to go critical (i.e., sustain a chain reaction) only under very limited conditions**

because the ratio of fissile to non-fissile material is low. Hence

for use as a nuclear fuel uranium is usually enriched in the fraction of ^{235}U to

perhaps 3%. The criteria for the choice of coolant, moderator and structural

materials then become less stringent since fewer neutrons are abosorbed by ^{238}U.

When a nucleus absorbs a neutron without fissioning, it is converted into

another isotope of the same element. This may be itself a fissile nucleus of it

may indirectly result in one by a short term decay process. In this way the

uranium isotope ^{238}U is transformed into ^{239}U " which after emission of an electron

(beta particle) becomes ^{239}Np. This in turn decays by a beta emmission to become

^{239}Pu. In a similar way, the thorium isotope ^{232}Th are said to be fertile. Reactors

can be fueled with any of the fissile isotopes and supplied with fertile material

to breed more fuel.

The fuelelements of nearly **all** power reactors contain both ^{238}U and $_U235$. In

normal operation, some of the ^{238}U is converted to ^{239}Pu, some of which is fissioned.

If the plutonium in the spend fuel is less than the ^{235}U in the fresh fuel, the

reactor is called a converter or breeder. A sustainer reactor would be one which

produces the same amount of fissile material as it consumes. Some reactors, known

as breeders, produce more fuel from fertile isotopes than they use in operation.

Reactors operate on one of two major fuel cycles. The one used in most

reactors today is the uranium-plutonium cycle where the initial fissile material

is ^{235}U and plutonium is generated from the fertile ^{238}U. The other cycle is

thorium-uranium where ^{233}U is fissile and ^{232}Th is fertile. The major nuclear

reactions for these cycles are shown in Figure 1.

A neutron emitted by a fissioning atom has a high velocity and is referred

to as a fast or high energy neutron. As it strikes nuclei in its path, it loses

energy and slows down. It is then referred to as a slow or thermal neutron.

$$N + \boxed{^{232}Th} \longrightarrow {}^{233}Th \xrightarrow[\text{22.2 min.}]{\beta^-} {}^{233}Pa \xrightarrow[\text{27.0 day}]{\beta^-} \left(^{233}U\right)$$

$$N + \left(^{233}U\right) \longrightarrow \text{fission}$$
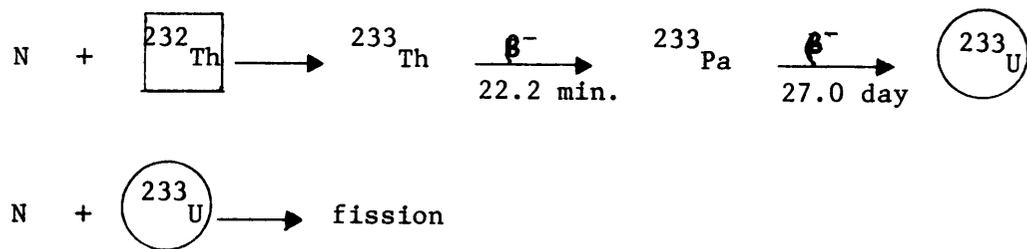
(b) Thorium - Uranium

Figure 1.   Uranium and Thorium Fuel Cycles.   (Nuclides in the circles are "fissile," that is, they readily undergo fission in nuclear reactors. Nuclides enclosed in square boxes are fertile. They undergo very little fission themselves, at least in thermal-neutron reactors, but are converted by neutron capture into fissile nuclides.)

The efficiency of breeding of fertile material is dependent upon the neutron energy spectrum. In general, thorium is bred more efficiently by thermal, or low energy neutrons while 238_U is bred more efficiently with fast or high energy neutrons. Present fast breeder reactors are generally based on the U-Pu fuel cycle although there is interest in a Thorium fast breeder. Thermal breeders must be based on the Th-U fuel cycle. The reactor neutron energy spectrum is mainly determined by the type of coolant and/or moderator.

As illustrated by the above discussion, there are many character-istics which define a nuclear reactor. Figure 2 illustrates a fission power reactor characterization tree. Since the reactor is the dominant part of the fuel cycle, the fuel cycle itself is generally characterized by the reactor. Typically, the fuel cycle is expected to contain those elements depicted in Fig. 3. At present, fuel is not being reprocessed, and the cycle ends with spent fuel storage. The component which are most vulnerable to diversions are uranium enrichment, spent fuel processing and the transportation of their products. Each of the major elements will be discussed. The reactor concept will be identified

2.2 M I N I N G

Uranium is the principal fuel required for present nuclear reactors. It has been estimated that uranium constitutes 2-4 ppm of the earth's crust. Most of it, however, is such low grade (less than 0.001% U_3O_8) that its extraction may not be economical. Presently, commercially

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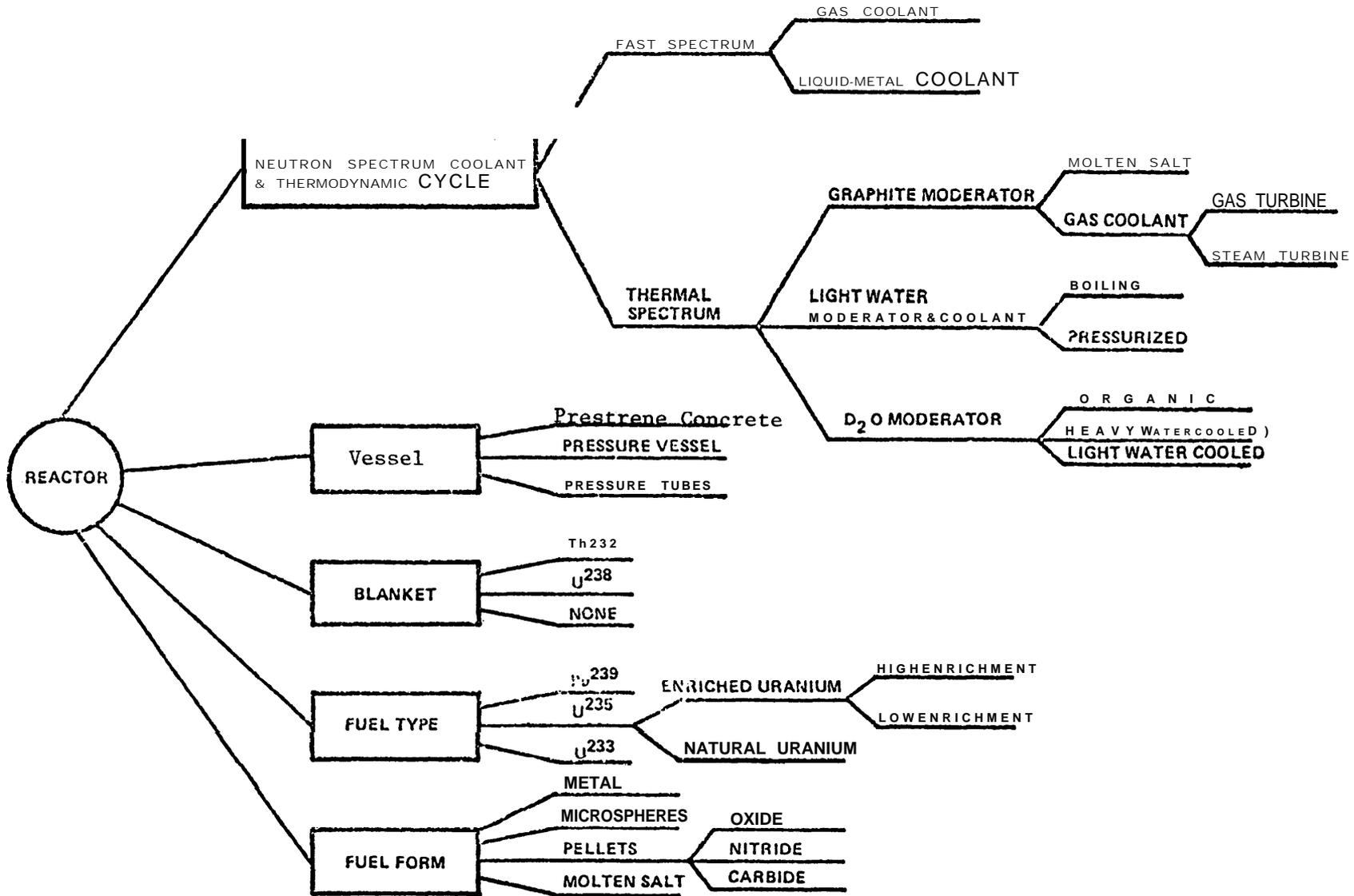
Fig. 2 . Fission Power Reactor Classification

REACTOR

NEUTRON SPECTRUM COOLANT & THERMODYNAMIC CYCLE

FAST SPECTRUM
- GAS COOLANT
- LIQUID-METAL COOLANT

THERMAL SPECTRUM
- GRAPHITE MODERATOR
  - MOLTEN SALT
  - GAS COOLANT
    - GAS TURBINE
    - STEAM TURBINE
- LIGHT WATER MODERATOR & COOLANT
  - BOILING
  - PRESSURIZED
- D_2O MODERATOR
  - ORGANIC
  - HEAVY WATER COOLED
  - LIGHT WATER COOLED

Vessel
- Prestrene Concrete
- PRESSURE VESSEL
- PRESSURE TUBES

BLANKET
- Th232
- U^{238}
- NONE

FUEL TYPE
- Pu^{239}
- U^{235}
- U^{233}
- ENRICHED URANIUM
  - HIGH ENRICHMENT
  - LOW ENRICHMENT
- NATURAL URANIUM

FUEL FORM
- METAL
- MICROSPHERES
- PELLETS
- MOLTEN SALT
- OXIDE
- NITRIDE
- CARBIDE

Fig. 3 The Generalized Fuel Cycle

attractive ores must contain at least 0.05% U$_3$O$_8$. **Deposits mined currently**
contain between 0.1 to 0.5%. In some cases however, uranium ore is a
byproduct of the recovery processes of other minerals such as gold or
phosphates. About 12,000 metric tonnes (MT) of U$_3$O$_8$ are produced annually
in the U.S. Proven reserves in this country are 500,000 MT. The total
recoverable resource is probably several times this. World resources
are discussed in Appendix VIII, but it should be noted that most nations
have at least some low grade ores. A typical reactor requires about 150 MT
of U$_3$O$_8$ per year.

Thorium is generally estimated to be three to five times more abundant than
uranium and is found in veins, sedimentary rocks and sands. Most of the
thorium currently produced in the U.S. (see Fig. 7) is a byproduct of
rare earth extraction. The annual production is about 150 MT. There
appears to be more than 100,000 MT of ThO$_2$ at \$10/pound in the U.S.; world
supplies are five to ten times this amount.

A typical uranium or thorium mine may process about 1,000 MT ore/day.
This would yield about 1,000 MT U$_3$O$_8$ or 1,500 MT of ThO$_2$ per year.
The ratio of overburden to ore ranges from 1 to 10. The capital cost
for the mine would range from \$10 to \$20 million, with an operating cost
of \$1 million (about \$0.50/lb for operations). Required equipment is
similar to that required for other mining operations. Mines are either
open pit or underground depending on depth. Underground mines are more expen-
sive to develop but are more secure from surveillance than pit mines if

clandestine operation is required. The usual hazards of underground mines
are augmented by the presence of radioactive radon gas, which can in the
long term cause cancer in the miners.

2.3 MILLING

In the milling operation uranium is recovered from the ore and
purified in preparation for subsequent fuel fabrication operations or
conversion for enrichment processing. The product of the milling
operation is a uranium salt called yellowcake, which contains between
70% and 90% U_3O_8 The established milling industry in the U.S. has
a capacity to produce about 20,000 tons U_3O_8 annually in 16 mills. Individual
production capacity ranges from 400 tons of ore per day to 7,000 tons of
ore per day.

The unit operations at a mill include crushing, grinding, leaching,
solids separation, extraction and yellowcake precipitation. The specific
methods vary with the composition of the ore mined. A general flow sheet
for a uranium mill using the acid leach-solvent process is shown in Fig. 4.

Major plant features include an ore storage and blending area; a
crushing building; a mill building containing grinding equipment, leaching
tanks, precipitation tanks, drying and packaging equipment; a solvent
extraction building; a tailings retention system; a sewage treatment
system, and several auxiliary buildings needed for offices and maintenance.
A typical mill could process 1,000 MT ore per day, requiring a capital
investment of about $10 million with operating costs of about $10 per
ton of ore (about $1.00/lb U_3O_8). The land required for the mill is about
300 acres. The equipment is similar to that in other ore milling industries.

ATMOSPHERE

STOCKPILED

t

FILTER

SCREENING AND CRUSHING

3/4 IN. SIZE

ATMOSPHERE

HOT WATER

WET GRINDING

SCRUBBER

SMALL MESH

H20

H2S04

NoC10$_3$

ACID LEACHING

ISL URRY

SOLID WASTES TO TAILINGS

COUNTER-CURRENT RECANTATION

PREGNANT LIQUOR

FLOCCULANTS

CLARIFIER AND FILTERS

PREGNANT LIQUOR

RAFFINATE RECYCLE

r

4

ORGANIC SOLVENT

SOLVENT EXTRACTION

RAFFINATE TO TAILINGS

IORGANIC IRECYCLE

SALT SOLUTION

PREGNANT ORGANIC PHASE

STRIPPING CIRCUT

STRIP SOLUTION RECYCLE

PREGNANT AQUEOUS PHASE

NH^3

PRECIPITATION AND THICKENING

ATMOSPHERE

AMMONIUM DIURANATE $(NH_4)_2U_2O_7$

FILTER

b

CALCINATION AND PACKAGING

U30B PRODUCT

Figure 4. Generalized Flow Sheet for Uranium Milling

STOCKPILED ORE

H_2O → CRUSHING GRINDING → FILTER

↓ SMALL MESH

H_2SO_4 → ACID LEACHING

↓ SLURRY

RESIDUE
TAILINGS L   COUNTER- CURRENT DEcANTATION H_2O

PREGNANT LIQUOR

F LOCCULANTS → CLARIFIER 81 FILTER

STRIPPING
SOLUTION

PREGNANT LIQUOR

LOADED ORGANIC

SOLVENT EXTRACTION   STRIPPING

RAFFINATE
TO RARE EARTH RECOVERY   STRIPPED ORGANIC

LOADED STRIP
SOLUTION

No~s204
Na_2CO_3 → REDUCTION
NEUTRALIZATION

FILTRATE
TAILINGS ← FILTRATION ← WASH SOLUTION

CAUSTiC DIGESTION — No OH

FILTRATE
TAILINGS — FILTRATION — WASH WATER

FILTER CAKE(Th (OH) ~)

Figure 5. Thorium Milling Flow Sheet

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Recovering thorium from its ore requires milling and refining. The

milling process extracts the ore and upgrades it to predetermined specifications.

The techniques for current production of thorium from monazite sand are proprietary.

The thorium mill may employ an acid leach-solvent extraction process, as depicted

in Fig. 5. The major steps in this milling operation include crushing, leaching,

decanting, solvent extraction and precipitation to from a crude mill product.

The second part of the thorium recovery process is to refine this mill

product into material suitable for nuclear reactor fuel. One requirement is

that the uranium content must be less than 10 parts per million because of isotopic

dilution of the U-233 formed in the reactor.

The large neutron cross section of the lanthanides requires that their

concentration be reduced to 1-5 ppm. The most economical method of purification

is considered to be the counter--current solvent extraction process. A flow

diagram of the refinement is given in Fig. 6.

The capital and operating costs for thorium processing are likely to

be similar to those for uranium.

2.4 CONVERSION

For those reactors requiring enriched uranium, i.e. concentration of

U-235 greater than . 71%, the U_3O_8 must be converted to uranium hexa-

fluoride (UF_6), the only compound of uranium which is gaseous at a

temperature low enough for easy handling. In a conversion plant, the

yellowcake is purified and converted to approximately 99.9% Pure U^F6 and

FILTER CAKE

```
                    ┌─────────────────────────┐
                    │                         │        HN0₃
                    │        DISSOLVER        │ 4  ─────────
                    │                         │        ~O
                    └─────────────────────────┘
```

$$\text{DISSOLVER} \quad 4 \; \frac{HNO_3}{\sim O}$$

THORIUM NITRATE SOLUTION I

LOADED ORGANIC   **STRIP SOLUTION**

```
   ┌──────────────────┐         ┌──────────────────┐
   │     SOLVENT      │         │      SALT        │
   │   EXTRACTION     │         │   STRiPP!NG      │
   └──────────────────┘         └──────────────────┘
```

STRIPPED ORGANIC

RAFFi NATE TAILING S

LOADED STRIP SOLUTION

```
                ┌──────────────────────┐
                │                      │
                │        DRYER         │
                │                      │
                └──────────────────────┘
```

t
THORIUM NITRATE CRYSTALS
(NUCLEAR GRADE)
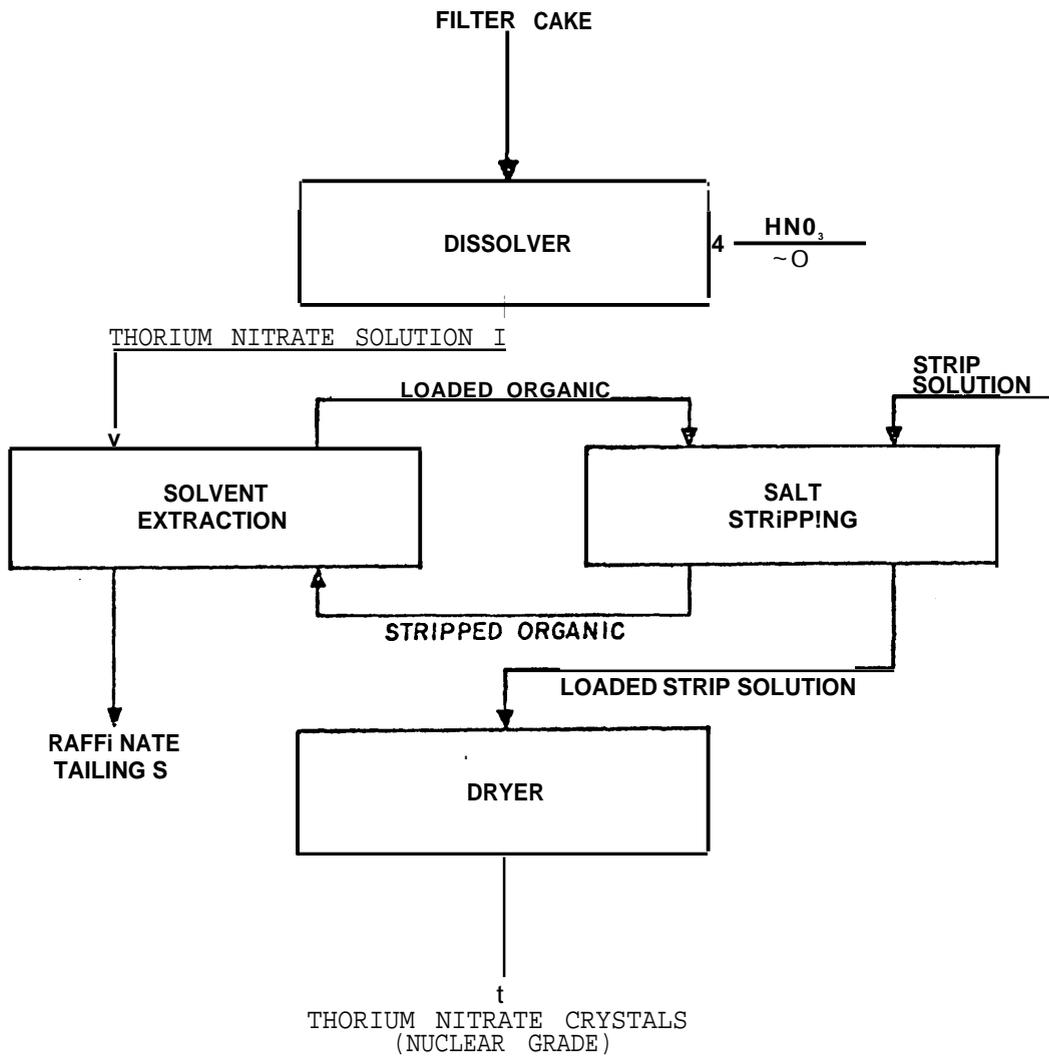
Figure 6.   Thorium Refining

ROASTED URANIUM CONCENTRATE

$'3°8$

$'2$

REDUCTION

$'2$

UO_2

ATMOSPHERE

.HF

HYDRO -
FLUORINATION

SCRUBBING

UF_4

LIQUID WASTE

$^F2$

FLUORINATION

UF_6

SOLID
WASTES
BURIED

COLD TRAP

UF_6

DISTILLATION
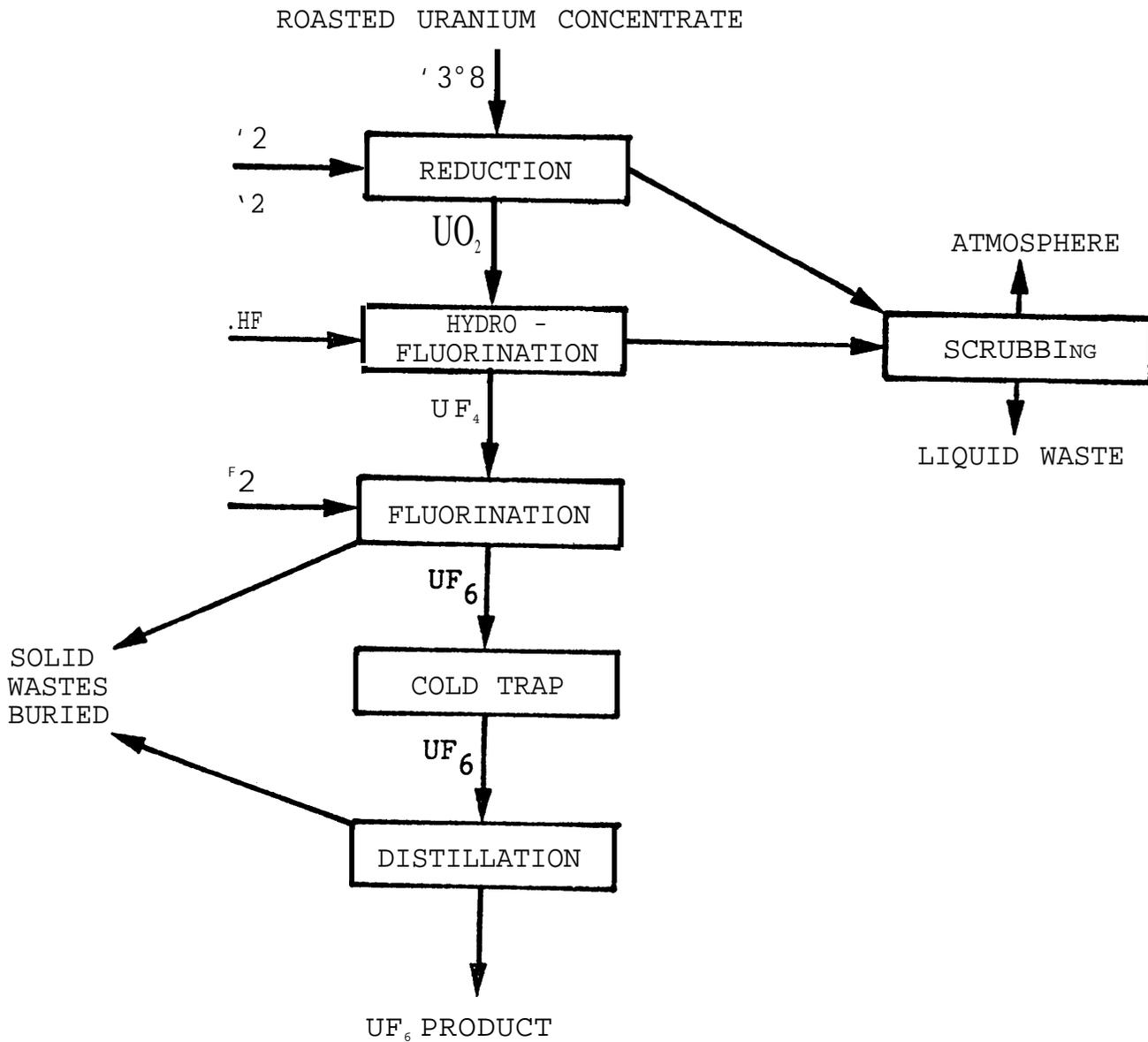
UF_6 PRODUCT

Figure 7.   UF6 Production-Dry Hydrofluor Process,
            Simplified Block Flow Diagram

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is shipped in special cylinders to the enrichment facilities.

Conversion plants in the U.S. use both a dry (hydrofluor) and **a** wet process. Total production capacity is about 20,000 MT of uranium per year as UF_6, with the hydrofluor process accounting for about 75% of this total. A simplified flow diagram of the hydrofluor process is shown in Fig.

A typical plant would have a capacity to convert 5,000 MT of uranium to UF_6 per year. This plant capacity requires approximately 1,000 acres and a capital cost of about $35 million. Operating costs for a plant of this type run about $1.25 per pound of uranium. Typical chemical process equipment is required.

If uranium is recycled from the reprocessing plant, it must be converted to UF_6 for reenrichment. The reconversion is performed in a plant specifically dedicated to that task; it converts UO_2 $(NO_3)_2 \cdot 6H_2O$ to UF_6. A typical plant might have a capacity of 1200 MT/year, with capital costs of about $50 million and operating costs of about $1.00 per pound of uranium.

There is no counterpart process for thorium because it contains no fissile isotope to be enriched.

2.5 ENRICHMENT

In none of the fuel cycle steps considered so far is the uranium in a form suitable for use in weapons; the concentration of the fissile isotope U-235 is far too low. The next step for most uranium reactor types except heavy water reactors is to enrich the uranium. Light water reactors (LWR) require 2-4% U-235 (still not suitable for weapons), but some power and research reactors use "fully enriched" uraniums which contains 93% U-235. Because of its potential for producing the highly

enriched uranium required for nuclear weapons, the enrichment plant is of

great concern in preventing proliferation. Even if a plant is built to

produce low-enriched fuel for LWR's, it could be restructured, possibly

even clandestinely, to produce weapons-grade uranium.

For all enrichment techniques, a key parameter is the separative

work unit (SW), which is a measure of the work to obtain a certain degree

of separation. It is defined in reference to a kilogram of uranium.

An enrichment plant will use 1 kg of SWU in processing 2.35 kg of

natural uranium feed (0.71% U-235) to provide 1 kg of product enriched

to 1.4% 235U (twice the natural enrichment) and 1.35 kg of waste

(called "tails") with a concentration of 0.2% ^{235}U. This is illustrated

in Figure 8. The SWU required to enrich uranium depends in a complex way on the

concentrations of 235U in the feed, product and trails. In general, a higher

tails assay requires fewer SW'S and more feed. If resource utilization were

not important a given quantity of enriched uranium could be obtained with less

work by raising the tails assay.

The quantity of natural uranium feed and the separative work required for a

product enriched to a specified level with 0.2% or 0.5% Tails assay is shown

in Figure **9**. For example, with 0.2% Tails, 4.3 SWU and 5.5 kg of Feed are required

for 1 kg of 3% product. For one kilogram of 90% ^{235}U, 227 SWU and 176 kg of feed

are required. Less than twice as much separative work is required to produce

weapons grade material as is needed to enrich uranium to 3% from a given amount of

feed.

Enrichment plants are designed for a specific SWU capacity. Other

factors can be adjusted fairly easily however. Increasing the tails

assay and the feed permit one to raise either the output or the enrichment

level. The Chinese may have produced their first uranium bomb by converting

a U.S.S.R. supplied low enrichment plant. The use of partially enriched

uranium as feed would also increase the output of the plant. A given

amount of separative work can, of course, be achieved by a small plant

over a long period of time or by a large plant working for a short period

2.35 kg Natural Uranium (0.7% $^{235}U)$
FEED

Enrichment Plant

1 Separative Work Unit

WASTE
1.35 kg Uranium
Tails (0.2% ^{235}U)

PRODUCT
1.0 kg Enriched
Uranium (1.4% ^{235}U)
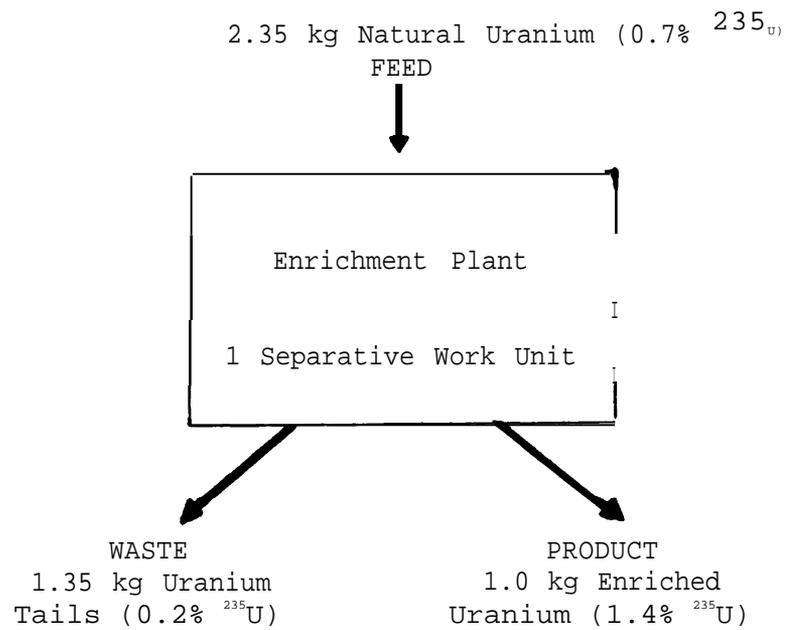
Fig. 8 Example of One Separative Work Unit
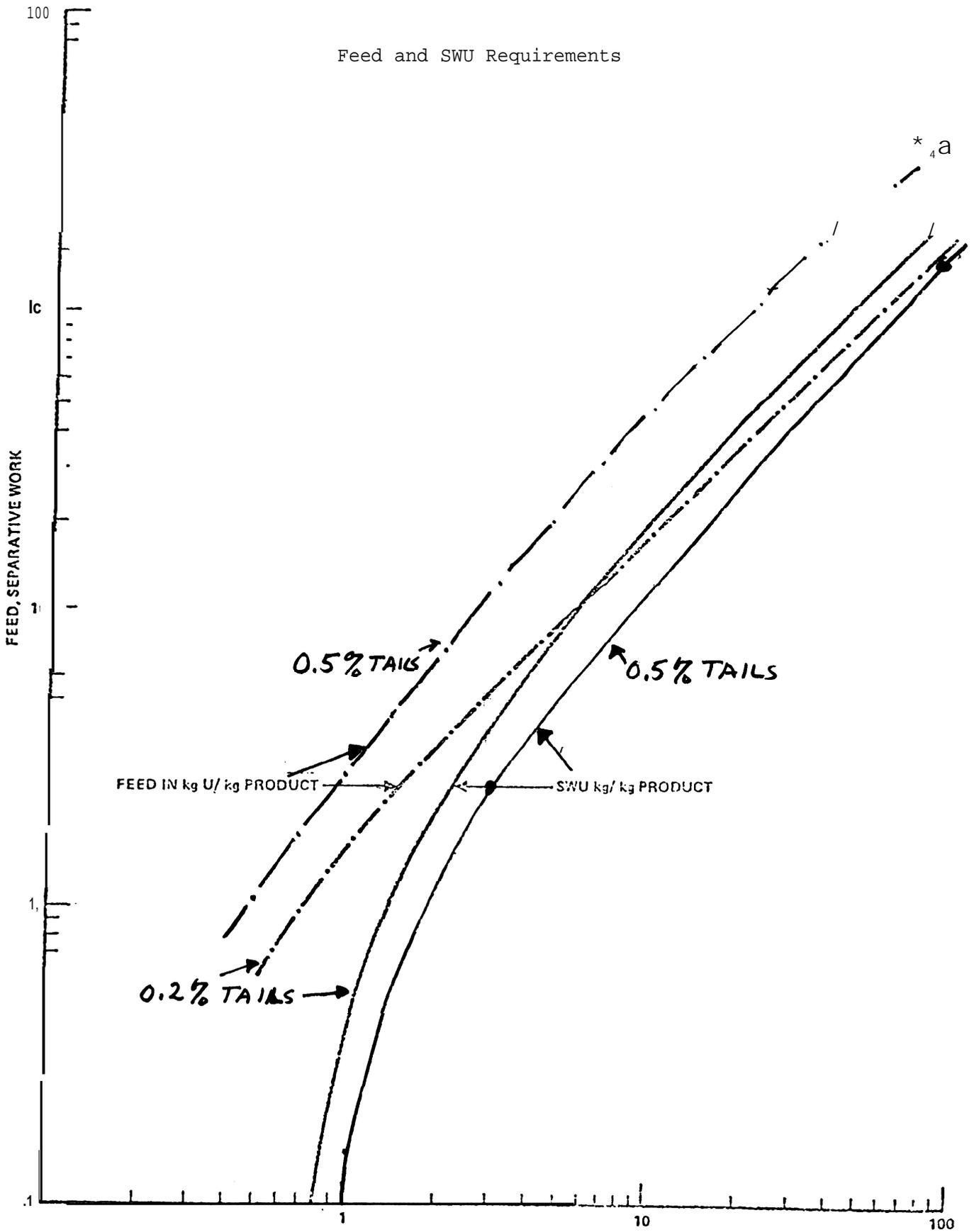
Fig. 9. % ^{235}U in Product

of time. **It is for this reason that plant capability is expressed in separative work per unit time.**

Although many processes have been investigated for potential applications to uranium enrichment, only relatively few are now considered to be serious candidates for practical applications. Thermal diffusion and electromagnetic separation were developed to the pilot plant stage during World War 11 and were found to be very inefficient. Techniques for laser isotope separation and plasma centrifuges are now under development but more work must be done before feasibility can be properly assessed. Isotopic enrichment by ion exchange processes is another candidate for possible future development.

All large scale enrichment facilities currently operating utilize gaseous diffusion. Recently the gas centrifuge has been attracting more attention because of its potentially smaller size and lower power consumption. Several pilot plants have been built, and larger facilities are planned by several nations, including the U.S. (See Table 2.) The jet nozzle is another technology under development but is attractive only where electrical power is abundant and inexpensive. The characteristics of various enrichment techniques will be discussed below.

2.5.1 Gaseous Diffusion

The gaseous diffusion process depends upon the separation effect that arises from the phenomenon of molecular effusion (i.e., the flow of gas through small openings). In a mixture of two gases, the molecules of the lighter gas have a higher velocity at a given temperature and therefore will strike the walls of the vessel more frequently, relative to its concentration. If the walls of the container (the barrier) are porous

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with openings large enough to permit the passage of individual molecules, but sufficiently small so that bulk flow of the gas as a whole is prevented (i.e., with opening diameters approaching the mean free path dimension of the gas), then the lighter molecules will pass through the barrier more readily than heavier ones. and this gas will be enriched with respect to the lighter component of the mixture. In this method, the degree of separation is determined by their relative velocities which depends upon the square root of the ratio of the masses of the isotopes. For UF_6 the maximum separation per stage, that is, the ratio of final to initial concentrations of U-235, is 1.00429. If one-half the input flow passes through the barrier and one-half is recycled to a lower stage, the theoretical separation factor is 1.0030. In practice, the properties of the barrier are not ideal. Back-diffusion through the barrier and some bulk flow through pores reduce the separation.

UF_6 is introduced as a gas and made to flow along the inside of a porous barrier tube containing thousands of submicroscopic openings per square inch. Through molecular effusion, the diffused stream is slightly enriched with respect to ^{235}U, the lighter uranium isotope, and the stream that has not been diffused is depleted. The enriched UF_6 in the outer cylinder is removed for input to the next stage. The process is illustrated in Figure 10.

Because the separation factor (ratio of final to initial ^{235}U content) is . highest at low throughput, it is necessary to use many parallel, connected units, each with the same composition of feed material. **This group of units is called a stage. Because the separation factor in a single stage (1.003) is very small, it is necessary to utilize many stages in series. A series of stages is called a cascade. The large**

**More concentrated mixture
to higher stage**

**Hex gas /
(UF.)_feed,
pumped
in
under
pressure**

uranium-235

uranium-238

uranium-235

Porous nickel barrier

**More dilute mixture recycled to
lower stage to recover
remaining uranium-235**
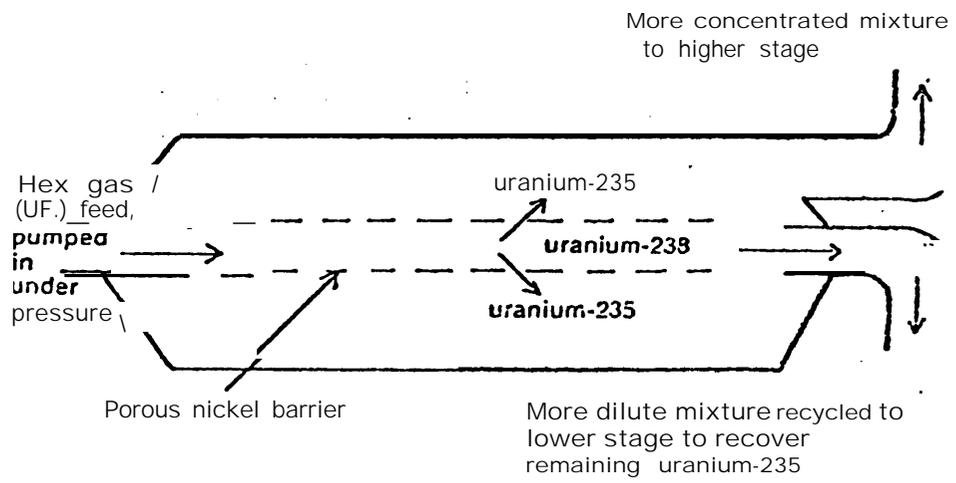
Figure 10. Gas Diffusion Barrier

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number of stages makes the minimum economical size of a gaseous diffusion plant quite large. A large amount of electric power   is also required to pump the UF_6 through the barrier in each stage. A cascade is shown in Figure 11.

The need to use uranium hexafluoride as a working gas has a large impact on the design of the plant. This substance is a gas at a pressure of one atmosphere at 60°C. It reacts strongly with most materials. The system must be kept absolutely air tight so that it does not decompose to form particulate uranium dioxy-difluoride which will block the pores. Since corrosion rates must be low to insure a long life only a limited number of metals such as nickel or aluminum which form stable fluoride layers or fully fluorinated plastics can be used for the construction of the barrier in an element of a gaseous diffusion plant.

Table 1 gives the characteristics of the three operating gaseous diffusion plants in the U.S. A new enrichment facility (now expected to be a centrifuge plant) is projected to have a 9 million SWU/year capacity, operating at .3% tails and if based on gaseous diffusion, would require 2,500 MW of electricity. Capital costs would be on the order $3 billion, or $333/sw (about the same for gaseous diffusion or centrifuge). This could produce 1800 Tonnes of uranium enriched to 3% per year, enough to provide fuel for fifty 1000 megawatt power reactors.

2.5.2 Centrifuge

A centrifuge is a means for applying a high artificial gravitational field to separate fluids of different weights which would otherwise remain mixed because of the thermal motions of the molecules. A cylinder filled with uranium hexafluoride turns about its axis at high speed. The centrifugal field establishes a radial pressure gradient which results in an enrichment of the lighter isotope at the center and the heavier isotope at the wall.

LEGEND: ▨ ENRICHED MATERIAL
⊠ DEPLETED MATERIAL
▨ FEEDSTOCK
--- BARRIER

Figure 11. Cascade

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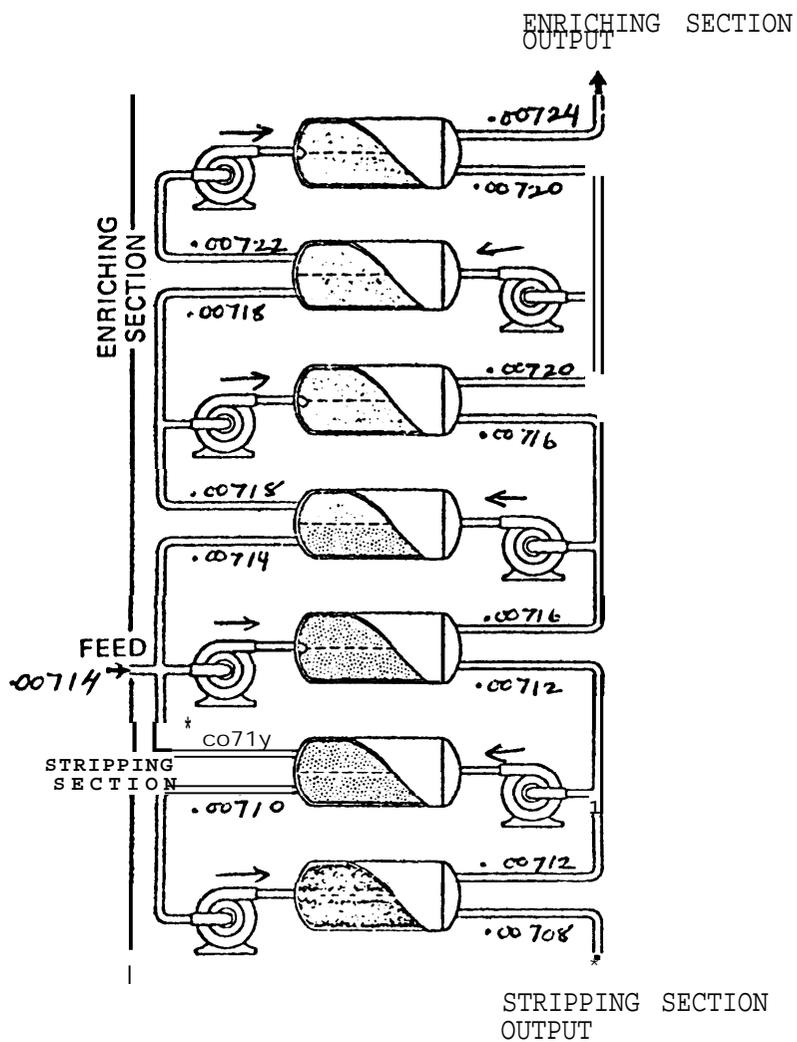
Table 1. Characteristics of existing U.S. enrichment plants

Location	Oak Ridge	Paducah	Portsmouth
Completion Dates	Aug. 1945 - June 1954	Jan. 1953 - Dec. 1954	Nov. 1955 - Feb. 1956
Separative barrier stages	5,104	1,812	4,080
Feed Range (^{235}U)	0.3-1.1	0.3-0.711	0.3-1.1
Enrichment Range (^{235}U)	0.7-4	0.7-1.1	0.74-97.65
Capacity, 1970 (million SWU)	4.730	7.310	5.190
Disturbed area (acres)	640	345	515
Ground coverage (acres)	105	74	93
Electrical Power (MW(e))	1,600	2,550	1,900
Recirculating Water System (gpd)	400,000,000	500,000,000	450,000,000

A schematic diagram of a gas centrifuge suitable for use in an enrichment facility is given in Fig. 12. The rotor might be 40 to 300 cm long with a radius of 6 to 22 cm. A distinguishing feature of this counter-current gas centrifuge is the toroidal internal circulation of gas in the axial direction -- an upward flow in the center of the rotor and downward along the walls. As the gas moves up the rotor core, the ^{238}U diffuses outward in the centrifugal field. The gas arrives at the rotor top as a stream enriched in ^{235}U. Similarly, the peripheral downflow stream arrives at the rotor bottom enriched in ^{238}U.

A counter current flow may be induced either by establishing a small temperature-difference between the ends or by introducing a frictional mechanism (such as the scoops used to withdraw the product end waste). Due to the recirculation of gas within the tube, the separation achieved is greater than that expected for a single element.

An attractive feature of the centrifugal process is that the degree of separation depends upon the difference of masses of the isotopes rather than on their ratio as with gaseous diffusion. For a heavy element such as uranium, the ratio is close to unity. Thus **a** much larger separation factor per stage is possible with the centrifuge method.

The major challenge has been to produce high speed centrifuges suitable for large-scale operations, because the separation factor for a centrifuge varies with the fourth power of the peripheral speed of the rotor. A major research problem has been to find materials for rotors that can withstand such high rates of rotation. Maximum rotor speeds vary from **300** m/see for aluminum alloy or high-strength steel, to a potential of 700 m/see for a carbon fiber/resin rotor. Separation factors of greater than 1.1 per stage are feasible.

speeds vary from 300 m/see for aluminum alloy or high-strength steel, to a potential of 700 m/see for a carbon fiber/resin rotor. Separation factors of greater than 1.1 per stage are feasible.

If the speed is doubled the theoretically predicted separating power will increase by sixteen times. A twenty percent increase in speed, will result in a doubling of its performance.

WASTE LINE

FEED LINE

PRODUCT LINE

VACUUM SYSTEM

UPPER SUSPENSION
SUBASSEMBLY

ROTOR SUBASSEMBLY

CASING SUBASSEMBLY

POST-SUBASSEMBLY

DRIVE SUBASSEMBLY

LOWER SUSPENSION
SUBASSEMBLY
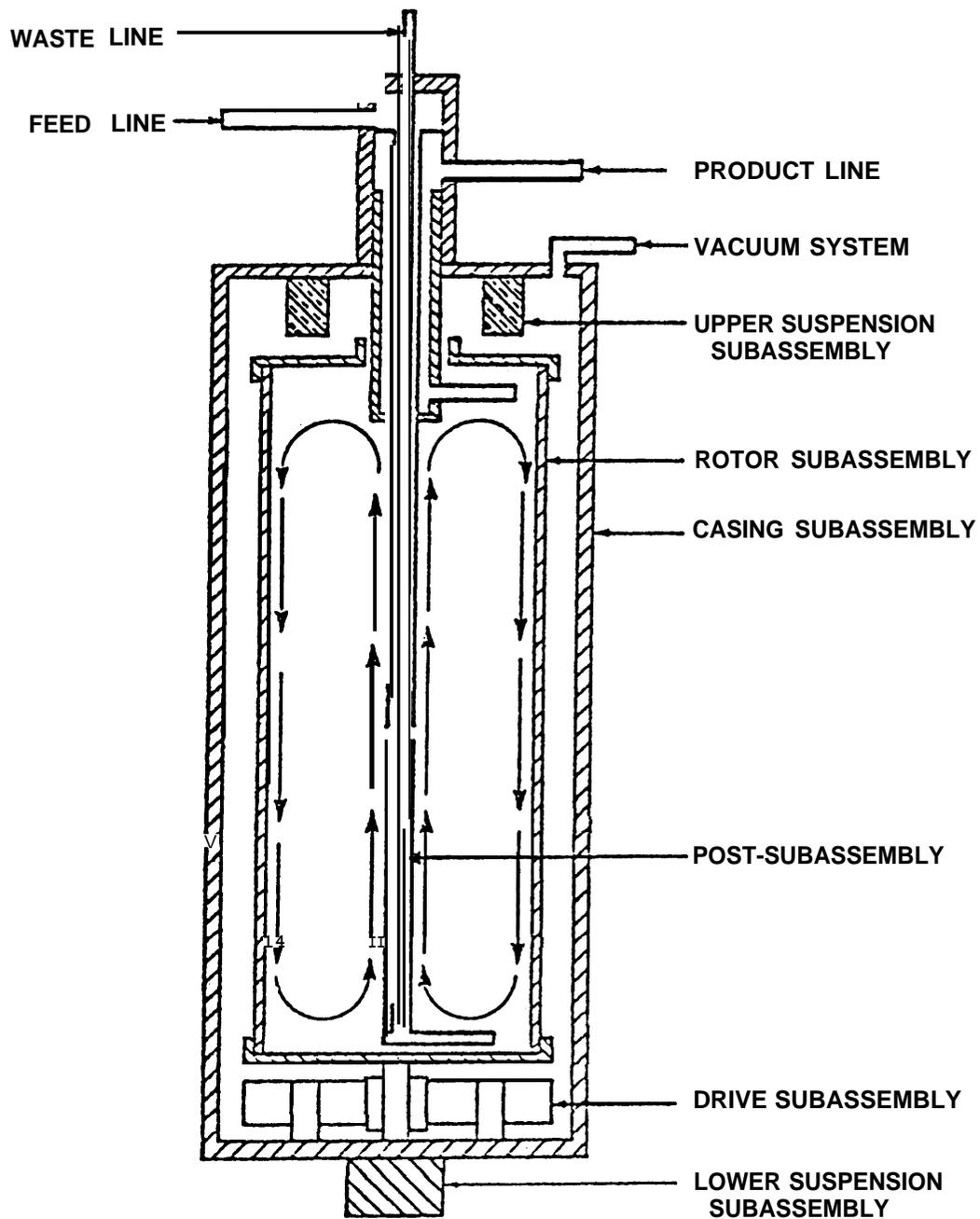
Figure 12.   Schematic of Gas Centrifuge

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The separation achieved also depends on the length to diameter ratio of the rotor. Long rotors can be expected to **give** increased performance. However, design of the unit may have to make provision for it to pass through "critical speeds" (corresponding to renounces of the tube) before it reaches operating speed.

A second difficulty is the fact that each centrifuge can handle only a small feed even though the separation per stage is high. An individual centrifuge can produce 2 to 5 kg of SWU per year. This means that 100,00 machines would be needed for a 300.000 SWU/year plant. Table 2 gives the number of machines per stage for a plant of this size. The estimated fractional cost of the plant is .3 for the machines, **.35** for the plant, .10 for machine replacements, 15 for power consumption and .10 for operation and maintenance. An Anglo-German-Dutch enrichment group, Urenco, has successfully demonstrated the first cascades of two small centrifuge plants each with a planned capacity of about 200,000 kg. Separative work (SWU) per year at Capenhurst, England and Almelo, Holland. No full size production plants have yet been build. However, both the United States and Urenco have announced plans to build them.

The major advantages of centrifuge plants over diffusion plants are that they are expected to cost less to build per SWU, can be much smaller without losing economies of scale and have power requirements which appear to be about one-fifteenth as great. Difficulties can result from the thousands of complex mechanical units operating at very high rpm. Machine failure rates of less than 2% per year have been achieved, however, and it is believed that a plant can operate economically with failure rates as high as 2.5% per year.

2.5.3 <u>Other Enrichment Processes</u>

There are several other techniques which have been used in the past, demonstrated technically or show promise. An aerodynamic process, known as the jet or Becker nozzle process, has been under active development for the

Table 2. Centrifuge machines for 300,000 SWU/year enrichment
          plant to produce 2.8% enriched uranium with .2% tails

<u>Centrifuges per Stage</u>

	2,160	Product
	4,850	
	8,190	
	12,360	
F e e d	17,570	
	15,990	
	14,020	
	11,580	
	8,540	
	4,740	Tails

past decade. It utilizes the pressure gradient developed in a curved expanding

supersonic jet of a mixture of uranium hexafluoride and hydrogen to achieve a

separation of the uranium isotopes. As the expanding jet traverses the curved path

the heavier components tend to diffuse preferentially toward the curved

outer wall. A knife edge placed relatively near the outer wall divides

the jet stream into two fractions, the inner one enriched in ^{235}U and the

outer one enriched in ^{238}U. The two streams are then pumped off separately. The

placement of the knife edge in the jet stream is critical with respect to separation

performance. The diameter of the curved deflecting wall is on the order

of 0.1 mm and the spacing between the knife edge and the outer wall may

be about 10_P m, with a tolerance of $\pm 1_P$ m. The process is illustrated in

Fig. 13.

Because of the higher separation factor a jet nozzle plant will require about

one-third the number of stages in a gaseous diffusion facility which will provide

the same degree of enrichment. At the present time, the specific energy consumption

estimated for the separation nozzle process is larger than that for gaseous diffusion.

However, significant progress has been made. The specific energy consumption projected

for the process has been reduced in recent years and may be further reduced to the

present level of the gaseous diffusion process within the next few years.

A manufacturing process has been developed by a German firm for the

mass production of the separation nozzle slits with the required tolerances,

thereby leading to reduced capital costs. The development group at Karlsruhe

is confident that the process technology will be advanced to the point where

its unit cost for separative work will be equal to or less than that for

gaseous diffusion by 1977. A joint development program has been arranged with

Brazil, which is scheduled to lead to a full scale plant. This plant will

take advantage of the otherwise unuseable cheap hydroelectricity in a remote

region of the Amazon. South Africa has developed a similar process-and is now

constructing a production plant,

Feed Gas

5 percent UF$_6$
95 percent hydrogen

Light
Fraction
Enriched
in U^{235} and
H$_2$

Heavy Fraction
Depleted
in U^{235}
and H$_2$

Cross-section
of the separation
nozzle system
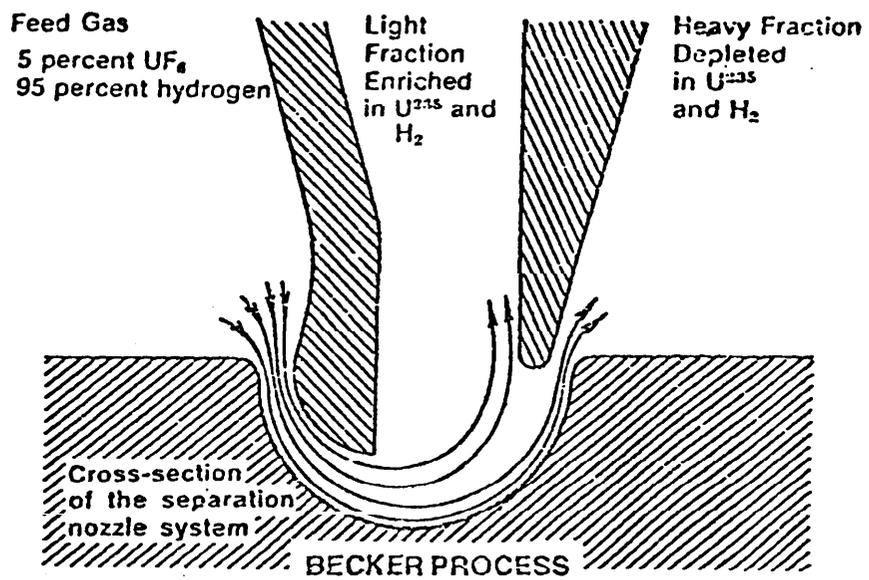
BECKER PROCESS

Figure 13. Becker Nozzle

The calutron process is one of the older separation methods. A compound of uranium is vaporized in an electrically heated container. The vapor passes through slots into an arc chamber where it is ionized by an electron beam. The ionized uranium is accelerated by electrodes in another slot. The high velocity stream then enters a vacuum tank where it is forced into a 180° curve by a large electromagnet. The ^{235}U and 238U follow slightly different paths because of the different centrifugal forces and are collected separately in properly spaced graphite receivers. The graphite reacts with the uranium ions to form uranium carbide (UC). The receivers are processed chemically to obtain the separated isotopes. The calutron, although considered the best of the electromagnetic processes, was not economically competitive with the gaseous diffusion process for large-scale enrichment of ^{235}U. A pilot plant was built during World War II and found to be very inefficient , although it was adequate to produce much of the enriched uranium used in the Hiroshima bomb.

During the past thirty years there have been many advances in technology which are relevant to the development of electromagnetic separation of uranium on a large scale. These include magnets pumps, controls and apparatus for carrying out the related chemical operations.

A significant contribution may be the techniques and hardware which have been developed for ion propulsion of spacecraft. It would be necessary, of course, to modify the systems to provide very intense focused beams of singly charged uranium ions instead of broad diffuse beams of lighter elements such as cesium. Some progress has been made in the development of electrohydrodynamic sources in which ions are extracted directly from the surface of a liquid metal. A reduced accelerating potential would permit the use of lower intensity magnetic fields of limited size.

If the many scientific and engineering problems can be solved, it seems

possible that an electromagnetic isotope separator based on this technology can efficiently produce enriched uranium. Because individual units are small and are able to effect a rather high degree of separation of isotopes this process may be suitable for the production weapons grade uranium.

Within the past several years, two additional concepts for isotope separation have shown considerable promise, the plasma centrifuge and laser isotope separation (LIS). The former is similar to the gas centrifuge. As the name implies, the feed material is converted to a plasma, and centrifugal action is achieved electromagnetically. Theoretically, much greater rotational speeds can be achieved in the plasma centrifuge than in the gas centrifuge because no rotating parts are involved. Another advantage is that solids may be used as feed materials to the plasma. This concept is in the early stage of its development and no published experimental evaluation of its feasibility is available.

The feasibility of LIS, on the other hand, has been demonstrated on a microscale and it has been stated that a pilot plant could be built within five years. This process differs completely in principle from the physical separation mechanisms of the other methods. In this case, separation depends upon the ability to activate, in a specific manner, one of the isotopic species to be separated. A beam of uranium atoms (another LIS process uses uranium hexafluoride molecules) is generated in an oven, collimated and then directed through an evacuated region. In this region, two photon beams are applied; one laser beam selectively excites one of the uranium isotopes, while the other laser beam ionizes the previously excited uranium isotope. The ionized isotope is then removed from the atomic beam by an electric or magnetic field and collected on a plate. The process is still in the laboratory stage, where only minor quantities of uranium have been enriched. The ultimate industrial feasibility and economic

practicality of this technology has not yet been fully defined and demonstrated.

Laseer separation plants of commercial size would require individul lasers with at

least 1 to 10 kw average power, a level significantly beyond the present state of the

aret. One advantage that lasers have over most other enrichment methods is that

extremely high levels of enrichment can be achieved in a single pass. Separation

factors of nearly 100 may be feasible. Numerous material problems must be solved

before this method can be applied on a large scale basis.

Some of the other processes which have been or are being studied are

phase equilibrium processes -- such as gas-liquid chemical exchange; exchange

chromatography or ion-exchange; diffusion processes -- such as thermal

diffusion or sweep diffuson; aerodynamic Processes'‾such as Fenn-shock

process; molecular flow processes, and nuclear spin processes. None now appear

likely to become economically competitive with either the gaseous diffusion

or gas centrifuge processes in the near future.

2.5.4 Uranium Recycle

The Uranium spent fuel from an LWR contains about 0.9% ^{235}U. If it is to be

recycled, it must be reenriched or blended with more uranium of much higher enrich-

ment. This recycled uranium will contain traces of various radioactive fission

products, actinides and many uranium isotopes. Facilities for the reenrichment of

recycled uranium may require special traps such as cobaltous fluoride to remove

these contaminants. Other uranium isotopes, such as ^{232}U and ^{236}U will "contaminate"

both the product and the tails. "The ^{232}U may present a radiation hazard. Both

^{232}U and 236U will reduce the worth of the enriched material because they absorb

neutrons in the reactor. It is expected that an enrichment facility will be ded-

icated to the reenrichment of recycled uranium if reprocessing is carried out.

2.6   FUEL FABRICATION

Depending upon the specific reactor type, the fabrication of many types of fuel elements is required. Light water reactors use slightly enriched uranium; gas cooled reactors require the fabrication of highly enriched fuel, while breeder reactors require cores containing depleted uranium or thorium as fertile material and ^{235}U and plutonium fuels.

2.6.1 <u>Light Water Reactor Fuel</u>

In the U.S. , the existing LWR fuel fabrication industry[*] consists of nine commercial plants, each of which performs part or all of the fuel fabrication operation. These facilities and their locations are listed in Table 3. Three of the facilities produce complete light water reactor fuel assemblies using enriched uranium hexafluoride (UF_6) as the feed material, while two other plants start with uranium dioxide (UO_2) powder or UO_2 pellets to produce fuel assemblies. The four remaining facilities produce only UO_2 powder or pellets from enriched UF_6 as feed for fuel assembly plants. Current capacity of the industry is about 3000 metric tons of uranium as fuel assemblies per year.

The dominant process used by the commercial facilities for production of UO_2 fuel for an LWR reactor is basically a three-phase operation:

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[*] Final Environmental Statement, LWBR Program, ERDA 1541, June 1976

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Table 3   LWR Fuel Fabrication Plants

Licensee	Plant Location	Plant Feed Material	Plant Product
Babcock & Wilcox	Lynchburg, Va.	UO_2 Pellets	Fuel Assemblies
Comb us t ion Engineering	Hematite, Mo.	UF_6	UO_2 Powder or Pellets
Combustion Engineering	Windsor, Corm.	UO_2 Powder	Fuel Assemblies
General Electric	Wilmington, N.C.	UF_6	Fuel Assemblies
Exxon Nuclear	Richland,	UF_6	Fuel Assemblies
Kerr-McGee	Crescent, Okla.	UF_6	UO_2 Powder or Pellets
Nuclear Fuel Services	Erwin, Term.	UF_6	UO2 Powder or Pellets
B&W Nuclear Materials Division (Formerly NUMEC)	Apollo, S.D.	UF_6	Fuel Assemblies
Westinghouse	Columbia, S.C.	UF_6	Fuel Assemblies

1.   Chemical conversion of feed material to powder,

2.   Mechanical processing of materials into solid fuel
     pellets, and

3.   Scrap recovery and recycle.

2.6.1.1 Chemical Conversion

Enriched UF_6 is the feed material used in the fabrication of LWR fuels. The enriched UF_6 gas is converted to UO_2 powder before being formed into pellets. The principal method employed to convert UF_6 to UO_2 is the wet process which involves the use of ammonia to form an intermediate ammonium diuranate (ADU slurry) compound prior to processing to UO_2 powder. The ammonium diuranate process shown schematically in Figure 14 involves :

1.   Volatilizing and hydrolysis of the enriched UF_6
     to form uranyl fluoride solution,

2.   Precipitating ammonium diuranate by the addition
     of ammonia,

3.   Dewatering the ammonium diuranate by centrifuging
     or filtering, and

4.   Drying and reducing the ammonium diuranate to UO_2
     powder in a hydrogen atmosphere.

There are two alternative processes used to convert uranium for fuel fabrication. These are the pyrohydrolysis and the Perclene methods.

In the pyrohydrolysis process, a continuous flow of gaseous UF_6 enters into a fluid bed conversion unit where the UF_6 combines with steam to form solid particles of uranyl fluoride. The uranyl fluoride particles then overflow the reaction bed and are collected in hoppers. In a batch-type process, the uranyl fluoride powder is placed in a second fluid bed reactor where it is reduced to UO_2 by the action of a fluidizing gas consisting of hydrogen and steam. The off-
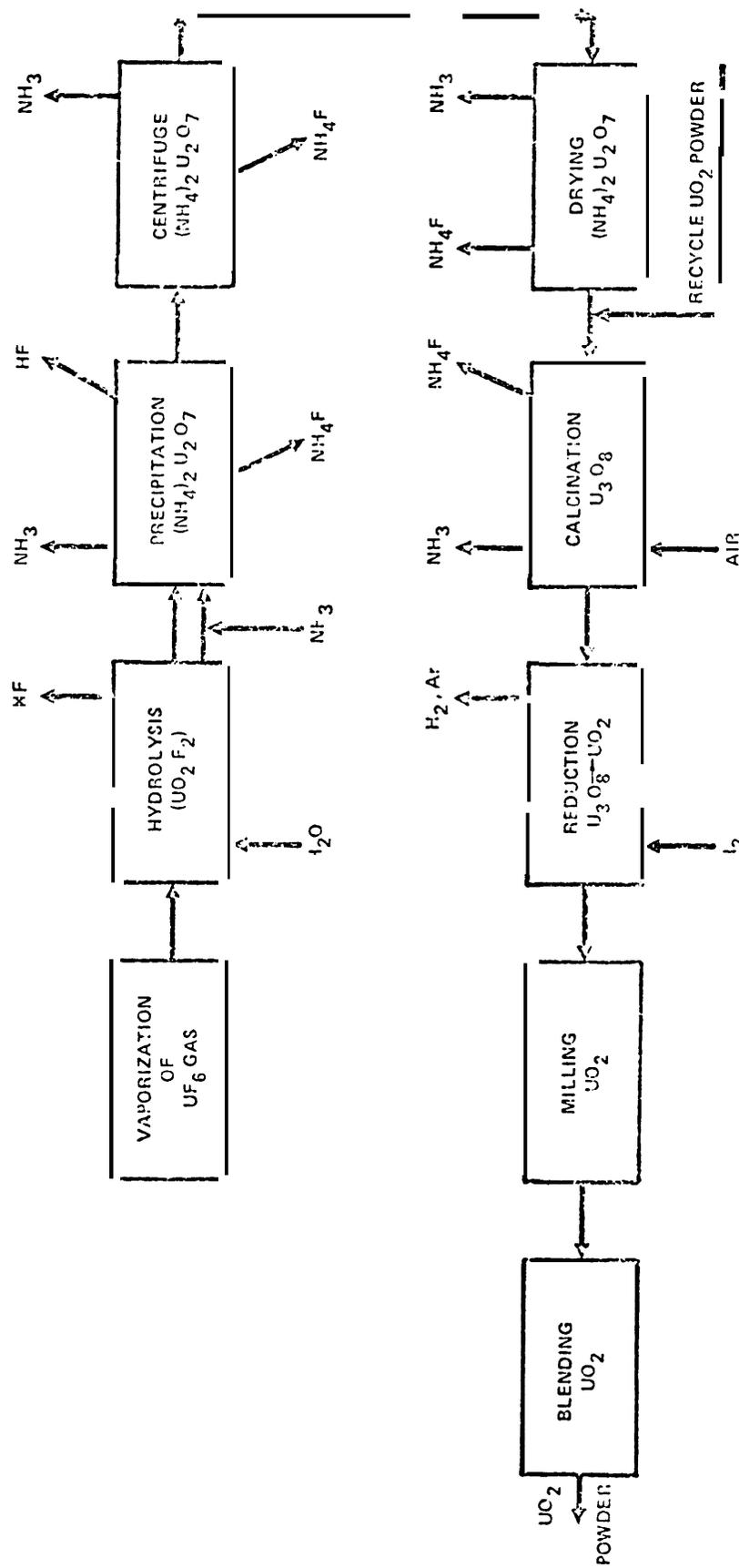
Fig. 14. Schematic Flow Sheet of Uranium Dioxide Conversion Process

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gases, consisting of hydrogen fluoride, hydrogen, uranyl fluoride and UO_2 particles pass through a centrifugal collector and a metallic filter to remove uranium-bearing particles which are then returned to the fluidized beds.

The perclene process involves the reaction of UF_6 with perchlore.ethylene to form tetrachlorodifluorethylene and insoluble uranium tetrafluoride (UF_4). The UF_4 is removed by filtration and pyrohydrolyzed into UO_2. This process requires the recovery and recycle of perchloroethylene and the recovery and disposal of contaminated tetrachlorodifluoro-ethylene gas, in addition to recovery, neutralization and solidification of hydrogen fluoride.

2.6.1.2 Mechanical Processing

In the mechanical processing of the uranium oxide powders to a specific "fuel form, the principal process steps are shown in Figure 15.

The steps utilized in forming fuel elements from the oxide powders are similar for all heavy metals. These steps are:

1. <u>Powder Prepress</u> - In the powder prepress or slugging operation, the powder is prepressed into short wafers to increase the bulk density of the material and to reduce the amount of entrapped air in the powder.

2. <u>Powder Granulation</u> - The short wafers are conveyed to the granulator where the material is granulated and screened through approximately a 14-mesh screen. The granulation process yields a standard agglo-merate size of material for feed to the pellet press, which is important in obtaining a uniform die cavity fill. The amount of oxide granules in each die fill affects the pellet length and density parameters.

3. <u>Pellet Pressing</u> - The granulated powder is auto-matically fed into the die cavity at the pellet press where pellets of uniform density and size are pressed. A die lubricant, approximately 0.2 weight percent sterotex, is applied to the surface of the die walls and punches during pellet pressing. The sterotex is vaporized from the pellets during the sintering step and is expelled with the furnace off-gas.
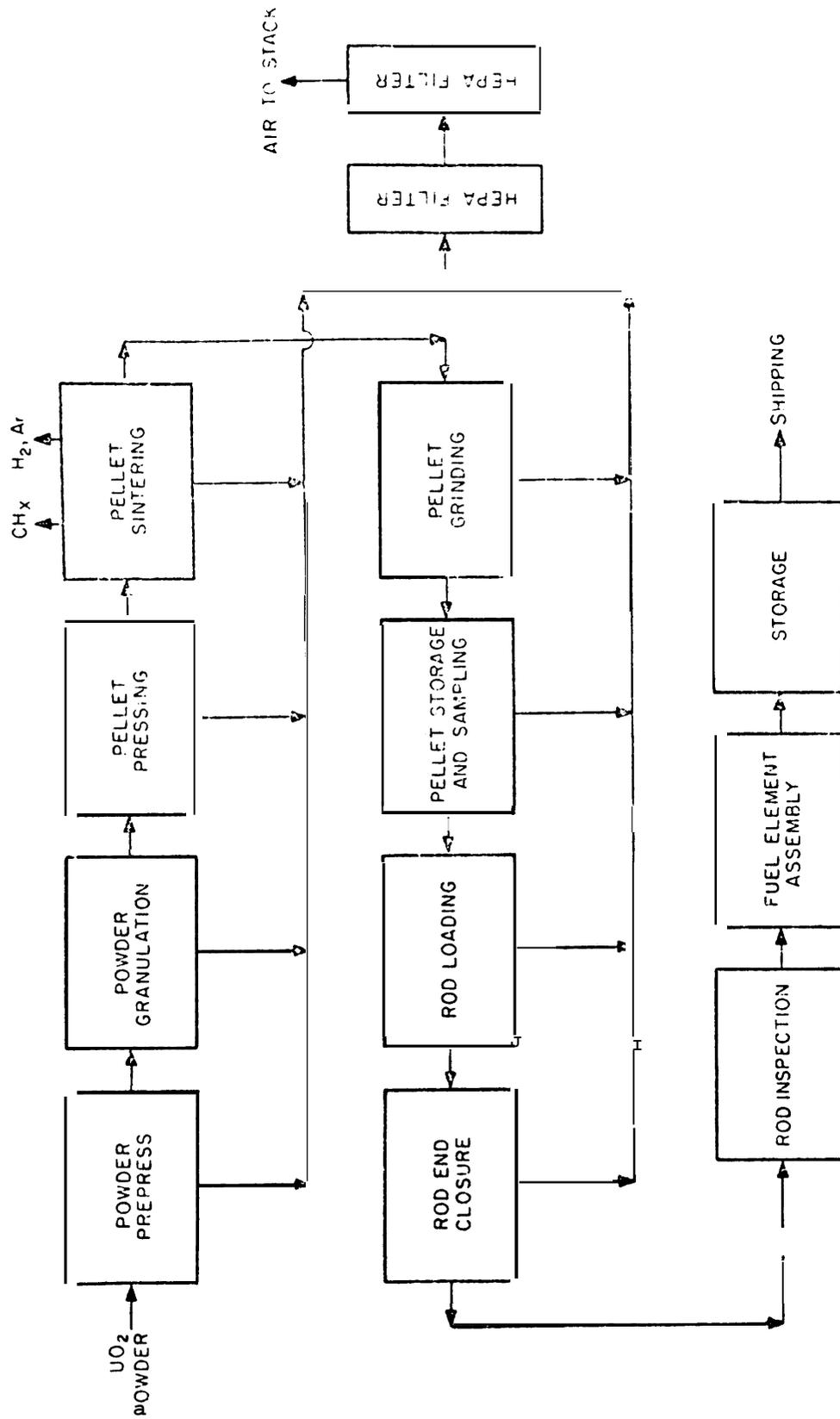
Fig. 5. Schematic Flow Sheet of Uranium Dioxide Fue Fabrication Process

4. Pellet Sintering - The pellets are transferred from
    the penet pressing operation to the sintering furnace
    complex in molybdenum trays. The pellets are then
    sintered to the required density at a temperature of
    approximately 1700 C for approximately 12 hours in a
    hydrogen atmosphere. The exit flow of hydrogen from
    the furnace sintering atmosphere is diluted (with
    argon) to less than the explosive concentration prior
    to passage through the HEPA filter system and dis-
    charge at the stack.

5. Pellet Grinding - The sintered pellets are checked
    for correct density and dimensions and then trans-
    ferred to the centerless grinder. The sintered
    pellets are dry-ground to a specified diameter.

6. Pellet Storage and Sampling - The dioxide pellets are
    statistically sampled, analyzed, and inspected to
    assure that pellet specifications have been met prior
    to further Processing.

7. Rod Loading - The fuel rod loading operation includes
    the receipt of sintered pellets, tubes with welded
    bottom plugs, springs, and top plugs. Dioxide pellets
    released by quality control are mechanically pushed
    into the empty tubes. Each fuel rod contains a pellet
    column length and weight which has been recorded and
    is in compliance with the specification requirements.
    A spring is then inserted into each rod and a top
    end plug is pressed into place.

8. Rod End Closure - The end plug welding is performed
    in a welding chamber with an inert helium gas atmos-
    phere. The welding chamber is pressurized with
    helium gas and the rod is seal welded.

9. Rod Inspection - Each fuel rod is subjected to
    various inspections, including helium leak test,
    rod assay, visual, dimensional) fluroscopic~
    cleanliness, and X-ray.

10. Fuel Element Assembly - After final inspection and
    quality control release, the fuel rods are stored
    in critically safe arrays prior to mechanical
    assembly into modules of reactor core.

11. Storage - The fuel assembly modules are inspected
    and held in storage in critically safe arrays until
    shipment.

2.6.1.3 Scrap Recovery and Recycle

Chipped or broken pellets and pellets that do not meet density or dimensional standards are recycled as oxide powder. Some of the material, however, is not suitable for dry scrap recovery and must be recycled through a solvent extraction process.

2.6.2 Highly Enriched Fuels

Many research reactors and High Temperature Gas-cooled Reactors (HTGR) require highly enriched uranium fuel. Research reactor fuel may be of the MTR. plate type, the HFIR plate type, or TRIGA rods. The fabrication process of the uranium oxide fuels is similar to that discussed in 2.6.1. HTGR and TRIGA fuels are unique and have custom fuel manufacturing facilities In handling highly enriched fuels, particular attention must be given to fuel geometry so that all operations are Performed within geometries that do not allow the accumulation of critical masses of material.

The HTGR fuel elements consist of a graphite block which serves as the reactor moderator. Each block is 79.3 cm high with a hexagonal cross section that is 35.9 cm across the flats. The graphite block is drilled lengthwise with two sets of holes: one allows the passage of the helium coolant, while the second accommodates the fuel rods. Fuel rods are formed by molding selected blends of fuel particles with a graphitic pitch; each fuel rod is 5.1 cm in length and has a diameter of 1.58 cm. Fuel particles are the basic material for the rods and elements, and have a core of either uranium dicarbide (highly enriched in ^{235}U or recycle ^{233}U $_{\circ}ThO_2$ (thorim oxide). Particle diameters are 500 to 800 um.

TRIGA fuel elements are fabricated from an alloy of enriched uranium and zirconium. The alloy is produced by simultaneous vacuum arc melting of small pieces of uranium and zirconium into an ingot about 5 cm in diameter and 50 cm long. This ingot is jacketed to prevent oxidation during further processing, forged and rolled into a thin strip. The jacket is removed from the strip, the surface cleaned by pickling and the strip is chopped into small pieces. These pieces are remelted and cast again into an ingot. The double melt technique is necessary to provide the required uranium-zirconium alloy homogeneity.

The remelt ingot is pickled and machined to approximate size. The ingot is then heated to about 900°C in a heat-treating furnace with a hydrogen atmosphere to form zirconium hydride. The hydrided ingots are machined to final size and inserted into either zircalloy or stainless steel tubes with one end cap already welded in place. The partial fuel rod assembly is swagged (to improve the mechanical contact between the cladding and the fuel), the assembly is evacuated and the second end cap welded.

2.6.3 Breeder Fuels and Blankets

The fabrication of depleted uranium oxide elements for breeder blankets follows essentially the same steps as discussed in 2.6.1.* Thorium fuels are required for thermal breeder blankets.

Processes similar to those used for uranium dioxide fabrication are used to produce ThO_2. For powder conversion, the feed material most commonly in use is in the form of nitrate crystals. The oxalate process used in the conversion of throium nitrate crystals to ThO_2 powder, shown in Figure 16 involves:

1. Dissolution of thorium nitrate crystals,

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*Environmental Statement - LMFBR Program, WASH 1535, December 1974.

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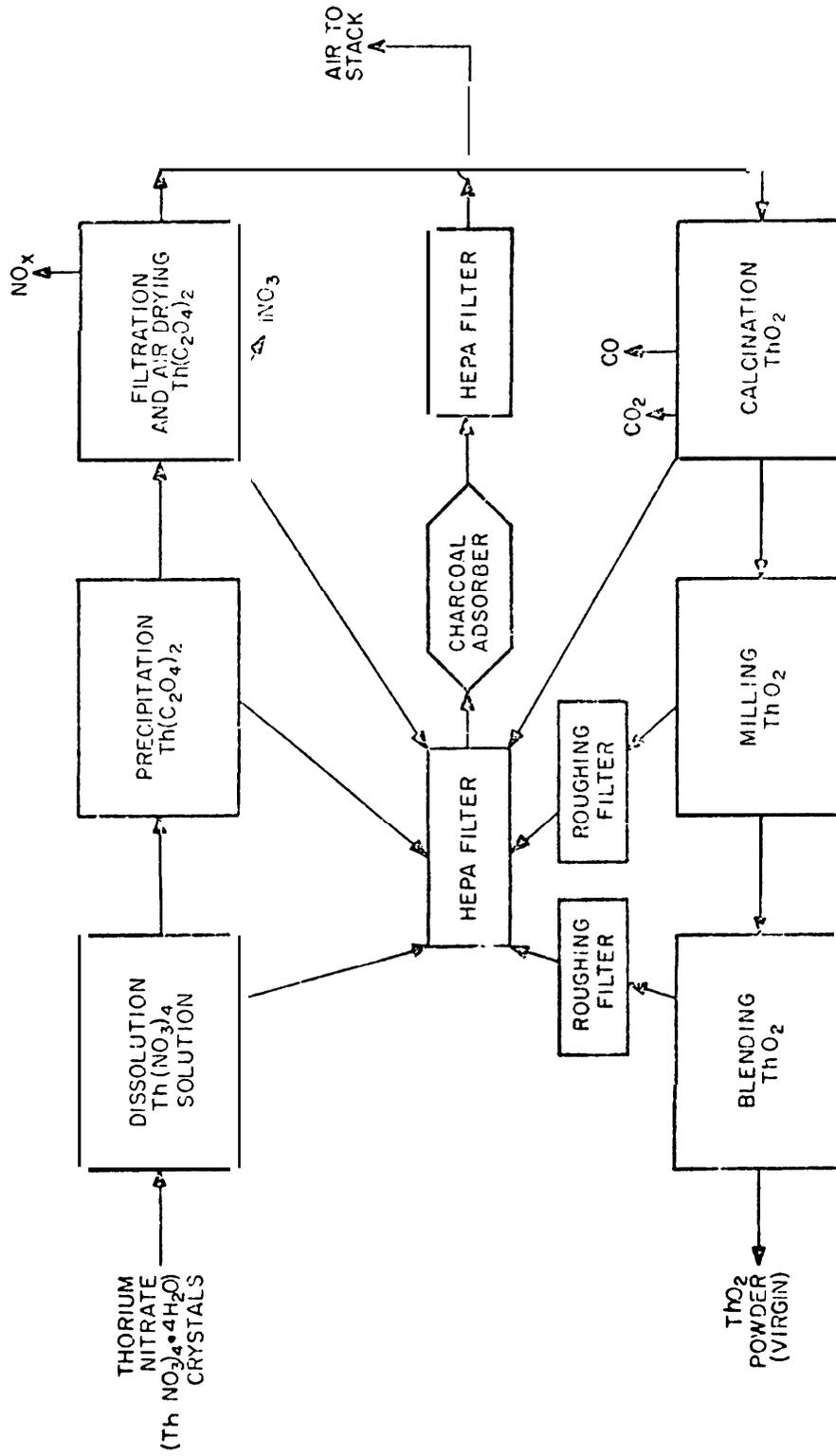
Figure 16. Schematic Flow Sheet of ThO$_2$ - Conversion Process

2. Precipitation of thorium with the addition of oxalic acid,

3. Filtration of the thorium-oxalate slurry,

4. Drying of the moist filter cake, and

5. Calcination of thorium-oxalate to ThO_2 powder.

The mechanical processing of ThO_2 powder to a specific form is identical to that used in the fabrication of UO_2 fuels. The recovery of scrap is similar, with somewhat different chemicals used when solvent extraction is needed.

The fabrication of ^{233}U fuel from a thorium blanket requires special consideration due to the ^{232}U contamination. Only a few hundred parts per million of ^{232}U is a sufficient quantity to prevent contact fabrication and handling techniques. Thus, a ^{233}U fabrication facility must be designed for remote operation and maintenance. The processes are the same as those shown in Figure 17.

Plutonium fuel is normally fabricated as an oxide mixed with uranium. Because of the toxicity, all operations are performed in multiple enclosures to prevent releases to the atmosphere. Thus, glove boxes enclose all processes and, if recycled plutonium is fabricated, the glove boxes must be shielded. plutonium dioxide (PuO_2) powder is mechanically blended and milled with the UO_2 powder in the desired ratios. The mixed oxide powder is then pelletized and the remainder of the process is the same as shown in Figure 15 (See section 4.3 for plutonium metal production.)

Some applications may involve metal or carbide fuels. For these fuels the fabrication techniques will be different from those previously discussed, with the differences depending upon the specific applications. The fabrication of uranium metal can be by conventional means, including casting, rolling,

extrusion, forging, swaging, drawing, and machining. Hot-rolling of the alpha phase is a useful method for forming the metal. Because of the ease with which uranium oxidizes, especially at higher temperatures, it must be protected from air during fabrication either by means of a fused salt or by an inert gas atmosphere. The metal can be machined moderately easily if suitable lubricants and coolants are present to prevent oxidation. Uranium parts can be joined by welding or brazing. Fusion welding is achieved by using a Heliarc torch in an inert atmosphere.

2.7 FUEL STORAGE

It is necessary to store several types of fuel -- fresh fuel that is waiting to be loaded into the reactor, spent- fuel which has been irradiated and reprocessed fuel.

2.7.1 Fresh Fuel

Fresh fuel may be in many different forms, depending upon the type of reactor in which it is to be utilized. This fuel may be stored at the fuel fabrication facility) at the reactor facility, or in both locations. The length of storage time depends on schedules and operating history of the reactor but would probably be at least 30 days at each location. For power reactors, the fuel elements are generally very large (a LWR fuel element may be 4 meters long and weigh 300 to 700 kg). Fuel elements for research reactors may be less than a meter long and weigh less than 50 kg.

2.7.2 Spent Fuel

Irradiated (spent) fuel is removed from the reactor and stored on site to allow the fission products to decay. This storage time varies considerably and in the near future will depend on the availability of reprocessing facilities or high level waste repositories. Due to the radioactivity and heat

generated, the spent fuel must be handled remotely and shipped in shielded casks or stored in shielded facilities. Storage may occur. at the reactor site, at the reprocessing plant or both. Light water reactors replace about one-third of their fuel each year and must be shut down for up to 30 days to perform refueling. Some reactors, such as the CANDU; can be refueled on line, while other reactors, such as the Molten Salt Breeder, have continuous refueling.

The specific characteristics of spent fuel depend upon the integrated exposure (burnup) frequently quoted in mega-watt days/metric ton (MWD/MT). The isotopic content depends on the neutron energy spectrum and the burnup. See the discussion on reactor types for specific information (Section 3) .

2.7.3 Reprocessed Fuel

Reprocessed fuel elements may yield plutonium, ^{233}U and ^{235}U with enrichments which depend upon the initial loadings. This "raw fuel" is different from the enriched UF_6 leaving an enrichment plant, in that it may be stored as a liquid or as a solid (oxide). Storage requirements are dominated by the necessity to prevent the arrangement of material in a critical geometry, the heat dissipation rate, and the physical security requirements. This fuel may be stored at the reprocessing facility or the fuel fabrication facility for periods up to months, depending upon the overall fuel cycle used.

2.8 REPROCESSING

A spent fuel reprocessing plant is a complex of facilities designed to recover fissionable material and to process radio-active wastes. Reprocessing of spent reactor fuel has had numerous problems in the past. There are, however, several plants which have operated (see Table 4). The

TABLE 4

NUCLEAR FUEL REPROCESSING PLANTS

Country	Type of Fuel	Start of Operation	Feed Capacity (Tonne U/yr)	Pu Product/yr[a] at Capacity (kg)	Comments
Argentina[1]			200 kg/yr	--	
Belgium (Mol)[4]	Metal/LWR	1968(5)	80	516 (1077)[a]	167 tonnes U have been processed
Eurochemic	METR[b]	1966	40	--	Eurochemic is not expected to process any more fuel
France[4]					
Marcoule	Metal	1958	500	2150	French military and civilian reactors
La Hague	LWR	1975/78	400	2580	Will increase production gradually until 1978
Germany[4]					
WAK, Karlsruhe	LWR	Sept 1971	36	232(206	32 tonnes U have been processed
KEWA	LWR	1988/89	1400	9030	
India[1]					
Trombay	HWR	1967	100	230	
Tarapur	HWR & LWR	1979	150	968	Assume all LWR fuel
Italy[4]					
Eurex 1	MTR	1970	5	--	
Japan[2]	LWR	1975	10	64	
Tokai-Mura	LWR & Nat U	1978	200	290	Assume all LWR fuel
Spain[1]					
Moncia	MTR		100 kg/yr	—	Small pilot plant
Taiwan[3]					

TABLE 4 (Cent)

NUCLEAR FUEL REPROCESSING PLANTS

Country	Type of Fuel	Start of Operation	Feed Capacity (Tonne U/yr)	Pu Product/yr [a] at Capacity (kg)	Comments
United Kingdom[4]					
Windscale 1	Metal Nat U	1964	2500	10,750	
Windscale 2	LWR	1970(76)	400	2580 (645)	Shut down 1973 after processing 100 Te will restart 1976 at 200 Te/yr and 1977 400 Te/yr
		1982	400	2580	
Dounreay[1]	Highly Enriched U and Pu		1	----	

(a) For LWR fuel we use an average production rate between LWR and BWR fuel. We assume a mature fuel cycle which produces 6.45 kg of fissile plutonium per tonne of uranium reprocessed. Heavy Water Reactor (HWR) fuel is assumed to have an average recovery of 2.3 kg Pu/TeU. Metal Fueled Reactors (that are used for power production) are assumed to produce 4.3 kg of fissile Pu per tonne of unanium.

(b) MTR - Materials Test Reactor uranium aluminum alloy fuel. Usually enriched to 20% or higher in ^{235}U normally produces very little Pu.

(c) Assumes all 167 tonnes of uranium that have been processed were LWR fuel.

Reprocessing References

1. The Nuclear Industry, 1970, p 264.
2. Nuclear Engineering International, February 1955, page 82, World Digest.
3. Epstein, William, "The Proliferation of Nuclear Weapons", Scientific American, Vol. 232, Number 4, April 1975, p 18.
4. Schuller, Walter, "Reprocessing in Europe", ANS/CNA Joint Topical Meeting on Commercial Nuclear Fuel Technology Today, April 28-30, 1975, Toronto, Canada.
5. Science Vol. 184, No. 4144, p. 1315, June 28, 1974.

plant performs five major operations: (1) the receipt and storage of spent fuel assemblies, (2) the processing of the fuel assemblies to separate the fissionable materials from the other fuel assembly materials, (3) the conversion of the recovered uranium to UF_6 for return to an enrichment facility, (4) the conversion of the recovered plutonium to plutonium dioxide (PuO_2), and (5) the processing of radioactive wastes into an appropriate form for transfer to a waste repository if uranium and plutonium are recycled, special design Considerations must be given to the added neutron fluences, heat loads and criticality issues.

A simplified block flow diagram of the Purex-type reprocessing plant is shown in Figure 17* The significant features of the process are described below:

1. Irradiated fuel elements are received at the reprocessing site in shielded casks via rail or truck. Fuel is removed from the shipping casks and stored under water until it is ready to be processed. The irradiated fuels are cooled for at least 150 days to assure the decay of short half-lived radionuclides.

2. Fuel awaiting processing is stored in the storage pool where fuel storage canisters limit fuel element placement to an array which is always safe from a criticality standpoint.

3. The uranium spent fuel rods are transferred to the Purex separations facility where they are chopped by a shear into short lengths (approximately 1 inch) to expose the core material and then charged directly to a dissolver. A semi-continuous dissolution of the oxide cores is performed to minimize, as well as control, the peaking of off-gas release.

4. A soluble nuclear poison is used in the nitric acid dissolvent to-assure nuclear safety in the dissolver.

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'proposed Final Environmental Statement - LMFBR Program WASH-1535, December 1974.
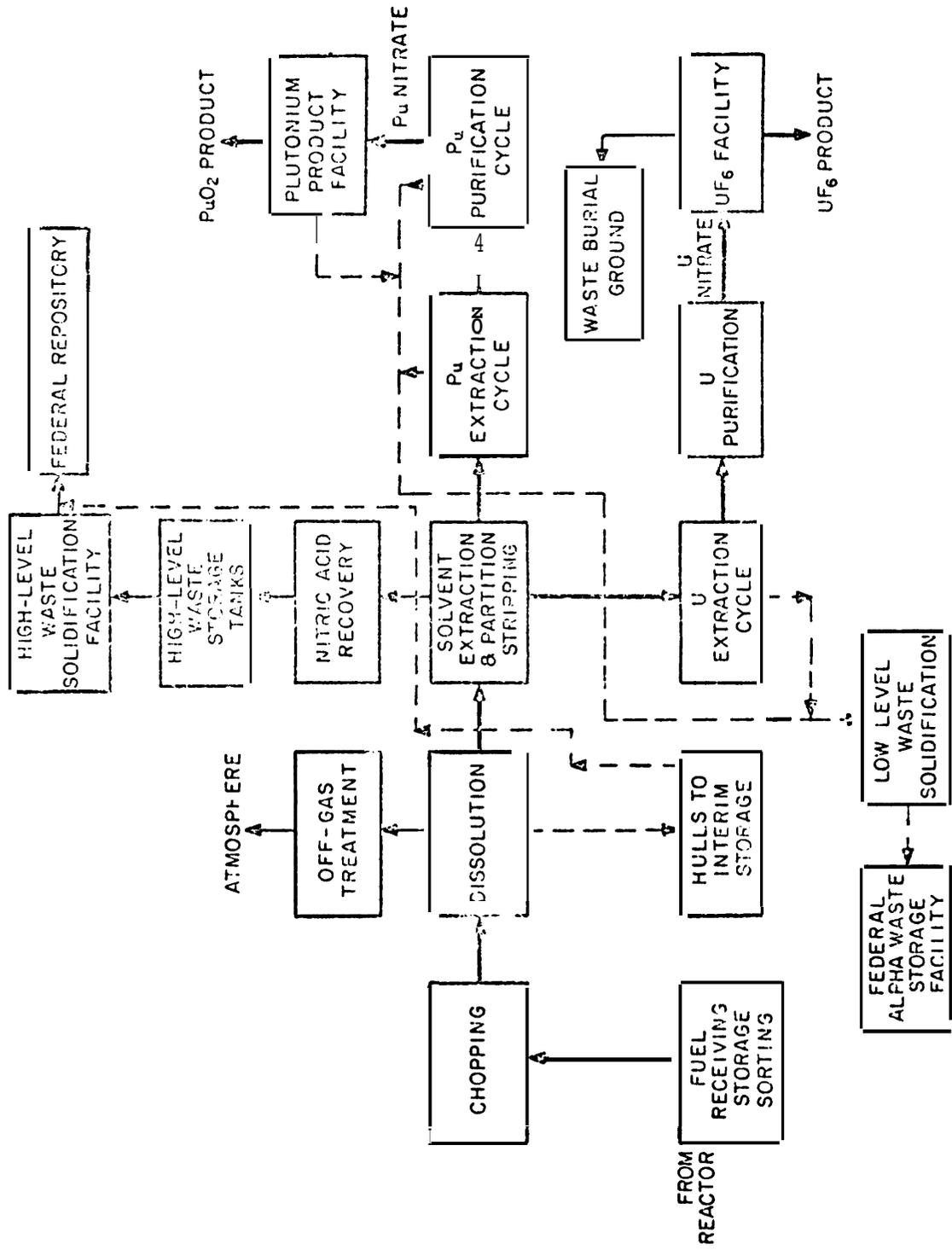
Figure 17. Simplified Flow Diagram of Purex Reprocessing Plan Complex.

5. Centrifugation is used to remove any suspended solids in the extraction feed.

6. A centrifugal contactor is used for the first cycle extraction where uranium and plutonium are separated from bulk fission products.

7. Pulsed columns are used for the partitioning (separation) of plutonium from uranium in the first cycle extraction.

8. Plutonium and uranium are processed simultaneously and separately in continuously operating solvent extraction columns. Uranium solutions are given a final silica gel filtration adsorption for removal of any residual zirconium. Final solutions of these plant products are concentrated prior to storage and/or further processing, such as UF_6 and PuO_2 generation.

9. Solvents used in fuel recycling operations are treated in two parallel solvent treatment systems before reuse.

10. All aqueous raffinates containing small quantities of fissile material (except solvent treatment wastes and the high activity waste stream) are passed through a recovery extraction system prior to concentration and storage. All potential fissile-containing organic raffinates are recycled through the partitioning column prior to routing to solvent treatment.

11. The combination of iodine and $^{14}CO2$ scrubbers and inorganic adsorption beds give multiple assurance of effective iodine and $14CO_2$ cleanup of gas discharged to the atmosphere through the stack.

12. Tritium is released as water vapor from an evaporator through the stack. Nitric acid is recovered and reused.

13. The high-level wastes are chemically denitrated to a nitric acid concentration of 1-5 molar prior to interim liquid storage in cooled stainless steel tanks. High-level waste is transferred from the original tanks as the requirements for cooling decrease due to the decay of the heat-producing radionuclides. The storage of acidic high-level liquid wastes is an interim measure to

allow flexibility regarding their ultimate dis-
position. Multiple cooling systems for the tanks
provide back-up cooling in case the primary cooling
system fails.

A separate plant to convert the uranyl nitrate to UF_6
for return "to the enrichment cascade may be included within
the facility. The capacity of the UF_6 facility would be
compatible with the output of the Purex separations facility.
Figure 18 contains a schematic flow diagram of the conversion
process.

The solid waste (the spent fluorinator beds) contain the
bulk of the radioisotopes entering the process, including the
residual fission products and plutonium not removed in the sepa-
rations facility. These are periodically replaced with fresh
inert bed material. The spent material must be monitored for
activity, packaged in suitable containers and transferred
with other solidified high-level waste to a Federal reposi-
tory.

The Purex separations plant also includes a plutonium
product plant to convert recovered plutonium nitrate to
plutonium oxide powder and to provide storage for the pro-
duct. A chemical process, the oxalate process, may be used
for this purpose. Figure 19 shows a block flow diagram for
the principal steps involved in the oxalate process to pro-
duce plutonium oxide powder from the plutonium nitrate solu-
tion. An alternative process, Coprecipitation, involving
the introduction of uranyl nitrate into the plutonium nitrate
stream, results directly in a mixed oxide. Figure 20 con-
tains a schematic for this process.

The processing of ThO_2 fuels containing uranium may
utilize the Acid Thorex process (see Figure21).

Feed solution for these processes will be formed by
reacting chopped thoria-based element material with a solution

```
                    ┌──────────┐
                    │ STORAGE  │
                    └──────────┘
                         │
                         ▽
                    ┌──────────────┐
                    │ CONCENTRATOR │────▶ TO ACID RECOVERY
                    └──────────────┘
                         │
                         │
STEAM ──────┐            ▽
            ┌──────────────┐
            │  CALCINER    │──── TO ACID RECOVERY
            └──────────────┘
HEAT  ──────┘    │
                 │ UO3
H₂ + N₂ ────┐    ▽                                    TO ATMOSPHERE*
            ┌──────────┐    ┌──────────┐    ┌────────┐
            │ REDUCTOR │──▶ │ SCRUBBER │──▶ │ FILTER │──┐
            └──────────┘    └──────────┘    └────────┘  ▲
                 │ UO2
                 │ UO₂
HF ─────────┐    ▽                                      TO ATMOSPHERE *
            ┌──────────────────┐  ┌──────────┐  ┌────────┐
            │ HYDROFLUORINATOR │─▶│ SCRUBBER │─▶│ FILTER │──┐
            └──────────────────┘  └──────────┘  └────────┘  ▲
                 │ UF4
F₂ ─────────┐    ▽
            ┌──────────────┐
            │ FLUORINATOR  │───▶ SOLIDS TO BURIAL
            └──────────────┘
                 │ UF6
```

*(VIA REPR'G PLANT STACK)

TO ATMOSPHERE*

```
            ┌─────────────┐   ┌──────────┐   ┌────────┐
            │ COLD TRAPS  │──▶│ SCRUBBER │──▶│ FILTER │──┐
            └─────────────┘   └──────────┘   └────────┘  4
                 │
                 ▽ UF₆ PRODUCT
```
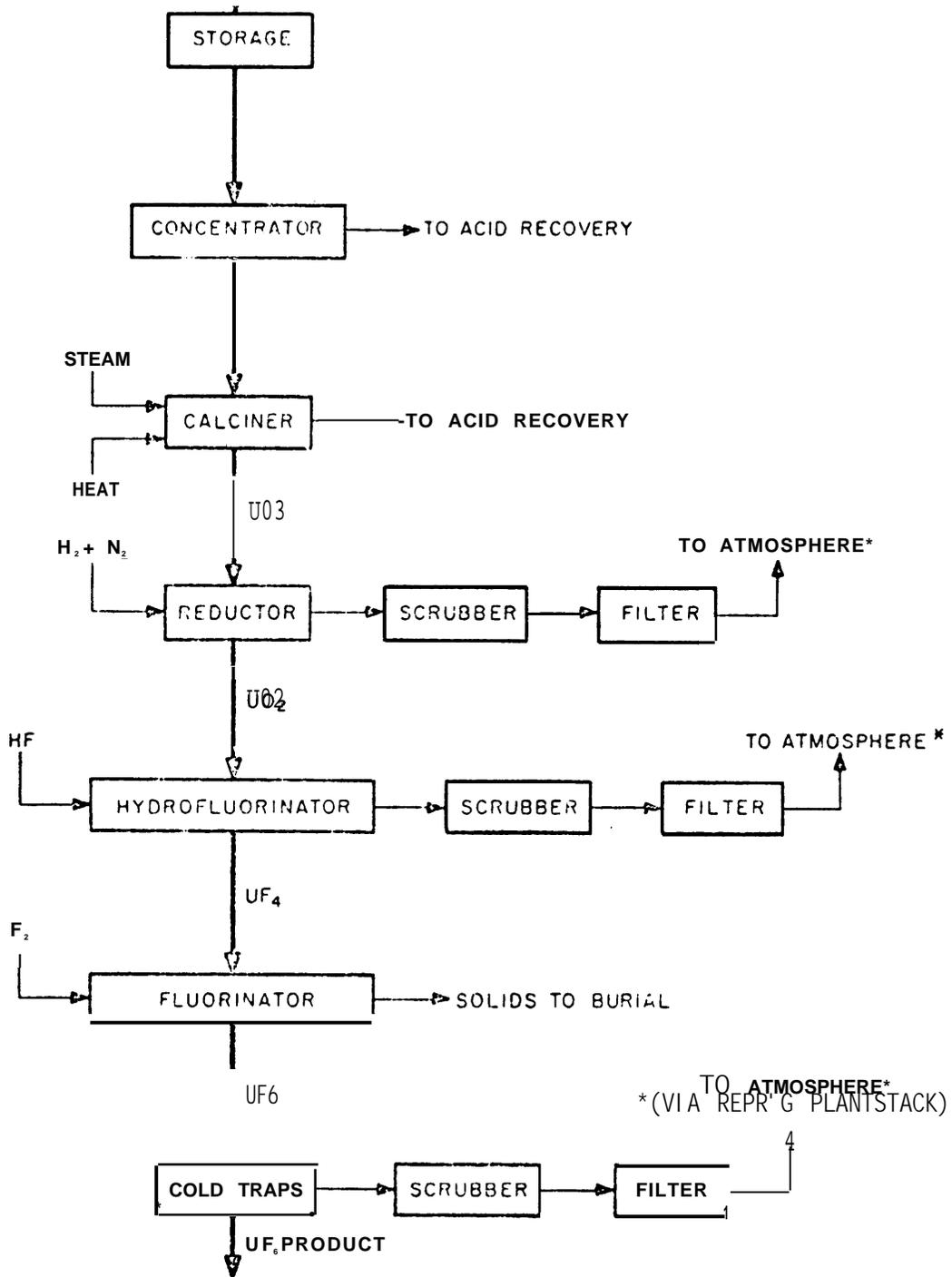
Figure 18. UF_6 Conversion   Plant

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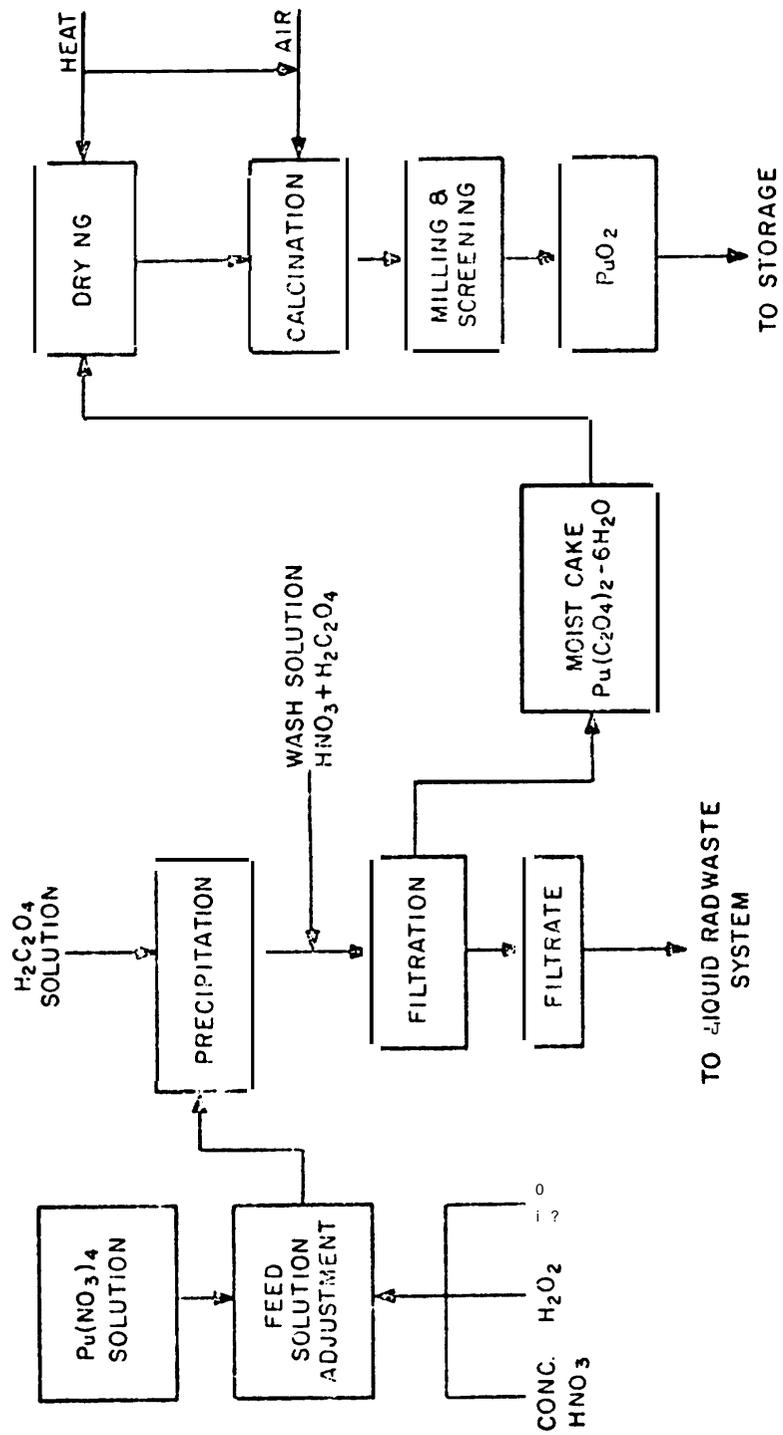
Figure 19. Oxalate Conversion Flow Sheet for PuO_2.

Figure 20. Coprecipitation Conversion Flowsheet for Fixed $(U, Pu)O_2$
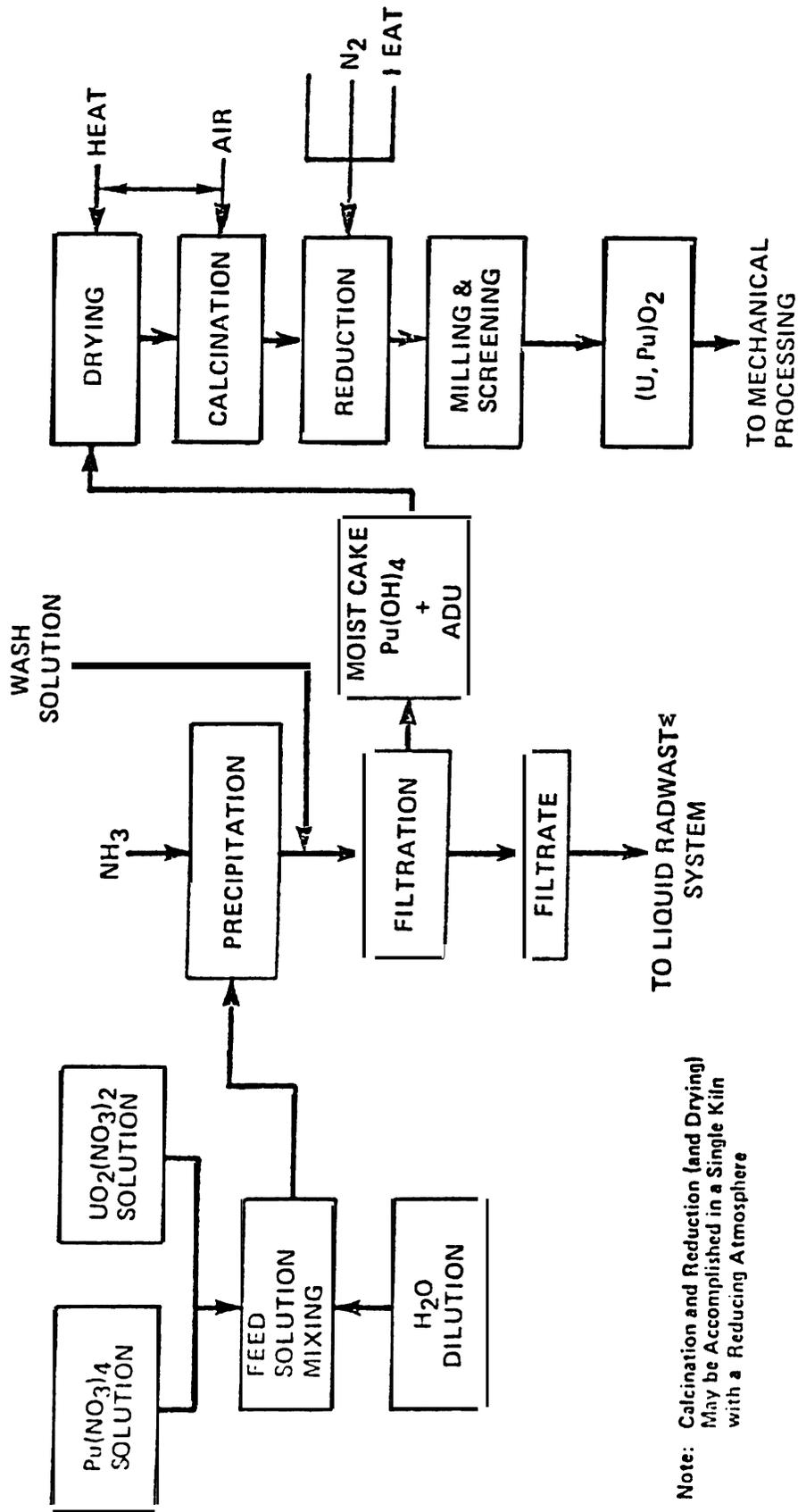
Note: Calcination and Reduction (and Drying)
May be Accomplished in a Single Kiln
with a Reducing Atmosphere

53
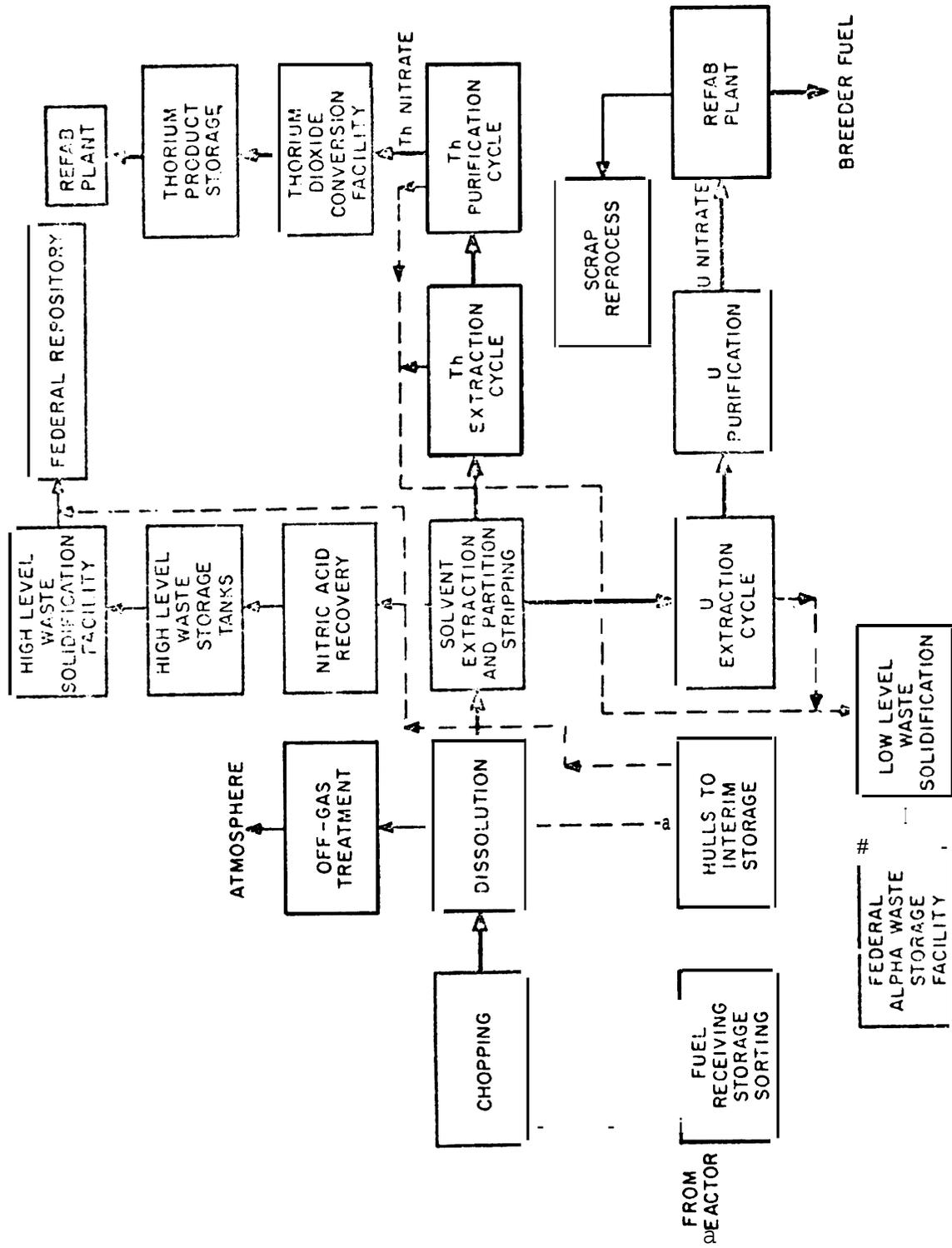
Figure 21. Simplified Block Flow Diagram of the Acid Thorex/Modified Acid Thorex Reprocessing Facility.

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containing nitric acid, hydrofluoric acid, aluminum nitrate
and a neutron absorbing material, such as boron or cadmium.
The resultant solution can be chemically adjusted to solvent
extraction flowsheet specifications and transferred into a
feed tank for the first cycle solvent extraction.

The first extraction cycle serves to separate the
uranium and thorium from the bulk of the fission products in
the aqueous feed solution. A simplified block flow diagram
of feed preparation and first cycle extraction is shown in
Figure 22. In the extraction-scrub column, uranium and
thorium will be extracted into the organic solvent and
scrubbed with nitric acid to remove fission products.

In the stripping column, the organic solvent phase con-
taining thorium and uranium will be stripped from the solvent
using dilute nitric acid. In the solvent scrub column, the
aqueous uranium-thorium solution will be contacted with kero-
sene, and concentrated, by evaporation, to about 1.5 molar
thorium.

The uranium-thorium solution from the first extraction
cycle will be fed into an extraction-scrub column where the
thorium and uranium will be extracted into the solvent and
transferred to the partitioning column. In the partitioning
column, the thorium will be selectively stripped from the
solvent with dilute nitric acid. The thorium solution will
then contact fresh solvent to re-extract any remaining
uranium. If the uranium content of the thorium solution is
sufficiently low, it will be concentrated by evaporation and
transferred to the ThO_2 conversion facility, where the nitrate
solution will be precipitated using oxalic acid, air dried,
and then calcined to the oxide prior to storage.

A typical reprocessing plant will process 1,500 MT/year
of fuel, with capital costs for the reprocessing plant, waste
solidification and PuO_2 conversion of \$1.5 billion. The

FROM FIRST EXTRACTION CYCLE

Extraction-Scrub Column → DILUTE FISSION PRODUCT WASTE (HNO₃, SOLVENT)
→ Partitioning Column → THORIUM SOLUTION (HNO₃, SOLVENT)
→ Stripping Column → SOLVENT TO CLEANUP (HNO₃)
→ Solvent-Scrub Column → KEROSENE TO DISPOSAL (KEROSENE)
→ Evaporation → H₂O TO WASTE → URANIUM PRODUCT

FROM HEAD END

Feed Adjustment → HNO₃ (H₂O)
→ Extraction-Scrub Column → FISSION PRODUCT WASTE (HNO₃, SOLVENT)
→ Stripping Column → SOLVENT TO CLEANUP (HNO₃)
→ Solvent Scrub Column → KEROSENE TO DISPOSAL (KEROSENE)
→ Evaporation → H₂O TO WASTE → TO SECOND EXTRACTION CYCLE

Figure 22. Block Flow Diagrams of Feed Preparation and First Cycle Extraction and Second Cycle Extraction-Acid Thorex Process

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equipment required is typical of the chemical industry and the processes can be scaled down to very low throughput. Remote handling and maintenance are required.

Reprocessing of other fuels, such as the HTGR fuels, requires some unique processes* Figure 23 is a simplified block-flow diagram for HTGR fuel reprocessing. The spent fuel elements are mechanically crushed and then burned to remove the fuel element graphite and the pyrolytic carbon coatings from the fuel particles. Leaching permits separation of the fissile particles (those originally containing ^{235}U) and the fertile particles (those originally containing only thorium but now containing thorium and ^{233}U) because the fissile particles have a silicon carbide coating which remains intact during burning and leaching, while the all pyrolytic-carbon coatings on the fertile particles are burned away. Attainment of a perfect separation of the two particle fractions is not vital, but minimizing the loss of ^{233}U is important. The leach solution is treated by solvent extraction to remove fission products and to separate the bred ^{233}U from the thorium.

The silicon-carbide-coated fissile particles are mechanically crushed to expose the fuel and are burned to remove carbon and oxidize the fuel material; the ash is leached to separate the fuel and fission products from the coating hulls. The ^{235}U is then separated from the fission products by solvent extraction.

The acid thorex solvent extraction process is used to decontaminate and purify the ^{233}U and thorium and to separate the ^{233}U from the thorium:

Some fuels, such as those utilized in low power reactors, might consist of aluminum clad uranium metal. This type of fuel will be much easier to reprocess than the zirconium clad

‘death, C. G., and Spaeth, M. E. "Reprocessing Development for HTGR Fuels", Proceedings of Joint Topical Meeting on Commercial Nuclear Fuel Technology Today, ANS & CNA, April 1975.
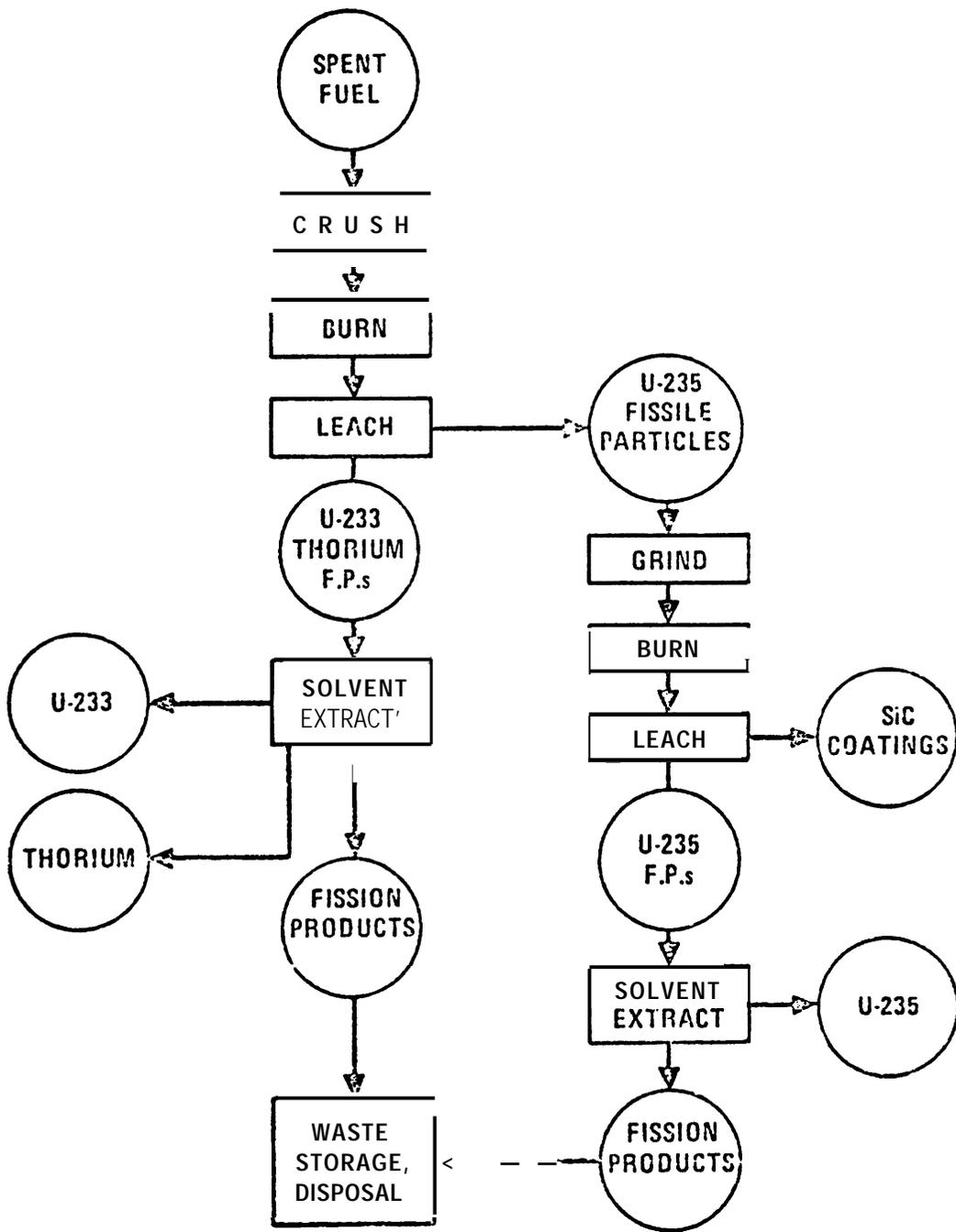
Figure 23.   Block Flow Diagram for HTGR
Reprocessing

oxide fuel of the light water reactors. Sodium hydroxide can be used to dissolve the clad and the uranium metal can be easily dissolved with nitric acid. Typically, the burn-up will be lower for the aluminum clad fuel, so that the fission product activity will be less than that of power reactor fuels.

2.9 WASTE STORAGE

Radioactive wastes are generated at every step in the fuel cycle. The high level wastes containing fission pro-ducts and the actinide wastes cause the greatest concern, however. For the most part, these wastes are contained in the spent fuel elements. The properly packaged spent fuel can be considered waste or it can be reprocessed to recover the fissile and fertile materials. The reprocessing waste must be solidified and properly packaged. This solidified high and intermediate level waste may occupy a volume approx-imately half that of the spent fuel.

The characterization of nuclear waste has received much attention. Low level waste has been stored in shallow land burial sites, and some has been dumped into the ocean. High level wastes have been stored as liquids awaiting deci-sions on ultimate disposal. Programs currently underway will lead to the solidification of the liquid wastes for perma-nent disposal. Ultimately, most radioactive waste will be stored at a Federal repository.

Design for a Federal repository is just beginning. This repository will be capable of handling high level waste, actinides, low level wastes and spent fuel elements. Criteria for packaging these wastes and for repository design are still under development. Although several repositories will be built, the first ones will probably be located in deep stable geologic formations.

A single repository will be capable of annually receiving up to 360m^3 of high level waste, 2100 m^3 (15,000 fuel assemblies) of spent fuel and 6000 m^3 for transuranic waste. Capital investment for a repository will be $300 million, with operating costs of $10 million/year.

2.10 TRANSPORTATION

The fuel cycle facilities which support light water reactors are widespread, and a broad transportation network exists to tie them together. Most shipments of nuclear material occur in routine commerce, using conventional transport equipment; for some nuclear material, however, specially designed containers, packaging, and transport equipment is necessary. Typical shipments for each element of the fuel cycle are presented in this section for U.S. operations.

2.10.1 Mine

Uranium ore is mined and shipped in bulk by open truck to nearby mills. Ore shipments require no specialized containers since the low concentration of naturally occurring radionuclides poses no contamination threat. Approximately 30 MT of ore per vehicle can be transported. The nominal distance in the U.S., from the mine to the mill, is 5 miles, and the average shipment takes about 1/2 hour.

2.10.2 Mill

The uranium concentrates from the milled ore are shipped to the UF$_6$ conversion plant in 55-gallon steel drums. Approximately 15 MT of U$_3$O$_8$ is transported per truck and 38 MT per rail car. The nominal distance in the U.S., from the mill to the conversion plant is 1000 miles. The average shipment takes 5 days by truck and 10 days by rail.

2.10.3 Conversion Plant

Uranium ore concentrates are processed and converted to UF_6, which is packaged in 2.5-, 10-, or 14-ton capacity steel cylinders and shipped by truck or rail to an enrichment plant. Unenriched UF_6 is handled by typical bulk material techniques for industrial chemicals whose primary hazard stems from the chemical rather than the nuclear properties of the material. Most shipments are made by truck with one 14-ton, two 10-ton, or four 2.5-ton cylinders per vehicle. The nominal distance for the shipment is 500 miles and the average shipment time is about 10 hours.

2.10.4 Enrichment Plant

The low-enrichment UF_6 product is shipped to fuel fabrication plants in 30-inch-diameter cylinders which are placed inside protective structural packages designed to protect the enriched UF_6 from impact and fire. Commercial vehicles can accommodate up to five 2.5-ton units at one time. Such shipments are transported an average distance of 750 miles and take about 1-1/2 days. The transportation activity is the same for shipments to a fresh fuel fabrication plant or to a mixed-oxide (MOX) fuel fabrication plant.

2.10.5 Fresh Fuel Fabrication Plant

For shipment to nuclear power plants, unirradiated fuel assemblies are packaged in special containers designed to prevent occurrence of a self-sustaining nuclear reaction in the unlikely event that a sufficient number of assemblies become separated from their shipping package, are arranged in a particular geometric pattern, and flooded with water. A nominal truck shipment contains 32 BWR fuel assemblies or 12 PWR fuel assemblies per truck. The average distance for the shipment is about 1000 miles and takes about 3 days.

2.10.6 Reactor

Irradiated fuel discharged from nuclear power reactors is stored onsite for several months before it is loaded into an irradiated-fuel shipping cask and transported to a fuel reprocessing plant. Casks are available for both truck and rail transport, and most can accept either BWR or PWR fuel assemblies. The cask must provide radiation shielding as well as a means to dissipate the large amount of heat (approximately 10 kw) produced by radioactive decay. The nominal distance from the reactor to the fuel reprocessing plant is assumed to be 1000 miles. The average transit time is assumed to be 5 days by truck and 10 days by rail.

2.10.7 Fuel Reprocessing Plant

Shipments from the fuel reprocessing plants will be of two types. Recovered fissionable materials will be shipped to the fuel fabrication or enrichment plant, and waste products will be shipped to a federal repository.

The plutonium shipped from the fuel reprocessing plant to the MOX fuel fabrication plant will be in the form of PuO_2 or uranium-plutonium dioxide. Present plutonium oxide packages contain a few kilograms of the oxide sealed in metal cans placed in an inner, gasketed steel container which is supported within an outer steel drum of 10 to 110 gallon capacity. The gasketed steel cylinder is supported within the steel drum by thermal and shock insulating material, such as cane fiberboard, vermiculite, or foamed phenolic plastic. Mass limits up to 4.5 kg plutonium per package are defined in present package designs primarily by heat dissipation requirements. Designs have been developed for plutonium product packages, or canisters, which can hold up to 18 kg of plutonium. Four of these cannisters can be packed into a primary pressure vessel, to provide additional containment. These pressure vessels must be transported in

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a special semi-trailer vehicle. This vehicle, termed an Integrated Container Vehicle (ICV), is a cylindrical steel secondary pressure vessel capable of carrying seven of the primary pressure vessels. With this system, the vehicle payload will be about 500 kg of plutonium oxide. The distance for this shipment may be 1000 miles, with a transit time of about 5 days.

The equipment for the shipment of reprocessing plant waste has not yet been developed. It is expected that the shipping containers will resemble the casks currently used for spent fuel transport. The cask will be designed to hold about 60 to 75 cubic feet of solidified waste. The containers for shipping low-level plutonium waste are expected to be 55-gallon steel drums packed in a secondary steel container, resulting in a total waste capacity of 1000 cubic feet. A similar package is expected for the contaminated cladding fuels, with a resulting container payload of about 125 cubic feet. If fuel is not reprocessed, and the waste is in the form of the fuel elements themselves, the spent fuel shipping casks may be used for transport. It is expected that shipments from the fuel reprocessing plant to the federal waste repository will be made by rail. Shipments are expected to take about 10 days, with a travel distance of 1500 miles.

> 10.8 <u>Mixed-Oxide Fuel (MOX) Fabrication Plant</u>

MOX fuel assemblies will be shipped to the reactor in packages similar to those used for shipping fresh fuel assemblies. These packages will be modified to provide neutron shielding. The transport capabilities and requirements are like those identified for fresh fuel elements. More detailed package designs will be developed as the need arises.

2.11 PROSPECTUS OF FUEL CYCLE COMPONENTS

A country may wish to establish commercial nuclear power fuel cycle components within its borders to support its own nuclear reactors and possibly to compete in the world market. An important question is what unit sizes make sense and what" design, construction, and production lead times are required before the facility begins to operate at design capacity.

Table 7 lists characteristics of some U.S. designed fuel cycle facilities along with the estimated capital cost in 1976 dollars. Of course, these cost estimates pertain to U.S. economics and industrial capabilities and may differ sub-stantially with cost estimates for other specific countries. In addition, implicit in these estimates are U.S. environmental considerations on effluent control, and U.S. radiation safety requirements.

The fuel cycle components that are of primary interest in a proliferation assessment are the enrichment facility, the recycle fuel fabrication facility and the reprocessing facility. The parameters for the enrichment facility are for a gaseous diffusion plant, the present primary uranium enrichment method. A severe economic penalty must be paid for smaller sized dif-fusion plants. However, centrifuge enrichment plants can be scaled down more or less linearly with capacity. The total capital costs for a 9,000 MT SWU centrifuge plant are expected to be approximately equal to a simularly sized gaseous diffusion plant .*

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*Environmental Statement "Expansion of Us. Uranium Enrichment Capacity", ERDA-1543, April 1976.

TABLE 5

CHARACTERISTICS OF FUEL CYCLE FACILITIES

Component	Unit Capacity/Yr	Design Construction Lead Time (yr)	Production Lead Time (yr)	Estimated Capital Cost (106 $)	Approximate LWR'S Supplied/Yr
Mining/Milling	1000 MTU$_3$O$_8$	3	2	20	5
Conversion	5000 MTu	3	1.5	35	29
Enrichment	9000 MT SWU	8	1	3,000	77
UO$_2$ Fuel Fabrication	900 MTU	3	0.5	70	31
Recycle Fuel Fabrication	200 MTHM	3	0.5	45	15
Reprocessing	1500 MTHM	8	2	1,500(')	51

(1) Includes Plutonium Nitrate conversion to oxide
            UF6 conversion
            Waste treatment

MT+metric tonnes

MTU + metric tonnes uranium

SWU + separative work units

MTHM + metric tonnes heavy metal

Reprocessing plants also exhibit a non linearity in capital costs.   A recent Savannah River Laboratory report* indicates that a 3000 MTU plant has a capital cost approximately 1.5 times the 1500 MTU plant.   It appears the higher throughput plants are necessary to offset the required costs for remote operations-, high shielding, etc.

An interesting feature of Table 5 is the relatively long lead times required to bring a commercial enrichment plant or commercial reprocessing plant into production.   These sub-stantial lead time periods create pressures for accurate pro-jections for the need of these facilities.   On the other hand these time periods insure that commercial enrichment and re-processing facilities will not proliferate the world in a short time period without an indication that they will be built.

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*"Light Water Reactor Fuel Recycle"" Savannah River Laboratory
Quarterly Report DPST-LWR- 76-1-1 Jan-Mar 1976.

3. REACTORS

As noted in Section 2.1, there are many ways to characterize current reactor systems. With the growing concern over uranium ore supplies, enrichment facilities and plutonium recycle, there are numerous studies which consider combinations of fuel cycles, such as increased use of thorium-^{233}U, the use of lower enrichments for the HTGR, the use of mixed oxides, and other possible alternatives for fast breeders. This section presents the characteristics of generic reactor types and the current or near term fuel cycles. The many alternative fuel options are also considered in this section and in Section 4. In Section 3.8, future systems are considered from the standpoint of their ability to produce fissile material.

For each reactor type described, a detailed flow sheet depicting material flow throughout the fuel cycle is given. All power reactors are normalized to 1000 MWe with a 75% capacity factor assumed. To obtain material flows for another capacity factor , Z, and power level, Y, from the model plant data, X_M (1000, 75%), the following relation should be used:

$$x \ (Z,Y) \ = X_M \ (1000,75\%) \frac{Y}{} \quad \frac{Z}{75}$$

Material flows for research reactors are based on 10 MWth .

3.1 LIGHT WATER REACTORS*

The dominant nuclear power reactor in use today is the light water moderated and cooled reactor (LWR) . There are two basic types -- one, the PWR, in which the coolant is pressurized so that the water does not boil in the reactor, and the other, the BWR, in which the reactor coolant is used to drive a steam turbine directly. Both reactors utilize slightly enriched (2-4%) UO_2 fuel clad in zircalloy. The UO_2 pellets are inserted into the zircalloy tubing, with the small void regions filled with a gas such as helium. The enrichment, fuel management scheme, and burnup are dependent on whether the LWR is a BWR or a PWR. Both types of reactors must be shut down for refueling, which may take as long as 30 days.

3.1.1 Pressurized Water Reactors

PWR vessels are made of steel, are typically 20 meters high, and about 5 meters in diameter, and have walls that are about 20 cm thick. The hemispherical head is bolted into place, but must be removed for refueling. The coolant pressure is about 2250 psi and the outlet temperature is about 320°C. The fuel elements are typically 3.5 to 4 meters long and the core contains around 190 fuel assemblies. Each assembly contains approximately 250 rods and each rod contains about 250 pellets. The assemblies are approximately 20 cm square and 400 cm long. The fuel pellets are generally .8 cm in diameter and 1.3 cm long. The enrichment level of the UO_2, depends on the specific fuel management scheme, but will typically be 2 to 3%. Reload fuel may contain fuel enriched to 3.3% ^{235}U. There may be several core regions of uniform enrichments. Burnable neutron poisons are utilized to provide higher burnups and to balance power density.

Approximately one-third of the fuel elements are replaced each year. Frequently, the refueling schedule is dictated by other plant maintenance requirements and not necessarily by the estimated burnup. Burnup variations of 25% may exist within a single fuel element, and from fuel element to element. A typical burnup appears to be close to $.8E \times 10^4$ MWD/MT, where E is the enrichment of the fuel. This value may be altered by the burnable poisons and by the fuel management scheme. Experience tends to indicate burnups of 20,000 to 25,000 MWD/MT. With this type of reactor, a fuel element suffering clad failure may be removed prior to achieving full burnup by removing the entire assembly.

When the fuel is removed from the reactor, it is stored on site for at least 150 days to permit partial decay of-the fission products. The spent fuel is stored in racks in a water pool at least 5 meters deep to provide the required shielding. When the fuel is shipped, it is transported through a canal and loaded into a shipping cask. A typical reactor facility can store about 3 core loadings in the spent fuel pool. Table 6 gives representative characteristics for a PWR.

3.1.2 Boiling Water Reactor

BWR vessels are about 20 meters high, 6.5 meters in diameter, with wall thicknesses of about 15 cm. The coolant is pressurized to about 1000 psi which permits boiling at around 240°C. The steam-water mixture leaves the core and flows through steam separators before leaving the reactor vessel. The hemispherical head is bolted to the pressure vessel, so both the head and the steam separators must be removed for refueling.

The fuel assemblies are about 450 cm long and are 13.8 cm square. There are over 730 assemblies which may contain

either 49 or 64 fuel rods. Each fuel rod contains about 350 pellets. The enrichment varies by zone and can range from 1.5 to 2.3%. Reload fuel enrichment levels can be as high as 2.8%.

Approximately one-fourth of the fuel assemblies are replaced annually. A typical burnup is about 25,000 MWD/MT for equilibrium conditions and less for the initial loading. As with the PWR, refueling of the BWR may be dictated by other schedules and full burnup may not be achieved. Also, the burnup may vary by 25% within a fuel element and from element to element. Refueling will require about 30 days and the reactor head and steam separators must be removed and the core flooded. The spent fuel is stored on site for at least 150 days. Table 6 gives general characteristics for a BWR.

3.1.3 Material Flow in Light Water Reactors

The material flow (and particularly the discharge) depends upon the burnup level achieved in the fuel. As noted above, burnups of 33,000 MWD/MT are design goals. Table 7 contains the material flows for this burnup under the title PWR 1. Also included, under PWR 3, are data for a 23,000 MWD/MT equilibrium burnup. As experience is gained, the average burnup will probably be somewhere in between these two values.

One current fuel cycle issue is the recycle of plutonium. Table 7 also contains, under PWR 2, the material flows for a typical plutonium recycle case. It should be realized that there are many options and that this may or may not be typical. Figures 24-26 schematically illustrate the material flows for these equilibrium cycles.

Table 8 contains fuel cycle requirements for a BWR with and without plutonium recycle. This information is schematically shown in Figures 27 and 28.

It should be noted that the material flows assume uranium recycle. Depending on future decisions, uranium recycle may not occur and the uranium requirements would be made up for uranium ore.

3.2 HEAVY WATER REACTORS

3.2.1 Introduction

The use of heavy water, D_2O, in reactors has been considered for many years. Presently, only Canada, the West Germans and, more recently, the British have actively pursued the concept for commercial power. Only the Canadians are marketing heavy water reactors at this time.

Heavy water is desirable as a reactor moderator 'due to its satisfactory neutron slowing power and its very small neutron absorption cross section. These factors allow natural uranium to be used as a fuel. However, the need for large quantities of heavy water partially offsets the advantages of not needing enriched uranium. Heavy water costs are around $50 per pound with an enrichment of 99.8% D_2O. Even at the .2% light water impurity level, the light water absorbs as many neutrons as the heavy water. About one tonne of heavy water is needed per MWe of installed capacity.

The greatest advantage of heavy water moderated reactors is their ability to use natural uranium fuels or fuels of near natural enrichment with high neutron economy, long reactivity duration, and, therefore, high fuel utilization. Burnups in the neighborhood of 10,000 Mwd/T of natural uranium fuel are possible in heavy water reactors. Higher burnups have been achieved in other reactors, but only with enriched fuels. Natural uranium can also be used in graphite-moderated reactors, but the burnups there are comparatively low due to physics and metallurgical reasons. There is a great incentive, therefore, to develop heavy water reactors, particularly for those countries with no fuel-enriching facilities. The capture-to-fission reaction rates in heavy water reactors make it possible to more fully utilize the natural uranium.

The strength of the economic incentive to develop heavy water reactors depends upon the different methods used for estimations and projections. In Canada, for example, it is

believed that natural uranium-heavy water moderated reactors can produce power at a lower cost than enriched-fuel reactors. Canada has, therefore, concentrated on developing this type of reactor. These Canadian-type reactors are now being built in such countries as India and Pakistan. In the United States, however, it is believed that cheaper power can be obtained from enriched-fuel, light-water reactors and that, even if heavy water is used, power would be cheaper if enriched fuels are used. The factors that influence this decision are the higher capital costs, the large, expensive heavy water inventory needed, and the availability of large enrichment facilities.

Heavy water reactors can use either metallic or oxide natural fuels. Metallic fuels are more desirable when parasitic neutron absorption is considered, while the oxide fuels are desirable from the standpoint of resistance to radiation damage. The oxide fuel consideration favors the use of UO_2 in power reactors where higher burnups are sought. Cladding materials of low neutron absorption must be used in all natural uranium reactors. Zircaloy, beryllium, and beryllium magnesium alloys are suitable for the higher-temperature natural uranium power reactors. Aluminum may be used in low-temperature reactors.

Heavy water moderated reactors require large moderator-to-fuel volume ratios. Such reactors, therefore, require large-diameter reactor cores. Because of this large diameter, large power reactors operating at high temperatures and pressures require larger, thicker, and costlier pressure vessels than ordinary-water reactors of comparable output. Both pressure vessel and pressure-tube designs have been used. The latter design allows the use of lower-pressure, less costly vessels but adds the expense of constructing a leaktight calandria vessel, free of differential expansion. It also

results in the separation of the coolant and moderator. Operational problems associated with D_2O reactors are the loss, by leakage, of the expensive D_2O and the high activity associated with the decay of tritium formed in the reactor.

Presently, the dominant heavy water reactor concept is the CANDU-PHW (Canada Deuterium Uranium-Pressurized Heavy Water) reactor in which D_2O is utilized as both the moderator and the coolant. Reactors in the range of 500-750 MWe are currently operational. Reactors of 850 MWe capacity are under construction. HWR reactors cooled with light water or organic materials are also possible. A prototype station (Gentilly 1), in which a light water coolant is allowed to boil in the pressure tubes, has been in operation since 1972. This reactor, CANDU-BLW (Canada Deuterium Uranium-Boiling-Light Water), is very similar to the Steam Generating Heavy Water Reactor (SGHWR) now being developed in Great Britain as their next generation of power reactors. The organic cooled reactor concept (OCR) has the potential for achieving high temperatures. Table 9 summarizes several characteristics of the various heavy water reactor concepts.

3.2.2   CANDU-PHW*

Atomic Energy of Canada Limited (AECL) is presently the only commercial manufacturer of HWRs. A 600 MWe unit has been selected by AECL as its standard model. The so-called CANDU 600 units are being installed in Canada (at Gentilly and Lepreau) as well as in Korea and Argentina.

The uranium oxide fuel is supported in a suitable spatial arrangement in the heavy water-moderator which is contained in a vessel called a "calandria"" This spatial arrangement is provided by a system of tubes which pass through the calandria in a regular pattern (lattice) . Due to the moderating

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McIntyre, H. C., "Natural Uranium Heavy-Water Reactors", Scientific American, Vol. 223, No. 4, October 1975.

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characteristics of D_2O, the optimum lattice spacing of the fuel is relatively large compared to the lattice spacing in a light water reactor.

To permit removal of the nuclear heat, the fuel bundles are contained in pressure tubes which pass concentrically through the calandria tubes, but are separated from them by an insulating gas gap. The nuclear heat is removed by a coolant which is pumped through the pressure tubes. The heavy water coolant transports the heat, in a closed, high-pressure circuit, to heat-exchanger boilers where it generates steam to drive the turbine.

A principle feature of the CANDU reactor is the complete separation of the moderator system from the heat transport system. The moderator system is a cool (non-boiling) system maintained at substantially atmospheric pressure. The typical primary coolant system, on the other hand, operates at a reactor outlet temperature of approximately 300°C and a pressure of 100 kg/cm^2. The CANDU coolant is contained inside the 10 cm diameter pressure tubes as it passes through the calandria. This separation of systems reduces the severity of the design basis accident and some believe that the HWR is, therefore, safer than the LWR. The large heat sink, which exists in the form of the relatively cool heavy-water moderator in the calandria, minimizes the consequences of pressure-containing component failure within the reactor core.

The reactor is fueled with natural uranium in the form of compacted and sintered cylindrical pellets of uranium dioxide (UO_2). Approximately 30 of these UO_2 pellets, stacked end-to-end, are sealed in a zirconium alloy sheath to form a fuel element. Thirty-seven of these elements are welded to two end-plates to form the cylindrical bundle. The elements are separated by split spacers.

The reactor is refueled by two remotely controlled fuel-ing machines, one at each end of the horizontally-tubed reac-tor. The fueling machines, working at opposite ends of the same fuel channel, insert new fuel and remove spent fuel while the reactor continues to operate. The spent fuel is transferred under water, through a canal and transfer lock to the spent fuel bay.

At the present time, it is not economically feasible to reprocess the spent fuel to recover the plutonium. Storage space adequate for accumulation of 10 reactor-years of spent fuel is a design requirement. It is not expected, however, that Canada will have a reprocessing capability within 10 years and must, therefore, provide either additional storage facilities at the reactor sites or a rational program of interim storage. Three concepts have been proposed for long term spent fuel storage; one wet storage concept and two dry storage concepts utilizing concrete canisters and convection vaults.

Several studies have analyzed the use of thorium and mixed oxide fuels in the HWR. The feasibility of the use of these fuels depends strongly upon the long term uranium prices and the feasibility of reprocessing. A conversion ratio of .9 is feasible if enriched uranium or bred fissile fuels are used.

3.2.3 Heavy-Water Moderated Boiling Light Water Reactor

The HW-BLW reactor is a conceptual 1000 MWe design developed jointly by AECL, and Sargent and Lundy. The design features a vertical pressure tube calandria-type reactor, cooled with boiling light water. The coolant enters the bottom of the reactor at about 1,000 psi and exits at the top as 30% quality steam. The steam, after separation, goes

directly to the turbine. The fuel assemblies for this
reactor consist of 19-rod, Zr-4 clad oxide pellets. Five
assemblies, each 1.5 m long, are stacked in each of 688
pressure tubes. Burnups of about 8,000 Mwd/MT are possible.

3.2.4 Heavy Water Moderated-Organic Cooled Reactor

The fuel cycle for a 1000 MWe natural uranium carbide
fuel HWOCR is based on a design developed at ORNL. There
appears to be little current interest in this approach. The
main asset of an organic cooled reactor is an increased
plant efficiency, resulting from a higher temperature opera-
tion.

3.2.5 Material Flow in Heavy Water Reactors

Table 10 summarizes the material flow in the various
heavy water reactors. It is noted that the ^{235}U content of
the discharged fuel from the CANDU-PHW is .22%, which is
less than the current tails assay from the enrichment plant.
Figures 29-31 illustrate the fuel cycle material flows.

3.3 GAS-COOLED REACTORS

3.3.1 Introduction

The attractiveness of gas cooling lies in the fact that,
in general, gases are safe, are relatively easy to handle,
have low macroscopic neutron cross sections, and may be oper-
ated at high temperatures without pressurization. The main
disadvantages are the lower heat-transfer and heat-transport
characteristics of gases, which require large contact sur-
faces and flow passages within the reactor and heat exchangers,
and their high pumping requirements (between 8 to 20 percent
of plant's gross power) .

To partially overcome the inherent disadvantages of gas
coolants and, at the same time, to obtain attractive thermo-
dynamic efficiencies, it is necessary to operate the fuel
elements at high temperatures (commensurate with metallurgy)
and to permit a high gas-temperature rise in the reactor by
reducing the gas mass-flow rate and pressurizing the gas.
Because the fuel operates at high temperatures, fuel-element
and cladding-material choice and fabrication in gas-cooled
reactors present major problems, and the trend seems to be
toward using ceramic fuels in such reactors. Because gas-
cooled reactors are inherently large, they are particularly
suited to large-capacity power plants, but the reactor itself
may impose structural and foundation problems. The size of
the units can, of course, be reduced to a certain extent by
increasing the fuel enrichment.

Significant gas-cooled reactor development and commer-
cialization programs have been undertaken by Great Britain,
France, West Germany, the United States, and the USSR. His-
torically, the British led the way with their natural uranium,
carbon-dioxide cooled and graphite moderated reactors. Since
the fuel was in the form of metallic uranium rods, canned
within a magnesium alloy cladding, these plants became known
as MAGNOX reactors. In an attempt to improve the steam
conditions by raising the coolant temperatures, a second
generation of gas-cooled reactors evolved in Great Britain.
These advanced gas-cooled reactors (AGR) are characterized by
their carbon dioxide coolant, graphite moderator, and stain-
less steel clad rods of slightly enriched uranium dioxide.
Continued efforts in raising the coolant temperatures have
resulted in a class of high temperature reactors. The two
outstanding examples are General Atomic's HTGR and the West
German thorium high temperature reactor (THTR). Both are

graphite moderated and helium cooled, and operate on the uranium-thorium fuel cycle. Presently, 300 MWe demonstration plants for both the HTGR and THTR are nearing commercial operation, and large commercial plants (in the neighborhood of 1000 MWe) have been designed.

The basic differences between the HTGR and THTR lie in the fuel design and refueling procedures. In the HTGR, microsphere of fuel are mixed with a graphite binder to form fuel rods which are subsequently inserted into prismatic blocks of graphite. Annual refueling is anticipated for the HTGR. On the other hand, the THTR is a pebble-bed concept, designed for continuous on-line refueling. A design for even higher coolant temperatures is referred to as the very high temperature reactor (VHTR). Finally, a gas-cooled fast breeder reactor (GCFR) has been proposed. The basic idea is to combine the helium coolant technology from the HTGR program with fuel development from the liquid metal fast breeder program and produce a GCFR with a minimum of additional research effort. In Table 11 the general characteristics of each type of gas-cooled reactor are given.

3.3.2 High Temperature Gas-Cooled Reactor**

The HTGR is a thermal reactor characterized by a helium coolant and a uranium-thorium fuel contained in graphite blocks which serve both as moderator and core structural material. The entire nuclear steam supply system, which includes the reactor core, steam generators, helium coolant circulators, control rod drives, and the auxiliary core cooling system, is housed in a pre-stressed concrete reactor vessel. The unique material requirements demand about 2 million cubic feet of helium and over 50 cubic meters of graphite per core for a 1160 MWe plant.

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"Development Status and Operational Features of the High Temperature Gas-Cooled Reactor", Electric Power Research Institute, April 1976, EPRI NP-142

The basic fuel element in the HTGR is a graphite block with a hexagonal cross-section. Each element is 35.6 cm across the flats and 78.7 cm long. The fuel is in the form of coated particles of uranium dicarbides and oxides as the fissile material and thorium oxide as the fertile material. These are bonded in a graphite matrix to form fuel rods which are located in vertical blind holes in the fuel elements. Vertical coolant holes are provided for helium flow through the fuel elements. The core is formed by stacking these graphite blocks into 493 columns, each eight blocks high. The core is divided into 73 fuel regions. Each region is composed of a central control fuel column surrounded by six columns of standard fuel elements, except at the core periphery, where reflector columns replace some of the fuel columns. Each group, called a refueling region, rests on a graphite support block and is located directly below a refueling penetration that houses a control rod drive assembly. The refueling regions are grouped into four segments for refueling purposes, and one segment is refueled each year.

The fuel cycle for HTGRs is based upon the 93% enriched ^{235}U-thorium fuel cycle, with recycle of bred ^{233}U. This fuel cycle can involve two different modes of operation over the lifetime of the plant. These are:

1. Non-recycle operation, in which fuel removed from the core is placed in storage awaiting processing and recycle. Core operation is sustained by the introduction of additional fresh fully enriched fuel.

2. Recycle operation in which the fuel removed from the core is reprocessed and the ^{233}U is fed back into the core along with sufficient ^{235}U.

The utilization of ^{233}U has advantages because, 1) ^{235}U yields about 10% fewer neutrons per absorption than ^{233}U,

2) the fission rate per atom of ^{235}U is much more temperature sensitive than ^{233}U and 3) significant amounts of ^{236}U are formed by radiative neutron capture in ^{235}U. Since ^{236}U is an undesirable neutron poison) the continued use of the uranium fissile material may be limited to one recycle.

Two types of fuel particles will be used. The recycled ^{235}U, as well as the highly enriched uranium feed, will be contained as UC_2 in the kernel of the fissile particle which has a special coating. The thorium will be contained as ThO_2 in the kernel of the fertile particle. The fuel particles are blended in suitable proportions and formed into fuel rods using a graphite matrix as binder material. The fuel rods are about 16 cm in diameter and 5 to 6 cm long. The fuel rods are loaded into a graphite block to make a completed fuel element. Each fuel block contains only one of the three types (initial or makeup, highly enriched uranium; recycle ^{233}U, and once irradiated ^{235}U) as well as 'he fertile thorium.

Refueling must be performed when the reactor is shut down, the PCRV* depressurized to slightly subatmospheric pressure, and the core inlet temperature reduced to about 120°c. The fuel elements and replaceable reflectors are installed or removed through penetrations located in the top head of the PCRV; these penetrations also serve as control rod drive supports. During refueling, the control rod drives are removed and the fuel handling equipment mounted directly over the penetration. Removed elements are placed in a transfer cask, which is shuttled to the fuel storage area. During a normal refueling year approximately 18 regions are visited and 1000 elements are replaced. The total **shutdown** time is believed to be about 20 days.

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*Pre-stressed concrete reactor vessel

3.3.3 Thorium High Temperature Reactor*

The pebble bed reactor is an alternative to the GA-HTGR as a viable high temperature gas-cooled reactor concept. Pebble bed reactors are characterized by a mixture of fuel and some or all of the moderator, which is fabricated into spherical "pebbles". The pebbles are then randomly packed into a suitable vessel, or bed, to form the reactor core. Core cooling is provided by gas flowing through the space between the pebbles. Figure 32 schematically compares the THTR and the HTGR.

Development of pebble bed reactors has occurred principally in West Germany. The initial result of this effort is the 15 MWe helium cooled pebble bed reactor at Jülich. A 300-MWe THTR (thorium high temperature reactor) is currently under construction at Uentrop. Designs for 1000 MWe THTRs have also been initiated.

The Uentrop THTR primary system is integrated into a pre-stressed concrete reactor vessel. The core, i.e., the pebble bed, is enclosed in a round graphite structure which is 5.6 m in diameter and approximately 6 m high. The bed contains 674,200 fuel spheres, each 6 cm in diameter. Control rods enter from above the core. A bank of 42 rods may be inserted pneumatically directly into the core and a total of 36 control rods may be inserted vertically into the reflector surrounding the core. Average core power density is 6 MWt/m^3

The spherical fuel element of the THTR contains 200 gm of graphite and 33,000 uranium-thorium oxide kernels, coated with a layer of pyrolytic carbon which has a minimum thickness of 0.5 cm. Each kernel is 0.4 mm in diameter and is enclosed in two layers of pyrolytic carbon 0.18 mm thick. The metallic content of an element is 0.96 gm ^{235}U and 9.62 gm ^{232}Th. In order to equalize the radial power and helium

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* Oehme, H. , "Comparative HTGR Designs", ANS Topical Meeting Gas Cooled Reactors: HTGR and GCFBR CONF-740501, May 1974.

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* Oehme, H. , "Comparative HTGR Designs", ANS Topical Meeting Gas Cooled Reactors: HTGR and GCFBR CONF-740501, May 1974.

outlet temperature, there will be two concentric core enrichment zones. Average fuel burn-ups of 110,000 MWd/MTM are anticipated.

A continuous refueling scheme has been adopted for the THTR. Fuel elements which are discharged from the bottom of the core are monitored for burn-up and either returned to the core or disposed to waste storage. A pneumatic tube mailing system is installed beneath the core to perform the refueling. An average throughput rate of 1.8 times per year is anticipated. Thus, an average of 5-8 passes through the reactor are made by each element during its 3 year life.

The design principles of the Uentrop THTR can only be conditionally applied to large pebble bed reactors of 1000 MWe rating. The increase in the number of fuel elements required for higher thermal power cannot be accommodated by the larger scaling of the core dimensions. Thermodynamic and physical considerations limit the core height. Thus, a relatively flat core with an enlarged diameter of over 10 m is envisioned. Additionally, for the 1000 MWe system, a new on-line refueling scheme is proposed. The fuel elements would pass through the reactor only once. This scheme is known as OTTO for Once-Through-Then-Out. The pebbles are inserted through the top of the core by gravity discharge through 24 tubes and withdrawn at the bottom through 3 discharge tubes. Since the pebbles are not recycled, the pneumatic fuel handling facility and the burn-up measurement system are excluded. The power distribution is shifted towards the cold upper region of the core, which reduces the maximum fuel temperature and increases the reactivity value of the shut-down and control rods in this core region.

Detailed fuel cycle information suitable for a 1000 MWe plant is not available.

3.3.4 Advanced Gas-Cooled Reactors*

The Advanced Gas-cooled Reactor (AGR) represents the second generation in Great Britain's development of graphite-moderated and CO_2 cooled reactor plants. The objectives of the AGR program are to construct nuclear power stations that supply steam at conditions comparable with those in modern fossil-fueled power stations and with a degree of integrity which permits siting nearer population centers. Five commercial-sized AGR plants are in various stages of construction and operation.

These plants are very similar and each plant consists of 2 reactors. The reactors have a thermal output of nearly 1400 MWt, with reactor thermal efficiencies of 45.3% and plant net efficiencies of 41.6%. Coolant temperatures at the core inlet and outlet are approximately 300°C and 650°C, respectively. Steam conditions at turbine inlet are about 550°C at 170 kg/cm^2.

The reactor core is a 16-sided structure constructed from polygonal graphite blocks arranged in a square lattice. The blocks are interconnected by graphite keys to provide stability and to maintain the correct pitch. Large vertical bores through the blocks form the vertical fuel channels. Square interstitial graphite blocks are placed between the polygonal blocks and contain coolant channels and control rods.

To maximize the temperature of the coolant while main-taining fuel pin integrity, a stainless steel cladding is used. This cladding requires low enrichment fuel (~2% ^{235}U): The fuel elements consist of 36 pins containing hollow UO_2 pellets. The pins are arranged in three rings within a graphite sleeve. Eight such elements are linked together by a tie bar to a fuel unit extending to the top of the refueling standpipe and terminated with a pressure closure. Fuel

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*
 "Hinkle Point B, A Survey of Design and Construction," Nuclear Engineering International, Vol. 13; No. 147, August 1968.

and control rods are replaced on line. Burnups of 18,000
MWd/MT are anticipated. The fuel cycle does not generally
require axial shuffling of fuel elements.

Refueling of the twin reactors is accomplished with a
single refueling machine that runs on a moving gantry span-
ning both reactors. A central service block contains all
the shielded cells required for the assembly and dismantling
of fuel stringers, for the maintenance of control rods, and
for storing complete fuel stringers as necessary. During a
refueling operation, a stringer of spent fuel is removed
from the core and allowed to decay 10 to 12 hours before
being lowered into the Irradiated Fuel Dismantling Cell.
After the fuel stringer is disassembled, the spent fuel
elements are stored in a cooling pond. Provisions are avail-
able for sealing the fuel elements within a stainless steel
bottle which has an inert atmosphere before discharge to the
pond. It is anticipated that the spent fuel will be shipped
by rail and road to Windscale for reprocessing.

3.3.5   Gas-Cooled Fast Reactor*

This concept requires that the neutron spectrum not **be**
degraded by a moderator so that the resonance capture of
neutrons ^{238}U is maximized. Consequently, there is no
graphite moderator and the reactor core is similar to the
LMFBR.

The major characteristics of the current reference con-
cepts of a gas cooled fast reactor (GCFR) are based on mini-
mizing development work. This results in the selection of
the steam cycle for power conversion and of bundle type fuel
elements containing oxide ceramic fuel pellets in steel cans.
Concepts have been proposed by General Atomic (GA), the Gas
Breeder Reactor Association (GBRA) , and Kraftwerk Union (KWU) .
The development of the HTGR and the AGR provides the required

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 "Development, Status and Operational Features of the High
 Temperature Gas-Cooled Reactor", Electric Power Research Institute,
 April 1976, EPRI NP-142

background for designing a nuclear steam supply system
entirely housed in a pod type prestressed concrete reactor
vessel (PCRV) in all three cases. The major difference from
the LMFBR fuel element design is the need to withstand a
substantially higher coolant pressure in the GCFR
for adequate heat transfer. The current concepts are based,
therefore, on equalizing the pressure between the interior of
the individual fuel pin and the ambient coolant. Also arti-
ficial roughening of the fuel cladding is proposed to improve
heat transfer.

Considerable effort has been made to develop a 300 MWe
demonstration plant. Preliminary analyses for a 1500 MWe
commercial plant have been performed and are summarized in
the following paragraphs.

The reactor core consists of 271 hexagonal, vented fuel
elements, 2.5 m long and 21.36 cm across the flats. There
are 27 similar control elements. The core is arranged into
four enrichment zones and a radial blanket.

The fuel elements are made up of 331 individual, .696 cm
diameter fuel rods, fabricated from 316 stainless steel clad-
ding with a wall thickness of about .037 cm. In this respect,
these rods are similar to LMFBR fuel rods except that these
employ surface roughening to enhance the heat transfer by a
factor of two and to reduce clad surface temperature. The
roughness results in a factor of three increase in friction
losses and, thus, increases the coolant pumping requirements.

Each fuel rod contains mixed (U,Pu) oxide annular fuel
pellets in the form of a right circular cylinder. The pellets
have a center hole to prevent center line melting. The U/Pu
fraction is such that the initial average fissile loading is
approximately 18 percent. Each fuel rod has a depleted UO_2
axial blanket below and above the stack of core pellets and

an individual alumina thermal shield and 3-inch activated
carbon fission-product trap.   The radial blanket fuel rods
are similar to the core rods except that they are larger,
1.98 cm o.d., are not roughened, and contain only pellets
of depleted UO_2 without a center hole.   Alternatively, the
radial blankets may be loaded with ThO_2 for the Production
of 233_U.   Because of the larger size, only 61 rods are
contained in the 126 blanket fuel elements.

Refueling is carried out with the reactor shut down and
at atmospheric pressure, with either air or helium in the
vessel.   The fuel transfer machine is placed in the plenum
space beneath the core by raising it through a port in the
bottom of the vessel.   This machine has a vertical receptacle
tube which can be positioned under any core or blanket ele-
ment.   Spent fuel elements are removed from the core by
lowering them into the receptacle tube by means of reach rods
which extend through nozzles in the top of the pressure
vessel.   The elements are then transferred to a spent fuel
removal port in the bottom of the vessel, through which they
are discharged and moved to a storage pit.   Cooling of the
fuel is provided during all stages of the fuel transfer.

3.3.6 Material Flows in the Gas Cooled Reactors

The fuel cycle requirements for the HTGR and AGR are
given in Table 12. They are depicted schematically in Figures
33' and 34 for both the start-up and equilibrium cycle.   Data
for the pebble bed reactor were not available, but they are
expected to be similar to the HTGR cycle.

There are several options in the GCFR fuel cycle which
depend upon the use of thorium or uranium blankets and the
use of high or low burnup plutonium in the core.   Table 13
contains fuel cycle information for the uranium blanket.
This information is shown schematically in Figures 35 and 36
Figure 37 shows similar results if a thorium blanket is used
with low burnup plutonium.

3.4 LIQUID METAL FAST BREEDER REACTOR*

The liquid metal fast breeder reactor (LMFBR) is the most widely used breeder reactor concept under development throughout the world today. This concept utilizes a liquid metal coolant, no moderator and is based on the uranium-plutonium fuel cycle. Plutonium from LWR'S is needed to provide initial fuel loading but after startup the LMFBR generates all the plutonium needed for subsequent refueling. Depleted uranium (the enrichment plant tails) is used as the fertile material. (There are some 200,000 tons of depleted uranium stored in the U.S.) ^{238}U is fissionable by fast neutrons and about 20% of the fissions in the LMFBR are from the ^{238}U.

The central core contains the fissile material and provides the main source of energy. The core is surrounded by radial and axial blankets of depleted uranium. The coolant is generally liquid sodium and both loop and pot concepts have been used. The loop concept is similar to the PWR coolant system and the pot concept utilizes a large reservoir of sodium which contains the heat exchanger, primary pumps and the reactor. Some believe, since the loop concept is more susceptible to a loss-of-coolant accident, that the pot concept is inherently safer.

The decision to pursue a LMFBR program has been made by every nation having a nuclear development program, with the exception of Canada. Figure 38 shows the various LMFBR's which have been built throughout the world along with the

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planned higher power reactors. Major difficulties center around the steam generator and material problems. The Clinch River Breeder Reactor Plant, a 380 MWe demonstration facility, is the lead fast reactor in the U.S.

The fissile loading of the core at the beginning of the life cycle is essentially all ^{239}Pu and ^{241}Pu. Depleted uranium, containing from .2 to .3% ^{235}U, is utilized as the fertile material. The core undergoes burnups of 70,000-100,000 MWD/MT and the blankets will undergo from 5,000 to 20,000 MWD/MT. The axial blankets are generally part of the core fuel element and will not be reprocessed separately.

The fuel is generally UO_2-PuO_2 clad in stainless steel. The core for a 1,000 MWe plant will be about 1 meter high and 2.0 to 3.0 meters in diameter and contain about 200 to 300 fuel assemblies. The radial blanket will contain another 100 to 150 assemblies, and may contain two or more radial zones. The number of pins per assembly may range from 200 to 300 for the core, and from 50 to 100 in the blanket. The plutonium loading in the core will probably vary from 10 to 20% Pu over several regions. The breeding ratio will be around 1.2 to 1.3.

Refueling will occur once a year, replacing about one-half of the core and one-third of the radial blanket. The spent fuel will be stored on site for more than 30 days before shipment to a reprocessing facility. The reactor must be shut down for refueling but the fuel elements are with-drawn through the top of the reactor vessel. The spent fuel will be stored in hot cells or in sodium-cooled decay tanks on site.

3.4.1 Material Flow in the Liquid Metal Fast Breeder Reactor

Table 14 depicts the material flow in the nuclear fuel cycle for a "typical" 1,000 MWe LMFBR. This material flow is shown schematically in Figure 39. It is assumed that the depleted uranium is recovered from the waste tails of the enrichment plant, so that no mining or milling is required.

3.5 LIGHT WATER BREEDER REACTORS*

The light water breeder reactor (LWBR) relies extensively upon the LWR technology and has the major purpose of producing as much fissile material as it uses. The present concepts are based on the pressurized water reactor (PWR) and may be implemented by placing a different reactor core and control system in present PWR reactor plants. The reactor concept is being studied in the U.S. and a demonstration operation in the Shippingport reactor is scheduled for the late 1970's.

The LWBR is a thermal reactor which would convert thorium to 233_U. Because the breeding (conversion) ratio is near 1, prebreeders are required to produce enough ^{233}U for the first few breeder cores. The prebreeder cores have different neutron requirements. The basic core design utilizes the seed-blanket concept, in which each fuel module contains fissile regions (seeds) and a fertile blanket. A low water content in the core is required to minimize neutron capture in the hydrogen and a water-to-metal ratio of about 1/10 that of the standard PWR has been proposed.

To minimize parasitic neutron capture in control rods, various designs utilize either fissile or fertile materials as the control element. Fertile blankets increase the size of the core but utilize the leakage neutrons. For a given PWR reactor vessel, a LWBR core could produce only about 70% of the power of the PWR. A prebreeder would not result in a significant derating of power.

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*Final Environmental Statement LWBR Program, ERDA 1541, June 1976

Burnup of the core will be about 20,000-50,000 MWD/MT. It is expected that the reactor will be refueled in a manner similar to the LWRS. The reactor will be shut down for a period of up to 30 days, and the pressure vessel head removed to retrieve a portion of the fuel.

3.5.1 Prebreeder

The prebreeder will be obtained by placing a new core in an existing PWR. For example, the Westinghouse PWR core module could be replaced by one containing UO_2 and ThO_2, rods. About 190 modules will fill the 360 cm high, 360 cm diameter core. Each module will contain 240 UO_2 rods .75 cm in diameter and 100 ThO_2 rods 1.7 cm in diameter. The uO_2 **is** enriched to 10-13% [235]U.

3.5.2 Breeder

A larger reactor vessel is necessary for a breeder core to prevent derating an existing PWR vessel. The core will be about 450 cm in diameter and have an active height of 320 cm, with a reflector region about 20 cm on both the top and bottom and contain about 74 fuel modules. The [233]UO_2-ThO_2 seed region of the fuel module will contain about 620 rods, .91 cm in diameter, and the blanket region about 445 rods, 1.7 cm in diameter.

3.5.3 Material Flow in Light Water Breeder Reactors

The material flow for both a prebreeder and a breeder are shown schematically in Figures 40 and 41 . Table 15 provides more detailed information on the fuel cycles. Since detailed data have not been published for a commercial size plant, the data given should be applied with caution.

3.6 MOLTEN SALT BREEDER REACTOR*

The molten salt breeder reactor (MSBR) concept is based on the use of a liquid fuel which circulates between the reactor vessel and a heat exchanger. The fuel is a complex salt ($LiF-BeF_2-ThF_4-UF_4$) in the ratio of 71.7-16-12-.3 mol percent. This chemically toxic salt melts around 500°C and serves as the fuel for the reactor. The salt flows through channels in a graphite moderator in the 6.6 meter diameter, 6.0 meter high reactor vessel. The coolant flow is about 4 m^3/sec and leaves the core at about 700°C. The heat is transferred to another molten salt in a heat exchanger. A fraction of the fuel (about 3 liters/minute) may be continuously removed for chemical removal of fission products. The entire fuel inventory is processed about every 10 days. The thorium, uranium and plutonium are not separated.

The major disadvantages of this reactor concept are related to the containment and continuous processing of very radioactive, toxic, and corrosive materials. Maintenance of the system must be a dominant design goal. Most of the processes involved are not utilized elsewhere in the nuclear industry, so the development costs will be high.

3.6.1 Material Flow in the Molten Salt Breeder Reactor

As in other reactor concepts, there are several options available. These options include the use of batch or continuous reprocessing, (this affects the breeding ratio), the use of plutonium as an initial salt, the replacement of the graphite, etc. Figure 42 schematically illustrates the fissile and fertile material flow. Table 16 provides additional characteristics of the reactor.

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*"The Use of Thorium in Nuclear Power Reactors," USAEC WASH-1097, June 1969.

3.7 RESEARCH AND MARINE REACTORS*

In this section research reactors and the low-power reactors designed for propulsion of merchant ships are discussed. There are many types of research reactors operating throught the world -- See Appendix B2. As with power reactors, it is possible to categorize them in various ways -- by type of fuel, type of moderator, power level, type of coolant, etc. Some general characteristics can be determined without regard to detailed features. For example, Fig. 43 indicates the annual natural uranium fuel requirements of an enrichment plant serving various types of 10 MWt reactors. As expected, regardless of fuel enrichment, between 1 MT and 10 MT of fuel are required. The SWU requirements are given in Fig. 44 and Fig. 45 shows the annual uranium fuel requirements for the reactors. It is noted that for fully enriched uranium, about 10 kg of uranium per year are required. (Marine reactors (non Navy) are also shown in these figures) . Figure 46 summarizes the plutonium production, per MWt per year, for 75% operation. The graphite reactors produce about .9 gm ^{239}Pu per day, per MWt. For high burnup cores, the plutonium production rate may be high due to the assumption, in our analysis, that none of the plutonium is fissioned. In the operation of research reactors it is usual practice to replace a portion of the core and to shuffle the remaining fuel elements in order to achieve a higher uranium burnup.

3.7.1 Heavy Water Moderated Research Reactors

These reactors are generally tank-type reactors, with the heavy water acting both as moderator and coolant. A tank-type reactor has a closed, pressurized primary coolant system which transfers reactor heat to a light water, secondary coolant system. Fuel enrichments may vary from natural uranium to fully enriched. The natural uranium fuel elements have relatively low

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*"Power and Research Reactors in Member States," 1974 Edition
 International Atomic Energy Agency, Vienna, 1974.

burnup (600-1500 MWD/Ton) and are usually aluminum clad uranium
metal rods.   Low enriched fuel elements include higher burnup
aluminum clad uranium - molybdenum alloy rods and clad UO_2 fuel.
Highly enriched fuel is generally dispersion type, of which
the Material Test Reactor, fuel element is a typical example. Straight place
elements are also in use.   The MTR fuel element has aluminum
clad plates, with an aluminum-uranium dispersion fuel.
The MTR fuel element may contain between 10-19 curved or straight
fuel plates, with a variety of fuel enrichments (generally 20-93%)
and fuel loadings (8 to 19 gm ^{235}U per plate).   Due to its
flexibility, the MTR fuel element is widely used in heavy water
and light water research reactors.

Dispersion fuel reactors characteristically have high
burnup.   Twenty percent enriched MTR-fuel heavy water reactors
have demonstrated 15,000-26,000 MWD/MT, while 90-93% enriched
heavy water reactors have reached burnups well above 200,000-
250,000 MWD/MT. The National Bureau of Standards is currently operating with
fuel burnups as high as 50-55% of fissionable material, which
corresponds to a burnup greater than 400,000 MWD/MT.   Figure 49
illustrates the material flow for a 10 MWt heavy water moderator
research reactor.

3.7.2 Graphite Moderated Research Reactors

These reactors are generally air cooled, graphite pile.
Light water coolant may be required for high power levels. Fuel
is generally natural uranium slugs clad in aluminum. Some gra-
phite reactors, such as the Brookhaven Graphite Reactor, have
operated with fully enriched uranium fuel. The natural uranium
fuel has a burnup in the range of 600-2000 MWD/MT. Fuel load-
ings are very high on the order of tens or hundreds

of metric tons. Typically, the BR-1 reactor, operating at 4 MWt, required approximately 24,000 kg of natural uranium.

High plutonium production rates are achieved with natural uranium fuel. These reactors are easily fueled on line. Figure 47 illustrates the fuel cycle feed/discharge characteristics of graphite moderated reactors.

3.7.3 Light Water Moderated Research Reactors

These reactors may be pool-type (generally low power, < 5 MWt), tank-type or pressurized water type (generally higher power, > 5 MWt). This is a very simple type of reactor, without the multitude of supporting systems and secondary coolant loops typical of the higher power, tank-type reactors and PWRs.

The fuel is enriched (10-93%) uranium in dispersion type elements. Amont the types in widespread use are the MTR type, previously discussed, and the TRIGA-type. TRIGA elements are usually   20% or 70% enriched; however, 93% enriched elements are contemplated for use in the latest and largest TRIGA core designs.   The fuel is a uranium-zirconium hydride matrix, clad with aluminum (on the elements with the lowest ^{235}U loading) or stainless steel. Uranium loadings vary from 37 to 53 grams of 20% enriched uranium, to 136 grams of 70% enriched uranium in each fuel element.   A TRIGA core consists of 85-100 fuel elements. Burnable poison is incorporated in the 70% enriched fuel elements, which contain 1.6 W/o Erbium to compensate for the high uranium loading. See Figure 48 for a 10 MWt research reactor fuel cycle.

3.7.4 Critical Facilities

The assembly-machine type of critical facility provides the designer the opportunity to investigate subcritical and critical reactor configurations with a wide variety of

lattice arrangements, fuel loadings and fuel types. The flex-
ibility of a plutonium critical facility is well illustrated by
the Zero Power Plutonium Reactor (ZPPR) which can accommodate
fourteen fuel types, including plates of unclad uranium metal,
clad Pu-A1 and clad U-Pu-Mo alloys and rods of clad $(U-Pu)O_2$
and UO_2. A wide range of enrichments of both Pu and U are
utilized.

An example of a ZPPR core is a 6000 liter, 2-zone core,
measuring approximately 120 cm in height, with an inner zone
diameter of 180 cm and an outer zone diameter of 250 cm. The
fuel is Pu-U-Mo, with an approximate critical mass of 2260 Kg,
including 2083 Kg ^{239}Pu, 144 Kg ^{241}Pu and 33 Kg ^{235}U.

Critical facilities are generally characterized by very
low power levels, on the order of hundreds of watts, to a few
kilowatts. As a result, there is negligible fuel burnup or
fission product generation. In addition, there is negligible
plutonium buildup in uranium fueled cores and fuel loading is
generally accomplished by manually loading clad plutonium or
uranium fuel elements. Because of the flexibility required
in performing critical experiments, several researchers
will   have access to the fuel.

3.7.5 Marine Reactors

To date, marine reactors (non-Navy) have been character-
ized by relatively low enrichment (4-6.5%), high burnup cores,
using UO_2 type of fuel elements similar to those used in com-
mercial LWRS. The more recent designs, typified by the B&W
CNSG design, provide for extensive burnable poison zoning in
order to achieve high fuel burnup. The average CNSG core
burnup is approximately 36,000 MWD/MT. This is on the same
order of magnitude as the current generation of commercial
PWR power reactors.

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Current marine reactors are PWRs with stainless-steel clad fuel. Their design may differ significantly from that of land based PWRs.

The marine reactor fuel cycle is generally designed to provide complete core replacement after a 3 to 4 year service life. Because of this, the spent fuel elements are characterized by high fission product inventory, high ^{235}U burnup and high plutonium buildup. Due to the long residence time in the core, there is substantial burnup and transmutation of ^{239}Pu. At the end-of-life, the 313 MWt B&W CNSG reactor contains a total plutonium loading of 103 Kg, including 85.5 Kg of fissile plutonium. Operating Marine reactors are considerably smaller than the current CNSG design. The U.S. Savannah reactor was rated at 70 MWt and the German Ship Otto Hahn operates with a 38 MWt reactor plant.

3.8 ADVANCED CONCEPTS

In this section, several specific topics are presented. These include the use of the tandem fuel cycle, fissile material production in fusion reactors and electric breeding.

3.8.1 Tandem Fuel Cycle

A variation on the basic light water reactor fuel cycle concept is a tandem fuel cycle. This fuel cycle is based on the use of nuclear fuel in LWR's and, after mechanical refabrication, using it" as fuel first in LWR's and, after mechanical refab-fuel cycle option does not involve the chemical reprocessing of spent nuclear fuel from LWR's. Most of the useful fissile material content of the LWR fuel is utilized without separating the uranium and plutonium from the radioactive fuel elements.

Because deuterium is a more efficient neutron moderator than light water, HWRs can utilize fuel with a lower fissile material content. Studies have indicated that PWR spent fuel could generate an additional 30% to 45% more electricity if the spent fuel is placed in a HWR. These analyses were based on a 33,000 MWD/MT PWR fuel burnup, and indicated that an additional 10,000 MWD/MTU burnup is achievable in a HWR. (Current experience with LWR fuel indicates a burnup of only about 25,000 MWD/MT, indicating that more than 10,000 MWD/MT may be possible with the HWR.) This additional energy is roughly equivalent to the energy which would result from the reprocessing of spent PWR fuel and the recycle of both recovered uranium and plutonium. Table 17 summarizes the changes in fuel composition during the tandem fuel cycle.

TABLE **17**

FUEL COMPOSITION

	Fuel Charged (K/MT)	LWR Discharge (33 ,000 MWd/MT)	HWR Discharge (45 ,000 MWd/MT)
U-235	32 (,3 . 2% U-235)	7. 2 (@.72% U-235)	2.8 (~0. 28%)
Pu Total	0	9.1	8.7
Pu 239 + 241	0	6.4	4.8

It is interesting to note that the final ^{235}U enrichment of HWR fuel is at or below the depleted uranium level from enrichment facilities and that the total plutonium contained by the HWR spent fuel is almost as great as that in the PWR spent fuel but with a reduction in the thermally fissile content. The significance of these observations are as follows:

- The utilization of uranium resources by the tandem fuel cycle is approximately equal to that of the light water reactor with recycle.

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- Because of the depleted nature of the resultant uranium and the low fissile content of the plutonium, the spent HWR fuel is probably economically unsuited for reprocessing as a source of fissile material for the LWR fuel cycle.

- The spent HWR fuel represents a source of plutonium which could be useful in future LMFBR development. The spent HWR fuel could be stored as spent fuel which would discourage unauthorized use.

The commercialization of the tandem fuel cycle requires the resolution of several technical problems, detailed analysis of economic potentials, and formulation of a new set of regulations and regulatory procedures. All of these issues are inter-related and must be resolved.

The first technical step is the verification of the preliminary reactor calculations both by additional calculations and actual reactor demonstration. The major technical difficulty is the mechanical refabrication of LWR spent fuel so that it can be utilized by a HWR. The LWR fuel has a different configuration than current HWR fuel, and the LWR spent fuel has a high temperature history, variations in ^{235}U and fissile plutonium content, and brittle fused chunks of fuel.

There are several approaches to the conversion of LWR fuel for use in HWRS. Some are listed below:

1. Mechanical disassembly of the LWR fuel with a subsequent rejacketing and swaging into a HWR configuration.

2. Mechanical disassembly of the LWR fuel followed by grinding of the LWR fuel with subsequent reformation and fabrication in a suitable geometry.

3. Modification of HWR designs to utilize the LWR fuel with minor modifications. This entails recladding the fuel, as a minimum, and possible power derating to allow for nonuniformity of fissile content.

Problems associated with irregularities in fuels causing low thermal conductivity, non-uniform power generation and poor bonds between fuel and cladding must be addressed in any utilization" scheme.

The fuel from three or four 1,000 MWe LWRs would supply fuel for one 1,000 MWe HWR. Economic considerations include the capital cost of constructing a significant number of HWRs to accept the fuel from LWRs (HWRs are about 10-20% more expensive than LWRs) , and the capital cost of facilities required to refabricate LWR fuel to HWR as compared to the LWR fuel cycle capital costs. Finally, the operating costs of safeguards and security must be considered. These will most likely be less than for a LWR recycle. The regulatory problems of such a fuel cycle center around the licensability by NRC. The basic Canadian HWR is not currently licensed in the U.S. , although it is proven and used in several other countries. The problems of licensing, considering the fact that the fuel will have a unique and variable history, may increase the cost of reactor construction.

3.8.2 Fission-Fusion Systems

One of the major efforts now underway to develop long term energy sources is in the area of controlled fusion. At present, the major objective of the fusion program is the development of a commercial electric power reactor. The potential also exists for using controlled fusion to produce fissile material although this is currently being pursued at a low level in this country. Of the several possible reactions for a fusion reactor, the most promising is that of the two hydrogen isotopes, deuterium and tritium (the D-T reaction). This reaction produces a neutron and a helium nucleus, and releases 17.6 million electron volts of energy- The kinetic energy of the neutron accounts for about 75% of the energy releases by the reaction.

In a fusion device using this reaction, the neutrons would have two functions. First, they would provide a source of heat to generate steam and second, **a** fraction of the neutrons would be used to produce tritium, to fuel the reactor, by reacting with lithium. The latter occurs in an assembly, called a blanket, which surrounds the chamber in which the fusion reactions take place. The lithium moves through the blanket as a liquid metal. Only a small fraction of the lithium is converted to tritium by reactions with the neutrons and the remainder is heated upon absorbing the neutron's kinetic energy. This heated lithium is extracted from the blanket and passes through a heat exchanger where steam is produced.

As indicated, however, there are other ways to use the energy and neutrons produced by the fusion reaction. Because it is a source of fast neutrons, the possibility exists of producing fissile material to be used **i**n a fission reactor. This could be accomplished by placing fertile material in the blanket which would absorb a fraction of the fusion neutrons.

There have been several studies of fissile material production in potential fusion reactors. Three candidate systems have emerged. The first produces only fissile material and no energy is extracted to produce elec-tricity. The second is a hybrid system that produces both fissile material and electric energy from the fusion reaction. The third is also a hybrid system but the energy production results from fission reactions in the blanket and from the fusion reactions. In this case a portion of the neutrons from the fusion reactions are used to produce fissile material which is in a critical assembly, while the remaining are used to produce tritium and heat the lithium. The scientific conditions that must be achieved so that the system has energy gain are theoretically less stringent than needed for a fusion reactor by itself because of the potential energy yield of the fissile material. For a given sized device, however, the total energy yield would

be less than if one could achieve conditions needed for a fusion power reactor so if the latter works, it should be less expensive. The conditions required for successful operations appear to be less stringent for fission-fusion systems, however, so they may become available sooner.

Of course, the fissile material produced in a hybrid system can also be used for weapons, Therefore, fusion-fission devices would be subject to problems of proliferation and safeguards considerably more severe than would be the case for a fusion system alone. It is possible that the latter could be modified after construction to produce fissile material but this would involve placing fertile material in the blanket after it was constructed. In all probability this would mean an entirely new blanket assembly since it is unlikely that the original could be modified. The requirements of the blanket assembly in terms of structure and neutron reactions with the lithium for a given reactor means that it would have to be redesigned if fertile material were also to be included in order for the fusion reactor to continue opera-tion. Since the blanket assembly will be a large cost item for any potential reactor and since it will not be easily accessed, it is probable that there would be easier ways to obtain fertile material than to either modify an existing fusion reactor or build a hybrid system.

A fertile blanket may contain natural or depleted uranium or thorium. Due to the rapid buildup of fissile material, the blanket lifetime is limited by the power density resulting from the fissioning of the bred material. Calculations indicate that about 1.5 kg of fissile material are produced, per metric ton of fertilemetal, per 100 days in a blanket. A 600 MWe fusion reactor might utilize a blanket containing 100 tons of uranium, so nearly

500 kg of Pu could be produced per year. Due to the high
energy neutrons and relatively low burnup, the plutonium
would contain greater than 95% fissile isotopes.

    Experiments to test the design analyses of fertile
blankets are believed to be some of the goals of the Russian
T-20 fusion test reactor. The blanket design is probably one
of the least difficult portions of a fusion-fission system
and a fertile blanket could be incorporated into a fusion
reactor without a major development effort. Blanket refuel-
ing and heat removal are two major potential problems with
this system.

3.8.3 Electric Breeders

    When high energy protons strike a high Z target, such as Tangsten,
many neutrons result. These neutrons could be used in a depleted
uranium or thorium assembly to produce fissile material. Alternatively
these neutrons could be directed into a reactor containing spent fuel
and convert some of the remaining fertile material into fissile material.
This concept may be viewed as a re-enrichment process and if metallurgical
issues can be resolved, the fuel could reenter a fission reactor without
reprocessing. Thus, an accelerator could be used to reenrich spent fuel,
burn actinides in the spent fuel, or convert fertile to fissile material. For
example, ^{233}U could be produced form thorium for a denatured LWR fuel cycle
On the other had, the device could also be used to produce fissile material
for weapons. Hence the technology presents the same proliferations dichotomy
as advanced enrichment techniques.

    This concept has been pursued more vigorously by the
Canadians than by other countries. Technologies utilized in
producing neutron beams in facilities such as the Los Alamos
Meson Physics Facility can be used to investigate this concept
in the U.S. An 800 MeV proton incident on depleted uranium

will produce about 25 neutrons with an energy spectrum not much different from that due to fission. About 50 MeV of energy will be depposited in the uranium target, per proton, and this energy could be reconverted to electricity through a steam cycle turbine. A 30 milliamp proton beam produces about 100 kg of plutonium per year. The heat removal for the high power demsity targets and the improvements on the current maximum beam currents of about 1 ma are two major problems inherent to this concept.

In the United States there has been recent interest in electric breeding utilizing 500 MeV deuterons incident on lithium targets surrounded by uranium or thorium. For a 375 ma beam of 500 MeV deuterons about 1000 kg of plutoniun can be produced per year at a cost of approximately $100/gram.

TABLE 6

REPRESENTATIVE PWR AND BWR CHARACTERISTICS

Reactor Characterization	PWR	BWR 1
Date of Information	1974	1970
Representative Reactor		
Reactor Thermal Power (MWt)	3250	3293
Net Electrical Power (MWe)	1050	1053
Net Plant Efficiency (%)	32.3	32.3
Average Burnup (MWd/MTM)		
Initial	15,000	19,000
Equilibrium	25,000	25,000
Core Inventory (MTM)	80.12	148.50
Core Height (m)	3.65	3.66
Neutron Flux (n/cm^L/see)		
Peak Thermal	7.8×10^{13}	3×10^{13}
Average Thermal	2.6×10^{13}	7.8×10^{13}
Peak Fast	NA	NA
Average Fast	3×10^{14}	1×10^{14}
Fuel Description		
Number of Assemblies (core/blanket)	193/ --	764/ --
Dimensions of Assembly (hxd)m	4. 06x0. 215	4. 47x0. 138
Number of Rods per Assembly (core/b lanket)	204/ --	49/--
Chemical Composition (core/blanket)	UO_2/--	uo_2/ --
Cladding Material/Thickness (cm)	Zr- 4/ 0.062	Zr-2/O. 08
Enrichment (%) (Initial/Equilibrium)	(2. 8/3. 3) U^{235}	(2. 25/2 .60) U^{235}
Control Material	Ag-In-Cd rods B in solution	B_4C H_2O flow regulations
Control Rods/Assembly	20	1/4 cruciform rod 1/2 curtain rod
Refueling Interval	1 year	1 year
Fraction Reload per Cycle	1/3 core batch	1/4 core batch
Conversion Ratio (Initial/Equilibrium)	.5/ .6	0. 6/0.6

TABLE 7

FUEL CYCLE REQUIREMENTS (100 MWe, 75% LOAD FACTOR)*

REACTOR CHARACTERIZATIONS	PWR1-U Fueled		PWR2-Pu nat U Fuel		PWR3-U Fueled	
	Initial Load	Equilibrium Cycle	Initial Load	Equilibrium Cycle	Initial Load	Equilibrium Cycle
Mining (10^3 MT)	61,672	19,860	13,916	3,837	66,359	23,346
Material Removed						
Uranium 0.2% concentration	208.14	67.029	46.966	12.950	223.961	78.792
Thorium 0.2% concentration	0	0	0	0	0	0
Milling (MT U_3O_8)	466.355	150.184	105.230	29.015	501.083	176.538
(MT Th O_2)	0	0	0	0	0	0
Conversion (=Enrichment Feed) (MT UF_6)	581.947	223.012	0	0	621.181	272.059
Enrichment						
SWU (MT)	301.52	125.15			322.36	163.31
Tails (MTU at 0.2%)	316.45	125.011			339.55	146.61
Enriched Product (MTU)	77.07	25.78			83.86	37.35
Assay (% U-235)	2.8	3.3			2.78	3.4
Feed nat U (MTM)	393.50	126.722			423.41	148.96
Fuel Fabrication						
Input (MT)	77.07	25.78			83.86	37.35
Output (MT)	76.31	25.52			83.03	36.62
No. Assemblies/year	184	61			200	88
Composition	UO_2 enr.	UO_2 enr.			UO_2 enr.	UO_2 enr.
Weight of Assembly MTM	0.415	0.415			0.415	0.415
Used in Reactor	core	core			core	core
Reactor Load in Quantity (MTM) U/Pu/Th	76.31/0/0	25.52/0/0	87.912/2.598/0	24.24/1.281/0	83.03/0/0	36.62/0/0
Isotopics (%)						
U-233	0	0	0	0	0	0
U-234	0.03	0.0352	0	0	0.035	0.035
U-235	2.80	3.3	0.71	0.71	2.78	3.4
U-236	0	0	0	0	0	0
U-238	97.17	96.56	99.29	99.29	97.18	96.56
Pu-239	0	0	67.7	56.4	0	0
Pu-240	0	0	18.8	24.0	0	0
Pu-241	0	0	10.0	11.24	0	0
Pu-242	0	0	2.51	3.92	0	0

TABLE 7 (Cont)
FUEL CYCLE REQUIREMENTS (1000 MWe, 75% LOAD FACTOR) *

REACTOR CHARACTERIZATIONS	PWR1-U Fueled		PWR2-Pu nat U Fuel		PWR3-U Fueled	
	Initial Load	Equilibrium Cycle	Initial Load	Equilibrium Cycle	Initial Load	Equilibrium Cycle
Reactor Discharge						
Quantity (MTM) U/Pu/Th	24	39/0.229/0		23.67/0.833/0		35.46/0.266/0
Isotopics						
U-233		0		0		0
U-234		0.012		0.007		0.019
U-235		0.83		0.327		1.44
U-236		0.44		0.079		0.30
U-238		98.72		99.59		98.24
Pu-239		58.4		38.68		67.7
Pu-240		24.0		27.53		18.8
Pu-241		11.2		18.08		10.0
Pu-242		3.92		11.53		2.51
Th		0		0		0
Fission Products (MTM)		0.891		0.891		0.891
Net Plant Thermal Efficiency (%)	32.5	32.5	32.5	32.5	32.5	32.5
Burnup (Avg. MWd/MTM)	14,040	33,000	14,040	33,000	14,700	23,000
Spent Fuel Storage						
Decay Heat (W/kg)		28		24		26
Days Holdup		150		150		150
Activity at Discharge From Pool (10⁶ Curies F.P.)		113		116		79
Fuel Reprocessing						
Input (MT)		25.52		25.52		36.62
Recycle Uranium (MT)						
U-233		0		0		0
U-235		0.200		0.077		0.505
U-236		23.833		23.34		34.488
Plutonium Production (MT)						
Pu-239		0.133		0.356		0.178
Pu-241		0.026		0.166		0.026

TABLE 7 (Cont)

FUEL CYCLE REQUIREMENTS (1000 MWe, 75% LOAD FACTOR *)

REACTOR CHARACTERIZATIONS	PWR1-U Fueled Initial Load	PWR1-U Fueled Equilibrium Cycle	PWR2-P nat U Fuel Initial Load	PWR2-P nat U Fuel Equilibrium Cycle	PWR3-U Fueled Initial Load	PWR3-U Fueled Equilibrium Cycle
Mixed Oxide Fuel Fabrication						
Input (MT) U/Pu		0	88.791/2.624	24.482/1.294		00
Output (MT)		0	90.51	25.52		0
No. Assemblies/year		0	201	57		
Composition			nat.UO$_2$+PuO$_2$(2.85%)	nat.UO$_2$:PuO$_2$(5.02%)		
Weight of Assembly (MTM)			0.450	0.450		
Used in Reactor			core	core		
Waste Disposal						
High Level Wastes						
(HLW+Cladding hulls)						
Volume (ft^3)		115		115		115
Activity (10^6 Curies)		18.5		15.3		18.5
Activity of Pu (10^3 Curies)		20.5		189.0		20.5
Canisters (a 3.5 ft^3)		33		33		33
Low Level Wastes						
Volume (ft^3)		8,600-16,000		18,600-46,000		8,500-16,000
55 Gallon Drums		640-1,650		2,040-5,850		640-1,660
Burial or Repository Space ft^3		900-2,200		2,900-8,200		900-2,200

*Note: All reactors are assumed to have uranium recycled. This is not a necessary condition only an example of possible fuel schemes

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TAILS = 125.011 MTU
SWU = 125.015 MT
(0.2% U–235)

ENRICHMENT

FUEL FABRICATION

MIXED OXIDE FUEL FABRICATION

223.012 MT UF$_6$

150.8 L

25.78 MTU (3.3% U–235)

PLUTONIUM URANIUM STORAGE

227 kg Pu

0.5% WASTE LOSS

CONVERSION TO UF$_6$

25.52 MTU (3.3% U–235)

24.146 MTU (0.83% U–235) RECYCLE TO ENRICHMENT

REPROCESSING

1% WASTE LOSS

84 MT

\$ O*

52 MTM SPENT FUEL

115 ft^3 HLW
0.89 MT FP

5% WASTE LOSS

URANIUM MINING & MILLING

19.86 x 10^6 MT MATERIAL MINED

REACTOR

WASTE STORAGE

Fig. 24. PWR 1 Equilibrium Cycle (U Recycled)

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5% WASTE LOSS

URANIUM MINING & MILLING

3.84 x 10^6 MT MATERIAL MINED

CONVERISON TO UF_6

29.015 MT U_3 O_8 = 24.482 MTU

25.52 MT FUEL (0.71% U-235 +5.02% Pu)

FUEL FABRICATION

MIXED OXIDE FUEL FABRICATION

1% WASTE LOSS

1,294 kg Pu
875.3 kg FISSILE Pu

PLUTONIUM URANIUM STORAGE

407.2 kg FISSILE Pu MAKEUP FROM CENTRAL STOCK

REACTOR

25.52 MTM SPENT FUEL

TO STORAGE 23.43 MTU (0.325% U235)

TO STORAGE 824.7 kg Pu 468.1 kg FISSILE Pu

REPROCESSING

1% WASTE LOSS

115 ft^3 HLW 0.891 MT FP

WASTE STORAGE

Fig. 25. PWR 2 Equilibrium Cycle (Pu Recycle)

115

TAILS = 146.61 MTU
SWU = 163.31 MT
(0.2% U-235)

ENRICHMENT

1% WASTE LOSS

FUEL FABRICATION

MIXED OXIDE FUEL
FABRICATION

37.75 MTU
(3.4% U-235)

272.059 MT UF$_6$
183.96 MTU

PLUTONIUM
URANIUM STORAGE

36.62 MTU (3.4% U-235)

0.5% WASTE LOSS

CONVERSION TO UF$_6$

35.105 MTU
(1.44% U-235)
RECYCLE TO
ENRICHMENT

263.3 kg Pu

REPROCESSING

1% WASTE LOSS

176.538 MT
U$_3$O*

5% WASTE LOSS

URANIUM MINING & MILLING

23.35 x 10^6 MT
MATERIAL
MINED

REACTOR

36.62 MTM
SPENT
FUEL

115 ft^3 HLW
0.891 MT FP

WASTE STORAGE

Fig. 26. PWR ≡ Equi ibrium Cycle Recycled)

116

TABLE U3

FUEL CYCLE REQUIREMENTS (1000 MWe   75% LOAD FACTOR)

REACTOR CHARACTERIZATIONS	BWR1-U Fueled		BWR2-60% U Fueled and 40% nat.U+Pu Fueled	
	Initial Load	Equilibrium Cycle	Initial Load	Equilibrium Cycle
Mining (10^3 MT)				
Material Removed	89,564	18,432	20,213	12,376
Uranium 0.2% concentration	302.277	67.208	68.232	41.770
Thorium 0.2% concentration	0	0	0	0
Milling (MT U_3O_8)	672.276	139.381	152.876	93.589
(MT Th O_2)	0	0	0	0
Conversion (=Enrichment Feed) (MT UF_6)	845.150	219.217	0	143.970
Enrichment				
SWU (MT)	307.03	109.84		70.29
Tails (MTU at 0.2%)	429.03	115.390		77.24
Enriched Product (MTU)	142.44	32.84		19.513
Assay (% U-235)	2.25	2.61		2.61
Feed nat U (MTM)	572.47	117.507		66.55
Fuel Fabrication				
Input (MT)	142.44	32.84		19.513
Output (MT)	141.03	32.2		19.32
No. Assemblies/year	725	166		102
Composition	UO_2 enr.	UO_2 enr.		UO_2 enr.
Weight of Assembly (MTM)	0.194	0.194	0.185	0.185
Used in Reactor	core	core	60% core	60% core
Reactor Load in				
Quantity (MTM) U/Pu/Th	741.03/0/0	32.3/0/0	127.704/3.437/0	31.615/°.585/0
Isotopics (%)				
U-233	0	0	0	0
U-234	0.03	0.235	0	0.147
U-235	2.25	2.61	0.71	1.873
U-236	0	0.065	0.	0.040
U-238	97.72	97.09	99.29	97.94
Pu-239	0	0	67.7	42.05
Pu-240	0	0	18.8	27.35
Pu-241	0	0	10.0	15.35
Pu-242	0	0	2.51	11.83

TABLE 8 (Cont)
FUEL CYCLE REQUIREMENTS (1000 MWe, 75% LOAD FACTOR)

REACTOR CHARACTERIZATIONS	BWR1-U Fueled		BWR2-60% U Fueled and 40% nat.U+Pu Fueled	
	Initial Load	Equilibrium Cycle	Initial Load	Equilibrium Cycle
Reactor Discharge Quantity (MTM) U/Pu/Th		29.025/0.276/0		29.606/0.574/0
Isotopics				
U-233		0		0
U-234		0.121		0.076
U-235		0.822		0.631
U-236		0.425		0.292
U-238		98.631		99.00
Pu-239		60.55		43.58
Pu-240		23.93		25.97
Pu-241		10.66		15.88
Pu-242		3.25		1.12
Th		0		0
Fission Products (MTM)		0.891		0.891
Net Plant Efficiency %	32.5	32.5	32.5	32.5
Burnup (Avg. MWd/MTM)	19,250	26,160	19,250	26,160
Spent Fuel Storage				
Decay Heat (W/kg)		16		17
Days Holdup		150		150
Activity at Discharge From Pool (106 Curies F.P.)		89		92
Fuel Reprocessing Input (MT)		32.2		32.2
Recycle Uranium (MT)				
U-233		0		0
U-235		0.252		0.191
U-238		30.295		29.997
Plutonium Production (MT)				
Pu-239		0.165		0.248
Pu-241		0.029		0.090

TABLE 8 (Cont)
FUEL CYCLE REQUIREMENTS (1000 MWe. 75% LOAD FACTOR)

REACTOR CHARACTERIZATIONS	BWR1-U Fueled		BWR2-60% U Fueled 40% nat.U+Pu Fueled	
	Initial Load	Equilibrium Cycle	Initial Load	Equilibrium Cycle
Mixed Oxide Fuel Fabrication				
Input (MT) U/Pu	0	0	128.936/3.472	12.413/0.591
Output (MT)	0	0	131/141	12.68
No. Assemblies/year	0	0	709	72
Composition	-	-	nat.UO_2+PuO_2 2.62%)	nat.UO_2+PuO_2(4.54%)
Weight of Assembly (MTM)			0.185	0.185
Used in Reactor			100% core	40% core
Waste Disposal				
High Level Wastes (HLW+Cladding hulls)				
Volume (ft^3)		115		115
Activity 10^6 Curies)		18.5		15.8
Activity of Pu (10^3 Curies)		20.5		189.0
Canisters (a 3.5 ft^3)		33		33
Low level Wastes				
Volume (ft^3)		8,600-16,000		18,600-46,000
55 Gallon Drums		640-1,660		2,040-5,260
Burial or Repository Space ft^3		900-2,200		2,900-8,200

TABLE 9
HEAVY WATER REACTORS

Reactor Characterization / Date of Information / Representative Reactor	PHW 1973 CANDU-600 Gentilly 2	HW-BLW 1972 CANDU-BLW Gentilly	SGHWR 1967 UKAEA Design	HWSLW 1967 WASH-1085	HWOCR 1968 NUC-WASH 1083
Reactor Thermal Power (MWt)	600	250	625	965	1118
Net Electrical Power (MWe)	600 UY	50.0			NA
Net Plant Efficiency (%)			32.9	29.8	34.2
Average Burnup MWD/MTM — Initial	NA	NA	12700	NA	NA
Average Burnup MWD/MTM — Equilibrium	7500	7000	20700	8000	8000
Core Inventory (MTM)	85.82	65	100.1	X9	C4
Core Height (m)	5.94	5.00	4.68	7.62	5.42
Fuel Description — Number of Assemblies (core/blanket)	4560/-- 12 assemblies/channel	308	532	3440 5 assemblies/channel	assemblies/channel
Dimension of Assembly (h×d)m	.495×.102	.50×.102	3.08×.13	1.52×.102	1.12×.094
Number of Rods per Assembly (core/blanket)	37/--	18/--	36/--	19/--	19/--
Chemical Composition (core/blanket)	UO$_2$/--	UO$_2$/--	UO$_2$/--	UO$_2$/--	UC/--
Cladding Material / Thickness (cm)	Zr-4 / .049	Zr-4 / .059	Zr-4 / NA	Zr-4 / .076	SAP / .076
Enrichment (%) (Initial/Equilibrium)	Natural Uranium	Natural Uranium	(2.139 avg./2.105 avg.)	Natural Uranium	Natural Uranium
Control Material	21 SS adjuster rods 4 cadmium absorber rods 20 cadmium-SS rods gadolinium in solution	7 cadmium rods 16 booster rods flex regulation	boric acid	90 absorber rods 24 booster rods flow regulation	158 B$_4$C SS shutdown rods 32 B$_4$C SS rod clusters
Control Rods/Assembly					
Refueling Interval	Continuous	Continuous	1/2 year	Continuous	Continuous
Fraction Reload per Cycle	.96 core/yr J=92% C.F. 2 core/yr at 80% C.F.	Continuous	.046	0.5 core/year	.49 at 80% C.F.
Conversion Ratio (Initial/Equilibrium)	(--/0.6)	(NA/NA)	(.56/NA)	(NA/.74)	(NA/0.70)

120

TABLE 10

REACTOR CHARACTERIZATIONS

	CANDU-PHW		HW-BLW		HWOCR(nat UC)	
	Initial Load	Equilibrium Cycle	Initial Load	Equilibrium Cycle	Initial Load	Equilibrium Cycle
Mining (10^3 MT)						
Material Removed	22530	19430	39600	18260	33740	15700
Uranium 0.2% concentration	76.05	65.59	133.7	61.63	113.9	62.0
Thorium 0.2% concentration	0	0	0	0	0	0
Milling (MT U$_3$O$_8$)	170	147	300	129.1	255.1	118.7
(Th(NO$_3$)$_4$)	0	0	0	0	0	0
Conversion (=Enrichment Feed) (MT UF$_6$)	0	0	0	0	0	0
Enrichment						
SWU (MT)	0	0	0	0	0	0
Tails (MTU at 0 %)	0	0	0	0	0	0
Enriched Product (MTU)	0	0	0	0	0	0
Assay (% U-235)	0	0	0	0	0	0
Feed nat U (MTM)	0	0	0	0	0	0
Fuel Fabrication						
Input (MT)	144.5 U	124.6 U	254.0 U	117.1 U	216.4 U	100.7 U
Output (MT)	143.0 U	123.3 U	252.0 U	115.9 U	214.3 U	99.66 U
No. Assemblies/year						
Composition	UO$_2$	UO$_2$	UO$_2$	UO$_2$	UC	UC
Weight of Assembly (MTM)						
Used in Reactor	.019	.019	.071	.071	.052	.052
Reactor Load in						
Quantity (U/Pu/Th) (MTM)	143.0/-/-	123.31/-/-	252.0/-/-	115.9/-/-	214.3/-/-	99.68/-/-
Isotopics (%)						
U-233	0	0	0	0	0	0
U-234	0	0	0	0	0	0
U-235	.711	.711	.711	.711	.711	.711
U-238	0	0	0	0	0	0
U-239	99.29	99.29	99.29	99.29	99.29	99.29
Pu-239	0	0	0	0	0	0
Pu-240	0	0	0	0	0	0
Pu-241	0	0	0	0	0	0
Pu-242	0	0	0	0	0	0

TABLE o (Cont)

REACTOR CHARACTERIZATIONS	CANDU-PHW		HW-BLW		HWOCR (nat UC)	
	Initial Load	Equilibrium Cycle	Initial Load	Equilibrium Cycle	Initial Load	Equilibrium Cycle
Reactor Discharge						
Quantity (U/Pu/Th) (MTM)		121.8/.41/--		114.5/.421.--		98.46/.367/--
Isotopics						
U-233						.208
U-234						.0812
U-235		.22		.247		
U-236				.078		
U-128		99.78		99.7		99.7
Pu-239		67.0		55.1		59.9
Pu-240		26.0		29.5		28.9
Pu-241		4.7		11.2		8.72
Pu-242		1.6		3.22		2.46
Th		0.0		0.0		0.0
Fission Products (MT)		973		.971		.842
Net Plant Efficiency (%)		29.6	29.8	29.8	34.2	.842
Burnup (Avg. MWd/MTM)		7500	NA	8000	8030	8030
Spent Fuel Storage						
Decay Heat (W/kg)		--	--	--	--	--
Days Holdup		indefini[a]		indefinite		indefinite
Activity at Discharge From Pool (10^6 Curies F.P.)		--		--		--
Fuel Reprocessing (Possible)						
Input (MT)		122.5		115.3		99.68
Recycle Uranium (MT)						
U-233		--		--		--
U-235		.268		.252		.205
U-238		121.5		113.0		97.27
Fissile Plutonium Production (MT)						
Pu-239		.280		.234		.220
Pu-241		.020		.047		.032

TABLE 10 (Cont)

REACTOR CHARACTERIZATIONS	CANDU-PHW		HW-BLW		HWOCR (nat UC)	
	Initial	Equilibrium	Initial	Equilibrium	Initial	Equilibrium
Mixed Oxide Fuel Fabrication						
Input (MT) U/Pu						
Output (MT)		NO MIXED OXIDE FABRICATION				
No. Assemblies/year						
Composition						
Weight of Assembly (MTM)						
Used in Reactor						
*Waste Disposal						
High Level Wastes						
(HLW+Cladding hulls)						
Volume (ft^3)						
Activity (10^6 Curies)						
Activity of Pu (10^3 Curies)						
Canisters (a 3.5 ft^3)						
Low Level Wastes						
Volume (ft^3)						
55 Gallon Drums						
Burial or Repository						
Space (ft^3)						

124

TAILS = 115.390 MTU
SWU = 109.84 MT
(0.2% U-235)

ENRICHMENT

FUEL FABRICATION

MIXED OXIDE FUEL
FABRICATION

219.217 MT UF$_6$
148.321 MT U

0.5% WASTE LOSS

CONVERSION TO UF$_6$

139.381 MT
O*
3

5% WASTE LOSS

URANIUM MINING & MILLING

18.43 x 10^6 MT
MATERIAL
MINED

REACTOR

32.2 MTU (2.61% U-235)

30.714 MTU
(0.822% U-235)
RECYCLE TO
ENRICHMENT

PLUTONIUM
URANIUM STORAGE

273.2 kg Pu

REPROCESSING

1% WASTE LOSS

32.2 MTMT
SPENT
FUEL

115 ft^3 HLW
0.891 MT FP

WASTE STORAGE

Fig. 27. BWR Equilibrium Cycle (U Recycled)

125
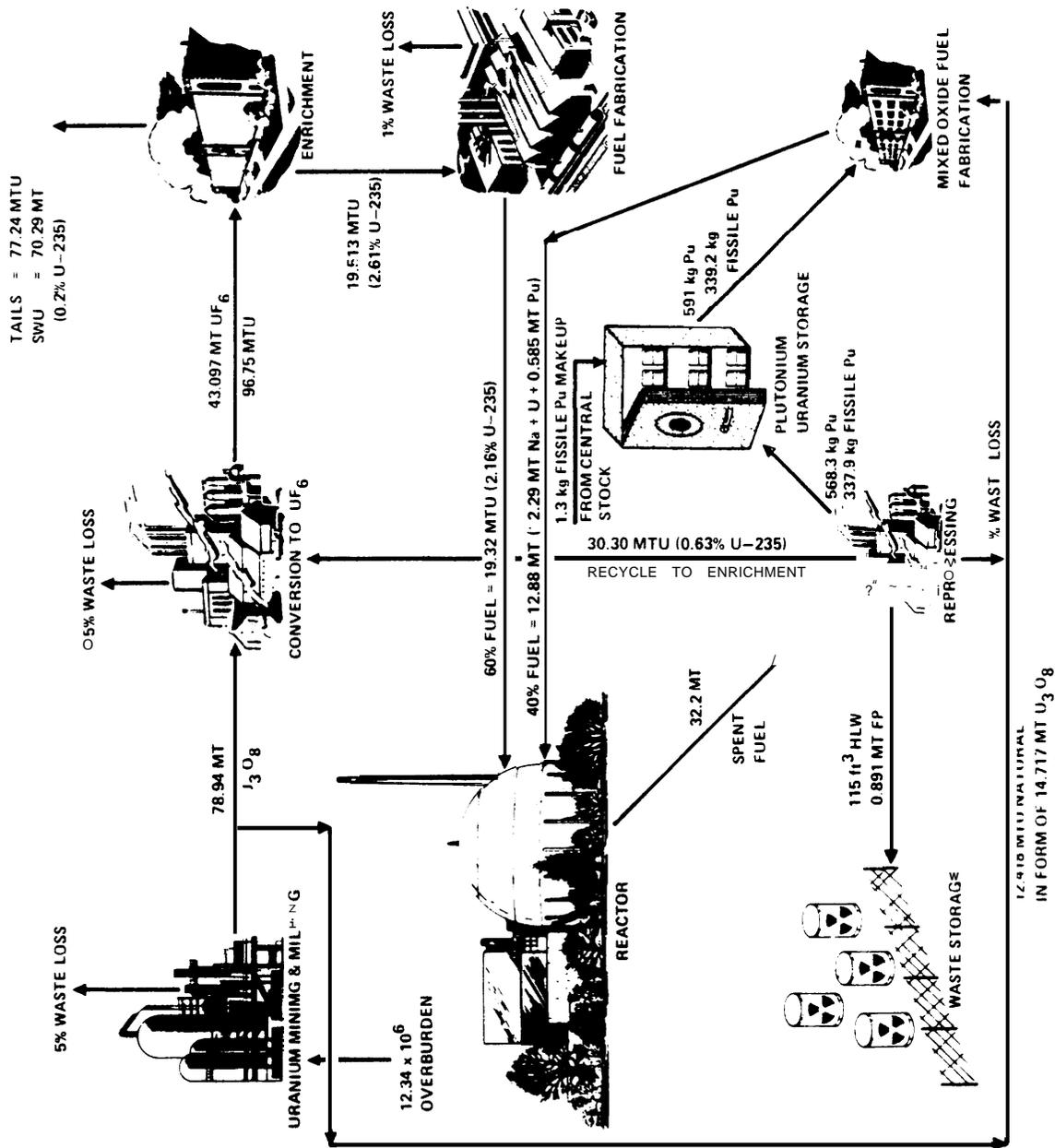
Fig. 28. BWR 2 Equilibrium Cycle (U and Pu Recycle)

TAILS = 77.24 MTU
SWU   = 70.29 MT
          (0.2% U-235)

ENRICHMENT

1% WASTE LOSS

FUEL FABRICATION

19.513 MTU
(2.61% U-235)

MIXED OXIDE FUEL
FABRICATION

591 kg Pu
339.2 kg
FISSILE Pu

43.097 MT UF$_6$

96.75 MTU

1.3 kg FISSILE Pu MAKEUP
FROM CENTRAL
STOCK

PLUTONIUM
URANIUM STORAGE

60% FUEL = 19.32 MTU (2.16% U-235)

40% FUEL = 12.88 MT (2.29 MT Na + U + 0.585 MT Pu)

568.3 kg Pu
337.9 kg FISSILE Pu

% WAST LOSS

05% WASTE LOSS

CONVERSION TO UF$_6$

30.30 MTU (0.63% U-235)
RECYCLE TO ENRICHMENT

REPROCESSING

?"

78.94 MT

U$_3$O$_8$

32.2 MT

SPENT
FUEL

115 ft^3 HLW
0.891 MT FP

5% WASTE LOSS

URANIUM MINING & MILLING

12.34 x 10^6
OVERBURDEN

REACTOR

WASTE STORAGE

IN FORM OF 14.717 MT U$_3$O$_8$

126

ENRICHMENT

1% WASTE LOSS

FUEL FABRICAT

MIXED OXIDE FUEL
FABRICAITON

CONVERSION TO UF$_6$

144.5 MTU

SPENT FUEL
STORAGE

NATURAL URANIUM
143.0 MTU

REPROCESSING

5% WASTE LOSS

URANIUM MINING & MILLING

22.53 x 10^6 MT OVERBURDEN

REACTOR

WASTE STORAGE

Fig. 29. CANDU-PHW Startup Loading

127

ENRICHMENT

1% WASTE LOSS

FUEL FABRICATION

MIXED OXIDE FUEL FABRICATION

CONVERSION TO UF$_6$

124.56 MTU

123.31 MTU (NATURAL)

SPENT FUEL STORAGE

REPROCESSING

5% LOSS

URANIUM MINING & MILLING

19.43 × 10^6 MT

CANDU 600 1000 MWe

REACTOR

121.6 MTU .973 MT F.P. .41 MT Pu

WASTE STORAGE
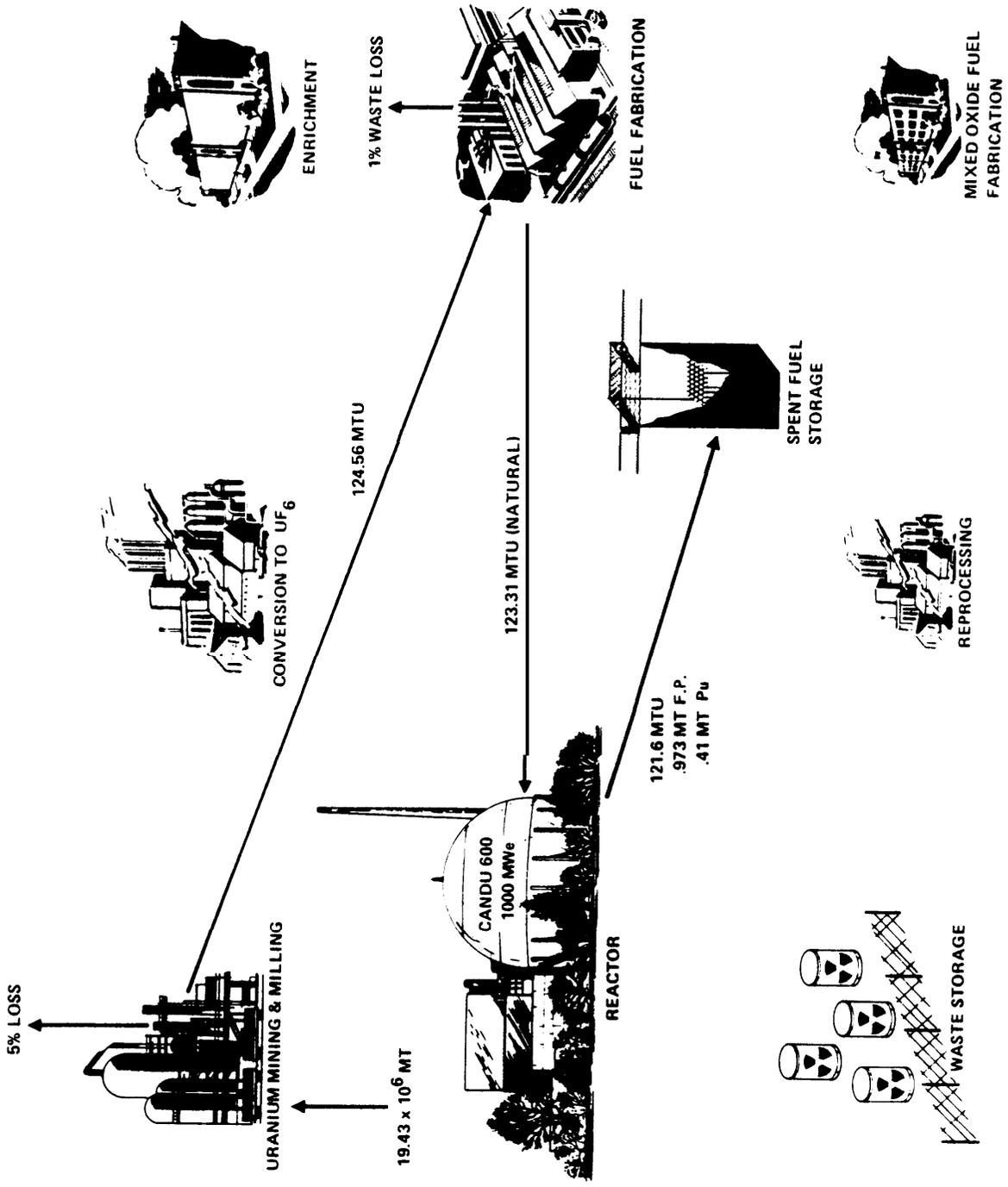
Fig. 29. (Cont). CANDU-PHW Equilibrium Cycle

128

ENRICHMENT

1% WASTE LOSS

FUEL FABRICATION

MIXED OXIDE FUEL
FABRICATION

CONVERSION TO UF$_6$

117.1 MTU

SPENT FUEL
STORAGE

REPROCESSING

115.9 MTU
NATURAL URANIUM

114.5 MTU
.421 MT Pu
.971 MT F.P.

5% WASTE LOSS

URANIUM MINING & MILLING

18.2 × 10^6
OVERBURDEN

REACTOR

WASTE STORAGE

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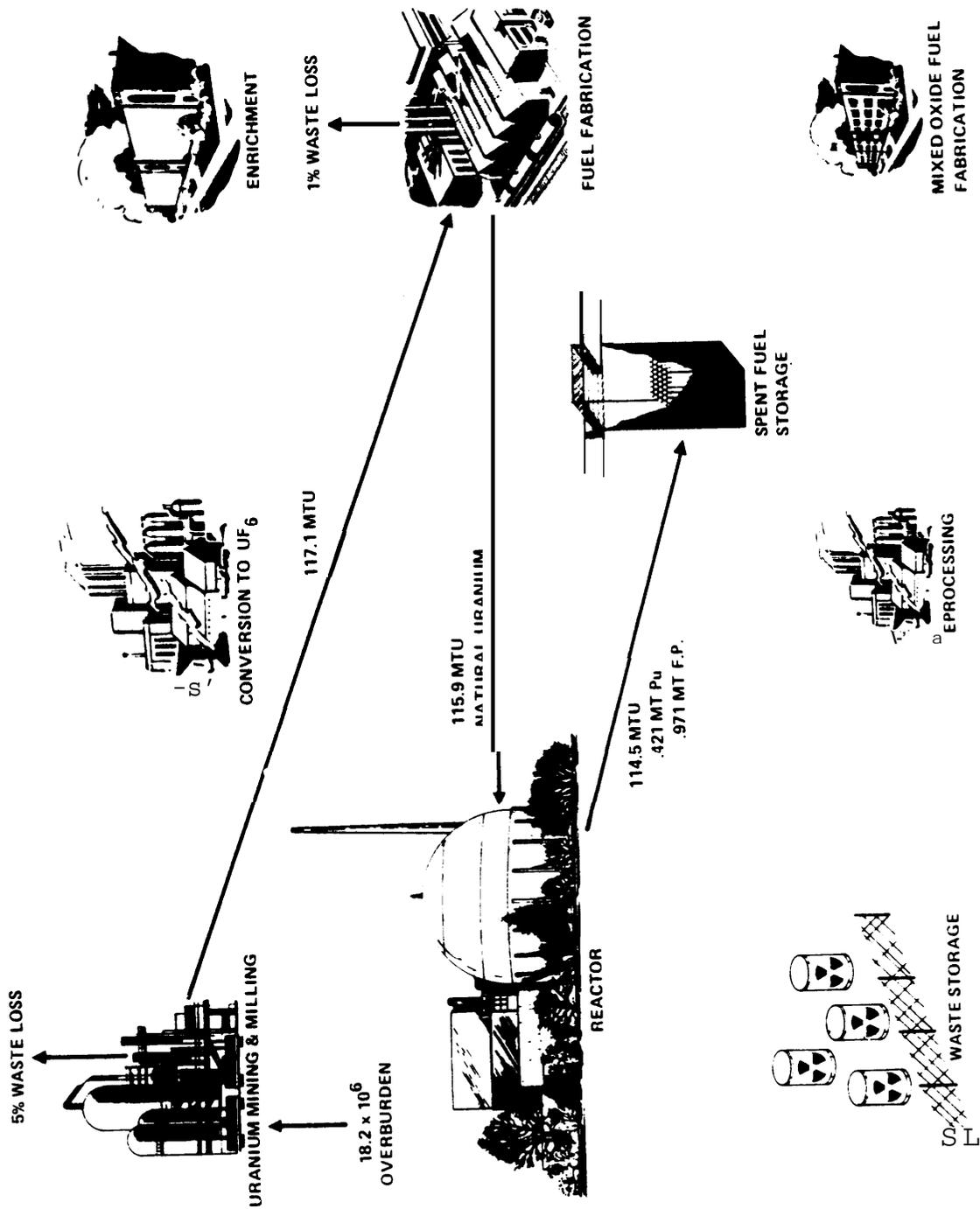
Fig. 30.   HW-13LW Equilibrium

129

ENRICHMENT

1% WASTE LOSS

FUEL FABRICATION

MIXED OXIDE FUEL FABRICATION

100.7 MTU

CONVERSION TO UF

99.7 MTU

SPENT FUEL STORAGE

98.46 MT U
.367
.842 MT F.P.

REPROCESSING

5% WASTE LOSS

URANIUM MINING & MILLING

15.70 x 10⁶ MT OVERBURDEN

REACTOR

WASTE STORAGE

Fig. 31. HWOCR Equilibrium (Natural Uranium Carbide Fuel)

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TABLE 11
GAS-COOLED REACTORS

Reactor Characterization	GCFR (34) 1976	HTGR (30) 1974	AGR (21) 1976	THTR (38,39) 1972
Date of Information				
Representative Reactor	GA 1500 MWe Design	Fulton Station	Hinkley-Point B	THTR Uentrop
Reactor Thermal Power (MWt)	3997	3000	1493	750
Net Electrical Power (MWe)	1500	1160	621	300
Net Plant Efficiency (%)	37.5	38.6	41.6	40.0
Average Burnup (MWd/MTM)				
Initial	NA	NA	11,000	NA
Equilibrium	88,400/C200/2100	98,600	18,000	113,000
Core Inventory (MTM)	125.2	39.22	114	6.56
Core Height (m)	2.59	6.30	6.3	5.1
Neutron Flux (n/cm²-sec)				
Peak Thermal	NA	NA	13×10^{13}	19×10^{13}
Average Thermal	NA	NA	8×10^{13}	13×10^{13}
Peak Fast	NA	NA	NA	NA
Average Fast	NA	NA	NA	NA
Fuel Description				
Number of Assemblies (core/blanket)	271/126	3944 0 assemblies of 493 columns	2464 assemblies 8 assemblies per channel	74,200
Dimension of Assembly (hxt)a	2.50x.214	.79x.36	1.039x.238	6 cm diam. spheres
Number of Rods per Assembly (core/blanket)	331/61	102/--	36/--	NA
Chemical Composition (core/blanket)	UO_2/PuO_2/UO_2	UC-UO_2-ThO_2/--	UO_2/--	UC-UO_2-ThO_2/--
Cladding Material/ Thickness (cm)	SS 316 0.37 core .057 radial blanket	TRISO/BISO coating .17/.09	Stainless Steel .038	Pyrolytic Carbon .50
Enrichment (%) (Initial/Equilibrium)	15.4/ 5.4	93 U^{235}	(1.46,1.75/2.10,2.54)	93 U^{235}
Control Material	B_4C	B_4C	Boron	B_4C
Control Rods/Assembly	37	2 in each of 73 channels	Z control channels	36 control rods 42 safety rods
Refueling Interval	Annually		Continuous	Continuous
Fraction Reload = Cycle	1/3 core 1/4 radial blanket	1/4 core	≈ 1½ channels/week	1.a cores/year
Conversion Ratio (Initial/Equilibrium)	(NA/1.44)	(NA/.67)	(0.6/0.5)	(NA/0.53)

HTGR

PEBBLE BED
REACTOR
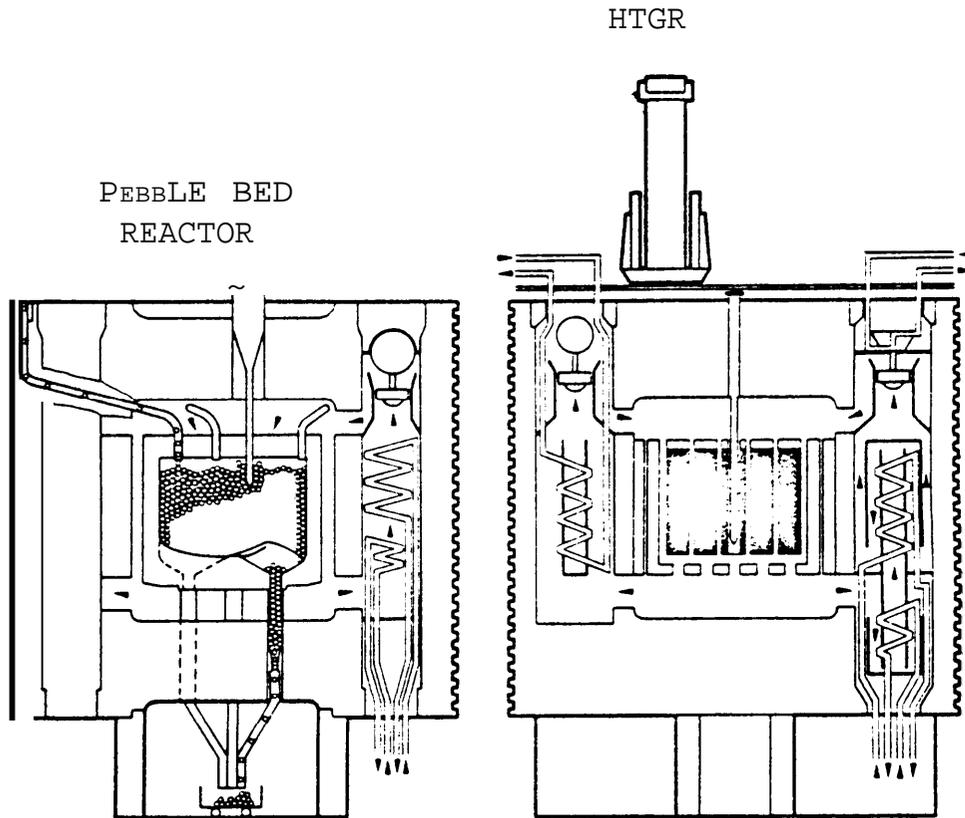
Fig. 32.   Comparison of the main systematic dif -
            ferences of two typical designs with
            pebbles and prismatic fuel.

TABLE 12

FUEL CYCLE REQUIREMENTS (100⁰ MWe, 75% LOAD FACTOR)

REACTOR CHARACTERIZATIONS	HTGR		AGR	
	Initial Load	Equilibrium Cycle	Initial Load	Equilibrium Cycle
Mining (10³ MT)				
Material Removed	44,940	12,335	78,000	20,750
Uranium 0.2% concentration	136.4	37.613	263.3	70.04
Thorium 0.?% concentration	16.27	3.066		
Milling (MT U₃O₈)	303.4	84.71	590.0	156.9
(Th(NO3)4)	63.38	15.52	0.0	0.0
Conversion (=Enrichment Feed) (MT U)L9	378.1	105.6	735.6	195.6
Enrichment				
SWU (MT)	332.1	32.72	247.3	89.5
Tails (MTU at 0.2%)	254.6	71.09	311.8	99.0
Enriched Product (MTU)	1.408	.393	186.1	33.3
Assay (% U-235)	93.12	93.12	1.46/1.75	2.10/2.54
Feed nat U (MTM)	256.0	71.48	497.9	132.4
Fuel Fabrication				
Input (MT)	d.408U/30.62 Th	.292U/4.011 Th	186.1	33.3
Output (MT)	1.394U/30.31 Th	.389U/3.92 Th	184.2	33.0
No. Assemblies/year	3188	800	4000	720
Composition	ThO₂-UC	ThO₂-UC	UO₂	UO₂
Weight of Assembly (MTM) Used in Reactor	.010	.010	.075	.075
Reactor ——m in				
Quantity (U/Pu/Th) MTM	1.394/--/30.31	.794/--/7.349	184.2/--/--	33.0/--/--
Isotopics (%)				
U-233	0	21.49	0.0	0.0
U-234	.74	5.19	0.0	0.0
U-235	93.12	62.63	1.46/1.75(a)	2.10/2.54
U-236	.28	4.07	0.0	0.0
U-238	5.96	6.62	93.54/98.25(a)	97.9/97.46
Pu-239	0	0	0.0	0.0
Pu-240	0	0	0.0	0.0
Pu-241	0	0	0.0	0.0
Pu-242	0	0	0.0	0.0

TABLE 2 Cont

FUEL CYCLE REQUIREMENTS (1000 MWe, 75% LOAD FACTOR)

REACTOR CHARACTERIZATIONS	HTGR Initial Load	HTGR Equilibrium Cycle	AGR Initial Load	AGR Equilibrium Cycle
Reactor Discharge Quantity (U/Pu/Th MTM)		.498/-/6.901		32.15/ 16/-
Isotopics				
U-233		38.75		--
U-234		12.81		--
U-235		10.67		.59/.78
U-236		29.55		--
U-238		8.21		99.41/99.22
Pu-239		0		74.
Pu-240		0		17.
Pu-241		0		8.
Pu-242		0		0.4
Th		100		
Fission Products (MT)		.744		.690
Net Plant Efficiency (%)		38.7	41.6	41.6
Burnup (Avg. MWd/MTM)		87,000	13,000	20,000
Spent Fuel Storage				
Decay Heat (W/kg)		1220		10.6
Days Holdup		180		130
Activity at Discharge From Pool (106 Curies F.P.)		78.5		135
Fuel Reprocessing Input (MT)		3.53 U		33.0 (a)
Recycle Uranium (MT)				
U-233		.174		--
U-235		.048		.216
U-238		.018		31.93
Plutonium Production MT)				
Pu-239	0	0		.120
Pu-241				.013

TABLE 12 (Cont)

FUEL CYCLE REQUIREMENTS ≠1000 MWe. 75% LOAD FACTOR)

REACTOR CHARACTERIZATIONS	HTGR		AGR	
	Initial Load	Equilibrium Cycle	Initial Load	Equilibrium Cycle
Mixed Oxide Fuel Fabrication				
Input (MT) U/Pu		.387U/3.412 Pu		— —
Output (MT)		.386U/3.378 Pu		— —
No. Assemblies/year		NA		— —
Composition		ThO_2-UO_2		—
Weight of Assembly (MTM)				
Used in Reactor		.010		HH
Waste Disposal				
High Level Wastes				
(HLW+Cladding hulls)				
Volume (ft^3)		71.		218.7
Activity (105 Curies)		35		NA
Activity of Pu (103 Curies		.26		NA
Canisters (a 3.5 ft^3)		20.3		62.49
Low Level Wastes				
Volume (ft^3)		10,000-30,000		10,000-30,000
55 Gallon Drums		1,400-4,200		1,400-4,200
Burial or Repository		2,000-6,000		2,000-6,000
Space (ft^3)				

Notes: (a) No plutonium recycle is assumed in the AGR fuel cycle. However, quantities of recycle plutonium are presented to indicate the amounts present.

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Fig. 33.   HTGR Start-up

Labels within figure:

254.6 MTU TAILS
SWU = 332.1 MT

ENRICHMENT

1.408 U

1% WASTE LOSS

INITIAL AND MAKEUP
FUEL FABRICATION

REFABRICATION

≡78.1 MT UF$_6$

30.62 Th

0.5% WASTE LOSS

0.5% LOSS

CONVERSION TO UF$_6$

THO$_2$ MICROSPHERE FABRICATION

303.4 MT U$_3$O$_8$

63.38 MT
th (NO$_3$)$_4$

1.394 MT U
30.31 MT Th

5% WASTE LOSS

5% WASTE LOSS

URANIUM MINING & MILLING

THORIUM MINING & MILLING

42.8 × 10^6 MT
OVERBURDEN

5.04 × 10^6 MT
OVERBURDEN

REACTOR

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Fig. 33. (Cont). HTGR Ninth Year Conditions

11.09 MTU TAILS
SWU = 92.71 MT

ENRICHMENT

.393 MTU
AT 93.12% U235

INITIAL AND MAKEUP
FUEL FABRICATION

1% LOSS

3.41 MT Th

REFABRICATION

1% WASTE LOSS

105.56 MT UF$_6$

4.014 MT Th

7.42 MT Th

.358 MT U

0.5% WASTE LOSS

CONVERSION TO UF$_6$

ThO$_2$ MOCROSPHERE
FABRICATION

3.378 MT Th
.386 MT U

REPROCESSING

1% WASTE LOSS

84.71 MT U$_3$O$_8$

15.5 MT
Th(NO$_3$)$_4$

3.97 MT Th
.389 MT U

.498 MT U
6.901 MT FP
.744 MT FP

5% WASTE LOSS

URANIUM MINING & MILLING

5% WASTE LOSS

THORIUM MINING & MILLING

69 MT FP
6.9 MT Th

11.21 x 10^6 MT
OVERBURDEN

1.18 x 10^6 MT
OVERBURDEN

WASTE STORAGE

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Fig. 34. AGR Initial Loading

TAILS = 311.8 MTU(.2% U 235)
SWU = 247.3 MT

1% WASTE LOSS

ENRICHMENT

FUEL FABRICATION

MIXED OXIDE FUEL FABRICATION

735.6 MT UF$_6$
497. MTU

186.1 MTU
119.0 MTU(1.46% U−235)
67.1 MTU(1.76% U−235)

0.5% WASTE LOSS

CONVERSION TO UF$_6$

SPENT FUEL STORAGE

REPROCESSING

590 MT U$_3$O$_8$

184.2 MTU

5% WASTE LOSS

URANIUM MINING & MILLING

REACTOR

WASTE STORAGE
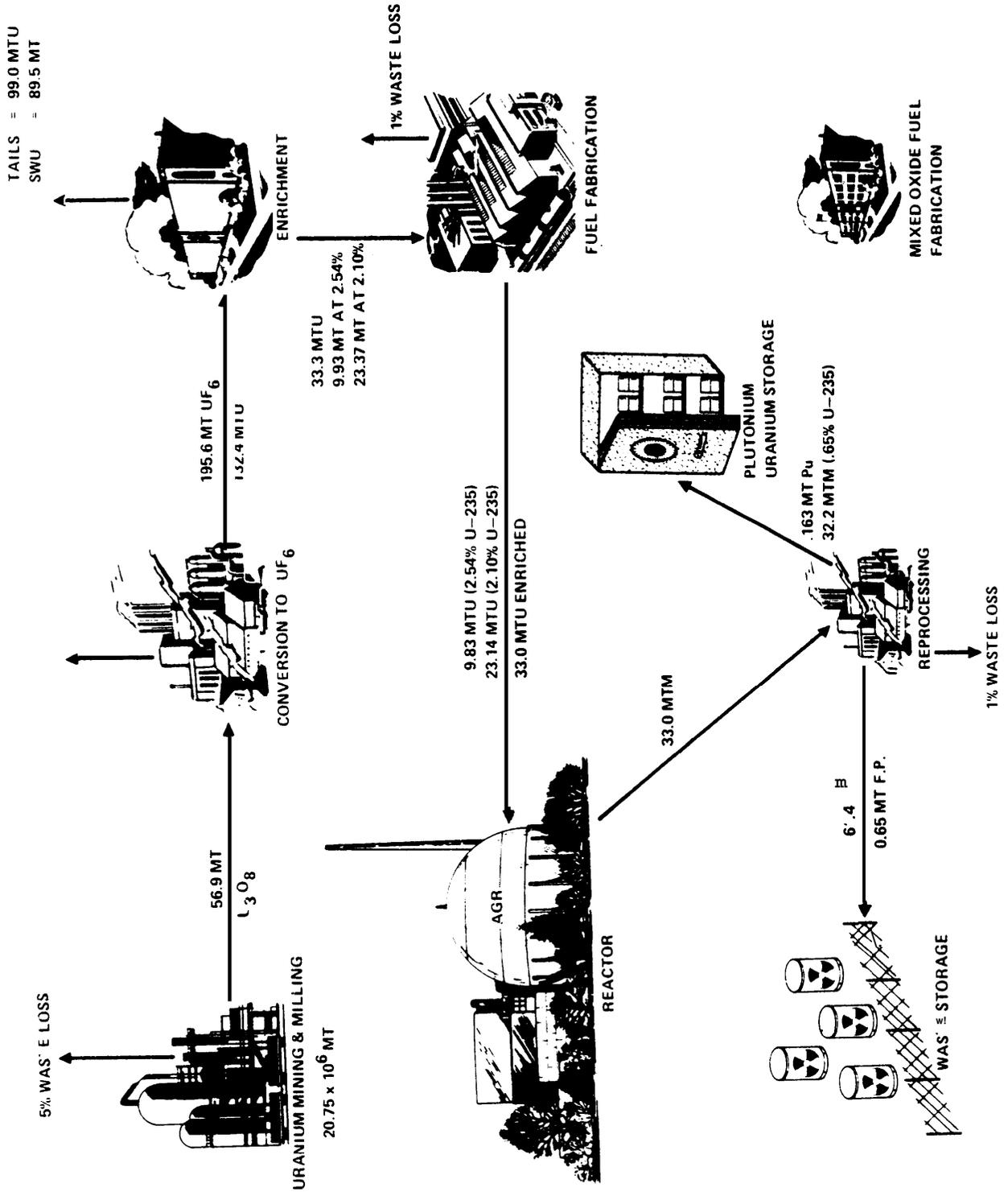
Fig. 34. (Cont) Equilibrium AGR Fuel Cycle (No Recycle is Assumed)

TAILS = 99.0 MTU
SWU = 89.5 MT

ENRICHMENT

1% WASTE LOSS

FUEL FABRICATION

MIXED OXIDE FUEL FABRICATION

195.6 MT UF$_6$
152.4 MTU

33.3 MTU
9.93 MT AT 2.54%
23.37 MT AT 2.10%

CONVERSION TO UF$_6$

56.9 MT
U$_3$O$_8$

5% WASTE LOSS

URANIUM MINING & MILLING
20.75 x 10^6 MT

9.83 MTU (2.54% U−235)
23.14 MTU (2.10% U−235)
33.0 MTU ENRICHED

PLUTONIUM URANIUM STORAGE

.163 MT Pu
32.2 MTM (.65% U−235)

33.0 MTM

AGR

REACTOR

REPROCESSING

1% WASTE LOSS

6.4 m
0.65 MT F.P.

WASTE STORAGE

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TABLE 13.

REACTOR CHARACTERIZATIONS

GCFR Uranium Blanket[4]

	High Burn-up Plutonium		Low Burn-up Plutonium	
	Initial Load	Equilibrium Cycle	Initial Load	Equilibrium Cycle
Mining (10^3 MT)				
Material Removed	0	0	0	0
Uranium 0.2% concentration	0	0	0	0
Thorium 0.2% concentration	0	0	0	0
Milling (MT U$_3$O$_8$)	0	0	0	0
(Th(NO$_3$)$_4$)	0	0	0	0
Conversion (=Enrichment Feed)	0	0	0	0
(MT UF$_6$)				
Enrichment				
SWU (MT)	0	0	0	0
Tails (MTU at 0.2%)	0	0	0	0
Enriched Product (MT)	0	0	0	0
Assay (% U-235)	0	0	0	0
Feed nat U (MTM)	0	0	0	0
Fuel Fabrication				
Input (MT)	54.86	13.71	35.96 U	8.87 U
Output (MT)	54.31	13.58	35.60 U	8.78 U
No. Assemblies/year	180	60	NA	NA
Composition	UO$_2$	UO$_2$	UO$_2$	UO$_2$
Weight of Assembly (MTM)	.287	.292	NA	NA
Used in Reactor				
Re or Load in				
Quantity (U/Pu/Th) (MTM)	41.38/4.94/--	25.94/1.65/--	68.70/4.041/--	19.81/1.35/--
Isotopics (%)				
U-233	0.0	0.0	0.0	0.0
U-234	0.0	0.0	0.0	0.0
U-235	0.25	0.25	0.25	0.25
U-236	0.0	0.0	0.0	0
U-238	99.75	99.75	99.75	99.75
Pu-239	54.8	54.8	66.9	6.9
Pu-240	25.0	25.0	26.1	26.0
Pu-241	14.1	14.1	5.0	5.0
Pu-242	6.1	6.1	2.0	2.0
Th	0.0	0.0	0.0	0

TABLE 13. (continued)

GCFR Uranium Blanket

REACTOR CHARACTERIZATIONS	High Burn-up Plutonium		Low Burn-up Plutonium	
	Initial Load	Equilibrium Cycle	Initial Load	Equilibrium Cycle
Reactor Discharge				
Quantity (U/Pu/Th) (MTM)	24.86/1.96/--			18.69/1.69/--
Isotopics				
U-233		--		--
U-234		--		--
U-235		.19		.19
U-236		--		--
U-238		99.81		99.81
Pu-239		65.6		71.46
Pu-240		21.8		23.57
Pu-241		7.3		3.30
Pu-242		5.3		1.68
Th		0.0		
Fission Products (MT)		.77		.77
Net Plant Efficiency ( )	37.5	37.5	37.5	37.5
Burnup (Avg. MWd/MTM)	NA	88,400 in core	NA	88,400 in core
Spent Fuel Storage				
Decay Heat (W/kg)		160		NA
Days Holdup		30		30
Activity at Discharge From Pool (10^9 Curies F.P.)		400		400
Fuel Reprocessing				
Input (MT)		27.59		21.16
Recycle Uranium (MT)				
U-233		0.0		0.0
U-235		.048		.034
U-238		24.56		18.47
Plutonium Production (MT)				
Pu-239		1.28		1.20
Pu-241		.143		.055

TABLE 13. (continued)

REACTOR CHARACTERIZATIONS

	GFCR Uranium Blanket			
	High Burn-up Plutonium		Low Burn-up Plutonium	
	Initial Load	Equilibrium Cycle	Initial Load	Equilibrium Cycle
Mixed Oxide Fuel Fabrication				
Input MT U/Pu	37.44U/4.96Pu 37.07U/ 4.91Pu	12.49U/1.67Pu 12.37U/1.65Pu	37.45U/4.99Pu 37.08U/4194Pu	11.14U/1.36Pu 11.03U/1.35Pu
Output (MT)	181	46		NA
No. Assemblies/year Composition	UO_2-PuO_2	UO_2-PuO_2	UO_2-.9...	UO_2-PuO_2
Weight of Assembly (MTM Used in Reactor	.235	.238	N^4	N^4
Waste Disposal				
High Level Wastes (HLW+Cladding hulls)				
Volume (ft³)		215		215
Activity 10⁶ Curies)		525		525
Activity of Pu (10³ Curies)		~70		61
Canisters (a 3.5 ft³)		62		62
Low Level Wastes				
Volume (ft³)		10,000-30,00ᵖ		10,000-30,000
55 Gallon Drums		1,400-4,200		1,400-4,200
Burial or Repository Space (ft³)		2,000-6,000		2,000-6,000

Fig. 35. GCFR Startup High Burn-up Pu)

Fig. 35. (Cont). GCFR Equilibrium Cycle Uranium Blanket, High Burn-up Pu)

URANIUM MINING & MILLING

CONVERSION TO UF$_6$

ENRICHMENT
26.20 MTU TAILINGS

1% WASTE LOSS

FUEL FABRICATION

13.71 MTU

12.49 MTU

1% WASTE LOSS

MIXED OXIDE FUEL
FABRICATION

RADIAL BL 2×FT 1358 MT

CORE AND AXIAL BLANKET
2.36 MTU
1.65 MT Pu

REACTOR

24.86 MTU
1.96 MT Pu

1.67 MT Pu

PLUTONIUM
URANIUM STORAGE

1.94 MT Pu
24.61 MTU

REPROCESSING

1% WASTE LOSS

.769 MT a
73 cu ft

WASTE STORAGE

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ENRICHMENT

69.39 MTU
.25% U-235
TAILS

FUEL FABRICATION

35.96 MTU

33.43 MTU

1% WASTE LOSS

MIXED OXIDE FUEL
FABRICATION

1% WASTE LOSS

1% WASTE LOSS

CONVERISON TO UF$_6$

RADIAL BLANKET   35.60 MTU

CORE AND AXIAL BLANKET   4.04 MT Pu

33.096 MTU

PLUTONIUM
URANIUM STORAGE

4.082 MT Pu

REPROCESSING

URANIUM MINING & MIL

REACTOR

WASTE STORAGE

Fig. 36.   GCFR Startup (Low Burn-up Pu, Uranium Blankets)

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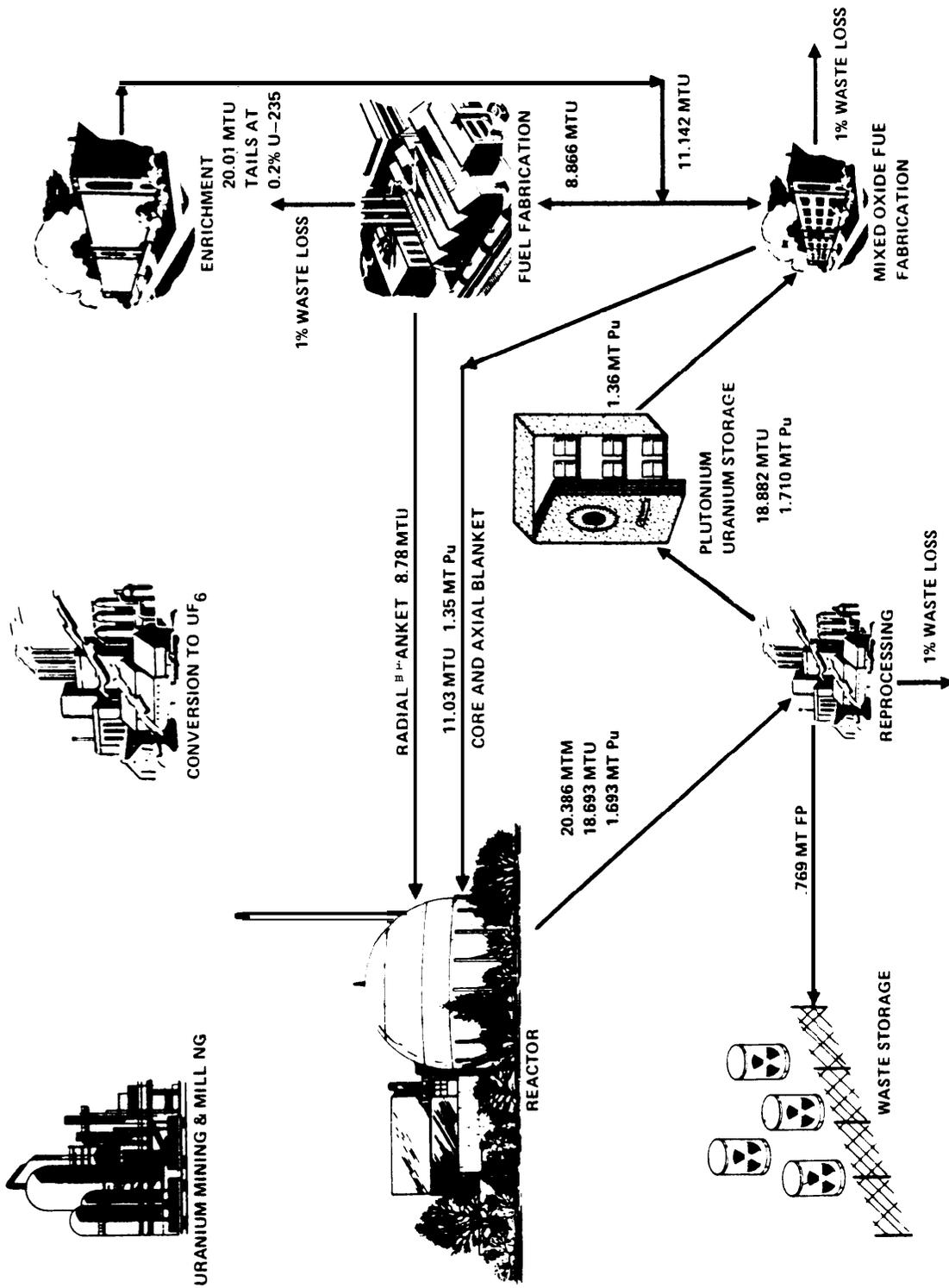
Fig. 36. (Cont). GCFR Equilibrium Cycle (Uranium Blanket. Low Burn-up Pu)
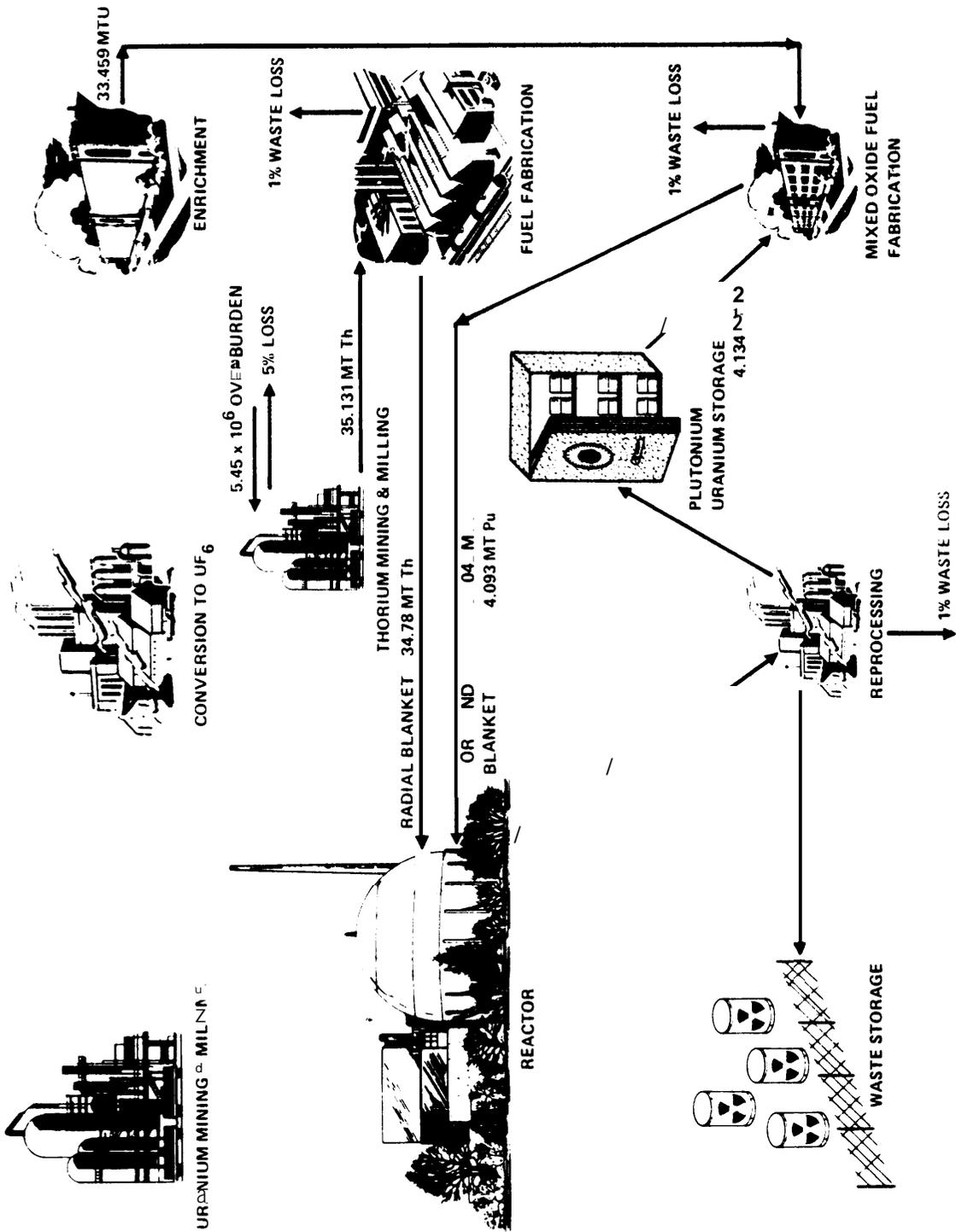
Fig. 37. GCFR Startup (Thorium Blanket Low Burnup Pu)
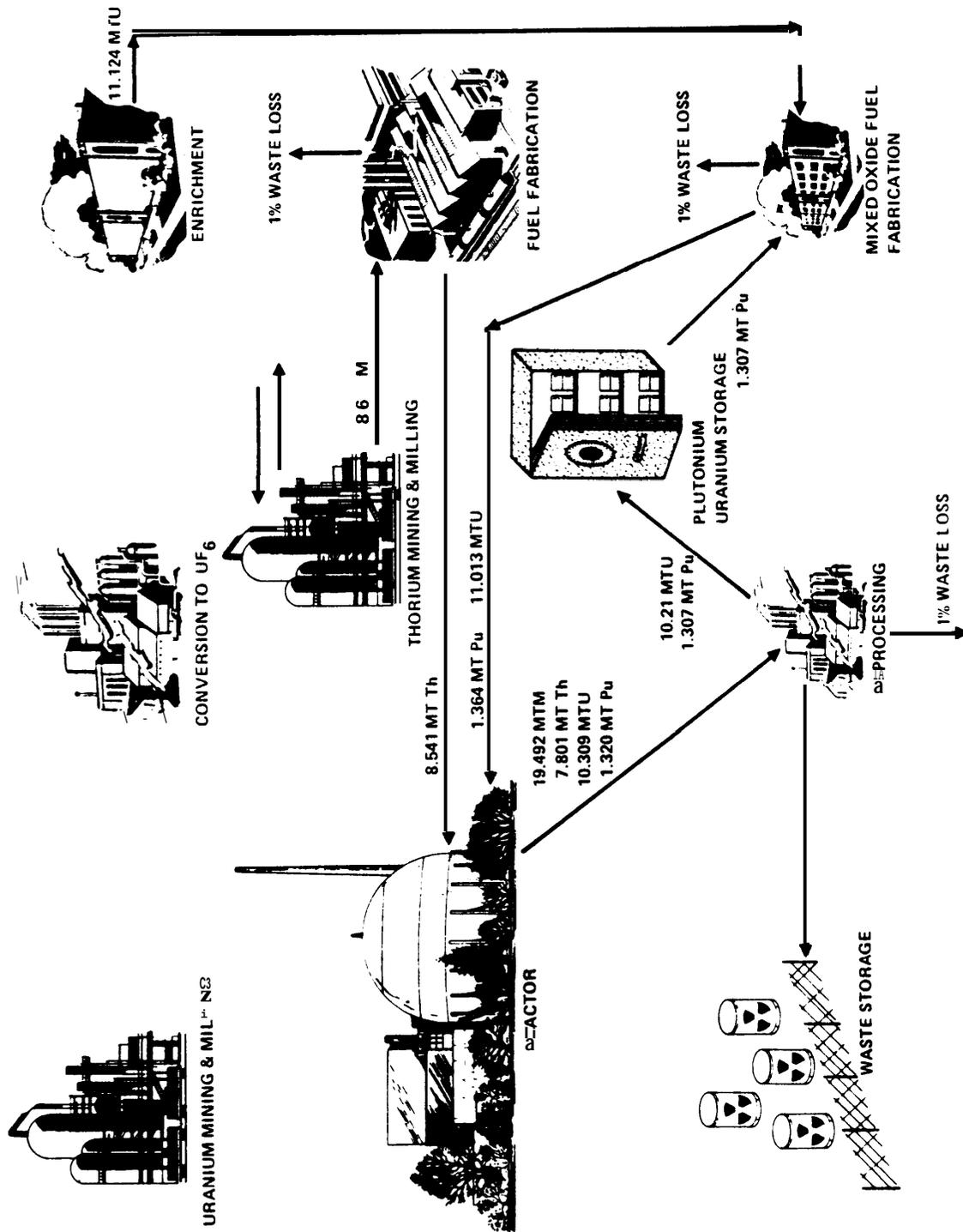
Fig. 37.(Cont). GCFR Equilibrium (Thorium Blanket Low Burnup Pu)

URANIUM MINING & MILLING

CONVERSION TO UF$_6$

ENRICHMENT

11.124 MTU

1% WASTE LOSS

FUEL FABRICATION

THORIUM MINING & MILLING

8.6 M

8.541 MT Th

1.364 MT Pu   11.013 MTU

REACTOR

19.492 MTM
7.801 MT Th
10.309 MTU
1.320 MT Pu

PLUTONIUM
URANIUM STORAGE

1% WASTE LOSS

MIXED OXIDE FUEL
FABRICATION

1.307 MT Pu

10.21 MTU
1.307 MT Pu

REPROCESSING

1% WASTE LOSS

WASTE STORAGE
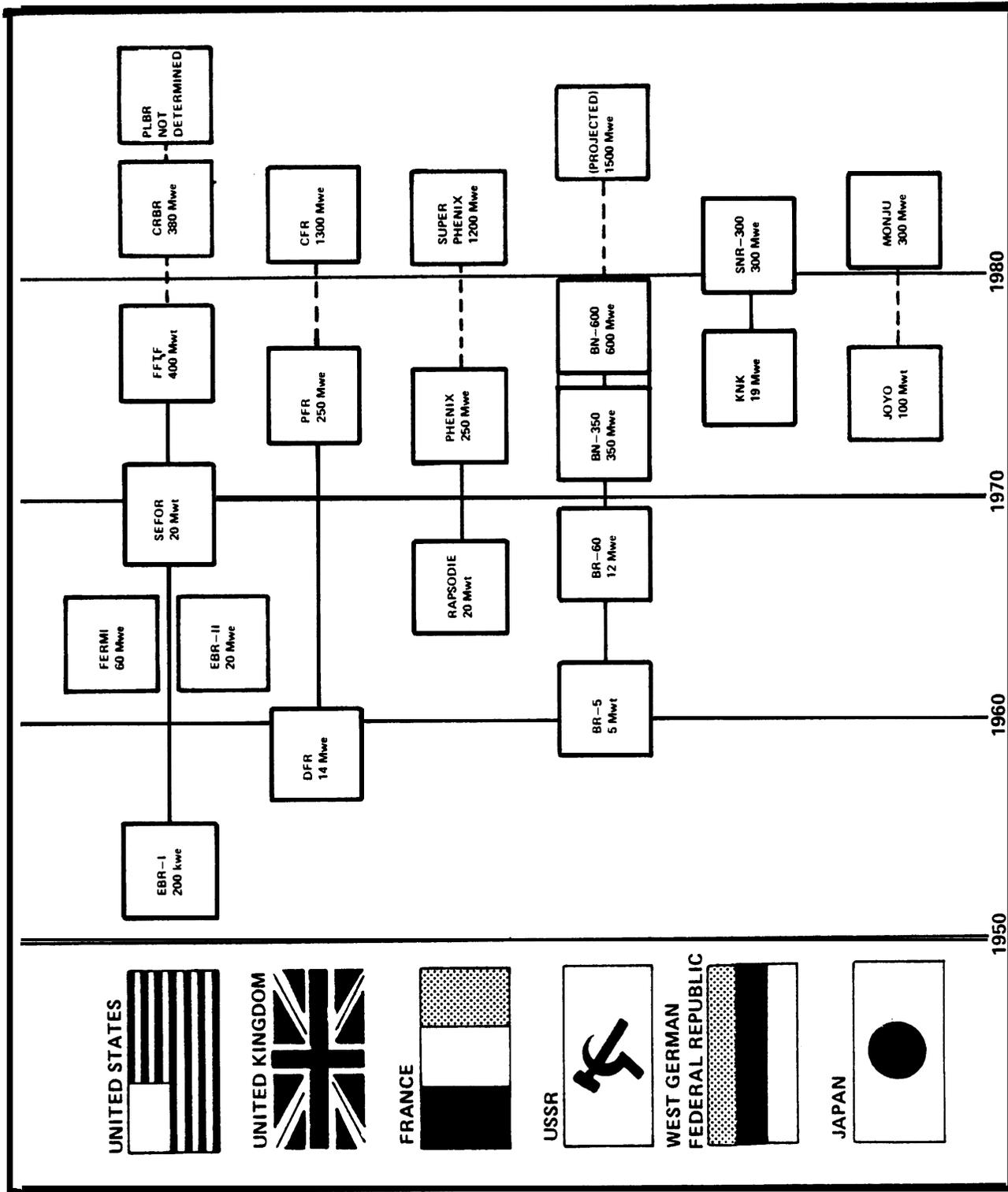
Fig. 38. International LMFBR Progress

TABLE 14

FUEL CYCLE REQUIREMENTS (1000 MWe, 75% LOAD FACTOR)

REACTOR CHARACTERIZATIONS	LMFBR1-AI Reference oxide	
	Initial Load	Equilibrium Cycle
Mining (10^3 MT)		
Material Removed		
Uranium 0.2% concentration		
Thorium 0.2% concentration		
Milling (MT U_3O_8)		
Conversion MT (= Enrichment **Feed**)		
(MT UF_6)		
Enrichment		
SWU (MT)		
Tails (MTU at 0.2%)		
Enriched product (MTU)		
Assay (% U-235)		
Feed nat U (MTM)		
Fuel Fabrication		
Input (MT)	15.453	**2.558**
Output (MT)	15.3	**2,533**
No. Assemblies/year	138	**23**
Composition	depl.UO_2	depl.UO2
Input U (MTM)	43.733	1.302
From Enrichment tails	(0.3%U235)	(0.3% U-235)
Weight of Assembly (MTM)	**0.1109**	**0.1109**
Used in Reactor	radial,bl.	radial bl.
Reactor load in		
Quantity (MTM) U\Pu\Th	**43.3/4.3/0**	16.25/1.528/0
Isotopes (%)		
u-233		o
u-234		0
u-235	**0.3**	**0.063**
u-236		**0.051**
u-238	99.7	**99.896**
Pu-239	71.5	**71.5**
Pu-240	25.2	**25.18**
Pu-241	2.4	**2.40**
Pu-242	0.9	**0.902**

TABLE 45 (Cont)

FUEL CYCLE REQUIREMENTS (1000 **MWe**, 75% LOAD FACTOR)

| | LMFBR1-AI reference oxide | |
REACTOR CHARACTERIZATIONS	Initial Load	Equilibrium Cycle
Mixed Oxide Fuel Fabrication		
Input (MT) U\Pu	28.28/4.343	13.728/1.545
output (MT)	32.3	15.120
No. Assemblies/year	274	128
Composition	depl.uO$_2$+PuO$_2$	depl.uO$_2$+Puc$_2$
Weight of Assembly (MTM)	0.'1179	0.1179
Used in Reactor	core & axial bl.	core & axial bl.
Waste Disposal		
High Level Wastes		
(HLW+Cladding hulls)		
Volume (ft3)		220
Activity (106 Curies)		24.3
Activity of Pu (103 Curies)		83
Canisters (a 3.5 ft^3)		63
Low Level Wastes		
volume (ft^3)		16,000-42,000
55 Gallon Drums		2,240-5,880
Burial or Repository Space (ft^3)		3,200-8,400

TABLE 14(cont)

FUEL CYCLE REQUIREMENTS (1000 MWe, 75% LOAD FACTOR)

REACTOR CHARACTERIZATIONS	LMFBR1-A1 reference oxide	
	Initial Load	Equilibrium Cycle
Reactor Discharge		
Quantity (MTM) U/Pu/Th		15.135/1.815/o
Isotopics		
U-233		**o**
u-234		**0**
U-235		**0.042**
U-236		**0.056**
U-238		**99.902**
Pu-239		**71.7**
Pu-240		**25.1**
Pu-241		**2.38**
Pu-242		**0.76**
Th		**0**
Fission Products (MTM)		0.690
Net Plant Thermal Efficiency (%)	**41.8**	**41.8**
Burnup in core (Avg. MWd/MTM)	**40,000**	**67,600**
Spent Fuel **Storage**		
Decay Heat **(W/kg)**		149
Days Holdup		30
Activity at Discharge From Pool (106 Curies F.P.)		301
Fuel Reprocessing		
Input **(MT)**		17.653
Recycle Uranium (MT)		
U-233		**0**
U-235		0.006
U-239		14.967
Plutonium Production (MT)		
Pu-239		1.2883
Pu-241		0.0429

CENTRAL STOCK
OF DEPLETED U
(0.3% U—235)

ENRICHMENT

1.302 MTU
(0.3% U—235)
FOR RADIAL
BLANKET

CONVERSION TO UF$_6$

URANIUM MINING & MILLING

2.533 MTU RADIAL BLANKET
(0.15% U—235)

15.120 MTU CORE & AXIAL BLANKET
13.592 MTU (99.9% U—238)
1.528 MT Pu (73.9% FISSILE Pu)

REACTOR

FUEL FABRICATION

1% WASTE LOSS

MIXED OXIDE FUEL
FABRICATION

1545 kg Pu
1495 kg
FISSILE Pu

PLUTONIUM
URANIUM STORAGE

13.730 MT U—238 RETURN

1,797 kg Pu
1,331 kg FISSILE Pu

REPROCESSING

1% WASTE LOSS

17.653 MT
SPENT
FUEL

220 ft^3 HLW
0.690 MT FP

WASTE STORAGE

Fig. 39 . LMFBR := Equilibrium Cycle

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Fig. 40. Prebreeder Fuel Cycle

154

1% LOSS

FUEL FABRICATION.

2.8 MT ^{232}TH
5.79 MT TH(NO$_3$)$_4$

91.1 MT ^{232}TH
1.94 MT ^{233}U
0.2 MT ^{235}U
93.2 MT
TOTAL

1% LOSS

MIXED OXIDE FUEL
FABRICATION

5% LOSS

THORIUM MINE AND MILL
UO2 CONC. (1470 MT OVERBURDEN)

THORIUM/URANIUM FUEL ASSEMBLIES (ThO$_2$/U
TOTAL { 93 MT ^{232}TH
95.14 MT { 1.94 MT ^{233}U } 2.14 MT U
{ 0.2 MT ^{235}U }

94.16 MT

92.00 MT ^{232}TH
1.96 MT ^{233}U } 2.16 MT U
0.2 MT ^{235}U }

LWBR

REACTOR

THOREX REPROCESSING

1% LOSS
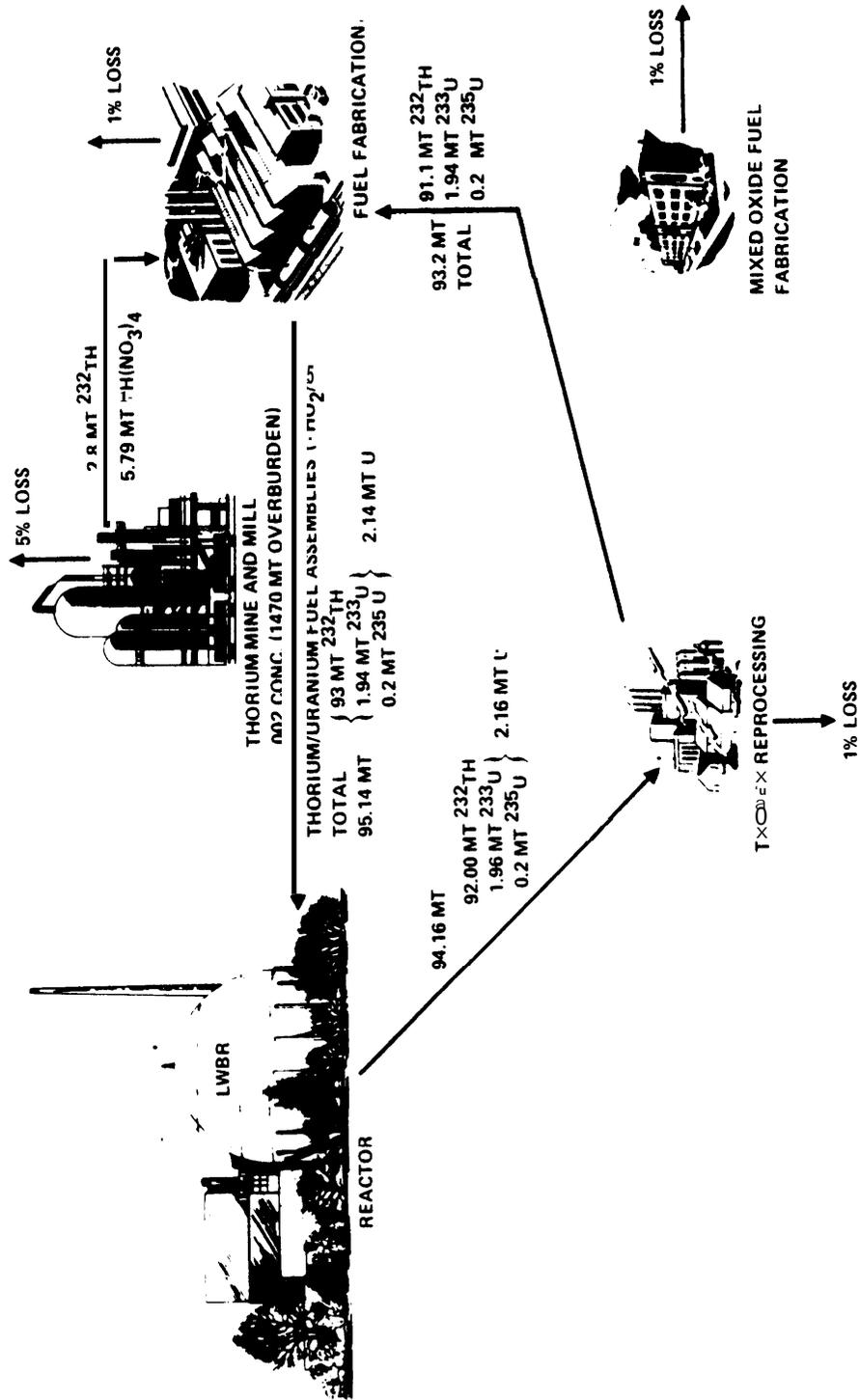
Fig. 41. Breeder Fue Cycle

TABLE 15

LWBR FUEL CYCLE REQUIREMENTS (1000 MWe, 75% LOAD FACTOR)

REACTOR CHARACTERIZATIONS

	PREBREEDER	BREEDER
Mining (10^3 MT)		
Material Removed		
Uranium 0.2% concentration	76.5	0.0
Thorium 0.2% concentration	27.1	5.79
Milling (MT U_3O_8)	155	
(Th $(NO_3)_4$)	54.2	
Conversion =Enrichment Feed) (MT UF_6)	191	0.0
Enrichment		
SWU (MT)	158	N/A
Tails (MTU at 0.2%)	118	N/A
Enriched Product (MTU)	11.2	N/A
Assay (% U-235)	13%	N/A
Feed nat U (MTM)	129	N/A
Fuel Fabrication		
Input (MT)	37.4	
Output (MT)	37.0	
No. Assemblies/year		96
Composition	UO_2/ThO_2	
Weight of Assembly (MTM)		
Used in Reactor		
Reactor Load in		
Quantity (U/Pu/Th)(MTM)	11.1/--/25.9	2.14/--/-/93
Isotopics (%)		
U-233	0.0	90.7
U-234	0.0	0.0
U-235	13.0	9.3
U-236	0.0	0.0
U-238	87.0	0.0
Pu-239	0.0	0.0
Pu-240	0.0	0.0
Pu-241	0.0	0.0
Pu-242	0.0	0.0

TABLE 15 (Cont)

LWBR FUEL CYCLE REQUIREMENTS (1000 MWe, 75% LOAD FACTOR)

REACTOR CHARACTERIZATIONS

	PREBREEDER	BREEDER
Reactor Discharge		
Quantity (U/Pu/Th) (MTM)	10.5/.11/25.2	2.2/--/92
Isotopics %		
U-233	2.3	90.7
U-234	--	--
U-235	7.6	9.3
U-236	--	--
U-238	89.6	--
Pu-239	--	--
Pu-240	--	--
Pu-241	--	--
Pu-242	--	--
Fission Products (MT^n		
Net Plant Efficiency (%)	32.5	33.5
Burnup (Avg. MWD/MTM)	51,400	16,300
Spent Fuel Storage	--	--
Decay Heat (W/kg)	--	--
Days Holdup	--	--
Activity at Discharge		
From Pool (10^6 Curies F.P.)	--	--
Fuel Reprocessing		
Input (MT)		
Recycle Uranium (MT)		93.2
U-233		1.94
U-235		.2
U-238		--
Plutonium Production (MT)		
Pu-239		0.0
Pu-241		0.0
Thorium		
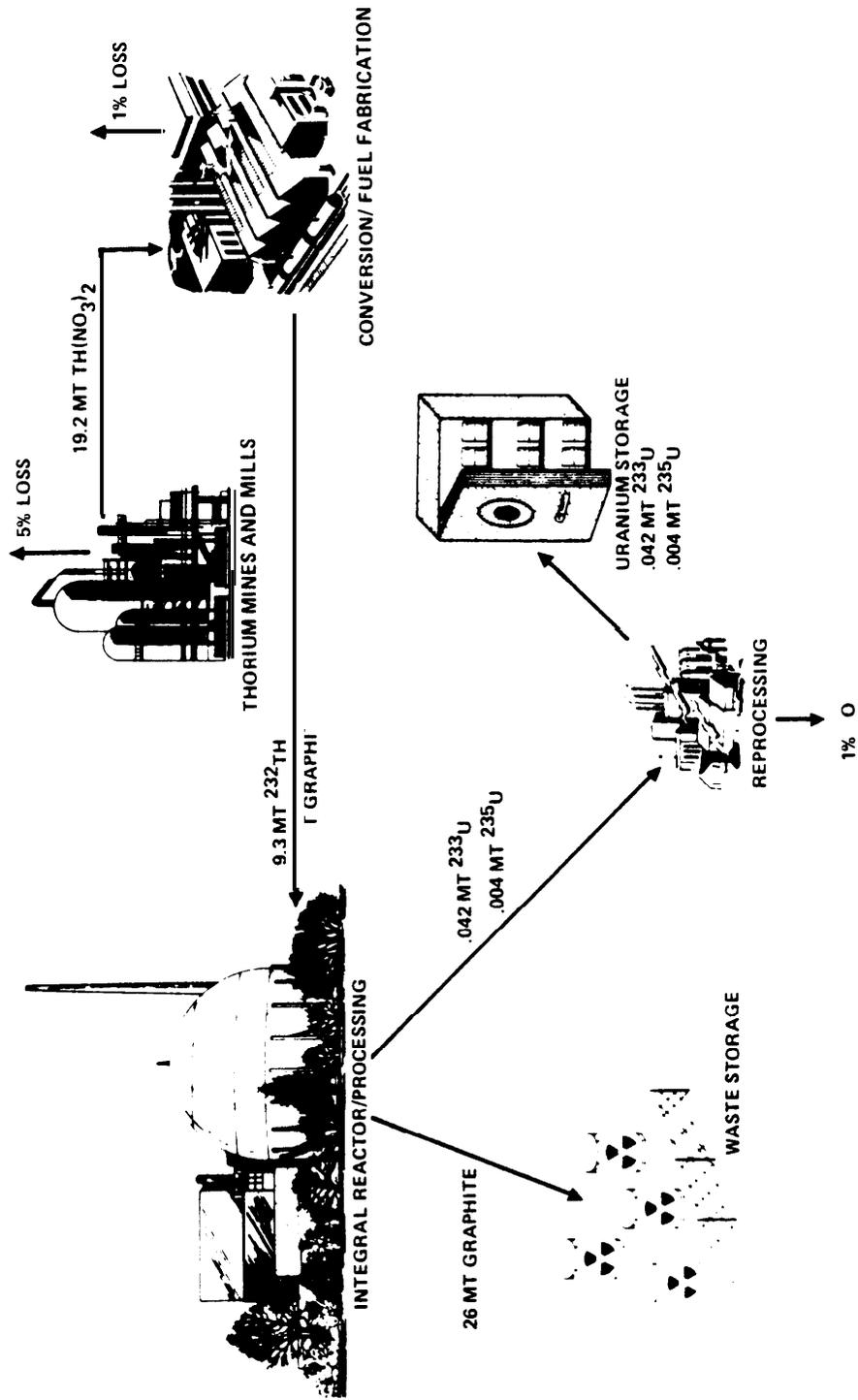
Fig. 42. Material Flow for the Molten Salt Breeder Reactor

1% LOSS

CONVERSION/ FUEL FABRICATION

19.2 MT TH(NO$_3$)$_2$

5% LOSS

THORIUM MINES AND MILLS

URANIUM STORAGE
.042 MT ^{233}U
.004 MT ^{235}U

9.3 MT ^{232}TH

GRAPHITE

REPROCESSING

1% O

.042 MT ^{233}U
.004 MT ^{235}U

INTEGRAL REACTOR/PROCESSING

26 MT GRAPHITE

WASTE STORAGE

Table 16

MSBR (1000 MWe 75% Capacity Factor)

	Start Up	Equilibrium
Mining (10^3 MT)		
Uranium 0.2% concentration	234	
Thorium 0.2% concentration	73	10.1
Milling (MT U_3O_8)	521	
(MT Th $(NO_3)_4$	281	19.2
Conversion (=Enrichment Feed)		
(MT $UF6$)	650	
Enrichment		
SWU (MT)	**570**	
Tails (MTU at 0.2%)	**436.9**	
Enriched Product (MTU)	**2.4**	
Assay	**93%**	
Feed nat U (MTM)	**439**	
Reactor Load in		
Quantity (U/Pu/Th) (MTM)	2.419/ - / 136	
Isotopics %		
u-233	0	
U-234	0	
U-235	93	
u-236	0	
U-238	7	
Reactor Inventory		
Quantity (U/Pu/Th) (MTM)		2. 3/-/ 136
Isotopics %		
U-233	0	68
U-234	0	17
U-235	93	7
U-236	0	8
U-238	7	
Net Plant Efficiency (%)	43	43
Reload in kg/yr		
Th-232	9281	9281
Graphite	26219	26219
Reload out kg/yr		
U-235	4.2	4.2
U-233	42.8	42.8
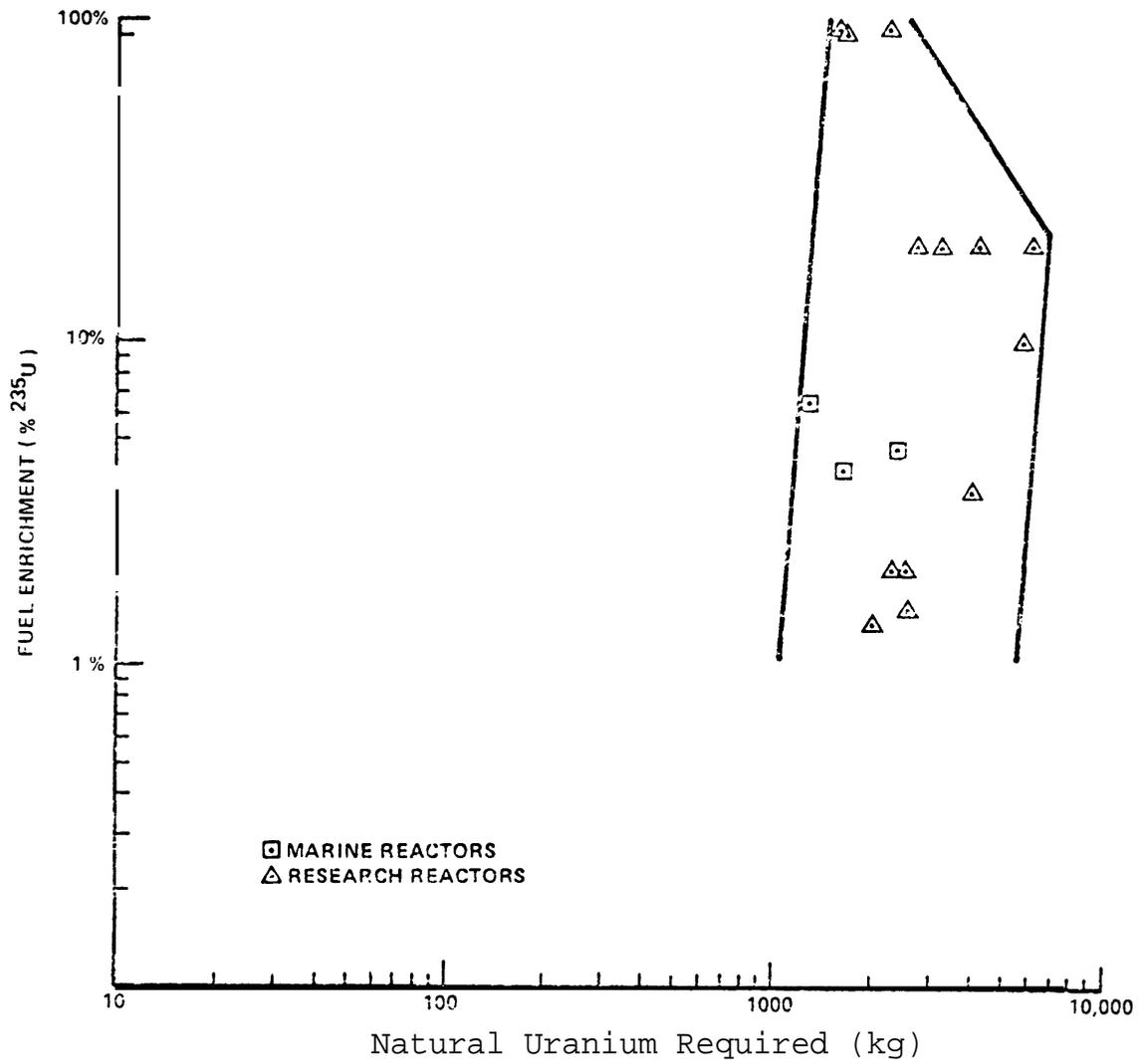
Fig. 43. Annual Natural Uranium Requirements to Enrichment Plant to Provide for 10 Mwt Reactor Annual Fuel Requirements
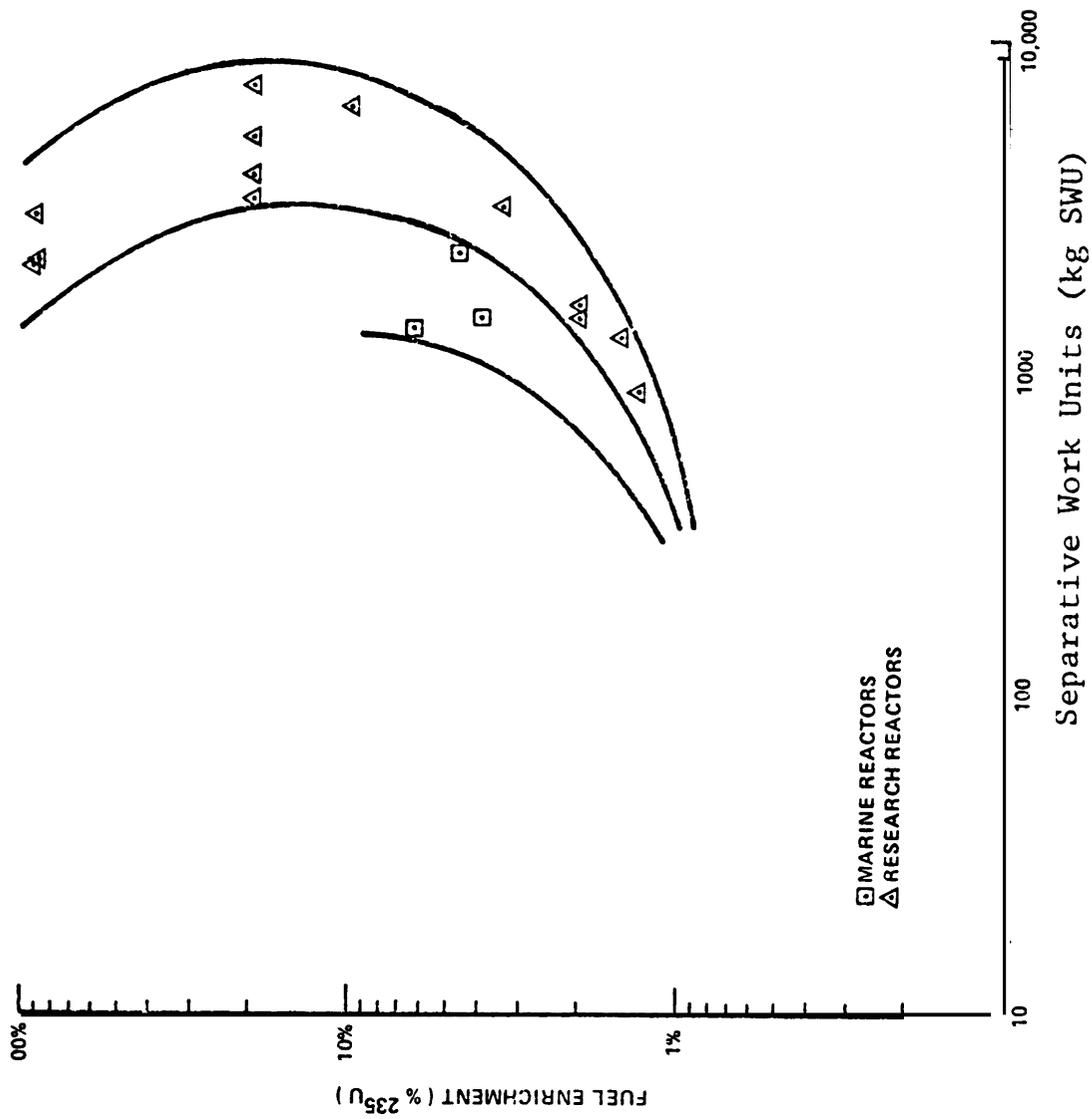
Fig. 44. Seprative Work Units Requirements to Provide for 10 Mwt
Reactor Annual Fuel Requirements

161
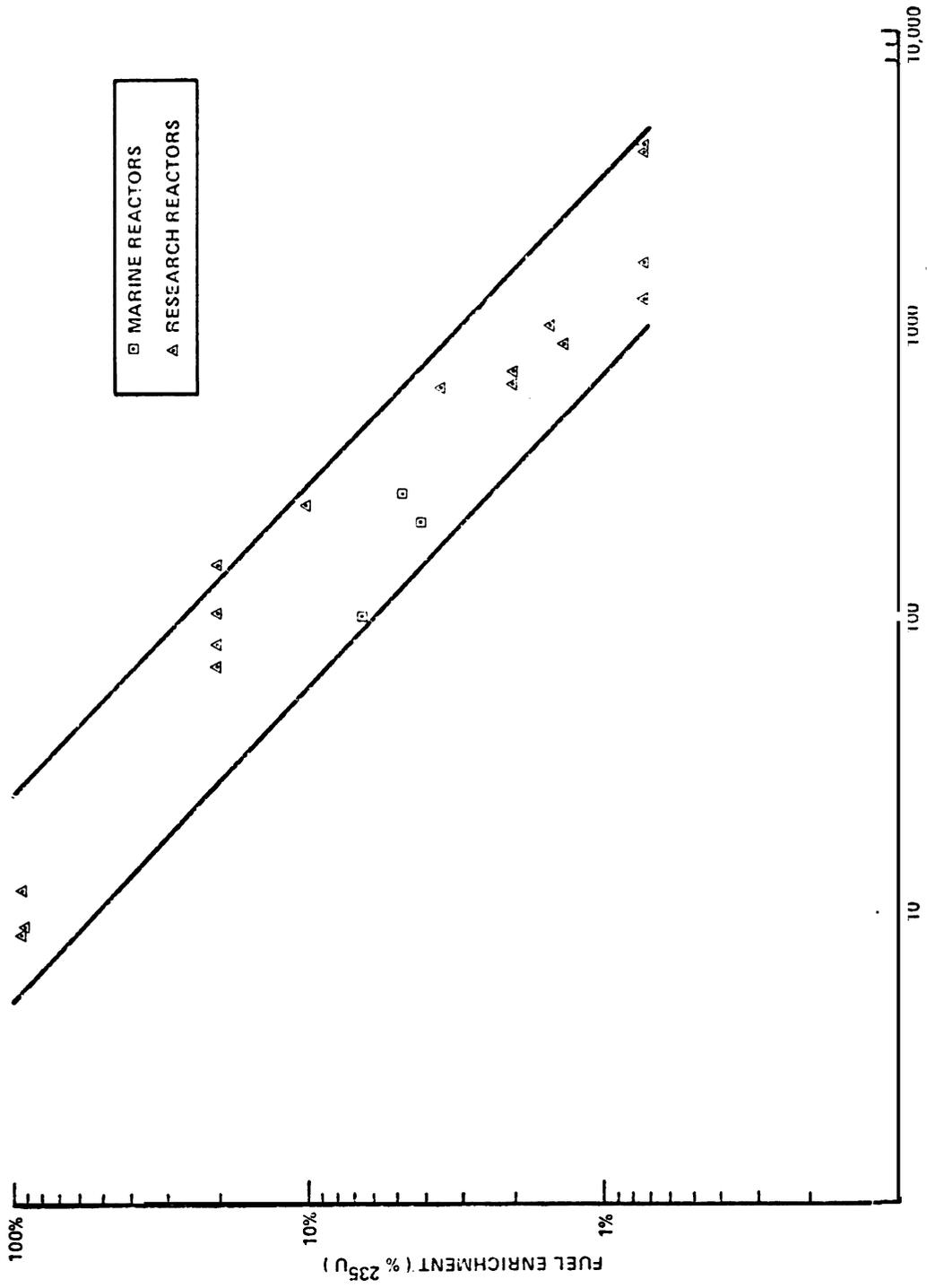
Fig. 45. Annual Uranium Fuel Requirements for a 10 Mwt Reactor
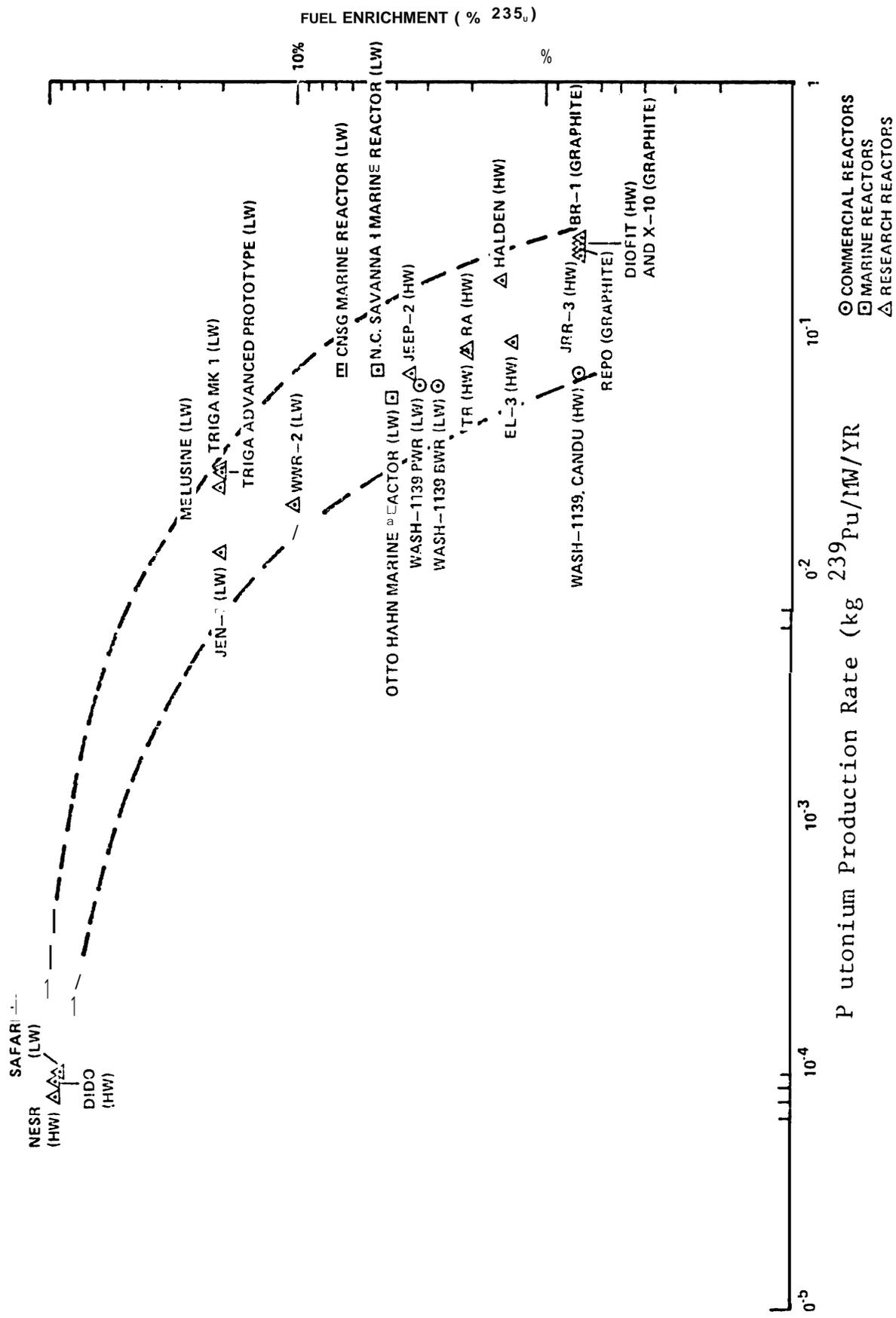
Fig. 46. Estimated Plutonium Production for Various Research and Power Reactors

NATURAL URANIUM
FUEL ASSEMBLIES
1255 Kg NATURAL URANIUM
(1246 Kg ^{238}U, 9 Kg ^{235}U)

URANIUM MINING & MILLING

1394 Kg
U_3O_8

CONVERSION
[10% LOSS]

NATURAL
URANIUM
METAL

1255 Kg

FUEL FABRICATION

7
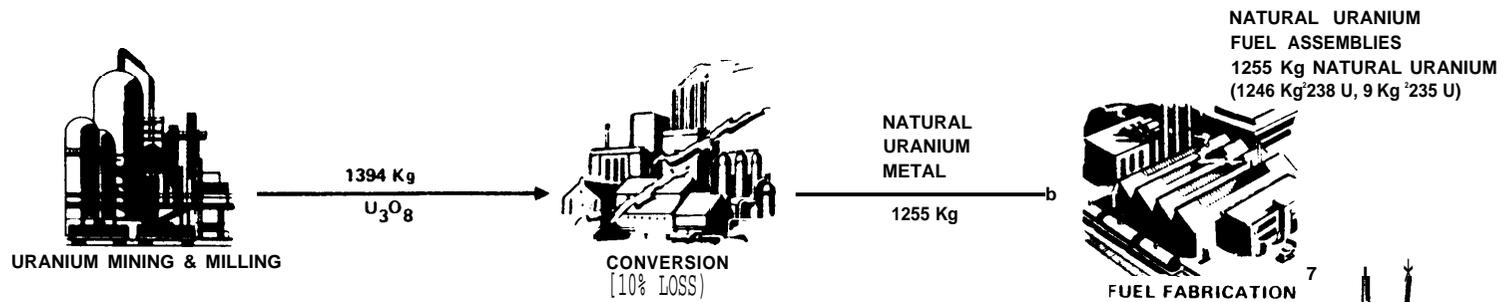
ANNUAL THROUGHPUT REQUIREMENTS FOR A 10 MWT BULK GRAPHITE OR
HEAVY WATER MODERATED AND COOLED TANK TYPE, NATURAL URANIUM
METAL FUELED RESEARCH REACTOR.

**MIXED OXIDE FUEL
FABRICATION FOR
POWER REACTOR FUEL**

REACTOR
1249.5 Kg URANIUM DEPLETED TO .49%
(1243.3 Kg ^{238}U, 6.2 Kg ^{235}U, AND 2.35 Kg PU)
SPENT FUEL

THE ABOVE VALUES OF THROUGHPUT ARE BASED ON BR–1,
WHICH IS A HIGH BURNUP (2000 MWD/TON URANIUM) GRAPHITE
REACTOR. URANIUM THROUGHPUTS FOR NATURAL URANIUM
FUELED GRAPHITE OR HEAVY WATER MODERATED REACTORS,
WITH DIFFERENT BURNUPS, CAN BE APPROXIMATED BY MULTI–
PLYING THE ABOVE VALUES BY THE RATIO $\left(\dfrac{2000 \text{ MWD/TON}}{(X) \text{ MWD/TON}}\right)$.
PLUTONIUM PRODUCTION RATE CHARACTERISTICALLY REMAINS
IN THE RANGE OF 0.20 TO 0.25 Kg ^{239}PU/MWT/YR BASED ON FIVE
NATURAL URANIUM FUELED REACTORS (JRR–3 AND DIORIT (BOTH
TANK TYPE) BEPO, X–10, AND BR–1 (ALL GRAPHITE MODERATED)).
HEAVY WATER MODERATED PRESSURE TUBE REACTORS HAVE
THROUGHPUTS THAT CAN BEST BE SCALED FROM THE VALUES
FOR THE CANDU POWER REACTORS.

REPROCESSING AND PU STORAGE

WASTE STORAGE

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Fig. 47.   Material Flow for Heavy Water Research Reactors

URANIUM MINES AND MILLS

U_3O_8
1784 Kg

CONVERSION
(10% LOSS)

NATURAL URANIUM
1606 Kg

ENRICHMENT
(.2% TAILS)

90% ENRICHED URANIUM
9.3 Kg

FUEL FABRICATION
(NO LOSSES)

URANIUM FUEL ASSEMBLIES

9.3 Kg 90% ENRICHED URANIUM
(.9 Kg ^{238}U : 8.4 Kg ^{235}U)

REACTOR

7.2% URANIUM DEPLETED TO 87%
(.9% ^{238}U, 6.3 Kg ^{235}U)
9.4×10^{-4} Kg ^{239}PU

SPENT FUEL

REPROCESSING AND PU STORAGE

MIXED OXIDE FUEL FABRICATION

WASTE STORAGE
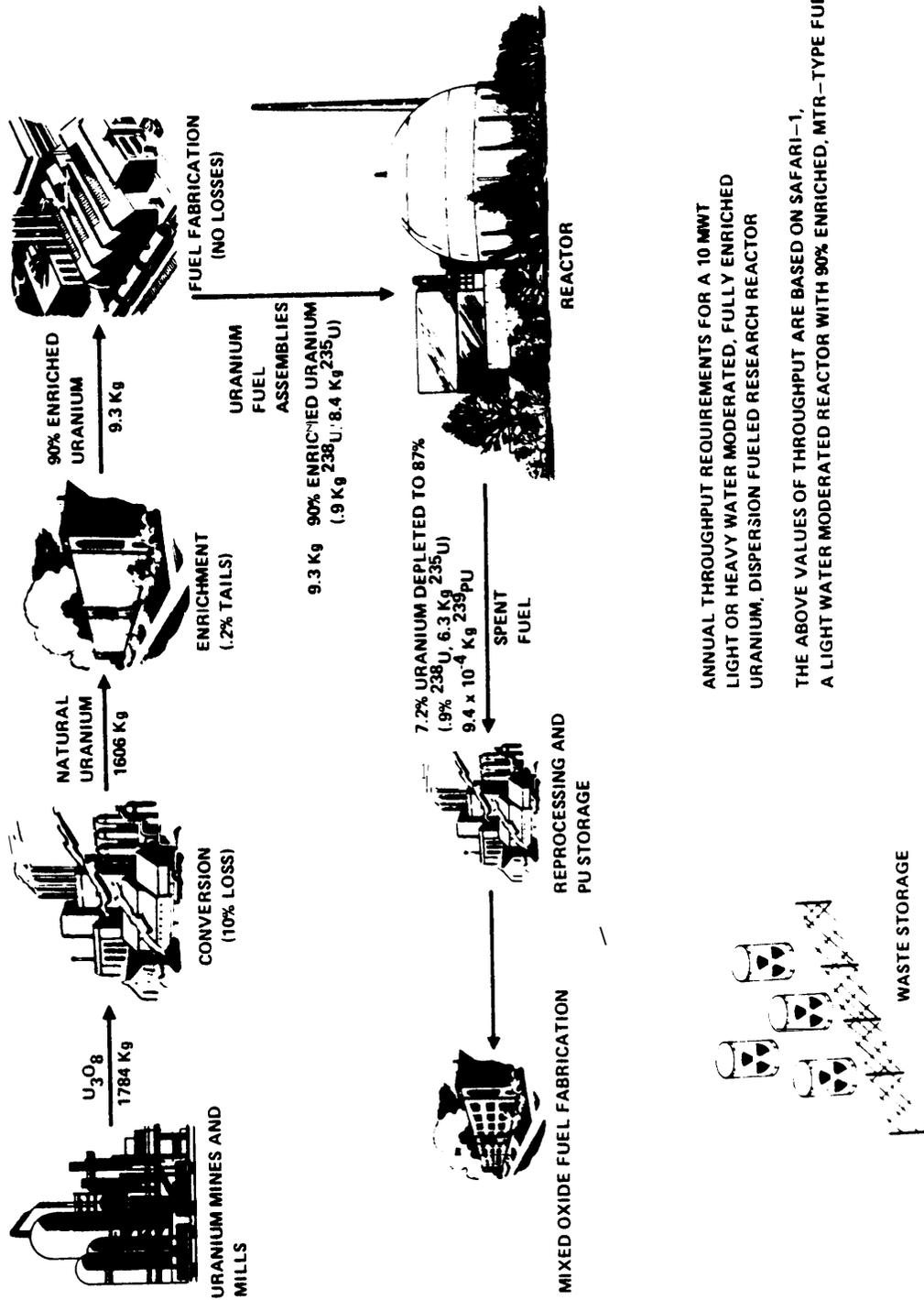
ANNUAL THROUGHPUT REQUIREMENTS FOR A 10 MWT
LIGHT OR HEAVY WATER MODERATED, FULLY ENRICHED
URANIUM, DISPERSION FUELED RESEARCH REACTOR

THE ABOVE VALUES OF THROUGHPUT ARE BASED ON SAFARI—1,
A LIGHT WATER MODERATED REACTOR WITH 90% ENRICHED, MTR—TYPE FUEL.

Fig. 48. Material Flow for Light or Heavy Water, Fully Enriched Uranium, Dispersion Fueled Research Reactors.

4. DIVERSION POTENTIAL OF REACTOR FUEL CYCLE MATERIAL**

The proper perspective on nuclear weapon proliferation re -
quires knowledge about nuclear weapon material requirements and
their relation to reactor fuel cycle material. Since a detailed
analysis of nuclear fission weapon parameters would require a
classified report, the unclassified details reported below only
define the order of magnitude of the weapon requirements. In
any case, the exact amount, type, and geometrical configuration
of material depends upon a specific design for a desired explosive
yield

The isotopes of particular interest for the construction of nuclear
explosives are ^{233}U, ^{235}U, and ^{239}Pu.   (However, it should be noted that all
Pu isotopes are fissile to fast neutrons.) Since each of these isotopes have
different nuclear properties, differing amounts of these mate-
rials, in pure form, are required to sustain a critical reaction.
Inside an infinitely thick, heavy metal-reflector, approximately
5 kilograms of ^{239}Pu or 233U, or 15 kilograms of ^{235}U are re-
quired for a critical mass [1,2]

None of the nuclear power reactor fuel cycles can be con-
sidered "ideal" sources for weapons material since they do not
contain isotonically pure, metallic* fissile material at any

1. H. C. Paxton, "Los Alamos Critical-Mass Data", Los Alamos
    Scientific Laboratory report LAMS-3067, April 1964.

2. W. R. Stratton, "Criticality Data and Factors Affecting
    Criticality of Single Homogeneous Units", Los Alamos
    Scientific Laboratory, July 1964.

*   HTGR fuel comes relatively close, in that highly enriched
    uranium (93% 235U) is utilized.

** See Appendix A for discussion of diversion potential from
    critical fuel   cycle facilities.

point in the fuel cycle. The fuel cycles do, however, contain fissile material in other chemical forms such as oxides, carbides, or nitrates, typically mixed with other isotopes from the same element.

The other isotopes (^{238}U, ^{240}Pu, ^{242}Pu) generally found in fuel cycle material act as impurities as far as weapons are concerned

The inclusion of these isotopes does not preclude the use of the material for the construction of weapons. Material enriched with 20% or more ^{235}U and any isotopic form of plutonium found in a power reactor fuel cycle could be converted to an explosive without enriching the isotopic fissile content of the material. The larger the percentage of these isotopes, however, the less attractive the material becomes in terms of weapon parameters, such as explosive yield per gram, weapon size, spontaneous radiation from the nuclear material, and predictability.

The stable chemical forms of fissile material found in fuel cycles include the oxides, carbides and nitrate solutions. Theoretically, pure oxide and carbide fissile material could be used directly in a weapon core[3]. As before, the oxygen and carbon act as "foreign" elements by changing the fissile material density and scattering properties of the material. The nitrate solutions could not be used directly in a weapon. All of these chemical" forms can be converted to metallic form, however, through standard chemical techniques found in open technical literature. On the national threat scale, the chemical form of the fissile material presents no particular difficulty. This may or may not be true for a terrorist or subnational threat.

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3. Mason Willrich and Theodore B. Taylor, "Nuclear Theft: Risks and Safeguards", Ballinger Publishing Co. , 1974.

4.1 EXPOSURE OF STRATEGIC SPECIAL NUCLEAR MATERIAL

An educational, but imprecise, technique for qualitatively classifying reactor fuel cycles is to look at the fuel cycle components for the various reactors having strategic special nuclear material (SSNM) exposed at some point in the process. (Exposed is defined to mean that the material exists in a pure chemical form separated from other compounds and SSNM indicates that no further isotopic enrichment is necessary to produce weapons grade material.) This taxonomic procedure is useful but does <u>not</u> necessarily rate the fuel cycles in terms of proliferation potential. A systematic consideration of the threat, plus specific scenarios on the fuel cycle components, require additional information for a fuel cycle proliferation rating. For example, in a country which contains only a reactor and no other fuel cycle components, the proliferation potential of-each of the fuel cycles is nearly the same. The only proliferation difference exists in the difficulty of separating fresh fuel SSNM versus the separation of SSNM in irradiated fuel.

In Table 18 the exposure of SSNM for the representative reactor fuel cycles is shown for the various fuel cycle components. A quick perusal of Table 18 indicates that no ideal reactor fuel cycle exists -- they all contain SSNM. 1n addition, SSNM is generally exposed in the facilities in the back end of the fuel cycle (i.e. , reprocessing and recycle fuel fabrication) .

Before discussing additional details relative to the exposure matrix table, the typical exposure modes for fuel cycle processes should be mentioned. In an enrichment plant, the SSNM may appear as the output of the plant -- it is the end

TABLE 18

STRATEGIC SPECIAL NUCLEAR MATERIAL EXPOSURE MATRIX

REACTOR TYPE*

FUEL CYCLE COMPONENT	LWR NO RECYCLE	LWR WITH RECYCLE	HWR	HTGR	LMFBR	LWBR	MSBR	RESEARCH REACTORS		
								MTR	TRIGA	D$_2$O
ENRICHMENT	3	3	–	1	– or 1	3	1	1	1	–
FUEL FABRICATION	3	3	3	1	1	3	–	1	1	3
REACTOR	2	2	2	2	2	2	2	2	2	2
REPROCESSING	–	1	–	1	1	1	–	–	–	–
RECYCLE FABRICATION	–	1	–	1	1	1	–	–	–	–
WASTE DISPOSAL	2	2	2	2	2	2	2	2	2	2

LEGEND

1 I STRATEGIC SPECIAL NUCLEAR MATERIAL (SSNM) EXPOSED DURING THE SPECIFIC FUEL CYCLE COMPONENT PROCESS

2 SSNM EXISTS IN THE PROCESS BUT MUST BE SEPARATED FROM OTHER MATERIAL

3 III SSNM DOES NOT EXIST IN THE FUEL CYCLE PROCESS

– III THIS FUEL CYCLE COMPONENT IS NOT CONSIDERED TO BE A STANDARD COMPONENT OF THE REACTOR FUEL CYCLE

LWR – LIGHT WATER REACTORS
HWR – HEAVY WATER REACTORS
HTGR – HIGH TEMPERATURE GAS–COOLED REACTOR
LMFBR – LIQUID METAL COOLED FAST BREEDER REACTOR
LWBR – LIGHT WATER COOLED BREEDER REACTOR

MSBR – MOLTEN SALT BREEDER REACTOR
MTR – MATERIALS TEST REACTOR (URANIUM ALUMINUM ALLOY FUEL)
TRIGA – RESEARCH REACTOR MANUFACTURED BY GENERAL ATOMIC
D$_2$O – HEAVY WATER MODERATED REACTOR (NATURAL URANIUM FUEL)

*THE INITIAL AND EQUILIBRIUM CORE LOADINGS ARE CONSIDERED TOGETHER

product of the process. The chemical form of the SSNM is uranium hexafluoride or after conversion, uranium oxide (UO_2). This material is input to the fuel fabrication facility, where it is processed into fuel elements which are, in turn, input to the reactor. After sufficient reactor operation, the irradiated fuel may be reprocessed; here the SSNM is exposed as the reprocessing plant product. The typical chemical form for the reprocessing plant product is a nitrate or after conversion, an oxide. This material may be then input to a recycle fuel fabrication plant and refabricated into reactor fuel and fed back to the reactor.

The light water reactor fuel cycle without recycle does not expose SSNM in its normal operation. This is the present mode of operation in the United States, with the irradiated fuel temporarily stored until a reprocessing or permanent disposal decision is made. The irradiated fuel contains significant amounts of plutonium. In the LWR fuel cycle with recycle, SSNM (plutonium) is exposed at the reprocessing and recycle fuel fabrication facilities. The rationale for recycle is more efficient utilization of the energy content in uranium fuel. The economic basis for recycle is unclear, since total recycle costs have not evolved into a predictable value. Uranium resources would be extended by recycle by about 30%. It must be noted however, that there are alternative methods of extending the uranium resources to about the same fractional increase. The use of more efficient converter reactors, such as the HTGR, the development of a high burnup, throwaway LWR fuel **cycle,** thorium fuel cycles, the tandem fuel cycle, or the development of laser enrichment (with a decreased tails enrichment) would all tend to extend the uranium resources. The economic value and practicality of these alternatives has yet to be proven.

An alternative reprocessing scheme is also available and technically proven. In this scheme the uranium and plutonium are never separated but are decontaminated from the reactor poisons (fission products) only. Some advantages of this scheme include the following:

- A simplified reprocessing flow chart, with fewer safety problems and improved efficiency.

- Improved recycle fuel because of the intrinsic homogeneity of the coprecipitated uranium and plutonium oxides.

- Improved safeguards because the uranium and plutonium would only contain approximately 1.5% fissile material. A diverter would have to divert approximately 1000 kg of coprecipitated material to separate out 5 kg of fissile plutonium for a nuclear weapon. The safeguard advantages are primarily against a subnational threat, since a national entity with large resources could, with relative ease, separate the plutonium from the uranium.

The major disadvantage of the coprecipitation reprocessing scheme is the recycle fabrication plant. Large amounts of mixed oxide fuel have to be handled (increasing plant size), and all waste and scrap streams are contaminated with plutonium. In addition, the mixed uranium-plutonium oxide has to be enriched to roughly 4% fissile content. These disadvantages incur economic penalties which tend to detract from the advantages.

Heavy water reactors, like the Canadian CANDU, have a rather simple fuel cycle, with no reprocessing and no exposure of SSNM. Even though the SSNM is not exposed during the "standard" CANDU fuel cycle, this reactor does have proliferation liabilities. Because of the on-line refueling capabilities of the pressurized tube design, fuel management can be optimized to produce weapons grade plutonium with no power production penalty. Of course, the fuel throughput must be increased accordingly; this, however, is a relatively minor economic penalty.

There are heavy water reactors which utilize a pressure vessel design and consequential, off-line refueling. A notable example of this is the German built reactor in Atucha, Argentina.

The high-temperature gas-cooled (HTGR) reactor promoted by the General Atomic Company in the U.S. is the **only** operational civilian power reactor concept that utilizes fully enriched uranium (93% ^{235}U) as fuel. The exposure of SSNM in the front end of the HTGR fuel cycle requires careful safeguards considerations, particularly if this concept becomes popular on a world-wide scale. The manufactured HTGR fuel does have some intrinsic protection against a sub-national threat, since recovery of the uranium is not a trivial task.

The backend of the HTGR fuel cycle has not been fully developed, not even to the incomplete extent of the LWR fuel cycle. Consequently, a substantial uncertainty exists as to the relative economic merit of various HTGR reprocessing and recycling programs. A recent report[6] indicates that the most economic schemes involve mixing recovered ^{235}U and ^{233}U for one recycle back through the reactor. After the one recycle, the remaining ^{235}U and ^{233}U is retired. The indications are that the HTGR recycle has a significant economic advantage over a throw-away fuel cycle. HTGR recycle may be limited to a one-time recycle because of the build-up of ^{236}U -- a reactor poison.

6. N. D. Holder, V. H. Pierce, and M. P. Rothstein, "An Economic Analysis of U-235 Recycle in the HTGR", General Atomic report GA-A13836, July 15, 1976.

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HTGR recycle material has built-in protection against a subnational diversion, because of the in-breeding of ^{232}U -- an active precusor of a highly radioactive chain of daughter products. For example, 5 kg of ^{233}U, containing 1000 parts per million of ^{232}U, would have a radiation dose rate near 10 Rem per hour, one foot from the material, after a 20 day delay period following uranium separation. For longer delays, the radiation dose builds up to saturation level roughly 10 years after separation.

There are alternative HTGR fuel cycle concepts to the present fully enriched uranium-thorium fuel cycle. A 1968 General Atomic report[7*] concluded that a low-enriched uranium fuel cycle, possibly as low as 6% ^{235}U, would have fuel cycle costs comparable to a throw-away thorium cycle. A more recent economic analysis tends to support this conclusion. The effect of utilizing a low-enriched fuel cycle would lower the conversion ratio substantially (it would now be roughly the same as LWRS), require more uranium fuel and introduce substantial quantities of plutonium into the fuel cycle. It would, however, remove the exposure of SSNM from the front end of the HTGR fuel cycle.

The liquid metal-cooled fast breeder reactor (LMFBR) has SSNM exposed throughout its fuel cycle. The enrichment facility is only needed in the fuel cycle, if the initial core load is made of enriched uranium and not plutonium. Since the LMFBR is a breeder reactor (it produces more fissile material than it consumes) the reprocessing and recycle fuel fabrication plants are required components of the fuel cycle. An important feature of an LMFBR is the amount of plutonium involved in the fuel cycle. An initial load in a 1000 MWe reactor would involve some **3** to **4** metric tons of plutonium. Annual reload requirements are roughly 1 ton of plutonium.

7* P. U. Fischer, S. Jaye, and H. B. Stewart, "Alternate Fuel Cycles for the HTGR," Gulf General Atomic report GA-9010, October 4, 1968.

The light water breeder reactor (LWBR) necessarily exposes SSNM in the backend of its fuel cycle. Since it operates on a thorium fuel cycle, the comments on the in-breeding of ^{232}U in the HTGR fuel are also relevant here. An interesting feature of the LWBR is that it is just barely self-sustaining. Removal of significant quantities of SSNM from the fuel cycle shut down the reactor for power production or require a corresponding importation of fresh fuel.

The molten salt breeder reactor (MSBR), with its unusual fuel cycle, does not expose SSNM during its operational cycle, except for the possible production of roughly 3% excess fuel per year. The initial core would most likely start up with 93% enriched uranium.

Research reactors are normally characterized by their small core fissile inventory, an uncertain operational schedule and lack of a commercial backend of the fuel cycle. These features are generally considered positive assets for safeguards purposes. On the other hand, research reactors are often sold on the basis of their flexible fuel management options and relative ease of refueling. This flexibility gives the operators many options.

MTR (Material Test Reactor) type fuel is often used to generically describe uranium-aluminum alloy fuel clad in aluminum. Fuel enrichments for research reactors using MTR type fuel vary from slightly enriched to fully enriched (93% ^{235}U), with typical enrichments at 20% and 93%. The MTP--type fuel plate is relatively easily processed to separate out the uranium. The chemistry involved is well known and available in open literature.

TRIGA fuel elements consist of uranium-zirconium alloy. The enrichment may be 20%, 70% or 93%. The reprocessing and recovery of uranium from these types of elements is much more difficult than for MTR fuel.

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Heavy water moderated and cooled research reactors using natural or slightly enriched uranium fuel, appear to be the most easily safeguarded reactor. They are, however, capable of producing plutonium on a small scale, as India proved in May 1974.

4.2 DIVERSION PATHWAYS FOR SSNM

Assuming that a political/technical decision has been made to develop the most effective nuclear weapons possible by diverting material from a reactor fuel cycle, the most cost effective, least detectable and lowest impact pathway to acquiring the necessary SSNM must be determined. There is no simple, unique manner to make this determination. It depends upon the threat (large country, small country, sub-national group), the resources available, and the specific reactor fuel cycle under attack. In this section, we shall examine a few representative fuel cycles to illustrate the impact on the fuel cycle (power production, fuel throughputs, etc.) and the additional facilities required to produce nuclear weapons from fuel cycle material. Since fuel cycle plutonium is not optimum SSNM for nuclear weapons production, we shall also discuss fuel cycle tampering to produce fore favorable isotopic concentrations of SSNM.

At the bottom of Figure 49, the fuel cycle for a 1000 MWe CANDU-type heavy water reactor is depicted with annual equilibrium cycle fuel flows. The most obvious, straightforward method for achieving nuclear weapons capability without affecting the power production is to reprocess spent fuel containing approximately 0.4 tons of plutonium (72% fissile Pu) into roughly 57 nuclear weapons. This assumes a 1% Pu loss to reprocessing and SSNM rework and fabrication. Each of the weapons would contain 5 Kg of fissile Pu and nearly 2 Kg of ^{240}PU + ^{242}PU.

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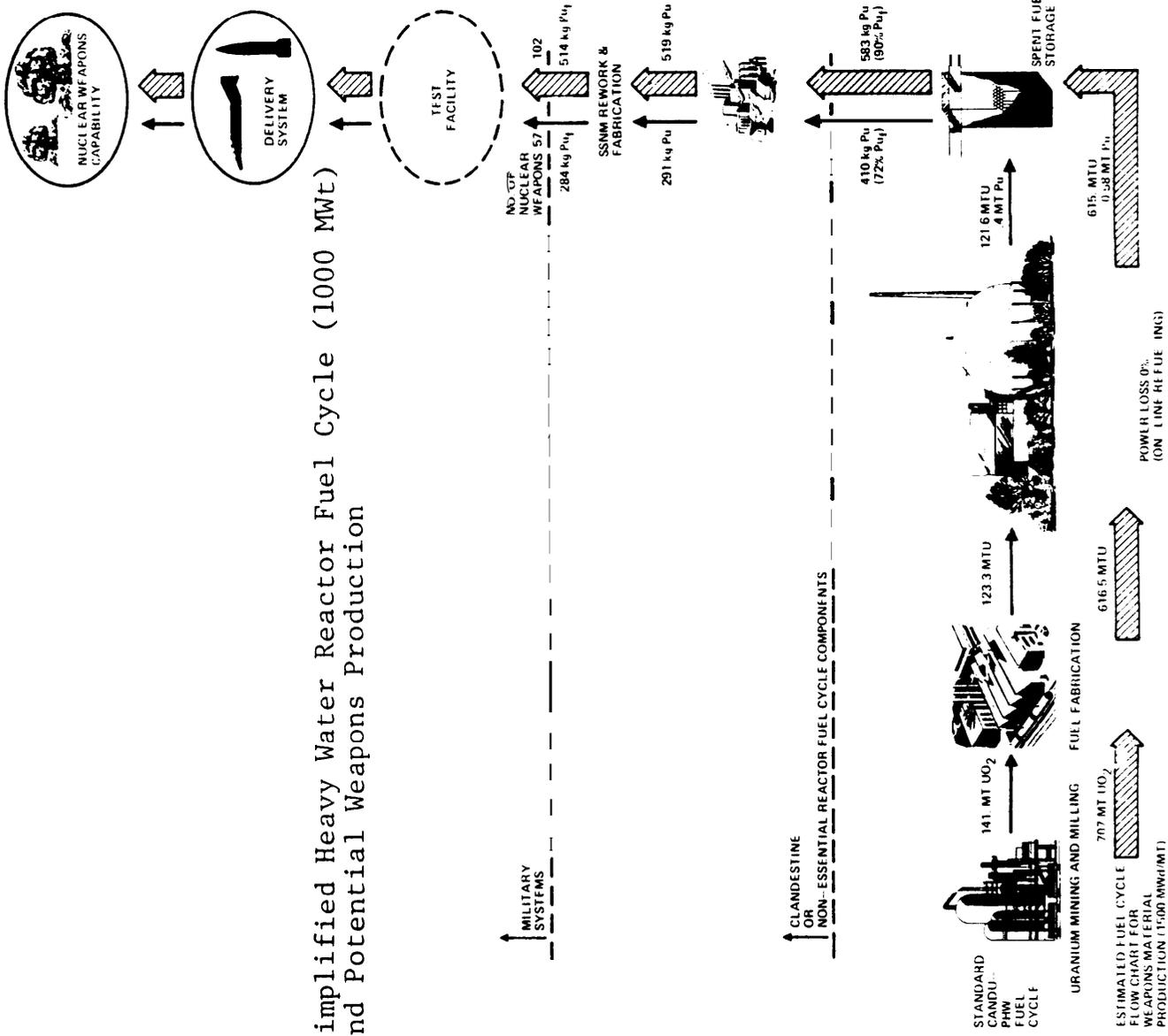
Fig. 49. Simplified Heavy Water Reactor Fuel Cycle (1000 MWt) and Potential Weapons Production

MILITARY SYSTEMS

NUCLEAR WEAPONS CAPABILITY

DELIVERY SYSTEM

TEST FACILITY

NO. OF NUCLEAR WEAPONS 57   102

284 kg Pu$_f$ 514 kg Pu$_f$

SSNM REWORK & FABRICATION

291 kg Pu   519 kg Pu

CLANDESTINE OR NON—ESSENTIAL REACTOR FUEL CYCLE COMPONENTS

410 kg Pu (72% Pu$_f$)   583 kg Pu (90% Pu$_f$)

SPENT FUEL STORAGE

12.6 MTU 4 MT Pu

615 MTU 0.38 MT Pu

POWER LOSS 0% (ON LINE REFUELING)

123.3 MTU

616.5 MTU

STANDARD CANDU-PHW FUEL CYCLE

141 MT UO$_2$

URANIUM MINING AND MILLING   FUEL FABRICATION

707 MT UO$_2$

ESTIMATED FUEL CYCLE FLOW CHART FOR WEAPONS MATERIAL PRODUCTION (1500 MWd/MT)

Plutonium with a higher percentage of ^{239}Pu
could, in principle, be achieved in a CANDU fuel cycle without
causing any power production losses. Instead of burning the
fuel to 7500 MWd/MT, the fuel
times faster, producing more total plutonium (.58 MT Pu) , with
a more favorable fissile content (90%). This procedure would
ultimately produce roughly 102 weapons per year for a 1000
megawatt reactor. The only tampering indication would be the
increased fuel throughput. The key safeguards management
point in the fuel cycle is the spent fuel. An account of all
the spent fuel emanating from the reactor counters this method
of clandestine weapons production.

A surveillance technique for fuel removal from a CANDU reac-
tor has, in fact, been demonstrated.* The on-line refueling
machine, of course, complicates the accounting of fuel bundles
discharged from the reactor; however, the transfer channel can
be monitored (for high activity gamma ray sources) to count the
number of irradiation fuel bundles as they pass by on their way
to the storage bay. The item count of the stored bundles in the
bay correlated with the tamper proof continuous surveillance
monitor could -_ assure an inspector that all irradiated CANDU
fuel is accounted for. Monitoring of the spent fuel discharged
from CANDU reactors is not presently required.

In Figure 50 a simplified 1000 MWe LWR fuel cycle with no
recycle is depicted. This fuel cycle is representative of the
majority of power reactors operating in the world today.
Assuming 75% power production (i.e., 75% capacity factor) for
the year, this equilibrium cycle pressurized water reactor
produces approximately 230 Kg of plutonium each year. After

_____

*
 D.B. Siden, J.G. Hodgkinson, J.W. Cornbell, H.D. Kosanke, "Test-
 ing of Techniques for the Surveillance of Spent Fuel flow and
 Reactor Power at Pickering Generating Station," International
 Atomic Energy Synposium on Safeguarding Nuclear Materials, 20-24
 October 1975, Vienna, IAEA-SM-201/67.

Fig. 50. Simplified LWR Fuel Cycle (No Recycle) and
Potential Impact of Weapons Production

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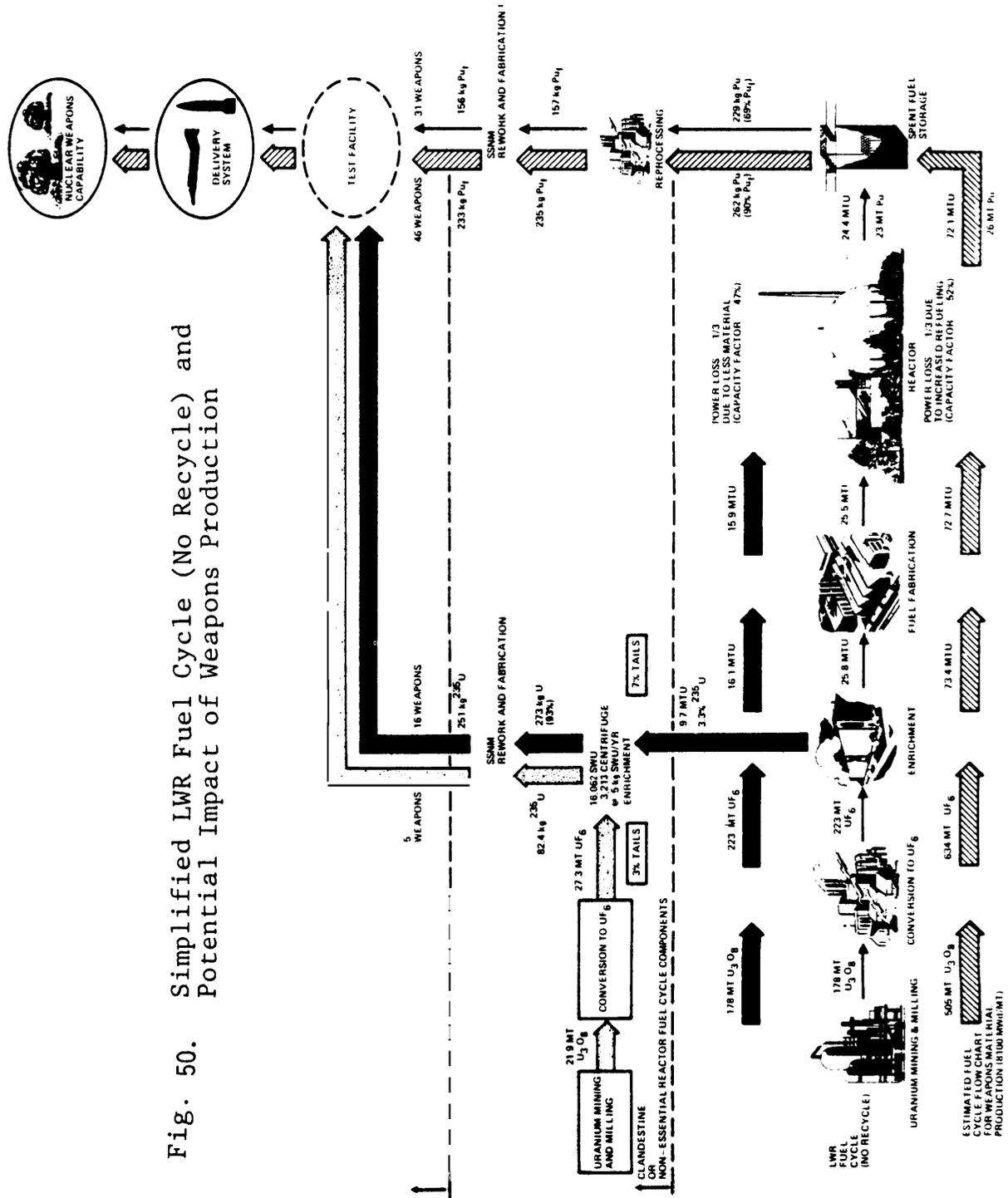
accounting for losses, this amount of Pu could produce up to 31 nuclear weapons.

The isotopic concentration of the plutonium could be made more attractive for weapons (i.e., from 69% fissile Pu to 90% fissile Pu) by lowering the fuel burnup from 33,000 MWd/MT to 8100 MWd/MT. This, in effect, increases the fuel throughput in the fuel cycle. For LWRs, however, the refueling must be performed off-line. Consequently, a power production penalty must be paid for the additional refueling required by a lower burnup, higher fuel throughput. For the situation depicted in Figure 50 the power loss is almost 1/3 the yearly power production, assuming that four refueling with typical down times[8] are required. An additional indicator that an abnormal fuel cycle is in operation is the roughly 3 fold increase in fuel. If this material is being imported (i.e., the country has no enrichment capability), then the abnormal situation is discernible. The lower burnup fuel can produce more plutonium than the normal fuel cycle (even with less power production), with higher Pu isotopic concentration. The resultant weapons production increase is almost 50%.

There are, of course, other LWR fuel management schemes available to a reactor operator which produce weapons grade plutonium. The impact on power production, fuel requirements, and weapons production may differ in detail from those depicted above. The sense of the impacts is apparent, however. The critical safeguards management point for LWR plutonium production is accounting for the spent fuel emanating from the reactor (as in the case for the HWRs).

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8. A. Fattah, R. Skjoeldbrand, "Performance Analyses on Nuclear Power Plants from Operating Experience Data", IAEA-SM-195/36, Symposium on Reliability of Nuclear Power Plants, April 14-18, 1975.

LWR fuel assemblies have identifiable serial numbers that can be utilized by safeguards inspectors for accounting purposes. However, BWR assemblies can and occasionally are disassembled to remove fuel rods. This practice, potentially, could be subverted to produce optimum weapons material if a clandestine source of fuel rods is available. A detection counter to this hypothetical threat is the use of tamper proof seals on BWR and PWR fuel assemblies. These seals are presently under test and development.*

There is another credible weapons production possibility associated with the LWR fuel cycle. The diversion of a portion of the slightly enriched fuel to a clandestine centrifuge enrichment plant could facilitate the production of nuclear weapons. Assuming that a power production loss of roughly 1/3 is acceptable (to an overall capacity factor of 47%), then-almost 10 tons of the 3.3% enriched uranium could be diverted to the centrifuge plant. With a 0.7% tails enrichment, some 16 MT of separative work units (SWU) are required to produce 273 Kg of 93% enriched uranium. This corresponds to roughly 16 nuclear weapons. The number of centrifuges, at 5 Kg SWU/year capacity, required to further enrich the LWR fuel is over 3,000, These same centrifuges could be utilized to enrich natural uranium to weapons material at 93% enrichment. This type of clandestine operation could produce approximately five uranium weapons, less than 1/3 the number produced by the fuel cycle diversion method.

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S.J. Crutzen, R. Haas, P.S. Jehenson, A. Lamourox, "Application of Tamper-Resistant Identification and Sealing Techniques for Safeguards," International Atomic Energy Agency symposium on Safeguarding Nuclear Materials, 20-24 October 1975, Vienna, IAEA

4.3 CONVERSION OF FUEL CYCLE MATERIAL TO WEAPONS MATERIAL

A country desiring an efficient, reliable nuclear weapons capability, particularly one that would be handled by sophisticated delivery systems, would have to convert the fuel cycle material into weapons material. This section summarizes the resources and physical facilities required to convert either plutonium nitrate or plutonium oxide to plutonium metal. The conversion of highly enriched uranium to weapons material is somewhat similar, except that less attention would have to be paid to the containment of the uranium within the process. The emphasis is on well-known and proven production processes as opposed to R&D or laboratory scale operations. Other processes do exist and may be more efficient. The required techniques and equipment are more complex, however.

Assumptions

1. The simplest operational facility will be scoped.

2. No seismic or tornado design requirements will be imposed on the design.

3. Minimum contamination containment systems will be provided (i.e. only one level of filtration of process cell air will be considered).

4. No scrap recycle or recovery will be provided. All scrap material will be treated as waste.

5. All solid waste will be disposed of by shallow land burial.

6. All liquid waste streams will be disposed of by cribbing (shallow land disposal).

7. All remote operations will be done in glove boxes.

8. Recovery rates as low as 85% of the original material as metal are acceptable.

9. All unit processes are batch type.

10. The two most important criticality control Parameters will be batch size and equipment design.

There are two separate process steps to be considered. The first is the conversion of plutonium nitrate to plutonium oxide; the second is the conversion of plutonium oxide to plutonium metal. It is possible to go directly from the oxide to the metal or to by-pass the oxide on the way to the metal, but these processes will not be considered because of the assumptions stated **above.**

There are two well-known, dependable processes for the conversion of plutonium nitrate to plutonium oxide. These processes are referred to as plutonium peroxide precipitation and plutonium oxalate precipitation. The flowsheet concepts are the same for both cases. The plutonium feed streams are first adjusted for plutonium and acid concentrations. Next, the precipitation agent, either oxalic acid or hydrogen peroxide, is added. This is followed by a digestion period, then filtration of the plutonium precipitate. The plutonium cake is then decomposed to plutonium dioxide by calcining the cake. Figure 51 outlines a typical plutonium oxalate precipitation flowsheet.

The conversion of the oxide to the metal is a two step process. The plutonium oxide is first converted to plutonium tetrafluoride by reacting the oxide with dry hydrogen fluoride. Secondly, the tetrafluoride is reduced to metal by the high temperature, high pressure reaction with metallic calcium.

plutonium metal can be prepared in the massive state by reducing any of the several plutonium halides with an appropriate alkali or alkaline earth metal. In practice, plutonium fluoride is used, principally because it is nonhygroscopic. Figure 51 shows a typical material balance for reduction of plutonium fluoride.

Operational Cycle

All unit operations are assumed to be batch type separations. This allows the design of the equipment to be simple and manually operated. Criticality control is also important in the design.

Based on a batch operation with approximately 5 Kg of plutonium being processed, the following operational time cycles might be expected:

1.	Feed adjustment, precipitation, and digestion	1.5 hours
2.	Filtration	0.75 hours
3.	Drying, calcination, and hydrofluorination	4.5 hours
4.	Reduction	5.0 hours
		11.75 hours

FEED

P u o IOOG/LITER(IOO%)
H⁺ 5M
N O 5.8M
10 VOLUMES

OFF GAS

O* 150 VOLUMES
(S. l. P.)

STRIKE SOLUTION

50% H_2O_2 0.5 VOLUMES
1 M$H_2C_2O_4$ 10.5 VOLUMES

PRECIPITATION VESSEL

SLURRY

p_u4+ 0.5 G/LITER
Pu(c₂0₄)2 O.2M
H⁺ 3.0M
NO; 2.8 M
$C_2O: -$ 0.1 M
DIG TIME 1 HR.
DIG TEMP 55° c
21 VOLUMES

WASH

HN0₃ 2M
H_2C2O4 0.05M
5 VOLUMES

FILTER

CAKE

$Pu(C_2O_4)_2 6H_2O$
Pu 99%
HN0₃ 0.8M
H_2O 20M
2 VOLUMES

WASTE
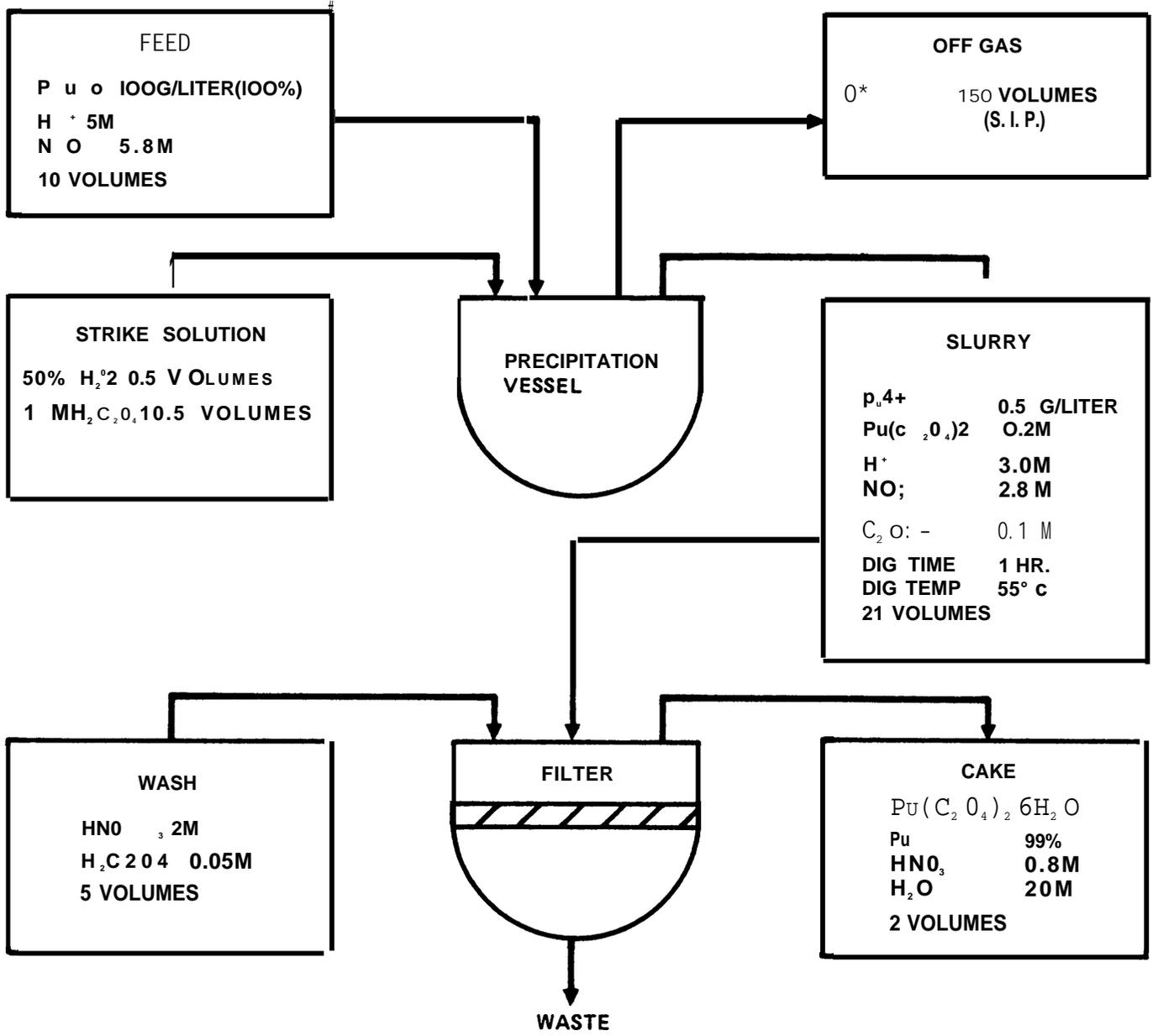
Fig. 51   Plutonium Oxalate Precipitation Flowsheet

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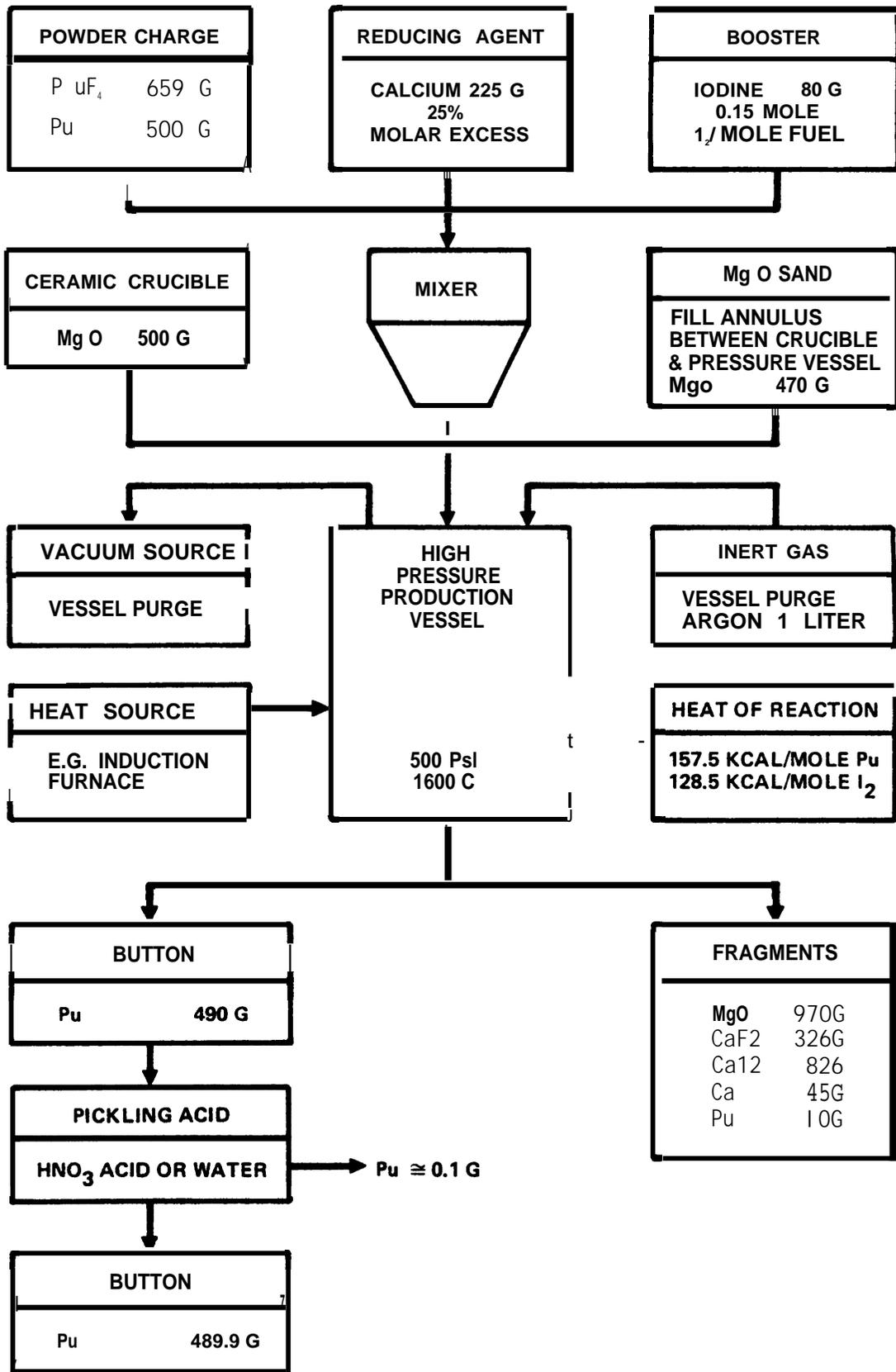
Fig. 52. Plutonium Fluoride Reduction Material Balance

The total cycle time for a one 5 Kg batch of plutonium metal is approximately 12 hours. But it is important to note that there are two independent operations which require a majority of the process time. These are the hydrofluorination step and the reduction step. This means that both operations could be conducted at the same time, allowing a 5 Kg batch of plutonium to be produced every six hours. This means a total of 20 Kg of plutonium metal per day, with a single process line using all of the equipment designed to handle 5 Kg batches of plutonium.

The selection of 5 Kg batch sizes is based on criticality considerations, in that it is reasonably easy to design critically safe equipment for 5 Kg batch sizes. Larger batch sizes present more of a problem.

For a single process line, it is conservatively estimated that all of the process and support areas could be contained within a 3,600 ft^2 building. The actual glove box process area would be no larger than 350 ft^2 with operational glove ports in the front and maintenance access through the back.

Cost Estimates

The following order-of-magnitude estimate is for a **one** process line operation.

Process Equipment	$ **750,000**
Building	**350,000**
Piping and Instruments	**150,000**
Glovebox system	**75,000**
Building Ventilations	**45,000**
Personnel Support Systems	**50,000**
TOTAL	$ 1,420,000

These costs include all direct and indirect costs, such as design, engineering, construction and startup. The numbers are only representative of the simplest operating facilities.

There are three important time increments in any construction schedule. These time periods are:

- Scoping and design

- Procurement

- Construction

The controlling scheduling factor for the construction of a facility such as the one being discussed is procurement. The reason for this is that several major items of equipment must be fabricated to specific design and require somewhat unusual materials. The three longest lead items for procurement are the hydrofluorination vessel, the hydrofluorination furnace, and the reduction furnace. Following the detailed designs of these items, procurement would probably require 14 to 24 months fabrication.

It is important to note that detail design and construction of the building and operational support systems can be underway during the procurement phase. With good coordination between all phases of the project, operational startup could be between **24** to **36** months after the start of the project.

5. SUMMARY OF REACTOR CHARACTERISTICS

There are many characteristics pertinent to power reactors that are decisively important to countries that wish to purchase a reactor. The previous sections discussed in detail the general technical and quantitative aspects of the power reactor fuel cycles. This section will summarize some of these technical detail-s concentrating on the resource utilization and safeguards characteristics. Many of the other characteristics are often subjective by nature or specific to a particular situation and not amenable to a generalized technical discussion.

A partial list of reactor characteristics that might be considered by a country embarking upon a nuclear power program should include the following:

1. Remaining R&D problems and requirements
2. Resource requirements
3. costs
4. Fuel cycle independence
5. Design available in size desired
6. Environmental effects
7. Safeguard characteristics

Remaining R&D problems and requirements are listed first on the list, not necessarily because it is the number one consideration, but because it focuses attention on the power reactor fuel cycles that would be considered by a country that wishes to <u>purchase</u> a reactor. At the present time only the light water reactors and the CANDUS are available in essentially off-the-shelf designs.

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The other reactor concepts require considerable R&D to achieve
a commercial status. We will limit the remaining discussion to
the reactors that might be commercialized within the next decade:
PWR, PWR with recycle, BWR, BWR with recycle, CANDU-PHW, HTGR,
AGR and LMFBR.   (Over the next decade isotopic enrichment methods
are likely to advance faster than the other reactor concepts --
consequently in 10 years, uranium enrichment may have the clear
edge as a potential proliferation pathway.)

Table 19 summarizes the equilibrium fuel cycle material
flow characteristics along with other pertinent characteristics.
The material utilization is normalized to 1000 MWe and a 75%
capacity factor.   The requirements for a BWR recycle reactor is
included in Table 19, however, it must be noted that there are
many recycle fuel management plans that could be utilized, some
of which differ significantly from that listed in Table 21.

The net fissile material utilized to produce 1000 MWe is
substantially less for a heavy water reactor like the CANDU
than for other reactor types (approximately 33% less than for
a PWR).   This efficient use of fissile material is a direct
manifestation of the superior neutron economy of heavy water
reactors.   If the amount of fissile material remaining in the
tails of the enrichment process is included this difference is
even higher.   On the other hand, the amount of energy necessary
to supply the heavy water is not included.   In addition, a more
efficient utilization of fissile material could be achieved in
the LWR's via the use of a tandem fuel cycle, recycle fuel cycle
management plan, or an optimum throw-away fuel cycle.   There is
little room for improvement in a heavy water fuel cycle without
substantial reactor design changes.

Table 9. SUMMARY OF REACTOR MATERIAL FLOW

	CANDU-PHW	HTGR	AGR	PWR	BWR	BWRRC	LMFBR
REACTOR TYPES							
Mil ing (MT)*	125	72	155	130	1..	79	--
SWU (MT)	--	93	89	128	114	70	--
Uranium Enrichment (%)	.71	93	2.1-2.5	3.3	2.61	--	--
Fiss @ in (MT)	.88	.66	.76	.84	.85	.98	1.14
F x le out (MT)	.56	.25	.35	.36	.45	.54	1.34
Total Fissile (MT) (including tails)	.32	.55	.61	.74	.65	.59	-.2
Burnup MWD/MT	7,500	87,000	20,000	33,000	26,160	26,160	67,600
Initial Fissile Inventory (MT)	1.02	2.96	2.14	3.17	4.5	3.59	
Net Plant Efficiency	29.6	38.7	41.6	32.5	32.5	32.5	41.8

*Includes recycle

costs are certainly an important point in considering a reactor purchase. However, the total capital cost often is determined by the financial arrangements, We note here that the total capital costs of LWRs are thought to be at least 10% less than for an equivalently sized CANDU.

Natural uranium reactors such as the CANDU offer fuel cycle independence to those countries not having enrichment facilities. In addition, they are generally available in smaller sizes (600 MWe is to be a standard export CANDU whereas most present LWR designs are over 1000 MWe) . The smaller size power station is often more compatible with a country's electric power grid. (Because of the necessary shutdowns for refueling and maintenance, no single power station should be more than approximately 15% of the total grid capacity.)

The environmental effects of the various reactors do differ. The lower efficiency reactors such as the CANDU promote more thermal pollution. Some reactors produce more of certain types of radioactive isotopes. However, none of these considerations are decisive in a proliferation potential discussion.

In Table 20 some of the reactor safeguards considerations relative to various threats are summarized. If a country manufactures fuel rods from raw materials then the LWR with recycle, the HTGR, and the LMFBR could be credibly threatened by any of the groups listed. If a country receives only fresh fuel assemblies then the threat diminishes somewhat for these fuel cycles. All of the irradiated fuel from the various reactors contain strategic special nuclear material. A credible threat can be directed against this material if the group can put together the necessary facility to separate this SSNM from the highly radioactive fission products.

Table 20 REACTOR SAFEGUARDS CONSIDERATIONS

MATERIAL EXPOSURE

Reactor Fuel Material	LWR	LWR with Recycle	CANDU-PHW	HTGR	AGR	LMFBR
Fuel (Raw Materials)	0	3	0	3	0	3
Fuel Assemblies	0	2	0	2	0	2
Irradiated Fuel	1	1	1	1	1	1

0   No Strategic Special Nuclear Material (SSNM) involved

1   SSNM involved but must be separated in reprocessing plant

2   SSNM can be separated by standard chemical and mechanical processes (no high radiation)

3   SSNM could be converted with little or no processing into a weapon

MATERIAL EXPOSURE RATING VERSUS THREAT

	Rating			
	0	1	2	3
Subnational Group				Credible Threat
Nation Desiring a Quick Response Capability				Credible Threat
Small Nation			Credible Threat	
Large Nation		Credible Threat		

REACTORS VERSUS THREAT CREDIBILITY

	LWR	LWR with Recycle	CANDU-PHW	HTGR	AGR	LMFBR
Subnational Group		c		c		c
Nation Desiring a Quick Response Capability		c		c		c
Small Nation	u	c	c	c	u	c
Large Nation	c	c	c	c	c	c

C— credible threat

u — credible but unlikelv

It seems obvious that exposure of SSNM in the power reactor fuel cycle either through the use of reprocessing and/or SSNM in fresh fuel assemblies makes the fuel cycle somewhat more vulnerable to a broader spectrum of threats. The decision as to the relative significance of this increased vulnerability might better be answered by other considerations such as:

1. Can effective safeguard measures be incorporated to counter this increased vulnerability?

2. How significant is this vulnerability relative to the use of dedicated facilities?

Extensive programs are being conducted in the United States by ERDA and NRC, and throughout the rest of the world to arrive at acceptable answers to these questions.

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APPENDIX A

DIVERSION POTENTIAL OF CRITICAL
FUEL CYCLE FACILITIES

Fuel cycle facilities that may contain strategic special nuclear material in a separated form are of particular interest to a safeguards and proliferation assessment. The exposure matrix (Table 18) lists the critical facilities for the various power reactor fuel cycles. Enrichment plants, reprocessing and possibly to a lesser extent recycle fuel fabrication plants merit special attention as explicitly implied in Table 20. The capability of an enrichment plant and reprocessing plant to generate separated strategic special nuclear material makes them particularly vulnerable to a diversion threat.

Reprocessing Plant

There have been a number of reasons expressed by various countries for acquiring a reprocessing plant. Included among these reasons are typically the following:

1) Going to light water recycle fuel to more economi-cally operate LWRs, con-serve uranium resources, and to gain more control over their LWR fuel cycle.

2) Eventually expect to go to a breeder fuel cycle which must include reprocessing. The reprocessing plant is needed to gain the necessary experience.

3) Spent LWR fuel assemblies have to be reprocessed to generate an acceptable waste disposal form.

All of these responses have a base of validity; however, the firmness of these bases is uncertain at the present time. For example, the economic gain in using LWR recycle fuel is a rather sensitive function of uranium prices and the cost of reprocessing. Neither of these costs are firm at the moment and they have a rather large uncertainty for a time period eight years in the future (following the decision to

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build a plant) when a reprocessing plant might come on line.
The conservation of resources argument is certainly true
with a most likely uranium savings of 25%. However, as
discussed in Section 4.1,   there are alternative
ways of achieving roughly the same gain in resources without
resorting to the use of recycle LWR fuel. The other argu-
ments (2 and 3) also have rather large associated uncertain-
ties which preclude them from being definitive statements
on a perceived positive requirement for a reprocessing plant.

For example, it is clear that any breeder fuel cycle
requires reprocessing. However, it is not clear that breeder
reprocessing will utilize the PUREX process that is standard
for production metal fuels and the oxide fuels of LWRs.
Consequently, experience in reprocessing LWR fuel may only
apply in a limited way toward the reprocessing of LMFBR fuel
because of the higher burnup, higher plutonium throughput,
and possible fuel dissolution problems with LMFBR fuel.

The West Germans have repeatedly stated that reprocessing
of spent LWR fuel gives them a flexibility in developing waste
disposal forms that will be acceptable for disposal in their
country. Until recently, U.S. policy has assumed a base
case fuel cycle that includes reprocessing and waste treatment
prior to disposal. The alternative of directly disposing
spent fuel assemblies (after some treatment) has been assumed
to be technically feasible. There has not been an extensive
experimental program to demonstrate this, however.

An additional argument against a country developing an
indigenous reprocessing capability is that analysis indicates*
that large plants are by far the most economical. Large plants
mean a capacity of 1500 to 3000 MT (roughly 15 to 30 MT of Pu)
capacity per year, or fifty to one hundred 1000 MWe nuclear

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*
 "Light Water Reactor Fuel Recycle," Savannah River Laboratory
 Quarterly Report, DPST-LWR-76-1-1, January-March, 1976.

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power reactors.   Few countries outside the U.S. are expected
to develop a nuclear power reeactor electric generating capacity
of this magnitude.   Moreover, there are design problems
associated with large capacity reprocessing plants (particu-
larly for those greater than 1500 MT capacity) that have not
been completely resolved.   These problems are primarily
related to ensuring that a sufficient criticality safety
factor is achievable under all credible circumstances.

Reprocessing plants have one obvious proliferation poten-
tial when viewed in the context of a national threat.   If the
host country abrogates the Non-Proliferation Treaty and refuses
to allow IAEA inspection of the operating reprocessing plant,
then the country can rather overtly proceed to generate
weapon material from the stockpile of spent fuel.   (The likely
amount of Pu that could be separated from spent power reactor
fuel is listed in Appendix B1 for the various countries.
One result of   an overt proliferation attempt might
be the shutoff of the imported fresh LWR fuel from supplier
countries.

An assessment of covert diversion of plutonium from a reprocessing plant
by the host country or by a subnational group requires some
consideration of the material form and flow through the re-
processing plant.

Figure Al* is a schematic illustration of the principal
physical areas and average daily material flow through a model
1500 MT\year reprocessing plant.   The principal areas are:

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* G. Bray and H. Kendrick, 'Spent Fuel Reprocessing Plant
  Characteristics Important to an Integrated Safeguards Design".
  INMM 17 Annual Meeting, Seattle, Washington. January 22-24,
  1976.   PP. 485-494.

Figure A1. AVERAGE DAILY PLUTONIUM FLOW FOR MODEL 1500 MT HEAVY METAL / YEAR SPENT−FUEL REPROCESSING PLANT

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- Spent fuel receiving and storage
- plutonium, uranium and fission product separation
- Plutonium nitrate blending and storage
- Plutonium nitrate-to-oxide conversion
- Plutonium oxide storage and shipping

The average daily plutonium flow between plant areas is shown in the figure. The daily plutonium flow is expressed in kg Pu/day and as a percentage of the average 50 kg Pu/day assumed coming into the plant. Plutonium **assay** stations and accountability tanks, which are required for material balance accounting are also shown in Figure Al. The data and physical arrangement shown is intended to be representative of present design criteria.

The cress-hatched enclosures of the spent fuel receiving and storage area and the separations area indicates the highly radioactive nature of these areas. Material in these areas is not attractive for diversion.

The plant processes manifest certain key characteristics which relate to the <u>attractiveness</u> of the material to potential diverters, and to the <u>accessibility</u> of the material. The attractiveness of the material can be expressed in terms of the form (chemical composition, physical form, packaging, etc.), concentration (grams plutonium per gram of material), and the presence or absence of hazardous levels of radioactivity. The accessibility of the material is related to its form, the type of processing or material-handling equipment used, the degree of automation of that equipment, and the quantity of plutonium available in a given location.

Key process characteristics have been determined for all areas of a model reprocessing plant, including both mainstream and sidestream material flows, and are shown in Figure A2

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Figure 2A(a) – Reprocessing Plant Characteristics

Figure 2A(b) – Reprocessing Plant Characteristics

Quantity Normally in Process Area

Kg Pu

100 200 300 400

REPROCESSING PLANT

SPENT FUEL RECEIVING AND STORAGE — 1000

SEPARATIONS

MAINSTREAM
Pu NITRATE BLENDING AND STORAGE — 2000
Pu NITRATE-TO-OXIDE CONVERSION
Pu OXIDE STORAGE — 7000
Pu OXIDE SKIPPING — (0)

SIDESTREAM
ANALYTICAL LABORATORY — 8
WASTE ASSAY AND STORAGE — (0)

QU

WEIGHT % Pu

25 50 75 100

POSSIBLE RANGE

POSSIBLE RANGE

POSSIBLE RANGE

DIFFERENT MATERIAL FORMS

SPENT FUEL ASSEMBLIES

AQUEOUS AND ORGANIC Pu SOLUTIONS

Pu NITRATE SOLUTION

Pu SOLUTIONS, SLURRIES, OXIDE POWDER

Pu OXIDE POWDER IN SEALED CONTAINERS

Pu OXIDE POWDER IN SEALED CONTAINERS

Pu SOLUTIONS AND POWDERS

WASTE IN SEALED CONTAINERS

Handling and Process Techniques

	MANUAL	GLOVE BOX	REMOTE	AUTOMATED
			x	
			x	
			x	
		x	x	
		x	x	
	x		x	
		x		
	x	x		

Scrap and Waste Generation

Kg Pu/DAY

.1 .2 .3 .4

(TRACE)

0.75

1.5

s

(0)

(0)
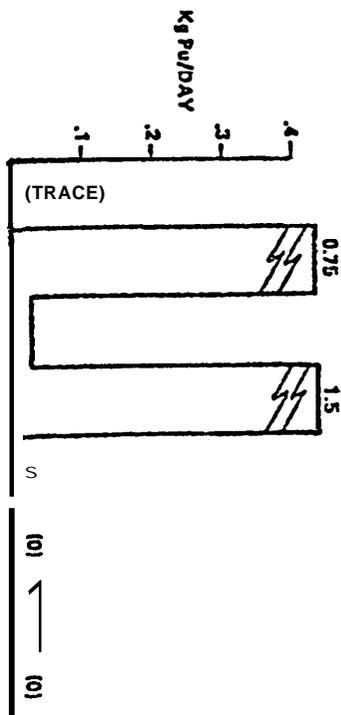
Time to Acquire 100 Rem Dose at 1 meter from 1 Kg

DAYS

.01 .1 1 10 100 1,000 10,000

REPROCESSING PLANT

SPENT FUEL RECEIVING AND STORAGE

SEPARATIONS

MAINSTREAM
Pu NITRATE BLENDING AND STORAGE
Pu NITRATE-TO-OXIDE CONVERSION
Pu OXIDE STORAGE
Pu OXIDE SHIPPING

POSSIBLE RANGE

SIDESTREAM
ANALYTICAL LABORATORY
WASTE ASSAY AND STORAGE

POSSIBLE RANGE
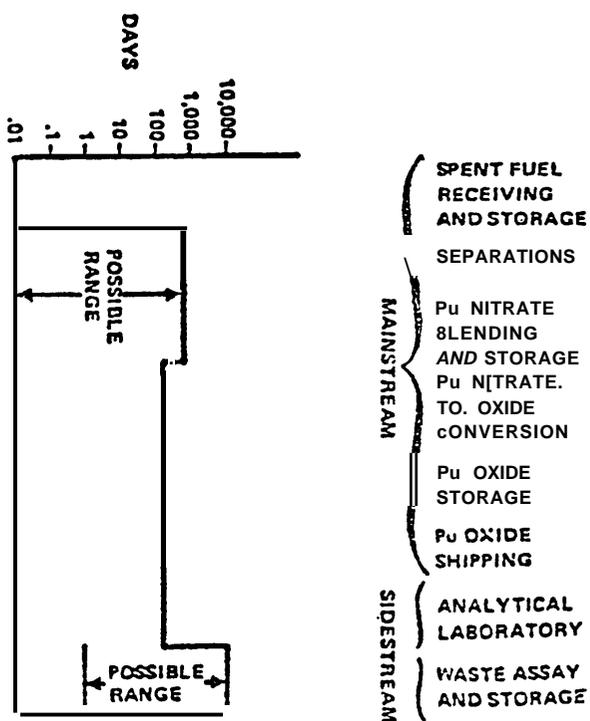
These figures illustrate the important differences in process characteristics across the plant. The figures 'on the left side of Figure A2 represent the maximum quantity of plutonium that normally might be present in the various plant areas, the chemical form of this material, and the possible range of plutonium concentration in this chemical form. Note that the high points on the figures represent the more attractive material from a diversion standpoint.

The three figures on the right side of Figure A2 are measures of the accessibility of the material in the various plant areas. The top figure on the right illustrates the time necessary to acquire a 100 REM dose at a one-meter distance from one kilogram of material. As before, the high points on the figure represent the more accessible material (i.e., less radioactive) . Scrap and waste (middle figure, right side of Figure A2) generally is considered to be more accessible material partly because it may indicate a process upset and often is more difficult to accurately measure. Even though the scrap may not meet the customer's specifications, it may still consist of relatively concentrated Pu material that can be reclaimed and transformed to weapon material. In addition, scrap and waste represent outgoing material streams from the main process line. The lower righthand figure illustrates the handling and process technique. The accessibility to the Pu-containing material is rated from manual through automatic.

Considering all of the characteristics of the various areas delineated in Figure A2, the Pu nitrate blending and storage area, the Pu nitrate to oxide conversion area, and the analytical laboratory all would appear to be critical areas for material measurements and accounting. An indication of an accounting system detection capability for the conversion area of our model reprocessing plant that is operating with presently available measurement equipment and an accounting interval

period of two months (U.S. NRC requirement) is that an individual or group has a 50 per cent chance of avoiding detection by the accounting system if they divert 13.2 kilograms of plutonium. For the analytic laboratory, the SO per cent probability of detection amount is 1 kg of Pu for the two-month accounting period. If these systems were upgraded to state-of-the art measurement instrumentation, approximately 8.7 kg of Pu from the conversion area, or 0.5 kg from the analytical laboratory, or 11.7 kg Pu from the nitrate blending and storage area could be diverted with a 50 per cent probability of detection by an accounting system. It is highly likely that these upgrade numbers represent the most optimistic detection capability since many of the assumed measurement capabilities are laboratory results that may degrade when introduced into a production facility. The larger numbers quoted above are achievable with proven measurement hardware and procedures.

Accounting procedures monitor nuclear material in the main process stream, the associated sidestreams and, of course, the sealed, item count material in storage. Undetected diversions from these areas are statistically possible for measurement and accounting systems as discussed above. However, the material must then be removed from the process line or storage area. Now a number of containment and surveillance procedures come into play that are designed to detect the removal of the nuclear material from its authorized location. Many of these surveillance procedures can utilize tamperproof hardware that can be left unattended by IAEA inspectors. A partial listing of these procedures include the following:

- portal monitors for nuclear material
- CCTV surveillance
- door or glove box access alarms
- motion detectors
- random personnel or package searches
- clothing changes
- randomly located radiation detectors

The applicability of specific procedures varies according to the particular area; however, it should be possible to design and implement a set of redundant protective measures to meet a specified subnational threat (for example, six adversaries with two being inside employees) to the material access areas. Consequently, an undetected diversion of nuclear material would have to overcome the containment and surveillance systems in addition to the measurement and accounting systems.

The reprocessing diversion potential as described above differs if an alternative reprocessing scheme is employed. For example, if a coreprocessing scheme (see Section 4.1) is used so that none or only some of the uranium is separated from the plutonium, then the model plant is changed along with measurement procedures. Most significantly, the amount of material a divertor would have to remove to obtain a strategic amount of special nuclear material is increased. In addition, the divertor would have to chemically separate the plutonium from the uranium. For subnational groups, the required chemical separation could be a significant obstacle to overcome.

An enrichment plant is the only other nuclear fuel cycle facility besides a reprocessing plant capable of generating separated strategic special nuclear material. A significant difference exists, however, in that a reprocessing plant normally handles an intrinsic strategic special nuclear material--plutonium; whereas an enrichment plant built to service a LWR fuel cycle would only generate slightly enriched uranium (approximately 3% ^{235}U) . Consequently, the diversion/proliferation potential of an enrichment plant exists in its <u>potential</u> capability for generating highly-enriched uranium (typically 90% or greater ^{235}U) .

Since most commercial enrichment plants will only be designed and operated to produce slightly enriched uranium, the proliferation threat to the facility is limited somewhat. It is not credible that a small group of adversaries (particularly outsiders) could subvert the normal plant operation to produce highly-enriched uranium. A credible threat would have to include most of the plant management and many of the operating personnel. Thus, a diversion threat directed by the operating country is of most concern for an enrichment facility.

The remainder of this section on enrichment plants will be primarily directed toward centrifuge enrichment plants. There are two main reasons for focusing this discussion of the proliferation potential of enrichment plants on centrifuge techniques and facilities rather than the present dominant enrichment technique of gaseous diffusion:

1) Centrifuge plants can be initially constructed with a modest separative work capacity and then added to as the demand for enrichment services grows. Gaseous diffusion plants make economic sense only in very large capacity such as 9,000,000 kg SWU\year.

2) Gaseous diffusion plants require approximately 15 times more electric **power per** SWU than centrifuge plants.

(The Becker nozzle process which requires somewhat fewer stages but consumes more than twice as much electric **power as gaseous diffusion is also not an attractive choice for production of weapons material.)** These **points are discussed in more detail in Section** 2.5 and 2.11. The primary effect on the above is that small enrichment plants, particular those build with a limited amount of capital, will undoubtable be centrifuge plants. The advanced isotope separation processes such as laser enrichment techniques are not likely to be available for use before 1985.

Before concentrating on the centrifuge technique, it is of interest to note a few technical differences between gaseous diffusion and centrifuge enrichment that relate to their diversion potential.

- Enrichment Limits

    The amount of enrichment obtainable from a single barrier in a gaseous diffusion plant (GDP) is small and relatively fixed. Consequently, all barriers are connected in series, thus limiting the maximum enrichment that can be obtained with natural uranium feed. The amount of enrichment from a single centrifuge is much larger and variable as discussed below.

- Electric Power Usage

    The amount of separative work performed in a GDP is proportional to the power used to pump the UF6 through the plant. Thus, the power usage is a direct and external indicator of the SWU actually performed. The power used to drive the centrifuges is relatively small and a poor separative work indicator.

- process Inventory

    The process inventory of a GDP is orders of magnitude larger than that of a centrifuge enrichment plant. Since there is always a rather large uncertainty associated with the process inventory, this could create a rather large MUF in GDPs.

To facilitate a coherent discussion of centrifuge enrichment plants, we shall develop a reference centrifuge plant following Kouts.* The characteristic parameters listed are nominal values traceable to results reported in the literature; however, there may be significant differences between these reference values and those actually utilized in a specific centrifuge plant. For example, European centrifuge development (Urenco) is known to be concentrating on developing highly reliable (failure rate less than 2 per cent per year), low-capacity centrifuges, whereas American development efforts are reported to be directed toward higher separative work capacity machines with a reduced reliability. Hopefully, the reference characteristic numbers are somewhere in between these values.

Table A2 lists the design characteristics of the reference centrifuge enrichment plant. Note that this is a very small plant (when compared with U.S. diffusion plants), representative of the initial enrichment plants expected to come on-line in Europe and Japan. A mature LWR industry would require much larger plants with at least an order of magnitude or more additional capacity.

Table A3 lists the design material flow through the individual cascades. Table A4 illustrates the cascade that produces 4% enriched ^{235}U. Note there are 29 stages with varying numbers of centrifuges in each stage. The other cascades that produce a lower enrichment product are similar in nature with fewer stages.

---

*
Herbert Kouts, "Reference Uranium Enrichment Plant," Technical Support Organization, Brookhaven National Laboratory, December 6, 1972.

TABLE A2

REFERENCE CENTRIFUGE PLANT

Total Separative Capacity MT SWU/Yr	200
Unit Separative CapacitY kg SWU/Yr	5
Number of Cascades	5
Number of Centrifuges Per Cascade	8,000
Total Feed Rate MTU/Yr (natural uranium)	325.2
Total Product MTU/yr (various enrichments)	60.8
Total Tails MTU/Yr at .25% ^{235}U	264.4
Centrifuge Floor Area ft^2	200,000
Centrifuge Building Area ft^2	320,000
Nuclear Power Industry Supported 1000 MW(e)	about 1.5

TABLE A3

MATERIAL FLOWS IN THE CASCADES

Product Enrichment (% U -235)	Feed (Tonnes U/Yr)	Product (Tonnes U/Yr)	Tails (Tonnes U/Yr)
2.0	79.4	20.9	58.5
2.5	68.7	14.1	54.6
3.0	62.6	10.5	52.1
3.5	58.7	8.4	**50.3**
4.0	55.8	6.9	48.9
TOTAL	325.2	60.8	264.4

TABLE A4
8000 CENTRIFUGE
(numbers represent centrifuge per stage)

4% ^{235}U Product

Stage
12
26
41
57
75
95
116
139
164
193
223
257
293
334
378
426
480
538
602
569
532
492
449
401
348
291
228
159
82

Enriching
Stages

25% ^{235}U Tails
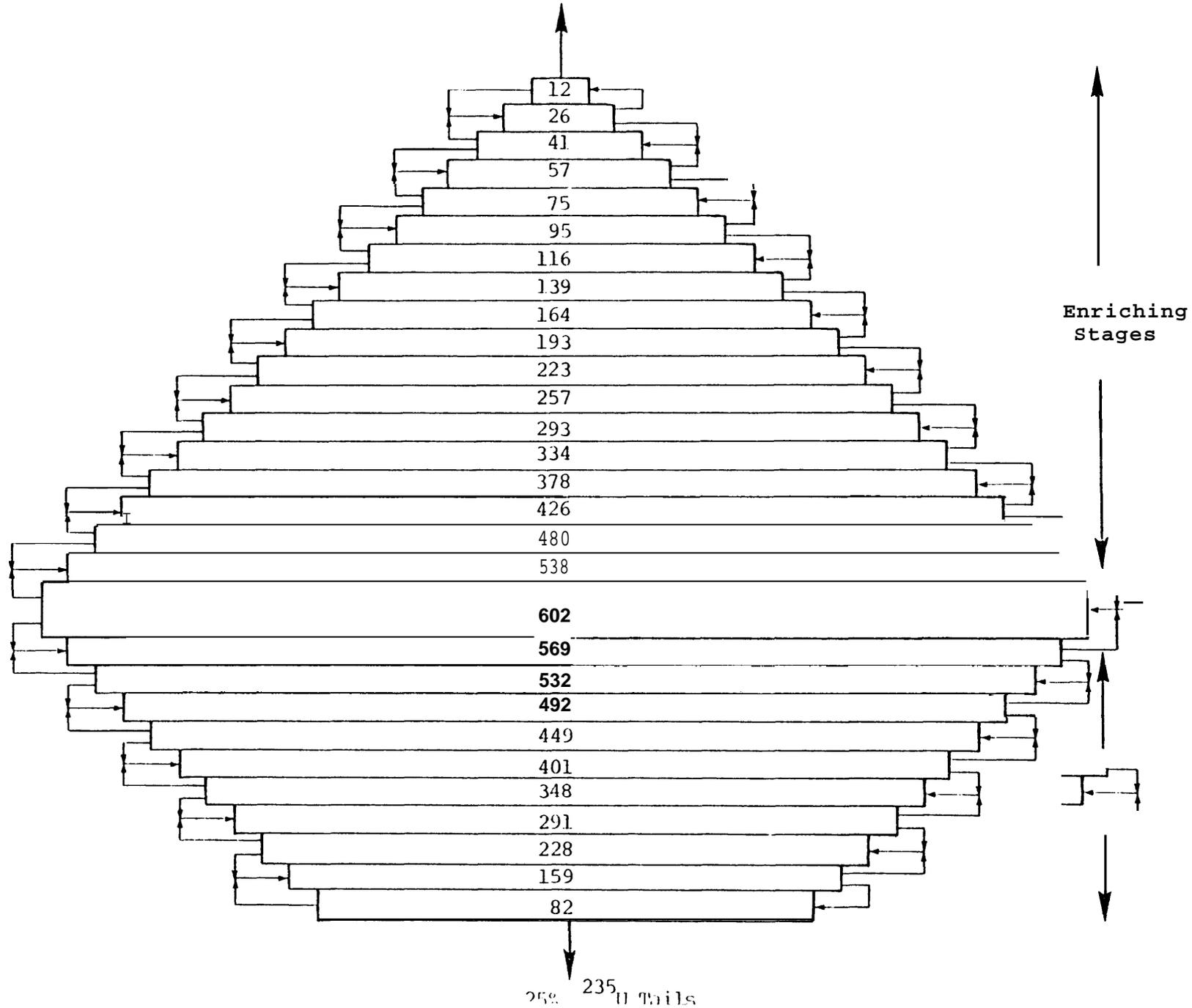
An important feature to note about the above example is that centrifuge enrichment plants are inherently versatile--much more so than gaseous diffusion plants.

This versatility that is inherent with the overall configuration of centrifuges in an enrichment plant also extends to the operation of individual centrifuges. Figure A3 illustrates the characteristic curies of enrichment for a gaseous centrifuge. The top figure shows the variation in separative work with feed rate and the bottom figure the variation in the enrichment factor with the feed rate. The implication of these curves are that changes in the plant operation can produce an enriched product that is higher than the design enrichment.

Measurements and material accounting in enrichment plants that produce slightly enriched uranium can be viewed from a different prospective than for reprocessing plants because the normal product material does not contain strategic special nuclear material. Nevertheless, material accounting can be important for some diversion scenarios and it is of interest to consider the various loss mechanisms as causes of inventory uncertainty. A comprehensive list of uranium loss mechanisms would have to include:

> traps in vacuum system
> centrifuge failure
> centrifuge maintenance
> accidental losses
> wet air inleakage
> reaction of UF_6 with impurities
> intermetallic diffusion
> surface absorption of UF_6
> active chemisorption of UF_6

The importance and absolute gram value of each of these mechanisms is difficult to predict in the absence of experimental data from a production centrifuge plant. The total loss might be comparable with that of a GDP which is 0.5 per cent of the product (for 4 per cent enrichment) .
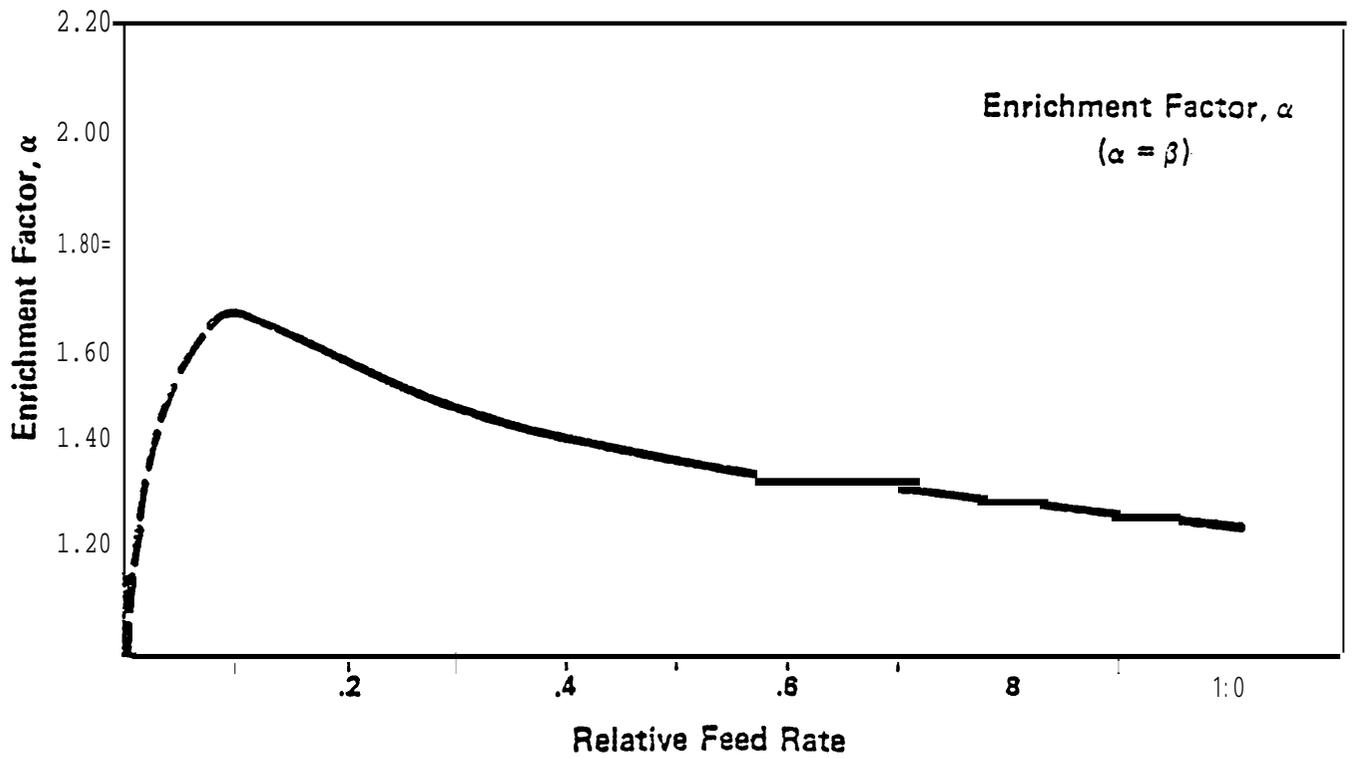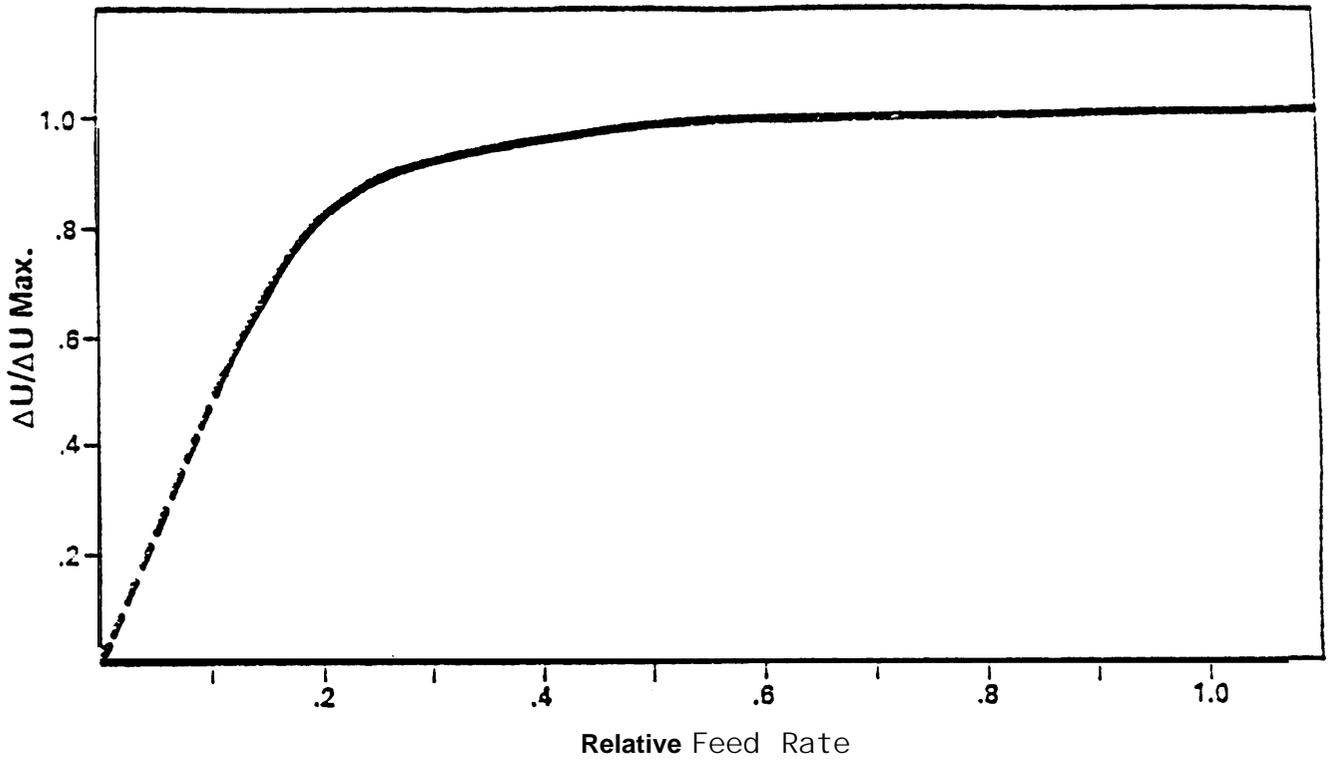
Figure A3. Assumed Centrifuge Characteristics

A quick consideration of material diverted within the material **measurement and accounting uncertainly indicates that this amount of** material is not very important for a plant the size of the reference plant. Assume a physical inventory every three months. The expected accuracy of material accounting (IAEA) should be .2 per cent of the throughput for this period.   This corresponds to roughly 1 kilogram of highly-enriched uranium every three months if all this could be diverted and enriched to **90%** ^{235}U.

-Table A5 shows typical inventories inside the fenced area.   We note that the major inventories are in the feed, product, and tails.   These can be measured quite accurately .2% and thus contribute a rather small total uncertainty to the plant inventory.

To complete the reference plant model, a layout of the plant is shown in Figure A4.   The cascade area (which at present is not accessible to IAEA inspection) also contains the sensitive areas associated with the plant such as the centrifuge maintenance, decontamination, and test areas as well as the control room.   In the U.S. and possibly else-where an outside perimeter fence would surround the entire plant.   The key measurement points (KMP) for flow, inventory, and surveillance are as indicated on the figure.

As stated above the prime diversion potential associated with these types of enrichment plants is related to an un-declared upgrading of material beyond the maximum enrichment designed into the plant.   There are a number of ways that this could be accomplished:

1)   Reconfiguration of unit cascades
2)   Recycle of plant product
3)   Off-design cascade operation
4)   Internal cascade recycle

TABLE A5

TYPICAL IVENTORIES IN FENCED AREA

Location	kg u	kg U-235
Gas Phase in Centrifuges	40	0.4′
Traps	100	1.0
Deposition in Centrifuges	2000	20.0
Feed*	2300	16.4
Product**	3800	114.0
Tails***	3800	9.5
Waste Recovery	Nominal	

---

*   Assumes 1 half-full cylinder feeding all cascades
      from feed purification, 1 cylinder waiting.

**   Assumes equivalent of half-full cylinder at each
      product withdrawal point.

***Assumes tails from separate cascades sequestered.
      Equivalent of half-full cylinder at each tails
      withdrawal point.
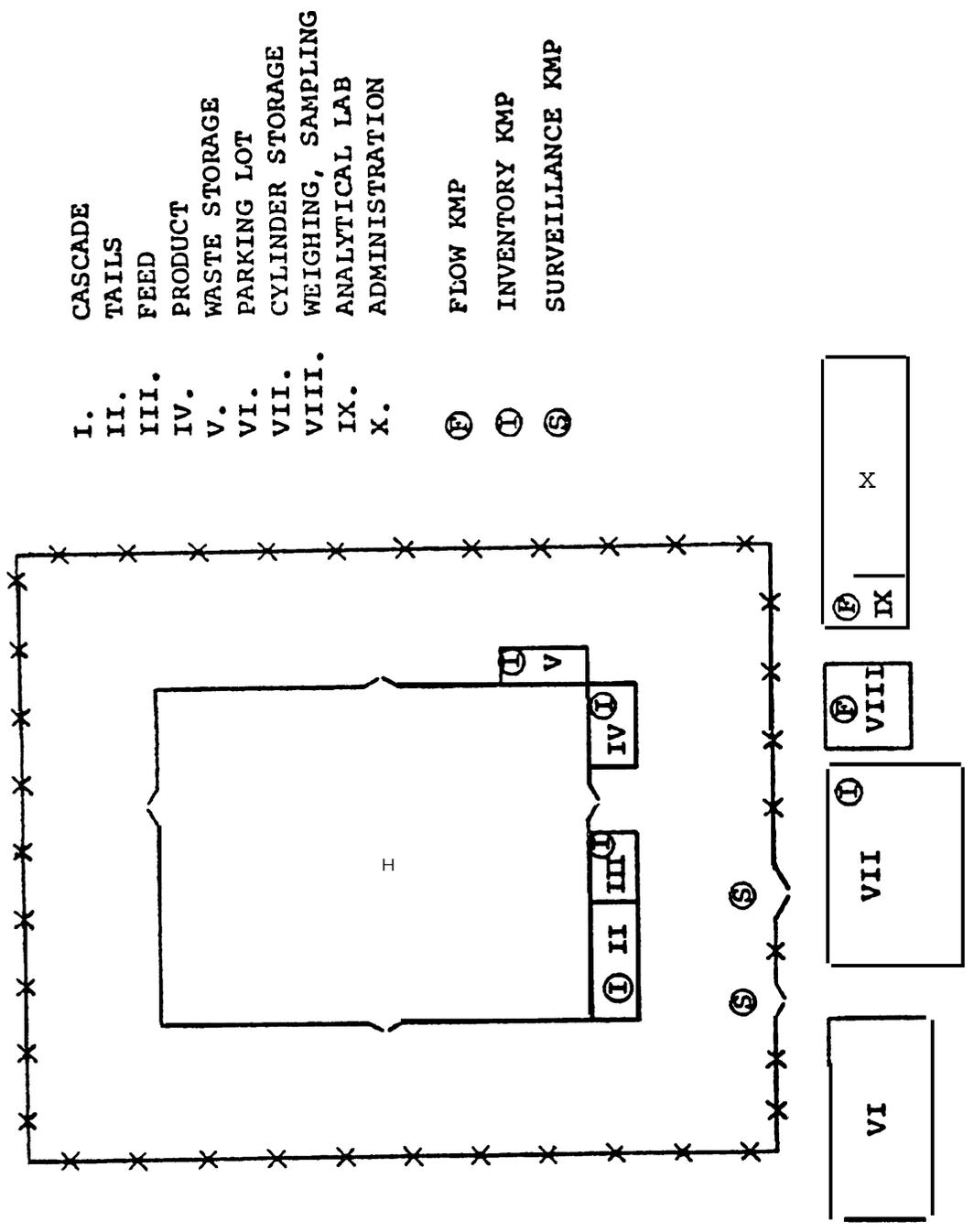
I.   CASCADE
II.   TAILS
III. FEED
IV.   PRODUCT
V.   WASTE STORAGE
VI.   PARKING LOT
VII. CYLINDER STORAGE
VIII. WEIGHING, SAMPLING
IX.   ANALYTICAL LAB
X.   ADMINISTRATION

Ⓕ   FLOW KMP
Ⓘ   INVENTORY KMP
Ⓢ   SURVEILLANCE KMP

Figure A4

LAYOUT OF REFERENCE PLANT

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The implications or details of each of these will be discussed in turn.

1) The centrifuges in one or several of the cas-
    cades could **be** reconfigured to take natural
    uranium feed to a 90 per cent enriched product.
    A cascade that would accomplish this might
    consist of 77 enriching stages and 10 stripping
    stages. Each stage could, of course, contain a
    minimum of one centrifuge. Consequently, this
    covert cascade could conceivably be constructed
    from one of the 8000-unit reference cascades.

2) If the 4 per cent product were fed into
    another covert cascade that had a natural
    enrichment tails, then 58 enriching stages
    would be required with 18 **stripping stages.**
    Again, this cascade could be configured from
    one or a portion of one of the model cascades.

3) The feed rate, product rate, and reflux ratio
    (the relative amount of interstage circulation
    to product flow) could be varied to produce a
    more highly enriched product. The maximum
    enrichment obtainable requires a detailed
    analysis; however, it is not likely to exceed
    20% 235U. Thus, this material would require
    further enriching to achieve a 90 per cent
    product. In addition, the off-design opera-
    tion would lead to significant inefficiencies
    in the cascade operation.

4) Each cascade could be equipped with lines to
    recycle cascade product and tails into the
    feed stream. If only the product were fed back
    with a continuing tails withdrawal the product
    assay would rise. The maximum rise might be to
    roughly a 10 per cent enrichment requiring a
    further enrichment for weapons grade material.

TO accomplish these undeclared upgrading operations? a corresponding source of material would have to be developed. Techniques to accomplish this might (1) overstate the MUF; (2) **overstate** the inventory; (3) have a stored undeclared feed; and (4) have a steady undeclared feed and takeoff.

Inspection techniques that are presently proposed for enrichment plants include input-output monitoring of the cascade area (basically considers the cascade area to be black box with a detailed monitoring of all input and output of material) .

Figure A5 shows a schematic of all inputs and outputs from the cascade area of a centrifuge enrichment plant. At present new equipment is an undeclared path. This creates a problem since conceptually unaccounted feed material could be introduced via this feed stream.

```
                              New
                  People   Equipment
                    in        in
                    |          |
                    |          |
                    v          v1
              +-------------------------+
              |                         | ----> Product
   F e e d    |       CASCADE           |
              |        AREA             | ----> Tails
              |                         |
              +-------------------------+
                    |          ^
                    v          |
                  People   Decontaminated
                   out     Equipment out
```
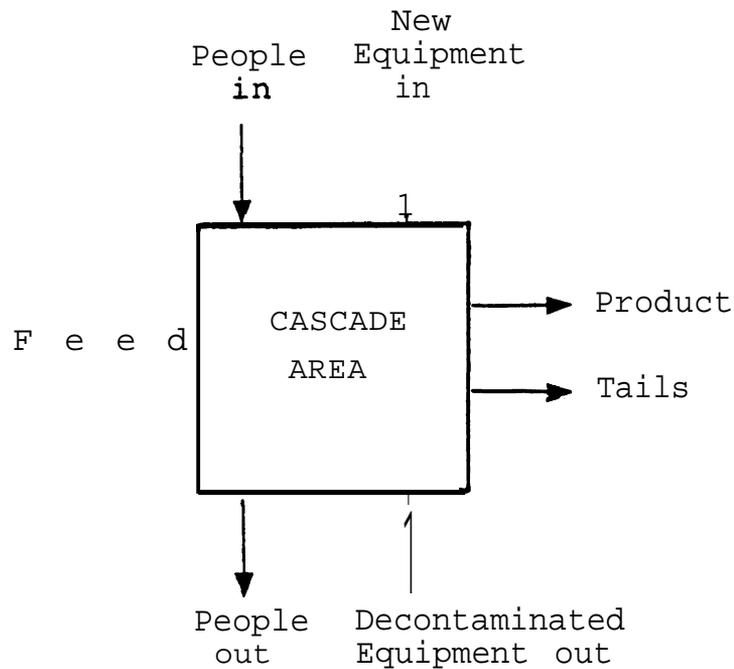
Figure A5

INPUT-OUTPUT INSPECTION STREAMS

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For input-output to be effective, <u>all</u> streams need to be monitored. Physical inventory and surveillance methods need to complement the continuous input-output monitoring. Other complementary inspection techniques would include enrichment monitoring (particularly checking the tails enrichment) and the use of the minor isotope technique (311ST) to check the product and tails U-235/U-234 ratio. Table A6 shows how this would vary for the design product.

The effectiveness of these inspection techniques applied in an integrated way to a centrifuge enrichment plant has not been demonstrated as yet. One improvement that would make all of these inspection methods more credible would be to allow the inspectors access to the cascade area upon demand. Inspection accessibility to the cascade area would certainly not insure that the plumbing changes required for an unde-clared cascade reconfiguration would be detected in the mass of necessary piping in the cascade area. However, it **could** **act** as a significant deterrent for any country wishing to conduct a covert operation. In addition, unde-clared feed, product, or tails takeoff would also have a higher probability of detection.

TABLE A6

MINOR ISOTOPE CONTENT OF STREAMS

Product Enrichment (% U-235)	Product U-235/U-234 ratio	Tails U-235 U-234 ratio
2.0	114	231
2.5	112	235
3.0	111	239
3.5	110	243
4.0	109	245
AVERAGE :	112	238

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APPENDIX B

WORLD POWER AND
RESEARCH REACTORS

Table B-1

WORLD POWER REACTORS

LOCATION	TYPE	Mwe	REACTOR SUPPLIER	START DATE(1)	PU PROD Kg/Yr	69	74	79	84	99
ARGENTINA										
Atucha 1	PHWR	319	SIEMENS	6/74	70	0	35	385	735	1,085
Cordoba	PHWR	600	AECL	1/82	132	0	0	0	395	1,056
Atucha 2	PHWR	600	AECL	1/84	132	0	0	0	132	792
					Total	0	35	385	1,263	2,933
AUSTRIA										
Tullnerfeld 1	BWR	692	KWU/AEB	10/76	118	0	0	374	964	1,554
Stein-St. Pantaleon 1	PWR(2)	1300	-	1/83	221	0	0	0	442	1,547
					Total	0	0	374	1,406	3,101
BELGIUM										
Doel 1	PWR	390	W-ACEC-COCKERILL	2/75	70	0	0	344	694	1,044
Doel 2	PWR	390	"	1/75	70	0	0	292	642	992
Doel 3	PWR	900	FRAMATOM-ACEC-COCKERILL	2/81	162	0	0	0	635	1,445
Tihange	PWR	870	"	9/75	157	0	0	680	1,464	2,250
CNW 1	PWR	1000	W-ACEC-COCKERILL	1/83	180	0	0	0	360	1,260
CNW 2	PWR	1000	"	5/82	180	0	0	0	480	1,380
Tihange	PWR	910	FRAMATOME-ACEC-COCKERILL	4/81	162	0	0	0	611	1,426
					Total	0	0	1,316	4,886	9,797
BRAZIL										
Angra 1	PWR	626	W	3/79	113	0	0	85	650	1,215
Angra 2		1245	KWU	1/85	216	0	0	0	0	1,080
Angra 3		1246	KWU	7/86	216	0	0	0	0	756
					Total	0	0	85	650	3,051
BULGARIA										
Kozloduy 1	PWR	440	AEE	1/75	79	0	0	395	790	1,165
Kozloduy 2	PWR	440	AEE	1/76	79	0	0	316	711	1,106
Kozloduy 3	PWR	440	AEE	1/81	79	0	0	0	395	790
Kozloduy 4	PWR	440	AEE	1/81	79	0	0	0	316	711
					Total	0	0	711	2,212	3,792

WORLD POWER REACTORS

	Type	MW		Date	Total					
CANADA										
NDP	PHWR	25	AECL	6/62	6	45	75	105	135	165
Douglas Point	PHWR	206	"	9/68	45	60	285	510	735	960
Pickering 1	PHWR	514	"	8/71	113	0	386	951	1,516	2,081
Pickering 2	PHWR	514	"	1/72	113	0	339	904	1,469	2,034
Pickering 3	PHWR	514	"	7/72	113	0	283	843	1,413	1,978
Pickering 4	PHWR	514	"	7/73	113	0	170	735	1,300	1,865
Gentilly 1	BLWHWR	250	"	6/72	55	0	142	417	692	967
Bruce 1	PHWR	746	"	8/77	164	0	0	389	1,194	1,999
Bruce 2	PHWR	746	"	4/77	164	0	0	451	1,271	2,091
Bruce 3	PHWR	746	"	6/78	164	0	0	0	1,052	1,872
Bruce 4	PHWR	746	"	6/79	164	0	0	0	888	1,708
Bruce 5	PHWR	769	"	10/83	169	0	0	0	211	1,056
Bruce 6	PHWR	769	"	7/84	169	0	0	0	85	930
Bruce 7	PHWR	769	"	4/84	169	0	0	0	127	972
Bruce 8	PHWR	769	"	1/86	169	0	0	0	0	845
Gentilly 2	PHWR	600	"	1/81	132	0	0	0	528	1,188
New Brunswick	PHWR	600	"	10/80	132	0	0	0	561	1,221
Pickering 5	PHWR	516	"	4/81	114	0	0	0	428	998
Pickering 6	PHWR	516	"	1/82	114	0	0	0	342	912
Pickering 7	PHWR	516	"	1/82	114	0	0	0	257	827
Pickering 8	PHWR	516	"	7/83	114	0	0	0	171	741
					Total	105	1,680	5,310	14,375	27,410
CZECHOSLOVAKIA										
Bohunice 1A	GCHWR	112	AEE	1/72	25	0	75	200	325	450
Bohunice 2A	PWR	440	AEE	7/78	79	0	0	119	514	909
Bohunice 2B	PWR	440	"	7/79	79	0	0	0	435	830
Czechoslovakian 3	PWR	440	"	1/81	79	0	0	0	316	711
Czechoslovakian 4	PWR	440	"	1/82	79	0	0	0	237	632
					Total	0	75	319	1,827	3,532
DENMARK										
Unnamed	LWR	900	-	1/83	162	0	0	0	324	1,134
EGYPT										
First	PWR(2)	600	-	1/83	108	0	0	0	216	756

Table B-1 (Cont'd.)

WORLD POWER REACTORS

FINLAND											
Louisa 1 (C)	PWR	420	AEE	1/77	76	0	0	228	608	985
Louisa 2 (C)	PWR	420	AEE	2/79	76	0	0	70	450	830
TVO 1 (C)	BWR	660	AA	8/79	112	0	0	47	607	1,166
TVO 2 (C)	BWR	660	AA	9/81	112	0	0	0	383	943
				Total	0	0	345	2,048	3,927	
GERMANY (Democratic Republic)											
Rheinsberg 1	PWR	80	AEE	5/66	14	51	91	161	231	301
Nord 1-1	PWR	440	"	1/74	79	0	79	476	869	1,264
Nord 1-2 (C)	PWR	440	"	1/76	79	0	0	316	711	1,106
Nord 2-1 (O)	PWR	440	"	7/78	79	0	0	119	514	909
Nord 2-2 (O)	PWR	440	"	7/79	79	0	0	40	435	830
Magdeburg 1	PWR	440	"	7/81	79	0	0	0	237	632
Magdeburg 2	PWR	440	"	7/81	79	0	0	0	237	632
				Total	1	170	1,112	3,234	5,674	
GERMANY (Federal Republic)											
Gundremmingen-Block A	BWR	237	GE	4/67	40	110	310	510	710	910
Lingen	BWR	256	AEG	6/68	44	70	290	510	730	950
Karlsruhe Mzfr	PHWR	52	SIEMENS	9/65	11	48	103	158	213	268
Obrigheim	PWR	328	SIEMENS	3/69	59	49	344	639	934	1,229
Stade	PWR	630	SIEMENS	5/72	113	0	301	866	1,431	1,996
Wurgassen	BWR	645	AEG	3/72	109	0	309	854	1,400	1,944
Biblis a	PWR	1150	SIEMENS	6/74	207	0	121	1,156	2,191	3,226
Biblis 2	PWR	1240	"	8/77	223	0	0	539	1,654	2,769
Brunsbuttel	BWR	771	AEG	8/76	131	0	0	448	1,103	1,758
Neckar GKN	PWR	805	KWU	4/77	145	0	0	495	1,220	1,945
Isar KKI	BWR	870	AEG/KWU	5/77	148	0	0	395	1,235	1,875
Phillipsburg 1	BWR	864	KWU	1/78	147	0	0	294	1,029	1,764
Phillippsburg 2	BWR	864	KWU	1/93	147	0	0	0	294	1,029
Thtr-Uentrop	THTR	300	HRB	7/78	66	0	0	99	429	759
Kalkar SNR-300	LMFBR	282	INTERATOM-NERA	3/82	40	0	0	0	113	313
KKN-Niederaichback	CCHWR	100	SIEMENS	1/76	22	0	0	88	198	308
Krumel	BWR	1260	AEG	3/79	214	0	0	178	1,248	2,318
Unterweser	PWR	1230	SIEMENS	1/77	221	0	0	663	1,768	2,373

Table B-1 (Cont'd.)

WORLD POWER REACTORS

Gundremmingen-Block B	BWR	1249	KWU	7/83	212	0	0	0	318	1,373
Gundremmingen-Block C	BWR	1249	"	7/82	212	0	0	0	530	1,590
Kernkraftuerk Sud.	PWR	1300	KWU	7/80	234	0	0	0	1,053	2,223
Kaerlich	PWR	1228	BBR	5/79	221	0	0	147	1,252	2,357
Grohnde	PWR	1294	KWU	2/32	233	0	0	0	214	1,378
KKG (Grafenrheinfeld)	PWR	1225	KWU	9/30	221	0	0	0	956	2,063
Neckar (GKN 2)	PWR	805	KWU	1/83	145	0	0	0	290	1,015
Brokdork	PWR	1300	KWU	1/85	239	0	0	0	0	1,195
Hamm Ventrop (KKH)	PWR	1300	KWU	1/83	239	0	0	0	478	1,673
Biblis 3	PWR	1228	KWU	1/83	239	0	0	0	478	1,673
					Total	277	1,778	8,039	23,369	44,779
HUNGARY										
Paks 1	PWR	440	AEE	1/81	79	0	0	0	316	711
Paks 2	PWR	440	AEE	1/82	79	0	0	0	237	632
Paks 3 (PAKS)	PWR	440	AEE	1/85	79	0	0	0	0	395
Paks 4 (PAKS)	PWR	440	AEE	1/85	79	0	0	0	0	395
					Total	0	0	0	553	2,133
INDIA										
Tarapur 1	BWR	220	GE	10/69	34	9	179	349	519	689
Tarapur 2	BWR	200	GE	10/69	34	9	179	349	519	689
Rajasthan 1	PHWR	200	AECL	1/74	40	0	44	264	484	704
Rajasthan 2	PHWR	200	AECL	3/77	44	0	0	125	345	565
Madras 1	PHWR	220	DAE	1/79	52	0	0	52	312	592
Madras 2	PHWR	220	DAE	1/81	52	0	0	0	208	468
Narora 1	PHWR	220	DAE	1/83	48	0	0	0	96	336
Narora 2	PHWR	220	DAE	1/84	48	0	0	0	48	283
					Total	18	402	1,139	2,531	4,326
IRAN										
Iran-1	PWR	1200	KWU	1/82	216	0	0	0	648	1,728
Iran-2	PWR	1200	KWU	1/33	215	0	0	0	432	1,512
Iran-3	PWR	900	FRAMATOME	1/84	162	0	0	0	162	972
Iran-4	PWR	900	"	1/85	162	0	0	0	0	810
					Total	0	0	0	1,242	5,022
ISRAEL										
Unnamed	LWR	600	-	1/84	108	0	0	0	108	648

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Table B-1 (Cont'd.)

WORLD POWER REACTORS

ITALY										
Latina	GCR	150	TNPG	1/64	20	120	220	320	420	520
Garigliano	BWR	150	GE	6/64	26	145	275	405	535	665
Trino Vercelese	PWR	247	W	1/65	44	220	440	660	880	1,100
Cirene	LWCHR	40	AMN	1/80	10	0	0	0	50	100
Enel 4	BWR	840	AMN-GETSCO	1/76	142	0	0	572	1,287	2,002
Enel 5	PWR	952	ELECTRONUC-ITAL	1/83	171	0	0	0	342	1,197
Enel 6	BWR	982	AMN	1/83	167	0	0	0	334	1,169
Enel 7	PWR	952	ELECTRONUC-ITAL	1/84	171	0	0	0	171	1,026
Enel 8	BWR	982	AMN	1/84	157	0	0	0	167	1,002
				Total	Total	485	935	1,957	4,166	8,781

JAPAN										
Fukushima 1	BWR	460	GE/TOSHIBA	3/71	78	0	299	689	1,079	1,469
Fukushima 2	BWR	784	GE	7/74	133	0	67	732	1,397	2,062
Shimani	BWR	439	HITACHI	3/74	75	0	63	438	813	1,188
Mihama 1	PWR	320	W	11/70	58	0	242	532	822	1,112
Mihama 2	PWR	470	MHI	7/72	85	0	213	638	1,063	1,488
Tsuruga	BWR	340	GE	3/70	58	0	280	570	860	1,130
Tokai 1	GCR	159	GEC	7/66	21	74	179	284	389	494
Tokai 2	BWR	1067	GE	1/78	181	0	0	362	1,267	2,172
Takahama 1	PWR	781	W	11/74	141	0	242	729	1,434	2,139
Takahama 2	PWR	781	MHI	11/75	141	0	101	588	1,293	1,998
Genkai 1	PWR	559	MHI	3/82	101	0	0	430	934	1,439
Ohi 1	PWR	1122	W	6/78	202	0	0	320	1,330	2,340
Ohi 2	PWR	1122	W	7/79	202	0	0	101	1,111	2,121
Mihama 3	PWR	781	MHI	1/77	141	0	0	423	1,433	2,443
Fugen	LWCHR	200	HITACHI	7/77	48	0	0	120	360	600
Ikata	PWR	538	MHI	4/77	97	0	0	267	752	1,237
Hamaoka 1	BWR	516	TOSHIBA	3/76	38	0	0	337	777	1,129
Hamaoka 2	BWR	840	TOSHIBA	9/78	138	0	0	184	874	1,564
Fukushima 3	BWR	784	TOSHIBA	3/76	133	0	0	510	1,174	1,840
Fukushima 4	BWR	784	HITACHI	10/78	133	0	0	166	831	1,496
Fukushima 5	BWR	784	TOSHIBA	4/78	133	0	0	234	898	1,563
Fukushima 6	BWR	1100	GE	10/79	187	0	0	31	966	1,901
Monju	LMFBR	300	-	1/84	42	0	0	0	42	252
Onagawa	BWR	500	TOSHIBA	1/82	85	0	0	0	255	600
Genkai 2	PWR	1300	BBR	1/85	239	0	0	0	286	791
				Total	Total	74	1,686	8,685	22,440	36,648

Table B-1 (Cont'd.)

WORLD POWER REACTORS

	Type	Capacity	Mfr	Date						
KOREA (South)										
Ko-Ri 1	PWR	564	W	11/77	102	0	0	221	731	1,241
Ko-Ri 2	PWR	600	W	1/85	108	0	0	0	0	540
Nae-Po 1	PHWR	600	AECL	1/84	132	0	0	0	132	792
Nae-Po 2	PHWR	600	AECL	1/85	132	0	0	0	0	660
Chang-Yung 1	-	800	-	1/86	176	0	0	0	-	-
Chang-Yung 2	-	800	-	1/87	176	0	0	0	0	-
					Total	0	0	221	863	3,233
LUXEMBOURG										
Remerschen	PWR	1300	BBR	1/85	239					
MEXICO										
Laguna Verde 1	BWR	654	GE	3/81	111	0	0	0	426	981
Laguna Verde 2	BWR	654	GE	3/82	111	0	0	0	315	870
					Total	0	0	0	741	1,351
NETHERLANDS										
Dodewaard	BWR	55	GE/GKN	3/69	9	8	53	98	143	188
Borssele	PWR	477	KWU	10/73	86	0	100	530	960	1,390
					Total	8	153	628	1,103	1,578
PAKISTAN										
Kanupp	PHWR	125	CGE	12/72	28	0	61	201	341	431
Chasma-Barrage	-	600	-	1/83	132	0	0	0	264	924
					Total	0	61	201	605	1,405
PHILLIPINES										
Bagac 1	PWR	626	W	1/84	113	0	0	0	113	678
Bagac 2	PWR	626	W	1/84	113	0	0	0	113	678
					Total	0	0	0	226	1,356
PORTUGAL										
Unnamed	LWR	600	-	1/83	103	0	0	0	216	756
Unnamed	LWR	600	-	1/85	108	0	0	0	0	540
					Total	0	0	0	216	1,296
POLAND										
Zarnowieckie	PWR	440	AEE	1/85	79	0	0	0	0	396

Cont'd.)

WORLD POWER REACTORS

Plant	Type	MW	Mfr	Date							
ROMANIA											
Olt	PWR	440	AEE	1/82	79	0	0	0	237	632	
SOUTH AFRICA											
Koeberg 1	LWR	922	FRAMATOM	1/83	166	0	0	0	332	1,162	
Koeberg 2	LWR	922	FRAMATOM	1/84	166	0	0	0	166	996	
				Total		0	0	0	493	2,158	
SPAIN											
Santa Maria de Garona	BWR	440	GE	3/71	75	0	281	656	1,031	1,406	
Vandellos	GCR	430	GC	7/72	62	0	155	465	775	1,085	
Jose Cabrera	PWR	153	W	8/69	28	12	152	292	432	572	
Asco 1	PWR	900	W	7/79	162	0	0	81	891	1,701	
Asco 2	PWR	900	W	1/80	162	0	0	0	810	1,620	
Almaraz 1	PWR	900	W	7/77	162	0	0	405	1,215	2,025	
Almaraz 2	PWR	900	W	7/78	162	0	0	243	1,053	1,863	
Confrentes	BWR	930	GE	1/81	158	0	0	0	632	1,422	
Lemoniz 1	PWR	901	W	3/79	162	0	0	135	945	1,755	
Lemoniz 2	PWR	901	W	4/80	162	0	0	0	770	1,550	
Cabo Cope	BWR	930	GE	1/83	158	0	0	0	516	1,106	
Valde Caballeros 1	BWR	937	GE	1/83	159	0	0	0	337	1,183	
Valde Caballeros 2	BWR	937	GE	1/83	159	0	0	0	169	1,014	
TRILLO 1	PWR	997	KWU	7/82	180	0	0	0	270	1,170	
				Total		12	568	2,277	9,646	19,502	
YUGOSLAVIA											
Krsko	PWR	615	W	7/79	111	0	0	55	610	1,165	
SWEDEN											
Oskarshamn 1	BWR	440	ASEA-ATOM	2/72	75	0	212	587	962	1,337	
Oskarshamn 2	BWR	580	"	1/75	99	0	0	495	930	1,485	
Ringhals 1	BWR	760	"	2/75	129	0	0	634	1,273	1,924	
Ringhals 2	PWR	809	W	2/75	146	0	0	681	1,411	2,141	
Ringhals 3	PWR	900	W	1/78	162	0	0	324	1,134	1,944	
Ringhals 4	PWR	900	W	1/80	162	0	0	810	1,620	2,430	
Barsebeck 1	BWR	580	ASEA-ATOM	7/75	99	0	0	446	941	1,436	
Barsebeck 2	BWR	580	"	7/77	99	0	0	248	743	1,238	
Oskarshamn 3	BWR	1060	"	1/85	180	0	0	0	0	901	

Table B-1 (Cont'd.)

WORLD POWER REACTORS

	Type	MW	Mfr.	Date						
Forsmark 1	BWR	900	ASEA-ATOM	7/78	153	0	0	230	995	1,760
Forsmark 2	BWR	900	"	1/81	153	0	0	0	612	1,377
Forsmark 3	BWR	1000	"	1/84	180	0	0	0	170	1,020
					Total	0	212	4,455	10,791	18,993
SWITZERLAND										
Beznau 1	PWR	350	W	12/69	63	5	320	635	950	1,265
Beznau 2	PWR	350	W	3/72	63	0	178	493	808	1,123
Muhleberg	BWR	306	GETSCO	10/72	55	0	124	349	574	799
Geosgen	PWR	920	KWU	4/78	166	0	0	291	1,121	1,951
Graben	BWR	1140	GETSCO	1/81	194	0	0	0	970	1,940
Leibstadt	BWR	955	"	1/80	160	0	0	0	800	1,600
Kaiseraugst	BWR	932	"	1/81	157	0	0	0	471	1,256
Ruthi	-	900	"	-	153	0	0	0	-	-
					Total	5	622	1,768	5,694	9,934
TAIWAN										
Chin-Shan 1	BWR	604	GE	1/78	103	0	0	206	721	1,236
Chin-Shan 2	BWR	604	"	1/80	103	0	0	0	515	1,030
Kuosheng 1	BWR	951	"	4/81	162	0	0	0	540	1,350
Kuosheng 2	BWR	951		4/82	162	0	0	0	446	1,256
Nr. 5	PWR	907		4/84	163	0	0	0	122	937
Nr. 6	PWR	907		4/85	163	0	0	0	0	774
					Total	0	0	206	2,344	6,583
					TOTAL	1,035	8,397	39,580	120,444	237,526

Table B-2

RESEARCH REACTORS ≥ 1 MW(t)

Location	Type	MW_{th}	Operating Date	^{239}Pu Prod. kg/yr	^{239}Pu Accumulated (kg) Through				
					69	74	79	84	89
Argentina RA-3	Tank	8	5/67	Enr. U					
Australia HIFAR	Tank, D_2O	10	1/58	93% Enr U	.0009 (3)	.011	.016	.020	.025	.029
Austria ASTRA	Pool	12	9/60	90% Enr. U	.001 (8)	.01	.015	.02	.025	.03
Belgium BR-1	Graphite	4	5/50	Nat. U	.94	12.9	17.6	22.3	27.0	31.7
BR-2	Tank	100	7/61	90% Enr	.01 (8)	.085	.14	.19	.24	.29
Brazil IEAR-1	Pool	5	9/57	20% Enr. U					
Bulgaria IRT-SOFIA	Pool	1	8/61	10% Enr. U					
Canada NRX	Tank, D_2O	33	7/47	Nat. U	2.18 (7)	49.1	50.0	70.9	81.8	92.7
NRU	Tank, D_2O	110	11/57	Nat. U	7.26 (7)	88.3	124.6	160.9	197.2	233.5
MNR	Pool	2	2/50	90% Enr. U	.0002 (8)	.002	.003	.004	.005	.006
NR-1	Tank, Organic, D_2O	60	11/65	2.4% Enr. U					
Chile CHEN	Pool	5	/73	Enr. U					
Czechoslovakia WWR-S	Tank	4	9/57	10% Enr. U	.077 (4)	.95	1.34	.72	2.11	2.49
Denmark DR-2	Pool	5	12/53	90% Enr. U	.0005 (8)	.006	.008	.011	.013	.016
DR-3	Tank, D_2O	10	1/60	93% Enr. U	.0009 (3)	.009	.014	.018	.023	.027
Italy ISPRA-1	Tank, D_2O	5	3/59	20% Enr. U	.0007 (8)	.007	.010	.014	.017	.021
AVOGADRO RS-1	Pool	7	9/59	90% Enr. U	.026 (5)	.25	.38	.51	.64	.77
RC-1 (Triga)	Pool, U-ZrH	1	6/60	20% Enr. U					
GALILEO RTS-1	Pool	5	4/63	90% Enr. U	.0005 (8)	.0033	.0058	.0083	.011	.013
ESSOR	Tank, Organic D_2O	40	3/67	Nat. & 90% Enr. U					
PEC (C)	Fast	130							

Table B-2 (Cont'd.)

RESEARCH REACTORS ≥ 1 MW(t)

Location	Type	MW$_{th}$	Operating Date	Fuel	^{239}Pu Prod. kg/yr (2)	^{239}Pu Accumulated (kg) Through				
						69	74	79	84	89
Japan										
JRR-2	Tank, D$_2$O	10	10/60	90% Enr. U	.001 (8)	.01	.015	.02	.025	.03
JRR-3	Tank, D$_2$O	10	9/62	Nat U	2.04	15.0	25.2	35.4	45.6	55.8
KUR	Tank	5	6/64	90% Enr. U	.0005 (8)	.003	.005	.008	.01	.013
JRR-4	Pool	25	1/65	90% Enr. U	.0025 (8)	.013	.025	.038	.05	.06
JMTR	Tank	50	3/68	90% Enr. U	.005 (8)	.008	.033	.058	.083	.11
JOYO (C)	Fast	50	75							
Mexico										
RCN (Triga)	Pool, U-ZrH	1	11/68	20% Enr. U	.026 (5)	.004	.13	.26	.39	.52
Netherlands										
HFR	Tank	45	11/61	90% Enr. U	.0045 (8)	.036	.059	.081	.10	.13
HOR	Pool	2	4/63	90% Enr. U	.0002 (8)	.0013	.0023	.0033	.0043	.0053
KSTR	Aqu. Homog.	1	4/74							
Norway										
Halden	Tank, D$_2$O	25	6/59	1.5% Enr. U	4.46	40.1	62.4	84.7	107.0	129.3
JEEP-2	Tank, D$_2$O	2	12/66	3.5% Enr. U	.13	.40	1.05	1.70	2.35	3.00
Pakistan										
PARR	Pool	5	12/65	90% Enr. U	.005 (8)	.002	.0045	.007	.0095	.012
Phillippines										
PRR-1	Pool	1	8/63	20% Enr. U						
Egypt										
WWR-C	Tank	2	7/61	10% Enr. U	.038 (4)	.32	.51	.70	.89	1.08
Germany, Fed. Republic										
FRM	Pool	4	5/68	20% Enr. U						
FRG-1	Pool	5	10/58	20% Enr. U						
FR-2	Tank, D$_2$O	44	7/66	Nat. U	10.3 (3)	36.1	87.6	139.1	190.6	242.1
FRJ-1	Pool	10	11/71	80% Enr. U	.002 (3)	.004	.014	.024	.034	.044
FRJ-2	Tank, D$_2$O	23	12/67	93% Enr. U	.0015 (8)	.003	.010	.017	.025	.033
FRG-2	Pool	15	2/68	90% Enr. U	.0001 (8)	.0002	.0007	.0012	.0017	.0022
FMRB	Pool	1	10/67	90% Enr. U						
FRN (Triga)	Pool, U-ZrH	1	8/72	20% Enr. U	.026 (5)	-	.063	.19	.32	.45
BER-2	Pool	5	12/73	Enr. U		-				
Greece										
Democritus (GRR)	Pool	5	7/61	20% Enr. U	.038 (4)					
Hungary										
WWR-C	Tank	2	3/59	10% Enr. U	.038 (4)	.41	.60	.79	.98	1.17

Table B-2 (Cont'd.)

RESEARCH REACTORS >1 MW(t)

Location	Type	MWth	Operating Date	Fuel	239 Pu Prod kg/yr (2)	239Pu Accumulated (kg) Through				
						69	74	79	84	89
India APSARA	Pool	1	8/56	46% Enr. U						
CIRUS	Tank, D₂O	40	7/60	Nat. U	9.4 (6)	89.3	136.3	183.	230.3	277.3
FBTR (C)	FBR	50	/75	Thorium		-				
Iran UTRR	Pool	5	11/67	20% Enr. U						
Iraq NWR-C	Tank	2	1/63	10% Enr. U	.038 (4)	.27	.46	.55	.84	1.03
Israel IRR-1	Pool	5	6/60	90% Enr. U	.0005 (6)	.005	.007	.010	.012	.015
IRR-2	Tank, D₂O	26	12/63	Nat U	6.11 (6)	37.2	67.8	93.3	128.9	159.4
Poland EWA (NWR-C-WARSAW)	Tank	10	1/53	10% Enr. U	.19 (4)	2.28	3.23	4.18	5.13	6.08
MARIA (C)	Pool	30	Unknown	Enr U						
Portugal RPI	Pool	1	4/61	Enr U						
Romania NWR-C (Super Triga)	Tank	3	7/57	10% Enr. U	.06 (4)	.75	1.05	1.35	1.65	1.95
	Tank, U-Zrh	14	77	Enr. U						
South Africa SAFARI-1	Tank	20	3/65	90% Enr. U	.002	.01	.02	.03	.04	.05
Spain JEN-4	Pool	3	10/58	20% Enr. U	.04	.45	.65	.85	1.05	1.25
Sweden R-2	Tank	50	5/60	90% Enr. U	.005 (8)	.048	.073	.098	.12	.15
R-2-0	Pool	1	6/61	90% Enr. U	.0001 (8)	.0008	.0013	.0018	.0023	.0028
Switzerland SAPHIR	Pool	5	4/57	20% Enr. U	4.47	42.1	64.4	86.3	109.1	131.5
DIORIT	Tank, D₂O	20	8/60	Nat U						
Taiwan TRR	LWCHWR	40	1/73	Nat U	2.64 (7)	-	5.3	18.5	31.7	44.9
Thailand TRR-1	Pool	1	10/62	90% Enr. U	.0001 (8)	.0007	.0012	.0017	.0022	.0027

Table B-2 (Cont'd.)

RESEARCH REACTORS $\geq$1 MW(t)

Location	Type	MW_{th}	Operating Date	Fuel	^{239}Pu Prod. kg/yr (2)	^{239}Pu Accumulated (kg) Through				
						69	74	79	84	89
Turkey TR-1	Pool	1	1/62	90% Enr. U	.0001 (8)	.0008	.0013	.0018	.0023	.0028
Uruguay RUDI	Pool	1	73	Enr. U						
Venezuela RV-1	Pool	3	10/65	20% Enr. U						
Yugoslavia R-A	Tank, D_2O	10	12/59	1.5% Enr. U	82	8.27	12.37	16.47	20.57	24.67
Zaire TRICO (Triga)	Pool, U-ZrH	1	72	20% Enr. U	.026 (5)	-	.078	.14	.27	.40

March 2, 1977

Ms. Audrey Buyrn
Office of Technology Assessment
Congress of the U.S.
Washington, D.C. 20510

Dear Ms. Buyrn:

As requested in your telephone conversation of February 17, 1977,
I am enclosing two schedules reflecting total United States exports,
by country, of enriched uranium of 20 percent or greater U-235 since
1968 and of all plutonium since the beginning.

Data on uranium enriched to 20 percent and above prior to January 1,
1968, has not yet been computerized, and accordingly, we are unable
to readily furnish you exports prior to 1968. This data is now being
recovered from historical files and will be automated for recovery
by October of this year.

If I can be of further assistance, please call.

Sincerely,

Thomas J. Haycock, Jr.
Assistant Director for Information
   Support
Division of Safeguards and Security

Enclosures:
As stated

Schedule **A**

Plutonium
Exported by the United States
From Beginning Through December 31, 1976

Country	Grams
Argentina	5
Australia	6,577
Austria	296
Belgium	16,349
Brazil	80
Canada	4,928
China (Taiwan)	683
Columbia	80
Denmark	81
Eurochemic	14
Finland	2
France	41,442
Germany	767,126
Greece	192
IAEA	44
India	82
Iran	112
Iraq	16
Ireland	16
Israel	605
Italy	129,097
Japan	111,227
Korea	8
Mexico	164
Netherlands	790
New Zealand	80
Norway	1,083
Pakistan	117
Philippines	32
South Africa	159
Spain	6
Sweden	9,143
Switzerland	1,502
Thailand	80
Turkey	368
United Kingdom	22,417
Uruguay	80
Venezuela	**10**
Total Exported	**1,115,093**

SCHEDULE B
ENRICHED URANIUM
EXPORTED BY THE UNITED STATES
FROM JANUARY 1, 1968 THROUGH DECEMBER 31, 1976
(Transfer of 500 grams or more - enriched to 20% or more)

COUNTRY	KILOGRAMS	
	ELEMENT	ISOTOPE
Argentina	31	28
Austria	2	1
Belgium	23	13
Brazil	6	6
Canada	619	575
China (Taiwan)	5	5
Colombia	1	1
Finland	11	2
France	2,371	2,087
Germany	3,543	2,775
Greece	6	6
Israel	9	8
Italy	164	138
Japan	1,707	
Korea	7	3
Netherlands	2	2
Portugal	8	7
South Africa	25	23
Switzerland	2	2
United Kingdom	1,119	1,040
Yugoslavia	5	3
TOTAL EXPORTED	9,666	7,405

SCHEDULE B
ENRICHED URANIUM
EXPORTED BY THE UNITED STATES
FROM JANUARY 1, 1968 THROUGH DECEMBER,R 31, 1976
(Transfer of 500 grams or more - enriched to 20% or more)

COUNTRY	KILOGRAMS	
	ELEMENT	ISOTOPE
Argentina	31	28
Austria	2	1
Belgium	23	13
Brazil	6	6
Canada	619	575
China (Taiwan)	5	5
Colombia	1	1
Finland	11	2
France	2,371	2,087
Germany	3,543	2,775
Greece	6	6
Israel	9	8
Italy	164	138
Japan	1,707	
Korea	7	3
Netherlands	2	2
Portugal	8	7
South Africa	25	23
Switzerland	2	2
United Kingdom	1,119	1,040
Yugoslavia	5	3
TOTAL EXPORTED	9,666	7,405