

XA0007200

INFCE

International
Nuclear
Fuel
Cycle
Evaluation

INFCE/DEP/WG.5/35

FAST BREEDER FUEL CYCLE

INFCE WG. 5/B

FAST BREEDER FUEL CYCLE

July 6, 1978

Contribution from Japan

- B. 1. Short review of the fast breeder fuel cycle
 B. 1. 1 Reference design

Item		Unit	FBR-A					
Electrical Power Out put		MW	1.000					
Thermal Power Out put		MW	2.500					
Thermal Efficiency		%	40					
Average Linear Heat Rating		W/cm	213					
Average Specific Power		Kw/Kg	84					
Average Power Density		Kw/l	272					
Average Fuel Burnup		MWD/T	About 75.000					
Breeding Ratio			1.2					
Refuelling Interval		year	1					
Refuelling Batch Number		Core/Blanket	3/4					
Plant Life Time		year	25					
			Loaded Fuel			Discharge Fuel		
			Core	Blanket	Total	Core	Blanket	Total
Initial load	HM	t.t/y	23.16	32.18	55.34	10.64	13.14	23.78
	U	t.t/y	19.24	32.18	51.42	8.73	12.91	21.64
	Uf	t.t/y	0.06	0.10	0.16	0.01	0.01	0.02
	Pu	t.t/y	3.92	-	3.92	1.91	0.23	2.14
	Puf	t.t/y	3.06	-	3.06	1.44	0.21	1.65
Refuelling load	HM	t/y	11.03	13.17	24.20	10.13	13.11	23.24
	U	t/y	9.10	13.17	22.27	8.19	12.69	20.88
	Uf	t/y	0.03	0.04	0.07	0.01	0.01	0.02
	Pu	t/y	1.93	-	1.93	1.94	0.43	2.37
	Puf	t/y	1.50	-	1.50	1.39	0.41	1.80
End of Reactor Life	HM	t				21.90	32.09	53.99
	U	t				17.83	31.49	49.32
	Uf	t				0.04	0.07	0.11
	Pu	t				4.07	0.60	4.67
	Puf	t				2.99	0.57	3.56

* FBR-A will be used till 2.000 and FBR-B will be used after 2.000.

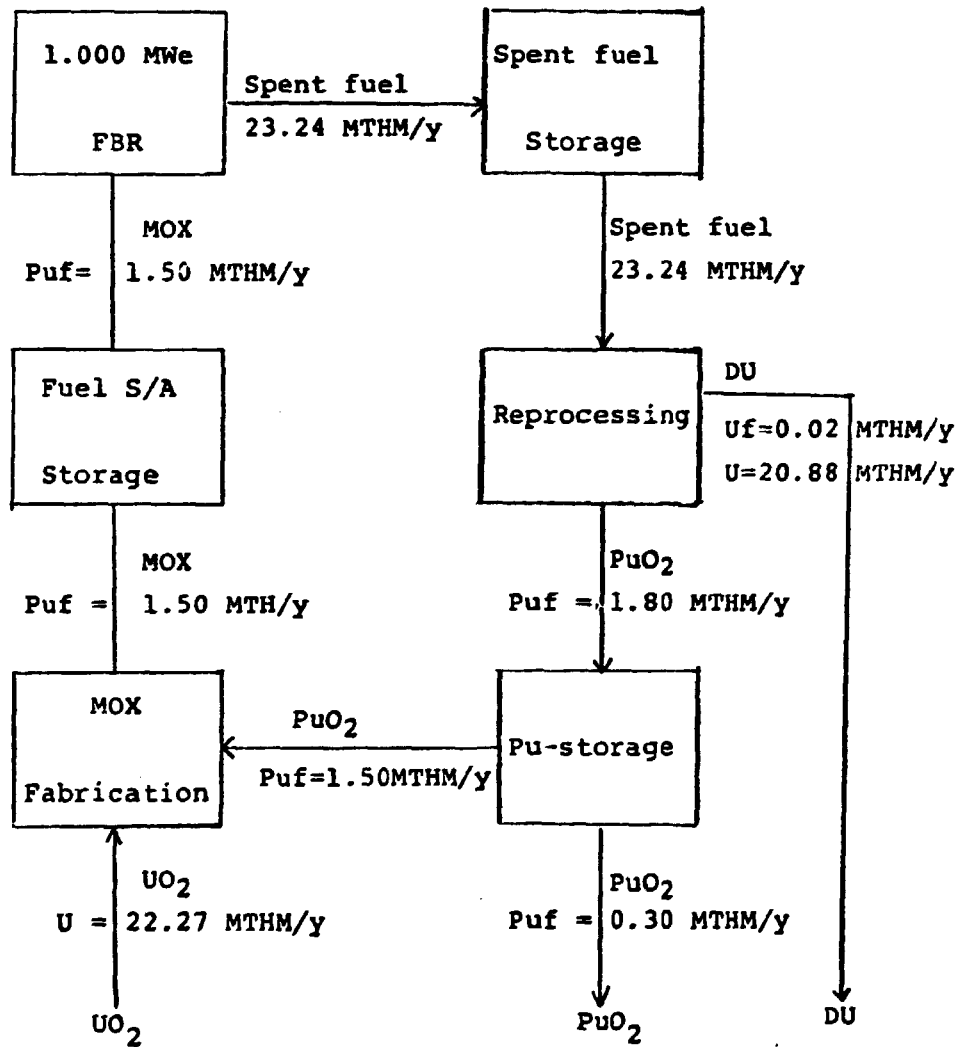
SECTION 1

Feeder fuel cycle

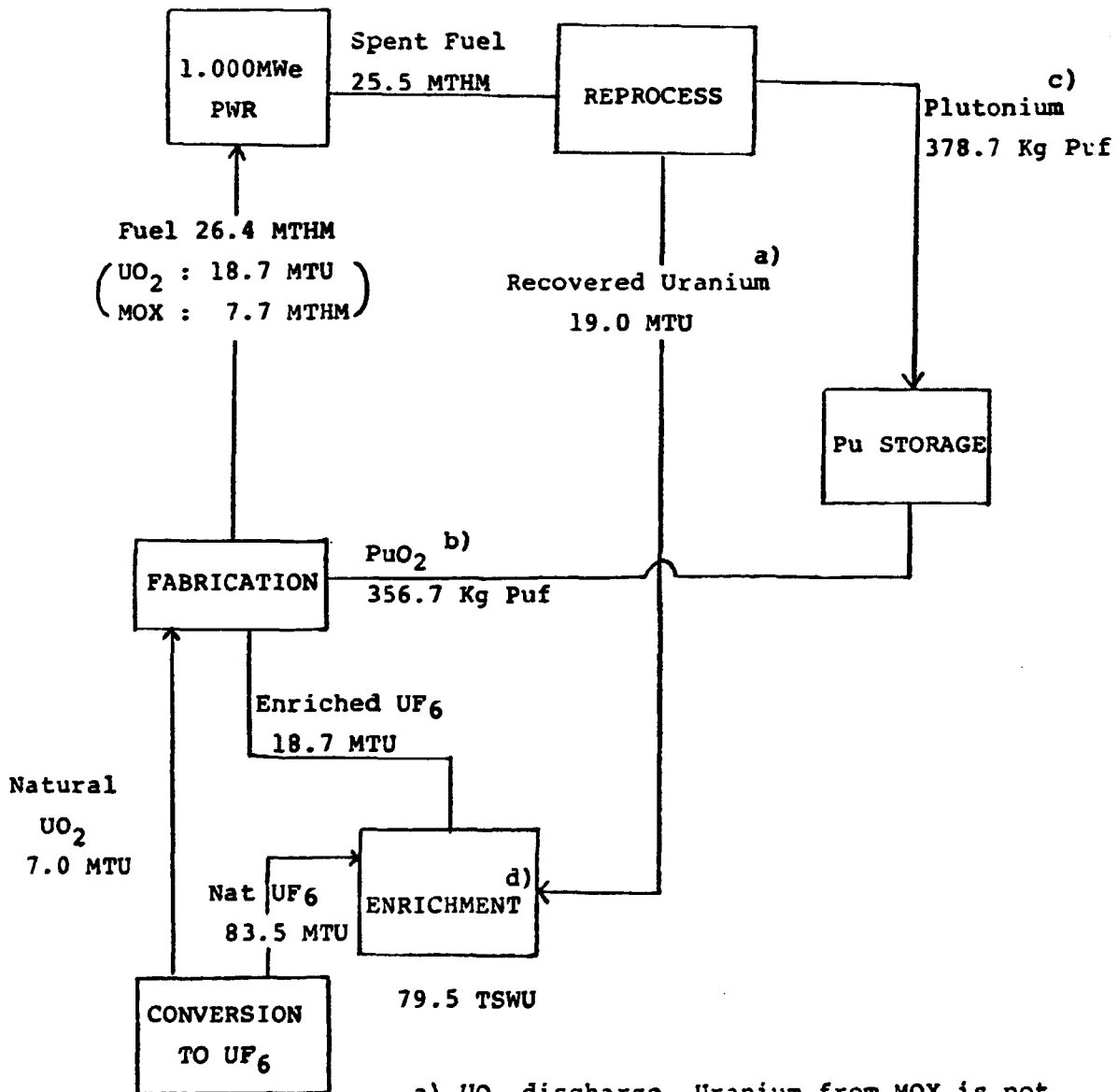
FBR-A						FBR-B					
1.000						1.000					
2.500						2.500					
40						40					
213						250					
84						105					
272						319					
About 75.000						About 90.000					
1.2						1.3					
1						1					
3/4						3/4					
25						25					
Loaded Fuel			Discharge Fuel			Loaded Fuel			Discharge Fuel		
Core	Blanket	Total	Core	Blanket	Total	Core	Blanket	Total	Core	Blanket	Total
23.16	32.18	55.34	10.64	13.14	23.78	20.93	32.52	53.45	9.53	13.56	23.09
19.24	32.18	51.42	8.73	12.91	21.64	17.49	32.52	50.01	7.91	13.33	21.24
0.06	0.10	0.16	0.01	0.01	0.02	0.05	0.10	0.15	0.03	0.01	0.04
3.92	-	3.92	1.91	0.23	2.14	3.44	-	3.44	1.64	0.24	1.88
3.06	-	3.06	1.44	0.21	1.65	2.68	-	2.68	1.26	0.23	1.49
11.03	13.17	24.20	10.13	13.11	23.24	9.97	13.59	23.56	9.20	13.50	22.70
9.10	13.17	22.27	8.19	12.69	20.88	8.34	13.59	21.93	7.56	13.06	20.62
0.03	0.04	0.07	0.01	0.01	0.02	0.03	0.04	0.07	0.03	0.01	0.04
1.93	-	1.93	1.94	0.43	2.37	1.63	-	1.63	1.49	0.51	2.00
1.50	-	1.50	1.39	0.41	1.80	1.27	-	1.27	1.16	0.50	1.66
			21.90	32.09	53.99				20.51	32.34	52.85
			17.83	31.49	49.32				16.91	31.73	48.64
			0.04	0.07	0.11				0.04	0.07	0.11
			4.07	0.60	4.67				3.60	0.61	4.21
			2.99	0.57	3.56				2.77	0.59	3.36

and FBR-B will be used after 2.000.

B. 1. 2 Identification of critical points (from a proliferation point of view) in the fuel cycle.
 Comparison with LWR cycle.



* Annual Fuel Cycle Requirements for 1.000 MWe FBR-A
 (Equilibrium-Core)



a) UO_2 discharge, Uranium from MOX is not recycled.

b) 4th Recycle discharge.

c) 5th Recycle discharge.

d) Tails Assay 0.2 w/o.

* Annual Fuel Cycle Requirements for 1.000 MWe
PWR for 5th (near-equilibrium) Plutonium Recycle (SGR)

Fig B. 2. 1 Discharge Fuel Radiation Level
(discharge from core)

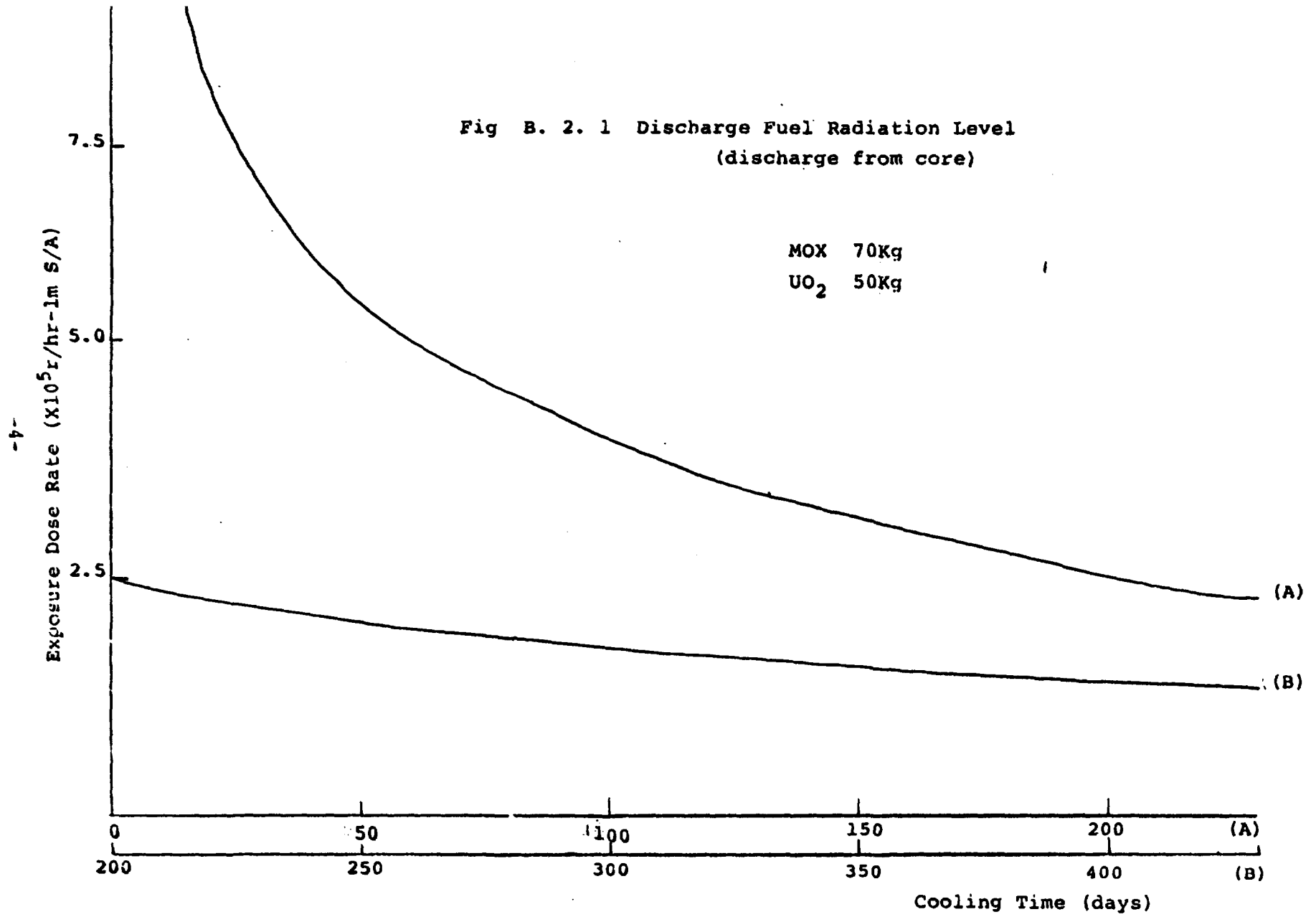
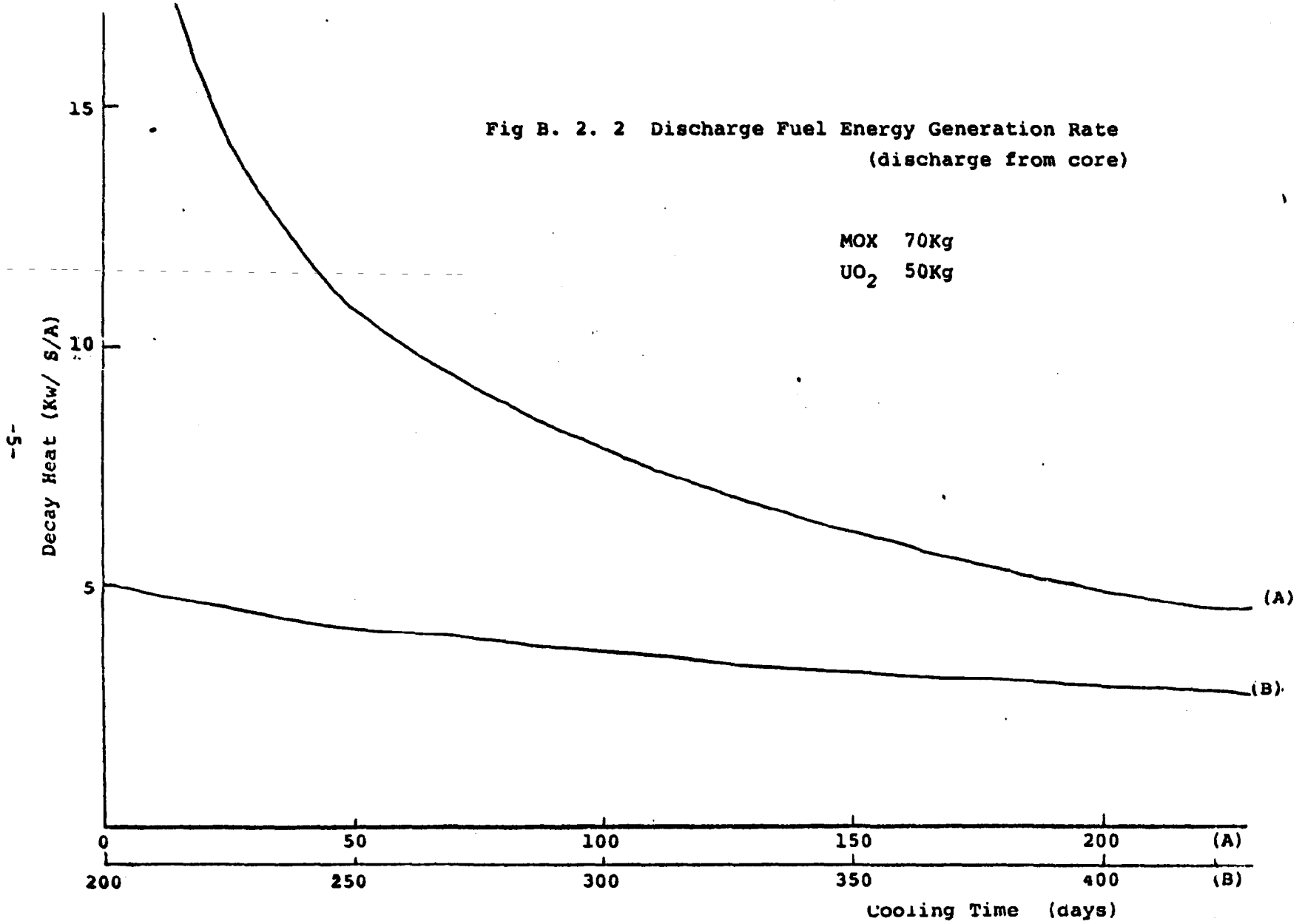


Fig B. 2. 2 Discharge Fuel Energy Generation Rate
(discharge from core)

MOX 70Kg
UO₂ 50Kg



B. 2. Fuel Reprocessing

B. 2. 1 Basic FBR fuel reprocessing scheme

B. 2. 1. 1 Composition of irradiated fuel

(1) Composition of fresh fuel

$^{239}\text{Pu} : ^{240}\text{Pu} : ^{241}\text{Pu} : ^{242}\text{Pu} = 65 : 20 : 13 : 2$

(2) Composition of irradiated fuel in core, oxial blanket, with regard to fission products/actinides determination of actinides/heat generation.

FBR-A

<u>Initial load</u>	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu
Core	67	21	8	4
Blanket	94	6	0	0
<u>Equilibrium core</u>				
Core	63	25	8	4
Blanket	96	4	0	0

FBR-B

<u>Initial load</u>	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu
Core	69	19	8	4
Blanket	96	4	0	0
<u>Equilibrium core</u>				
Core	69	19	9	3
Blanket	98	2	0	0

Activities and heat generation of a core fuel assembly are illustrated in Figs B. 2. 1 and B. 2. 2, respectively.

Those data are calculated from the basis of ;

Weight of core fuel ($\text{PuO}_2 - \text{UO}_2$) 70Kg

Oxial Blanket fuel (UO_2) 50Kg

Burnup of core fuel 75.000 MWD/T

B. 2. 1. 2 Basic FBR fuel reprocessing scheme

(1) Main differences with LWR reprocessing

1) Presence of sodium metal

Sodium used as a coolant must be inactivated so as to avoid its explosive reaction to nitric acid. Inactivation should also be considered for sodium which has penetrated into a fuel pin through a hair crack developed on the cladding.

2) Increased heat dissipation

Because of the high decay heat arising from the higher fuel burn-up and a shorter cooling period of spent fuel as compared with light water reactor fuel, there is a constant need for forced cooling during the spent fuel handling. Designs to remove heat will have to be provided on dissolution, solution adjustment, co-decontamination, treatment of high-level radioactive waste and storage processes.

3) Higher plutonium concentration

While the plutonium content of LWR fuel is 1-2%, FBR fuel has the plutonium content of 10-20%, and even when mixed with blanket fuel, it still has about 10% of plutonium content. Therefore, criticality limitations grow in severity in all stages from the dissolution tank to the denitrating reactor in the conversion process.

4) Presence of transplutonium actinides in larger amounts

Am and Cm in spent fuel may amount to as much as 2Kg/MTM, and the increased rays and neutrons due to such high concentration of transplutonium make that shielding and remote handling become important problems in the waste treatment process.

B. 2. 1. 3 Basic Sequence of Operation

1) Sodium removal

Sodium used as a coolant will be removed and inactivated by soaking in or spraying with N₂, Ar and CO₂ mixed with steam, or by alcohol.

2) Dismantling--shearing

Wrappers and entrance nozzles are cut and dismantled in order to get fuel pins out of fuel assemblies. Fuel pin bundles are sent into a shearing machine to be cut into small pieces.

3) Pulverization

Fuel cut into small pieces is pulverized by supplying with high-temperature, oxygen-containing gas, together with mechanical strength applied.

4) Dissolution

Pulverized fuel can be dissolved by the boiling nitric acid with applying oxygen as an agitator.

5) Removal of insolubles

After dissolution, pulse filter is used to separate insoluble residue, which is moved into a re-dissolution tank where nitric acid and fluoric acid are used to re-dissolve it, with aluminium nitrate as a masking agent, so that it is separated by filter.

6) Extraction of (U-Pu) and removal of fission products

The Purex process is applied to separate fission products from uranyl plutonium-nitrate. This comes with the use of the pulse-column to ensure criticality safety, to avoid moving elements and to prevent solvent deterioration.

7) Partition : U/Pu separation

Plutonium is reduced through the application of plutonium reducing agent or electrolytic reduction method so that it can be counter-extracted into the water phase. Uranium remaining at the organic phase is counter-extracted by dilute nitric acid.

8) Purification of U and Pu

Plutonium nitrate after valence adjustment and acid density adjustment is put into the pulse column extraction tower for extraction and cleaning, to be followed by dodecane cleaning.

Similarly, uranyl nitrate after liquid supply adjustment is put through the process of extraction tower and dodecane cleaning.

9) Conversion of U and Pu into oxide

Plutonium nitrate and uranyl nitrate are concentrated by evaporation before they are put into the fluidized bed denitrating tower for denitration, and PuO_2 and UO_2 are stored.

B. 2. 1. 4 Special handling measures

(1) Safety measures to avoid criticality condition

Plutonium concentration is so high that all apparatus should be designed to take the critically safe geometry, with a flow pattern

developed to be operated in such a manner as not to let plutonium deposits build up in the apparatus.

Equipment designed to take the critically safe geometry should also be furnished with neutron absorbing materials to make sure of safety.

(2) Dissolution

1) Resistance to dissolution of PuO_2 in HNO_3

The high Pu content and high burn-up fuel is likely to leave 1-10 insoluble particles at the dissolution process. These particles are mainly nuclear fission products (Ru, Mo, Rh, Pd, Tc) containing plutonium, and the amount of insoluble particles increases with the increasing fuel burn-up.

2) Pu recovery from insolubles

Pulse filter and centrifugal separator are used to separate insoluble particles, and HF-added nitric acid is used for re-dissolution before plutonium is recovered.

3) Problems related to presence of noble metals

Noble metals remain insoluble even after re-dissolution and plutonium remaining with them contributes to a loss of nuclear material. These particles building up at the interface of two-phases in the extracting apparatus have an adverse effect on liquid-liquid extraction, form the third phase and aggravate solvent damage, thus causing a lowering of DF.

(3) Liquid extraction

1) Avoiding criticality condition

Geometrical criticality control makes for ease for the use of extracting equipment, in which it should be made easy to put in neutron absorbent. The system should be operated as continuously as possible, with using a limited number of intermediate storage tanks, so that Pu and other sediment will not occur in the system.

2) TBP irradiation damage

Use of a centrifugal extraction machine which requires little contact time, or of a pulse column should be combined with measures for complete solvent regeneration and cleaning, in order to remove the radiation decomposed materials and polymer compounds.

3) Crud formation

Due to the corrosion of equipment materials and radiation decomposition, the crud will be formed in the co-decontamination process. Therefore, apparatus should be selected from among those which allow little crud to build up and which can easily be cleaned.

4) Special problems in extraction/partition cycle due to high Pu concentration. Alternatives reduction methods.

If plutonium is reduced through the use of a chemical reductant, the amount of reductant increases in proportion to the amount of plutonium and the amount of waste increases. The increase in amount to be treated poses a serious problem of criticality safety. An alternative method that can be used to resolve this problem is to reduce plutonium electro-chemically.

(4) Storage of purified Pu

1) Ingrowth of ^{241}Am

Due to ^{241}Am growing from ^{241}Pu in storage, it is necessary to set up gamma shield and to treat remotely.

2) Neutron emission due to presence of ^{238}Pu and ^{240}Pu

Since ^{238}Pu and ^{240}Pu fission spontaneously in addition to their alpha decay, it is necessary to use a shield against neutrons. In case of long-term storage, it is necessary to build a shield against gamma rays emitted from accumulated fission products.

B. 2. 3 Technical modifications to conventional aqueous product

B. 2. 3. 1 Modification of the end product composition: the master blend concept

(1) Composition of an acceptable master blend

Acceptability for a master blend can be determined in consideration of:

- The Pu fissile content in the FBR fuel to be made,
- The impact on manufacturing costs,
- Convenience for fuel fabricating technology -- the percentage of U addition in order to control Pu and U fissile contents,
- A U/Pu ratio appropriate for non-proliferation.

Since the amount of Pu in FBR stands at 10-20% (U/Pu \approx 9-4), the U/Pu ratio must be considerably lower than that.

(2) Methods to obtain a master blend

1) Mechanical blending

As reprocessing products, pure PuO_2 and UO_2 are manufactured and are blended later by ball mill.

2) Co-precipitation

The mixed nitric acid solution of U and Pu after removal of fission products can be co-precipitated into $\text{PuO}_2\text{-UO}_2$ powder by using $\text{NH}_4(\text{OH})$ or oxalic acid.

B. 2. 3. 2 Modification of other aqueous reprocessing steps

(1) Short cooling time

When spent fuel is reprocessed after short cooling time, the following problems have to be taken into consideration with regard to the processes of receiving, storage, mechanical pre-treatment, dissolution, liquid supply adjustment, co-decontamination, high-level waste treatment and off-gas treatment:

-- Method for the removal of decay heat -- Shortening the 90-day cooling period to 30 days will increase decay heat by some 1.6 times.

-- Increased radiation damage associated with the increase of specific radioactivity.

-- The increase of the required coefficient of decontamination in regard to iodine, etc. -- Reducing the 90-day cooling period to 30 days gives rise to the need to take the DF figure up by two orders.

-- A thicker cell shield against γ rays.

-- Increased amounts of volatile nuclides such as Xe, Cs, and Ru.

(2) Modified extraction -- partition step

To move all of Pu and part of U from the TBP to the water phase through the use of a reducing counter-extraction column, it is necessary to adjust the flow rate and the acidity.

(3) Incomplete transplutonium actinide removal

If the actinide removal process is removed from the extraction process, there remain Zr, Nb and Ru together with actinide, and Zr, Nb and Ru cannot be removed unless the process is changed. It is necessary to develop a process for minimizing loss of Pu when using this concept.

B. 2. 3. 3 Evaluation of impact on economics of various modified schemes

(1) Impact of master blend

An application of co-extraction and master-blend results in an increase in the amount of Pu to be handled in every process -- increasing in proportion to the U/Pu ratio, thus resulting in a higher cost. The economic impact in this respect will be on:

- Storage of (Pu-U) nitric acid solution in the reprocessing process,
- The amount to be handled in the conversion facility,
- Storage of mixed oxide powder,
- The amount to be transported from reprocessing facility to fuel fabricating facility.

(2) Impact of short cooling time

1) Impact on facility

Measures for the removal of heat have to be taken because the decay heat of spent fuel increases, and the transport and receiving at the reprocessing plant have to be furnished with high heat capacity cooling systems. These requirements demand increased expenses for the development of processes and equipment. The need for the removal of iodine and other off-gases and for shielding against increasing γ rays is also expected to demand more spending.

2) Impact on operation

Expenses increase as increasing γ rays result in an increase in radiation damage.

(3) Impact of incomplete transplutonium actinide removal

The impact of actinide remaining in $\text{Pu}(\text{NO}_3)_4$ solution in the process of reprocessing can be considered to take the following forms:

1) Shielding of the storage facility of $\text{Pu}(\text{NO}_3)_4$ + actinide solutions in the reprocessing facility, and countermeasures against radiation exposure

When pure $\text{Pu}(\text{NO}_3)_4$ is taken as a reference, the facility cost will increase at least by two times if actinide is left to remain.

2) Shielding of conversion facility and countermeasures against radiation exposure

There is a need for remote handling in the hot cell, and when this is compared with the conversion of a pure $\text{Pu}(\text{NO}_3)_4$ into PuO_2 , facility and other costs will increase about three times more.

- 3) Shielding of the PuO₂ + actinide storage facility and countermeasures against radiation exposure
- 4) Shielding of the fuel fabrication facility and countermeasures against radiation exposure

When fuel is fabricated through the use of actinide-containing PuO₂, it requires remote operation in the hot cell. When compared with the case of PuO₂ being handled independently, facility and operation costs will increase about three times more.

What is more important is that large sums of money are needed to develop equipments for remote operation in hot cell and that the long period of development (up to 15 years) resulting in a delay for the entire project.

- 5) Shields for shipping and countermeasures against radiation exposure
- 6) Waste treatment and waste storage

What affects the storage and treatment of high active waste is the decay heat and activity of actinides and fission products. The gamma activity and decay heat of waste remain under the influence of fission products for hundreds of years after reprocessing.

However, the presence of actinide in waste will have a prominent effect more than 1,000 of years after reprocessing.

- 7) Impact of actinide on reactor concept

Expenses for technological development and equipment will have to be increased in order to verify the irradiation behavior of actinide-containing fuel, to provide shields for receiving and transport operations at the reactor sites and to give facility for remote operation and remote maintenance.

B. 3. 2 Impacts of modified reprocessing on fuel fabrication

B. 3. 2. 1 Impacts of master blend concept

(1) Technical

- 1) Adjustment and mixing of master blend powder

Since master-blended powder has Pu fissile content, as well as U fissile content, varying from lot to lot, measurements of the isotopic ratios of U and Pu must be taken each time for each lot. When U added to it to make it meet fuel specifications, it will be difficult and time consuming operation. If the amount of U to be added to meet the specification is too small compared with (U + Pu)O₂

powder, it brings on a problem of mixture uniformity. Therefore, the fissile content (U + Pu) in the master-blended powder has to be considerably larger than envisaged by fuel specifications.

2) Powder characteristics

The characteristics of the powder must be adjusted in such a manner as to make it easy for pellets to be fabricated.

(2) Economical

Storage capacity of plant

In the storage house as a whole, when the U/Pu ratio increases as 1. 2. 3....., the storage house space needs 1.4, 1.8, 2.2.....times against U/Pu=0 case.

B. 3. 2. 2 Technical impacts of various types of coprocessing on fabrication processes

(1) Co-precipitation

One co-precipitation method is adding oxalic acid or NH_4OH to a mixed nitric acid solution of Pu and U. The oxalic acid-based method has yet to be developed in point of the valence control of Pu and U, and this has a possibility for selective precipitation of Pu. The one based on NH_4OH has already been in use. Although this shows good powder characteristics, it is necessary to consider ensuring technical safety from explosion of NH_4NO_3 products.

(2) Co-denitration

Continuous fluidized bed denitration and other methods are available for co-denitration, but these are techniques now under development.

(3) Gel-precipitation

This method involves making $\text{PuO}_2\text{-UO}_2$ particles directly out of mixed nitric acid solution of Pu-U, with sphere-pac or vibro-compaction method to make fuel. The wet process is the main part of this process and, therefore, it may be difficult, for reasons of criticality control, to handle large quantities of fuel. In addition, it has not been proven of its irradiation performance as yet.

B. 3. 3 Denial of unauthorized use of fresh fabrication fuel

B. 3. 3. 1 Spiking

(1) Concept

When extracting Pu-U in the course of reprocessing, this method involves leaving actinides and fission products there or adding high-level radioactive material, so that Pu-U will not be handled unless it is heavily shielded.

(2) Method

1) Additives

Pu-U can be mixed with the following three substances:

- fission products,
- actinides,
- ^{60}Co or other high γ nuclides.

2) Adding method

The foregoing additives can be mixed by following methods:

- incomplete extraction of fission products at reprocessing plant,
- incomplete partitioning of actinides at reprocessing plant,
- deliberate addition of radioisotopes at Pu conversion process.

(3) Impact on fuel cycle

1) Spiking on pure Pu

Since shielding and remote operation are required for all the processes that come before fuel is loaded onto the reactor, time (about 15 years) and money will be needed to develop the necessary apparatus and processes. Equipment expenses and operating costs will also be prohibitively high.

-- In a reprocessing plant, there is a need for the remote operation and maintenance of Pu storage tanks.

-- In a Pu conversion plant, shielding and remote operation and maintenance are required for all processes.

-- In a fabrication plant, the remote operation and maintenance of all processes are required. Since it is impossible for remote manufacturing techniques to keep up the accuracy which is now

being maintained, new techniques will have to be developed for analysis, inspection, quality control and so on. New developments will also be required for the prevention of corrosion by fission product and for the handling of scrap recovery.

-- Transpot and fuel handling services must be rendered on a level equal to the standards for spent fuel.

-- There is no great impact on reactivity in the reactor. For example, if all the fission products produced at 2.000 MWD/T burn-up would be added to new fuels, the reduction of reactivity due to this added fission products would amount to about 1%Δk/k.

2) Impact of spiking of (Pu + U) co-process fuel

The same problem arises as with the spiking of a pure Pu. In addition to this, increasing volumes of handling in all processes from reprocessing through fuel fabrication is added problems that confront the co-process. Therefore, development will require more time and more money.

B. 3. 3. 2 Pre-irradiation

(1) Concept

Irradiation at fuel fabrication plant makes fresh fuel so radioactive that diversion of Pu will be blocked from taking place.

(2) Method

The time when irradiation is performed may be divided into three stages:

- Powder -- raw powder, mixed powder or pellets,
- Fuel pins,
- Fuel assemblies.

(3) Impact on nuclear fuel cycle

This cannot change the course of the nuclear fuel cycle as a whole, but, as is the case with spiking, it is necessary to handle it like spent fuel.

1) Impact on fuel fabrication plant

- Irradiation
 - Irradiation in the powder process

Handling after irradiation should be the same as that of spiking fuel.

- ° Irradiation at the fuel pin stage
 - Assembling is a difficult job.
 - There is no point in preventing diversion from the fuel fabrication plant unless irradiation is started from the raw powder stage.
 - To prevent diversion at the stage of transport from fuel plant to nuclear reactor, irradiation can be started after assemblies have been made.
 - In this respect, irradiation at the fuel pin stage has little significance in the light of non-proleferation.
 - ° Irradiation of fuel assemblies
 - There is a need for remote operation of loading in assembly storage houses and onto transport vessels.
 - There is a need for the installation of irradiation equipment (reactor), in the fuel fabrication plant.
 - How much irradiation should be given is a big question.
 - ° Impact on transport and reactor
 - This is the same as with spiking.