

LIMITATIONS OF ACTINIDE RECYCLE AND WASTE DISPOSAL CONSEQUENCES A GLOBAL ANALYSIS

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ABSTRACT

The paper emphasizes the impact of LWR-MOX introduction on the subsequent Actinide management and the fate of reprocessed and depleted uranium. The spent fuel from LWR-MOX contains in principle 75 % of the initially produced plutonium. This new source term has to be considered together with the minor actinides (M.A.) from the conventional reprocessing. Subsequent LWR-MOX reprocessing is the first step in a very long term Pu + M.A. management.

Recycling of Pu + M.A. in fast reactors to significantly reduce the Pu + M.A. inventory (e.g. a factor of 10) is a very slow process which requires the development and operation of a large park of Actinide Burner Reactors during an extended period of time.

The overall feasibility of the P & T option will greatly depend on the massive introduction during the next century of fast neutron reactors as a replacement of the present LWR generation of NPP's.

LIMITATIONS OF ACTINIDE RECYCLE AND WASTE DISPOSAL CONSEQUENCES

A GLOBAL ANALYSIS

1. INTRODUCTION

Very much emphasis has been put on transmutation of minor actinides and long lived fission products as an alternative or additional waste management option. This approach supposes that quantitative recycle of uranium and plutonium can be achieved and that the minor actinides and the fission products constitute the only residual radionuclide stream to be reckoned with in the long term [1,2].

However the fuel cycle with reprocessing and recycle of plutonium in LWR's is only a partial solution to the decrease of the cumulative plutonium inventory resulting from LWR operation and leaves in the most advantageous case of a double recycle scenario a residual plutonium inventory of $\sim 50\%$ which remains with the LWR-MOX spent fuel or, as a separated (and degraded) plutonium product.

The depleted uranium inventory and the separated uranium from reprocessing are in terms of mass, large bodies which have to be stored in the short term; up to now very little attention was given to these U inventories [3].

Recycle of U_{REPRO} in LWR's is one way to make proper use of an otherwise unused product-stream, it results in a decrease of fresh uranium demand with its environmental impacts during mining and milling. Depleted uranium can only be used to a very limited extent in LWR-MOX fuel. If this product-stream from reprocessing is to be exploited energetically, the use of fast reactors is a necessity otherwise it will have to be stored as a strategic resource or disposed as a waste.

In order to reduce significantly the Pu and M.A. source term the use of a fast neutron spectrum device is indispensable. MOX fuelled fast reactors or Advanced Liquid Metal cooled Reactor (ALMR's) constitute a first choice for the decrease

of the Pu-M.A. source term. However as far as present developments are concerned burnup maxima of 15 at.% and mean values up to 12 at.% have currently been achieved. Consequently a large number of recycle operations have to be scheduled in order to reduce the Pu-M.A. inventory of LWR-MOX spent fuel to e.g. 10 % of the initial Pu-M.A. content from UO₂LWR's.

Accelerator driven transmutation is a potentially interesting but very complex technology which bears promises to reach much higher burnups than achievable in fast reactors and to accelerate the actinides' transmutation.

The nuclear power production in Western Europe amounts to 120 GWe and in the US about 100 GWe; the other countries make up a joint capacity of 120 GWe. Throughout the world about equal capacities (~ 130 to 140 GWe) are committed to the fuel cycle with reprocessing and to the direct disposal option.

In order to normalize the data in this paper the quantities will be expressed in appropriate units per Ton Heavy Metal (THM) and summarized per 100 GWe installed capacity with a load factor of 0.8 and a mean fuel burnup of 40 Gwd/T. Table I summarized the output of a 100 GWe NPP park.

Table I
Annual Fuel Cycle Output from a 100 Gwe NPP park

Load factor : 0.8	Burnup : 40 Gwd/T
Electricity production	700 TWh
Discharged fuel	2200 THM
Uranium content	2088 THM
Plutonium content per THM	10.24 kg/THM
Total plutonium inventory	22.53 THM
Total Minor actinide (M.A.) inventory	2.58 THM
HLW	330 m ³ (88 t Fiss. Pr. + 2.58 T M.A.)

This table shows clearly the magnitude of the residual actinide problem (U+Pu+M.A.) which has to be resolved by an appropriate waste management strategy.

2. ONCE-THROUGH-STRATEGY [4]

According to this scenario the entire output of spent fuel is to be stored and eventually disposed of in deep geologic structures. Since a disposal site is limited (in the US) to 70 000 THM according to the present US legislation, every 32 years a new disposal site has to be made available. The Pu + M.A. inventory of such a disposal site would amount to ~ 800 THM Pu which is a worrisome prospect. Taking into account that depending on the cooling time 57-66 % of the Pu present is fissile material, the loss of resources amounts to 408-473 T Pu fissile and long term criticality issues have to be investigated before adopting such strategy.

3. REPROCESSING AND Pu RECYCLE STRATEGY

The reprocessing strategy with reuse of Pu in LWR-MOX fuel with partial reuse of reprocessed uranium and with vitrification of high level waste is historically the most important venture in Europe.

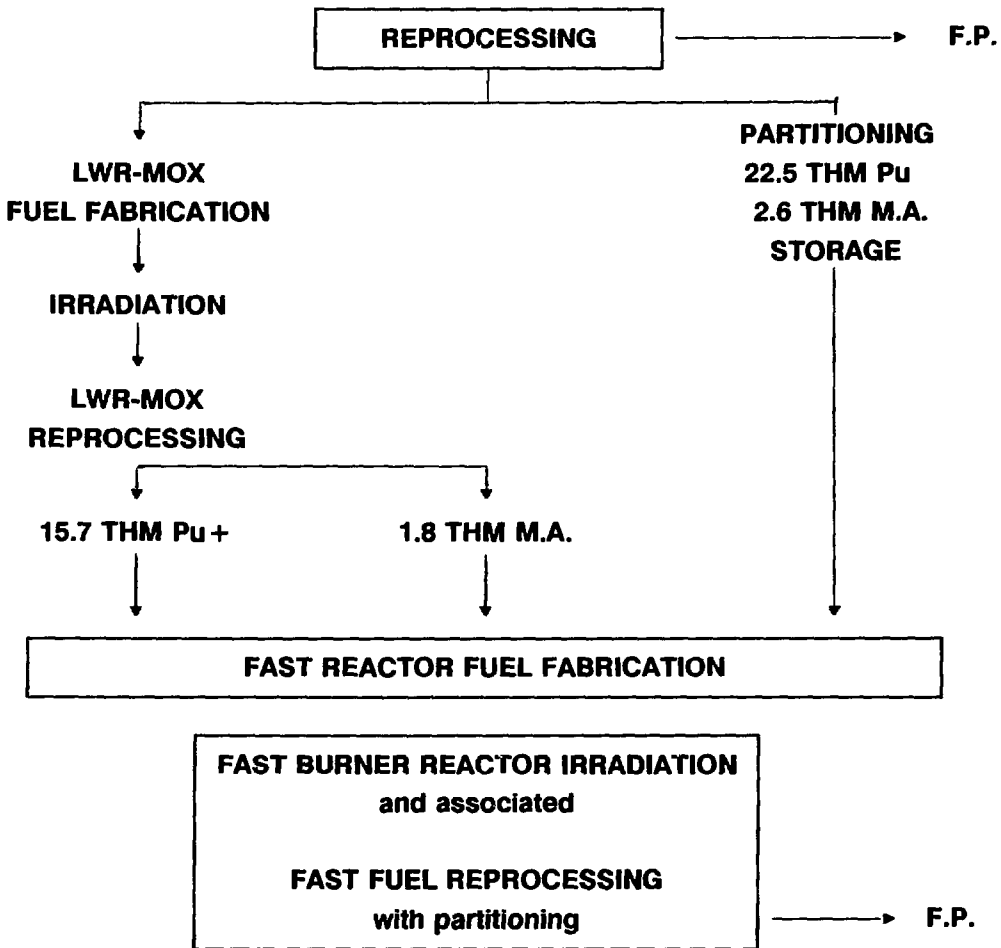
The quantity of Pu produced by 100 GWe NPP park amounts annually to 22.53 T Pu. This quantity is sufficient to provide 20 % of a second core reload i.e. a park loaded with 80 % UO₂ and 20 % MOX fuel. During this second core irradiation Pu is produced in the UO₂ core and depleted in the MOX fuel. The overall balance of electricity, fissile or fertile products, and waste streams is given in Table II.

The economic benefit results from the surplus electricity production (140 TWh) which has to compensate the costs of reprocessing, transport and MOX fuel fabrication.

If we compare the data of Table I and II it is obvious that the total Pu inventory after two "once through" cycles would amount to 45 THM Pu while it is reduced to 33.7 THM Pu if a combination is made of one UO₂ and one UO₂-MOX cycle with intermediate reprocessing and partial MOX use in the second cycle. The overall Pu inventory reduction is important (25 %) but does not change the long term waste management approach since a substantial fraction i.e. 46 % (or 15.68 THM Pu) remains occluded in the discharged MOX spent-fuel elements (Fig. 1).

Figure 1: Pu + M.A. Incineration Flowsheet

100 GWe = 2200 THW/year



YEARS	BURNUP %	CYCLES	INCINERATION %
182	15	14	86.4
130	20	10	86.3
104	25	8	86.5

RESIDUAL Pu + M.A.
Reactor Fuel Inventory
 10 % of initial Pu + M.A. throughput

Table II
Output of a 100 GWe NPP park with 20 % MOX Fuel at a
Global Burnup of 40 Gwd/T after 7 Year Cooling

Electricity	700 TWh (560 TWh from UO ₂ 140 TWh from MOX)
Fresh fuel	2200 T (1760 T HM UO ₂ , 440 T HM MOX)
Discharged fuel	2200 T
1760 THM (UO ₂)	1669 T U, 70.4 T FP, 18.02 T Pu, 2.06 T M.A.
440 THM MOX	404.5 T U, 17.6 T FP, 15.68 T Pu, 1.8 T M.A.
Totals	2073 T U, 88 T FP, 33.7 T Pu, 3.86 T M.A.
Reprocessing yields of UO ₂ fuel	1660 TU 18.0 T Pu, 250 m ³ HLW (70.4 T FP + 2 T M.A.)
Spent MOX fuel	440 T HM (with 15.68 T Pu, 1.8 T M.A.)

3.1. Reprocessing of MOX fuel

Reprocessing of a spent MOX fuel load is technically feasible but much more difficult than conventional UO₂ spent fuel. Table III shows a comparative spent fuel composition of UO₂ and MOX fuel at the same burnup, 7 years after discharge and the resulting compositions 100 and 500 years after discharge if no reprocessing would have been done.

The technological difficulties associated with reprocessing of MOX fuel are the following.

- Industrial reprocessing of large quantities (440 THM) of MOX fuel requires a very advanced equipment to cope with the increased criticality risk during MOX dissolution. The construction of dedicated head-end facilities has to be envisaged. Small quantities of MOX fuel can be "diluted" with UO₂ fuel.
- The high concentration of α emitting nuclides particularly Pu-238 and Cm-244 will seriously increase the solvent degradation. The overall α level is 7 times higher with recycled MOX fuel than with equivalent UO₂ fuel of the same burnup. The use of centrifugal extractors in the first extraction cycle

might perhaps mitigate the solvent degradation due to Cm-244, but the Pu-238 influence remains until the loaded TBP has been stripped in the BX column.

Table III

Evolution of Pu and M.A. content in UO₂ and MOX spent fuel of 40 Gwd/T compared to reprocessing residues of UO₂ fuel [5,6]

SPENT FUEL CASE - QUANTITIES IN Ci/THM AND g/THM							
DECAY TIME		t = 7 y		t = 100 y		t = 500 y	
		UO ₂	MOX	UO ₂	MOX	UO ₂	MOX
Pu-238	Ci	3.48 10 ³	1.59 10 ⁴	1.67 10 ³	7.66 10 ³	66.8	305.3
	g	203	934	97	448	3.9	17.9
Pu TOT	Ci	1.07 10 ⁵	5.20 10 ⁵	3.79 10 ³	1.68 10 ⁴	9.84 10 ²	3.68 10 ³
	g	10,243	35,644	9,140	30,175	8,826	29,065
	Ci α	4.67 10 ³	1.95 10 ⁴	3.78 10 ³	1.11 10 ⁴	9.84 10 ²	3.68 10 ³
Np-237	Ci	0.4	0.166	0.49	0.6	0.56	2.28
	g	569	237	705	848	810	3,947
Am-241	Ci	1.5 10 ³	8.2 10 ³	4.2 10 ³	2.43 10 ⁴	2.20 10 ³	1.25 10 ⁴
	g	438	2,400	1,230	7,110	645	3,675
Am-243	Ci	25.6	198	25.3	196	24.4	189
	g	128	996	127	987	122	950
Cm-243	Ci	21.1	152.7	2.2	15.9	-	-
	g	0.408	2.96	0.042	0.308	-	-
Cm-244	Ci	2.75 10 ³	3.25 10 ⁴	78.6	927	-	-
	g	34.1	403.2	0.97	11.5	-	-
Cm-245	Ci	0.27	14.1	0.268	14.0	0.26	13.5
	g	1.59	46.18	1.58	45.82	1.52	44.3
Cm-246	Ci	0.56	1.67	0.55	1.647	0.52	1.55
	g	1.83	5.43	1.8	5.35	1.7	5.04

- The occurrence of (900-1000 g Pu-238/THM) will complicate the Pu purification, particularly the reextraction/stripping steps, due to a higher level of TBP degradation products especially DBP which strongly complexes Pu species.
- The heat emitted by Pu-238 (0.5 W/g) will call for increased precautions during the storage and transportation of Pu containers. This phenomenon will become much more important if multiple MOX recycle would be considered.

The use of plutonium resulting from MOX recycle will mostly affect the operations in the fuel fabrication plant where increased levels of Pu-238 and Am-241 will boost the neutron and gamma radiation around the glove boxes.

- The neutron emission increases with 30 % when Pu from spent MOX fuel would be reworked in a fuel fabrication plant. This will mostly affect the powder mixing and grinding steps where large quantities of Pu bearing materials are handled simultaneously. The multiple recycle of MOX fuel as would be required in a P & T scenario will further increase the handling difficulties and call for the development of fully "remote handled" fuel fabrication facilities with remotely operated maintenance.

Delayed reprocessing of spent MOX fuel seems at the present state of the art of the technology as the best alternative to direct disposal. A storage of 50 years, time period also considered for vitrified fission products, is beneficial as it would decrease the Cm-244 content of the spent MOX fuel with a factor of 7 and ease further separations and purification operations. The choice between disposal and reprocessing after that time interval will depend on the availability of fast reactors or accelerator driven transmutation systems for burning massively the Pu and residual M.A..

3.2. Minor Actinides and Fission Products' Partitioning

Reprocessing is an indispensable step in making Pu available for reuse in LWR-MOX fuel as discussed above but this operation leads to the generation of HLLW which contains M.A. and fission products. The fission products are a "short term" concern. Each tonne of spent UO₂ fuel contains ~ 5000 Ci α emitting M.A.. In order of importance Cm-244, Am-241 and far behind Np-237.

Vitrification of fission products is the last step before long term storage and disposal. This implies that the partitioning operation of M.A. ought to be carried out shortly after reprocessing. But during that time interval (7 to 10 years) Cm-244 is the strongest α emitter which will influence the methods and equipment to be used for these partitioning steps.

Successively the partitioning operation will require the following separations:

- Extraction of M.A. + Rare Earths from HLLW.
- Separation of M.A. from Rare Earths.
- Conditioning Np, Am and Cm for long term storage.

The strongly radioactive Rare Earths and the highly-active isotope Cm-244 (α and neutron emitter) interfere with the use of complex and fragile organic molecules which undergo radiation damage. By combining the TRUEX process [7] with a Actinide/Rare Earth separation (CMPA, DTPA) Am and Cm can both be removed together from the HLLW.

The Am-Cm fraction has to be stored indefinitely or transferred to a transmutation scenario.

The Rare Earths are recombined with the other Fission Products (Cs-137, Sr-90, Ru-106, ...) and transferred to the vitrification facility. The vitrified products have to be stored during ~ 50 years before disposal because of their high heat output.

By delaying the reprocessing step, Am-241 instead of Cm-244, becomes the most important α emitting radionuclide. Some long lived *fission products* are of importance in a future partition strategy. Table VI contains the most important long lived ($t_{1/2} \geq 30$ y) fission products. The data for UO₂ fuel are very similar and are not treated separately.

As can be deduced from the data of Table IV, Cs-137 and Sr-90 determine the decay pattern of the fission products up to 500 years. Beyond that period some long lived fission products draw the attention: Tc-99 and Sm-151 make up 70 % of the total residual FP activity but Tc-99 is the most important one because of its very long half life.

Table IV
Concentration of long lived fission products in Ci/THM
at different cooling times from MOX fuel irradiated at 40 Gwd/T

NUCLIDE	t $\frac{1}{2}$ /(y)	Cooling times after discharge			
		7 y	50 y	100 y	500 y
Se-79	10 ⁴	0.4	0.4	0.4	0.4
Sr-90	28.5	3.62 10 ⁴	1.27 10 ⁴	3.77 10 ³	0.29
Y-90	equil.	3.62 10 ⁴	1.27 10 ⁴	3.77 10 ³	0.29
Tc-99	2.1 10 ⁵	15.7	15.7	15.7	15.6
Zr-93	1.5 10 ⁶	1.5	1.5	1.5	1.5
Nb-93m	3.6 10 ⁷	1.42	1.42	1.42	1.42
Pd-107	6.5 10 ⁶	0.32	0.32	0.32	0.32
Sn-121	50	0.41	0.22	0.11	-
Sb-126	~ 10 ⁵	1.43	1.43	1.43	1.43
Sn-126	equil.	1.26	1.26	1.26	1.26
I-129	1.6 10 ⁷	0.05	0.05	0.05	0.05
Cs-135	2 10 ⁶	0.8	0.8	0.8	0.8
Cs-137	30.1	1.07 10 ⁵	3.97 10 ⁴	1.25 10 ⁴	1.21
Ba-137	equil.	1.01 10 ⁵	3.97 10 ⁴	1.25 10 ⁴	1.14
Sm-151	93	9.08 10 ²	6.54 10 ²	4.54 10 ²	20.45
TOTAL		4.06 10 ⁵	1.19 10 ⁵	3.25 10 ⁴	46.3

Though very low in absolute value I-129 plays an important role in the dose to man because of its biologic concentration in the thyroid gland. According to present practice I-129 is discharged into the sea or stored on land as AgI.

According to new trends in the reprocessing technology *Tc-99* [8] can be separated from the other fission products by an intermediate extraction process between the HS and BX column of the first PUREX extraction cycle. However this "partitioning" has not been conceived for environmental reasons but because Tc-99 which is present in the F.P. mixture at a level of ~ 926 g/THM interferes with the quantitative Pu (III) stripping from the loaded TBP as it destroys hydrazine, a reductor used as valency stabilizer. The partitioning of Tc-99 is

important as this very long lived nuclide could interfere with the long term disposal criteria of HLW in oxidative conditions. In reducing geologic disposal conditions (e.g. clay). Tc-99 is stable as a Tc-TcO₂ compound with very low solubility. From environmental point of view only **Kr-85 and I-129** are dispersed in the air and in the oceans. Kr-85 with 10.7 y half life is not a long term threat but I-129 with its extreme long half life is a critical nuclide on the very long term. If direct disposal is chosen as fuel cycle strategy the presence of I-129 may become the limiting factor of the repository.

Since Sm-151 is a member of the Rare Earth family and as its half life is 93 years, it vanishes within the scheduled period of disposal (i.e. between 1000 and 10 000 years).

The case of C-14 deserves special attention though it is an activation product. During the dissolution of fuel it is discharged with the dissolver off gases but can be trapped if deemed necessary (BNFL). The concentrated ¹⁴CO₂ fraction can be stored as BaCO₃. The ultimate destination of such C-14 concentrate is the geologic repository [9].

4. LIMITS OF ACTINIDES' RECYCLE AND IMPACT ON DISPOSAL

Up to very recently the recycle of long lived Actinides meant only : the recovery of M.A. from HLLW, the conditioning of M.A. for long term storage and the preparation and irradiation of M.A. targets in nuclear reactors or accelerator-driven transmutations. This way of thinking resulted from the underlying philosophy that after reprocessing the "valuable" elements, U and Pu were quantitatively recycled. In reality neither reprocessed uranium (U_{REPRO}) nor Pu are quantitatively recycled.

Whatever optimistic recycle scenario chosen large amounts of unused uranium will have to be stored or disposed of if there is no massive break-through of the fast reactor technology.

4.1. Residual uranium characteristics and their consequences

The radiological significance of residual uranium may not be neglected especially because large masses are involved. Table V summarizes the data.

A nuclear capacity of 100 GWe demands a uranium feedstock of 17 000 T/year which is transformed into 2200 THM U 3.7 % [10]. The ore body containing the extracted uranium contains 46.750 Ci α which remains on the mining site in the tail ends. This activity decreases with the half life of Ra-226 (1640 y).

Table V
Comparative radiologic characteristics of U products in Ci/THM

<u>Nuclide</u>	$t_{1/2}(y)$	Series	α/At	Nat. U	Depl. U	UO ₂ U _{REPRO}	MOX U _{REPRO}
U-232	73.6	4 n	7	-	-	0.0475	0.0337
U-234	2.74 10 ⁵	4 n+2	7	(0.3131)	-	0.0885	0.6264
U-235	7.1 10 ⁸	4 n+3	7	0.0154	0.0035	0.0166	0.0069
U-236	2.39 10 ⁷	4 n	8	-	-	0.311	0.0469
U-237	1.33 10 ⁻⁵	4 n+1	7	-	-	(2.58)	-
U-238	4.5 10 ⁹	4 n+2	8	0.3303	0.3303	0.314	0.3048
TOTAL				0.3457	0.3356	0.77	1.0187
TOTAL (α) AT GEOLOGIC EQUILIBRIUM				2.75	2.67	6.05	7.556

About 14 800 THM depleted U is accumulated per year and will reach its initial activity level over a very extended period of time. This uranium "residue" ought therefore to be considered as a strategic stockpile of uranium for a future fast reactor era.

The spent fuel-uranium mass (2200 THM/100 GWe year) will contain 3.35 Ci/THM after reprocessing and this α activity level will increase up to 6.05 Ci/THM at secular equilibrium. This corresponds to 7370 - 13 300 Ci α for the total uranium inventory of the discharged spent UO₂ fuel. The α activity of U in UO₂ contained in MOX spent fuel is of the same order of magnitude.

Some minor actinides e.g. Np-237 which are at 0.16 to 2.28 Ci α /THM spent fuel contribute only with 350 to 5000 Ci to the overall α activity. Such comparisons ought to be further investigated in order to establish the effective contribution of M.A. to the overall radiotoxicity in order to better understand the potential benefit of a given recycle scenario.

Recycling U from spent UO_2 fuel involves that, apart from reprocessing which is a necessity, a series of additional operations have to be carried out:

- fluorination of UO_2 to UF_4 and UF_6
- enrichment of U-235 from 0.8 to 3.7 %
- conversion to U_3O_8 and UO_2 .

Enrichment of U_{REPRO} has to be carried out in a centrifuge-enrichment-plant in order to avoid permanent build up of Th-228 in the diffusion membranes of a diffusion plant. During enrichment the concentration of lighter isotopes (U-232, U-234) increases in the same proportion as U-235 (a factor of 7) and multiplies with the same factor the α activity of recycled and reenriched uranium. The ingrowth of Th-228 reaches 30 % of the equilibrium value after 1 year, 50 % after 1.9 year, 84 % after 5 years and nearly equilibrium after 10 years.

The utilities should therefore be encouraged to recycle U_{REPRO} quickly in order to avoid this buildup. However economic factors have led to the accumulation of large inventories of U_{REPRO} in secular equilibrium with the Th-228 decay series.

In the present economic circumstances of very low uranium market prices, recycle of uranium has to be justified by other arguments e.g. decrease of dose to the workers at the mining and milling facilities, or supported by waste minimisation arguments or justified as a strategic resource preservation.

It is therefore not excluded that significant uranium inventories containing radiotoxic impurities will be considered as a waste stream rather than as a resource and will be conditioned for final disposal. In such a scenario the separation and recycle of Np becomes from radiologic point of view unnecessary.

4.2. Recycle and incineration of plutonium versus conventional disposal options

The first result of Pu recycle in LWR's is its "concentration" in a smaller number of fuel elements reducing the MOX elements to 1/5 of the initial UO_2 spent fuel elements. Secondly the total Pu quantity is reduced by 25 % in a single recycle.

It is not excluded to consider a second LWR recycle but in that case "early"

reprocessing has to be provided for, in order to avoid the loss of important Pu-241 quantities. However as explained above the reprocessing and fuel refabrication operations are in that case complicated by the presence of Cm-244 and Pu-238. Long term storage of spent MOX fuel (~ 50 years) with reprocessing and recycle of Pu and M.A. in a fast reactor park is probably the most realistic scenario which transfers the Pu + M.A. inventory from a storage facility to a reactor which produces at the same time electric power.

However the combustion also called "incineration" of Pu + M.A. is only a partial process. At the present time the burnup i.e. the percentage of fission, varies from 12 to 15 at. % which means that at the end of the cycle the fuel still contains 85 to 88 % of the initial actinides. A strategy of Pu + M.A. incineration involves obviously the gradual replacement of the thermal reactor park by fast reactors. Such replacement will take decades to be accomplished.

A typical 1000 MWe fast actinide burner reactor has an initial loading of 6 T Pu + A.M. and is operated over a 4 year cycle to reach burnups of 12-15 %. The initial enrichment is 18 to 25 % total Pu depending on the isotopic composition. The annual output coming from the LWR's is 22.5 T Pu which can be reduced to 15.68 T Pu if a single passage to LWR-MOX fuel is inserted.

The fast reactor park necessary to take up the large quantities resulting from a 100 GWe LWR park is of course very impressive. In case of UO₂ fuel cycle 16 GWe FN Burner Reactors and in the MOX case 12 GWe have to be constructed and operated to "use" the residual Pu + M.A. as a fuel. Each cycle takes 4 years to be accomplished followed by at least 7 years cooling and two years for reprocessing and fuel-refabrication. At the end of a 13 year's cycle of that limited reactor park the initial Pu + M.A. load would be reduced to about 85 % of the initial inventory. From then on a second strategic decision must be taken with respect to the further elimination of the residual Pu + M.A. load by fissioning. In order to reduce the annual output to e.g. 10 % of the initial LWR output i.e. 2.5 T Pu + M.A. per year an additional fast reactor park almost equal to the initial LWR-NPP park has to be constructed and operated during 14 cycles of each 13 years of the shrinking Pu + M.A. inventories (Fig. 2). The required "incineration" capacity strongly depends on the attainable burnup. Table VI shows some typical values. The residual content of the cores 13-15 % of the input have to be further fissioned or disposed of any way in a geologic repository.

Pu + M.A. INCINERATION CASCADE

UO₂ FUEL

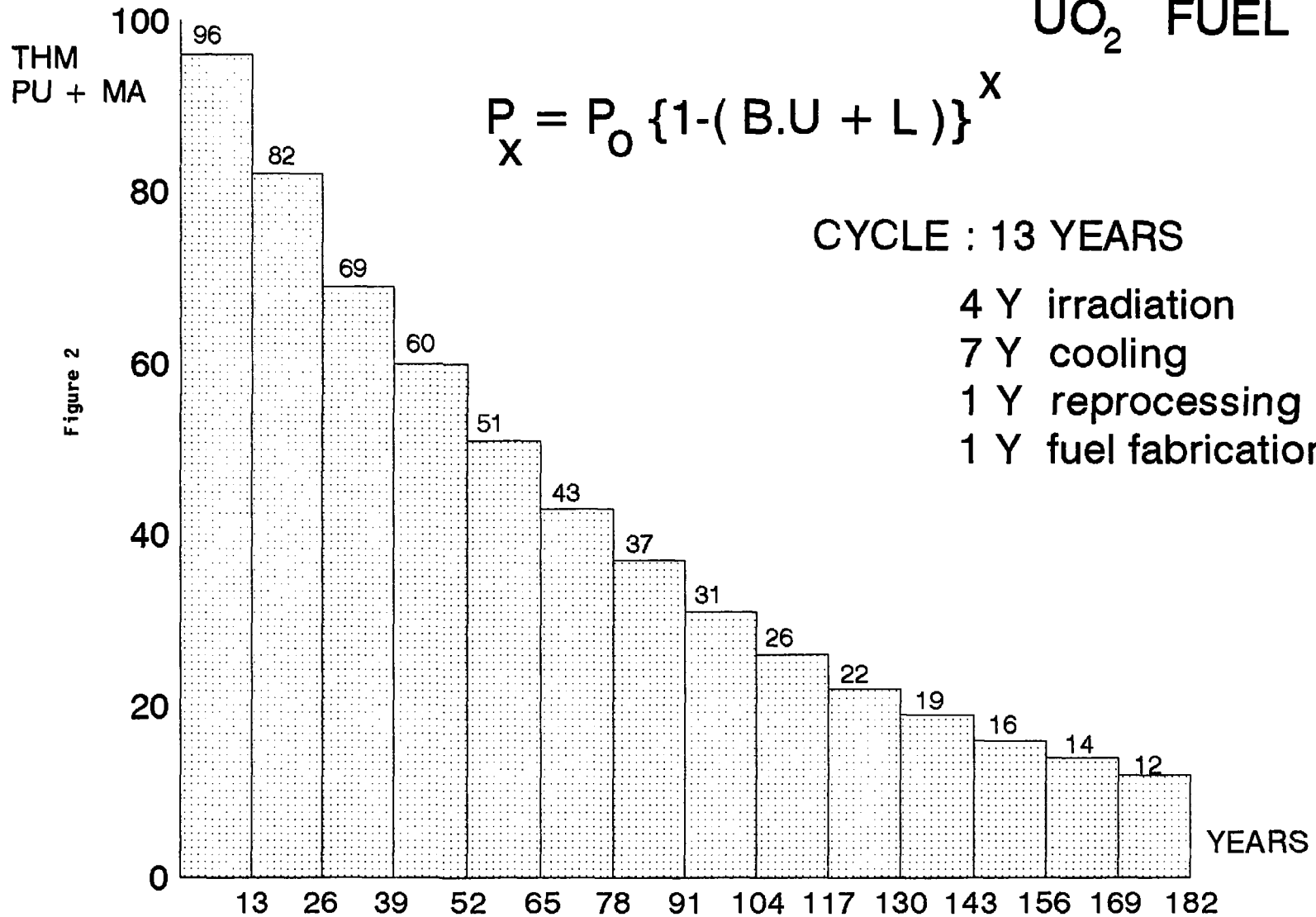


Figure 2

Table VI
Actinide reduction yields

Burnup %	Cycles	Years	F.R. Capacity GWe	Pu + M.A. reactor	Reduction factor
15	14	182	72.6	13.6 %	7.35
20	10	130	52	13.7 %	7.29
25	8	104	44	14.5 %	6.9
30	6	78	36	15.8 %	6.3

Unless a real fast reactor-breeder economy is going to replace the present LWR-UO₂ era it does not look very attractive to consider such massive investments just to destroy Pu + M.A. with a factor of 7.

Accelerator driven transmutation is a potentially more interesting technology since it can burn actinides to a much higher burnup than achievable in a fast reactor. The bottle neck of increasing burnup lies with the cladding behaviour but in this field considerable progress has been made with the ferritic steels. If for example the burnup could be doubled to e.g. 25-30 % the elimination time will be considerably shortened and the number of recycles significantly reduced to 5 or 6. It may take several decades to develop this technology from the conceptual design at present to a fully industrial plant in the first quarter of the next century.

However for the same throughput the investment is also doubled and the technological complexity of the accelerator-transmuter device may jeopardize the operability of such a plant on a year round, 24 h a day, basis.

4.3. Disposal option of MOX fuel

In case of direct disposal 2200 THM spent LWR-UO₂ fuel has to be disposed of every year. In case of a generalized use of LWR-MOX fuel the Pu inventory would be reduced to 75 % and even to 56 % in case of two consecutive recycles. But in this case the quantity of M.A. particularly Am and Cm will become much higher and will dominate the radiological impact during the first 4000 years. A 100

GWe NPP park with partial MOX loading produced as shown above about 440 THM/year of spent fuel.

The disposal of Pu + M.A. rich fuel will necessitate a particular conditioning operation by which any danger of criticality is eliminated e.g. by embedding the fuel elements in refractory type neutron absorbing materials (B_4C). The other precautions to be taken are nearly the same as for vitrified HLW.

However it must be emphasized that the decision to dispose of Pu + M.A. is more of an ethical nature rather than of technical or radiological nature. A long term retrievable surface storage of spent LWR-MOX fuel is therefore the most adequate decision to be taken within the current decade.

4.4. Disposal of waste

Conventional reprocessing of spent LWR- UO_2 fuel produces apart from U and Pu a HLLW solution which contains all the fission products and the most important M.A. Presently this HLLW is stored for a few years and then vitrified. The glass canisters are then stored for ~ 50 years till the heat dissipation has decreased to levels acceptable for underground disposal.

As shown in Table I the net-volume of HLW to be stored amounts to 330 m^3 /year in case of a LWR- UO_2 scenario and 250 m^3 in case of a LWR-MOX scenario. Table IV shows the specific activity of each of the important radionuclides, and Table III gives an overview of the M.A. concentrations in spent fuel. The data shown for the M.A. are also valid for HLW except the Am-241 levels which are lower in case of reprocessing. The Am-241 levels in HLW after reprocessing of UO_2 or MOX fuel are for completeness given below.

	t + 7 y		t = 100 y		t = 500 Y	
	UO_2	MOX	UO_2	MOX	UO_2	MOX
Am-241 Ci/THM	1500	8200	1290	7066	673	3681
Am-241 g/THM	438	2400	377	2066	196	1076

In earlier publications it has clearly been shown that Cs-137, Sr-90, Am-241 and Cm-244 dominate the radiotoxic potential during the first 200 years. Gradually

However the concurrent production of additional power from these processes is a condition sine qua non for a favourable outcome. This scenario implies the deployment of a very large fast reactor park or of large accelerator driven transmutation devices.

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