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^{231}Pa and ^{232}U production in a fusion breeder to aid nonproliferation with thorium fission fuel cycles

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Abstract

^{231}Pa is made especially copiously in a fusion reactor blanket by n,2n reactions on ^{232}Th owing to fusion's uniquely high neutron energy being well above the reaction threshold unlike fission neutrons. The ^{231}Pa can be extracted for use in making thorium cycles more proliferation resistant or left in the fusion reactor's blanket to produce ^{232}U for self-protection of the produced ^{233}U or some of both. Typical fusion production rates of ^{231}Pa and ^{233}U are of order 0.1 and 2 kg per full power year per MW_{fusion}, respectively. Neutrons captured in ^{231}Pa produce ^{232}U that contributes to making ^{233}U a nonproliferant. ^{231}Pa revenues per $W_{nuclear}$•y range from 0.08 $ to 0.5 $ depending on blanket design and market value of isotopes. By comparison, the electricity revenues is typically 0.1 $ at Q=2 and falling for Q<2 (Q=fusion power/input power).

1. Introduction

This note describes a fusion breeder designed to produce ^{231}Pa, ^{232}U and ^{233}U for use in molten salt reactors that are described in a companion note.[1] Special emphasis is given to nonproliferation of weapons useable materials. As opposed to a fission breeder reactor, a fusion breeder reactor can produce far more fissile material for the same amount of nuclear power by an order of magnitude so that we can think of a fusion breeder being located in a site and supplying the isotopes (^{231}Pa, ^{232}U and ^{233}U) to dozens of fission reactors at various sites. We prefer to use molten salt in both the fusion breeder and the "client" fission reactors for cost and safety reasons. The plan to satisfy nonproliferation concerns is two fold:

1-spike the fissile material ^{233}U with the 2.6 MeV gamma emitting ^{232}U so that the standard of "self-protection"[2] is achieved and

2-apply safeguards, transparency and openness of all operations

If needed, ^{232}U can be produced in situ in a fission reactor from neutron capture in ^{231}Pa that is supplied by the fusion breeder in a unique way because the neutron reaction that

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1. R. W. Moir, "Nonproliferation role of ^{231}Pa and ^{232}U in a Molten Salt Reactor on the Th-^{233}U fuel cycle," Vallecitos Molten Salt Research Report No. 6, Rev.1 (September 9, 2014).

2. When the gamma radiation activity is 1 Sv/h one meter from 5 kg of ^{233}U if the ^{232}U content =2.4% one year after removal of all decay products, the IAEA states the material is so harmful (deadly in a few hours) to be near by that it is designated to be "self-protected". J. Kang and F. N. von Hippel, "U-232 and the proliferation resistance of U-233 in spent fuel," *Science & Global Security*, 9 (2001) 1-32.

produces ^{231}Pa has a threshold at ~6 MeV and the 14 MeV fusion neutrons are above this threshold.

2. Background

Fusion technology is not known well enough to build a fusion breeder today to meet the requirements that will be spelled out in this note and its companion.[1] However, the fusion technology requirements are not far from today's state of the art and significantly short of the goal of the international fusion development goal of commercial electrical power production. Therefore, we can contemplate the possibility of a second generation molten salt reactor utilizing fusion produced isotopes whereas the first generation could use enriched uranium and thorium. The first generation of reactor would use nonweapons grade uranium, that is ^{235}U <20% of ^{235}U +^{238}U as its startup and supplied makeup fuel. Then very similar designs could use fusion produced ^{233}U along with the nonproliferants ^{232}U and ^{231}Pa that are the subject of this note.

3. Production rates and revenues for ^{233}U and ^{231}Pa

The number of reactions $F_{n,2n}$, $F_{n,3n}$ and $F_{n,\gamma}$ and usually tritium production is 1.1 tritons all per source 14.1 MeV neutron. The production rate PR is then:

$$PR = F \cdot P_{fusion}(MW)\frac{232.038 \times 1.66054 \times 10^{-27} kg \times 3600 \times 24 \times 365.25}{17.58\ MeV / fusion \times 1.602 \times 10^{-19} J / eV}$$

$$= 4.318 \frac{kg}{MW_{fusion} \cdot y} F \cdot P_{fusion}(MW) \qquad (1)$$

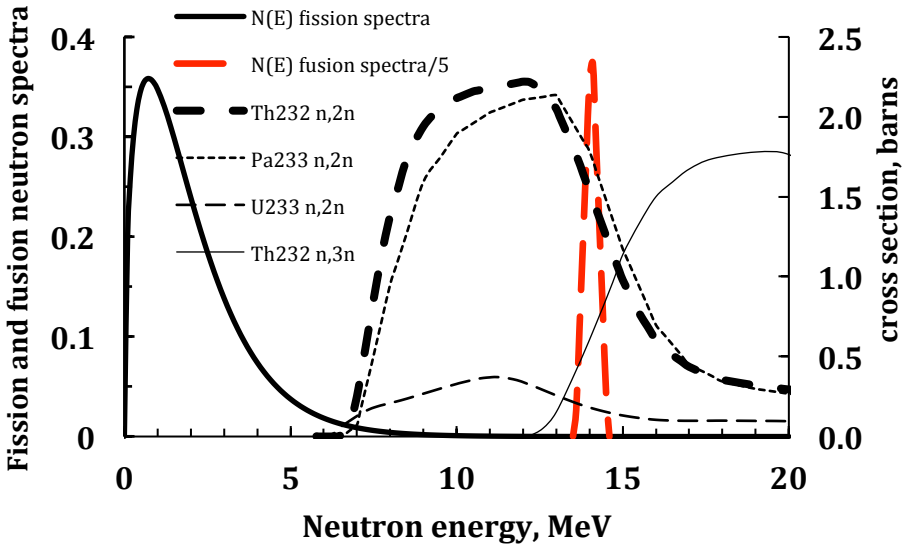

Fig. 1. Neutron spectra from fission and fusion and n,2n cross sections relevant to producing ^{231}Pa: ^{232}Th(n,2n)^{231}Pa.

For the **Li/MS case** (M=1.37, where M is the blanket energy released produced by each fusion neutron divided by 14.1 MeV, $P_{nuclear}/P_{fusion}$=0.2+0.8M=1.296) where Li is the neutron multiplication material and molten salt is the thorium carrier we have approximate reaction numbers[3] (from p 15 of Table 3 of the reference). The geometry is cylindrical but toroidal or spherical geometry should give results within <±20%.

$F_{n,2n}$ = 0.0246	^{231}Pa	32,800 y half-life
$F_{n,3n}$ = 0.00229	^{230}Th	75,400 y half-life
$F_{n,\gamma}$ = 0.515	^{233}Pa	27.0 d half-life, decays to ^{233}U

$$PR = 0.1062 \quad \frac{kg\ ^{231}Pa}{MW_{fusion}\cdot y} P_{fusion}(MW) \rightarrow \frac{\$106{,}000}{MW_{fusion}\cdot y} P_{fusion}(MW)\ for\ \$1000/g\ of\ ^{231}Pa$$
$$= 0.009887 \frac{kg\ ^{230}Th}{MW_{fusion}\cdot y} P_{fusion}(MW)$$
$$= 2.224 \quad \frac{kg\ ^{233}Pa}{MW_{fusion}\cdot y} P_{fusion}(MW) \rightarrow \frac{\$130{,}000}{MW_{fusion}\cdot y} P_{fusion}(MW)\ for\ \$60/g\ of\ ^{233}U \qquad (2)$$

The example market values used here for ^{231}Pa and ^{233}U are derived in a companion note,[1] ^{233}U usually has 2.4% ^{232}U, enough to satisfy IAEA standard for "self-protection" in our example cases.

The production rate and revenues per unit of fusion power can be converted to per unit nuclear power. $P_{nuclear}/P_{fusion}$=0.2+0.8M=1.296. The revenues then become 0.0819 $/W$_{nuclear}$•y for ^{231}Pa and 0.100 $/W$_{nuclear}$•y for ^{233}U for a total of 0.182 $/W$_{nuclear}$•y.

For the **Be/MS case** (M=2.1, $P_{nuclear}/P_{fusion}$=0.2+0.8M=1.88) where Be is the neutron multiplication material and molten salt is the thorium and lithium carrier we have approximate reaction numbers[1]:

$F_{n,2n}$ = 0.0267	^{231}Pa
$F_{n,3n}$ = 0.0054	^{230}Th
$F_{n,\gamma}$ = 0.780	^{233}Pa

$$PR = 0.1153 \quad \frac{kg\ ^{231}Pa}{MW_{fusion}\cdot y} P_{fusion}(MW) \rightarrow \frac{\$115{,}300}{MW_{fusion}\cdot y} P_{fusion}(MW)\ for\ \$1000/g\ of\ ^{231}Pa$$
$$= 0.02331 \frac{kg\ ^{230}Th}{MW_{fusion}\cdot y} P_{fusion}(MW)$$
$$= 3.368 \quad \frac{kg\ ^{233}Pa}{MW_{fusion}\cdot y} P_{fusion}(MW) \rightarrow \frac{\$202{,}000}{MW_{fusion}\cdot y} P_{fusion}(MW)\ for\ \$60/g\ of\ ^{233}U \qquad (3)$$

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3. R. W. Moir. Production of U-232 and U-233 in a fusion-fission hybrid, Vallecitos Molten Salt Research Report No. 3 (December 21, 2010), 37 pages. http://www.ralphmoir.com/media/moirProdu232_12_21_2010.pdf

The revenues then become 0.0613 $/W$_{nuclear}$•y for ^{231}Pa and 0.107 $/W$_{nuclear}$•y for ^{233}U for a total of 0.169 $/W$_{nuclear}$•y.

In this case the ^{231}Pa production rate decreases by 30% but the ^{233}U production is about the same compared to the Li/MS case for the same nuclear power.

Other blanket designs have different ^{231}Pa production rates. Higher thorium concentrations increases $F_{n,2n}$ but typically lowers $F_{n,\gamma}$.
For example an **all molten salt blanket** (M=2, $P_{nuclear}/P_{fusion}$=0.2+0.8M=1.80) with no separate Li or Be multiplier yields the following rates[1]:

$F_{n,2n}$ =0.188 ^{231}Pa
$F_{n,3n}$ =0.051 ^{230}Th
$F_{n,\gamma}$ =0.236 ^{233}Pa

$$PR = 0.812\frac{kg\ ^{231}Pa}{MW_{fusion}\cdot y}P_{fusion}(MW) \rightarrow \frac{\$812{,}000}{MW_{fusion}\cdot y}P_{fusion}(MW)\ for\ \$1000/g\ of\ ^{231}Pa$$
$$= 0.220\frac{kg\ ^{230}Th}{MW_{fusion}\cdot y}P_{fusion}(MW)$$
$$= 1.019\frac{kg\ ^{233}Pa}{MW_{fusion}\cdot y}P_{fusion}(MW) \rightarrow \frac{\$60{,}000}{MW_{fusion}\cdot y}P_{fusion}(MW)\ for\ \$60/g\ of\ ^{233}U \quad (4)$$

The revenues then become 0.451 $/W$_{nuclear}$•y for ^{231}Pa and 0.033 $/W$_{nuclear}$•y for ^{233}U for a total of 0.484 $/W$_{nuclear}$•y.

In this case the ^{231}Pa production rate increases 5.5 times (~7 times when including the ^{230}Th production) but the ^{233}U production rate drops a factor of 3.2 compared to the Li/MS case for the same nuclear power.

^{231}Pa production can be increased significantly with modest reduction in ^{233}U production by placing some molten salt containing thorium in front of the lithium in the Li/MS case. If thorium metal were used the enhancement would be even stronger as can be inferred from the following example.

For an **ideal infinite media of Th/^{6}Li case** (M=3.7, $P_{nuclear}/P_{fusion}$=0.2+0.8M=3.16) (83.24 a% ^{232}Th and 16.76 a%^{6}Li) where tritium breeding is the usual 1.1 per source neutrons, we have approximate reaction numbers:

$F_{n,2n}$ = 0.568 ^{231}Pa
$F_{n,3n}$ = 0.235 ^{230}Th
$F_{n,\gamma}$ = 1.417 ^{233}Pa

$$PR = 2.453\ \frac{kg\ ^{231}Pa}{MW_{fusion}\cdot y}P_{fusion}(MW) \quad \rightarrow \quad \frac{\$2{,}453{,}000}{MW_{fusion}\cdot y}P_{fusion}(MW)\ for\ \$1000/g\ of\ ^{231}Pa$$
$$= 1.1015\frac{kg\ ^{230}Th}{MW_{fusion}\cdot y}P_{fusion}(MW)$$
$$= 6.118\ \ \frac{kg\ ^{233}Pa}{MW_{fusion}\cdot y}P_{fusion}(MW) \quad \rightarrow \quad \frac{\$367{,}000}{MW_{fusion}\cdot y}P_{fusion}(MW)\ for\ \$60/g\ of\ ^{233}U \qquad (5)$$

The revenues then become 0.776 $/W$_{nuclear}$•y for ^{231}Pa and 0.117 $/W$_{nuclear}$•y for ^{233}U for a total of 0.892 $/W$_{nuclear}$•y. Remember, this is an ideal case with no structure and should not to be considered achievable but rather should be used for understanding ultimate limits. Even so, the ^{231}Pa production rate per nuclear power is 9.5 times that of the Li/MS case.

4. Protactinium extraction

^{231}Pa can be extracted for export to use in fission reactors. ^{233}Pa and an exceedingly small amount of ^{232}Pa (1.31 d half-life) will also be extracted depending on the hold up time before extraction to allow for decay. Resulting ^{233}U can be fluorinated out or left to carry along with the ^{231}Pa. The ratio of ^{232}Pa /^{231}Pa and ^{233}Pa /^{231}Pa is typically 0.0009 and 0.9 in a fusion breeder blanket based on Ref. 1 case Li/MS after six years of operation when ^{232}U/^{233}U=2.4%. The molten salt is removed from the reactor and after 4 half-lives of 27 days of hold up the ^{233}Pa /^{231}Pa ratio is 0.056.

For the Li/MS case, a 1000 MW$_{fusion}$ plant would produce 106 kg of ^{231}Pa per full power year. If half of this were extracted it would be 53 kg of ^{231}Pa per year; for use in fission fuel cycles. The other 53 kg of ^{231}Pa would be left in the fusion breeder. 2,220 kg of ^{233}U with 2.4% ^{232}U can be extracted yearly. From the example above with a holdup of 4 times 27 days before Pa processing, the ^{233}Pa carryover would be 0.056 x 53.1 kg =2.99 kg of ^{233}Pa per year. The ^{233}Pa decays into ^{233}U and is not self protected but is less than a bare sphere critical mass. Waiting one more half-life (to 5) would reduce the carryover to 1.49 kg of ^{233}Pa per year. The processed salt could be fluorinated one more time after a suitable further hold up time to remove ^{233}U at the plant before export.

There are alternative blanket designs that could be considered. The Be/MS blanket produces about 30% less ^{231}Pa and about the same ^{233}U per nuclear power. The all molten salt blanket produces about 5.5 time more ^{231}Pa and 3 times lee ^{233}U for the same nuclear power.

Production of ^{231}Pa from ^{230}Th

Production of ^{230}Th adds to the production of ^{231}Pa once a reaction ^{230}Th(n,γ)^{231}Th--->e- + ^{231}Pa takes place. One source of ^{230}Th is the reaction ^{232}Th(n,3n)^{230}Th. Another source

of ^{230}Th is chemical extraction from fertilizer mineral deposits[4] where ^{230}Th exists in secular equilibrium with ^{238}U.

^{238}U--->^{234}Th + ^{4}He; ^{234}Th--->2e- + ^{234}U; ^{234}U--->^{230}Th + ^{4}He.

$$
\begin{array}{ll}
^{238}\mathrm{U} \xrightarrow[4.47\times10^{9}\mathrm{y}]{} {}^{234}\mathrm{Th} + {}^{4}\mathrm{He} & {}^{218}Po \xrightarrow[3.1\,m]{} {}^{214}Pb + {}^{4}\mathrm{He} \\
{}^{234}\mathrm{Th} \underset{24.1\,d}{\rightarrow} {}^{234}Pa + \mathrm{e}^{-} & {}^{214}Pb \xrightarrow[27\,m]{} {}^{214}Bi + \mathrm{e}^{-} \\
{}^{234}Pa \underset{6.69h}{\rightarrow} {}^{234}U + \mathrm{e}^{-} & {}^{214}Bi \xrightarrow[19.9\,m]{} {}^{214}Po + \mathrm{e}^{-} \\
{}^{234}\mathrm{U} \xrightarrow[2.46\times10^{5}\mathrm{y}]{} {}^{230}\mathrm{Th} + {}^{4}\mathrm{He} & {}^{214}Po \xrightarrow[163.7\times10^{-6}\,s]{} {}^{210}Pb + {}^{4}\mathrm{He} \\
{}^{230}\mathrm{Th} \xrightarrow[7.54\times10^{4}\mathrm{y}]{} {}^{226}Ra + {}^{4}\mathrm{He} & {}^{210}Pb \xrightarrow[19.9\,m]{} {}^{210}Bi + \mathrm{e}^{-} \\
{}^{226}Ra \xrightarrow[1.6\times10^{3}\mathrm{y}]{} {}^{222}Rn + {}^{4}\mathrm{He} & {}^{210}Bi \xrightarrow[5.01\,d]{} {}^{210}Po + \mathrm{e}^{-} \\
{}^{222}Rn \xrightarrow[3.8235\,d]{} {}^{218}Po + {}^{4}\mathrm{He} & {}^{210}Po \xrightarrow[138.38\,d]{} {}^{206}Pb + {}^{4}\mathrm{He}
\end{array}
\tag{6}
$$

$$
\frac{^{230}Th}{^{238}U} = \frac{7.54\times10^{4}\mathrm{y}}{4.47\times10^{9}\mathrm{y}} = 1.69\times10^{-5} = 17\ ppm \tag{7}
$$

$$
\frac{^{234}U}{^{238}U} = \frac{2.46\times10^{5}\mathrm{y}}{4.47\times10^{9}\mathrm{y}} = 5.50\times10^{-5} = 55\ ppm \tag{8}
$$

$$
\frac{^{226}Ra}{^{238}U} = \frac{1.6\times10^{3}\mathrm{y}}{4.47\times10^{9}\mathrm{y}} = 3.58\times10^{-7} = 0.36\ ppm \tag{9}
$$

We can estimate how much ^{230}Th each 1000 MWe reactor would need to produce ^{231}Pa at the same rate it is lost by neutron capture in making ^{232}Pa in steady state.

$N_{Th230}\sigma_{Th230}\Phi = N_{Pa231}\sigma_{Pa231}\Phi; \quad \sigma_{Th230} \approx 23\ b, \sigma_{Pa231} \approx 200\ b$

$N_{Pa231} = 25.5kg\ for\ 1000MWe \quad ref.1$

$$
N_{Th230} \approx 25.5kg\frac{\sigma_{Pa231}}{\sigma_{Th230}} \approx 25.5kg\frac{200\ b}{23\ b} \approx 25.5kg\times8.7 \approx 220\ kg
$$

As a check of this 220 kg inventory of ^{230}Th, we can in steady state roughly estimate the amount of ^{232}Th needed. Assume in steady state all neutron captures in ^{232}Th result in a ^{233}U that are lost by fission and capture.

4. Bruce Hoglund, private communications note, Dec. 2012.

$$N_{Th232}\sigma_{Th232}\Phi = N_{U233}\sigma_{U233}\Phi; \quad \sigma_{Th232} \approx 7.37\ b, \sigma_{U233} \approx 531 + 46 = 577\ b$$

$$N_{U233} = 1500\ kg\ for\ 1000 MWe \quad ref.1$$

$$N_{Th232} \approx 1500\ kg \frac{\sigma_{U233}}{\sigma_{Th232}} \approx 1500\ kg \frac{577}{7.37} \approx 117{,}000\ kg$$

This is about twice the inventory of ^{233}Th in Molten Salt Breeder Reactor (MSBR) and about equal to that of Denatured Molten Salt Reactor (DMSR) whose fissile inventory is about twice as much fissile as 1500 kg of ^{233}U but explained by being denatured.

Table 1. Inventories per 1 GWe

	kg	#/^{232}Th	kg/y	#/^{232}Th
^{230}Th	220	0.0019	6.5	0.006
^{232}Th	117,000	1	1090	1
^{231}Pa	25.5	0.00022	6.5	0.006
^{232}U	36	0.00031	6.5	0.006
^{233}U	1500	0.013	1090	1

An external source of ^{230}Th is not further considered. However, if it were available in quantities of a few hundred kg per 1000 MWe, it could play a role in nonproliferation and not need a fusion breeder. Especially interesting would be a source of thorium with 0.2% ^{230}Th.

Use of ^{231}Pa in a fission reactor

The reader is referred to Ref. 1 for a discussion of this topic.

5.0 Relationship between revenues from breeder and fusion Q.

There are fusion concepts that might be able to produce ^{231}Pa soon enough for possibly anticipated Molten Salt Reactors (MSRs) base on plasma containment or performance already demonstrated or nearly so. For example, Q ($Q=P_{fusion}/P_{input}$)~1 that has already been demonstrated in neutral beam driven fusion devices[5] and this performance might be sufficient for expensive ^{231}Pa production or nearly so but not enough for commercial fusion power production. The design for the International Thermonuclear Reactor (ITER) calls for Q=10 with operation expected in 2025.[6] Still there is considerable engineering R&D needed to achieve a highly reliable, industrial fusion breeder.

5. D. Meade, "50 years of fusion research," Nuclear Fusion 50 (2010). Already in 1993 Tokamak Fusion Test Reactor (TFTR) in discharges of ~1 s produced about 11 MW of fusion power with Q≈0.3 and with similar discharges Joint European Tokamak (JET) in 1997 produced about 16 MW of fusion power with Q≈0.65.
6. ITER, http://en.wikipedia.org/wiki/ITER

In Fig. 2 we plot revenues from sale of electrical power and breeding revenues of 0.1 $/W$_{nuclear}$•y.[7,8] Our example blankets in Section 3 have breeding revenues estimated at up to 0.182 $/W$_{nuclear}$•y for the Li/MS case, 0.169 $/W$_{nuclear}$•y for the Be/MS case and 0.484 $/W$_{nuclear}$•y for the all MS case based on 60 $/g of ^{233}U with 2.4% ^{232}U and 1000 $/g of ^{231}Pa.

Assume commercial market or commercial competitiveness is achieved at some specific revenues in units of 0.1 $/W$_{nuclear}$•y. Then the breeding design is equally competitive at a lower value of Q depending on the value of the bred products. Another conclusion is that for high enough Q the bred products increase the revenues over that from electrical power by a significant amount depending on the value of the bred products.

The breeding mission enables fusion technology to be commercial at an earlier stage of development or performance measured for example by Q, because at any given stage of development breeding adds revenues hence more return on investment.

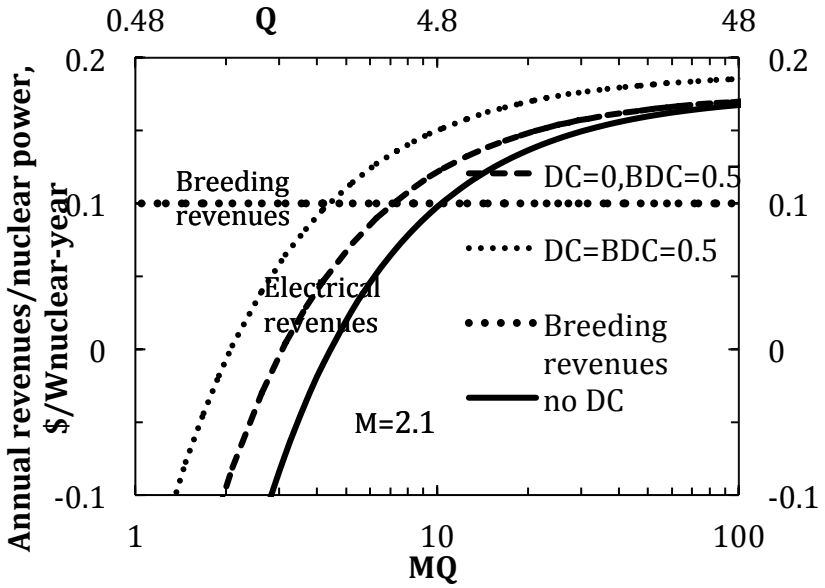

Fig. 2. Revenues for producing electrical power and for various direct conversion assumptions and for one specific breeding rate in $/W$_{nuclear}$•year.

The lower curve has no direct conversion. The next curve includes direct conversion at 50% efficiency for unneutralized beams (labeled BDC) and the top curve also includes direct conversion at 50% for end leakage plasma (labeled DC). M is the blanket energy released divided by 14.1 MeV.

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7. The assumptions and calculations in this section and the next are described more fully in a draft note in preparation (2014).
8. R. W. Moir, N. N. Martovetsky, A. W. Molvik, D. D. Ryutov, T. C. Simonen, "Mirror-based hybrids of recent design," FUNFI, Workshop on Fusion for Neutrons and Sub-critical Nuclear Fission, Villa Monastero, Varenna, Italy, September 12-15, 2011, AIP Conference Proceedings1442, 43-54 (2012). http://www.ralphmoir.com/media/varenna2011_2R.pdf

In Fig. 3 we plot the sum of the electrical revenues and various fuel breeding revenues.

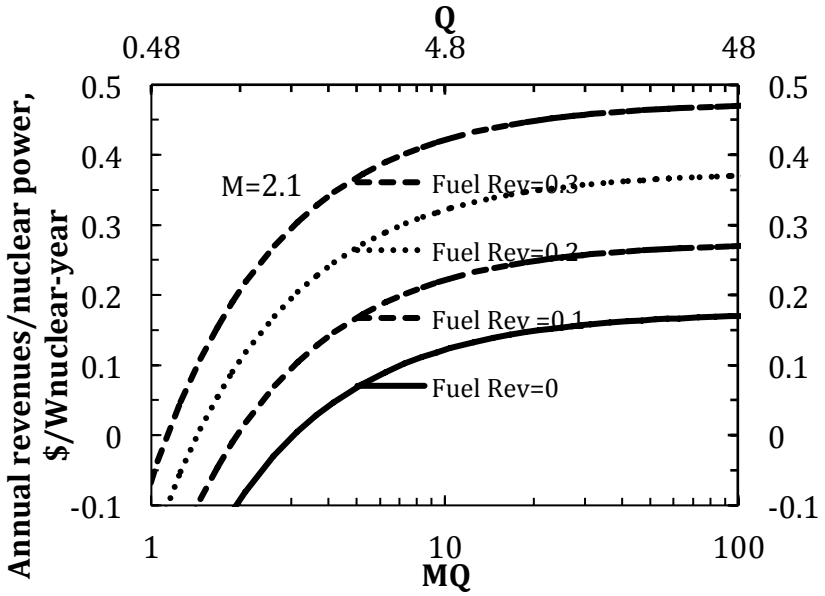

Fig. 3. Revenues for producing electrical power and for various breeding rates giving a range of fuel revenues in $/W$_{nuclear}$•year. Direct conversion at 50% efficiency is assumed for unneutralized beams but no direct conversion for end leakage plasma.

6. Cost of isotopes from the fusion breeder

The cost of isotopes or various materials produced can be calculated with a number of assumptions. Assume we purchase electrical power at 50 $/MWe•h at low Q and when net electrical power is produced we sell at the same price. We assume the capital cost is a parameter ranging from 1.25 $ to 2 $/W$_{nuclear}$ independent of Q. The fissile production F=0.5 appropriate to the Li/MS case discussed earlier. Annual capital charges are 0.1 times capital cost and annual operations and maintenance costs are 0.03 times capital costs. The results are shown in Fig. 4 and 5.

At Q <<1 the cost of fissile is dominated by electrical power cost proportional to 1/Q. At Q>>10 the cost of fissile is independent of Q and is an increasing function of capital cost. Direct conversion of leaking plasma significantly lowers the cost of fissile fuel. Direct conversion of leakage plasma[9] might be feasible for some fusion concepts such as the open-end systems, an example of which is the Gas Dynamic Trap (GDT)[10] but unlikely to be feasible for a closed field line system such as a tokamak.

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9. R. W. Moir, and W. L. Barr, "Venetian-Blind Direct Energy Converter for Fusion Reactors" *Nuclear Fusion*, **13**, 35-45 (1973). http://www.ralphmoir.com/wp-content/uploads/2012/10/venBlnd.pdf, other direct conversion references. http://www.askmar.com/Direct_Energy.html

10. A A Ivanov and V V Prikhodko, "Gas-dynamic trap: an overview of the concept and experimental results," Plasma Phys. Control Fusion 55 (2013).

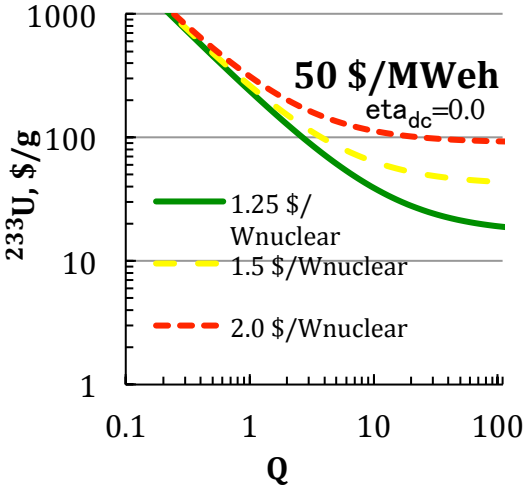

Fig. 4. Calculated cost of fissile fuel, $/g of ^{233}U with no direct conversion of leaking plasma.

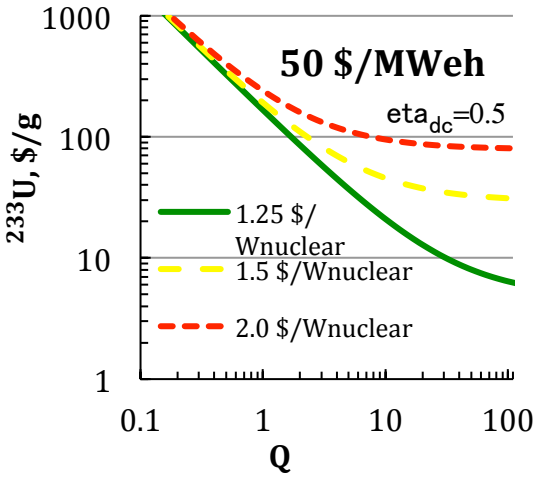

Fig. 5. Calculated cost of fissile fuel with 50% efficient direct conversion of leaking plasma.

The cost of ^{233}U with 2.4% ^{232}U is shown in Fig. 4 & 5 not taking into account the value of ^{231}Pa. When we assign a value to the sales of ^{231}Pa in $/g we need to have the sum of the revenues from both ^{233}U and ^{231}Pa add up to the same amount as in Fig. 4 & 5. The revenues are proportional to the production rates and the value in $/g of each isotope:

$$F_{n,\gamma} \cdot {}^{233}U\$/g + F_{n,2n} \cdot {}^{231}Pa\$/g = F_{n,\gamma} \cdot {}^{233}U\$/g(no\ Pa)$$

$$0.5 \cdot {}^{233}U\$/g + 0.0246 \cdot {}^{231}Pa\$/g = 0.5 \cdot {}^{233}U\$/g(no\ Pa)$$

$${}^{233}U\$/g + 0.0492 \cdot {}^{231}Pa\$/g = {}^{233}U\$/g(no\ Pa)$$

The results are plotted in Fig. 6. To illustrate how to use the results of Fig. 6 consider an example. Suppose in Fig 4 ^{233}U only is sold at 100 $/g at Q=2.5 from the lower curve. Using 100 $/g curve (middle one) in Fig. 6 we see that if we can sell ^{231}Pa at 1000 $/g then the price of ^{233}U need only be 50 $/g to cover expenses.

P. A. Bagryansky, Yu. V. Kovalenko, V. Ya. Savkin, A. L. Solomakhin, D. V. Yakovlev, “First results of auxiliary electron cyclotron resonance heating experiment in GDT magnetic mirror,” to be published (2014).

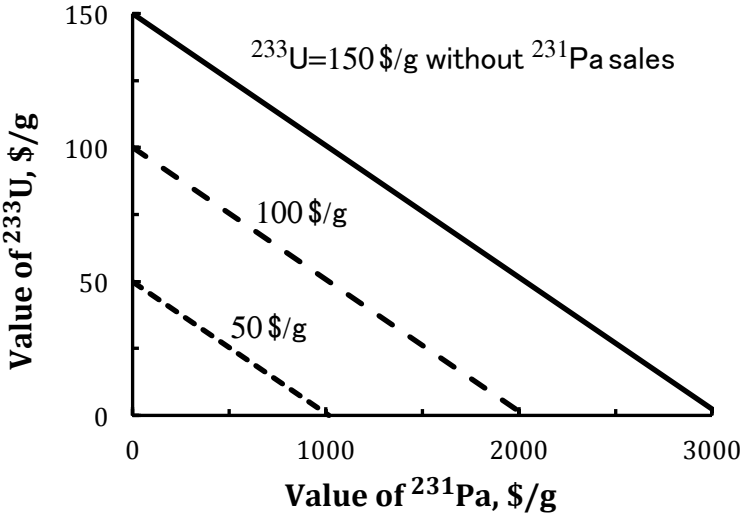

Fig. 6. The value of ^{233}U from Fig. 5 & 6 can be adjusted for the ^{231}Pa sales.

$$^{233}U\$/g = {}^{233}U\$/g(no\ Pa) - 0.0492 \cdot {}^{231}Pa\$/g = 100\ \$/g - 1000\ \$/g = 50\ \$/g$$

Fig. 4, 5, 6 give examples of computed cost of isotopes from the fusion breeder. We treat three isotopes (^{233}U, ^{232}U, ^{231}Pa) and electricity revenues.

7. Protactinium processing rate

The purpose of the fusion breeder is to produce ^{233}U for use in fission reactors with ^{232}U/^{233}U=2.4% for "self-protection." The uranium would be extracted by fluorination. A side stream would be held up for a time[11] to allow ^{233}Pa to substantially decay to ^{233}U. After this decay time this molten salt would be processed to remove protactinium that is mostly ^{231}Pa that is an appropriate product for export for use in reactors to in situ "self-protect" bred ^{233}U.

We now set about calculating the protactinium processing rate needed to cap the ^{232}U/^{233}U=2.4% so the excess ^{231}Pa can be extracted for export. From Ref. 1 p 6 we get:

$$C_{P1} = \frac{^{231}Pa\,atoms}{^{232}Th\,atoms}; \quad \frac{dC_{P1}}{dt} = R'_{Th}KC_{Th} - R'_{P1}KC_{P1} + R''_{Th230}KC_{Th230} - \lambda_{P1}C_{P1} - F^{Pa}\eta_{Pa}C_{P1}$$

11. There is a logic problem that needs clearing up. Processing should not begin until the ^{232}U/^{233}U ratio builds up to 2.4%. For the Li/MS case of Ref. 3, Fig. 9c with no processing, ^{232}U/^{233}U=2.4% after about 4.5 y when the ^{233}U/^{232}Th=0.012 and substantial fissioning is occurring. From fig. 9a with 14.4 m^3/d processing, ^{233}U/^{232}Th=0.001 after 0.5 y but ^{232}U/^{233}U=0.2% only. The precursor to ^{232}U, ^{231}Pa has not built up enough until about 6 y. This operational logic problem needs further study. The Be/MS blanket builds up to ^{232}U/^{233}U=2.4% in 1.5 years greatly reducing this problem.

with terms defined in Ref. 3.

Neglecting decay and ^{230}Th and setting $\frac{dC_{P1}}{dt} = 0$ after a time at which the

^{232}U/^{233}U=2.4% which is about 6 y for the Li/MS case and 1.5 y for the Be/MS case, the protactinium process rate is:

$$F^{Pa}\eta_{Pa} = R'_{Th}KC_{Th} / C_{P1} - R'_{P1}K$$

Li/MS blanket ($F_{n,\gamma}$ = 0.515)

The Li/MS blanket consists of 0.5 m zone of liquid lithium followed by 0.5 m of molten salt. The Li/MS blanket has a production rate of ^{233}U proportional to Fn, γ = 0.515. The fusion power was 3000 MW and the nuclear power was 3888 MW.

$$F^{Pa}\eta_{Pa} = R'_{Th}KC_{Th} / C_{P1} - R'_{P1}K = 0.0246 \times 2.76 \times 10^{-10} / 1.0 \times 10^{-3} - 12.91 \times 2.76 \times 10^{-10}$$
$$= 6.789 \times 10^{-9} - 3.563 \times 10^{-9} = 3.22 \times 10^{-9} / s$$

$V_{inside}+V_{outside}$=1152 m^3 The Li/MS blanket has a large volume of salt that would be desireable to reduce for cost reasons, possibly by adding graphite in the molten salt region.

$$F^{Pa}\eta_{Pa} \cdot V = 1152\ m^3 \times 3.22 \times 10^{-9} / s = 3.709 \times 10^{-6}\ m^3 / s = 0.32\ m^3 / d$$

For a 4 x 27 d holdup time, the held up volume of salt in the process system is:

$V_{holdup} = 0.32\ m^3 / d \times 108\ d = 34.6\ m^3$ That is only 3% of the salt inventory.

Be/MS blanket

The Be/MS blanket consists of 0.5 m zone of 10 mm dia Be soheres with 17 mm steel tubes among the pebbles carrying molten salt. The Be/MS blanket has a production rate of ^{233}U proportional to Fn, γ = 0.78. The fusion power was 3000 MW and the nuclear power was 5640 MW.

$$F^{Pa}\eta_{Pa} = R'_{Th}KC_{Th} / C_{P1} - R'_{P1}K = 0.0267 \times 11.4 \times 10^{-10} / 7.0 \times 10^{-4} - 23.64 \times 11.4 \times 10^{-10}$$
$$= 4.35 \times 10^{-8} - 2.695 \times 10^{-8} = 1.65 \times 10^{-8} / s$$

$V_{inside}+V_{outside}$ =170 m^3

$$F^{Pa}\eta_{Pa}\cdot V = 170\ m^3 \times 1.65\times 10^{-8}\ /\ s = 2.805\times 10^{-6}\ m^3\ /\ s = 0.242 m^3\ /\ d$$

$$V_{holdup} = 0.242\ m^3\ /\ d \times 108\ d = 26.1\ m^3$$ That is 15% of the salt inventory.

All MS blanket

The All MS blanket has a production rate of ^{233}U proportional to Fn, $\gamma = 0.24$. The fusion power was 500 MW and the nuclear power was 900 MW.

$$F^{Pa}\eta_{Pa} = R_{Th}^{'}KC_{Th}\ /\ C_{P1} - R_{P1}^{'}K = 0.188\times 3.4\times 10^{-10}\ /\ 2\times 10^{-2} - 1.45\times 3.4\times 10^{-10}$$
$$= 3.196\times 10^{-9} - 4.93\times 10^{-10} = 2.703\times 10^{-9}\ /\ s$$

The all MS with C reflector and ^{231}Pa/Th=0.02 and ^{232}U/U=0.09 corresponding to 10 years on Fig. 5 of Ref. 3, where the uranium processing rate was 2.5 m^3/d (the figure erroneously labeled it m^3/s).

$V_{inside}+V_{outside}$ =95.3 m^3

$$F^{Pa}\eta_{Pa}\cdot V = 95.3\ m^3 \times 2.703\times 10^{-9}\ /\ s = 2.576\times 10^{-7}\ m^3\ /\ s = 0.022256\ m^3\ /\ d$$

$$V_{holdup} = 0.0223\ m^3\ /\ d \times 108\ d = 2.40\ m^3$$ That is 2.5% of the salt inventory.

In Fig 5 of Ref. 3, ^{231}Pa is still rising with time at 10 years and is 0.02 of Th. By processing Pa we remove ^{231}Pa and lower the ratio of ^{232}U/^{233}U from 0.09 to 0.024.

^{231}Pa/Th=0.006 and ^{232}U/^{233}U =0.024 corresponding to 3 years on Fig. 5 of ref. 1.

$$F^{Pa}\eta_{Pa} = R_{Th}^{'}KC_{Th}\ /\ C_{P1} - R_{P1}^{'}K = 0.188\times 3.4\times 10^{-10}\ /\ 6\times 10^{-3} - 1.45\times 3.4\times 10^{-10}$$
$$= 1.065\times 10^{-8} - 4.93\times 10^{-10} = 1.016\times 10^{-8}\ /\ s$$

$$F^{Pa}\eta_{Pa}\cdot V = 95.3\ m^3 \times 1.016\times 10^{-8}\ /\ s = 9.68\times 10^{-7}\ m^3\ /\ s = 0.0836\ m^3\ /\ d$$

$$V_{holdup} = 0.0836\ m^3\ /\ d \times 108\ d = 9.03\ m^3$$ That is 9.5% of the salt inventory.

The processing rate of Pa is 30 times less than the processing rate of U.

Example Li/MS case at 6 years of operation

The 4 x 27 day holdup of volume of salt of 34.6 m^3 contains the following inventories: $LiF+BeF_2+ThF_4+PaF_4+UF_4$

Table 2. Salt inventories in the held up volume, Li/MS at 6 y.

	Mol%	mol ratio to ^{232}Th	Kg metal/m^3	At time of extraction Kg HM in 34.56 m^3	After 108 days holdup Kg HM
^{6}LiF	0.662	5.5E-02	2.199E+00	7.610E+01	7.610E+01
^{7}LiF	71.338	5.9E+00	2.765E+02	9.567E+03	9.567E+03
BeF_2	16	1.3E+00	7.974E+01	2.759E+03	2.759E+03
$^{228}ThF_4$	4.8E-06	4.0E-07	5.071E-04	1.753E-02	1.753E-02
$^{230}ThF_4$	1.2E-03	1.0E-04	1.279E-01	4.420E+00	4.420E+00
$^{232}ThF_4$	1.2E+01	1.0E+00	1.290E+03	4.458E+04	4.458E+04
$^{231}PaF_4$	1.2E-02	1.0E-03	1.284E+00	4.439E+01	4.439E+01
$^{232}PaF_4$	6.0E-06	5.0E-07	6.450E-04	2.229E-02	0
$^{233}PaF_4$	4.8E-03	4.0E-04	5.182E-01	1.791E+01	1.119E+00
$^{232}UF_4$	2.9E-04	2.4E-05	3.096E-02	1.070E+00	1.092E+00
$^{233}UF_4$	1.2E-02	1.0E-03	1.296E+00	4.477E+01	6.157E+01
$^{234}UF_4$	1.2E-04	1.0E-05	1.301E-02	4.497E-01	4.497E-01
$^{235}UF_4$	2.0E-06	1.7E-07	2.221E-04	7.677E-03	7.677E-03
Total	100.03		1.652E+03	5.708E+04	5.708E+04

*The numbers in this table are scaled from 1290 kg/m^3 for ^{232}Th from Ref. 3 that appears to be in error and inconsistent with density of 3350 kg/m^3 but are used here to be consistent with Ref. 3. In the future this error needs to be corrected in Ref. 3. The heavy metal densities will go up and the process rates will go down with little predicted change in breeding rates.

8. Protactinium processing

Mechanically separated

This product could be exported as is for use in a fission reactor with the assays given in Table 1 assuming the fissile concentration is sufficient. If not the molten salt could be irradiated longer before removal to allow buildup to a higher concentration. Actually no separation is involved, just physically transferring from fusion breeder to the MSR.

Fluorination

If we want to export fissile uranium, the blanket molten salt can be sent to a fluorination process to remove uranium as shown in Figs. 7 & 8 to remove for export 6810 kg/y of

^{233}U for use in fission reactors. The extracted uranium would be in the form of UF_6 that would be reduced back to UF_4 in the presence of the solvent suitable for use in the MSR such as $LiF+BeF_2$ or $NaF+BeF_2$.

Protactinium separation by fluorination followed by reductive extraction

If we want to export ^{231}Pa for use as a nonproliferant additive to a thorium fuel cycle reactor, then after fluorination, the reductive extraction process can be used on the 34.6 m^3 containing 17.91 kg of ^{233}Pa, 0.022 kg of ^{232}Pa and 44.39 kg of ^{231}Pa as shown in Table 2. The problem with this plan is the ^{233}Pa decays into 17.91 kg of ^{233}U with only 0.022 kg of ^{232}U and is a proliferation concern.

To solve this problem we add a holdup tank to Fig. 7 as shown in Fig. 8 to allow the ^{233}Pa to decay to ^{233}U in the presence of considerable ^{232}U. We have to modify the flow sheet by adding a holdup tank (4x27 = 108 days of hold up or 34.56 m^3) before the second fluorinator. Carry over of thorium is acceptable to some extent. The extracted Pa will grow ^{233}U up to 1.12 kg depending on time. This is a proliferation issue. We could diminish the amount of ^{233}U by holding up longer than four half lives.

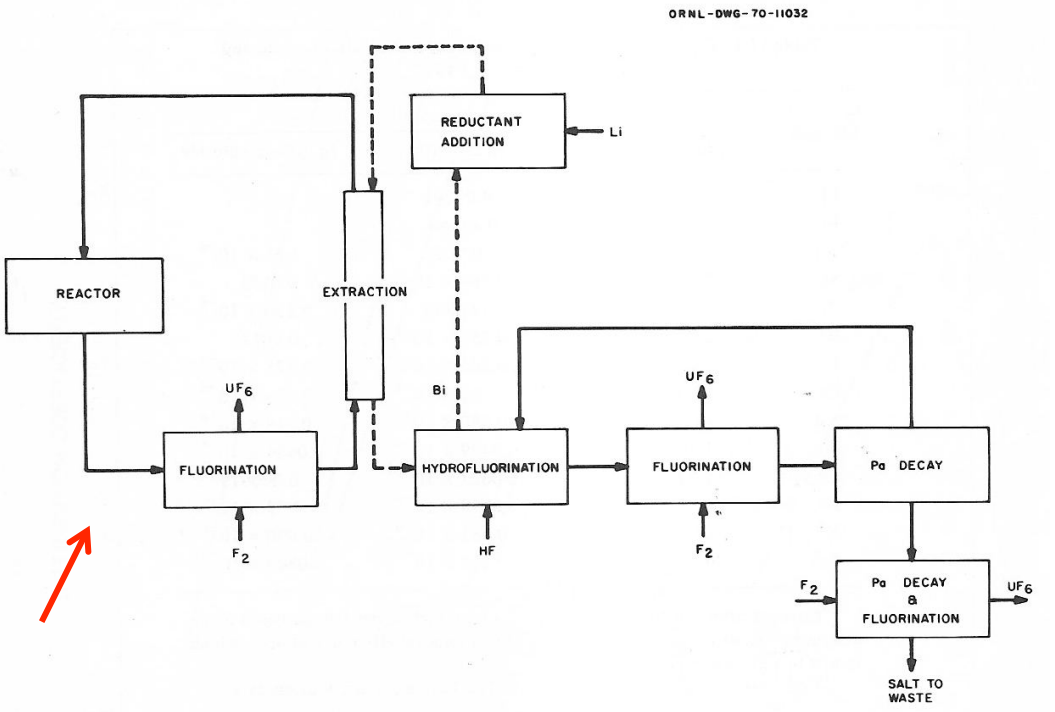

Fig. 11.4. Flowsheet for isolation of protactinium from a single-fluid MSBR by fluorination-reductive extraction.

Fig. 7. Protactinium separation by fluorination-reductive extraction.[12] A holdup tank of many times 27 day half-lives is added where the red arrow shows—see Fig. 8.

12. M. W. Rosenthal, P. N. Haubenreich, R. B. Briggs, "The development status of molten-salt breeder reactors (MSBR)," ORNL-4812 (1970), p338, Fig. 11.4.

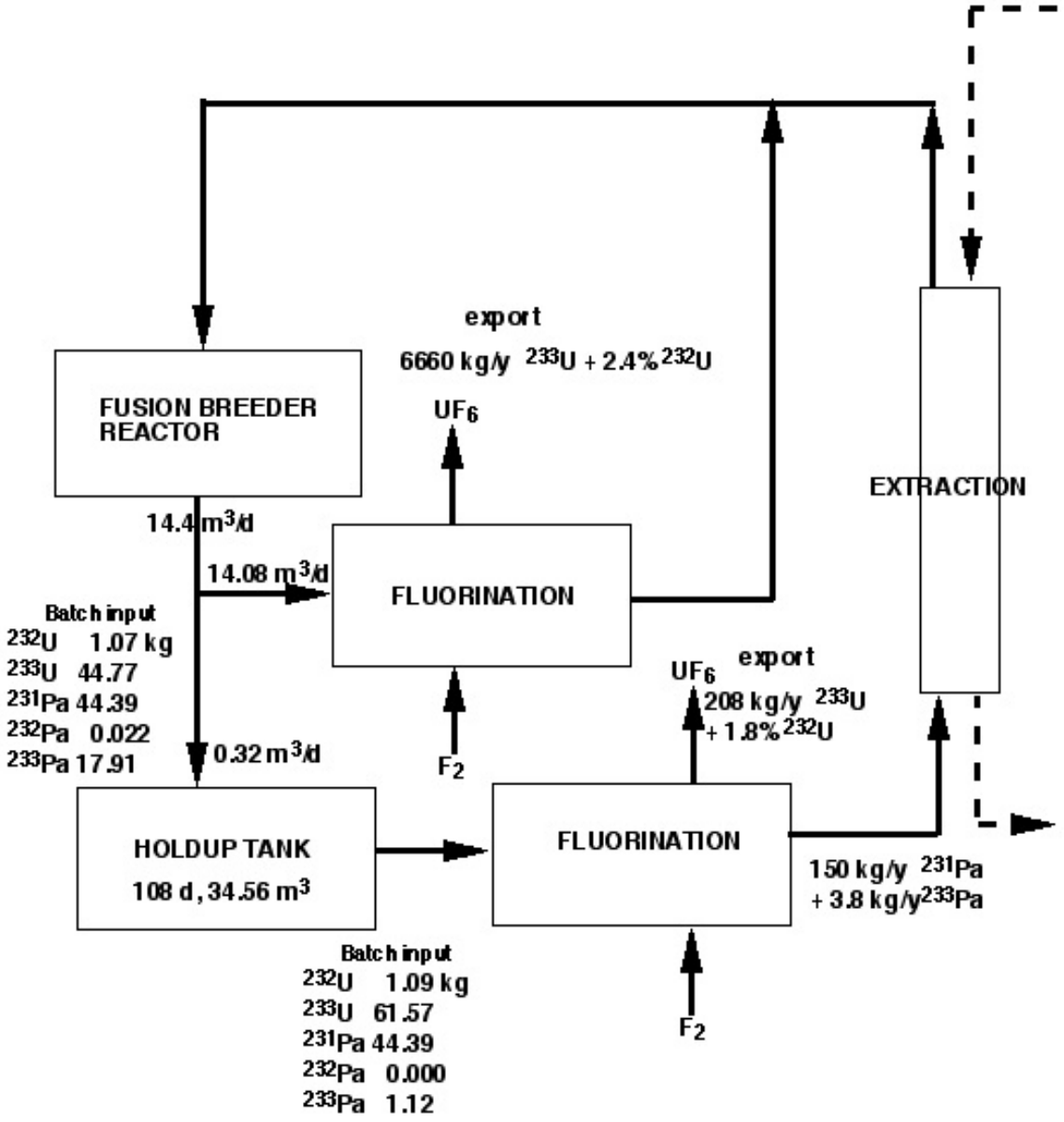

Fig. 8. The holding tank of Fig. 8 is shown added. 6810 kg/y of ^{233}U with 2.4% ^{232}U and 150 kg/y of ^{231}Pa can be exported for use in fission reactors.

The 3.8 kg/y of ^{233}Pa going into the extraction system is of concern because this will decay into ^{233}U without any ^{232}U. Perhaps the holdup time in the decay tank should be extended to reduce the carryover of ^{233}Pa. Two more months of holdup would reduce the 3.8 kg to about 1 kg/y of ^{233}Pa that will decay into 1 kg/y of ^{233}U.

Modifications to this flow sheet could result in production of significant quantities of ^{233}U with little ^{233}U and would be of considerable proliferation concern. This possibility would likely require a well-guarded facility. The amount of ^{233}U production in kg/y and

its accompanying ^{232}U level, the extent of modifications and time delay (warning time) are straight forward extensions of this work but left to future work.

9. Proliferation assessment of fusion breeder

The example discussed based on the Li/MS blanket and shown in Fig. 8 material flow sheet is based on 3000 MW_{fusion}. This is probably three times larger than needed, however, it will be used for the example. For simplicity let us assume the fluorination and reductive extraction steps are 100% effective. With normal operations, nonproliferation conditions are met:

2.22 kg/MW_{fus}•y	6660 kg/y ^{233}U with 2.4% ^{232}U	conditions met
0.069 kg/MW_{fus}•y	208 kg/y ^{233}U with 1.8% ^{232}U	small short fall-to be delt with
0.0013 kg/MW_{fus}•y	3.8 kg/y ^{233}U with 0% ^{232}U	weapons grade but small amount

The large production of ^{232}U comes mostly from ^{231}Pa, which starts out at zero concentration in the fusion breeder. At the beginning of life the $^{232}U/^{233}U$ ratio is 6×10^{-4} and builds up linearly to 0.024 after 6 years. The facility will have to be sufficiently safeguarded for this 6 years start up period.

Assume for the time being the above normal operations are acceptable from a nonproliferation point of view and that safeguards are sufficient to maintain such operations. By modifying the operations thereby violating safeguards ^{233}U can be made with less than the "self-protected" 2.4% ^{232}U. Should any operation change, high alert would be sounded. **Future study** is needed to examine quantitatively a number of modifications that could be made. Such a facility would probably be located in a country that already has nuclear weapons.

10. Conclusions

In the example case of Li/MS we discuss a fusion breeder that can export 6660 kg/y of ^{233}U with 2.4% ^{232}U and in a separate stream 206 kg/y of ^{233}U with 1.8% ^{232}U and in a third stream 150 kg/y of ^{231}Pa and 3.8 kg/y of ^{233}Pa for use in fission reactors to improve their nonproliferation features. One fusion breeder can supply 6.4 kg/y ^{231}Pa to keep 15 molten salt reactors self-protected of equal nuclear power. The fusion power for this case was 3000 MW and nuclear power was 3890 MW. Other fusion breeder designs increase ^{233}U production at the expense of ^{231}Pa production and vice versa. ^{230}Th can also be obtained from mineral deposits.