

Recommendations for a restart of molten salt reactor development

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Abstract

The concept of the molten salt reactor (MSR) refuses to go away. The Generation-IV process lists the MSR as one of the six concepts to be considered for

extending fuel resources. Good fuel utilization and good economics are required to meet the often-cited goal of 10 TWe globally and 1 TWe for the US by non-carbon energy sources in this century by nuclear fission. Strong incentives for the molten salt reactor design are its good fuel utilization, good economics, amazing fuel flexibility and promised large benefits. It can:

- use thorium or uranium;
- be designed with lots of graphite to have a fairly thermal neutron spectrum or without graphite moderator to have an epithermal neutron spectrum;
- fission uranium isotopes and plutonium isotopes;
- produces less long-lived wastes than today's reactors by a factor of ten to one hundred;
- operate with non-weapon grade fissile fuel, or in suitable sites it can operate with enrichment between reactor-grade and weapon-grade fissile fuel;
- be a breeder or near breeder;
- operate at temperature >1100 °C if carbon composites are successfully developed.

Enhancing ^{232}U content in the uranium to over 500 ppm makes the fuel undesirable for weapons, but it should not detract from its economic use in liquid fuel reactors: a big advantage in nonproliferation.

Economics of the MSR are enhanced by operating at low pressure and high temperature and may even lead to the preferred route to hydrogen production. The cost of the electricity produced from low enriched fuel averaged over the life of the entire process, has been predicted to be about 10% lower than that from LWRs, and 20% lower for high-enriched fuel, with uncertainties of about 10%. The development cost has been estimated at about 1 B\$ (e.g., a 100 M\$/y base program for ten years) not including construction of a series of reactors leading up to the deployment of multiple commercial units at an assumed cost of 9 B\$ (450 M\$/y over 20 years). A benefit of liquid fuel is that smaller power reactors can faithfully test features of larger reactors, thereby reducing the number of steps to commercial deployment. Assuming electricity is worth \$50 per MWe•h, then 50 years of 10 TWe power level would be worth 200 trillion dollars. If the MSR could be developed and proven for 10 B\$ and would save 10% over its alternative, the total savings over 50 years would be 20 trillion dollars: a good return on investment even considering discounted future savings.

The incentives for the molten salt reactor are so strong and its relevance to our energy policy and national security are so compelling that one asks, "Why has the reactor not already been developed?"

Keywords: molten-salt reactor, thorium, nonproliferation, economics, startup fuel, deployment scenario

1. Introduction

The molten salt reactor (MSR) is well described in the paper by Furukawa et al.[1]. Undergrounding, interim onsite storage of wastes, proliferation and other aspects are described in Ref. [2]. The development needs for the molten salt reactor have been described by Forsberg [3]. An earlier design by Engel et al. of a Denatured Molten-Salt Reactor (DMSR) without processing for its 30-year graphite lifetime needed only 150 kg ^{235}U per year at 20% enrichment of makeup fuel[4]. Another design by Engel et al. also using 20% enriched fuel but with processing, needed no refueling for its 30-year life [5]. Processing of molten salt fuel is discussed in Ref.[6]. This paper elaborates on how the MSR can burn thorium while producing all or most of its fuel at a lower cost than today's LWRs. An example shows why its late deployment start means the MSR can have little market penetration (<100 GWe) by 2050 but dominance to 10 TWe by 2100.

The role of ^{233}U in the context of nonproliferation is discussed as it relates to startup fueling with mined and enriched ^{235}U , and with ^{233}U from accelerators or from fusion. The Global Nuclear Energy Partnership (GNEP) model is adopted for our thorium reactors where proliferation-vulnerable facilities are in guarded centers and reactors can be located outside these centers with proliferation resistant fuels. The difference from GNEP is its use of plutonium whereas we use thorium and potentially have better economics. The MSR's advantages are so many and large that we wonder why the reactor has not already been developed and put into service. Perhaps low cost, plentiful uranium and natural gas was the reason but costs for these commodities are expected to continue to rise. A reexamination and restart of an MSR development program is recommended. This paper presents my recommendations for a restart of the MSR development based on research during the past 30 years.

2. Economic motivation and predictions

Oak Ridge National Laboratory (ORNL), known for its conservatism, made capital and operating cost estimates that were based on detailed conceptual studies of various MSRs. The results were given in laboratory reports but never published in the open literature. Maybe this was due to the controversy surrounding the demise of the MSR program in the US at the time of the decision to concentrate on the Liquid Metal Fast Breeder Reactor (LMFBR) and due to the modesty with which ORNL people treated future economic projections of this infant technology. In any case the present author recently published a brief summary of the cost of electricity (COE) projections based on ORNL reports for 20% enriched uranium compared to contemporary PWRs and coal of 1978 vintage and then converted to year 2000 dollars [7]. The reactor was 1000 MWe and used reactor grade uranium (20% enriched) as fuel and did not reprocess the fuel for its 30-year life that was taken to be the damage lifetime of the graphite. The 100% enriched case is given in Ref. 8. The results are given in Table 1 with the assumptions in the references. The COE of the MSR (38 \$/MWe•h) being 7%

lower than the PWR and 8% lower than coal is a hopeful result even considering the uncertainties were judged by the ORNL people to be approximately $\pm 10\%$.

Table 1
Economic comparisons of cost of electricity
in units of \$/MWh, 2000\$

	MSR, 20% enriched	MSR, 100% enriched	PWR	Coal
Capital	20.1	20.1	20.7	15.8
O&M	5.8	5.8	11.3	8.0
Fuel	11.1	4.0	7.4	17.2
Waste disposal	1.0	1.0	1.0	0.9
Decom	0.4	0.4	0.7	--
Total	38.4	31.3	41.1	41.9

The 20% enriched case requires about 150 kg/y of ^{235}U makeup per year for 1 GWe. At 90% capacity factor used in Table 1 for the MSR cases and costing 60 \$/g of ^{235}U this makeup fuel would cost \$1.1/MWe•h. To get the total fuel cost in Table 1, one adds initial fuel charge \$2.7/MWe•h, enriched Li and other costs. The \$11/MWe•h (\$4.6/MWe•h in 1978 dollars) fuel cost for the 20% enriched case may be too high today owing partly to commodity prices and enrichment costs that have not increased nearly as much as the 2.4 escalation from 1978 to 2000 dollars that was assumed.

If we operate without much ^{238}U the reactor produces just enough fuel to make its own ^{233}U from ^{232}Th and the predicted COE without processing is \$31/MWe•h, which is 24% lower than for a PWR of 1978 vintage. There will be extra cost associated with extra security measures resulting from use of highly enriched uranium as its fuel. This extra cost might be largely offset owing to the large amount of ^{232}U content (hundreds of ppm) as will be discussed later.

If the MSR does online processing then fuel self sufficiency after startup can be achieved at some yet to be determined cost and at an enrichment varying all the way down to non-weapon grade ($<20\%$ ^{235}U or $<12\%$ ^{233}U). There are trade-offs between economics, processing and proliferation resistance. Likely, investment in safeguarding will be more than offset by the savings in COE. The MSR discussed here is argued to be 10-20% lower in cost of energy than an LWR but the proper comparison would be a fuel self-sufficient system envision by GNEP type reactors system. However, past examples of fast reactors on the plutonium cycle have had projected economics several tens of percents more than the LWR. I recommend more analyses on the economics of MSRs. If the trends shown in Table 1 turn out to be correct then the argument in favor the MSR will be even more compelling.

3. Development arguments

When the MSR development program was shut down in the 1970s about 1 B\$ of development was judged to be required on materials, on components, on processes, on tritium control techniques and other items. By now some of this development has already been done. This program might consume \$100 M/y and take ten years. Added to this would be a series of reactors leading up to the deployment of commercial units. The first reactor might be an electricity producing version of ~10 MWe modeled after the successful MSRE that operated at ORNL at 8 MWth. The fluid-fuel nature of the reactor makes such a test reactor relevant to much higher power reactors because the local conditions are almost indistinguishable from that of a much higher power reactor just so long as the power density is the same. The next reactor might be a demonstration of a future commercial reactor operating at a hundred or a few hundred MWe. These two steps might cost \$9 B (\$450M/y for 20 y). In 30 years such a program on a non-crash basis could start deploying commercial units if the demand were to emerge. It is clear that this new technology (actually quite old but not fully tested) cannot make significant market penetration (>10%) by 2050. One wonders why some well-known studies [9] concentrated on a time or goal of 2050. Were they designed to forestall development of Generation-IV options that could have a large impact by 2100 but not by 2050? The deployment rate can be significant without violating the norms of growth and investment rates in new industries.

I have constructed a deployment scenario for the sake of discussion rather than as a prediction. This scenario follows and modifies an earlier deployment scenario by Furukawa [1]. The assumptions are incremental deployment shown in Fig. 1 followed by a growth limit then slowing down and finally after 2100 I arbitrarily show a steady state or a modest growth where the MSR system on thorium can operate for thousands of years. I also show the possibility of a transition to some new technology, maybe fusion or solar power.

The assumptions in the deployment scenario are: start-up a 10 MWe unit in 2012, five years later in 2017 a 100 MWe unit begins operation, in 2023 I assume a 500 MWe unit operates, a 1 GWe unit or two 500 MWe units in 2027, in 2029 and in 2031. The number of new units per year and the growth rate are shown in Fig. 1. This scenario is aggressive but not a crash program in its first six plants. From then on market forces are assumed to demand a large growth rate. As the industry grows in size its growth rate is assumed to drop down to the robust rate of 10% per year from about 2050 to 2080. This hypothetical deployment to 10 TWe in 100 years scenario assumes only molten salt reactors are deployed, however, it could have assumed any high conversion ratio reactors such as the liquid metal fast reactors or high temperature gas cooled graphite reactors or a combination.

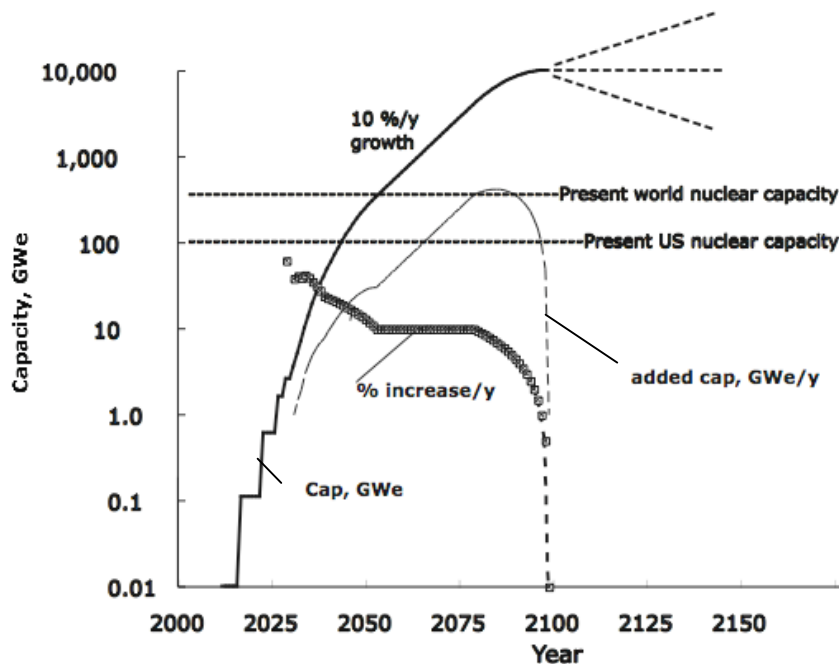


Fig. 1. Hypothetical worldwide deployment scenario of new MSRs illustrates only a doubling of nuclear power by 2050 but twenty five-fold increase by 2100.

Will there be enough uranium to start-up 10 TWe (10,000 MWe) of new capacity based on self-sustaining MSRs? At 3 tons of ^{235}U per 1 GWe, 10 TWe of new capacity will require 30,000 tons of uranium-235. If 0.5% of uranium can be separated as ^{235}U , then 6 million tons of uranium would be needed for start-up fuel. The world's resources are considerably more than this so startup on mined uranium looks feasible. If the specific start-up inventory is different than 3 kg/MWe the results can be scaled. An important incentive for thermal reactors is their low fissile inventory that can be an order of magnitude lower than that of fast reactors (see discussion later). However, proliferation concerns over use of highly enriched uranium suggest strategies to fully use the proliferation resistance coming from use ^{232}U and ^{238}U as will be discussed in the next section.

There are some major conclusions to be drawn from this hypothetical deployment exercise:

- 1-10 TWe are possible by the end of the century with a plausible growth rate, even more with an aggressive deployment
- 2-However, by 2050 only a few hundred GWe seem possible for this new or for practically any new fission technology, which is about equal to the present world nuclear capacity.
- 3-A surprisingly fast growth rate seems possible early on due to the small size of relevant steps; 10 MWe and 100 MWe steps are relevant to larger commercial units.
- 4-These deployment limits apply to many new nuclear technologies.

5- The number of Light Water Reactor (LWR) can grow some but quickly is limited by fuel limitations and waste management.

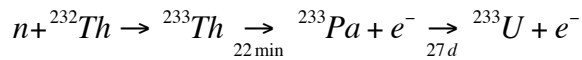
Closing the fuel cycle with new reactor types will require long lead times so early work on prototypes is recommended and should not be postponed. The LWR system lend itself to evolve to a system with good characteristics such as the MSR system for the tens of TWe deployment level if we process the spent fuel and use some of it as startup fuel for MSRs. The LWR's low conversion ratio and expensive processing and fuel fabrication necessary for recycle appears to present a barrier to increasing their number by more than a factor of two from present whereas we need several tens of times more power than the present LWR "fleet"

4. Nonproliferation arguments and the role of ^{232}U

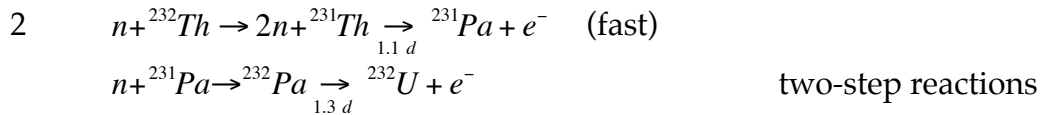
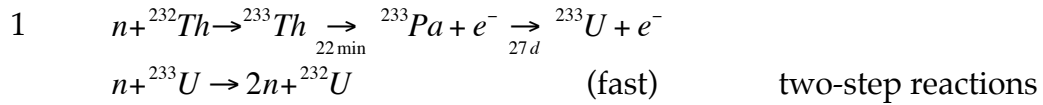
Kang and von Hippel discuss proliferation resistance emphasizing the role of ^{232}U in the thorium fuel cycle [10]. Suppose a reactor that started up on ^{233}U enriched to 50% or over but that could operate self-sufficiently from then, on was acceptable from a nonproliferation point of view. Possibly this material would be valuable enough that the economics would put a premium on such material with sufficiently high ^{232}U content such as >1000 appm.

The production of ^{232}U produced while making ^{233}U has a 2.6 MeV gamma radiation daughter that might makes this fissile material sufficiently resistant to nuclear weapons proliferation that a fuel cycle using thorium- ^{233}U could be acceptable for large scale worldwide use.

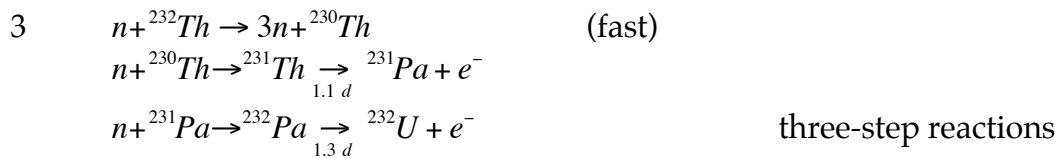
When we expose thorium to neutrons we produce the fissile material ^{233}U .



The following important reactions shown in Fig. 2 lead to ^{232}U :



There are more remote low probability reactions that also lead to ^{232}U :



This third, three-step reaction was included because of the possibility of a natural source of ^{230}Th . ^{230}Th is naturally occurring in uranium as a product of secular decay of ^{238}U (17 ppm) [Info thanks to B. Hoglund [11], who thinks it might be possible to separate ^{230}Th from uranium ore tailings and add to thorium to enhance ^{232}U production for extra proliferation resistance. He further points out that removing ^{230}Th from the tailings would remove the largest very long term radiation hazard of tailing piles.] It would be worthwhile knowing what concentration of ^{230}Th in ^{232}Th would be needed to significantly enhance the ^{232}U production and whether there are any important reactions resulting in ^{232}U omitted from the above list.

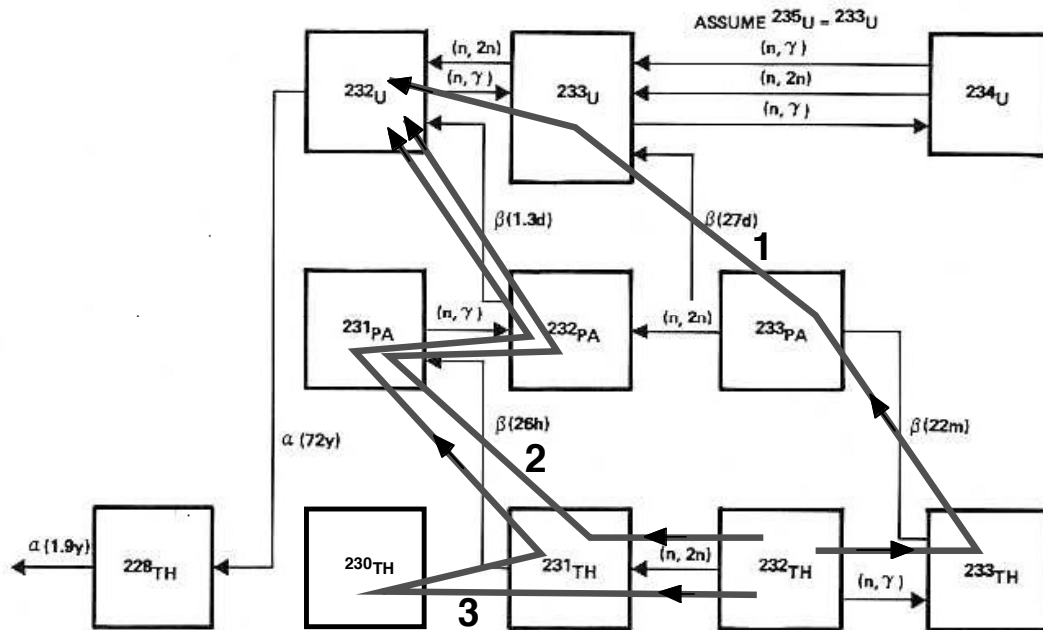


Fig. 2. Neutron reaction paths leading to ^{232}U production (from Ref. 12, Fig VII.C-1).

Past studies have shown that the primary route to ^{232}U production is by reaction path 1 until long irradiation time of thorium when ^{231}Pa builds up and path 2 becomes the primary path. Since ^{231}Pa builds up (half-life of 33,000 y), this set of reactions depends on irradiation time of thorium even after ^{233}U is removed. Long irradiation times are useful and the Pa needs to be left in during processing as happens with molten salt fluoride volatility-only processing. Studies [12] of molten salt blankets with ThF_4 irradiated by fusion neutrons obtained the $^{232}\text{U}/^{233}\text{U}$ ratio varying from 1600 to 2400 appm between year 1 and year 7 of exposure and is 3,500 appm after 30 y and 10,000 after 90 y of exposure. Similarly interesting results should be obtained using accelerator neutrons. At 10,000 appm

(1%) the dose rate at 0.5 m from 5 kg one year after separation of daughter products is 127 rem/hr (1.27 Sv/h) and concentration of 2.4% would satisfy IAEA standards for reduced physical protection [10]. The subject, "Fuel Cycle Technologies," is further discussed in section VII of Ref.[12]. The $^{232}\text{U}/^{233}\text{U}$ concentration in a molten salt fission reactor is quoted as 500 appm shortly after startup but the concentration is increasing with time and is quoted without spiking with non-proliferants, such as ^{231}Pa or ^{230}Th [13]. Clearly the buildup of ^{231}Pa and potential spiking is important and deserves special attention.

The fuel cycle that the MSR can accommodate varies all the way from use of reactor grade enrichment (<20% $^{233}\text{U}/^{238}\text{U}$) up to fully enriched fuel. Because breeding improves with higher enrichment, less demanding or very infrequent processing as in every 30 years at the graphite change-out time might be possible. The role of small amounts of ^{238}U (but still $^{238}\text{U}/^{232}\text{Th} \ll 1$) and significant but small amounts of ^{232}U should be studied in the context of nonproliferation including safeguards, spiking, and energy parks.

I recommend adopting the GNEP model with secure well-guarded centers, where uranium could be fully enriched and used to start-up molten salt reactors including processing. After a time (few years) its fuel might have enough ^{232}U to make it suitable to transport to startup new outlying reactors at less well-guarded sites. The guarded reactor could repeat this operation over and over.

5. Startup fuel for MSRs

MSRs can be fueled with ^{233}U , ^{235}U or ^{239}Pu . The preferred fuel from a neutron economy point of view for thermal and epithermal neutron spectra reactors is ^{233}U with new fuel produced from ^{232}Th neutron captures. ^{235}U is the fissile fuel available today and used in present reactors but more is required due to nonfission captures. Based on the previous discussion of proliferation resistance we might want to dilute the fissile uranium with ^{238}U and this leads to ^{239}Pu by neutron captures. Even more Pu is required because of the even larger nonfission captures. Therefore we need to consider all three fissile isotopes. For fast spectra the preferred order of fissile material is ^{239}Pu , ^{233}U and then ^{235}U , but the inventory is up to an order of magnitude larger. A reactor with solid fuel can have a large amount of inventory tied up outside the reactor in the recycle steps. The cost of this inventory can be an important factor in economics. The inventory of fissile material for the molten salt reactor is relatively quite low.

To start-up many MSRs we need economical sources of fissile material. We will consider three sources. Mined uranium enriched in ^{235}U , ^{233}U produced from accelerator made neutrons captured in ^{232}Th and ^{233}U produced from fusion produced neutrons captured in ^{232}Th . There are other possible sources such as ^{233}U produced from other fission reactors such as LWRs or CANDUs some of whose neutrons can be captured in ^{232}Th [14,15,16]. It is even possible to burn

transuranics from LWRs in the MSR and this may be a surprisingly economical way of starting up early MSRs while helping solve a waste management problem.

For a basis of discussion we will make some cost estimates. Fissile material has a value based on the cost of mining and enriching uranium-235. For our discussion we will assume the value of all three fissile isotopes are the same but we will remember there is a correction to be made as mentioned above because of the difference in neutron utilization.

5.1 Mined and enriched ^{235}U

The price of ^{235}U depends on mined uranium cost, enrichment and other processing cost and market factors. Today the cost is about 30 \$/g for uranium costing 50 \$/kg but increased projected usage and market forces might drive the cost to the region of 75 \$/g at uranium costs of 200 \$/kg. The startup inventory of 3000 kg/GWe would cost 230 M\$ at 75\$ for 1000 MWe.

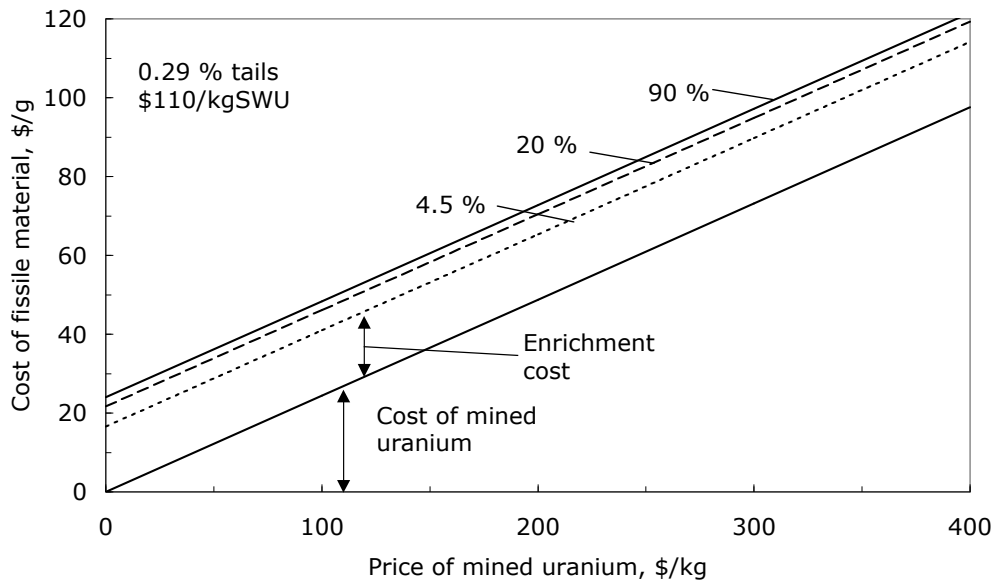


Fig. 3. Cost of mined and enriched ^{235}U .

5.2 Accelerator produced ^{233}U

We would now like to compare the cost of ^{233}U made in accelerators and fusion to the above mined ^{235}U . This is not easy to do because no such facilities exist and even designs are uncertain and in the case of fusion, feasibility has yet to be proven. Accelerators are feasible, although issues remain around high average power, cost and reliability. Nevertheless a critical item for accelerators is expected to be the cost of electricity to power the accelerator and the same is true for fusion. Let us assume the owner of the facility buys electricity and leases out that part of the facility that recovers any heat generated and converts it into electricity. This

electricity that is sold may come close to paying for the capital and operating cost of the facility or maybe make a modest profit.

We take our numbers from a study where a 0.3 A, 1 GeV proton beam is directed into a simple molten salt vortex chamber producing 36 to 58 neutrons (0.82 to 1.33 Ton $^{233}\text{U}/\text{y}$).¹⁷ There is considerable fission taking place in the vortex without which the number of neutrons produced would be closer to 25. We can estimate that portion of the cost of produced fissile material, ^{233}U , due to electricity costs with the following assumptions: electricity is purchased for \$50/MWh and converted to beam power at 50% efficiency, each 1 GeV proton makes 47 neutrons in a molten salt vortex reaction chamber and each neutron makes one ^{233}U ; the rest of the numbers in the equation below are the usual unit conversions.

$$\frac{\frac{50\$}{\text{MWh}} \times \frac{1 \text{ GeV}}{0.5} \times 1.6 \times 10^{-19} \text{ J/eV}}{3600 \text{ s/h} \times \text{J/W} \cdot \text{s} \times 47 \text{ n} \times 233 \text{ amu} \times 1.67 \times 10^{-27} \text{ kg/amu}} = \$240/\text{g}$$

From an extrapolation of Fig. 3 the cost of mined U would have to approach 900 \$/kg for accelerator produced neutrons to be competitive with these assumptions. This is much higher than the projected cost of ^{235}U from mined uranium. The startup inventory of 3000 kg/GWe would cost 0.7 B\$ at 240 \$/g. Means to lower the cost of accelerator produced material will be important for the accelerator technology to compete with mined uranium. It is possible that the sales of electricity will more than offset the capital and operating costs and therefore lower the cost of fuel produced. Using deuterons or tritons rather than protons results in more neutrons per unit of energy. The fact, that ^{233}U is more valuable than ^{235}U and more so with its concomitant ^{232}U , will help offset some of the high cost.

5.3 Fusion produced ^{233}U

Studies of magnetic and inertial fusion power plants designed to produce fissile material have been carried out. One such study [18] produced the results in Fig. 4. The cost of produced fuel is plotted as the solid line with dashed lines at 0.67 and 1.5 times this to illustrate uncertainties. The cost of electricity dominates the cost of material when the driver efficiency, η times fusion gain, Q is < 1 , where Q =fusion power or energy / driver power or energy.

The cost of electricity to drive the magnetic fusion fuel producer is given below:

$$\frac{\frac{50\$}{\text{MWh}} \times \frac{17.6 \text{ MeV}}{\eta \cdot Q} \times 1.6 \times 10^{-19} \text{ J/eV}}{3600 \text{ s/h} \times \text{J/W} \cdot \text{s} \times 0.6 \text{ n} \times 233 \text{ amu} \times 1.67 \times 10^{-27} \text{ kg/amu}} = \frac{170}{\eta \cdot Q} \$/\text{g}$$

where we assume as before, electricity cost 50\$/kWh and each fusion produced neutron can produce 0.6 ^{233}U atoms as well as replace the tritium consumed in the D-T reaction. The portion of the cost of fuel, due only to the cost of electricity to drive the fuel producer, is plotted in Fig. 4 as the heavy dashed line.

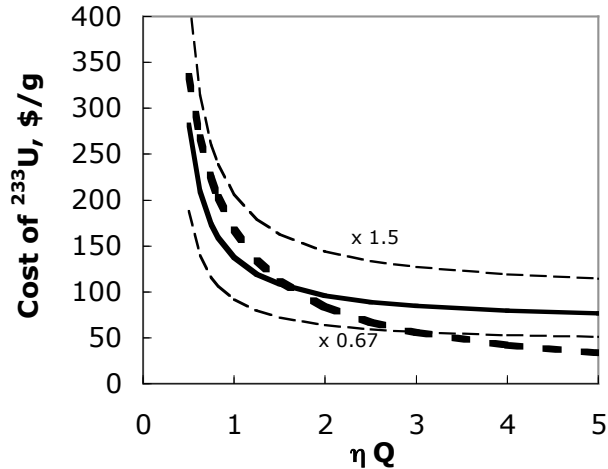


Fig. 4. Predicated cost of fusion-produced material versus driver efficiency times Q .

Typical driver efficiencies in magnetic fusion approach 0.5 and 0.1 to 0.2 in inertial fusion. The gain of about 1 has been demonstrated in magnetic fusion facilities and is expected to be about 10 in the next generation fusion facility, the International Thermonuclear Experimental Reactor, (ITER). In inertial fusion a goal of a gain of 10 is hoped for by about 2012 or a few years later in the National Ignition Facility (NIF). Fusion produced fissile material from a facility with $\eta Q > 4$ might be competitive with mined uranium at about \$100/kg. The possibility of producing economical fuel with fusion technology may not be so far off and coincide with the possible strong demand at mid century as shown in Fig. 1. Also this application of fusion is much less demanding than is the goal of producing commercial electricity. As fusion technology improves, the D-D reaction will become more interesting as it produces far more neutrons than the D-T reaction.

To summarize: accelerator produced ^{233}U startup fuel along with its concomitant production of proliferation resistant ^{232}U has a nonproliferation advantage over mined and highly enriched uranium ^{235}U but is costly. Highly enriched ^{235}U could be used as start-up fuel in reactors in well-guarded centers and after ^{232}U builds up could be used to start-up less well-guarded reactors outside these centers. Fusion produced ^{233}U might substantially lower the startup fuel cost over that of accelerator produced fuel and of mined and enriched uranium while simultaneously producing proliferation resistant ^{232}U .

5.4 LWR produced ^{233}U

Adding thorium to LWR or CANDU fuel can be a source of fissile material for starting up MSR. There are processes especially appropriate that result in liquid fuel rather than reconstituted solid fuel that may avoid some of the high cost impediments known for thorium fuels. Such a use of LWRs would make maximum use of the large investment in this technology while transitioning to the much large "fleet" of high conversion ratio reactors.

6. Thermal (moderated) versus unmoderated MSR

The MSR can be designed with a low power density and therefore a low enough neutron flux, that the graphite radiation damage lifetime is manageable, perhaps 30 years. The graphite reflector protects the vessel walls from radiation damage. If the MSR's operating temperature increases and the fission products are removed the negative temperature coefficient can become positive and aggravate control as pointed out by Lecarpentier and colleagues [19]. This fact contributed to the French studies tending towards unmoderated versions of the MSR an example of which is given in Ref. [20,21] and in Russian work [22]. Both these studies emphasize the thorium cycle with startup on transuranics from LWRs. However, reflector control rods that are automatically operated by temperature should be able to maintain a strong negative temperature coefficient [23]. The graphite-moderated reactor should be able to operate at a higher temperature than the unmoderated reactor, depending on the reflector material temperature limit because graphite can handle temperatures higher than almost any other material.

Power flattening reduces breeding but increases graphite lifetime and increases the fuel inventory. Peaking the power profile reduces graphite lifetime but increases breeding and reduces the fissile inventory. Load following results in reduced average power and reduced revenues but longer life until graphite damage requires shut down for replacement.

6.1 Thermal neutron (graphite moderated) MSR

Trade-offs are important in considering and comparing thermal neutron (with neutron moderator) with epithermal neutron molten salt reactors without moderator. In the thermal neutron MSR, once the power density is specified, the lifetime of the graphite determined by radiation damage is known. The power density along with the core size (diameter) sets the power. The core diameter is d and height is h .

$$P = \frac{P}{V} \pi d^2 h / 4 = \frac{\pi P}{4 V} d^3 \quad \text{if } h = d$$
$$\tau = \frac{\text{damage criterion}}{C \frac{P}{V} CF}$$

P/V is the power density in the reactor and is typically ten times higher in the fuel for a typical graphite to fuel ratio of 10.

The *damage criterion* is in units of n/cm^2 for energy above 50 keV, $3 \times 10^{26} \text{ n/m}^2$. The damage criterion is chosen such that $\tau = 30 \text{ y}$, $P = 1000 \text{ MWe}$, $d = 10 \text{ m}$ and $CF = 0.75$. $C = 1.047 \times 10^{25} \text{ nm/yMWe}$, $P/V = 1.27 \text{ MWe/m}^3$, at 43% thermal efficiency, $P/V = 2.96 \text{ MWth/m}^3$.

$C \cdot P/V$ is the damage rate in units of $4.225 \times 10^{17} \text{ n/cm}^2\text{s}$.

CF = capacity or power factor. Typically for a base load plant today CF is 0.9 and for load following mode of operation might be about 0.5.

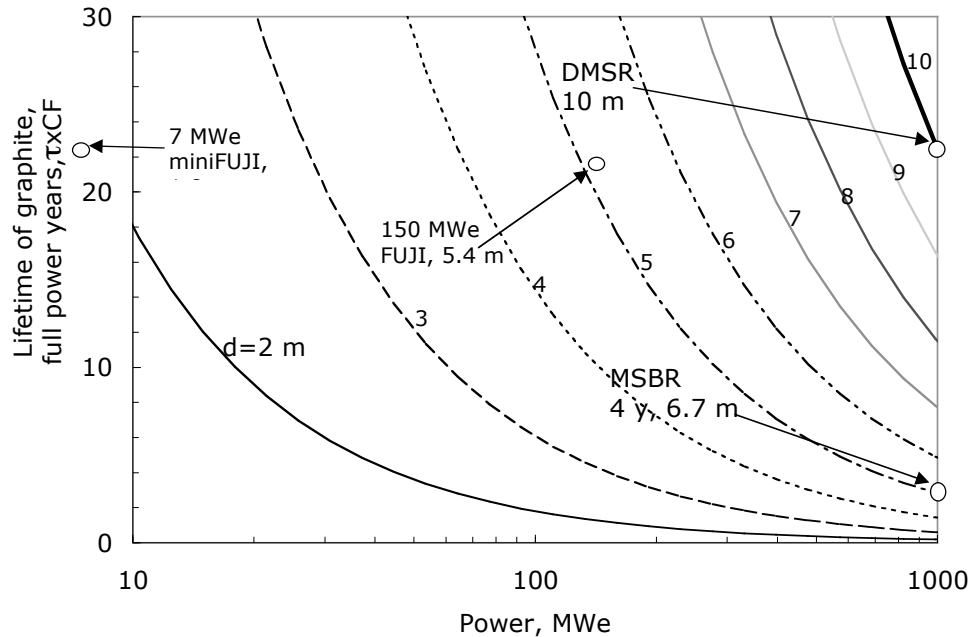


Fig. 5. The core graphite lifetime is related to power and core size.

When the graphite reaches its damage limit (swelling and or cracking) the reactor is shut down for an extended time to replace the graphite. This is costly and results in loss of power sales that might be minimized with use of advanced robotics.

The core size is related to factory manufacture of the vessel and transportation to the reactor site, which is a favorable factor for diameters up to about 5 m. At 10 m the construction is likely an expensive field operation or more elaborate transportation to the site by barge for example.

A base load plant can have 30 years graphite lifetime for a diameter of about 10 m at 1 GWe or 5 m at 150 MWe. A load following plant can have a diameter of about 5 m at 230 MWe. Core design can change these numbers somewhat.

In the course of development of the MSR and during the early deployment of the first few power plants, the size of the vessel will not be a concern, but extensive deployment is expected to be highly driven by “market pull” resulting in a tendency towards larger vessels. Economic competition will be intensive with mature LWRs whose sizes today are approaching 2 GWe for economy of scale reasons. This means the MSR might have to deliver large plants in order to be competitive. Of course smaller markets away from transmission grids will still have a market for small plants.

Another aspect of the thermal neutron MSR is its low fissile inventory: 3.5 kg of ^{235}U /MWe for the thorium-uranium-plutonium fuel cycle Denatured Molten Salt Reactor (DMSR) with 30-year graphite lifetime, 1.6 kg ^{233}U /MWe for the Molten Salt Breeder Reactor (MSBR) (with 4 year graphite life). Some parameters of several MSR designs are given in Table 3, particularly size and core lifetime.

Table 3
Parameters for various MSR designs

	Vessel diam, m	Power	CF	Graphite lifetime
MSRE	1.4	8 MWth	0.4	3.3 y*
miniFUJI	1.8	7 MWe	0.75	30 y
	5.4	150 MWe	0.75	30 y
MSBR	6.7	1000 MWe	0.75	4 y
DMSR	10	1000 MWe	0.75	30 y

*Operating time June 1965 to October 1968, 11,500 full power hours.

The MSR can operate on any fuel, in a variety of mixtures, and those mixtures can be changed while it operates; i.e., the MSR has extraordinary fuel flexibility.

6.2 Epithermal neutron (unmoderated) MSR

The reactor design is simplified by removing the graphite moderator in the core thus eliminating the graphite damage lifetime limitation. The harder neutron spectra will reduce actinide captures thus enhancing fissioning of transuranics. However, the reflector material inside the reactor vessel sees a large neutron flux, and will have to be replaced periodically. The reflector protects the vessel walls from neutron damage. This radiation damage will set the power density and lifetime to replacement similar to the graphite moderated reactor discussed above. Then the power and diameter of the vessel can be determined, but this topic is beyond the scope of the present paper. The material of the reflector might set a limit to the operating temperature. More attention is needed on the problem of radiation damage to the walls for the unmoderated MSR.

The fissile inventory of an epithermal spectrum unmoderated MSR is higher owing simply to the fact that the fission cross-section is lower at higher neutron energies than at low energies. Unmoderated MSRs could also use plutonium and other Transuranics (TRU) more efficiently, as TRUs have higher probabilities of fissioning in higher energy neutrons spectra. As discussed earlier, the cost of electricity due to the initial fuel inventory can be considerable and is clearly an economic issue. However, the use of the currently classified TRU wastes and the lack of having to make expensive TRU fuel elements, may make the unmoderated MSR fuel inventory a revenue generator versus an inventory cost.

7. Carbon-composite material development

If an MSR test reactor were built in the next ten years it would almost surely be constructed of the nickel alloy, Hastelloy. However, carbon composites (C/C) are undergoing rapid development that might allow use with molten salts. Operating temperatures could exceed 1100 °C that, in turn, increases the solubility of plutonium molten salts and permits increased TRU concentrations. Higher temperatures would also allow a closed cycle gas turbine power conversion system (Brayton cycle) with the benefit of higher efficiency, lower cost, easier tritium recovery and minimization of freeze-up problems. Handling tritium and

keeping it from getting into the environment might be easier than with a steam cycle. Another application that high temperature might enable is hydrogen production.

For carbon composites to become practical we must:

1-develop leak-tight composites or design to accommodate a porous/leaky material

2-develop means to join C/C vessels and pipes

3-develop means to repair C/C components

A flat plate heat exchanger might be made from C/C slabs (1 m × 2 m nominal) ~2 mm thick with channels milled in the panels for coolant [24]. Alternatively C/C slabs ~1 mm thick could have a thin silicon sandwich that is reacted to form a SiC layer whose purpose is to reduce gas leakage.

An experimental development program could begin by building a small carbon composite vessel with a diameter of about 0.1 m to mock up full size units. Full size units might be 2 m dia for ~10 MWe and 5 to 10 m dia for a few hundred MWe to a GWe. Tests could use the surrogate molten salt NaCl +MgCl₂ that is nearly identical thermo-chemically to LiF +BeF₂. A vacuum oven would be needed to bake out gases, especially oxygen and hydrogen. Carbon composite research is recommended for molten salt reactor application.

8. Future of MSRs and discussion

The MSR needs a demonstration to reawaken the potential of the concept and its advantages. An NGO (Non-Government Organization) or venture capitalist or philanthropist could fund the ~\$1B needed for some development and construction of a small reactor. China and India could take on the task to their and the world's great benefit. The major reactor development nations have passed on the MSR probably discouraged by the legacy of "...there must be something wrong with the concept because the Americans stopped their work completely three decades ago...." and distracted by the hope of liquid metal cooled fast reactors becoming economical even after two dozen such plants have been built.

Relevant technology useful for the MSR is, nevertheless, being developed. For example, carbon composites are finding applications in many places. The South African development of the high temperature graphite pebble bed reactor with Brayton cycle closed cycle turbines replacing the steam cycle might be useful to an eventual MSR. The high temperature capability should make the MSR a competitive candidate for hydrogen production. Strategies for avoiding CO₂ production will help the MSR. Better graphite will be useful. Finally the concern over global warming might contribute to restarting the MSR work.

9. Conclusions

The MSR has so many favorable features, many discussed here that one is at a loss to explain why the reactor has not already been developed. Once a program has been killed there is a stigma attached that creates a legacy of its own. Several decades ago reactor accidents, low number of orders for new reactors, low uranium prices and low-cost natural gas have discouraged reactor development such as the MSR but all these things have reversed. I strongly recommend

independent thinkers to relook at and invest in the MSR. China and India could take on this task to their great advantage. Even a philanthropist or venture capitalist could breathe new life into this concept – a small 10 MWe (or even much smaller) test reactor would provide relevant information useful to proceeding on to a commercial power reactor and represents a low risk, low cost first step.

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