COMPACT FUSION TECHNICAL NOTE

ON CROSS BREEDING OF FUEL BETWEEN GT-MHR FISSION REACTORS AND SMALL SPHERICAL TOKAMAK REACTORS

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Overview

Supplies of mined Uranium are likely to decline after 2035 unless remarkable discoveries are made to more than double the current IAEA Redbook's estimated endowment of about 20Mt. This will stall any aggressive programme of nuclear deployment (~3000 reactors) by 2050 as Fast Fission Breeders have a breeding ratio of about 1.2, which is too low to build the Fast Reactor fleet and support the existing thermal reactors at the same time. Fusion can breed up to 20 times as much fissile fuel for the same energy output.

We examine the possibility that Small ST Fusion (SSTs) reactors with outputs below 100 MW-thermal could breed sufficient Plutonium from depleted Uranium to supply makeup fissile fuel to GT-MHR reactors which would, in turn, be breeding Tritium to run the SSTs. This was first suggested in a wide ranging web paper on 'Fission & Fusion Futures'.

Fission reactors are also slow at breeding Tritium to start and fuel the fusion breeders. Nevertheless, simple considerations show that an SST Fusion Hybrid breeder with output power around 62MW-thermal can indeed make top-up Plutonium fuel for a GT-MHR plant with output power of 2400 MW-th, running on Tritium made by the fission plant. The low Fusion power required means that neutron wall and coil loadings can be low and it should be possible to build the reactor very soon with existing and very near-term technologies. It also means that the SST breeding blanket need only produce Plutonium.

Fusion is also a very efficient breeder of Tritium, once started, with a breeding ratio of about 1.6 and a doubling time of about 2 years. Other types of SST breeding only Tritium can expand the SST fleets rapidly to allow Plutonium production for standard light water cooled reactors like the French EPR or the Toshiba AP1000. By separating the fusion breeding of Tritium and Plutonium the blanket designs are simplified and the total engineering challenges reduced.

The calculations here are the simplest kind of estimates to give guidance to detailed design efforts required for both reactor systems. Fortunately, many studies have been performed with the full neutronics of all the components and we can therefore quote achieved design numbers where necessary.

1. **Energy Production from 1 tonne of fuel.**

   One tonne of Uranium-235 or of an equal mix of Deuterium and Tritium contains N nucleons of mass m=1.66 10^-24 grams, or N/235=2.56 10^27 Uranium nuclei, or N/5 D-T pairs. The mass difference between neutrons and protons is a small correction to this.

   Fission of a Uranium-235 nucleus produces 190MeV of energy in fission fragments, an average of 2.5 neutrons with an average energy of 1 MeV, gamma-rays, and beta emissions. Another 10 Mev is released in neutrinos which are never seen again. The available total from each tonne of U-235 is 7.8 10^16 Joules. The main constituent (95.1%) of the Uranium fuel is U-238 which does not fission at neutron energies below ~1 MeV.

   We convert this to a reactor scale measure of energy, the Gigawatt-day, or 1000 MW per hour for 24 hours. The energy output from a tonne of U-235 is then 902.96 GWth-days and from Plutonium 948 GWth-days.

   In a typical thermal reactor, EPR or AP1000, with about 4.9% of U-235 in the Uranium oxide fuel rods, some 60% of the neutrons are absorbed by the 95.1% of U-238 and are transmuted into the fissile Plutonium-239, with emission of a further 2MeV per transmutation. This leaves 1 neutron over to cause another U-235 fission. The reactor neutronics is finely controlled to leave very few spare neutrons for, say, breeding Tritium. This balance can be adjusted. About 1/3rd of the energy produced is from fission of some of the Plutonium bred in the process, so the spent fuel contains U-235 and Pu-239, some of their higher isotopes , 5% of radioactive fission products, and some higher Actinides.

Fusion of a Deuterium-Tritium pair yields one neutron with 14.1 Mev and a Helium-4 nucleus with energy of 3.5MeV. The energy output from one tonne of DT fuel is (17.6/5)*(235/190)* 903 GWth-days, or 3931 GWth-days.

2. Breeding rates for GT-MHR Tritium.

The GT-MHR uses 16-20% enriched fissile fuel pellets, but we assume the Pu breeding ratio is still about 60%. This leaves 40% of the fissile material to be replaced in the refuelling cycle.

The GT-MHR is designed in small modules of 600MWth power each. The duty cycle is to be 90%. It therefore burns the equivalent of 304kg of U-235 per annum and leaves 182 kg of fissile material in the spent fuel. Some 121kg of fissile fuel is needed to make up that burned, or 116kg of Pu as this produces a net 200MeV of usable energy per fission.

If a fraction, f_T, of a neutron is assigned to breeding Tritium by the fission of pellets containing Li-6, then the mass of tritium produced is proportional to the mass ratio, 3/235, of the fusion and fission fuels. With f_T =0.1, a GT-MHR600 should yield 0.387 kg of Tritium. With f_T =0.2 we get 0.774kg. and a power station of 4-6 MHR600s has a total output of 3.1- 4.6 kg, sufficient to fuel an SST Plutonium breeder.

The GT-MHR reactor is favoured for this breeding application because the Tritium produced is contained in the breeding pellets till harvested. Extraction is then done at the fuel recycling plant.

Whatever the output of Tritium, the GT-MHR or other enriched fuel reactor are essential to produce at least the initial loads for the SSTs. Tritium production must therefore begin 3-5 years before any SSTs are brought online. Since Tritium decays to Helium-3, the stable nuclide with atomic number 3, with a half life of 12 years then only 4 SST loads would be available after 5 years. The Helium-3 is still a potential fusion fuel, but at much higher temperatures, and should be reserved.

3. **Breeding rates for SST Plutonium**.

Fusion breeding blankets are usually designed with three layers – a neutron multiplier layer (possibly Beryllium-9) to slow down the neutrons and to make 2 neutrons per fission of the Be. The second layer would absorb one of these to fission Lithium-6 and make Tritium, or Li-7 and recover the neutron. The next layer would be of fertile material like Uranium-238 (or Thorium-232 in a Th-U233 fuel cycle).

We would like to minimise the size of the SST and omit the Tritium breeding completely and keep all the neutrons for Plutonium breeding. This means that the Pu breeding ratio for each DT-neutron, b_{Pu} , can be bigger and the Fusion power needed to generate enough Pu for a GT-MHR power station could be lowered with attendant engineering benefits.

The Pu breeding ratio for fusion neutrons penetrating an infinite medium of depleted Uranium is around b_{Pu} =4 per fusion neutron. The very energetic 14.1 MeV neutrons can cause some fissions of the U-238 which is therefore also a neutron multiplier. In engineered blanket systems, neutron losses, absorption in structural materials, finite blankets, and other factors reduce this. Calculated Pu breeding ratios for blankets on large Tokamaks are about b_{Pu} =1.8. The mass of Pu produced is proportional to the mass ratio of the fission fusion fuels, (239/5).

A 62MWth SST , with a 90% duty cycle, would burn about 3.1kg of Tritium per annum. The resulting neutrons would produce 463kg of Plutonium with a b_{Pu} = 1.87, which is sufficient makeup fuel to support a 4 unit GT-MHR power station.

It remains for detailed neutronics calculations to find the best blanket performance for this application. There are many options. Depleted Uranium is currently stored as Uranium Oxides or as solid Uranium Hexafluoride. The Oxide could be packaged with Beryllium in TRISO particles for the blanket, using high temperature (900^{o}C) Helium coolant for high efficiency of energy conversion to electricity. This shares technology with the GT-MHR. Another design option is to simply mix UF with BeF as a molten salt breeder and coolant. An advantage of this is that extraction of Pu can be done online without shutting down the SST. This shares technology with current Fast Reactor designs.

A depleted Uranium blanket, with .03% of U-235, would be quite heavy, wrapped about the SST, around 150 tonnes. The Pu produced would enrich it to only 0.463/300 or 0.15%, or double that on a two year recycling programme, far below the level at which this could reach reactor criticality. The Pu is extracted by 99.9% recovery chemical and thermal processes in the UREX system, not by isotope separation in centrifuges.

4. SST Reactor Parameters

Self consistent parameters for the Design of an SST with an output of 62MW are calculated to agree with the IPB-98 scaling laws for plasma containment. The ST research group at Culham find some enhancement of containment over this scaling by a factor H of about 1.3 or higher. We use H=1.3 for that model. The Princeton NSTX group have found a much more aggressive scaling with magnetic field strength, though only over a limited range. We also use that scaling to give some idea of an upper limit on the performance. We adjust the toroidal plasma current at various toroidal magnetic fields to achieve an energy gain factor of Q=2 for the Culham scaling and Q=3 for the Princeton scaling.

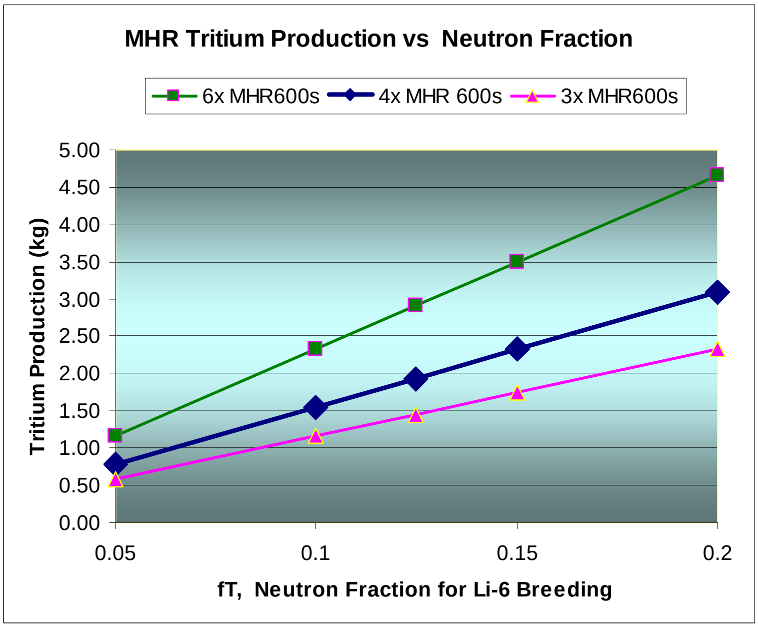
The reactor parameters are shown below and demonstrate just how small this reactor is at 1/15th. the volume of JET and especially compared with a 2400MWth fission plant.

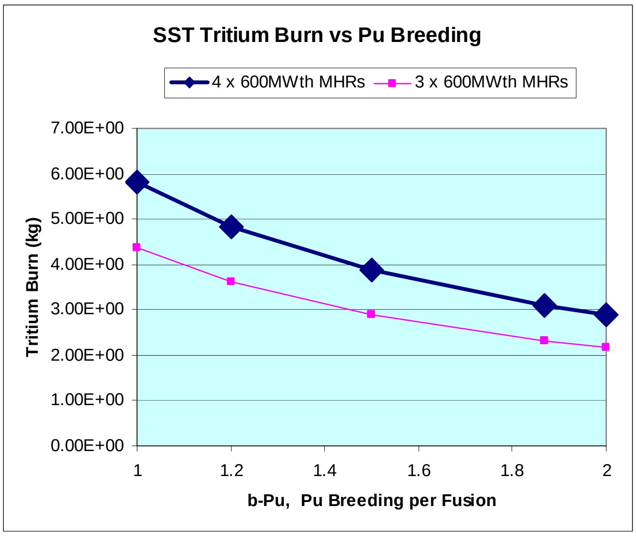
Plasma Size	**IPB-98 Scaling**	**NSTX Scaling**
Major Radius R (m)	**1.5**	**1.5**
Minor Radius a (m)	**0.98**	**0.98**
Ellipticity (b/a)	**3**	**3**
Volume (m^3)	**48.8**	**48.8**
Plasma Parameters		
Ion Temperature T_i (keV)	**8.5**	**8.5**
Electron Temperature T_e (keV)	**8**	**8**
Plasma pressure/Magnetic Pressure β %	**12**	**12**
<Electron particle density> n (10^{20} / m^3)	**2.24**	**2.24**
Fusion Power (MW)	**62**	**62**
Input Power (MW)	**31**	**20.7**
Wall Loading (MW/ m^2)	**0.77**	**0.77**
Performance Parameters		
Toroidal Magnetic Field B_T (T)	**3.5**	**3.5**
Toroidal Plasma Current I_P (MA)	**8.26**	**4**
Confinement Factor H	**1.3**	**1**
Energy Confinement Time τ_E (sec)	**0.94**	**1.4**

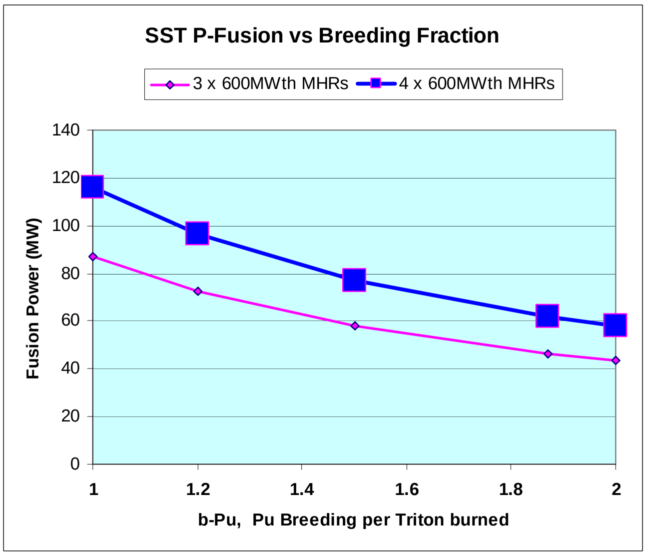
5. Cross breeding Performance

The ability of a GT-MHR power station to supply its SST breeder depends upon the allowed neutron fraction, f_T. The SST breeding performance depends upon the ratio, b_{Pu} , and the Tritium consumption depends on the fusion power needed to make 463kg Pu per annum.

The charts below show the results for these elementary calculations:







We see that the 4x MHR600 station can make sufficient Tritium at f_T. = 0.2 to run an SST with b_{Pu} =1.87 and a power output of 62MWth. This is apparently achievable by GT-MHRs.

Clearly, there is much detailed work to be done, but this simple approach shows plausible parameters.

References

McNamara B., 'Fission & Fusion Futures', http:/gt-mhr.ga.com

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