Fusion breeding for mid-century, sustainable, carbon free power

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1. Introduction

The magnetic fusion project, for more than half a century now, has hoped to achieve a true nirvana, a carbon free, ‘infinite’ energy source without any side pollution or radioactive waste. ‘Infinite’ here means capable of supplying electrical power at tens of terawatts for thousands of years. However noble the objective, progress has been glacially slow. The effort is now focused on the ITER project (International Tokamak Experimental Reactor) now being constructed by an international coalition in France. But even if ITER is successful, the world will still be a very, very long way from this nirvana. As we will show, there are enormous obstacles between a successful ITER and an economic reactor, obstacles that may or may not be possible to overcome. The conventional fusion approach may be letting perfect be the enemy of good enough. An alternative is to use a fusion reactor to breed fuel for conventional thermal nuclear reactors, that is fusion breeding. The requirements of a fusion breeder are greatly relaxed from the requirements for a pure fusion reactor. For instance, ITER, the large tokamak being built by an international consortium in France, could well be the basis of an economical fusion breeder, but would have to clear many more scientific and technical hurdles before it could become the basis for a pure fusion reactor; hurdles it may or may not be able to clear. Even if it clears them, ITER is unlikely to evolve into an economical pure fusion power supply this century. A fusion breeder as could be alternate approach to speed the delivery of economical fusion power.

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breeding is given in [7, 9], and a discussion of many different hybrid fusion proposals is given in [8]. Another discussion has recently been published [11].

Hybrid fusion was studied in the United States and other places in the late 1970's and early 1980's, but was then abandoned in favor of 'pure fusion', namely using only the kinetic energy of the fusion neutrons. Much of this information is archived in a web site [12]. This site contains many early LLNL (Lawrence Livermore National Lab) and PPPL (Princeton Plasma Physics Lab) reports, which would be difficult to access any other way. Generally, these reports considered a fusion device surrounded by a uranium or thorium blanket which provided a 'fusion kick' to the power produced. Today, Russia, possibly with Chinese help, has announced plans to build such a fusion fission reactor at the Kurchatov institute, a tokamak reactor called T-15MD [13]. Peter Khvostenko is one of the project leaders.

"The hybrid tokamak is now called the T-15MD, which is a large installation," said Khvostenko. "At the end of the year we have to assemble it on the site of the old T-15 which we dismantled in order to build a new one on its foundation." [13].

He added that in 2020 there would be a physical launch of a new facility, and scientists will work on the technologies that "are necessary for a thermonuclear neutron source precisely for a hybrid reactor" [13].

China has long advocated hybrid reactors, mentioning among many advantages long fuel life, use of 238U as fuel, and running the fission part of the reactor subcritical. Since such a reactor is burning mostly the fission fuel, the additional Q to the provided to the fusion part can be large [14]. Professor Hongwen is the deputy director of the project.

"Professor Wang Hongwen, the deputy director of the hybrid reactor project, said that the key components would be built and tested around 2020, with an experimental reactor complete by 2030." [14]

However, few details for either regarding schedule, budget, milestones etc. have been announced. The ITER web site does mention it, but also gives virtually no details [15].

Shortly we will discuss the drawbacks of the 'fission kick' concept. For one thing, we have known how to build critical nuclear reactors safely for 70 years now, so to this author, a subcritical fission reactor does not seem to be an advantage worth all the additional complications of the hybrid reactor.

In these earlier studies, fusion breeding, the use of fusion neutrons to breed nuclear fuel for other free-standing fission reactors was hardly ignored [16, 17, 18], but was certainly not emphasized either. This author sees the main justification for fusion breeding as combating a potential shortage of nuclear fuel [3]. Furthermore, using fusion simply to breed fuel for stand-alone fission reactors fits in much better with current nuclear infrastructure, than does a fusion hybrid reactor, where every reactor is something very different from the what is used at present. In short, it is not only possible, but likely, that fusion breeding could fill a pressing future need.

Summarizing some of this work earlier is a classic Physics Today Article called The Fusion Hybrid by none other than Hans Bethe [19], a giant of 20th century physics.

So, what has changed in the intervening 40 years to make it worthwhile to revisit the issue? Probably the main thing that has changed is that the predictions of fusion progress back then grossly underestimated the difficulty of the plasma physics task and overestimated the variety of potential plasma configurations. For instance, here is a quote from Bethe:

"Currently a large version (tokamak), the TFFT, is being built in Princeton: It will probably be completed in 1982 and may begin operation the following year. Its designers expect the Princeton tokamak to reach Q ~ 1". (Q here is defined as the fusion power divided by the power driving the tokamak, i.e. Ohmic heating and/or microwave heating and/or beam heating).

Forty years after Bethe’s article, this still has not happened. In fact, one of the main points of this article and its predecessors [7, 9] is that pure magnetic fusion is almost certainly too difficult, at least for providing economic power this century; fusion breeding most likely is not.

TFTR did a Q ~ 0.3 experiment in a DT plasma, but only in a 'hot ion' mode that could not be sustained [20]. JET, the European tokamak did a similar experiment; it also achieved a Q ~ 0.6 in a hot ion mode which could not be sustained, and a Q ~ 0.2 in a thermal plasma which could be sustained for the length of the pulse. Figure 1 is a plot of neutron power for both modes from different shots of JET [21].

While these results are justifiably regarded as triumphs, tokamaks have produced no other 14 MeV DT neutrons in the intervening 20 years. The tokamak program has largely been becalmed in the last 20 years, at least as regards to operating with decent Q's in DT plasmas. TFTR was disassembled shortly after its DT run, and JET has never repeated its DT experiments.

However, tokamaks have produced a small number of DD neutrons in deuterium plasmas. From the DD neutrons, it is a relatively simple matter to estimate what the plasma would have produced if it were a DT plasma. The Japanese tokamak JT-60 has produced an equivalent DT neutron rate in a sustained discharge, to give a DT equivalent Q ~ 1.25 with its W shaped divertor [22, 23, 24, 25, 26]. Shown in Figure 2 from [23] is a
Tokamak JT-60 1.6
TAE (Padua) 3
ST (NSTX) 5
RFO (Padua) 3.10^{17}
Mirror (gas dynamic trap) 1.2 \times 10^{17}
TAE (field reversed configuration) 6 \times 10^{16}

The asterisk refers to data from Wendelstein, no asterisk refers to data from LHD.

Table 1. Figures of merit for a variety of fusion devices. The left most column is the device; the second is the triple fusion product; that of ion density in m^{-3} times the ion temperature in keV times the energy confinement time in seconds; the third column is the contained plasma energy in Megajoules. The first 3 rows are numbers taken from web sites or published articles, the last 3 are estimated from approximate numbers given in publications or presentations.

| Device                      | nTτ | Confined energy |
|-----------------------------|-----|-----------------|
| Tokamak JT-60               | 1.6 \times 10^{20} | 8.6             |
| Stellerator LHD, Wendelstein | 5 \times 10^{19} * | 1.4             |
| ST (NSTX)                   | 5 \times 10^{18} | 0.15            |
| RFO (Padua)                 | 3 \times 10^{17} | 0.006           |
| Mirror (gas dynamic trap)   | 1.2 \times 10^{17} | 0.03            |
| TAE (field reversed configuration) | 6 \times 10^{16} | 0.01            |

Figure 3. The development of the triple fusion product for tokamaks over the years and for a variety of tokamak devices. The total support for the tokamaks over the years, worldwide, have easily been in the many tens of billions of dollars.

graph from JT-60 of the Q as a function of current with and without the W shaped divertor.

Recently, D-3D, the tokamak at General Atomics has produced an equivalent Q of ~0.15 in a sustained discharge by finding a new, more quiescent mode of operation called the Super H mode [27]. Note that D-3D is a much smaller tokamak than JT-60 so one would not expect nearly the same Q in a sustained discharge.

Again, from Bethe:

“Another promising device is the tandem mirror being developed by the Lawrence Livermore Laboratory. . . . Its Q may also approach 1.”

However, while the gigantic field coils were manufactured and set up for their tandem mirror experiment called MFTX in the early 1980’s, the device was never turned on and was quickly disassembled.

Hence something happened; the progress on tokamaks turned out to be much, much less than Bethe anticipated, and the mirror machine was never even turned on before it was disassembled. However, perusing the earlier reports from PPPL and LLNL, the emphasis seemed to be more on hybrid fusion mirror machines than on hybrid fusion tokamaks. To some extent this is natural, mirror machines would have a much simpler geometry, it would be much easier to wrap a blanket around one of these than on a tokamak. However, first one needs the fusion whatever the device.

While there are those who advocate various different plasma configurations, they should consider realities. Tokamaks are far ahead of any of the competition and are likely to remain there for the foreseeable future. There are two metrics for any magnetic fusion device which we use here, the triple fusion product, the product of ion density times temperature times energy confinement time; and the contained plasma energy in the machine. (We discuss the triple fusion product in more detail shortly, it is roughly proportional to the Q.) Reference [7] gave a table for these two metrics for a variety of different devices. The first 4 rows of Table 1 here are reproduced from Ref [7], but with information from the Wendelstein stellarator added; the last two rows from other literature [28, 29]. Regarding mirrors, the gas dynamic trap in Novosibirsk has received attention as a possible neutron source for materials testing, if not as a fusion device itself [28]. The last row contains results for a field reversed configuration by TAE (originally Tri Alpha). TAE is one of a variety of different companies which have sprung up in the last decade or so. They claim imminent fusion and are supported by the private sector [29]. Many have gotten high powered individuals to serve on their boards, and have gotten very wealthy individuals to support them financially [30]. A skeptical look at these was recently assembled by Daniel Jassby [31]. All other configurations are way behind the tokamak. Hence, this paper concentrates on the tokamak.

The next issue concerns the development of the tokamak. Just what did it take to achieve its current performance? Figure 3 shows a plot of the triple fusion product as a function of years as the various tokamaks have developed.

Notice that TFTR came on line at about 1985, not too long after Bethe predicted, but it took another 10 years before it achieved a triple fusion product sufficient to give the neutron production quoted. The fusion effort points with pride to the fact that it has its own “Moore’s law”; namely the product doubles every 1.8 years.

Because of this success, the nations doing major MFE research (the USA, Europe, Russia, China, South Korea, Japan, and India) have joined together to build a much larger Tokamak, ITER. It was originally designed to have a major radius of 8 m, a minor radius of 2.7 m and cost $10B for construction and $1B per year to operate. This ITER, which we now call Large ITER here was designed to achieve Q~10 and generate 1.5 GW of neutron power in a 400 s pulse and have a current of 20 MA [32]. Because of the high cost, the USA pulled out. It rejoined when a smaller, less powerful version (ITER instead of Large ITER) was substituted, one having a major radius of 6 m, a minor radius of 2 m, and had half the cost. It was designed to generate 500 MW of neutron power, still with Q ~ 10 for 400 s and have a plasma current of ~15MA [33].

ITER is an enormous project; the construction cost was originally estimated at ~5B, but is now estimated to be at least $25B, possibly higher. It was originally scheduled to be turned on in 2016, but this date has now slipped to 2025, with DT experiments to begin about 2035.

Quoting from the ITER web site, its most important goal is

1) Produce 500 MW of fusion power

The world record for fusion power is held by the European tokamak JET. In 1997, JET produced 16 MW of fusion power from a total input heating power of 24 MW (Q = 0.67). ITER is designed to produce a ten-fold return on energy (Q = 10), or 500 MW of fusion power from 50 MW of input heating power. ITER will not capture the energy it
produces as electricity, but—as first of all fusion experiments in history to produce net energy gain—it will prepare the way for the machine that can.

However, there are two facts to consider as regards Figure 3. First, the actual Moore’s law allowed profitable devices at every point along the red line, MFE has to advance along the blue curve a considerable distance before it earns a dime. Secondly, if MFE could sustain its own Moore’s Law, it would have produced ITER by about 2005. Instead, ITER is now expected to come on line in about 2025, probably produce an equivalent Q ~ 10 plasma around 2030, and begin DT operation around 2035. In other words, the final doubling time is no longer 1.8 years, but more like 12 years!

Furthermore, this author has shown in the past [7, 9], and will show here, that there are significant obstacles between successful operation of ITER and an economical power plant; obstacles that might well not be possible to overcome.

However, a fusion device like Large ITER (or even ITER) is fine as an economical fusion breeder, even if not as a stand-alone fusion power plant. Hence what is needed in the MFE community is a realization that economical fusion breeding is very likely achievable, whereas economical pure fusion might well prove not to be. Fusion breeding could produce a product having real economic value, something the world might actually need and use on a large scale, and do so around mid-century. One of the author’s earlier reviews have given the argument (insisted upon by the reviewer) that the best hope for midcentury carbon free power, on a scale large enough to power modern economies, is nuclear rather than solar [9].

Even in a best-case scenario, where economic pure fusion does prove to be possible, the use of fusion breeding would not only be usable for decades, while pure fusion is being developed, but the experience with fusion breeders would aid in the development of pure fusion.

Let us turn to nuclear energy, a discussion that will be very brief. So far virtually all nuclear reactors in the world are light water reactors (more commonly measured in terawatt (TW) years, in a worst-case scenario, this would only supply world needs for 12 years!)

There has been a great deal of research into more optimum reactors including the CANUD, which uses deuterium instead of hydrogen as a moderator, the gas cooled pebble bed reactor, and the molten salt reactor (MSR). Several of these more modern reactors are designed to be passively safe; that is the reactor cools down by itself when the cooling power is turned off, or if the reactor is unpowered for any other reason. It is interesting that a Korean company, Thorcon, is now advertising modular MSR reactors, which it claims will be cheaper than coal [34].

If thermal nuclear reactors are to be the power supply of choice by mid-century or somewhat later, an important issue is how much nuclear fuel there is. There is almost certainly more fossil fuel than a nuclear fuel than a fissile material if the time comes when the maximum IFR breeding rate is insufficient to meet demand.

If breeding is necessary, the question is why go to fusion breeding? Why not just use fission breeding, which has a much shorter development path? The reason is the fusion breeding has enormous advantages over fission breeding. Two IFR’s, at maximum breeding rate, can fuel one LWR of equal power. However, a single fusion breeder can fuel about 5 LWRS of equal power. Bethe also pointed out this advantage of fusion breeding [19]. Furthermore, a fission breeder needs a great deal of fissile material just to get started [37], a fusion breeder needs none. In addition, the figure of merit for a fission breeder is the doubling time, that is the time to double the fissile material in the reactor. This doubling time is generally measured in years or decades. Doubling time is not a consideration or a constraint for a fusion breeder.

The reason fusion breeding is so much more prolific as a breeder is very simple. Whether the reaction is a fission or fusion reaction, each reaction produces 2–3 neutrons (in the fusion reaction this is after neutron multiplication, which is possible because the fusion neutron has a much higher energy than the fission produced neutron; more on this shortly, see Table 3). In fission, one of these neutrons is needed to continue the chain reaction; in fusion one is needed to breed the tritium from lithium, so in either case one or two neutrons are available for other purposes. Of course, in either case there are losses, so probably somewhere between half and one neutron per reaction is available for breeding 233U from 232Th, or 235Pu from 238U. However, the fission reaction produces about 200 MeV, while the DT fusion reaction produces only about 20 (actually a 14 MeV neutron and a 3.5 MeV alpha particle). Hence for reactors of equal power, a fusion reactor generates about 10 times more neutrons, and therefore breeds about 10 times more nuclear fuel than a fission reactor does. In other words, a fusion reactor is neutron rich and energy poor, while a fission reaction is energy rich and neutron poor, a perfect match. While there is some speculation about neutron free fusion reactions (for instance p-11B, [29]), not only are these reactions much more difficult to produce and sustain, but this article takes the attitude that neutrons are our friend, not our enemy.
Here we consider breeding of $^{233}$U since a mixture of $^{233}$U and $^{238}$U obviously is less of a proliferation risk than a mixture of $^{239}$Pu and $^{238}$U. When a $^{232}$Th absorbs a neutron, it becomes $^{233}$Th. But this decays in a two-step beta decay. First it rapidly decays to $^{233}$Pa (protactinium) which has a half-life of about a month. It then decays to $^{233}$U.

In a fusion breeder it is essential to use a liquid blanket which flows through the region of neutron flux. That way, thorium can be inserted at the input, and protactinium can be extracted at the output.

It seems extremely unlikely that a small number of fission breeders could supply a large number of thermal reactors. However, it is possible to have an energy infrastructure where each reactor is a breeder for itself alone, in other words each has a breeding ratio of unity [38, 39, 40]. A 60 MW thermal thorium breeder has been built and operated at Shippenport for 5 years, and at the end had slightly more fissile material than it began with [41]. It started burning the $^{238}$U it was fueled with, and ended up burning the $^{238}$U it produced from the thorium. This could obviously develop into an energy infrastructure competitive with fusion breeding. However, it has drawbacks as well. To attain a breeding ratio of unity, the fuel must be reprocessed often. Also, to achieve a breeding ratio of unity, the fertile material is assumed to be pure thorium. However, the thorium and $^{233}$U can easily be separated chemically, creating a real proliferation risk. To mitigate this, some $^{238}$U could also be added to the fertile material, reducing the breeding ratio. Whether fusion breeders and thermal reactors with breeding ratios of less than one, or thermal fusion breeders with conversion factors of unity ultimately, prove optimum is unknowable at this point. Most likely there would be a role for both. This author believes both should both be developed, and let the chips fall where they might. Interestingly, in our correspondences, neither Dan Meneley nor George Stanford ever mentioned the possibility of reactors with breeding ratios of unity. Furthermore, fusion breeding, in addition to itself being economically viable, could also serve as an intermediate step on the path to pure fusion.

While fast neutron reactors such as the integral fast reactor are unlikely to fuel a large number of thermal reactors, they do have a vital role in the energy architecture envisioned here. These fast reactors can burn any actinide by using its fast neutron spectrum of $\sim 2$ MeV neutrons. Hence a view of the ultimate energy infrastructure begins to emerge. As we will see shortly, one fusion breeder can supply about 5 current LWR’s of equal power. However, after a year, the LWR releases about 25% of one year fuel load as transuranic elements. Hence a single IFR of equal power could burn the actinide wastes of 5 LWR’s. If the thermal reactor were of an advanced type, one that had a higher conversion ratio, one fusion breeder might fuel ten thermal reactors of equal power. If the fertile material in the thermal reactors were in part thorium instead of $^{238}$U, there would be fewer transuranic elements produced, so perhaps one IFR could clean up after 10 thermal reactors. Bethe also made this point in his Physics Today article. It is worth noting that fusion breeding, plus actinide burning with a fast neutron reactor like the IFR is much more compatible with current nuclear infrastructure than are thermal breeders with a conversion factor of unity. There are already $\sim 400$ LWR’s world-wide, and many more are under construction or are planned.

In Section 2, we discuss plasma physics aspects of fusion breeding. Section 3 discusses the nuclear aspects. Section 4 discusses the ‘Energy Park’, a proposed sustainable, carbon free mid and late century energy infrastructure which is economically and environmentally sound and has little or no proliferation risk.

2. Plasma and magnetic fusion aspect

2.1. Fusion reactions

To orient ourselves, we list the most common possible fusion reactions. The most important of these is the DT reaction, it has the highest reaction rate and requires the minimum plasma temperature.

\[
D + T \rightarrow n(14.1MeV) + He(3.5MeV)
\]

A similar reaction uses helium 3 instead of tritium, but because of the additional Coulomb repulsion, requires higher plasma temperature and has a lower reaction rate.

\[
D + He \rightarrow p(14.7MeV) + He(3.6MeV)
\]

One problem with each of these reactions is that neither tritium nor helium 3 (in usable quantities) exists on earth. Tritium must be bred, and helium 3 exists on the surface of the moon. The fusion project is considering only breeding tritium. Tritium can be bred from lithium, and there are two possible breeding reactions. The first is exothermic:

\[
n + Li \rightarrow T(2.75MeV) + He(2.05MeV)
\]

The second possible reaction is endothermic, taking 2.47 MeV away from the reacting particles:

\[
n + ^3Li \rightarrow T + He + n
\]

Clearly this reaction requires an energetic neutron. However, depending on the breeding blanket and reactions used to breed the nuclear fuel, it may be worth the energy price to price to preserve the extra neutron.

A reaction not requiring any breeding is the DD reaction, which may proceed along one of two paths with equal probability for each.

\[
D + D \rightarrow 2n(2.5MeV) + ^3He(0.8MeV)
\]

or

\[
D + D \rightarrow p(3MeV) + T(1MeV)
\]

This reaction produces less energy and requires still higher plasma temperature. However, if one looks at it not as a reaction to produce energy, but to breed tritium and helium 3, it could be a viable reaction path, but only after the DT reaction is fully developed.

In Figure 4 are shown reaction rates for the three fusion reactions. Clearly the DT reaction rate is largest and requires the lowest plasma temperature to proceed. The reaction rate maximizes at a temperature about 50 keV. However, the total reaction per unit volume goes as $n^2 <\sigma v>$. If the pressure is constrained to some certain value, i.e. the density is this pressure divided by the temperature, then the reactions maximize at the temperature where $<\sigma v>/T^2$ maximizes, or at about 16–17 keV, where the reaction rate is about $<\sigma v> \sim 3 \times 10^{-22}$.

2.2. The triple fusion product

An important measure of the capability of any magnetic fusion device is the triple fusion product $nT$, where $n$ is the density in m$^{-3}$, T is the temperature in keV, and $\tau$ is the energy confinement.
time in seconds. At fusion temperatures, the DT fusion reaction rate \( <\sigma v> \), is roughly proportional to \( \mathrm{n}^2 \tau^2 \). where the \( n \)'s are the deuteron and triton number density, and \( W \) is the fusion energy per reaction, 14 MeV for the neutron and 3.5 for the alpha particle, this power density is roughly proportional to \( n^2 \tau^2 \). However the input power density is simply \( n^2 \tau^2 \), so the ratio of fusion power to input power, the Q of the device is roughly proportional to \( n^2 \tau^2 \).

Table 1 enumerated the triple fusion product of a number of magnetic fusion devices. The largest value is \( 1.6 \times 10^{23} \), by the Japanese tokamak JT-60. For a stellarator, the largest value up to now is from the Wendelstein stellarator LHD and is \( 5 \times 10^{19} \). Every other fusion device has a triple fusion product at least two and a half, and half, and even as much as 5 orders of magnitude below what JT-60 has achieved.

Figure 3 showed the rate at which the triple fusion product has advanced for tokamaks, currently the optimum configuration.

It seems very unlikely that any of the other devices mentioned could catch up to the tokamak in any reasonable time. The tokamaks were designed and built by a large, international coalition, by the best designers and builders of fusion devices at the time. The total effort cost tens of billions of dollars over the 50 or so years of development. Even if these other devices could maintain the same rate of advance, it would take the stellarator more than a decade to catch up with the tokamak, and if these other con

2.3. The tokamak program

The development of fusion has proven to be extraordinarily difficult. In its early days, many seemingly promising concepts were carefully considered and rejected before settling on the tokamak, now the most highly developed device. A tokamak is a toroidal plasma, which is confined by both, a toroidal magnetic field provided by external coils and a poloidal field produced by the plasma current. The plasma has cylindrical symmetry about the vertical axis passing through the center of the horizontal torus.

There have been three large tokamaks, JET in England (set up by the European community), TFTR in Princeton in the United States, and JT-60 in Japan. Large here means that 40 MW of external power, mostly neutral beams, have been used to power them, and that the major radius is at least 2.5 m. Each operates with Mega Amp currents, magnet fields of 3–5.5 T and aspect ratios of about 3 (i.e. about a 3 m major radius and a 1 m minor radius). Maximum electron density in tokamak discharges is typically about \( 10^{20} \text{ m}^{-3} \) and the temperature is around 10 keV. In neutral beam heated discharges, the ion temperature is greater, than the electron temperature. However, in a reactor, which would have larger size, the temperatures are assumed to be about equal.

Two dimensionless parameters characterizing tokamak operation are the \( \beta \) and the \( q \). The former is the plasma pressure divided by the magnetic pressure

\[
\beta = \frac{n_0 n_B}{B^2} \left( T_e + T_i \right) \frac{1}{B} \tag{1}
\]

A \( \beta \) related parameter which has proven to be useful is the normalized beta,

\[
\beta_n = 100 \frac{n}{\mu \beta B/I} \tag{2}
\]

Here \( n \) is the electron number density in units of \( \text{m}^{-3} \), \( T \) is the temperature in units of keV, and \( B \) is the magnetic field in Teslas, \( \beta \) is the plasma minor radius in meters along the midplane, and \( I \) is the current in Mega Amps (MA). Note that these are not consistent, standard units, for instance cgs, but rather are units which are commonly used in tokamak physics.

Since the minor radius \( a \) is always fixed in a particular tokamak, and the magnetic field \( B \) almost always is, the \( \beta \) is maximized by maximizing the current in a device with a given \( \beta_0 \). However, there are limits on the current also, limiting the beta to typically the range of a few percent up to perhaps 4 percent.

The other dimensionless parameter characterizing tokamak discharges is the \( q \). In the simplest case where the plasma cross section is circular in the poloidal plane, the \( q \) is simply defined as

\[
q(r) = \frac{r B}{R B_\theta(r)} \tag{3}
\]

where \( B \) is the major radius in meters, \( r \) is the minor radius in meters, \( B \) is the toroidal field, and \( B_\theta(r) \) is the poloidal, both in Teslas. Note that as the current increases, so does the poloidal field, and therefore \( q \) decreases. Note that \( q \) is dependent on minor radius and varies across the plasma cross section. If the cross section is not circular, \( q^{-1} \) is the number of loops the poloidal field makes in the poloidal plane when the toroidal field goes around once.

While \( B \), the toroidal field is relatively constant, the density and temperature generally have a strong variation with \( r \), the minor radius. For instance, shown in Figure 5 is an ion temperature, \( q \), and relative
Figure 6. Plots of the normalized beta in a sustained discharge in A: JT-60 [25], and B: D-3D, the time axis for D-3D [27] in B goes up to 4.5 s.

pressure plot from JT-60 [42]. Notice that these plots shows various regions, the main fusion core, and also what is called the pedestal, a region of the plasma near the separatrix at the edge, and having much lower temperature and density than the main plasma. Notice that in the core plasma, the ion temperature is a rather peaked function of minor radius. The $q$ has a strong variation with $r$, much of it coming from outside the main heated and current carrying plasma. However Figure 5 shows that at the edge of the hot plasma, the $q$ is about 3.

The tokamak is what one calls a two-dimensional configuration (a purely cylindrical plasma has no bootstrap current) [43]. Both JT-60, D-3D and other tokamaks have been successful in running quiescent discharges in fusion relevant regimes as long as their power supplies and or volt seconds have held out. Recall that neither of these use superconducting magnets, so the magnetic field requires an enormous amount of power. Figure 6 show plots of $\beta_n$ for a long (i.e. 30 second) discharge in JT-60 [25, 26] and a long (5 s) discharge in D-3D [27].

Another problem the tokamak confronts is disruptions; this is the sudden release of the plasma and poloidal magnetic field energy in some uncontrolled manner. A particular potential problem is that a portion of this energy might reside in a decoupled high current, high energy circulating electron beam. This could be generated if the current is driven Ohmically initially, when the density is low. If this beam is present when the plasma disrupts, there is virtually nothing that can stop the energetic electron beam before it hits a wall somewhere, and does great damage. Tokamaks have had a long history of being damaged by disruptions, especially if the damage mechanism is the electron beam, where even one Joule dumped on the wall has, on occasion, done serious damage.

In JET, about 10 MJ of plasma energy (about 5 pounds of TNT) can be released; the poloidal field energy is also released so typically this 10 MJ becomes about 20. A great deal of progress has been made in avoiding disruptions, as we have seen, JT-60 has demonstrated disruption free operation for 30 s, the maximum time of their pulse power, in fusion breeding relevant regimes. However just because the tokamak has run disruption free this long does not mean the problem has been solved, a fusion reactor after all, has to run disruption free for months or years. JET has published research on what is called the disruptivity, the frequency of disruptions [44] (in inverse of the time between major disruptions) as a function of various parameters. Figure 7 shows the disruptivity as a function of the reciprocal of $q$ and of the density as a fraction of the Greenwald limit (more on this limit shortly).

While the disruptivity looks low ($\sim 10^{-2}$) for sufficiently low current ($i.e. q^{-1}$) and density, we note that a disruptivity of $\sim 10^{-2}$ still means a disruption every 2 min; obviously intolerable for a reactor.

Note also that ITER has a stored magnetic energy, at the design field of $\sim 5$ T, of about 7000 Megajoules, or nearly the energy of about a two-ton bomb. Large ITER would have about twice the stored energy. If the $\beta$ is about 3 %, assuming it maintains a normalized $\beta$ of about 2.5 as has been achieved (more later), the plasma energy is about 200 Megajoules. However, the energy available for a disruption also includes the poloidal magnetic field energy, which is roughly equal to the plasma energy. Hence the energy available for a disruption is about 400 MJ, or the energy of a 200-pound bomb, more than the energy in the warhead of the Exocet missile that sank the Sheffield cruiser in the Falkland Island war. It is an enormous energy to release in the confined space of ITER.

But that is only not the only risk. If the disruption, or anything else, should generate and uncontrolled quench of the superconducting magnets, the energy released would be about that of 2 tons of TNT, and this would be enormously destructive. An uncontrolled quench did occur in CERN a few years ago, putting the machine out of commission for over a year. Yet CERN is in a tunnel 10’s of km in circumference. An uncontrolled quench in the confined volume of ITER would almost certainly destroy the building and much else. Many measures are taken to prevent uncontrolled quenches, and they occur only very rarely, but they can, and have occurred [45]. ITER explores an entirely new range of energy. Its 5 T field has a stored energy of about 10 MJ/m$^3$, so in a 700 m$^3$ machine that

Figure 7. Disruptivity of JET as a function of $q^{-1}$ and fraction of Greenwald density [44].
2.4. L, H and super H modes of tokamak operation

In the earliest days of the tokamak research, the confinement time was regarded as unsatisfactory. There were a variety of phenomenological formulas for the confinement time, as a function of various parameters; the details are unimportant to present here. However, in 1982, the ASDEX tokamak in Germany found that with some increase in beam heating power, the confinement time began to increase, roughly by about a factor of 2 in the best circumstances. Every other tokamak has confirmed this. The original mode was called the L mode, for low confinement; the new mode was called the H mode for high confinement. The numerical multiplicative factor characterizing the improvement over the original formula for confinement time was defined as H, and is typically about a factor of 2.

Figure 8 from [24] shows a plot of the H factor for JT-60 as a function of density divided by the Greenwald density (more on Greenwald in the next section). Clearly, as the density approaches the Greenwald limit, the H value decreased.

Virtually all planning of future tokamak operation, especially for ITER, assumes H mode operation.

More recently, two American tokamaks, Alcator (as its final experimental campaign before being closed down) and D-3D discovered a new mode called the super H mode [27], which once again roughly doubled the confinement time. The key was increasing the pedestal density and temperature, which they did both by exploiting code predictions, and finding a serpentine path in parameter space which would get them there.

To envision an extremely simple model for the super H mode, imagine that the energy transport is dominated by some gradient driven mode, for instance the trapped electron mode, which, in certain temperature regimes, has a growth rate proportional to \(-dT/dr/T\) (the temperature gradient is assumed negative), the inverse temperature gradient scale length. Let us also assume that there is some stabilizing feature, for instance shear of ion Landau damping, which might stabilize the instability or reduce its growth rate. Denote the sum of these as \(\Omega\). Let us also envision that there is another parameter, called \(\Pi\), which in some way measures how much above instability threshold the profile is. Note that if \(\Pi = 0\), the profile is at marginal stability. In any case, the equation for the temperature profile is

\[
\frac{dT}{dr}/T = -\left[\frac{1}{\Omega} + \frac{1}{\Pi}\right]
\]

Notice that Eq. (4) is only an equation for the relative temperature profile. To the get actual temperature profile, the temperature must be anchored at one point. Let us assume that this point is at the edge of the pedestal. In that case, the higher the pedestal temperature, the higher the central temperature.

This is then a greatly oversimplified insight into why the higher pedestal pressure can have a great effect on the temperature profile. Perhaps it is worth pointing out that 40 years ago, the author developed a theory for tokamak profiles and energy confinement based on these very points [48].

Experiments show that not only does the plasma perform better in the super H mode, the profiles are also much less peaked. Shown in Figure 9 are temperature and density profiles from both Alcator and D-3D in the super H mode, as well as the magnetic surfaces. The picture is taken from [27], which is published open access.

2.5. Conservative design rules

Conservative design rules are well-known constraints on tokamaks operation; they are not controversial, they have been known for years, and have been well confirmed experimentally. They are discussed in detail in [6, 7, 9]. Furthermore, more recently they have been discussed (under a different name) by Freidberg et al [49]. However, the tokamak community has been loath to put them all together and see that they provide serious constraints on what a tokamak reactor can, and cannot do.

We will do things a little differently than in Refs [6, 7]. There, a parabolic profile was assumed for the density and temperature going out to the limiter or separatrix radius. Here we will assume that the super H mode can be made even more super, and assume a flat density and temperature profile, at an ion temperature of 16 keV, the temperature for maximum fusion in a pressure limited plasma. Other than that, the rules and calculations are as in Refs [6, 7].
First of all, the q at the plasma edge, must be greater than 3, otherwise tearing modes will be excited [50, 51, 52].

Second the normalized beta must be less than 2.5, otherwise ballooning modes will be destabilized [53, 54].

Third the average density must be less than 0.75 times what is called the Greenwald limit. This is an empirical law, but tokamaks have always obeyed it. The Greenwald limit is $n_G < \frac{2I}{\pi a^2}$ [55, 56].

The consequence of violating these limits is usually a major disruption, something intolerable in a functioning reactor.

These limits have been well confirmed experimentally. Figure 7 showed the disruptivity of JET as a function of both $q^{-1}$ and $n/n_G$. Notice that there is a large jump in disruptivity just where conservative design rules would predict. Furthermore, data from JT-60 shown in Figure 8 shows that even without disruption, the plasma loses confinement as one approaches the Greenwald limit. JT-60 has some even more interesting and useful data. An example of such a long shot was shown in Figure 6. Shown in Figure 10 are the statistics of such a large number of shots [22, 23, 24, 25, 26]. The hollow squares are shots that are long lasting, the solid squares represent shots that terminate abnormally and early. Clearly the conservative design rules $\beta_n < 2.5$ and $q > 3$ are confirmed here. Of course where $\beta$ is proportional to $I$ (or $q^{-1}$), those points on the left are of most interest.

Now let us see what this means for neutron power production. In Refs. [6, 7], the maximum power was calculated by calculating the maximum current from the minimum q ($=3$) at the plasma edge. However, calculating this minimum q is only simple for circular poloidal cross section. Furthermore, even for circular cross section, the position of the plasma edge could be uncertain as Figure 5 showed. Here we use...
a slightly different approach. Assuming that the maximum stable current is given, as it nearly always is for the particular tokamak, and that it is low enough that the profile is tearing mode stable, we can get the average beta by assuming a maximum βN of 2.5. For all the devices we have been considering, the maximum beta calculated this way is less than 4%.

In Table 2 is a list of parameters of D-3D, TFTR, JET, JT-60, ITER, Large ITER, and ARC, a high field tokamak proposed by MIT [57]. Where the B, I, and a are all known, the β can easily be calculated assuming that βN = 2.5, the maximum value. As we can see, the maximum value of β is just under 4% for ITER. To make the most optimistic assumptions possible, let us assume that the β is 0.04. Furthermore, for a pressure limited plasma, we have seen that the plasma temperature for maximum fusion is about 16 keV. We assume here that the electron and ion temperatures are equal, as one would expect for a large reactor. Using Eq. (1) and setting β = 0.04 and T0 + Ti = 32, we find that n = 3.13 × 10^{19} B^2. Assuming a DT plasma with equal densities of D and T, we find

\[ n_D = n_T = 1.6 \times 10^{16} B^2 \]  

(5)

Using the reaction rate at 16 keV, \( <\sigma v> = 3 \times 10^{-22} \), we find that the number of neutrons per cubic meter per second is 7.3 × 10^{23} B^4. Since each neutron has an energy of 2.2 × 10^{−18} Mega Joules, the neutron power is 1.6 × 10^{23} B^4 Megawatts per meter cubed. To get the total power, one simply multiplies by the tokamak volume, 2π K^2 R^5 A^2, where K the vertical elongation of the assumed elliptical cross section. TFTR has circular cross section, but every other tokamak listed in Table 2 is vertically elongated, and we have assumed a K of 1.6 for each of these.

Hence, conservative design rules predict a maximum power for a tokamak of

\[ P_{\text{MW}} < 3.2 \times 10^{-3} \pi^2 K R^3 A B_4 \]  

(6)

This maximum power is listed in the column in Table 2 as \( P_{\text{cdr}}(\beta = 0.04) \). The actual neutron power of TFTR in the hot ion mode, and JET in the thermal mode is also listed. Furthermore, the design power of ITER, Large ITER, and ARC are also listed.

It is likely that in the hot ion mode, TFTR has a very nonthermal ion distribution, and many of the neutrons are produced by beam plasma reactions. In a large thermal reactor, even if heated with neutral beams, this is unlikely to occur, as the beam slowing down rate greatly exceeds the beam ion fusion rate. In any case, in TFTR, the hot ion mode does not seem to be able to access the high power possible in a thermal plasma. Notice the TFTR, with its higher magnetic field than JET, and the B^4 scaling of power would seemingly be able to achieve considerably higher power. Unfortunately, the machine was taken down before this advantage could be further investigated and possibly exploited.

While the hot ion mode is almost certainly not a thermal, Maxwellian plasma, it could be possible to exploit non thermal plasmas in other ways. If the fusion alpha particles can be convinced to heat the ions, rather than electrons, this would increase the fusion rate. This has been explored by Fisch et al [58, 59] and is called alpha channeling. Some rf is injected into the plasma in such a way as to transfer energy from the alphas to the ions. If the ions can be maintained at twice the electron temperature, i.e. \( T_i = 16\text{keV} \) and \( T_e = 8\text{keV} \), then at the pressure limit, the electron density would be increased by a factor of 4/3, and the fusion power by 16/9.

However, the conservative design rules give rise to a real limit on neutron power. For the JET thermal plasma, the conservative design rules predict more than twice the actual neutron power, and both the design powers of ITER and ARC are well under what conservative design rules would allow. However, play with Eq. (6) for the neutron power any way you want; it is difficult to see how a tokamak can ever be an economical stand-alone fusion reactor as long as tokamaks are constrained by conservative design rules, as they have been for their entire existence.

2.6. Pure fusion’s scientific dilemma

To see pure fusion’s scientific dilemma, let’s stipulate the best possible outcome from ITER. Say it achieves Q~10, producing 500 MW of fusion power and had the plasma heated and current driven by 50 MW of beams and/or microwaves soon after 2035. While ITER is an experimental device, not a power plant, let us imagine a power plant having its parameters. Since electricity is typically produced with an efficiency of ~1/3, the device would produce 170 MWe. However, it needs 50 MW to drive it. But beams and microwaves are not produced with 100% efficiency either, again 1/3 is a better estimate, so 150 MWe is needed to drive the tokamak, leaving all of 20MW for the grid! Of course, one could calculate

Table 2. The columns are the particular tokamak, The magnetic field in Teslas, the current in MA, the minor radius along the horizontal plane in meters, the β, assuming \( \beta_N = 2.5 \), the neutron power as predicted by conservative design rules (Eq. (6)), and the actual or design power.

| Tokamak   | B   | I   | A   | B   | \( P_{\text{cdr}}(\beta = 0.04) \) | \( P_{\text{a or d}} \) |
|-----------|-----|-----|-----|-----|--------------------------------|-------------------|
| D-3D      | 2.2 | 2   | 0.7 | 0.032 |                                |                   |
| TFTR      | 5.6 | 2.7 | 0.9 | 0.013 | 63                             | 10                |
| JET       | 3.5 | 4.8 | 1.25| 0.03 | 11                             | 4                 |
| JT-60     | 4   | 2.75| 1.1 | 0.016 |                                |                   |
| ITER      | 5   | 15  | 2   | 0.038 | 800                           | 500               |
| ARC       | 9   | 8   | 1.1 | 0.02 | 1400                          | 300               |
| Large ITER| 5.7 | 20  | 2.8 | 0.031 | 3600                          | 1500              |

\(^a\) For TFTR the power is given for the hot ion mode

\(^b\) For JET the power is given for the thermal mode.
a higher estimate by stipulating higher efficiencies. In fact, higher efficiency power plants have been designed, but is a question of a tradeoff between their cost and their higher efficiency Depending on a new, higher efficiency power plant just to make fusion viable does not sound like a very good argument for either fusion or the new power plant. Up to now, 1/3 is really about right and corresponds to nearly all experience. Furthermore, the total beam and microwave systems used to heat the plasma and drive its current, struggle to reach even that efficiency. ARC [52] assumes an electrical generating efficiency of 40% and a heating and current drive power efficiency of 42% to achieve its 300 MWe. However, drop these numbers to a more conservative estimate of 33% and its delivered electrical power drops very significantly. Also regarding ITER, given the size and cost of ITER, even if it were fully ignited and took no external power, its size and cost for 170 MWe would still render the device totally uneconomical. To make pure fusion economically feasible, first of all, ITER’s Q would have to be increased by at least a factor of 3 or 4. Secondly, the device would have to be made smaller and cheaper while increasing the power by at least a factor of 5 or 6 (a typical power plant has about 3 GW thermal and about 1 GW electric power). This would almost certainly mean that the tokamak would have to operate well beyond the limits of ‘conservative design rules’, something no tokamak has yet been able to do in over half a century of tokamak research and development. Finally, since the device would be both smaller and more powerful, the neutron wall loading, and the plasma divertor loading, would be at least an order of magnitude greater. These are not minor details; almost certainly, they would take decades and decades, and tens and tens of billions of dollars to achieve, assuming they could be achieved at all. At best pure fusion could be a 22nd century power source. But the need could well be for carbon free power much sooner. As we will see shortly, a fusion reactor having Large ITER, or even ITER like parameters is fine for an economical fusion breeder, if not for an economical pure fusion reactor. In addition to these scientific difficulties, pure fusion has other problems regarding time and dollars which the author has discussed elsewhere [9].

3. A brief discussion of nuclear matters

3.1. Thermal, fast, and energetic neutrons; and their nuclear reactors

There are three neutron energy ranges of interest to fission and fusion. From lowest to highest, these are thermal, about room temperature to about 1000 °C; fast, about 2 MeV; and energetic, about 14 MeV. To see the importance of the first 2, in Figure 11 is a graph of the collision cross section [60] for nuclear fission and nuclear absorption as a function of neutron energy for two target atoms, $^{235}$U and $^{238}$U. The curves are typical both for odd atomic weight atoms, $^{235}$U or $^{239}$Pu as well as $^{238}$U, these atoms are called fissile; and for even atomic weight atoms, $^{232}$Th, these are called fertile.

Notice that for $^{235}$U, the fission cross section is larger than the absorption cross section and has a strong maximum at lower energy. When a $^{235}$U undergoes fission, it breaks up into two lower Z atoms, called fission fragments, and also emits 2 or 3 neutrons with energy of about 2 MeV. The fission fragments together have energy of about 200 MeV. Notice that for $^{238}$U, absorption is the most important process at all energies below about 1 MeV. When a $^{238}$U absorbs a neutron, it becomes first a $^{239}$U. However, this is unstable to a double beta decay, first decaying once becoming neptunium, and then decaying a second time to become $^{239}$Pu. Notice that since this has an odd atomic weight, its collision cross section looks roughly like that of $^{235}$U. That it becomes a fissile atom. Since the $^{238}$U has the potential to become a fissile atom, it is called fertile.

Since fission produced neutrons are produced at ~2 MeV, in a thermal fission reactor, it is crucially important to slow down the neutrons. This is done via multiple elastic collisions with light atoms, for instance hydrogen, deuterium or carbon. The most common reactors, called light water reactors, or LWR’s, use the hydrogen in water to slow down the neutrons, and a then also use the water as a coolant for the reactor. The reactor is fueled with about 100 metric tons of uranium enriched to about 4% $^{235}$U mixed in with about 96% $^{238}$U, and about a quarter of this is replaced every year. This is a fuel mixture which is dilute enough in the $^{235}$U that there is no criticality risk, there is no proliferation risk without isotope separation. As the nuclear burn progresses, fission fragments accumulate in the fuel, some of which, like xenon, are strong neutron absorbers. After about a year, the reactor must be refueled. It discharges roughly 24 tons of $^{233}$U, now enriched at about 1% $^{235}$U; about 200 kg of plutonium, with a half-life of 24,000 years; higher actinides; as well 700 kg of various fission fragments, most of which are highly radioactive, but with half-life of 30 years or less. At the start of the year, the reactor burns only $^{235}$U, at the end it burns some combination of $^{235}$U and $^{239}$Pu which has been generated by the $^{238}$U. The breeding ratio of a standard LWR is about 0.6 [37].

This vastly oversimplified description gives an idea of the fundamentals of an LWR. There are of course numerous complicating aspects and it is well beyond the purpose of this paper to get into them. Among the many potential neutron loss mechanisms, one is the absorption of a neutron by a proton to form a deuterium atom.

One way to avoid this particular loss is to use deuterium (i.e. heavy water) instead of hydrogen as the moderator. This is the basis of the CANDU (Canada Deuterium Uranium) reactor. Since the losses are less than in the LWR, it can run on natural uranium, or even the depleted uranium once used by an LWR. As the losses are less, the breeding ratio of a CANDU is usually about 0.8. However, there is a similar, but less potent loss channel. The deuterium can absorb a neutron and become tritium. Thus, the Canadian nuclear program produces tritium as a byproduct. It is likely that in the earlier DT fusion experiments, before fusion produces its own tritium, the program will purchase the necessary tritium from the

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**Figure 11.** The fission (red) and absorption (green) cross section in barns (i.e. $10^{-24}$ cm$^2$) for a neutron collision with $^{235}$U and $^{238}$U as a function of neutron energy [60].
Canadians. However, this tritium does not accumulate very long in Canada. The tritium nucleus is unstable with a half-life of 12 years. It then decays to a stable \(^{3}\)He nucleus by a single beta decay.

Today there are about 440 thermal nuclear reactors worldwide. About 400 of these are LWRs, about 100 in the United States; and about of these are 40 CANDU’s, about 20 in Canada. There are also about 50 thermal reactors under construction and in the planning stage, mostly in Asia. In addition, new types of reactors like the molten salt reactor, discussed in the Introduction; the high temperature gas reactor, the pebble bed reactor, are on the drawing board. Most new concepts, whatever they are, are designed to be passively safe, that is, the reactor shuts down if the power goes off for any reason. However, whether the power is on or off, there is always a risk of excess heat and radiation from the accumulation of reactor produced fission fragments in the reactor.

Now consider the next neutron energy region, energy of around 2 MeV. Figure 11 shows that at these energies, \(^{235}\)U and \(^{238}\)U are not that different as regards fission. However, the fission cross section of \(^{235}\)U at thermal energies is much greater than the cross section of either at 1 MeV. The former has a fission cross section of about 200 b at a neutron energy of 0.1 eV (i.e. \(1000\) °C), while the latter cross section at 1 MeV is about 1 b. However, a carefully designed reactor with fast neutrons can be made to work. In fact, the decay from fast neutron fission produces slightly more neutrons than does fission with thermal neutrons. Hence a fast neutron reactor can be run in a breeder mode of producing additional fissile material from the fertile background than it uses. As mentioned in the introduction, it would take two breeder reactors at maximum breeding rate to fuel a single LWR of equal power. It is also important to note that a fast neutron reactor can operate in a pure burner mode. In this mode, it can burn any actinide, \(^{232}\)U or \(^{238}\)U; it makes little difference.

A fast neutron reactor is much more complicated and expensive than is a thermal reactor. Since the fission cross section is so low, the neutron travels further between collisions, so the fuel mass has to be much larger. That is, it takes much more fissile material to start the fast reactor than it does to start a thermal reactor. Furthermore, since these fast neutrons cannot be allowed slow down as they collide, it is necessary to use as coolants, as well as any other material a fast neutron might interact with, only materials that the neutrons hardly interact with. The choices are few, basically they are sodium and lead.

There are several fast neutron reactors that have been built, and so far they have used liquid sodium as the coolant. These are Super Phenix \([37]\) in France, and the Integral Fast Reactor, IFR, in the United States \([60, 61, 62, 63, 64, 65]\). The British are now in the process of constructing a 600 MW IFR, which they call PRISM, for the purpose of treating their large plutonium inventory \([66]\). Further information on the thermal and fast neutron reactors are given in \([7]\), as well as in their references, as well as in many standard references on nuclear reactors such as \([37]\).

Although fast neutron reactors were designed principally for fuel production, not burning actinide waste, as we will see, a fusion breeder fueled economy envisions an important role for them as actinide waste burners. One important fusion ‘waste’ product, \(^{239}\)Pu has a half-life of 24,000 years. Quotation marks are used because while viewed as waste, \(^{239}\)Pu is actually valuable fuel for a properly designed reactor. If it takes 20 half-lives to decay to a safe level, this means that storing it in say a Yucca Mountain, would impose an obligation on our decedents for the next 20,000 generations (!) to care for this incredibly dangerous stuff, stuff which might be produced for only a short period of total human history. One could argue that imposing this burden on our descendants is simply immoral.

However, this \(^{239}\)Pu, and other higher actinides, are perfectly good fuel for a fast neutron reactor. Furthermore, unlike a thermal reactor, (as in the French reprocessing program) it just burns the actinides, it does not ‘burn’ them and produce still more. It is unlikely that The British with their PRISM reactor are thinking of ‘burning’ their plutonium ‘waste’. PRISM is designed for 600 MWe; it takes 1GW\(e\) a year to burn a ton of nuclear material, and they have 100 tons of stored plutonium, which they propose to ‘treat’ in 5 years. Likely by ‘treating’, they are thinking of reducing the proliferation risk of the plutonium by mixing it with other non-fissile plutonium isotopes and/or mixing it with highly radioactive fission products so as to reduce the proliferation danger. But whatever they are doing, they are making a start on an important aspect a nuclear economy.

We now consider the third energy range, energetic neutrons around 14 MeV produced by fusion. At this energy, the energetic neutrons can produce additional neutrons, called spallation neutrons, depending on the target. Figure 12 is a plot of neutron cross section for producing one additional neutron (denoted \((n,2n)\)), or two additional neutrons (denoted \((n,3n)\)) in a lead target, \([60, 67, 68]\). Notice that a 7MeV neutron is necessary for the former, and a 15 MeV neutron is necessary for the latter reaction.

Of course, lead is not the only neutron multiplier. Others are \(^{239}\)U, \(^{232}\)Th and Be. Table 3 gives the minimum neutron energy for neutron multiplication for each, as well as the number total number of neutrons produced by a 14 MeV neutron as it slows down in this material \([17]\).

In a fusion reactor, one neutron is needed to produce the tritium. However, since the fusion reaction produces only a single neutron, some neutron multiplication is necessary even in pure fusion so as to cover the inevitable losses. If additional neutrons are required, as for fusion breeding, a well-designed blanket can produce perhaps as many as 3–4 neutrons for each fusion reaction.

### 3.2. \(^{235}\)U, \(^{237}\)U, \(^{239}\)Pu, \(^{238}\)U and \(^{232}\)Th

In any thermal reactor there are three possible fissile materials (those listed in the title with odd number) and 2 possible fertile materials (those listed in the title with even number). In many ways, they all act the same, but there are important differences, especially as regards proliferation risk and radioactive waste.

Light water reactors use \(^{235}\)U as the fissile material (with an enrichment of ~4%) and \(^{239}\)U as the fertile material. As such, the raw fuel provides maximum protection against proliferation, quite a bit of isotope separation would be required before enough \(^{235}\)U would be available to produce a nuclear weapon. However, since the atomic number of the \(^{238}\)U is high, it absorbs neutrons, first to produce \(^{239}\)Pu, and then it absorbs more to produce the higher actinides, americium, etc.

If \(^{232}\)Th were the fertile material, since it has a lower atomic weight, it has to absorb more neutrons to reach the various plutonium and americium isotopes. The thorium reactors once they got started would use \(^{233}\)U as their fuel, since that is what the thorium breeds. Secondly \(^{239}\)Pu has a much higher absorption cross section than does \(^{235}\)U. Figure 13, from \([60]\) is a graph of the neutron absorption cross section for the various reactor relevant elements.
Notice that as regards the two possible fertile materials, at low energy, $^{232}\text{Th}$ has about twice the capture cross section as does $^{238}\text{U}$. This means that thorium has an easier job producing $^{233}\text{U}$ than does uranium in producing $^{235}\text{Pu}$, so using thorium as the fertile material ought to give a higher breeding ratio than $^{238}\text{U}$. Also note that $^{233}\text{U}$ has an absorption cross section about a factor of 2 or 3 times lower than $^{235}\text{U}$, and about an order of magnitude lower than $^{239}\text{Pu}$. Therefore, a reactor with $^{233}\text{U}$ as the fissible material, and $^{232}\text{Th}$ as the fertile material will produce many fewer high actinides in its waste stream. Hence a reactor with $^{233}\text{U}$ as the fissible and $^{232}\text{Th}$ as the fertile material would seem ideal.

However, this is not ideal for at least one very important reason. The raw fuel is an enormous proliferation risk, as the fissile uranium and fertile thorium can be easily separated. In fact, the one reactor the US produced using this fuel and fertile combination [41] had as its goal submarine propulsion. Obviously, proliferation was hardly a consideration, where the fuel would have been in the hands of the Navy and on submarines. Ultimately this approach was abandoned and submarine reactors use $^{235}\text{U}$.

In his article, Bethe [19] suggested a compromise:

“In a future advanced converter reactor the fissile material may be $^{233}\text{U}$ and the fertile a combination of $^{232}\text{Th}$ with some $^{238}\text{U}$. (The latter is added so that chemical separation of uranium from thorium would still leave the uranium unsuitable for bomb manufacture.)”

In Bethe’s proposed compromise, the breeding ratio would be less than if the fertile material were pure thorium; there would still be actinides in the waste stream, but less than if the fertile material were pure $^{238}\text{U}$. Furthermore, while there would be proliferation protection from the raw fuel, there would be less than if the fertile material were pure $^{238}\text{U}$.

There is one other important advantage to using $^{233}\text{U}$ as the fissile material. Inevitably as it is produced, some $^{232}\text{U}$ creeps into the mix; in fact, Moir [16, 69] has discussed a scheme where this amount could be engineered with some precision. The advantage of this is that $^{232}\text{U}$ is not only radioactive, but it has a high energy gamma in its decay chain. This means that the raw fuel could only be handled remotely, making it extremely unlikely that a terrorist group or other non-state group could steal this fuel.

In the next section, we will briefly summarize the use of fusion neutrons to breed fuel for thermal reactors. As we have seen here, there are a number of possibilities. We consider only a single one, that is producing $^{233}\text{U}$ to mix with $^{238}\text{U}$ to fuel LWR’s, although we recognize that ultimately this might not be the optimum choice. However, it does have certain advantages. First of all, it fits in best with today’s nuclear infrastructure. Balancing all the advantages and disadvantages of a particular reactor, the nuclear industry world-wide, has decided almost exclusively on LWR’s. Secondly this fuel mix provides maximum proliferation protection. Thirdly, this is the most difficult task for fusion breeding; if fusion breeding can accomplish this, then it can certainly achieve any of the lesser tasks (i.e. fueling thermal reactors with higher breeding ratio).

### Table 3. The minimum neutron energy for neutron multiplication in a variety of materials, and the total number of neutrons produced by a 14 MeV as it slows down in that material. Notice that a fusion neutron might lose as little as 2 MeV in generating a spallation neutron from beryllium. Hence it might be possible for a single fusion neutron, with an appropriately designed blanket, to breed still more neutrons in a blanket containing beryllium.

| Element | $^{232}\text{Th}$ | $^{238}\text{U}$ | Be |
|---------|-----------------|-----------------|----|
| Minimum neutron energy | 7.5 | 6.5 | 9 | 2 |
| Number of neutrons produced | 1.7 | 2.5 | 4.2 | 2.7 |

![Figure 13. The neutron capture cross section for 6 nuclear relevant elements as a function of the neutron energy [60].](image)
dollars later, assuming it is possible at all. All one needs to do is go from ITER to Large ITER, a machine which has already been designed. If a fission reactor with a higher breeding ratio is used, even ITER might be sufficient to fuel 5 advanced reactors.

While this author is no expert on the economics or finance of the nuclear industry, References [7, 9] give a very rough estimate of the cost of fusion bred $^{233}$U of $\sim 1$–3 cents per kilowatt hour. Mined uranium fuel now costs $\sim 0.5$–1 cent per kilowatt hour. Either cost is very little compared to the total cost of nuclear power.

4. The energy park

Fusion breeding envisions an energy infrastructure called ‘The Energy Park.’ We give a very brief summary here. More details of the energy park are given in the linked references ([7] and [9], as well as in [4, 5, 6] and [8]).

In the energy park, there is one fusion breeder fueling a large number of thermal fission reactors, and one fast neutron reactor burning the actinide discharge of these reactors. For our example we will consider this fusion reaction to be based on the parameters of Large ITER, the fission reactors to be 1GWe LWR’s, and the fast neutron reactor to be a 1 GWe IFR. As a pure reactor Large ITER was designed to generate about 1.5 GW of neutron power. Adding in the alpha particle power, and assuming an M of somewhat less than 2, the total thermal power of the fusion breeder is about 3 GW of thermal power, or 1 GWe.

Around the reactor there is one or more regions of the blanket in which the lithium and thorium flow through the fusion neutron stream. They could either be pebbles carried along with a flow, or could be dissolved in the flow, most likely a molten salt. The salt FLiBe has been discussed for this purpose. Thorium, protactinium and uranium are all soluble in it. The input to the flow has thorium and lithium dissolved, the output has some protactinium and tritium dissolved in it, and this is removed from the exiting fluid. A rough schematic, where the lithium and thorium enter the blanket separately is shown in Figure 14, taken from [16].

Then the 1.5 GW of neutron power breeds about 15 GW of nuclear fuel, or about 5 metric tons of fissile material (we consider $^{233}$U). In other words, it supplies power to 5 1GWe LWR’s. The fusion breeder produces $^{233}$U but this is immediately diluted to about 4% with $^{238}$U so there is no proliferation or criticality risk from the fuel.

Now consider the LWR’s. After about a year, 25 tons of used fuel is discharged from each reactor. In the spent fuel are about 200 kg of plutonium and other actinides (that is elements with atomic number greater than 92), and about 800 kg of fission products. These are elements of intermediate atomic number, for instance cobalt 60, strontium 90, barium 137 etc. These typically have half-life of 30 years or less [37]. The 24 metric tons of $^{238}$U, now enriched at $\sim 1\%$, mostly just goes along for the ride.

As the wastes from these LWR’s are discharged every year, the transuranic elements, those with atomic number greater than 92, principally plutonium and americium, but others as well (i.e. those with proliferation risk) are separated out and burned in a single fast neutron reactor of about equal power (1GWe), for instance the integral fast reactor (IFR, or PRISM), which has been developed at the Argonne National Lab and is under development by the British. Note that 5 LWR’s produce just enough actinide ‘waste’ to fuel a single IFR, or PRISM reactor of equal power.

In the energy park, there is neither long-term storage, nor long distance travel of any material with proliferation potential; it is all burned or diluted in the park behind a high security fence. Only fission products, virtually all with half-life 30 years of less, would be retained there, for instance in cooling pools. Some of these fission products have commercial value and would be separated out and sold. The rest would be stored for 300–500 years until they become inert. This is a time scale that...
human society can reasonably plan for it. It is far different from storing for instance plutonium, in say Yucca Mountain, where one must be concerned with storage for half a million years (the half-life of 239Pu is 24,000 years). One could certainly make a strong case that it is immoral for the next few generations to saddle the next 20,000 generations with caring for stored plutonium.

Figure 15 is a schematic of the energy park. It is more than a dream, but much less than a careful plan.

Once the energy park has been developed, the world could decide whether it wished to continue with research into DD fusion. DD fusion is not much of an energy producer, but it is a very prolific breeder (the various reactions produce a total of ~4 MeV). The reaction breeds both tritium and 3He, both of which are excellent fuels for a pure fusion reactor. If the reactor can burn deuterium, it could surely burn DT or D3He. There is no need to separately breed any fuel (T); or to go to the moon for it (3He). Using deuterium from the world’s oceans, the DD reactor could provide power for millions of years. It is a genuine infinite energy source. However, the decision on whether to continue with research into DD fusion, or to just settle for DT fusion breeding and energy parks for the foreseeable future, is not ours to make. Large, great, great, great … grand children can decide this. Our generation could provide maximum help to them by developing fusion breeding now.

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