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Reprocessing free nuclear fuel production via fusion fission hybrids

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ABSTRACT

Fusion fission hybrids, driven by a copious source of fusion neutrons can open qualitatively "new" cycles for transmuting nuclear fertile material into fissile fuel. A totally reprocessing-free (ReFree) Th^{232}–U^{233} conversion fuel cycle is presented. Virgin fertile fuel rods are exposed to neutrons in the hybrid, and burned in a traditional light water reactor, without ever violating the integrity of the fuel rods. Throughout this cycle (during breeding in the hybrid, transport, as well as burning of the fissile fuel in a water reactor) the fissile fuel remains a part of a bulky, countable, ThO_2 matrix in cladding, protected by the radiation field of all fission products. This highly proliferation-resistant mode of fuel production, as distinct from a reprocessing dominated path via fast breeder reactors (FBR), can bring great acceptability to the enterprise of nuclear fuel production, and insure that scarcity of naturally available U^{235} fuel does not throttle expansion of nuclear energy. It also provides a reprocessing free path to energy security for many countries. Ideas and innovations responsible for the creation of a high intensity neutron source are also presented.

1. Preamble and introduction

A fusion fission hybrid (Hybrid) harnesses fusion-generated neutrons to greatly augment fission reactions [1], and thus, allows the implementation of nuclear fuel cycles that may not be readily accessible to fission-only systems. This paper has two main objectives; (1) an exposition of scientific explorations of ideas and innovations that may lead to a near term technically feasible hybrid, and (2) a discussion of the more significant and unique applications of the hybrid, in particular, in the arena of nuclear fuel breeding – that is breeding fissile fuel (Pu^{239} and U^{233}) from the naturally occurring fertile materials (U^{238} and Th^{232}) to feed the standard nuclear reactors (for example the light water reactors (LWR)).

Amongst the breeding cycles that we have investigated, a particular Thorium based fuel cycle that avoids reprocessing altogether (to be referred as Reprocessing-Free = ReFree), will be emphasized here. The ReFree cycle may be not only the most proliferation resistant of the known fuel producing schemes (including alternative fuel production via FBRs [2] and enrichment via centrifuges [3]), it is also found to be very efficient in the sense that a single hybrid would suffice to fuel ~3.5–4 LWRs of the same thermal power; such a high support ratio will translate into good economics.

Today, nuclear energy offers, perhaps, the most mature, tried and tested option available for supplying base-load carbon-free electricity. Driven by a variety of considerations, many countries are building, or are planning to build a large numbers of nuclear power plants. The primary reason for this activity may be simply to cope with the rapidly growing energy needs but the fact that nuclear power plants can directly replace base load coal power plants (whereas intermittent renewables cannot readily do so in the foreseeable future) may also have played a decisive role. In addition, several countries pursue nuclear energy as a means to become energy self-sufficient through nuclear fuel breeding.

Most commercial nuclear plants, likely to be built in the next several decades, will be predominantly [4] Light Water Reactors (LWRs) that are cooled and moderated by ordinary water and use enriched Uranium as a fuel. Since the percentage of the fissile isotope U^{235} is only ~0.7% of the natural occurring Uranium (most of it is U^{238} that is not fissile in the thermal neutron spectrum), the standard mode of nuclear power production extracts less than a percent of the total energy stored in Uranium [5].

Because of the scarcity of U^{235} (the only naturally occurring fissile isotope), the nuclear economy may, eventually, face fuel shortages, particularly, if nuclear energy were to substantially replace base load coal power over the next thirty to forty years. In addition, since a typical modern LWR is designed, and is being licensed for over 60 years of operation, (ultimate lifetimes could well be 80 or even 100 years for some designs), the resurgence and sustenance of nuclear economy in the next several decades will demand a guaranteed supply of fuel up to, and extending past 2100 – manufacturers would not build and utilities would not buy unless they are sure that adequate fuel will be available for the expected life time of the reactor. Fueling requirements for the

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lifetime of this new generation of reactors (past 2100), therefore, could become a serious issue in the much nearer term.

The fuel supply problem, in principle, is technically solvable within the realm of nuclear fission. There can be two straightforward approaches towards this end:

(1) The first approach will directly use the much more abundant U^{238}. Although the isotope U^{238} cannot fuel the typical LWR (because of its negligible fission cross section in the LWR neutron spectrum), it can be used as fuel in reactors in which the fission neutrons are not moderated (and cooled). Such fast spectrum reactors (FRs), that can directly utilize U^{238}, have been investigated for a long time in several countries. Commercial experience with FRs is limited but not zero. Unfortunately, to date, most commercial FBRs have had serious problems with reliability, and economics; they have also been prone to accidents. One must appreciate, however, that even if conversion to FRs were possible at some future date, it will create a totally new parallel FR-based nuclear economy. It will still beg the question of fuel supply needed for the fleet of current and future LWRs built to satisfy the need for near term de carbonization of energy production. One must note that for a long time to come, *almost all of the growth in nuclear energy sector will be based on thermal reactors that will need fissile fuel*.

(2) The fast reactor can be used in another mode, that is, as a breeder of fissile fuel – that is, as a producer of fuel in addition to the amounts it needs to maintain its own criticality. In this incarnation, it is aptly called a fast breeder reactor (FBR), and development of FBRs is very much on the agenda of many countries. From the two naturally occurring fertile isotopes – Uranium (U^{238}) and Thorium (Th^{232}), one may breed excess fissile fuels – Plutonium (mainly Pu^{239}) and U^{233} respectively – and keep any LWR fleet running almost endlessly. However an FBR based nuclear economy has excited much controversy; more pertinent objections may be summarized as: (1) in order to extract the fissile fuel to be sent to an LWR, copious amount of reprocessing and handling of Pu^{239}/U^{233} is required creating serious problems on the nuclear weapon proliferation front, (2) FBRs are not efficient as producers of excess fissile fuel; an FBR cannot even supply enough excess fuel for a single LWR of comparable power, (3) the time required to build and fuel a large number of FBRs (they require a large amount of initial fissile loading) could be long, possibly too long to meet the challenge of future growing energy needs or evolving shortfalls of natural fissile fuel for a fleet of LWRs and other thermal reactors, and finally (4) FBRs, especially with fuel reprocessing, have not been found to be commercially competitive with LWRs, at least so far.

One is thus forced to conclude that the FR (FBR) path may not be a very attractive solution for assuring a dependable and economic fuel supply for the steadily growing nuclear industry firmly based on thermal spectrum reactors. It is, then, prudent to explore other options – including the ones not wholly within fission – lest the fuel scarcity becomes the Achilee's heel of nuclear power sorely needed as a near time, low carbon replacement for coal. Demonstrating the workability of just such an alternative – fuel production using fusion fission hybrids – provided the theme as well as the prime motivation for this work.

The basic idea of a Hybrid is as old as nuclear reactors [1]. All these years, one could not quite marshal it to help fission because the intense fusion neutron sources, required for effective breeding, were not on the horizon. Recent advances in fusion research, strengthened by several new ideas and innovations spanning fusion, fission (and their coupling), however, have set the stage for the conceptual design of a technically credible Hybrid driven by a compact fusion neutron source (CFNS). Although the main purpose of this paper is to present a critique of the Hybrid as a fuel-breeder, some description of the CFNS-Hybrid will also be given.

It may be some interest to make a small digression here. A Hybrid, just like an FR, does have another incarnation; instead of being an excess fuel breeder, it can function as a direct fission reactor. What is different and important is that unlike a standard fission reactor (LWR or FR), the Hybrid can safely operate with very bad fuel. The fusion neutrons can provide the extra neutrons needed to burn the "difficult to fission" isotopes like the minor actinides. Since these minor actinides comprise most of the very long lived radio toxicity in the spent nuclear fuel (the so-called nuclear waste) coming out of a thermal-spectrum reactor, the Hybrid has unique capabilities as a waste destroyer. Once in action, the Hybrid can help rid the nuclear industry of the three cardinal drawbacks associated with nuclear power – the accumulation of very long lived hazardous waste, proliferation concerns and future fuel scarcity.

Fusion fission hybrids are highly complex systems in which a fusion neutron source and a fission blanket must be optimally coupled to produce the best results. In order to investigate the efficiency and efficacy of the Hybrid (working, for instance, as a fuel maker), we have done detailed neutronic calculations in a system that incorporates realistic geometries of the fusion and fission parts. Distinguishing qualitative features of our reference ReFree cycle may be summarized as:

(1) Because the ReFree cycle requires no fuel reprocessing, it has major proliferation advantages. Fuel rods of the fertile Th^{232} are exposed to neutrons in the hybrid to produce an appropriate small percentage of the fissile material. *These fuel rods are then taken, with no further modification, to an LWR and used as fuel*. In the thermal spectrum of water-moderated reactors (LWRs), this small percentage of U^{233} is quite sufficient for criticality, though in the faster spectrum of the hybrid breeder, the rods are strongly sub-critical. In fact the Hybrid is always run in a safe highly subcritical mode. Note that the fissile material is protected, at all times, by a strong radiation field from *all* fission products generated by fission reactions occurring in the hybrid during the breeding phase. In these "charged" rods, the fissile material is highly diluted and embedded in a very bulky matrix (many discrete, easily countable fuel rods) of fertile material. *The ReFree mode of fuel production is qualitatively different from the advertized FBR fuel cycle; the proliferation problems characteristic of the FBR cycle[2] are completely absent.* In fact, this method of fuel production, in several respects, has substantial proliferation advantages over the gas centrifuge enrichment technology that is used to make the enriched U^{235} fuel of today[3]. The fuel cycle components that have historically lead to proliferation – enrichment and reprocessing – are absent.

(2) A Hybrid, supplied with fusion neutrons, is far more efficient as a fuel breeder. A single Hybrid in the ReFree mode can support at least 6 times more LWRs than a comparable thermal power FBR, so relatively few hybrids would be required to sustain commercially proven thermal reactors. Two crucial advantages result from high support ratio: (1) the cost of fuel production is considerably reduced, and (2) the Hybrid does not fundamentally change the nature of the current, successful nuclear economy that will remain predominantly based on LWRs, and in the future may use thermal reactors on the drawing board that are cheaper and safer. The (advanced reactor) Hybrid, unlike the (fission only) advanced reactor FBR is only a perturbation (though with crucial impact) on the entire energy production system. Both the Hybrid and the FR have uncertain costs, but cost of the overall nuclear energy system is much less sensitive to the cost of a Hybrid.

(3) Fuel production using Hybrids might be ramped up more quickly than through FBRs for two main reasons: Far fewer Hybrids are required, and the Hybrids have no start-up fuel requirement. This can be particularly significant for a country like India (committed to building up an eventual Thorium economy) with very little Pu^{239} stockpile for initial loading of the FBRs.
(4) The ReFree cycles dispenses with the need, and hence the expense of building an enormous infrastructure for reprocessing and fuel re-fabrication. For example, the cost of the most recent major reprocessing facility, in Rokkasho-Mura Japan, was surprisingly high [4]. This may be particularly significant for the thorium cycles in which the fissile (U^{233}) is inevitably tainted by the radioactive isotope (U^{232}) making fuel re-fabrication both challenging and expensive.
(5) The neutron multiplication factor in the ReFree fusion-fission hybrid is very low ~ 0.3. This effectively eliminates the concern that unexpected fuel rearrangement can lead to a criticality accident, which is a safety issue in FRs.

Finally, in the ReFree Hybrid-LWR cycle, the utilization of Thorium is an order of magnitude higher than for today's Uranium fed water-cooled reactors translating into a highly extended lifetime for the given natural resources. For example, the high grade thorium reserves in the US, utilized in a ReFree breeding cycle, could fuel its entire present electricity supply for nearly a century; for India, it could be much longer.

It is also pertinent to remark that the Thorium utilization could be considerably extended if one were to add reprocessing to the breeding cycle, should that become necessary in the far future. It is worth emphasizing that Hybrid based cycles, tailored for this scenario, could reduce the weaponizable fissile throughput in reprocessing by nearly two orders of magnitude compared to conventional FR based approaches; this aspect of the hybrid cycles will be described in future work.

At this stage, one may wonder why should fusion, instead of assisting fission power production, not be directly harnessed for power production. In our opinion there is no contradiction between the two goals and they could and should be pursued simultaneously. The hybrid described here is based on a Compact Fusion Neutron Source (CFNS). In fact the creation of a neutron source of the CFNS variety is considered to be an integral part of the main fusion program (proposed by many as the component testing facility CTF). However, since the CFNS need not be a net power producer (the Hybrid, through fission, is), it is a much less demanding undertaking then a pure fusion power source that must be a strong net energy producer; many key technical requirements for the CFNS are much easier to satisfy than for a pure fusion power plant. We believe that, fortified by several recent innovations, a tokamak-based Hybrid could be realized in much less time and with less risk than a pure fusion power plant.

The subject of fissile fuel production via fusion fission hybrids has been quite thoroughly studied by various groups. In almost all of them, the need for reprocessing was automatically assumed. *To the best of our knowledge the ReFree fuel producing Thorium cycle, with a strong emphasis on minimizing proliferation risks (by avoiding reprocessing altogether), is unique.*

To determine the efficiency of fuel utilization in this unconventional fuel cycle, our detailed neutronic calculations and analysis follow the fuel rod from its Hybrid original breeding to its burning as a fuel in an PWR, recharging of the partially spent rod in the hybrid and re burning it in the LWR. Crucial computed parameters include the criticality of the fuel in the LWR, the neutron damage in the cladding, and the accumulation of neutron poisons during both the LWR and the hybrid phases. A full core model of a Pressurized Water Reactor (PWR) is used, with multiple zones of fuel rod lattices. (Details at the assembly level are omitted for simplicity – for example, control rod guide tubes and water tubes.) This simplified model should give a reasonable estimate of the burn-up possible with the fuel. We have used a conventional Pressurized Water Reactor (PWR) for these calculations, though we expect results would be fairly similar for a Boiling Water Reactor (BWR).

In the present paper, we recharge the fuel rod only once. However, our calculations indicate that the fuel rods maintain their "integrity" by the end of the cycle, and might be recharged several times. This would substantially increase the Thorium utilization efficiency, and also increase the support ratio, the number of comparable power PWRs than could be fueled by the hybrid. "Multiple charging" cycles will be pursued in future work.

In the ReFree fuel cycle, the fuel rods are fabricated out of natural thorium. Thus, the fuel rod construction process does not have to contend with any U^{232}, or other highly radioactive isotopes. This is a particularly significant advantage when breeding from thorium, where U^{232} is an inevitable byproduct – the highly penetrating (2.6 Mev) gamma rays from daughters of U^{232} can greatly complicate fuel fabrication. Eliminating the need to handle U^{232} while making fuel rods could be a major engineering plus for the Hybrid created ReFree cycle.

In order to attain a suitable fissile concentration in the fuel rods while avoiding reprocessing, the fissile breeding must be done in a fast neutron spectrum. We have selected gas cooling for the hybrid to minimize: (1) neutron moderation, (2) coolant interactions with a magnetic field, and (3) possible corrosion issues in the Hybrid. The choice of gas cooling (for example helium) reduces corrosion of the cladding in the hybrid. The gross volumetric power densities in the fission blanket of the Hybrid are several times lower than in an LWR, so gas cooling should be adequate. The parameters for the fusion components are also consistent with gas cooling.

For the fuel rods, we have adopted high chromium ferritic steel cladding that is being currently developed for high burn up fuels in both fast reactors and water-moderated reactors [6,7]. Such cladding should be capable of sustaining much higher neutron damage than traditional zirconium fuel rods, and should be compatible with the Hybrid-PWR ReFree cycle. Silicon carbide cladding, presently under development for high burn-up PWR fuels, is another a promising option. [8,9].

The fusion neutron source required for this fuel cycle can be rather modest-our reference Hybrid is driven by a spherical tokamak with 400 MW fusion power. Compared to a pure fusion power plant, the fusion source considered here has far more modest requirements in many areas. The technical requirements are very similar to those of fusion component test facilities that have been proposed for relatively near term implementation [10,11] In some important respects – they are even less demanding. The reference CFNS, presented here, is a copper based, non-superconducting spherical tokamak. This choice was predicated by the desire to have a compact, relatively low weight, self-contained, replaceable fusion module. These features are essential in order to make a workable Hybrid that could be engineered in near term.

Compactness and high power density (high neutron flux) are needed for economy as well as for good neutronic coupling between the fusion and the fission parts. A new magnetic geometry – the Super-X Divertor (SXD) [12–14] – had to be invented in order to handle the enormous heat and particle fluxes peculiar to high power density fusion devices; an intense but compact source must perforce have high power density. It must be stressed that without a divertor of the SXD caliber, it is difficult to conceive of a compact high power density fusion machine. The SXD geometry will be tested on the MAST tokamak [15] in UK in the next few years.

In Fig. 1 we show the architectural plan for a Hybrid with a removable/replaceable CFNS module. The replaceability option, that is, the provision that the fusion module can be vertically

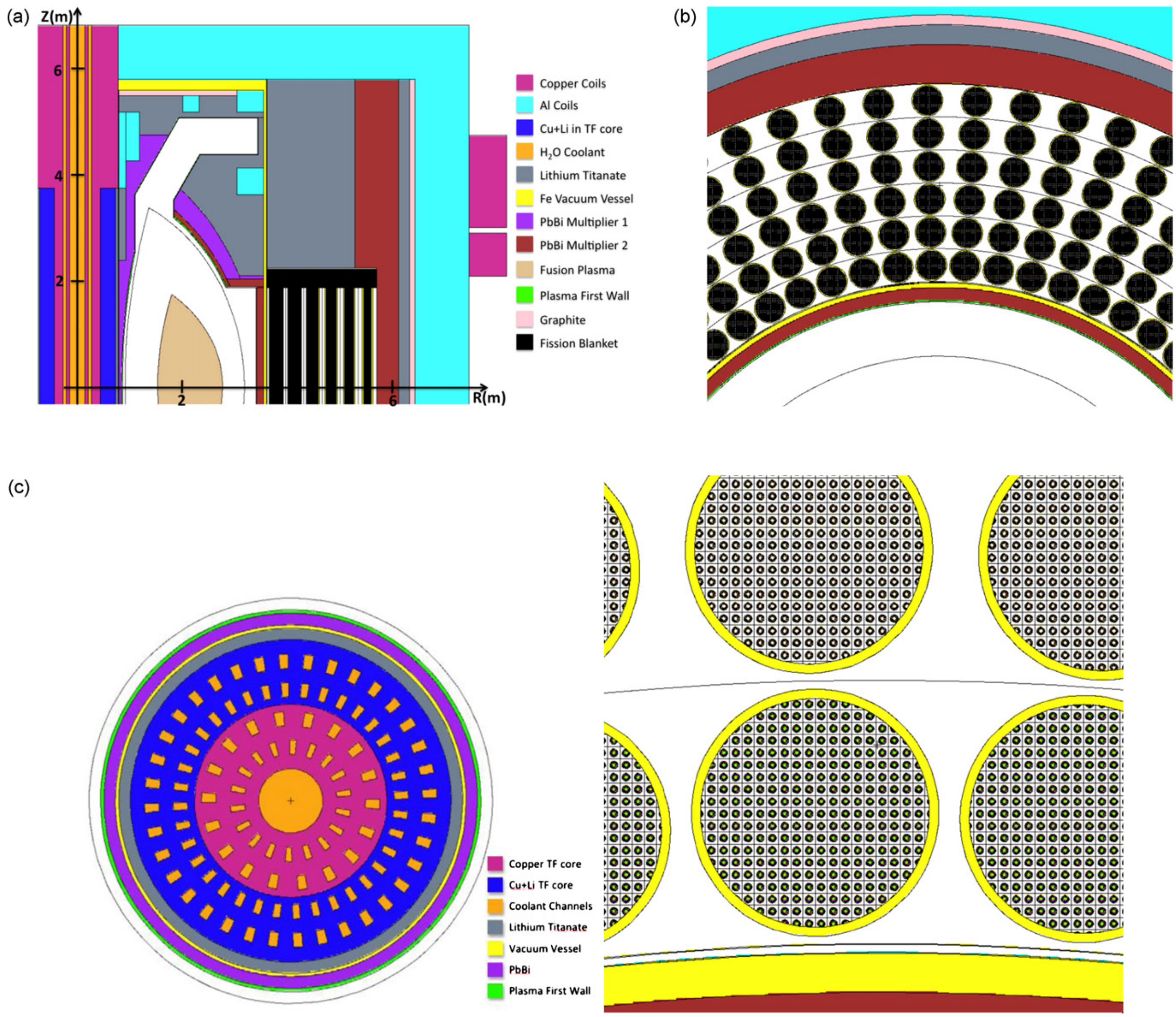

Fig. 1. (a) Cross sectional view of the hybrid. See Sections 2.4 and 2.5 for more details. (b) Top view of the fission blanket region. See Sections 2.4 and 2.5 for more details. (c) and (d) Clos-ups of the CFNS centerpost (left) and the fission blanket showing the fuel rods (right.) See Sections 2.4 and 2.5 for more details.

removed [16] and replaced by a duplicate, is a critical engineering idea that may allow us to beat the enormously difficult material constraints (damage caused by 14 Mev fusion neutrons) plaguing the entire enterprise of fusion power; the high energy fusion neutrons, because of their high cross sections for (n,α) reactions, cause much more material damage than fission neutrons. Designs for fusion devices have envisaged vertical replacement of components of comparable mass to the CFSN [16,17] (Fig. 1).

The replaceability option addresses another related, but equally important challenge – the challenge of a Hybrid obtaining high overall availability in a device driven by a fusion neutron source. Devising a fusion neutron source with very high availability is likely to be an extraordinarily difficult technical undertaking. Nonetheless, high availability is an obviously desirable requirement; one would want the fertile material to be exposed to neutrons nearly continuously. Having two separate CFNS units – one producing neutrons, while the other is in a remote maintenance bay being refurbished, provides a neat solution to this problem. Though the availability of an individual CFNS is somewhat less than 50%, the duo manages to (collectively) irradiate the fertile material much closer to 100% of the time. At intervals of about 1–2 years, nuclear plants typically shut down for periodic maintenance (such as reshuffling fuel rods). During this same time, the fusion module could be replaced. Some components of the fusion module (e.g. the first wall) might be replaced as frequently as every 1–2 years – a replacement time much shorter than possible in a Hybrid of conventional design, or a putative fusion power plant. For such relatively shorter times, the Hybrid components directly exposed to 14 Mev neutrons should likely be able to withstand the accumulated neutron damage with existing or near term materials. *The replaceability option may be the technical key to the near term realization of the Hybrid.*

A compact, low weight CFNS, that is self contained and physically separable from the fission blanket (located outside the vacuum vessel), is optimally suited for exercising the replaceability option. The self contained, physically separable CFNS module brings another additional advantage: failure modes in the fusion system are likely to have much less effect on the fission system. For example, we expect that off-normal events in the fusion module should not severely affect the fission blanket – which is likely not true for a traditional tokamak hybrid design, where fission and fusion components are intimately intertwined. This relative independence of

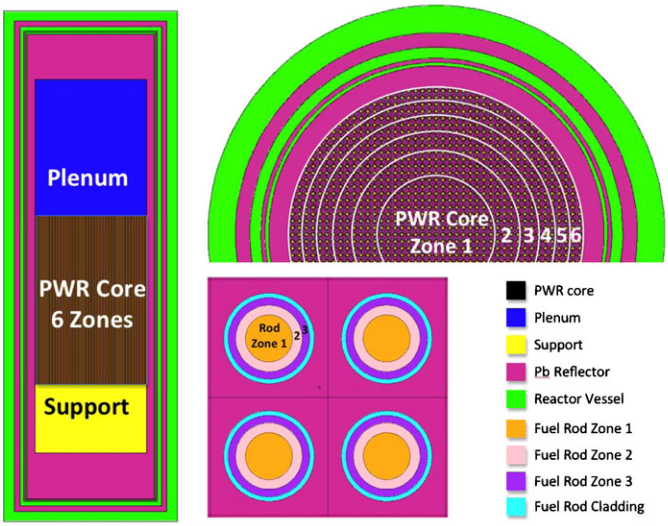

Fig. 2. Model of the PWR, showing the six radially concentric zones, and the three separate zones in each fuel rod. The plenum and support zones have a smeared volume fraction of stainless steel to simulate structure, and a steel core barrel around the core.

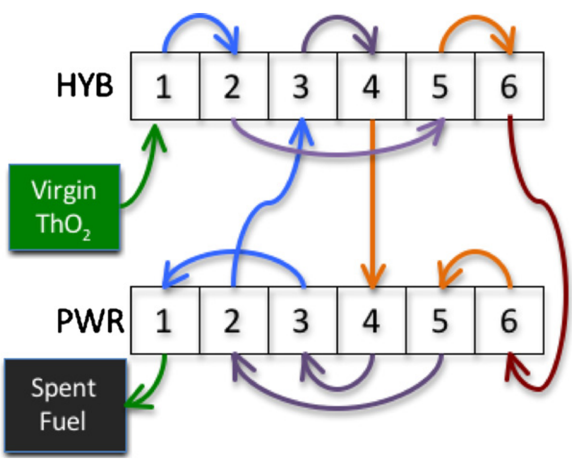

Fig. 3. Fuel shuffling pattern showing the path of fuel in the hybrid and PWR.

the fusion and fission aspects is very desirable to circumvent new modes of failure when two advanced and complicated technologies are combined.

Before the fusion neutrons get to the surrounding fission blanket, they must go through a neutron multiplier. In our design, we employ PbBi to multiply the neutrons via (n, 2n) reactions. This stage serves two important purposes:

(1) During the passage through the PbBi blanket, the neutron energy is rapidly reduced (by inelastic processes) to ~1 Mev; the peaked fusion spectrum (at 14 Mev) is transformed to a much more fission like spectrum. In this much lower energy range, cross sections for (n,α) reactions, the main reason for the intense material damage wrought by 14 Mev neutrons, are strongly reduced. Hence, most components of the CFNS, due to the mediation provided by the Pb–Bi multiplier, are exposed to a neutron spectrum that is much closer to a fission spectrum. The cross sections for parasitic losses are also greatly reduced. Hence, the neutrons can be transported through the vacuum vessel to the fission blanket with relatively low losses. Our neutronic calculations quantify these aspects. Structural materials under development for FRs can likely last considerably longer than materials exposed directly to 14 Mev neutrons, and would boost up the replacement time interval.

(2) The multiplier is an essential and integral part of the neutron source. Since for every fusion neutron, we need one to generate tritium for any self- sufficient fusion system, the only "extra" neutrons that we need to augment nuclear reactions (in the fission blanket) must come from the neutron generation in the multiplier.

In order to insure a containment boundary with minimum neutron damage, we will use the return leg of the toroidal field coil as such a boundary for both the fusion and the fission systems. This has some conceptual similarities to the ARIES ST fusion reactor design, where the TF coil return legs were used as the primary vacuum vessel [18], and hence as a containment boundary. It also has conceptual similarities to "guard containment" vessels proposed for gas cooled fast reactors [19].

Finally, before giving a detailed technical description we must put the Hybrid project in perspective.

For the Hybrid to make headway, it must perform (much) better than its fission only counterpart (FR, FBR), since it has additional costs and complexity. For the two primary applications, fuel breeding and waste reduction, it is possible to define a quantitative figure of merit – the so-called thermal support ratio – it is the number of LWRs of the same thermal power that an advanced reactor – the Hybrid or the F(B)R – can support. In the context of fuel breeding (the main thrust of this paper), the thermal support ratio could be measured by $S = P_{LWR}/P_{Hyb}$ where P_{LWR} is thermal nuclear energy produced in an LWR from bred fuel, and P_{Hyb} is the amount of thermal energy (fission plus fusion) generated in the hybrid to create that fuel.

Previous authors have shown that fusion breeders that strongly "suppress" fission are capable of obtaining high support ratios. In the breeding part of the ReFree cycle (free of reprocessing), we cannot fully "suppress" fission because we must breed a high enough concentration of fissile material so that the rods will be critical in a thermal spectrum reactor. And for such levels of concentration (not critical in the fast spectrum of the Hybrid), neutron physics limits fission suppression. Support ratio is also reduced due to the replaceable module design in which the fission blanket is not in intimate contact with the fusion system, but rather is significantly isolated from off normal events. Therefore, to gain the obvious qualitative advantages of the ReFree process, and of the replaceable module concept, a price has to be paid; thermal support ratio cannot be as high as in hybrid cycles with large amounts of reprocessing, and where the fission and fusion aspects are intimately interconnected; we estimate it to be 2–3 times less in the initial ReFree design offered here, but we also believe that future refinement can narrow this gap considerably.

Following this lengthy preamble, the rest of the paper is organized as follows. Section 2 is devoted to a description of the Hybrid system with 2.1 delineating general overall system features, 2.2 giving details about the invoked Pressurized Water Reactor model, and 2.3 describing the device as well as the plasma aspects of the CFNS. We address the engineering and technical challenges of the hybrid with 2.4 devoted to the fusion part, 2.5 describing the details of the fission blanket. In section 3, we describe the computational tools employed to work out the "neutronics" of the Hybrid. In Section 4, we describe the representative results of our neutronic calculations spanning both fusion and fission parts of the Hybrid – support ratios, material damage, tritium breeding etc. A short concluding summary is given in Section 5 and Appendix I–IV, respectively, give some details of the plasma model, current drive schemes for the CFNS, details of the material composition of the Hybrid parts, and magnet cooling.

2. Hybrid description

2.1. General device features and motivation

The principal new component of a Hybrid is the neutron source, CFNS. For a variety of Physics and Engineering considerations, we have chosen the compact spherical tokamak configuration as the basic building block of the CFNS. Our decision was, partially, influenced by the fact that the spherical tokamak was the choice of several groups working on building near term Component test facilities (CTF) [10,11] that have a lot of commonality with the CFNS.

We have chosen water-cooled copper–aluminum magnets to provide the toroidal field for the CFNS; the center-post will be all copper, whereas the return legs will be aluminum (similar to the ARIES-ST fusion reactor design [18]). Copper magnets, in distinction to superconducting ones, will allow an enormous reduction in size and weight for the fusion module, so that it can be replaced as a unit for rapid maintenance. Following numerous reactor designs, the structural material will be ferritic steel [17,20–23], and all nuclear components will be gas (helium) cooled [17,21–23].

As mentioned earlier, behind the first wall, there is a surrounding neutron multiple blanket composed of material with a substantial (n,2n) cross section for DT neutrons. Candidate materials for this include Be, and lead based eutectics such as PbBi and PbLi. Although we have studied Be to some extent, we will concentrate wholly on the PbBi option in this paper. Nuclear operating experience with it as a coolant for fast spectrum fission reactors, lends considerable weight to this option. PbBi has further advantages over either Be or PbLi: (1) it is a liquid that can be drained to reduce the weight of the module when it is removed, and (2) it is quite safe because it does not have any chemical reactions with water that generate hydrogen, and (3) it does not generate tritium, and so does not lead to a host of engineering difficulties that arise when the primary coolant loop is strongly contaminated by tritium. The liquid PbBi would be cooled by Helium flowing through tubes, conceptually like the model AB of the European fusion power plant conceptual studies [23].

As mentioned earlier, every fusion system has to be self-sufficient in tritium. Therefore, tritium-breeding blankets are an integral part of the CFNS. We will simply adopt one of the standard choices of fusion reactor designers – a Helium cooled Li_2TiO_3 (a Li [6] enriched (to 90%) ceramic) blanket [24].

A structural fraction of ferritic steel (15–16% of the volume) is included in both the neutron multiplier, and the breeding blanket for tritium. It is represented as a smeared fraction, with details given in Appendix II.

Fissile fuel breeding occurs in a separate, outer breeding – blanket, located outside the vacuum vessel of the CFNS. In the Th^{232}–U^{233} cycle, the breeding-blanket consists of thorium oxide fuel rods, with a steel or SiC cladding. We have avoided the standard Zr cladding used to make the fuel for thermal spectrum reactors like PWR; it is liable to unacceptable damage from exposure to the fast spectrum in the Hybrid breeder. A desirable cladding must be simultaneously corrosion resistant for PWR operation, and capable of tolerating neutron damage from a fast spectrum. Fortunately several claddings, being developed for fission reactors, appear to be suitable for the complete ReFree cycle. High chromium alloy ferritic steels, possible with Oxide Dispersion Strengthening (ODS), for example, being investigated for high burn-up fuels in supercritical water reactors (SCRs), and for lead cooled FRs [6,7]. A cladding that resists corrosion in an SCR is very likely to be suitable for a PWR, since supercritical water is far more corrosive than the lower temperature water in a conventional PWR. Ferritic steels also have very good resistance to neutron damage, including He generation, which is why they are often chosen as structural materials for FRs [6,7,20] and fusion reactor designs [21–23]. Silicon carbide composite, under active development for PWR cladding (since it shows considerable promise for allowing high fuel burn-up [8,9], and is extremely resistant to water corrosion), offers another possible Hybrid- PWR relevant cladding. Note that it is also being examined for use in gas cooled FRs [25], and in fusion reactors. Our calculations show that either material has acceptable neutronic properties. It is even conceivable that stainless steels, already used in PWRs in the past, and in sodium cooled FRs today, might be acceptable.

Table 1
PWR parameters.

Active core radius	1.22 m
Active core height	3.65 m
PWR power	900–1500 MW
Fuel rod diameter	1.0 cm
Fuel cladding thickness	0.05 cm
Thorium oxide density	9.3 g/cc
Pitch to diameter ratio	1.3
Water temperature	300 C
Water density	0.7 g/cc

The choice of a coolant is the next crucial decision for the Hybrid design; the choice must be, first, dictated by the fact the Hybrid has both fusion and fission systems. The “special pathway” of the ReFree fuel cycle – breeding a given concentration of fissile material that will be critical in the PWR while remaining highly subcritical in the Hybrid spectrum during breeding – will also be a principal determinant. The Hybrid spectrum, by necessity, must be fast since the fission cross sections in the fast spectrum are lot lower.

The hybrid coolant, then, must not be a strong moderator. In addition, the coolant must not cause alterations in the surface of the cladding that impairs its ability to function in a PWR. Finally, the coolant – magnetic field interaction must be minimal – both for reliable cooling and for avoiding large forces in the fissile fuel during, for instance, plasma disruptions. A gas coolant like, helium, appears ideal to satisfy all these requirements. Helium is particularly desirable because of its superior chemical inertness. Gas cooling constrains the fission blanket to a relatively low gross volumetric power density. For example, gas cooled fast reactor designs typically have power density in the range 50–100 Mw/m^3 [19,25], which is far lower than that in the sodium cooled fast reactors. The hybrid design given here has a volumetric power density of only about 15 MW/m^3. We, therefore expect that gas cooling would be found to be feasible in a more detailed engineering design.

2.2. Pressurized water reactor model

The final destination for the Hybrid bred fuel rods is a pressurized water reactor (PWR). Most of our calculations are done for a PWR with a core size in the intermediate range of conventional reactors of today, with a thermal power of 1500 MW. Specific parameters are given in Table 1. The core dimensions are also very similar to new “modular” PWR designs, such as IRIS [26], one of several such “modular” concepts [27] that are expected to be commercial in the 2015–2020 time frame. This new generation of “modular” PWRs is noteworthy for several reasons. *They have multiple advanced safety features compared to PWR designs of older generations. One example of these is that they can be passively cooled from natural circulation processes (no pumps or other active components are required).* Modular reactors may allow factory construction with substantially reduced on site construction time, which improves both economics and the speed with which plants can be built. These features could be very desirable to enable nuclear power to be used to reduce greenhouse emissions in a timely fashion. These reactors operate a lower core power density than conventional PWRs. The IRIS design operates at a thermal

Table 2
Parameters of the CFNS plasma.

Major radius	2 m
Aspect ratio	1.8
Elongation	3
Triangularity	0.5
Plasma current	15.3–17.6 MA
Current drive power to plasma	45–68 MW
B_0 = vacuum B field at plasma geometric center	2.7–2.9 T
$\langle\beta_N\rangle = 2\mu_0\langle p\rangle/\langle B^2\rangle$	3–4
$\beta = 2\mu_0\langle p\rangle/B_0{}^2$	26–33%
H mode enhancement factor H_H	0.96–1.19
Plasma temperature: $\langle T\rangle$ and $T(0)$	18 keV and 31 keV
Plasma density $\langle n_e\rangle$	1.08×10^{20} m^{-3}
Greenwald number	0.22–0.29
Fraction of bootstrap current	0.35–0.52
B at centerpost	7.1–7.9
Ohmic dissipation in TF coil	158–194 MW

power of approximately 1000 MW, similar to our lower power cases.

Conventional PWR fuel management is assumed, with each fuel rod shuffled to a new location in the core three times (after which it is removed from the PWR core). The full core is simulated, in six concentric radial zones of equal volume. In conventional PWR operation, fuel is periodically shuffled between these zones to obtain the maximum fuel burn-up while keeping the power peaking to an acceptable value. The time interval between shuffling is taken here to be 1½ year, a value typical of operation today. (This is shorter than nominal values for some modular reactor proposals). Also, as is conventional, one third of the fuel is replaced upon each shuffling.

For simplicity, we do not further delineate assembly level structures in the PWR, such as control rods and guides, water zones, etc. (see Fig. 2). This treatment of the PWR core, though simplified, should give a reasonable approximation of the burn-up that could be achieved with the fuel.

In the PWR, the fuel is burned till the whole core k_{eff} drops to 1.02, at which point, the fuel is re-shuffled. Old fuel that has been burned for three cycles is removed, and is replaced by new fuel. This cutoff was chosen to reproduce burn-ups conventionally found in the US PWR fleet with standard UOX fuel (39 MWd/kg for a U^{235} enrichment of 3.75% [28,29]).

Several shuffling schemes among the zones were examined. In addition to optimizing energy production, the fuel must not produce a highly non-uniform power distribution. The radial power peaking of the six zones was computed, and kept below the conventional value of 1.26. One shuffling pattern was found to slightly outperform the others, and is described in the results section.

2.3. The CFNS – Tokamak plasma description

For a whole variety of compelling reasons, some of them already delineated, the reference CFNS is based on a spherical tokamak (ST) plasma configuration (fortified by the SXD). We repeat that the compact STs require a TF coil with relatively low mass hugely facilitating the replaceable module concept. As an aside we should point out that our initial calculations indicate that “neutronically”, a standard aspect ratio tokamak (not discussed further) should also be just as capable of driving a Hybrid with the ReFree cycle.

The fusion power level of the reference CFNS for the fuel breeding application is 400 MW. The machine and plasma parameters are shown in Table 2. It is possible to produce 400 MW with a range of plasma equilibria, as explained in detail below

We choose geometrical parameters that are similar to the ones proposed for the ST based component test facilities and Powerplants [10,21]. Similar to ST reactor studies [18,30,31], by choosing an aspect ratio of 1.8, we have left room for some shielding ~10 cm thick, on the inboard side.

We have chosen an elongation that is similar to that in conceptual designs of other ST fusion devices, such as a Component Test Facility [10], and fusion power reactors [17,21]. A detailed assessment of the control requirements for this device (such as the placement of feedback coils) is beyond the scope of this investigation.

Plasma operation consistent with results found in H-modes on present ST experiments, has been assumed. Representative plasma equilibria were generated with the MHD equilibrium code VMEC [32] for a range of plasma currents and plasma beta values (the ratio of plasma pressure to the magnetic pressure). Pressure profiles were taken to be H-mode like, with a pressure peaking $p(0)/<p> = 2$, consistent with NSTX experience [33] of profiles with good MHD stability. We also take the density profile to be rather flat. The fraction of bootstrap current is computed to be rather modest, 0.35–0.52, so the rest must be driven by external power. To minimize this power, we choose an average plasma temperature of 18 keV, and a peak temperature of 31 keV. These temperatures are comparable to the ones adopted by tokamak reactors studies that operate with a bootstrap fraction similar to ours. In one set of fusion reactor studies with bootstrap fractions ~43–45%, temperatures with an average value of ~20–22 keV, and peak value ~30–33 keV were chosen [34]. Steady state scenarios in ITER have similar temperatures. Parameters of various MHD equilibria that produce 400 MW are shown in Table 3.

Details of H-mode-like pressure profiles used for this study are given in Appendix I. A well known drawback of operation at high plasma temperature is that the alpha particles slow down less rapidly, and hence, end up contributing a significant fraction of the total pressure. Consequently, for a given total pressure, the alpha pressure reduces the DT pressure, causing a significant reduction in plasma fusion reactivity. This “loss” is accounted for in our calculations as described in Appendix I.

Two methods of RF current drive are potentially feasible for the ST parameters considered here – Fast Wave Current Drive (FWCD) and Electron Cyclotron Current Drive (ECCD) resonating on the inboard of the torus. Neutral beam current drive is also possible. However, it requires large horizontal ports that must penetrate the fission blanket and the containment boundary. In addition, these beam ports would allow neutron losses that are deleterious to the fission performance, and also subject the beam injectors to considerable activation. Current drive with RF is, therefore, more desirable. Our estimates show that adequate efficiencies should be possible.

Let us first consider FWCD. For an ST, the CD efficiency of FWCD degrades sharply away from the axis, due to trapped particle effects [35]. This degradation is a sensitive function of the ratio of the phase velocity of the wave to the electron thermal velocity, which we denote by w. If the phase velocity is chosen so that w is near the value for maximum absorption on axis ($w\sim2$) [36], an acceptable CD efficiency is possible in the inner third of the CFNS plasma. Details of the calculations are given in Appendix II. Inside of one third of the minor radius, the efficiency is reduced by less than 50%, and the volume averaged efficiency reduction inside the half radius is also roughly 50% of the value on axis. Here, we will estimate the profile averaged efficiency to be half the value on axis, on the expectation that the large majority of current will be driven inside the half radius due to the choice of w. As noted in Appendix II, because of antenna coupling effects, the RF input power could be larger, perhaps, by a factor of 1.5 for FWCD, than the power absorbed in the plasma core.

As is well known, ECCD avoids the coupling issues associated with FWCD.

In Appendix II we make a rough estimate of the power requirements for second harmonic ECCD, and find that it should be close to our estimate of the FWCD power. For the parameters here, waves launched from the outboard side can propagate near the midplane

Table 3
Parameters of possible CFNS plasma equilibria.

$\langle\beta_N\rangle$	β_N	β (%)	I_p (MA)	$B(T)$ plasma	H_H	Boot-strap fraction	CD power (MW)	TF power (MW)	$B(T)$ at centerpost
3	4.42	27.4	20.5	2.87	0.99	0.35	64	190	7.77
3	4.54	30.9	20.4	2.70	0.96	0.33	68	168	7.31
4	5.06	26.8	18.9	2.90	1.07	0.52	45	194	7.85
4	5.56	32.7	17.1	2.62	1.19	0.47	46	158	7.11

to the resonance region on the *inboard* side (at about $r/a \sim 0.2$). This may seem surprising, since present ST experiments are highly "overdense" to such waves – the plasma frequency $\omega_{pe} \gg \omega_{ce}$, the electron cyclotron frequency, so the waves do not propagate. However, note that $\omega_{pe}^2/\omega_{ce}^2 \sim n/B^2 \sim \beta_e/T_e$. The value of β_e for the hybrid ST is comparable to the β_e in present experiments, but T_e is far higher for the hybrid ST. For the hybrid, ω_{pe}/ω_{ce} is much smaller than for present experiments – in fact it is only about 1.2–1.3 on axis. As indicated in Appendix I, it appears that waves that are resonant at the second harmonic, at a position slightly on the inboard side, could be successfully launched from the outboard. Detailed calculations are in progress, but it appears that current drive efficiency for second harmonic ECCD would be comparable to the FWCD values estimated above, based on the absorbed power. Since there is almost no coupling inefficiency with ECCD, the over-all efficiency of ECCD could be significantly higher. The wave frequency is ~170–180 GHz, so that the gyrotron technology being developed for ITER could be used.

Results for the current drive (CD) power are given in Table 3. The values quoted are for the power absorbed by the plasma.

Operation below the no-wall limit is one of the options among operational regimes displayed in Table 3. Detailed MHD stability calculations are left to future work, but previous analysis of ST MHD stability [37] find that the no-wall stability limit allows $\langle\beta_N\rangle$ to be slightly greater than 3, for cylindrical safety factors greater than 2, just the range we consider here. Operation below the no wall limit requires substantial driven current, and hence, substantial current drive power – 64–68 MW, which is roughly comparable to the total current drive power expected for steady state operation of ITER. Because of the high current, only a modest enhancement of confinement above the value predicted for an H-mode is required. Employing the H-mode scaling law ITER 98(y,2) recommended to predict ITER confinement [38], the necessary enhancement factor H_H is ~1.0–1.2. (This surprisingly modest value pertains because the hot alpha particle pressure should not be included in the stored energy, when employing ITER98 (y,2).) Such levels of H_H have been attained in current ST experiments, and are expected to be attainable for the CFNS parameters. Since ST experiments have found a substantial improvement in H_H as collisionality decreases [39,40], the CFNS plasmas with much lower collisionality than present experiments, is likely to have higher H_H. Experiments have been successful in operation at parameters up to 1.5 times than the no-wall limit [41]. Operation at $\langle\beta_N\rangle = 4$, then, seems reasonable; we also include such cases in Table 3.

Hence, the required parameters of the plasma core lie in a range that appears consistent with present experimental results and reasonable near term extrapolations from them. The main extrapolation is in the area of steady state operation using non-Ohmic currents. In fact the quest for a steady state operation is the driving force behind the next generation of tokamak experiments, both for the normal aspect ratio and for upgrades of low aspect ratio tokamaks.

2.4. The CFNS – engineering description

Materials in the respective zones are assumed to have appropriate smeared fractions – for example, the zones for the ceramic breeder and PbBi multiplier have significant volume fractions of steel and helium. These values used here are given in Appendix III.

We expect that the ceramic breeder would be very similar to conceptual designs for fusion reactors, with tritium removed from the ceramic by a separate purge gas stream. The PbBi neutron multiplier is cooled using pipes with flowing He, conceptually similar to the cooling scheme of PbLi in the reactor model AB of the EU Power Plant Conceptual Study [23]. The PbBi would be drained for module replacement. We also attempt to make use of neutrons that are normally lost via parasitic capture by Cu; it is done by adding sealed tubes of Li enriched material to the outer zone of the centerpost. This increases the TBR by a few %. The tritium could either be removed continuously with a purge gas, or be separated and used in a batch type mode when the centerpost is replaced.

We had chosen the aspect ratio of 1.8 to allow room for ~10 cm thick shielding on the inboard side (similar to ST reactor studies [18,30,31]). We find that this shield moderately reduces damage and nuclear heating to the centerpost. We also include a thin ceramic tritium breeding region, which improves the neutron utilization.

The replaceable module concept is related to maintenance schemes that have been proposed for tokamak CTFs [9,10,43] and ST based power plants [16,17]. The novel aspect of the concept developed here is the replacement of the entire tokamak, including the vacuum vessel, which should speed the operation, and the use of *two* CFNS modules; when one module is in operation, the other is being refurbished in a remote maintenance bay. This greatly reduces the accumulated material damage on a unit, and also eliminates the need for high availability of a single module. The fusion module could be switched out during normal plant maintenance outages, that occur every 1½ years in nuclear fission plants. Below we argue that the replaceability option may be the crucial element to enable a near term realization of the Hybrid. Here we assume a vacuum vessel thickness of 5 cm, and examine the resulting overall neutronic performance; we also examine the neutron damage rates to the vacuum vessel.

We remind the reader that the material damage caused by 14 MeV neutrons is one of the biggest obstacles in the path of thermonuclear fusion. A common metric for assessing the desirability of a given material is the number of MW yr/m^2 that it can withstand. A pure fusion reactor will probably require materials that can withstand 10–20 MW yr/m^2. This stringent requirement stems from the fact it is impractical to replace the first wall frequently enough in traditional "designs" of tokamak fusion reactors because this highly time consuming task will greatly reduce the reactor availability. However, with a replaceable module, the time consuming operation of replacing the first wall can be done in a remote maintenance bay. In the present hybrid, if the module is replaced every 1½ years; the first wall materials need only withstand ~3 MW yr/m^2, a requirement that is far less challenging.

Furthermore, in situ maintenance of the internal components of a traditional tokamak reactor is extremely difficult because of very poor access. To attain high availability in such designs, it will be necessary to develop technologies for very rapid maintenance in the small percentage of time that the reactor is down. This is a very difficult and unsolved challenge. However, with a replaceable module, a single module can be down about half the time, and the device can be disassembled. Each module only needs to be available ~40–45% of the time in order for the plant availability to be 80–90%. The next generation of fusion devices – CTFs [10,11] – are expected have availability in the range of 30%. It is much easier to extrapolate from this to a module availability of 40–45%, than to a fusion device availability of 80–90%.

Previous design studies have adopted vertical replacement of large components as the preferred maintenance concept. In the case of the ARIES analysis for an ST power plant the entire 4000 ton power core was replaced as a single unit [16]. In the ST power plant study of Voss et al., the 800 ton centerpost was vertically replaced as a unit [17]. The 400 MW (thermal) CFNS is far smaller and lighter than a typical fusion power plant designed to produce 3100–3300 MW thermal power. The entire CFNS module, including the vacuum vessel and everything within it, weighs about 800 tons (when the PbBi is drained out). Hence, the replaceable module concept is no more challenging than cases previously considered.

For replacement, the vacuum ducts, coolant lines, and power lines must be severed. A more detailed design would, hopefully, find that it is possible to replace the module by severing these connections in areas that have relatively low activation – that is, areas relatively removed from the high neutron flux regions. We believe it is likely to be possible to connect these ducts and lines substantially behind the neutron absorbing, tritium breeding, blankets – ideally, from the top and bottom. This could make the process of severing these connections faster. An important feature of our concept is that components that are, inevitably, extremely highly activated (such as the plasma first wall, fusion related blankets, and outboard vacuum vessel) do not need to be cut and reassembled in situ. We expect that this should lead to a significantly faster replacement process.

A major concern for a high power density machine is the heat flux on the plasma facing components. If 30% of the total heating power is radiated from the main plasma, the average first wall heat will be <0.3 MW/m^2. This value is quite low compared to 0.5–1 MW/m^2 [34], the range pertinent to most fusion reactor designs [34]. The remaining heating power would be sent to the divertor. This would not be feasible with a traditional divertor, but the Super-X Divertor can handle much higher divertor power than other approaches [12–14]. Calculations are in progress with SOLPS, and initial results indicate that, for a 3 mm SOL width, and with modest gas puffing and slight impurities, the outer divertor heat flux is ~4 MW/m^2, and the inboard divertor heat flux is ~1 MW/m^2. Details of these simulations will be presented elsewhere.

A structural and thermo-hydraulic analysis of the TF magnets is in progress but has not been completed. However, the CFNS appears to be comparable to other design studies for which such an analysis was done. Primary stresses on the centerpost are proportional to the TF field at the coil, which will be in the range 7.1–7.8 T here. These values are similar to those in ST reactor, and ST CTF design studies (7.6 T for one ST reactor design [30], 7 T another [18], and ~8 T for ST CTFs [10,11]. It is also less than the limiting value (8.5 T) found in another engineering analysis [31]. The cooling challenges and secondary stresses for a centerpost increase with the current density. The CFNS center post has a current density of only 1.5–1.7 kA/cm2, which is several times less than values in studies of ST Component Test Facilities [10,11]. It is also significantly less than limiting values found in another engineering analysis of an ST CTF [31], including both primary and secondary stresses. Thus, we conclude that the stresses and cooling requirements of the CFNS center post are likely acceptable.

In order to compute TF coil losses, the engineering assumptions used for the TF coil are given in Appendix IV.

We have not pursued a detailed design concept for the vacuum system and fueling system. We expect that vacuum ports could be located at the top of the CFNS in a region that have good access and relatively low neutron flux – however, it must be verified in a more detailed design that there is adequate space available. Presently, simulations are in progress with SOLPS, to better estimate the pumping and fueling requirements. Preliminary results indicate that it is desirable to perform some amount of gas puffing in the private region to ameliorate divertor heat fluxes. The possibility of using gas puffing on the outboard side to fueling the plasma is also being assessed with SOLPS. We note that plasmas on JET, with a roughly comparable minor radius and density, can be fueled by gas puffing alone. The ST geometry might also facilitate the penetration of neutrals to the core – that is, inside the pedestal – since, for STs, the pedestal width appears to be narrower than on conventional aspect ratio devices. Pellet fueling is another option, but it adds an additional port to be severed for maintenance. Detailed results from SOLPS on these questions will be presented in the future.

A thermo-hydraulic analysis is beyond the scope of this effort. However, the CFNS cooling concept is essentially the same as in previous helium cooled fusion reactor studies [22,23]. The neutron wall loading of the CFNS is comparable to the value in those studies. Hence, we expect that a helium coolant pressure in the range of 5–10 MPa would be employed. The helium volume fractions in the "smeared fraction" neutronic model of the ceramic breeder and neutron multiplier are consistent with previous fusion reactor design studies.

2.5. Fission blanket

One engineering difficulty of hybrids is that neutrons from the fusion source must pass through the walls that enclose the fissionable material. This leads to neutron damage of the vessel walls that are meant to serve two important functions – containing the fission coolant, and acting as one "containment boundary" in case of serious fission fuel damage during off-normal events. The coolant function, of course, must be maintained even under off-normal conditions, e.g. to remove afterheat.

Since these walls must be thin enough to readily allow neutrons to pass through them, we cannot make them arbitrarily thick to compensate for material degradation causing by entering neutron. The present design addresses the "enclosing wall" problem in two ways: (1) We have broken the fission blanket into many moderate radius pressure tubes (~10 cm) that contain bundles of fuel rods and gas coolant. The walls of these tubes are thin enough (~1 cm) to readily transmit neutrons, but still thick enough to contain the coolant pressure. (2) We have also invoked an innovative concept somewhat similar to one proposed for gas cooled fast reactors: a "guard containment" vessel around the fuel [19] to maintain a reduced pressure gas to be used for cooling in off-normal events, and to act as a containment boundary.

In the present case, the guard containment vessel is the TF return legs. Coolant circulated inside this region can allow heat removal. We have chosen the radius of the pressure tubes to be small enough so that, we suspect, the afterheat can be conducted to its surface without excessive fuel temperatures. Estimates indicate that passive radiation could transfer afterheat from the pressure tubes to heat pipes (with passive internal circulation), which could transfer the heat to the TF coils or even outside the TF coils. Thus, further analysis of a more detailed design may show that passive decay heat removal is possible.

We note that this geometry has an additional major advantage: it minimizes the eddy currents in the fission blanket from a tokamak plasma disruption. Outside the vacuum vessel, the disruption caused transient electric field is, primarily, in the toroidal direction. The pressure tubes do not have a toroidal current path. Hence, the eddy currents and electromagnetic forces on the fission assemblies should be greatly reduced.

The neutronic results presented here employ the geometry displayed in Fig. 1. The fissile material is inside of the pressure tubes. The TF coils have a layer of Lithium Titanate breeder around them, for neutron shielding as well as for capturing leaking neutrons for tritium breeding.

Table 4
Fissile fuel output per GigaWatt year of thermal energy production.

Hybrid feeding 900 MW PWR no reprocessing	1160 kg
Hybrid feeding 1200 MW PWR no reprocessing	934 kg
FBR (U^{238}–Pu^{239}), BR 1.5 with FBR reprocessing	<196 kg
FBR (U^{238}–$Pu^{239}$and Th^{232}–U^{233}) BR 1.35 with FBR reprocessing	<141 kg

3. Computational tools and model

The suite of neutron codes – MCNP, ORIGIN, and MONTEBURNS was employed for investigating the entire ReFee cycle. MCNP was used to calculate neutron transport in the hybrid and in the PWR model. The codes ORIGEN and MONEBURNS were used to compute the evolution of the materials in the fuel rods due to fission and neutron capture. Scripts were written to employ these codes to compute fuel movement and shuffling through both the hybrid and the PWR. Several different shuffling patterns were attempted, with six radial zones in the hybrid and six in the PWR.

The longer the fuel remains (during a shuffling time) in the hybrid, the more fissile material is generated. Hence, the fuel shuffling time in the hybrid was varied; it was made long enough to insure that the k_{eff} in the PWR stays above 1.02.

The most significant fission products were kept during the evolution of the materials in the fuel. MONTEBURNS automatically keeps the most significant fission products, with the "importance fraction" parameter determining the stringency of the criterion for inclusion. A rather stringent value of 0.002 was used for this parameter. Varying this parameter did not, significantly, change the answers.

An overall measure of the Hybrid-PWR system efficiency is the support ratio defined as the amount of thermal energy generated by the PWR divided by the amount of thermal energy generated in the hybrid to create its fuel. If there are many hybrids and many PWRs, complications due to a discrete number of reactors can be ignored, and the support ratio can be computed simply. It is just the ratio of the Megawatt days generated per kilogram generated by the fuel in the PWR, divided by the number of Megawatt days per kilogram needed to generate the fuel in the hybrid. In steady state, this is the same as ratio of the thermal power generated by PWRs divided by the thermal power of the hybrids that fuel them. This is the definition of support ratio used here. Note that the total number of megawatt days in the hybrid includes the nuclear power from both fission and fusion. We will quote only the steady state value of support ratio here, after several cycles when the initial transient effects have damp out.

4. Results and discussion

All results in this section are quoted for hybrid cases that produce one tritium per tritium burned. We note that traditional control rods in the PWR could be replaced by Li [6] bearing rods, which would, then, produce several percent more tritium [44]. We quote steady state results, after several cycles, to allow "initialization" transients to damp out.

Since the fuel is intended for a thermal spectrum reactor, the important output from the Hybrid are the isotopes that fission in a thermal spectrum (U^{233}, U^{235}, etc.). This output is enormously larger than the excess fissile output from an FBR. The hybrid produces ∼1100 kg fissile per Gigawatt yr (GWyr) of thermal energy output. Fast Breeder Reactors attain the highest breeding ratio when using U^{238}–Pu^{239} breeding. For a unit with thermal power equal to 1200 MW (the FBR size in the literature that has been analyzed in detail and is closest our hybrid value of ∼700 MW), a breeding ratio of 1.5 is theoretically attainable by FBR [45]. This is likely generous, as described below. But even making the generous assumption that the breeding ratio (BR) is 1.5, and that the entire excess output were fissile in a thermal spectrum, the output is only 196 kg per GWyr, 5–6 times less than the hybrid. Comparative fuel production numbers are displayed in Table 4.

Even this several fold excess of Hybrid fissile production over the FBR understates the overall advantage of the Hybrid as a fuel supplier for PWRs. The BR of 1.5 assumes metallic fuel without Zr, which is very challenging in a practical system due to fuel swelling, low fuel melting temperature compared to peak fuel temperature, etc. For a conventional Zr fraction, the BR is ∼1.3. The breeding ratio averaged over the length of the cycle is also significantly lower [46]. Finally, pure Pu^{239} is a significantly poorer fuel for a thermal spectrum reactor than U^{233} – only about 74% of thermal neutron captures by Pu^{239} result in fission, as compared to 92% for U^{233}.

Clearly, a Hybrid can sustain far more thermal spectrum reactors than a comparable power FBR on the U–Pu cycle. It is quite possible for FBRs to breed U^{233}, if thorium is included as a fertile material. However, in an FBR spectrum, U^{233} produces less excess neutrons per fission, so the breeding ratio FBR emplying Th^{232}–U^{233} is lower. Even if the U–Pu and Th–U are combined, the BR is lower. A value of 1.35, optimistically, may be attainable [46,47].

Tritium breeding, necessary for a self-sufficient fusion system, is a huge neutron drain on the Hybrid. For every fusion neutron, only ∼0.4 fissile nuclei are produced. For the ReFree cycle investigated here, the fission power multiplication is modest, so that the total fission power output hybrid is ∼0.6 of the fusion input power. Even so, the hybrid has a far more prolific fissile output for other reactors, compared to an FBR.

The support ratio of the hybrid depends on the fuel burn-up in the PWR (see Table 5). The support ratio is highest at low fuel burn-up. This might be due to the fact that Th^{232} fuel has significant conversion to U^{233} in a PWR. It is known that the PWR conversion ratio (bred U^{233}/fissioned U^{233}) is higher at low burnup [48]. At low burnup, the PWR may make a larger fraction of its own fuel, giving a boost to the support ratio.

If the hybrid fuel were reprocessed after its final pass through the PWR, it could be refabricated into new fuel. We estimate that this could likely increase the Hybrid support ratio by a factor of 1.5–2, to values of 7 and above. Hybrid fuel cycles can be optimized to recover this by reprocessing, while minimizing proliferation, by denaturing the U^{233} with U^{238}. This option is not available to a critical FR because the requirements of criticality do not allow nearly sufficient U^{238} for meaningful denaturing. This investigation will be the subject of a future paper, however.

Without reprocessing, the cumulative fraction of Th^{232} that is eventually converted to energy is ∼6–8%. This is about an order of magnitude better than the ultimate utilization of natural uranium in a conventional PWR (which is about 0.7–0.8%, taking into account Pu breeding in the PWR, and also the fact that some residual U^{235} is left with U^{238} in the enrichment process, and it is uneconomic to extract and burn [5].)

In the ReFree fuel cycle, the lifetime neutron damage in the PWR cladding is above the likely tolerable limits for the traditional cladding material – Zircalloy. Hence, the question of fuel cladding must be addressed. We display, in Table 6, the calculated values of the two crucial "measures" of neutron damage; the results show that steel and SiC are very promising. Claddings of steel with high chromium content are currently under development for high burnup fuels in both FR and water reactors [6,7,20]. Fuel in an FR is

Table 5
PWR support ratios, fuel burn-up and fission multiplication in the hybrid.

Hybrid case	Support ratio	PWR fuel burn-up per pass (=3 shuffles) (MWd/kg)	Fission power multiplication in Hybrid	Total fraction of Th fissioned (%)
Hybrid feeding 900 MW PWR	4.1	25	0.54	6
Hybrid feeding 1200 MW PWR	3.0	33	0.71	8

Table 6
Neutron damage to materials.

Material	dpa	appm He
Steel fuel cladding: lifetime dose in hybrid only (900 MW PWR)	17	26
SiC fuel cladding: lifetime dose in hybrid only (1200 MW PWR)	28	44
SiC fuel cladding: lifetime dose in hybrid only (900 MW PWR)	18	111
CFNS fist wall (outboard) per FPY	24	210
CFNS first wall (inboard) per FPY	19	6
CFNS vacuum vessel per FPY	8	29
Fission blanket tubes per FPY	4	7

expected to sustain ~100 dpa, and a roughly comparable number of appm He. We see that the calculated neutron damage for this fuel cycle is well below these values. (Note that ferritic steels resist He damage particularly well.) Furthermore, high chromium ferritic steels are under development for high burnup fuels in environments that are far more corrosive than a PWR (such as supercritical water reactors [6,7,20]), and in lead based FRs [7,20]. We expect that high chromium steels, of some appropriate alloy, would make claddings suitable for the Refree cycle. Even the traditional stainless steels may be able to tolerate the expected neutron damage, and the corrosion in the PWR environment (early generation commercial PWRs operated with stainless steel cladding, and they are also used in sodium FRs, where they sustain neutron damage well above the values in the ReFree cycle). Similarly, the neutron damage for silicon carbide cladding is considerably less than is expected in FR applications of SiC composites (~80 dpa [25] and He appm several times this). Composites of SiC are currently under development as attractive claddings for high burnup PWR fuels [8,9].

If the damage to the cladding is sufficiently low (as it seems to be), we can consider sending the fuel rods back to the hybrid to be recharged multiple times. This will increase both the Thorium utilization fraction, and the support ratio. Multiple Hybrid charging – PWR burn up cycles will also be the subject of future work.

The neutron damage to the fusion first wall, however, is quite high. The distinguishing feature of fusion neutron damage is the very high ratio of He production to dpa damage, and the values of that ratio found here (~10) are typical of steel in fusion systems. High He production is, in fact, acknowledged to be biggest challenge to developing first wall materials that have a prolonged lifetime in a fusion device. However, the material damage constraints for a Hybrid (with replaceable fusion module) are much less stringent- the components need to withstand neutron impact for a much shorter span – 1½–2 ½ Full Power Years (FPY). The neutron damage requirements, then, are very comparable to those for an FR (with much less dpa but commensurately more He appm). Consequently, materials that have already been developed in the FR and fusion program, or are likely to be developed in the near term, might well suffice to serve as the first wall of the replaceable CFNS module driving a Hybrid. It is the replaceable module concept, allowing the first wall to be replaced within a "manageable" interval, that could be crucial to near term realization of a Hybrid.

The replaceable module concept, however, does bring the vacuum vessel into the line of fire of neutrons, which, in principle, creates the potential for a number of deleterious effects. A significant result of our calculations is that these neutronic effects appear to be tolerable since the vacuum vessel is protected from direct fusion neutron bombardment by the intervening PbBi neutron multiplier that greatly reduces the energy of the neutrons. The energy reduction caused by the multiplier translates into two important benefits – it reduces the damage to the vacuum vessel, and it reduces parasitic losses in the vacuum vessel. As seen in Table 6, the vacuum vessel damage has ~3 times less dpa and ~7 times less He appm. It is, thus, likely that the vacuum vessel and fission blanket components, made from the type of materials considered for the FR program- ferritic steels-, could last for many FPY before requiring replacement. Calculations show, in addition, that removing the vacuum vessel improves the breeding performance by only a few %. *The replaceable module concept, therefore, does not appear to have any neutronic showstoppers.*

Several fuel shuffling patterns were examined. The shuffling adopted here, keeps the radial power peaking limited to the range 1.2–1.26 that is expected tolerable. With this restriction, the shuffling pattern in Fig. 3 was found to give the best support ratio. The hybrid and PWR are divided into six radially concentric zones. These are labeled 1–6, in ascending order from the innermost radial zone (=1), up though the outermost zone, (=6).

Finally, we note that the Hybrids described should produce at least enough electricity to operate themselves. Clearly, the amount of power needed to run the hybrid is largest for the MHD equilibria with the most conservative physics assumptions. In those cases, essentially all of electricity would likely be consumed within the hybrid itself. For equlibria with higher $\langle\beta_N\rangle$, a significant fraction of the electricity could be sold to the grid.

We note that preliminary results indicate that by increasing the fusion power to 800 MW from 400 MW, most of the Hybrid produced electricity would be available for sale to the grid even for conservative MHD equilibria. We also note that an 800 MW CFNS would have some advantages with neutron coupling, and we believe it would significantly improve the support ratio. Quantifying these expectations is left to future work.

5. Concluding summary

This paper has attempted to make a case for a fusion fission hybrid as a vehicle for bringing fusion research to a near-term application via the exploitation of fusion neutrons to augment and "fortify" fission. Based on the existing achievements of fusion research and experience, aided by several crucial recent innovations, we have, first, developed a credible pre conceptual design for an intense but compact source of neutrons. To take care of the very stringent material problems, the high power density compact fusion neutron source, CFNS, is conceived as a copper based, light weight, removable and replaceable module (that can be pulled in and out of a surrounding fission blanket) so that neutron damaged parts of the module can be periodically replaced in a remote bay. Once such a copious source of neutrons is available, it can be readily harnessed to solve two major problems associated with fission energy – destruction of the difficult to fission isotopes of nuclear waste, and the future scarcity of the nuclear fuel to feed the most successful thermal spectrum nuclear reactors. Concentrating on the fuel production alone, we have developed a reprocessing free Thorium breeding cycle (ReFree) that is qualitatively as well as quantitatively more desirable than the cycles available to fission only fast breeder reactors; the ReFree cycle is not only much more efficient (in the sense that a single hybrid can breed fuel for many more reactors (of the same thermal power) than the FBR, it is also highly proliferation resistant. Unlike the FBR mode of fuel production, in the ReFree Hybrid cycle, the integrity of the fuel rods is never broken, and the fissile fuel is never separated from a bulky matrix consisting of large countable components, and protected (at all times) by radiation from the fission products. Such a highly efficient and proliferation resistant total cycle (from fuel production to fuel utilization) could underwrite a safe and long lasting nuclear fission energy era, including energy self-sufficiency, until pure fusion energy becomes commercially available.

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Appendix I

An H-mode like pressure profile is assumed, with $p(0)/\langle p\rangle = 2$, as shown below (see Fig. A1.)

We use the common modeling assumption that the plasma temperature is computed from the pressure profile from $T(\psi) = T(0)[p(\psi)/p(0)]^{\varepsilon}$, where we take $\varepsilon = 0.8$. If the energetic alpha particle pressure was ignored, the plasma density would be $n(\psi) = n(0)[p(\psi)/p(0)]^{1-\varepsilon}$. However, the thermal pressure is diluted by the presence of hot alpha particles. The alpha fraction can be calculated by balancing its energy production with its energy loss rate. It is well known that the fraction of alpha pressure is a function only of the plasma temperature. For computation, we use the Maxwellian averaged *DT* fusion rate, assume an equal density of *D* and *T*, and employ standard formulas for high energy particle slowing down on electrons and ions. The result for the fraction of the total pressure contributed by high energy alpha particles, α_P, is given in Fig. A2.

Since the average alpha energy is many times higher than the thermal energy, the plasma density can be approximated by $n(\psi) = n(0)\{[1-\alpha_P(T)]p(\psi)/p(0)\}^{1-\varepsilon}$.

In addition, it is assumed that the plasma has a 4% thermal He fraction (comparable to ITER projections), and $Z_{eff} = 1.3$ due to the presence of Ar impurity, injected to enhance divertor radiation.

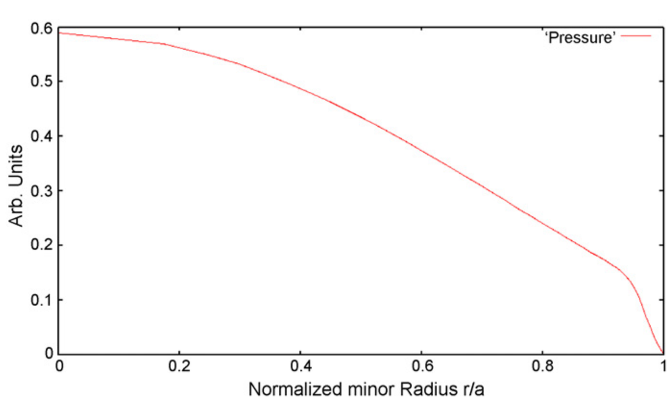

Fig. A1. Pressure profile assumed in equilibrium calculations.

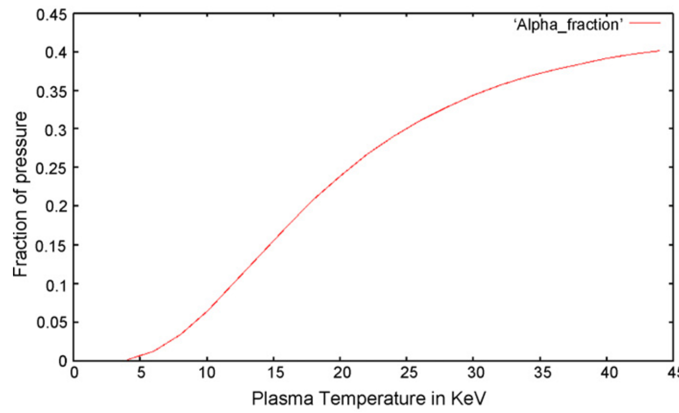

Fig. A2. The fraction pressure in high energy alpha particles, α_P, versus Temperature in keV, from classical slowing down.

Appendix II

In general geometry, the definition of the current drive (CD) efficiency η is defined as [35]:

$$\frac{\langle j_{RF}\,B\rangle}{\langle B^2\rangle} = \frac{V'\,\eta\,\langle p_{RF}\rangle}{L} \tag{A1}$$

where $\langle\rangle$ is a flux surface average, V' is the derivative of volume V with respect to poloidal flux ψ_P, and L is the distance along a magnetic field line to make a complete poloidal surface. Near the magnetic axis, we can make a number of approximations: the magnetic field is nearly constant and equal to the toroidal field B_T, the toroidal flux $\psi_T = B_T A$ (where A is the area in the poliodal plane), $L = 2\pi qR$ (where q is the safety factor), $\langle j_{RF}B\rangle/\langle B^2\rangle = j_{RF}/B$. Furthermore, we can use $j_{RF}dA = dI_{RF}$ (where dI_{RF} is the differential toroidal RF current in a flux surface), $dP = dV\langle p_{RF}\rangle$, and recall the general identity $q = d\psi_T/d\psi_P$, to arrive at

$$dI_{RF} = \frac{\eta\, dP_{RF}}{2\pi R} \tag{A2}$$

where R is the major radius. Expressions η can be written $= \eta_0 \mathrm{MC}R_{RF}$, where η_0 is the CD efficiency neglecting trapped particle effects, and the factors $\mathrm{MC}R_{RF}$ give the reduction in CD efficiency for finite aspect ratio [35]. This reduction is a sensitive function of the ratio of the phase velocity of the wave to the electron thermal speed w, and larger w results in less trapped particle reduction. Maximum wave absorption occurs at w approximately equal to 2 [36], and quickly decreases for larger w. In Fig. A3, we graph the product $\mathrm{MC}R_{RF}$ for two cases: the CFNS under consideration, and for parameters approximately equal to those of the HHFW experiment on NSTX, which had $w = 1.5$ in the core.

Notice that the current drive efficiency is uniformly higher for CFNS as compared to NSTX. It stems from two factors: the higher aspect ratio of the CFNS, and the assumption of a somewhat larger w. This combination leads to a good current drive efficiency in a considerably large region of the plasma. In the region spanned by the inner third of the minor radius, the product $\mathrm{MC}R_{RF}$ for the CFNS parameters is between 0.5 and 1. If one chooses the wave frequency and wave number to be 2 on axis, one would expect that most of the power would be absorbed near the axis. A detailed ray tracing analysis is beyond the scope of this paper. A volume average of the efficiency degradation inside of $r/a = 0.5$ gives a value slightly greater than 0.5. Here we simply take the CD efficiency to be 0.5 of the value on axis, where we assume $w = 2$. With this assumption, using Ref [42] (with $Z_{\mathrm{eff}} = 1.3$), the CD efficiency becomes:

$$I = \frac{2.76 \times 10^{18}\, T_e(0) P}{R\, n_e(0)} \tag{A3}$$

where I_p is in toroidal current in amps, R the major radius in m, $n_e(0)$ is the central density in $10^{20}\,\mathrm{m}^{-3}$, $T_e(0)$ the central temperature in keV, and P the RF power in watts.

This result above describes the power deposited in the plasma. However, the RF transmitter power will likely be larger due to several effects [49]. Real antennas have sidebands in the wave number, whose magnitude depends on the number of antenna elements. As a rough estimate, these might reduce the efficiency by 20–30%. Since the power levels here are rather high, and the power per antenna is limited by various factors, suitable antennas for a CFNS might have somewhat more antenna elements than lower power systems, so we hope we might have a somewhat "more favorable" wave number spectrum. In addition, coupling of the antenna power into the plasma might introduce another 20–30% loss of useful power. Hence, RF transmitter power might be larger than estimated in Eq (A3) by about a factor of ~1.5 for it to transfer the required power to the plasma.

Next, we consider the feasibility of second harmonic ECCD resonating on the inboard side. In Fig. A4, we plot $\omega/\omega_{\mathrm{pe}}$ and $\omega/\omega_{\mathrm{ce}}$ versus minor radius along the midplane, for a frequency resonant at about $r/a = 0.2$ on the inboard. As can be seen, the wave frequency is significantly above the plasma frequency, and no other resonances are encountered for a wave propagating from the outboard. (Resonances occur when an integer m satisfies $m = \omega/\omega_{\mathrm{ce}}$.) Using the cold plasma dielectric, we find that inwardly propagating waves do indeed exist along the midplane, all the way from the resonance to the outboard boundary. We estimate that a wave (either an X or O mode) launched from the outboard midplane would reach the inboard resonance, following a path that deviates only slightly above or below the midplane. In short, the inboard resonances are accessible to both X2 and O2 current drive. At this inboard position, there are no trapped particles, so there should be no degradation of the current drive efficiency that is found from outboard resonances. Previous calculations for ITER [50] have computed the ECCD efficiency near the axis from X2, with a central temperature similar to the hybrid (30 keV for ITER). That efficiency is about 20% lower than the average efficiency estimated above for FWCD. However, it is well known that ECCD efficiency increases with increasing plasma β_e at the resonance point [51]. The β_e in the ST hybrid is several times higher than in the case of ITER, so we anticipate the current drive efficiency would be comparable to, or better than, the efficiency we have estimated for FFCD. More detailed calculations are in progress to quantify this efficiency.

It is worth while, at this stage, to point out a great advantage for ECCD; ECCD power can be coupled to the plasma very efficiently, without the challenges encountered by fast waves, and with antenna structures further from the plasma.

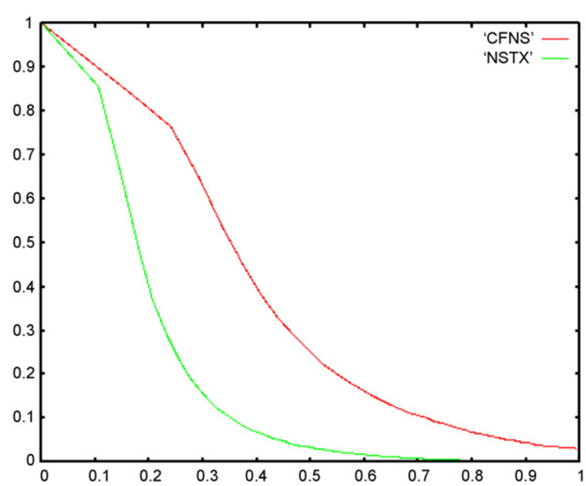

Fig. A3. CD current drive degradation factor versus minor radius r/a for the CFNS and for an equilibrium similar to an NSTX experiment with FWCD.

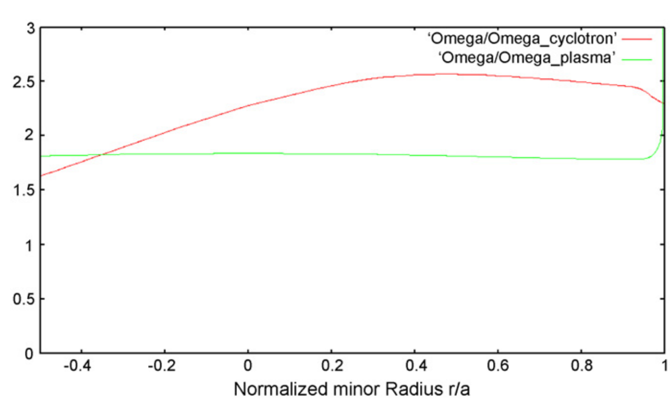

Fig. A4. The ratio of the wave frequency to the electron cyclotron frequency, $\omega/\omega_{\mathrm{ce}}$ and the wave frequency to the electron plasma frequency $\omega/\omega_{\mathrm{pe}}$, versus normalized minor radius. Values of minor radius less than zero correspond to locations inboard of the magnetic axis.

Appendix III

The material compositions for the hybrid are as follows. We use volume fractions similar to blankets in other fusion engineering studies [52]. For the tritium breeding region, the breeder has a volume fraction of 15% ferritic steel, 10% helium coolant, and 75% Lithium Titanate breeder Li_2TiO_3. The Lithium Titanate has an additional 35% of its volume for helium purge gas, so the total helium volume fraction is 36.25%, and the Li_2TiO_3 volume fraction is 48.75%. The PbBi neutron multiplier zone has 16.7% ferritic steel (by volume fraction), 12.7% Helium, and the 70.6% PbBi eutectic. The eutectic is 55.5% Bi, with the remainder being Pb.

The first wall is 35% He coolant and 65% ferritic steel, and is 1.5 cm thick.

For the neutronic calculations, we use a helium density corresponding to 10 MPa pressure at a temperature of 500 °C.

Several ferritic steels are being examined for use in fast reactors and in accelerator windows for the accelerator based transmutation reactors (ADS). Ferritic steels, the preferred structural material for a fusion reactor DEMO, are also under active development in the fusion program. We note that ADS applications have an even higher He generation per dpa than fusion neutrons. A detailed justification for a specific alloy composition is beyond the scope of this paper. However, we must choose a specific alloy to run MCNP. The neutronic results should be relatively insensitive to the specific alloy composition, since they all are mainly Iron with 9–12% Chromium, and the differences are only in the remaining 1–2% of the alloying agents. But since we must make a choice, we choose an alloy under consideration for FRs and ADS called EM-10, since there is evidence that it suffers significantly less of an increase in the ductile to brittle transition temperature than other ferritic steels under neutron bombardment [53].

Appendix IV

Preliminary thermo-hydraulic calculations [54] indicate that the centerpost will operate at an average temperature of 100–150 C. The temperature of the centerpost is determined by the following factors: the coolant temperature rise, the temperature difference across the coolant-channel interface, and the temperature difference within the Cu centerpost. In the parameter range being investigated, the second factor plays the largest role. The coolant fraction is primarily determined by the requirement that the total surface area of all coolant channels is sufficient to keep the interface heat flux to about 1.2 MW/m^2. This keeps the interface temperature drop to a moderate value (at a coolant flow rate of 10 m/s), so that the centerpost temperature does not exceed 150 C. We assume coolant channels 2 cm in diameter. For the cases in Table 3, the coolant volume fraction of the centerpost is about 4%. To estimate the average conductivity, we assume a temperature of 150 °C for the entire centerpost. The centerpost material, CrCu, has a resistivity of 3.28×10^{-8} at the specified temperature (To put this in perspective, this is about 53% of the electrical conductivity of pure copper at 20 °C.). Finally, the centerpost height is 1.75 times the height of the plasma. Optimization of these results using detailed thermo-hydraulic calculations is in progress – we are trying: (1) flaring the centerpost to a larger diameter towards the ends in order to significantly reduce the Ohmic losses, and (2) using a larger number of coolant channels with a smaller diameter (with more total surface area, and the same coolant volume) to lower the average centerpost temperature, and hence increase the conductivity.

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