NEAR TERM FUSION-FISSION HYBRIDS

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Abstract. Two distinct approaches to a fusion-fission hybrid are discussed. For a near term realization, the University of Texas design emphasizes a rather light weight, modular, separable (fusion and fission parts), and replaceable compact high power density fusion neutron source that vastly reduces the material and engineering demands and allows relatively easy maintenance and high availability. The Georgia Tech design emphasizes using ITER physics and technology as a basis for their machine. Both approaches combine the fusion systems with standard fission reactors. Also studied are the unique abilities of a hybrid as a destroyer of nuclear waste and as a breeder of fissile fuel.

1. Introduction

Two breakthroughs are necessary for a successful renaissance of fission nuclear energy: 1) a dependable and economic solution to the nuclear waste problem for environmental and social acceptability, and 2) an efficient procedure for creating fissile fuel since a nuclear resurgence could eat up known exploitable resources in the not too distant future. A Fusion-fission hybrid (Hybrid), in which neutrons from a fusion source significantly assist nuclear transmutations, can provide a solution to both waste and fuel issues that have limited or likely to limit the widespread use of fission power. Though an old idea, the Hybrid research has gained great momentum in the last few years because fusion research has matured to the level that a realistic fusion neutron source can be readily conceived. Within the context of magnetic confinement approach to fusion, two distinct pre-conceptual designs of fission-fusion hybrid reactors, with multiple novel elements to enable nearer term realization, have attracted attention:

1) The University of Texas hybrid (UT Hybrid[1]) driven by an ST based, replaceable, modular high power density Compact Fusion Neutron Source (CFNS[2]). The enormous power exhaust of the high power density CFNS will be handled by the recently invented Super-X divertor geometry[3]. The UT hybrid program is based on several innovations both in fusion and fission.

2) The second design (SABR, developed at Georgia Tech) invokes a standard tokamak based on ITER physics and technology.

Both designs, self-sufficient in tritium breeding, choose dimensionless plasma physics parameters (table1) in the range anticipated in ITER. Since the fusion architecture as well as the intended mode of operation and utilization of the two hybrid scenarios is quite different, they will be described, along with their conclusions, under separate headings.

Table 1 Neutron Source Parameters

Parameter	SABR nominal	SABR extended	CFNS	ITER	ARIES-AT
Major Radius R, m	3.75	3.75	1.35	6.2	5.2
Current I, MA	8.3	10.0	10-12	15.0	13.0
P_{fus}, MW	180	500	100	400	3000
Elongation κ	1.7	1.7	3.	1.8	2.2
Plasma B field, T	5.7	5.7	2.8	5.3	5.8
B_{TFC}/B_{OH} at coils, T	11.8/13.5	11.8/13.5	7	11.8/13.5	11.4/?
Confinement H_{IPB98}	1.0	1.06	1.0-1.2	1.0	1.0
Normalized Beta β_N	2.0	2.85	2.5	1.8	5.4
Plasma Energy Mult. Q_p	3	5	2	5-10	>30
CD eff., γ_{cd}, 10^{-20} A/Wm^2	0.61	0.58	0.15-0.2		
Bootstrap cur. frac f_{bs}	0.31	0.26	0.3		0.91
Neutron Γ_n, MW/m^2	0.6	1.8	1.0	0.5	4.9

2. The UT Hybrid

We will break up the description into parts – the fusion part and the fission part. The first part will gives the main guiding principals and essential characteristics of the tokamak-based high power density CFNS and its coupling to a fission blanket, while the latter part will be devoted to a discussion of several novel fission fuel cycles that are either enabled by or vastly facilitated by the copious supply of CFNS neutrons.

i) The CFNS and the fission blanket:

Since the physics parameters of CFNS are smack in the range of what is being demonstrated in fusion laboratories today, many aspects of its design were dictated by considerations that will provide appropriate connectivity between the fusion and the fission systems. Strong neutronic coupling is a must for efficiency, but mechanical and electromagnetic independence may be necessary so that 1) the hybrid can make maximum use of the existing, in development programs in fusion and fission 2) the combined fusion-fission system may not suffer from new possible interactive modes of failure 3) maintenance of the fission and fusion aspects is possible with adequate availability. The reference CFNS has the following characteristics:

a) It is designed to produce fusion power of order 100-200 MW in a compact machine that fits inside a sub-critical fission assembly as a remotely handled, replaceable module. Compatibility with standard economic fission assemblies requires the CFNS plasma to have a major radius plus minor radius < 2.5 m.

b) A tokamak with an aspect ratio (A=1.8) was chosen for the CFNS because such a conceptualized machine captures the best combination of physics basis, ease of coupling to a fission assembly, potential for high power density, and ease of maintenance. The geometry for MCNP neutronic calculations is shown in Fig.1, and nominal fusion parameters are displayed along in Table 2. A substantial theoretical basis and computational tool set has been developed for thermonuclear level tokamaks as a prelude to ITER; this powerful machinery is being used to design and test the CFNS design.

Attempts to construct a neutron source of this power and compactness would have been impossible without the impressive achievements of the worldwide fusion research. Two recent innovations, however, were necessary for the architectural design of the CFNS.

A	R (m)	κ	$<\beta>_N$	β %	neutrons MW/m^2	n_e $10^{20}m^{-3}$	n_e/n_G	P_{CD} (MW)	Ip MA	B_{Coil} T	B_{Plas} T
1.8	1.35	3	2.5	22	1.0	1.2	0.2	50	12	7	2.6

Table 2. Fusion Core parameters for the 100 MW CFNS-Hybrid of Fig.1.

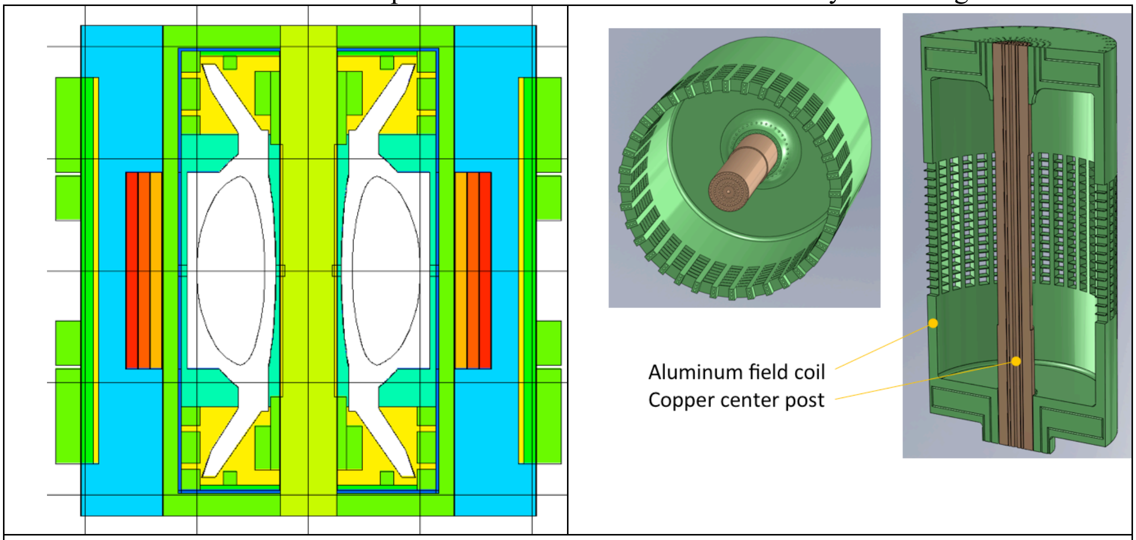

FIG 1. Veiws of the CFNS. One the right, a 2 dimensional schematic cross section of the geometry used in neutronic calculations. On the left, 3 dimensional representations of the coils used in thermo-mechanical calculations. Slots in the TF coils allow neutrons to stream from inside the CFNS to the fission blanket.

1. A new magnetic divertor geometry, the **Super-X divertor (SXD)** (Fig.2 shows the MAST upgrade[4]) with power exhaust capacity boosted up by a factor of ~5 (over conventional alternatives) to withstand the enormous heat fluxes anticipated in a CFNS. The Super-X divertor pulls the divertor plasma channel to a large radius where the heat flux naturally decreases and spreads onto a larger area: the plasma cools and its radiative capacity is enhanced over the standard divertors. The SXD will allow the CFNS core plasma to operate in readily accessible, demonstrated and conservative dimensionless parameter space. By substantially shielding the highly stressed divertor components from neutrons as well as heat, the SXD geometry is crucial in making near-term divertor technology adequate for CFNS design goals

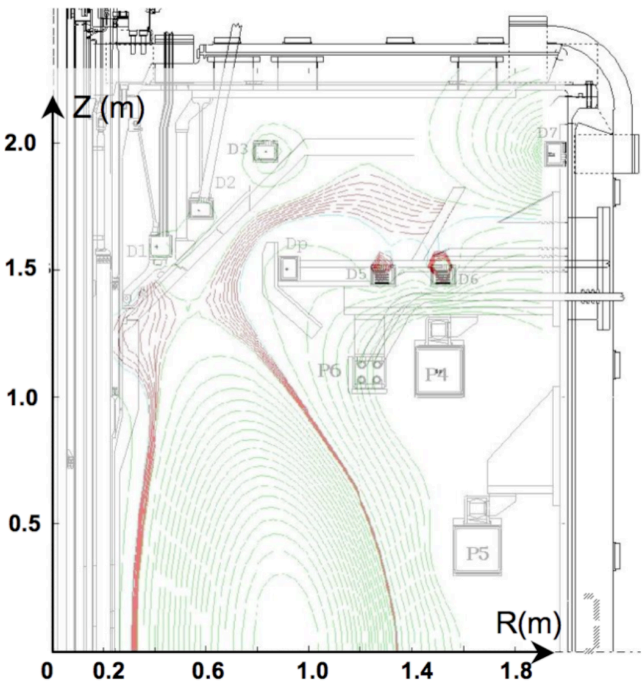

Fig. 2 Super-X engineering design for MAST-upgrade

2. Due to its compactness, the CFNS can be designed as an independent ***replaceable fusion module*** that fits within a fission blanket, but is *not physically connected to*

it. The replaceable module concept greatly minimizes the impact of problematic issues that could arise on the integration of fission with fusion. In addition, the replaceable fusion module concept greatly reduces the fusion technology requirements for the hybrid, and allows it to be developed much sooner than a pure fusion DEMO. Specifically:

a) The fusion driver may be (if needed) replaced at the same time the fission blanket is reshuffled - both maintenance operations could hopefully be carried out simultaneously in several weeks. The replaceable option makes the material constraints much less stringent; the driver components, then, have to withstand exposure to fusion neutrons for only about 1-2 years compared to the ~5 required for a conventional Hybrid or a pure fusion reactor. Cumulative damage from the fusion neutrons at 14 MeV is greatly reduced to a level many times below the requirement for pure fusion power plants.
b) Preliminary calculations with the state of the art tools indicate that the peak temperature in the magnets can be kept below 150 degrees C. The mechanical stresses (primary and secondary) are well below the yield strength. The stresses are likely below the allowables even for the embrittled copper centerpost.
c) Electromagnetic simulations show that placing the fission blanket outside the TF brings an order of magnitude reduction in the impact of plasma transients on the fission blankets; the TF coil structure will act as an electromagnetic shield against disruptions, for instance. There is an even larger reduction in MHD drag forces on the liquid metal coolants (used in fast spectrum fission blankets).

Attempts to create maximum compatibility between the fusion and fission assemblies dictate crucial design aspects - for instance, the choice of the best current drive modes in the CFNS. Since large penetrations of the fission blanket must be avoided for safety and "complexity" reasons, neutral beams do not remain an attractive option for heating and current drive; RF current drive becomes highly desirable. Preliminary investigations indicate that electron cyclotron current drive appears to have adequate efficiency. EBWCD efficiency, for Te(0)=20 keV, pressure profile consistent cases, was found in the range ~100-240 kA/MW at a frequency ~90 GHz implying absorption near 0.8 of minor radius. This is for launch at about 60 deg up from the mid plane. Equatorial launch shows that greater penetration can be obtained, but with degraded CD efficiency. ECCD and Fast Wave based current drive are also being explored. A rather limited amount of neutral beam current drive may be possible, if it is found to be indispensable.

The science and technology needed for a CFNS has a great deal of overlap with research and development planned for CTF, a component test facility for a pure fusion reactor. Although the development stages for the two are quite similar, the most time consuming stages in the CTF mission will not be relevant to the Hybrid; the hybrid mission can be accomplished on much shorter time scales because the neutron fluence requirement is much lower for the replaceable CFNS module ((1-3 (1-3 MW yr/m^2).) as compared to the CTF (~6 (MW yr/m^2).

Liquid lithium on a porous substrate is a highly attractive PFC option. Rough calculations indicate that capillary action could replace the entire surface on an hour long time scale. This liquid, continuously replaceable surface would greatly mitigate PFC lifetime concerns, dust production, tritium retention, etc. In addition, this liquid surface should provide great resilience to transient heat pulses. The use of lithium could also have beneficial effects on plasma performance. However, the lower edge recycling could increase the divertor heat flux. This possible issue is presently under investigation, but since the SXD gives a low heat flux of only 3-4 MW/m^2 for the CFNS with conventional recycling[2], there is considerable margin to accommodate an increased heat flux.

ii) Waste Destruction and nuclear Fuel production:

Given a high intensity source of neutrons, nuclear reactions in a fission blanket can be strongly affected. We envisage two very important applications of the hybrid 1) destruction of nuclear waste from the power producing standard light water reactors, 2) breeding fissile fuels from fertile materials like U238- Th232.

Since a hybrid is so much more complicated than a fission reactor (thermal spectrum or the fast reactor, FR), it must bring serious advantages over highly investigated fission only paths for these applications. It happens through the use of innovative fuel cycles with high support ratios that become safe and possible with fusion neutrons allowing the sub critical running of the fission reactors. The support ratio is defined as the total number of standard light water reactors (LWR) with the same thermal output, that can be supported by a single advanced reactor like the hybrid. Neutronic calculations show that a waste destruction scenario in which the LWRs are harnessed to burn as much as is practically possible, and then the leftover, highly toxic and long term radioactive, difficult to fission, collection of transuranics is burnt in the hybrid. Support ration as high as 30-50 are possible in such a scenario. Neutronic calculations indicate that a highly proliferation resistant, Th232 based, fuel production methods are possible that entirely avoid reprocessing; the fuel is protected at all times by the combination of bulk, denaturing, and radioactivity. The basic steps are:

1) Breed fissile U233 in a Th (Th Oxide) rod to 3-4% - preferably denatured by U238- in the Hybrid-Fast, epithermal and thermal spectrum scenarios are being investigated
2) Burn the bred rods in a standard commercial thermal- spectrum reactor (an LWR or perhaps a TRISO based version when ready)
3) If materials considerations allow, another round is possible, where the discharged fuel rod from LWRs is recharged again in a hybrid
4) Take the recharged rod back to the commercial reactor for further burning

Notice that the fissile fuel is never isolated from a denatured radiation/bulk protected matrix. As distinct from the critical FR path, no Pu processing, or initial Pu loading is needed – isolated Pu does not figure in the entire cycle. The hybrid, in addition, can produce 3-4 times as much fuel per unit of fission power as compared to quite optimistic estimates for fast breeder reactor (breeding ratio 1.5).

A hybrid based on the UT design, with its modular, separable, replaceable, compact high power density fusion neutron source that vastly reduces the material and engineering demands and allows relatively easy maintenance and high availability, has the potential for propelling fusion research towards an exciting near term application to the nuclear energy sector.

3. SABR

Conceptual design, fuel cycle and safety studies have been performed at Georgia Tech with the objectives of identifying the physical and performance characteristics and potential advantages (relative to critical reactors) of fast burner (transmutation) reactors that could be operated sub-critical with a tokamak fusion neutron source, for the purpose of fissioning the transuranics (TRU) remaining in the spent nuclear fuel discharged from nuclear power reactors. Both gas-cooled[5-12] and liquid metal cooled reactors[13-18] have been examined. The most highly developed fast reactor type, the sodium-cooled reactor, has been chosen as the reference concept because of the greater experience and more advanced state of development of this concept and the associated facilities. Likewise, the most developed magnetic fusion concept, the tokamak, has been chosen for the neutron source[19,20], and the design has been based on fusion physics and technology that will be demonstrated in ITER. It seems technically feasible to plan deployment of such SABRs beginning in 2040.

A. Fission Fast Burner Reactor

A design concept for a Subcritical Advanced Burner Reactor (SABR) that would be fueled with 100% TRU fuel to maximize net TRU burnup and that would operate subcritical with a variable neutron source to achieve deep TRU burnup has been developed. SABR is a loop type sodium-cooled fast reactor fueled with transuranics (TRUs) cast into a TRU-Zr metal fuel pin. The annular SABR core is adapted from previous ANL (Argonne National Laboratory) fast reactor designs and consists of 918 hexagonal fuel assemblies arranged in 4 concentric rings. Each assembly is 15.5 cm across flats and contains 271 wire-wrapped fuel pins of radius 3.63 mm (2mm fuel radius surrounded in turn by a 0.83mm thick Na-bond, a 0.5 mm ODS stee; cladding and a 0.3mm $LiNbO_3$ electrical insulator (to inhibit MHD pressure drop effects)). The design was based on the "fresh" TRU fuel (40Zr-10Am-10Np-40Pu) being developed at ANL. The initial TRU loading is 30 MT, which achieves $k_{eff} \approx 0.95$ with all fresh TRU fuel. The 4-ring annular core, with an inner core radius of 5.0 m, a thickness of 0.62 m and an active fuel height of 2 m (plus a 1 m upper fission gas plenum), encircles the plasma of a tokamak D-T fusion neutron source. See Fig. 3.

B. Tokamak Neutron Source Plasma Physics[19,20]

Conservative ITER-like physics has been adopted for the design of the SABR tokamak neutron source. Fusion power of P_{fus} = 100-500 MWth is required to support P_{fis} = 3000MWth under the range of subcritical operation envisioned. A reference normalized beta β_N = 2.0-2.5% was chosen, although operation at β_N values up to 2.5-3.0% could be justified on the basis of present experience. A confinement multiplier H = 1.0-1.1 relative to the IPB98(y,2) energy confinement scaling was adopted. The line average electron density was fixed at 75% of the Greenwald density limit to avoid confinement degradation at higher densities. An edge safety factor q_{95} = 3 was specified to avoid MHD kink instabilities.

For a R = 3.75 m tokamak a range of operating parameters are possible[15]. Detailed parameters for I = 8.3 and 10.0 MA are given in Table 1. For comparison, the corresponding design parameters for ITER and ARIES-AT also are given in the table. The requirements on β_N and confinement are within the range routinely achieved in present experiments, and the requirements on β_N, confinement, energy amplification Q_p, and fusion power level are at or below the ITER level and well below the ARIES-AT level. The requirement on the current-drive efficiency, after calculation of bootstrap current fraction using ITER scaling, is only somewhat beyond what has been achieved to date (γ_{CD} = 0.45 in JET and 0.35 in JT60-U).

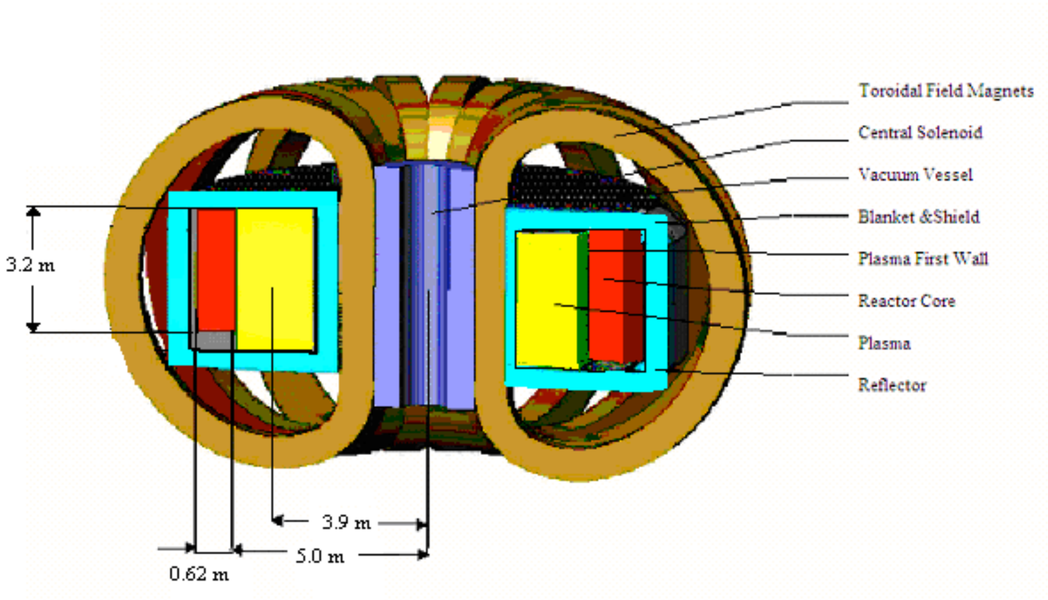

Fig. 3. Configuration of the SABR

C. Fusion Neutron Source Technology for SABR[19,20]

The ITER single null divertor (not shown in Fig. 3) and first wall were adapted for sodium coolant by scaling down to the SABR dimensions with the same coolant channels. The heat removal capability was confirmed by detailed FLUENT code calculations. The ITER Lower Hybrid (LH) heating and current drive system was adapted to provide 100 MW of heating and to drive 7.5 MA of plasma current

The TF and CS superconducting magnet systems for SABR were directly adapted from the ITER cable-in-conduit Nb_3Sn conductor surrounded by an Incoloy 908 jacket and cooled by a central channel carrying super-cooled helium, with maximum fields of 11.8 and 13.5 T, respectively. The dimensions of the CS coil were constrained by the requirement to provide inductive startup and to not exceed a maximum stress of 430 MPa set by matching ITER standards and Incoloy properties. The dimensions of the 16 TF coils were set by conserving tensile stress calculated as for ITER, taking advantage of an Incoloy 908 jacket for support.

D. Transmutation Fuel Cycle Analysis

The fuel used in SABR is TRU from the spent fuel discharged from light water reactors (LWRs). This fuel is burned in SABR in a 4-batch fuel cycle; the fuel is first loaded into the outermost of 4 fuel rings surrounding the plasma, burned for one burn cycle, then moved inward one ring, etc. until it has been burned for a residence time consisting of 4 burn cycles, one in each fuel ring. The fuel is then removed from the reactor and reprocessed (using the pyro-processing method being developed at ANL), combined with fresh TRU, refabricated and recycled in SABR again, etc. The fuel residence time in the reactor of 2800 EFPD was determined by the radiation damage limit to the clad of 200 dpa. The fission products are separated from the TRU (99% efficiency) and sent to the geological repository.
The neutron transport and fuel burnup calculations were performed with the ERANOS code[21], using JEFF2.0 cross sections in 33 energy groups from 20 MeV down to 0.1 ev. A lattice cell calculation in P1 transport theory and 1,968 energy groups was performed on the fuel assembly, the energy groups were collapsed to 33 groups and the assembly was then homogenized. A 2D-RZ S8 discrete-ordinates neutron transport calculation of the entire neutron source, reactor and shield was then performed.

Two different fuel cycle analyses have been performed. In the first *TRU Burner Fuel Cycle*[16], it is assumed that all the transuranics in the spent nuclear fuel discharged from LWRs are reprocessed for SABR fuel, and a fuel composition based on the metal fuel being developed at ANL[22] (which contains the average composition of all spent nuclear fuel that has been discharged from LWRs in the USA) was used as the initial fuel feed. The fusion power (neutrons) required to maintain 3000MWth fission power varied from 170 to 400 MW over this fuel cycle, and the rate of TRU fission (destruction) was 1060 kg/EFPY. Since a representative 1 GWe LWR fission reactor produces 250 kg/yr of TRU, one SABR operating at 75% availability would “support” (fission the TRU produced by) 3.2 LWRs of 1 GWe. Since 99% of the TRU discharged from SABR is recycled, only the fission products and 1% of the TRU at each reprocessing step go to HLWR repositories, the number of Yucca Mountain type repositories thaht would otherwise be required for LWR spent nuclear fuel could be reduced by a factor of 10 by replacing a 100% LWR nuclear fleet by one that is 75% LWRs (GWe) and 25% SABRs (GWe).

The second *MA Burner Fuel Cycle* analysis emphasized fissioning the minor actinides (MA) in spent fuel while setting aside the plutonium for other uses, as specified in the European studies of reactors to burn minor actinides[23]. Using the European “MA-rich” fuel required a slight redesign of the SABR fast reactor fuel assembly. The fusion power required to maintain 3000MWth fission power varied from P_{fus} = 200-500 MW in this fuel cycle, and the rate of MA fission (destruction) was 850 and 675 kg/EFPY, for metal and oxide forms of the fast reactor fuel, respectively. Since a representative 1 GWe LWR produces 25 kg/yr of MA, one SABR operating at 75% availability would “support” (destroy the MA produced by) 25.5 or 20.3 1-GWe LWRs.

E. Dynamic Safety Analyses

Initial analyses with the RELAP5 code[24] of the response of the coupled (neutron source)-(fast reactor)-(Na heat removal) system to various accident initiation mechanisms indicate that SABR can survive loss of power, up to 50% loss of primary coolant flow, and up to 50% loss of system secondary flow without core damage[14]. It is anticipated that SABR may in the future be predicted to survive 100% loss of flow in either primary or secondary coolant systems when account is taken of the negative reactivity effects of fuel bowing and expansion.

References

1. Kotschenreuther, M., Valanju, P.M., Mahajan, S.M., Schneider, E.A., Fusion. Eng. Des. 84, 83 (2009)
2. Kotschenreuther, M., Valanju, P.M., Mahajan, S.M., Zheng, L.J., Perlstein L.D., Bulmer, R.H., Canik, J., Maingi, R., Nuclear Fusion 50, 035003 (2010)
3. Valanju, P.M., Kotschenreuther, M., Mahajan, S.M., Canik, J., Phys. Plasmas 16, 056110 (2009)
4. www.ccfe.ac.uk/MAST_upgrade.aspx
5. W. M. Stacey, D. Tedder, J. Lackey, J. Mandrekas, et al., *Nucl. Technol*., 150, 162 (2005).
6. W. M. Stacey, D. Tedder, J. Lackey, et al., *Nucl. Technol*, 156, 9 (2006).
7. W. M. Stacey, J. Mandrekas and E. A. Hoffman, *Fusion Sci. Technol*. 47, 1210 (2005).
8. W. M. Stacey, D. Tedder, J. Lackey, et al., *Nucl. Technol*, 156, 9 (2006).
9. W. M. Stacey, *Fusion Sci. Technol.* 52, 719 (2007).
10. W. M. Stacey, C. de Oliviera, D. W. Tedder, et al., *Nucl. Technol*. 159, 72 (2007).
11. E. A. Hoffman and W. M. Stacey, *Nuclear Technol.* 144, 83 (2003).
12. J. W. Maddox and W. M. Stacey, *Nuclear Technol,* 158, 94 (2007).
13. W. M. Stacey, J. Mandrekas, E. A. Hoffman, et al., *Fusion Sci. Technol.* 41, 116 (2001).
14. W. M. Stacey, W. Van Rooijen, et al., *Nucl. Technol.* 162, 53 (2008).
15. W. M. Stacey, *J. Fusion Energy* 38, 328 (2009).
16. C. M. Sommers, W. M. Stacey and B. Petrovic, *Nuclear Technol*. 172, 48 (2010).
17. W. M. Stacey, C. M. Sommer, T. S. Sumner, et al., 11th OECD/NEA Info. Exchg Mtng on Actinide and Fission Product Partitioning/Transmutation, San Francisco, 1-5 November (2010).
18. T. S. Sumner, W. M. Stacey and S. M. Ghiaasiaan, *Nuclear Technol*. 171, 123 (2010).
19. J-P. Floyd, et al., *Fusion Sci. Technol*, 52, 727 (2007).
20. W. M. Stacey, *Nucl. Fusion* 47, 217 (2007).
21. G. Rimpault, et al., in: Proc. Int. Conf. New Frontier of Nuclear Technol.: Reactor Physics, Safety and High-Performance Computing, PHYSOR, Seoul, South Korea, 7-10 October 2002.
22. Personal Comm., Stephen Hayes and Mitchell Meyer, Argonne National Laboratory (2007).
23. V. Romanell, C. Sommer, M. Salvatores, W. Stacey, et al., 11th OECD/NEA Info. Exchg Mtg. Actinide and Fission Product Partitioning/Transmutation, San Francisco, 1-5 November (2010).
24. "RELAP5-3D Home Page", Idaho National Laboratory, www.inl.gov/relap5/ (2009).

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