Energy Policy ∎ (∎∎∎∎) ∎∎∎–∎∎∎

Contents lists available at ScienceDirect

Energy Policy

journal homepage: www.elsevier.com/locate/enpol

Nuclear fusion: Status report and future prospects

Jef Ongena [a,*], Yuichi Ogawa [b]

[a] Plasma Physics Laboratory – Royal Military Academy, Renaissancelaan 30, 1000 Brussels, Belgium
[b] Graduate School of Frontier Sciences, University of Tokyo, 5-1-5 Kashiwanoha, Kashiwa, Chiba Prefecture, 277-8568, Japan

H I G H L I G H T S

- Renewable energy sources need to be complemented by clean and environmentally friendly backup energy sources.
- Controlled nuclear fusion has the potential to be a major player in future energy systems.
- Magnetic fusion research is entering a new research phase with the construction of ITER, that, once in operation, will be the largest magnetic fusion device in the world.

A R T I C L E I N F O

Article history:
Received 12 February 2016
Received in revised form
21 May 2016
Accepted 24 May 2016

Keywords:
Magnetic fusion
Inertial fusion
Sustainable energy

A B S T R A C T

The paper gives an overview of fusion research in the world. The prospects for fusion as an energy source for the future are reviewed. Environmental compatibility, safety and resources are discussed.

1. Introduction

Perhaps the greatest challenge facing our modern world is to develop the necessary technology for an affordable, clean and sustainable energy production. Currently more than 85% of the primary energy production in the world is originating from fossil fuels. The disadvantages are well known: risk of irreversible changes to the climate system, limited reserves, dependency in supply. The number of conceivable non-fossil candidates that could replace the current massive use of fossil fuels is very limited: renewables, nuclear fission and fusion. Fusion is the least developed of the three, but has particularly valuable environmental and safety advantages and has virtually inexhaustible resources. It could prove very important as a backup energy source to cope with the variability of renewable energy sources, thus able to cover long cloudy and/or wind still periods. This paper discusses the current status of worldwide fusion research, resources, safety, environmental and economic aspects of fusion energy.

Fusion research is a worldwide effort. There are currently about 100 fusion research labs scattered in nearly all continents. The EU, Japan, the Russian Federation and the USA undertake a large research effort, with fast growing contributions from China India and South Korea. Other countries like e.g. Brazil and Australia are participating as well with substantial investments.

2. Nuclear fusion: principles

Replicating the fusion reaction in the sun would be a first possible approach to realise fusion on earth. However, the p-p reaction in the sun essentially converts 4 protons into a Helium-4 (^{4}He) nucleus containing 2 neutrons. This reaction requires thus the conversion of protons into neutrons, via inverse beta decay with a very low probability and therefore not suited for an economical process on earth. A much more 'simple' solution is offered using hydrogen isotopes, already containing the necessary numbers of neutrons and protons from the start, thus resulting in a reaction where essentially a rearrangement of the nuclides takes place, with a 10^{24} times higher reaction rate than the p-p process in the sun. From all possible reactions involving H isotopes, the least difficult fusion reaction is the one between the hydrogen

* Corresponding author.
E-mail addresses: j.ongena@fz-juelich.de (J. Ongena), ogawa@ppl.k.u-tokyo.ac.jp (Y. Ogawa).

http://dx.doi.org/10.1016/j.enpol.2016.05.037
0301-4215/

isotopes deuterium (D) and tritium (T):

$$D + T \rightarrow {}^{4}He\ (3 \cdot 5\ MeV) + n\ (14 \cdot 1\ MeV)$$

To produce sufficient fusion reactions, the core temperature of a D-T plasma has to be about 150–200 million C. This is about 10–15 times larger than the temperature in the centre of our sun, estimated to be about 15 million C.

The reaction products are a 3.5 MeV helium nucleus (alpha particle) and a 14.1 MeV neutron, i.e. in total about 17.6 MeV is released per fusion reaction. This could be converted into heat in a blanket and then into electricity using conventional technology (Carnot cycle). The huge energy release from the fusion reaction also results in a minimal fuel consumption: the deuterium contained in 1 l of sea water (about 30 mg) and used in D-T reactions will produce as much energy as burning 250 l of gasoline.

Other fusion reactions of interest are:

$$D + D \rightarrow {}^{3}He\ (0 \cdot 82\ MeV) + n\ (2 \cdot 45\ MeV)$$

$$D + D \rightarrow T\ (1 \cdot 01\ MeV) + H\ (3 \cdot 02\ MeV)$$

$$D + {}^{3}He \rightarrow {}^{4}He\ (3 \cdot 6\ MeV) + H\ (14 \cdot 7\ MeV)$$

They are more difficult to realise, as they need even higher temperatures, but the lower neutron energy or even absence of neutrons is important benefit.

3. Status of fusion research: magnetic and inertial fusion

From the above, it is clear that the first two major challenges in fusion research are: (i) heating the fuel to several tens of million degrees, which is several times hotter than in the centre of the sun, and (ii) confining the hot fuel in some kind of ‘bottle’. This cannot be a material ‘bottle’ as the highest know melting point is around 3000 C. Thus the ‘bottle’ must be necessarily ‘immaterial’. The solution of these two seemingly impossible requirements necessitates evidently solutions that are radically different from all we know in daily life. There exist presently two approaches: inertial and magnetic fusion (Chen, 2011). Magnetic fusion makes use of magnetic fields. Strong magnetic fields are used to keep the hot particles away from the walls of the confinement device. This is possible because of the property of charged particles to follow a helical path around magnetic field lines caused by the Lorentz force and possible movements perpendicular to the field are thereby highly restricted. This line of research into controlled fusion is being funded in a large number of countries around the globe.

In inertial fusion a small pellet (containing a 50/50 mix of D and T) is compressed using lasers or particle beams and the fuel reacts in the very short time before the pellet is blown apart.

Two main classes of toroidal devices are in use in magnetic fusion research: tokamaks and stellarators. In a tokamak, a set of coils placed around the doughnut-shaped plasma chamber produces the main toroidal magnetic field (Fig. 1). The conducting plasma ring itself serves as the sole secondary winding of an enormous transformer. A current pulse in the primary winding induces a large current in the secondary, i.e. in the plasma ring itself. This induced plasma current generates a poloidal magnetic field. The combination of this poloidal field with the main toroidal field results in a helical magnetic field. The magnetic structure thus generated consists of an infinite set of nested toroidal magnetic surfaces, each with a slightly different twist, reducing further the leakage of particles and heat from the plasma. These surfaces fit into each other just like the puppets in a Russian doll. On each of these surfaces, the plasma pressure is constant and each field line

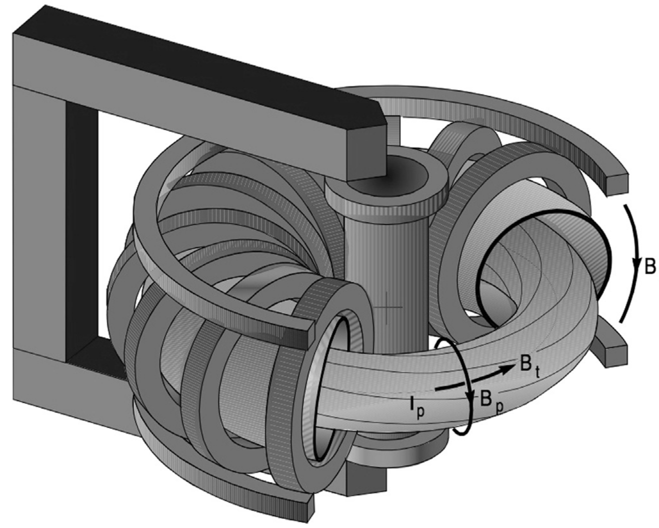

Fig. 1. Schematic representation of a tokamak. The main components are: (i) transformer (yokes and primary windings around the central column) to induce the plasma current; (ii) set of coils around the plasma vessel for the toroidal magnetic field; (iii) planar coils for the vertical field. The poloidal field associated with the plasma current adds to the toroidal field resulting in a helical field, needed for stability. The vertical field is needed to cope with hoop forces originating from the plasma current and the stored energy in the plasma.

lies on one such surface. The tokamak is a pulsed device, since the transformer that induces the plasma current needs a steadily increasing current. For practical applications, continuous operation of such a device would clearly be a great advantage and it is an object of present research.

One way to obtain continuous operation of a fusion device is to avoid the need for the (pulsed) plasma current. The stellarator exploits this idea by relying on currents external to the plasma. Extra helical coils around the toroidal plasma provide the necessary additional twist to the toroidal magnetic field generated by the main field coils. These helical windings around the plasma ring, however, complicate the construction of a stellarator. In addition, their presence renders the accessibility to the device more difficult than in the case of the tokamak. This is why the latest generation of stellarators is based on a new concept: a set of specially shaped coils (Fig. 2) generates the necessary twisted magnetic field configuration directly and eliminates the need for the extra helical coils. Owing to advanced research and calculational efforts, encouraging results are now obtained with stellarators of the current generation. Their performance, however, lags one or two generations behind that of the tokamak. Only research on larger stellarators will show whether these devices

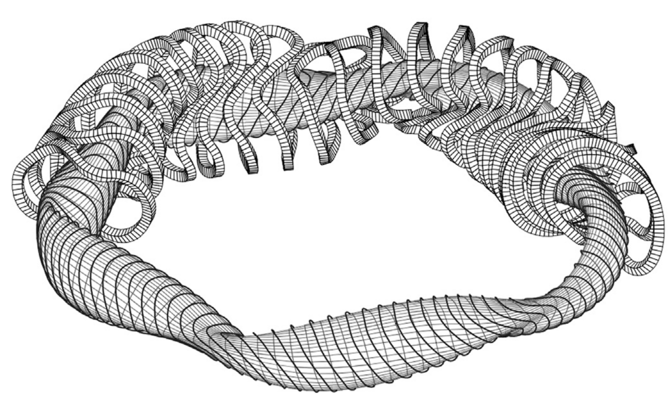

Fig. 2. The complex coil system used in a modern optimized stellarator like W7-X. The twisted plasma shape is also shown.

Please cite this article as: Ongena, J., Ogawa, Y., Nuclear fusion: Status report and future prospects. Energy Policy (2016), http://dx.doi.org/10.1016/j.enpol.2016.05.037

Table 1
Main parameters of past, existing and futurelarge tokamaks around the world: DIII-D (General Atomics, San Diego, USA), JT-60U (Japan Torus 60 Upgrade, Naka Fusion Institute, Ibaraki Prefecture, Japan), TFTR (Tokamak Fusion Test Reactor, Princeton University, Princeton, USA), JET (Joint European Torus, Culham, UK), ITER (International Thermonuclear Experimental Reactor, Cadarache, France).

Tokamak name	DIII-D	JT-60U	TFTR	JET	ITER
Land/Organization	USA	Japan	USA	UK/EURATOM	International
Plasma shape	Elliptical (D)	Elliptical(D)	Circular	Elliptical (D)	Elliptical (D)
Minor radius a (m)	0.67(hor) 1.74(vert)	1.1(hor) 1.4(vert)	0.96	1.25(hor) 2.1(vert)	2.0 (hor) 3.9(vert)
Major radius R (m)	1.67	3.4	3.1	2.96	6.2
Toroidal magnetic field B_t (T)	2.2	4.2	6.0	4.0	5.2
Plasma current I_p (MA)	3.0	5.0	3.0	7.0	15(17)
Pulse length (s)	10	10	7	60	400–3000
Power amplification Q (in D-T discharges)	N.A.	N.A.	0.25	0.7	10
Injection of neutral particle beams (MW)	20	40	40	24	73 in total
Injection of electromagnetic waves (MW)	8	15	12	42	

hold promise for future fusion reactors and several such devices are in operation at this moment. The largest stellarator is Wendelstein 7-X, in Greifswald (Germany). A stellarator-like device of similar size, the LHD (Large Helical Device) of the National Institute for Fusion Science (NIFS, near Nagoya, Japan) is in operation since 1998.

The plasma current, which in the largest tokamaks can amount to several million amperes, serves a double purpose. It is necessary to obtain the correct magnetic configuration for the stable confinement of the hot plasma as explained above. In addition, the plasma current is used to heat the plasma. Since the heating results from the finite resistance of the plasma, just as in an electrical heating element, this is referred to as ohmic heating. The plasma resistance, however, decreases with increasing plasma temperature, and at some millions of degrees, the efficiency of ohmic heating becomes too low to be useful for further heating and at these temperatures the plasma becomes a better conductor than copper at room temperature. Stellarators do not require a plasma current by definition. Thus auxiliary means have to be used to reach the temperatures required for fusion.

One additional heating method consists of injecting energetic particle beams of neutral hydrogen or deuterium into the plasma. This is possible because neutral particles can penetrate the magnetic fields needed to confine the plasma. To this end, ions are created in a plasma source, accelerated with voltages up to 150,000 V and sent through a cloud of neutral gas. The accelerated ions take electrons from atoms in this neutral gas cloud, and become energetic neutral particles that enter the hot plasma ring without impediment from the confining magnetic fields of the fusion device. Once in the hot plasma, they are almost immediately ionised again and deposit their energy via collisions to the rest of the plasma particles. Powers of up to several million watts per neutral injector can be delivered in this way. The acceleration voltage mentioned above is sufficient for tokamak device like JET, with a major radius of about 3 m and a volume of about 80 m^3. In order to deposit the particles close to the centre of the plasma in ITER, the next step device in construction in Cadarache (see below) with a major radius of about 6 m and a volume of about 800 m^3, voltages are needed of the order of 1,000,000 V. This requires a totally different ion generation and acceleration technique, using negative ions instead of positive ions, to optimize the neutralization of the accelerated ions. To this end a specialized laboratory in now in construction in Padova (Italy) (Sonato et al., 2013; Hemsworth et al., 2009; Grisham et al., 2012; Sonato et al., 2009), that should come into operation very soon.

A second heating method introduces electromagnetic waves into the plasma. The electromagnetic energy is delivered to the plasma by antennas or waveguides at the plasma edge. The principle underlying this method is similar to that of a microwave oven: the energy from the waves is most easily absorbed if the frequency used is equal to cyclotron frequency of the particles to be heated. Thus, two different heating systems exist: Ion Cyclotron Resonance Heating (ICRH) and Electron Cyclotron Resonance Heating (ECRH), depending on whether ions or electrons are to be heated. Ion cyclotron frequencies are in the MHz range (20 MHz and upwards), while electron cyclotron frequencies are approximately a 1000 times higher (up to 200 GHz), due to the smaller mass of the electrons. Heating powers for high frequency systems range from 100 kW to several tens of MW. Again to illustrate the challenges in this field of research, ICRF antennas for ITER are currently being designed that need to deliver a heat flux of about 10 MW/m^2 (Messiaen et al., in preparation) in order to reach an input power of 20 MW from a port with a surface of about 2 m^2. This is of the same order of magnitude as the heat flux at the surface of the sun, being about 60 MW/m^2. Another system is Lower Hybrid Current drive with intermediate frequencies in the GHz range, and is mainly used to generate (non-inductive) currents in the plasma. Systems exist with an output power density of 24 MW/m^2 (Mayoral et al., 2009). An overview of the main parameters for several magnetic fusion devices systems in the world is given in Table 1.

As stated in the introduction, inertial fusion consists of micro-implosions (and subsequent explosion) of small spherical fuel pellets by means of powerful lasers or particle beams. Confinement of the fuel is based on the inertia of the pellet fuel mass: the inertia of the imploding fuel keeps it confined (typically about 1 ns) before it is blown up. To achieve sufficient fusion reactions in this short time requires a density of about 1 kg/cm^3, a 10,000-fold increase over the original density and about 100 times the density of lead or 1000 times that of water (Tabak et al., 1994). This approach is being investigated in far fewer countries than is magnetic fusion.

The outermost layers of the pellet consist of material that ablates easily. The intense external radiation vaporizes this layer nearly immediately, and consequently an inward-propagating spherical shock wave is generated by the evaporating material that compresses the central part of the pellet. Two techniques are being investigated to obtain a symmetrical irradiation of the fuel capsule: direct-drive and indirect-drive. In direct-drive one attempts to obtain uniform irradiation of the fuel pellet using a large number of laser or ion beams directly striking the spherical pellet. In indirect drive the pellet is contained within a small, high Z metal cylinder (so-called hohlraum, Fig. 3) with open ends. The hohlraum itself is irradiated on its inner surface by laser or ion beams. The intense beams transform the metal cylinder into superhot plasma that radiates mostly X-rays. Those X-rays are subsequently absorbed by the surface of the fuel pellet, imploding it in the same way as if it had been hit directly with lasers or particle beams. The advantage of the second method is that the radiation is hitting the fuel pellet in a far more isotropic way, ensuring a much

Please cite this article as: Ongena, J., Ogawa, Y., Nuclear fusion: Status report and future prospects. Energy Policy (2016), http://dx.doi.org/10.1016/j.enpol.2016.05.037

Fig. 3. Illustration of the size of a hohlraum as used in inertial fusion research.

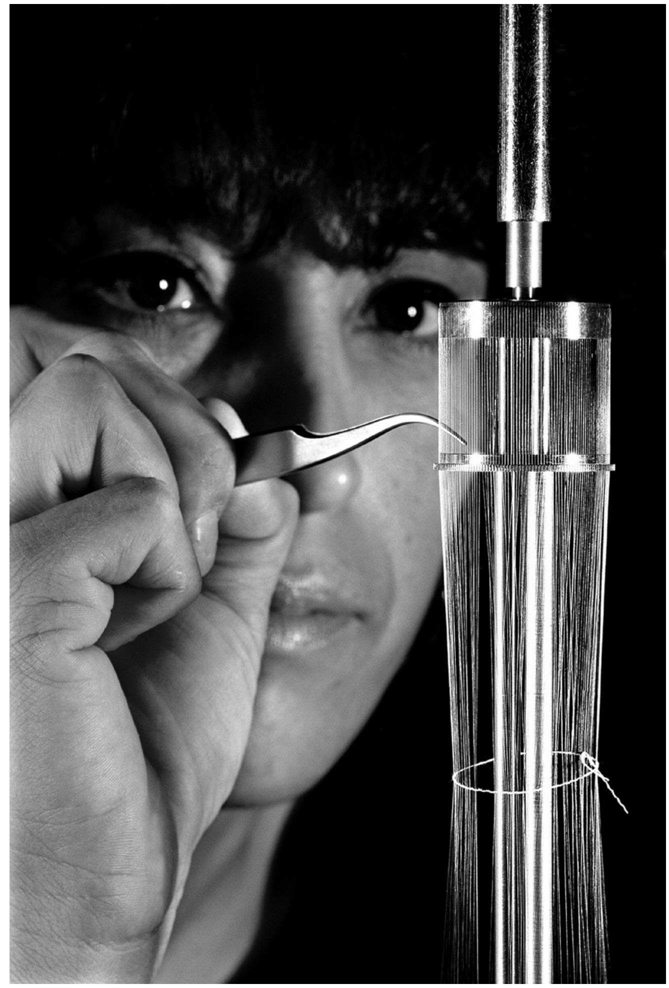

Fig. 4. A liner consisting of 300 tungsten wires, in use in Z-pinch research.

higher quality spherically symmetrical implosion.

The amount of laser energy needed to effectively compress the capsules to ignition conditions using indirect drive is in the MJ range. Separating the compression phase from the ignition trigger can lead to higher gains. This is applied in the fast ignition and shock ignition approaches. In fast ignition (Tabak et al., 1994), the pellet is first compressed to hundreds of times solid density as described above. When the implosion reaches maximum density, a second, ultra-short pulse ($\sim$10 ps) from a PW (10^{15} W) laser system is delivered on one side of the compressed core. It acts like a kind of “spark”: the photons from this pulse interact with the fuel, generating a shower of high-energy (several MeV) relativistic electrons that are driven into the fuel and thus create a hot spot on one side of the dense core. If this is sufficiently localised, it is expected to heat the dense centre of the compressed capsule well beyond ignition conditions. In “shock ignition” the fuel is first irradiated symmetrically, but driven at lower velocity than for central ignition: the temperature of the hot spot generated at the end of the implosion is below the ignition threshold. At an appropriate time, toward the end of the implosion, the fuel capsule is irradiated by intense (quasi-symmetric) pulses, which drive a strong converging shock wave, with an initial pressure of about 300 Mbar. This pressure is amplified by convergence, and further amplified as the converging shock collides with the outgoing shock bouncing from the centre. The hot spot then undergoes additional heating and ignites. One could understand shock ignition as shock-assisted central ignition. The duration of the shock is between 200 and 500 ps with an energy of $\sim$80 kJ corresponding to a power of 150–400 TW (Betti et al., 2007). This new concept has been experimentally studied at the OMEGA laser facility (Theobald et al., 2008, 2009), demonstrating the potential of a properly timed final shock to significantly enhance the neutron yield. Yet another approach is to make use of magnetic fields also in inertial fusion. This is the so-called Magneto-Inertial fusion (MagLIF): the applied magnetic fields reduce the electron thermal conductivity. For more details on this technique we refer to (Gotchev et al., 2015, Cheng et al., 2011).

Getting the laser light onto the small compressed spot (with a size of about 30 μm) is a subject of further research. Two fast ignition approaches are in under study: (i) the “plasma bore-through” method and (ii) the “cone-in-shell” method. In the first approach, successfully tested at the OMEGA device (see below), the ultra short pulse PW laser burns a hole through the compressed plasma heating the dense fuel inside. The second approach, tested successfully on the GEKKO XII laser in Japan (see below), uses a small gold cone that cuts through a small area of the capsule shell; on heating no plasma is created in this area, leaving a hole that can be aimed into by shining the laser into the inner surface of the cone.

A far different approach uses a plasma z-pinch. A z-pinch is the radial implosion of a cylindrical or annular plasma under the influence of a strong magnetic field produced by a current flowing down the length of the plasma. One of the principal z-pinch systems in the world is the Z machine located at the Sandia National Laboratory, Albuquerque, New Mexico, USA (Ryutov et al., 2000). The Z machine sends an extremely large electrical current (up to 18 MA) during about 100 ns into an array consisting of several hundred thin (diameter of $\sim$10 μm), parallel tungsten or steel wires called a liner (see Fig. 4). The high electrical current vaporizes the wires, which are transformed into a cylindrical plasma curtain that implodes under the force of its own radial magnetic field. The imploding plasma generates a high temperature and an X-ray pulse which creates a shock wave in a target structure located on the axis of the cylindrical configuration. The shock both compresses and heats the target. Fusion of deuterium using this system was observed in 2003. A refurbishment of the Z machine to increase its power by 50% was completed in 2007. This newer Z

Please cite this article as: Ongena, J., Ogawa, Y., Nuclear fusion: Status report and future prospects. Energy Policy (2016), http://dx.doi.org/10.1016/j.enpol.2016.05.037

Table 2
Main parameters of ICF laser systems in the world. The power for the separate extreme high power ultrafast fast ignition laser beam (f.i), if available, is listed separately.

Device name	NIF	LMJ	GEKKO-XII	ISKRA-5	OMEGA / OMEGA-EP
Land/Organization	USA	France	Japan	Russian Federation	USA
Energy on target	1.8 MJ	1.8 MJ	10–12 kJ 0.4 kJ (f.i.)	30 kJ	30 kJ 2.6 kJ (f.i.)
Power	500 TW	500 TW	10–20 TW 0.5 TW (f.i.)	100 TW	60 TW 1 PW (f.i.)
Wavelength (nm)	350	350	532	1315	351, 1053
Pulse length (ns)	1–2 ps	1–2 ps	1–2 ns few ps (f.i.)	250 ps	2 ns@351 nm 1–100 ps @1053 nm(f.i.)

machine (called the ZR machine) can now produce up to 27 MA in 95 ns. The radiated power has been raised to 350 TW and the X-ray energy output to 2.7 MJ. Sandia's roadmap includes another Z machine version called ZN (Z Neutron) to test higher yields of fusion power. ZN is planned to produce between 20 and 30 MJ of fusion power at a rate of one shot per hour. The next step planned would be the Z-IFE (*z*-inertial fusion energy) test facility, the first true z-pinch driven prototype fusion power plant. Sandia has recently proposed a conceptual 1 PW Z-pinch power plant, where the electric discharge would reach 70 MA. An overview of Z-pinch research can be found in (Cuneo et al., 2006).

An overview of the main parameters for several ICF devices/ large laser systems in the world is given in Table 2. Further information on heavy ion ICF research can be found in (Logan et al., 2008).

The fundamental physical challenge in fusion research is the necessity to avoid all kind of instabilities and turbulences in the heated plasma fuel. In magnetic fusion, this is the consequence of having to realise an enormous temperature gradient of several tens of millions of degrees per meter; e.g. in JET with a minor radius of ~2 m, a temperature gradient of about 100 million degree C per m has to be maintained. Understanding plasma turbulences is not an easy task, as the movement of the charged plasma particles and the structure of the confining magnetic field are coupled, thus leading to a highly non-linear description. In inertial fusion the Rayleigh-Taylor instability destroys the spherical symmetry of the imploding plasma and thus the ability to reach sufficient fusion reactions. This instability occurs when a heavier liquid is on top of a lighter one: fluctuations in the top layer can easily grow and penetrate the lighter one, with a mix of the two liquids as a result: here the top layer is the ablator material, which is mixing with the fusion fuel. Simulating and understanding such plasmas is a major challenge, with steady progress being obtained in the last decade with the advent of ever faster and larger supercomputers (Krommes, 2012; Howard et al., 2015; Thomas and Kares, 2012). Still, further steps need to be taken before being able to use such calculations as a means to predict the properties of a hot plasma with sufficient confidence, and thus also to predict the engineering parameters (dimensions, plasma density, external heating required, etc.) before arriving at a detailed design of an electricity producing future fusion power system.

A practical way to find out how a next step device could behave is by using so-called "scaling laws". This is a technique much like the windtunnel approach in avionics. E.g. in magnetic fusion several important plasma parameters are collected from experiments on devices with increasing size. Scaling laws are then derived from these data by applying non-linear fits using several engineering parameters as variables (e.g. major and minor radius of the toroidal plasma, plasma density, isotopic composition etc). With these scaling laws extrapolations can then be made to predict the behaviour of 'next step' devices within a certain error bar (Kardaun et al., 2007). Such extrapolations have to be done in a prudent stepwise approach, and each step necessarily needs experimental verification on 'next step' devices with resulting possible adjustments. In addition, each next step size device needs the development of new technologies (e.g. heating systems with higher power and longer pulse length, new materials, new diagnostics…) and the time needed is obviously difficult to plan: how to predict when a important step forward will be made? Moreover, such a planning also depends heavily on international decisions for research budgets, siting of research infrastructures, management of such organizations etc. They are often the result of a compromise, and therefore do not necessarily reflect the most efficient organizational/scientific solutions, causing further delays. It is thus clear that fusion cannot be other than a long term undertaking, and any prediction of when an economical reactor will be build can only be a very rough estimate. To optimize as much as possible the timing for the construction of a first commercial fusion reactor, the strategy is to have as much as possible parallel developments: material research, fusion physics research on both tokamaks and stellarators, inertial fusion research, etc.

Despite all challenges fantastic progress has been made in magnetic fusion in the last decades. Three generations of tokamaks with doubling of characteristic dimensions at each step led to a 10,000 times higher value of the fusion triple product (density times temperature times confinement time, characterizing performance of the fusion reaction) in the last 40 years. Since the start of controlled fusion research, a 10 million-fold improvement in the fusion triple product has been obtained approaching reactor conditions (Wesson, 2004). First large-scale deuterium-tritium experiments took place in the early 1990s. Several MW of fusion power have been released for the first time in a controlled way in deuterium-tritium experiments in JET (Joint European Torus, Culham, UK) and TFTR (Tokamak Fusion Test Reactor, Princeton, USA). Peak values of about 16 MW have been obtained on JET in 1997 corresponding to values for the power amplification Q_{DT} (i.e. the ratio of the power released from deuterium-tritium fusion reactions to the power applied to heat the fuel) close to 0.7. A summary of what has been achieved in high power D-T experiments in EU and USA is shown in Fig. 5.

In addition, ion temperatures up to 45 keV, which is about 30 times higher as that in the centre of the sun, have been achieved in Japanese tokamak JT-60U (Japan Atomic Energy Agency, Naka Fusion Institute, Naka, Japan) (Ishida et al., 1997). Magnetic fusion research has thus now arrived at the point where large amounts of fusion energy can be produced in a controlled way. The next step is to maintain a steady power output from fusion reactions in long pulses. To make this possible, superconducting coils for the magnetic fields will have to be used, because they are not subject to the huge resistive losses of classical copper coils. The use of superconducting coils has been very successfully demonstrated e.g. in Tore Supra (CEA-Cadarache, France) and the Large Helical Device LHD (National Institute for Fusion Science, Toki, Japan). Both machines have record length pulses at relevant plasma parameters in hydrogen and deuterium plasmas: 6 min and 30 s on Tore Supra in 2004 (Bucalossi et al., 2005; Giruzzi et al., 2005) and 54 min 28 s on LHD in 2007 (Mutoh, 2007). Recently (10 Dec 2015) the

Please cite this article as: Ongena, J., Ogawa, Y., Nuclear fusion: Status report and future prospects. Energy Policy (2016), http://dx.doi.org/10.1016/j.enpol.2016.05.037

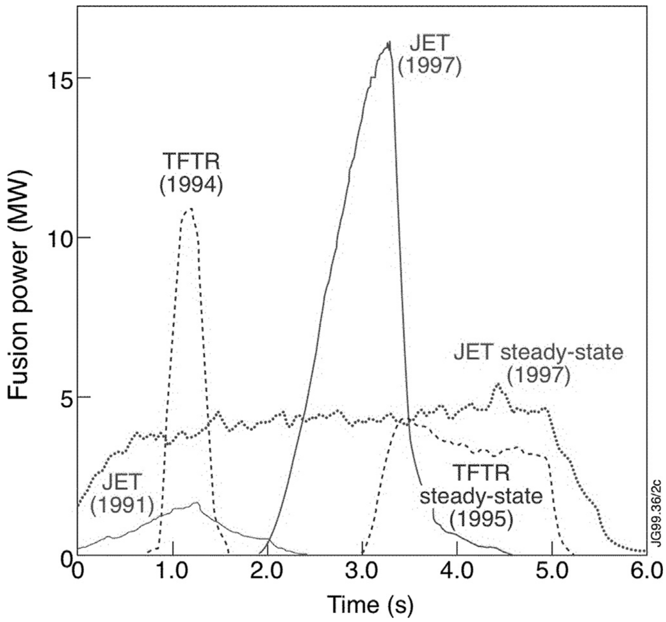

Fig. 5. Fusion power development in the D-T campaigns of the tokamaks JET (Jacquinot and the JET Team, 1999) (full and dotted lines) and TFTR (Hawryluk et al., 1995, McGuire et al., 1995) (dashed lines).

stellarator Wendelstein 7-X (W7-X), (Bosch, 2013) also equipped with superconducting coils, started plasma operation at the Max Planck Institute for Plasmaphysik in Greifswald, Germany. In its final configuration (foreseen in the coming years) the device should be capable of maintaining a hot plasma during 30 min.

Magnetic fusion research entered a new era in 2005 with the international agreement (28 June 2005) on the construction site of ITER at Cadarache (close to Aix-en-Provence in France). Seven large nations are participating in the project: Europe, Russia, India, China, Japan, South Korea and the USA. Construction started in August 2010. First (H or ^{4}He) plasmas are projected for the first half of the next decade and full power D-T plasmas a few years later. Long pulses in D-T plasmas are planned for ITER, with 500 MW of fusion power in pulses of 400 s and longer, and corresponding Q_{DT} values ranging between 10 and 30.

In addition, Europe and Japan have launched a joint project called "Broader Approach" for projects to explore the next step device after ITER, which should be a first prototype of a fusion power reactor and is called DEMO. The "Broader Approach" includes three research projects (Nishitani et al., 2014):

(1) The Engineering Validation and Engineering Design Activities (EVEDA) for the International Fusion Materials Irradiation Facility (IFMIF), a dedicated laboratory to be built in the near future to test candidate first wall materials/alloys for a future fusion power reactor. IFMIF consists of an accelerator-based, D-Li neutron source to produce neutrons with an energy peaking around 14 MeV at sufficient intensity and irradiation volume. The EVEDA phase consists of validating the main technological challenges of IFMIF, accelerator, target and test facility, with the construction of full scale prototypes: a deuteron accelerator at 125 mA and 9 MeV (a reduced version of the 40 MeV, 125 mA accelerator for IFMIF); three different lithium loops (Brasimone (ENEA), Oarai (JAEA) and Osaka University); a High Flux Test Module and He cooling gas prototype in the Karlsruhe Institute of Technology (KIT) and Small Specimens Test Technique in Japanese Universities (Knaster et al., 2013, 2015).
(2) The International Fusion Energy Research Centre (IFERC) in Rokkasho, close to Aomori, in Japan, for DEMO design and R&D. Main aims of this project are to coordinate the DEMO design and R&D, to manage the computational simulation centre and the ITER Remote Experimentation Center.
(3) an upgrade of the tokamak JT-60U, termed JT-60 Super Advanced or JT-60SA (Kamada et al., 2013), which will be the largest superconducting fusion device after ITER. The current construction schedule foresees first plasma experiments in 2019..

To lead a coordinated effort in the EU towards DEMO, the Power Plant Physics and Technology Department (PPP&T) has been set up. The aims of the DEMO studies in Europe are: (i) to quantify key physics and technology prerequisites for DEMO; (ii) to identify the most urgent technical issues that need to be solved in physics and technology and (iii) to plan and implement supporting physics and technology R&D.

Two DEMO design options are currently being investigated by PPP&T.

(1) DEMO Model 1: a "conservative baseline design" that could be delivered in the short to medium term, based on the expected performance of ITER with reasonable improvements in science and technology i.e. a large, modest power density, long-pulse inductively supported plasma in a conventional plasma scenario.
(2) DEMO Model 2: an "optimistic design" based upon more advanced assumptions which are at the upper limit of what may be achieved, leading to a steady state plasma scenario where a large fraction of the plasma current is induced non-inductively, i.e. without making use of the transformer.

Similar reactor studies are on going in other participating countries. In Japan, in order to reinforce the strategy towards the developments for a DEMO reactor, a special team, the so-called Joint-Core Team was set up in the fusion community to study critical issues for a DEMO reactor. In 2014 the Joint-Core team has published a report on the basic concept of DEMO together with structural and technological issues (http://www.mext.go.jp/b_menu/shingi/gijyutu/gijyutu2/056/shiryo/__icsFiles/afieldfile/2015/01/26/1354643_5.pdf). This Joint-Core team has sorted out tasks regarding the development of the design of DEMO, and set up research and development programs to resolve key issues. Based on the guidelines of the report of the Joint-Core team, a Joint Special Design Team was established in 2015 to design a DEMO fusion reactor. The first option for DEMO is a tokamak, but reactor designs based on helical magnetic configurations and laser systems (inertial fusion) are also under study. For example, since steady-state operation is an intrinsic property of helical plasma systems, designs for helical reactors like the FFHR (Force Free Helical Reactor) series are ongoing in the National Institute for Fusion Science (NIFS, Toki city close to Nagoya) in Japan (Sagara et al., 2012).

Progress in inertial fusion research has equally been impressive. In the US, the National Ignition Facility (NIF, Livermore, California), hosts one of the most powerful laser systems in the world. The NIF laser system is designed to deposit 1.8MJ in a few pico seconds, using 192 individual beamlets and was completed in 2009. To characterize the performance of inertial fusion experiments one uses the fuel energy gain G, defined as the ratio of the energy output produced in fusion reactions to the energy input delivered to the plasma. A fuel energy gain of about 2 has been reached recently in NIF (Hurricane et al., 2014) with an energy of 17.3 kJ from fusion reactions compared to an input energy of 8.5–9.4 kJ delivered to the DT plasma. Note that the definition of fuel energy gain G only takes into account the energy coupled to the DT pellet plasma and not the total laser input energy. The

Please cite this article as: Ongena, J., Ogawa, Y., Nuclear fusion: Status report and future prospects. Energy Policy (2016), http://dx.doi.org/10.1016/j.enpol.2016.05.037

European counterpart to NIF is the Laser Megajoule (LMJ), an experimental ICF device being built near Bordeaux, France by the CEA (Commissariat à l'Energie Atomique et aux Energies Alternatives). Laser Mégajoule plans to deliver about 1.8 MJ of laser energy to its capsules, making it about as powerful as NIF. Laser Mégajoule is the largest ICF experiment being built outside the US and uses a series of 240 laser beam lines, organized into eight groups of 30 beams. GEKKO XII is a high-power 12-beam laser system at the Institute for Laser Engineering in Osaka, Japan in use since 1983. The 12 beams of the GEKKO laser are capable of delivering about 10 kJ per 1–2 ns pulse (i.e. 10–20 TW). The Osaka group has proposed a new concept, called fast ignition, using a Petawatt laser, and is now promoting this as the FIREX project (Azechi et al., 2013). The difference between central ignition and fast ignition can be compared to the difference between a diesel and a gasoline engine one from the viewpoint of the ignition mechanism.

Future plans in high power short pulse laser systems are ELI (Extreme Light Infrastructure) in Europe and GEKKO EXA in Japan. Both systems are intended to deliver 0.2 ExaWatt (2×10^{17} W) in femto (10^{-15}) or atto (10^{-18}) second pulses. The European system plans a pulse repetition rate of 1 min, while the system in Japan will have a repetition rate of 2 h, however at a 10 fold reduced cost. Concerning laser fusion reactor design, the group of Osaka University is strongly promoting KOYO-F, a fast-ignition laser fusion reactor (Norimatsu et al., 2009).

4. Fusion fuel resources

Deuterium, a non-radioactive isotope of hydrogen is extremely plentiful as it can be obtained from ordinary water (about 30 g from 1 t (Friedman, 1953)) with cheap extraction techniques using conventional technology. Complete burning of deuterons and the first generation fusion products (T and ^{3}He) results in the overall equation:

$$6D \rightarrow 2\ {}^4He + 2H + 2n + 43 \cdot 3\ MeV$$

providing 350×10^{15} J/ton D. The amount of deuterium in seawater on earth is estimated at 4.6×10^{13} t, equivalent to about 5×10^{11} TWyr.

Tritium is the radioactive isotope of hydrogen. It decays to ^{3}He by emission of an electron:

$$T \rightarrow {}^3He + e^- + 18 \cdot 7\ keV$$

with the rather short half-life of 12.3 years. The quantities available in nature are not sufficient for technical applications. The neutrons produced in the fusion reactions will be used to breed it by bombarding a blanket around the burn chamber containing a lithium compound, according to:

$${}^6Li + n \rightarrow {}^4He\ (2 \cdot 05\ MeV) + T\ (2 \cdot 73\ MeV)$$

$${}^7Li + n \rightarrow {}^4He + T + n - 2 \cdot 47\ MeV$$

Thus the real consumables in the D-T fusion process are D and Li, while T is an intermediate product burned in the fusion reaction.

Lithium, like deuterium, is a widely available element. There are two isotopes ^{6}Li and ^{7}Li, which occur naturally (7.5% and 92.5% respectively). ^{6}Li is the most useful isotope as it reacts with neutrons in the lower energy range ($E < 1$ MeV). Together with the energy released in D-T fusion reactions, it would result in 22.38 MeV per ^{6}Li atom. Thus if one would only take ^{6}Li as resource for T, the energy content of natural Li (for use in the D-T reaction) would therefore be about 27×10^{15} J/ton. Estimated reserves of natural Li are somewhat less than 29 million tons in known ore deposits and brines (Evans, 2008) and about 200 billion tons dissolved in sea water (0.1–0.2 ppm) (Evans, 1978) (but evidently more difficult to extract), equivalent to about 2.5×10^4 and 1.7×10^8 TWyr. With an annual primary energy consumption of currently about 18 TWyr, this would thus correspond to about 1400 years resp. 9.4 million years of primary energy supply to the world. If consumption of ^{7}Li for the production of T could be included (partially or totally) the energy reserves made available by fusion would be further increased.

Since only one neutron is produced in each fusion reaction and since each new tritium nucleus to be bred from Li requires one neutron, it will be necessary to provide an additional neutron source to balance losses in the breeding blanket and the fact that only a fraction (20–40%) of the neutrons undergo a reaction with Li. Possible suitable neutron multipliers are beryllium and lead, using the (n, 2 n) reaction:

$${}^9Be + n \rightarrow 2\,{}^4He + 2n - 1 \cdot 57\ MeV$$

$${}^{208}Pb + n \rightarrow {}^{207}Pb + 2n – 7 \cdot 4\ MeV$$

5. Safety aspects

5.1. Inherent and passive safety

First, the amount of fuel available at each instant is sufficient for only a few tens of seconds, in sharp contrast with a fission reactor where fuel for several years of operation is stored in the reactor core. Second, fusion reactions take place at extremely high temperature. The fusion process is not based on a neutron multiplication reaction. Note however, that a carefully chosen amount of neutron multiplier will have to be incorporated in the design to compensate for unavoidable neutron losses. However this will, by design, not lead to a multiplication of neutrons as in the case of fission, where for every reaction ~3 extra neutrons are generated. With any malfunction or incorrect handling the reactions will stop. An uncontrolled burn (nuclear runaway) of the fusion fuel is therefore excluded on physical grounds. As a fusion system is essentially a gas or pellet burner, operations can also be stopped immediately whenever the need would arise by cutting the gas or pellet supply. Even in case of a total loss of active cooling, the low residual heating excludes melting of the reactor structure (Cook et al., 2001).

5.2. Minimal radioactivity

The basic fuels (D and Li) as well as the direct end product (^{4}He) of the fusion reaction are not radioactive. However, a fusion reactor will require confinement and control of radioisotopes since it has a radioactive inventory consisting of (i) tritium and waste contaminated by tritium and (ii) reactor materials activated by the neutrons of the fusion reaction. Studies (Cook et al., 2001) indicate, however, that an adequate choice of the latter can minimise the induced radioactivity such that recycling should become possible after some decades to a century. Thus, radioactivity does not have to be inherent to nuclear fusion, in contrast to nuclear fission where the fission reaction itself leads to dangerous long-lived radioactive products.

The tritium cycle is internally closed, and the total tritium inventory in the fusion power plant will be on the order of a few kg. An evacuation of the public might not even be needed in case of an accident if a proper detritiation system is implemented. Special permeation barriers will have to be used to inhibit discharge into

Please cite this article as: Ongena, J., Ogawa, Y., Nuclear fusion: Status report and future prospects. Energy Policy (2016), http://dx.doi.org/10.1016/j.enpol.2016.05.037

the environment of tritium diffusing through materials at high temperature (Cook et al., 2001). As tritium is chemically equivalent to hydrogen, it can replace normal hydrogen in water and all kinds of hydrocarbons. It could thus contaminate the food chain when released in the atmosphere. The absorption of tritium contaminated food and water by living organisms is a potential hazard. However, possible damage is reduced owing to the short biological half-life of tritium in the body of about 10 days.

5.3. Reduced proliferation risk

The operation of fusion reactors is not accompanied by the production of fissile materials required for nuclear weapons. Only a significant modification of the fusion reactor – the introduction of a special breeding section containing fertile material - would make the production of weapons grade fissile materials possible. However, the presence of such a section (in an environment where none at all should be present) could be easily discovered by qualified inspectors (IAEA, 1990).

6. Environmental aspects

6.1. Environmental pollution?

The primary fuels (D and Li) and the direct end product (^{4}He) are not radioactive, do not pollute the atmosphere, and do not contribute to the greenhouse effect or the destruction of the ozone layer. Helium is in addition chemically inert and indispensable for superconducting applications. There are no problems with mining (Li) and fuel transportation. No ecological, geophysical and land-use problems exist such as those associated with biomass energy, hydropower and solar energy.

Measures for tritium containment and detritiation of substances contaminated with tritium will have to be taken. During normal operation the dose for the public in the neighbourhood of the plant will only be a fraction of the dose due to natural radioactivity.

6.2. Dangerous waste?

An important advantage of fusion is the absence of direct radioactive reaction products, in contrast to fission, where radioactive waste is unavoidable since the products of the energy releasing nuclear reaction are radioactive.

Adequate disposal of radioactive waste is especially difficult if the products are volatile, corrosive or long-lived. The neutron-activated structural materials of a fusion reactor would not pose such problems and because of their high melting point and their low decay heat, will not necessitate active cooling during decommissioning, transport or disposal. Studies (Cook et al., 2001) show that over their life time, fusion reactors would generate, by component replacement and decommissioning, activated material similar in volume to that of fission reactors but qualitatively different in that the long-term radio toxicity is considerably lower (no radioactive spent fuel).

Fusion could be made even more attractive by the use of advanced structural materials with low activation as e.g. vanadium alloys or silicon carbides. These materials offer in principle the prospect of recycling in about 100 years after the shutdown of the reactor as the radioactivity would fall to levels comparable to those of the ashes from coal-fired plants (Cook et al., 2001) (which contain always small amounts of thorium and other actinides). As already discussed above, there is a large research programme set up jointly between the EU and Japan to prepare the experimental investigation of properties of candidate alloys, with the IFMIF/EVEDA program.

7. Economic aspects

It is obviously difficult to estimate with any useful precision the cost of a system that will only be put into service several decades from now. In comparison with other energy sources, environmental and safety-related advantages and the virtual inexhaustibility of the fuel sources should be taken into account, as well as the evolution of the cost of electricity based on (exhaustible) resources. Present studies, embodying many uncertainties, produce cost estimates, which are close to those of present power plants. Investment costs (reactor chamber, blanket, magnets, percentage of recirculating power…) will probably be higher, but the fuel is cheap and abundant. Fusion is likely to be a centralised energy source. On the basis of present knowledge, technologically sophisticated power plants will probably have an electrical output larger than 1 GW to be economic. The fast neutrons produced in the D-T reaction could be used to produce fissile material in fusion-hybrid breeder reactors. This complementary role for fusion might improve system economics compared with pure fusion systems; however, it would increase societal concerns related to safety, environment and weaponry.

References

〈http://www.mext.go.jp/b_menu/shingi/gijyutu/gijyutu2/056/shiryo/__icsFiles/afieldfile/2015/01/26/1354643_5.pdf〉.
Azechi, H., et al., 2013. Present status of fast ignition realization experiment and inertial fusion energy development. Nucl. Fusion 53, 104021.
Betti, R., et al., 2007. Phys. Rev. Lett. 98, 155001.
Bosch, H.S., et al., 2013. Nucl. Fusion 53, 126001.
Bucalossi, J. et al., 2005. 32nd EPS Conference on Plasma Phys. Tarragona, 27 June – 1 July 2005. ECA vol. 29C, O-4.005.
Chang, P.Y., et al., 2011. Phys. Rev. Lett, 035006.
Cook, I. et al., Safety and Environmental Impact of Fusion. Report EFDA-S-RE-1, EUR (01) CCE-FU/FTC 8/5. April, 2001.
Cuneo, M.E., et al., 2006. Plasma Phys. Control. Fusion 48, R1.
Evans, R.Keith, 1978. Lithium reserves and resources. Energy 3, 379–385.
Evans, R. Keith, 2008. An abundance of lithium. July 2008. 〈www.worldlithium.com〉.
Chen Francis, F., 2011. An Indispensible Truth. Springer Verlag, Berlin, Germany, ISBN: 978-1-4419-7819-6.
Friedman, I., 1953. Deuterium content of natural water and other substances. Geochim. Cosmochim. Acta 4, 89–103.
Giruzzi, G., et al., 2005. Plasma Phys. Contr. Fusion 47 (12B), B93.
Gotchev, O.V., et al., 2015. Phys. Rev. Lett 107, 016105.
Grisham, L.R., et al., 2012. Recent improvements to the ITER neutral beam system design. Fus. Eng. Des. 87, 1805.
Hawryluk, R.J., Adler, H., Alling, P. et al., 1995. Plasma Physics and Controlled Nuclear Fusion Research. In: Proceedings of the 15th International Conference Seville, Spain, 1994, vol. I, p. 11, Vienna: IAEA.
Hemsworth, R., et al., 2009. Status of the ITER heating neutral beam system. Nucl. Fusion 49, 045006.
Howard, N., et al., 2015. Plasma Phys. Control Fusion 57, 065009.
Hurricane, O.A., et al., 2014. Fuel gain exceeding unity in an inertially confined fusion implosion. Nature 506, 343.
IAEA, 1990. Status Report on Controlled Thermonuclear Fusion, Executive Summary and General Overview, prepared by the International Fusion Research Council (IFRC), International Atomic Energy Agency, Vienna.
Ishida, S. et al., 1997. High performance experiments in JT-60U high current divertor discharges. In: Proceedings of the 16th International Conference Plasma Physics and Controlled Nuclear Fusion Research. Montreal, Canada, 1996, IAEA, Viennna, vol. 1, p. 315.
Jacquinot, J., 1999. and the JET Team, 1999. Deuterium-tritium operation in magnetic confinement experiments: results and underlying physics. Plasma Phys. Control. Fusion 41, A13–A46.
Kamada, Y., et al., 2013. Progress of the JT-60SA project. Nucl. Fusion 53, 104010.
Kardaun, O., et al., 2007. Nucl. Fusion 47, S1–S414.
Knaster, J., et al., 2013. IFMIF: overview of the validation activities. Nucl. Fusion 53, 116001.
Knaster, J., et al., 2015. The accomplishment of the Engineering Design Activities of IFMIF/EVEDA: the European–Japanese project towards a Li(d,xn) fusion relevant neutron source. Nucl. Fusion 55, 086003.
Krommes, J.A., 2012. Annu Rev. Fluid Mech. 44, 175–201.
Logan, B.G., Perkins, L.J., Barnard, J.J., 2008. Phys. Plasmas 15, 072701.
Mayoral, M.-L., J. Ongena et al., 2009. In: Proceedings of the 18th Topical Conference on Radio-Frequency Power in Plasmas, AIP Conf. Proc. 1187, p. 39.
McGuire, K.M., Adler, H., Alling, P., et al., 1995. Phys. Plasmas 2, 2176.

Messiaen, A. et al., 2016. Contribution of LPP/ERM-KMS to the modern developments of ICRH antenna systems, under review for Fusion Engineering and Design (in preparation).
Mutoh, T., et al., 2007. Steady-state operation and high energy particle production of MeV energy in the large helical device. Nucl. Fusion 47, 1250–1257.
Nishitani, T., et al., 2014. DEMO activities in the broader approach and beyond. Fusion Sci. Technol. 66, 1–8.
Norimatsu, T., et al., 2009. Activities on the Laser Fusion Reactor KOYO-F in Japan. Fusion Sci. Technol. 56 (1), 361–368.
Ryutov, D.D., Derzon, M.S., Matzen, M.K., 2000. Rev. Mod. Phys. 72 (issue 1), 167–223.
Sagara, A., et al., 2012. Design activities on helical DEMO reactor FFHR-d1. Fusion Eng. Des. 87, 594.
Sonato, P., et al., 2009. The ITER full size plasma source device design. Fus. Eng. Des. 84, 269.
Sonato, P. et al., 2013. Status of PRIMA, The test facility for ITER neutral beam injectors. In: Proceedings of the 3rd Int. Symp. on Negative Ions, Beams and Sources (NIBS 2012, 3–7 Sept. 2012, Jyväskylä, Finland), AIP Conf. Proc. 1515, pp. 549–558.
Tabak, M., et al., 1994. Phys. Plasmas 1, 1626.
Theobald, W., et al., 2008. Phys. Plasmas 15, 056306.
Theobald, W., et al., 2009. Plasma Phys. Control. Fusion 51, 124052.
Thomas, V.A., Kares, R.J., 2012. Phys. Rev. Lett. 109, 075004.
Wesson, J., 2004. Tokamaks, third edition. Clarendon Press., Oxford (Oxford Science Series Nr. 118) ISBN-0–19-850922-7.

Please cite this article as: Ongena, J., Ogawa, Y., Nuclear fusion: Status report and future prospects. Energy Policy (2016), http://dx.doi.org/10.1016/j.enpol.2016.05.037