Ion Sources for Neutral Beam Heating of Fusion Plasmas

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Abstract: In modern Fusion experiments one of the most powerful heating methods is the injection of energetic neutral hydrogen (Neutral Beam Heating). The heart of any neutral beam injector is the ion source. These are large sources which must deliver multiple Amperes of beam current at voltages over 50 kV for time scales of seconds to hours. The current machines are based on positive ion injection for the most part but in the future negative ion injection will play a more substantial role. This paper will give an introduction to the techniques of neutral beam injection. Following this introduction will be details of the RF sources used currently in ASDEX Upgrade and Wendelstein 7X experiments. These are modern state of the art ion sources that can deliver 2.5 MW of neutral beam power for pulses up to 10 s long. The ITER Tokamak will use negative ion based neutral beam injection. After explaining the reasoning behind the switch from positive to negative hydrogen the productions methods and challenges these present for ion source operation will be discussed. IPP Garching has developed the prototype sources for the ITER NBH system. The state of the current experimental efforts at IPP Garching plus the implementation of neutral beam heating on ITER will be covered in brief.

I. I. Introduction

Magnetically confined Fusion research is focused now on building machines intended to show the proof in principle of a fusion reactor, and demonstrate that the technical challenges of the construction and operation of fusion reactor are possible to overcome. The two leading experiments currently under construction in the world that are next generation machines are Wendelstein 7X [1, 2] and ITER [3]. Wendelstein 7X is a Stellerator [4] intended to demonstrate both 30 min plasma pulses and the reactor relevance of the Stellerator concept. ITER, a Tokamak, will demonstrate hour long pulses, operation with significantly more fusion power produced (500 MW) then utilized by the machine and its support systems (50 MW). Both machines need to overcome significant technical challenges both in construction and operation to meet these goals. If it can be successfully realized thermonuclear fusion has the potential to deliver a supply of electrical energy from a fuel source that is, in practical terms, inexhaustible. The cost per kW-hr is estimated to be competitive with all other sources of electrical power and there are no significant amounts of radioactive waste produced that will produce a toxic legacy for future generations. It is for these reasons that significant international effort has been expended in the last 60 years to realize, on a commercial scale, fusion power. Unfortunately, it has not been easy to reproduce in the laboratory the centre of our local star.

The fusion of four protons into a helium atom is what fuels our sun and enables life to exist on our planet. The process that occurs is; however, not a single reaction but instead a series of reactions in the *p-p* chain [5]. The starting point of this chain is the reaction $p(p,e+\nu)d$ and this reaction is mediated via the weak force. Thus it has an overall low probability to occur as weak interactions are very short ranged but this also means that the hydrogen burning stage of a star lasts a long time since the star will not consume its nuclear fuel quickly. As protons are the most common constituent of a stellar interior the most likely second step reaction is $d(p,\gamma)^3$He. From here the chain splits into three branches depending on the abundance of He, density, temperature and chemical composition of the star but in all cases the net result is the reaction $4p \rightarrow {}^4\mathrm{He} + 2e^+ + 2\nu + Q_{eff}$ (chain I:

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26.2 MeV, Chain II: 25.7 MeV, and Chain III: 19.2 MeV). Unfortunately, for the purposes of fusion power generation none of the natural reactions are suitable due to producing a low energy density and the smaller scale of a fusion reactor to the sun cannot tolerate this. Although many potential reactions based on electromagnetic or strong force interactions are possible the reaction that is intended to be used to initially fuel reactors is: $d(t,n)^4He$. Deuterium is available from sea water and tritium can be synthesized via the $^6Li(n,^4He)t$ reaction so fuel availability is assured. The energy of the deuterium and tritium required (the plasma temperature) is also technologically achievable so the reaction rate is sufficient to viably produce energy.

Magnetic Confinement Fusion works on the principle of trapping the particles in a magnetic bottle to ensure the energy required to have a fusion reaction in the first place is not transmitted directly to the wall at a rate which is greater than the wall materials can survive. Of all the approaches that tried to implement this principle in practice the most successful have been the Tokamak and Stellerator. Both concepts avoid the problems with end losses of mirror machines by bending the plasma into a torus shape the difference between them is in how the total magnetic field is made, in particular how the polodial confining field is produced. In the Tokamak this field is produced by inducing via the transformer effect a current in the plasma. In the Stellerator the plasma is given a helical twist as it proceeds around the torus, this is accomplished by shaping of the primary field coils in W7X but other methods have been used successfully. The Tokamak is the more technically mature concept and the plasma current that induces the polodial confinement field also is a source of heating to the plasma. Unfortunately, due to the need to induce this current via an external coil the Tokamak is by its nature a pulsed machine, additionally, it is highly sensitive to plasma conditions and is open to violent disruptions (sudden loss of plasma confinement) leading to substantial energy fluxes that can damage the inner vessel components. The Stellerator is due to the lack of an externally induced current intrinsically long pulse capable and the plasma current, if any, is completely under the control of the experiment. Unfortunately the coils are of a complex shape and are expensive to design and fabricate.

The ITER Tokamak is an international project involving countries all around the world. The site for the experiment is in Cadarche, France. When completed the ITER Tokamak will be the largest in the world with a major radius of 6.2 m, a minor radius of 2 m and an overall plasma volume of 840 m^3. It is planned to achieve a plasma pulse of one hour by the end of the experiment. Auxiliary heating will total 50 MW (33 MW Negative Neutral Beam Heating, and the remainder split between Electron Cyclotron Resonance Heating, and Ion Cyclotron Resonance Heating). The machine is expected to generate a plasma current of 15 MA (the ASDEX Upgrade Tokamak [6] in Garching has between 0.6 and 1.5 MA) and has a toroidal magnetic field of 5.3 T using superconducting coils. It is planned to achieve at least 500 MW of fusion power (or Q>10) by the end of the operation of the machine. Due to the high level of activation expected in the machine all components both in-vessel and located near the torus will have to be remote handle able and repairable.

Wendestein 7X is the last in series of machines developed in Germany by the Institut für Plasmaphysik in collaboration with other laboratories from around the world. W7X will have a major radius of 5.5 m, a minor radius of 0.53 m and a plasma volume of 30 m^3. The initial plasma pulse duration for W7X will be 10 seconds (comparable to a typical Tokamak experiment) but a further upgrade is planned after commissioning to extend that to the originally envisaged plasma pulse duration of 30 minutes. Auxiliary heating will total initially 20 MW (10 MW Neutral Beam Injection, 10 MW Electron Cyclotron Resonance Heating) but of that only the 10 MW ECRH power will be initially long pulse capable, the NBI system is available only for a maximum of 10 seconds ever 3 minutes due to a variety of technical constraints on the cooling available to the injector components. The toroidal field is 3 T and is generated by specially shaped coils. Activation will be much less of an issue in W7X then in ITER and so remote handling is not a requirement.

Table 1. A comparison of the primary machine characteristics between Wendestien 7X and ITER.

Machine Parameter	**ITER**	**W7X**
Major Radius (m)	6.2	5.5
Minor Radius (m)	2.0	0.53
Plasma Volume (m^3)	840	30
Plasma Current (MA)	15	~0
Toroidal Magnetic Field (T)	5.3	3
Auxiliary Heating Power (MW)	50	20
Estimated Fusion Power (MW)	500	N.A.

II. Neutral Beam Heating

Neutral beam Heating is a straight forward way to couple power into a plasma. Ions are built in an ion source, extracted and accelerated up to a specific energy (~100 keV in the case of W7X, 1 MeV in the case of ITER), sent through a gas target to neutralize the beam to allowing it to penetrate the magnetic confinement field of the machine. The particles which exit the neutralizer still as ions are bent out of the beam by a magnet (this is typical but in ITER an electric field will be used) and are sent to an ion dump. Once the neutral beam enters the plasma the neutral particles undergo charge exchange and ionizing collisions with the plasma particles. This transfers energy to the plasma particles (depending on the plasma parameters the amount given to the electron and ions varies). The particles in the beam are also a source of fuel for the plasma and the momentum of the beam particles can be used to generate a current inside the plasma. More details on the process involved can be found in [7].

All neutral beam injector systems have the following components: an ion source, an extraction system, a gas cell neutralizer, a residual ion separation system, an ion dump, a calorimeter, and lastly the beam must pass through a duct before entering the plasma chamber of the machine. The positive ion NBI system used on W7X [8] is a copy of the system currently in use on the ASDEX Upgrade Tokamak [9] with some aspects changed due to the differences in the machines. The ion sources used at W7X are state of the art RF driven large extraction area hydrogen ion sources that are identical to those used by ASDEX Upgrade [10]. The main parameters of the source can be found in Table 2, and the details of the grids can be found in Table 3. The most significant changes for the injector system of W7X are in the magnetic shielding required to reduce the impact of the stray field of the Stellerator and the fast pumping system. Due to the different orientation of the stray magnetic field of the W7X, the planned use of several different field configurations and the need to minimize influence of the shielding material on the main field itself a careful analytical study was made in order to optimize the design of the magnetic screening. The final configuration was optimized in terms of influence on the main field of W7X and beam power density of the residual ions in the ion dump [11]. In W7X the field of the machine will always be present during the experimental day, this field would interfere with the standard operation of the Titanium Sublimation pumps used at AUG.

Table 2. Source parameters for the NBH system of W7X.

Source Parameter	Hydrogen	Deuterium
Applied RF Power	80 kW @ 1 MHz	
Estimated Neutral Beam Power injected into Torus	6.9 MW	9.5 MW
Acceleration Voltage	55 kV	60/90 kV
Extracted Current	90 A	78/60 A
Energy Mix (E^0:$E^0/2$:$E^0/3$)	51:30:19	65:25:10
Typical Gas Flow	25 mbar-l/s	
Pulse Length	5 s	8 s

Table 3. Details of the Grids for the NBH system of W7X.

Beam Area	390 cm^2	
Dimension of Beam	50.6x22.8 cm	
Aperture Diameter	8 mm	
Number of Apertures	774	
Transparency	37%	
Gap Plasma to Decel Grid	60 kV: 8 mm	100 kV: 15.3 mm

As the plasma of ITER is both more dense and larger in cross section than current Tokamak plasmas the energy of the beam required to give good penetration is about an order of magnitude higher (1 MeV). Unfortunately at 500 keV per nucleon the neutralization efficiency for a positive ion is essentially zero. This is due largely to the fact that neutralization of a positive ion requires the capture of an electron at rest by a rapidly moving proton. The neutralization cross section of a negative hydrogen ion (binding energy 0.7 eV) on the other hand is in this energy range essentially 60% and the energy dependence is flat. This is primarily due to the outer most electron being so weakly bound. The reduced neutralization efficiency made the use of negative ion sources unavoidable. But negative hydrogen ion sources while suited to use as accelerator sources produce in a large area source a very low negative hydrogen ion density (~0.1 that of positive hydrogen ion density plasmas) and extraction of a negative ion requires extraction of the electrons in the plasma. This means that a negative ion source for heating fusion plasmas is significantly larger than a positive ion source for the same beam current. The parameters of the ITER ion source [12] are given in table 4.

Table 4. Main paramenters of the negative ion source proposed for the ITER NNBH system.

Source Dimensions	1.95x1.55 m	
Operating Pressure	≤0.3 Pa	
RF Frequency	1 MHz	
Applied RF Power	100 KW/Driver: 8 Drivers	
Extraction Voltage	10 kV	
Accelerating Voltage	1 MeV	
Extracted Current	H: 70	D: 58 A
Accelerated Current	40 A	
Co-extracted Electron to Ion Ratio	H: <0.5	D: <1
Aperture Diameter	14 mm	
Number of Apertures	1280 in 16 groups	
Total Extraction Area	2000 cm^2	

III. Current Status of the W7X NBH System

The NBH system is being built currently and is on schedule and will be ready for the commissioning phase of W7X in 2015. Figure 1 shows the ion source planned to be used while Figure 2 shows a computer drawing of the injector system itself. The injector boxes are being assembled with all internal components (ion dumps, magnetic shielding) plus the external water pipes for cooling components. The two calorimeters are being leak tested as well. A collaboration with IPJ Swierk (Poland) is providing the support structures for the boxes, the second ion removal magnet and gate valve. The port protection is designed and the various components are under construction. The high voltage services are under assembly and will be installed into the Torus hall this year. The cooling water system including pumps, heat exchanger, and distribution piping network will be built as a collaboration with IPJ Swierk and with the assistance of the W7X assembly team. The current status is that the contract and system specification is under preparation. Planning is well advanced for the control and data acquisition systems. The only part of the injector system that has significant potential to force the NBI system off the schedule is the fast pumping system. At this time no definitive proposal exists to provide the required $2x10^6$ L/s pumping speed that is required for operation of the injector system. Three options exist: using the existing AUG pumps but with AC heating of the wire, using DC heating but with a new design of the evaporation rod, or using cryogenic pumps designed and built in collaboration with another laboratory. Both technical and financial considerations constrain the choice and at this time the way forward is unclear.

IV. Negative Hydrogen Ion Source Development

Since 2008 the ITER reference design ion source has been based on the development work carried out at IPP Garching on the Type VI RF driven ion source. The source consists of three regions: the driver, the expansion chamber and the extraction region. The driver is where the RF power is coupled in from an antenna wound around a ceramic cylinder with internal Faraday screen. The expansion chamber is an area where the plasma from the multiple drivers combines and cools on the way to the extraction region. The extraction region is the area just above the plasma grid and is separated from the expansion chamber by a magnetic filter. The magnetic filter serves to both reduce the overall plasma density in front of the plasma grid but to also reduce the electron temperature to ~1 eV. These two actions substantially reduce the destruction of any H- present in the extraction region due to either direct stripping reaction with energetic electrons or mutual neutralization with positive ions. Details on the development of the type VI source can be found in [13, 14].

Negative hydrogen ions are produced by two methods in the ITER ion source: volume production and surface production. Volume production or dissociative attachment is the capture of an electron by an highly vibrationally excited hydrogen molecule and its subsequent break up into a neutral atom and negative ion. This occurs throughout the ion source but as the mean survival path of the ion is relatively short and the optimal

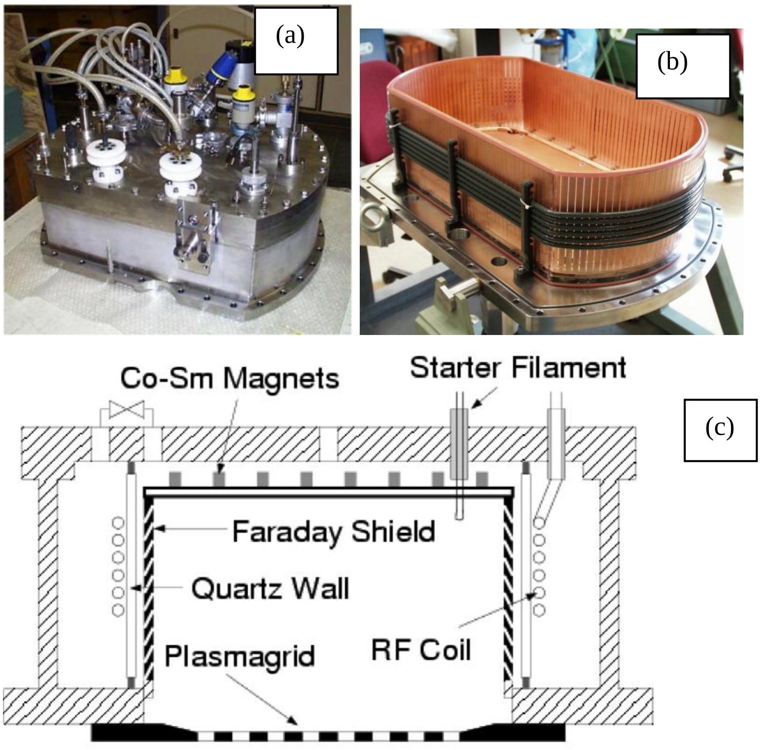

Figure 1. (a) shows a picture of the assembled ion source, (b) shows a picture of the back plate of the source with RF coil, quartz insulator and Faraday screen visible, and (c) shows a sketch of the source with all major components labelled.

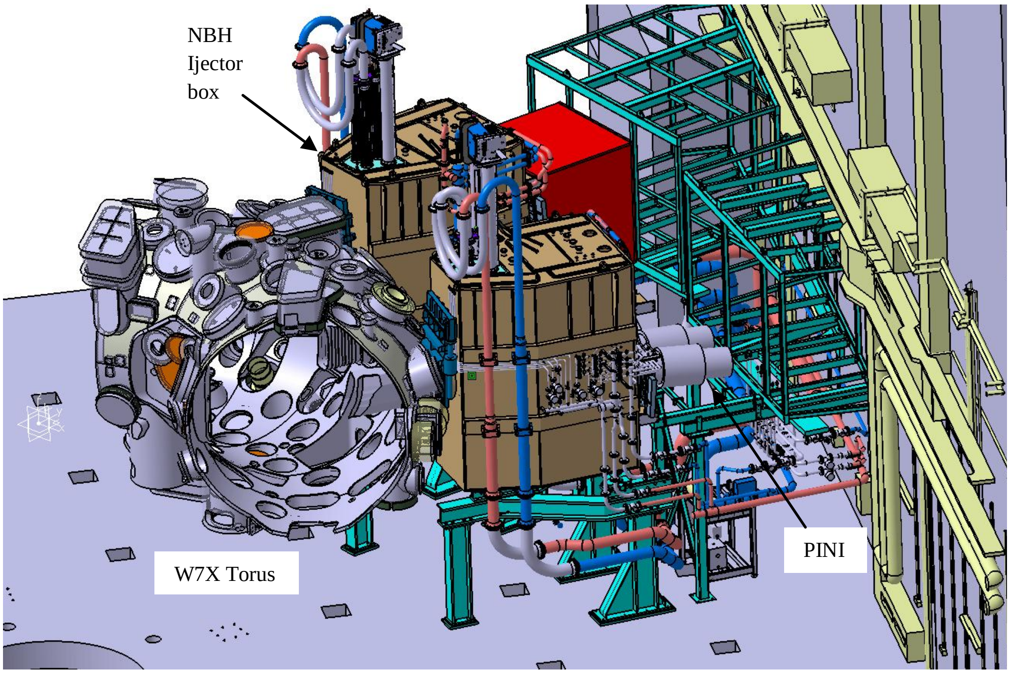

Figure 2. Shown above is a CAD model of the W7X NBH Injector system.

conditions for it to occur are near the plasma grid so overall it contributes only a small fraction of the extracted beam current (~10%). The primary source of negative ions relies on the conversion of atoms and ions on a low work function surface. The best way to produce this surface is to deposit caesium on the plasma grid from a caesium oven. The current designs utilize an oven installed on the rear wall of the ion source but other options are under investigation at IPP Garching [15].

In contrast to the straight forward production and extraction that is present for positive ion sources the situation for negative ion sources is much more complex. Production is a two step process and is strongly dependent on a variety of influences: temperature of the plasma grid and source walls, rate of caesium evaporation, work function of the plasma grid, residual source pressure, chemical interactions on the surface, location and strength of magnetic fields, plasma density and profile, the density of negative ions, applied RF power, caesium transport from the walls, ratio of atomic to molecular hydrogen, and source input gas flow.

Extraction of the negative ions is significantly more complex than in the case of a positive ion source as the vast bulk of the negative ions start on trajectories that are directed towards the centre of the source and the survival length of negative ions is relatively short for the plasma conditions of the ITER source. Thus an additional influence, either a charge exchange collision with an atom or else bending of the ions trajectory in the magnetic fields present in the extraction region is necessary. Theoretical modelling efforts are hampered by both a lack of solid experimental data for the input parameters but also due to the complexity of the situation, which requires treating the plasma in either two or three dimensions thus increasing the computational time and power requirements.

Despite these challenges it has been possible to meet the ITER requirements for the heating beam source and an active research and development program is being carried out at IPP Garching focused on source physics investigations at the test stand BATMAN and a new test stand ELISA which has a source of the same width but half the height of the ITER source and is intend to demonstrate large scale beam extraction and long pulses as well as assist in the technical realization of the ITER NNBH [16] system. A test bed for the full ITER NNBI system (MITICA) is under design in Padua Italy [17, 18] and together with its satellite ion source development test bed (SPIDER) will ensure that ITER will have a functional neutral beam heating system available to it.

V. Conclusion

This paper has given a very general introduction to both the topic of neutral beam heating and the neutral beam injectors used to provide it. Some details on the planed capabilities and current stand of the construction of the NBH system for W7X are given. An introduction into the complex development behind the NBI system for ITER was also provided.

Significant challenges remain for the realization of fusion power on a commercial scale. However, NBH is a well established source of heating, current drive and plasma fueling for the existing large fusion experiments including ITER and W7X. Positive ion based NBH is an established technology and there is essentially no new development in it being undertaken. In contrast to this negative ion based NBH is still in its infancy and significant work will be required to reach the same reliability that existing positive ion systems now have.

It is expected though that whether positive or negative ion based NBH systems will provide vital contributions to the development of fusion as an energy source for the future.

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