R &D OF ENERGY TECHNOLOGIES

ANNEX A
VI-NUCLEAR FUSION ENERGY

- TABLE OF CONTENTS -

AVI-1 Research and Development Opportunities for Fusion Energy

Raymond Fonck
University of Wisconsin – Madison

The goal of the world fusion energy research programs is the development of practical energy production from the fusing of light nuclei, e.g., deuterium and tritium, in a hot ionized gas (or plasma). Fusion research is still in an overall concept development activity, and has not reached the stage for building a demonstration reactor. Its ultimate success of producing an economically attractive new energy source lies well into the future.

The realization of an attractive fusion reactor concept will depend on future developments in fusion science and technology, economics, energy needs, national priorities, etc. Many outstanding scientific and technical issues have to be resolved, and it is premature to define the exact features required for an attractive energy source based on fusion. Nevertheless, some goals for achieving an attractive fusion energy concept are readily identified: optimize the plasma pressure and the energy confinement time; minimize the recirculating power needed for sustainment; and develop a simple and reliable plasma confinement configuration. Achieving these goals involve both understanding the plasma science and developing dvanced technologies.

Sustained and rapid progress in increasing the fusion power output from laboratory experiments has been achieved over the past few decades of fusion research, and there is broad consensus that fusion research is ready to cross a major threshold: the production and study of a burning plasma, wherein substantially more energy is released from fusion reactions than is used to heat the fusion plasma fuel. Such a step appears likely for magnetic confinement fusion in the proposed international burning plasma fusion facility ITER, which would operate in the middle of the next decade, and for new inertial confinement facilities under construction in the U.S. and France.

AVI-1.1 Status and Opportunities for the Conventional Tokamak Configuration

The majority of effort in fusion energy development worldwide is centered on magnetically confined fusion plasma concepts, wherein a hot plasma with density and temperature sufficient for fusion burn is confined continuously in a toroidal magnetic bottle. The Tokamak plasma confinement concept is the most technically advanced approach for magnetic confinement fusion energy (MFE) research. As such, it has received the bulk of attention for moving to the burning plasma regime and for an eventual demonstration reactor. It can be considered to be the baseline approach to achieve controlled fusion conditions. The proposed ITER device is based on a Tokamak design. R&D activities needed to prepare for and exploit the ITER project are being addressed in existing large tokamak facilities, such as JET (UK) and JT-60U (Japan) as well as smaller devices such as the DIII-D facility (U.S.) and others throughout the world.

- Scientific issues that need investigation prior to and during the operation of ITER include:

- Understanding turbulence and turbulent transport in toroidal plasmas
- Stability limits to plasma pressure
- Plasma behavior and organization in self-sustaining systems
- Control of a sustained burning plasma
- Control of power and particle exhaust

These issues map to more specific technical topics that are being addressed by the Tokamak research communities. These include: development of stabilization techniques for large-scale instabilities which arise from plasma pressure gradients; development of an understanding of the plasma edge region where large pressure gradients and steep profiles strongly influence the overall plasma performance through both steady-state properties and intermittent rapid bursts of heat and particles to the plasma facing components; limits to operation at high density; turbulent transport of energy and particles; avoidance and mitigation of large-scale plasma termination events when stability limits are exceeded; and development of more advanced Tokamak operating regimes with enhanced confinement and stability limits.

Other issues needing investigation include nuclear technologies relevant to burning fusion plasmas and long-range issues for reactor applications. Relevant to the next-stage burning plasma phase of fusion energy R&D, they include:

- Test and performance evaluation of tritium breeding blankets
- Superconducting magnet technology
- High heat-flux component development
- Remote handling technology
- Material integrity at low-to-modest energetic neutron fluence

Additional required areas of technology developments include advanced plasma technologies such as plasma heating and/or control technologies, plasma chamber technologies, and fusion materials. Plasma technologies, such as radiofrequency wave antennae, feedback control coils, diagnostics, and plasma facing components, must be developed to survive the heat-fluxes and radiation levels in the reactor wall region. Novel chamber technologies may be required to handle the higher heat loads in future fusion systems, especially as they become more compact.

A critical technology issue for fusion energy is the development of low-activation materials that can survive in a sustained fusion environment. The availability of appropriate materials impacts the performance, safety, costs, and overall attractiveness of an eventual fusion system. Hence, this is an issue of great importance to the eventual development of a D-T fusion concept. At least one separate facility, such as the proposed International Fusion Materials Irradiation Facility (IFMIF), will be required. The U. S. fusion community also foresees the need for a component test facility for testing materials and components at near full scale.

AVI-1.2 Innovation and New Paths for Magnetic Fusion Energy R&D

Further improvements beyond the conventional Tokamak will likely be required for an attractive energy system. To varying degrees, these improvements include efficient stable sustainment to steady-state and higher fusion power density at increased plasma pressure. The world fusion program is pursuing several research directions to achieve these improvements while understanding outstanding scientific issues. This requires R&D on a variety of magnetic confinement configurations to address fundamental issues such as:

- Optimization of plasma pressure limits and control of high-pressure plasmas
- Self-organization and sustainment of plasmas
- Role of magnetic field symmetry in confinement and sustainment
- Energetic particle behavior
- Stochastic magnetic fields and plasma organization

Practically addressing these issues requires studies of a range of differing magnetic configurations to develop the knowledge base for future fusion systems. These include:

- Developing a path to ***Advanced Tokamak*** regimes, with steady-state operation and enhanced confine~~n~~ment.
- Exploring the potential for the ***Spherical Torus*** configuration to provide a compact system at very high pressure for potentially reduced costs.
- Exploring sustainment and improved confinement in the ***Reversed Field Pinch***, a low-field system with magnetic fields generated mainly by internal plasma currents.
- Exploiting underlying magnetic symmetries in 3-D ***Stellarators*** for passive stability and steady-state operation.

On a smaller scale, research in speculative emerging concepts explores the possibility of even more innovative solutions to the magnetic fusion problem. For example, some configurations are spherical in topology rather than toroidal and hence offer considerable advantages in engineering simplicity if several scientific challenges can be successfully resolved.

The technology needs of fusion concepts are similar for most magnetic confinement configurations. The degree of similarity varies, of course, with each potential concept. For example, since the stellarator is inherently a steady-state concept, it has much less need of new sustainment techniques or technologies. Any of the innovative concepts which lead to more compact systems, such as the spherical torus or advanced Tokamak, increase the need for more innovative plasma chamber and materials solutions.

AVI-1.3 Status and Opportunities for Inertial Confinement Fusion

Inertial fusion energy (IFE) employs a very short pulse of high energy to rapidly compress and heat targets to produce small dense plasmas for a brief fusion burn. The driver of this implosion can be either direct, wherein the energy is directly deposited on the target for compression, or indirect, wherein the energy is converted to X-rays which then compress the plasma. Challenges for IFE include the development of knowledge of target physics in order to design high-gain targets and the development of repetitive driver technology.
Laser-driven indirect drive presently receives the bulk of attention in inertial fusion research, due to its relatively advanced level of development and its relevance to national security interests. X-ray holhraums are used to indirectly implode the target capsules to fusion conditions. Such experiments will provide the critical physics basis for IFE target development in the future, and are relevant to all future IFE drivers of interest. A particular goal is the development of target physics leading to ignition and high-yield/high-gain

targets for IFE applications. Investigations along these lines are in progress in the US, Japan, and Europe, and will culminate in future facilities such as NIF in the US and LMJ in France in the next decade. These latter new facilities will address for the first time the production of a burning plasma through a properly stable symmetric implosion of a target.

AVI-1.4 Challenges and R&D opportunities: concept development and optimization for IFE

Advances are required with new concepts for realizable IFE. The general challenges are similar to those for magnetic fusion, but the criterion of economics is driven more by the development of driver technologies than efficient high pressure magnetic configurations. Issues include development of high repetition-rate driver technologies, target physics, target fabrication, target injection/placement, power focusing, and chamber technologies.

Drivers for IFE can be pursued in 3 directions: heavy ion beams (indirect); high average power lasers (direct); and Z-pinch X-ray production (indirect). Challenges for the heavy ion beam include ion source development, beam transport, and integration of the beam to placement on a suitable target. The laser driver requires the laser development itself and direct-drive target production. Issues of target injection and tracking are common to both approaches. The Z-pinch offers challenges of recyclable transmission lines and applicable holhraum design.

Fast ignition is a concept wherein short intense energy pulses are introduced (e.g., via petawatt lasers) to create a local hot spot to initiate the fusion burn, thereby increasing the gain of the fusion implosion. This enhancement is compatible with all drivers under consideration, and is being actively explored in Japan and the US.

Chambers must be designed to withstand the intense flux of high-energy neutrons on a repetitive basis, and thick liquid wall concepts are under study for heavy-ion and Z-pinch concepts. An integrated test facility will likely be required to demonstrate high yield targets for the chosen IFE drivers, and produce repetitive pulses for component testing relevant to future demonstration plant concepts.

AVI-2 FUSION ENERGY USING TOKAMAKS

P.K. Kaw
Institute for Plasma Research
Bhat , Gandhinagar India 382428

Fusion energy is the energy of stars. The most likely fusion reaction to be used in the first generation fusion reactors on earth (because of its relatively high cross sections) is the deuterium – tritium fusion reaction :

$$D + T = He + n + 17.6 \text{ MeV}$$

The stakes for the early development of fusion reactors are high because of several inherent merits of such systems. Firstly the fuel for this process is limitless and will last humanity a long time. Deuterium is a heavier isotope of hydrogen, and is a component of heavy water, which is naturally distributed in the ratio 1 part in 6000 in sea water. Tritium can be bred from Lithium which is widely distributed in the earth's crust and the oceans. Both deuterium and tritium will be readily accessible to most nations giving widespread energy security. Secondly, the energy is very clean. There is no atmospheric pollution, no green house gas emissions etc. Thirdly, the radioactivity from waste products in fusion is negligible and in principle, can be totally eliminated. This is because, unlike fission, the reaction products themselves are non radioactive. Radioactivity is induced in the surrounding materials by the fast neutrons coming from the fusion reaction. One can reduce its hazards by utilizing low activation materials, which are currently a major area of development. One can also totally eliminate it by the use of advanced fusion reactions like proton - Boron reactions, Deuterium- Helium 3 reactions etc, where energy comes out mostly in the form of charged particles. Fourthly, fusion reactions are inherently safe; the reaction is difficult to ignite and there is no possibility of a chain reaction or a melt down. Lastly, there are no dangers of proliferation, no worries that some rogue nation or terrorist group will steal strategic material for a nuclear weapon. Thus fusion looks like the ultimate solution to the energy problems of mankind. Major programs for its development are going on around the world.

Since nuclei carry a positive electrical charge, fusion can take place only if the nuclei are given sufficient kinetic energy to overcome the forces of repulsion. This can readily be accomplished by working with the reacting materials at a high temperature, typically millions of degrees .This is the basis of thermonuclear fusion and typically puts matter into the plasma state. Thus the problem of controlled thermonuclear fusion is directly related to the problem of production, heating and confinement of a thermonuclear plasma. In the basic D – T reaction the neutron comes out with 14 MeV and the Helium nucleus (or alpha particle) with 3.6 MeV. The neutron being a neutral particle comes out of the plasma and can be trapped outside in a moderator blanket where its kinetic energy gets converted to heat and steam. The steam may then be used to run a usual steam turbine cycle for the generation of electricity. The Helium nucleus on the other hand, being charged, is trapped and slowed down by the plasma, thereby keeping the plasma hot. Thus, once the plasma reaches fusion temperatures, its temperature can be maintained by the alpha particles and we need only supply the fuel in the form of deuterium tritium mixture and remove the ash in the form of slowed down Helium nuclei. Such a thermonuclear plasma which maintains itself hot is known as an ignited plasma. For a successful deuterium tritium fusion reactor the thermonuclear plasma has to be maintained at a temperature of about 100 million degrees (~ 10 keV). At a fundamental level the fusion reaction must produce more energy than the energy required to maintain the plasma at fusion temperatures. If we say that the fusion temperature is to be maintained by alpha particle heating alone we find the critical condition for DT ignition as

$$n \,.\, T \,.\, \tau > 5 \times 10^{21}\ m^{-3}\ keV\ sec$$

where n is the plasma density as measured by number of particles in a cubic meter, τ is the energy confinement time of the plasma in seconds and T is the plasma temperature in keV. Most potential schemes for fusion reactors aim at fulfilling the above mentioned criterion.

Magnetic confinement, as opposed to inertial confinement, relies on the utilization of magnetic fields to provide plasma confinement. We may thus work with sub-atmospheric plasma densities (say 10^{20} per m^3) at fusion temperatures (T ~ 10 keV) and try to achieve plasma confinement times of the order of a few seconds. This is being attempted in Tokamaks and stellarators. Tokamaks are magnetic bottles in which the magnetic cage confining the plasma is produced by a combination of external coil currents and plasma currents . In a stellarator, on the other hand, the confinement configuration is produced by external coils only. We shall now review the work on Tokamak fusion systems.

Tokamaks were invented by Russian scientists, Sakharov and Tamm[1]. Fig. 1 shows the schematic of a typical Tokamak.

The toroidal vacuum chamber is evacuated to a base pressure of about 10^{-8} torr and filled with fuel mixture (say D – T gas) at a pressure of 10^{-4} torr. A toroidal field of few Tesla is produced in the chamber by external toroidal field coils. An ohmic transformer in the central hole of the torus is used to induce a breakdown voltage in the toroidal direction. The gas breaks down and a fully ionized D – T plasma is produced. The ohmic transformer voltage next drives a toroidal plasma current of order a few Mega Amperes which produces the Tokamak configuration. Additional coils with toroidal currents are used to assist the plasma equilibrium against its natural tendency to expand in major radius due to electromagnetic hoop forces. These coils may also be used to shape the cross section of the plasma current filament, giving it geometrical features like elongation and triangularity, which are useful from the point of view of stability and confinement. With shaping coils, one also often creates the plasma divertor configuration for impurity control ; in this configuration surfaces with magnetic x – points separate closed magnetic surfaces from those which have open field lines that are led away from the hot plasma towards an independently pumped region with specially prepared target plates. The plasma current in aTokamak not only provides the equilibrium, but also heats the plasma to temperature of the order of 10 to 20 million degrees due to resistive Joule heating mechanism.

Conventional Tokamaks face several difficulties from the point of view of reactor operation. Firstly, since the current in the plasma is driven by an ohmic transformer which soon runs out of stored flux , one cannot keep the current going indefinitely.

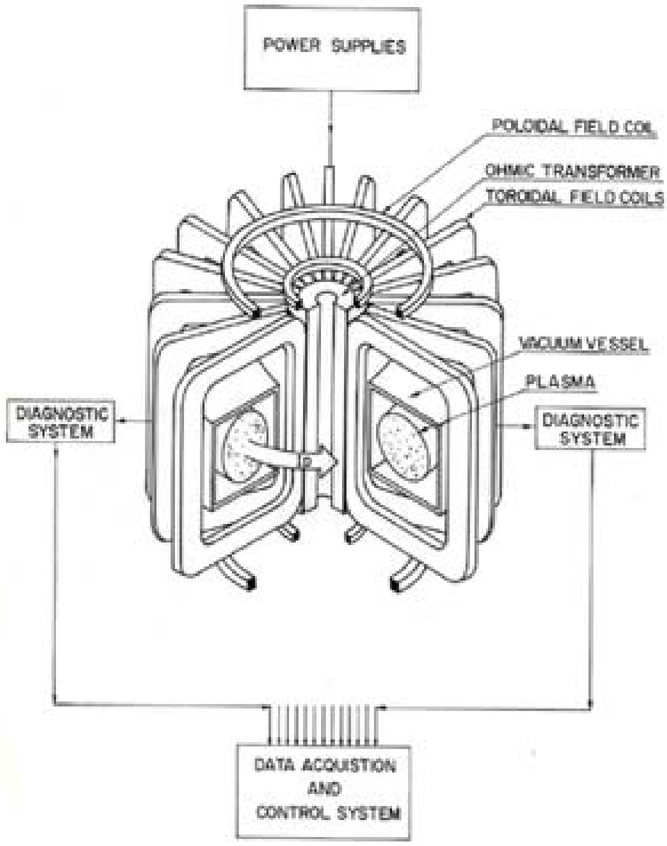


Figure 1- Schematic of a Tokamak system

Thus conventional Tokamaks are necessarily pulsed devices. Secondly, heating due to Joule heating becomes inefficient as the plasma temperature rises and so it becomes difficult to heat Tokamak plamas to fusion temperatures (~ 100 million degrees) by this process alone. Thirdly , the confinement of heat and particles in a Tokamak plasma is anomalous and is determined by turbulent processes rather than collisional transport. This has made it essential that the physics of microinstabilities in Tokamaks be understood sufficiently well so that the confinement may be improved. The first two of these difficulties have been removed by the utilization of radio frequency waves and neutral particle beams to produce heating and sustained current drive in Tokamak plasmas[2]. Energetic neutral beams of deuterium or tritium particles coast across the magnetic fields, get ionized by charge exchange collisions and become energetic ions trapped by the magnetic fields; these energetic particles then slow down by friction on the plasma electrons and ions thereby delivering their energy and momentum to the plasma. In this manner, neutral beams with tens of Megawatts of power have been coupled to Tokamak plasmas and have been used to produce temperatures up to 400 million degrees; these have also been used to drive significant amount of plasma current. The radio frequency waves propagate around several characteristic frequencies in the plasma. The lowest frequencies of interest are the ion cyclotron range of frequencies (ICRF) which are typically in the range of tens of MHz and the highest frequencies are the electron cyclotron frequencies which are close to 100 GHz. Significant amount of power (typically ~ 10 MW can be coupled at these frequencies and leads to intense electron and ion heating in the plasma. Selective and localised electron heating can also be used to influence the current profile in the plasma since the plasma conductivity is influenced by electron temperature. Current drive has mostly been accomplished by utilizing a wave at an intermediate frequency known as the lower hybrid frequency; this is typically in the range of a few GHz. Lower hybrid waves have been used to drive megaamperes of current for long periods of time showing the feasibility of a steady state Tokamak reactor[2]. .

Early experiments on Tokamaks were largely done in Russia. In 1968, a team of Western scientists made temperature measurements on the T – 3 device using the Laser Thomson scattering technique and verified that the Soviet claims of relatively high temperature in this device were correct. This opened the floodgates of modern research in tokamak physics. New and larger devices like PLT, ALCATOR, TFTR, D-III D (in the US), JET, ASDEX, TORE SUPRA (in Europe) and the JT-60U and other devices (in Japan) were constructed and experimented upon. A number of new auxiliary heating and current drive methods were tried and new regimes of tokamak operation and performance were discovered. The progress as measured by the rise in the critical figure of merit viz. the n Tτ criterion has been phenomenal (Fig. 2).

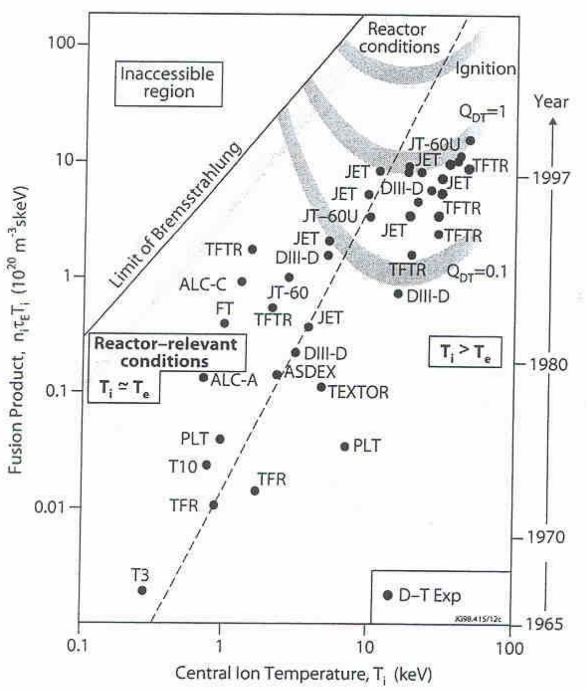


Figure 2- Progress in the critical triple product in fusion experiments

This parameter has increased by about 5 orders of magnitude. Scientific breakeven point has been crossed and one is within a factor of 3 – 5 from the final ignition criterion. Experiments have been done with D – T plasmas and fusion powers of up to 16 MW have been sustained for several seconds. Temperatures up to 400 million degrees and plasma currents up to several mega amperes in reactor size plasmas (~ several meters in minor diameter) are being routinely obtained and manipulated[2].

During this research a great deal of understanding of the physics of Tokamak plasmas has also been achieved[2] and performance accordingly optimized. Thus it is evident that the magnetic cage confining the plasma must operate in a parameter space where the fast growing magnetohydrodynamic (MHD) instabilities are absent .Therefore a lot of attention has been paid to the study of macroscopic MHD stability of Tokamak plasmas (to internal and external kink modes and ballooning modes). These instabilities are now fully understood and codes are routinely used to shape the plasma current filament cross section and aspect ratio to maximize the plasma beta, viz. ratio of plasma pressure (which determines the fusion power density) to magnetic field pressure (which determines the capital cost). The slow macroscopic modes such as resistive wall and neoclassical tearing modes are tolerated and can be kept under control by feedback and rf current drive methods. From the confinement point of view, the microscopic instabilities which drive up the level of fluctuations that determine the cross field transport from the magnetic cages (such as the ion and electron temperature gradient mode in the core , the resistive ballooning and interchange mode in the edge region etc), have been identified and regimes of operation found where they give acceptable transport. The overall data base has led to important scaling laws for energy confinement which can be used to design new experiments. An understanding of the nonlinear saturation mechanisms of relevant microinstabilities has been achieved and modeling of bifurcation transitions from regimes with high fluctuations (L mode) to that with low fluctuations and self consistently generated velocity shear (so called H mode) is in an advanced state. Advanced confinement modes with reversed magnetic shear in the core, an internal transport barrier with near collisional confinement for ions and a high percentage of self consistently generated bootstrap current have also been discovered and sustained for several seconds in large plasmas. The performance of the edge and divertor region can also be modeled by special multidimensional codes which retain atomic physics, radiative effects, neutral dynamics, plasma surface interaction effects like recycling, erosion etc and these are also improving by comparison with experiments. Today one can say that the overall modeling of Tokamak plasmas has reached a very high degree of sophistication .

The next set of Tokamak experiments must address questions related to steady state operation, operation with burning plasmas and the selection of appropriate fusion materials. The longest duration of a Tokamak discharge has been carried out with the TRIAM experiment in Japan where a current of less than 50 kiloamperes was maintained in a small device for about 3 hours with the help of lower hybrid current drive. Long pulse experiments with duration of the order of 100 seconds at higher plasma parameters (current ~ Mamps , temperature of order ~ few keV) have been carried out in the TORE SUPRA device in France. Both of these devices use superconducting toroidal field coils and have no divertors. They are therefore able to do a limited amount of particle control at the boundaries. New experiments with superconducting toroidal and poloidal field coils which will produce shaped plasmas and a divertor configuration and be able to address key questions of steady state operation, namely, fuel and impurity particle control at the boundaries , heat removal in a steady state at the boundaries etc are currently under construction in India (SST1 device), China (EAST experiment) and Korea (KSTAR). These experiments should start yielding useful experimental information within 1-2 years time. The earlier DT experiments were carried out in the TFTR device in Princeton and the JET experiment in Europe. These experiments showed that fusion power up to 16 MW could be generated for several seconds and also highlighted the role of several novel physics effects.

It was demonstrated that the alpha particles generated by the fusion reactions lose their energy by friction with the background plasma particles through the Coulomb scattering mechanism and should thus play their expected role in keeping an ignited plasma hot . There was a preliminary indication that the energetic alphas may be driving Alfven eigenmodes unstable; however, the deleterious effects of these eigenmodes on the confinement of alphas was not observed. This is something which will have to be carefully examined in the future experiments, especially in plasmas where the pressure of energetic alphas has a significant value. Another good feature which was observed in the early experiments is the improvement in the confinement time with the introduction of tritium because of a favorable mass dependence of the anomalous transport processes. Burning plasma experiments will also have to be analyzed for problems due to ash (helium) accumulation, influence of alpha particle internal heating on the L to H transitions (as opposed to transitions induced by external heating sources), temperature control in the presence of fusion related thermal instabilities and so on. Another major area of experimentation has to do with the choice of fusion reactor materials for the first wall, vacuum vessel,blanket modules, coil insulations etc; the key areas of concern are the performance in the presence of intense neutron , energetic particle and heat fluxes , desire to have low activation, design requirements of low erosion and recycling from first wall materials and so on. Since as yet we have no major facilities in which materials can be suitably tested under fusion neutron energy and flux conditions, there is considerable amount of materials research which needs to be done. This will have to be done in large scale fusion experiments like ITER and/or in specially constructed new facilities like the IFMIF (International Fusion Materials Irradiation Facility) which is currently under active discussion.

Progress in Tokamaks has culminated in the design of the next step experiment[3], namely ITER (International Thermonuclear Experimental Reactor), which will tackle many of the residual physics and technology issues for fusion reactors, such as, physics of burning plasmas , steady state operation, choice of fusion materials and their development etc. The basic parameters of ITER are toroidal field 5.3 T, plasma current 15 MA, major/minor radius 6.2m/2.0m, elongation 1.7, triangularity 0.33 , output fusion power 500 MW , neutron flux at first wall 0.57 MW/m^2, plasma duration > 300 seconds. Auxiliary heating power up to 73 MW will be available for heating and current drive. For the main chamber, beryllium is chosen as the first wall material, and for the divertor region CFC and tungsten will be used . Tritium breeding experiments will be attempted with special blanket modules containing lithium ceramic breeding materials. ITER (Fig 3) is an international experiment involving China, Europe, Japan, Korea, Russia and the U.S. ITER construction is likely to start around 2006 and operation is likely in 2013.

Considerable amount of work has already been done on fusion power plant design based on conventional Tokamaks[4]. Typical plants are for 1-2 GW Systems and give a cost of electricity in the range of 6-8 US cents/ KWh around the year 2050. It is envisaged that between ITER and the commercial plants there will be a prototype demonstration plant DEMO. Traditional fusion program assumes that commercial plants will be ready and economically viable by the year 2050. The longest lead time seems to be needed for the development of appropriate fusion materials. If this depends on the use of ITER, it will get delayed by about 15-20 years. There is a fast track fusion program that has been examined in Europe[5], which envisages a parallel development of fusion materials on facilities like the IFMIF and deployment of fusion reactors commercially as early as the year 2035.

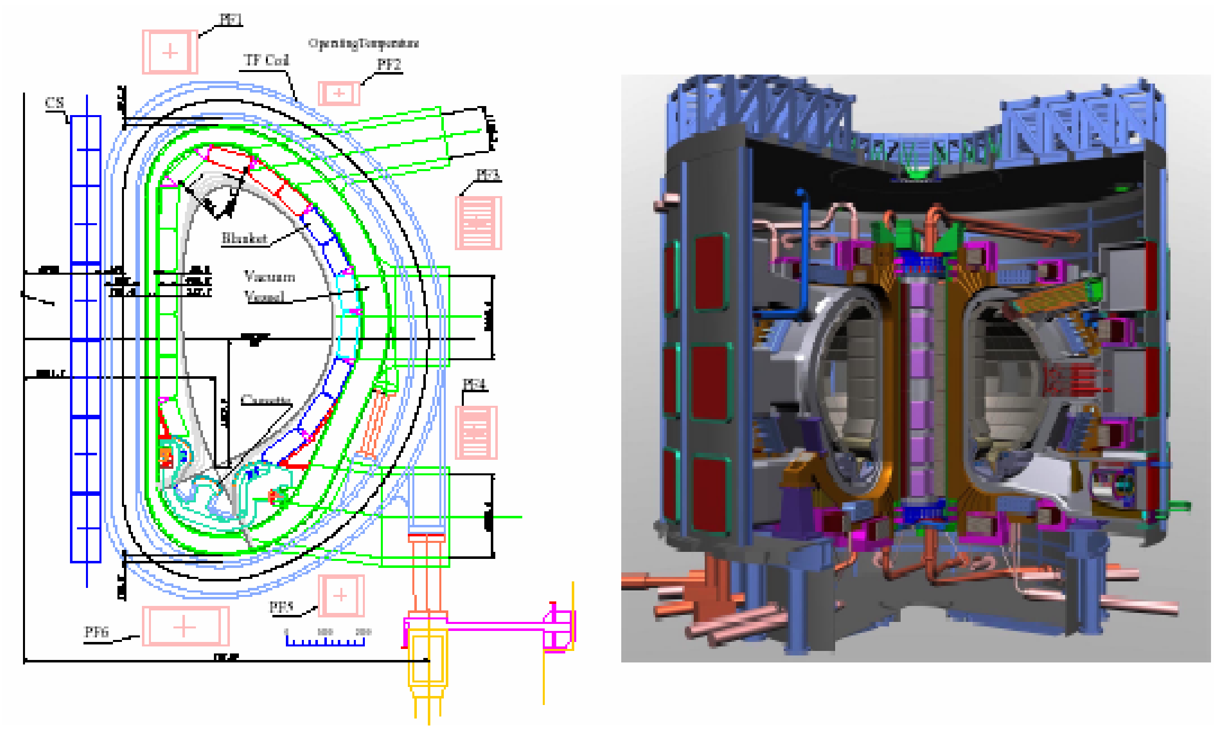


Figure 3- Schematic and cut out view of ITER

References

[1] Kadomtsev, B.B. *Tokamak Plasma : A Complex Physical System*, Institute of Physics Publishing, Bristol (1992) ; Wesson , J. *Tokamaks* (second edition), Oxford University Press, Oxford (1997) .

[2] ITER Physics Experts Group et al , Nucl. Fusion, **39** ,(1999) 2137 .

[3] Aymar, R. et al , *Fusion Energy 2000* (Proc. 18th Int. Conf. Sorrento, 2000), IAEA, Vienna (2001), Paper OV/1 .

[4] See e.g. Andreani, R. , Nucl. Fusion, **40** (2000)1033 .

[5] Fusion fast track experts meeting – conclusions web.gat.con/iter-ga/eu_fast_track_101201.pdf

AVI-3 PHYSICS ISSUES ON MAGNETICALLY CONFINED FUSION PLASMAS: STELLARATOR DEVICES

Carlos Hidalgo
Laboratorio Nacional de Fusión
EURATOM-CIEMAT, 28040 Madrid, Spain

The idea behind controlled fusion is to use magnetic fields to confine high temperature plasma of deuterium and tritium. One way to do this is to use a Tokamak, a doughnut-shaped vessel in which a strong helical magnetic field guides the charged particles around it. At present, the first burning fusion plasma experiment, the international Tokamak experimental reactor (ITER) is now ready for construction. This project, which is design to produce 500 MW of fusion power, is based on the continuous progress of many years on experimental and theoretical fusion research. As a consequence, the emphasis of present day fusion plasma physics research is shifting towards the issues of ITER related plasma physics in the following areas:
- plasma confinement.
- magneto-hydrodynamic stability and plasma control.
- power and particle exhaust.
- plasma heating and current drive.
- research on magnetic confinement concepts;
- plasma diagnostics;
- long term technology.

Concept improvements are physics and technological activities aimed to improve the basic development of fusion devices, for the conceptual definition of the demonstration power plant which would be the final stage before the construction of a commercial power fusion station. Although next step magnetic confinement devices such as ITER will be based on the tokamak idea, other magnetic confinement concepts should be explored for performance in the advanced regimes. It is not clear that a unique magnetic configuration will be the answer to the various possible applications of fusion energy. From this perspective of particular interest for both performance and understanding is the investigation of confinement in stellarator devices.

Tokamaks and stellarators differ in the way the magnetic poloidal magnetic field is produced. Tokamaks devices create the main toroidal magnetic field using coils located around the torus. They rely on the transformer effect to induced a toroidal current in the plasma that, in turn, creates the poloidal magnetic field; thus in the tokamak concept the plasma itself contributes to its confinement via a strong toroidal current. Stellarator relies on external coils to create all components of the magnetic field.

Both concepts have advantages and disadvantages that are mainly linked to the different ways of producing the poloidal magnetic field[1,2]. In Tokamaks the participation of the plasma (via plasma current) in its confinement leads to stability problems and the need for plasma position control. Furthermore the sustainment of the plasma current requires additional efforts to get steady-state operation. Stellarators offer direct solutions to those problems. Since they do not need a plasma current, stellarators are per-se intrinsically steady-state devices. The intrinsic potential of stellarator for steady state operation is their prime contribution to concept improvement studies.

Plasma disruption is a sudden loss of magnetic confinement. In this process the plasma stored energy is released, in a short time scale, to the plasma facing components and device structures. Furthermore, during the disruption process electron can be highly accelerated causing metallic component to melt. Although disruption mitigation techniques applicable in next step magnetic confinement devices have been recently successfully tested in different Tokamaks, further research on improved characterization of disruption processes and demonstration of routine disruption avoidance and mitigation techniques should be pursued in the Tokamak concept. The absent of plasma current in stellarators removes an important energy source of free energy in the plasma and hence the associated instabilities and disruptions.

However, these advantages of the stellarator concept are not for free. The 3-D stellarator magnetic topology leads to large particle orbit deviations from flux surfaces and, as a consequence, to severe limitations on energy confinement and α – particle (helium nucleus) confinement. For the technological point of view, the three dimensional nature of stellarators makes it more difficult to construct the coils that create the magnetic field. In a future reactor the heat removal system (divertors) and lithium blanket systems would be more complicated in stellarators than in Tokamaks.

In spite of those problems there is a significant revival in the international stellarator program. Medium size stellarators are at present in operation in USA (e.g. HSX, at the Wisconsin University), Japan (e.g. H-J, at the Kyoto University) Europe (e.g. TJ-II, at the Centro de Investigaciones Energéticas Medioambientales y Tecnológicas in Madrid), Russia (L-2M, at the Institute of General Physics in Moscow) and Australia (H-1, at the Australian National University). The superconducting Large Helical Device (LHD) is in operation at the National Institute for Fusion Science (Japan); the large superconducting stellarator Wendelstain 7-X, under construction at the Max Plank Institute in Germany, is expected to start operation in 2010.

The basic limitations of stellarator devices (like insufficient confinement of α – particles) and technical limitations have been, at least partially, removed. This was due to the development of powerful supercomputers and advances in technology. Supercomputers have allowed to design optimised magnetic traps with the physical properties to confine the plasma at relevant density (n_i), temperature (T_i) and confinement times (τ_E) in stellarators.

The main goal of fusion research is to reach the triple product of density, temperature and confinement time needed for ignition ($n_i . T_i . \tau_E > 5 \times 10^{21}$ m^{-3} keV s). From this perspective stellarators have a long way to go before they can get the performance of Tokamaks in terms of the triple product ($n_i . T_i . \tau_E$). Even the expected performance of superconducting stellarators (like W-7X and LHD) will be about a factor of ten below break-even. This lag of stellarators behind the Tokamak line is by more than one development step (Fig. 1).

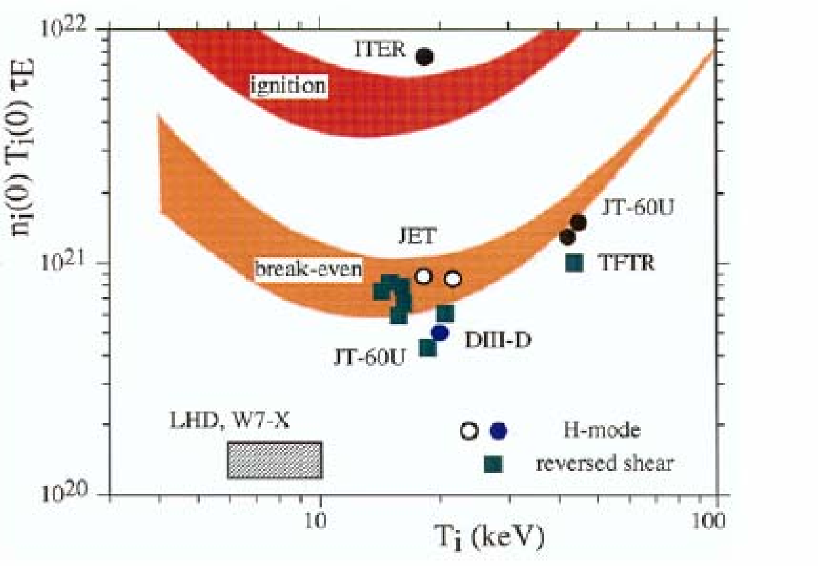


Fig. 1- Triple product for superconducting stellarators and existing Tokamaks. The stellarator W7-X is still under construction.

Encouraged by all the progress made in the last decades, one of the greatest scientific challenges confronting fusion research is still open: the development of models to fully explain the mechanisms underlying transport of energy in fusion plasmas. From this perspective stellarators will continue to help in the development of Tokamaks. In particular the scaling laws predicting the performance of fusion devices (like the energy confinement time) are very similar in large Tokamaks and much smaller stellarators (Fig. 2).

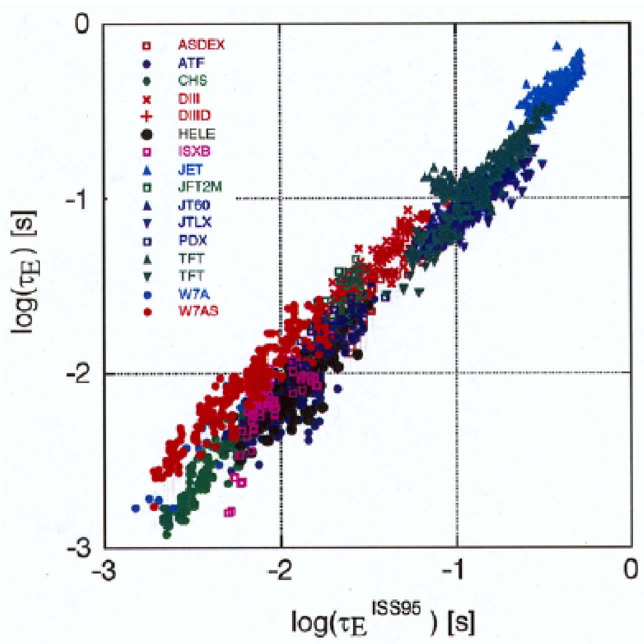


Fig. 2- Confinement times of Tokamaks and stellarators plotted against the expected value from the international stellarator scaling law[1]. The fact that all data points lay on an approximately straight line mean that the same physical mechanisms are at work in both confinement devices.

This result clearly indicates similar underlying processes in both devices. Particularly important, both from the perspective of the basic understanding of system far from thermal equilibrium and for the final goal of fusion energy, is the observation of transition with strong reduction of plasma turbulence with a concomitant increase in the confinement time (τ_E) in Tokamaks and stellarators. The understanding and development of methods for controlling plasma turbulence have opened a new path in plasma physics research.

Two additional import goals for fusion is the operation at high beta and the control and particle and energy exhaust. A critical parameter determining reactor feasibility is the maximum ratio of plasma pressure to magnetic field pressure (ß). This value should be sufficiently high for economic operation of a fusion reactor. A critical issue is whether instabilities driven by fast particles (e.g. Alfvén type instabilities) will become unstable and their impact on plasma performance in next step magnetic fusion devices. So far, no evidence of limitation in achieved beta value due to magneto hydrodinamic instabilities (MHD) has been observed in present stellarator experiments. A very promising method to attack the heat removal and particle control problems (both in Tokamaks and stellarators) is the use of a divertor (an edge magnetic configuration which allows to locate the principal plasma-surface interaction remote from the main plasma). Different types of divertors are presently being investigated in helical devices. Although the divertor concept has been investigated in few stellarators the knowledge of divertor concept has reached already a fairly solid basis.

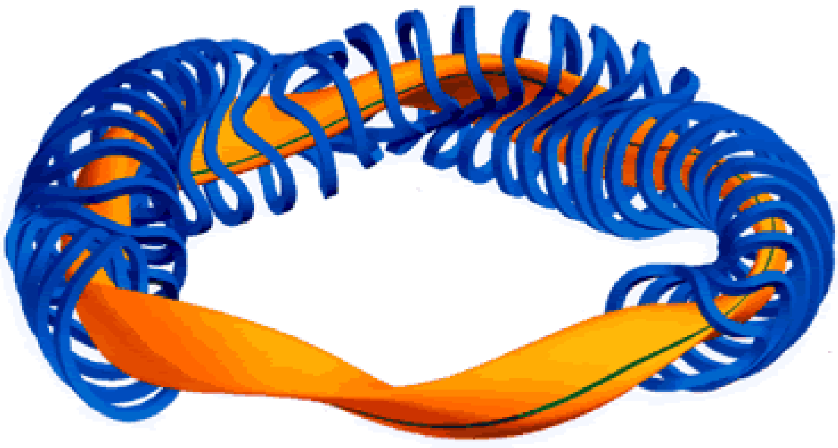


Fig. 3-Wendelstein 7 – X superconducting project currently under construction in Germany. The sketch shows (in blue) the modular superconducting coils and (in orange) the plasma.

Particles in high temperature plasmas have shown very long mean free paths between collisions (typically a factor 1000 longer than the torus circumference). An important issue is to get good confinement properties in these low collision regimes. The confinement properties of stellarators can be optimised by an appropriate 3-D shaping of the equilibrium magnetic surfaces with quasi-symmetry (quasi-helical and quasi-poloidal symmetry) to prevent the particle orbit loss. The discovery of quasi-helical symmetry provided the first possibility of building stellarator with collisionless particle confinement. Supercomputers have enabled to design the coils needed to produce the magnetic field required to achieve these conditions.

A quasi-helical stellarator is at present in operation in USA and the superconductor W7-X stellarator (under construction) is based in the concept of quasi-poloidal symmetry (Fig. 3).

The inherent capability of stellarators for steady-state operation and disruption-free operations opens the possibility of development attractive fusion devices which would allow to produce self-sustaining fusion reaction to release useful energy in this century. A convincing technical and scientific demonstration is possible only a matter of time and, last but not least, economical support[3].

References

[1] F. Wagner, Plasma Phys. Control Fusion **39** (1997) A23.
[2] C. Alejaldre, Phys. World **9** (1996) 25.
[3] Europhysics News **29** (1998) Fusion special issue

AVI-4 STATUS REPORT ON INERTIAL FUSION ENERGY

Burton Richter

AVI-4.1 Introduction

Inertial confinement and magnetic confinement are the foundations of the two technologies that are being pursued as possible routes to obtaining energy from fusion reactions. In fusion, two isotopes of hydrogen, deuterium and tritium, are combined to yield helium and a neutron with an energy release of about 14 MeV per reaction. If a viable and affordable way could be found to produce these reactions on a large scale, a new, relatively clean, large-scale, energy source would be available to help meet the rapidly expanding demands of society for energy to fuel economic growth.

To produce the fusion reaction, deuterium and tritium ions must be brought close enough together so that the nuclear reaction takes place in spite of the strong electric repulsion from the like charges of the two nuclei that push them apart. Magnetic and inertial fusion takes two different approaches. In the better known magnetic fusion a dilute gas of deuterium and tritium is heated to a temperature of about 100 million degrees and the plasma created is confined in some sort of magnetic bottle. At the high temperatures, the thermal motion of deuterium and tritium ions can overcome the electric repulsion and produce the desired reaction. Part of the energy released goes into maintaining the temperature of the gas, and the energy generation is continuous. The required size of the plasma is very large to produce a sufficient reaction rate for a useful energy source.

Inertial confinement fusion takes a route almost opposite in approach. It starts with a millimeter-sized capsule of frozen deuterium and tritium. The capsule's surface is struck with a beam of some kind and heated to a high temperature and begins to evaporate. The material coming off the surface creates a kind of rocket effect and, if the surface is hot enough and uniform enough, the walls of the capsule are driven inward symmetrically, compressing the fuel. The densities reached at the maximum of the compression are far above those of ordinary solid materials and the heat generated during compression heats the material to roughly the same 100 million degrees as is required in magnetic fusion. The fusion reaction begins and the capsule explodes. Nearly all of the energy is in the neutrons produced in the reaction as is the case in magnetic fusion.

Recently, the Department of Energy's Fusion Energy Science Advisory Committee (FESAC) has produced a report on the state of research toward Inertial Fusion Energy (IFE). In the next section, parts of the report are reproduced since they give a very good evaluation of the state of the program. (The entire report can be found at www.ofes.fusion.doe.gov/More_HTML/FESAC/IFE PanelReport.pdf) This report will conclude with my own views of the potential of the program as an energy source.

Briefly put, that potential is very low. The reason for this is that for decades almost all of the financial resources for inertial fusion have been put into the development of "drivers" that can produce intense pulses of neutrons for the nuclear weapons program.

These fusion capsule explosions are primarily to allow the study of matter and radiation flow at temperatures and pressures like those achieved in nuclear weapons. Little consideration was given in the R&D program to the development of devices capable of the necessary high-repetition rate and efficiency required for a useful energy source. The largest of the current generation of drivers is the National Ignition Facility (NIF) laser at Lawrence Livermore National Laboratory in the U.S. (a similar machine is being built in France) which will soon come into operation but is limited to a few shots per day compared to the few shots per second required for an energy source. A similar repetition rate limitation exists for a different kind of device (the Z-pinch) at Sandia National Laboratory in the U.S.

The most promising driver for an energy source, beams of heavy ions accelerated to moderate energies in conventional particle accelerators, has received little financial support. Particle accelerators can easily achieve efficiencies of 40%-50% and routinely run at the repetition rate required for an inertial fusion driver. Although reviews going back to the late 1970s have consistently rated its energy potential higher than alternative approaches, it has been starved of funds.
Given the current directions of the program and the allocation of resources, the potential to achieve the ignition of a capsule is high, while the potential to realize a useful energy source is low.

AVI-4.2 Section of the FESAC Report

AVI-4.2.1 INERTIAL FUSION ENERGY (IFE) (subheadings are those in the report)

The inertial confinement approach has the potential to be an attractive path to fusion energy. That potential is being explored, and these IFE research activities are the subject of this report. The motivation for fusion energy research includes the promise of an energy source with no greenhouse gas emissions, and with a virtually inexhaustible fuel supply that is widely available.

While these and other features of IFE are attractive, an earlier FESAC report "recognized that difficult scientific and technological questions remain for fusion development." [2] A diversified basic and applied science research portfolio is required to prepare for the realization of the ultimate goal of fusion energy production and to reduce developmental risk. IFE and MFE (magnetic fusion energy) are pursued because they present major opportunities for advancing both science and fusion energy. While they share a common goal, IFE and MFE have significantly different scientific and technological challenges and opportunities. Because of this diversity, the parallel pursuit of IFE and MFE broadens the contributions to science while reducing developmental risk.

Conceptually, IFE can be harnessed to generate electricity and other useful products from a steady sequence of inertial confinement fusion (ICF) events. Each ICF event involves placing a small capsule of fuel in a chamber and then compressing and heating it to ignition by some type of "driver" that generates intense pressure on the outside of the capsule.
Fusion-power system studies indicate that the energy released per event could range between hundreds of megajoules(MJ) and several gigajoules(GJ). The corresponding

repetition rates would range from several per second to about once every ten seconds for a 1-GW(e) power plant.

Four principal technical requirements must be achieved in a cost-competitive, environmentally attractive manner for IFE to be successful:

1) **High Energy Gain and Efficiency**: The efficiency of the driver in converting energy from the electrical power grid to the energy needed to compress the capsule, coupled with the energy "gain" of the capsule (ratio of energy released to the energy needed to compress and heat the fuel) must be sufficient to yield substantial net energy.

2) **Repetition Rate**: The driver, target (which includes the fuel capsule) fabrication, and reaction chamber must operate at a repetition rate that is sufficient to produce economically useful power. The chamber must be restored to a sufficiently quiescent state after each shot to allow insertion of the next target, and for the transmission and focusing of the next pulse of energy from the driver to that target.

3) **Energy Conversion and Tritium Breeding**: The energy released from the burning deuterium-tritium (DT) fuel is mainly in the form of energetic ions, neutrons, and x rays. This energy must be absorbed by the chamber and converted into "high-grade" thermal energy that can be efficiently used to drive electric generators. The chamber must also utilize the emitted neutrons to breed sufficient new tritium (from lithium) to sustain the fuel supply.

4) **Durability**: The components in an IFE system must carry out the above functions with sufficient durability for the high capacity factors required in an attractive energy system.

AVI-4.2.2 Approaches to IFE.

This complex set of interrelated requirements for IFE has motivated the study of novel potential solutions. Three types of "drivers" for fuel compression are presently studied: high-average-power lasers (HAPL), heavy-ion (HI) accelerators, and Z-Pinches. The need for efficient coupling of energy from these drivers to the capsule has motivated different conceptual designs for the "targets," which contain the capsule of fuel.

In broad terms the targets can be categorized as "direct drive" or "indirect drive" types. Both types of targets have been used in the ICF program for various physics studies. Fusion capsules for both target types are conceptually similar, and are envisaged to consist of a tiny, spherical, cryogenic solid shell of DT fuel or DT-wetted foam, coated with plastic or other materials. In the case of direct drive, the driver beams (e.g., lasers) are focused directly on the surface of this coating. For indirect drive, the driver energy is delivered to the interior of a high-Z (heavy element) enclosure (hohlraum) so that the material at or near the inside surface of the hohlraum is heated to 2 to 3 million degrees. This creates blackbody x-ray radiation that impinges on the capsule at the center of the hohlraum.

The impinging energy (directly from driver beams or from x-rays) heats up the coating on the outer surface of the capsule. The heated coating ablates radially outward, generating an inward momentum impulse very much like a standard rocket engine. Driven by this "rocket" the shell implodes, reaching high velocities, and then slows down as the pressure builds up in the fuel. As the shell slows down, its kinetic energy is converted into internal energy (i.e. pressure) of the material enclosed by the imploding shell. The rapidly converging fuel creates a "hot spot" at the center that reaches its maximum temperature and pressure when the shell stagnates. At pressures of hundreds of gigabars and a temperature3 of about 10 KeV the hot spot ignites and a fusion burn front propagates outward through the bulk of the compressed fuel.

System studies indicate that each driver type is best matched with a specific target design and to a specific reaction chamber (including energy conversion) design. Two general types of chambers are being examined to accommodate the specific needs of the different drivers and targets. Drywall chambers have armor to withstand the energetic radiation and debris from the targets. The thick liquid-wall chambers have the advantage of continually replacing the surface exposed to this radiation and debris but with the added complexity of managing the flowing liquid. The three main driver types (HAPL, HI, and Z-Pinch), coupled with their corresponding target and chamber types, become the three main approaches to IFE. The following paragraphs briefly describe the present concepts for these three approaches.

High-Average-Power Lasers.

The HAPL approach involves research on both kryptonfluoride (KrF) gas lasers and diode-pumped solid-state lasers (DPSSL). The main-line approach for either laser is to use direct-drive targets to maximize energy coupling efficiency to the capsule. Advances in laser techniques indicate that the spherical uniformity of illumination
required by these targets may be achievable. Dry-wall chambers for energy conversion appear to be most compatible with the final optics and the penetrations in the chamber wall needed for spherical illumination. Dry walls are also compatible with the relatively low x-ray output from direct-drive targets, as compared with that from indirect-drive targets.

Heavy-Ion Accelerators.

The HI-accelerator approach plans on indirect-drive targets because the large ICF target physics database for indirect drive allows good definition of HI beam requirements, and because two-sided illumination is compatible with thick liquid protected chambers. The loss in efficiency compared to direct drive should be offset by the higher efficiency that is expected from HI accelerators. Liquid wall techniques for absorbing the energy while protecting structural material appear usable with the HI approach due to geometrically limited wall penetrations (spherical illumination not needed) and less demanding requirements for protecting final focusing optics from chamber and target debris compared with lasers.

Z-Pinches.

The Z-Pinch approach produces x-rays from an imploding cylindrical array of current-carrying wires that stagnate on a low-density-foam cylinder. One or two of these assemblies provide the x-rays inside of a hohlraum for the indirect-drive-target design. The high x-ray-generation efficiency of the pulsed-power driver and the z-pinch should compensate for the lower efficiency of this indirect-drive target design. The Z-Pinch driver, in contrast with the other two IFE approaches, is physically connected to the target by transmission lines. Lower repetition rates than HAPL or HI, and hence higher fusion yields per target, are envisioned to allow for replacing the transmission lines on each shot. Liquid walls are the baseline chamber approach, as in the HI approach, and they should be useful in dealing with the larger fusion yields.

The development of high-energy petawatt (PW) lasers raises the possibility of reducing the compression-driver requirements for any of the three approaches. The potential improvement relies on the extremely short pulse (~10 ps) of a petawatt laser to heat and ignite a small portion of the fuel near the edge of a compressed capsule. This "Fast Ignition" concept separates the functions of compression and ignition, and potentially relaxes the constraints on compression driver energy and target uniformity because no central hot spot is required. This flexibility may allow higher-gain capsules, and simplification of chamber and driver specifications.

AVI-4.3 Status

AVI-4.3.1 Programs and Facilities

Only three countries have large-scale ICF programs; France, Japan and the United States. Of these, the U.S. programs is by far the largest and is more than 90% devoted to weapons-related work and high-energy density physics (HEDP). The French program is large and also mainly devoted to weapons-related work and HEDP. The Japanese program is exclusively devoted to energy research.

Large, low-repetition rate, lasers have been the main instruments used to understand target behavior. France and the U.S. are both building lasers capable of delivering around 2 MJ of laser energy to target. Theoretical extrapolations from lower energy experiments indicate that this should be enough to achieve ignition. Both lasers should be ready within the next few years. In addition, a 2 MJ Z-pinch driver is being developed in the U.S.

There are several lasers in the tens of kJ class in France, Japan and the U.S. In Japan a 10 KJ petawatt laser is being added to the 10 KJ laser at Osaka for fast ignition studies.

There are no heavy-ion drivers although there is little doubt that MJ-class facilities could be built if desired.

AVI-4.3.2 Driver Issues:

Lasers

The main laser issues are efficiency and repetition rate. Krypton-fluoride and diode-pumped solid-state lasers are the main approach and the hope is to reach a laser energy efficiency of around 7% with repetition rates of several per second. This relatively low efficiency has led to an emphasis on direct-drive targets, because of the improved coupling of laser light to the targets compared to the indirect drive case. There is, in addition, a requirement for a large number of beams with a high degree of symmetry to avoid instabilities in the capsule implosion.

Z-Pinch.

The pulse-power system has a high efficiency but the driver must have an electrical connection to the target to establish the pulsed current. The targets are indirect drive and concept studies are underway on how to quickly produce the required connection after it has been destroyed by the previous target explosion.

Heavy Ion
The main issue here is in the low-energy end of the accelerator where space-charged forces are worst and tend to defocus the beams. This is a well understood problem whose solution is to begin acceleration with a reduced line charge density and compress the beam longitudinally after it reaches higher energies. Electrical efficiencies are expected to be 40-50%. The ion beam must eventually be propagated across the reaction chamber to reach an indirect drive target. Here too, space-charged forces tend to defocus the freely propagating beam. The solution is to use a pre-ionized gas channel as a focusing system. No demonstration projects are planned.

Fast Ignition.
If fast ignition works out, it has the potential to ease driver requirements for any of the three main approaches. It is too early to say whether it will be useful.

AVI-4.3.3 Reaction Chambers.
A 1-GW electric power plant requires about 3 GJ/sec of primary energy for a 33% efficient system. If that primary energy is delivered by five explosions per second, each capsule will deliver a blast equivalent to about 150 kg of TNT. Both z-pinch and heavy-ion approaches use a liquid wall chamber (a thick waterfall of lithium, for example) to absorb the neutrons. While the waterfall is disrupted by each blast, it can be re-established by the time of the next shot.

Direct-drive laser systems studies have focused on dry wall chambers because of the many ports required to achieve symmetry, a task much too difficult for a wet-wall system. The chamber walls must absorb the blast.

AVI-4.3.4 Other Issues.
A mass-production target system, a target-injection tracking system, and a tritium separation system also have to be developed. No more than concept studies have been done on any of these issues.

AVI-4.4 Conclusions

Megajoule-class laser and Z-pinch drivers are nearing completion in the U.S. and in France. Within the next three to four years it should be known if the theoretical extrapolations that predict successful ignition are correct, as all expect them to be. It will probably be several more years before it is know if fast ignition can ease driver requirements.

However, a demonstration of ignition is only the beginning of a much longer road to a practical energy source. The repetition-rate issue needs to be addressed, and only the non-funded heavy ion approach is known now to be capable of the necessary repetition rate. For lasers, an entirely new class of high-energy, high-efficiency, rep-rated lasers are required. For the Z-pinch, a quick, simple, and low-cost reconnection scheme needs to be developed.
The reaction chamber requirements, will really not be understood until there is more information on target gains and on the effect of fast ignition.
Successful achievement of ignition of a deuterium-tritium target pellet seems highly likely. Achieving a cost-effective energy system is a much more problematic.

AVI-5 Report on Laser Fast Ignition for Inertial Fusion Energy

Kunioki Mima
Institute of Laser Engineering, Osaka University, 2-6 Yamada-oka, Suita , Osaka, Japan, 565-0871

The present status and future prospects of the laser fusion research and related laser plasma physics are reviewed. In laser fusion research, giant lasers for ignition and burn by imploding DT fuel pellets are under construction at LLNL(Lawrence Livermore National Laboratory) and CEA, France. In Japan , the Gekko XII and the PetaWatt laser system have been operated to investigate the implosion hydrodynamics, fast ignition, and the relativistic laser plasma interactions and a new project; FIREX(Fast Ignition Realization Experiment) had started toward the ignition and burn at the Institute of laser Engineering of Osaka University. In the FIREX, multi 10kJ petawatt laser is used to heat compressed DT fuel to the ignition temperature. Recently, heating experiments with cone shell target have been carried. The thermal neutron yield is found to increase by three orders of magnitude by the pet watt laser injection to the cone shell target.

AVI-5.1 Introduction - Basic idea and Research Status of Laser Fusion in the World

In mid 90's, construction of ignition facilities of central hot spark ignition, started at LLNL(USA) and CEA, Bordeaux (France). In Japan, fast ignition and implosion hydrodynamics have been studied with GEKKO-XII and Petawatt laser system since 1996 at the Institute of Laser Engineering (ILE) of Osaka University.

In laser fusion, a tiny fusion fuel pellet is irradiated by intense laser to be imploded as shown in Fig.1. When the pellet is irradiated by laser, the surface is strongly heated, ablated and corona plasma is formed. The corona plasma density is typically one tenth of solid density and the temperature increases from 1eV to 1KeV. Namely, the pressure increases from 1Mbar to 100Mbar in a few nano second laser pulse. The fuel pellet consists of a plastic capsule which contains a DT cryogenic spherical shell. The tailored high pressure pulse causes multiple shock heating and compression and then the compressed fuel shell imploded. When the fuel shell is imploded spherically symmetric, the strong spherical shock wave is formed at the center of the target to generate a hot spark which initiates a fusion burn wave as shown in Fig. 1.

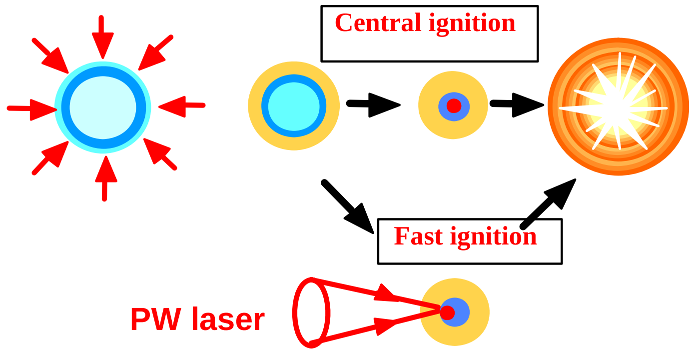
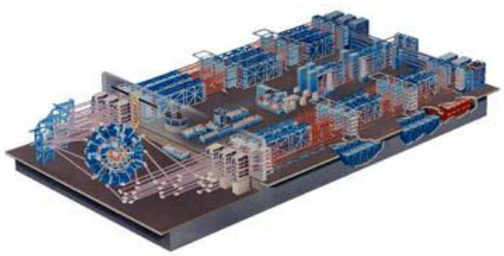


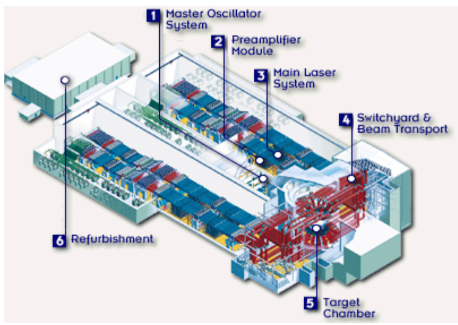
Figure 1- Implosion and burning processes of central ignition and fast ignition and expected fusion gain v.s. laser energy

This concept is called the central hot spark ignition. One of the drawback of this concept is the sever requirement on the spherical uniformity of the implosion since the spherical shock formation and the following adiabatic compression of hot spark are essential for the ignition. Because of the thick mixing layer on the hot spark-main fuel contact surface, total fuel mass is required to be large and the driving laser energy is predicted to be higher than 1MJ. This is another drawback.

On the other hand, in the case of fast ignition, the hot spark is generated by the external heating as shown in Fig.1. Namely, at the maximum compression, a heating laser with several tens of KJ pulse energy is injected to heat the core plasma. This concepts was proposed recently[1], when the high energy petawatt laser became available. Since the fast ignition does not need the central hot spark formation, the compressed fuel density profile is uniform and the required area mass density can be achieved with relatively small driver laser energy.

Therefore, the expected energy gain will be higher than that of central hot spark ignition for a same driver energy. The critical issue of this concept is the relativistic ultra-intense laser plasma interaction and the energy transport from the interaction region to the core plasma.





(a) (b)

Figure 2- (a) Overview of the OMEGA laser at University of Rochester (40KJ in 0.35 µm wavelength, 60 beams), (b) NIF laser overview (1.8 MJ in 0.35µm, 192 beams)

At present, the world largest laser fusion facility for the integrated implosion research is the OMEGA laser at LLE, University of Rochester. The overview of the laser is shown in Fig.2(a). This facility demonstrated the fusion gain of 1% and the cryogenic hydrogen pellet implosion is under investigation. Toward the demonstration of ignition and burn of DT pellet, two multi-Mega joule lasers are under construction in USA and France. The schematic view of the National Ignition Facility (NIF) at LLNL, USA is shown in the Fig. 2(b). The NIF will be completed in September of 2008 and the ignition will be demonstrated before the end of 2010. The current largest laser fusion facility in Japan is the Gekko XII laser and the petawatt laser for fast ignition experiment which are shown in Fig.3(a) and (b). The high energy petawatt laser (10kJ/10 ps); Laser for Fast ignition Experiment (LFEX) is under construction for the fast ignition realization experiment (FIREX) as shown in Fig.3(c).

(a) (b) (c)

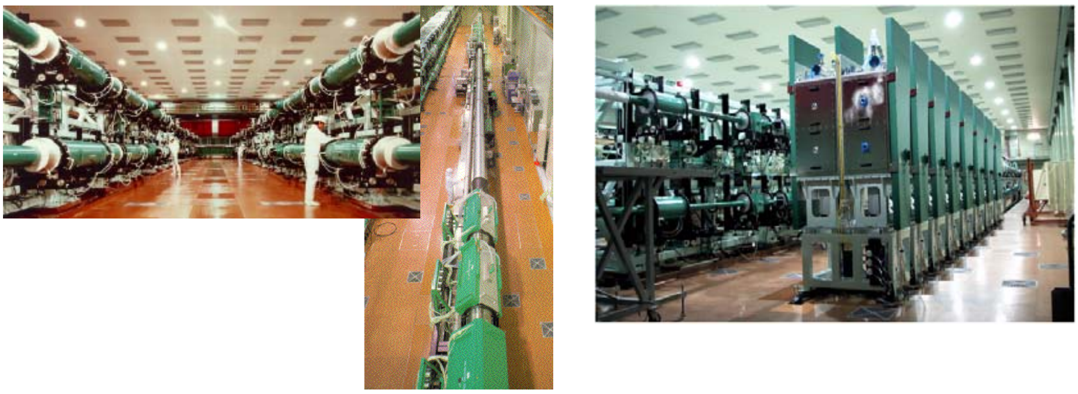


Figure 3- (a) Gekko XII laser at Osaka University (15 kJ in 0.53 μm , 12 beams), (b) Peta watt laser (500J/0.5ps), (c) main amplifier of LFEX for FIREX-I (10 kJ/10ps)

AVI-5.2 Fast ignition experiments and simulations

AVI-5.2.1 Fast ignition integrated experiment

The critical issues of the fast ignition research are

- ultra intense laser interactions with high density plasmas,
- physics of transport and energy deposition of laser produced relativistic electron and demonstration of high coupling efficiency of heating laser energy to core plasma thermal energy, and
- heating of dense plasmas to 10 keV and demonstrations of hot spot formation, fusion ignition, and burn.

Since 1996, 0.1~1 petawatt short pulse lasers (Fig.3 (b)) have been installed in the GEKKO laser system to investigate the fundamental physics of the relativistic laser plasma interactions related to fast ignition and heating processes of imploded plasmas at ILE [2][3]. The 50 ~ 500J /0.5ps laser pulses were injected into solid targets and imploded plasmas[2,3].

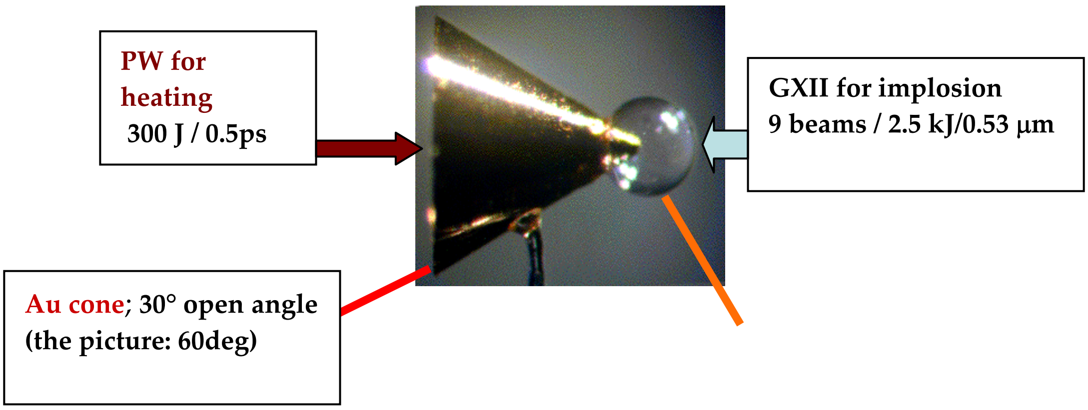


Figure 4- Gold cone target with CD shell which was used for the fast ignition experiments

When the short pulse laser is injected into a imploded cone shell target as shown in Fig.4, the neutron yields reach a value between 10^5 and 10^7, with a heating energy of 100 J/0.5ps and 400 J/0.5ps respectively as shown in Fig.5. They are 10 times and 1000 times higher than that of the non-heating case. This indicates that the thermonuclear fusion is enhanced by heating of the ultra-intense short pulse laser. The temperature increase was obtained with the neutron energy spectrum and the neutron yield enhancement shown in Fig.5. In the best shot, the temperature increased by 130 eV with 100J/0.5ps injection and by 500 eV with 400 J/0.5 ps. We found from these results that the 20-25% of the input laser pulse energy is deposited in the core plasma[2,3].

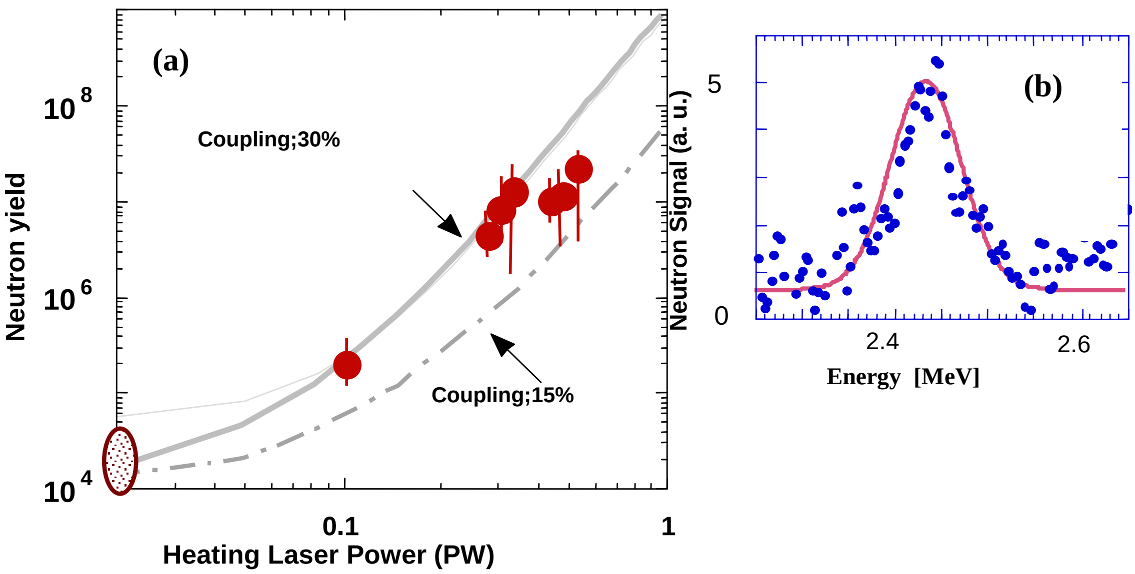


Figure 5-a DD fusion neutron yield enhancement and 5b neutron energy spectrum[1]

Since the focused laser energy included in the μm diameter is less than 30% the, observed coupling efficiency of 20-25% means that the energy in the wing of the laser spot will be coupled to the core heating. Namely, it is likely that the energy in the halo of the laser spot is focused by the guide cone and transferred to the core plasma. In Fig.5, scaling curves are shown, where the temperature increase is assumed proportional to the short pulse laser energy and the coupling efficiency is assumed the same as the results of the cone target PWM experiments. This indicates that the coupling efficiency for 400 J case is almost the same as for 80 J case as far as neutron yields are concerned. This scaling law has been used for planning the fast ignition experiment (FIREX). Further analysis of the cone target experiments are ongoing with PIC simulation and 2D hydro simulation including the Fokker Planck simulation as discussed in the following[6]

When we inject the petawatt laser directly into a spherical implosion plasma, we found that the ultra-intense laser pulse penetrates into the over-dense region and heats the plasma heating. Actually, we observe the neutron yield enhancement in the direct petawatt laser injection into the imploded plasmas[5]. However, the neutron spectra of imploded plasmas directly irradiated by the intense short pulse laser were broad. Therefore, the enhanced D-D fusion reaction is considered mainly due to the high-energy deuterium ions generated in the coronal plasmas where the plasma is locally heated by the relativistic laser. Recent simulation indicates that the intense relativistic electron flux deposits the energy locally to the back ground plasma through the anomalous resistivity and high energy ions of a few hundred keV are generated.[3]

AVI-5.2.2 Electron transport simulation and experiment

The relativistic electron propagation in solid density plasmas has been widely investigated by experiments, simulation and theory[4]. Those experiments and simulations indicate that the intense relativistic electron energy flow is self-organized and focused to the core plasma. The related simulations show that the cluster of the filaments are merged and self-pinched. Recently, it became clear by the experiments and simulations that the filamentation and the self-pinch of the relativistic electron flow are sensitive to the target electrical conductivity and thermoelectric processes in dense plasmas[4].

Recent 3D PIC simulation results show that the relativistic electron current is well organized and confined to a small radius in the over-dense plasma[3]. This indicates that the hot spot is produced efficiently by the petawatt laser heating. In the PIC simulations, we found that small-scale magnetic field fluctuations generated by the Weibel instability are inversely cascaded to longer wavelength fluctuations and subsequently self-organization of the relativistic electron flow occurs.

In the cone target PIC simulations, ultra-intense laser light is partially reflected on the cone inner surface and guided to the top of the cone while the relativistic electrons are generated both on the inner wall and on the wall of the top of the cone. Since the electrons are accelerated along the laser propagation direction, strong current is driven on the sidewall and the relativistic electron flows are pinched by the magnetic fields to the top of the cone. Those unique features of the laser interaction with the cone contribute to enhance the coupling efficiency of core plasma heating.

We carried out Fokker Planck simulations related to the cone target experiment for predicting the neutron yield. In the F.P. simulations, the relativistic electron energy spectrum was taken from the cone target PIC simulation, where the spectrum has two kinds of slope temperature, 0.5 MeV and 2.0 MeV.

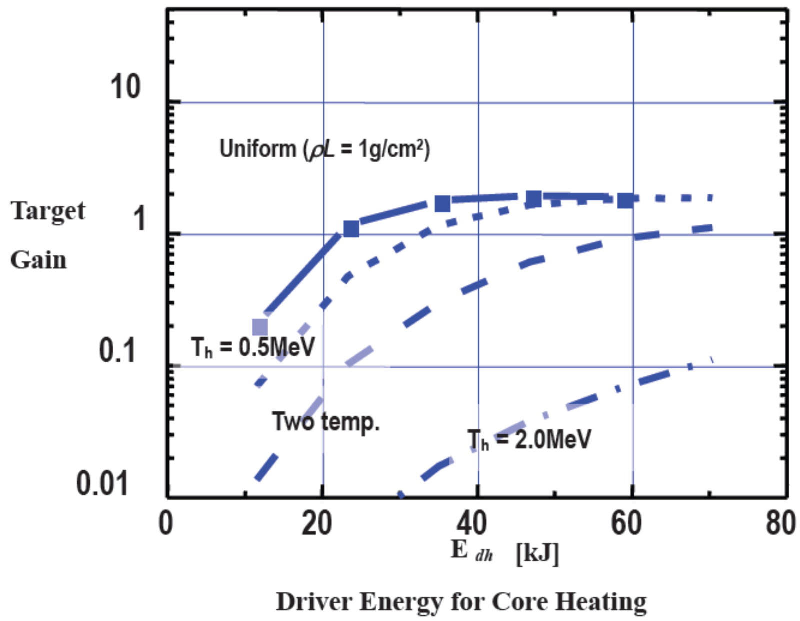


Figure 6- Fast ignition burning simulation with Fokker Planck code. Gain curves for FIREX-I (ρR = 0.7g/cm^2). The gain is sensitive to the fraction of low energy electron component

The Fokker Planck simulation shows that the plasma heating and the neutron yield depend on the 0.5 MeV electron heating. This softening of the electron spectrum is also one of the advantages of the guide cone.

AVI-5.2.3 Cone- Shell Target Implosion Hydrodynamics

Non-spherical cone-shell target implosions have been investigated on OMEGA and GEKKO-XII. We found that the fuel shell is partially stagnated and then the stagnated plasma expands toward the cone tip and the top of the cone is shocked to be heated up. These experiments indicate that the imploded plasma is not smaller than that of the corresponding spherical CD shell implosion, as was presented by H.Shiraga[8]. The PINOCO hydro simulation has been carried out to see the cone shell implosion and found that the ρr of plastic shell reaches 0.2 g/cm^2 . It can be concluded that the implosion of the cone shell target is spherically non-uniform, but the density can be as high as spherically symmetric target.

AVI-5.3 Future Prospects: FIREX-I Project

The fusion gain of the fast ignition scheme has been evaluated by the Fokker Planck simulation which is coupled with the hydro-burning code for the FIREX-I. The results are shown in Fig.6 which indicates that the gain expected in the FIREX I is 0.1 according to the Fokker Planck –PINOCO hydro simulation when the electron slope temperature is assumed 0.5MeV for the 10 kJ/10ps heating laser pulse and the fuel ρr =0.7g/cm^2. When the electron spectrum has two components, namely, 50% of total energy is in the 0.5 MeV component and the other component has a 2.0 MeV slope temperature. The energy deposition is classical (without magnetic field and/or electrostatic field effects), more than 15 KJ heating energy is necessary to achieve the gain of 0.1. However, the recent 2D Fokker Planck simulation shows that the effective energy deposition rate in the core plasma is enhanced by the self-generated magnetic fields and electrostatic sheath fields. So, the requirement on the heating laser may be relaxed. In the FIREX II, both the implosion and heating lasers are upgrated to 50 kJ. In this case, the ρr reaches 1.2 g/cm^2 and the hot spark will be greater than 0.5 g/cm^2. So, the gain will be higher than one and the ignition will be achieved.

In the Fig.7, the fusion plasma parameters achieved by the GEKKO XII and PW lasers and NOVA are plotted together with the goals of the FIREX-I and –II and NIF. The fast ignition condition will be clarified by the FIREX-I experiment and FIREX-II will demonstrate burning plasma in the fast ignition scheme. The Laser for Fast-ignition Experiment (LFEX) for the FIREX-I has been designed to deliver 1kJ/10ps with 1ps rise time. The 10 kJ short pulse laser consists of 4 segmented beams which are compressed by 2X2 segment dielectric gratings. One of the critical technical issues of the FIREX-I laser is the coherent combination of the 4 segmented beams. The laser construction will be completed and the plasma experiment will start before 2007[8] .The present view of the LFEX laser is shown in Fig.3c. High power amplification test of the LFEX will start in August 2004

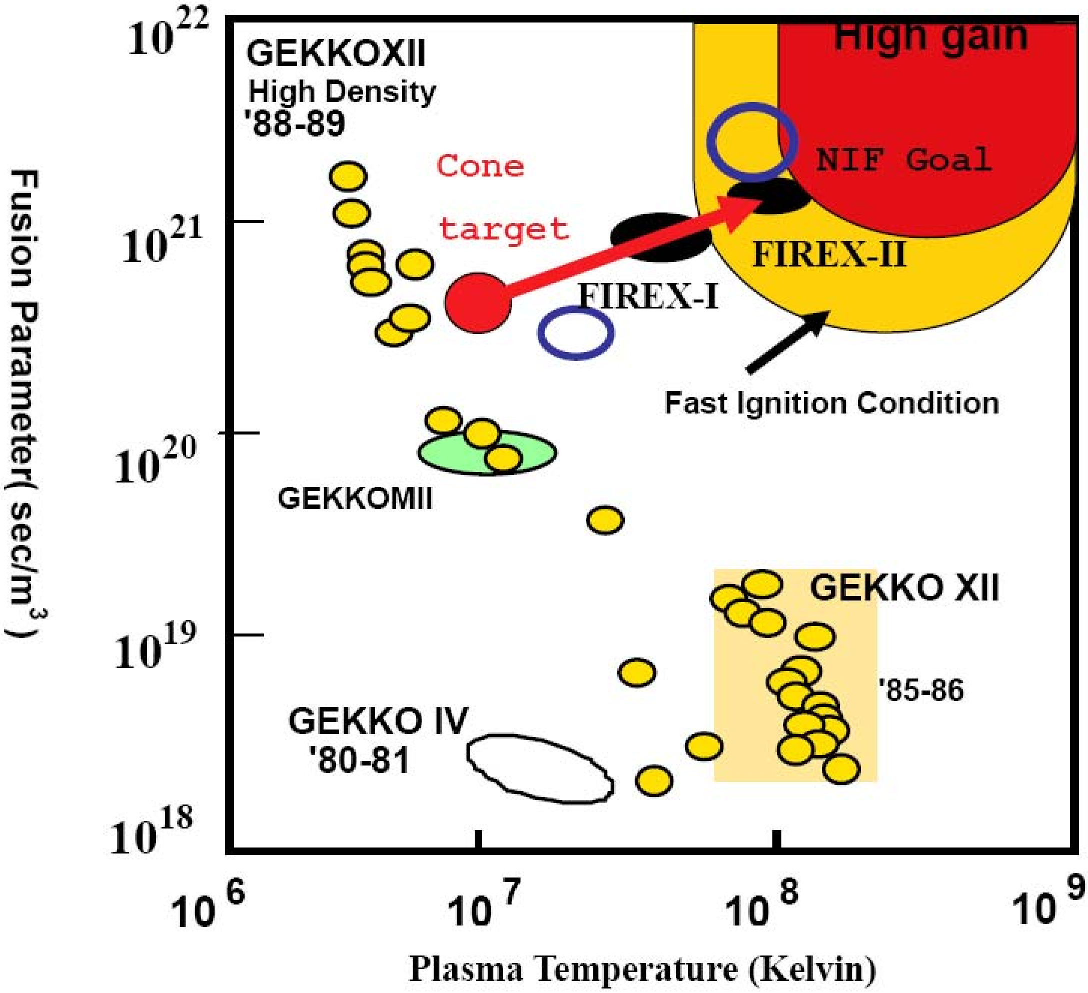


Figure 7 - Fusion parameter achievement by Gekko lasers and NOVA. Ignition condition for fast ignition and goals of FIREX-I and –II , and NIF project are indicated.

Acknowledgements

When I prepared this manuscript, my colleagues of the Institute of Laser Engineering of Osaka University helped me so much. I would like to express many appreciations to the member of laser technology, plasma experiment, theory and simulation groups at ILE, for providing me many materials to prepare this paper. Finally, I acknowledge that the part of this work is supported by the grand in aid of scientific research (Grand number 15GS0214).

References

[1] M.Tabak, etal, Physics of Plasmas, vol. 1,1626(1994)
[2] R.Kodama, P.Norreys, K.Mima, et al., Nature, **412**, 798(2001)
[2] R.Kodama, et al., Nature, **418,** 933, (2002)
[3] Y. Sentoku, K.Mima, et al., Phys. Rev. Lett., **90**, 155001(2003),
R.Kodama, etal, IFSA'03 Proceedings, 2004ANS, Inc. Edited by B.A. Hammel, D.D.Meyerhofer, J.Meyer-ter-Vehn, and H.Azechi, Mp1.2, p.333(2004)
[4] M.Key,,, etal, IFSA'01 proceedings (Edited by K.A.Tanaka, D.D.Meyerhofer, J. Meyer-ter-Vehn) Elsevier, Paris(2002)P.357 , S.D.Barton etal., ibid. p.375
[5] Y.Kitagawa, Physics of Plasmas, **9,** 2202 (2002)
[6] T.Johzaki, H.Nagatomo, etal. IFSA'03 Proceedings, 2004ANS, Inc. Edited by B.A. Hammel, D.D.Meyerhofer, J.Meyer-ter-Vehn, and H.Azechi,, Wpo3.16, 474 (2004)
[7] H.Shiraga, etal. IFSA'03 Proceedings, 2004ANS, Inc. Edited by B.A. Hammel, D.D.Meyerhofer, J.Meyer-ter-Vehn, and H.Azechi, WP3.4, 438(2004)
[8] N.Miyanaga, IFSA'03 Proceedings, 2004ANS, Inc. Edited by B.A. Hammel, D.D.Meyerhofer, J.Meyer-ter-Vehn, and H.Azechi, ThF1.1, 507(2004)