The cover picture shows an inside view of the coils of Wendelstein 7-AS. 
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FOREWORD

The International Atomic Energy Agency Conferences on Plasma Physics and Controlled Nuclear Fusion Research are the largest and most significant conferences in the field. The 1992 conference in Würzburg was the 14th in a series of meetings which began in 1961 and which, since 1974, have been held on a biennial basis. The conference was highlighted by reports of recent results from all of the major fusion facilities around the world, including the milestone experiment at JET in which tritium was introduced for the first time into a tokamak fuel mixture.

The conference was organized in co-operation with the Max-Planck-Institut für Plasmaphysik, Garching, to which the IAEA wishes to express its appreciation and deep gratitude. The conference was attended by around five hundred participants representing some thirty countries and two international organizations.

The opening session of the conference was highlighted by a round table discussion on ITER and its Relationships to Ongoing Fusion Programmes and by the traditional Artsimovich Memorial Lecture, which was given by Professor P.K. Kaw. During the technical sessions, over two hundred papers were presented. Contributions were made on tokamak experiments, inertial confinement, non-tokamak confinement systems, magnetic confinement theory and modelling, plasma heating and current drive, ITER, and technology and reactor concepts.

These proceedings include all the technical papers and five conference summaries. For the first time, the summary talks are being published as a separate volume before the rest of the proceedings.

The IAEA contributes to international collaboration and exchange of information in the field of plasma physics and controlled nuclear fusion research not only by organizing these biennial conferences but in a number of other ways as well. The International Fusion Research Council is sponsored by the IAEA and provides advice to the Agency on all matters related to fusion. The Nuclear Fusion journal has been published continuously by the IAEA for over 32 years. The IAEA organizes and maintains databases of nuclear, atomic, molecular and plasma-material interaction data for applications in fusion research and engineering. It also regularly organizes co-ordinated research projects, technical committee meetings, workshops, consultants meetings and advisory group meetings on relevant topics. Through all these activities, the IAEA hopes to contribute towards achievement of the long range goal of controlled fusion as a future energy resource.
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(Sessions B-1 to B-3 and Poster Session B-4)

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Paper IAEA-CN-56/B-1-1 was presented
by G.M. Chenevert as Rapporteur
RECENT PROGRESS IN THE U.S.
INERTIAL CONFINEMENT FUSION PROGRAM

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Presented by G.M. Chenevert

Abstract

RECENT PROGRESS IN THE U.S. INERTIAL CONFINEMENT FUSION PROGRAM.

The United States Department of Energy has recently completed a refocusing of the U.S. inertial confinement fusion (ICF) program to integrate the individual program elements into a cohesive national effort. Central to this effort is the establishment of the scientific and technology basis for an ICF ignition demonstration early in the next decade. The paper presents an overview of recent results of both the target physics and driver development parts of the program. In target physics, both direct- and indirect-drive approaches are being pursued for ICF capsule implosions, with notable progress.

1. INTRODUCTION

The goal of the U.S. inertial confinement fusion program, as shown in Fig. 1, is to produce small thermonuclear explosions in the laboratory. For defense applications, fusion yields of 200 to 1000 megajoules are required in a Laboratory Microfusion Facility (LMF). As contained in the National Energy Strategy [1], the long-term goal for civilian applications is commercial electric power generation. Both defense and civilian end uses require the technical demonstration of DT fusion fuel ignition and gain in the laboratory as a necessary intermediate step. As recommended by the 1990 National Academy of Sciences' review committee [2], the U.S. ICF program is preparing for an ignition demonstration using indirect drive with a 1-2 MJ glass laser.

The primary objective of the ICF program during the next several years is to complete the science and technology base to construct an ignition facility capable of successfully achieving ignition and moderate gain within the next decade. A theoretical and experimental program is being conducted to resolve the remaining target physics issues for x-ray driven, indirect-drive implosions with the Nova glass laser [3]. These include a Hohlraum and Laser Physics (HLP) campaign, which will
define the radiation environment necessary to symmetrically drive a capsule, and a complementary Hydrodynamically Equivalent Physics (HEP) campaign, which addresses capsule implosion physics in the absence of alpha particle deposition. In parallel, the compact, multipass amplifier design proposed for the ignition laser will be validated at full aperture and high fluence on a prototype beamlet.

The direct-drive approach can potentially provide high gain at a smaller drive energy than the indirect-drive approach. To realize this potential, it is necessary to produce extremely uniform laser irradiation of capsules to control hydrodynamic instabilities that degrade capsule performance. The current emphasis is in developing techniques to achieve symmetric, uniform illumination on the Omega Upgrade, presently under construction. If the experiments with this laser are sufficiently encouraging, a direct-drive option could be added to the ignition facility near the end of the decade.

The ICF program will develop, in addition to the glass laser, the KrF laser and the light-ion accelerator to establish the technical basis for an LMF driver decision and as alternative driver paths to ignition. A multielement, multiparticipant effort is essential for a successful ICF program.
2. TARGET PHYSICS

Theoretical estimates suggest that fusion ignition requires that the DT fuel be compressed and heated to simultaneously obtain a central hotspot with density of 100 to 500 times solid density having ion temperature $>5$ keV and a main fuel region of 1000 to 5000 times solid density at low temperature. To achieve these conditions efficiently requires a near-spherical implosion with ablation pressures uniform to 1-2%. Direct-drive capsules are particularly sensitive to low-order illumination nonuniformities during the startup "foot" portion of the power pulse. Although the mechanism of driver energy coupling to the target is driver dependent, many aspects of the capsule physics are independent of driver type, and there is a high degree of synergism in this area.

The HLP campaign involves a series of experiments and the associated computational modeling that explores laser-hohlraum coupling, generation and transport of x-rays, and the development of energy-efficient hohlraums. A primary objective is to show that relatively precise control of individual beam positioning can be maintained over time to ensure that the symmetry expected for ignition-style hohlraums will be adequate for ignition capsules. A second objective is to show that laser-plasma coupling instabilities will not significantly degrade energy coupling in ignition-style hohlraums. Indirect-drive experiments with appropriately shaped laser pulses have already demonstrated high hohlraum radiation temperatures, high laser energy absorption, and efficient x-ray conversion with low levels of both plasma instability and hot electron production. Detailed integrated hohlraum modeling has shown good agreement with a variety of experimental configurations. Experiments to measure time-resolved x-ray drive symmetry and capsule response inside a laser-driven hohlraum have begun.

The complementary HEP campaign will explore capsule implosion physics in regimes that are hydrodynamically equivalent to ignition capsules. The objective of these experiments is to demonstrate that hydrodynamic instabilities and consequent mix of materials that degrades burn performance are understood and are at a tolerable level. The campaign will culminate in the implosion of capsules that have hydrodynamic instability growth factors, convergences, and asymmetries comparable to ignition capsules.

The glass laser, direct-drive target physics program is investigating laser plasma interactions and plasma instabilities at
scalelengths (mm-sized) and temperatures (> 1 keV) comparable to the coronal region of direct-drive, high gain capsules. Techniques to control the laser coherence and obtain spatially and temporally smooth beams have shown much reduced growth of parametric instabilities, including SRS and SBS. Other experiments measure ablatively stabilized Rayleigh-Taylor growth rates and determine the effects of beam smoothing on unstable hydrodynamic behavior with directly driven planar foils and converging shells. Single-frequency ablation-stabilized growth rates have been measured with planar foils at approximately 50% of classical with smoothed beams of the Nova laser, in agreement with the Takabe formula in the linear regime. Harmonic evolution has shown good agreement with computer simulations. Techniques to reduce the growth rate and vary the implosion isentrope with pulse shaping and high-Z-doped ablators are being developed. Finally, hydrodynamic implosion experiments, to be conducted on the Omega Upgrade near the end of the decade, will examine the behavior of energy-scaled targets with convergence ratios, implosion velocities, and Rayleigh-Taylor growth factors characteristic of high gain implosions.

A primary objective of the Nike KrF laser program is to explore the utility of laser beam smoothing for controlling the growth of hydrodynamic instabilities in direct-drive targets. The experiments will accelerate flat and corrugated planar foils to velocities about one-third of those achieved in imploding ignition capsules. The effect of beam smoothing and pulse shape on laser absorption and hydrodynamic stability will be determined. Colliding foil techniques and x-ray sidelighting will be used to measure accelerated target nonuniformities. In addition, diagnostics will measure related parameters, such as mass-ablation and plasma temperature nonuniformities.

Intense ion beams exhibit favorable deposition characteristics, including efficient conversion to thermal radiation and the absence of beam-target instabilities. In addition, radiation smoothing and time-dependent symmetry are expected to be improved in ion hohlraums, since ions deposit their energy volumetrically. Initial light-ion target experiments have been performed with a 3.5 TW/cm², 5 MeV proton beam [4]. This beam provides a specific power deposition of 100 TW/gm in the target material. Initial experiments observed ion deposition, heating, and x-ray generation in a "thermal source" target consisting of a thin-walled gold cylinder filled with a low-density hydrocarbon foam. The near-term objective is to achieve a focal intensity of 10 TW/cm² with a 9 MeV lithium beam, which will increase the specific power deposition to 3800 TW/gm, due to the much shorter range of the more massive lithium ion.
3. DRIVER DEVELOPMENT

It has been calculated that to attain ignition conditions in few-millimeter-sized capsules with indirect drive requires a driver capable of delivering more than 1 MJ of energy (for indirect drive) in a 5-15 ns shaped pulse at a focal intensity in excess of 100 TW/cm$^2$ with a minimum of fuel preheat. Glass laser technology is the most well developed and has demonstrated the flexibility to be modified to meet the evolving requirements for ICF (short wavelength, broad bandwidth, optical smoothing, pulse shaping, etc). KrF lasers offer potentially improved optical smoothing for direct-drive targets, the shortest laser wavelength, very broad bandwidth, high efficiency, and high repetition rate capability, since the lasing medium is gaseous. However, the gas laser technology has not been demonstrated on a large scale. Light-ion beam drivers have favorable energy deposition characteristics, but focusing ion beams to the high intensities necessary for ICF remains an obstacle.

The present glass laser program includes both indirect- and direct-drive approaches. The Nova glass laser is the primary facility for investigating the indirect-drive approach to inertial fusion. The Precision Nova project is a broad-based effort to improve the existing Nova facility's experimental capability to meet and diagnose the energy, power balance, pulse shaping, and pointing accuracy necessary to complete the target physics campaign and achieve target conditions similar to ignition and gain implosions.

As presently envisioned, the ICF ignition facility will be a 1-2 MJ (at $3\omega$) glass laser capable of driving a capsule to ignition and moderate gain ($G = 1-10$) with indirect drive. The laser configuration envisioned for the ignition facility incorporates a number of cost-reducing features in a compact, modular design [5]. A current concept contains 12 beamlines, each consisting of a 4x4 array of optically independent beamlets. The design incorporates fiber-coupled front-end pulse injection, image-relayed multipass amplification, active switch-out pulse extraction, increased flashlamp pumping efficiency, multisegmented optics, and high damage threshold optical materials. Key components have been demonstrated separately in the laboratory. Small-signal gain uniformity of 98% over a 30 cm$^2$ optical aperture has been obtained with a multisegment amplifier. High fluence damage testing of the plasma electrode Pockels cell switch has indicated a polarization rotation efficiency of 99.9% and a damage threshold of >25 J/cm$^2$ (@1$\omega$) and no performance degradation after more than 40,000 shots. A 5 kJ (@ 3 ns and $3\omega$), 30x30 cm$^2$ prototype beamlet is under
development to demonstrate integrated system performance at full aperture and high fluence. The beamlet project will be sufficiently flexible to investigate alternate layouts, amplifier slab and switch placements, alignment techniques, performance optimization, operational issues, and engineering subsystem prototypes. The beamlet is expected to be completed in 1994.

Construction of the 30 kJ (@ 3ω), 60 beam Omega Upgrade laser [6] is proceeding. Recently, the design goal of 1 kJ (740 ps) output energy in the infrared has been achieved through a 20 cm single segment amplifier with an input energy of only 140 J. The goal of the Omega Upgrade project is to validate the performance of direct-drive capsules which are hydrodynamically similar to ignition capsules. Achieving these objectives requires improvements in pulse shaping and irradiation uniformity. Spatial multiplexing of the laser beam aperture and co-propagation of separate foot and main pulses through the power amplifiers will be used to achieve 10^3 dynamic range pulse-shaping capability with minimal energy loss through the frequency conversion crystals. The objective of the irradiation uniformity program is to reduce both low and high spatial frequency nonuniformities with control of beam-to-beam power balance and single-beam smoothing, respectively. Following activation in 1995, the Omega Upgrade will have a residual laser irradiation nonuniformity of 6-7%. Good power balance can be achieved through control of gain saturation, self-focusing effects, and harmonic conversion. Improving the individual beam uniformity will require lossless, wavefront-insensitive phase conversion and reduction of both the time instantaneous and time integrated speckle contrast. With the implementation of angularly dispersed terahertz bandwidth smoothing by spectral dispersion (SSD) and various hybrid techniques, the goal is ultimately to reduce the residual drive irradiation nonuniformity to less than 1% rms total and optimize the spectrum to minimize the growth of the Rayleigh-Taylor instability in the target.

The 2-3 kJ (@ 0.25 μm, and 3 ns) Nike laser is being constructed to investigate KrF gas laser technology and target physics issues relating to direct-drive ICF [7]. In particular, Nike will evaluate the utility of the echelon-free induced spatial incoherence (ISI) technique for meeting the stringent single-beam uniformity requirements for direct drive. Although the instantaneous focal distribution is highly nonuniform with ISI, the profile is averaged on a timescale that is short compared to the target hydrodynamic timescale due to the broad oscillator bandwidth and resulting short laser coherence time (1 ps). In principle there is no residual laser nonuniformity for sufficiently long averaging times with the ISI beam-smoothing technique.
The Nike laser will include a broad-bandwidth (>1 THz), spatially multimode KrF oscillator, discharge-pumped preamplifiers, and two e-beam pumped amplifiers that use 56 optical angularly multiplexed beams for efficient energy extraction. Chromatic aberrations are minimized by using reflective optics and high f-number lenses throughout the optical train. Installation has been completed through the first, 20 cm e-beam amplifier. Tests with this amplifier have produced 175 J of laser energy output in 150 ns (>1.1 GW). Atmospheric propagation tests in the 60 m long, temperature controlled laser bay have demonstrated negligible air distortion and acceptable vibration. However, simulations have suggested the propagation bay may need to be filled with a noble gas to avoid stimulated rotational Raman scatter in air and maintain the optical uniformity on target. The design goal is to achieve a 1% rms target ablation pressure uniformity in the low spatial frequency modes at a focal intensity of 200 TW/cm^2. The Nike laser should be completed in 1994.

Light-ion experiments are primarily conducted on the Particle Beam Fusion Accelerator II (PBFA-II), which consists of 36 radial pulse-forming lines connecting self-magnetically insulated vacuum transmission lines that feed a common, cylindrically symmetric, applied magnetic field ion diode. These experiments have produced a 9 MeV lithium beam with focal intensities near 1 TW/cm^2 with a "passive" LiF ion source. Further improvements in beam focusing are primarily dependent on reducing the lithium ion beam divergence from the present 26 mrad to 12-14 mrad. Electromagnetic particle-in-cell simulations using the 3D QUICKSILVER code have suggested the primary cause of this divergence is electromagnetic fluctuations in the anode-cathode (AK) acceleration gap of the applied-B ion diode. Experimental observations of beam energy/momentum correlations have indicated this mechanism exists for both proton and lithium beams. Simulations have indicated that the time evolution of the instability can be suppressed by controlling the electron charge evolution in the diode, for example, by varying the ion source size, increasing the insulating magnetic field, or using physical "electron limiters" in the AK gap. To realize this divergence reduction and enable the ion diode to couple more efficiently to the accelerator power pulse will likely require the development of a preformed "active" ion source. Recent spectroscopic evidence from the Laser Evaporation Ion Source (LEVIS) has indicated the production of a high purity lithium anode plasma having an intrinsic ion source microdivergence of less than 8 mrad.
The favored approach for future applications employs multiple, positive polarity inductive voltage adders powering separate extraction-type ion diodes and long distance beam transport. This technology allows the possibility of pulse shaping and beam bunching to improve target coupling efficiency. Many of these pulse power technologies can be developed on the 10 stage, 10 MV, 2.5 TW Sabre linear induction accelerator.

4. CONCLUSION

The U.S. ICF program is preparing for an ignition demonstration with a 1-2 MJ glass laser, using indirect drive, within the next decade. During the next several years, the program will establish the target physics and laser technology base necessary to proceed toward this goal. The 60 beam Omega Upgrade laser will validate the performance of direct-drive capsules that are hydrodynamically similar to ignition capsules. If these experiments are successful, a direct-drive option could be added to the ignition facility. The Nike KrF laser will be used to investigate the effect of beam smoothing and pulse shaping on the hydrodynamic response of planar foils. Techniques to reduce the beam divergence have been suggested which should increase the lithium beam focal intensity on PBFA-II and facilitate target experiments with high specific power depositions.

REFERENCES


DISCUSSION

R.J. HAWRYLUK: What is the relationship of the National Ignition Facility (NIF) to the proposed Nova Upgrade machine at the Lawrence Livermore National Laboratory?
G.M. CHENEVERT: The NIF may be different from the Nova Upgrade, an earlier proposal by the Lawrence Livermore National Laboratory which appears in the literature. The proposal has interesting features. However, the NIF will be based on a conceptual design which looks at all requirements. It is within this context that the Nova Upgrade will be considered.

A. GIBSON: Can you summarize the approval status and intended construction dates for the Low Gain and High Gain Demonstration projects?

G.M. CHENEVERT: The ICF programme strategy is to demonstrate ignition and moderate gain (G > 1) in the NIF. This facility may be upgraded to a Laboratory Microfusion Facility (LMF) (G > 50), depending upon the outcome of the NIF conceptual design, which has not yet started but is expected to start in 1993. Specific construction dates for these facilities will depend upon the outcome of the conceptual design and continued technical progress.

B. COPPI: Could you quantify the minimum goals of the NIF?

G.M. CHENEVERT: The minimum goal of the NIF is to achieve ignition and moderate gain (G > 1). Thermonuclear yield greater than the laser beam energy (e.g. 1–2 MJ) is necessary to achieve that goal. However, for programme applications, fusion yields of 200–1000 MJ would be required in an LMF. Consequently, if the NIF conceptual design study finds that the NIF should be upgradable to the LMF, this criterion may be included in the minimum goal.

C. YAMANAKA: What is US policy with regard to international collaboration in inertial confinement fusion research?

G.M. CHENEVERT: The policy on international collaboration in inertial confinement fusion is currently being reviewed by the US Department of Energy. Possible international involvement in the NIF has not yet been considered.
FERMI DEGENERACY OF HIGH DENSITY IMPLODED PLASMA AND STABILITY OF HOLLOW SHELL PELLET IMPLOSION

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Abstract
FERMI DEGENERACY OF HIGH DENSITY IMPLODED PLASMA AND STABILITY OF HOLLOW SHELL PELLET IMPLOSION.

Recent progress in the understanding of implosion physics, the parameters of imploded plasma achieved and the advance of the relevant technologies have made inertial confinement fusion a promising and rational energy development programme. In the paper the physical feasibility and the technical requirements for achieving ignition and high gain through a series of qualified implosions and sophisticated plane target experiments with GEKKO XII as well as numerical simulations corroborated by experimentally proven physics models are discussed.

1. INTRODUCTION

The basic areas of inertial confinement fusion (ICF) technology are: (1) fuel pellet design and fabrication; (2) driver; and (3) reactor or implosion chamber with diagnostics. The development programme and milestones to be passed on the way towards inertial fusion energy (IFE) are shown in Fig. 1.

As to the implosion, we have demonstrated a high temperature implosion which produced 10 keV and a high density implosion with 600 times compression over the initial solid density of the hollow shell pellet. These achievements have clarified the requirements for ignition. Implosion optimization is being pursued by investigating implosion stability, coupling efficiency and entropy control. The key technologies for quantitative implosion experiments are pellet design and fabrication (uniformity and
FIG. 1. Development programme towards inertial fusion energy.
layer structure) and driver (irradiation uniformity and pulse shape) control. The required laser energy for ignition and breakeven is estimated to be 0.1–1 MJ/pulse under well tailored conditions.

The demonstration of reactor scale implosion with gain requires the investigation of the physics of a propagating burn ignited by a central hot spark. The driver energy for this is estimated to be of the order of 1 MJ/pulse.

For the experimental test reactor, \( \eta_d Q \approx 10 \) is required with the capability of repetitive operation of \( \sim 10 \) Hz; \( \eta_d \) is the driver efficiency.

The conceptual design studies for an IFE power plant reveal many features that are essentially different from those of a magnetic fusion reactor. The most important and decisive issue which might affect the realization of an IFE reactor is the high efficiency driver \( \eta_d Q \approx 10 \) with repetitive operation, preserving the capability of high performance, i.e. beam focusability and uniform illumination, as well as the pulse shape.

We note that, in designing the strategy, the elementary technical components of IFE pellet, driver and reactor can be developed separately and independently and then be assembled to form a functioning power plant. The required period of development can be minimized by proper organization of the collaboration.

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2. FERMI DEGENERACY IN HIGH DENSITY COMPRESSION

High density compression 600 times the initial solid pellet density was reported at the last IAEA Conference in Washington [1]. The compressed density was evaluated from the \( \rho R \) values measured by a neutron activation technique [2]. The \( \rho R \) values are also consistent with those independently measured by secondary reaction [3], knock-on deuteron [4] or proton [5] techniques. At this Conference, the results of directly measuring the density \( \rho \) are reported; these are consistent with the previous results of the density derived indirectly from the \( \rho R \) values.

The high density compression achieved, together with the quantitative characterization of the laser and pellet parameters and properties, yields information on the stability of the implosion. The analysis of the experimental results demonstrates the existence of several smoothing/stability mechanisms for the implosion.

2.1. Experiment on Fermi degeneracy

The implosion experiments were performed on GEKKO XII essentially under the same conditions as the previous experiments [1]. The laser energy was 8–10 kJ of 0.53 \( \mu \)m wavelength, with a nominally 1.7 ns (full width at half maximum, FWHM) flat-top pulse with a rise time equivalent to a 1–1.3 ns Gaussian pulse (FWHM). The laser beams, after passing through a random phase plate, were focused through \( f/3 \) lenses behind the target centre at a distance of five target radii.
These irradiation conditions allowed 65% of the laser energy to be irradiated on the target. The non-uniformity of the laser irradiation was calculated, from measurements of the intensity profile in the target plane, to be \( \sim 20\% \) root mean square (rms), which is the square root of the quadrature sum of all \( \ell \) modes considered (\( \ell = 1-180 \)).

The plastic hollow shell pellets were used with various chemical composites required for the diagnostics. To measure the Fermi degeneracy, deuterized polystyrene (CD) pellets were used. The sphericity and the uniformity of the thickness were measured to an accuracy better than 0.3%.

In previous measurements, the compressed core density was estimated to be 600 g/cm\(^3\), from \( \rho R \) values. As a cross-examination of high density compressions, we used an independent technique based on Fermi degeneracy [6]. The electron number density is calculated to be \( 2 \times 10^{26} \text{ cm}^{-3} \), from the above given density, at an experimentally evaluated temperature of \( \sim 0.3 \text{ keV} \). The corresponding average Coulomb energy of the ions and the Fermi energy of electrons are \( \sim 1.4 \text{ keV} \) and \( \sim 1 \text{ keV} \), respectively. These compressed plasmas are hence expected to be strongly coupled (average Coulomb energy/thermal energy \( \approx 5 \)) and partially degenerate (thermal energy/Fermi energy \( \approx 0.3 \)).

The stopping power of charged particles in such a partially degenerate electron system may be reduced compared to the classical value for a Maxwell–Boltzmann plasma. In highly compressed plasmas with areal densities greater than the triton

![Graph](image_url)

**FIG. 2.** Yield ratio of secondary DT neutrons to primary DD neutrons, \( Y_{2nd,n}/Y_{1st,n} \), versus the areal mass density of CD plasma, \( \rho_{CD} R \), at a temperature of 0.3 keV with density as a parameter. Solid circles denote calculated ranges of 1 MeV tritons.
range, the yield ratio of the secondary DT neutrons to the primary DD neutrons (the DT/DD yield ratio) is approximately proportional to the range of 1 MeV tritons in the compressed fuel, as is shown in Fig. 2. The DT/DD yield ratio can, therefore, be a measure of the triton range and, therefore, of the degree of electron degeneracy.

The overall results of the density measurements are shown in Fig. 3 as a function of the shell thickness of the hollow shell pellet. The consistency of the results obtained by different diagnostics proves the existence of high density compression under the present experimental conditions of laser and pellet non-uniformity.

2.2. Stability of implosion and Rayleigh–Taylor instability

In order for the target to be compressed to high density, the shell has to cope with the instabilities, at least in the acceleration phase.

There are two sources of shell distortion that might be caused by irradiation non-uniformities. The first source is the Rayleigh–Taylor (RT) instability leading to
an exponential growth in time. The second source is 'secular' growth of distortion with the square of time if there is no instability. These two classes of distortion must be coupled and, therefore, we used a forced RT equation for the shell distortion $\xi$:

$$\frac{d^2 \xi_k}{dt^2} = \gamma_k^2 \xi_k + \delta g_k$$

where $\delta g_k$ is the gravity perturbation, $\gamma_k$ is the RT growth rate, and $k$ is the wavenumber of the perturbation, which is related to the mode number $\ell$ and the radius of the ablation front $R_0$ by $k R_0 = [\ell (\ell + 1)]^{1/2}$. We used the Takabe formula [7] for the RT growth rate including the ablative stabilization effect, $\gamma_k = 0.9 \sqrt{k g - 3 k v}$, where $v_a$ is the ablation velocity in the frame moving with the shell. We have assumed that the perturbation generated at the absorption surface decays exponentially away from this surface as $\exp (-kD)$, where $D$ is the distance between the ablation front and the absorption surface at which maximum absorption occurs.

The 'mix width' at the ablation surface due to the RT instability was evaluated as $\sqrt{2} \xi_{rms}$, where $\xi_{rms}$ is the rms perturbation of all modes given by $\xi_{rms} = (\sum \xi_k^2)^{1/2}$. This is shown in Fig. 4 with the solid line of 'mix thick' as well as the variation of the shell thickness as a function of time. The shell breaks at 1.6 ns, an early instant of time before which the shell has only moved one third of its initial radius.

There are several possible mechanisms that are not included in the present model but that might relax the RT constraints: (a) more effective thermal smoothing than that of the simple argument of $\exp (-kD)$; (b) a density gradient at the ablation front where by the mixing itself the density gradient at the ablation front may decrease; (c) more effective ablation stabilization than that given by the Takabe formula; and (d) dynamic stabilization, where a target surface 'feels' an oscillating

![Fig. 4. Time variation of shell thickness, thickness of mixing layer ('mix thick') due to forced RT instability and 'mix thick' with enhanced smoothing as functions of time during implosion. Increase of smoothing/stabilization yields results consistent with high density compression.](image-url)
pressure perturbation due to the movement of the target in the non-uniform intensity field.

With all these effects included, the mix thickness was calculated to be less than one half the shell thickness at any time, as is shown in Fig. 4 by the dotted line. In this re-evaluation, the thermal smoothing exponent increases by a factor of ~2, an ablative stabilization factor $\beta = 4$ is used instead of $\beta = 3$, and a simulated density gradient at the ablation front is introduced.

We have also examined several ignition designs (high velocity and moderate gain) using this 'optimistic' parameter set. Stability analysis suggests that the laboratory ignition/burn can be demonstrated with a driver energy of a few hundreds of kilojoules.

3. FUNDAMENTAL EXPERIMENT ON SMOOTHING/STABILIZING MECHANISMS

Several sophisticated experiments have been designed and performed to investigate the above mentioned physics and to examine quantitatively the role of these stabilizing mechanisms in spherical implosions. A reduction in the non-uniformities

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**FIG. 5.** D/L dependence of thermal smoothing factor and comparison with Manheimer and cloudy day models. D is the standoff distance from HISHO and Manheimer, and L is the modulation wavelength.
of mass ablation rate and ablation pressure was observed by using an intensity modulated beam [8]. As is shown in Fig. 5 the measured smoothing factor, $\Gamma$, was compared with the cloudy day model in which $\Gamma$ is estimated as $\exp(-kD)$, as was mentioned previously. More effective smoothing [9] than would follow from the simple cloudy day model is observed.

The RT growth rate $\gamma$ was evaluated by using an initially corrugated plane target with various perturbation wavenumbers $k$. The gravity in the Takabe formula was measured by X ray side light methods by which the accelerated trajectory of the target was observed. The temporal behaviour of the perturbation amplitude was measured from the opacity of the backlighted X rays. The energy shift of a 3 MeV proton beam passing through the perturbed target was also used to cross-evaluate the amplitude.

The results suggest a larger coefficient of ablation stabilization than 3 in the Takabe formula and the existence of some additional stabilization in the saturated phase.

4. DEVELOPMENT OF KEY TECHNOLOGIES OF THE IMPLOSION EXPERIMENT FOR IGNITION

The improvement of laser irradiation uniformity is one of the most important issues in high density implosion. The introduction of the random phase plate (RPP) into the implosion experiment has greatly improved the performance. Several new technologies for uniformity improvement have been developed and evaluated on the GEKKO XII system.

Fabrication of pellets of desired structure and material with good sphericity and uniformity is another key issue. Various kinds of low Z fuel pellets such as plastic shells, foam shells and cryogenic pellets have been developed.

Simulation codes based on the experimentally verified physics model and high resolution diagnostics of implosion parameters have been developed for a quantitative discussion of the implosion process.

5. SUMMARY

The formation of Fermi degenerated plasmas by high density implosion of hollow shell pellets was experimentally verified by the GEKKO XII glass laser at the Institute of Laser Engineering. The density achieved shows the feasibility of laser fusion which is approaching the ignition point.

The database accumulated recently by various basic experiments on the relevant physics is introduced into the analysis of high density implosion and into the design of high gain pellets [10].
The engineering progress in pellet fabrication, in particular cryogenic targets [11] and high power lasers, together with reactor design studies [12], have enabled a strategic programme to be set up along a route consisting of steps each of which is characterized by a well defined goal.

Laser fusion research comprises numerous aspects of various technologies concerning fuel pellets, drivers, and reactor or implosion chamber; these technologies can be developed separately and independently. Therefore, a collaboration in the research and development programmes of different fields should be most effective in furthering the development of inertial fusion energy.

REFERENCES


DISCUSSION

R.L. McCORORY: The stopping power of fusion reaction products such as tritons is strongly dependent on the electron temperature. In an imploded core there are also effects due to $\nabla T_e$ at the edge of the fuel region at the time of peak yield. Do you measure the time of peak yield? How well do you diagnose the time and space evolution of $T_e$ in the core? How confident are you that you are really seeing the effects of Fermi degeneracy in the high density experiments?

S. NAKAI: The experimental results obtained confirm that we are indeed seeing the effect of Fermi degeneracy in the high density experiments. Measurements are taken automatically at the time of maximum neutron yield. Evaluations based on neutron time of flight measurements show us that $T_e$ in the compressed core is lower than $T_i$. This leads to a lower estimate of density by degeneracy evaluation than the
actual density value, even with some spatial distribution of $T_e$. The effect of the temperature gradient $\nabla T_e$ at the edge of the fuel region is minor because the expected triton range is small compared with the compressed core $\rho R$, and the particles observed outside are neutrons which do not suffer any edge effect when they leave the fuel region.

J. COUTANT: What is the spectral bandwidth of your ASE laser source?

S. NAKAI: The spectral bandwidth of the ASE generator is about 60 Å. The bandwidth of the transported beam can be controlled by guiding optics.

M.G. HAINES: Have you carried out any two dimensional Fokker–Planck calculations to verify the enhanced thermal smoothing?

S. NAKAI: No, not yet.
ICF LASER PROGRAMME AT CEL-V

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Abstract

ICF LASER PROGRAMME AT CEL-V.

The laser inertial confinement fusion programme developed at the Centre d’études de Limeil-Valenton relies on the indirect drive scheme, in which the laser radiation is converted into a soft X radiation that drives the implosion of a D-T filled capsule. Experiments are performed with the Nd glass laser facility Phebus, which delivers 5 kJ in 1.3 ns at λ = 0.35 μm. The objectives are to investigate the critical elements in target physics that determine ignition and gain. The experiments address primary topics such as irradiation symmetry and hydrodynamic instabilities as well as other subjects relevant to hohlraum physics: X ray conversion, radiative transfer and atomic physics in X ray driven plasmas. Theoretical work and numerical simulations have shown that fuel ignition conditions and moderate gain (≈ 10) would be obtained by the indirect drive approach with about 1–2 MJ of laser energy.

1. INTRODUCTION

The principle of ICF is to compress to up to a thousand times the liquid density a small sphere of D-T which ignites and burns before being decompressed. High gain (i.e. ≈ 100) can be obtained from a ‘hot spot’ structure, formed by an ignition spark (areal density > 0.3 g/cm², temperature > 5 keV) where α particles are reabsorbed — which generates a self-sustained thermonuclear burn wave — surrounded by a large amount of cold compressed fuel (areal density > 3 g/cm²) [1–3]. An important intermediate step to be attained is to fulfil the ignition conditions of the D-T capsule, if possible with a small gain.

High densities require an ablative regime, with elaborate target structures and tailored laser pulses. The key issues of such a process are:

— An irradiation uniformity of ≈ 1%, to preserve the hydrodynamic stability of the implosion;

† Deceased.
— A D–T compression mode close to an isentrope (a fuel preheat of less than a few electronvolts).

To realize the most efficient energy transfer to the target and overcome these difficulties, the Centre d'études de Limeil-Valenton (CEL-V) has chosen to explore the indirect drive approach, in which the laser radiation is first converted into thermal X rays.

2. X RAY DRIVEN IMPLOSIONS

The indirect drive (or X ray driven) scheme involves a high Z cavity (or hohlraum) enclosing the fusion target. Laser beams irradiate the inner side of the wall through small holes. Focusing conditions are defined in order to optimize the conversion into soft X rays; confinement concludes in the generation of a nearly Planckian X radiation [4] (equivalent black body radiation temperature \( \approx 200 \) eV) which ablates and implodes the pellet.

The efficiency of energy transfer to the microballoon is smaller than in the direct drive case, and parasitic effects due to the filling up of the cavity by expanding plasma can affect the implosion. However, embedding the pellet in a black body radiation ensures a high irradiance uniformity without a drastic constraint on laser beam quality; besides, X ray driven ablation is favourable for lessening the development of hydrodynamic instabilities [5].

Experiments at CEL-V have been conducted with Octal (1.5 TW at \( \lambda = 1.06 \mu \text{m} \), and 0.4 TW at \( 0.35 \mu \text{m} \)) and with Phebus (5 kJ in 1.3 ns at \( \lambda = 0.35 \mu \text{m} \)).

2.1. Irradiance uniformity

In exploding pusher implosions, the symmetry of the compressed core was evaluated from the shape of time integrated X ray pinhole pictures recorded in the photon range \( h\nu > 1 \) keV. The spatial distribution of core emission can be related to the on-target irradiance through analytical models and numerical simulations. Numerical simulations are performed with the FCI2 code, a two dimensional Lagrangian code with limited heat flux conduction, multigroup radiation diffusion, and a non-LTE package for gold; post-processors can be coupled to simulate the diagnostics. As an example, a comparison between experimental and numerical X ray pictures is presented in Fig. 1.

In this figure the core emission has been partly occulted by the cavity but appears to be nearly spherical and corresponds to a non-uniformity of irradiance \( \Delta\phi/\phi < 15\% \text{ RMS} \); in such a case a convergence ratio (ratio of the pusher initial ratio to the final one) can be defined. The higher values of convergence ratio obtained in ablative experiments range around \( 15 \pm 5 \), in good agreement with numerical simulations.
The implosion symmetry can be controlled by the hohlraum structure. Modelling of the radiation distribution inside the cavity indicates that the on-target irradiance uniformity can be improved by lowering the capsule radius $R_c$ with respect to the characteristic dimension $R_h$ of the cavity. This effect is illustrated in Fig. 2, with a decrease of the surface of the microballoon by a factor of 4, necessarily leading to a decrease of the coupling efficiency.

FIG. 1. Core X ray emission: comparison between experimental time integrated X ray pinhole picture ($h\nu > 1$ keV) and FCI2 numerical simulation (same space scale and colour levels).

FIG. 2. Improvement of implosion symmetry resulting from a lowering of $R_c/R_h$. Left to right: decreasing $R_c$ for fixed $R_h$. 
FIG. 3. Phebus implosion experiments: comparison with performance at other laboratories.

FIG. 4. Experimental neutron yield normalized to one dimensional simulations.
2.2. Compression measurements

A series of experiments have been conducted with Phebus, varying the initial pusher thickness in order to progress towards high fuel densities. The density of the compressed core, deduced from the values of final pusher areal densities, is plotted versus the neutron yield in Fig. 3. Results are close to those presented by other laboratories for similar experiments (X ray driven implosions, plastic coated glass microballoons filled with D-T gas) [3, 6, 7]. This figure shows that a compressed D-T density close to $100\rho_0$ ($\rho_0$ is the liquid D-T density) has been obtained.

The neutron yields measured in these experiments have been compared with one dimensional predictions; the evolution of the ratio of the measured to the predicted neutron yield versus the calculated convergence ratio is presented in Fig. 4. Good agreement is obtained for exploding pusher type implosions (lower convergence ratio), while a difference of two orders of magnitude is observed for higher convergence ratios. This discrepancy can be attributed both to non-uniformities of drive flux and to the development of hydrodynamic instabilities.

3. HYDRODYNAMIC INSTABILITIES

Owing to the drastic consequences hydrodynamic instabilities can have on implosion performance, we are conducting experiments on directly and indirectly driven planar targets, in which mixing is expected to occur [8].

The principle is to accelerate a layered target involving a light (Al)–heavy (Au) interface. As the shock breaks through, the situation of a light fluid pushing a heavy one is similar to the slowing down of the pusher by the fuel, and mixing can develop. After some delay, the mix is detected by a spectroscopic method: the rear side of the accelerated target is heated by a probe beam, set to ablate the heavy material over a depth less than the initial thickness. The eventual detection of X ray lines specific to light material (Al He-a) is the signature of mixing.

For direct drive measurements, the Al He-a emission revealed early mixing with a deep penetration of aluminium into gold, even for a low driving irradiance.

For indirect drive, the mixing appeared significantly smaller. Obviously, X ray drive offers a smoothness of irradiation much better than that with laser drive. Moreover, stabilization mechanisms can also operate in this experiment.

In both cases X ray face-on shadowgraphy observations performed on the same target configuration showed no bubble-and-spike structure within the spatial resolution (15 $\mu$m); mixing can thus be considered as homogeneous.

4. IGNITION AND GAIN

Theoretical work and numerical simulations have been developed to determine the laser energy needed to provide hot spot formation, ignition and gain within the indirect drive framework.
The crucial parameter linking target gain with laser energy is the radiation temperature inside the cavity, which defines the implosion velocity [9]. Its impact can be seen in Fig. 5. The two curves are deduced from a model taking into account thermonuclear burn, capsule X ray implosion and cavity radiation temperature physics. The circles are calculated by one dimensional numerical simulations. Each curve gives the accessible domain of a thermonuclear burn wave for a given implosion velocity. They differ by a factor of ≈1.5 between the radiation temperatures. Ignition and moderate gain can be achieved with a laser energy of about 1–2 MJ.

REFERENCES

S. NAKAI: You discussed neutron yield and normalized density as a function of initial pusher thickness. What are the physics characteristics for the transition to the implosion mode, and what is the thickness of the pusher?

J. COUTANT: The transition between an exploding pusher regime and a more ablative one occurs when the X photon mean free path becomes smaller than the pusher thickness. This thickness varied between one and a few micrometres (equivalent of silicon).

H. NISHIMURA: Although very non-uniform implosion structures (mostly dominated by a mode number of 4 non-sphericity) are found in both the X ray pinhole image of the compressed core and the simulated core image, the neutron yield, normalized by one dimensional calculation, does not decrease drastically as the radial convergence ratio increases. The slope of the curve appears similar to the result obtained at the Institute of Laser Engineering, Osaka (see paper B-2-4), where a more spherically uniform compressed core was observed at high convergence. How do you interpret this result?

J. COUTANT: Experimental X ray pinhole pictures and numerical simulation of this diagnostic were displayed with the same colour code (Fig. 1). This enhances the apparent non-uniformities. In the experiment presented, the non-uniformity of irradiation was less than 15%.
DIRECT-DRIVE LASER FUSION
TARGET PHYSICS EXPERIMENTS

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Abstract

DIRECT-DRIVE LASER FUSION TARGET PHYSICS EXPERIMENTS.

The goal of the direct-drive laser fusion program is to validate high-performance, direct-drive targets. A decision to construct a direct-drive capability on the proposed 1-2-MJ NOVA Upgrade laser will be based on target physics experiments conducted on the OMEGA Upgrade laser system now under construction at the Laboratory for Laser Energetics (LLE). The OMEGA Upgrade will provide up to 30 kJ of UV laser energy in precisely shaped pulses with irradiation nonuniformities in the range of 1-2%. An understanding of and predictive capability for direct-drive targets are required to assure reliable estimates of ignition and gain with 1-2 MJ of incident laser energy. The paper reviews the target physics efforts currently underway to assess the critical physics issues of direct-drive ICF and plans for the experimental program to be carried out on the OMEGA Upgrade laser, as well as progress in the design, development and construction of this facility.

The objective of the U.S. inertial confinement fusion (ICF) program is twofold: (a) to provide advanced research capabilities to the U.S. nuclear defense program; and (b) to develop ICF as a safe, potentially inexhaustible source of energy. The two general approaches to ICF are direct and indirect drive (see Fig. 1). In the direct-drive approach, a fuel capsule is directly irradiated with multiple overlapped laser beams, arranged to optimize the drive uniformity. Indirect drive involves the conversion of the energy of an incident driver to x rays inside a case made of high-Z material and the use of these x rays to drive the capsule. The two approaches offer complementary and uniquely different ways of achieving the ICF objectives, i.e., a product of capsule gain and overall system efficiency in excess of 10.

In its 1990 review of the U.S. ICF program, a National Academy of Sciences (NAS) review committee concluded that the direct-drive approach to ICF could potentially provide high gain at a smaller drive energy than the
indirect-drive approach [1]. To realize this potential, it is necessary to produce extremely uniform laser irradiation of targets in order to reduce the effects of hydrodynamic instabilities on capsule performance.

The NAS review placed highest priority, for the next decade in ICF research, on the demonstration of ignition and gain. Experiments underway at LLNL will examine the feasibility of the indirect-drive approach. A national ignition facility is planned by the end of this decade to demonstrate ignition and moderate gain using the indirect-drive approach. The role of LLE in the ICF program is to validate the feasibility of the direct-drive approach. If such a validation occurs, a direct-drive capability could be added to the ignition facility that would enable it to demonstrate gains higher than 50. Validation of the direct-drive approach will be carried out on the OMEGA Upgrade facility—a 30-kJ (UV) laser now under construction at LLE.

The critical target physics issues associated with direct drive that need to be examined are laser energy coupling and transport, irradiation uniformity, hydrodynamic stability, and hot spot and main fuel layer physics. Presently, the OMEGA and NOVA facilities are addressing some of these issues. In the future, the OMEGA Upgrade laser will be used to examine all of the above critical physics issues.

Plasma physics issues relevant to direct-drive targets are currently being examined using long-scale-length plasma experiments on OMEGA [2]. These experiments have three objectives: The first is to generate underdense plasma conditions having densities \((n<n_c/4)\), density scale lengths (~800 \(\mu m\)), and temperatures (>1 keV) comparable to regions in the corona of direct-drive, high-gain target designs (without requiring the very large laser energies associated with high-gain designs). The second is to measure the temporal and spatial characteristics of these plasmas and to compare the measurements to hydrodynamic code simulations. The final goal is to conduct interaction experiments with single- and multiple-beam irradiation to examine the behavior of several plasma instabilities under conditions similar to those of high-gain, direct-drive target designs.
Raman scattering and the two-plasmon decay (TPD) have been examined by an analysis of 3/2-harmonic emission for interaction beam irradiances near $10^{15}$ W/cm$^2$. In these interaction experiments the 3/2 harmonic emission (Fig. 2) has been found to have various origins: (1) It may be produced by Thomson (or Raman) upscattering of one of the beams from TPD plasmons produced by another beam (most easily seen in Fig. 2(c) and (d) around 1.6 ns), or (2) it may be the usual single-beam effect where the interaction beam produces the TPD plasmons and scatters to produce the 3/2-harmonic (strong features extending to 2.1 ns in Fig 2 (b), (c), and (d). The former is a typical pump-probe beam experiment and allows an accurate determination of the TPD threshold ($\sim 10^{13}$ W/cm$^2$), while the latter is the usual self-scattering, 3/2-harmonic emission. This emission has so far defied satisfactory and widely accepted explanation and may be indicative of the highly nonlinear phase of the TPD instability.

**FIG. 2.**
Typical sequencing of the primary, secondary, and interaction beams (a), and time-resolved 3/2-harmonic spectra for different timings of the interaction beam: (b) 1.9 ns (I$_1$), (c) 2.2 ns (I$_2$), (d) 2.2 ns (I$_2$), and (3) 1.9 ns (I$_1$). Shots (b) through (d) had no FM bandwidth, while (e) had an SSD bandwidth of $\Delta\lambda/\lambda = 3 \times 10^{-4}$. In (f) the time evolution of the peak on-axis density calculated by SAGE (for primary and secondary beams only) is shown with the TPD Landau cutoff indicated at $n_e = 0.2 n_c$. The interaction beam intensity is $\sim 10^{15}$ W/cm$^2$ for all figures but (d) for which it is $\sim 2 \times 10^{14}$ W/cm$^2$. The primary and secondary beam intensities are $\sim 5 \times 10^{13}$ W/cm$^2$. The wavelength of all beams is 351 nm. In (c) an additional 6 tertiary beams of $\lambda = 1054$ nm were added, with their peaks at 2 ns. The contour lines shown correspond roughly to the 50% and 20% of maximum intensity.
These experiments will be continued on the OMEGA Upgrade where density scale lengths of ~1 mm and electron temperatures in excess of 3.5 keV are predicted.

Issues associated with the Rayleigh-Taylor (RT) instability are currently being examined using both planar and spherical target experiments. The first phase of the planar hydrodynamic instability experiments is underway on the two-beam NOVA facility [3]. Directly driven planar foils subjected to known, single-wavelength mass perturbations are used to measure linear growth rates and early nonlinear behavior. Initial results from these experiments confirm the linear growth rate predictions obtained from 2-D ORCHID calculations. The next series of experiments to be conducted on the two-beam NOVA facility is currently being designed. The main objective of this series will be to modify (reduce) the linear growth rates by changing the foil isotrope by pulse shape control or high-Z dopants added to the polymer material. Figure 3 shows numerical simulations of the evolution of several unstable modes for a pure CH-foil and a foil composed of C₈H₇Cl. The growth rates are predicted to be lower for the C₈H₇Cl material, (when compared to pure-CH) due to an increase in the ablation velocity resulting from an increase in radiative heating (see Fig. 4).

On the current OMEGA, the first phase of spherical converging geometry RT experiments has investigated the effect of SSD beam smoothing [4] on the ablation surface instability [5]. Similar experiments will be continued on the OMEGA Upgrade. Both the planar and spherical experiments are critical to achieving an understanding of the limits imposed by the RT instability for ignition marginal targets.

FIG. 3. Numerical simulations of the evolution of several unstable wavelengths for foils composed of pure-CH and C₈H₇Cl and two different initial amplitudes, 2.0 μm peak-to-valley (a) and 0.2 μm peak-to-valley (b). These simulations show that the smallest possible initial amplitudes, limited by residual irradiation nonuniformities, will be required for the next series of experiments. The C₈H₇Cl foil thickness was adjusted to give equal accelerations for both materials under identical laser conditions.
FIG. 4. The ablation velocity for the C$_8$H$_7$Cl material is higher than the pure-CH foil, assuming identical laser conditions, due to an increase in radiative heating and subsequent reduction in the peak density of the C$_8$H$_7$Cl foil. Figure 4(b) shows the normalized growth rates for these foils assuming the dispersion relation shown in the legend of Fig. 4(b).

The objective of the OMEGA Upgrade HET experiments is to examine the hydrodynamic behavior of energy-scaled, cryogenic, high-gain, direct-drive targets using both noncryogenic and cryogenic targets. Many of the important physics issues associated with cryogenic target performance can be addressed using noncryogenic targets. This approach avoids the operational issues associated with cryogenic experiments and allows these experiments to continue while cryogenic target fabrication technology is developed.

The OMEGA Upgrade is a 60-beam system, which will produce 30 kJ of UV energy on target in shaped pulses with peak powers in excess of 40 TW. The upgraded system fits within the existing OMEGA building, allows for the maximum use of existing hardware, and satisfies budgetary constraints. The performance requirements for this system are given in Table 1.

<table>
<thead>
<tr>
<th>Table 1</th>
<th>OMEGA Upgrade Design Goals</th>
</tr>
</thead>
<tbody>
<tr>
<td>Energy on target</td>
<td>Up to 30 kJ (pulse shaped)</td>
</tr>
<tr>
<td>Wavelength</td>
<td>351 nm (third harmonic of Nd:glass)</td>
</tr>
<tr>
<td>Lasing medium</td>
<td>Nd-doped phosphate glass</td>
</tr>
<tr>
<td>Number of beams</td>
<td>60</td>
</tr>
<tr>
<td>On-target irradiation nonuniformity</td>
<td>1% to 2% rms</td>
</tr>
<tr>
<td>Diagnostic solid angle</td>
<td>3 $\pi$ sr</td>
</tr>
<tr>
<td>Repetition rate</td>
<td>1 shot/hr</td>
</tr>
</tbody>
</table>
The OMEGA Upgrade system will use rod amplifiers up to an aperture of 9 cm and disk amplifiers with apertures of 15 and 20 cm, respectively. Prototypes of both amplifiers were built and tested in the last year. Figure 5 summarizes the high-energy beam propagation tests carried out on an equivalent beam line of the OMEGA Upgrade using a beam line of the present OMEGA system as an input beam. The test beam exceeded the required OMEGA Upgrade beam energy goal of 1 kJ at 1054 nm.

ACKNOWLEDGMENT

This work was supported by the U. S. Department of Energy Office of Inertial Confinement Fusion under agreement No. DE-FC03-85DP40200 and by the Laser Fusion Feasibility Project at the Laboratory for Laser Energetics, which is sponsored by the New York State Energy Research and Development Authority and the University of Rochester.

REFERENCES

DISCUSSION

H. TAKABE: You said that by using the picket fence high gain design you can expect enhanced ablative stabilization as compared with the continuous pulse case. In your intended design, will you expect only ablative stabilization or do you expect another effect which stabilizes the implosion dynamics?

R.L. McCORY: First, it is possible to use either a continuous or a picket fence pulse to operate near \( \alpha = 4 \). There are two effects we hope to exploit in the higher \((\alpha = 2-4)\) isentrope designs: higher ablation velocities (ablative stabilization) and gradient length stabilization. In the formula for the Rayleigh–Taylor linear growth rate,

\[
\gamma = \alpha \frac{\Delta(+) \text{kg}}{\sqrt{1 + kL}} - \beta kV_A \ (\alpha \sim 0.9, \beta \sim 3),
\]

both \( L \) and \( V_A \) are affected in the pulse shaped designs. \( L \) increases owing to radiative preheat and \( V_A \) increases because \( V_A \sim \dot{m}/\dot{\rho} \), and \( \dot{\rho} \) decreases owing to radiative preheat. Picket fence pulses generally can be frequency-tripled on solid state lasers a little more efficiently than a continuous pulse shape.

H. TAKABE: In your e-folding value of the Rayleigh–Taylor instability, do you include the contribution made by the hydrodynamic instability in the stagnation phase?

R.L. McCORY: Yes, the total growth factor includes the ablation surface instability growth feedthrough to the inner surface and internal growth which is not ablatively stabilized during the stagnation phase.

R.J. GOLDSTON: A simple ‘bottom line’ measure of success would seem to be the neutron deficit compared with 1-D simulation versus convergence ratio. J. Coutant, in presenting paper B-1-3, showed perhaps a factor of 30 deficit at a convergence ratio of about 12. What are the OMEGA results when measured this way?

R.L. McCORY: The OMEGA results plotted this way are essentially identical to the results from both the Institute of Laser Engineering and Lawrence Livermore National Laboratory. It is more meaningful to plot the observed/calculated yields as a function of e-foldings for the Rayleigh–Taylor instability. Plotting the results in this way for OMEGA experiments, we find nominal ‘one dimensional’ performance when the number of e-foldings is less than 8. It is for this reason that the HET (hydrodynamically equivalent target) campaign uses pulse shapes and targets designed to operate in this range.

C. YAMANAKA: Is the 30 kJ of the OMEGA Upgrade sufficient to ignite the fuel or is it enough only to simulate the compression experiment?

R.L. McCORY: 30 kJ at 0.35 \( \mu \text{m} \) should allow us to carry out ignition scaling and experimental tests of spark-plug physics. Optimistically, it may be possible to ignite, but since high or even moderate (\( \sim 20 \)) gain would not be possible with only 30 kJ, that is not the primary motivation for the OMEGA Upgrade. If we can show adequate uniformity and high enough compression to achieve \( \langle \rho R \rangle \) values between
0.15 and 0.40 g/cm² at ion temperatures in excess of 3.5 keV, a direct drive option for the NOVA Upgrade with gains of ~50 appears attractive.

S.E. BODNER: What is the status of experimental development of random polarizer plates?

R.L. McCORY: We have fabricated large aperture (90 mm) polarizer plates where the polarization varies linearly across the aperture in one dimension. Experimental tests show the $1/\sqrt{2}$ improvement expected. The fabrication of random polarization plates is more difficult and we have shifted our main interest to other techniques where the instantaneous improvement in uniformity will be more than $1/\sqrt{2}$ (zero correlation masks are but one example).
INERTIAL CONFINEMENT FUSION RESEARCH AT THE LOS ALAMOS NATIONAL LABORATORY

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Abstract

INERTIAL CONFINEMENT FUSION RESEARCH AT THE LOS ALAMOS NATIONAL LABORATORY.

Research on inertial confinement fusion (ICF) at Los Alamos is focused on resolving the key target physics issues associated with capsule ignition using indirect drive. In collaboration with the Lawrence Livermore National Laboratory, an extensive series of experiments on the Nova laser to examine drive symmetry, hohlraum plasma dynamics, laser-plasma interaction physics and other topics are being conducted. The theoretical group has constructed self-consistent models of indirectly driven targets that agree well with the experimental data. The development of precision target fabrication technologies, including the beta layering method for the production of uniform layers of DT within ICF capsules, has been continued. KrF laser technology is also progressing with the construction of the Mercury laser and, with the authors' technical support for the Nike KrF laser, at the Naval Research Laboratory.

1. INTRODUCTION

Advances in target physics and laser technology offer the potential for thermonuclear ignition and burn in the laboratory with the next generation laser driver. The Los Alamos inertial confinement fusion (ICF) programme is concentrating on resolving the key remaining target physics issues associated with the indirect drive approach to ICF.

There are three requirements for indirect drive. First, one must achieve a sufficiently uniform X ray drive on the capsule to produce a symmetric implosion. Second, one must be able to control the complex plasma hydrodynamics inside the hohlraum. Third, one must be able to implode the capsule such that hydrodynamic instabilities and wall/fuel mixing do not prevent ignition and burn.

2. HOHLRAUM SYMMETRY STUDIES

We have performed a large number of experiments on the Nova laser to evaluate implosion symmetry versus beam pointing and hohlraum design. As is shown in Fig. 1(a), multiframe gated X ray cameras are used to image the capsule implosion
near stagnation both perpendicular to the capsule axis and through the laser entrance hole. To examine the time dependence of the drive, we modify the capsule design or truncate the laser pulse. We also use surrogate targets that can be imaged throughout the drive pulse.

In order to model these experiments we have developed large scale integrated models of indirectly driven targets that include as much physics as possible to treat the hohlraum in a self-consistent fashion. We have found a subtle but important interplay between complex plasma hydrodynamics inside the hohlraum and the generation and distribution of the drive radiation that is sensitive to details of the hohlraum such as the laser entrance hole and the geometry. Using as-built target geometries and laser shot histories, we are able to model the peak temperature and the time history of the drive to an accuracy of better than 10%.

3. HOHLRAUM PLASMA DYNAMICS

Hohlraum plasma dynamics has important consequences for the energy balance and ultimately the capsule performance of indirect drive targets. The interaction of the drive beams with the hohlraum wall causes substantial ablation that eventually fills the hohlraum with plasma. Stagnation of this blowoff plasma can result in refraction of the incident laser beams, additional radiation sources, paths for electron thermal conduction between the laser spots and the capsule, and hydrodynamic pressure. In collaboration with the Lawrence Livermore National Laboratory (LLNL), we have investigated several means of quantifying these effects, including the observation of stimulated Brillouin scattering and the use of openings in the hohlraum wall to image internal dynamics. Of particular interest are interactions between the capsule and hohlraum and the ability of indirect processes to influence the radiation field inside the hohlraum.
The Lagrangian codes often used for ICF calculations fail to describe effects associated with plasma interpenetration and stagnation. We have adapted our particle-in-cell code, ISIS, to study plasma motion in complex three-dimensional target geometry. At low densities we find substantial interpenetration of the plasma. As an ion approaches the opposite wall, however, its velocity is reversed as it is captured by the higher density counterstreaming flow. Eventually, the turnaround point approaches the centre of the target, and a dense stagnation feature appears. Since the stagnation process converts directed ion motion to thermal energy, the ion temperatures can rise to tens of kilovolts. An unresolved question is the electron temperature, which results from an energy transfer balance between the ions and the radiation field. Since the spectrum of this radiation will be non-Plankian, an accurate radiation transport model is required to resolve energy balance issues. In 1993, we plan a series of colliding plasma experiments to examine stagnation physics, including the role of plasma radiation.

4. IMPLOSION HYDRODYNAMICS

Control of hydrodynamic instabilities during implosions is essential to avoid shell breakup and unacceptable mixing of fuel and pusher. Although much progress has occurred in the understanding of instability growth in planar geometries, questions remain regarding the importance of mode coupling and other phenomena in convergent geometries. Time dependence in the capsule drive can result in shear flows (e.g. oblate to prolate capsule shape), which can interfere with familiar bubble and spike growth, perhaps resulting in the premature onset of turbulence. To assess these effects, we are conducting a preliminary series of experiments in cylindrical geometry, shown in Fig. 1(b), that will allow us to image instability growth with the effect of convergence.

High convergence implosions are taxing on hydrodynamics computer codes owing to approximations required to handle shocks, etc. It is useful to have analytic solutions to which computer calculations can be compared, especially for multidimensional calculations. We have developed a set of group invariant analytic solutions to the 3-D one-temperature perfect gas hydrodynamic equations including thermal conduction and energy sources that provide benchmarks for numerical calculations.

5. TARGET FABRICATION

Some ignition target designs contain a layer of solid DT as the innermost shell. The beta layering method may offer a technique for the production of such layers in situ. When DT is frozen, the natural beta decay of tritium results in non-uniform heating leading to enhanced sublimation in the thickest regions and eventually to an even distribution of frozen DT in the capsule. We have recently succeeded in producing a 73 µm thick DT layer at the interior of a 2 mm diameter cylinder. The layer
is uniform to within 4 µm, and there appears to be a clear path to improvement. Optical and neutron scattering are being used to determine the uniformity of the layer, the surface finish and the presence of helium bubbles associated with tritium decay products.

6. LASER SCIENCE

KrF lasers offer advantages for ICF because of their short wavelength, high efficiency, broad bandwidth, pulse shaping capabilities and ability to be repetitively pulsed. We are constructing a small (800 J) KrF laser called Mercury that will constitute a flexible KrF laser technology testbed. Mercury is a 48 beam angularly multiplexed machine that uses an all reflective optical system with two electron beam pumped amplifiers. Pulse length and shape will be variable from 200 ps to over 5 ns. With a spot size of 200 µm, focused intensities in excess of 10^{16} W/cm^2 are anticipated.

To assist in the development of target diagnostics and ideas for ICF experiments on larger facilities we have constructed a small two-beam glass laser called Trident. Trident has two frequency doubled beams of 100 J each in a 200 ps pulse with a third, lower energy, beam used for backlighting purposes.

Construction of an ignition class laser at an affordable price demands substantial reductions in optics fabrication costs. We have recently demonstrated a rapid pad polishing method that employs careful thermal management of the glass to reduce the time required to figure mirrors to a few per cent of standard industrial techniques. This process has been successful in producing many of the optics required for the Mercury and Nike KrF lasers.

7. CONCLUSIONS

A collaborative programme of target physics research involving LLNL and Los Alamos has led to rapid progress in our understanding of indirect drive laser fusion. Current emphasis is on hohlraum drive symmetry, hohlraum plasma dynamics and capsule hydrodynamics. Within the next few years we should be in the position to confidently proceed with the construction of a national ignition facility in the United States with the objective of thermonuclear burn in the laboratory.

ACKNOWLEDGEMENTS

This work was performed under the auspices of the United States Department of Energy by the Los Alamos National Laboratory. The authors wish to thank the members of the Los Alamos ICF programme for their assistance in the preparation of this overview.
DISCUSSION

H. TAKABE: In the case of the laser driven indirect drive scheme, the temperature of the plasma heated by laser light is in excess of 1–2 keV. In such a situation non-local-thermodynamic-equilibrium (non-LTE) atomic modelling is essential in dealing with radiation transport, etc. Did you include it in your ‘integrated model’ calculations? Furthermore, do you include the radiation trapping effect when calculating the atomic state of the cavity plasma?

S.M. YOUNGER: Our integrated model calculations routinely employ non-LTE with accurate radiation transport. Detailed line transport can be done with some codes, but can be adequately represented by models in more complex calculations.

S.E. BODNER: The Los Alamos National Laboratory has carried out extensive experiments and simulations of implosion asymmetry on NOVA. How would you characterize the results? Are they reproducible, controllable and predictable?

S.M. YOUNGER: At present, there is considerable scatter in the symmetry data — due, at least in part, to pointing and beam balance inaccuracies on the laser. We have noted that quality improves as these laser problems are overcome. The predictive capability for hohlraums has improved considerably. Recent calculations show only qualitative agreement with experiment, but this process has only just begun in earnest. Significant progress can be expected over the next year.

M.G. HAINES: Is it possible to use the radiation pressure to control the buildup of plasma in the hohlraum?

S.M. YOUNGER: This is unlikely since plasma blowoff occurs early in the laser pulse, when the radiation pressure is low. It is possible for the laser beams to form channels through the plasma, although simple plasma heating is important here.
Abstract

RECENT DIAGNOSTIC DEVELOPMENT FOR THE LOS ALAMOS INERTIAL FUSION PROGRAMME.

Diagnostics for indirect drive inertial confinement fusion (ICF) require significant advances in time, space and energy resolution. Two new diagnostics meant for permanent operation on the Nova Laser Facility are reported on: (1) an ion temperature diagnostic based on single hit neutron time of flight measurements for the study of implosion dynamics for d-d and d-t fusion targets, and (2) a multiframe gated X-ray camera for acquiring highly time resolved two dimensional target images.

1. ION TEMPERATURE DIAGNOSTIC

Neutron time of flight (TOF) detectors can provide important information on the fuel ion burn temperature in various inertial confinement fusion (ICF) target designs. Conventional current mode neutron TOF detectors measure the time history of the total light output from many neutron interactions in a scintillator. These detectors are useful for neutron yields above $10^{10}$, but are limited at lower yields by the finite emission time of the scintillator, the finite response time of the detector, and the statistics of neutron scattering and light production in the scintillator [1].

We are constructing an ion temperature ($T_i$) diagnostic based on timing measurements of single neutron interactions in many scintillators. This diagnostic is designed for low yield targets on the Nova ICF laser facility at Livermore (Fig. 1). The diagnostic measures the neutron arrival time distribution using an array of 960 scintillator-photomultiplier detectors with about 1 ns time resolution and operated in the single hit mode [2]. The arrival time distribution is constructed from the results of 100 or more detector measurements. The diagnostic will be located outside the Nova target chamber at a distance of about 28 m from the target.

The ion temperature is determined from the spread in neutron energy as denoted by the relation [3] $\Delta E_n(\text{keV}) = C_{dd}T_i(\text{keV})^{1/2}$, where $\Delta E_n$ is the energy spread (full width at half maximum, FWHM) and $C_{dd} = 82.5$ (for d-t reactions the coefficient is $C_{dt} = 176$). The energy spread is related to the time spread by $\Delta t/t \approx -(1/2)\Delta E_n/E_n$. Corrections to the raw data must account for the system time resolution (about 1 ns) and target burn time (about 100 ps).
FIG. 1. In situ neutron time of flight ion temperature diagnostic, as arranged in the Nova laser facility.

FIG. 2. Photomultiplier array for ion temperature diagnostic.
The neutron arrival times are detected by using a photomultiplier tube (PMT) to observe the photons produced by recoil protons in a plastic scintillator. The recoil proton energy depends on the proton recoil angle $\theta$ through the kinematic relation $E_p = E_n \cos^2 \theta$, where $\theta$ is measured with respect to the incident neutron direction. Since n-p scattering is isotropic in the centre of mass frame, the proton recoil energy distribution is uniform up to the neutron energy. The pulse height contains no information on the incident neutron energy, but can be useful for monitoring the system gain or in some pulse pile-up rejection methods.

The dynamic range of the diagnostic covers d-d neutron yields from $5 \times 10^7$ to $10^9$. This range can be obtained through variation of the number of array channel hits (from 100 to 500) and scintillator volume (from 0.8 cm$^3$ down to 0.2 cm$^3$). Operation at yields above $10^9$ can provide overlap with results obtained from current mode time of flight detectors [1] and below $5 \times 10^7$ with estimates of $T_i$ obtained from a first hit analysis [4] of data from the Large Neutron Scintillator Array (LaNSA) on Nova.

Another method of extending the dynamic range is by operating the array in the multiple hit regime. An acceptable fraction (10%) of the channels will provide single hit data even when the average number of hits per channel is three. Thus, a dynamic range of about 30 is possible provided that channels affected by pulse pile-up can be rejected. Data reduction based on first hit analysis [4] can also be considered.

The detectors are enclosed in a cylindrical steel chamber (Fig. 2), which provides magnetic shielding for the PMTs. A total of 1029 PMTs, including spares, are supported by fore and back planes made of black nylon. A spring behind each PMT base presses each PMT against its scintillator. A silicone pad is used to improve the optical coupling. Gamma sources are included for in situ calibration. The chamber is supported by a stand allowing it to be rotated for alignment or maintenance.

The electronics for each channel consist of the PMT voltage divider base, a discriminator, a multiple hit time to digital converter (TDC), and a gated charge sensitive analog to digital converter (ADC). The multiple hit capability of the TDC is valuable in observing the gammas from neutron interactions with materials along the neutron flight path and thus correcting for timing differences between channels. ADCs can be valuable for recording the pulse height distribution of the recoil proton scintillations to monitor the overall health of the PMT array.

Initial testing and application to Nova implosion dynamics experiments are planned for late 1992.

2. HIGH SPEED GATED X RAY IMAGER

Gated X ray detectors have been used for several years to image laser produced plasma events for the ICF community [5]. Optical gate times of $\leq 100$ ps are necessary to reduce motional blurring when spatial resolutions of 10 $\mu$m are required for plasma velocities of $1 \times 10^7$ cm/s. The ICF applications for gated X ray detectors
vary from observation of plasma blowoff to the dynamics of capsule implosion symmetry. We have completed a gated X ray camera that is now in routine use on the Nova laser system. A picture of the camera is shown in Fig. 3.

The compact 16-channel instrument constitutes a significant improvement over earlier generations. The new camera is based on a gated microchannel plate (MCP) operated in a stripline configuration [6]. The MCP has 4–12.5 Ω striplines deposited on the front surface photocathode. Each stripline has a $-880 \text{ V}$, 180 ps FWHM voltage wave that propagates across the MCP and creates the gate. Gating pulses are generated by the avalanche breakdown of 24 avalanche transistors which charge in parallel and fire in series. After the gate pulse has travelled across the stripline, it is directed to a monitor scope for timing and diagnostic utilization. Along with the gate pulse a variable negative DC bias voltage can be applied for gain control. Each of the 16 optical channels can be filtered differently for spectral discrimination by using a slip-in filter pack located just in front of the MCP. Delays between the four striplines are switch adjustable from 0 to 5.5 ns in 50 ps steps, while the delays between the four channels on each strip are fixed to 60 ps because of the propagation velocity of 15 mm/100 ps. Imaging is accomplished with 5 μm pinholes and selectable magnifications of 4, 8 and 12 times. The gold photocathode is sensitive from approximately 1 to 5 keV, with a maximum sensitivity at 3 keV. Temporal reso-
olution has been experimentally measured to be 90 ± 10 ps, by using a short pulsed (3 ps) UV laser. Spatial resolution is primarily limited by proximity focusing between the channel plate and the phosphor screen and is predicted to be about 7 μm at the object plane with 12 times magnification. Currently, images are recorded on Kodak 2484 film and then digitized for archival and analysis.

Results from several experimental campaigns using our new camera have shown it to be flexible and easy to maintain while producing high quality X ray images.

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KRYPTON-FLUORIDE LASER FUSION DEVELOPMENT IN THE USA*

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Abstract

KRYPTON-FLUORIDE LASER FUSION DEVELOPMENT IN THE USA.

The first phase of the Nike 5 kJ laser has been completed at the Naval Research Laboratory, consisting of the oscillator through the first E-beam amplifier, with 28 multiplexed laser beams. The individual laser components are all highly reliable. This laser subsystem is now undergoing laser tests. The energy through the first E-beam amplifier during preliminary measurements was 180 J; this meets the design requirements. Recent oscillator improvements, optical design studies and tests in the propagation bay all indicate that it should be possible to demonstrate a 2% nonuniformity in the total laser system, with 3000 J on the target. A new automatic alignment system demonstrates that there is no longer any difficulty in rapidly aligning multiple-beam laser systems. In a parallel effort at Los Alamos National Laboratory, the Mercury laser is being constructed to address KrF system engineering tests and advanced KrF technology concepts such as advanced pulse shaping, very broad bandwidth, and high repetition rates.

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1. INTRODUCTION

The primary near-term justification for developing KrF lasers in the U. S. is its potentially superior capability over a glass laser to produce a sufficiently-uniform laser beam for directly-driven pellet implosions. The primary long-term justification for KrF is of course its attractive potential for meeting the requirements of a reactor system, with high efficiency, moderate cost, scalability to megajoule size, short wavelength, very broad bandwidth, etc.

Before the invention of the first beam smoothing concepts, large glass lasers typically had on-target intensity nonuniformities and ablation pressure nonuniformities of 400%-1000%. With the invention of various optical smoothing techniques in the 1980s, the nonuniformities were typically reduced to about 10%, and occasionally to a few percent\(^1\). However achieving ignition and high gain with direct-drive pellets may require that the ablation pressure nonuniformities be reduced to ~ 1-2% peak to valley (less than 1% rms) for some of the spatial modes around the pellet. One potentially serious problem is an uncontrollable tilt or curvature in the individual laser beams. Although there are now design concepts for both KrF and glass lasers that may be able to reduce this type of distortion to the 2% level, it has not yet been experimentally demonstrated, and it is not yet clear which laser driver can really meet this requirement.

The Nike laser is now being developed at the Naval Research Laboratory, with the support of Los Alamos National Laboratory, to address this beam uniformity issue. The optical smoothing technique we are using is called "echelon-free induced spatial incoherence"; it is based upon a simple image relaying of a spatially-incoherent but uniform beam from the oscillator to the target\(^2\). This technique strongly favors the use of a gas laser such as KrF, to minimize nonlinear effects in glass which are proportional to \(\int n_2 I \, dz\). The first phase of Nike is now assembled and undergoing tests that are reported here. See Fig. 1. The laser should be completed in 1993, with target experiments in 1995 (funding limited). If we are successful, Nike will have the most uniform and controllable laser beam profile of any laser fusion facility. The detailed laser design is guided by the need to evaluate
FIG. 1. Amplifier and optical staging for the Nike laser.
methods of controlling the Rayleigh-Taylor hydrodynamic instability physics in directly-driven targets.

The Mercury laser is being developed at Los Alamos National Laboratory. Its highest priorities will be to serve as a test bed for KrF technology assessment and development. Such assessment and development will address both the issues of system engineering and the testing of advanced KrF technology concepts, both of which would be important for any potential future scaled systems. While Nike will concentrate on beam smoothness and specific direct-drive experiments, Mercury will emphasize testing of other KrF advantages such as very broad bandwidth, enhanced efficiency, and high shot rate. Mercury will also be used for some laser-target experiments.

2. NIKE LASER AT THE NAVAL RESEARCH LABORATORY

2.1 Optical Design and Optical System

The underlying principle of echelon-free ISI is to form a high quality image on the target by image-relaying an object aperture through the laser system. If this aperture is uniformly illuminated with spatially and temporally incoherent light, then the amplified image on the target may also be uniform. This design approach requires careful control of various aberrations that can distort the focal image. These distortions include systematic aberration (i.e. spherical and chromatic aberration, coma, astigmatism, hard apertures, etc.), random aberrations (optical imperfections and atmospheric turbulence), and nonlinear effects (nonlinear refraction, self-seeded broad band stimulated rotational Raman scatter in air, two-photon absorption in the lenses and windows). All of these issues, as well as automatic remote beam alignment and atmospheric absorption, have been addressed in the Nike optical system and are discussed below.

The systematic aberrations have been virtually completely corrected for with the Nike design. Ray trace calculations indicate diffraction-limited performance for the entire system at bandwidths up to 5 THz (0.25%) with primarily reflective optics.
The random aberrations have been designed to keep the laser to less than 10 times diffraction limited performance. Since the nominal operating conditions for Nike involve imaging a 60 times diffraction-limit beam, these random aberrations do not excessively affect the target beam profile. We have established an allowable surface error for a single optical element that is typically 12 nm rms. To date, all of the optics procured for Nike have been approximately 1.5 to 2 times better than this requirement.

Random aberrations due to thermal gradients in the atmosphere were a major design consideration for Nike, because the angularly multiplexed beams travel in excess of 300 meters. Small levels of atmospheric turbulence over this path length can cause image degradation. Therefore most of the beam propagation is in a sealed, environmentally-controlled room called the propagation bay. The quiescent air in this room is surrounded by three layers of walls, three layers of ceiling, and two layers of insulated and heated flooring. A computer controlled air conditioning system adjusts the system. Tests in this propagation bay have demonstrated diffraction-limited performance with a 10 cm diameter, 632 nm laser beam for a path length of 70 meters. Atmospheric distortion will not be a problem for Nike. In addition, ozone, hydrogen sulfide, and most hydrocarbons strongly absorb at 248 nm. Materials used in the construction of the Nike bay were chosen to minimize outgassing of hydrocarbons, and the air is filtered by HEPA and carbon charcoal filters. Initial measurements with unfiltered air during our current construction phase show absorption in the range of 10-17%. Air filtering should substantially reduce this number; our base design point for Nike allows for 10%.

Nonlinear distortion due to propagation of high laser intensities in air is a potentially serious problem. We predict severe and unacceptable degradation of the beam profile for B-integrals $B > 0.4$. Since Nike operating conditions are in the 0.4-0.7 range, we are planning to fill the propagation bay with an inert gas atmosphere (Ar or He). This would reduce the total B-integral to 0.2, which has a negligible effect on the beam quality. We plan experiments in the near future to better quantify the nonlinear effects before committing to an inert gas atmosphere.
To control thermal distortion, and to deal with the potential of an inert atmosphere, we have developed a remote alignment system that is capable of controlling the approximately 250 optical components in Nike. Commercial CCD cameras and image analysis software monitor the laser beam positions at various points in the system, and individual mirrors are then adjusted with custom-designed stepping motors and custom driver software with PC architecture. The system has demonstrated the ability to simultaneously align an array of 28 mirrors in approximately 10 sec. This should allow remote alignment of the entire Nike optical system in less than 15 min. We believe that we have successfully addressed the common concern that one could not easily align large numbers of laser beams.

A pointing stability goal of \( \pm 5 \, \mu \text{Rad} \) on target is the overall goal for Nike, although most experiments could be performed with \( \pm 10 \, \mu \text{Rad} \). This leads to an error budget for vibration with individual mirror stability of typically \( \pm 2 \, \mu \text{Rad} \). Measurements on the floor of the Nike propagation bay meet this requirement, and the various optomechanical structures are being designed and tested to minimize any amplification.

### 2.2 Oscillator, Pulse Slicing, and Uniformity Measurements

If we are to image a uniform and controllable laser profile at the target, we must first be able to produce the same profile in the object plane, which comes after the oscillator. Our oscillator system can now produce a 60 times diffraction limited flat intensity profile with \( \sim 1\% \) tilt, in a 4 ns laser pulse. The bandwidth is adjustable; currently it is 1.6 THz.

The oscillator uses a lens that images the gain medium into itself off of the back mirror. This makes the output appear to come from the entire laser medium, resulting in the high angular divergence. The oscillator is then split into two beams, one of which passes through a single telescope that inverts the image. The other beam is uninverted. When these two beams are then combined, the first-order tilts (which vary during the oscillator pulse) are greatly reduced.
The 4 ns pulse is produced with two Pockels cells in series. The energy contrast ratio was $\sim 3000$ to 1. The power contrast ratio has not yet been measured with the present system, but is expected to be higher.

A typical profile is shown in Fig. 2. The variation of the fit was 2% within the central 65% of the diameter. The shape is really typical, and is invariant if the slice is taken at other times during the oscillator pulse, and is invariant when the energy is decreased by a factor of 10, as will be necessary with our planned pulse shaping. We are now evaluating various further improvements, such as the use of a larger laser cavity to reduce the quadratic variation, and alternate Pockels cell designs for a still larger dynamic range in pulse shaping.

### 2.3 Electron-Beam Pumped Amplifiers

#### 2.3.1 20 cm Amplifier

The pulsed power performance of the 20 cm Amplifier has now met or exceeded all of our system requirements. It produces a flat-top power pulse. The
foil that separates the e-beam from the laser gas has proven to be relatively transparent, with measurements of Faraday cups and pressure rise showing that 46±8% of the cathode current is transmitted into the gas; this is rather high for this type of device. The total energy deposited in the gas exceeds our gain requirements. The total shot-to-shot jitter, from control-room jitter to rise time of the beam current, is less than 10 ns, which eliminates any timing concerns. And most important, the 20 cm Amplifier has proven to be very reliable – it has been fired at 60 shots per day with over 99% reliability!

We are just beginning to test the 20 cm Amplifier as a laser. Using a test optical system with six 16-ns FWHM input beams, we were able to extract 180 Joules from the amplifier. These first tests indicate that we will have sufficient energy to drive the 60 cm Amplifier.

2.3.2 60 cm Amplifier

This larger 60 cm Amplifier, like the 20 cm Amplifier, is a two-sided electron-beam pumped system, and accordingly we have incorporated many of the features that we developed for the 20 cm system, including laser trigger switches, the cathode structure, the Hibachi assembly, the laser cell, the guide magnet, and all of the support systems. The primary differences, in addition to the larger size, are the use of independent Marx generators to drive each side of the amplifier, and use of four coaxial lines on each side (instead of one). Each line is triggered by its own switch and fed through its own vacuum insulator to a common cathode. A 3.5 kG electromagnet guides the electron beam from the cathode, through the anode screen, through the Hibachi/foil structure, and then into the laser cell.

We expect the 60 cm Amplifier to produce two 671 kV, 540 kA electron beams. The calculated diode power is flat within ±5% for 250 ns, which is more than adequate to amplify the 240 ns long train of 56 laser pulses. We have designed in some flexibility in the diode parameters. We can raise the diode impedance up to 1.5 Ω if required for beam stability, or lower it to 1.0 Ω if required by laser deposition. Based upon our experience with the 20 cm Amplifier, we expect that approximately 50% of the electron beam will be deposited into the laser gas. With
0.5-0.6 MW/cc deposited in the gas, and with 60% of the total laser light reaching the target, we expect 3000 J on the target in the total of 56 four-ns laser beams. The laser energy is dependent upon uncertain laser kinetics, and the 3000 J may be an underestimate.

3. MERCURY LASER AT LOS ALAMOS NATIONAL LABORATORY

For several years until early 1991, Los Alamos assembled and tested a prototype KrF laser system, called Aurora, which was designed to test key concepts of KrF technology and to provide laser energy for inertial confinement fusion experiments. The results of these tests were generally successful, and key elements essential to the issue of KrF lasers for fusion research were demonstrated. These included angular multiplexing and large-volume electron-beam amplifier technology. Several features of the implementation were, however, of limited success, including the use of amplifiers in a single-pass geometry, a partially-refractive optical train, and a complex control system. Building upon the lessons learned from Aurora, and working within the confines of a modest budget, Mercury is a smaller system that makes use of as many Aurora components as possible, improving and modifying them as needed.

The Mercury design invokes a reduction in the number of amplifiers, and the remaining amplifiers are used in a double-pass configuration (as with Nike) resulting in considerably higher stage gains. A reduction in the charge voltages, currents, and pulse lengths results in improved reliability for the pulse-power systems. The refractive optical system from Aurora is replaced with an all-reflective system, which provides a much-improved beam quality. The front end has been rebuilt, and it can now generate adjustable pulse lengths from 200 ps to 5 ns, with arbitrary pulse shape. The combination of available shorter pulse length and improved focusability (<200 μm spot size) will make available power output up to > 4 TW with focal intensities up to > \(10^{16}\) W/cm\(^2\) for the nominal 1 kJ that Mercury is expected to generate when completed. That intensity level provides useful capability for both direct and indirect drive ICF experiments.
Mercury phase II
48 beams

FIG. 3. Amplifier and optical staging for the Mercury laser.
A schematic diagram of the conceptual design of the Mercury laser system is shown in Fig. 3. The front end consists of an oscillator with multiple Pockels cell switches, generating a single pulse of arbitrary shape and pulse length. Contrast ratio can be enhanced by the addition of more Pockels cells after the first preamplifier. The resulting beam is replicated 12-fold with angle and time encoding (5 ns beamlet spacing) by aperture division, and is then amplified in a double pass through a $12 \times 12 \times 100 \text{ cm}^3$ electron-beam pumped amplifier (A1). This 12 beamlet train is then further replicated 2-fold by amplitude division, and is angularly encoded before a double-pass through a $25 \times 25 \times 100 \text{ cm}^3$ intermediate electron-beam-pumped amplifier (A2). Finally, each of the two 12 beamlet envelopes is again replicated 2-fold by amplitude division, and is angularly encoded through a $55 \times 55 \times 200 \text{ cm}^3$ electron-beam-pumped amplifier (A3). The resulting 48 beamlets then pass through an optical "decoder" system, which removes their time delays and focuses all of the beams simultaneously onto the target.

A number of design verification tests have been performed, demonstrating the feasibility of key aspects of the design. These include the ability to extract energy efficiently at 200 ps pulse lengths, the ability to achieve focal spot sizes of $< 200 \mu\text{m}$ (90% encircled energy), specified performance from the amplifiers, and the absence of degrading nonlinear optical effects at the design intensities. At the time of this writing, the Mercury laser system is half-completed and several sub-system components have been successfully tested.

REFERENCES

DISCUSSION

J.M. ARAGONES BELTRAN: Can you comment on the pulse shaping capabilities of the KrF lasers at the Naval Research Laboratory and Los Alamos National Laboratory?

S.E. BODNER: Since the amplifiers are heavily saturated one can amplify ICF pulse shapes without excessive distortion. Los Alamos scientists have calculated that the 'Kidder' pulse shape can be used. Of course, some pulse shapes are harder to produce with a KrF laser, but they are less relevant to ICF.

B. COPPI: What, in your opinion, are the most significant factors that have turned the tide in favour of direct drive?

S.E. BODNER: First, the invention of optical smoothing techniques at Osaka University, the Naval Research Laboratory, the University of Rochester and elsewhere has allowed us to move from infrared lasers, which were required where one relies on thermal smoothing, to UV lasers, which satisfy the other constraints on ICF. Secondly, the demonstration at NRL, in the mid-1980s, that optically smoothed laser beams dramatically reduce all the deleterious laser–plasma instabilities.
LIGHT ION DRIVEN INERTIAL CONFINEMENT FUSION

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Abstract

LIGHT ION DRIVEN INERTIAL CONFINEMENT FUSION.
Advances are reported in theoretical and experimental research in light ion driven inertial confinement fusion. The first ion driven hohlraum experiments at >100 TW/gm and the first ion driven implosion experiments are reported. Achievement of a laser initiated preformed lithium plasma source and generation of a pure lithium beam from this source are reported. Development of the theoretical understanding of ion beam divergence and very recent results from experiments demonstrating reduction of ion divergence in both proton and lithium beams are also reported.

1. LIGHT ION DRIVEN INERTIAL CONFINEMENT FUSION TARGET EXPERIMENTS

Intense light ion beams are being developed to drive inertial confinement fusion (ICF) targets [1]. While ion driven targets have several features in common with laser driven targets, as depicted in Fig. 1, there are some differences. Two basic differences are that (1) no holes are required in ion driven targets since the ions can pass through the radiation case, and (2) the ion hohlraum is filled with a low density foam which serves as the absorber region. Recently, intense proton beams were used
to drive two different types of targets in experiments on the Particle Beam Fusion Accelerator (PBFA II). The experiments focused separately on ion deposition physics and on implosion hydrodynamics, as shown in Fig. 2. In the ion deposition physics experiments, a 30–50 kJ proton beam heated a low density hydrocarbon foam contained within a gold cylinder with a specific power deposition exceeding 100 TW/gm for investigating ion deposition, foam heating, and generation of X rays.

FIG. 1. Features of laser driven and ion driven targets.

FIG. 2. Separable issues of ion deposition and radiation conversion physics, and implosion physics.
FIG. 3. Configuration of applied magnetic field ion diode and beam and target diagnostics.

FIG. 4. Configuration of the ion deposition and radiation conversion experiments shown with a time integrated soft X ray image of the foam region.
The proton beam was generated by an applied magnetic field ion diode, shown schematically in Fig. 3, and focused to an intensity of approximately 3.5 TW/cm² at the target. The configuration of these experiments, along with a time integrated soft X-ray image of the foam region, is shown in Fig. 4. The significant results from these experiments included the following: (1) the foam enhanced the radiation output; (2) the foam provided an optically thin radiating region; and (3) the foam tamped the radiation case, retarding the motion of the gold [2].

In the spherical hydrodynamic target experiments, a direct drive exploding pusher configuration was used. The target consisted of a thin plastic shell, doped with a chlorine layer on the inner surface and filled with 1.2 atmospheres of deuterium. The ion beam azimuthal non-uniformity at the equator of the target was 15–20%. The configuration of these experiments, along with a time integrated hard X-ray image of the beam imaging cone, the shell motion and the imploded core, is shown in Fig. 5. The significant results from these experiments included the following: (1) chlorine inner shell radiation induced by the ion beam was sufficient to provide an X-ray measurement of the shell velocity; (2) the target compression was about 5:1, consistent with the level of beam non-uniformity; and (3) there were distinct differences in the X-ray emission between filled and unfilled targets as predicted by hydrodynamic simulations [3].

FIG. 5. Configuration of the implosion physics experiment shown with the time integrated hard X-ray image of the implosion and stagnation features.
2. SPECIFIC POWER DEPOSITION

An important figure of merit for ion driven targets is the specific power deposition of the ion beam in the foam absorber region. The specific power deposition (TW/gm) is determined by the ratio of ion beam intensity (TW/cm\(^2\)) to the ion range in the absorber (gm/cm\(^2\)). To increase the specific power deposition in the target, we are developing focused lithium beams. Since lithium ions have a much shorter range in target material than protons at comparable kinetic energy, the specific power deposition is much higher for lithium ions than for protons. As shown in Table I, a 9 MeV lithium ion beam at a power intensity of 1 TW/cm\(^2\) provides a specific power deposition of 380 TW/gm, more than three times the specific power deposition of 110 TW/gm provided by a 5 MeV proton beam at a power intensity of 3.5 TW/cm\(^2\). The peak values of beam intensity achieved in PBFA II experiments so far are 5 TW/cm\(^2\) for protons and 1 TW/cm\(^2\) for lithium. The significance of these levels of specific power deposition can be seen in Fig. 6. Ion deposition and equation of state physics can be studied in the range of 10–100 TW/gm. Between 100 and 1000 TW/gm, radiation physics can be investigated. Above 1000 TW/gm, significant radiation smoothing is possible, enabling indirectly driven implosion experiments [4].

<table>
<thead>
<tr>
<th>Ion</th>
<th>Ion energy (MeV)</th>
<th>Range (^a) (mg/cm(^2))</th>
<th>Intensity (TW/cm(^2))</th>
<th>Specific deposition (TW/gm)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Protons</td>
<td>5</td>
<td>32</td>
<td>3.5</td>
<td>110</td>
</tr>
<tr>
<td>Protons</td>
<td>5</td>
<td>32</td>
<td>5</td>
<td>157</td>
</tr>
<tr>
<td>Lithium</td>
<td>9</td>
<td>2.6</td>
<td>1</td>
<td>380</td>
</tr>
</tbody>
</table>

PBFA II capability today

Near term improvements in lithium focusing

| Lithium | 9 | 2.6 | 2 | 760 |
| Lithium | 9 | 2.6 | 5 | 1900 |
| Lithium | 9 | 2.6 | 10 | 3800 |

\(^a\) Range for CH foam.
Our long term goal for PBFA II is to perform target experiments using a lithium ion beam in the range of 5000–10,000 TW/gm by increasing the beam intensity and selecting the ion energy to optimize conversion of ion energy to X ray energy for driving fuel filled capsules. Our immediate goal is to push above 1000 TW/gm with a lithium beam. The specific power deposition at a lithium beam intensity of 10 TW/cm$^2$ will enable us to study deposition physics at a specific power deposition level (3800 TW/gm) comparable to that required for ignition. The specific power deposition for an ignition target requiring a 120 TW/cm$^2$ lithium beam at 30 MeV is 6200 TW/gm. For an ignition target, the much higher beam intensity is required to provide an adequate amount of energy within the hydrodynamic acceptance time of the target.

3. LITHIUM BEAM INTENSITY

The achievable lithium beam intensity in PBFA II is determined by three factors: (1) the amount of power coupled from the accelerator to the ion beam; (2) the lithium beam purity; and (3) the lithium beam divergence. The lithium beam divergence has an inverse quadratic effect on the beam intensity and is therefore most important.
Comparison with >3 h heating (left) and <1/2 h heating (right):

FIG. 7. Spectroscopic data showing the effect of substrate heating on the level of contaminants in the LEVIS lithium ion source.

4. LITHIUM BEAM GENERATION

Good coupling of power from the accelerator to the ion beam requires that the ion source be ready to produce ions as soon as the accelerating power pulse arrives; i.e., the source should be preformed [5]. Visible emission spectroscopic measurements [6] have recently confirmed the presence of a preformed anode plasma in PBFA II experiments using a two-step laser evaporation and ionization approach (LEVIS) [7]. By heating the LiAg substrate to 150°C for five hours, generation of a pure lithium beam from this source has recently been achieved. Spectroscopic data from the LEVIS source with >3 h heating and with <1/2 h heating are shown in Fig. 7. The heating produced a reduction of 3-5 in carbon line intensities, a reduction of 40 in hydrogen line intensity, and a >90% pure lithium beam. The preformed LEVIS source provides an important basis for experiments on reduction of the divergence of the lithium beam, a critical step in demonstrating the feasibility of light ion fusion.

5. ION BEAM DIVERGENCE

Simulations using the three dimensional electromagnetic particle-in-cell code QUICKSILVER have identified an early time diocotron instability in the electron flow in the diode [8]. Analytic calculations [9], which include a charge neutral region
in the diode following the beam acceleration gap, produce a calculated growth rate for the diocotron instability in good agreement with the simulations. In the QUICKSILVER simulations, the early time diocotron instability evolves during the pulse to a low frequency instability which couples energy to the ion beam on the ion acceleration time-scale. When this enhanced coupling occurs, the ion divergence increases dramatically.

A trend of decreasing ion beam divergence with decreasing ion current enhancement over the Child–Langmuir value is also found in the simulations [10]. The simulations suggest a potential means to reduce ion divergence. By providing better control over the evolution of the electron density in the anode–cathode gap, the simulations show that the low divergence phase can be extended, avoiding the strong coupling between electromagnetic instabilities and the ion beam. Specific solutions include providing the electron control with an increased magnetic field (about 5.8 T for protons and 6.3 T for lithium), increasing the ion emission area, and using an electron limiter to limit electron density near the anode. Very recent data have demonstrated that these measures work. In proton beam experiments in extraction diode geometry on the KALIF accelerator at Kernforschungszentrum Karlsruhe, the divergence of the proton beam produced by an active Pd coated hydrogen loaded titanium thin film source was reduced by approximately 30% at modest magnetic field [11]. In similar proton beam experiments in extraction diode geometry on the LION accelerator at Cornell University, comparable reductions in the proton beam divergence were obtained, and the effect increased with magnetic field [12], as predicted by particle-in-cell simulations. Very recently, the divergence of a lithium ion beam produced by a LiF source on PBFA II in radial diode geometry was reduced by a factor of two. In these experiments, the ion emission area was increased from about 600 to 900 cm$^2$ and the physical acceleration gap (anode–cathode gap) was increased from 15 mm to 20 mm. The image of the radially focused lithium ion beam

![Graph and Image](image-url)

**FIG. 8.** Ion pinhole camera image of a lithium beam on PBFA II with 20 mrad divergence.
taken by an energy filtered ion pinhole camera on PBFA II is shown in Fig. 8. The horizontal full-width-at-half-maximum intensity in the highest energy frame is about 6 mm, corresponding to a 20 mrad horizontal divergence. The intensity of the vertically unfocused beam is approximately 1 TW/cm². Vertical focusing of this beam is expected to increase the lithium intensity to about 3 TW/cm², a level which would enable target experiments at a specific power deposition exceeding 1000 TW/gm.

A proof-of-principle experiment has yielded important data on two-stage beam acceleration. Small scale experiments have demonstrated that the diode current of the second stage can be dominated by the current of the first stage. In addition, the 17 milliradian divergence of a proton beam in the first stage is reduced by post-acceleration in the second stage to 8 milliradians.

Together, these results with differing ion species (protons and lithium ions) and differing diode configurations (extraction diodes, radial diodes, single stage diodes, and multiple stage diodes) demonstrate that controlled reduction of ion divergence has been achieved. This breakthrough has substantially increased the prospects for ion beam focusing and the utility of intense ion beams with optimal range for ICF target experiments.

ACKNOWLEDGEMENTS

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[4] MEHLHORN, T.A., et al., to be published in the Proceedings of the 1990 Hirschegg Meeting. Figure 6 was adapted from Bock, Arnold and Meyer-ter-Vehn.
MULTISTAGE ION ACCELERATOR FOR INERTIAL FUSION ENERGY

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Abstract

MULTISTAGE ION ACCELERATOR FOR INERTIAL FUSION ENERGY.
Recent results of the two stage diode experiment on Reiden-SHVS are reviewed. The instability of the ion beam current and its stabilization, the evidence of charge stripping and the anomalous emittance reduction are discussed. The feasibility of this scheme for a commercial reactor driver is briefly treated.

1. INTRODUCTION

An important milestone in inertial fusion energy (IFE) research is to achieve ignition and high gain, which will be realized by a high power laser or particle beam machine in this decade. The next milestone is to develop the reactor driver with sufficient efficiency, low cost, long life and high repetition rate. The ignition driver might not meet the requirements of a driver for a commercial reactor.

From this point of view, the superhigh voltage source (SHVS) based on pulsed power technology has been proposed as a reactor driver candidate [1]. The characteristics of the SHVS are as follows: (1) Voltages of tens of MeV per stage can be generated by providing the inductive voltage of each unit with a magnetic insulation.
The voltage in each unit will be less than 1 MV; therefore conventional pulsed power technology will be applicable [2]. (2) The voltage waveform can be shaped to the bunching of the ion beam by controlling the voltage waveform of each unit [2]. (3) Heavier ions such as C, Na, etc. can be sufficiently accelerated up to the required energy. Especially, by using charge stripping, the acceleration gradient will be chosen to be very high, which implies compactness and lower cost of the driver [3]. This will be an essential advantage of the reactor driver over other devices.

In this paper, we review recent results of the two-stage diode on Reiden-SHVS and discuss the feasibility of a reactor driver.

2. STUDY ON TWO STAGE DIODE

2.1. Experiments on Reiden-SHVS

Figure 1 shows a schematic diagram of the two stage ion diode with charge stripping foil [4]. The energetic ions that are accelerated at the first diode are injected into the second diode through a thin charge stripping foil. The injected ions are stripped to higher charge states which are determined by the energy obtained in the first stage and are efficiently accelerated in the second diode.

The potential distribution in the ion beam injected diode is different from the conventional diode with a stationary anode plasma ion source. An ion beam of \( J < J_{10} \) generates the electric field that tends to extract the ion beam from the injection side of the second stage diode; \( J_{10} \) is the saturation current of the space charge limit [5]. In these conditions, a significant portion of the accelerated beam will be extracted from the second surface. This suggests operating the diode with \( J > J_{10} \) in order to avoid accelerating the undesirable ions generated from the anode electrode of the second stage. An ion beam of \( J > J_{20} \) induces the space charge instability [6],

![FIG.1. Two stage diode with charge stripping foil.](image-url)
which gives rise to the virtual anode potential and its oscillation. $J_{20}$ is the maximum steady state ion current density transmitted through the second stage. The ions are reflected back to the injection side, in this case.

The ions generated from the stripping foil degrade beam purity. An effect of the virtual anode is also to avoid undesirable ions accelerated from the stripping foil. We call this effect a virtual anode filter. The desirable operation regime is $J_1 < J < J_2$.

Experiments have been performed on the Reiden-SHVS induction adder accelerator [7]. It consists of eight induction adder units. Reiden-SHVS has a capability of providing 4–6 MV, 40 kA, 100 ns pulses to the single diode or to the two stage diodes. Positive and negative centre electrodes are extended from both sides. The diodes are located at the centre of the machine. The ground electrode connects the cathode of the first diode and an anode of the second diode. All diodes are powered by four induction cavities.

The anode of the first stage diode is made of aluminium to exclude the insulating magnetic field. Two types of ion source, a carbon flashboard plasma injection source located behind the anode and a paraffin groove anode, are used. Paraffin groove anodes are used to generate the proton beams in the experiments to measure beam trajectories and divergences. The outer and inner radii of the active anode are 8 and 5.7 cm, respectively. The beam cross-section is 100 cm$^2$. The diode gap and the insulation field are, respectively, $d_1 = 0.9$ cm and $B_1 = 7$ kG in the first diode. In the second diode, $d_2(B_2)$ ranges from 0.83 cm (9 kG) to 3.5 cm (6 kG). The diode currents are measured by B-dot probes, the ion current densities by biased charge collectors behind the second stage diode, and the beam trajectories and divergences by shadow boxes with $5 \times 3$ pinhole arrays located just behind the cathode of each diode.

Figure 2 shows measured B-dot signals of the first stage diode and their Fourier spectrum. Shot No. C107 is a reference shot operated without applied B-field and with no oscillation on the signal. The oscillations on the B-dot signals of the first diode are caused by space charge instabilities in the second stage diode through reflected ions from the large oscillating virtual anode. The data plot on the $V_1/V_c$ versus $V_2/V_c$ chart shows that shot Nos C114 and C145 are in the unstable region. The oscillation frequency and its dependence on the anode–cathode (AK) gap of the second diode are in agreement with the predictions from electron beam simulation [6]. By decreasing the AK gap of the second diode, stable operation is achieved (shot No. C119).

The charge stripping diodes are tested by using carbon flash board ion sources. A foil of approximately 0.1 $\mu$m thickness is used as stripping foil. Without stripping foil, the maximum energies of the $C^+$ and $C^{2+}$ ions are equal and about 1.1 MeV. An increase in the higher ion energy component is observed, obviously as a result of the diode with stripping foil. The charge state equilibration of the ions passing through the stripping foil is calculated from charge stripping and the charge recombination cross-section. The carbon ions $C^+$ and $C^{2+}$ are stripped (or partially recom-
FIG. 2. B-dot probe signal and its Fourier spectrum for various anode-cathode (AK) gaps and diode conditions. (a) $d_2 = 0.83$ cm; (b) $d_2 = 2.33$ cm; (c) $d_2 = 3.33$ cm; (d) no B field applied.
FIG. 3. Emittance plot of radial beam divergence in the first and second stages of a two stage diode.

(bined) after acceleration in the first stage diode ($V_1 = 0.6$ MV), and subsequently the acceleration energy is multiplied by the charge state times of the voltage of the second diode ($V_2 = 0.5$ MV). The maximum energy edges of all ion parabola traces are in agreement with the calculation. The lower energy part of the ions ($<0.5$ MeV) in the stripping foil case are lower than in the case without stripping foil. This indicates the effect of virtual anode filtering.

The decrease in the damage size in the second stage diode indicates an effective improvement in the beam divergence brought about by the second acceleration. Figure 3 shows the emittance plot of this shot when radical beam divergences are considered. The improvement in the emittance is better than the value predicted from the conservation of normalized beam emittance as

$$\epsilon_1/\epsilon_2 = 3.5 > \gamma_2 \beta_2/\gamma_1 \beta_1 = 2.2$$

The emittance of the first stage diode may be strongly influenced by the non-uniformity of the anode ion source implied by the non-uniform damage of the paraffin filled groove anode. The two stage diode may have an effect like an emittance filter and beam smoothing.

Small humps are observed in the damages. Beam trajectory calculations indicate that this may be due to reflected ions at the large virtual anode in the second stage diode. Even in these unstable, large virtual anode conditions, the remaining ion damages, due to unreflected ions, are less strongly affected by the virtual anode. Thus, it is interesting to use virtual anode operation in the second stage diode without beam degradation or deflection.

### 2.2. Simulations of the two stage diode

Simulations of the two stage diode are performed. The code is PIC 1.5D, with a non-linear circuit model for ion emission. The result corresponds quite well to the
experiments shown in Fig. 4. Oscillation of the ion beam current in the simulation is observed when \( J_i < 2J_{10} \), which is observed in the experiment. The simulation code is now being developed to include electron motion and sheath dynamics.

3. PROSPECTS FOR INERTIAL FUSION ENERGY DRIVER

In the reactor system, the power flow can be described as

\[
\eta_D G_i M \eta_f = \frac{P_{\text{out}} + P_{\text{reci}} + P_a}{P_{\text{reci}}},
\]

where \( \eta_D \) is the efficiency of the energy driver, \( G_i \) is the target gain and a function of the driver energy, \( M \) is a multiplication factor of the produced nuclear power in the reactor (when we use a Li waterfall, \( M \) is 1.1–1.3), \( \eta_f \) is the conversion efficiency, \( P_{\text{out}} \) is the output power of the power plant electricity, \( P_{\text{reci}} \) is the recirculated power for the operation of the energy driver, and \( P_a \) is the auxiliary power needed to run the reactor system. Here, we assume \( P_{\text{reci}} \ll P_a \).

The cost of electricity \( \Gamma \) per MW(e)-h in this system can be written as

\[
\Gamma = \left\{ [A \epsilon_b R_{\text{rep}}^{1/3} + B (\eta_f \epsilon_b MG_i R_{\text{rep}})^{1/3} (IR_{\text{rep}} \tau \times 365 \times 24)^{-1}
+ 3600 R_{\text{rep}} C_k^{1/3} \left[ \eta_f \epsilon_b R_{\text{rep}} MG \left( 1 - \frac{1}{G_i \eta_D \eta_f} \right) \right]^{-1}\right\}
\]
Here, $A$ is the driver system cost per MJ·Hz$^{1/2}$, $e_b$ is the ion beam energy in MJ, $B$ is the reactor system cost per (MW(e))$^{1/3}$, $C$ is the cost of the target per MJ$^{1/3}$, which includes the initial cost of the target fabrication factory, the fuel cost and the fabrication cost. Here, we estimate the cost of the driver system and the reactor system to increase with $e_l R_{rep}^{1/2}$ and $(\eta_s e_1 M G_{rep})^{1/3}$, respectively; $R_{rep}$ is the repetition rate of the energy driver, $f$ is the fraction of the capital cost in the electricity cost, and $R_{op}$ is the average operational rate in the plant lifetime $\tau$ (in years).

Figure 5 shows the relation between $T$ and $e_b$ for several cases listed in Table I. The gain is estimated by the gain curve [8, 9]. We choose $\eta_D = 0.3$ to 0.1.

**FIG. 5. Electric power cost as a function of driver energy for various parameters indicated in Table I.**

**TABLE I. PARAMETERS FOR IFE REACTOR**

<table>
<thead>
<tr>
<th>Case</th>
<th>$A$ (yen)</th>
<th>$B$ (yen)</th>
<th>$C$ (yen)</th>
<th>$R_{rep}$ (Hz)</th>
<th>$\eta_s$</th>
<th>$\eta_D$</th>
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<tr>
<td>1</td>
<td>$2 \times 10^{10}$</td>
<td>$3 \times 10^{10}$</td>
<td>50</td>
<td>10</td>
<td>0.4</td>
<td>0.3</td>
</tr>
<tr>
<td>2</td>
<td>$2 \times 10^{10}$</td>
<td>$3 \times 10^{10}$</td>
<td>50</td>
<td>5</td>
<td>0.4</td>
<td>0.3</td>
</tr>
<tr>
<td>3</td>
<td>$5 \times 10^{10}$</td>
<td>$3 \times 10^{10}$</td>
<td>100</td>
<td>5</td>
<td>0.3</td>
<td>0.2</td>
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<tr>
<td>4</td>
<td>$5 \times 10^{10}$</td>
<td>$5 \times 10^{10}$</td>
<td>100</td>
<td>5</td>
<td>0.3</td>
<td>0.2</td>
</tr>
<tr>
<td>5</td>
<td>$5 \times 10^{10}$</td>
<td>$5 \times 10^{10}$</td>
<td>100</td>
<td>5</td>
<td>0.3</td>
<td>0.1</td>
</tr>
</tbody>
</table>

$R_{rep} = 0.7; \ f = 0.3; \ t = 20 \ years; \ M = 1.2; \ \text{conservative gain curve to driver energy.}$
A and C are estimated from the update technology, and B is estimated from the cost of the nuclear power plant system.

The results show that we need high repetition rate, low cost, high efficiency and long life of the driver. A cheap target is also strongly required. The driver cost must be less than $2 \times 10^4 \text{ yen/J}$, which can be found from Fig. 5. The cost of Reiden-SHVS is $3 \times 10^4 \text{ yen/J}$. It will not be difficult to overcome the cost restriction by this driver, which is the most important item from the commercial point of view. An efficiency of 30% has already been achieved. Compared with other drivers, this system is very compact, which will be an additional advantage for the power plant.

The important issues will be repetition, lifetime, beam focusing and transport in the reactor chambers. More theoretical and experimental studies are required for the next step.

4. SUMMARY

Experiments have been performed on the induction adder accelerator Reiden-SHVS. Ion beam current oscillations due to the space charge instability are observed and studied with a view to their stabilization. Charge stripping is demonstrated to obtain a higher acceleration gradient. Virtual anode filtering is observed, which improves the emission significantly.

Preliminary studies of the requirements of an inertial fusion energy driver have been performed. SHVS will be able to meet the cost and efficiency requirements which are the most important issues for a commercial reactor. Beam transport and focusing as well as system repetition rate and lifetime are important topics to be studied.

REFERENCES

THEORETICAL STUDIES OF LIGHT ION DRIVERS FOR INERTIAL CONFINEMENT FUSION*

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Abstract

THEORETICAL STUDIES OF LIGHT ION DRIVERS FOR INERTIAL CONFINEMENT FUSION.

Three important issues in three critical elements of a light ion inertial confinement fusion (ICF) system are addressed: (1) scaling of the ion beam divergence caused by collective instabilities in applied B diodes; (2) effect of magnetically insulated electron flow on the operation of multigap inductive accelerators of the Hermes III type operated in positive polarity mode; and (3) the effect of self-fields on the focusing of intense ion beams with plasma/gas filled solenoidal lenses.

1. INTRODUCTION

At present, inertial confinement fusion (ICF) by light ions envisages applied B ion diodes powered by electrostatic accelerators, such as PBFA II or multigap inductive accelerators such as Hermes III [1]. A number of Li$^{3+}$ beams are brought to focus on a D–T pellet in a reactor chamber by magnetic lenses to deliver 100 TW. We have studied three important issues in three different critical elements of such a proposed ICF system.

2. ION BEAM DIVERGENCE RESULTING FROM COLLECTIVE INSTABILITIES IN APPLIED B DIODES

There is experimental evidence indicating that the bulk of ion beam divergence originates in the fluctuations of the electric field in an ion diode because of the spread in ion energy and momentum of the accelerated ions. Numerical simulations [2], employing the 3-D particle-in-cell code QUICKSILVER, have shown such fluctuations. In the early stages after the electrical power has been applied to the diode, these fluctuations are identified with the diocotron mode of the cathode electron sheath. As a result of the diocotron activity, the electron sheath extends over the entire region of the anode–cathode gap. Soon after this has occurred, the total number of electrons

* The present work was supported by Sandia National Laboratories under Contract No. 63-4881. The computations were performed at the Cornell National Supercomputer Center.
rise in the gap; there is a sharp drop in frequency of the dominant fluctuations accom­
panied by a rapid increase in the ion beam divergence. We identify an electrostatic
electron plasma oscillation driven unstable by ion streaming with the low frequency
mode observed in QUICKSILVER simulations [2]. Unlike previous 2-D calcula­
tions [3], we consider a 3-D mode with phase variations also along the insulating
magnetic field. The electrons dynamics is treated by the relativistic drift kinetic equa­
tion and the ion dynamics in the cold, unmagnetized, non-relativistic fluid limit. The
resulting equation governing the mode amplitude of the fluctuating electrostatic
potential \( \phi \) is

\[
\begin{aligned}
(d/dx - i\omega/u)^2 \{ \nabla^2 - [\omega_e^2/(\Omega^2 - k_y^2 v_d^2)](k^2_y - d^2/dx^2) + \lambda_d^2 W(\xi) \} \phi \\
+ (\omega_e^2/u^2) \phi = 0
\end{aligned}
\]  

(1)

In deriving Eq. (1), the insulating magnetic field \( \vec{B} = Bz \), the electron and ion densi­
ties and the \( \vec{E} \times \vec{B} \) electron drift \( \vec{v}_d = v_d \hat{y} \) are taken to be uniform; \( u \) is the ion
beam velocity; \( \omega_e \) and \( \omega_i \) are the electron and ion plasma frequencies, respectively;
\( \Omega \) is the electron gyrofrequency; \( \lambda_d = 7\pi/4\pi S_e^2 \) is the electron Debye length;
\( W(\xi) = \pi^{-1} \int_0^\infty du/u e^{-u^2/(u - \xi)} \) with \( \xi = [(\omega - k_y v_d)/k_y(2T/m)^{1/2}] \), \( T = mv_e^2 \), \( v_e \)
is the electron velocity spread along the magnetic lines of force and \( mc^2/\gamma \) is the total
mean electron energy. Equation (1) with suitable boundary conditions at the cathode
surface, \( x = d \), and anode surface, \( x = 0 \), is solved for \( T = 0 \) by perturbation tech­
niques [4], employing \( \omega_e^2 d^2/u^2 \) as an expansion parameter. To lowest order, \( \tilde{\phi}(x) \)
\( = \tilde{\phi}_0 \sin(n\pi x/d), n = \pm 1, \pm 2, \ldots \), and \( \omega_0 = k_y v_d + |k_z| d \omega_e/\pi. \) To next order,

\[
\omega_1 = |k_z| d \omega_e (\omega_e^2 d^2/u^2) g^n(\tilde{\omega}_0)
\]

with \( \omega_0 = \omega_d d/u \),

\[
\text{Im } g^n(\tilde{\omega}_0) = \frac{2\omega_0(\tilde{\omega}_0^2 + n^2 \pi^2)}{n\pi(\tilde{\omega}_0^2 - n^2 \pi^2)^2} \left[ (-1)^n \cos(\tilde{\omega}_0) - 1 \right]
\]

(2)

The dominant unstable mode has \( n = \pm 1, \omega_0 = \pm 2.105(d/u) \) and \( \text{Im } g^n(\tilde{\omega}_0) \)
\( = 0.0174. \) The predicted frequency matches the low frequency observed in the simu­
lations and so does \( k_y \), which is given by \( k_y v_d = \pm (2.105(u/d)) - d|k_z| \omega_e/\pi \) and
\( k_z = \pi/L_z \), where \( L_z \) is the anode length along the magnetic field.

Numerical solutions of Eq. (1) confirm the analytical results but in addition
reveal the stabilization of the mode with electron temperature \( T \) through Landau
damping. Figure 1 shows that the maximum growth rate decreases with the electron
thermal velocity, \( v_e = (T/m)^{1/2} \), and ultimately stabilizes at \( v_e/u = 11.6 \) for \( n = -1. \) Even if the electrons start out initially with \( T = 0 \), non-linear effects heat the
FIG. 1. Maximum growth rates of the strongest two modes as a function of electron thermal velocity, $v_e$ (solid line: $n = -1$, dashed line: $n = -2$). The growth rate is a maximum over $k_y$ with the parameters $(\omega_d/d/u)^2 = 3.4$, $m_i/m_e = 2571$, $v_d/u = 10.0$, $k_zd = 0.5$ held fixed.

electrons to stabilize the mode. Quasi-linear calculations of this mode yield a saturation amplitude given by

$$\phi_s^2 = \sum_k |\phi_k|^2 = 0.06n_0^2e^2d^4$$

where $n_0e$ is the electron charge density in the gap. Thus, the saturation amplitude is proportional to the column electron charge per cm$^2$ of the anode surface which fits in very well with QUICKSILVER results.

Given the saturation amplitude of the fluctuations, it is straightforward to obtain the ion beam divergence $\Delta \theta$, the energy spread $\Delta E$, and the momentum spreads which scale as:

$$\Delta \theta \propto (\phi_d/V)(d/L_x), \quad \Delta \theta_y \propto (\phi_d/V)(d/L_y)$$

$$\Delta E/E \approx (\omega_0d/u)(\phi_d/V)$$

$$\Delta p_y/p_x \approx (\phi_d/V)(d/L_y)$$

where $V$ is the gap voltage. For numerical values of the diode employed in Ref. [2] we obtain $\Delta \theta_x \sim 19$ mrad and $\Delta \theta_y \sim 43$ mrad, which are in approximate agreement with the simulations.
3. DYNAMICAL BEHAVIOUR OF MAGNETICALLY INSULATED ELECTRONS IN MULTIGAP INDUCTION ACCELERATORS OPERATED IN THE POSITIVE POLARITY MODE

A 2-D ($r,z$) electromagnetic particle-in-cell code, MASK$^1$, is employed to simulate the dynamics and collective behaviour of electrons emitted in three accelerating gaps of the inductive accelerator (6 MV, 300 kA) SABRE at Sandia National Laboratories. Figure 2(a) shows the emitted electrons in such a configuration where the inner surface is at positive polarity. The electron flow from the leading edge of each cathode in the accelerating gap is unsteady, producing an intermittent train of electron vortices of approximate size $\lambda = c/\omega_e$; $\omega_e$ is the local electron plasma frequency. Each vortex drifts with the local $\vec{E} \times \vec{B}$ flow of the self-magnetically insulated electrons. A quasi-static vortex equilibrium from the relativistic 2-D equations of motion is obtained for diamagnetic vortices which includes both self charge and current. A maximum radius of $2 \sqrt{2} \lambda$ is predicted which checks with numerical simulations with MASK$^1$. The predicted reduction of the self-magnetic field at vortex centre also agrees with numerical observations.

Our main attention is focused on predicting the behaviour of the multicomponent mean flow. In an $N$ gap accelerator there will be $N$ electron flows with different energies in the last segment. However, we make the ansatz that, because of vortex formation in the unsteady flow discussed above, $N - 1$ flows mix together. Thus, to study mean flow conditions only the flow from the last gap retains its identity; all the electrons from previous gaps are represented by a single flow. A model which makes a similar assumption has previously been developed [5] at Sandia. We have refined this model by accounting for the space charge of the launched flow.

The new model is used to construct an equivalent circuit for the simulated three gap accelerator. A set of non-linear algebraic equations are obtained that describe the interaction of the power supplies, the MITL sections and the load region. The circuit model of the load region (anode current, $I_a$, as a function of load impedance, $Z_{LOAD}$) is determined empirically from the simulations. The cathode currents of the first, second and third gaps, $I_{c1}$, $I_{c2}$, and $I_{c3}$, predicted by the circuit model (dashed lines), compare favourably with the cathode currents from simulation (data points) as shown in Fig. 2(b).

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$^1$ The MASK code was provided to us by Adam Drobot of SAIC McLean, Virginia, USA.
4. FOCUSING OF ION BEAMS

The focusing of 30 MeV, Li$^{+3}$ beams by a plasma filled solenoidal magnetic lens at an intensity of 5 kA/cm$^2$ is studied with a 2-D (r,z) hybrid PIC code in which the electrons are treated as a fluid and the beam ions as particles. Since the beam pulse times $\tau_b \ll R/V_A$, the background plasma ion motion is neglected; R is here the beam radius, and $V_A$ the Alfvén speed. Quasi-neutrality is assumed. The model electron magnetohydrodynamic (EMHD) equations evolve the poloidal flux function $\psi$ and the azimuthal magnetic field $B_\phi$ as a function of space and time. The electron density, $n_e$, and the plasma conductivity, $\sigma$, are system parameters. The beam acts as a source driving currents in the plasma which can distort the applied lens field and result in poor focusing.

Two distinct regimes of the beams have been identified. If the plasma is highly resistive then electron return currents decay and the self-field of the beams is exposed. Conversely, when the plasma is highly conductive, the beam acts as a diamagnetic body and drags the lens field with it. We have qualified the latter behaviour by establishing an upper bound on $\sigma$. For the magnetic field to diffuse rapidly enough through the beam, we require that the electron drag velocity be less than the diffusive velocity. For EMHD plasmas this is equivalent to $R_m < 1$, where $R_m$ is the magnetic Reynolds number. Assuming near current neutrality, then

$$v_e \approx n_b v_b / n_e < \frac{d}{dt} \left( \frac{c^2 t}{4 \pi \sigma} \right)^{1/2} \approx 1/2 \eta \tau_b^{-1/2}$$  (5)
FIG. 3. Schematic of solenoidal magnetic lens system. The applied flux strength $\psi$ across the beam annulus is $\sim 1.2 \text{ MG} \cdot \text{cm}^2$ ($1 \text{ G(auss)} = 10^{-4} \text{T}$).

FIG. 4. Normalized energy $E/E_0$ versus $\alpha/\alpha^*$.

FIG. 5. Power density time integrated for 3 ns as a function of radius at three different times for (a) $\alpha/\alpha^* = 0.1$ and (b) $\alpha/\alpha^* = 2.0$. 
Thus, we require

\[ \sigma < \sigma^* = \frac{1}{8} \pi \left( \frac{n_e}{n_b} \right)^2 \left( \frac{c}{v_b} \right)^2 \tau_0 \]

A schematic of a magnetic lens system is shown in Fig. 3. The lens is 30 cm long, and a 1 cm radius disk is located 1.5 m downstream at the focal plane. All beams have zero initial divergence and the above mentioned parameters. They are 4 cm wide annuli with an inner radius of \( r_1 = 8 \) cm and a square beam pulse length of \( \tau_0 = 10 \) ns. Since we expect the electron conductivity to be adequate for current neutralization outside the lens, we only evolve the field equations within the lens region. Outside this region we assume that the beam drifts ballistically to the target without any self-fields.

In Fig. 4, the time integrated total energy \( E \) deposited at the focus (normalized by the value \( E_0 \) without self-field interaction) is plotted for various values of \( \sigma/\sigma^* \). The maximum at \( \sigma/\sigma^* = 0.1 \) supports condition (6). At lower values of \( \sigma \), the electron diffusion becomes important. In Fig. 5 the power density at the focus as a function of radius is plotted for three different times. For \( \sigma/\sigma^* = 0.1 \) the focal spot size is better maintained throughout the pulse duration, and the peak power is nearly two times greater near the end of the pulse than in the \( \sigma/\sigma^* = 2.0 \) case.

In order to quench the filamentation instability it is necessary that \( \sigma \) exceed a threshold which is \( \sim 10^{14} \) s\(^{-1} \) for the above quoted beam parameters. The criterion established by relation (6) suggests that \( n_e/n_b > 10^3 \) in order to limit the effects of electron advection on good focusing. At such high conductivities, resistive diffusion will not be as important. If the lens plasma is created by beam induced ionization of a gas, such electron densities will not be obtained until after a few nanoseconds, by which time significant distortions will have occurred. This implies the need to ionize the gas in the lens before beam propagation.

REFERENCES

ILSE: THE NEXT STEP TOWARDS A HEAVY ION INDUCTION ACCELERATOR FOR INERTIAL FUSION ENERGY*

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Abstract

ILSE: THE NEXT STEP TOWARDS A HEAVY ION INDUCTION ACCELERATOR FOR INERTIAL FUSION ENERGY.

LBL and LLNL propose to build, at Lawrence Berkeley Laboratory, the Induction Linac Systems Experiments (ILSE), the next logical step towards the eventual goal of a heavy ion induction accelerator powerful enough to implode or 'drive' inertial confinement fusion targets. ILSE, although much smaller than a driver, will be the first experiment at full driver scale in several important parameters. Most notable among these are line charge density and beam cross-section. Many other accelerator components and beam manipulations needed for an inertial fusion energy (IFE) driver will be tested. The ILSE accelerator and research programme will permit experimental study of those beam manipulations that are required of an induction linac inertial fusion driver which have not been tested sufficiently in previous experiments, and will provide a step towards driver technology.

1. INTRODUCTION

Research programmes throughout the world using high power lasers and light ion accelerators, along with the Halite/Centurion programme conducted at the US Nevada Test Site and using nuclear explosives, have already put to rest fundamental questions about the basic feasibility of achieving high gain with inertial fusion

targets. However, the efficiency and cost goals that mark the difference between laboratory science and economically competitive power plants must also be achieved. It now appears possible to develop and build a demonstration inertial fusion energy (IFE) power plant for energy production by the year 2025. Plans for this development are outlined in the newly formulated US National Energy Strategy. A multigap heavy ion accelerator has unique advantages as an IFE driver because it is simultaneously high in repetition rate, electrical efficiency, reliability and lifetime. Thus it has become the leading driver candidate for commercial IFE, as recommended by the Fusion Policy Advisory Panel [1] and described in the National Energy Strategy [2].

The fundamental requirements for an HIF driver are to produce a short beam pulse (about 10 ns) with an instantaneous power of more than $10^{14}$ W and enough quality to be focused to a spot with a radius of about 3 mm. To minimize the space charge forces and other collective effects associated with high currents, it is desirable to achieve the required beam power by maximizing the ion kinetic energy (voltage) while minimizing the beam current. An upper limit on the ion kinetic energy is set by the desired target penetration. For the target to achieve high energy gain, the ion range must be limited to $\sim 0.1 \, \text{g/cm}^2$. For light ions such as protons, this requirement limits the kinetic energy to about 10 MeV, and a total beam current of 40 MA would be required to produce $4 \times 10^{14}$ W on target. For heavy ions, the limit determined by the ion range is about 10 GeV, leading to a current requirement of about 40 kA. Consequently, the space charge forces and collective effects are much smaller and the beams more manageable.

A schematic diagram of a generic induction accelerator designed to produce 40 kA of 10 GeV ions is shown in Fig. 1. To achieve this current, 40 beams, initially

![FIG. 1. Schematic of conceptual accelerator/driver delivering $4 \times 10^{14}$ W of heavy ions for 10 ns. The total length of the facility is expected to be 5–10 km.](image-url)
at a current of \(-0.5\) A apiece, are electrostatically focused and accelerated to an energy where magnetic focusing becomes preferable (\(-100\) MeV). The 40 beams are combined transversely to ten beams and accelerated to 10 GeV. During acceleration the total current within the accelerator is amplified from 40 to 4000 A by increasing the beam velocity (by a factor of \(-30\)) and decreasing the bunch length (by a factor of \(-4\)). Downstream the accelerator, power increases by an additional factor of ten, reaching \(4 \times 10^{14}\) W, because the ‘velocity tilt’ imparted in the accelerator causes additional compression in the drift distance from accelerator to target.

Since 1984, the US Heavy Ion Fusion Accelerator Research (HIFAR) Programme at LBL and LLNL has been concentrating on the multiple beam induction accelerator. (A parallel programme in Europe is exploring RF linacs and storage rings.) A series of increasingly sophisticated experiments has explored, in scaled parameters, the accelerator physics of the induction approach, developed the relevant accelerator technology and estimated the capital costs and potential economics of induction drivers.

Early HIFAR experiments investigated fundamental aspects of ion induction accelerators. One of the first experiments demonstrated that ion sources of adequate intensity and beam quality could be fabricated [3]. A single beam transport experiment showed that it was easy to transport intense ion beams at the brightness needed [4]. The recently completed four beam accelerator experiment, MBE-4, demonstrated that it is possible to amplify the current of ion beams during acceleration — an important driver feature — without appreciably degrading them [5]. In order to study most of the other beam manipulations required of a driver, we have proposed an accelerator and a sequence of experiments collectively called the Induction Linac System Experiments (ILSE).

2. ILSE PHYSICS DESIGN

A principal design criterion is that the beams must be of the same line charge density as is expected in a full scale driver. As a consequence, the size of the beams and the strengths of the focusing fields in the accelerator will be directly relevant to the low energy end of a driver. Thus, the ILSE accelerator and experimental programme will allow realistic experimental investigation of many key issues, providing the base of knowledge needed for the next step.

Figure 2 presents a block diagram of the physics design of the ILSE accelerator and a possible arrangement of some of the experiments. Four beams from a 2 MV injector are matched to an electrostatic transport system and accelerated to 5.0 MV. The four beams are then combined into one and matched into a magnetically focused linac for further acceleration to 10 MV. A relatively light ion, potassium, is used to permit magnetically focused beam transport at energies approximately 20 times lower than those in a driver and therefore at a fraction of the cost. The electrostatic
and magnetic focus accelerator sections of ILSE each contain 32 accelerating cells, which are physically grouped into blocks of eight. Each cell block is separated from the next by a full lattice period, which provides diagnostic access to the beams. Along the machine, as the focusing system becomes more efficient with increasing ion velocity, the length of the focusing lattice period increases from 66 cm in the first three blocks to 82 cm in the fourth block, and then to 1 m from the beginning of the combining section to the end of the accelerator.

Amplification of current and control of longitudinal bunch length through the accelerator are essential components of our studies of induction linacs for HIF. These aspects require the use of carefully shaped accelerating voltage waveforms. One method of finding waveforms for accelerating the beams, ignoring longitudinal space charge effects, has been described by Kim and Smith [6]. In this self-replicating scheme, the profile of current versus time at a fixed location is preserved throughout the accelerator, and the magnitude increases as the bunch shortens in time. We have adopted this scheme for the ILSE physics design. Solutions for the current and the accelerating waveforms at every accelerating gap can be constructed easily.

3. ISSUES TO BE ADDRESSED EXPERIMENTALLY WITH ILSE

By building and commissioning ILSE itself, and by subsequently performing an experimental programme, we will examine most driver issues either directly or in scaled form. While ILSE will initially use ions in the range of neon to potassium,
### TABLE I. PAST LBL EXPERIMENTS COMPARED TO ILSE AND A DRIVER

<table>
<thead>
<tr>
<th>Line charge density</th>
<th>Initial current per beam</th>
<th>Final kinetic energy</th>
<th>Initial $T_p$</th>
<th>Final $T_p$</th>
</tr>
</thead>
<tbody>
<tr>
<td>CS ion source</td>
<td>0.5–1 $\mu$C/m</td>
<td>1 A</td>
<td>2 MeV</td>
<td>~0.1 eV</td>
</tr>
<tr>
<td>SBTE</td>
<td>0.03 $\mu$C/m</td>
<td>20 mA</td>
<td>200 keV</td>
<td>~0.1 eV</td>
</tr>
<tr>
<td>MBE-4</td>
<td>0.01 $\mu$C/m</td>
<td>~5–10 mA</td>
<td>1 MeV</td>
<td>~0.1 eV</td>
</tr>
<tr>
<td>ILSE</td>
<td>0.25 $\mu$C/m</td>
<td>~1 A</td>
<td>10 MeV</td>
<td>~0.1 eV</td>
</tr>
<tr>
<td>Driver</td>
<td>~0.25 $\mu$C/m</td>
<td>~0.5 A</td>
<td>~10 GeV</td>
<td>~0.1 eV</td>
</tr>
</tbody>
</table>

$^a$ $T_p$ is the beam temperature perpendicular to the direction of beam propagation.

$^b$ Not applicable.

The results will be scaleable to ions with different charge-to-mass ratios, such as the mass 100–200 ions typical of a driver. We list here ten experiments and system wide issues in the ILSE programme and comment on the comparable features as incorporated in a driver.

#### 3.1. Experiments within the linear accelerator

1. **Performance of the four beam, 2 MV injector.** Close in size to the injector needed for a driver, it will generate beams at full driver line charge density.
2. **Acceleration of four beams with electrostatic focusing.** This models the first 400 m of a driver and was investigated in MBE-4, but with smaller beams.
3. **Transverse beam combining or merging of four beams into one.** Studies show that this beam manipulation, although not essential, permits significant cost savings at driver scale.
4. **Acceleration of intense beams with magnetic focusing.** Acceleration of multiple beams with magnetic transport will be used in more than 90% of a driver. To date, acceleration of space charge dominated ion beams using magnetic quadrupole transport has not been studied experimentally.
5. **Pulse shaping and longitudinal control.** To achieve high gain, a fusion target requires a properly shaped pulse. ILSE will allow us to test various acceleration 'schedules' to control pulse shape.
6. **Alignment and steering.** ILSE will allow us to examine the practical accelerator trade-offs between steering and alignment. Steering with time dependent voltages will be investigated just after the electrostatic focus accelerator, whereas steering with time independent focusing will be studied in the magnetic focus accelerator.
3.2. Experiments downstream from the accelerator

(7) Magnetic bending of an intense ion beam. With most high gain target designs, intense final beams that are dominated by space charge must be transported in bending magnets and turned through angles as large as 270°.

(8) Recirculating experiments. The ILSE linac can be used as an injector for a recirculating induction accelerator, a scheme that, in conceptual studies, has shown promise for reducing the cost of heavy ion driver. However, this type of accelerator is not as well understood as the induction linac. A more thorough examination of the physics of the recirculator is in progress. If successful, this device could increase the total beam energy of ILSE by a factor of ten.

(9) Drift compression current amplification. This manipulation amplifies the beam power just before the target. This is a fundamental and vital test of new beam physics in which the beam energy tilt is removed, rather precisely, by longitudinal collective accelerating forces at the beam head and decelerating forces at the tail. In other words, the velocity tilt causes beam compression over the drift distance, as previously mentioned, but eventually the space charge forces remove the velocity tilt. The challenge is to arrive at the final focus with a highly compressed beam in which space charge forces have reduced the velocity tilt to zero (thus avoiding chromatic problems in final focus) but have not yet caused beam blow-up. The ILSE beam will have enough line charge density to remove an energy tilt of more than 10%, compared to an expected tilt of about 5% in a driver.

(10) Focusing, with or without neutralization, onto a small target spot. Although there is extensive theory, relatively little is known experimentally about the behaviour of space charge dominated beams in the target chamber. The higher perveance of the final ILSE beam will allow us to experiment under conditions more demanding than those in a driver.

4. PROSPECTS FOR ILSE

After receiving US Department of Energy approval to begin construction, we would expect to build and commission ILSE within five years, at a total estimated cost of construction of roughly $60 million (1992 US dollars).

In summary, ILSE represents a step beyond existing experimental facilities both in quantitative parameters and in the variety of the work that can be performed. It will have a highly flexible experimental capability with which we can address most of the remaining induction accelerator issues for inertial fusion.

REFERENCES

DISCUSSION

B. COPPI: Could you give an estimate of the cost of the 'final' machine you propose (4 kA/beam, 10 GeV) for an ignition experiment?

R. BANGERTER: The cost estimates for a driver generally fall in the range US $500-2000 million. The cost of a full scale driver would almost certainly be less than the cost of ITER. The 4 kA/beam, 10 GeV accelerator is not simply for an ignition experiment. It would produce enough target gain — gain of about 100 — for commercial energy production.

K. IMASAKI: Could it be that a beam breakup instability, or something similar, is induced? It would seem that a large number of cavities is required to accelerate the ions to 10 GeV with long pulse and high current.

R. BANGERTER: The beam breakup instability can be important in electron induction linacs. We do not believe that it is important for heavy ion fusion. The most important instability for heavy ions appears to be a longitudinal bunching instability, but the growth rate is slow enough for us to believe that this instability can be controlled by feedback. Acceleration to 10 GeV does require many cavities. As each cavity supplies only a small fraction of the energy, the components are not highly stressed. The low stress leads to long life and good reliability.

D.H. CRANDALL: You emphasized that the theory for high ion beam performance is strong. Is it really necessary to do the ILSE experiments?

R. BANGERTER: Although agreement between experiment and theory has been very good, it is always necessary to check the theory. No theory is perfect.
HIGH CONVERGENCE UNIFORM IMPLOSIONS BY CANNONBALL TARGETS WITH GEKKO XII BLUE LASER

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Abstract

HIGH CONVERGENCE UNIFORM IMPLOSIONS BY CANNONBALL TARGETS WITH GEKKO XII BLUE LASER.

Systematic studies on the radiation hydrodynamics of an X ray confining cavity and a fuel capsule have led to remarkable progress in the last few years. This makes it possible to analyse quantitatively the energy transfer processes from the laser to the fusion capsule and to find a uniform condition for the irradiation of the fusion capsule by X rays. As a result, reproducible and stable implosions of the fuel capsule have been achieved. — A series of experiments has been carried out by using intense blue laser light (351 nm wavelength and 5 kJ energy) from the GEKKO XII Nd:glass laser system. Quantitative understanding of X ray confinement and radiation redistribution inside a cavity, the propagation of a radiative heating wave in medium-Z material and the re-emission characteristics of matters heated by X rays has been obtained. A fuel capsule placed in a cylindrical gold cavity has been imploded by irradiation with X rays. Good agreement between experiment and numerical simulations has been obtained for the implosion parameters such as the fuel $\rho R$ (product of density and radius) has been obtained up to a radial convergence ratio of $\sim 15$, on the assumption of perfect spherical symmetry.

1. INTRODUCTION

In radiation driven cannonball targets, a fuel capsule is placed inside a cavity made of high-Z material and is heated by incoherent thermal X rays in the cavity generated with intense laser light. Owing to the geometrical smoothing effect, shot scale irradiation non-uniformities on the fuel capsule associated with small structures involved in the laser beam profile are considerably reduced [1]. In addition to this advantage, the mass ablation rate obtained by X ray irradiation higher than that
reached by direct laser irradiation can reduce the growth of hydrodynamic instabilities due to ablative stabilization [2]. The main disadvantage is inefficient energy transfer from the laser to the fusion capsule. This fact can, however, be compensated for, to a considerable extent, by radiation confinement in a cavity [3, 4]. On the assumption of quasi-stationary conditions for the flux balance at the surface of the cavity wall and the fuel capsule, the efficiency of power transfer, which is defined as the net heat power transferred to the inner capsule divided by the radiation power of the source, is approximately given by [5, 6]

\[
\eta_{\text{trans}} = m^{-1} \frac{1 - r_t}{N^{-1} + n^{-1} + m^{-1} - m^{-1} r_t}
\]

where \( n^{-1} \) is the fractional hole area on the cavity wall, \( m^{-1} \) the fractional capsule area with respect to the total area (including holes) of the cavity, \( r_t \) the re-emission coefficient of the capsule, and \( N \) the quality factor of confinement (equal to \( r_c/(1 - r_c) \), where \( r_c \) is the re-emission coefficient of the cavity wall). This relation states that the transfer efficiency is given by the geometrical coupling factor \( m^{-1} \) multiplied by the X-ray enhancement factor. Thus it is important for a quantitative evaluation of the X-ray re-emission coefficients and the spectra of various materials heated by X-ray radiation.

2. X RAY REDISTRIBUTION, RE-EMISSION AND CONFINEMENT

A three-dimensional model for the calculation of the X-ray intensity distribution on a radiation driven target has been developed [7]. The model includes energy transfer processes such as conversion of laser light to X-rays, radiation re-emission from the X-ray heated wall of a cavity and the influence of a fuel capsule on radiation redistribution. To confirm the validity of the model, the intensity distribution of the X-rays inside a cylindrical cavity heated by laser light was investigated by measuring a burnthrough signal from a diagnostic foil integrated into the cavity (see Figs 1(a) and (b)). As is shown in Fig. 1(c), the experimental result is well recovered by the model calculation when the X-ray re-emission from the X-ray heated wall is taken into account. The model calculation shows that, under optimum conditions for the GEKKO XII facility, we can expect an irradiation non-uniformity of 2% (mainly due to the mode number of 4), an energy transfer efficiency of 11% and an X-ray intensity on the capsule of \( 1 \times 10^{14} \text{ W/cm}^2 \), including the enhancement of the average intensity by a factor of 1.5 due to radiation confinement in a cavity.

The properties of X-ray transmission and self-emission were investigated in detail for various materials heated by thermal X-ray. The dependence of the re-emission coefficient on the atomic number ranging from 6 to 79 has been measured at a maximum irradiance of \( 0.8 \times 10^{13} \text{ W/cm}^2 \) by using the simple experimental layout shown in Fig. 2(a). An X-ray source was generated with a gold plate facing another sample plate, and the time evolution of the re-emission spectrum from the
FIG. 1. Experimental set-up for measuring radiation redistribution inside a cylindrical cavity and experimental results: (a) Burnthrough signal from a diagnostic foil (0.47 μm gold) heated by radiation confined in the cavity, detected by an X ray streak camera and a transmission grating spectrometer. (b) Output image from an X ray streak camera. Intense burnthrough signals at both ends of the foil appear at an early time followed by a weaker burnthrough signal at the centre. The earlier signals originate from the directly laser heated region and the later signals stem from the indirectly thermal radiation heated region. (c) Comparison of model calculation with experimental result indicated by closed circles. Thick and thin solid lines show the calculation results including and excluding the effect of re-emission on radiation redistribution, respectively.
FIG. 2. Experimental layout for measuring the X ray re-emission from X ray heated materials and the experimental results: (a) X ray source generated by one arm of the GEKKO XII laser (500 J energy, 0.7 ns duration) irradiating a gold plate. The time evolution of the re-emission spectrum from the sample is measured absolutely with a transmission grating spectrograph coupled with an X ray streak camera. Signals arising from expanding gold plasma generated by laser irradiation were spatially and temporally eliminated. (b) Measured X ray re-emission coefficients of various materials versus atomic number. These values are obtained at an X ray irradiance of $0.8 \times 10^{13} \text{ W/cm}^2$ on the sample, roughly corresponding to 0.7 ns after the onset of X ray irradiation.

sample material was measured absolutely. The measured re-emission coefficients at maximum intensity of the source X rays are given as a function of the atomic number of the sample material in Fig. 2(b). The experimental results are under continuing investigation by a numerical calculation with ILESTA [8, 9], in which an improved version of the screened hydrogenic, average ion model is used.

3. CANNONBALL IMPLSION WITH GEKKO XII BLUE LASER

After testing several types of the cannonball targets with different configurations and parameters, we are making a systematic study of a cylindrical cavity target. The experimental conditions and some important results are described in Ref. [10]. The convergence ratio of $R_i/R_f$ (where $R_i$ and $R_f$ are the initial and final radii) was varied by changing the filling pressure of the D–T gas from 1 to 20 atm. The implosion time measured with an X ray streak camera and the ion temperatures determined from the velocity spread of the neutrons were weakly dependent on the initial filling pressure. The fuel $\rho R$ estimated from deuteron knock-on measurements increases monotonically from 0.1 to 2 mg/cm$^2$ with increasing fuel pressure. The size of the
FIG. 3. Dependence of fuel $\rho R$ (product of density and radius) on the initial D-T fill pressure which is varied systematically in order to change the radial convergence ratio of the compressed fuel. The $\rho R$ values are evaluated from deuteron knock-on measurements (denoted by □) and from the imploded core size (symbol ○). Good agreement between the two values is seen at an initial pressure exceeding 2 atm. For comparison, we show the calculated $\rho R$ values at two different phases defined by the behaviour of the shock waves propagating through the compressed fuel. The time at which the shock wave reflected at the target centre first encounters the imploding pusher is represented by '1st', and the time at which the shock wave hits the pusher again is represented by '2nd'.

Compressed fuel was evaluated from the X ray images of the D-T filled glass microballoon capsules taken with an X ray framing camera and time integrating pinhole cameras. The convergence ratios increased from 4.5 to 20 when the initial pressure was reduced from 20 to 1 atm. The fuel $\rho R$ derived from this convergence ratio, on the assumption of perfect spherical convergence, agreed with the values derived from the knock-on measurement at an initial pressure exceeding 2 atm (corresponding to a convergence ratio of less than 15 as shown in Fig. 3). The discrepancy found at the lower pressures may be ascribed to the difference in the observation times of the two measurements; as deduced from a computer simulation, the X rays effective for core imaging are emitted from the pusher region shortly (a few tens of picoseconds) after the moment of maximum neutron production which is closely related to the knock-on result. At the later time the pusher is still converging. The other possibility is disruption (i.e. pusher and fuel mixing) of the compressed core throughout the deceleration phase.

As is shown in Fig. 3, numerical simulations with the one-dimensional fluid code ILESTA reproduce the experimental $\rho R$ value obtained from knock-on measurement. The measured fuel temperature is also consistent with the simulations for initial pressures above 2 atm. The ratio of the experimental neutron yield to the simulation value is significantly below unity (0.3–0.03) for all pressures. The simulation
shows that, owing to the temperature variation throughout the fuel, the fusion reaction mainly occurs in a compressed fuel layer adjacent to the pusher. Then, the disagreement in the neutron yield may suggest the occurrence of pusher–fuel mixing at the contact surface. Further details of the implosion stability and its influence on the implosion parameters are under investigation.

On the basis of these results, target designs aiming at ignition and high gains in the fusion reactions are being undertaken by using the code ILESTA. As a preliminary result, a pellet gain of 30 to 40 has been achieved for 1 MJ driver energy, on the assumption of an energy transfer efficiency of 30% from the driver to the fuel capsule and adopting a target with very low initial aspect ratio target \((R/\Delta R \approx 2.5)\) to suppress the growth of fluid instabilities and radiation preheat.

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DISCUSSION

J.M. PERLADO: Could you explain where you applied the 3-D formulation you presented to calculate the transfer efficiency?

H. NISHIMURA: First of all, the cavity and the fuel capsule are divided into wall elements. Assuming flux balance at each wall element and using the source X ray flux converted from the laser, the radiation flux at each wall element can be solved in a self-consistent manner. To obtain the re-emitted flux from the heated cavity, we use the self-similar solution obtained by Pakula and Sigel for the ablative heat wave driven by thermal radiation. In this way, we obtain the time evolutions of the radiation fluxes at the fuel capsule, and hence the transfer efficiency is calculated. The intensity obtained, which is directly related to the transfer efficiency, shows good agreement with the values given by the experimentally observed implosion velocity.
D.L. COOK: What is the best time dependent X ray drive uniformity you have achieved in these experiments?

H. NISHIMURA: From analysis of the X ray framing shadow image we assess an intensity non-uniformity of several per cent at such low mode numbers as two or four.

S.E. BODNER: What non-uniformity will be acceptable for an ignition or high gain pellet, and do you think it can be achieved?

H. NISHIMURA: To attain ignition and high gain burn with realistic laser energy, we will have to achieve a non-uniformity level of less than 1% at each non-uniformity mode. This is not easy but can be done if we obtain high re-emission at the cavity wall on the cannonball scheme. To this end research is being undertaken into high re-emission materials and a more closed geometry cavity design.
RADIATIVE ENERGY TRANSFER IN A CYLINDRICAL CAVITY

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Abstract

RADIATIVE ENERGY TRANSFER IN A CYLINDRICAL CAVITY.

Results from experiments are presented that demonstrate the process leading to isotropization of the radiation field in a cavity heated by thermal X rays. Cylindrical cavities in gold, 500 µm in diameter, were heated from one side by thermal radiation produced by a laser plasma. The temperature profile along the cylinder was obtained by measuring the radiation emanating from a small slit parallel to the axis. The results are in good agreement with a theoretical model that predicts the axial temperature profile due to multiple absorption and re-emission of thermal radiation by the cavity wall.

The symmetrization of the implosion of the pellet containing the fuel is a central issue in the so-called indirect drive inertial confinement fusion (ICF). It is achieved by positioning the pellet inside a high-Z cavity ('hohlraum'), which is heated by laser or ion beams. The radiative energy transfer occurring inside the cavity results in the isotropization of the radiation field which is confined in it. As a consequence, the temperature variation on the cavity wall is reduced with time and the pellet is irradiated by incoherent and nearly isotropic radiation. In this work we have experimentally measured the spatial temperature variation in a cylindrical cavity heated by X rays from a laser produced plasma and compared the results with theoretical predictions. The cylindrical geometry has proven to be most relevant to ICF implosion experiments [1, 2].

The experimental scheme is shown in the upper part of Fig. 1. Pulsed laser light is injected into a gold cylinder with a diameter of 500 µm. The front end or converter section of the cylinder, where the laser heats the wall, is separated from the main part of the cylindrical cavity by a thin (~1000 Å) gold foil. The gold foil prevents laser light from entering the cylinder but at the same time serves as a source of soft X rays for the cylinder. This is because the thermal radiation generated in the converter section rapidly heats and diffuses through the low mass foil. Laser heating was accomplished by using 200 J/300 ps laser pulses with a wavelength of 0.44 µm generated by the ASTERIX laser.

The heating of the tube was investigated with spatial and temporal resolution through a slit in the main section of the tube with the help of an X ray streak camera (see Fig. 1) and with spatial and spectral resolution using a transmission grating spectrometer. As can be seen in Fig. 1, two phases of the X ray emission from the cavity
FIG. 1. Cylindrical gold capillary heated by X rays generated in the laser heated converter section and passing through a thin gold foil (upper part). The lower part shows the intensity of the X-ray emission through the observation slit as a function of time.
may be distinguished. The first phase, called the re-emission phase, is determined by
the heating from the source and by energy exchange through re-emission among the
wall elements of the tube. During this phase the tube is still free of plasma. It is fol-
lowed by the collision phase [3], where the hot material from the wall implodes onto
the axis of the tube and causes a second burst of X ray emission. The time delay
between the two phases increases with the diameter of the tube. The information con-
tained in the experimental data set delivered by the two diagnostics has been unfolded
to obtain the time integrated spectrum and the corresponding radiation temperature
from each axial location of the slit. Comparison with the theoretical model requires
the determination of the radiation temperature during the re-emission phase. This is
accomplished with the help of the X ray streak camera data (Fig. 1) which give the
fraction of radiated energy during the re-emission phase as well as during the colli-
sion phase for each axial location. Using this information, a time averaged radiation
temperature profile for the re-emission phase is deduced. Details will be given in
Ref. [4]. The results of the analysis of several shots for selected axial locations are
depicted in Fig. 2.

We have calculated the temperature distribution at the end of the re-emission
phase by using the model described in detail in Ref. [5]. This model is based on the
scaling laws for the optically thick ablative heat wave [6] which is expected to form
in the cylinder wall and locally determines its temperature and re-emissivity (from
the Stefan-Boltzmann law). The heat exchange between the wall elements of the
cylinder is then calculated numerically in a self-consistent manner. We note that this
method can be applied to any cavity with arbitrary three dimensional shape. It gives
the evolution of the temperature variation on each wall element of a cavity and calcu-
lates the degree of isotropization achieved after a given period of heating and for a
specific geometry. In the calculations we used previously obtained opacities [7].

FIG. 2. Comparison of calculated (solid line) and measured (points) temperature distribution along the
tube. The dashed line indicates the theoretical prediction when no re-emission is taking place.
Taking into account the wall re-emissivity, we have calculated the temperature distribution along the cylinder at the time of maximum emission. As is evident from Fig. 2, there exists good agreement between the calculated (solid line) and the measured (points) temperature distribution. Furthermore, it is seen that by suppressing the re-emission of the wall [5], the theoretical model gives a temperature distribution (dashed line) which is considerably lower than the measured distribution. This result confirms our conception of radiative energy transfer which, using the same opacity and material input data, had previously led to satisfactory modelling of the average temperature in spherical cavities [8]. However, more importantly, the present results show that it is possible to predict quantitatively the temperature evolution in space and time in cavities for ICF purposes and to measure it.

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RADIATIVE ABLATION OF LOW-Z MATERIAL

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Abstract

RADIATIVE ABLATION OF LOW-Z MATERIAL.

A 0.5 \( \mu \text{m} \) thick beryllium foil is heated by thermal X rays from a laser irradiated converter foil up to temperatures of a few 10 eV. During the transformation of the solid beryllium into an ionized gas time resolved absorption spectra have been measured in the region 100 eV \( \leq \hbar \nu \leq 250 \text{ eV} \). The measurements agree well with calculations of radiation transport for which electron impact line broadening was found to be essential.

The energy delivered by a high power laser can with high efficiency be converted to intense thermal X rays [1]. This makes it now possible to generate hot and dense matter by radiative heating and to study radiation transport in a temperature and density regime hitherto inaccessible in a laboratory. This subject is of great interest in indirect drive inertial confinement fusion where the pellet is driven by soft (sub-keV) X ray heating of a low-Z ablator [2].

In this paper for the first time we report a detailed experimental study of the penetration of laser generated soft X rays into low-Z material (in the following to be called heating wave; see Ref. [3]) using time resolved X ray spectroscopy. The results were obtained with beryllium, i.e. with low-Z material of simple atomic structure. This choice allows a detailed comparison with numerical simulations.

In order to study the heating wave in low-Z material we used the experimental set-up shown in Fig. 1. A thin gold foil (~3000 \( \text{Å} \) gold on ~1 \( \mu \text{m} \) polypropylene (\( \text{CH}_2 \)) substrate) is heated by a laser pulse (~15 J, 3 ns, \( \lambda = 0.53 \ \mu \text{m} \), spot diameter 300 \( \mu \text{m} \), intensity \( 10^{13} \ W/\text{cm}^2 \)). The foil serves as a converter of laser light into thermal X rays [4]. The intensity of the X ray flux on the sample can be varied through the distance \( d \) between converter and sample in the range \( \leq 10^{12} \ W/\text{cm}^2 \). The radiation transmitted through the sample was analysed with temporal (50 ps) and spectral (0.5 eV) resolution by an X ray streak camera coupled to a flat field grazing incidence spectrometer. More details of the experiment are given in Ref. [5].

Figures 2 and 3 show results obtained with Be as sample material. The spectral transmission is obtained by comparison with the source spectrum recorded without Be sample (Fig. 2(a)). The source spectrum has no pronounced spectral features; the temporal modulation is due to a modulation of the laser pulse. If the Be sample is placed at a large distance (\( d = 350 \ \text{mm} \), Fig. 2(b)) it remains cold and the absorption
spectrum shows a sharp edge at \( \approx 111 \) eV, the K-edge of the cold Be metal. If the Be sample is brought close to the source (\( d = 130 \) \( \mu \)m, Fig. 2(c)) it is transformed into a hot, dense gas of multiply ionized Be atoms. The transmission recorded at \( t = 2.6 \) ns is shown separately in Fig. 3. The dominant ion species is \( \text{Be}^{2+} \), and its absorption lines \( 1s^2-1s2p \) and \( 1s^2-1s3p \) are clearly seen. The more energetic lines of this series are not observed. This is attributed to continuum lowering due to the high density of the material (although it expands during the heating), which shifts the K-edge of the isolated \( \text{Be}^{2+} \) ion at 154 eV down by about \( \sim 10 \) eV. In fact, this edge and its slow motion to higher photon energies as the Be plasma expands with time are clearly visible in Fig. 2. In addition to \( \text{Be}^{2+} \), in Fig. 3 also absorption lines of \( \text{Be}^{3+} \) and \( \text{Be}^{1+} \) are observed.

FIG. 1. Experimental scheme for the study of heating wave formation in X-ray heated low-Z material.

FIG. 2. Time resolved spectrum obtained with (a) the source alone, (b) with a cold and (c) an X-ray heated 0.5 \( \mu \)m thick beryllium sample.
The spectral transmission at $t = 2.6$ ns through the X-ray heated Be foil. Thick line: measured transmission; thin line: computer modelling.

The hydrodynamic motion of the Be absorber has been simulated with the MULTI code, a one-dimensional hydrocode with multigroup radiation transport [6] (80 spectral groups were taken here). For the opacity we used local thermal equilibrium (LTE) values calculated with the SNOP code [7]. Modelling of line broadening is of particular importance because radiative transport preferentially takes place between the absorption lines in a region where the line wings overlap. We used electron impact broadening with a Lorentz profile as the dominant mechanism (for simplicity we set the Gaunt factor equal to 1) [8]. As input for the heating X rays the measured temporal and spectral dependences were employed. The calculations are performed in plane geometry.

The simulations typically show a heating wave penetrating into the foil. It is generated because a large fraction of the incident X rays in the spectral region above the K-edge is initially absorbed in the front layers of the foil. Subsequently, the X rays penetrate more deeply into the foil since the opacity decreases with heating. For the conditions of Fig. 1(c), the heating wave reaches the rear side of the foil after about 2 ns. At $t = 2.6$ ns, the time for which we plotted the spectrum in Fig. 3, the values for the average temperature and density in the foil are $\sim 25$ eV and $\sim 5 \times 10^{-3}$ g/cm$^3$, respectively.

The absorption spectra were calculated by a post-processor which solves the radiation transport equation for the density and temperature profiles calculated by MULTI. This procedure has been employed in order to achieve the same spectral solution as in the experiment. The comparison of the simulations with the observed spectra (Fig. 3) shows quite satisfactory agreement. We note that the simulations depend sensitively on the line broadening. The good agreement achieved confirms the fact that the employed electron impact broadening is the dominant mechanism for the conditions here.
Summarizing, we may state that we have presented an appropriate method of studying the ablation of a low-Z wall by X rays, which appears of general interest for inertial confinement fusion.

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DESIGN OF THE NOVA UPGRADE LASER SYSTEM FOR IGNITION FUSION EXPERIMENTS

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Abstract
DESIGN OF THE NOVA UPGRADE LASER SYSTEM FOR IGNITION FUSION EXPERIMENTS. A 1–2 MJ, 500 TW 0.35 μm solid state laser has been proposed to demonstrate the scientific feasibility of inertial confinement fusion (ICF) at laboratory scale for national security applications and to develop ICF technology and science for long term commercial electric power generation and for basic scientific applications. In the paper the design of this facility, designated by LLNL as the Nova Upgrade, is described. This facility consists of a frequency converted neodymium doped glass laser utilizing advanced optical components, a cost effective multipass architecture and a target area capable of safely containing thermonuclear fusion yields of up to 45 MJ.

1. INTRODUCTION

The primary scientific requirement for realizing the full military and civilian applications of inertial confinement fusion (ICF) is the achievement of a target gain (the ratio of fusion energy produced to driver energy) of \( \geq 50 \). The critical concept underlying the achievement of high gain is the tailored compression of fuel which results in a fuel assembly consisting of two effectively distinct regions at nominally equal pressure: a hot spot of modest density, but high fuel ion temperature, comprising approximately 3–5% of the total fuel mass, and the remaining fuel mass, compressed to a very high density but at a low temperature. The fuel ignites in this hot spot, resulting in a vigorous fusion burn that propagates into, and consumes approximately 30–40% of, the high density fuel [1]. Once an ICF capsule ignites, the propagation of the fusion burn to produce high gain is relatively straightforward. Consequently, a laboratory demonstration of ignition and propagating burn serves as the final requirement to establish the scientific feasibility of ICF.

During the past decade, using both the Nova laser and underground nuclear tests, the ICF programme at the Lawrence Livermore National Laboratory (LLNL) has made substantial progress in target physics, target diagnostics and fabrication, and laser science and technology. Extensive experiments and modelling have established much of the physics necessary to validate the basic concept of ignition and ICF target gain in the laboratory [2].
FIG. 1. The Nova Upgrade will be constructed within the existing Nova building at LLNL. Approximately half of this building was built in 1976 for the Shiva laser fusion system, and the remainder was constructed in 1982 to house the larger Nova laser and experiment area. The ability to upgrade Nova to 1–2 MJ of energy within the existing facility results from significant advances in solid state lasers technology made in recent years at LLNL.
The next major step in the national ICF programme is the experimental demonstration of ignition and modest gain \((G \leq 10)\) \([3, 4]\). To achieve this objective, LLNL has proposed an upgrade to the Nova laser from its present on-target 30-50 kJ, 30-50 TW output to 1-2 MJ, 300-700 TW at 0.35 \(\mu\)m \([5]\). This facility will incorporate many of the scientific and technological advances in laser technology that have occurred since the construction of Nova so that it will achieve these performance goals for a total estimated construction cost (TECC) of US \$400 million (as-spent dollars) for a four year construction period beginning in 1995.

Before proceeding with the construction of such a facility, further physics and technical milestones must be achieved. Target experiments utilizing the Nova laser must further demonstrate our quantitative understanding and modelling of the key target physics and our ability to scale this understanding to targets capable of ignition and gain. In addition, laser and optical manufacturing technologies must be further developed to improve the technical and cost basis for the Nova Upgrade facility.

The goals and present status of the Nova target physics programme have been summarized elsewhere \([5-8]\). In the following sections we shall describe the present Nova Upgrade facility and discuss the associated ongoing technology programme that underlies the design.

2. NOVA UPGRADE LASER DESIGN

Requirements for the laser are determined by ignition target requirements and are met by the Nova Upgrade, a nominally 12-18 beamline, flashlamp pumped, frequency converted Nd:glass laser. The laser is a compact multipass design, fully relayed, with 4 \(\times\) 4 segmented optical components, each with nominal aperture of 30 to 35 cm. Each beamline is thus composed of 16 beamlets, giving a total of 192 to 288 for the entire laser. The beamlets are individually pointed at the target for maximum control of illumination uniformity. Multiple independent pulse generation systems are also envisaged giving maximum experimental flexibility. All beamlines are exact duplicates of one representative beamline, leading to economy of design and construction. This design is compact, efficient and expandable. These features allow the Nova Upgrade to be built in the existing Nova facility (as shown in Fig. 1) and to utilize many of the existing site and Nova resources.

A low energy pulse of the required format is produced by a laser oscillator in the pulse generation section. This pulse passes through beam conditioning devices that provide the desired initial energy, temporal shape and bandwidth. The pulse gains energy and power in the main amplifier section, which is traversed several times. After transport to the experimental area, the 1.05 \(\mu\)m laser beam is frequency converted to 0.35 \(\mu\)m and focused and directed onto a target. Additional sections may be employed after the frequency converters to modify beam coherence as required by the targets.
The multipass architecture is key to this design and allows greater energy extraction from the amplifier modules [5]. Optimization in the staging of multipassed components has ensured a compact design. Higher damage thresholds throughout the design have kept overall aperture sizes tolerable.

The ability to operate the Nova Upgrade at peak average design fluences substantially greater than possible with Nova is due to improvements in the last several years in thresholds for optically induced damage. Since the cost of a laser system scales approximately with the aperture area, these increases in damage threshold are a major reason for the ability to achieve the 1–2 MJ Nova Upgrade performance at the TECC goal [9].

Given the architecture to be described below, we combine the energy storage, extraction, propagation and frequency conversion models to accurately predict the performance of the laser.

This combined model specifies how much energy and power are available from any individual system, within constraints of flashlamp lifetime, damage to components, non-linear optical effects and pulse distortion. The cost models can then be used to derive the affordable beam count and, therefore, the total power and energy available within a specified budget.

3. NOVA UPGRADE LASER SUBSYSTEMS

As described previously, each beamlet, which constitutes the basic building block of the Nova Upgrade laser, consists of three main elements: a pulse generation system, a large aperture multipass amplifier and beam conditioning and transport. The facility also includes a target chamber designed to handle the multimegajoule yields from targets with fusion gains of ~10–20. These elements will be briefly described below.

3.1. Pulse generation

The pulse generation subsystem of the laser consists of the oscillators, pulse shapers, preamplifiers and beam control hardware up to the optical system that injects a properly shaped and timed input pulse into each beamlet of the multipass power amplifier.

The pulse generation system injects a shaped 1 to 10 J pulse into each beamlet. The exact shape and energy depend on the operating conditions, and several different pulse shapes can be delivered to groups of beamlets during a single target shot.

The baseline conceptual design for the pulse generation system uses integrated optics modulators and fibre optics distribution [10]. The initial laser pulse is produced by an array of diode pumped continuous wave (cw) oscillators whose output is modulated in amplitude and phase by low voltage electro-optic modulators. These pulses are distributed into different single mode optical fibres, each with the proper temporal
shape and phase modulation. The pulses go through amplifier and splitter states and then into a fibre optic 'switchyard'.

Much of the technology required for this concept has been demonstrated in a high peak power, picosecond pulse glass laser system at LLNL [11]. This system uses a switched multipass preamplifier to amplify the pulse from a single mode fibre by a factor of $10^{10}$ at a pulse repetition rate of 1 Hz. For the Nova Upgrade, a somewhat longer pulse and higher energy are required, but less bandwidth. The development path is straightforward.

Figure 2 shows the output beam at full performance from a prototype pulse generation system which incorporates all features described above. The excellent beam quality ($1.2 \times$ diffraction limit) and fill factor (0.87) have met the specifications for the Nova Upgrade front end.

### 3.2. Multipass amplifier

Nova uses image relaying and spatial filtering to reduce beam non-uniformity arising from diffraction and non-linear refractive index effects [12]. Figure 3 shows a schematic layout of the image relayed multipass final amplifier stage for the Nova Upgrade. On one side of the cavity there is the amplifier, and on the other side the plasma-electrode Pockels cell (PEPC) [13], which is fired to rotate the polarization of the laser pulse so that the thin-film polarizer extracts the pulse from the cavity on its final pass. A prototype PEPC has been constructed and demonstrated both the required performance and operational lifetime. Switching efficiencies exceeding $99\frac{1}{2}\%$, uniformity <1%, and switching times of <20 ns have been demonstrated with a lifetime exceeding 40 000 shots. The baseline design has two reflections from M1 and one from M2. In one design variation, the output pulse then goes through a second amplifier, which boosts the energy to the full output value.
FIG. 3. Image relayed multipass final amplifier for the Nova Upgrade, including a beam dump to protect the cavity if the switch fails.

The compact array amplifier for the Nova Upgrade has individual components of the amplifier that are not significantly different from those on Nova or other large glass lasers; each of the 16 beamlets has a nominal aperture dimension of 30–35 cm. However, the compact array has much fewer mechanical components, flashlamps and connections, occupies much less space than a comparable quantity of more conventional hardware and is more efficient. These features reduce the cost of laser and facility.

The use of a multipass amplifier requires a high degree of gain uniformity to avoid the buildup of low spatial frequency intensity non-uniformities which could damage optical components and reduce the laser focusability and frequency conversion efficiency.

A prototype amplifier, shown in Fig. 4, has demonstrated gain uniformity < 1% across a full aperture of 35 cm, meeting the performance specifications. The experimental two dimensional gain map is shown as an insert in Fig. 4.

As part of the National Academy of Sciences recommendations, a scientific prototype of one of the Nova Upgrade beamlets is in progress, scheduled for completion in 1994. This prototype will integrate all components at a full scale and will produce > 5 kJ of 0.35 μm light in a 3 ns pulse. The beamlet is illustrated in Fig. 5.
FIG. 4. Demonstrated gain uniformity across the full aperture of the prototype amplifier.
3.3. Beam transport and target illumination

After amplification, the laser pulse enters a beam transport system composed of mirrors and an output relay lens pair that forms an image of the last amplifier near the frequency conversion crystals. The frequency converter consists of two KDP crystal plates of approximately 1 cm thickness in series, as on Nova.

Single pairs of crystals on Nova have demonstrated 80% energy conversion to the third harmonic under conditions appropriate for the Nova Upgrade, and 3 × 3 arrays have demonstrated energy conversion efficiencies as high as 70% [14]. These values equal or exceed the Nova Upgrade design specification of 70%.

The proposed method of final pointing and alignment is to orient the last two mirrors in the beam transport system to point each beamlet at the target chamber centre and then to make fine pointing adjustments on each beamlet separately by using translations of the final focusing lens. This method is similar to the final pointing and alignment done on Nova. The standoff distance of 5.5 m between the last optical element (a blast shield) and the target is sufficient for fusion yields below 45 MJ [15].

Beam smoothing techniques would make the laser produced X ray source within the hohlraum more spatially homogeneous and potentially more efficient. These techniques should also reduce levels of unwated beam-plasma instabilities, such as beam filamentation and stimulated Raman and Brillouin scattering, which can occur during high power beam propagation through large, underdense plasmas in hohlraums.
Several techniques for beam smoothing including efficient multistep phase plates are being developed and specific beam smoothing requirements for the Nova Upgrade will be determined by ongoing target interaction experiments on Nova. The beam smoothing techniques being developed for solid state lasers will enable the Nova Upgrade to irradiate direct drive targets. A collaborative effort on both direct drive target physics and laser technology is underway between LLNL and the University of Rochester. This effort, with the success of the target programme envisaged for the Omega Upgrade at LLE, will allow a second, direct drive target chamber to be installed for the Nova Upgrade in the future.

4. NOVA UPGRADE TARGET CHAMBER

The target chamber of the Nova Upgrade is designed for sophisticated ICF and weapons physics research; it is intended to contain fusion yields of up to 45 MJ and to provide a safe environment for LLNL staff and the public.

The achievement of ignition and yields of up to 20 MJ will result in emissions of radiation $10^6$ times greater than previously achieved at an ICF facility. To assure maximum safety to the public and the research support staff, as well as flexibility and experimental access, Nova Upgrade design specifications substantially exceed

![Target chamber diagram](image-url)

FIG. 6. Evaluation view of target chamber area.
regulatory safety standards [16]. To meet these objectives, materials used in the experiment area will be carefully selected and tested to minimize activation.

The baseline design for the target chamber, shown in Fig. 6, is a 5 cm thick spherical aluminium shell with a 4 m inner radius. The bare aluminium first wall serves as the vacuum barrier, absorbs the majority of the X rays and debris emitted from the capsule and contains the most severe shrapnel impact anticipated (10 mg at $10^5$ cm/s) without being pierced.

The baseline design includes a 35 cm thick shield of high density polyethylene immediately outside the Al shell. This self supporting shield, constructed from large interlocking blocks, attenuates both prompt neutrons from the target and decay γ rays from the Al wall. As discussed above, a second target chamber for direct drive could be also added. This chamber, outside the illumination geometry, would be similar to the baseline chamber design.

5. SUMMARY

Recent advances in ICF target design and experiments have made possible the achievement of ignition and gain with 1–2 MJ, 500 TW of 0.35 μm light. An advanced Nd:glass laser utilizing cost effective multipass architecture can generate the required drive output at a cost of US $500 million. Substantial progress in developing the components and demonstrating the architecture of such a facility has been made. A scientific prototype, the beamlet of which will integrate all components at full scale, is on schedule for completion in 1994.

ACKNOWLEDGEMENT

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DISCUSSION

C. YAMANAKA: Pellet uniformity and irradiation uniformity are very important in attaining high compression. In the indirect drive case with high irradiation uniformity, the compressed density of 16 g/cm³ you give seems very low. What kind of fuel pellet do you use for these experiments? Is there any explanation for this low compression density?

E.M. CAMPBELL: Our experiments are designed to measure integrated and detailed capsule performance rather than a simple density milestone. The impressive aspect of this work is the agreement between experiment and simulations, for all the observables including neutron yield, with the high contrast pulse shaped drive.

S. NAKAI: What is the present cost estimate for the NOVA Upgrade, and what cost reductions are expected in the future?

E.M. CAMPBELL: Our goal, for a project start in 1995 or 1996, is a construction cost of approximately US $400 million. A technology effort with industry,
which would focus on the optical, mechanical and electronics aspects, and on pulsed power, is required to reduce the construction cost. Multipass amplifiers and economy of scales reduce the present projected cost per joule by ~3. A further reduction by ~2–3 is required, and a path to achieve this with industry has been designed but is not yet funded.
PROGRESS AND PROSPECTS FOR INDIRECT DRIVE INERTIAL CONFINEMENT FUSION

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Abstract

PROGRESS AND PROSPECTS FOR INDIRECT DRIVE INERTIAL CONFINEMENT FUSION.
The paper reviews recent progress on the Nova laser at Lawrence Livermore National Laboratory (LLNL) and the performance expected with the Nova Upgrade, which is an LLNL proposal for the so-called National Ignition Facility endorsed by the National Academy of Science and the Fusion Policy Advisory Committee.

During the past few years, a great deal of progress has been made toward demonstrating the requirements for ignition and high gain inertial confinement fusion (ICF) targets. Because of this progress, the 1990 National Academy of Sciences (NAS) and Fusion Policy Advisory Committee (FPAC) reviews [1, 2] recommended that the US National ICF Programme focus on the physics of ignition. Subject to successful completion of a series of experiments to be carried out on the Nova laser at Lawrence Livermore National Laboratory (LLNL), these reviews advocated the construction of a 1 to 2 MJ glass laser, whose purpose would be demonstration of ignition and modest gain ICF targets within about a decade. The LLNL proposal for this National Ignition Facility, which was endorsed by NAS and FPAC as the most timely and cost effective path to this goal, is referred to as the Nova Upgrade. This paper reviews recent progress on the Nova laser and the performance expected with the Nova Upgrade.

In general, ICF implosions can be categorized as either direct or indirect drive [3]. In direct drive, the laser beams (or charged particle beams) are incident on the fusion capsule. In the indirect drive (or X ray) approach to ICF, the capsule is imploded by X rays. The laser beams (or charged particle beams) are first absorbed in a high Z enclosure, a hohlraum, which surrounds the capsule. Shown schematically in Fig. 1 are examples of a laser and a heavy ion driven hohlraum. For planar targets, 70 to 80% of this absorbed energy can be converted to X rays. While such conversion efficiencies have been demonstrated with lasers, in order for X rays to be efficiently generated by ion beams, minimum irradiances of $10^{14}$ to $10^{15}$ W/cm$^2$ are required for typical ion energies of 30 MeV to 10 GeV, depending on the ion.
FIG. 1. Schematic of (a) laser and (b) heavy ion indirect drive targets.

FIG. 2. Schematic of proposed ICF capsules including initial and imploded fuel configurations. Critical parameters are also defined with representative values.
The primary challenge for ion beams has been, and continues to be, achieving the required focused intensity. The purpose of the Nova Upgrade would be demonstration of ignition and modest gain using indirect drive ICF targets.

High gain ICF targets have features similar to those shown in Fig. 2. These capsules consist of a spherical shell filled with low density ($\leq 1.0 \text{ mg/cm}^3$) equimolar deuterium-tritium (D-T) gas. The shell is composed of an ablator and an inner region of D-T which forms the main fuel. Energy from a driver is rapidly delivered to the ablator which heats up and expands. As the ablator expands outward, the rest of the shell is forced inward to conserve momentum. The capsule behaves as a spherical ablation driven rocket. The efficiency with which the fusion fuel is imploded typically lies in the range of 5-15%. In its final configuration, the fuel is nearly isobaric at pressures up to $\sim 200 \text{ Gbar} (\sim 2 \times 10^{16} \text{ Pa})$ but consists of two effectively distinct regions — a central hot spot, containing $\sim 2-5\%$ of the fuel, and a dense main fuel region comprising the remaining mass. Fusion initiates in this central region, and a thermonuclear burn front propagates radially outward into the main fuel, producing high gain.

The efficient assembly of the fuel in this configuration places stringent requirements on the details of the driver coupling, including the time history of the irradiance and the hydrodynamics [4]. In this implosion process, several features are important. The in-flight aspect ratio (IFAR) is defined as the ratio of the shell radius $R$ as it implodes to its thickness $\Delta R$, which is less than the initial thickness because the shell is compressed as it implodes. Hydrodynamic instabilities [5] similar to the classical Rayleigh–Taylor fluid instability impose limits on this ratio which result in a minimum pressure or absorbed driver irradiance. For IFAR values of 25–35, the pressure must be $\sim 100 \text{ Mbar} (\sim 10^{13} \text{ Pa})$ and the intensity $\sim 10^{15} \text{ W/cm}^2$. These minimum values depend on the required implosion velocity which is determined by the capsule size. Minimum velocities are in the range of $(3-4) \times 10^7 \text{ cm/s}$. The convergence ratio is defined as the ratio of the initial outer radius of the ablator to the final compressed radius of the hot spot. Typical values of the convergence ratio are 30–40. To maintain a near spherical implosion, we require implosion velocities uniform to about 1%. Control of the implosion symmetry and Rayleigh–Taylor induced mix is crucial to the successful formation of the central hot spot. Departures from spherical symmetry and mix can result in failure to achieve the conditions required for ignition during the final stages of the capsule implosion.

The NAS recommended a series of experiments on Nova as a prerequisite to the construction of a National Ignition Facility. These Nova experiments, in hohlraum and laser plasma interaction physics (HLP) and hydrodynamically equivalent physics (HEP), constitute what is referred to as the Nova Technical Contract (NTC). The NTC experiments are being carried out as a co-operative effort by LLNL and the Los Alamos National Laboratory (LANL).

Demonstration of symmetry control, in hohlraums, with the pulse shape and internal structure scaled from hohlraums required for ignition on Nova Upgrade,
is one of two primary objectives of the HLP experiments. Experiments on Nova using constant power 1 ns pulses as well as temporally shaped pulses have demonstrated that time integrated fluxes uniform to a few per cent can be achieved inside hohlraums. This uniformity was demonstrated by imaging the compressed fuel region of an X ray driven implosion. Experiments which use a variety of pulse shapes and hohlraum configurations and which also obtain information about time variations in symmetry are now under way.

In order to achieve this control of symmetry, it is necessary to have reproducible beam propagation conditions in the laser channel. In addition, these conditions must be consistent with accurate placement of the laser beams. To accomplish this, plasma parametric instabilities such as stimulated Raman scattering (SRS) and stimulated Brillouin scattering (SBS) must be kept below the 5–10% level. Both SRS and SBS can cause a redirection of energy in the hohlraum. In addition, high energy electrons generated from Landau damping of the plasma wave in SRS can result in preheat and reduced compressibility of the fuel. Experiments using 0.355 mm light on Nova have all had SRS levels of a few per cent or less. Some experiments on Nova have had SBS levels in the 5–20% range. Investigation of the control and scaling of these instabilities to the larger plasmas of Nova Upgrade hohlraums is the second major objective of the HLP experiments.

The minimum capsule size required to achieve ignition and propagating burn into the main fuel depends strongly on the achievable implosion velocity. This velocity is, in turn, primarily determined by the peak pressure that can be generated and by the degree of hydrodynamic instability of the implosion process.

Recent experiments on Nova have shown that the single mode Rayleigh–Taylor instability in the presence of ablation and density gradients is well modelled by existing numerical simulation codes. For both direct [6] and indirect drive [7], recent experiments and numerical simulations [8, 9] are well modelled by the dispersion relation:

$$\gamma = \sqrt{\frac{ka}{1 + kL}} - \alpha kV_{abl}$$

In this equation, $\gamma$ is the growth rate, $k$ is the modal wavenumber, $a$ is the acceleration, $L$ is the density gradient scale length in the ablation front and $\alpha$ is a constant between 1 and 3. $V_{abl} = \dot{m}/\rho$ is the ablation velocity, i.e. the velocity with which the ablation front moves through the shell ($\rho$ and $\dot{m}$ are the shell density and mass ablation rate, respectively). In addition, significant progress has been made toward developing experimental and modelling techniques which can be used to evaluate mix in ICF capsules. Mix is the interpenetration of materials on two sides of an unstable interface. A number of such interfaces can exist, but the one of primary interest is that between the hot fuel and the pusher or cold main fuel. Verification that mix can be accurately predicted, in Nova capsules which have hydrodynamic instability growth and a convergence ratio approaching that of Nova Upgrade ignition capsules, is the goal of the HEP experiments on Nova over the next
two to three years. Completion of these experiments requires about a factor of two improvement in the pointing accuracy, up to about 30 μm, and in the power balance up to 5–10%, depending on the pulse shape. The 'Precision Nova' activities to achieve these levels are scheduled for completion at the end of fiscal year 1993.

Using present experimental and theoretical information, sophisticated numerical simulations can be used to predict capsule performance at larger incident driver energies than currently exist in the laboratory. These results are shown in Fig. 3, which presents gain curves for indirect drive at two different implosion velocities [10]. These curves are calculated under the assumption of a fixed hohlraum coupling efficiency of laser energy to a capsule. At any given velocity, capsules below a certain size will fail to ignite because the hot spot does not achieve an adequate ρr and temperature. The shaded band which defines the minimum driver size for ignition, at the left of each set of curves, corresponds to the uncertainty in the achievable capsule surface quality. The left hand edge corresponds to the gain achievable for perfectly uniform implosions. The right hand edge of the band corresponds to the gain for targets with surface finishes of 500 to 1000 Å (used in present experiments). As the driver energy or capsule size increases, the minimum implosion velocity required to ignite a capsule decreases. If we exceed the minimum velocity at any driver size, the capsules will still ignite. However, there is a performance penalty for operating above the minimum velocity. The gain will drop
because we will implode less mass and obtain lower yield for a given energy. Hence, the optimum strategy implies operation at the minimum implosion velocity consistent with the desired yield or driver size. This optimum is given by the dashed line through the two curves.

Given the constraints of hydrodynamic instability, achieving a higher implosion velocity requires a higher hohlraum temperature. In early 1990, using the recently increased power and energy available from Nova, LLNL demonstrated hohlraum drive temperatures consistent with achieving the implosion velocity of $4 \times 10^7$ cm/s required for demonstration of ignition and burn propagation using a 1–2 MJ laser. By using the compact Athena laser amplifier architecture being developed at LLNL [11], it is possible to fit a 1–2 MJ upgrade to Nova into the existing Shiva–Nova building. High gains at lower driver energies would be achievable at higher implosion velocities, but we believe that plasma physics constraints on hohlraums will limit the achievable implosion velocity to approximately $4 \times 10^7$ cm/s.

The demonstration of ignition and burn propagation in the laboratory would complete the basic target physics objectives of the ICF programme. Such a demonstration would set the stage for high confidence development of the applications of ICF. Although it has not yet endorsed construction of the Nova Upgrade, DOE has begun implementing the NAS recommendations, including the Nova Technical Contract and demonstration of the laser technology which would be required for Nova Upgrade. The next few years promise to be a very exciting time in the inertial confinement fusion programme. With continued success of the technical programme and the required funding for the Nova Upgrade, it should be possible to achieve ignition and fusion burn propagation soon after the turn of the century.

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EXPERIMENTAL DETERMINATION OF THE HYDRODYNAMIC INSTABILITY GROWTH RATES IN INDIRECT AND DIRECT DRIVE ICF

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Abstract

EXPERIMENTAL DETERMINATION OF THE HYDRODYNAMIC INSTABILITY GROWTH RATES IN INDIRECT AND DIRECT DRIVE ICF.

A comprehensive set of measurements of the growth of ablatively stabilized Rayleigh-Taylor instabilities in laser driven direct drive and indirect drive planar foils is presented. For both cases, imposed single Fourier modes are seen to grow as a planar foil is accelerated. Harmonics of the imposed wavelength are observed as the sinusoidal perturbation becomes a classic bubble and spike. Independent measurements of the foil acceleration are made by sidelighting the foils. For small amplitude initial perturbations, a shaped indirect drive has been used to achieve growth rates as large as 80. Two-dimensional simulations of the foils demonstrate excellent agreement with the measured growth rates and with the non-linear saturation. For both direct and indirect drive, the small amplitude growth rates are seen to be in agreement with simple dispersion relations $\gamma = \alpha \sqrt{k} - \beta k \nu$, albeit with different values of the constants for the two cases. Further experiments with radiatively accelerated planar foils with two imposed modes of wavenumber $k_1$ and $k_2$ show mode coupling with sum and difference modes, again in agreement with theory. Experiments have also been performed measuring the modulations in areal density of unperturbed foils. The early time non-uniformity of the laser beam can be varied by changing the bandwidth of the SSD smoothing scheme. As the smoothing of the laser drive beam is increased, the degree of non-uniformity of the accelerated foil decreases. For indirect drive there is no measurable non-uniformity of an initially smooth accelerated foil. Indirect drive experiments need a rough foil with a 4 $\mu$m surface finish to see any growth of Rayleigh-Taylor non-uniformity.

In ICF designs, a high aspect ratio shell is required to increase the ablation pressure of drive. However, the growth of the dominant mode of hydrodynamic instability of a shell during an implosion increases with the shell's aspect ratio, requiring an accurate modeling of the instability so that it
does not dominate the ignition physics. In this paper measurements of the Rayleigh-Taylor hydrodynamic growth rates for both indirect and direct drive acceleration of a planar foil are presented, and show good agreement with calculations of both the linear growth rate and the rate of mode coupling. For radiative drive, these measurements are made for hydrodynamic growth factors similar to high gain designs and give confidence that the modeling of the critical hydrodynamic instabilities for ignition designs is correct. In addition, the growth of instability seeded by laser beam non-uniformity and surface roughness are measured for the first time.

Designs of ICF implosions are possible despite the Rayleigh-Taylor instability because of two ameliorating effects. First, ablative stabilization reduces its growth rate to significantly below the classical value of $\gamma = \alpha \sqrt{k_g}$. Numerical studies [1] have shown that a simple dispersion relation $\gamma = \alpha \sqrt{k_g} - \beta k_g^a$ fits the calculated linear growth rate with different values for $\alpha$ and $\beta$ for indirect and direct drive. For values of drive appropriate to ICF capsules, a reduction of the linear growth rates to 60-70% of classical is predicted, with a concomitant large increase in the acceptable in-flight aspect ratio of a capsule. Second, non-linear effects [2] become important (but are not dominant) requiring an accurate description of the weakly non-linear phase of the hydrodynamic instabilities. Experiments described in this paper with both indirect (x-ray) drive and direct drive have verified that the calculations of the growth and mode coupling of the hydrodynamic instabilities are accurate. Preliminary measurements of the growth of multi-mode instabilities seeded by the early time laser beam non-uniformity of a smoothed laser beam are also presented.

To measure the growth of Rayleigh-Taylor instabilities, planar foils with small initial perturbations have been accelerated by x-rays [3] or by direct drive [4]. Face-on radiography, as illustrated in Fig. 1, for the indirect drive experiments,
FIG. 1. Experimental setup and sample raw data. (a) Schematic of the experimental setup (not to scale): the foil is mounted on the front wall of a cylindrical gold hohlraum with surface perturbation facing inwards, and drive beams entering the ends of the hohlraum generate a thermal x-ray drive. As the foil accelerates by x-ray ablation toward the 22× magnification x-ray microscope, a backlighter beam striking a Rh disk generates a back-illumination of x-rays which travel through the hole on the back wall of the hohlraum, and subsequently through the accelerating foil. Modulations in foil area density then translate to modulations in exposure at the x-ray camera. (b) One-dimensional image of the optical depth modulation of an accelerated foil "streaked" vertically versus time. The overall increasing brightness with time is from foil thinning due to bubble formation and from hohlraum emission (see text), and the abrupt cessation of the brightness occurs when the backlighter beam turns off. (c) Two-dimensional x-ray "snapshot" of the same accelerating foil during a 100 psec window centered at 2.6 nsec after the start of the drive. Little if any transverse distortion or bowing is observed.
accurately measures the growth and non-linearity of small perturbations, after
careful account is taken of the backlighting x-ray spectrum and the spatial
resolution of the several x-ray microscopes used. For the indirect drive
experiments most of the measurements were made with a Wolter (magnification
= 22X) x-ray microscope coupled to an x-ray streak camera. The time
dependence of the growth in the modulation in x-ray optical depth, normal to the
direction of the initial modulations, is then measured along the x-ray streak. For
the direct drive experiments an 80 psec gate time, multi-aperture x-ray pinhole
camera [5] was used to record a series of two-dimensional framed radiographs.
This novel technique allows the growth in modulation to be observed by frame
to frame comparison and moreover allows two-dimensional modulations to be
measured. In both cases the object plane spatial resolutions were 10 μm.

Indirect drive experiments [3] have been conducted with eight beams of the
Nova laser using 16 kJ of 0.35 μm light in a shaped pulse to produce an x-ray
drive to accelerate a planar foil. The use of a shaped laser pulse of length 3.2
nsec and contrast of 6:1 between the 1 nsec foot of the pulse and the 13 TW
peak of the pulse, is chosen to produce an x-ray drive that keeps the planar
accelerated foil on a low adiabat and at a high density. Most of the experiments
were performed with fluorosilicone or brominated plastic foils, both chosen for
their high x-ray opacity leading to high in-flight density and consequently a
relatively high hydrodynamic growth rate. X-ray backlighting was performed by
focusing another beam of Nova at 0.53 μm in a 5 nsec pulse onto disks of
materials chosen to optimize the x-ray flux at a chosen wavelength. The x-ray
drive has been characterized in several independent ways. The acceleration of
the fluorosilicone (SiOC₄H₇F₃) foil is measured by side-on radiography at 6 keV
[6] and in agreement with measurements of the radiation temperature from both
the shock break out of wedged witness plates and the x-ray flux.
To measure the hydrodynamic growth of single modes, the planar packages were sinusoidally modulated on their drive side at wavelengths from 30 to 100 μm, with initial amplitudes from 0.16 μm to 4 μm. With the shaped pulse, the hydrodynamic growth factors were measured to be up to 80, and so for the 50 μm wavelength case, the 0.16 μm perturbations only grow to about 20% of the initial wavelength and so remain essentially linear. For larger modulations this is not so and harmonics are clearly seen to grow.

A backlit streaked face-on radiograph of the growth in the modulations of optical depth of the foil is also shown in Fig. 1. Fourier transforming this radiograph shows the initial growth of modulations. At later time the formation of higher harmonics of the initial modulations is seen as the amplitude of the modulation exceeds about λ/10. Fig. 2 shows an amplitude scaling study of this experiment. Three initial amplitudes 0.16 μm, 0.8 μm and 4.5μm were used. As long as the initial perturbation is small enough that there is little saturation, a growth of single modes by as much as a factor of 80, close to the values required for ignition designs, is measured. Calculations with an average atom model for opacity (XSN) are superimposed in Fig. 2 showing good agreement for the small initial amplitude case where a growth factor of 80 is measured. A super configuration transition array model for the x-ray opacity does not give such good agreement and differences in the modeling are currently being scrutinized. It is meaningful to compare the growth of the modulations with a simple dispersion relation \( \gamma = \alpha \sqrt{k} - \beta k \nu_a \) only if the measurements are for a linear phase and after the shock has passed through the foil. In this phase of the growth of modulations, the simple dispersion relation with \( \alpha \sim 0.9 \) and with \( \beta \) between 1 and 2 for the ablative stabilization term best fits the data. Under these conditions the measured and calculated growth rates are about 60% of classical.
FIG. 2. Coefficient of the fundamental mode (circles), second harmonic (squares), and third harmonic (triangles) from a Fourier transform of the ln (exposure) curves for accelerated thick fluorosilicone foils with large (a), intermediate (b), and small (c) initial amplitude perturbations. The data have not been corrected for the instrumental resolution (MTF). The smooth curves are the results of two-dimensional computer simulations, including the effect of the instrument MTF. The thick curves assume constant back-illumination from the Rh backlighter disk only; the thin curves include the additional transitory contribution from hohlraum emissions (see text). All curves are timed relative to the start of the drive lasers at $t = 0$. For the intermediate-amplitude foil, a calculation using a different opacity model, default XSN, is shown with the dot-dashed curves for the fundamental and the second harmonic. All other curves correspond to simulations using a modified opacity model.
In experiments when the initial perturbations become comparable in amplitude with their wavelength \((2\pi/k_0)\), harmonics \(k_0 + k_0 = 2k_0\) and \(k_0 + 2k_0 = 3k_0\) are clearly seen growing as the characteristic bubble and spike formation appears. In further experiments, two different initial perturbations \(k_1\) and \(k_2\) (wavelength 50 \(\mu\)m and 70\(\mu\)m), were imposed on an accelerated planar foil. Sum and difference modes, \(k_1+k_2\), and \(k_1 - k_2\), are clearly seen growing at a rate that agrees with simulations.

Similar hydrodynamic growth rate measurements have been made for directly driven planar foils, at the lower end of the drive pressures envisioned for ignition implosions. A single 5 kJ, 3 nsec 0.53\(\mu\)m laser beam smoothed by the technique of smoothing by spectral dispersion (SSD) [7] was split into nine and overlaid by steering wedges to provide a laser beam that had a 10 \% large scalelength non-uniformity over a 1mm focal spot at a peak intensity of \(8 \times 10^{13}\) W/cm\(^2\). The smoothing of the laser beam was varied by the bandwidth of the laser beam. Initially a large bandwidth of 0.14\% was used that produced a small scale anisotropic perturbation \(\delta l/l\) of 8\% at late time: at early times the smoothing is less with the smoothing time decreasing with increasing bandwidth. The anisotropic smoothing is clearly visible in Fig. 3, an equivalent plane photograph of the focused beam.

When this laser beam was incident on a 20 \(\mu\)m thick plastic foil an ablation pressure of approximately 9 Mbar was produced as measured by the acceleration from side-on radiography of the foil, and by measurements of the shock transit time through wedged witness plates. This is the pressure expected from simulations with the 80-100\% absorption expected from measurements of other authors at this intensity.
As in the indirectly driven case, we have measured the growth of 1\mu m initial amplitude, 20-70 \mu m wavelength perturbations on the foil. Up to 12 sequential gated radiographs, three of which are shown in Fig. 4, measured the perturbation growth by up to a factor of 10 on each experiment. Fig. 4 shows both the initial perturbation of wavelength 70 \mu m (in this case) growing as well as modulations that are imposed by the anisotropy of the laser beam. For the imposed wavelength, modal analysis shows the second, third and fourth harmonic growing as shown in Fig. 4. Two-dimensional simulations are also shown on Fig. 4 where there is good agreement with the experiments.

As well as the imposed perturbations on Fig. 4 the late time frame shows a perturbation that resembles the striated structure of the smoothed laser beam at
an angle of about 30° to the imposed perturbation. This observation prompted experiments where the degree of smoothing of the laser beam was varied by the bandwidth of the laser. Bandwidths from 0% to 0.15% were used and as the bandwidth and therefore laser smoothing was increased, the modulations of the areal density decreased drastically.

In conclusion this work has shown that the linear and quasi-linear simulation of x-ray drive and directly driven foils, at ablation pressures relevant to ICF capsules, and for growth factors similar to those expected for ICF capsules,
agree with experiments. As the hydrodynamic models of mix rely on the linear and quasi-linear growth rates, this work validates the accuracy of these models. Further work will directly measure the growth of a spectrum of modes.

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DISCUSSION

H. NISHIMURA: In order to apply the analytical model (the so-called Takabe formula), you need to know the density scale length and the ablation velocity. How did you derive and verify these values?

J.D. KILKENNY: For our comparison with the simple analytical model we used the simulation values. The most significant comparison, however, is with the classical growth rate ($g_{cl}^{1/2}$), for which no code values are needed as $g$, the target acceleration, is measured directly.

M.G. HAINES: You stated that transport is very important in these Rayleigh-Taylor experiments. Since they are two dimensional, I wonder whether you have used 2-D Fokker-Planck modelling?

J.D. KILKENNY: The direct drive experiments were performed at $8 \times 10^{13} \text{ W/cm}^2$ with 0.53 $\mu$m laser light. This gives a low value for $\Gamma\lambda^2$, so that heat fluxes will be much lower than the flux limits. Our modelling, which so far is simple flux limited, should adequately describe the phenomena.

M.G. HAINES: 2-D Fokker-Planck calculations (Rickard and Bell) have shown marked, new effects, such as the heat flux having a component up the density gradient as well as down a temperature gradient, and this can modify thermal smoothing.

J.D. KILKENNY: At higher values of $\Gamma\lambda^2$ a more complex transport formulation will be needed in the modelling. Hydrodynamic growth may be a powerful tool for understanding electron transport in these conditions.
THEORETICAL STUDIES ON NON-LINEAR STAGES OF HYDRODYNAMIC INSTABILITY IN LASER DRIVEN IMPLOSION

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Abstract

THEORETICAL STUDIES ON NON-LINEAR STAGES OF HYDRODYNAMIC INSTABILITY IN LASER DRIVEN IMPLOSION.

Three topics related to the non-linear stages of hydrodynamic instability in laser driven implosion are studied: (1) With the help of a two dimensional Lagrangian hydrocode, the generation of a turbulence state through mode-mode coupling in the Rayleigh-Taylor instability has been demonstrated. (2) Anomalous energy diffusion due to turbulence is described by a non-linear diffusion model, explaining the degradation of neutron yield in a high density implosion experiment. (3) Fundamental studies have been performed theoretically and numerically on linear and non-linear stages of the Richtmyer-Meshkov instability to obtain and confirm a universal relation describing its evolution.

1. INTRODUCTION

To achieve inertial confinement fusion (ICF) with the use of realistic driver energy, the fuel must be compressed up to $10^3$ times the solid density. Spherical implosion of the shell structured fuel is essential for ICF. Hydrodynamic uniformity appears to be the most important subject in ICF research. Technical improvements of the uniformities of drivers (lasers) and targets have been carried out, and the physics of the hydrodynamic stability has also been studied theoretically and experimentally.

Several institutes [1–3] have reported that the experimental fusion yields have been far below the values predicted by one dimensional hydrodynamic simulations.
Even in the case of implosion experiments with plastic targets characterized by relatively thick shells, the achieved yield is two orders of magnitude lower than predicted although 600 times solid density has been measured [4]. This degradation of the fusion yields is mainly due to the non-uniformity of implosion, whose amplitudes are increased by the hydrodynamic instability in the course of the implosion.

In the present paper, we study the multimode Rayleigh–Taylor instability related to turbulent mixing in the laser implosion dynamics. In addition, a fundamental study has been performed on the Richtmyer–Meshkov instability.

2. NON-LINEAR DEVELOPMENT OF FLUID INSTABILITY AND RESULTANT TURBULENCE STAGE

In order to study the non-linear development of the multimode Rayleigh–Taylor instability and the generation of turbulence, we have carried out a two dimensional model simulation of a plasma at rest in a gravitational field. To resolve small amplitude perturbations, a Lagrangian code is used without artificial viscosity. In the simulation, initial perturbations are imposed at the discontinuous surface with random number. The time development of the amplitude of each Fourier mode is shown in Fig. 1, where $k \xi_k$ is the product of wavenumber and displacement of each mode. We see that initially each mode grows linearly for $t < 4.5$ ns. Taking the gradient of each mode of growth yields the linear growth rate.

![FIG. 1. Time evolution of the amplitudes of Fourier expanded perturbations.](image-url)
At later times when the amplitudes of modes with larger growth rates become sufficiently large, the mode-mode coupling between them tends to make other modes with smaller growth rates grow abruptly. This corresponds to the rapid growth seen for $t > 4.5$ ns in Fig. 1. Actually, a model equation for an unstable mode can be written as

$$\frac{d}{dt} v_k = \gamma_k v_k - \sum_{k' \pm k'' = k} v_{k'} \nabla v_{k''}$$  \hspace{1cm} (1)

Here, the first term on the right hand side yields linear growth, and the second term stems from the non-linearity of the convection term. In the present simulation, the $k'$ and $k''$ modes with larger growth rates play the role of a force term to the $k = k' \pm k''$ modes with smaller growth rate. This force term makes the modes with smaller linear growth rates grow abruptly through a mode-mode coupling process [5].

In Fig. 2, the time development of the energy spectra of Fig. 1 is shown as a function of $k^{1/2}$. It is clear that in the linear regime ($t < 5$ ns) the spectrum evolves according to the $k$ dependence of the growth rate $\gamma_k$, while the wings of the smaller and larger $k$ regions both start to grow abruptly after 5 ns, and finally the spectrum becomes almost that of 'white noise', i.e. a turbulent state is formed. The critical time

**FIG. 2. Turbulence energy spectra as functions of wavenumber $k$ (the numbers assigned to the traces represent the time in nanoseconds).**
when the mode-mode coupling becomes important depends, of course, on the amplitudes of the initial perturbations.

Moreover, at later times the short wavelength modes grow up to the amplitude satisfying the condition $k\xi_k \approx O(1)$, and the second non-linear stage starts. In the second stage a bubble-and-spike structure can be seen, and each mode enters the non-linear phase as a single mode. Then, the bubble tends to grow slowly as $\xi_k \propto t$ instead of $\xi_k \propto \exp(\gamma t)$. As a result, modes with relatively small $k$, i.e. large bubbles, tend to be dominant [6] in the turbulent mixing region.

Phenomena consisting of larger bubbles becoming dominant at later times may also occur because of a bubble coalescence process, which is essentially the same as the inverse cascading process due to mode-mode coupling [6].

3. NON-LINEAR DIFFUSION BY TURBULENT MIXING

Anomalous mixing of material, momentum and energy is induced once turbulent fluid motion has been generated. On the basis of the quasi-linear theory, a non-linear diffusion model [6] has been introduced into the one dimensional fluid code so that the degradation of the fusion yield in the plastic shell implosion can be studied. On the assumption that the Rayleigh-Taylor instability is strongly induced in the stagnation phase, the anomalous diffusion model explains the experimental results consistently.

In order to introduce the effect of turbulent mixing into a one dimensional code, we have added a turbulent diffusion term of the form

$$\frac{df_0}{dt} = \nabla \cdot (D \nabla f_0) + \text{other terms}$$

(2)

where $D$ is the turbulent diffusion coefficient given, in spherical geometry, by

$$D = \sum_{l,m} \xi_{l,m} \frac{d}{dt} \xi_{l,m}$$

(3)

Here, $\xi$ is the displacement, and the summation is taken over all modes expanded by the spherical harmonics. Linear development according to the effective saturation model by Haan [7] is assumed for $\xi_{l,m}$. When $f_0$ is the temperature, $D$ in Eq. (2) represents an anomalous thermal conductivity stemming from the material convection generated by the hydrodynamic instability.

In the plastic shell implosion experiment the conventional one dimensional code allows the formation of a high temperature, low density hot spot region at the centre while in the experiment no hot spot seems to be formed [4]. It is suggested that the turbulent mixing is generated in the stagnation phase and, consequently, anomalous thermal conduction does not allow the temperature to increase at the core centre.

In order to explain such phenomena numerically, we have introduced the anomalous diffusion term into the 1-D code. The neutron yield and the product of
Density and radius for a typical plastic shell implosion are shown by E with an error bar in Fig. 3. It is possible to reproduce one of these two values by modifying the radiation transport model without invoking the anomalous diffusion term. Since silicon with which the target is doped for diagnostic purposes may preheat the target, the radiation transport model with lower preheat results in points A and C in Fig. 3, while the model with enhanced preheat yields point B. By including the anomalous diffusion term in the stagnation phase, we have obtained point D. In the simulation, we start including the diffusion term when void closure takes place. A total 15% displacement of $\ell = 1-100$ is imposed.

In the case without mixing, the hot spot region has sufficiently high pressure to bring about stagnation in the surrounding high density region. Since no conduction or energy loss processes can cool the hot spot drastically, excess neutrons are produced in this region. In contrast, the inclusion of the mixing process allows a rapid cooling of the central region due to anomalous energy diffusion. As a result, the central region cannot sustain the surrounding material, and no stagnation process is seen. It is noted that in the case of mixing, the density and temperature structures near the compression maximum are rather flat.
4. NON-LINEAR EVOLUTION OF THE RICHTMYER-MESHKOV INSTABILITY

In the previous sections, we have studied turbulent mixing due to a multimode process. In contrast to this, when an instability with a given wavelength is induced predominantly, it enters the self-non-linear regime when $k \xi \approx O(1)$ is satisfied. Detailed studies have been conducted on the non-linear stages by using 2-D and 3-D fluid codes [8]. Here, we describe the evolution of the Richtmyer-Meshkov (R-M) instability with a given wavenumber.

The linear and non-linear time evolution of the R–M instability is investigated by theoretical analysis and numerical simulations [9]. The theoretical model gives the amplitude $a(t)$ for a perturbation of $k = 2\pi/\lambda$ in the form

$$\frac{a(t)}{\lambda} = F(\xi)$$

Here, $F(\xi)$ is a universal function and $\xi$ is an effective time given by

$$\xi = \frac{a^*}{\lambda} + \frac{a^*}{\lambda^2} 2\pi A \Delta \nu t$$
where $a^*$ is the initial perturbation amplitude, $A$ the Atwood number and $\Delta V$ the velocity difference after shock passage.

By varying initial amplitude, wavenumber, Atwood number and Mach number of the incident shock waves, a two dimensional simulation has been carried out and the resultant normalized amplitudes of Eq. (4) are plotted in Fig. 4, from which we have obtained a universal function of the form

$$F(\xi) = c \log[(\xi + c)/c]$$

(6)

where $c = 0.794$ provides the best fit to Fig. 4.

It can easily be derived from Eq. (6) that if we have a multimode situation initially, the dominant mode shifts from a shorter to a longer wavelength mode as time passes on.

5. CONCLUSIONS

By using numerical procedures, we have found that

(1) in spite of the non-linear mode--mode coupling process, a turbulent state appears in the case of a multimode Rayleigh--Taylor instability;

(2) anomalous diffusion due to turbulent mixing in the stagnation phase can explain the degradation of neutron yield in the high density implosion experiment; and

(3) a universal function describing the evolution of the Richtmyer–Meshkov instability can be derived theoretically and good agreement with the numerical results is obtained.

REFERENCES


DISCUSSION

R.L. McCRORY: Have you considered anomalous transport effects (expected for irradiances of \(10^{14} - 10^{15} \text{ W/cm}^2\)) on the universal formula you derived for the Richtmyer–Meshkov instability?

H. TAKABE: No, but maybe this fitting formula for the amplitude of the Richtmyer–Meshkov instability could be compared with that proposed by S. Haan in the case of the Rayleigh–Taylor instability. It would be interesting to use the formula to see the mixing by the Richtmyer–Meshkov instability.

S. ATZENI: You showed that there is evidence of mode coupling in the nonlinear evolution of the Rayleigh–Taylor instability and then you developed a model for the non-linear diffusion caused by the mixing. This was based on Haan’s model for the evolution of the Rayleigh–Taylor instability (Phys. Rev. A, 39 (1989) 5812), which does not assume any mode coupling. Could you comment on this?

H. TAKABE: You are correct. Haan’s model includes non-linearity through the self-non-linear effect but no mode coupling. Whether the effect of mode coupling is essential to the mixing depends on the non-uniformity level of the targets and the laser non-uniformity condition. In order to maintain the consistency of the mixing mode, we ought to improve our model to include mode coupling.

P.K. KAW: Do you have any feeling for the scaling laws of the turbulent diffusion coefficient and their impact on the proposed ignition experiments?

H. TAKABE: This is an important question. Our eventual aim is to apply the turbulent diffusion model to see the fusion yield degradation in the ignition scalings. Up to now, the model has been applied only to the high density experiment, simply to provide a possible explanation.

J.D. LINDL: How are the mode amplitudes calculated for the diffusive mixing mode? Do you include feedthrough from the ablation surface as well as effects of surface finish and non-uniform laser intensity?

H. TAKABE: First, we include the perturbation amplitude obtained by means of a perturbation code in which linearized equations are solved numerically by coupling with the one dimensional implosion code. Non-uniformity of laser absorption is included in the simulation. In this case, the plastic shell shows strong diffusion in the acceleration phase and no high density is seen. In our simulation, therefore, only the stagnation phase has been treated by including the growth of the Rayleigh–Taylor instability with the classical growth rate. Our standpoint on the 600 times liquid density implosion experiment is that there is something missing which stabilizes the acceleration phase in addition to the well known ablative stabilization mechanism.
IMPLOSION EXPERIMENTS INVOLVING CRYOGENIC FOAM TARGETS WITH PLASTIC ABLATOR

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Abstract

IMPLOSION EXPERIMENTS INVOLVING CRYOGENIC FOAM TARGETS WITH PLASTIC ABLATOR.

A cryogenic target fabrication system has been developed for an implosion experiment using the GEKKO XII glass laser system. Deuterium gas of up to 130 atm ($= 1.3 \times 10^7$ Pa) can be filled into a plastic capsule in situ at the experimental target chamber. A low density foam shell with a plastic ablator has been developed, which provides a uniform spherical fuel layer even in the liquid phase. The mass density of the foam layer is as low as 90 mg/cm$^3$, and the cell size is less than 0.5 $\mu$m. A cryogenic deuterium target is sensitive to preheating because of hot electrons and a shock wave. The preheating could be suppressed to a great extent by a plastic layer coating. Cryogenic targets containing liquid or solid deuterium have been imploded with the use of 530 nm GEKKO XII laser light.

1. INTRODUCTION

In inertial confinement fusion (ICF) experiments remarkable achievement has been made in the last few years, such as reaching a compressed density of up to 200 [1] to 600 [2] times liquid density, which is close to the value required for igniting the fuel with moderate laser energy. In order to achieve further progress in ICF, several issues have to be studied or developed. The development of cryogenic (deuterium or deuterium–tritium) targets is certainly one of these, aiming at the achievement of both high temperature and density so that a considerable fraction of thermonuclear energy is released from the target. The present studies include both
technical and physical issues for direct drive laser fusion: (1) fabrication of fuel containers; (2) development of a cryogenic system adaptable to laser irradiation of the target; (3) coupling and energy transport; and (4) implosion performance.

Cryogenic targets were developed in the past by using glass microballoons [1, 3, 4]. The glass microballoons are no suitable containers of deuterium–tritium fuel because of their high average atomic number and their high mass density. A plastic shell is, however, one of the best candidates for an ICF fuel capsule because of its low atomic number and its low mass density. We have developed a cryogenic system that can be applied to plastic shells. The system has been employed at the experimental target chamber of GEKKO XII. Plastic shells with a low density foam layer are developed in which a solid and a foam layer act as ablator and sustainer of liquid or solid deuterium, respectively.

Energy transport focusing on the preheating and implosion performance of a cryogenic deuterium spherical target has been studied by using green light of the GEKKO XII glass laser system. A solid deuterium target sustained in a planar plastic foam is shown to be sensitive to preheating by high energy electrons and shock waves. The preheating is, however, found to be reduced by coating the deuterium foam layer with a solid plastic layer. In the implosion experiments, $10^8$ D–D neutrons and a density–radius product of the deuterium fuel, $\rho R = 8$ mg/cm$^2$, have been observed for liquid deuterium foam targets by irradiating 6–12 kJ laser energy at 1.6–1.8 ns pulse width. Three states (solid, liquid, gas) of the fuel were effectively controlled in the implosion experiments.

2. CRYOGENIC SYSTEM AND TARGET

A polystyrene spherical shell [5] and a low density plastic foam shell with a solid plastic layer have been developed for use as a cryogenic deuterium (or deuterium–tritium) fuel capsule. A low density plastic foam shell with a solid plastic layer has been fabricated by coating the foam shell with a solid plastic layer. Plastic foam shells are produced by MMA–TMPT copolymerization using the water–oil–water (W/O/W) emulsion method [6], where MMA and TMPT are short for methylmethacrylate and trimethylpropane trimethacrylate, respectively. The organic materials MMA and TMPT are solved in oil with a polymerization initiator. Overcoating the foam shell with a solid plastic layer is performed by means of interfacial polycondensation of hydroxyethyl cellulose or poly (p-vinyl phenol) and isophthaloyl chloride [7]. In order to have the chemical reaction just on the surface of foam shells, the oil–water (O/W) emulsion method is used where O is the foam shell with the isophthaloyl chloride solution and W is the water solution of hydroxyethyl cellulose with NaOH as an acceptor of the acid produced by the chemical reaction.

An electron scanning microscope (ESM) picture and an interference pattern of the shell are shown in Fig. 1. The shell quality can be studied by normal optical methods. The structure of the plastic foam is a three dimensional network, and the
cell or mesh size is less than 0.5 μm. The foam density could be controlled to be as low as 90 mg/cm³. The surface smoothness as estimated from the ESM picture is less than 0.2 μm. The sphericity and uniformity are better than 98%. A small amount of Si is doped into the foam for X ray spectroscopy. The average atomic number $<Z>$ is 3.64, and $<Z^2>$ is 23.2.

FIG. 1. (a) Electron scanning microscope picture of the cross-section of a plastic foam shell with solid layer; (b) interference pattern of the dry shell obtained by He-Ne laser.

FIG. 2. Schematic illustration of cryogenic target preparation and following laser irradiation: (a) After filling gas into the plastic shell at room temperature, the cryostat shroud with the optical windows is cooled by liquid He and the deuterium is replaced by helium gas. A uniform solid or liquid layer is fabricated with the aid of the uniform foam layer. (b) The pressure vessel inside the shroud is lowered so as to be ready for laser firing. (c) The shroud is pulled down about 70 ms before the laser. The laser irradiates the target before the shroud reaches the bottom of the chamber.
The cryostat can fill D\(_2\) gas of up to 130 atm (\(\approx 1.3 \times 10^7\) Pa) at room temperature [8] into the plastic shell at the laser firing position. The gas is filled by diffusion with a small pressure difference being kept across the shell. The system consists of two cryostats installed at the top and the bottom of the GEKKO XII target chamber, whose diameter is 1.8 m. The main function of the upper cryostat is to supply the plastic shell to the laser firing position and that of the lower cryostat is to fill the target with fuel gas and to cool it down to freezing temperature. Figure 2 shows schematically how to prepare the cryogenic target and how to shoot it. The target position and the uniformity of the fuel layer are measured by optical target position monitors with He–Ne laser interferometers. The temperature of the cryogenic targets and the vapour pressure at laser irradiation are calculated theoretically by using the temperature of the target just before taking off the cryogenic shroud and the temperature of the target chamber which determines the radiation temperature around the target without the cryogenic shroud.

3. PREHEATING OF CRYOGENIC TARGET

The cryogenic target is sensitive to preheating by both hot electrons and shock wave since it has low heat capacity, low average atomic number Z and low mass density. These characteristics may require careful selection of experimental conditions in order not to excite a high level of corona instabilities. Planar solid deuterium targets have been irradiated by 530 nm, 1 ns Gaussian laser pulses in order to study preheating. Two beams of GEKKO XII irradiate the target from angles of ±32° to

![FIG. 3. Temperature at the rear side of the target versus target areal density \(\rho\Delta r\). The solid plastic layer of CMC reduces the preheat temperature of the target.](image-url)
the target normal. The laser intensity is $3 \times 10^{14}$ W/cm$^2$, and the spot size is $0.6 \times 0.7$ mm$^2$. Solid deuterium is sustained in a 50–100 mg/cm$^3$ polystyrene foam plate [9]. The preheat temperatures of the rear target surface are estimated by using temporally resolved UV emission (wavelength 250–400 nm) on the assumption of black body radiation.

Figure 3 shows the dependence of the preheat temperature on the target areal mass density, $\rho \Delta r$, obtained at a laser intensity of $3 \times 10^{14}$ W/cm$^2$. The preheating results from the hot electrons and the shock wave. The preheat temperature due to the hot electrons decreases exponentially with increasing target thickness whereas that due to the shock wave is almost independent of the target thickness. The preheat temperature due to the hot electrons is 15 eV at $\rho \Delta r = 1.5$ mg/cm$^2$, corresponding to 70 $\mu$m thick solid deuterium suspended in the polystyrene foam plate; the total preheat level is 35 eV. The preheat temperature decreases considerably for carboxylmethyl-cellulose (CMC) targets where the target is overcoated with 5 $\mu$m sodium CMC at the laser irradiation side. On the basis of these experimental results the plastic coating on cryogenic fuel is found to be very important in suppressing the preheating due to both hot electrons and shock wave, at least for 530 nm laser.

4. IMPLOSION PERFORMANCE

Deuterium cryogenic foam targets with a plastic ablator have been imploded by the green laser from the GEKKO XII twelve beam system. The typical laser energy is 8 kJ on the target. The random phase plates (RPPs) are inserted in front of the focusing optics. The focusing conditions are adjusted for obtaining the best implosion performance. The energy balance among the twelve beams is as good as 2% standard deviation. The diameters of the targets are $\sim 600 \mu$m. The thickness of the solid plastic layer is 4 $\mu$m, which is chosen to be equal to the ablated thickness before maximum compression. The thickness of the foam layer is $\sim 10 \mu$m. Just enough deuterium gas is filled in to saturate the foam layer in liquid or solid state. The uniformity of the deuterium layer with foam is better than 98%. Three states of $D_2$ — solid, liquid and gas — have been imploded.

The neutron yield is measured by using a silver activation counter and plastic scintillators. The density–radius product $\rho R$ of the deuterium fuel is estimated from the ratio of the secondary D–T neutron yield and the primary D–D neutron yield, i.e. by the secondary reaction method. The energy spectrum of the primary D–D protons is also measured to estimate $\rho R$. For liquid and solid fuel targets, $\rho R$ is estimated under the following assumptions: Some levels of the deuterium gas corresponding to the vapour pressure are in the liquid and solid targets, and the shock wave is driven from the converging shell. The compressed core plasma consists of the hot plasma which is initially in the gas state, and the cold dense plasma surrounding the hot plasma, which is initially liquid. The primary D–D neutrons are produced mainly in the hot plasma, while the secondary D–T neutrons are generated in the cold plasma.
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since the secondary reaction rate is roughly proportional to the inverse square root of the electron temperature and also most deuterium ions are in the cold plasma. Dynamics and symmetry of the imploding shells are measured by using X ray streak cameras and X ray framing cameras.

The trajectories of the ablation front measured by the X ray streak camera agree with those of the 1-D simulations until the shock wave collides with the liquid or solid shell fuel. After that, the position of the ablation front no longer changes. The convergence ratios estimated from the sizes obtained by the X ray streak camera images lie between two and three and do not depend on the target.

Figure 4 shows the neutron yield $N_y$ and the density–radius product $\rho R$ normalized with the 1-D simulation results by HISHO, in which the effects of hot electron preheating are taken into account from the planar target experiments. The data at the convergence ratios of 5, 8–12 and 13 correspond to the gas, liquid and solid state targets, respectively. The observed D–D neutron yields are $(1-3) \times 10^6$ for the solid state targets and $3 \times 10^7$–$2 \times 10^8$ for the gas and liquid targets. The observed ion temperature is 1–2 keV, measured by time of flight neutron spectroscopy. The normalized $N_y$ values are $10^{-1}$, $\sim 5 \times 10^{-3}$ and $10^{-2}$ for the gas, liquid and solid targets, respectively. The observed yields for the three fuel target states are consistent with those in the simulation when the shock wave collides at the target centre for the second time although the normalized values are different for each target. The observed $\rho R$ values of the deuterium ions are 5–8 mg/cm$^2$ for the gas and liquid targets. They are 80% of the simulation results at the time when the neutron yield

![Graph showing implosion performance of cryogenic targets. Neutron yield ($\bullet$) and $\rho R$ ($\star$) normalized by 1-D simulation results. $\Delta$ designates the $\rho R$ value normalized by 1-D simulation results at the time of neutron yield, corresponding to the neutrons observed.](image-url)
becomes equal to the observed values. The $\rho R$ values estimated from the compressed core size are slightly lower than those found by the secondary reaction method.

One possible scenario explaining the results is as follows:

According to the simulation of the spherical shell target with gas fuel inside, the gas fuel temperature becomes high enough to produce thermonuclear neutrons by shock heating, but the shell remains at low temperature. If the Rayleigh–Taylor instability in the deceleration phase occurs when the shock having bounced from the target centre hits the shell, some fractions of the shell fall freely into the hot plasma with a speed prevailing just before the shock hits the shell. Then the temperature of the mixing region becomes lower than that required to produce neutrons. After the shock has initiated this mixing region, the neutron production region in the hot gas is squashed by this cold region. Then the neutron emission rate may decrease to a negligible level. The $\rho R$ value of the shell at this time is that measured by the secondary reaction method. This value could be slightly smaller than the 1-D simulation value and slightly larger than the value estimated by using the compressed core size obtained from the streaked X ray image.

5. SUMMARY

Implosion techniques involving cryogenic targets with plastic capsules have been established. A low density plastic foam shell with plastic coating has been developed and successfully introduced into the implosion experiments. A cryostat creating a cryogenic target with plastic capsules has been developed for use in the GEKKO XII system. Three states (solid, liquid, gas) of the fuel are effectively controlled for the implosion. The cryogenic targets are found to be sensitive to preheating due to hot electrons and shock waves. Solid plastic layer coating can suppress preheating. The observed neutron yield and $\rho R$ values are consistent with the simulations when the shock waves collide at the target centre for the second time.

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INERTIAL CONFINEMENT FUSION RESEARCH AT THE SERC CENTRAL LASER FACILITY, UNITED KINGDOM

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Abstract

INERTIAL CONFINEMENT FUSION RESEARCH AT THE SERC CENTRAL LASER FACILITY, UNITED KINGDOM.

The paper reviews experimental and theoretical work relevant to inertial confinement fusion which was carried out in connection with the high power laser programme at the SERC Central Laser Facility. The main effort of recent research concentrated on studies with improved laser illumination uniformity generated by various optical smoothing techniques. It was observed that non-uniformities are produced at the ablation front caused by the initial laser imprint on the cold target surface at the beginning of the laser pulse even when optically smoothed laser beams were used whereas uniform plasmas were generated with soft X ray irradiation. To combat the initial imprint problem, a novel indirect/direct drive scheme is proposed in which the target is irradiated with intense soft X ray radiation ahead of the optical laser pulse. The soft X ray radiation is generated by irradiating a converter foil with optical laser light. The interaction, transport and hydrodynamic efficiency of intense soft X ray pulses are studied in thin foil targets. The propagation of a supersonic ionization front produced by soft X ray heating is investigated in low density foam targets. The effects of various beam smoothing techniques on the Rayleigh–Taylor instability are studied, and the first experimental observations of mode coagulation are made in which short wavelength modes couple into long wavelength ones. The growth of the Rayleigh–Taylor instability in the deceleration phase is calculated with a 3-D hydrodynamics code.

Uniformity of illumination to a very high degree (< 1%) is essential for the direct drive laser fusion scheme. Small perturbations on the laser beams, for example, may grow because of self-focusing and filamentation, resulting in localized heating of the target and causing non-uniformities in the ablation pressure that drives the implosion. The non-uniformities may also be a seed for hydrodynamic instabilities such as the Rayleigh–Taylor instability which can lead to a breakup of the imploding target shell during the acceleration phase at the ablation surface and/or during the deceleration phase at the inner surface of the fuel shell. For these reasons various
beam smoothing techniques were recently developed to improve the spatial intensity profile of the laser beams. We report here on some physics issues which are critical for the direct drive approach. In particular, the effects of various laser beam smoothing techniques on several physical processes were investigated. It was observed that the initial laser imprint on the cold target surface at the beginning of the pulse produces considerable density perturbations at the ablation surface. The find a solution to this startup phase is one of the key issues for the direct drive approach. A novel indirect/direct drive scheme is proposed as a possible solution.

The levels of parametric instabilities including laser beam filamentation, stimulated Raman scattering (SRS) and stimulated Brillouin scattering (SBS) were studied in large underdense preformed plasmas for either coherent laser light or optical beams smoothed by random phase plate (RPP) arrays or the induced spatial incoherence (ISI) technique. Millimetre sized cylindrical underdense high temperature plasmas were produced by irradiating thin foil targets with a number of laser beams in a line focus geometry. A separate laser beam delayed between 1 and 3 ns interacted axially with the preformed plasma [1, 2]. The focal spot profiles of the interaction pulse were calculated with an interference code and compared to either time integrated or framed equivalent focal spot images. Direct experimental evidence was obtained that SRS is predominantly generated in laser filaments [3]. The ‘anomalous’ SRS spectral features so commonly reported in time resolved studies are shown to be characteristic emission of SRS from filaments. The effectiveness of the ISI in suppressing SBS was to be strongly dependent on the laser wavelength and intensity for the interaction regime investigated. Significantly higher levels of SBS were recorded for infrared ($\lambda_0 = 1.05 \mu m$) as opposed to green ($\lambda_0 = 0.53 \mu m$) laser wavelength interactions. A correlation can be made between the SBS level and the strength of the filamentation instability. In addition, the first observations of the occurrence of thermal whole beam self-focusing with ISI laser light were obtained [4]. Side on X ray pinhole imaging with a multiframe camera was used to record the propagation characteristics of the interaction beam through the preformed plasma. The self-focusing growth length was measured directly.

The uniformity of the overdense plasma of laser irradiated targets was investigated by using a novel time resolved X ray imaging technique with submicron spatial resolution [5, 6]. Two-dimensional spatially resolved images show that laser beam non-uniformities imprint themselves onto the cold target surface at the beginning of the laser pulse, generating considerable density perturbations which persist throughout and after the laser pulse with no evidence of smoothing [7]. This imprint phenomenon is interpreted in terms of poor thermal smoothing under the conditions prevailing at the beginning of the laser pulse. On the other hand, it was observed that the plasma blowoff is uniform when the targets are irradiated with intense soft X ray pulses. The plasma production was characterized interferometrically with the use of an optical probe beam with a wavelength of 350 nm. Electron density profiles were obtained for soft X ray heated thin wire targets during and after the pulse, allowing the distance of the critical density for an infrared laser beam ($n_c$...
The efficiency of thermal smoothing was investigated by interacting an infrared laser beam with a uniform plasma produced by an X ray pulse. A periodic structure was imposed on the beam with wavelengths between 15 and 50 μm and an intensity modulation of 2:1. It was observed, by using side on soft X ray radiography, that the ablation surface was uniform within the sensitivity of the imaging system when the distance between critical and ablation surfaces was larger than the imposed periodic structure on the laser beam. The production of uniform plasmas with soft X rays and the process of thermal smoothing are utilized in the indirect/direct drive approach. The scheme was tested by irradiating thin foil targets with a soft X ray pulse followed by an SSD/RPP smoothed optical pulse. No breakup of the accelerating foil was observed which is in sharp contrast with results in which the foil was driven directly with either an SSD/RPP or an ISI/RPP optical pulse. In this case a periodic breakup with a wavelength consistent with the initial imprint structure was observed.

The interaction, heating and dynamics of low Z foil targets irradiated with intense, approximately Planckian, soft X ray pulses have been investigated. These studies are not only intrinsically interesting but are also important for the indirect/direct drive approach. The soft X ray pulses, generated from separate laser irradiated converters consisting of 1 μm thick CH which is overcoated with 750 Å of gold, were used to irradiate planar plastic foils. The global transport of soft X ray radiation through thin foil low atomic number targets was studied by using time resolved X ray ultraviolet (XUV) spectroscopy [8]. The X ray heating was investigated by measuring the temperature histories of chlorinated tracer layers buried at

![Graph](https://via.placeholder.com/150)

**FIG. 1.** Radiation hydrodynamic simulations showing that the ablation pressure strongly depends on the shape of the soft X ray pulse.
different depths in the targets. The temperature diagnostic was a novel time resolved XUV absorption spectroscopy technique using chlorine L-shell transitions. The temporal temperature profiles were reasonably well reproduced by radiation hydrodynamic simulations [9]. The dynamics of accelerated targets were diagnosed by using a high resolution soft X ray imaging system. A long pulse backliter was used, and the rear surface of the accelerating foil target was time resolved with an X ray streak camera. The probe wavelength was about 60 Å. Similar results were obtained on targets driven by optical smoothed laser radiation. The experimental results were simulated with a radiation hydrocode which showed good agreement [10]. In addition, thin foil targets were driven either with shaped soft X ray or optical pulses or a combination thereof. It was observed that the efficiency of the soft X ray drive was significantly reduced later in the shaped pulse, owing to absorption of radiation in the plasma blowoff. These results are in distinct contrast to similar targets which were driven with shaped optical laser radiation. Figure 1 presents hydrodynamic simulations showing the ablation pressure for targets driven by a single 1 ns (full width at half maximum, FWHM) and a shaped soft X ray pulse, respectively. Evidently, the ablation pressure generated by the latter part of the shaped soft X ray pulse is significantly lower than the pressure produced by a single pulse with a similar incident X ray flux.

The propagation of an ionization front produced by soft X ray heating was observed in cylindrical foam targets with a density between 30 and 50 mg/cm$^3$. The position of the ionization front was observed either in absorption or emission at various times during and after the pulse using a 2-D soft X ray imaging technique with a probe wave of about 50 Å. In absorption, the heated material becomes transparent to the probe beam whereas the cold material ahead of the front is opaque. The use of low density foam attached to the outside of the fusion target may be another solution to the initial laser imprint problem.

Growth rates of the Rayleigh–Taylor instability were measured in thin foil targets with imposed sinusoidal modulations irradiated by optically smoothed laser beams. A hybrid optical smoothing scheme utilizing ISI and RPP was used. The enhancement in the modulation depth during acceleration was observed with time resolved transmission radiography using a soft X ray backlighting source. The wavenumber dependence and non-linearity of the Rayleigh–Taylor growth were investigated by using a range of modulation periodicities and depths. The measurements were compared with 2-D hydrocode simulations [11, 12]. The effects on the Rayleigh–Taylor instability and the secular target breakup were studied by using different illumination schemes including coherent, ISI/RPP smoothed green and soft X ray radiation. Side on images with a probe wavelength of about 50 Å were obtained for various targets with modulation wavelengths between 20 and 50 µm. Figure 2 shows a series of images of targets which were irradiated with the three different illumination schemes. Targets with a 50 µm periodicity were used for this study. As can be seen the target driven with coherent laser beams shows both Rayleigh–Taylor growth and target breakup caused by non-uniformities in the laser
Modulation period = 50 pm
Probe wavelength = 50Å
Modulation depth = 2.7 μm
Target thickness = 15 μm

FIG. 2. Side on soft X ray radiographs of premodulated CH foil targets driven by coherent, ISI/RPP smoothed laser beams and soft X ray radiation. The frames were recorded 3 ns after the start of the driving 1 ns pulse.

illumination. The target driven by ISI/RPP smoothed irradiation was accelerated more uniformly, showing bubble and spike formation. It moved furthest at the centre of the beam where the laser intensity was highest. The third image taken on a target which was driven entirely uniformly by an intense soft X ray pulse clearly shows the target breakup caused solely by the Rayleigh–Taylor instability. Again bubble and spike formation is observed. In addition, the first measurements of the Rayleigh–Taylor instability at short wavelength were obtained. Thin foil targets with a modulation periodicity of 2.6 μm and an initial amplitude of 0.2 μm were driven by ISI/RPP irradiation. Side on images recorded 3 ns after the start of a 2 ns laser pulse show that the target is broken up with wavelengths between 5 and 50 μm. The longer wavelengths are seen on the part of the target which was driven furthest caused by the most intense part of the laser beam (which had a Gaussian spatial profile). These observations indicate that the short wavelength modes couple into longer perturbations.

Three dimensional simulations of the Rayleigh–Taylor instability in the deceleration phase of ICF implosions have shown that (a) non-linear growth is about 25% faster in 3-D than in 2-D; (b) growth is faster for thin shells; and (c) shell integrity is increased by driving the target more strongly under the beams than at the beam intersections. Two topologically non-linear modes are apparent in 3-D, the spike valley and the bubble ridge, their occurrence depending on the initial conditions [13].
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DISCUSSION

L.J. DHARESHWAR: In earlier work, you reported on plasma jets which had no relevance to laser beam uniformity. Have you done any recent conclusive experiments on the role these jets play in the ablative acceleration physics?

O. WILLI: We have seen that the initial laser non-uniformities of the laser beam imprint themselves on the cold target surface. We have not yet carried out experiments which show that the plasma jets occur at the ablation surface.

L.J. DHARESHWAR: In our work, we have observed that plasma jetting is pronounced for gold targets. In your opinion, would these plasma jets have a detrimental effect on indirect drive schemes where gold converters are being used?

O. WILLI: We have only used converter foils to generate soft X rays. We have not yet investigated the gold foil, and we have no experimental data on plasma jets in the indirect drive scheme.
STUDY OF ABLATION PROFILE SMOOTHING AND STABILITY OF LASER DRIVEN HIGH Z DOPED PLASTIC FOIL TARGETS

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Abstract

STUDY OF ABLATION PROFILE SMOOTHING AND STABILITY OF LASER DRIVEN HIGH Z DOPED PLASTIC FOIL TARGETS.

The hydrodynamics of plastic foil targets doped with a high Z material such as tungsten have been studied using a time resolved optical shadowgraphy technique. It has been observed that these targets not only exhibit a higher ablation pressure at a given laser intensity but also give rise to a complete smoothing of the ablation profile when ablatively accelerated by a non-uniform laser profile. These tungsten doped plastic foils have shown remarkably stable target motion, with no foil breakup even up to 12-14 ns after the peak of the laser pulse. It is concluded that an enhanced lateral energy transport due to X rays emitted by the high Z atoms in the plastic targets leads to a complete ablation pressure profile smoothing. Experiments were performed with a neodymium glass laser pulse of 5 ns duration and the intensity on target was in the range $10^{12}-10^{13}$ W/cm$^2$. The optical probe was a second harmonic (wavelength 0.53 $\mu$m) pulse of 1 ns duration, derived from the main laser pulse.

1. INTRODUCTION

A crucial problem in direct drive inertial confinement schemes involves the achievement of an ablation pressure uniformity of better than a few per cent [1]. In the direct drive scheme, asymmetry of pressure at the ablation surface is mainly due to non-uniformity of the laser irradiation. Lateral diffusion is classically known to reduce irradiation non-uniformity. However, the extent of this smoothing is dependent on the distance between the energy deposition and the ablation surface and the non-uniformity scale length and laser wavelength [2]. Schemes have been evolved whereby the irradiation symmetry could be improved by induced spatial incoherence [3] and the use of random phase plates [4]. Earlier work [5, 6] has shown that gold
coated plastic foil targets result in uniform hydrodynamic motion even in the presence of severe non-uniformities in the laser intensity profile. In gold coated targets smoothing of the ablation profile has been explained to take place owing to intense X ray emission in the region between the absorption and the ablation surface. The radiative energy transport helps in the attainment of deep penetration of the thermal front into the target. However, in gold coated targets, firstly, an excessive forward X radiation loss \([7, 8]\) from the high Z blow-off plasma results in a lower ablation pressure \([6]\). Secondly, if the thickness of the gold layer exceeds the ablation thickness, a poorer smoothing results owing to self-absorption of X rays in the high Z material itself \([5]\), thus effectively reducing the penetration of the radiation induced thermal front. It was therefore decided to work on a target scheme which could overcome these drawbacks. This target scheme was devised to provide adequate smoothing of the ablation profile through X ray assisted lateral transport, without causing a reduction in ablation pressure. This was achieved by doping a low Z plastic target foil with a high Z material such as tungsten. The level of dopant concentration was adjusted to be just sufficient to provide smoothing without resulting in high X ray emission losses and consequently a lower ablation pressure. The X radiation losses in this target would be lower, since the average target atomic number is small compared with that of a pure high Z target. In fact, enhanced radiative transport in comparison to electron conduction has been reported in tungsten doped glass target experiments \([9]\). A high Z seeded plastic pusher known as TaCHO (tantalum doped plastic) has also been proposed in some target designs, as a preheat shield \([10]\).

2. EXPERIMENTS

These experiments were performed with a 17 \(\mu m\) thick plastic (Zapon, \(C_6H_7O_1N_3\)) target doped 22\% by weight with tungsten powder of 3 \(\mu m\) particle size. The areal mass density of the tungsten doped plastic target was \(24 \times 10^{-4}\ g/cm^2\). The performance of these doped targets was compared with that of an undoped plastic target of identical areal mass density. These foil targets were ablatively accelerated by a 20 J, 5 ns neodymium glass laser beam focused to an intensity in the range \(10^{12}-10^{13}\ W/cm^2\) over a spot diameter of about 160 \(\mu m\) on the target. The time resolved hydrodynamic motion of the target foil and target foil velocity were measured by an optical shadowgraphy technique \([11]\). Optical shadowgrams were recorded using a 1 ns duration, 0.53 \(\mu m\) (second harmonic) probe pulse derived by splitting a fraction of the main beam. The pulse duration was reduced from 3.5 ns to 1 ns using a pulse compression technique described elsewhere \([12]\). In order to observe the target foil motion at various instants of time, a variable optical delay was used in the path of the probe pulse. The experimental set-up is shown schematically in Fig. 1. The ion expansion velocity was measured using a Langmuir probe placed at a distance of 15 cm from the target, making an angle of 45° with respect to the normal to the target.
3. RESULTS AND DISCUSSION

Using a rocket model for plasma ablation [13], ablation pressure is given by the expression \( P = \rho t V / \tau \), where \( \rho \), \( t \) and \( V \) are respectively the target foil density, thickness and velocity, and \( \tau \) is the laser pulse duration. The scaling of the ablation pressure with absorbed laser intensity is shown in Fig. 2. It is observed that there is almost 30% improvement in ablation pressure for a doped plastic target compared with an undoped target with an identical areal mass density. This improvement in ablation pressure is attributed to a corresponding decrease in ion expansion velocity for a tungsten doped plastic target. At an absorbed laser intensity of \( 5 \times 10^{12} \text{ W/cm}^2 \), the average ion expansion velocities measured for doped and undoped plastic targets were \( 1.5 \times 10^7 \text{ cm/s} \) and \( 2 \times 10^7 \text{ cm/s} \), respectively. The lower expansion velocity for the doped target is as expected, owing to the higher average ion mass. This results in a higher ablation pressure \( P \), since for the rocket model, \( P = 2l_a / U \), where \( l_a \) and \( U \) are the absorbed laser intensity and the expansion velocity, respectively. This observation thus indicates that there is no significant lowering of absorbed laser intensity due to radiation losses. Measurements of relative X ray emission losses from undoped, tungsten doped and pure tungsten foils were performed using two different diagnostics — an X ray calorimeter [14] and an X ray PIN diode with a 25 \( \mu \text{m} \) beryllium filter [15]. At an incident laser intensity of \( 8 \times 10^{12} \text{ W/cm}^2 \), it was observed that X ray emissions from pure tungsten and tungsten doped plastic were, respectively, 10 and 2 times higher than that from a pure

![FIG. 2. Scaling of ablation pressure with absorbed laser intensity. Open circles: tungsten doped plastic targets; solid circles: undoped plastic targets.](image-url)
plastic target. This result indicates that there is a substantial reduction in X-ray emission in the case of a tungsten doped plastic target as compared with a pure tungsten target.

Ablation smoothing in tungsten doped plastic foil targets was studied by recording target foil motion when the incident intensity profile was deliberately made non-uniform, as shown in Fig. 3(a). The shadowgram of an undoped plastic target foil at an average incident laser intensity of $5 \times 10^{12}$ W/cm$^2$ and recorded 4 ns after the peak of the main laser pulse is shown in Fig. 3(b). A clear double lobe structure of

![Diagram](image)

**FIG. 3.** Ablation pressure profile smoothing. (a) Spatial intensity profile of the incident laser beam; (b) target motion in undoped target; (c) uniform motion in tungsten doped target showing smoothing of the ablation pressure profile. Incident laser intensity: $5 \times 10^{12}$ W/cm$^2$, shadowgrams recorded 4 ns after the peak of the laser pulse.
FIG. 4. Shadowgrams recorded for: (a) undoped plastic foil showing complete breakup; (b) tungsten doped plastic foil exhibiting very uniform motion with no foil breakup. Incident laser intensity: $5 \times 10^{12}$ W/cm$^2$, shadowgrams recorded 8 ns after the peak of the laser pulse; (c) tungsten doped plastic foil recorded 14 ns after the peak of the laser pulse at a laser intensity of $1.5 \times 10^{13}$ W/cm$^2$. 
the rear foil ejecta shows the non-uniformity of the ablation pressure profile due to the non-uniform laser intensity profile. The shadowgram of a tungsten doped foil irradiated at an identical laser intensity and recorded 4 ns after the peak of the laser pulse is shown in Fig. 3(c). Here it is observed that the profile of the rear foil ejecta is uniform, indicating a complete smoothing of the ablation pressure profile.

Shadowgrams recorded 8 ns and 12 ns after the peak of the laser pulse are shown in Fig. 4, in which (a) shows the shadowgram of an undoped plastic foil target irradiated with a laser intensity of $5 \times 10^{12}$ W/cm$^2$. This shadowgram was recorded 8 ns after the peak of the laser pulse and shows a near complete breakup of the foil. However, the tungsten doped foils at an identical laser intensity and recorded 8 ns after the laser peak exhibit a remarkably uniform foil motion, as shown in (b). The shadowgram in (c) was recorded 12 ns after the peak of the laser pulse for a doped foil target at an incident laser intensity of $1.5 \times 10^{13}$ W/cm$^2$. Even at this high laser intensity, the foil motion is seen to be very uniform.

It is well known that the high atomic mass number targets such as tungsten emit copious soft X rays which lie typically in the 0.1–1.5 keV range. Components of the plastic, that is, carbon, nitrogen and oxygen, have k absorption edges in the range 0.3–0.6 keV [16]. Thus the X rays emitted by tungsten atoms within the range 50–60 Å are strongly absorbed by the surrounding plastic. Owing to the isotropic nature of these X rays, they tend to heat and ablate even those regions on the target which are not directly heated by the laser, enhancing lateral energy transport. In addition, owing to the deeper penetration of these X rays, the effective length of the conduction zone is enhanced, providing the further advantage of lateral transport smoothing. The breakup of the undoped plastic target shown in Fig. 4(a) can be attributed to the inherent small scale intensity non-uniformities in the laser beam profile. At moderate laser intensity, such small scale non-uniformities are not smoothed out by thermal conduction and therefore could lead to the growth of fluid instabilities and ultimately to target breakup. However, an enhanced lateral energy transport due to X rays could result in these non-uniformities being completely smoothed out and lead to an extremely stable target motion, as seen in the photographs of Fig. 4. Tungsten doped targets could be attractive in the ICF reactor grade experiments where stable motion of the targets has to be achieved for several nanoseconds.

In conclusion, it can be said that plastic targets doped with high atomic number elements such as tungsten or gold could offer advantages over pure plastic targets. An enhanced ablation smoothing, coupled with a higher ablation pressure, makes them attractive as targets for inertial confinement schemes.

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SYMMETRY AND STABILITY ISSUES IN TARGET DESIGN FOR HEAVY ION FUSION

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Abstract

SYMMETRY AND STABILITY ISSUES IN TARGET DESIGN FOR HEAVY ION FUSION.

Results on several topics concerning the symmetry and the stability of the implosion of heavy ion inertial confinement fusion (ICF) targets are reported. The Rayleigh-Taylor instability of some direct drive targets, during inward acceleration, is studied by means of 2-D radiation hydrodynamics simulations; it turns out to be unavoidable. Radiation generation, confinement and symmetrization in some hohlraum targets for indirect drive ICF, including cylindrical radiation converters, is studied by 2-D and 3-D codes. It appears that at least six converters are needed to provide adequate radiation symmetrization and efficient radiation coupling to the fusion capsule. Preliminary results of studying a new concept of radiation driven target, including a spherically symmetric converter, are presented. In the final section some results are given on high resolution 2-D studies of the turbulent mixing induced by Rayleigh-Taylor instability.

1. INTRODUCTION

An assessment of accelerator and target requirements for energy production by heavy ion (h.i.) inertial confinement fusion (ICF) is under way, following recent advances in accelerator conceptual design [1]. In this framework, several topics concerning the implosion symmetry and stability of h.i. ICF targets have been studied numerically and analytically. Indeed, the minimum driver energy required for ignition, and for sufficiently high gain, depends crucially on symmetry and stability constraints, as is also evidenced by simple models [2].

While numerical simulation studies ([2, 3] and references therein) have given semi-quantitative indications on the long scale properties of irradiation required by some reactor size targets with moderate hot spot convergence, the problems of the stability of direct drive h.i. targets and those of the generation of an adequately symmetric radiation field in indirect drive h.i. targets have received little attention so far. These topics are dealt with, respectively, in Sections 2 and 3 of this paper. It turns out that the direct drive targets studied are unstable to short wavelength

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perturbations. On the other hand, simple, two side, axially symmetric irradiation systems are not adequate to radiation symmetrization in the hohlraums considered so far (a fully 3-D, six spot target is required instead). These findings have motivated the study of a new indirect drive target (Section 4), including a spherically symmetric converter, in order to relax the focusing and pointing requirements.

Some results on turbulent mixing in Cartesian geometry are presented in Section 5.

The hydrodynamics simulations reported in this work have been performed by means of the three-temperature codes IMPLO-upgraded (1-D) and DUED (2-D) [4].

2. STABILITY OF DIRECT DRIVE TARGETS

Recently, it has been shown that some direct drive h.i. targets are subject, during acceleration, to violent Rayleigh–Taylor instabilities at the absorber–shield

![Diagram of target structure and radius versus time flow chart, ion beam power versus time, and instability of the acceleration of the planar analogue of the above target. Isodensity contours are plotted at several times.](image)

**FIG. 1.** Direct drive target with gain $G \equiv 100$. (a) Target structure and radius versus time flow chart; (b) ion beam power (4 GeV Bi) versus time; (c) instability of the acceleration of the planar analogue of the above target (here Z takes the place of R in spherical geometry; the ion beam comes from the right hand side). Isodensity contours are plotted at several times. (1) $\rho = 3 \, \text{g/cm}^3$; (2) $\rho = 1 \, \text{g/cm}^3$; (3) $\rho = 0.6 \, \text{g/cm}^3$; (4) $\rho = 0.4 \, \text{g/cm}^3$; (5) $\rho = 0.2 \, \text{g/cm}^3$.}
interface, even in the case where the absorber and the shield have the same density (about ten times higher than the fuel density) [5]. This has prompted an analogous study, referring to the target shown in Fig. 1(a) [2], which does not include any high density shield and in which the density of the absorber is close to that of the fuel. For simplicity, the study has been performed by using the planar analogue method [5]. The simulations show that this target is also subject to a Rayleigh–Taylor type instability, less violent than in the case studied in Ref. [6], but nevertheless posing a threat to the integrity of the fuel layer. Indeed, despite the absence of initial discontinuities, density and pressure gradients favourable to the development of the instability are inevitably produced, and stabilization mechanisms are not effective. Both steady beam non-uniformities and target defects can seed the instability. The evolution of a typical case, with the initial perturbation in the mass distribution and with a wavelength of the order of the initial thickness of the fuel layer, is shown in Fig. 1(c).

3. RADIATION SYMMETRIZATION AND COUPLING IN HOHLRAUM TARGETS

Radiation symmetrization has been studied by using a 3-D static model [6], and with reference to hohlraum targets (see Fig. 2(a)) consisting of a spherical gold casing, containing, at its centre, the fusion capsule (with a carbon ablator layer) and also some cylindrical generators of radiation (converters) [6]. The area ratio a (casing to capsule area) of the configurations considered is limited to a < 10–15, in order to ensure good energy coupling [7].

![FIG. 2. (a) Cross-section of a typical hohlraum target considered in the symmetrization study; (b) asymmetry of X ray deposition on the capsule (at time t = 5 ns after the beginning of constant power radiation emission from the converters) versus length L of the converters, for targets with cavity radius $R_2 = 1.16$ cm, capsule radius $R_1 = 0.367$ cm and converter radius $r_0 = 0.1$ cm.](image)
It is found that simple hohlraums with two converters (allowing for two side, axially symmetric irradiation) and $a = 10-20$ cannot achieve the uniformity regarded as necessary for ICF applications (rms asymmetry well below 2\% [2, 3]); six or more, thin, short converters are instead needed, as is shown in Fig. 2(b); the results for thicker converters are somewhat worse. In the two converter case, replacing the spherical casing with a cylindrical or an ellipsoidal one does not lead to any improvement.

The conditions for the efficient generation of radiation by cylinders heated by beams of heavy ions (such as shown in Fig. 2(a)) have also been studied analytically and by 2-D simulations. The results are detailed elsewhere [8].

4. INDIRECT DRIVE TARGETS WITH SPHERICALLY SYMMETRIC CONVERTER

Hohlraum targets such as those considered above imply accurate pointing and focusing of the h.i. beams on at least six spots of 1–1.5 mm radius. Such demanding requirements motivate the search for targets driven by radiation (to ensure adequate ablative stabilization), but with a spherical converter, thus allowing a large focal spot and relaxed target positioning requirements [1]. It would also be desirable to use a temporally unshaped beam pulse (‘box’ pulse).

A preliminary concept of such a target (similar to a concept developed by Basko [9]) is shown in Fig. 3. The ions are stopped in the tamper and in the outer portion of the absorber–converter plastic layer. The radiation is then transported up

FIG. 3. Structure and radius versus time flow chart of the radiation driven target discussed in Section 4. The couples of numbers in parentheses indicate, for each material, the mass density (in g/cm$^3$) and the total mass (in mg), respectively. (We note that gold can be replaced with other suitable high-Z elements).
to the ablative surface by a supersonic heat wave (SHW). The target parameters are chosen, according to the theory developed in Ref. [10], in such a way that the transit time (from the edge of the absorbing layer to the ablative surface) $t_i$ of the SHW is shorter than the heating time $t_h$ of the converter, but $t_i > t_{IE}$, where $t_{IE}$ is the time at which the SHW overcomes the shock wave generated at the beginning of the interaction. The latter condition is essential to ensure thermal smoothing of the initial beam non-uniformity (notice that the absorber-converter is marginally optically thick while it is transparent in Basko's design [10]). One-dimensional, three-temperature simulations show that, when a box pulse is employed, a 'pusher' consisting of a layer of high density material, interposed between the fuel and the absorber, is necessary to achieve ignition (it also serves as a shield against high energy photons). According to 1-D, three-temperature computations, the target of Fig. 3 achieves a gain of $G = 50$, upon irradiation with 10 MJ of 8.6 GeV Bi ions, in a pulse with constant power of about 700 TW. The hot spot convergence is about 35.

Two-dimensional simulations performed with the code DUED [4], have, however, shown that the pusher-ablator interface is Rayleigh–Taylor unstable and, after having moved a distance of the order of 100 µm, already strongly distorted. This result indicates the need for envisaging a variant of the target above, not requiring the high density pusher. This goal could probably be achieved by employing a temporally shaped ion beam pulse and ions with somewhat lower energy. The fuel could be shielded from high energy photons by a layer of low-Z material, doped with high-Z atoms.

5. TURBULENT MIXING

The impossibility of resolving short wavelength phenomena in full target implosion simulations (where evidence is found for the Rayleigh–Taylor instability) has motivated a 2-D, Cartesian, high resolution study (with up to $10^5$ mesh points) of the instability of superposed perfect gases [4]. Of particular relevance to ICF is the case where a multiwavelength, small amplitude initial perturbation (in the short wavelength portion of the spectrum) is superposed on one or a few large amplitude, long wavelength modes. In this case, on the time-scale of interest to ICF, the short wavelength modes give rise to the usual self-similar mixing layer, which grows around the interface deformed by the independent evolution of the longer wavelength modes. In the self-similar mixing regime, nearly independent of the detailed initial conditions, the height of the mixed layer grows quadratically with time ($h \equiv 0.05 \sqrt{Ag} t^2$, where $h$ is the maximum penetration of the lighter fluid into the heavier one, $A$ is the Atwood number and $g$ is the acceleration), and the turbulent energy grows as the fourth power of time; in addition, it has been verified (by spectral analysis) that the wavelength $\lambda_d(t)$ of the 'dominant mode' roughly scales with time $t$ as $\lambda_d(t) \propto h \propto gt^2$. The modes involved in the turbulent mixing are, at any time $t$, those with $\lambda \leq \lambda_d(t)$. 
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EXPERIMENTAL INVESTIGATION
OF INTERACTION PROCESSES
AND HYDRODYNAMIC EFFICIENCIES
IN LASER DRIVEN IMPLOSIONS

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Abstract

EXPERIMENTAL INVESTIGATION OF INTERACTION PROCESSES AND HYDRODYNAMIC EFFICIENCIES IN LASER DRIVEN IMPLOSIONS.

The first part of the paper shows that spatial incoherence strongly inhibits non-linear laser-plasma interaction. Filamentation is reduced in agreement with theoretical prediction. Stimulated Brillouin scattering (SBS) and stimulated Raman scattering (SRS) instabilities are also strongly damped. In the second part, hydrodynamic efficiencies as good as 10% are shown to be achieved in implosion experiments made with 0.26 μm laser light. Spatial incoherence was investigated and did not improve neutron emission. The persistence of dominant geometrical modes associated with the illumination configuration could explain this result.

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The direct drive approach to laser inertial fusion presents significant advantages over the indirect drive scheme if the efficiency transfer between laser and core energies is larger. Direct drive could, however, raise some difficulties: preheat by fast electrons generated by non-linear laser–plasma interaction, non-uniformity of energy deposition resulting from beam filamentation or from target illumination by multiply coherent beams. We have investigated some of these problems.

In a long scale length plasma approaching reactor conditions, parametric instabilities can develop for coherent laser beams: stimulated Brillouin scattering (SBS) or stimulated Raman scattering (SRS), as well as filamentation. We have investigated the effect of spatial incoherence on these instabilities, using random phase plates (RPPs) [1]; these plates indeed provide spatial incoherence in addition to a better illumination uniformity. Long density scale length cylindrical plasmas were produced, irradiating a 0.6 μm thin plastic foil by one beam of the Nd glass laser of LULI, at 1.06 μm with a rectangular focal spot. An interaction beam at 0.53 μm was then focused along the plasma axis 600 ps later, with the intensity varying from $2 \times 10^{14}$ to $2 \times 10^{15}$ W/cm$^2$. At the time of interaction, the plasma density scale length along the axis was 300 μm, and the electron temperature 0.5 keV.

The SBS and SRS scattered light level was measured. The presence of RPPs strongly reduced the scattered light level by nearly two orders of magnitude as is shown in Fig. 1 for SRS. This result has been compared with the theoretical predic-

![Fig. 1. SRS conversion efficiency.](image-url)
tion. For SBS and SRS the angular spread of the wavenumber of the interaction beam (characterized by a lens aperture angle of $2\Delta \theta_0$) results in linear mismatches, $\Delta_1$ and $\Delta_2$, under the usual three-wave resonance condition. If the laser beam is incoherent enough for these mismatches $\Delta_\alpha$ to exceed the linear damping $\nu_\alpha$ of wave $\alpha$ and the parametric growth rate $\gamma_0$, a significant reduction in the instability growth is expected. For SBS we obtain $\Delta \theta_0 > 3.5 (\nu_s/\omega_s)^{1/2}$, where $\nu_s/\omega_s$ denotes the ratio of the sound wave damping to its angular frequency; in our case, $\Delta \theta_0 > 1.1$. For SRS we find:

$$\Delta \theta_0 > 1.2 \times 10^{-1} \left( \frac{(n/n_\infty)^{3/2} Z eff \log \Lambda_{ei}}{T^5 \lambda_0 D(n)} \right)^{1/2}$$

with $D(n) = (1 - (n/n_\infty))^{1/2}(1 - 2(n/n_\infty)^{1/2})^{1/2}$, $T_\infty$ in keV, $\lambda_0$ in $\mu$m; in our case, $\Delta \theta_0 > 0.41$. These two conditions were not satisfied in our experiments where $\Delta \theta_0 \approx 0.18$. Hence, the dramatic reduction of SBS and SRS seems to demonstrate that other non-linear processes interact with these two instabilities; among them, a filamentation instability has been invoked. Our results, concerning reduction of filamentation with RPPs, to be presented below, support this conjecture.

Two mechanisms may give rise to filamentation instability: a ponderomotive force and thermal effects. In both cases the spatial incoherence reduces the filamentation growth if the incoherence is large enough for the inequality $X_{sp} < X_{opt}/2^{1/2}$ to be satisfied [2]. Here, $X_{sp}$ denotes the speckle size and $X_{opt}$ the transverse wavelength maximizing filamentation growth in the transverse direction. In our experiments, $X_{sp} = 2.9 \mu$m, and $X_{opt} = 4.7 \mu$m and 8.1 $\mu$m, for ponderomotive and thermal filamentations, respectively. So, filamentation should be significantly reduced in our experiments; this was indeed observed. Beam filamentation was studied by time resolved photography of the beam cross-section in the plasma. For coherent beams, filamentation appears, at the beginning of the pulse, as very short bursts randomly distributed in space. Then, rather stationary filamentary structures occur, showing that the beam has been subdivided into beamlets that are self-trapped in plasma channels. With RPPs, there was no longer lasting evidence of these structures, demonstrating stabilization by spatial incoherence as was explained theoretically just above.

The second part of our work deals with the analysis of laser implosion hydrodynamics. Experiments are conducted on high aspect ratio targets, $D/e \sim 100$ to 200, D–T filled, with diameters ranging from 200 to 400 $\mu$m. The targets are irradiated at 0.26 $\mu$m laser wavelength, by the six-beam Nd glass laser of LULI frequency quadrupled. Focusing on target is performed by f:1 aperture lenses. The overall UV energy pulse of 150 to 200 J has a duration of 500 ps.

The implosion dynamics has been studied: the shell R–t diagram, the implosion velocity and the implosion time. Fairly good agreement is obtained with the one dimensional hydrodynamic code FILM, for moderate flux inhibition as given by delocalized heat transport theory. The laser energy transfer efficiency to the core has also been evaluated at two stages of the implosion.
FIG. 2. Kinetic efficiency versus initial shell thickness (µm); • experiment; ○ simulation.

FIG. 3. Thermal efficiency as a function of the convergence ratio obtained at shock reflexion; • experiment; ○ simulation.
First, we have the mechanical transfer which is the kinetic energy imparted to the unablated imploding shell. This energy is given by \( E_k = 2\pi R_f^2 \rho \Delta R V_s^2 \), where \( \rho \Delta R \) was obtained from the \( \alpha \) particle slowing down in the shell; \( R_f \), the final radius, and \( V_s \), the shell velocity, are determined from X ray pinhole and streak photographs. A comparison between the experimental kinetic efficiency and the numerical simulations shows that, apart from a few shots for which the irradiation conditions were very poor, fairly good agreement is found with the simulations. Mechanical efficiencies in the range of 10% are obtained (Fig. 2).

The second stage for which energy transfer has been evaluated is the shell deceleration phase. The main diagnostic tool is the fuel temperature obtained from the \( \alpha \) particle spectrum, which yields the temperature of the region where the fusion reactions occur. This temperature is in relatively good agreement with the numerical simulations for the average temperature of the region where 90% of the neutrons are generated, at least at an early stage, i.e. when the initial shock is reflected on the imploding shell. At this time, a significant deceleration of the shell has already occurred. The comparison between numerical and experimental results, shown in Fig. 3, is expressed in terms of the thermal energy transfer efficiency in the region of the fusion reactions. The efficiency appears to be small because only a small fraction of the compressed core is involved. If we take, from numerical data, the remaining core at a lower temperature into account the efficiencies achieved range from 2 to 5%. This shows that the core is somehow heated by the deceleration of the shell which transfers nearly 20 to 50% of its kinetic energy to it. However, the deceleration is not complete and mixing with fuel occurs promptly, bringing the compression to an end. This is the generally agreed explanation for the end of the implosion process and the reduced neutron yield, one thousand times smaller than that given by the numerical simulation.

An improvement of the illumination uniformity has been attempted by the use of RPPs. A fairly uniform illumination was obtained with a speckle size \( \lambda_{sp} \) of the order of the wavelength. It is assumed that these non-uniformities are strongly smoothed by thermal conduction between critical layer and ablation front which are separated by a distance much larger than the laser wavelength. The remaining non-uniformity is therefore mostly associated with beam imbalance and beam superposition.

Use of RPPs results in a much higher quality of the macroscopic implosion symmetry as is observed on X ray streak photographs of the implosion dynamics. However, no reliable \( \rho R \) and temperature data could be obtained for these shots, because of the strong reduction in the number of emitted neutrons and \( \alpha \) particles. The reduction of the fusion products can only partly be explained by the reduction of energy on the target caused by the transmission efficiency of the RPPs. Such an effect was also observed in the Rutherford experiments [3]. One could conjecture that, for coherent illumination, the presence of multiple hot spots generates an initial shock which is much stronger than that obtained with RPPs. This shock creates more effective fuel preheating before the compression by the shell. The neutron yield,
which is small compared to the numerical simulation, confirms that with RPPs shell fuel mixing still occurs at the very early stage of the deceleration phase. This indicates that in our experiments shell breaking and fuel mixing are dominated by long wavelength unstable modes associated with the illumination geometry [4].

In conclusion, we may state that our experiments have demonstrated the stabilization of filamentation by spatial beam incoherence, in agreement with theoretical predictions. They also show a dramatic reduction of SBS and SRS, in a regime where the theory would not predict a large effect. Although this could be partly due to a modification of the plasma parameters, it gives some confidence that SBS and SRS mainly take place in hot spots and that beam smoothing should result in very low SBS and SRS activities. High hydrodynamic efficiencies of 10% have also been demonstrated for direct drive. The main problem remaining is, however, the final implosion stability which in our case seems to be dominantly related to the illumination geometry.

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PROGRESS IN INERTIAL FUSION PHYSICS AND TECHNOLOGY AT DENIM, SPAIN


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Abstract

PROGRESS IN INERTIAL FUSION PHYSICS AND TECHNOLOGY AT DENIM, SPAIN.

Since the 1990 IAEA Conference in Washington, progress on numerical models (hydrodynamics, atomic physics, radiation hydrodynamics) and the understanding of the inertial fusion target physics and the performance of the first wall materials of the inertial fusion reactors has been made. A new two dimensional hydrodynamics code, using finite differences and a staggered mesh, has been successfully tested in the description of some hydrodynamic instabilities. A new model for obtaining local thermodynamic equilibrium (LTE) opacities of high-Z materials has been developed, which strongly improves the computational performance by allowing larger and more refined studies. Non-LTE atomic physics has also been extended, including average ion and detailed configuration accounting solutions. The 1-D multigroup radiation hydrodynamics model has simulated X ray conversion experiments performed in the PHEBUS laser facility of CEA Limeil (France). Kidder type laser pulses are used to implode high gain targets, improving the hydrodynamic stability (low in-flight aspect ratio) with sufficient energy gain (> 100), and reasonable energies (~3 MJ). Heavy ion, compressed experimental and high gain targets have been studied. Low activation materials have been extensively revised. SiC and vanadium alloys demonstrated good performance, i.e. as the ceramic material, and the effect of their impurities has been observed. A comparison with the magnetic fusion neutronics environment has been performed.

1. TWO DIMENSIONAL HYDRODYNAMICS

Our previously reported [1] finite difference method for shock hydrodynamics using a staggered mesh, RMFCT, has been successfully tested with 1-D Lagrangian and Eulerian planar geometry problems [2]. The method has been extended to 2-D hydrodynamics including features such as interface tracking. By using this capability results on the evolution of the Richtmyer–Meshkov (RM) instability have been reported [3]. That instability has been recognized to produce fuel–pusher mixing in the stagnation free ICF targets at times before the maximum temperature. A problem with density ratios (in the contact discontinuity, before and after the passage of the shock wave) of 10.0 and 6.48 has been analysed, the accuracy in the description of
the RM growth rates with different mesh spacings being taken into account. An excellent agreement with linear theory was found in all cases in the temporal range (early times) where this theory is applicable. However, the expected saturation at final times is only reproduced when a higher number of zones is used. For all cases, damped oscillations of the transmitted shock are clearly observed.

Simulations with phase reversal (heavy to light shocking) and the final leftward state consisting in a rarefaction wave have also been performed.

2. ATOMIC PHYSICS MODELS

Our reported [1] local thermodynamic equilibrium (LTE) atomic physics models can be classified in average ion (AI) and detailed configuration accounting (DCA). In the case of high-Z materials, the DCA model is not recommended because of the large number of excited configurations to be computed. We have developed a new model [4] which obtains the more probable ion using the AI model, and then the configurations with probabilities higher than $10^{-5}$, reducing very significantly the computational effort and the computer time. We have also included the unsolved transition array (UTA) method, assuming j–j coupling in calculating the interaction energy. The lines in each transition array are strongly overlapping, producing an absorption band due to the electron–electron and the spin–orbit interactions which allows a better description of the multifrequency opacities. Results on gold at normal density and 750 eV have been obtained by using both a large number of configurations (DCA) and a small number with the UTA model. Large differences in the multifrequency and mean opacities have been observed.

Non-LTE situations are treated by using both non-LTE average ion and non-LTE DCA models. The non-LTE DCA model considers several configurations both in the ground and the excited states. Our model includes an original solution [5] to calculate atomic properties in hot dense plasmas, which solves the time dependent rate equations for each configuration, assuming a total coupling between the excited states of adjacent ionization levels. In this procedure, most of the elements of the matrix to be solved are non-zero. A simple calculation from the neutral to the fully ionized atom, comprising all the ground state configurations, is performed initially. Once the more abundant ions are known, the model automatically selects the number of excited states for them. Our method can be used together with a radiation transport code as a postprocessor.

3. SIMULATION OF X RAY CONVERSION EXPERIMENTS

Our 1-D radiation hydrodynamics model, SARA [6], has simulated the X ray conversion experiments performed in the PHEBUS laser facility at CEA Limeil (France), which consist of the illumination of gold disks with different widths and
different laser intensities [7]. The atomic model used for the emissivities and opacities is a non-LTE average atom (AA) model based on the radiative–collisional equilibrium including dielectronic recombination. The scaling of the conversion efficiency with the laser intensity (Fig. 1) shows good agreement between our computed values (SARA) and the experimental results (PHEBUS).

A good measure of the radiation transport modelling is the calculation of the energy emitted by the rear side in the form of X rays. Reasonable good agreement with experiments is also found in this quantity [3]. However, the spectra emitted by the rear side present some discrepancies with the experiments; in particular, the attenuation of the M-band photons (2–3 keV), which are not completely damped at thicknesses of about 1 μm, disagrees. Some uncertainties in the Au opacities and emissivities, probably related to the modelling of the line profiles, are at the origin of these discrepancies.

4. TARGET DESIGN

The evolution of low initial aspect ratio (IAR) fuel capsules with 1 mg of D–T and different masses has been simulated by using a Kidder type pulse shaping in direct laser illumination [3] instead of the previous Gaussian simulations [1]. An IAR of 12 has been considered, with an optimum pusher mass of 2.3 mg, which gives an
TABLE I. CRITICAL ISOTOPES AND CONCENTRATION LIMITS FOR THE
MOST SIGNIFICANT Metallurgical ELEMENTS AND Impurities IN
LIQUID PROTECTED IFE FWs [10]
(NL stands for 'not limited')

<table>
<thead>
<tr>
<th>Elements</th>
<th>Critical isotopes</th>
<th>Concentration limits</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>NSB</td>
</tr>
<tr>
<td>N</td>
<td>$^{14}$C</td>
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<td>$^{26}$Al</td>
</tr>
<tr>
<td>Fe</td>
<td>—</td>
<td>$^{60}$Co</td>
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<tr>
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</tr>
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<td>$^{60}$Co</td>
</tr>
<tr>
<td>Cu</td>
<td>—</td>
<td>$^{60}$Co</td>
</tr>
<tr>
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<td>$^{108m}$Ag, $^{113m}$Cd</td>
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<td>$^{154m}$Eu, $^{158m}$Tb, $^{166m}$Ho</td>
<td>$^{154}$Eu, $^{154}$Eu</td>
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<td>$^{154}$Eu, $^{166m}$Ho</td>
</tr>
<tr>
<td>Tb</td>
<td>$^{166m}$Ho</td>
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<td>Ho</td>
<td>$^{166m}$Ho</td>
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<td>Er</td>
<td>$^{166m}$Ho</td>
<td>$^{166m}$Ho</td>
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<td>Bi</td>
<td>$^{208m}$Bi, $^{210m}$Bi</td>
<td>$^{207}$Bi, $^{208m}$Bi, $^{210m}$Bi</td>
</tr>
</tbody>
</table>
almost pusherless implosion with a velocity of $4.6 \times 10^7$ cm·s$^{-1}$. The pulse has a maximum power of 830 TW, an intensity of $\sim 10^{15}$ W·cm$^{-2}$, and a total energy of 3 MJ. The gradient scale length of the corona (L) is that corresponding to the reactor target regime, $L/\lambda \approx 10^4$ and $\lambda^2 \approx 10^{14}$ W·μm$^2$·cm$^{-2}$, which is below the threshold of the convective stimulated Raman scattering (SRS) but may be above that of the absolute SRS. D-T carries away $\approx 80\%$ of the implosion kinetic energy at the end of the pulse. The maximum in-flight aspect ratio (IFAR) is <100, with a consistent hydrodynamic efficiency of 6%. A fuel convergence ratio of 23 is attained, together with good ignition conditions ($\approx 2$–3 keV, 3 g·cm$^{-2}$).

Experimental and high gain targets driven by heavy ions have been studied [8, 9]. Hollow targets made of a lithium coated shell of D-T are considered. Ignition was demonstrated to be feasible if fuel stagnation is avoided, and the dose rate delivered in lithium by the driving pulse is the parameter governing the fuel acceleration. Results leading to an identification of the types of experiment to be driven by beams with power and energy levels much lower that those expected to produce high energy yields have been performed, and two different implosion and ignition regimes (high density or high temperature) with these low masses ($\sim 0.1$ mg) have been identified.

5. ACTIVATION OF MATERIALS

Long term activation responses (hands on and remote recycling, and near surface burial (NSB)) of elements and materials components of liquid (LiPb) protected FWs of inertial fusion experimental (IFE) reactors have been evaluated at the end-of-life fluences [10] (Table I). Some valuable elements (C, Si, V, Ti, Cr, Mn) are not limited for waste management, but severe compositional restrictions are imposed on other metallurgically important elements such as Fe, Ni, Mo or W. SiC is the most attractive material in IFE; as to the impurities, remote recycling should be questioned by the presence of Co, hands on recycling by Co and Cd; in particular, Nb must be reduced below the ppm range because of NSB considerations. Vanadium alloys, except VANSTAR, exhibit an excellent long term activation performance, taking advantage of the good radiological properties of V, Cr, Ti and Si, but Fe should be avoided in the alloy. Charged particle activation — sequences from (n, charged particle) reactions — is not expected to deteriorate the V-alloy performance. Steels, in general, and, in particular, those based on Cr-W, show the worst performance and are not suitable as FW materials event if the fluence does not exceed two years of irradiation. Important differences can be found when these IFE results are compared with those in magnetic fusion environments [11].

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ANALYSIS OF HOHLRAUM TARGETS FOR HEAVY ION FUSION

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Abstract
ANALYSIS OF HOHLRAUM TARGETS FOR HEAVY ION FUSION.
Symmetrization of non-uniformity by thermal radiation in spherical hohlraum targets is systematically studied for indirectly driven inertial confinement fusion. The numerical calculations have shown that the effect of X ray re-emission on the illumination uniformity is quasi-linear. It is then derived on the basis of a linear theory that a non-uniformity of each mode of the X ray source is considerably smoothed out by three different effects: The first is the geometrical effect, which accounts for the configuration and the number of X ray converters. The second is the single emission effect, which depends only on the area ratio. The third is the multire-emission effect, which is equal to the reciprocal of the average circulation number of the radiation in a hohlraum target. As a result, $N_C = 6$ (hexahedron) is shown to be necessary and sufficient to ensure tolerably symmetrical illumination ($\leq 1-2\%$ rms).

1. INTRODUCTION
Symmetrical implosion is a most critical issue in inertial confinement fusion (ICF). It is expected that the symmetry problem can be solved with indirectly driven (so-called hohlraum) targets [1,2]. This scheme seems most appropriate to achieving controlled thermonuclear fusion, in particular, for heavy ion fusion (HIF)[3-7]. In HIF, the beam properties strongly affect the target performance. For example, deposition power of the order of $10^{16}$ W/g is necessary for high conversion of ion beam energy into thermal X rays [8]. There is another aspect to be clarified, i.e. an optimum number of X ray converters to meet ICF requirements. These requirements to the beam properties from the point of view of target design are expected to define areas for accelerator design in HIF [9,10].
In this paper, we focus on the mechanism of radiation symmetrization. In the following sections, it is shown that the non-uniform radiation field is smeared out by three different smoothing factors. Further, the analysis is applied to multi-converter systems.

2. SMOOTHING FACTORS

In this section, we describe the smoothing effect in a concentric spherical hohlraum target. As pictured in Fig.1, the capsule and the outer wall have radii of \( r \) and \( R \) and are labelled by 1 and 2 as subscripts in the following analysis, respectively. Two simplified forms of such a converter—line and point converter—are illustrated as an example; we actually deal with multiple converters, but only an element among these is seen in Fig.1.

When the capsule is not directly illuminated by the source, the \( n \)-th rms component \( \sigma_n \) of illumination non-uniformity is expressed as the product:

\[
\sigma_n = C_n G_n S_n M
\]  

where \( C_n \) denotes the source non-uniformity.

![FIG. 1. Schematic picture of spherical hohlraum target. Two different types of X ray converter — line and point source — are shown as examples.](image-url)
2.1. Geometrical factor

Skupsky and Lee [11] have shown for direct drive that the beam overlapping effect is accounted for by the geometrical factor $G_n$, which concerns such converter configurations as their individual positions and the number $N_c$, given by

$$G_n = \left[ \sum_{k=1}^{N_c} \sum_{k'=1}^{N_c} P_n(\hat{Q}_k \cdot \hat{Q}_{k'}) \frac{\tilde{r}_{a1}^{(k)} \tilde{r}_{a1}^{(k')}}{\tilde{I}_T^2} \right]^{1/2}$$

(2)

Figure 2 shows the geometrical factor for different numbers of converters: $N_c = 2$ (dipole), 4 (tetrahedron), 6 (hexahedron), 8 (octahedron), and 12 (dodecahedron), where we set all average intensities equal.

2.2. Smoothing factor by single emission

We consider here a single type of transport of radiation, which is emitted by the outer wall and shines directly on the capsule.
FIG. 3. Single emission smoothing factor for even modes.

FIG. 4. Multi-re-emission smoothing factor. Thin lines represent the results of numerical calculations for different modes, solid lines are predictions given by a simple analytical model.
Concerning this single transport, the smoothing factor has already been reported in Refs[12, 13]. When an X ray source irradiates the outer wall, whose profile is the n-th Legendre mode, a similar absorption pattern appears on the capsule, but with different amplitude. The factor $S_n$ is defined by the amplitude ratio between these quantities.

Figure 3 shows the absolute values of the smoothing factor for even modes as a function of $r/R$. We see a striking feature: $S_n$ drops quickly with decreasing $r/R$ and with increasing $n$. For $n \geq 5$ and $r/R \leq 0.3$, the non-uniformities are significantly reduced by a factor of about hundred.

2.3. Smoothing factor by multiple re-emission

Non-uniformities of the source are only symmetrized by radiation transport through the cavity. In the preceding subsection, we have derived the smoothing factor by single transport, $S_n$, between the outer wall and the capsule. Then, the smoothing effect by multiple re-emission is accounted for as the complementary part of the radiation symmetrization. One can analytically derive the smoothing factor $M$ as the reciprocal of the average circulation number of radiation in the cavity:

$$M = \frac{1}{N_{\text{cir}}} = \frac{A(1 + N_1) + N_2}{A(1 + N_1)(1 + N_2)}$$

(3)

Figure 4 shows a comparison of $M$ between the detailed numerical calculations for different mode numbers (thin lines) and the simple analytical model (thick line), Eq.(3). Here, the re-emission factor of the outer wall is fixed to be $N_2 = 10$, which corresponds to times of several ns for gold; $N_1 = 0, 1$ and 10 are assumed as numbers of the capsule composed of perfect absorber, carbon, and gold (also at several ns), respectively. For $N_1 = 0$, the model and the numerical calculations agree quite well. Meanwhile, for $N_1 = 1$ and 10, the deviations between them grow with increasing $r/R$ and depend on the mode number (dispersive). But these deviations converge monotonically to the model line with increasing mode number as can be seen in Fig.4. Besides, we are interested in the region of smaller radii ($r/R \leq 0.5$) from the point of view of ICF application. In this region, the factor $M_n$ is almost independent of the mode number. We can thus conclude that the average circulation number of radiation exclusively determines the degree of the multi-re-emission smoothing.

3. APPLICATION TO MULTI-CONVERTER SYSTEMS

Figure 5 shows the rms non-uniformities versus the normalized distance $D/R$ (compare Fig.1) for the two types of source. Direct illumination of both capsule and outer wall is taken into account. The
labels in Fig. 5 denote the converter number N\textsubscript{C}. With the line converters, it is notable that the non-uniformities for N\textsubscript{C} = 6, 8, and 12 stay well below 1%, neighbouring each other at D/R > 0.5. Under direct illumination of the capsule, high uniformity better than 1-2% rms does not seem to be achievable for N\textsubscript{C} = 2 and 4, at least for the typical converter configurations shown here.

4. SUMMARY AND CONCLUSIONS
We have systematically studied the symmetrization of a radiation field in a hohlraum target as a linear system. The non-uniformities are smeared out by three effects, i.e. the geometrical smoothing effect G, the single emission smoothing effect S and the multi-re-emission smoothing effect M. The S effect is strongly enhanced with increasing mode number and with decreasing radius ratio. In contrast, the M effect is almost independent of the mode number and equal to the reciprocal of the average circulation number of radiation. In addition, only M evolves with time.

The numerical calculations have shown that increasing converters bring about a drastic improvement in uniformity and that the number N\textsubscript{C} = 6 (hexahedron) is necessary and sufficient to assure tolerable uniformity meeting the ICF requirement (≤1-2% rms).
However, the converter design is still not trivial; by suppressing direct illumination of the capsule by the source, the non-uniformity can be improved by a factor of 3-10. Even with $N_C = 2$, a high uniformity may be obtained. In this case, however, such a source pattern should be made carefully so that a lowest dominant mode is, at least, $n_{\text{min}} = 4-6$ without direct illumination of the capsule.

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ASHURA KrF LASER DEVELOPMENT AND TARGET SHOOTING EXPERIMENT*

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Abstract

ASHURA KrF LASER DEVELOPMENT AND TARGET SHOOTING EXPERIMENT.

The ASHURA KrF laser (660 J, six beams) is being upgraded by adding a 10 kJ amplifier and 12 beam optics. The new system, Super-ASHURA, is expected to yield 8 kJ on target with shaped waveform and spatially smoothed profile. In the target shooting experiments with the present ASHURA, absorption rate, mass ablation rate and backscattering reflectivity have been measured for long scale-length plasmas produced by 10 ns flat-top laser pulses in the power density range of $10^{13}$ to $10^{14}$ W/cm$^2$.

1. INTRODUCTION

The kryptonfluoride (KrF) laser is considered to be a promising candidate as driver for an inertial confinement fusion (ICF) reactor, because of its short wavelength (248 nm), broad spectral width (1 nm), high efficiency and its capability of high repetition rate operation. At the Electrotechnical Laboratory (ETL), a high power KrF laser system, ASHURA (six beams, 660 J), has been developed in order to establish the technology for a KrF laser driver and to perform target shooting experiments with long duration (10 to 20 ns) UV laser pulses [1, 2]. Recently, upgrading of the system to a 10 kJ class system, Super-ASHURA, has started. The Super-ASHURA is going to be constructed by adding, to the present ASHURA system, a 10 kJ amplifier, the optics for 12 pulse multiplexing and pulse shaping and irradiation smoothing functions. In parallel with the system development, plane target shooting experiments are being performed with 10 ns flat-top pulses yielding a maximum focused intensity of $2 \times 10^{14}$ W/cm$^2$.

2. DESIGN OF THE SUPER-ASHURA

The Super-ASHURA aims at a maximum focused energy of 8 kJ on target with 12 beams. The layout of Super-ASHURA is shown in Fig. 1. The left hand half of

* Work supported by Atomic Energy Bureau, Science and Technology Agency, Japan.
FIG. 1. Layout of Super-ASHURA, a 10 kJ class KrF laser system. The left hand half is the present ASHURA (six beams, 660 J). The main amplifier (Amp-4), a 12 beam encoder, a decoder and a target chamber are going to be added.

The layout is the present ASHURA system. A main amplifier, Amp-4 (10 kJ, 270 ns), and an optical system for 12 pulse angular encoding and decoding are to be added. Temporally shaped pulses will be amplified through the multiplexing optical system, and the intensity profiles on target are to be smoothed by the broadband random phase (BRP) method [3].

The main amplifier, Amp-4, is designed, in the same way as the Amp-3 of ASHURA, to have a high gain length product \((g_0L > 10\) for single pass) to achieve a high output intensity \((> 10\) MW/cm\(^2\)), high intrinsic energy efficiency (10%) and sufficient amplified spontaneous emission (ASE) suppression at the same time. The gain volume of the Amp-4 (60 cm diameter, 2 m length) is pumped at a pumping density of 0.7 MW/cm\(^3\) by electron beams arranged cylindrically around the gas cell. The arrangement has given the Amp-3 a fairly uniform pumping distribution (less than 20% variation) without external magnetic field. Improvement in the electron beam injection efficiency can be expected by raising the electron beam voltage (from 500 to 750 kV) and the diode impedance (from 4 to 7 Ω), because they decrease the electron absorption loss in the anode and the pressure sustaining foils as well as electron backscattering by the laser gas. At full power operation, scheduled for 1994, an output energy of 10 kJ can be expected by using full aperture and full gain duration.

For the 12 time angular pulse multiplexing, an 88 ns pulse train \((22\) ns \(\times 4\)) amplified by Amp-2 and Amp-3 (both having a gain duration of 100 ns) is again encoded to form a 264 ns pulse train \((22\) ns \(\times 12\)) to fill the whole gain duration of
the Amp-4. The geometrical filling factor of the Amp-4 will be limited to 80% only because the distance between input/output mirror array and the amplifier is limited to 19 m. By using strong fluoride coated mirrors, the diameter of the final output beams can be reduced to 13 cm (5 J/cm², on average) before the decoder. Focused by an \( F = 10 \) spherical lens, and without irradiation smoothing, the smallest spot size of 100 \( \mu \)m can be obtained with the present beam quality of 30 times diffraction limited. A maximum focused power density of \( 5 \times 10^{15} \) W/cm² can be expected with 12 beams.

The BRP technique [3] is planned to be used for smoothing the intensity distribution on target. The technique uses a broad laser spectral width (>0.3 nm) and a dispersing thin wedge (\( \theta = 5^\circ \)) in order to smooth out the ripples appearing in the usual random phase irradiation scheme. An advantage of the BRP technique is that delicate control of the beam profile all the way from the front end to the target chamber is not critically important because the intensity distribution on target is mainly determined by the optics near the focusing lens. Although a spectral width of 0.2 nm has already been obtained, further spectral broadening is being planned.

A new front end which consists of a bandwidth variable oscillator and an optically gain switched preamplifier (Fig. 2(a)) has been installed to control the width and shape of the broadband pulse without jitter. The pulse width can be varied from 2.5 to 15 ns, and the spectral bandwidth can be changed from 0.003 to 0.2 nm. The output from the front end has been six times multiplexed and amplified through the amplifier chain of ASHURA. The amplified and temporally overlapped 10 ns pulse shape is shown in Fig. 2(b). The pulses are focused on to a single spot on target by \( F = 10 \) lenses with a positioning accuracy of 20 \( \mu \)m, and a maximum power density of \( 2 \times 10^{14} \) W/cm² has been achieved. The ratio of the prepulse, which is mainly

![Diagram](image-url)

**FIG. 2.** Layout of (a) front end and (b) output waveform of ASHURA. At the front end, the pulse width of the main output pulse is only determined by the delay between the main and the switching pulses. The waveform obtained by temporally overlapping amplified six pulses has 2 ns rise time and 6.5 ns flat part.
due to cross-talk between the pulses, is typically $10^{-7}$ on target. The shaped pulse is also being generated in the front end by suppressing the rising edge of the main pulse by the switching pulse and is going to be amplified by the 12 time multiplexing optical system of the Super-ASHURA.

3. INTERACTION OF 10 ns FLAT-TOP PULSES WITH LONG SCALE-LENGTH PLASMA

In parallel with the laser development, the interaction between an intense UV laser pulse and a long scale-length plasma has been investigated [2]. This is important because a high gain pellet will be surrounded by an underdense plasma which extends over a millimetre when the pellet is irradiated by shaped pulses with an intensity of $10^{13}$ to $10^{15}$ W/cm$^2$ and a duration of a few tens of nanoseconds.

In the experiments, the present ASHURA has been used to irradiate plane targets with an intensity of $10^{13}$ to $10^{14}$ W/cm$^2$. The laser pulse shape has 2 ns rise time (10 to 90%), 6.5 ns flat-top (maximum 20% variation) and 10 ns half maximum width as is shown in Fig. 2(b). The focal spot size is, typically, 120 μm. In this intensity region, the laser absorption rates measured by a calorimeter array arranged around the targets are higher than 95% for various materials. The pulse shape of the X ray emissions (> 1 keV) as measured by a fast response X ray diode follows the laser pulse shape, which means that the temperature in the laser absorbing region remains constant and steady state ablation is maintained during the flat part of the incident laser pulse. The mass ablation rates obtained from signals of the charge collector array are shown by open circles in Fig. 3 as a function of the incident laser intensity. The mass ablation rate obtained from X ray signals from burnthrough foils is also shown in the figure by a closed circle. The ablation rates are slightly higher

![FIG. 3. Mass ablation rate obtained by 10 ns flat-top pulse irradiation. Open circles correspond to values calculated from charge collector signals. The closed circle refers to a value obtained from X ray burnthrough signals.](image-url)
than the values obtained with shorted pulse width (<1 ns) and show an $I^{0.6}$ dependence, where $I$ is the incident laser power density. The space resolved and time integrated X ray spectra suggest that hot plasmas ($T_e \sim 1$ keV at $10^{14}$ W/cm$^2$) extend over 400 µm (e-folding length) from the initial target surface.

The backscattered light from the plasma to a focusing lens was measured with temporal and spectral resolution. With narrow laser bandwidth ($\Delta \lambda = 0.003$ nm), red shifted (0.15 to 0.25 nm) spectra caused by stimulated Brillouin scattering (SBS) were observed at laser intensities of over $2 \times 10^{13}$ W/cm$^2$ per beam. The intensity of the backscattered light has a deeper temporal modulation than the laser pulse but the wavelength shift stays constant during the plateau of the laser pulse. The dependence of the backscattered light reflectivity on the incident laser intensity is shown in Fig. 4 for various irradiation conditions. The reflectivity shows slight saturation at laser intensities of over $4 \times 10^{13}$ W/cm$^2$ per beam, and the maximum reflectivity is estimated to be 0.5%. With broadband ($\Delta \lambda = 0.2$ nm) laser pulses, backscattered light can still be observed, and the difference from the narrow bandwidth cases is not significant. Although the reflectivity observed so far is low enough, the saturation level of the SBS reflectivity in higher laser intensity regions ($>10^{14}$ W/cm$^2$) and the effect of the laser bandwidth must be investigated to greater detail.

4. SUMMARY

The development of the Super-ASHURA KrF laser system, which aims at 8 kJ total energy on target with 12 beams, has been started. A 10 kJ amplifier for the system is being constructed. The present ASHURA system is producing six flat-top pulses whose spectral width can be broadened up to 0.2 nm. By overlapping the pulses on a plane target, a maximum power density of $2 \times 10^{14}$ W/cm$^2$ has been
obtained. With a long scale-length plasma produced by a 10 ns flat-top pulse, absorption rate, mass ablation rate and backscattering reflectivity have been measured in the power density range of $10^{13}$ to $10^{14}$ W/cm$^2$.

REFERENCES

ACCURATE METHODS FOR CALCULATING ATOMIC PROCESSES IN HIGH TEMPERATURE PLASMAS

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Abstract

ACCURATE METHODS FOR CALCULATING ATOMIC PROCESSES IN HIGH TEMPERATURE PLASMAS.

A technique for computing monochromatic X-ray absorption is described and compared with experimental data. Calculations of power loss from carbon plasmas with comprehensive new datasets confirm that the direct inclusion of metastable states can noticeably decrease the calculated power loss.

1. X-RAY ATTENUATION

Atomic physics plays several roles in inertial fusion targets. For some applications, radiation transport and radiative energy deposition are important. These require accurate monochromatic absorption coefficients. Until recently, most plasma spectroscopic data were emission spectra, which can be very difficult to properly interpret if non-LTE effects are present. Deviations from LTE not significant enough to grossly affect overall ionization balance can still profoundly perturb the emission. Transmission measurements are much more germane to opacity and energy deposition issues.

Figure 1 shows a niobium transmission measurement made at NOVA, by T. Perry and associates [1]. Given the kT = 47 eV temperature and kilovolt photon energy range, one observes transitions from the inner shells into the valence subshells and into Rydberg levels. The isolated transitions at approximately 2200 and 2300 eV are 2p3/2 to 3d5/2 and 2p1/2 to 3d3/2 arrays respectively. The "background" absorption in this region, which results in a transmission of about 0.9, is the M shell continuum. Near 2500 eV are transitions from the L shell into n=4 and higher principal quantum number states. The 2p photoelectric edges and their many associated Rydberg series occur between 2500 and 2800 eV. At this temperature and density (47 eV and 1.7e-02 gm/cm³), the dominant ion is about 13 times ionized (3d10 ground state configuration). At somewhat lower temperatures, most
ions present in the plasma would have a full 3d subshell, and these isolated 2p to 3d transitions would weaken and eventually disappear. Thus for a limited range of temperature, the appearance and strength of these particular transitions are sensitive to ionization balance. Figure 1b shows a calculation of the niobium transmission under these conditions.

The basic atomic model is a relativistic atom-in-jellium model, which incorporates plasma effects on atomic properties (such as energy levels and transition moments) in an ab-initio fashion [2].

The (fictitious) average ion occupancies, which follow from the above model, provide a good indication of the average state of plasma ionization. Integer occupancy variations around these average ion populations provide a realistic indication of the ionic configurations likely to exist in the plasma. Calculation of Stater integrals and electron-electron interaction energies, from the average ion wavefunctions, permits the atomic configurational energies to then be calculated using first order perturbation theory [3]. Configurational probabilities can then be calculated from Gibbs statistics.

Structure arising in non-trivial transitions such as $2p_{3/2} - 3d_{5/2}$ (say, or $2p_{3/2} - 4d_{5/2}$) often is associated with distinct core configurations ($3d^{10}$, $3d^9$, $3d^8$ etc.). Thus, for a particular one electron transition, a number of probability weighted subvariances of transition energy are computed and used.
in assumed Gaussian distributions as a representation of the one electron transition. The calculated transmission spectrum (Fig. 1b) was obtained in this fashion. Photoionization cross sections are calculated from the ionic potential. Although these calculations are not intended to have spectroscopic accuracy, the overall agreement is quite good, especially since all the atomic data was generated internally.

2. POWER LOSS CALCULATION

In magnetic confinement schemes, radiative power losses are an important consideration.

At Los Alamos, considerable effort has recently been devoted to improving our ability to systematically calculate large numbers of radiative and collisional cross sections required in general non-LTE problems. These efforts spring from and extend [4] the original work of Robert Cowan

![Graph](image)

**FIG. 2.** Comparison of radiative power loss from ADPAC [5], configuration average calculation, fine structure calculation with plane wave Born (PWB) cross sections and fine structure calculation with selected PWB cross sections replaced with distorted wave (DW) cross sections.
(Hartree-Fock atomic structure) and Joseph Mann (distorted-wave collisional cross sections etc.). Calculated processes include photoionization, electron impact ionization and excitation, autoionization, and photoexcitation. Inverse processes are computed using the principle of detailed balance. Thus cross sections have been used to compute steady-state population distributions to predict carbon power loss. Bound and autoionizing lines are treated explicitly in the rate equations. Autoionizing levels are populated by collisional excitation, ionization, and dielectronic recombination. Branching ratios are not needed because radiative stabilization is treated explicitly.

Figure 2 shows an example of the calculated radiative power loss from a carbon plasma under a range of temperatures. The configuration average is the simplest assumption, and yields results generally similar to the Princeton calculation. The next series of calculations used intermediate coupling ($10^3$ levels). In one case, the collisional cross sections are plane wave Born, which are contrasted with a calculation where a subset of more computationally demanding distorted wave cross sections were used as a sensitivity test.

In the configuration average, the power loss from boron and beryllium-like ions is overestimated. At the intermediate coupling level, many of the energy levels are metastable, leading to a lower effective power loss. The importance of the metastable states has been noted by Marchand et al. [6]. These calculations confirm that the presence of metastable intermediate coupling levels can drastically reduce the calculated radiative power loss, compared to configuration averaged approximations.

We are indebted to T. Perry for providing the niobium data.

REFERENCES

APPLICATIONS OF GROUP INVARIANT ANALYTIC SOLUTIONS TO INERTIAL CONFINEMENT FUSION

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Abstract

APPLICATIONS OF GROUP INVARIANT ANALYTIC SOLUTIONS TO INERTIAL CONFINEMENT FUSION.

The fundamental behaviour of inertial fusion targets is described by the equations of radiation hydrodynamics. By using the technique of Lie group analysis, these partial differential equations can be reduced to ordinary differential equations which are much easier to solve. Particular solutions in 1, 2 and 3-D are described which are relevant to the evolution of inertial fusion capsules. Such analytic solutions can provide: (i) numerical benchmark problems; (ii) the basis for analytic models; and (iii) insight into more general solutions.

The quest for controlled thermonuclear fusion requires the study and solution of the equations of magnetohydrodynamics and/or radiation hydrodynamics. These systems of non-linear partial differential equations describe the dynamics of plasma evolution in both magnetic and inertial fusion devices. In practice, the solutions are explored numerically through computer simulations of specific plasma configurations. A parallel effort searches for relevant analytic solutions to these equations. Such analytic solutions can provide (i) numerical benchmark problems, (ii) the basis for analytic models, and (iii) insight into more general solutions.

The evolution of inertial fusion targets is described by the equations of radiation hydrodynamics. Useful analytic solutions to the equations of hydrodynamics have been found through the use of dimensional analysis and other insightful ansätze. An alternate approach has been the use of invariance of the differential equations under Lie group transformations [1]. Lie group invariance provides a systematic and deterministic technique to search for special, group invariant solutions. The equations of 1-D hydrodynamics and radiation hydrodynamics have been extensively explored by using Lie group analysis, and a large number of analytic solutions have been found [2, 3]. Recent emphasis has been on multidimensional (2-D and 3-D) solutions, which require a more detailed examination of the structure of the allowed Lie groups [4]. The solutions discovered can be used as models for certain aspects of inertial fusion target performance.
The equations considered in this particular study are the 3-D, one temperature perfect gas hydrodynamic equations including thermal conduction and an energy source $S$:

\[
\begin{align*}
\rho_t + \mathbf{u} \cdot \nabla \rho + \rho \nabla \cdot \mathbf{u} &= 0 \\
\mathbf{u}_t + \mathbf{u} \cdot \nabla \mathbf{u} + \frac{1}{\rho} \nabla (\Gamma \rho T) &= 0 \\
T_t + \mathbf{u} \cdot \nabla T + (\gamma - 1) \nabla \cdot \mathbf{u} - \frac{\gamma - 1}{\Gamma \rho} \nabla \cdot \kappa \nabla T - \frac{\gamma - 1}{\Gamma} S &= 0
\end{align*}
\]

The invariance transformation groups are calculated for these equations and used to introduce new sets of independent and dependent variables under which these equations take a simpler form [4]. These special combinations of variables represent a reduction in phase space to special classes of solutions which evolve for certain initial/boundary conditions. The partial differential equations (PDEs) can be reduced to ordinary differential equations (ODEs), which must then be solved to provide explicit analytic solutions. The ODEs can often be solved through a judicious choice of functional forms of the solutions.

One specific application of these solutions to inertial fusion is the identification of a variety of drive source time profiles that provide an isentropic implosion of a 1-D spherical capsule [3]. These profiles generally take the form where the pressure $\propto \left(\frac{r^n}{t^n}\right)^a$, where $a = 2.5$ for $n = 2$ and $a = 2, 2.5$ or $5$ for $n = 1$. The particular values of the exponents depend on the details of the initial conditions.

A two dimensional flow example provides an analytic solution of an axisymmetric implosion with a $P_2$ asymmetric drive. This solution is a consequence of three independent scaling groups (space, time and density) allowed by the differential equations [4]. These groups provide the similarity variables

\[3c^2: \rho = H(\theta) r^a t^b, \quad T = G(\theta) r^2 t^2, \quad u_r = U(\theta) \frac{r}{t}, \quad u_\theta = V(\theta) \frac{r}{t}\]

which reduce the PDEs to ODEs. A particular analytic solution is found through the ansatz $U = a + b \cos^2 \theta$ and $V = V_0 \sin \theta \cos \theta$ and is given by

\[\rho(r, \theta, t) = \rho_0 t^{-3 + 3(\beta + 1)(\gamma - 1)/(\gamma + 1)}(r \cos \theta)^\beta\]

\[u_r(r, \theta, t) = \frac{1}{t} \left(1 - 3 \frac{\gamma - 1}{\gamma + 1} \cos^2 \theta\right)\]

\[u_\theta(r, \theta, t) = 3 \frac{\gamma - 1}{\gamma + 1} \frac{r}{t} \sin \theta \cos \theta\]

\[T(r, \theta, t) = \frac{6(\gamma - 1)(2 - \gamma)}{\Gamma(\gamma + 1)^2(\beta + 2)} \left(\frac{r}{t}\right)^2 \cos^2 \theta\]
Another choice of groups, space and density scaling, along with time translation and a projective group, form the similarity variables

\[ \mathcal{C}_{1\pm}: \rho = \frac{H(\theta)r^{\beta-3}}{(a^2 \pm t^2)^{\beta/2}} \psi(t; a, \alpha), \quad T = \frac{G(\theta)r^2}{(a^2 \pm t^2)^2}, \quad u' = \frac{U(\theta) + t}{a^2 \pm t^2} \]

\[ u^\theta = \frac{V(\theta)r}{a^2 \pm t^2} \]

with \( \psi_+(t; a, \alpha) = \exp \left( \frac{\alpha}{a} \tan^{-1} \frac{t}{a} \right) \)

and \( \psi_-(t, a, \alpha) = \left( \frac{a - t}{a + t} \right)^{a/2a} \)

again reducing the 2-D PDEs to ODEs.

For the 1-D case \((V = 0, H, G, U \text{ constants})\), the reduced ODEs can be solved for two types of solutions

\[ \rho(r, t) = \rho_0 r^{\beta-3}(a^2 \pm t^2)^{-\beta/2}, \quad u(r, t)(= u') = \pm \frac{rt}{a^2 \pm t^2} \]

\[ T(r, t) = \frac{\pm a^2 t^2}{(\beta - 1)(a^2 \pm t^2)^2} \]

The material trajectories for these two solutions are given by \( r = R_0 \sqrt{a^2 \pm t^2} \), which are either hyperbolae (+) or ellipses (−) in the \( r-t \) plane, shown in Fig. 1. The positive time portion of the elliptical branch \((\mathcal{C}_{1-})\) can represent an implosion of an ICF target from initially zero velocity. A form of this solution, which is valid only for \( \beta > 1 \), was found earlier by Kidder and others [3]. The negative time portion of this branch represents an expansion phase from a point explosion which then turns around at \( t = 0 \). The hyperbolic branch, \( \mathcal{C}_{1+} \), has material moving in towards the origin that stagnates at \( t = 0 \) and moves out for \( t > 0 \). This solution, requiring \( \beta < 1 \), can be used to represent the stagnation and subsequent explosion phases of an ICF target.

The reduced ODEs for \( \mathcal{C}_{1\pm} \) can also be solved for two dimensional flow \((V \neq 0)\) on the assumption that \( U = 0 \). For \( \beta \neq 1 \), the equations become

\[ G = \frac{V^2 \mp a^2}{\beta - 1}, \quad V^2(V^2 \mp a^2)^3 = \frac{c}{\sin^2 \theta}, \quad \frac{H'}{H} = \mp \frac{\alpha}{V} - \frac{1}{\tan \theta} \]

\[ \times \left( \frac{3V^2}{4V^2 \mp a^2} \right) = \frac{(\beta + 1)V^2}{(4V^2 \mp a^2)\tan \theta} \]
When $\alpha = 0$, these relations, with $f = V/a$, become

$$\beta = -4, \quad G = \frac{a^2}{5} (\pm 1 - f^2), \quad f^2 (\pm 1 - f^2)^3 = \frac{c}{\sin^2 \theta}$$

$$\frac{H'}{H} = \frac{-3f^2}{(4f^2 + 1) \tan \theta}$$

and we find that only the hyperbolic ($3C_{1+}$) branch is allowed. At this point we must resort to a numerical solution. We first solve the implicit equation to obtain the function $f(\theta)$, which then provides $G(\theta)$. The function $f$ is then used in the remaining ODE for $H(\theta)$. The dominant parameter is the constant $c$ which determines the extent in $\theta$ of the solution, extending from $\theta_s < \theta < \pi - \theta_s$, $\theta_s = \sin^{-1}(\sqrt{256c}/27)$. Figure 2 shows these solutions for $c = 0.05$ ($a = 1$), and we find two independent upper and lower solution branches.

A three dimensional solution with axisymmetry but non-zero $\phi$ velocity can be found as a consequence of the above mentioned scaling groups. The similarity variables are

$$\rho = H(\theta)t^{a^2}e^{c\phi}, \quad u^r = U(\theta)t^{-1}, \quad u^\phi = V(\theta)t^{-1}, \quad u^\phi = W(\theta)t^{-1}, \quad \Gamma T = G(\theta)t^2$$
FIG. 2. Numerical solution for $\mathcal{K}_{1+}$ showing two possible solution branches for each choice of the arbitrary constant $c$. 
The resulting ODEs can be solved with the ansatz \( V = 0, H = H_0 + H_1 \sin \alpha \theta \) to obtain the analytic solution

\[
\rho(r, \theta, \phi, t) = t^{-\frac{b+3}{2}}\nu^b(\rho_0 + \rho_1 \sin^a \theta) \\
u^r(r, \theta, \phi, t) = \frac{r}{2t} \\
u^\theta(r, \theta, \phi, t) = 0 \\
u^\phi(r, \theta, \phi, t) = \pm r \left[ \frac{4\Gamma_0(b + 2)(\sin \theta)^{b+2}}{\rho_0 + \rho_1 \sin^a \theta} \frac{\alpha\rho_0}{4(\rho_0 + \rho_1 \sin^a \theta)(b + 2 - a)} \\
+ \frac{\alpha}{4(b + 2 - \alpha)} \right]^{1/2} \\
T(r, \theta, \phi, t) = \left( \frac{r}{t} \right)^2 \left[ \frac{\Gamma_0(b + 2)(\sin \theta)^{b+2}}{\rho_0 + \rho_1 \sin^a \theta} \frac{\alpha\rho_0}{4\Gamma(\rho_0 + \rho_1 \sin^a \theta)(b + 2)(b + 2 - \alpha)} \\
+ \frac{1}{4\Gamma(b + 2 - \alpha)} \right]
\]

Here we find a solution where the material flows on fixed cones in \( \theta \) and spins in \( \phi \) around the z axis. This particular solution can relate an initial non-spherical velocity component to an eventual 'mixing' state of the imploded target.

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X RAY IMAGING AND SPECTROSCOPIC MEASUREMENTS OF IMPLOSIONS

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Abstract

X RAY IMAGING AND SPECTROSCOPIC MEASUREMENTS OF IMPLOSIONS.

A large variety of X ray measurement techniques are used to address important physics issues in indirectly driven implosion experiments on Nova. X ray imaging, through the use of pinholes and ring apertures, provides information on the symmetry of the imploding capsule. X ray spectroscopic measurements of high-Z dopants in the deuterium fuel and plastic pusher are used to infer conditions in the compressed core.

Time resolved X ray measurements are essential in the investigation of laser driven inertial confinement fusion (ICF), where neutron and X ray emission are the only observable signatures of the compressed core conditions. High speed detectors, available for X ray measurement, provide a means of measuring the rapidly evolving conditions in imploding capsules on picosecond time-scales. We address a wide range of issues in our indirectly driven implosion experiments on Nova, with a large variety of X ray measurement techniques. Critical issues include symmetry of the compressed core, fuel density and temperature and hydrodynamic mix at the pusher/fuel interface.

X ray imaging, through the use of pinholes and ring apertures, provides information on the symmetry of the imploding capsule. These techniques can be applied to self-emission from the capsule, or X ray transmission through it from backlighters. Imaging systems used in conjunction with gated microchannel plate detectors routinely provide < 10 μm spatial resolution in the object plane and ~ 100 ps temporal resolution.
FIG. 1. Sequence of five coded images (a) and their unfolds (b) generated by the gated RAM. The temporal interframe spacing is approximately 60 ps. Enlargements of the two peak emission core images (c) show asymmetries in the core shape.
Pinhole cameras are the most common way of measuring self-emission from an imploded ICF capsule. Arrays of pinholes are used in conjunction with gated microchannel plate detectors to provide up to 16 separate images at different times though the implosion. Coded imaging, however, can produce substantial improvements in the signal-to-noise ratio (SNR) over pinhole imaging. Experiments comparing ring aperture imaging to pinhole imaging have demonstrated a twentyfold improvement in SNR in good agreement with simple theory [1].

We have constructed a simple ring aperture microscope (RAM) for routine use in the Nova laser facility [2]. The principal component in this device is a 1 mm diameter, 5 μm wide annulus fabricated in 9 μm thick gold, which is placed at ~4 cm from the target chamber centre. The projected ring image is recorded on a stack of X ray film with filters, providing several energy bands of sensitivity. The recorded images are digitized and unfolded by using a Wiener filtered deconvolution. The spatial resolution of this instrument has been tested by using a resolution target backlighted with a laser driven titanium X ray source (4.7 keV) and has found to be ≤5 μm full width at half maximum.

Recently, we began testing a time resolved version of the RAM. This instrument is like our gated X ray pinhole cameras, except that the array of pinholes is replaced with a 4 × 3 array of 5 μm wide, 250 μm diameter annuli. The instrument is operated at a magnification of ~10, providing a spatial resolution of 5 μm. A sequence of five coded images and their unfolds, from a directly driven implosion of a deuterium filled glass capsule at the University of Rochester's Omega laser facility, is shown in Fig. 1. The sequence shows an initial diffuse glow from the partially imploded glass shell, followed by core emission. The core diameter is approximately 20 μm, and the constant intensity contours exhibit ℓ = 3 or 5 asymmetries that appear to be well correlated with the laser beam illumination pattern. The core is beginning to disassemble in the final image.

X ray measurements are also crucial in inferring conditions in the compressed fuel [3, 4]. By doping the deuterium fuel with Ar (0.1 at.%), we measure K shell emission in the 3–4 keV spectral region, inferring the time dependent fuel density from measurements of the Stark broadened emission line profiles, and the electron temperature from line ratios. Spectroscopic measurements are made by using two streaked crystal spectrometers with resolving power (λ/Δλ) in the range of 500–3000 and ~35 ps temporal resolution [5]. Typical measured line shapes of the He β line (n = 3–1), for two different laser pulse shapes, are shown in Fig. 2. The inferred electron densities are 1.0 × 10²⁴ cm⁻³ for capsules driven with a 1 ns square drive (19 kJ of laser light at 3ω) and 2.0 × 10²⁴ cm⁻³ for those driven by shaped laser pulses (a low foot followed by a peak with 3:1 contrast). The shaped drive results in higher fuel densities, because of the lower entropy in the fuel. Fuel electron temperatures, which are inferred from the Ly β to He β ratio, for shaped and unshaped drives, are 1.0 and 1.2 keV, respectively. We also use spectroscopic techniques to study the hydrodynamic mix at the pusher fuel interface. Capsules with intentionally rough surface finishes show enhanced emission from pusher dopants (Cl and Fe) when the doped pusher material mixes into the hot fuel.
FIG. 2. (a) He $\beta$ line profile for 1 ns and (b) 3:1 contrast pulse shaped drive showing increased broadening due to higher peak fuel densities. Measured line profiles are shown as solid lines while calculated line profiles are dashed lines. Calculations assume $n_e = 1.2 \times 10^{24}$ cm$^{-3}$, $T_e = 1.2$ keV for the 1 ns drive and $n_e = 2.0 \times 10^{24}$ cm$^{-3}$, $T_e = 1.0$ keV for the shaped drive and include the effect of satellites of the type $1$s$213l' - 1$s$21$ and $1$s$3l' - 1$s$3l$. 
Future high performance targets that utilize more extreme laser pulse shaping have increased pusher opacity, forcing us to study the emission from higher Z elements whose spectroscopic features appear at higher photon energies. In preparation for these conditions, we have studied L shell (n = 3–2) emission from D–D filled capsules doped with Xe (0.02 at.%), in the 5–6 keV spectral region. In Fig. 3, we show Xe spectra from an experimental series where the fuel electron temperature was varied by varying the radiation drive. These two spectra were obtained from implosions driven by 28 kJ (‘high drive’) and 19 kJ (‘medium drive’), respectively, of 0.35 µm laser light in a 1 ns duration square pulse incident into the hohlraum. We observed clear differences in the spectra showing increased F like emission for the high drive case. Measurements of the ionization balance, though not as precise as the line ratio technique used for the Ar spectra, will serve to specify a range of temperatures of the fuel [6]. Measurements of fuel density from via line broadening will be difficult until high densities are achieved (>30 g/cm³), because of the relatively small pressure broadened widths (<10 eV for the 4d–2p line) at lower densities. Electron impact broadening dominates Stark broadening of the Xe 4–2 lines; the role of ion impact broadening may be significant and is under investigation.

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LASER-PLASMA INSTABILITIES AND COHERENCE CONTROL


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Abstract

LASER-PLASMA INSTABILITIES AND COHERENCE CONTROL.

The effect of random phase plate (RPP) beam smoothing on stimulated Brillouin scattering (SBS) backscattered light in exploding foil plasmas is investigated experimentally. The experiments consist of measuring the backscattered SBS emission excited by an unsmoothed and a RPP smoothed interaction beam impinging on a preformed plasma. A Ti or CH plasma is performed with typically 2500 J of 0.527 \(\mu\)m RPP smoothed laser light with a square pulse of 1 ns. The interaction laser focuses on the centre of the preformed plasma with a duration of 1 ns, 0.527 \(\mu\)m wavelength, ~2500 J, and an adjustable delay time. At least 15 times less backscattered SBS energy is measured with an RPP.

1. INTRODUCTION

Beam smoothing on parametric instabilities is important to inertial confinement fusion (ICF). Of particular relevance are the scattering instabilities, stimulated Brillouin and Raman scattering (SBS and SRS), because they can scatter large fractions of the incident energy and reduce the laser coupling efficiency. Beam smoothing techniques can have a strong effect on the growth and saturation of SBS and SRS, as well as filamentation, with the potential of stabilizing these processes. We are, at present, evaluating the implications of our beam smoothing techniques on these scattering instabilities [1].

Two schemes have been used to smooth beams: (1) spatial smoothing which breaks the beam up into a fine scale structure which the target is better able to further smooth by thermal conduction, and (2) temporal smoothing which rapidly varies the fine scale structure with time, giving a beam with time averaged smoothness. The spatial approach was first implemented in 1984 by Kato et al. [2] at Osaka University, who used a random phase plate (RPP) to break up the beam. Temporal smoothing was first introduced by the technique called induced spatial incoherence (ISI). This was conceived by Lehmberg and Obenschain [3] at the Naval Research Laboratory (NRL) in 1983 and implemented at NRL in 1985. A second temporal method,

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called smoothing by spectral dispersion (SSD), was demonstrated at the University of Rochester's Laboratory for Laser Energetics (LLE) in 1989 by Skupsky et al. [4].

This paper reviews recent experiments at Livermore designed to characterize the effect of beam smoothing on scattering instabilities and discusses their growth under conditions of limited spatial and temporal coherence. Beam smoothing on Nova consists of a combination of RPPs and SSD. The experiments use the two-beam chamber facility; one beam preforms a plasma, and the second is an interaction beam. This arrangement allows independent control of the plasma conditions and of the interaction beam. The target materials are Ti or CH foils. The plasma parameters at the time of arrival of the interaction beam are chosen so as to assure growth of SBS and SRS. The electron density is about 10–15% critical density, and the electron temperature is about 2 keV. The plasma size is of the order of 1 mm. An RPP is always used on the plasma forming laser beam to provide uniform illumination.

2. EXPERIMENTAL APPARATUS

Backscatter light collected by the Nova focusing optics is directed to the diagnostic station by a full aperture beam splitter and focusing optics (Fig. 1). This sta-

![Diagram](https://via.placeholder.com/150)

FIG. 1. Optical set-up used to measure backscattered light from interaction beam.
tion contains a high dispersion spectrometer coupled to a streak camera, providing temporally resolved SBS spectra ($\Delta \lambda \approx 3\text{Å}, \Delta \tau \approx 0.08\text{ ns}$). A similar set-up, but with a lower dispersion, is used to record the SRS spectra. Photodiodes and calorimeters measure absolute levels of reflectivity. An array of photodiodes at different locations in the chamber monitors the angular distribution of the scattered light. An RPP controls the spatial coherence in the interaction beam with hexagonal cells of 7 mm, producing an average intensity on the focal spot of approximately $4 \times 10^{15}\text{ W/cm}^2$.

**FIG. 2.** Time resolved spectra of preform and backscattered light for (a) Ti target without RPP; (b) Ti target with RPP; (c) CH target without RPP; (d) CH target with RPP.
3. OBSERVATIONS

Figure 2 shows the time resolved spectra of SBS backscattered light from a Ti and CH foil without RPP smoothing (Fig. 2(a) and (c)) and with RPP smoothing (Fig. 2(b) and (d)). The figures show the transmitted light from the preform pulse that begins at time zero and lasts for approximately 1 ns. The interaction pulse turns on at about 1.4 ns for Ti and 1.1 ns for CH and lasts about 1 ns. There is a spectrally broad blue shifted flash when the interaction beam is turned on which lasts about 100 ps. This flash is followed by a 300 ps long feature with a red shift between zero and about 20 Å. This feature appears only in cases without RPP smoothing. The average intensity of a smoothed beam is about a factor of five less than an unsmoothed beam. We emphasize this point in the figures by writing the total energy in the interaction beam and the average intensity in the interaction laser spot at the top of each figure. We have made a very recent effort to compare smoothed and unsmoothed cases with the same average intensity by reducing the total energy in the unsmoothed interaction beam (to be published).

4. DISCUSSION

The effectiveness of an RPP in reducing the level of scattering instabilities is determined by analysing the data shown in Fig. 2 to obtain the time averaged plasma reflectivity. What we mean by 'time averaged reflectivity' is the ratio of the total energy in the backscattered signal to the total incident interaction beam energy. The backscattered light is separated into two components. The first component is the fast feature observed with or without beam smoothing. The second component is the diffuse feature observed only without beam smoothing which exhibits the red shift characteristic of SBS backscattered light from stationary plasma.

The average reflectivity indicates that beam smoothing using RPP reduces the SBS backscattered light (second component) by at least a factor of 16 to 17 in Ti (0.5% to 0.03% reflectivity reduction) and CH (0.24% to 0.015% reflectivity reduction) plasmas. The actual level of reduction may be higher than this since we cannot detect higher levels of reduction with the present experimental set-up. This level of reduction is in agreement with previous experimental results reported by Jalinaud et al. [5], Labaune et al. [6], and Willi et al. [7] using smaller scale plasmas. Preliminary results of recent experiments on Nova show that when higher electron densities are present in the plasma, $n_e > n_{cr}/4$, RPP smoothing has only a minor effect. RPP beam smoothing only causes about a factor of two decrease in the first component for both Ti (0.2% to 0.1% reflectivity reduction) and CH (0.16% to 0.08% reflectivity reduction) plasmas. Recent experiments also indicate that this first component is only observed when the maximum plasma density is below $n_{cr}/4$ and when the interaction beam turns on after the preform beam has been shut off.
ACKNOWLEDGEMENT

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REFERENCES


ITER AND NEXT STEP DEVICES

(Session F)

Chairman

T. TSUNEMATSU
Japan
NET PREDESIGN OVERVIEW

NET TEAM
(Presented by F. Engelmann)

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Abstract

NET PREDESIGN OVERVIEW.

NET aims at demonstrating the scientific and technological feasibility of fusion power production in a device based on the tokamak principle. It is designed to achieve controlled ignited burn for a pulse duration of 1000 s and to allow testing of components for a reactor such as systems to exhaust power and particles from the plasma as well as the tritium breeding blanket; it will also demonstrate the availability of technologies essential for a reactor such as superconducting magnets and remote maintenance. Component testing will primarily address functional tests to be done in the basic performance phase of NET, together with the physics experiments. In this phase an average neutron wall load of about 0.8 MW/m² will be available for an integral burn time of 2500 h. If considered appropriate, on the basis of the experience gained in this phase, endurance and material tests could be done up to a fluence of about 1 MWa/m² in a subsequent, extended performance phase as the device allows 40 000 full performance discharges. Only for the latter phase would a tritium breeding blanket be installed. The NET parameters were derived from the performance requirements. The earlier choice of the values of the plasma current (25 MA), the plasma elongation (b/a ~ 3) and the aspect ratio (R/a ~ 3) has been confirmed. However, the longer pulse duration of 1000 s now adopted, and the use of more conservative design criteria partly as a result of the outcome of the R&D activities undertaken in the meanwhile, has resulted in an increase of the major radius to 7.3 m. The NET predesign, now completed, and the accompanying R&D programme in physics and technology support launching of the engineering design phase.

1. INTRODUCTION

During the NET predesign phase, the NET design was developed in depth in several critical areas [1], starting from the basis led out in the conceptual design phase [2,3]. The work was supported by a comprehensive R&D programme in physics and technology. In this paper an overview of the results of the NET predesign is given with emphasis on the NET performance characteristics and parameters.
2. TECHNICAL OBJECTIVES AND DESIGN PRINCIPLES

The technical objectives of NET are:

- to demonstrate reactor plasma performance and, in particular, to achieve controlled ignited burn of a DT plasma and to generate long burn pulses;
- to demonstrate the availability of technologies essential for a future reactor, such as superconducting magnets and remote maintenance;
- to test components for a future reactor, such as systems to exhaust power and particles from the plasma, and the tritium breeding blanket.

The guiding principle in the design of NET has been to ensure maximum technical simplicity, to use prudent criteria for the physics basis and for the technologies to be applied, and to give a prominent role to safety and environmental considerations. Only technologies were adopted which are either already in hand or can be demonstrated prior to construction of NET. Areas where reliance on the development potential of tokamak physics is necessary are power and particle exhaust as well as disruption control.

3. PLASMA PERFORMANCE

The operating regime of the plasma in a tokamak reactor is characterized by a low recirculation of power back into the plasma, a long burn pulse length, and a fusion power density as high as the physics constraints allow. Quantitatively, a reactor plasma will be operated at

\[ C = \frac{P_\alpha}{P_{\text{tot}}} > 0.9 \]

with \( P_\alpha \) being the fusion \( \alpha \)-particle power and \( P_{\text{tot}} \) the total heating power of the plasma, i.e., effectively in an ignited mode; the burn pulse length \( t_{\text{burn}} \) will be at least several thousand seconds, much longer than the time \( t_{\text{res}} \).
characterizing the resistive diffusion of the profile of the plasma current; and the plasma will be operated not far from the pressure (or equivalently beta) limit, typically at

$$\beta / \beta_{T_{\text{royon}}} > 0.8$$

and in the vicinity of the density limit.

From an operational point of view these constraints imply that in a reactor a high-beta plasma burning at $C > 0.9$ must be maintained in steady state conditions with respect to all time-scales characterizing the various dynamical processes appearing in the plasma. These time-scales are related to the power and particle balances as well as to the diffusion of the plasma current profile; in addition, there may be further characteristic time-scales related to the interaction of the plasma with the walls.

NET is conceived to allow demonstration of this reactor plasma performance, minimizing at the same time the extrapolation from presently operating tokamaks as well as the investment cost. This is possible because the physics information that is needed can be obtained without generating discharges having all the features of a reactor plasma simultaneously although an option is maintained for operation with an extended performance closer to the operating regime of the reactor. In fact, it is sufficient to show separately

(i) that the power and particle balance can be controlled in the ignited mode under steady state conditions with respect to the transport processes of energy and particles. For this, a minimum burn time $t_{\text{burn}} = 30 \tau_{E}$ is needed ($\tau_{E}$ being the energy confinement time of the plasma) and operation at $C > 0.9$ is required; and

(ii) that it is possible to operate routinely at high plasma beta ($\beta / \beta_{T_{\text{royon}}} > 0.8$) for at least $t_{\text{burn}} = 3 t_{\text{res}}$. For this it is permissible to work at lower $C$, e.g., at $C \geq 0.7$ and, therefore, to use external power for active current profile control, if needed, and for sustaining the power balance at
reduced plasma current and/or magnetic field (to increase $\beta/\beta_{\text{Troyon}}$).

Typical numbers for the two time-scales $\tau_\text{E}$ and $t_\text{res}$, under reactor conditions as well as in NET, are $\tau_\text{E} = 3$ s and $t_\text{res} = 300$ s.

NET is, therefore, conceived to achieve operation at $C > 0.9$ for at least 100 s, and in addition it is able to operate for burn pulses of at least 1000 s at $C > 0.7$.

The objective of operation at $C > 0.9$ determines the energy confinement capability required and, effectively, the plasma current needed. The aspect ratio of the NET plasma was taken to be $A = 3$. This choice minimizes the uncertainties in predicting the confinement properties of the plasma as present large tokamaks all have aspect ratios around this value. The option of an appreciably larger aspect ratio and correspondingly lower plasma current does not offer significant advantages in cost or in performance, in particular with respect to the power exhaust conditions.

NET has the capability for driving the plasma current inductively for a pulse length of 1000 s to be able to operate the discharge at the highest possible $C$ and at high plasma density.

High-recycling conditions at the divertor plates can then be established, which is a prerequisite for providing acceptable working conditions for the power and particle exhaust system. While in recent experiments a divertor plasma with a temperature in the eV range has been obtained under such conditions, the demonstration of an operating regime extrapolatable to NET is still outstanding. In particular, there remain uncertainties with respect to the effect of edge localized modes. To ensure a satisfactory lifetime of the plasma-facing components, it is essential that the frequency of occurrence of hard plasma disruptions be reduced with respect to present experiments. The availability of efficient disruption control is therefore assumed.

For attaining a pulse duration of 1000 s, control of the profile of the plasma current is anticipated to be necessary. This requirement drives the need for external
power and, hence, limits the value of $C$ accessible in this regime. However, the external power actually needed for this aim cannot yet be definitely quantified. It appears that a broad current profile must be maintained to keep the sawtooth mixing radius sufficiently small, typically less than one third of the plasma minor radius. Such a profile also makes operation at higher values of the plasma beta possible. For this, about 30%, or more, of the plasma current has to be driven non-inductively by injection of external power and by the bootstrap effect; on this basis a need for an external power of about 70 MW is estimated which, for $P_{\text{fus}} \approx 1$ GW, corresponds to $C \approx 0.75$.

4. COMPONENT TESTING AND STAGED OPERATION

The unique feature of NET, as far as component testing is concerned, is that it provides a "fusion reactor environment", namely the appropriate combination of surface loads (heat and particles) and of volume loads (heat, neutrons and electromagnetic forces) as present in a DT burning device. Only NET will therefore offer the possibility to test these interacting aspects of complex components, and it is for this purpose that it will be primarily used.

A reactor will operate without blanket replacement for at least five full power years at a wall loading of 2-4 MW/m$^2$, corresponding to 10-20 MWa/m$^2$, i.e. 100-200 dpa in steel at the first wall. However, the average neutron wall load, in a Next Step tokamak, cannot exceed 2 MW/m$^2$, and its availability, taking into account the down-times imposed by the requirements of the testing programme itself and of maintenance/ replacement of components, is estimated to be only 10-15%. The neutron fluence on the first wall, therefore, accumulates rather slowly (~2 dpa per year) and at high cost (~400 M$ per year).

Under these operating conditions, extrapolation to a reactor, with acceptable risk, of the testing results of components having a weak interaction with plasma and
neutrons, such as superconducting magnets, remote handling, tritium systems, and heating/current drive systems, is possible. However, for the in-vessel components, i.e. the plasma-facing components and blanket, the testing capability is more limited.

The performance aspects of these latter components, to be tested, are:

i) Ability of the plasma-facing components to sustain the loads under conditions of long burn at reactor-relevant power flux and in the presence of disruptions;

ii) Ability of the blanket to breed and release tritium, and to satisfy the key requirements of a reactor (e.g., breeding ratio larger than 1, high temperature operation, low tritium inventory);

iii) Ability of the in-vessel components to resist a fluence of 14 MeV neutrons that approaches the reactor level and, at the same time, minimizing inventories of activated materials.

These issues must be addressed in the above order of priority, because only when the in-vessel components have shown that they perform satisfactorily their functions as described in i) and ii), the endurance questions of point (iii) can be tackled. Furthermore, the testing of the functional requirements of the plasma-facing components (mission i)) and of the blanket (mission ii)) are consistent with a limited fluence (about 0.3 MWa/m²) and availability, while for mission iii) an enhanced wall loading, approaching the reactor level, and improved availability are needed. In addition, only this latter mission necessitates breeding a large part of the tritium needed for operation in the device itself, therefore requiring a driver blanket.

For these reasons, the technical objectives of NET include the functional tests (missions i) and ii)), but mission iii) is left to an optional second phase of operation. Hence, NET operation will be carried out in stages. During the basic performance phase, the functional testing will be performed. This requires 2500 h of integral burn time [1], including the physics investigations for which about 500 h are needed. Only during the basic performance phase, on the basis of the operating experience gained and of the component test
results, will it be decided whether it is appropriate to continue operation into an extended performance phase to do endurance tests up to fluence of about 1 MWe/m². For this latter phase, a tritium breeding blanket would be installed. Anyway, for material characterization, an intense 14 MeV neutron source will be required in parallel to NET.

FIG. 1. Three-dimensional view of the NET device.
5. NET MAIN PARAMETERS AND OPERATIONAL CHARACTERISTICS

The parameters of NET follow from the requirements to allow operation at $P_{\alpha}/P_{\text{tot}} = 0.9$, to provide a transformer capability for driving the plasma current inductively for 1000 s, and to ensure a lifetime of the basic device of 40000 full performance shots. The choice of the values of the plasma current (25 MA), the plasma elongation ($b/a = 2$) and the aspect ratio ($R/a = 3$) as made for NET in 1988 [3] has been confirmed. However, the longer pulse duration now adopted, and the use of more conservative design criteria partly as a result of the R&D activities undertaken, resulted in an increase in major radius of about 15% over the value given in Ref. [3].

The basic parameters of the NET device (see Fig.1) and its operational characteristics are:

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Plasma current, I (MA):</td>
<td>25</td>
</tr>
<tr>
<td>Plasma major radius, R (m):</td>
<td>7.3</td>
</tr>
<tr>
<td>Plasma minor radius, a (m):</td>
<td>2.43</td>
</tr>
<tr>
<td>Aspect ratio, R/a:</td>
<td>3</td>
</tr>
<tr>
<td>Elongation, b/a *):</td>
<td>2</td>
</tr>
<tr>
<td>Magnetic field on axis, B (T):</td>
<td>5.2</td>
</tr>
<tr>
<td>Safety factor, $q_{\psi}$ *):</td>
<td>3</td>
</tr>
<tr>
<td>Fusion power, $P_{\text{fus}}$ (GW):</td>
<td>1.2</td>
</tr>
<tr>
<td>Average neutron wall load, $L_{\text{wall}}$ (MW/m²):</td>
<td>0.8</td>
</tr>
<tr>
<td>Troyon coefficient, g ($10^{-2}$Tm/MA):</td>
<td>1.5</td>
</tr>
<tr>
<td>Average plasma temperature, $&lt;T&gt;$ (keV):</td>
<td>10</td>
</tr>
<tr>
<td>Average plasma density, $&lt;n&gt;$ ($10^{20}$ m⁻³):</td>
<td>1</td>
</tr>
<tr>
<td>Maximum allowable helium concentration (%):</td>
<td>10</td>
</tr>
<tr>
<td>Impurity contamination (effective ion charge), $Z_{\text{eff}}$:</td>
<td>&lt;2</td>
</tr>
</tbody>
</table>

*) at 95% of the magnetic flux
The confinement properties of the NET plasma have been quantified assuming that NET operates in the H-regime with edge localized modes. In this regime, plasma energy losses are comparatively low and at the same time accumulation of impurities, in present experiments, can be kept within acceptable limits. The confinement scaling developed during the ITER Conceptual Design Phase for this case was adopted [4]. However, as there are still significant uncertainties in the extrapolation of this regime to NET conditions, a nominal margin of more than 2 for reaching the target of $C = 0.9$ was kept when applying the confinement scaling.

The reference plasma equilibrium was selected to be a double null configuration because of the flexibility of this configuration. Slight deviations from up-down symmetry allow to generate asymmetric double null ("semi-double null") as well as single null configurations with a vertical offset of the magnetic axis up to about 0.5 m. The plasma current, for the same value of the safety factor $q_y$ at the edge, is only slightly lower in these configurations, by about 5% in the most extreme case. The variety of configurations offering comparable confinement properties thus is larger then in a device designed around a strongly asymmetric single null configuration with a large axis offset. Furthermore, in an up-down symmetric plasma configuration the heat loads and particle fluxes are distributed over four plates (rather than two), and plasma position control is simplified as horizontal and vertical perturbations of the position are only weakly coupled.

For the basic performance phase, an external power, $P_{\text{ext}}$, of 70 MW is foreseen to allow for current profile control. This power also covers the needs for heating to burn conditions ($\approx 50$ MW), for burn temperature control (about 25 MW), plasma initiation, and soft discharge shutdown. The design of the NET device is compatible with adopting combinations of RF systems (electron cyclotron, lower hybrid, ion cyclotron waves) and high energy neutral beam injection.
TABLE I. NET OPERATING POINTS

<table>
<thead>
<tr>
<th></th>
<th>Full Current</th>
<th>Reduced Current</th>
<th>(\beta)-limit</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>full ignition</td>
<td>burn control</td>
<td>full ignition</td>
</tr>
<tr>
<td>I (MA)</td>
<td>25</td>
<td>25</td>
<td>23.8</td>
</tr>
<tr>
<td>B (T)</td>
<td>5.2</td>
<td>5.2</td>
<td>5.2</td>
</tr>
<tr>
<td>q((\psi=95%)</td>
<td>3</td>
<td>3</td>
<td>3.1</td>
</tr>
<tr>
<td>(P_{\text{ fus}}) (GW)</td>
<td>1.2</td>
<td>1.2</td>
<td>1.2</td>
</tr>
<tr>
<td>C (=P_\alpha/P_{\text{tot}})</td>
<td>1</td>
<td>0.9</td>
<td>0.9</td>
</tr>
<tr>
<td>(L_{\text{wall}}) (MW/m(^2))</td>
<td>0.8</td>
<td>0.8</td>
<td>0.8</td>
</tr>
<tr>
<td>(t_{\text{burn}}) (s)</td>
<td>1000</td>
<td>800</td>
<td>1000</td>
</tr>
<tr>
<td>(I_{\text{BS}}/I)</td>
<td>0.12</td>
<td>0.11</td>
<td>0.13</td>
</tr>
<tr>
<td>(I_{\text{CD}}/I)</td>
<td>0</td>
<td>0</td>
<td>0</td>
</tr>
<tr>
<td>g</td>
<td>1.6</td>
<td>1.6</td>
<td>1.6</td>
</tr>
<tr>
<td>(\beta) (%)</td>
<td>3.2</td>
<td>3.1</td>
<td>3.1</td>
</tr>
<tr>
<td>(\beta_{\text{pol}})</td>
<td>0.4</td>
<td>0.39</td>
<td>0.43</td>
</tr>
<tr>
<td>(&lt;T&gt;) (keV)</td>
<td>11.7</td>
<td>9.6</td>
<td>9.7</td>
</tr>
<tr>
<td>(&lt;\text{n}&gt;) ((10^{20}) m(^{-3}))</td>
<td>0.9</td>
<td>1.08</td>
<td>1.07</td>
</tr>
<tr>
<td>(Z_{\text{eff}})</td>
<td>1.8</td>
<td>1.71</td>
<td>1.71</td>
</tr>
<tr>
<td>(P_{\text{ex}}) (MW)</td>
<td>0</td>
<td>27</td>
<td>27</td>
</tr>
<tr>
<td>(P_{\alpha}) (MW)</td>
<td>241</td>
<td>242</td>
<td>241</td>
</tr>
<tr>
<td>(P_{\text{div}}) (MW)</td>
<td>123</td>
<td>131</td>
<td>131</td>
</tr>
<tr>
<td>(\tau_{\text{req}}) (s)</td>
<td>4.7</td>
<td>4.3</td>
<td>4.3</td>
</tr>
<tr>
<td>(\tau_{\text{H-ELM-90}}) (s)</td>
<td>6.7</td>
<td>6.5</td>
<td>6.3</td>
</tr>
<tr>
<td>(\tau_{\text{req}}) H-ELM-90</td>
<td>0.7</td>
<td>0.65</td>
<td>0.7</td>
</tr>
</tbody>
</table>

**definitions:**
- \(I_{\text{BS}}\) - bootstrap current;
- \(I_{\text{CD}}\) - current driven by external power;
- \(P_{\alpha}\) - \(\alpha\)-particle power;
- \(P_{\text{div}}\) - power exhausted through divertor;
- \(\tau_{\text{req}}\) H-ELM-90 - required and extrapolated energy confinement times.
Table I shows a selection of operating points of NET. The first four columns exemplify cases in which current profile control is assumed not to be necessary and, hence, no external power is used to drive current. The first and the fourth column refer to ideally ignited cases (C=1); the second and the third column correspond to C = 0.9, allowing for burn temperature control. The last three columns give examples of low C operation using external power for current profile control and for reaching high beta operation. Impurity seeding, to enhance radiation losses, is allowed for in the last three cases. At nominal plasma current (25 MA) there is a large margin with respect to the beta limit. Therefore, operation at appreciably higher wall loading keeping the plasma beta below the permissible limit (e.g., operating at 25 MA and g = 2.5, yielding $L_{\text{wall}} \approx 2 \text{ MW/m}^2$ and $P_{\text{fus}} \approx 3 \text{ GW}$) would in principle be possible in an enhanced performance phase.

6. DESIGN CHARACTERISTICS

The basic device, consisting of the superconducting (Nb$_3$Su) magnet systems and the containment structures, has been designed to operate through the basic and extended performance phases.

Vertical rather than horizontal access for maintenance of the blanket segments has been chosen for NET because it allows a better optimization of the affected systems; in particular the poloidal field coils can be located at positions which permit the plasma equilibria needed to be created more effectively.

The choice of the number of toroidal field coils is the result of a trade-off between toroidal magnetic field ripple considerations and maintenance requirements. Layout studies have led to the choice of 16 coils for NET: a higher number of coils would have made the maintenance procedures of the in-vessel components more difficult, a lower number would have unnecessarily increased the outer radius of the toroidal field coils and, consequently, the magnetic energy of poloidal and toroidal field systems.
For the plasma-facing components, which are subject to demanding operation conditions, remote replacement through horizontal ports in a minimum number of operations has been foreseen. For the basic performance phase, cooling by water at low pressure and temperature (< 3.5 MPa, < 150°C in normal operation) has been selected. The first wall structure can be integrated with or separated from the neutron shield. Two basic design concepts have been studied which differ in the way the low Z protective armour is provided. The first concept uses local limiters to take the loads due to fast particles, having tiles made of carbon fibre composites (CFC) brazed to heat sink tubes. These limiters are designed for heat loads of about 5 MW/m². The first wall proper, in this concept, would be covered by a low Z coating (B₄C or Be). In a second concept, CFC tiles are mechanically attached to the heat sink and are passively cooled via radiation or conduction. The divertor is made of CFC armour, 1 cm thick, of high thermal conductivity, brazed in "monobloc" geometry around cooling tubes in Mo-alloy or DS Cu. The divertor plates are designed to operate at a static peak divertor heat flux of up to 15 MW/m².

The device has been designed for full remote maintenance, with provisions for hands-on maintenance where possible. Three confinement barriers are present during operation to avoid release of radioactive material in case of accident.

Details of the design are documented in Ref. [1].

7. SUMMARY AND CONCLUSIONS

The primary role of NET is to establish a reactor plasma and to assess the basic performance of its components with a view to reactor applications. The guiding principles of the NET design have been a prudent extrapolation from present physics and technology data as well as maximum technical simplicity and safety considerations. The NET predesign has been completed,
supported by a comprehensive R&D programme in physics and technology. From this study the following main conclusions can be drawn:

- The priority chosen for the technical objectives, namely DT plasma ignition, extended burn and functional tests of main components, in the first phase of operation ("basic performance phase") is confirmed. The option of a second phase of extended performance in a more reactor relevant mode of operation should be pursued during the engineering design; the actual implementation of this phase will depend on the results of the basic performance phase.

- The design and the results from the R&D activity support the launching now of the engineering design as it was planned in the formulation of the EC Fusion Programme 1992/94.

- Key issues requiring an intense effort are the industrial fabrication and extensive testing of highly reliable and high performance superconducting magnets, the identification of a configuration of the plasma-facing components and of ways of power exhaust which are able to reduce the present high concentration of heat load on the walls facing the plasma.

- The preliminary safety analysis of the NET predesign indicates that the safety requirement of no disruption in community life in case of accidental release of radioactivity is achievable.

REFERENCES


T. TSUNEMATSU: Is it necessary to control such plasma parameters as the current density profile to sustain the plasma for periods of more than 1000 s?

F. ENGELMANN: Control of the plasma current profile will be necessary to ensure satisfactory MHD properties of the discharge. Extending the pulse duration to 300 s or more, that is, to a time comparable with, or larger than, the time characteristic for resistive diffusion of the current profile will require non-inductive current drive by external power. At present, it is not possible to definitely quantify the power needed for this purpose. In NET high density operation regimes, with 70 MW of external power, more than 10% of the plasma current can be driven non-inductively; the bootstrap current fraction is about 20%. This provides some capability for current profile control.
A HIGH-ASPECT-RATIO DESIGN OPTION FOR THE INTERNATIONAL THERMONUCLEAR EXPERIMENTAL REACTOR


U. S. ITER Home Team\textsuperscript{1},
United States of America

Abstract

A HIGH-ASPECT-RATIO DESIGN OPTION FOR THE INTERNATIONAL THERMONUCLEAR EXPERIMENTAL REACTOR.

Design features and performance estimates for HARD — the high-aspect-ratio ($A = 4$) International Thermonuclear Experimental Reactor (ITER) design variant developed by the U. S. ITER Team — are presented. Key physics and engineering considerations associated with increased aspect ratio are described. The HARD design will make it possible for ITER to achieve both the ignition/extended-burn and steady-state/technology-testing performance goals set forth in the ITER Terms of Reference. These capabilities are obtained in a device that is otherwise similar in concept, size and cost to the $A = 2.8$ ITER design defined by the Conceptual Design Activity. No significant physics or engineering difficulties associated with the increase in aspect ratio have been identified.

1. INTRODUCTION

The Programmatic Objectives defined in the ITER Terms of Reference\textsuperscript{1} encompass three major requirements: 1) achievement of ignition and extended ($t_{\text{burn}} \geq -200$ s) fusion burn, with steady-state as an ultimate goal, 2) an integrated demonstration of the technologies essential for a fusion reactor, and 3) operation to perform integrated testing of high-heat-flux and nuclear components required for future fusion reactors. The third objective is challenging, since it requires that ITER operate with high reliability and availability for extended periods to achieve a test-module fluence of $\geq 1$ MWa/m$^2$.

\textsuperscript{1} The activities of the U. S. ITER Team are supported by the U. S. Office of Fusion Energy. The lead author's U. S. ITER Team work at General Atomics and preparation of this paper are supported by Lawrence Livermore National Laboratory subcontract B089350.
The rationale for HARD (High Aspect Ratio Design), the high-aspect-ratio \((A = 4)\) ITER concept developed by the U. S. ITER Team as an alternate to the low-aspect-ratio \((A = 2.8)\) ITER design developed [2] during the ITER Conceptual Design Activity (CDA) lies in the third Programmatic Objective. The lower plasma current, increased bootstrap current fraction and higher on-axis toroidal field inherent in a high-A design gives HARD greater potential than ITER CDA for achieving the 1 MW/m² minimum fluence goal set by the third objective, and of proceeding on to the 3 MW/m² fluence goal set as an ultimate ITER design condition. HARD obtains this enhanced capability primarily by the relaxation of the operational constraints [3] imposed by the plasma beta limit and divertor power handling and erosion limitations [4]. The technology-testing gain is obtained without compromising ignition performance, and without undue increase in physics and engineering risk or device size and cost.

2. **DESIGN CONCEPT AND PERFORMANCE**

Except for higher aspect ratio, the HARD design is essentially identical in concept to the ITER CDA design, and is intentionally based on the same physics [4] and engineering [5] guidelines set for the CDA design. The cross-sections of the two designs are compared in Fig. 1. HARD employs the same

![Fig. 1. Cross-sections of HARD (left) and ITER CDA (right). Dimensions in meters.](image-url)
device layout, $k_{25} = 2$ double-null plasma configuration, and Nb$_3$Sn superconducting magnet technology [6] as ITER CDA, and also retains the same nuclear shielding concept and build and magnet construction and design allowables [6] as ITER CDA. Known deficiencies of the CDA design are duplicated in order not to obscure the effects of the change in aspect ratio.

The plasma size and current level for HARD are set by ignition requirements: a 15 MA current capability provides the same ignition performance (ignition margin evaluated against ITER-89P confinement scaling) as ITER CDA. Key device and performance parameters with inductively-driven ignition plasmas are compared in Table I. HARD achieves the same ignition performance as ITER CDA, but at lower plasma current and with much greater margin for inductive sustainment of the pulse. The inductive pulse length possible in HARD is comparable to the current profile relaxation time.

<table>
<thead>
<tr>
<th>TABLE I</th>
<th>ITER Designs Compared: Device Parameters and Ignition Performance</th>
</tr>
</thead>
<tbody>
<tr>
<td>Parameter</td>
<td>Units</td>
</tr>
<tr>
<td>A</td>
<td>—</td>
</tr>
<tr>
<td>R</td>
<td>m</td>
</tr>
<tr>
<td>B</td>
<td>T</td>
</tr>
<tr>
<td>$B_{Tf(max)}$</td>
<td>T</td>
</tr>
<tr>
<td>I</td>
<td>MA</td>
</tr>
<tr>
<td>$P_{fus}$</td>
<td>MW</td>
</tr>
<tr>
<td>$\Delta \Psi_r/F*$</td>
<td>V-s</td>
</tr>
<tr>
<td>$t_{burn}$</td>
<td>s</td>
</tr>
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</table>

*Total poloidal field system flux swing

<table>
<thead>
<tr>
<th>TABLE II</th>
<th>ITER Designs Compared: Steady-State Mode (1.3 MeV neutral beam current drive, $I_{ind} = 0$)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Parameter</td>
<td>Units</td>
</tr>
<tr>
<td>$P_{CD}$</td>
<td>MW</td>
</tr>
<tr>
<td>I</td>
<td>MA</td>
</tr>
<tr>
<td>$P_{fus}$</td>
<td>MW</td>
</tr>
<tr>
<td>Q</td>
<td>—</td>
</tr>
<tr>
<td>$f_{bs}$</td>
<td>—</td>
</tr>
<tr>
<td>$&lt;n_e&gt;$</td>
<td>$10^{20}$ m$^{-3}$</td>
</tr>
<tr>
<td>$T_{e,div}$</td>
<td>eV</td>
</tr>
<tr>
<td>$Q_{div}$</td>
<td>MW/m$^2$</td>
</tr>
<tr>
<td>$\Gamma_{n,test}$</td>
<td>MW/m$^2$</td>
</tr>
<tr>
<td>$t_{op}$</td>
<td>Years</td>
</tr>
</tbody>
</table>

* "Physics" heat loads and temperatures, w/o peaking factors
** Calendar time for 1 MWA/m$^2$, with 25% availability and 20% attenuation by test module first-wall
TABLE III
ITER Designs Compared: Hybrid Mode
(1.3 MeV neutral beam current drive, \( f_{\text{CD}} = 0.30 \))

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Units</th>
<th>CDA</th>
<th>HARD</th>
</tr>
</thead>
<tbody>
<tr>
<td>( P_{\text{CD}} )</td>
<td>MW</td>
<td>113</td>
<td>113</td>
</tr>
<tr>
<td>I</td>
<td>MA</td>
<td>15.6</td>
<td>10.2</td>
</tr>
<tr>
<td>( I_{\text{ind}} )</td>
<td>MA</td>
<td>9.2</td>
<td>3.9</td>
</tr>
<tr>
<td>( f_{\text{ind}}/f_{\text{bs}} )</td>
<td>—</td>
<td>0.59/0.11</td>
<td>0.38/0.32</td>
</tr>
<tr>
<td>( P_{\text{fus}} )</td>
<td>MW</td>
<td>1090</td>
<td>1080</td>
</tr>
<tr>
<td>(&lt;n_e&gt;)</td>
<td>(10^{20}) m(^{-3})</td>
<td>1.13</td>
<td>1.50</td>
</tr>
<tr>
<td>( T_{e,\text{div}} )</td>
<td>eV</td>
<td>11</td>
<td>17</td>
</tr>
<tr>
<td>( Q_{\text{div}} )</td>
<td>MW/m(^2)</td>
<td>5.0</td>
<td>2.5</td>
</tr>
<tr>
<td>( \Gamma_{n,\text{test}} )</td>
<td>MW/m(^2)</td>
<td>1.46</td>
<td>2.20</td>
</tr>
<tr>
<td>( t_{\text{burn}} )</td>
<td>s</td>
<td>3,100</td>
<td>22,000</td>
</tr>
</tbody>
</table>

**Pulses for:**

- 1 MWa/m\(^2\) — 8,500
- 3 MWa/m\(^2\) — 26,000

*“Physics” heat loads and temperatures, w/o peaking factors

**Average fluence at test module, with 20% attenuation by test module first-wall

With non-inductive current drive (113 MW of neutral beam injection at 1.3 MeV), HARD operates steady-state at lower plasma current (~11 MA) and higher plasma density and bootstrap current fraction than ITER CDA, is less constrained by beta limit and divertor considerations, and has increased peaking of the neutron wall load at the test module location (Table II).

The performance gain for HARD with non-inductive current drive arises directly from the increase in aspect ratio, which 1) allows an increase in on-axis field for the same superconducting magnet technology, and 2) reduces the plasma current, and 3) increases the fraction of the total current (\( f_{\text{bs}} \)) sustained by bootstrap current. The net result is a major reduction in the current (\( I_{\text{CD}} \)) that must be sustained by non-inductive drive. For fixed current drive power, this reduced current-drive requirement allows operation at higher plasma density, and hence mitigation of the divertor power loading and plasma temperature constraints that limit steady-state performance in ITER CDA.

Similar gains (Table III) accrue for hybrid operation, where 30% of the plasma current is driven non-inductively, with the balance sustained by bootstrap current and by inductive drive. The reduced inductive drive requirement and the increased inductive flux swing capability of the HARD central solenoid combine to provide pulse lengths of 22,000 s. The number of pulses needed to reach 1 MWa/m\(^2\) decreases to 800. This decrease in the number of
operational cycles reduces the concerns identified for the CDA design about the lifetime and reliability-reducing effects of mechanical and thermal fatigue in the magnet, plasma-facing components and in-vessel nuclear systems.

3. PHYSICS BASIS

HARD is based on CDA physics guidelines [4]. The same ITER 1989-P confinement scaling, confinement margins (H-factor), edge safety factor, plasma impurity model and operational limits — maximum beta and plasma density — are used. The only significant differences are the TF ripple, 0.5% peak-to-peak at the outer plasma boundary, rather than 1.5% for CDA, and the plasma internal inductivity, $0.65 \leq l_i(3) \leq 1.0$ rather than $0.55 \leq l_i(3) \leq 0.75$ as set for CDA. The reduced ripple follows from considerations of allowable $\alpha$-particle ripple loss at higher aspect ratio; the inductivity range follows from considerations of optimal MHD stability at high beta and higher aspect ratio.

The impact of $A = 4$ on plasma equilibrium and axisymmetric stability is acceptable. The PF coil system allows operation over the full range of profile parameters needed for all three operational modes, with only minor restrictions at high $\beta_p$ ($> \sim 2$). Total PF system magnetic energy increases only slightly relative to CDA: 16 GJ at end-of-burn for HARD versus 14.8 GJ for CDA. Vertical instability growth rate approximately doubles, to $55 \text{s}^{-1}$ for HARD, with a corresponding doubling of peak control power to ~2 GW. The same plasma control technology planned for CDA [7] is adequate for HARD, but inboard (small-R) and outboard (large-R) twin loop in-vessel resistive stabilizing structures are required to maintain acceptable stability properties over the range of plasma $\beta_p$ and inductivity expected during HARD operations. The need for inboard twin-loops in HARD (and also in ITER CDA) derives from analysis [8] of non-rigid plasma deformations, which increase the instability growth rate in both designs relative to the growth rate calculated from the rigid-deformation "wire-array" model [7] employed during the CDA.

Peaking of neutron wall loading at the outboard midplane where the test modules are located is 1.75 for HARD versus 1.46 for CDA. The increased peaking factor combined with higher average wall loading results in a 3.5-x increase in test-module wall loading for HARD relative to CDA (1.98 MW/m² versus 0.57 MW/m²).

4. ENGINEERING BASIS

The HARD TF and PF magnet systems employ the same Nb₃Sn cable-in-conduit superconductor as ITER CDA. The average current density in the TF winding pack is reduced to 29 A/mm², 18% lower than for ITER CDA, to allow the higher peak field needed for HARD. The structural concepts for the TF and PF magnets are identical those for ITER CDA. A finite-element stress
analysis of the HARD TF magnet system shows peak stress levels of 510 to 630 MPa, comfortably within the 800 MPa allowable.

The modest increase in TF coil outer leg radius (+0.55 m) required to meet ripple requirement facilitates tangential access for neutral beam injection. The same 1.3 MeV negative-ion beamline design as for ITER CDA [9] is used, with 12 beamlines grouped in triplets on 4 tangential injection ports. Assembly and maintenance of the in-vessel blanket/shield and divertor plate modules uses the same scheme as for ITER CDA. Facility requirements for HARD are essential identical to those for ITER CDA.

5. SAFETY, RELIABILITY AND AVAILABILITY

HARD is, on the balance, expected to be somewhat safer and more reliable than ITER CDA. From a safety viewpoint, reduction in the number of operational cycles and likely in the number of disruptions reduces the number of accident-initiating events. Thermal and magnetic loadings on in-vessel components and surfaces are also generally lower: this reduces the chances of component failure and the severity of the consequences. HARD ignition operation will also require less tritium inventory: this reduces at-risk inventory. These positive safety aspects of HARD must be balanced against the negative aspects of higher TF magnet stored energy (72 GJ for HARD versus 42 GJ for CDA) and the increased activation and decay afterheat that occur in the testing phase owing to the higher wall loading.

The reduced number of operational cycles and/or reduced stress levels of HARD operation can also be expected to improve reliability and availability relative to ITER CDA. This assessment is only semi-quantitative: future study of the effect of cyclic versus steady-state operation on plasma and component reliability is needed for better quantification.

6. COSTS AND R&D REQUIREMENTS

Total device direct cost is estimated to be approximately $4.2 billion, or about 10% higher than for ITER CDA. The cost increase is almost entirely from the more-massive magnet systems and their associated power supplies. Physics and technology R&D requirements are essential identical to those for ITER CDA. The principal exception is development of higher frequency (~180 GHz) gyrotron sources for electron cyclotron heating and current drive.

REFERENCES

DISCUSSION

R.W. CONN: At the previous IAEA conference in Washington in 1990, the Japanese group reported on the Steady State Tokamak Reactor (SSTR)\(^1\) and our US group reported on ARIES\(^2\). Both studies reported the advantages of higher aspect ratio, lower plasma current, and higher on-axis magnetic field for steady reactors. You now come to the same conclusion. Looking to the future, ITER may increase \(R\) and \(I_p\) to gain margin, but still at an aspect ratio of about 3. Given your conclusions, what direction do you recommend for the ITER Engineering Design Activities (EDA) phase?

J. WESLEY: The tendency of the ITER design to invoke higher \(R\) and \(I_p\) in order to increase margin is opposite to what one wants to improve steady state and high fluence testing capability. Attempting to reconcile the need for both more margin and future steady state performance in the same design raises difficult problems. I hope that both issues can be given objective consideration by the ITER Joint Central team and the national ITER home teams very early in the EDA, before the final choice of \(R\), \(A\), \(B\) and so on is made. I believe that some increase in aspect ratio to \(-3.5\) can be justified, even with some cost increase, as an investment for future success in ITER Phase Two.

Y. SHIMOMURA: The most important features of a future steady state reactor such as SSTR are high current density and high stress in the toroidal coils. If we assume the same engineering as in SSTR, we can also design a steady state reactor with a relatively low aspect ratio of about 3.

J. WESLEY: I agree that an \(A = 3\) steady state reactor can be designed if very high performance magnets are used. However, higher \(A\) reduces the need for improved magnets, so one must weigh up how much magnet technology improvement beyond 'ITER levels' is needed. For our HARD study, we intentionally avoided 'advanced magnet' technology.

H.L. BERK: Why do you use two different magnetic fields for the 'identical' set of tokamaks?

\(^2\) CONN, R.W., NAJMABADI, F., ibid., p. 659.
J. WESLEY: The high A design naturally optimizes at higher on-axis field (7.1 versus 4.85 T) and higher in-magnet field (13.3 versus 11 T). However, the underlying superconductor design allowables are the same. One readjusts the magnet design (\( \langle j \rangle \), superconductor/copper/steel/helium ratio) to take advantage of the greater radial build inherent in higher A.

H.L. BERK: Would increased magnetic field in the low aspect ratio tokamak give it better reactor characteristics?

J. WESLEY: Increased magnetic field in a low A design would relax the beta limit but would not affect the divertor constraint, which requires a reduction of \( I_p \). Increased magnetic field would also require ‘improved’ magnet technology, since the radial build is already fully utilized.

R.J. HAWRYLUK: You have performed a two point comparison (A = 2.8 and 4) and identified the relative advantages and disadvantages. Have you carried out an optimization study and determined the optimum aspect ratio?

J. WESLEY: The choice of the A = 4 design point for HARD was based on systems code modelling to find a design with adequate steady state performance and accessible ‘engineering difficulty’, i.e. vertical control. A = 4 seems close to an optimal choice (>4 becomes more difficult), but we did not do a formal optimization of this aspect. Studies that optimize (minimize) total device cost for given ignition performance and steady state or hybrid fluence tend to optimize at about A = 3.5, in designs that are slightly smaller than CDA or HARD. J. Perkins of Lawrence Livermore National Laboratory and J. Galambos of Oak Ridge National Laboratory have done such studies and can provide more details.

Y. SHIMOMURA: At JAERI we have also studied a high A option based on the same assumptions as those of the ITER CDA. We believe that vertical control is very important and therefore we employ the same stability condition. In a higher A device with a smaller minor radius, the relative distance between the plasma and the stabilizer structure is wider. Therefore, the elongation must be reduced in order to keep the same vertical stability condition. When this effect is included, the advantage of high A is not significant and we also encountered other engineering difficulties as in the high A option for ITER.

J. WESLEY: Your choice at JAERI of keeping the same stability margin as the CDA leads to lower elongation and less favourable overall conclusions about the benefits of increased A. We conclude from our study that it is possible to retain \( k_{95} = 2 \) and the double null configuration, albeit with higher control power. This approach avoids the poloidal field system difficulties you found in your design.
ITER: ANALYSIS OF THE H-MODE CONFINEMENT AND THRESHOLD DATABASES

H-MODE DATABASE WORKING GROUP

(Presented by O. Kardaun)

Abstract

ITER: ANALYSIS OF THE H-MODE CONFINEMENT AND THRESHOLD DATABASES.

The collaboration between the six tokamak teams that led to a joint global confinement database has been continued. In the paper some features of the second version of this database (ITERH.DB2) are discussed, and an analysis of the thermal confinement time scaling based on these data is provided, which may be interesting from a scientific point of view (the database and its analysis can serve as an empirical benchmark for various plasma physical theories and can also be useful for predicting the performance of future machines). In addition, a threshold database (ITERTH.DB1) has been assembled by the same working group, of which some preliminary analysis providing information complementary to the global confinement scalings is presented.

1. NEW FEATURES OF THE H-MODE CONFINEMENT DATABASE

In Refs [1, 2] the results of an international effort to establish a global H-mode confinement database from six tokamaks have been reported. The database (ITERH.DB1) has been released for general use, and several analyses have been performed [1–5]. From these analyses, also some limitations of the dataset became obvious. Hence, the working group decided to improve the dataset in order to extend, as well as to increase the precision of some of the relevant plasma parameters.

As is described in Ref. [6], the new database (ITERH.DB2) has been expanded with respect to ITERH.DB1 by adding ion cyclotron resonance heating (ICRH), combined ICRH and neutral beam injection (NBI), beryllium conditioned, and hot ion mode discharges from JET, a B scan, a low-q scan, double null, carbon conditioned, hot ion mode, and ECH discharges from DIII-D, several B scans with

1 Advanced Fusion Research Center, Kyushu University, Kasuga, Japan.

2 National Institute for Fusion Science, Nagoya, Japan.
ELM free and ELMy discharges under boronized wall conditions from ASDEX, and ELMy H-mode discharges produced with an ergodic magnetic field from JFT-2M. For PDX, additional confinement measurements ($W_{\text{dia}}$ and its derivative) have been provided. In comparison with DB1, the type of H-mode (ELMy/ELM free) and the isotope composition (H, D, and 'mixed') are distributed more equally over the six tokamaks.

Furthermore, simultaneous efforts have been made to arrive at better estimates of several quantities, such as the fast particle contents and their anisotropy, and the power losses due to charge exchange and unconfined orbits, so that more precise estimates of the thermal confinement times are provided than were available in ITERH.DB1.

2. SELECTION CRITERIA FOR THE STANDARD DATASET

The total ITERH.DB2 dataset consists of about 3400 H-mode time slices and 2600 Ohmic and L-mode reference points. A set of selection criteria [2] has been formulated to produce a 'standard subset' of H-mode time slices, called the standard dataset. With some extensions induced by the introduction of new possibilities, basically the same criteria are applied to form the standard dataset of DB2.

As in DB1, the standard selection criteria exclude time slices where the plasma is not in the H-mode and, in addition, exclude time slices with high radiation, $(P_{\text{rad}}/P_{\text{tot}} > 0.6)$, with high fast particle content $(W_f/W > 0.4)$, in transient phase $(\dot{W}/P > 0.35$ or $<-0.05)$, with suspected confinement degradation from sawtooth effects, $(B(T)/I(MA) < 1$ for DIII-D [7] and $q_{95} < 3$ for various other tokamaks), or close to the beta limit (for PDX and PBX-M). For DB2, no restrictions on wall material, evaporation, isotope composition or on heating method are imposed, except for the fact that helium shots from JET and ECH shots from DIII-D are not included in the standard dataset.

For the convenience of the prospective user, two variables have been added to the dataset, from which one can easily identify the shots that are in the standard dataset as well as shots that satisfy various individual selection criteria. The ranges of the main plasma parameters in the ITERH.DB2 dataset are presented in Table I. A more precise description of the selection criteria as well as many interesting descriptive tables and plots of the data can be found in Ref. [6].

The standard dataset is a practical compromise of the conflicting requirements of including a large range of the data and at the same time excluding portions that are suspected to scale differently from the rest. As such, the criteria should not be considered as sacred: some relaxations or additional restrictions may certainly yield meaningful scalings. However, the standard dataset serves as a sensible starting point in the discussion of various regression analyses.
TABLE I. DATA RANGES IN ITERH.DB2 (STANDARD DATASET, WITHOUT q-RESTRICTION) AND NUMBERS OF ELM FREE ($N_{el}$) AND ELMy ($N_{el}$) OBSERVATIONS IN THE STANDARD DATASET

<table>
<thead>
<tr>
<th></th>
<th>$I_p$</th>
<th>$B_t$</th>
<th>$P_{li}$</th>
<th>$M$</th>
<th>$R$</th>
<th>$\kappa$</th>
<th>$a/R$</th>
<th>$n_e$ ($10^{19}/m^3$)</th>
<th>$N_{el}$</th>
<th>$N_{el}$</th>
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</thead>
<tbody>
<tr>
<td>ASDEX</td>
<td>0.20-0.43</td>
<td>1.5-2.8</td>
<td>1.2-4</td>
<td>1.5-2</td>
<td>1.65</td>
<td>0.93-1.0</td>
<td>0.21-0.26</td>
<td>1.9-6.8</td>
<td>91</td>
<td>292</td>
</tr>
<tr>
<td>DIII-D</td>
<td>0.43-2.8</td>
<td>1.1-2.1</td>
<td>0-17.5</td>
<td>1-2</td>
<td>1.68</td>
<td>1.7-2.1</td>
<td>0.35-0.38</td>
<td>1.2-12.8</td>
<td>45</td>
<td>176</td>
</tr>
<tr>
<td>JET</td>
<td>1.5-5.1</td>
<td>1.1-3.7</td>
<td>0-18.2</td>
<td>1.5-2</td>
<td>2.8</td>
<td>1.6-1.9</td>
<td>0.34-0.41</td>
<td>1.2-6.4</td>
<td>437</td>
<td>142</td>
</tr>
<tr>
<td>JFT-2M</td>
<td>0.14-0.19</td>
<td>0.8-1.3</td>
<td>0.18-1.6</td>
<td>1-1.5</td>
<td>1.3</td>
<td>1.3-1.5</td>
<td>0.19-0.22</td>
<td>2.1-8.2</td>
<td>247</td>
<td>59</td>
</tr>
<tr>
<td>PBX-M</td>
<td>0.28-0.38</td>
<td>1.3-1.4</td>
<td>1.4-4.8</td>
<td>2-2</td>
<td>1.65</td>
<td>1.5-1.7</td>
<td>0.16-0.18</td>
<td>2.4-9.0</td>
<td>159</td>
<td>61</td>
</tr>
<tr>
<td>PDX</td>
<td>0.24-0.46</td>
<td>1.0-2.0</td>
<td>0.94-4.5</td>
<td>1.5-2</td>
<td>1.4</td>
<td>1.0-1.0</td>
<td>0.29-0.30</td>
<td>2.5-9.3</td>
<td>22</td>
<td>97</td>
</tr>
</tbody>
</table>
3. CONDITION OF THE STANDARD DATASET

Before performing regression analysis, it is good to have an impression of the stability of the regression estimates against measurement errors. This is also called the 'condition' of the database. For the random ('shot to shot') component of the errors, this is assessed by principal component analysis. The interested reader is referred to Ref. [2] for a more detailed discussion of such an analysis, applied to the ITERH.DB1 database. Here we state the main results for ITERH.DB2.

The eight main variables we will use in the confinement time analysis are plasma current $I_p$, magnetic field $B_t$, loss power $P_L$, line average electron density $n_e$, major radius $R$, elongation $\kappa$, inverse aspect ratio $a/R$ and effective atomic mass number, $M$, of the (main) plasma and beam isotopes. For these eight variables, the direction of the smallest principal component of the ELM free standard dataset in DB2 is proportional to an arbitrary power of $R^{0.3}B^{0.85}I_p^{0.3}(a/R)^{0.7}$. (We rounded to nearest value of 0.1 and left out exponents smaller than 0.2). One standard deviation of the data along this direction (measured on the natural logarithmic scale) is 5.5%, which is about five times the assumed measurement error [2] in that direction. Note that the smallest principal component does not involve density or loss power.

This result means that the ELM free standard dataset in DB2 is about equally well conditioned with the atomic mass number included as a regression variable, as the ELM free standard dataset in DB1 without the atomic mass number included.

It is noticed that the direction of the smallest principal component in DB1 corresponds to the direction of the second smallest principal component in DB2, which is proportional to an arbitrary power of $R^{0.9}B^{0.6}/R$. In this direction, the data variation (one standard deviation) in the new standard dataset is 8%.

The condition of the ELMy standard dataset is, as in DB1, slightly better than that of the ELM free standard dataset: the least principal component is proportional to $R^{0.5}B^{0.1}I_p^{0.4}(a/R)^{0.7}$, with a variation of 5.8%, and the second smallest principal component to $\kappa^{1.4}B^{0.85}I_p^{0.3}(Rn_e^{0.4})$, with a variation of 10%. If the ELM free and the ELMy standard datasets are combined, the condition does not change appreciably, the two smallest variations being 6% and 9.5%, respectively.

The result of this analysis is that, as before in DB1 without the atomic mass, the conditions of both the ELM free and the ELMy standard datasets are good enough to give a stable regression against random ('shot to shot') measurement errors of a more or less realistic size [2]. This does not mean, however, that the regression results would be stable against systematically omitting large parts of the data (e.g. all the data from one tokamak), nor that it will be stable against important systematic tokamak-to-tokamak differences in assessing the confinement time. These aspects have to be investigated separately.

The expression 'not stable' means in this context that easily shifts of, say, four or six standard errors can occur in the regression estimates. It does not necessarily imply that the prediction for ITER is affected by such an amount. The design parameters of the conceptual ITER design [8] are only one standard deviation away
from the centre of the database in the direction of the smallest principal component, and four standard deviations in both the directions of the second and the third smallest principal component. This means that the smallest principal component in DB2 is not a critical one for the prediction of ITER.

4. POWER LAW SCALINGS

4.1. Main scalings

In this section, we present ELM free and ELMy power law scalings for the thermal confinement time \( \tau_{E,\text{th}} = \frac{W_{\text{th}}}{P_L} \) that take the usual variables into account, while adjusting for the influence of some additional variables. The most important of these is the difference (open/closed) in divertor type. A closed divertor provides a better retention of neutral particles in the divertor. This is known to have a favourable influence on the confinement time in, at least, the smaller tokamaks \([9, 10]\).

The estimates of the thermal energy contents \( W_{\text{th}} \) are based on \( W_{\text{DIA}} \) for ASDEX and JET, \( W_{\text{MHD}} \) for DIII-D, PBX-M, and PDX, and both \( W_{\text{DIA}} \) and \( W_{\text{MHD}} \) for JFT-2M, taking into account Monte Carlo based estimates of the fast particle contributions and their anisotropy. Effects of plasma rotation are neglected. These choices are considered to yield the best practical estimates at present. They will be discussed in a forthcoming report in greater detail.

The variable \( P_L = P_{\text{aux}} + P_{\text{ohm}} - P_{\text{sh}} - P_{\text{cx}} - P_{\text{ol}} - \frac{dW_{\text{tot}}}{dt} \), with \( P_{\text{aux}} = P_{\text{inj}} + P_{\text{ICRH}} + P_{\text{ECH}} \) denotes the power \( P_L \) used in Ref. \([2]\) minus the power loss from charge exchange, \( P_{\text{cx}} \), and from unconfined orbits, \( P_{\text{ol}} \). As \( W_{\text{tot}} = W_{\text{th}} + W_{\text{fp}} \), the last term in this expression takes the rate of change of the energy in the plasma and in the beam into account.

For ELM free discharges, the thermal scaling ("ITERH92-P") is

\[
\tau_{E,\text{th}} = 0.032 \, I_p^{0.95} B_T^{0.20} P_L^{0.65} M^{0.45} R^{1.95} \alpha^{0.65} (a/R)^{0.05} n_e^{0.30}
\]  

(1)

The regression has been based on the standard ELM free dataset (\( N = 1001 \)). The units used are MA, T, MW, , m, , , \( 10^{19} \, \text{m}^{-3} \). The coefficients have been rounded to entire multiples of 0.05. A plot of the observed versus predicted confinement times in the dataset is presented in Fig. 1. The root mean square error (rmse) from the fit is 14.9%. The corresponding t-values, i.e. the ratios between the estimated regression coefficients and their standard deviations estimated from ordinary least squares can be found in Table II.

We notice a strong \( R \) dependence in Eq. (1). If no density dependence is assumed, the \( R \) dependence drops to 1.4, which is about the same \( R \) dependence as in Ref. \([5]\). However, the rmse increases to 16.2%. This topic will be discussed further in the next subsection.
FIG. 1. Observed versus predicted thermal energy confinement times of ELM free standard dataset.
TABLE II. GLOBAL H-MODE SCALINGS, BASED ON THE STANDARD DATASET (POWER LAW EXPONENTS). (A number in brackets denotes the ratio between exponent and its estimated standard error. The first row represents the basic $W_{\text{in}}$ scaling, the other rows represent ancillary scalings. Other scalings are obtained by substitution, i.e. by adding the corresponding exponents of an ancillary scaling to those of the basic scaling. Column C contains multiplicative constants.)

(A) ELM free, $N = 1001$

<table>
<thead>
<tr>
<th>C</th>
<th>$I_p$</th>
<th>$B_i$</th>
<th>$P_L$</th>
<th>$M$</th>
<th>$R$</th>
<th>$\kappa$</th>
<th>$a/R$</th>
<th>$n_e$</th>
<th>rmse</th>
</tr>
</thead>
<tbody>
<tr>
<td>$W_{\text{in}}/200$ MW</td>
<td>4.7</td>
<td>0.95</td>
<td>0.20</td>
<td>0.35</td>
<td>0.45</td>
<td>1.95</td>
<td>0.65</td>
<td>-0.05</td>
<td>0.30</td>
</tr>
<tr>
<td></td>
<td>(32)</td>
<td>(9)</td>
<td>(20)</td>
<td>(14)</td>
<td>(32)</td>
<td>(10)</td>
<td>(1)</td>
<td>(14)</td>
<td></td>
</tr>
<tr>
<td>$W_{\text{soc}}/W_{\text{th}}$</td>
<td>0.95</td>
<td>-0.12</td>
<td>+0.12</td>
<td>+0.13</td>
<td>+0.09</td>
<td>-0.15</td>
<td>-0.15</td>
<td>-0.15</td>
<td>-0.06</td>
</tr>
<tr>
<td></td>
<td>(-9)</td>
<td>(7)</td>
<td>(22)</td>
<td>(7)</td>
<td>(-5)</td>
<td>(-6)</td>
<td>(-3)</td>
<td>(-30)</td>
<td></td>
</tr>
<tr>
<td>$(P_L/P_L)^{0.35}$</td>
<td>1.05</td>
<td>-0.02</td>
<td>+0.04</td>
<td>-0.02</td>
<td>-0.05</td>
<td>+0.10</td>
<td>+0.13</td>
<td>+0.11</td>
<td>+0.06</td>
</tr>
<tr>
<td></td>
<td>(-4)</td>
<td>(5)</td>
<td>(-8)</td>
<td>(-10)</td>
<td>(8)</td>
<td>(13)</td>
<td>(14)</td>
<td>(17)</td>
<td></td>
</tr>
<tr>
<td>$(n_e/12.5 \times 10^{19})^{0.3}$</td>
<td>0.72</td>
<td>+0.13</td>
<td>-0.11</td>
<td>+0.09</td>
<td>+0.4</td>
<td>-0.55</td>
<td>-0.11</td>
<td>-0.12</td>
<td>6.2%</td>
</tr>
<tr>
<td></td>
<td>(11)</td>
<td>(-6)</td>
<td>(15)</td>
<td>(3)</td>
<td>(-24)</td>
<td>(-5)</td>
<td>(-6)</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

(B) ELMy, $N = 833$

<table>
<thead>
<tr>
<th>C</th>
<th>$I_p$</th>
<th>$B_i$</th>
<th>$P_L$</th>
<th>$M$</th>
<th>$R$</th>
<th>$\kappa$</th>
<th>$a/R$</th>
<th>$n_e$</th>
<th>rmse</th>
</tr>
</thead>
<tbody>
<tr>
<td>$W_{\text{in}}/200$ MW</td>
<td>3.8</td>
<td>0.90</td>
<td>0.05</td>
<td>0.35</td>
<td>0.40</td>
<td>2.1</td>
<td>0.80</td>
<td>+0.20</td>
<td>0.30</td>
</tr>
<tr>
<td></td>
<td>(29)</td>
<td>(2)</td>
<td>(24)</td>
<td>(14)</td>
<td>(31)</td>
<td>(17)</td>
<td>(4)</td>
<td>(15)</td>
<td></td>
</tr>
<tr>
<td>$W_{\text{soc}}/W_{\text{th}}$</td>
<td>0.92</td>
<td>-0.07</td>
<td>+0.05</td>
<td>+0.14</td>
<td>+0.11</td>
<td>-0.23</td>
<td>-0.30</td>
<td>-0.09</td>
<td>-0.25</td>
</tr>
<tr>
<td></td>
<td>(-5)</td>
<td>(4)</td>
<td>(24)</td>
<td>(8)</td>
<td>(-8)</td>
<td>(-15)</td>
<td>(-4)</td>
<td>(-30)</td>
<td></td>
</tr>
<tr>
<td>$(P_L/P_L)^{0.35}$</td>
<td>1.15</td>
<td>-0.05</td>
<td>+0.06</td>
<td>0.005</td>
<td>-0.09</td>
<td>+0.14</td>
<td>+0.21</td>
<td>+0.06</td>
<td>+0.04</td>
</tr>
<tr>
<td></td>
<td>(-11)</td>
<td>(14)</td>
<td>(2)</td>
<td>(-18)</td>
<td>(14)</td>
<td>(30)</td>
<td>(8)</td>
<td>(14)</td>
<td></td>
</tr>
</tbody>
</table>
To derive this scaling, we used $\tau_{E,th}/C_{\text{cond}}C_{\text{div}}$ in the regression analysis, where $C_{\text{cond}}$ and $C_{\text{div}}$ are estimated constants to adjust for the effects of wall conditioning and divertor type of particular machines in the database. As these effects may depend on the size of the machine, the constants are not intended and may not be valid for ITER. Their purpose is only to derive a scaling for open divertors under ‘standard’ conditions.

We estimated from the dataset that $C_{\text{cond}} = 0.85 \pm 0.03$ for the 20 JET shots from the December 1987 campaign, in which no discharge cleaning was performed. This value is stable under various changes in the regression and in accordance with Ref. [11], which was based on an analysis of JET data only. For all other data, $C_{\text{cond}} = 1$.

Following the approach in Ref. [3], since ITER and most of the tokamaks in the database are of an open divertor type, we ‘normalize’ the data from the closed divertor tokamaks (ASDEX and PDX). In part this has to be based on information external to the present database. We take $C_{\text{div}} = 1$ for open divertors, and estimate, from Ref. [12], that $C_{\text{div}} = 1.2$ for the closed divertor (DV-I) ELM free ASDEX discharges. For the second type of closed divertor (DV-II), with boronized wall conditioning, the confinement in the database is roughly 10% lower than for the first type of divertor (on the assumption of a square root isotope dependence). For these ASDEX discharge, we take $C_{\text{div}} = 1.1$. No normalization was made for PDX as in the present ELM free PDX dataset there is no clear dependence of the confinement on the ratio of the midplane and divertor $D_n$ signals, and it was uncertain whether the study by Fonck et al. [9] would also apply to ELM free data. As only 22 time slices are involved, the normalization is less important here than in the ELMy case with 97 time slices from PDX.

The effect of introducing the two normalizing constants is that it changes the non-normalized scaling by $0.99B^{-0.05}R^{0.05}\kappa^{0.15}$, while reducing the rmse by 0.4%, and the prediction for ITER by 5%. Note that the kappa dependence is affected by a fair amount as ASDEX is a circular machine.

An application of the regression to the ELMy data in the standard dataset ($N = 833$) gives the scaling

$$\tau_{E,th} = 0.034 I_p^{0.90}B_t^{0.05}P_L^{0.65}M^{0.40}R^{2.1}\kappa^{0.80}(a/R)^{0.20}n_c^{0.30}$$

with an rmse of 14.6%. There are no ELMy JET shots from the December 1987 campaign. In the regression we used $\tau_{E,th}/C_{\text{evap}}C_{\text{div}}$ with, from Ref. [10], $C_{\text{div}} = 1.5$ as a rough estimate of the closed divertor effect for ELMy ASDEX discharges, and, from Ref. [13], $C_{\text{evap}} = 0.85$ for the carbonized ASDEX shots. The last factor can also be extracted from the present dataset. We used $C_{\text{evap}} = 0.9$ for the boronized DV-II shots from ASDEX. (We do not know whether this factor is due to the evaporation or to the difference between DV-I and DV-II, but this is not important for the present purpose.) For PDX, the ratio of the divertor and midplane $D_n$ signals is available in the database and can be used as a measure of the neutral particle retention.
H-MODE SCALING
(ELMy)

\[ 0.034 L^{0.9} T^{0.05} P_{L}^{-0.65} M^{0.4} R^{2.1 \kappa} (a/R)^{0.8} r_e^{0.3} \]

FIG. 2(a). Observed versus predicted thermal energy confinement times of ELMy standard dataset.
of the divertor. Specifically, we applied $C_{\text{div}} = (D_{\alpha, \text{div}}/3D_{\alpha, \text{MP}})^{0.4}$. The exponent, $0.4 \pm 0.04$, has been derived from regression of the ELMy PDX data in the database and is in agreement with Ref. [9], where the dependence on the neutral gas pressure has been reported. The ratio between the two $D_{\alpha}$ signals from PDX varies in the database between, roughly, 2 and 6. The ratio of 3 has been set, somewhat arbitrarily, to correspond to an open divertor.

Now, the effect of normalization is obviously larger than in the ELM free case: the non-normalized ELMy scaling is changed by $1.141^{0.2}B_{T}^{-0.2}R^{-0.5}\kappa^{0.4}(a/R)^{-0.2}\alpha_{e}^{-0.1}$, while the rmse is reduced by 0.7%. Such a reduction in rmse is statistically significant.

It is noted that normalization reduces the rmse of both the ELM free and the ELMy scaling, and that after normalization the ELMy scaling is more nearly proportional to the ELM free scaling than before the normalization. After normalization,
the ratio between ELMy and ELM free scalings is close to \((B_T/\alpha)^{-0.15}T_p^{-0.05}\), which is not a very strong parameter dependence. See Fig. 2(b), where the ratio of the normalized ELMy confinement times, i.e. \(\tau_{E,EHL}/\tau_{E,ELM}\), and the ELM free scaling is plotted against this parameter. According to the non-normalized scaling, the ELMy prediction for ITER at 22 MA, 4.85 T, 200 MW, \(M = 2.5\), 6 m, 2.2, 2.15/6, \(12.5 \times 10^{19} \text{ m}^{-3}\) is 4.7 \((\pm 0.3)\) s, which is the same as the ELM free prediction (4.7 s). (In brackets \pm 1 standard deviation of the prediction, arising solely from ‘random’ measurement errors in \(\tau_{E,EHL}\), on the assumption that the power law model used is correct.) According to the normalized scaling, the ELMy prediction is about 3.8 \((\pm 0.3)\) s, which is about 15% smaller than the corresponding ELM free prediction (4.5 s).

Normalizing the closed divertors to open ones tends to yield scalings that correspond better to reported confinement losses due to the presence of ELMs. If ELMy H-mode operation is considered to be a serious candidate for ITER, then a more precise assessment of the difference in influence between open and closed divertors on the confinement than presented here, possibly not only via a multiplicative factor, is of predictive value. The above analysis is just one example for the interesting aspects opened by the present database.

4.2. Ancillary scalings and extensions

For purposes of comparison or other reasons, one is sometimes also interested in the scaling of confinement times that are defined differently, e.g. \(W_{th}/P_L\), or \(W_{tot}/P_L\), with \(W_{tot} = W_{th} + W_{fp}\), or \(W_{tot}/P_{L'}\), or \(W_{th}/P_{L'}\), where \(P_{L'} = P_L - P_{rad}\), etc. The many combinations possible can conveniently be constructed from Table II, which contains, for both the ELM free and the ELMy case, a basic thermal scaling and some ancillary scalings. These scalings have been obtained directly by regression of the variables in the standard dataset. The rmse of each separate regression is presented in the last column. Using Pythagoras’ theorem in \(N = 1001\) dimensions [14], one can derive the following useful interpretation: we simply have to add one or more ‘ancillary’ coefficients to the corresponding basic coefficients to obtain a scaling for \(W_{th}\) or \(W_{tot}\). Clearly, we have to subtract \(-1\) from the power exponent to obtain the scaling of the corresponding confinement time. (The rmse values are not provided with a + or \(-\) sign, and are not to be added.) From Table II, we see, for example, that the \(P_L\) exponent of the \(W_{tot}\) scaling is more favourable than that for \(W_{th}\) by an amount of 0.13 in the ELM free and 0.14 in the ELMy case. Similarly, the estimated \(R\) exponent of the \(\tau_{E,tot} = W_{tot}/P_L\) scaling for ELMy shots equals 2.1 - 0.23 + 0.14 = 2.01. Note that the scaling of \((P_L/P_{L'})^{0.35}\) has been given because 0.35 corresponds to the exponent of \(P_{L'}\) in the basic scaling.

For both ELM free and ELMy discharges, \(W_{tot}\) has more favourable \(P_L\) and \(M\), but less favourable \(R\), \(\kappa\), and \(n_e\) exponents than \(W_{th}\). We note that there is no density dependence in the scaling of \(W_{tot}\).
In Table II, the formal scaling of the electron density with the other plasma parameters has also been given. We see that in the database the density decreases with the major radius. This entails that a scaling without density dependence, but otherwise the same variables, has a smaller R exponent (1.40 instead of 1.95). However, the density exponent in the $\tau_{E,th}$ regression differs by 14 standard deviations from zero, in accordance with the increase of 1.3% in rmse in the total fit. We note that the scalings for $W_{tot}/W_{th}$ and $P_{L'}/P_L$ are determined to a higher accuracy than the basic thermal scaling. The scaling of $P_{rad}/P_L$ can be handled in the same way and will be derived as soon as the data are complete.

The scalings mentioned above have been derived from the standard ELM free and ELMy datasets. In the ELM free case, relaxing the q restriction does not lead to a large change of the scaling: all coefficients remain the same within one standard deviation, except those for $B_t$, $P_L$ and $\kappa$, which change by $+0.1$, $-0.05$ and $+0.1$, respectively, which is about two standard deviations. In a residual plot (figure not shown) of the low-q data against the standard scaling, one does not see a clear degradation. For ELMy data, a strong degradation at low B/I has been reported by DIII-D [7]. Possibly, the sawteeth combined with ELMs lead to a stronger degradation than without ELMs. Also, the confinement degradation by sawteeth during the H-mode may not be well parameterized by $q_{cyt}$. This topic is currently under investigation.

It is interesting to have the power law scalings expressed in dimensionless quantities. Various plasma physical theories stipulate the values of some of these dimensionless exponents. As has been indicated in Refs [13, 15, 16], this can be conveniently expressed and tested by a matrix formulation. Not unlike the case of, but more clearly than, ITERH.DB1 [15], the results for ITERH.DB2 are that the short wavelength constraint is well satisfied by the ELM free data and the long wavelength constraint is outside the statistical error bounds while the MHD constraints clearly disagree with the experimental data. It is worth while noting that, if the density dependence is postulated to be zero, then the R dependence decreases to 1.4 and the long wavelength constraint is satisfied. However, the dependence of $\tau_{E,th}$ on the instantaneous density is more than ten times the standard error. The density in the standard dataset is only weakly correlated with the current or other plasma parameters. (The strongest correlation, $r = 0.37$, is with the major radius.) Hence, such a constraint is not supported by the present data.

Finally, it must be mentioned that some departures from a simple power law model can be perceived from the data. A simple power law is a linear approximation on a logarithmic scale. If coefficients of quadratic terms on the logarithmic scale are statistically significant, then the coefficients of the linear terms, i.e. the exponents of a simple power law expression, are not constant but vary with the other plasma parameters. In regression analysis, this is called an interaction. In the ELM free standard dataset, there is a significant interaction between the normalized power, $P_L/n_eV$, where $V$ denotes the plasma volume, and the effective mass number $M$, the quadratic term being $(+0.25 \pm 0.08) \log M \log (P_L/n_eV)$. This means that the
power dependence is more favourable for a larger value of M. By regression it can indeed be verified that the power exponent is about $-0.6$ for D into D, $-0.7$ for H into D, and, with a somewhat larger uncertainty, $-0.8$ for H into H. (The D into H shots from JET have not been used to estimate this interaction. When they are included, the interaction becomes somewhat stronger. The normalization constants described above do not affect the interaction to any significant extent.) For the non-normalized ELMy standard dataset, the interaction is about the same ($+0.30$). For the normalized ELMy data, it is $+0.15 \pm 0.06$. The result of this analysis can also be expressed as follows: the effective isotope exponent is not a constant, but depends on the value of $P_L/n_e V$ (Fig. 3). Note that $P_L/n_e V$ can be re-expressed as the ratio
of plasma temperature to confinement time. In the ELM free case, there seems to be also a negative interaction between magnetic field and some normalized power: at higher normalized power, the magnetic field dependence is weaker. If this interaction is included, the interaction between $P_L/n_eV$ and $M$ becomes stronger (+0.5 ± 0.1). The topic of interactions has interesting connections with offset linear scalings. Both can be seen as extensions and refinements of simple power law scalings, and both deserve further attention in the future.

5. POWER THRESHOLD DATABASE

5.1. Description of the database and its objectives

As a natural extension of the confinement database, a power threshold database has been assembled by the working group from data of the same six tokamaks. Its present version (ITERTH.DB1) consists of about 2200 observations (time slices) with 124 variables per observation. The ranges of some global parameters of the database are presented in Table III. With respect to the confinement database, additional variables expected to be relevant for H-mode power threshold and L–H transition physics were introduced, in particular: (1) various geometrical parameters, such as separatrix–vessel distances, that provide an improved description of the magnetic configuration and (2) plasma edge parameters, such as temperatures and densities at $R_{95}$. In addition, the plasma phase descriptor has been refined (identifying, e.g. 'dithering' H-mode), and L–H transition times as well as identification flags for sawtooth triggering are provided. Some of these variables will be supplied in the next release of the database.

| TABLE III. NUMBERS OF NON-OHMIC TIME SCLICES AND DATA RANGES OF THE ITERTH.DB1 DATABASE |
|---------------------------------|---------|---------|---------|---------|---------|
| No. of time slices              | ASDEX   | DIII-D  | JET     | JFT-2M  | PBX-M   |
| L/H                            | 186/55  | 167/51  | 168/105 | 337/361 | 186/153 |
| $I_p$ (MA)                     | 0.15–0.46 | 0.87–1.9 | 2.1–5.0 | 0.12–0.31 | 0.27–0.38 |
| $B_t$ (T)                      | 1.3–2.8  | 1.0–2.1  | 2.4–2.9 | 0.7–1.4  | 1.25–1.43 |
| $q_{95}$                       | 2.2–5.0  | 2.3–6.7  | 2.3–4.2 | 2.2–5.1  | 3.3–5.9  |
| $n_e$ ($10^{19}$ m$^{-3}$)     | 1.4–9.0  | 2.9–8.3  | 0.3–6.5 | 1.2–7.3  | 1.2–9.6  |
| $B_p n_e$ ($10^{19}$ T·m$^{-3}$) | 2.7–19.0 | 3.8–17.2 | 0.7–18.7 | 1.0–9.2  | 1.6–12.5 |
| $S$ (m$^2$)                    | 23–27   | 60–70    | 150–180 | 15–20    | 22–25    |
| $P_{\text{tot}}/S$ (MW·m$^{-2}$) | 0.02–0.10 | 0.03–0.10 | 0.01–0.14 | 0.01–0.08 | 0.04–0.12 |
In order to demarcate better the boundary between the H-mode region in plasma parameters space and the L-mode region, and possibly the region of overlap, shots have been included that stayed in the L-mode. For the shots that went into the H-mode, the information was gathered at time points just before and after the L-H transition. For many shots, also Ohmic reference points have been provided.

From this set-up the following objectives are being pursued: (1) predicting the plasma parameter region where an H-mode can be expected in future devices (in practice, the minimum power to achieve H-mode, $P_{\text{thr}}$, is expressed as an explicit function of the other plasma parameters); (2) contributing to the L-H transition physics; (3) investigating pedestal aspects.

Here, the first objective is to find a simple and physically relevant (device independent) expression for the power threshold as a function of global plasma parameters. This serves, among others, as an information complementary to the confinement scalings derived above.

5.2. Analysis and preliminary results

Previous results for each device [17–22] helped define the relevant variables and provided guidelines for the first analyses presented in this section.

In order to facilitate the search for the intermachine relevant parameters, a simplified analysis of the database was performed on a restricted part of the database according to the following criteria: single null configuration with the ion drift towards the active X-point or DN if SN is not available, deuterium target plasmas with $n_e \geq 2 \times 10^{19} \text{ m}^{-3}$, and $q_{95} \geq 3$, and a separatrix-wall distance large enough to avoid any strong influence on the power threshold. In fact, the power threshold increases with decreasing value of the distance between the separatrix and the wall. What distance is considered to be most critical depends on the device: for example, for DIII-D it is the distance between X-point and divertor tiles whereas, for other machines such as JFT-2M and ASDEX, it is the distance between the separatrix and the outside wall. When the range in the data is sufficient, all machines observe a more or less pronounced increase in the power threshold with increasing density and toroidal field.

Local measurements of the edge $T_e$ indicate that a minimum value is required for the H-mode transition [17, 22]. Therefore, in an attempt to unify the data from the different machines, these initial investigations have concentrated on the heat flux available to heating of plasma edge. This edge heat flux can be approximated by the total input power inside the last closed flux surface divided by this surface area, $P_{\text{tot}}/S$, where $P_{\text{tot}} = P_{\text{ohm}} + P_{\text{abs}}$, with $P_{\text{ohm}}$ the residual Ohmic power and $P_{\text{abs}}$ the injected power less shinethrough and fast ion losses. Because of the incompleteness of this first database, the radiation losses, $P_{\text{rad}}$, have not yet been subtracted. The results are given in Fig. 4. In this figure, of those discharges transitioning into the H-mode, only time points close to the L-H transition have been selected, in order to avoid a blurring of the picture. For the JET discharges with ICRF heating, the total
RF power has been included. Although there is significant scatter in the data, a clear trend of increasing heat flux with increasing density and magnetic field is observed. We see rather good agreement between the boundaries for ASDEX, DIII-D, JET and PBX-M discharges. The threshold values of \( P_{\text{tot}}/S \) for JFT-2M are qualitatively more or less in agreement with the other values, but the slope versus \( B_t n_e \) is distinctly weaker than in the other devices.

It is well known that the H-mode power threshold depends on more than just \( n_e \) and \( B_t \). For example, a residual influence may exist of plasma-wall distance (not covered by the selection criterion mentioned above), profile effects and wall conditions. This might lead to the scatter observed in Fig. 4. It is noted that most of the H-mode phases exhibit a density increasing with time. This means that for time points later than immediately after the transition, the H-mode is sustained by a power lower than the threshold given in Fig. 4. This may be attributed to the increased edge confinement during the H-mode, sustaining, despite the high edge density, high edge temperatures in comparison with the L-mode. The fact that the power flux \( P_{\text{tot}}/S \) yields a coherent picture is consistent with the observation that the L—H transition takes place at the plasma edge and with the H-mode triggering by sawteeth.

Further work will aim at: (1) taking into account adequate radiation losses in order to represent the power flux at the edge more satisfactorily; (2) describing the influence of the plasma—vessel distance; (3) introducing new data, in particular \( B_t \) variations from JET and density variations from ASDEX. The difficult problem of the divertor configuration (open/closed) also needs to be addressed in the future.
ACKNOWLEDGEMENTS

The main merit of this work rests with the ASDEX, DIII-D, JET, JFT-2M, PBX-M and PDX tokamak teams. The following people are acknowledged for contributing to improvements in the final stage of assembling the dataset and/or for discussions with the authors: G. Becker, M. Corneliussen, F. Engelmann, M. Fahy, O. Gehre, H. Maeda, K. Riedel, P. Roney, A. Staebler, K.-H. Steuer, A. Teubel, T. Tsunematsu and F. Wagner.

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[20] JET TEAM (presented by CAMPBELL, D.J.), ibid., 141.
R.R. PARKER: Data presented in Session A-1 of this Conference indicate a
dependence of $\tau_E$ on $\xi$, but this trend does not appear in your global confinement
scaling. More specifically, let us consider a gedankenexperiment in which machine X
is operated with fixed power (and geometry) and density, and the magnetic field is
increased. This would tend to raise $\xi$ in an inductively driven machine and,
therefore, according to results from several machines presented in Session A-1,
$\tau_E$ would increase. The ITER-P scaling, however, predicts no change. Can you
comment on this?

O. KARDAUN: The present scaling should be interpreted as having a ‘natural’
current profile, given the global parameters used in the scaling and under stationary
conditions, unless explicit measures are taken to change the shape of the current pro­
file. In the present data set, $\xi$ can be predicted, within the accuracy of its determina­
tion, from the current, the magnetic field and the three geometrical parameters. In
your gedankenexperiment, if you were to raise $B_t$ adiabatically and slowly at con­
stant $I_p$, our prediction would be applicable, and the improvement would be given
only by the weak $B_t^{0.2}$ dependence. This is in reasonable agreement with experi­
ments where such variations have been made from shot to shot. When current or
magnetic field are changed non-adiabatically, one moves away from the class of
‘natural’ current profiles considered here, and our results are not directly applicable.
An ancillary scaling of the current profile effect would certainly be interesting,
but in order for it to be effective we would need both additional experiments with
explicit current shaping and improved estimates of $\xi$, or other current moments, in
our database.

R. TOSCHI: Can you comment on the aspect ratio dependence of your scaling?

O. KARDAUN: The aspect ratio exponent depends on whether $\tau_E$ or $nT\tau_E$ is
considered and on what other variables are used in the scaling. The geometrical part
of the ELM free scaling is $\tau_E \propto R^{1.9} A^{-0.05} k^{0.65}$, where $A = R/a$ denotes the aspect
ratio. For $nT\tau_E = P\tau_E/V$, it is found that $nT\tau_E \propto R^{0.8} A^{1.9} k^{-0.3}$ in the ELM free case,
and $nT\tau_E \propto R^{1.2} A^{1.6} k^{0.6}$ in the ELMy case. The standard error of the estimated
aspect ratio exponent, based on an error propagation of 15% random error in $\tau_E$, is
small, approximately 0.1. The sensitivity to systematic shifts of all the data derived
from a single tokamak is highest for PBX-M. An increase of 10% in all values of
$\tau_E$ for PBX-M in the ELM free standard data set leads to an increase of 0.34 in the
aspect ratio exponent of the $nT\tau_E$ scaling. A 10% increase of all the DIII-D or
JFT-2M confinement times in this data set diminishes the aspect ratio dependence
by 0.14. The sensitivity is lower in the case of the other tokamaks.

R.J. GOLDSTON: Could you explain the basis for deducting $\sim 30\%$ from the
ASDEX data? If we build ITER with a closed divertor, can be increase the confine­
ment projection by $\sim 30\%$? What happens to the aspect ratio scaling if you take the
ASDEX data at face value?
O. KARDAUN: The normalization of ASDEX from a closed to an open divertor machine was based on an empirical comparison of ASDEX confinement data during periods of open and closed divertor configuration reported by F. Ryter at the EPS Meeting in Innsbruck (our Ref. [18]) and by F. Wagner at the IAEA Conference in Washington (our Ref. [10]). The improved confinement in a closed, as compared with an open, divertor is usually attributed to better retention of neutral particles in the divertor chamber. As this effect may well become smaller as the machine size increases, one cannot extrapolate this improvement to ITER. The present scaling applies only to open divertor tokamaks. To make a prediction for a closed ITER-like device, one would need to know how the difference in confinement between open and closed divertors scales with machine size. At present, we do not have sufficient empirical data to derive such an ancillary scaling. The geometrical part of the ELM free scaling changes by $R^{+0.05}k^{0.25}$ if the unnormalized ASDEX data are used, and the geometrical part of the ELMy scaling changes by $R^{+0.5}(a/R)^{+0.2}k^{-0.4}$.

B. COPPI: Have you carried out an analysis of nTr scaling? I imagine that the product $L_pB_p$ ($L_p =$ total plasma current, $B_p =$ average poloidal field) would feature prominently and that this scaling would be better than $n_T$ at illustrating the differences between compact high field experiments and large scale, low or moderate field experiments.

Another point to consider is that in order to assess a plasma column's evolution toward D-T burn conditions, it is necessary to identify consistent and acceptable forms of the relevant transport coefficients and to employ appropriate transport codes. As the experiments presented by the General Atomics group in paper F-l-2 confirmed, confinement is strongly influenced by profiles (e.g. of the particle or the current density) and their evolution.

O. KARDAUN: Using direct substitution in ITERH92-P, we find that $nT\tau_E = P\tau_E/V$ scales as $I_{9}^{1.9}A^{0.99}k^{0.3}B_{1}^{0.4}n^{0.6}p^{0.3}M^{0.9}$, where $A = R/a$ denotes the aspect ratio. It is, of course, possible to factor out a power of the product $L_pB_p$. The form will still depend on the choice of the other parameters. Following your line of thinking, we would probably want to use the volume $V$ and the power density $p = P/V$ as dimensional parameters, obtaining the scaling $(L_pB_p)^{1.15}V^{0.25}A^{1.05}k^{-0.075}q^{0.4}n^{0.6}p^{-0.3}M^{0.9}$, where we have not only your expected $L_pB_p$ dependence, but also favourable volume and aspect ratio dependences and an unfavourable power dependence.

We certainly do not deny the importance of empirical local transport analysis, but we would like to point out that this did not lie within our scope. Transport codes are critically dependent on assumptions about transport coefficients. Although one important aspect is to integrate those assumptions with theoretical understanding, it is also essential to test them against the available experimental data. With improved estimates of the global thermal confinement time for all six tokamaks, the database provides at least one such experimental benchmark.

As to the profile dependence illustrated by the experiments you mention, one should realize that the database includes only steady state current discharges without

\[ \text{IAEA-CN-56/F-1-3} \]
pellet refuelling and without significant current drive or off-axis heating, so that the class of profiles is rather narrow and primarily determined by global parameters such as q. For this narrow class of ‘natural’ profiles, profile effects are implicit in the global parameter dependences. In this sense the database is degenerate. To eliminate this degeneracy effectively, current ramp or other ‘profile changing’ experiments will need to be added to the database.
PROGRESS IN THE TOKAMAK BASIC DEVICE DEVELOPMENT FOR THE CONSTRUCTION OF FUSION EXPERIMENTAL REACTORS

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Abstract

PROGRESS IN THE TOKAMAK BASIC DEVICE DEVELOPMENT FOR THE CONSTRUCTION OF FUSION EXPERIMENTAL REACTORS.

Significant advance has been achieved in the technology of constructing tokamak basic devices of the Fusion Experimental Reactor (ITER, FER) at the Japan Atomic Energy Research Institute (JAERI). In the development of superconducting magnet technology, an 1 m inner diameter pulsed coil (DPC-EX) using Nb₃Sn was developed for poloidal coils and successfully charged up to 7 T in 0.5 s for the first time. For toroidal coils, the world’s first Nb₃Al 40 kA, 12 T conductor was developed and tested successfully. With regard to plasma facing components, JAERI has succeeded in developing a new plasma facing component that first achieved a high performance operation enduring 20 MW/m², 30 s for 1000 thermal cycles. For the technology to assure the remote handling of in-vessel components such as divertors, a 1/5 scaled model of the vehicle system was developed and successfully operated. Time is ripe now for starting technology R&D with scalable models for ITER and FER.
1. SUPERCONDUCTING MAGNETS

For tokamak operation at a high plasma current of 22 MA, a rather high magnetic flux swing of 325 V·s should be supplied by its poloidal coils. This requires the use of an Nb$_3$Sn pulsed coil that can generate a high field of 12–13 T. However, there were many problems left unsolved. Examples of these issues were:

Is it really possible
1. to fabricate a pulsed coil with an Nb$_3$Sn conductor that has a potentially high hysteresis loss?
2. to make a joint of brittle Nb$_3$Sn conductors after activation of heat treatment?, etc.

In the Demo Poloidal Coil (DPC) programme [1], JAERI developed a Nb$_3$Sn pulsed coil (DPC-EX) with an inner diameter of 1 m [2]. A new 18 kA conductor employed a Ta layer in the superconducting strands and a 5 µm Cr plating on the surface of the strands, obtaining a low pulsed loss. This Cr plating reduced the pulsed loss by a factor of 1/1000 and achieved rapid charging up to 7 T in 0.5 s by poloidal power supply of the JT-60 as the world’s first experience with large Nb$_3$Sn coils. By this achievement, most of the fundamental problems as described previously were solved. On the basis of this technology, a further advanced Nb$_3$Sn conductor is now under development for the Central Solenoid Scalable Model Coil (CS-SMC) to be built under the ITER programme. The strand with a high critical current density of 800 A/mm$^2$ at 12 T and with 6 µm fine filaments has already been fabricated [3].

For toroidal coils, JAERI has developed the world’s first Nb$_3$Al conductor with an operating current of 40 kA at 12 T, as shown in Fig. 1. Nb$_3$Al had been known as an excellent material for high field and large coils; it was, however, impossible to fabricate a large conductor because of the necessity of a high activation temperature about 1500°C. JAERI has developed a new technology in which thin layers of Nb and Al with thicknesses less than 100 nm can be reacted to generate Nb$_3$Al at a temperature of 800°C that is similar to that of Nb$_3$Sn. The degradation of its critical current was measured by a collaboration of JAERI and Kernforschungszentrum Karlsruhe in Germany to be only 5% at a typical 0.4% strain under 12 T [4]. This degradation is one-sixth of that of Nb$_3$Sn conductors (30%) in the same conditions. The filament diameter of the Nb$_3$Al conductor is still large, i.e. 20 µm, and is suitable for DC operated coils. This new technology will make a significant contribution to increasing the reliability of the ITER and FER toroidal coils.

2. PLASMA FACING COMPONENTS

There are three major issues on the development of the divertor plate: (1) high heat flux removal technology; (2) reliable joining technology between armour and structural materials; (3) reliable divertor support structure. Most of the R&D effort
FIG. 1. Cross-section of the world’s first Nb₃Al 12 T, 40 kA conductor: (a) slight reduction; (b), (c) magnifications.
Concerning heat removal technology, critical performance tests of various heat removal concepts have been performed to validate the burnout safety margin. The swirl tube, which is a tube with a twisted tape insert, resulted in the highest critical heat flux at the tube surface of $38 \text{ MW/m}^2$ with a cooling water speed of 10 m/s and a pressure of 1 MPa [5], which corresponds to a burnout safety margin of about two for the ITER heat removal condition of $15 \text{ MW/m}^2$ [6].

With regard to the joining technology, various carbon fibre reinforced carbon composites (CFC), which are some of the primary candidates for ITER armour
material, have been brazed onto heat sinks made of oxygen free high conductivity copper (OFHC Cu) without failure [7]. The joined structure of high thermal conductivity CFCs and OFHC Cu with the swirl tube successfully endured 20 MW/m², 30 s for 1000 thermal cycles, as shown in Fig. 2. This result is the world's record and is achieved by optimization of an armour tile configuration and brazing.

Establishing the technological database on the divertor support structure is essential to perform the engineering design of the ITER divertor plate. Screening tests on the various structures will start soon, using about 1 m long divertor mock-ups.

**FIG. 3.** Full scale telescopic manipulator with rotating mechanism.
3. REMOTE HANDLING OF IN-VEssel COMPONENTs

Remote maintenance of tokamak basic devices activated by D–T operation is an essential technology. In particular, the plasma facing components such as divertor plates and armour tiles are the most critical parts since they may require scheduled replacement.

In ITER [8], the remote handling system is required to achieve final positioning within a few mm and posture control within 10–20 mm for handling the divertor module (3.5 m, 1.5 t). Therefore, high mechanical stiffness is essential to avoid excessive deflection and vibration during divertor handling. For this purpose, JAERI has developed a rail mounted vehicle concept in which the toroidal rail extended into the vacuum vessel and supported by ports each 90° provides a rigid guide structure for the vehicle with manipulator moving on it for divertor handling.

As a first step, JAERI has fabricated and tested a one-fifth scaled model of the vehicle system [9]. The results indicated that the vehicle system can satisfy all ITER requirements, and high reliability compared with the ordinary articulated boom can be expected. In addition, it has been demonstrated that various manipulators for different purposes can be operated on the common rail, and a newly developed rotating mechanism can provide 360° accessibility to viewing, inspection and maintenance of all in-vessel components. On the basis of these achievements, a full scale telescopic manipulator with the rotating mechanism has been fabricated as shown in Fig. 3, and the kinematics including dynamic control is being tested in combination with the divertor mock-up structure for the demonstration of the divertor maintenance.

4. CONCLUSIONS

Fundamental problems were solved concerning the high flux swing Nb$_3$Sn pulsed coils, the critical current degradation of the toroidal coils, the high heat flux plasma facing components, and the reliable and accurate remote maintenance by a vehicle system. Time is ripe now to start technology R&D with scalable models under the international collaboration of the ITER Engineering Design Activities started in July 1992.

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grateful to all the Japanese industries that have made significant contributions to fabricating test samples and components for the technology development of ITER/FER basic devices at JAERI.

REFERENCES


MISSION AND DESIGN OF THE TOKAMAK PHYSICS EXPERIMENT/STEADY STATE ADVANCED TOKAMAK (TPX/SSAT)*

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Abstract

MISSION AND DESIGN OF THE TOKAMAK PHYSICS EXPERIMENT/STEADY STATE ADVANCED TOKAMAK (TPX/SSAT).

The Tokamak Physics Experiment/Steady State Advanced Tokamak is being designed to extend advanced tokamak operating modes with high $\beta_p$, $T_e$, and $I_b/I_p$ to the steady state regime. Strong plasma shaping and high aspect ratio are provided to assist in accessing advanced modes. In addition, a flexible heating and current drive system gives detailed control over $j(r)$, while a pumped “slot” divertor geometry provides control over particle recycling. The high duty factor of TPX/SSAT allows optimization of high heat flux steady state divertor operation, as well as stringent tests of disruption control techniques. Demonstration of advanced tokamak operating modes in TPX/SSAT will contribute to the data base supporting a more attractive steady state fusion reactor. Experience with superconducting magnets, steady state pumped divertors, and in-vessel remote maintenance on TPX/SSAT will aid in the operations phase of ITER.

1. INTRODUCTION

An integrated U.S. national team is designing a superconducting Tokamak Physics eXperiment for the complementary missions of Steady State and Advanced Tokamak operation. TPX is equipped with a strongly shaped plasma, full non-inductive current drive with active control of the current profile, active particle recycling control, and high aspect ratio. These tools provide the device with the flexibility to explore advanced operating regimes with high $\beta_N$, $\tau_E$, and $I_{BS}/I_p$ that have been seen transiently in present experiments or predicted theoretically. Steady-state operation of TPX permits these studies to be extended to time scales far exceeding the global current-relaxation time and the plasma-wall equilibration time. The high duty factor of TPX will permit stringent tests of divertor erosion in operational conditions consistent with efficient current drive, and of disruption control techniques in advanced operating modes. Favorable results from TPX will point to a more attractive steady-state tokamak fusion reactor, with low recirculating power for current drive, smaller unit size, and/or simpler and less costly magnet systems.

The TPX program will contribute to the development of key technologies required for fusion power reactors. Both the toroidal and poloidal field magnet systems are composed of superconducting coils, which will need to withstand both normal start-up and disruptions without quenching. TPX can produce challenging divertor conditions, thus providing a facility to qualify high-heat-flux components in a reactor-like plasma environment. The device design will incorporate in-vessel remote maintenance because of its high duty factor operation in deuterium. The moderate internal activation levels during the first few years of operation provides a good opportunity to gain experience with remote maintenance. Experience with superconducting magnets, steady-state divertors, and in-vessel remote maintenance on TPX will contribute to the success of the operations phase of ITER.

2. MACHINE DESCRIPTION

An elevation view of TPX is shown in Fig. 1, while reference design and operating parameters are shown in Table I. The ranges indicated by arrows in Table I indicate device flexibility ($\kappa_x$), maximum parameter ranges under design consideration ($B_t$, $I_p$),
Table I. TPX Reference Design Parameters

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Major radius, $R_0$ (m)</td>
<td>2.25</td>
</tr>
<tr>
<td>Minor radius, $a$ (m)</td>
<td>0.50</td>
</tr>
<tr>
<td>Aspect Ratio, $R_0/a$</td>
<td>4.5</td>
</tr>
<tr>
<td>Elongation, $\kappa_X$</td>
<td>1.6 $\rightarrow$ 2.0</td>
</tr>
<tr>
<td>Triangularity, $\delta_X$</td>
<td>0.8</td>
</tr>
<tr>
<td>Toroidal field, $B_T$ (T)</td>
<td>3.35 $\rightarrow$ 4.0</td>
</tr>
<tr>
<td>Plasma Current, $I_p$ (MA)</td>
<td>1.8 $\rightarrow$ 2.0</td>
</tr>
<tr>
<td>Pulse Length (s)</td>
<td>1000 $\rightarrow$ $\infty$</td>
</tr>
<tr>
<td>Neutron Budget (#/year)</td>
<td>$6 \times 10^{21}$</td>
</tr>
<tr>
<td>Heating and Current Drive:</td>
<td></td>
</tr>
<tr>
<td>Neutral Beam (MW)</td>
<td>$8 \rightarrow$ 32</td>
</tr>
<tr>
<td>ICRF (MW)</td>
<td>$8 \rightarrow$ 18</td>
</tr>
<tr>
<td>Lower Hybrid (MW)</td>
<td>$1.5 \rightarrow$ 3.0</td>
</tr>
</tbody>
</table>

and device upgrade capabilities (pulse length, heating and current drive systems).

The superconducting magnet system consists of 16 toroidal field (TF) coils and 14 (up-down symmetric) poloidal field (PF) coils external to the TF. Nb$_3$Sn conductor is used in all but the outer ring coils in the PF system, in which NbTi is used. The TF outer leg position is determined by tangential neutral beam access, giving 0.4% ripple at the outer edge of the plasma. Large symmetric-trapping regions above and below the horizontal midplane prevent ripple-trapped fast particles from exiting the plasma, even at high q. Beam ion energy losses due to ripple are calculated to be <6%.

The PF system provides sufficient volt-seconds to inductively initiate the discharge and ramp up the plasma current to its maximum value while leaving 2 volt-seconds capable of sustaining the current flattop for ~20 seconds without non-inductive current drive. It is designed to support the wide range of plasma equilibrium conditions required to study advanced regimes: $\kappa_X$=1.6-2.0, $\beta_N$=$\beta/(I_p/aB)$ up to 6%, and $\ell$=0.5-2.0 (not all simultaneously). The PF coil system is also capable of providing single-null equilibria at near full parameters. All these equilibria are compatible with the "slot" divertor design.

Vertical control is an important issue in strongly shaped plasmas. Detailed calculations with the TEQ [1] and TSC [2] codes
show that vertical stability can be achieved with passive structure located inboard and outboard of the plasma, connected vertically to form saddle coils. The design of this structure, including its interaction with the remote maintenance system, is currently being optimized.

A budget of $6 \times 10^{21}$ neutrons per year has been established for TPX based on $2 \times 10^5$ seconds of operation at $3 \times 10^{16}$ neutrons/sec in
deuterium, which corresponds to a much larger total operating time in hydrogen and deuterium over a range of performance levels. A titanium alloy Ti-6Al-4V vacuum vessel, and lead oxide, B₄C, and water shielding between the vessel and magnets are used to limit the radiation dose rate external to the vessel to ≤10 mR/hr. This eliminates the need for major ex-vessel remote maintenance manipulators, though special procedures and tooling will be required. A full in-vessel remote maintenance system will be provided. Annual neutron yields will be limited in the early stages of operation to permit a mixture of hands-on and remote operations until remote maintenance procedures are fully established.

3. DIVERTORS AND DISRUPTIONS

The double-null poloidal divertor configuration has been selected in order to support a high-triangularity plasma shape. Theoretical analysis [3] and experimental results from both DIII-D [4] and PBX-M [5] indicate that the highest βₚ's and confinement enhancements are observed in highly triangular plasmas. The double null also reduces the maximum divertor heat loads compared to a single-null configuration. Based on data from DIII-D and JET [6], it is estimated that the heat flux to the outer strike point will be reduced from 40±12% of the total heating power in the single-null case to 26±9% at the hotter of the top and bottom divertor plates in the double-null case, while the heat flux to the inner strike point will be even more strongly reduced from 20±12% to 8±8%.

TPX is being designed for flexibility in the divertor target and baffle configurations in order to allow for modifications that may be needed to optimize divertor performance. Modeling of the SSAT divertor using the B2 code [7] assuming $\chi_e = 2 m^2/s$, $\chi_i = D = \frac{\chi_e}{3}$ and a 22° divertor target angle shows acceptable divertor operation with the 17.5 MW initial heating and current drive complement. The peak electron temperature at the worse outer divertor plate is calculated to be 42 eV, and the peak axisymmetric heat flux is 6.1 MW/m². With 32 MW of plasma heating more challenging conditions are found: $T_{e,\text{div}} = 54$ eV, $q_{\text{div}} = 10.4$ MW/m². The heat flux predictions are consistent with extrapolations from measured DIII-D divertor conditions. To allow for toroidal peaking factors, it will most likely be necessary to employ gaseous or radiative divertor operation with over 30 MW of heating power, although studies of these divertor modes will be performed with the initial installed heating.
Preliminary modeling suggests that substantial improvements in divertor conditions are possible with the additional neutral recycling expected if this divertor, which incorporates baffles to insure gas isolation and high recycling, operates in a successful gas-target mode. Estimates of impurity radiation indicate that 50–80% of the divertor power can also be radiated with an argon concentration of about 4%, but transport of argon out of the divertor region has yet to be modeled.

Disruptions may determine the availability factor of tokamak reactors. In TPX we expect to reduce the disruption frequency in part by extending the stable operating space through steady-state profile control. In addition active disruption avoidance techniques can be tested in advanced regimes (and at high duty factor) on TPX. Kink modes might be stabilized with feedback coils or with ponderomotive forces generated by ion Bernstein waves, while low-order tearing modes might be suppressed through modification of the current profile in the neighborhood of the mode-rational surface, using the lower hybrid system or an electron-cyclotron system dedicated to disruption control.

4. CURRENT PROFILE CONTROL

Current-profile control is a key element of the TPX program. From the point of view of energetics, $j(r)$ will be much easier to control in a reactor than $p(r)$, even in regimes with high $I_{bs}/I_p$. It is not at all clear, however, that methods will be available to provide strong $n(r)$ control. Recent experimental results suggest that the current profile has a substantial impact on tokamak confinement and $\beta$ limits. Current and elongation ramping studies on TFTR [8], DIII-D [9], and JET [10] show very favorable properties of high-$q_w$ high-$l_i$ plasmas. Confinement improvements range up to twice the expected (L or H-mode) performance of the base plasma, and $\beta_N$ values as high as 6 have been achieved [4]. Preliminary results on ASDEX [11] and Tore-Supra [12] indicate that similar improvements can be achieved with LHCD modification of $l_i$.

Another attractive current profile modification is the creation of reversed shear in the core of the plasma. The PEP mode in JET [13] shows strongly reduced transport within a region and time-period of $q'<0$, and analysis of very-high-beta plasmas in DIII-D [4] indicates that the $q'<0$ region is in the second-stable regime for ballooning modes. The Rebut-Lallia-Watkins transport model (based on
magnetic island formation) indicates that transport is greatly reduced when $q' < 0$ [14]. The drive for trapped-particle modes is also substantially reduced by the reverse toroidal precession arising from $q' < 0$ [15]. Long-pulse plasmas with active current-profile control are required to demonstrate these improvements in steady-state, and to provide enough detailed control to allow development of more complete physical understanding. These results might be extended to higher enhancements via larger regions of reversed shear, and/or to lower $q'_s$ with more subtle and stable methods of $j(r)$ control.

In TPX active control of the current profile will be accomplished with a combination of neutral beams for bulk current drive, fast waves for centrally peaked current drive, and lower hybrid for detailed current-profile control. The LH system design includes real-time $n_\parallel$ control, which will allow active current-profile control provided that the energetic tail electrons produced by the LH waves are well localized.

5. CONFINEMENT REQUIREMENTS

The minimum confinement performance of TPX is set by the combined requirement that the LH-tail electrons be well localized, and that the device operate in a reactor-like, low-collisionality regime, in order to have a realistic bootstrap current profile. Substituting $T \approx \beta N I_p B / n a$, the requirement for low collisionality can be written

$$n^3 < C_1 \beta N^2 (v*)_{req} I_p^2 B^2 / (R^2 5 q a^{0.5}) \equiv n_v^3$$

where the $C$'s here and below indicate constants. Experimental results on fast electron transport span a considerable range of uncertainty [16], but it is reasonable to estimate $D_{fast} = \tau_e$. To provide radial localization of lower hybrid current drive to $\sim a/4$ results in the requirement $\tau_E / \tau_{se} \sim 8$ [17]. Since $\tau_{se} \approx 1/n$, this amounts to a requirement on $n \tau_E$. Estimating $n \tau_E T \approx I_p^2 (R/a)^2 A_i$, we arrive at

$$n > C_2 \beta N (n \tau_E)_{req} a B / (I_p R^2 A_i) \equiv n_{LH}$$

Clearly $n_v / n_{LH}$ is a key figure of merit for an advanced tokamak experiment. For fixed bootstrap fraction and fixed $\beta_N$, we take $q \sim$
(a/R)^{1/2}. Evaluating $I_p$ in terms of the current-carrying capability of
the device at $q=3$, "$I_3$," we have $I_p = (3/q) I_3$, and

$$n_v / n_{LH} \approx I_3^{5/3}(R/a)^{13/6}A_i^{13/6}/P^{1/2}$$

At $B = 3.35T$ this advanced tokamak figure of merit is about three
times greater for TPX than it is for DIII-D, while it is about three
times smaller than for JET.

The scaling of the performance requirement for advanced
tokamak physics studies is quite similar to the scaling of $n\tau_E T$. The
scenarios described below show clearly, however, that a much more
modest $n\tau_E T$ is required for these studies than for ignition. It is
worth noting that the assumption $\tau_E \approx I_p R^{3/2}k^{1/2}A_i^{1/2}/P^{1/2}$, implicit
in the formula used for $n\tau_E T$, gives the result that a hydrogen-only
advanced tokamak experiment requires 52% larger size at fixed $B$, or
68% higher $B$ at fixed size, than a device capable of deuterium
operation. The power requirement to reach a given $\beta_N$ increases by a
factor of 3.16 and 5.65 in these two cases. The cost of the internal
remote maintenance system on TPX associated with deuterium
operation is far less than the costs that would be incurred by raising
the field or size of the device, and greatly increasing the auxiliary
heating power.

6. OPERATIONAL SCENARIOS

TPX is being designed to provide sufficient operational flexibility to
investigate a range of advanced tokamak scenarios, including full-
current steady-state operation close to the Troyon limit ($\beta \sim 3.5I_p/aB$),
operation with high bootstrap current fraction (\geq 66%) in the first-
stability regime, operation in the predicted second-stable regime, and
operation in a variety of advanced confinement modes (high $l_i$, $q' < 0$,
etc.).

As shown in Table II, the planned heating complement for initial
operation ($P_{NBI}=8$ MW, $P_{ICRF}=8$ MW, $P_{LH}=1.5$ MW) provides
enough power to reach the Troyon $\beta$ limit with VH-mode
confinement ($\sim 3x$ITER89-P). On the other hand, upgrades to much
higher power levels ($P_{NBI}=32$ MW, $P_{ICRF}=18$ MW, $P_{LH}=3.0$ MW) can
be accommodated. Up to the total of four 120 keV TFTR neutral
beam systems, each delivering 8 MW and upgraded for pulses >1000
Table II. Plasma Parameters for TPX Operational Scenarios

<table>
<thead>
<tr>
<th></th>
<th></th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>Toroidal field, $B_T$ (T)</td>
<td>4.0</td>
<td>4.0</td>
<td>4.0</td>
</tr>
<tr>
<td>Plasma current, $I_p$ (MA)</td>
<td>1.5</td>
<td>1.2</td>
<td>2.0</td>
</tr>
<tr>
<td>$q_{95}$</td>
<td>4.5</td>
<td>5.6</td>
<td>3.4</td>
</tr>
<tr>
<td>H-factor, $\tau_E/\tau_{ITER89-P}$</td>
<td>3</td>
<td>2</td>
<td>4</td>
</tr>
<tr>
<td>$\beta_N$ (%)</td>
<td>3.28</td>
<td>3.37</td>
<td>5.5</td>
</tr>
<tr>
<td>Central density, $n_{eo}$ (m$^{-3}$)</td>
<td>$1.1\times10^{20}$</td>
<td>$1.0\times10^{20}$</td>
<td>$1.3\times10^{20}$</td>
</tr>
<tr>
<td>Electron temperature, $T_{eo}$ (keV)</td>
<td>11.5</td>
<td>8.9</td>
<td>20.6</td>
</tr>
<tr>
<td>Ion temperature, $T_{io}$ (keV)</td>
<td>11.7</td>
<td>9.6</td>
<td>22.9</td>
</tr>
<tr>
<td>Energy confinement time, $\tau_E$ (s)</td>
<td>0.25</td>
<td>0.11</td>
<td>0.34</td>
</tr>
<tr>
<td>$\tau_E/\tau_{se}$</td>
<td>67</td>
<td>26</td>
<td>143</td>
</tr>
<tr>
<td>Bootstrap fraction, $f_{BS}$</td>
<td>0.65</td>
<td>0.67</td>
<td>0.73</td>
</tr>
<tr>
<td>BS collisionality correction factor</td>
<td>0.92</td>
<td>0.88</td>
<td>0.99</td>
</tr>
<tr>
<td>DD neutron rate, $S_{DD}$ (s$^{-1}$)</td>
<td>$2.2\times10^{16}$</td>
<td>$1.9\times10^{16}$</td>
<td>$1.2\times10^{17}$</td>
</tr>
<tr>
<td>Neutral beam, $P_{NBI}$ (MW)</td>
<td>8</td>
<td>16</td>
<td>16</td>
</tr>
<tr>
<td>ICRF, $P_{ICRF}$ (MW)</td>
<td>8</td>
<td>12</td>
<td>12</td>
</tr>
<tr>
<td>Lower Hybrid, $P_{LH}$ (MW)</td>
<td>1.5</td>
<td>1.5</td>
<td>1.5</td>
</tr>
</tbody>
</table>

sec, can be aimed tangentially ($R_{tan}=2.0$ m) to provide both heating and current drive. The ICRF launcher design features a phased array of 12 near-contiguous straps to deliver 8 MW through two ports for initial operation (upgradable to 18 straps delivering 18 MW through three ports) to support efficient fast-wave current-drive in the frequency range of 40-80 MHz. The 3.7 GHz lower-hybrid launcher array fits into a single midplane port, and is upgradable from 1.5 to 3.0 MW. Choices among these upgrade options will depend on results from ongoing experiments and from TPX itself. Even with installation of the full heating and current drive complement, adequate access is available for midplane diagnostics. The initial pulse length of TPX will be limited to 1000 sec due to auxiliary systems, which can be upgraded to fully steady-state operation if results indicate that this is required.

Table II presents fixed-profile calculations of high-bootstrap-fraction operating modes, similar to the operating mode of ARIES-I [18], which may be accessible in TPX. The VH-mode case shows that impressive plasma parameters can be achieved with the day-one
heating and current-drive complement, if the results already obtained in DIII-D can be extended to steady state. Note that the fast-electron localization, parameterized by $\tau_E/\tau_{se}$, is quite satisfactory, while the collisional correction to the bootstrap current represents only an 8% overall decrease from the fully collisionless case.

If only H-mode confinement ($\sim 2\times$ITER89-P) can be sustained in steady state, then power upgrades to $\sim 30$MW are required to reach the Troyon limit at full field. Note that in this case the fast electron radial localization is poorer, and the collisional correction to the bootstrap current is a stronger effect. At the higher $q_v$'s required for second-stability access, $\tau_E/\tau_{se}$ reaches marginal values $\sim 8$ in collisionless plasmas. If, on the other hand, advanced confinement and $\beta$-limits can be achieved at lower $q$, the very impressive parameters shown in the third column become accessible.

A wide range of other operating points have been investigated. For example high $q_v$ and $q_0$ operating points exist which are consistent with access to second stability. $q_{95}=3$ steady-state operating points close to the Troyon limit are accessible at $B \sim 3.5$T. For machine start-up steady-state operation at $I_p \sim 1$MA can be achieved using the day-one heating and current-drive complement, even in hydrogen L-mode.

7. UNIQUE FEATURES OF TPX

Many machines in the world fusion program have significant capability to address elements of the steady-state advanced tokamak mission. Indeed results from existing facilities, combined with the physics requirements for an attractive steady-state fusion reactor, have motivated the TPX design. However TPX will offer a unique combination of capabilities required for advanced tokamak physics studies. Unlike the other superconducting tokamaks, Tore Supra, T-15, and Triam-1M, TPX will have a poloidal divertor and capability for strong plasma shaping ($\kappa_x \leq 2.0, \delta_x = 0.8$) which are necessary to access regimes of high confinement and $\beta$. The device will have a dramatically longer pulse than resistive, diverted tokamaks of similar or greater confinement capability (e.g., DIII-D, Alcator C-Mod, ASDEX-U, JET, and JT-60U). This will allow TPX to demonstrate techniques for current profile and particle recycling control for times long compared to the global current relaxation time and the plasma-
wall equilibration time. In addition TPX will have a sufficient annual neutron budget to permit reasonably high-duty-factor operation in deuterium. This will allow stringent tests of divertor erosion and of disruption control techniques in advanced tokamak operating regimes. The TPX design includes space for an actively cooled and pumped divertor. The x-point to strike-point distance, measured in the poloidal plane, is 60 cm on the outboard and 35 cm on the inboard side, compared to a minor radius of 50 cm. The initial divertor design is based on a closed "slot" geometry at the outer leg. However the available space will allow other concepts to be tested. Finally, the high aspect ratio of TPX (A=4.5) permits high $I_{BS}/I_p$ operation at moderate $\beta_N$ and $q$, in a regime found attractive in fusion power reactor studies [17, 19].

8. CONCLUSIONS

TPX is designed to make key contributions to the development of a more attractive steady-state fusion power reactor. Improvements in $\beta_N$, $\tau_E$, and $I_{BS}/I_p$ should be accessible in steady state on TPX through strong plasma shaping, active current profile control, particle recycling control, and high aspect ratio. Successful results from TPX will point to a reactor with reduced recirculating power for current drive, smaller unit size, and/or simpler and cheaper magnet systems. Experience with superconducting magnets, steady-state pumped divertors, and in-vessel remote maintenance on TPX will aid in the operations phase of ITER.

REFERENCES

DISCUSSION

R.R. PARKER: I found your divertor concept interesting in that you propose to inject gas through a baffle which appears to be contiguous to the highest temperature region of the SOL. Have you estimated the lifetime of such a component or the contribution to $Z_{\text{eff}}$ from the impurities that will be sputtered from it?

W.M. NEVINS: You raise an important point which we will address when we finalize the shape and location of the divertor baffle and gas puffing. Sputtering by charge exchange neutrals is a generic problem in gas target divertors. In TPX/SSAT I expect that impurity generation, rather than erosion per se, will be the dominant concern, but the relatively high duty factor of TPX/SSAT will enable us to measure erosion rates in a range of different operating modes.

J.G. CORDEY: What are the predicted $n_T E_i$ and $Q$ in your device?

W.M. NEVINS: For the cases listed in Table II, $n_0 \tau_E T_{i0}$ ranges from $9 \times 10^{19}$ keV-s/m$^3$ to $9.3 \times 10^{20}$ keV-s/m$^3$. The thermonuclear DT equivalent $Q$ (excluding beam-target and beam-beam reactions) is estimated to range from 0.05 to 0.79.

M.L. WATKINS: Are the operating scenarios presented based on full transport calculations which (a) determine density and temperature profiles that are consistent with heating and fuelling rates, and (b) can be applied to a large aspect ratio reactor?
I ask this question because we have used a model that gives a similar dependence on R/a of $\tau_{\text{ITER}}^{\text{P}}$ and simulates the details of R/a scaling experiments on TFTR. We find that a high aspect ratio reactor would have a high bootstrap current fraction, would produce substantial fusion power and would need, for steady state operation, a current drive efficiency somewhat higher than that achieved at present. However, these efficiencies would have to be guaranteed at a high operating density ($> 2 \times 10^{20} \text{ m}^{-3}$).

W.M. NEVINS: We have done simulations and the most interesting feature we found was the complex and slow response of the plasma to non-inductive current profile control. These studies mainly focused on determining volt-second requirements for the TPX/SSAT poloidal field system. The form of the density and temperature profiles depends on both the sources and the transport model (which are uncertain). We felt most comfortable using prescribed profiles of n and T in developing the steady state scenarios in our paper. The optimal steady state operating density depends on aspect ratio, magnetic field strength, divertor capability, and the assumed transport model. We believe that the Rebut–Lallia–Watkins model tends to favour high density.
TECHNOLOGY AND REACTOR CONCEPTS

(Sessions G-1 and G-2 and Poster Session G-3)

Chairmen

R.W. CONN
United States of America

A. GIBSON
CEC
DIRECTIONS FOR
ATTRACTION TOKAMAK REACTORS:
THE ARIES-II AND ARIES-IV
SECOND-STABILITY DESIGNS

F. NAJMABADI, R.W. CONN
University of California,
Los Angeles, California,
United States of America

and the ARIES TEAM

Abstract

DIRECTIONS FOR ATTRACTION TOKAMAK REACTORS: THE ARIES-II AND ARIES-IV
SECOND-STABILITY DESIGNS.

ARIES is a research program to develop several visions of tokamak reactors with enhanced economic, safety, and environmental features. The ARIES study has developed four visions for tokamaks, each having a different degree of extrapolation in physics and technology. All four designs are steady-state, 1000-MWe (net) power reactors. The ARIES-II and ARIES-IV designs assume potential advances in plasma physics, such as second-stability operation, that are predicted by theory but are not yet well established experimentally. The two designs have the same fusion plasma but different fusion-power-core designs. There are only minor differences between the ARIES-II and ARIES-IV plasma parameters. ARIES-IV is a 1000-MWe reactor with an average neutron wall loading of 3 MW/m², and a mass power density of about 120 kWe/tonne of fusion power core. The reactor major radius is 6.1 m, the plasma minor radius is 1.5 m and the plasma elongation is 2, and the plasma triangularity is 0.67. The plasma current is low (6.8 MA), the magnetic field strength on-axis is 7.7 T (corresponding to a maximum field at the coil of 16 T), and the toroidal beta is 3.4% (corresponding to a Troyon coefficient of 6). The plasma operating regime is optimized such that most of the plasma current (~90%) is provided by the self-sustained bootstrap current. The ARIES-II reactor uses liquid lithium as the coolant and tritium breeder. The vanadium alloy, V-5Cr-5Ti, is used as the structural material so that the potential of low-activation metallic blankets can be studied. The ARIES-IV reactor uses helium as the coolant, a solid tritium-breeding material (Li20), and silicon carbide (SiC) composite as the structural material. The waste produced by neutron activation in both designs is found to meet the criteria allowing shallow-land burial under U.S. regulations. The cost of electricity for the ARIES-II-IV class of reactors is estimated to be about 20% lower than comparable, steady-state first-stability reactors (e.g. ARIES-I).

1. INTRODUCTION

The ARIES study was undertaken to determine the economic, safety, and environmental potential of steady-state tokamak fusion reactors and to identify physics and technology areas with the highest leverage for achieving attractive fusion power plants [1]. To this end, the ARIES team
Table I
Major Characteristics and Parameters of ARIES-II and ARIES-IV Second-Stability Tokamak Reactors

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Plasma major radius (m)</td>
<td>6.1 (5.6)</td>
</tr>
<tr>
<td>Plasma minor radius (m)</td>
<td>1.5 (1.4)</td>
</tr>
<tr>
<td>Plasma vertical elongation, $\kappa_X$</td>
<td>2</td>
</tr>
<tr>
<td>Plasma triangularity, $\delta_X$</td>
<td>0.67</td>
</tr>
<tr>
<td>Plasma current (MA)</td>
<td>6.8 (6.5)</td>
</tr>
<tr>
<td>Toroidal field on axis (T)</td>
<td>7.7 (8.1)</td>
</tr>
<tr>
<td>Toroidal field on the coil (T)</td>
<td>16</td>
</tr>
<tr>
<td>Toroidal beta</td>
<td>0.034</td>
</tr>
<tr>
<td>Poloidal beta</td>
<td>5.4</td>
</tr>
<tr>
<td>Electron density ($\times 10^{20}$ m$^{-3}$)</td>
<td>2.0 (2.2)</td>
</tr>
<tr>
<td>Plasma temperature (keV)</td>
<td>10.</td>
</tr>
<tr>
<td>Current-drive method</td>
<td>ICRF fast wave</td>
</tr>
<tr>
<td>Bootstrap-current fraction</td>
<td>0.9</td>
</tr>
<tr>
<td>Current-drive power (MW)</td>
<td>70</td>
</tr>
<tr>
<td>Impurity control system</td>
<td>Double-null</td>
</tr>
<tr>
<td>Fueling method</td>
<td>Pellet injection</td>
</tr>
<tr>
<td>Net electric output (MW)</td>
<td>1,000</td>
</tr>
</tbody>
</table>

has performed detailed reactor-design work and has explored reactor optimization and trade-off sensitivities using a cost-based systems code. The design effort has been directed to maximize the environmental and safety attributes of fusion through careful design and selection of low-activation material.

Four ARIES designs have been produced, each with a different degree of extrapolation in physics and technology, to yield a spectrum of tokamak reactor concepts based on various potential advances in physics and technology. In turn, this should provide a sensible basis to evaluate the potential of the tokamak as a commercial reactor.

All four ARIES designs are steady-state, 1000-MWe (net) power reactors. The ARIES-I design [2] assumes a minimum extrapolation in physics and, hence, is close to the present-day tokamak data base (e.g., ARIES-I operates in the first MHD stability regime [3]). It incorporates
Table I (Continued)

<table>
<thead>
<tr>
<th>Parameter</th>
<th>ARIES-II Fusion Power Core Parameters</th>
<th>ARIES-IV Fusion Power Core Parameters</th>
</tr>
</thead>
<tbody>
<tr>
<td>Coolant</td>
<td>Liquid lithium</td>
<td>Helium</td>
</tr>
<tr>
<td>Structural material</td>
<td>V-5Cr-5Ti</td>
<td>SiC composite</td>
</tr>
<tr>
<td>Tritium breeder</td>
<td>Liquid lithium</td>
<td>Li$_2$O</td>
</tr>
<tr>
<td>Neutron multiplier</td>
<td>None</td>
<td>Be</td>
</tr>
<tr>
<td>Blanket energy-multiplication ratio</td>
<td>1.4</td>
<td>1.2</td>
</tr>
<tr>
<td>Tritium breeding ratio</td>
<td>1.1</td>
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</tr>
<tr>
<td>Coolant inlet temperature (°C)</td>
<td>330</td>
<td>350</td>
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<tr>
<td>Coolant outlet temperature (°C)</td>
<td>610</td>
<td>750</td>
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<tr>
<td>Coolant pressure (MPa)</td>
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</tr>
<tr>
<td>Average neutron wall loading (MW/m$^2$)</td>
<td>3.1</td>
<td>2.9</td>
</tr>
<tr>
<td>Fusion power (MW)</td>
<td>1,950</td>
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<td>Thermal power (MW)</td>
<td>2,630</td>
<td>2,590</td>
</tr>
<tr>
<td>Gross thermal-cycle efficiency</td>
<td>46%</td>
<td>49%</td>
</tr>
<tr>
<td>Recirculating power fraction†</td>
<td>17%</td>
<td>21%</td>
</tr>
<tr>
<td>Net plant efficiency</td>
<td>38%</td>
<td>39%</td>
</tr>
<tr>
<td>Mass power density (kWe/tonne of FPC)</td>
<td>100</td>
<td>120</td>
</tr>
</tbody>
</table>

* Most of plasma parameters are similar for the two designs. If different, the ARIES-II parameters are given in parenthesis. † Includes 4% designated for auxiliary site power.
The ARIES-II and ARIES-IV designs assume potential advances in plasma physics (e.g., second stability operation [5]) that are predicted by theory but are not yet well established experimentally. These two designs have the same fusion plasma but different combinations of structural material and coolants.

The major characteristics and parameters of the ARIES-II and ARIES-IV designs are given in Table I. A cross section of the ARIES-IV fusion power core is shown in Figure 1. There are only minor differences between the ARIES-II and ARIES-IV plasma parameters, due to differences in the engineering design of the fusion power core (blanket energy multiplication, pumping power, thermal cycle efficiency, blanket and shield thickness). ARIES-IV is a 1000-MWe reactor with an average neutron wall loading of 3 MW/m², and a mass power density of about 120 kWe/tonne of fusion power core. The reactor major radius is 6.1 m, the plasma minor radius is 1.5 m, and the plasma elongation is 2, and the plasma triangularity is 0.67. The plasma current is low (6.8 MA), the magnetic field strength on-axis is 7.7 T (corresponding to a maximum field at the coil of 16 T), and the toroidal beta is 3.4% (corresponding
to a Troyon coefficient of 6 [3]). The plasma operating regime is optimized such that most of the plasma current (∼90%) is provided by the self-sustained bootstrap current [6].

The ARIES-II reactor uses liquid lithium as the coolant and tritium breeder. The vanadium alloy, V-5Cr-5Ti, is used as the structural material so that the potential of low-activation metallic blankets can be studied. The ARIES-IV reactor uses helium as the coolant, a solid tritium-breeding material (Li₂O), and silicon carbide (SiC) composite as the structural material. The blanket design is similar to that of ARIES-I [2], but several refinements have been incorporated in order to further improve the safety and environmental characteristics. The waste produced by neutron activation in both designs is found to meet the criteria allowing shallow-land burial under U.S. regulations [7]. The cost of electricity for the ARIES-II-IV class of reactors is estimated to be about 20% lower than comparable, steady-state first-stability reactors (e.g., ARIES-I).

In the following sections, we describe the physics and fusion power core engineering of the ARIES-II and ARIES-IV designs. Greater detail can be found in an extensive report of this research [2].

2. PLASMA ENGINEERING

Steady-state operation of tokamaks requires that plasma current be maintained by non-inductive means. The most likely candidates for external current-drive methods (i.e., neutral beam, ICRF fast wave, and lower hybrid) have projected low efficiencies at reactor-relevant plasma conditions [8]. Therefore, the optimum steady-state tokamak reactors operate at plasma regimes in which the total recirculating power is minimized by lowering the plasma current and maximizing the self-induced bootstrap-current fraction [6]. For tokamak reactors operating in the first MHD stability regime [3], such as ARIES-I and SSTR [9], this optimum regime of operation is at relatively high plasma aspect ratios (A ∼ 4 to 5), low plasma currents (I ∼ 10 MA), high poloidal betas (eβp ∼ 0.6), and relatively high safety factors, leading to a high bootstrap-current fraction (which is usually limited to ∼70% based on self-consistent MHD and bootstrap-current calculations). The choice to operate at lower plasma current in the first MHD stability regime then leads to a low toroidal beta (∼2% to 3%) and the need for high magnetic field in order to achieve an economical fusion power density.

In order to improve the reactor economics, the cost of the magnet system should be reduced (while maintaining or increasing the fusion
power density) through an increase in the plasma beta. Theoretical studies of the second MHD stability regime with higher plasma-beta values have been on-going [5]. Recent analyses [10], including those performed for the ARIES-III second-stability D-^3^He reactor [4], indicate that operation in second-stability requires an on-axis safety factor of ~2 with a corresponding increase in the edge safety factor. Furthermore, since second stability plasmas have a high poloidal beta, the bootstrap current fraction is high and can exceed the equilibrium plasma current. In this case, the current-drive system should drive the seed plasma current as well as canceling the bootstrap current-density overdrive in the outer regions of the plasma, which increases the current-drive power and associated costs. Thus, the optimum second-stability plasma equilibria are those in which the self-sustained bootstrap current density matches the required plasma equilibrium current. These equilibria have only moderate toroidal beta (~4% to 6%) (corresponding to a Troyon coefficient of 6).

Stability analyses of ARIES-II/-IV-type second-stability equilibria indicate that the plasma \( \beta \) decreases as the aspect ratio is increased (for \( A \sim 3 \) to 5). However, for a fixed maximum-field strength on the superconducting coils, the on-axis toroidal-field strength increases with increasing aspect ratio. As a result, the fusion power density (and the cost) remains insensitive to the value of aspect ratio (for \( A \sim 3 \) to 5). Given the engineering advantages of operating at higher aspect ratio and because preliminary indications that start-up at higher values of aspect ratio may be less difficult, a value of \( A = 4 \) has been adopted for the ARIES-II and ARIES-IV design.

Many second-stability equilibria have been examined for ARIES-II-IV reactors. These equilibria all have modest shaping, smooth pressure and surface-averaged current-density profiles, and a monotonic \( q \) profile. It was found that stability to \( n = \infty \) ballooning modes requires an on-axis safety factor, \( q_\text{o} = 2 \), a high edge safety factor \( (q*/q_\text{o} \gtrsim 2.4) \), and a high plasma triangularity \( (\delta_{95} \gtrsim 0.45) \). Unfortunately, none of the examined equilibria was found to be stable to \( n = 1 \) kink modes with no stabilizing walls. The reference equilibrium is, however, stable to \( n = 1 \) kink modes if a perfectly conducting wall is located at a radial distance of 0.25\( a \).

The effect of a finite conductivity wall on the kink stability of the plasma is unknown. For the ARIES-II-IV designs, it is assumed that the plasma rotation would compensate for finite wall conductivity. (An analogy can be drawn with the problem of a plasma ring which is translating inside a cylindrical wall with finite conductivity. The ring can
be stable if the plasma transit time is shorter than the wall skin time). Assuming a plasma rotational velocity of 50 km/s, the conducting wall for ARIES-II design (made of vanadium alloy) should be 7.5 cm thick. For the ARIES-IV design, a 2-cm thick zone of beryllium would be sufficient. These materials are present in both fusion power core designs and are located closer than 0.25*a to the plasma.

The average plasma ion temperature is 10 keV (central temperature of 26 keV) and the average ion density is $2 \times 10^{20} \text{ m}^{-3}$. About 90% of plasma current (6.8 MA) is provided by the bootstrap current. A central seed current of 0.6 MA is needed and is driven by ICRF fast waves [11] (19 MW at a frequency of 124 MHz with an efficiency of 23 mA/W). The bootstrap current density slightly exceeds the equilibrium current density in the middle of the plasma. The overdrive current in this region is 0.18 MA and lower hybrid waves are utilized for canceling the overdrive current in this region (12 MW at a frequency of 8 GHz with an efficiency of 11 mA/W). Because of collisional effects, the bootstrap current density decreases rapidly in the cold plasma edge. A total of 0.16 MA of current is supplied to this region by lower hybrid waves (39 MW at a frequency of 8 GHz with an efficiency of 3 mA/W).

The required energy confinement time is about 3 times the prediction of the ITER-89P L-mode [12] model. This value is comparable to those of VH-mode discharges in DIII-D and high-$\ell_i$ discharges both in TFTR and DIII-D. About 20% of plasma energy is radiated mainly as bremsstrahlung. An ash-particle to energy confinement ratio $\tau_p/\tau_E = 9$ is assumed. The alpha particle density is estimated to be $2 \times 10^{19} \text{ m}^{-3}$, corresponding to a burn-up rate of 18%. Loss of fast alpha particles due to stochastic ripple effect or prompt orbit loss has been analyzed and found to be negligible (the toroidal-field ripple is 0.5% on the outboard plasma edge).

Because of the high triangularity needed for second-stability operation, single-null divertors do not appear to be feasible. The ARIES-II-IV designs, therefore, utilize double-null divertor configurations. The reference designs incorporated gas-target divertors in which the plasma is terminated in a high-density gas [13]. Preliminary analyses indicate that the plasma temperature can be reduced to about 2 eV and the plasma energy is deposited more or less uniformly on the side wall of the divertor chamber (with a peaking factor of $\lesssim 1.5$). The average heat load on the outboard divertor wall is estimated to be $\sim 2.5 \text{ MW/m}^2$. Neutral-transport studies show that the leakage of divertor gas target into the main plasma is negligible. Because of low core-plasma temperature and
high density, a conventional divertor appears to be feasible using the same structural material and coolant for ARIES-II and ARIES-IV design. The surface of these divertors, however, should be coated with tungsten to limit sputtering erosion.

Plasma start-up is an important issue for second-stability plasma since plasma evolution to its final steady-state burn parameters should follow a stable access path to the second-stability regime. A preliminary start-up scenario has been identified which uses the current drive system both for heating and current drive. The total heating power required for ignition is about 90 MW, and, thus, additional 20 MW of auxiliary heating equipment may be needed for start-up. The current-drive power, however, appears to be adequate for current profile tailoring.

3. FUSION POWER CORE ENGINEERING

The ARIES-II-IV fusion power core is divided into 16 sectors (Figure 2). An entire sector is replaced as a single unit during a maintenance
operation. This approach is preferable to a piece-by-piece scheme in which components are removed sequentially through the space in between the TF coils (such as that proposed for ITER [12]) because it requires fewer connects/disconnects and joints. In ARIES-II-IV designs, all of the toroidal-field coils, most of the poloidal-field coils, and the associated support structures are located in a cryostat as shown in Figure 1. These coils remain in place during the maintenance operation. The outer leg of the toroidal-field coils is extended by 1.2 m so that a “pie-shaped” sector can be horizontally removed and replaced through the space between the TF coils.

The ARIES-II-IV toroidal-field (TF) coils represent a modest extension of present technology since the maximum field on the ARIES-II-IV toroidal-field (TF) coils is 16 T. The magnetic stored energy, however, is large because of the extended outer leg of the TF coils. The winding concept is similar to that of ARIES-III [4] and of a new concept proposed for ITER. This design has a very high capability to carry out-of-plane loads and, thus, no structure is needed in the outboard region of the torus, facilitating access for maintenance. Second-stability reactors require a lower toroidal-field ripple in order to maintain fast-\(\alpha\)-particle losses to an acceptable level (due to stochastic ripple effect or prompt orbit loss). The enlarged outer leg of the toroidal-field coils in ARIES-II-IV designs reduces the toroidal-field ripple on the outboard plasma surface to \(\sim 0.5\%\)—fast \(\alpha\) particle losses are negligible for this case.

3.1 ARIES-II Fusion Power Core

The ARIES-II fusion power core uses V-5Cr-5Ti vanadium alloy as the structural material and liquid lithium as the coolant and tritium breeder. The key design issue for the self-cooled liquid-metal blanket is the MHD pressure drop. Because of the relatively high magnetic-field strength at the inboard blanket location of the ARIES-II design, it appears that the MHD pressure drop is large and even exceeds the design-stress limit of the structural material. The ARIES-II fusion-power-core design assumes that a V-Ti-N coating will be formed on the blanket coolant-channel walls in order to reduce the MHD pressure drop. Several liquid-metal loop experiments [14-16] with liquid lithium and vanadium have reported the formation of this V-Ti-N coating on the V-Ti alloy surface. This coating is formed by the interaction of the nitrogen impurity in the liquid lithium with the alloy. The coating appears to be stable and self-healing. Long-term stability and electrical resistivity of this coating in a radiation environment is the critical R&D issue for the ARIES-II fusion power core.
Based on the electrical resistivity of this coating, which was recently measured at Argonne National Laboratory [17] to be about $5 \times 10^{-5}$ $\Omega$m, it is estimated that the MHD pressure drop can be reduced by a factor of 15 when compared to a similar design with uninsulated channel walls. Because the MHD pressure drop is reduced substantially by this coating, the ARIES-II blanket design has aimed at simplicity and optimum heat-transfer capability. The first-wall and blanket coolant channels are rectangular with cross sections of $25 \times 50$ mm and $55 \times 50$ mm, respectively. The channel wall thickness is 3 mm. The coolant inlet pressure is 0.5 MPa and the total MHD pressure drop is estimated at 0.3 MPa. The coolant inlet and outlet temperatures are, respectively, 330 and 610 °C and the maximum temperature of vanadium structure is 660 °C. Tritium recovery is based on a molten salt extraction technique and the tritium inventory in the primary lithium coolant is estimated at 140 g.

The total blanket and shield thicknesses in the inboard and outboard regions are, respectively, 1.1 and 1.6 m. The blanket is 0.2-m thick at the inboard, 0.5-m thick at the outboard, and contains 90% natural lithium and 10% vanadium structural material. The reflector and shield regions use vanadium-alloy structural material with stainless steel and B$_4$C fillers and are cooled by liquid lithium. The blanket energy-multiplication ratio is estimated at 1.38 and the tritium breeding ratio is 1.1. The end-of-life fluence for the vanadium alloy is estimated to be 16 MW-y/m$^2$ (based on degradation of mechanical properties by He embrittlement process). Thus, the shield lifetime is longer than 30 full power years.

Activation analyses indicate that the first wall, blanket, and shield qualify as Class-C shallow-land burial waste under U.S. regulations [7]. Loss of coolant analysis has been performed and the maximum structural temperature is estimated at 850 °C. The safety consequences of lithium fires are under study. Preliminary indications are that the ARIES-II reactor meets the requirements for passive safety (Level 3) as defined by ESECOM [18,19].

The ARIES-II reactor includes an intermediate coolant loop (with liquid lithium). An advanced steam power cycle, similar to those proposed for coal-burning power plants [20] is used for ARIES-II. The gross thermal efficiency is estimated at 46%.
3.2 ARIES-IV Fusion Power Core

Because of excellent safety and environmental features of the ARIES-I fusion power core [2], a similar design has been adopted for the ARIES-IV reactor (SiC-composite structural material and helium coolant). Several refinements, however, have been incorporated in order to further improve the safety and environmental characteristics. These include replacing the Li₂ZrO₃ tritium breeder of ARIES-IV with low-activation Li₂O and eliminating the tungsten coating of the divertor plates.

The ARIES-IV fusion power core employs a silicon-carbide-composite structural material to be manufactured as a large integrated piece utilizing techniques already in use or under development in the aerospace industry [21-25]. This composite retains many of the desirable features of bulk SiC ceramic, but the addition of SiC fibers greatly reduces the brittleness of the material and produces a high fracture toughness. The increase in toughness creates more freedom in engineering design and allows both tensile and compressive stress in the composites. Desirable features of SiC composite include high-temperature capability, high strength, extensive resource availability, and potentially good resistance to radiation damage. In addition, because the levels of induced activation and afterheat in SiC are quite low, the safety and environmental features of the reactor are enhanced.

The fusion-power-core design is based on a nested-shell configuration, with the coolant flowing through the coolant channels that are embedded in the shells. Each SiC-composite shell is manufactured as a single unit utilizing techniques already developed or under development by the aerospace industry. A sphere-pac of beryllium is located between the first two shells (total Be thickness is 4 cm). A sphere-pac of lithium-oxide (Li₂O) tritium breeder is located between the rest of the blanket shells. Tritium recovery is by a slow-flowing, low-pressure purge stream of helium between the shells. The reflector and shield components are also in the form of nested shells with the appropriate filler materials located between the shells.

The fusion power core is cooled by helium at 10 MPa. The coolant-flow configuration is in the poloidal direction, the coolant enters at the bottom of each shell and flows upwards. The coolant inlet and outlet temperatures are, respectively, 350 and 750 °C. The maximum temperature of the first-wall structural material is 950 °C. The blanket pumping power is 20 MW. An advanced Rankine power cycle (similar to that of ARIES-I) is adopted because the coolant outlet temperature is
sufficiently high, and because this cycle is planned for near-term, coal-fired power plants [20].

Neutronics analyses indicate that the total blanket and shield thicknesses in the inboard and outboard regions are, respectively, 125 and 160 cm. The blanket is 35-cm thick at the inboard and 60-cm thick at the outboard. Because of beryllium safety concerns (tritium production, reaction with air and water, and chemical toxicity), beryllium resource limitations, and high cost of beryllium, tritium breeding capability of the ARIES-IV blanket without any beryllium was considered. The “simulated” three-dimensional tritium breeding ratio was found to be ~1.05. It was estimated that a tritium breeding ratio of 1.1 is required to ensure adequate tritium breeding due to uncertainties in the nuclear data base and the differences between the neutronics model and actual blanket configuration. Therefore, a 4 cm beryllium zone is added to the front of the blanket (the total beryllium inventory is ~20 tonnes). The estimated tritium breeding ratio for the ARIES-IV reference design is 1.1 and the blanket energy-multiplication ratio is estimated at 1.18.

The reflector and shield regions also use SiC-composite structural material with SiC bulk material and B$_4$C fillers. The end-of-life fluence for the SiC composites is estimated to be 13 MW-y/m$^2$ (based on 3% burn-up limit). Thus, the shield lifetime is longer than 30 full power years. Activation analyses indicate that all fusion-power-core materials qualify as Class-C shallow-land burial waste under U.S. regulations [8]. Detailed analyses indicated that the ARIES-I design is passively safe (Level 2) as defined by ESECOM [18,19]. Because the major safety concern of the ARIES-I fusion power core (Li$_2$ZrO$_3$ breeder, tungsten coating of the divertor plate) is eliminated in the ARIES-IV design, ARIES-IV is expected to have better safety characteristics. Analyses are on-going to determine if ARIES-IV is inherently safe. This improvement in safety characteristics is expected to reduce the cost of electricity by 15%.

4. SUMMARY

The ARIES-II and ARIES-IV designs assume potential advances in plasma physics (e.g., second-stability operation) that are predicted by theory but are not yet well established experimentally. These two designs have the same fusion plasma but different combinations of structural material and coolants. Analyses indicate that the optimum second-stability plasma equilibria are those in which the self-sustained bootstrap current density matches the required plasma equilibrium current. These
equilibria have only moderate toroidal beta (\(\sim 4\%\) to \(6\%\)) (corresponding to a Troyon coefficient of 6). The improved physics performance of the ARIES-II-IV fusion power core are utilized both to lower the cost of electricity (by \(\sim 20\%\)) and to reduce the technological requirements (e.g., lower peak field on the coil, extended outer leg of TF coils to ease maintenance).

The physics of second-stability plasmas is yet to be explored even though experimental indications are encouraging. Although the ARIES-II-IV equilibria are robust for minor changes in the pressure and current density profile, the detail matching of bootstrap current density to the equilibrium current density requires careful control of plasma pressure and driven current profiles. Particle and energy transport in these plasmas are also unknown. Lastly, kink stability of second-stable plasma should be examined. Steady-state or long pulse (much longer than current skin time) experiments are required to thoroughly investigate the trade-offs of second-stability plasma and determine if their potential advantages, as highlighted by ARIES-II-IV research, can be realized.

As with other ARIES designs, the ARIES-II-IV design effort has been directed to maximize the environmental and safety attributes of fusion through careful design and selection of low-activation material. However, extensive fusion nuclear technology and material R&D are required so that the ARIES-II and ARIES-IV fusion power cores are realized. Because of the long lead time, this R&D (including the operation of a 14 MeV neutron source) should start as soon as possible so that these technologies would be available for the fusion demonstration power plant.

ACKNOWLEDGMENT

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DISCUSSION

F. ENGELMANN: Is there any analysis on how to get round the problem of instabilities caused by the presence of fast (fusion generated) ions in a tokamak reactor working with D–\(^3\)He?

F. NAJMABADI: Not to my knowledge.
D–^3^He TOKAMAK REACTOR

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Abstract

D–^3^He TOKAMAK REACTOR.
Two regimes — with modest and with high toroidal magnetic fields — of a D–^3^He tokamak reactor are studied on the basis of the plasma physics characteristics achieved at present. It is concluded that ITER can serve as an experimental base for a D–^3^He reactor.

1. INTRODUCTION

Today, the worldwide controlled fusion programme concentrates on employing the D–T fuel cycle because it is the reaction by which it is easiest to achieve fusion in the laboratory. The fusion performance in terms of $\eta E_T$ has been increased by two orders of magnitude in the last five years. The achieved plasma conditions are closest to those needed in the International Thermonuclear Experimental Reactor (ITER).

However, there are significant disadvantages with the D–T fuel cycle stemming mainly from the fact that 80% of the reaction energy is released in the form of neutrons and there is a need to breed, control and contain a large amount of radioactive tritium. The irradiation by 14.1 MeV neutrons causes severe damage to the structural components and induces a significant amount of radioactivity in the surrounding structures.

D–^3^He fuel is the most likely advanced fuel candidate because it produces only charged particles which are contained by the magnetic field. The fraction of the reactor power associated with neutrons from D–D reactions is much smaller than it is in the D–T reactors. Consequently, a D–^3^He reactor should have significant technological and ecological advantages over a D–T reactor.
The goal of the present paper is the investigation of D–^3^He tokamak reactors based on the plasma physics characteristics achieved at present and on the acceptable present day engineering solutions. A modest magnetic field D–^3^He reactor design and a high magnetic field design (Apollo) are studied, and a comparison of issues is carried out.

2. MODEST FIELD D–^3^He REACTOR DESIGN

On the basis of experimental data from the DIII–D tokamak with a poloidal divertor [1] and on the ITER scaling law for the energy confinement time, the reactor plasma parameters were optimized with respect to major radius R, aspect ratio $A = R/a$, elongation $k$, toroidal magnetic field $B_0$, and the plasma density and temperature profiles.

The optimal $^3^He/D$ density ratio in the plasma is equal to $n_{^3^He}/n_D \approx 0.7$ (41% $^3^He$, 59% D). The optimal temperature at the maximum Q value is equal to about 40 keV ($Q = P_{\text{fus}}/P_{\text{aux}}$, where $P_{\text{fus}}$ is the fusion power and $P_{\text{aux}}$ is the power injected into the plasma).

The effect of plasma density and temperature profiles was studied. The control over the plasma profiles allows the mode of reactor operation to be changed.

The major plasma and reactor parameters for this design are given in Table I. The value of $\beta_0$ is chosen to be equal to 18%, proceeding from the results achieved at DIII–D [1]: $\beta_0 = 44\%$, $\langle \beta \rangle = 11\%$. Steady state current drive is achieved by injection of the power $P_{\text{cd}} \equiv P_{\text{aux}}$ into the plasma, and the bootstrap current fraction is $0.56 I_p$.

The presence of a current between the divertor plates and the chamber walls on JET [2] and on DIII–D [3] confirms the opportunity to use direct energy conversion. At a fusion power of $P_{\text{fus}} = 2600$ MW the power of cyclotron and bremsstrahlung radiation was estimated to be $P_e + P_i \approx 1300$ MW, and the power carried by electrons and ions through the separatrix surface was estimated to be $P_s = P_s^e + P_s^i \approx 1300$ MW. At a $P_s^e/P_s^i$ ratio equal to 1/1, $P_s^e \approx P_s^i \approx 650$ MW. In principle, the ion power fraction going through the separatrix $P_s^i$ can be converted to electricity with high efficiency because of a locally non-ambipolar particle transport in the divertor layer across the magnetic field [4], as was observed in DIII–D.

The power $P_s^i \approx 650$ MW will enter the divertor plates. The ratio between the power to the outside divertor plate and that to the inside one will be $P_{\text{out}}/P_{\text{in}} = 2:1$, i.e. $P_{\text{out}} \approx 430$ MW and $P_{\text{in}} \approx 220$ MW. When a gaseous divertor is used, the power density to the divertor plates may not exceed 10 MW/m$^2$. This means that the area of a collecting surface should be $S_d = 2\pi R\Delta_d \approx 50\Delta m^2$, where $\Delta_d \approx 430$ MW/(10 MW/m$^2 \times 50$ m) $\approx 0.9$ m, i.e. the height of the divertor chamber should be about 1 m.

In the case of a 0.7:1 $^3^He/D$ density ratio, the total power in D–D and D–T neutrons is about 4–5%, and the average blanket power multiplication factor
TABLE I.  KEY PARAMETERS OF D–³He TOKAMAK REACTOR

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Modest field design</th>
<th>High field design (Apollo)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Direct conversion included</td>
<td>Direct conversion included</td>
</tr>
<tr>
<td>Major radius (m)</td>
<td>8</td>
<td>7.89</td>
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<tr>
<td>Aspect ratio</td>
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<td>Maximum field at TF coils (T)</td>
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<td>(T_e) (keV)</td>
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</tr>
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<td>1; 1</td>
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<td>Fusion power, (P_{\text{ fus}}) (MW)</td>
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<td>2600</td>
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<td>Synchrotron power (MW)</td>
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<td>200</td>
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<tr>
<td>Bremsstrahlung (MW)</td>
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<tr>
<td>Transport power (MW)</td>
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<td>1300</td>
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<td>(Q = P_{\text{ fus}}/P_{\text{ aux}})</td>
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<td>≥ 60</td>
</tr>
<tr>
<td>D–D neutron power (MW)</td>
<td>30</td>
<td>36</td>
</tr>
<tr>
<td>D–T neutron power (MW)</td>
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<td>Neutron wall loading (MW/m²)</td>
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<tr>
<td>Direct conversion efficiency (%)</td>
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<td>Thermal conversion efficiency (%)</td>
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<td>(1992 US $)</td>
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\(a\) Updated (1992) parameters for Apollo-L3.

\(b\) Thermal conversion only variation of Apollo-L3.
is \sim 2. Since breeding is not required and the neutron wall loading is low enough (\sim -0.03 \text{ MW/m}^2), a high temperature helium cooled shielding blanket and plasma facing components can be used to achieve high thermal cycle efficiency. Thus, for cyclotron and bremsstrahlung radiation and neutron components, a thermal cycle efficiency of about 40\% could be achieved. The power of the electron component will be converted with the same efficiency. The power of the ion component can be converted into electricity with an efficiency of \eta_i \approx 0.9. Thus, the total efficiency of fusion energy conversion into electricity may be about 0.5 (Table I).

For auxiliary equipment including coils and fuel cycle, 10\% on-site electrical power consumption was proposed.

3. HIGH FIELD, FIRST STABILITY, D-\textsuperscript{3}He REACTOR DESIGN

An alternative to the modest field, second stability design presented above is a low beta, high field design (Apollo) operating in the first stability regime [5, 6]. The design is more strongly based on the existing experimental database developed by a large number of tokamaks and, consequently, has different characteristics. Because of the higher magnetic field, synchrotron radiation constitutes a larger fraction of the total loss. This opens up the possibility of using synchrotron radiation to drive at least part of the plasma current and solid state technology to directly convert synchrotron radiation to electricity. In addition, the divertor power, first wall surface heating and neutron heating of the shield is converted to electricity by using a thermal cycle so that maximum electrical power is produced.

The major plasma and reactor parameters for this design are given in Table I. The on-axis magnetic field is 11 T, the plasma current is 53 MA, beta is 6.7\%, on using a Troyon coefficient of 0.035, and the average ion temperature is 57 keV. By using ITER-89P transport scaling with an H-mode enhancement factor of 4, the confinement product, nT_\text{E} = 3 \times 10^{21} \text{ s/m}^3 and the triple product, nT_\text{E}^3 = 1.6 \times 10^{23} \text{ s-keV/m}^3. The plasma current of 53 MA is composed of 23 MA of bootstrap current, 18 MA driven by synchrotron radiation, and 12 MA driven by neutral beam injection using 138 MW of absorbed neutral beam power (197 MW(e)).

Synchrotron radiation accounts for the dominant power loss because of the high magnetic field and low beta. A beryllium coating is used on the first wall to provide high reflectivity (98\%) for synchrotron radiation. About half the synchrotron power exits the plasma chamber through waveguides and is transported to rectifying antennas (rectennas) in a remote chamber for direct conversion to electricity. The waveguides are angled in the toroidal direction to remove toroidal momentum; the reaction on the electrons drives a plasma current. Rectennas potentially allow highly efficient conversion of synchrotron radiation to electricity. They perform well for 2.45 GHz microwave energy transmission; the concept is extrapolated here to very high frequencies (3–30 THz) as are encountered in synchrotron radiation. High energy conversion efficiency (80\%) is predicted but remains to be demonstrated,
although the technology lies well within state of the art integrated circuit dimensions. The rectenna geometry would be a dielectric slab with the antennas facing the synchrotron radiation, with coolant channels through the dielectric, and with the remaining circuit components on the opposite side of the slab, connected by strip line techniques. Since rectennas for this application require development, a backup option is to convert the synchrotron radiation to electricity by using the thermal cycle. A design using this option is also shown in Table I. Because of the lower efficiency (44%, using an organic coolant) for thermal conversion, this design is somewhat larger although the impact on the cost of electricity is only 10%. This backup option still uses synchrotron current drive.

In order to have a permanent first wall not requiring replacement during the lifetime of the reactor, the neutron wall loading has been constrained to be less than 0.1 MW/m². The neutrons come from the \(^2\text{H}(d,n)^3\text{He}\) and \(^3\text{H}(d,n)^4\text{He}\) reactions, where the tritium is produced by the \(^2\text{H}(d,p)^3\text{H}\) reaction. Neutrons from the D–T reactions account for about 70% of the total neutron power; about half the tritium diffuses out of the plasma before it reacts with deuterium. This produces tritium in the exhaust stream which is recovered and stored until it decays to \(^3\text{He}\). The resulting steady state inventory of tritium stored in a separate facility is 59 kg. The vulnerable tritium inventory in the reactor is 11 g. Neutron production in the plasma can be reduced by increasing the \(^3\text{He}/D\) ratio, but this reduces the power density in the plasma. The design operation point (\(^3\text{He}/D = 0.6\)) is deuterium rich in order to maximize the power output and reduce the cost of electricity while meeting the constraint on the neutron wall load. Lower neutron wall loadings could be obtained with larger (more costly) reaction chambers.

In order to keep the concentration of protons and alpha particles down to a tolerable level of 5% for each, the ash particle confinement time must be about equal to the energy confinement time. Whether the plasma will provide this naturally or will require some form of active ash removal is an open question, as will be discussed in Section 4.

4. COMPARISON OF ISSUES

For both D–\(^3\text{He}\) tokamak reactors presented here, the major physics issues are achieving adequate energy confinement, removing fusion ash and generating the plasma current. For the modest field case, a further issue is experimental verification of the second stability regime. The main technological challenges are handling the energy deposited by the thermal quench in a disruption and the steady state power deposited on the divertor. The thermal energy of a D–\(^3\text{He}\) tokamak plasma is about five times that in a D–T tokamak, but the magnitude of the divertor heat load is similar for both fuels. The concepts presented here also invoke advanced energy conversion technology for part of the fusion power, but these are desirable rather than necessary features.
The energy confinement time for the first stability reactor must be about four times better than that given by the ITER-89P L-mode scaling relation [7]. This value relies on future progress in reducing transport and is partly justified by recent VH mode results achieved in the DIII-D experiments [8]. The plasma current must also be high, ~ 50 MA, to increase both energy confinement and \( \beta \). In order to avoid choking the plasma with fusion ash, the ratio of ash particle confinement time to bulk energy confinement time must be relatively small, \( \tau_{\text{ash}} / \tau_{\text{bulk}} = 1-2 \). This regime appears to be present for L-mode operation in TEXTOR [9] and DIII-D [10], but it must be demonstrated in higher confinement operation, or an active means of enhancing fusion ash transport will be necessary.

Because of the high plasma current, even with a relatively good match of the bootstrap current radial profile to the desired plasma current radial profile, a large current must be driven. For the first stability case, the synchrotron radiation power drives about one-third of the required current, using a modification to the original Dawson and Kaw concept [11], in which the waveguide that transports the synchrotron radiation to the rectenna chamber serves the function of the original absorbing side of a fish scale wall. The efficiency of either neutral beam or fast wave current drive is low, requiring > 100 MW of injected power. More efficient current drive techniques would not only reduce this input power, but would also allow a larger parameter space operating region.

5. CONCLUSIONS

The two reactor regimes studied here, with modest and high toroidal magnetic fields, reveal the principal options for a D–\(^3\)He tokamak reactor based on the existing database in plasma physics. The required reactor parameters do not differ much from those adopted for the ITER design. An essential effect of the plasma density and temperature profiles on the reactor reactivity Q has been emphasized. Control over the plasma profile can be one of the main tools for the transition from low to high efficiency mode of operation.

The opportunity to use divertor layer plasma flow and cyclotron radiation direct energy conversion allows the net efficiency of the reactor to be increased by 10–15% over a D–T reactor.

The engineering problems of the D–\(^3\)He reactor design with magnetic field \( \leq 6 \) T are similar to those in the ITER design. Therefore, ITER can serve as an experimental base for the next step development, an experimental reactor with D–\(^3\)He fuel.

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THE D–³He FUELLED FIELD REVERSED CONFIGURATION REACTOR ARTEMIS-L

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Abstract

THE D–³He FUELLED FIELD REVERSED CONFIGURATION REACTOR ARTEMIS-L.

A neutron lean D–³He fuelled field reversed configuration fusion reactor is studied on the basis of the former high efficiency ARTEMIS design. Certain improvements such as effective axial contracting plasma heating and cusp type direct energy converters as well as an empirical scaling of the energy confinement are introduced. The resulting total neutron load onto the first wall of the plasma chamber is as low as 0.1 MW/m², which enables the life-time of the first wall or the structural materials to be longer than the whole life of the reactor. The attractive characteristics of the neutron lean reactor follow these of the ARTEMIS design: it is socially acceptable as far as radioactivity and fuel resources are concerned, and electricity is cheaper than that from a light water reactor. Critical physics and engineering issues of the ARTEMIS-L reactor are clarified.
1. INTRODUCTION

Progress in fusion research has mainly been achieved in the tokamak concept, which is the basis of a scientific feasibility experiment to be performed in the near future. Nevertheless, a number of engineering problems have to be solved before deuterium–tritium fuelled fusion will become acceptable in commercial power reactors. The engineering problems appearing in commercial D–T fusion reactors are attributed to the 14 MeV neutrons. These problems include neutron damage of the first wall and the structural materials of the reactor, and the disposal of large amounts of radioactive waste.

D–\(^3\)He fusion fuel is supposed to mitigate these engineering problems. With this fuel, the fraction of the power carried by neutrons in the total fusion power may be as small as a few per cent, and more than 70% of the fusion power is carried by charged particles such as 14.7 MeV fusion protons and the diffused thermal fuel component. By conducting these charged particles to highly efficient direct energy converters, we can achieve an environmentally sound, highly efficient and cheap fusion plant on the basis of D–\(^3\)He fuel. Very high \(\beta\) value and plasma temperature, good confinement of the plasma energy as well as accessibility of high power direct energy converters to utilize D–\(^3\)He fuel for a commercial fusion reactor are required. We know that plasma confined in a field reversed configuration (FRC) meet these requirements [1].

A comprehensive conceptual design of the D–\(^3\)He fuelled FRC fusion reactor ARTEMIS [2] has been carried out to examine the attractive characteristics of a combination of D–\(^3\)He fuel and FRC. The small neutron yields mitigate the engineering problems drastically, and the estimated cost of electricity from the plant is lower than that from a conventional light water reactor. The plasma parameters of ARTEMIS were, however, optimized so as to maximize the overall plant efficiency up to 60%; the resulting neutron load onto the first wall is still as large as 0.42 MW/m\(^2\). It may, therefore, be worth while examining another version of the D–\(^3\)He fuelled FRC fusion reactor by minimizing the neutron yield and retaining the attractive characteristics of ARTEMIS.

2. OPTIMIZATION OF D–\(^3\)He BURNING FRC PLASMAS

The figures of merit of a fusion reactor such as the overall plant efficiency, \(P_{\text{net}}/P_f\), or the ratio of the power carried by neutrons, \(P_n\), to the net electric power, \(P_{\text{net}}\), can be estimated from the particle number continuity equations of the respective species in the power balance equation. The averaged beta value \(\langle \beta \rangle\), is assumed to be 98%, and the particle confinement time, \(\tau_p\), is approximated as the double energy confinement time \(\tau_E\), which can be seen from FRC experiments [3]. Deviations of the ion velocity distribution from Maxwellian that are attributed to nuclear reactions are taken into account [4]. The power carried by the neutrons, \(P_n\), and
the overall plant efficiency are shown in Fig. 1(a) and (b), as functions of the density ratio \( n_{3\text{He}}/n_D \) and the averaged plasma temperature \( T \), respectively. The power ratio \( P_n/P_{\text{net}} \) takes a minimum value of 0.033, and the resulting overall plant efficiency is 55.8\% at a density ratio of \( n_{3\text{He}}/n_D = 1.35 \) and an averaged plasma temperature of \( T = 83.5 \text{ keV} \).

The required value of the confinement parameter, \( n_i\tau_E = 3.5 \times 10^{21} \text{ s/m}^3 \), can be obtained by choosing the plasma radius \( r_s \) according to the empirical confinement scaling [5]:

\[
\tau_E(s) \equiv 3.0 \times 10^{-5} \left( r_s(m)/\sqrt{\rho_0} (\text{m}) \right)^{2.7} T(\text{keV})
\]

as well as the external magnetic field. The quantities \( r_s \) and \( \rho_0 \) denote the plasma radius and 'gyroradius' estimated from the external magnetic field, respectively. Since a strong external magnetic field or a high plasma density yields a small plasma radius and, consequently, a large heat flux, we have used a maximum heat flux of 2 MW/m\(^2\). Thus we have a plasma radius of \( r_s = 1.7 \text{ m} \) for the optimized FRC plasma. The representative parameters of this D-\(^3\text{He} \) burning FRC plasma are shown in Fig. 2.
FIG. 2. Total view of D-³He fuelled FRC reactor ARTEMIS-L consisting of formation section, burning section and direct energy converter.
3. DESIGN OF ARTEMIS-L

An overview of the neutron lean D-\(^{3}\)He fuelled FRC reactor ARTEMIS-L is shown in Fig. 2. The reactor consists of a formation section, a burning section and a pair of direct energy converter sections, which are connected linearly in order to ease disassembly or repair of the device.

An initial FRC plasma is produced in the formation section. The chamber is arranged axisymmetrically so as to produce D-\(^{3}\)He/FRC plasmas reliably even for the case of very low gas pressure. A fast rising theta pinch discharge with a filling gas pressure of 0.05 Pa and a cusped bias field of 0.035 T produces an FRC plasma. During the last phase of the pinch discharge, an axial contraction of the plasma takes place [6], which induces collisionless shock heating leading to a plasma temperature \(T_i = 3\) keV. As the electron density \(n_e = 4.1 \times 10^{20}/m^3\) and the external magnetic field \(B_e = 0.7\) T, an ion temperature of 3 keV is obtained. The FRC plasma is then transferred to the adjacent burning section along the lines of force.

\[P_f = 1791\ MW\]
\[\text{Radiation} \quad 636\ MW \rightarrow 669\ MW \rightarrow \text{Thermal converter} \quad n = 0.30 \rightarrow 253\ MW \rightarrow \text{thermal converter} \quad n = 0.70 \rightarrow 414\ MW \rightarrow \text{P}^2 = 1042\ MW(e)\]
\[\text{Fuel component} \quad 545\ MW \rightarrow 411\ MW \rightarrow 480\ MW \rightarrow 131\ MW \rightarrow 29\ MW \rightarrow \text{P}^2 = 1042\ MW(e)\]
\[\text{Others} \quad 97\ MW\]
\[P_{aux} = 42\ MW\]

(a) Power Flow

\[\text{\(^3\)He: 63.7 kg/year} \]
\[\text{\(^3\)He: 0.50 \times 10^{-3} \text{ kg/min}}\]
\[\text{\(^3\)He: 3.34 \times 10^{-6} \text{ kg/min}}\]

(b) Particle Flow

FIG. 3. (a) Power and (b) particle flow charts of D-\(^{3}\)He fuelled FRC reactor ARTEMIS-L. The overall plant efficiency is estimated to be 56% with a helium fuel of 63.7 kg per year. The duty factor is assumed to be 75%.
A D-\(^3\)He burning FRC plasma is produced for 50 s after the transfer, by a combination of neutral beam injection of 1 MeV/100 MW, fuelling and slow magnetic compression up to \(B_e = 5.4\) T. For fuelling, we have introduced the 'Pac-Man' method: a small deuterium ice pellet with liquid \(^3\)He inside is injected into the burning section. At this moment, the FRC moves towards the pellet with a speed of \(5 \times 10^5\) m/s and places it deep inside the FRC before the pellet evaporates.

A large fraction in the D-\(^3\)He fusion energy is carried by charged particles along the lines of force to a pair of direct energy converters (DEC). A power of 577 MW carried by the diffused fuel components is led to positively biased plates located close to the line cusps, and a power of 545 MW carried by the 14.7 MeV protons is converted to electricity through the travelling wave direct energy converters. Another fraction of the fusion power amounts to 669 MW; it is converted to heat in the first wall and the neutron shielding blanket with borated heavy water and is ultimately converted to electricity.

The power and particle flows are illustrated in Fig. 3. The approximately equal amounts of thermal power, the power carried by the 14.7 MeV protons and the power carried by the thermal diffused fuel components are led to thermal converters, travelling wave direct energy converters and cusp type direct energy converters, respectively. The conversion efficiencies of the converters are estimated to be 30\%, 76\% and 65\%, respectively. Thus, we obtain a net electric power of 1000 MW(e). The estimated overall plant efficiency is approximately 55.8\%, which should be compared with the value of 60.1\% for the high efficiency ARTEMIS. Approximately 63.7 kg of \(^3\)He and 50.4 kg of deuterium are consumed for an operation of nine months per year. Diffused fuel \(^3\)He and deuterium are pumped out with turbomolecular pumps, together with the fusion products \(^4\)He, \(^1\)H and T. \(^3\)He and deuterium are separated from the other products and fed back to the fuel reservoirs. Tritium is oxidized to form T\(_2\)O, whose total content of 225 kg is stored in a water reservoir. A small quantity — \(3.3 \times 10^{-6}\) kg — of \(^3\)He is produced per minute in the water reservoir through the T(\(\beta^-\))\(^3\)He reaction and also fed back into the \(^3\)He reservoir.

4. DISCUSSION OF THE RESULTS

The entire engineering basis of our reactor design is conventional. No fuel breeding is necessary, and the neutron flux is as low as \(2.9 \times 10^{16}\) n·m\(^{-2}\)·s\(^{-1}\). The maxima of magnetic field and stress of materials applied are, respectively, less than 6.5 T and 350 MPa. No development of new material is hence needed to build the reactor. Straight structure and low radioactivity allow easy maintainability.

Because of the simple and compact structure of the D-\(^3\)He fuelled FRC reactor ARTEMIS-L, the total direct cost and the total plant capital cost — according to ESECOM studies — are estimated to be as low as US $1030 million and US $1800 million, respectively. The cost of electricity (COE) is estimated to be 30.5
mill/kW·h, which is lower than that from a light water reactor. We assumed a $^3$He fuel cost of 0.2 M$/kg. The fraction of the fuel cost in the COE is only 0.3%, and the uncertainty in the fuel cost does not affect the COE significantly.

Apparently the neutron lean D–$^3$He fuelled FRC reactor ARTEMIS-L is attractive as far as social acceptability and cost of electricity are concerned. Nevertheless, the development of highly efficient neutral beams of 1 MeV and high power direct energy converters is needed for the operation of the ARTEMIS-L reactor. The detailed physics of plasma transport and microinstabilities related to anomalous electron transport are important issues for further study.

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REACTOR STUDIES ON ADVANCED STELLARATORS

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Abstract

REACTOR STUDIES ON ADVANCED STELLARATORS.
The Helias configuration envisaged for the next stellarator experiment Wendelstein 7-X (W 7-X) in Garching has been extrapolated to reactor dimensions. The main parameters of the Helias reactor HSR are: major radius 20 m, plasma radius 1.6 m, magnetic field 5 T. The stored magnetic energy in the modular coils is 74 GJ. Stresses in the coils can be kept below 120 MPa; the superconductor is NbTi at 1.8 K. The operational window of the plasma in HSR is $n(0) = (3-4) \times 10^{20} \, m^{-3}$ and $T(0) = 15-17 \, keV$. Confinement properties, $\alpha$ particle behaviour and divertor geometry are discussed in the paper.

1. INTRODUCTION

The attractive features of a modular stellarator reactor are the inherent capability of steady state operation, the absence of disruptive instabilities, a low level of recirculating power and the modularity of the coil system. Stellarator reactor studies have been undertaken ever since the beginning of stellarator research. The first study of a modular stellarator reactor ASRA 5C [1] mainly concentrated on technical issues neglecting the limitations set by confinement and stability. In contrast, the Helias configuration [2] offers the chance to develop a self-consistent reactor concept where the plasma losses, MHD stability limits and $\alpha$ particle losses are not prohibitive to ignition. The aim of the present paper is to describe the main features of a Helias reactor (HSR).

2. DIMENSIONS OF A HELIAS REACTOR (Fig. 1)

The Helias reactor HSR is a scaled-up version of the magnetic field configuration envisaged for the Wendelstein 7-X (W 7-X) experiment [3]. The magnetic field

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is a five period Helias configuration. A toroidally periodic mirror with a value equal to the inverse aspect ratio must be provided for good confinement of high energy particles. The bootstrap current is eliminated. The lower limit on the aspect ratio \( A \geq 12 \) is mainly determined by stability considerations. To determine the dimensions of the reactor the following conditions and criteria have been imposed:

— The magnetic field on axis is limited to 5 T, which allows one to stay within NbTi technology.
— Sufficient space for shield, blanket and divertor systems has to be provided. This requires a distance of at least 1.1 m between the first wall and coil winding pack.
— The coil system consists of one set of modular coils. Extra coils as used in the W 7-AS and W 7-X experiments for parameter scanning are not necessary.
— Since the confinement time scales roughly with the plasma volume the minimum size is also determined by the ignition condition.

These conditions lead to minimum reactor dimensions as listed in Table I. It was found that the dominating criterion was to provide sufficient space for blanket and shield, the dimensions of which are rough estimates extrapolated from the ASRA 6C study (Fig. 2). The reference case HSR 5-8 has a major radius of 20 m and an average plasma radius of 1.6 m; the available space for blanket and shield is 1.1 m at the narrowest position on the inboard side. Several other versions with smaller and larger dimensions have also been investigated for comparison.

<table>
<thead>
<tr>
<th>TABLE I. DATA OF THE HELIAS REACTOR HSR 5-8</th>
</tr>
</thead>
<tbody>
<tr>
<td>Average major radius (m)</td>
</tr>
<tr>
<td>Average plasma radius (m)</td>
</tr>
<tr>
<td>Plasma volume (m³)</td>
</tr>
<tr>
<td>Surface of first wall (m²)</td>
</tr>
<tr>
<td>Magnetic field on axis (T)</td>
</tr>
<tr>
<td>Max. field on coils (T)</td>
</tr>
<tr>
<td>Stored magnetic energy (GJ)</td>
</tr>
<tr>
<td>Rotational transform on axis</td>
</tr>
<tr>
<td>Rotational transform on boundary</td>
</tr>
<tr>
<td>Number of field periods</td>
</tr>
<tr>
<td>Number of coils</td>
</tr>
<tr>
<td>Average coil radius (m)</td>
</tr>
</tbody>
</table>
3. THE MODULAR COIL SYSTEM

The magnetic field of the Helias reactor is generated by a single set of 50 modular non-planar coils; extra poloidal field coils are not necessary. The maximum magnetic field on the coils is 10.6 T; this would allow supercritical helium at 1.8 K to be used for cooling the NbTi coils. The coil aspect ratio is about 5; this value is in an optimum range in view of the maximum field at the coils and the mechanical stresses inside the coils. Because of the fivefold symmetry of the magnetic field and the symmetry within one field period, there are only five different coils types, which appreciably facilitates the manufacturing process. The dimensions of the largest coil
are 11 m × 6.5 m, and the weight is about 220 t. The coils can thus be transported to the site of the reactor. The reactor version HSR 5-8 with an average magnetic field $B_0 = 5$ T on axis has a stored magnetic energy of 74 kJ. The moderate helicity of the magnetic axis in the Helias configuration and the toroidally varying shape of the coil bores require a related helical geometry of the blanket and shield.

The magnetic forces on the modular coils are among the main technical problems in HSR 5-8. They determine the geometry of the support system and the mechanical stresses, thereby setting the technical limits of the coil system. The inhomogeneous field distribution inside the coils leads to an inhomogeneous force distribution and a resulting net force on each coil pointing not only in the radial direction but also in the vertical and lateral directions. The magnitude of the lateral force density is comparable to the radial force density. Some coils even see a force directed radially outward, while others experience a force in the vertical direction which leads to a torque on the field period. The maximum net coil force is about 130 MN; and for the whole field period the resulting force amounts to about 350 MN directed towards the torus centre.

Stresses depend on the support system of the coils, which must be optimized with regard to material limits, safety margins and minimization of the amount of structural material. The coils of HSR 5-8 must be surrounded by stainless steel housings. Studies with various dimensions of this housing and the location of intercoil support elements have been made; in the case considered here, a housing thickness of 35 cm at the outer coil face and of 20 cm on the three other sides of the coil is assumed. Stress maxima tend to arise where the coils are curved; therefore extra reinforcement is provided in these areas. Support elements are located between adjacent coils; these are mainly confined to the inner region, towards the torus centre, and leave the outer region accessible to heating. The maximum tensile stress in the winding pack is 120 MPa, and the maximum shear stress 30 MPa. The elastic data of the superconductor envisaged for W 7-X were used in these calculations [4]. In the coil casing the maximum stress is 520 MPa. These first results show that the stresses can be kept at a tolerable level although the support system is not yet fully optimized.

4. MAINTENANCE CONCEPT

The maintenance concept of the Helias reactor is based on the modularity of the coil system. The absence of poloidal field coils and other interlinked coil systems in the Helias reactor allows a modular structure of the reactor. In each period a module fo six coils will be displaced horizontally until the first wall and blanket are accessible from the open ends. The shield is considered as a semi-permanent component. This procedure of dismantling the reactor is only envisaged for replacing the blanket and first wall; repair of the first wall armour and the divertor target plates will be done through the portholes by remote handling. The critical components of this maintenance scheme are the interfaces between the removable modules.
5. DIVERTOR CONCEPT

In the Helias configuration two options for a divertor configuration exist:

(a) The last magnetic surface is surrounded by an ergodic layer without large magnetic islands. Plasma losses occur mainly along the helical edge where field line diversion is at its maximum. The target plates follow the helical edge and are appropriately shaped in order to minimize the local wall load (helical trough). This concept is rather insensitive to changes of the rotational transform and therefore well suited to an experiment where, for the sake of experimental flexibility, the transform may be varied [5].

(b) The second concept utilizes the existence of large magnetic islands at the plasma boundary; in the reference reactor five islands exist ($\iota = 1$). Streaming along the field lines the plasma crosses the X-point region of the islands and arrives

![Image](image-url)

**FIG. 3.** Diffusive broadening of SOL. The Poincaré plot of particles streaming along field lines is shown. The effective diffusion coefficient is 1 m$^2$/s. The current in the sweep coils is 50 kA. The figure shows how the position of the strike points is shifted by changing the sign of the currents. The average power load is reduced by roughly a factor of two.
at the rear of the island after four to five toroidal transits. Target plates are located in those regions where the plasma has reached the maximum distance from the main plasma after three to four toroidal transits.

The islands and the X-line helically encircle the plasma column, which also forces the target plates to follow this geometry. The target plates are located in regions of minimum separatrix angle $\gamma$ since there the outstreaming plasma is at the maximum distance from the main plasma. Small changes of the magnetic field by internal or external perturbations may ergodize the islands but this does not spoil the divertor action as long as the position of the O points stays fixed.

The target plates are located at the top and bottom of the plasma column following the X-line of the $\iota = 1$ island in the toroidal direction from $\varphi = -18^\circ$ to $\varphi = +18^\circ$. The dimensions of one plate are roughly $12 \times 1 \text{ m}^2$, and the overall area available is $120 \text{ m}^2$, about 50% ($60 \text{ m}^2$) being loaded by the plasma. If the diffusive broadening (Fig. 3) of the scrape-off layer (SOL) is too small to prevent excessive power load, the option of a sweep coil system exists as proposed for the W 7-X experiment. Since islands are resonance effects relatively small currents are needed to change the shape of the islands. This sweep coil system can also be used to control small field errors, which alter the position of the island and thus endanger the divertor operation. These arguments demonstrate the basic feasibility of an island divertor in a Helias reactor; the geometry of the target plates and the surrounding breeding blanket has not yet been optimized.

6. PLASMA PERFORMANCE

Because of the low value of Pfirsch-Schlüter currents in Helias configurations ($\langle |j||j| \rangle \approx 0.7$), the Shafranov shift in HSR is negligibly small. A consequence of this effect is a rather small variation of the magnetic surfaces and the rotational transform with rising plasma pressure, thus also leaving the topology of the boundary region nearly unchanged. Another result of the small Shafranov shift and the large aspect ratio of HSR is a nearly equal neutron wall load at the inboard and outboard sides of the first wall. In contrast to the ASRA 6C study where, owing to the large Shafranov shift, a peak neutron load of 2.5 MW/m$^2$ was found, in HSR this maximum wall load is only 1.6 MW/m$^2$.

MHD stability in Helias configurations is limited by the ideal ballooning mode. Numerical calculations for W 7-X show stability up to $\tilde{\beta} = 4.3%$; detailed calculations for HSR 5-8 have not yet been made; it is, however, expected that this limit can be pushed to $\tilde{\beta} \approx 5\%$ by a proper choice of the magnetic field. Neoclassical transport in Helias configurations has been investigated by Monte Carlo techniques [6] and other numerical methods [7]. The large superbanana losses, which have threatened former stellarator reactor concepts, do not occur in HSR. The neoclassical losses can be characterized by a small effective helical ripple of $\epsilon_{\text{eff}} \approx$
TABLE II. PLASMA PARAMETERS OF HSR 5-8 WITH R = 20 m AND a = 1.6 m.
The energy confinement times $\tau_E$ listed are those required for ignition; $\tau_{LGS}$, $\tau_{LHD}$ and $\tau_{GRB}$ follow from empirical scaling laws.

<table>
<thead>
<tr>
<th></th>
<th>A</th>
<th>B</th>
<th>C</th>
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</thead>
<tbody>
<tr>
<td>Abundance of $\alpha$ particles (%)</td>
<td>1.0</td>
<td>10</td>
<td></td>
</tr>
<tr>
<td>$Z_{eff}$</td>
<td>1.5</td>
<td>1.7</td>
<td></td>
</tr>
<tr>
<td>Peak density (10$^{20}$ m$^{-3}$)</td>
<td>3.0</td>
<td>3.0</td>
<td>4.0</td>
</tr>
<tr>
<td>Average density (10$^{20}$ m$^{-3}$)</td>
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<td>1.33</td>
<td>1.77</td>
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<tr>
<td>Peak temperature (keV)</td>
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<td>20.0</td>
<td>15.0</td>
</tr>
<tr>
<td>Average temperature (keV)</td>
<td>7.49</td>
<td>8.81</td>
<td>6.61</td>
</tr>
<tr>
<td>Volume averaged beta (%)</td>
<td>4.57</td>
<td>5.12</td>
<td>5.12</td>
</tr>
<tr>
<td>Plasma energy (MJ)</td>
<td>692</td>
<td>776</td>
<td>776</td>
</tr>
<tr>
<td>$\alpha$ heating power (MW)</td>
<td>682</td>
<td>576</td>
<td>600</td>
</tr>
<tr>
<td>Fusion power (GW)</td>
<td>3.34</td>
<td>2.82</td>
<td>2.93</td>
</tr>
<tr>
<td>Neutron power (GW)</td>
<td>2.66</td>
<td>2.24</td>
<td>2.33</td>
</tr>
<tr>
<td>Neutron wall load (MW/m$^2$)</td>
<td>1.2</td>
<td>1.1</td>
<td>1.1</td>
</tr>
<tr>
<td>Energy confinement time, $\tau_E$ (s)</td>
<td>1.16</td>
<td>1.65</td>
<td>1.75</td>
</tr>
<tr>
<td>$\tau_{LGS}$ (Lackner–Gottardi) (s)</td>
<td>1.27</td>
<td>1.46</td>
<td>1.81</td>
</tr>
<tr>
<td>$\tau_{LHD}$ (LHD scaling) (s)</td>
<td>0.79</td>
<td>0.91</td>
<td>1.15</td>
</tr>
<tr>
<td>$\tau_{GRB}$ (Gyro-Bohm) (s)</td>
<td>0.63</td>
<td>0.73</td>
<td>0.90</td>
</tr>
<tr>
<td>Fusion product $n_{DT}(0)\tau_E T(0)$</td>
<td>52.0</td>
<td>68.8</td>
<td>73.8</td>
</tr>
</tbody>
</table>

1–2.5%. Since the favourable parameter regime of the Helias reactor is high density and low temperature (see Table II), the collisionality is still high enough to keep transport coefficients at a tolerable level; although the trapped particles are in the $1/\nu$ regime the neoclassical thermal conductivities stay below 1 m$^2$/s. Therefore neoclassical diffusion in HSR is not prohibitive to ignition.

Since prompt losses of high energy $\alpha$ particles lead to a reduction of heating power and severe damage of the first wall the magnetic field of HSR has to be tailored to reduce these losses. The large mirror component ($\delta B/2B \geq 10\%$) confines a significant fraction of reflected $\alpha$ particles within a field period, and the poloidal drift, which increases with rising plasma pressure, leads to sufficiently good confinement of 3.5 MeV $\alpha$ particles up to 0.1 s [8]. This time is comparable with the slowing-down time. The confinement of thermal $\alpha$ particles is being investigated [9].
7. PLASMA PARAMETERS

A realistic prediction of the plasma parameters in a Helias reactor requires knowledge of transport processes and radiation losses of the fusion plasma close to the MHD stability limit. Since reliable information on these processes is not available, a more reasonable procedure is to postulate the parameters of an attractive stellarator reactor and check whether phenomena known from theory and experiment meet the requirements of the stellarator reactor. For the purpose of mass utilization the thermal power output of the Helias reactor should be around 3 GW. The MHD stability limit of $\beta = 4-5\%$ provides another condition on the plasma parameters. Since there is no disruptive density limit in stellarators, there is some freedom in selecting the density in the Helias reactor; confinement scaling laws favour the high density regime. In the present analysis the density is considered to be a free parameter with $n(0) = (2.0-4.0) \times 10^{20} \text{ m}^{-3}$. Furthermore, dilution of the fuel concentration by $\alpha$ particles, oxygen and carbon is taken into account, the maximum $Z_{\text{eff}}$ being 1.8.

Figure 4 shows the fusion power as a function of average $\beta$ for various dimensions of the reactor. The plasma profiles in these calculations are rather broad, being similar to those in today's stellarator experiments. The fusion power in a Helias reactor ($R = 20 \text{ m}$) ranges between 2 and 3 GW for $\beta = 4-5\%$. In the case of a stability limit of $\beta = 4\%$, the size of the reactor must be increased to 24 m to obtain a fusion power of 3 GW.

**FIG. 4.** Thermal fusion power versus average $\beta$. $B = 5 \text{ T}, n(0) = 3 \times 10^{20} \text{ m}^{-3}, Z_{\text{eff}} = 1.8$. ● HSR with $\beta = 5.1\%, P = 3 \text{ GW}$. ○ HSR with $\beta = 4\%, P = 1.8 \text{ GW}$. □ HSR with $\beta = 4\%, P = 1.2 \text{ GW}$. (The latter case is calculated with peaked profiles similar to those modelled for ITER.)
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In Table II the reactor plasma parameters are listed in detail. In column A a fraction of 1% α particles is assumed, which could be realistic in the startup and ignition phases. In the steady state phase, dilution by α particles and impurities may be higher ($f_\alpha = 10\%$). Columns B and C present plasma parameters with $Z_{\text{eff}} = 1.7$.

8. CONCLUSIONS

The extrapolation of a Helias configuration to reactor size leads to a minimum major radius of 20 m. In spite of this large radius the mass of the reactor core is 25 800 t, which is comparable to the mass estimated for the ITER tokamak. The stored magnetic energy is also comparable: 74 GJ in HSR, 63 GJ in ITER. The ITER test reactor is designed for 1 GW fusion power only.

The magnet system of HSR can be realized within NbTi technology if higher magnetic fields are not required for plasma performance. Any increase in the magnetic field or the reactor dimensions would lower the requirements on plasma stability and confinement. The mechanical stresses in the winding pack are slightly above those envisaged in W 7-X. The fusion power in HSR is limited by the MHD stability limit and the dilution effect by cold α particles. At a fraction of 10% α particles, the reactor must operate at a beta value of about 5% to deliver a fusion power of 2.9 GW. The density scaling of the energy confinement time ($\tau_E \propto n^{0.6}$) favours the high density regime ($n(0) = (3-4) \times 10^{20} \text{ m}^{-3}$), which is not limited by disruptions in HSR. In this regime a confinement time of 1-2 s is required for ignition. As is shown in Table II, this time is reached by LGS scaling, whereas the other two empirical scaling laws do not predict ignition; their confinement times are too small by a factor ≤ 2. Any improvement in confinement which may arise from the optimized Helias configuration or H mode operation is not yet accounted for. However, the aim of the present study was to demonstrate that — under rather conservative technical assumptions and with our present knowledge of plasma performance — the modular Helias configuration envisaged for W 7-X is a viable fusion reactor candidate and a serious competitor with other reactor concepts.

REFERENCES

DISCUSSION

A. GIBSON: In the Artsimovich Memorial Lecture (these Proceedings, Vol. 1, p. 3), Professor Kaw spoke of the need to reduce the time-scale for the development of a fusion reactor, and P.H. Rebut has spoken of the huge step from JET to ITER and the still larger step from ITER to a reactor. Considering that stellarators are now at least two large steps from a JET-like device and performance, what do you think could be a credible strategy and time-scale for the development of a stellarator reactor?

H. WOBIG: This is a difficult question. As a reasonable next step we propose the Wendelstein 7-X experiment. Many questions and problems of importance for the Helias reactor will be addressed in this experiment. The step from W7-X to a reactor is still so large that an intermediate step may be necessary. However, if the results and achievements of tokamaks are combined with those of stellarators, the big step from W7-X to a demonstration machine is justified.

S.A. HOKIN: Please comment on the auxiliary heating required to start up such a reactor.

H. WOBIG: In the startup phase the fraction of alpha particles is certainly less than 10%. I have found that the ignition curve comes down to a density of n(0) ≈ 2.5 × 10^{20} m^{-3}. In this case ignition could be achieved with 40 MW ECRH power at 5 T (140 GHz) if the cold alpha particle content is less than 5%.

R.J. GOLDSTON: I am surprised to see a favourable isotope scaling in your scalings since to my knowledge the published scalings and data do not support this.

F. WAGNER: Lackner-Gottardi scaling is T \propto A^{-0.2}. The positive A^{0.5} scaling was added in the light of experience on tokamaks, notably ASDEX.
RESULTS OF INERTIAL FUSION REACTOR DESIGN STUDIES: OSIRIS AND SOMBRERO

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Abstract

RESULTS OF INERTIAL FUSION REACTOR DESIGN STUDIES: OSIRIS AND SOMBRERO.

An 18-month study to evaluate the potential of inertial fusion energy (IFE) for electric power production has been completed. The study included the conceptual design of two inertial fusion power plants, one using an induction linear accelerator, heavy ion (HI) driver and the other using a krypton fluoride (KrF) laser driver. The study included the conceptual design of the reactors, power conversion and plant facilities, and drivers. Environmental and safety aspects, technical issues, technology development needs, and economics of the final point designs were assessed and compared. The paper summarizes the key results of the study.
1. **OSIRIS HEAVY-ION DRIVEN POWER PLANT**

Osiris is a 1000 MW, HIB-driven power plant design [1,2,3]. The operating parameters for Osiris are listed in Table I. The Osiris chamber design (Fig. 1) features a porous carbon fabric blanket that is filled with the molten salt Flibe (2LiF-BeF₂). A key feature of Osiris is the use of low-activation ceramics in a configuration in which brittleness and

<table>
<thead>
<tr>
<th>Operating Parameters:</th>
<th>Osiris</th>
<th>SOMBRERO</th>
</tr>
</thead>
<tbody>
<tr>
<td>Driver Energy (MJ)</td>
<td>5.0</td>
<td>3.4</td>
</tr>
<tr>
<td>Target Gain</td>
<td>86.5</td>
<td>118</td>
</tr>
<tr>
<td>Yield (MJ)</td>
<td>432</td>
<td>400</td>
</tr>
<tr>
<td>Pulse Repetition Rate (Hz)</td>
<td>4.6</td>
<td>6.7</td>
</tr>
<tr>
<td>Driver Efficiency (%)</td>
<td>28.2</td>
<td>7.5</td>
</tr>
<tr>
<td>Fusion Power (MW)</td>
<td>1987</td>
<td>2680</td>
</tr>
<tr>
<td>Thermal Power (MW)</td>
<td>2504</td>
<td>2849</td>
</tr>
<tr>
<td>Thermal Efficiency (%)</td>
<td>45</td>
<td>47</td>
</tr>
<tr>
<td>Gross Electric Power (MW&lt;sub&gt;e&lt;/sub&gt;)</td>
<td>1127</td>
<td>1359</td>
</tr>
<tr>
<td>Driver Power (MW&lt;sub&gt;e&lt;/sub&gt;)</td>
<td>82</td>
<td>304</td>
</tr>
<tr>
<td>Auxiliary Power (MW&lt;sub&gt;e&lt;/sub&gt;)</td>
<td>45</td>
<td>55</td>
</tr>
<tr>
<td>Net Electric Power (MW&lt;sub&gt;e&lt;/sub&gt;)</td>
<td>1000</td>
<td>1000</td>
</tr>
</tbody>
</table>

| Environmental and Safety Assessment:          |        |          |
| Routine Tritium Release (mrem/yr)             | 2.4    | 0.7      |
| Accident - WB Early Dose (rem)                |        |          |
| Reactor                                      | 0.1    | 2.2      |
| Target Factory                               | 2.8    | 3.0      |
| Waste Disposal Class                         |        |          |
| Chamber                                      | A      | A        |
| Breeder                                      | A      | C        |
| Shield                                       | A      | A        |

| Economic Assessment:                         |        |          |
| Direct Capital Costs (M$)                    |        |          |
| Land & Land Rights                           | 11.6   | 10.5     |
| Structures and Site Facilities               | 137.6  | 276.1    |
| Reactor Plant Equipment                      | 504.2  | 615.5    |
| Turbine Plant Equipment                      | 225.8  | 256.3    |
| Electric Plant Equipment                     | 66.2   | 70.0     |
| Miscellaneous Plant Equipment                | 18.5   | 19.9     |
| Heat Rejection Equipment                     | 44.8   | 52.0     |
| Driver Equipment                             | 587.5  | 579.1    |
| Total Direct Cost (M$)                       | 1596   | 1879     |
| Total Capital Cost (M$)                      | 3091   | 3640     |
| Unit Cost ($/kWe-gross)                      | 2743   | 2676     |
| Unit Cost ($/kWe-net)                        | 3091   | 3640     |
| Cost of Electricity (mills/kWh)              | 56     | 67       |
FIG. 1. Osiris uses a molten salt (Flibe) breeder/coolant in a carbon fabric blanket.

leak-tightness are not issues. A thin layer of liquid Flibe coats the carbon fabric first wall to protect it from x-ray and debris damage. Part of this protective layer is vaporized with each pulse. The vaporized Flibe condenses in a spray at the bottom of the chamber. Flibe circulates through the blanket and serves as the primary coolant and tritium breeding material. The blanket support structures and vacuum vessel are made of low-activation carbon/carbon composites. Liquid lead is used in the intermediate loop to transfer heat to a steam generator and a double reheat steam power cycle.

The 5 M J heavy ion driver uses singly charged (A = 131) Xenon ions [4]. The design approach is conservative in that it does not use beam combination, separation, or recirculation. Twelve beams are accelerated and transported through an array of compact, high-performance Nb,Sn quadrupoles. At the end of the accelerator the array of beams is split into two groups of six beams to give two-sided target illumination. The design maximizes component standardization. It uses a propagation mode in the accelerator with constant beam radius, quadrupoles with constant strength and length, a single quadrupole array design, and only two inductor cell designs, one each for low and high energy.
2. SOMBRERO KrF-LASER DRIVEN POWER PLANT

SOMBRERO is a 1000 MW, KrF-laser driven power plant design [1,2,5]. The operating parameters for the base case design are given in Table I. The SOMBRERO chamber (Fig. 2) is constructed of a low-activation carbon/carbon composite. The first wall is protected with a 0.5 torr of xenon buffer gas. Solid Li$_2$O particles flow by gravity through the blanket as the primary coolant and breeding material. This moving bed solid breeder blanket design has all the advantages of solid breeders but does not need a high pressure gas coolant or a separate He gas loop for tritium extraction. Helium is used to fluidize the particles for transport around the heat transfer loop. Liquid lead is used in the intermediate loop to transfer heat to a steam generator and a double reheat steam power cycle.

The 3.4 MJ KrF driver uses e-beam pumped amplifiers and angular multiplexing for pulse compression [6]. The laser uses relatively small (~60 kJ)
final amplifiers and a new plasma cathode technology for the e-beams in order to improve the laser system efficiency. Amplifiers are grouped in four-unit modules to minimize hardware requirements. Sixty beams are used to provide uniform target illumination. Grazing incidence metal mirrors are used as the final optical component to remove the dielectric focusing mirrors from the direct line of sight of high energy neutrons.

3. ENVIRONMENTAL AND SAFETY ASSESSMENT

Both power plants have attractive environmental and safety characteristics [7,8]. The results are given in Table I and summarized here. While remote maintenance will be required for the vessels, hands-on maintenance will be possible one day after shutdown in the regions outside of the shielding walls (in vicinity of intermediate heat exchangers).

The chamber and shield of both reactors qualify for shallow land burial as Class A low-level waste. Without reprocessing, Li\(_2\)O from SOMBRERO qualifies as Class C, and Flibe from Osiris qualifies as Class A low-level waste.

The dose to the maximally-exposed individual from the atmospheric routine release of tritium from both reactors is well below the 10 mrem/year EPA limit. The estimated off-site whole body (WB) doses from tritium and activation product releases in a conservative reactor accident scenario is 2.2 rem from SOMBRERO and 0.1 rem from Osiris. The dose from accidental release of tritium from the target factory and tritium recovery systems of either plant is ~3 rem. The WB early dose from any of these accidents is below the 5 rem level where evacuation plans are required. The very low off-site doses eliminate the need for N-Stamped nuclear grade components, which are only required if the dose exceeds 25 rem.

4. ECONOMIC ASSESSMENT

The results of the economic assessment are summarized in Table I [9,10]. The capital cost of the SOMBRERO plant is higher than the Osiris plant primarily because of the higher gross electric power (1359 MW\(_e\) vs 1127 MW\(_e\)) required to achieve a net power of 1000 MW\(_e\). The SOMBRERO reactor building is also much larger and more expensive than Osiris since it is sized to contain the dielectric final focusing mirrors, which are located 50 m from the center of the chamber. The resulting cost of electricity (COE) for SOMBRERO is about 20% higher than for Osiris.

Both of these plants have COEs that compare favorably with the most recent magnetic fusion energy (MFE) power plant designs. The ARIES-I tokamak reported a total capital cost of $4.4 B and a COE of 81 mills/kWh [11]. While the indirect costs and COE were calculated on a consistent basis...
in these two studies, more work is needed to ensure consistency in assumptions used to determine direct capital costs. Never-the-less, we believe the IFE advantages are real since the first walls, blankets, and vacuum vessels can be simpler, and radiation shielding requirements are much less. Also, the IFE driver costs (~$580-590 M direct) are not much more expensive than the sum of the plasma heating and magnet system costs for ARIES-I (~$450 M direct).

5. TECHNOLOGY ASSESSMENT

We made a preliminary assessment of development needs and priorities for the two IFE plant designs. In determining priorities we considered:
(a) The current state of technology maturity
(b) The criticality of the technology to plant performance
   (a technology is critical if no alternatives exist)
(c) Development cost
(d) Development lead time
(e) Fraction of the costs in experiments vs analysis

We concluded that driver development should have the highest priority. Target systems (design, production, delivery) ranked second in priority slightly ahead of reactor systems. We note, however, the danger of allocating virtually all R&D resources to driver development. A balanced program that includes materials development, reactor concept development and analysis, and work on beam delivery issues is needed since all these impact the overall system design including the characteristics required of the eventual driver.

6. CONCLUSIONS

The IFE Reactor Design Studies have helped to quantify the potential of IFE for electric power production. The conclusions are quite favorable and suggest the importance of continued development of IFE:
(a) IFE power plants can have very attractive environmental and safety characteristics. With the proper choice of materials, the routine and accidental releases of radioactivity are small, and the materials used in the reactors and shields will qualify for shallow land burial. In our studies, innovative designs using low-activation materials were developed for both KrF-laser and HI-driven power plants.
(b) IFE power plants can have attractive economic aspects. The overall plant economics compare favorably with MFE designs. Our driver architecture studies have led to higher laser efficiency and a smaller HI driver with lower costs than previous studies. Driver costs are comparable to the cost of heating and magnet systems for MFE power plants, while IFE reactor and shielding costs are substantially less.
(c) IFE development needs a balanced approach that includes not only driver development (the highest priority), but also research and development on target and reactor systems. We also conclude that many of the unresolved technical issues can be addressed in non-nuclear experiments (e.g., target injection and tracking) or in low-yield integrated experiments (e.g., first wall vaporization and condensation).

REFERENCES


DISCUSSION

R. TOSCHI: Could you specify the amount of waste, including the pellet/target debris, for instance as compared with a tokamak reactor?

W.R. MEIER: The Osiris carbon fabric first wall is replaced every year, but has a small total mass. The SOMBRERO first wall is replaced every five years, so the total waste should be comparable to ARIES-I. Materials used in the target are continually recycled, so the total mass of activated target material is not that large.

S. BODNER: What is the impact on cost of electricity of a higher pellet gain and of recycling of the waste heat in the KrF laser cell?

W.R. MEIER: Recycling of laser waste heat is already accounted for in the analysis. Our systems analyses indicate that for the given target gain curve, the minimum cost of electricity occurs at a slightly lower driver energy and gain. If the gain was a factor of two higher for a given energy, the recirculating power would be reduced, and the cost of electricity would be somewhat lower.

A. GIBSON: How far is the laser driver technology which you mentioned from the scale of your requirements (power, energy, convergence, repetition rate, etc.) for the reactor?

W.R. MEIER: Many of the component technologies have been demonstrated at nearly full scale. They have not been integrated into a multibeam, multimegajoule laser as described here. The plasma cathode diode technology for the e beam pumped amplifiers was invented by TEXTRON and has been demonstrated.
FEASIBILITY STUDY FOR AN INDUCTIVELY OPERATED DAY-LONG TOKAMAK REACTOR

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Abstract

FEASIBILITY STUDY FOR AN INDUCTIVELY OPERATED DAY-LONG TOKAMAK REACTOR.

A pulsed tokamak fusion power plant with only inductive current drive but with steady electric power output is described. The tokamak plasma performance is determined on the basis of the ITER database assuming the power law of the energy confinement time with an enhancement factor of 1.72 and a Troyon factor of 2.7. The divertor heat flux and temperature are manageable. The major and minor radii of the tokamak plasma are 10 and 1.87 m. The maximum field of the toroidal coils is 12 T (8.08 T at the plasma centre) and the plasma current is 13.9 MA. The operation time and the dwell time of the tokamak are taken to be 13 h and 300 s, respectively. Thermal power input to the turbine generator during the dwell time of the tokamak discharge is supplied by burning the hydrogen gas which is produced by an electrolysis cell throughout the power plant operation. The net electric output power is 1026 MW and the plant efficiency is 31%. The circulation power in the power plant, that is, the input power of the electrolysis cell necessary for producing a sufficient amount of hydrogen gas, is 32.4 MW. The power consumed in auxiliary systems is assumed to be 80 MW. The reactor is operated about $10^4$ times during the assumed lifetime of 30 years.

1. INTRODUCTION

Ultra-long-pulse tokamak reactors with the use of only inductive current drive have been studied by several design groups [1–3]. The advantages of such a reactor are simplicity, because of there being no non-inductive current drive system, and high plant efficiency due to the high energy multiplication factor of the reactor. The disadvantages are discontinuous electric power output and mechanical and thermal
fatigue of the reactor structure induced by repeated operation. The latter disadvantages, however, can be suppressed if the operation times are reduced to within the allowable level [4, 5]. Recently, cyclic tokamak operation has successfully been performed with the small tokamak STOR-1M [6] and the large tokamak JET [7]. Those experimental results show that the dwell time between successive operations of the tokamak can be made fairly short.

In this paper we consider a long pulse tokamak reactor, IDLT (inductively operated day-long tokamak), combined with an electric power regulation system [8] which makes up the electric power output during the dwell time of the tokamak. Thus continuous electric output from the power plant is maintained.

2. DESCRIPTION OF THE POWER PLANT SYSTEM

Figure 1 shows the power flow of the IDLT power plant. In the figure $P_1$ is the fusion power, $P_\alpha$ the alpha particle power, $P_n$ the neutron power, $P_{th}$ the thermal output power from the reactor, $P_0$ the gross electric power converted from the thermal power with the efficiency $\eta$, $P_{net}$ the net electric output power from the plant, $P_E$ the input power consumed in the water electrolysis cell for hydrogen gas production and $P_{aux}$ the power consumed in auxiliary systems such as vacuum pumps, coolant pumps and so forth. $M$ is the neutron energy multiplication factor in the blanket and $\epsilon$ is the fraction of $P_0$ used for plant operation. During the dwell time of the tokamak reactor the flow of reactor coolant in the primary loop is adjusted so as to keep the blanket and divertor plate temperature nearly constant, suppressing the

$\alpha$ particles $P_\alpha = P_1/5$

$P_n = 4P_1/5$

$\text{Plasma}$

$\text{Neutrons}$

$P_\alpha$

$P_n$

$\text{Blanket}$

$\text{MP}$

$\text{Thermal output}$

$P_{th} = P_\alpha + MP_\alpha$

$\text{Turbine}$

$\text{Generator}$

$P_{net} = (1-\epsilon)P_0$

$\epsilon P_0$

$\text{Deuterium reservoir}$

$\text{Hydrogen reservoir}$

$\text{Electrolysis cell}$

$P_E$

$\text{Auxiliary systems}$

$P_{aux}$

$\text{Boiler}$

$H_2$

$D_2$

$\text{FIG. 1. Power flow diagram of the IDLT reactor.}$
temperature rise due to the afterheat generated by induced activity. Thermal power equivalent to $P_n$ is supplied to the turbine generator from the hydrogen boiler through the secondary coolant loop, whereby the thermal stresses on the blanket and divertor plate structures and also on the turbine blades are minimized, and the turbine generator generates electric power continuously in spite of the shutdown of the tokamak reactor. The hydrogen boiler may be combined with a heat reservoir, if necessary, for quick startup. In the present analysis, however, the influence of the heat reservoir is neglected. The electrolysis cell produces deuterium fuel as a by-product.

3. DESIGN PARAMETERS

The tokamak plasma performance has been analysed using the ITER database in a way similar to that described in Ref. [3]. For the global energy confinement time, the ITER-89 power law multiplied by the factor 1.72 has been adopted. Defined and deduced parameters of the tokamak and plant system are summarized in Table I, with asterisks indicating the defined parameters. For the profiles of plasma temperature, density and current, parabolic profiles with different indices are assumed. We have chosen a high aspect ratio to generate a sufficient amount of bootstrap current for suppressing current profile peaking. The maximum toroidal field adopted in this design is 12 T, which is almost possible with the present technology. The deuterium production rate of the electrolysis cell is sufficiently larger than the gas feed rate of the fuelling system.

Figure 2 shows the dependences of the tokamak pulse length, the plant efficiency defined by $\eta_{pl} = P_{net}/P_{th}$, the plasma volume, the total volume of the winding pack of superconducting TF coils and operation times with the availability of 75% in 30 years. The other parameters used for calculation are given in Table I. In order to assure $\eta_{pl}$ larger than 30% we have taken the plasma major radius of 10 m and admitted a slight increase of the plasma volume and the volume of the winding pack of the TF coils, which we take as a measure of cost. Then from Fig. 2, one pulse length is 13 h and the number of operation times is $1.5 \times 10^4$.

The plasma temperature and the density at the operation point in the POPCON diagram [9] are at the stable point of thermal instability and sufficiently lower than the pressure limit with the Troyon factor $g = 3.0$. The plasma heating power necessary for D-T ignition at the startup time is 40 MW and the time for startup is less than 20 s. This level of heating power has already been achieved in the present experimental devices. The same heating system is used at the ramp-down phase of the plasma to avoid disruptions caused by rapid plasma cooling. Utilization of the ICRF heating system may be appropriate for its simple construction and low cost. The profile of the plasma current, which is the sum of the Ohmic current profile and the bootstrap current profile, is amenable to the MHD stability criterion [10]. The divertor plasma temperature and the heat flux on the divertor plate listed in Table I
<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
<th>Description</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>*Major radius, R (m)</td>
<td>10</td>
<td>Power to SOL, $P_{\text{SOL}}$ (MW)</td>
<td>443</td>
</tr>
<tr>
<td>*Minor radius, a (m)</td>
<td>1.87</td>
<td>Bremsstrahlung power loss, $P_{\text{B}}$ (MW)</td>
<td>67</td>
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<tr>
<td>Aspect ratio, A</td>
<td>5.35</td>
<td>Synchrotron radiation power loss, $P_{\text{i}}$ (MW)</td>
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<tr>
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<td>1.85</td>
<td>*Synchrotron power loss recovery fraction</td>
<td>0.9</td>
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<td>*Triangularity, $\delta$</td>
<td>0.4</td>
<td>Divertor heat flux, $P_{\text{dv}}$ (MW/m$^2$)</td>
<td>18-22</td>
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<tr>
<td>Plasma volume, $V$ (m$^3$)</td>
<td>1277</td>
<td>Divertor temperature, $T_{\text{dv}}$ (eV)</td>
<td>16-39</td>
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<tr>
<td>Plasma surface, $S$ (m$^2$)</td>
<td>999.6</td>
<td>Inboard blanket–shield thickness (m)</td>
<td>1.4</td>
</tr>
<tr>
<td>*Max. toroidal field, $B_{\text{max}}$ (T)</td>
<td>12</td>
<td>*OH coil field, $B_{\text{OH}}$ (T)</td>
<td>10</td>
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<tr>
<td>Toroidal field, $B_{\text{T}}$ (T)</td>
<td>8.08</td>
<td>OH coil flux, $2\Phi_{\text{OH}}$ (V-s)</td>
<td>2456</td>
</tr>
<tr>
<td>*Safety factor, $q_{95}$</td>
<td>3</td>
<td>OH coil energy, $W_{\text{f}}$ (GJ)</td>
<td>88</td>
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<tr>
<td>Plasma current, $I_p$ (MA)</td>
<td>13.9</td>
<td>TF coil energy, $W_{\text{f}}$ (GJ)</td>
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<tr>
<td>Bootstrap fraction</td>
<td>0.37</td>
<td>*TFC winding pack current density, $j_i$ (MA/m$^2$)</td>
<td>30</td>
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<tr>
<td>He concentration, $n_{\text{He}}$ (%)</td>
<td>10</td>
<td>TFC winding pack volume (m$^3$)</td>
<td>450</td>
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<td>*Effective charge number, $Z_{\text{eff}}$</td>
<td>1.5</td>
<td>*Energy multiplication factor in blanket, $M$</td>
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<td>*Average temperature, $\langle T \rangle$ (keV)</td>
<td>15</td>
<td>*Heat to electricity conversion efficiency, $\eta$</td>
<td>0.34</td>
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<td>Peak temperature, $T_0$ (keV)</td>
<td>25</td>
<td>Plant efficiency, $\eta_{\text{pl}}$</td>
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<td>Troyon factor, $g$ (%-m-T/MA)</td>
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<td>Toroidal beta, $\beta_t$ (%)</td>
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<td>Poloidal beta, $\beta_p$ (%)</td>
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<td>*Confined enhancement factor, $f_L$</td>
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<td>Energy confinement time, $\tau_E$ (s)</td>
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<td>Net electric output power, $P_{\text{net}}$ (MW)</td>
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<td>Gross electric output power, $P_{\text{g}}$ (MW)</td>
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<td>Fusion power, $P_f$ (MW)</td>
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<td>Neutron wall loading, $P_n$ (MW/m²)</td>
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<td>Thermal output power, $P_{\text{th}}$ (MW)</td>
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<td>Electric power fraction for plant operation, $\varepsilon$</td>
<td>0.099</td>
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<td>Electrolysis cell input power, $P_E$ (MW)</td>
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<tr>
<td>Efficiency of electrolysis cell, $\eta_E$</td>
<td>0.82</td>
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<td>*Efficiency of hydrogen boiler, $\eta_H$</td>
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<tr>
<td>Amount of H₂ gas in reservoir (MPa-m³)</td>
<td>9975</td>
<td></td>
<td></td>
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Note: * denotes parameters of interest.
have been estimated using the Borrass model [11] and Harrison and Hotston scaling [12]. In this case the ratio of the average density to the scrape-off layer density is about 2. A rather broad density profile is preferred to reduce the divertor heat flux.

4. CONCLUSION

A long pulse tokamak reactor combined with a hydrogen generator and boiler system is studied assuming the ITER technology level. It makes an attractive fusion power plant since most of its technology is well established, high plant efficiency is available and the reactor structure is simplified by the absence of a current drive system. The concept described here may be one of the shortest paths to the realization of a commercial fusion power plant, as long as a highly efficient and reliable current drive system has not been established.

ACKNOWLEDGEMENTS

The authors acknowledge the valuable advice on power plant construction given by Y. Fukai and T. Nakazato of Toshiba Corporation.
REFERENCES

[10] OKANO, K., et al., The $q_0$ criterion for a long pulse tokamak discharge assisted by bootstrap current (in preparation).

DISCUSSION

R. TOSCHI: What is the impact on your design of pulsed operation versus steady state?

N. INOUE: If long pulse length and short dwell time are achieved, the inductively driven tokamak reactor is superior to the non-inductively driven one. It has many advantages, such as high power efficiency, high reliability, low plasma current and simple structure. Fatigue in structural materials through repeated operation is avoided by reducing the burn cycles. Thermal stress and breaks in electrical power output are eliminated by using a hydrogen thermal power source in IDLT. As for the MHD stability, we have found a stable configuration for the day-long tokamak which does not use any additional current drive.
PILOT PLANT: A SHORTENED PATH TO FUSION POWER

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Abstract

PILOT PLANT: A SHORTENED PATH TO FUSION POWER.

Previous fusion reactor studies have focused on the characteristics of fusion reactors in a mature, commercial market, on full-scale "demonstration reactors" as commercial prototypes, and on other engineering development facilities. The projected large size and high capital cost of the development facilities present significant practical impediments to the development of fusion as a commercial power source. In other technologies, "pilot plants" have been constructed in advance of full-scale facilities. Such plants have had the characteristics of small size, low capital cost, and a limited set of objectives, while still having the integrated performance deemed necessary to gain experience with the operating characteristics of the new technology. A range of possible tokamak fusion pilot plants is considered, having as the primary objective providing requisite fusion power experience to an electric
utility prior to construction of a full-scale demonstration reactor. Two approaches are explored, having the characteristics of either net electricity production or only the production of high-grade heat. The effects of choices such as mode of plasma heating and normal versus superconducting coils are also examined. Since tokamak scaling laws do not seem to permit simply "miniaturizing" the DEMO, fusion pilot plant designs incorporate only certain essential features of a power plant, while leaving the development of other features to complementary, specialized facilities.

1. UTILITY INTERESTS

Electric utilities are most interested in areas for which they would have direct responsibility in building, operating and maintaining a reliable, economic electric power supply system. These would include the following areas:

(a) **Production and Extraction of High-Grade Heat.** A pilot plant [1] must at least be able to generate heat suitable for conversion to electricity through a proven steam cycle.

(b) **Operation and Maintenance.** This area is of greatest utility concern. The pilot plant should provide the utility with direct experience in monitoring, accessing and maintaining fusion-specific components, and provide data for the design of subsequent reactors.

(c) **Instrumentation, Control and Protection.** The utility will be especially interested in those aspects that differ significantly from their current experience.

(d) **Safety, Environment and Licensing.** Pilot plant licensing should bring out all the safety, environmental and licensing issues associated with operating a full fusion power system and allow utilities to compare these with fission power systems. Of particular interest are the inherent safety characteristics of fusion, the degree of applicability of existing nuclear power regulations, occupational and public exposure, public reaction to tritium, and waste management.

(e) **Fuel Cycle, Tritium Self-Sufficiency, Waste Management and Decommissioning.** Design and operation of the pilot plant should provide experience and data for future reactors.

The pilot plant should incorporate proven applicable utility materials control, quality assurance, and operating procedures. The utilities should provide and train the operating and maintenance staff and, if possible, the pilot plant should be built at a utility power plant site.

2. TOKAMAK FOR NET ELECTRICITY PRODUCTION

A pilot plant whose first priority is net electricity production [2] should have the following features:

(a) Fusion power should be as low as possible to reduce cost.
(b) Burn pulses should be long compared with the current ramp-up and ignition phases.
(c) Large aspect ratio is desirable in order to realize long burn pulses with minimal power.
(d) Non-inductive current drive is undesirable and superconducting magnets are essential to reduce power consumption.

Using the above guidelines, a 250 MWth reactor was scoped out with a major radius of 9 m, an aspect ratio of 9 and a plasma current of 3 MA, producing 40-50 MW of electricity, about half of which is required to operate the reactor itself. H-mode confinement is assumed and superconducting magnets with a maximum field of 15-16 T are used. About half of the 50 MW of alpha power is re-radiated by electrons, while the other half is transferred by the plasma to the walls. Only 5 kW per linear centimeter of the plasma column is transported to the walls, allowing use of a circular plasma cross section with a pump limiter instead of a divertor. It is sufficient to have 5-10 MW of auxiliary power for 15-20 s to reach ignition. The burn duration is about 1000 s using inductive current drive. Owing to the high aspect ratio and high poloidal beta, a significant part of the current can be produced by the bootstrap effect. Tritium consumption is estimated at less than 4 kg/y, so that there is no need for a tritium breeding blanket.

3. OPTIMIZATION FOR MINIMUM CAPITAL COST

If the requirement for net electricity production is dropped, smaller (15-50 MWth), less costly pilot plants become possible. In this case, the first priority is given to the continuous production of high-grade heat. Superconducting magnets are no longer required, as the plant is permitted to be a net consumer of electricity. Plasma ignition is also not required.

The TETRA systems code was used to explore a range of concepts, seeking the minimum-cost steady-state tokamak that achieves a prescribed wall load and satisfies the ITER constraints on H-mode confinement and beta limits [3]. The results are given in Fig. 1, showing the minimum cost trend and associated major radius versus the average neutron wall load for both the copper and the superconducting coil systems. Neutral beam injection, assumed for maintaining a steady-state plasma current, produces a significant amount of beam-target fusion. At all wall loads, the copper coil concepts result in lower cost, smaller reactors relative to the superconducting coil concepts. This is primarily due to the lesser shield distance between the copper coil and the plasma (30 cm) than in the superconducting case (77 cm). Because of the smaller size, up to 80% of the fusion power is produced by beam-target fusions for copper coil systems versus only about 10% for the larger superconducting coil systems. For wall loads near 0.5 MW/m², the copper coil systems with a major radius of about 2 m are estimated at about $1B in direct capital cost. The copper coil systems have lower plasma current, fusion power and injection power relative to the superconducting systems, all of which help reduce cost. However, the copper coils have large resistive power losses. Enhancement over H-mode confinement does not significantly reduce capital cost in either case, but beta enhancement can significantly reduce cost in the superconducting systems. The analysis (the upper two curves in Fig. 1) looked at the conventional range of aspect ratios (2.5 - 5), optimizing at around 4-5.
One particular embodiment has a major radius of 2 m and an aspect ratio of 4, with 2.62 MA plasma current (bootstrap fraction 32%), and a 4 T field on axis. The magnets are made of hard copper with demountable sliding joints and are designed for steady state operation with water cooling. The plant requires 15 MW of 500 keV neutral beams, 13 MW of ICRF, and produces an average wall load of 0.13 MW/m² (0.21 MW/m² peak). The power input for the toroidal coils is 125 MW.

The lower curve in Fig. 1(b) shows the effect of small aspect ratio (ST), smaller major radius, concepts [4]. Preliminary analysis, assuming physics consistent with initial results from the START experiments [5], suggests a system with a major radius of about 0.9 m and an aspect ratio of 1.6 - 2, with wall loads of 0.25 - 1 MW/m². The small size is made possible by eliminating shielding and insulation of the center leg of the copper toroidal field coils. This is expected to lead to large reductions in the fusion core cost, but the reduction of the balance of plant cost is expected to be less significant.

4. VARIATIONS

Consideration has been given to using ICRF rather than neutral beams for heating and current drive in order to increase the "reactor relevance" of the pilot plant technology. The main consequence to the plasma is the loss of beam-plasma fusion. Therefore, for the same device, switching to neutral beams roughly doubles the neutron wall load. For the same wall load, optimizing the device with neutral beams does not significantly affect capital cost, but can reduce the power requirements by perhaps 30%. Assuming fast wave current drive according to ITER correlations (but with a limited database), steady-state operation is achieved.

For neutral beam driven, copper coil systems, the desired wall load is attained primarily via beam-target fusion. This has a strong impact on reducing the required reactor size for a given wall load. If the copper coil shielding could be reduced, this advantage could be further accentuated.
Although analysis of tritium supply shows that there should be a sufficient stockpile after the year 2000 for the needs of both ITER and a pilot plant for a reasonable number of years, incorporation of a tritium breeding blanket or test module in the pilot plant may be desirable for mission reasons.

5. CONCLUSIONS

Fusion pilot plants could provide significant, relevant experience to utility personnel, and useful data for later demonstration and commercial fusion power plants. Analysis suggests such a plant could be constructed at less expense than the costs of other planned fusion development facilities.

REFERENCES


DISCUSSION

R.D. STAMBAUGH: You have brought to this conference a useful expression of the desires of our ultimate customer, the electric power utilities. The electric power infrastructure in the USA would like a low cost serial No. 1 pilot fusion device to try operating. I wonder whether the Japanese and European electric power infrastructures would also prefer a small pilot plant or whether they are comfortable with a serial No. 1 device as large as ITER?

S.O. DEAN: I am not the proper person to answer that question. You should ask people from Japan and Europe.

A. GIBSON: Your parameter lists give plasma currents of 3 MA. We have 3 MA tokamaks now, and despite our very best efforts they do not ignite. Do you think the utilities would be interested in such a device?

S.O. DEAN: I showed two pilot plant examples. The one that produces net electricity was calculated by B.B. Kadomtsev. Details are given in Ref. [2]. According to his calculation, it ignites at low current (3 MA) because of the assumed scaling law relating plasma current to aspect ratio. His aspect ratio is 9, whereas today’s tokamaks are in the range 3–4. In the ‘high grade heat’ case, the plasma is not ignited;
the tokamak is 'driven' in steady state at Q of about 1 and an aspect ratio of 3–4. In this case the plant is a net consumer of electricity.

The utilities are interested in such devices because they would provide early experience with a steady fusion heat source. This experience would allow them to participate in the design and planning of a full scale demonstration reactor.
REDUCED ACTIVATION STRUCTURAL MATERIALS DEVELOPMENT FOR DEMO FUSION REACTOR APPLICATIONS

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Abstract

REDUCED ACTIVATION STRUCTURAL MATERIALS DEVELOPMENT FOR DEMO FUSION REACTOR APPLICATIONS.

The proper selection and development of materials is a key to the achievement of the ultimate benefits of fusion as a safe, environmentally attractive and economically competitive energy source. The objective of the US fusion materials program is to develop high performance, reduced activation materials that will enhance the overall safety, environmental and economic features of fusion. The current program on the development of structural materials for DEMO first wall/blanket applications is focused on reduced activation compositions of ferritic/martensitic steels, selected vanadium-based alloys, and SiC/SiC composite materials. Significant progress has been made on the development of the ferritic steels and vanadium alloys, and a program to evaluate the critical feasibility issues for the SiC/SiC composites has been initiated.

1. INTRODUCTION

Fusion offers the potential for development of a central station energy source with important safety and environmental features, since the fusion reaction itself does not produce long-lived radioactive nuclides or reactive chemical wastes. As a consequence, the proper selection and development of fusion reactor materials is a key to the achievement of the ultimate benefits
of fusion as a safe, environmentally attractive and economically competitive energy source. The Blanket Comparison and Selection Study [1] and the ESECOM [2] study serve as a basis for the systems analysis and safety/environmental assessment of the structural materials program. The objective of the US fusion materials program is to develop materials that enhance these safety, environmental and economic features of fusion power. The goal of the program is the selection and development of materials that contribute to:

- Improved performance, lifetime and reliability so as to conserve materials, reduce maintenance requirements, and reduce the volume of recycled and/or radioactive waste material.
- Control the radionuclide inventory to simplify waste management and/or recycle.
- Reduced accident risk and consequences by limiting activation, afterheat, and volatility of the structural materials.
- Reduced inventory and release rates of tritium so as to decrease the consequences of an accident.

The current program on the development of structural materials for DEMO first wall/blanket applications is focused on reduced activation compositions of ferritic/martensitic steels, selected vanadium-based alloys, and SiC/SiC composite materials. The development of reduced activation manganese-stabilized austenitic steels has been terminated because of safety related problems associated with afterheat and manganese volatility, and limited heat load and compatibility performance. Significant progress has been made on the development of the ferritic steels and vanadium alloys, and a program to evaluate the critical feasibility issues for the SiC/SiC composites has been initiated.

2. STRUCTURAL MATERIALS REQUIREMENTS

The structural materials must be readily fabricable to produce the complex configurations typical of fusion reactor first wall blanket systems. Also, the material must maintain its structural integrity at elevated temperatures for acceptable lifetimes while exposed to high neutron fluxes, high primary and secondary (thermal) stresses, transient electromagnetic loads, a high temperature coolant and tritium breeding material, and hydrogen environments associated with the plasma. In addition, the structural material should contain only elements that exhibit low activation and decay afterheat as well as acceptable waste management and/or material recycle characteristics. It must also be possible to develop an adequate materials data base and design methods/codes in a time frame required for DEMO.
3. ACTIVATION CHARACTERISTICS OF CANDIDATE MATERIALS

The effects of radioactive products generated by the high neutron fluxes in the first wall/blanket region depend to a large extent on the half-life of the radioactive species and the type of radioactivity, e.g., $\beta$ or $\gamma$. The short half-life products typically contribute to afterheat problems and dispersible radioactivity in the event of an accident. Intermediate half-life products affect maintenance access and dispersible radioactivity, while the long half-life products affect primarily the waste management and/or material recycle considerations. Other properties of the materials such as volatility and

\[ \text{FIG. 1. Induced radioactivity after shutdown for selected elements exposed to first wall fluence of 12.5 MW/m}^2. \]
chemical reactivity are also important in the overall assessment of the safety and environmental characteristics of the various materials. Several elements typically present in structural materials are severely restricted since they produce very long half-life products after exposure to fusion neutrons. These include Cu, Mo, N, Nb, and Ni. Elements which exhibit low long term activation include C, Si, Ti, V, Cr, Fe, Ta and W. Only a limited number of structural materials appear to provide favorable performance characteristics while also exhibiting low activation properties. Figure 1 shows the induced activation as a function of time after shutdown for selected elements of interest after exposure to a typical first wall neutron spectrum [3]. Silicon and vanadium exhibit the lowest radioactivity for periods up to two days, while silicon and aluminum exhibit the lowest radioactivity for periods of two days to about ten years. Vanadium and chromium exhibit the lowest radioactivity for periods greater than 20 years. For the three classes of materials under consideration, these results indicate that the SiC/SiC composites will produce the lowest radioactivity for times up to about 20 years and the vanadium alloys will contribute the lowest radioactivity for times greater than 20 years. In all cases, trace amounts of impurities typically found in the candidate materials can contribute significantly to the radioactivity, particularly at longer times. However, the degree to which trace impurities contribute to the radioactivity can be controlled to a large extent by purification, albeit at increased cost.

4.REDUCED ACTIVATION FERRITIC/MARTENSITIC STEELS

Development of reduced activation ferritic/martensitic steels for fusion applications was initiated in the US in 1983. These alloys are based on the conventional 9-12% Cr steels and the 2-3% Cr steels which have been used in the fission reactor applications. The reduced activation compositions under development are obtained by removal of the nickel and substitution for the molybdenum and niobium by tungsten, vanadium and tantalum. Alloys based on Fe-Cr-2W-0.25V-Ta are currently considered as the leading candidates. Key features of the ferritic/martensitic steels include:
- Established technology of the conventional steels
- Radiation swelling resistance demonstrated in the fission reactor program
- Higher thermal stress figure of merit than austenitic steel
- Better liquid metal corrosion resistance than austenitic steels.

The major feasibility issue for this alloy system relates to an increase in the ductile-brittle transition temperature (DBTT) above room temperature and loss of fracture toughness during irradiation. Critical design constraints relate to interactions of the ferromagnetic structure with the magnetic field,
FIG. 2. Effect of irradiation on the DBTT for a 9Cr martensitic steel.

FIG. 3. Effect of fast reactor irradiation on the DBTT of a reduced activation 9Cr martensitic steel.
the need for post-weld heat treatment to retain desirable mechanical properties, and an operating temperature limit of about 500-550°C.

The current development program is focused primarily on the effect of irradiation on the DBTT and the fracture toughness. Figure 2 shows a significant effect of irradiation on the DBTT of a conventional 9Cr-1MoVNb steel that varies with the irradiation environment [4]. Limited results to date shown in Fig. 3 indicate that a reduced activation 9Cr-2WVTa martensitic steel exhibits a much smaller shift in the DBTT after a low fluence irradiation in the FFTF reactor [5]. Additional studies have shown that the unirradiated mechanical and physical properties and heat treatment requirements of the reduced activation compositions are essentially identical to those of the commercial 9Cr and 12Cr steels [6]. The phase stability, swelling resistance and tensile properties of irradiated materials and the liquid metal corrosion characteristics of the reduced activation alloys are also similar to the conventional steels.

Further work is required to investigate the effects of nuclear transmutations, viz., helium, on the swelling and mechanical properties of irradiated material. Other high priority work includes further evaluation of fracture and creep properties of base metal and weldments, further evaluation of corrosion/compatibility constraints, and development of fabrication/weld procedures.

5. VANADIUM BASE ALLOYS

Vanadium-base alloys with chromium and titanium additions offer several advantages as a structural material [7]. The current program is focused on alloys with (0-10%)Cr and (3-5%)Ti with minor additions of Si. A composition of V-5Cr-5Ti is currently identified as the leading candidate [8]. As indicated in Fig. 1, this class of alloys exhibit low long-term activation. Advantages of this alloy system include:

- High operating temperature capability (to about 700°C)
- High thermal stress figure of merit provides high surface heat load capability
- Good compatibility with high purity lithium and PbLi alloy
- Lower nuclear heating and lower He transmutation rate than for steels.

The current development program is focused on:

- development of weld procedures and characterization of weldments
- investigation of impurity control and impurity effects
- corrosion/compatibility including coolant/breeder, hydrogen interactions, and atmospheric oxidation
- irradiation effects on swelling, tensile properties, and fracture properties.
The key feasibility issue for this alloy system, as for all candidate alloys, relates to loss of fracture toughness during irradiation. The key design constraints relate to its chemical reactivity with oxygen and other nonmetallic elements.

Recent results indicate that certain vanadium alloys retain good fracture toughness after fission reactor irradiation (Fig. 4). The ductile-brittle transition temperatures for the leading candidate alloys are much below room temperature in the unirradiated state and limited data indicate a minimum in the irradiated condition [9]. Results of tests to about 100 dpa indicate that neutron induced swelling remains low. A simulation of the higher helium transmutation rates characteristic of the fusion neutron environment has been developed and conducted; however, the results of this experiment are not yet available. Preliminary experiments on weldments indicate that the mechanical properties are not significantly affected. Experiments have also shown that hydrogen redistribution in a vanadium-lithium system will not result in hydrogen embrittlement or unacceptable tritium inventory in the structure.
Further work is required to investigate the effects of nuclear transmutations, viz., helium, on the properties of irradiated material. Other high priority work includes further evaluation of fracture and creep properties, effects of nonmetallic elements on properties, and corrosion/compatibility constraints including development of protective coatings, as well as development of fabrication/weld procedures.

6. SiC/SiC COMPOSITES

The SiC/SiC composites have been identified in the ESECOM and ARIES [10] studies as attractive structural materials on the basis of safety and environmental considerations. The low activation characteristics of this material are illustrated in Fig. 1. Both silicon and carbon exhibit very low radioactivity immediately after exposure. $^{26}{\text{Al}}$ and $^{14}{\text{C}}$, both with very long half-lives, contribute to low levels of long term radioactivity. The fiber composite structure has significant advantages over a monolithic SiC structure with respect to fracture toughness, which is characteristically low for ceramics. Key features of the SiC/SiC composites for the fusion application in addition to the low activation characteristics include:

- high temperature capability to $\sim 1000^\circ\text{C}$
- high thermal stress figure of merit
- low nuclear heating.

Key feasibility issues identified for this material include [11]:

- irradiation effects on physical and mechanical properties
- fabrication and joining
- hermeticity in the complex geometries of a first wall/blanket
- compatibility with tritium breeding materials.

Important design constraints include development of design methods and codes for this type of structural material and projected costs for advanced composite materials. Also, of particular concern is the very high helium transmutation rate (about a factor of 20 higher than those of the candidate metal alloys) in the fusion neutron environment.

A program plan has been developed and research on these materials has been initiated. Initial experiments have been conducted primarily on available composite materials fabricated with a Nicalon fiber. These materials exhibit significant reductions in the thermal conductivities and degradation of the mechanical properties after modest fluences corresponding to about 1 MW y/m². These first results indicate that tailoring of these materials for the fusion environment will require improved fibers and interfaces. Advanced methods of fabrication may also be required.
7. SUMMARY AND FUTURE DEVELOPMENT REQUIREMENTS

Reduced activation structural materials that will enhance the attractiveness of the fusion power option can be developed for a DEMO to operate by 2025. However, an expanded and aggressive program will be required to meet this schedule. In particular, a high flux, 14-MeV neutron source for materials testing will be required by about year 2000. Although existing fission reactor irradiation facilities provide a basis for development of improved materials, materials for the high fluence requirements of a DEMO cannot be qualified without data with a fusion relevant neutron spectrum. Continued development activities on ferritic/martensitic steels and an expanded vanadium alloy development program are required to identify prime candidate alloys for DEMO by year 2000. An expanded effort on SiC/SiC composites is required to assess the feasibility of these materials by 1997.

REFERENCES


DISCUSSION

C.W. BARNES: You did not mention cost as a driver for choice of materials to be studied. Given the existence of plans to use reduced activation materials in experimental devices in the next decade, can you comment on the relative expense of your proposed structural materials?

D.L. SMITH: The cost of reduced activation ferritic/martensitic steels will be similar to that of austenitic steels, with some additional cost for purification. Vanadium alloys will be more expensive, but only modest performance benefits of a DEMO are required to offset the additional costs. The costs for the SiC–SiC composites will depend strongly on the fabrication procedure and are not known at this time. Cost has been identified as one of the critical issues.
R.L. McCROY: How would your favourable results for the vanadium alloys apply to inertial fusion energy applications, where the peak fluxes (pulsed nature of ICF) are very different from those of the magnetic fusion energy regime?

D.L. SMITH: The radiation damage effects for pulsed ICF conditions are believed to be quite different from those characteristic of magnetic fusion. However, I would expect the trends in performance for many of the issues to be similar.
HIGH POWER NEGATIVE ION BEAM DEVELOPMENT FOR HEATING AND CURRENT DRIVE IN FUSION PLASMAS

Part A: Development of high power negative ion sources at JAERI
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Part B: Joint CEA–JAERI experiment with a 2 A, 100 keV long pulse D− neutral beam injector
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Abstract

HIGH POWER NEGATIVE ION BEAM DEVELOPMENT FOR HEATING AND CURRENT DRIVE IN FUSION PLASMAS. Part A: DEVELOPMENT OF HIGH POWER NEGATIVE ION SOURCES AT JAERI; Part B: JOINT CEA–JAERI EXPERIMENT WITH A 2 A, 100 keV LONG PULSE D− NEUTRAL BEAM INJECTOR.

Part A. Negative ion source technologies for high energy and high power neutral beam injectors have made great progress during the last few years. After the development of a large negative ion source producing an H− beam of 50 keV, 10 A (with a current density of 37 mA/cm²) during 0.1 s, continuous production of a negative ion beam in a small scale H− source was demonstrated for a pulse length of 24 h at a beam of 50 keV and 0.3 A. Furthermore, the H− beam was accelerated up to a high energy of 300 keV, at 0.1 A and 0.5 s, in a caesium seeded ion source with a multi-aperture, two-stage accelerator. The beamlet divergence was measured with a single-aperture accelerator and found to be 5.5 mrad at 300 keV, 17 mA (11 mA/cm²). The current, current density, pulse length and beam divergence are close to the specifications required for the neutral beam injectors for JT-60U and ITER.

Part B. Intense D− ion beams of 2.2 A, transmitted on a calorimetric target at 6.4 m from the source and with energies up to 100 keV in multisecond pulses, have been obtained at Cadarache in a joint Euratom–CEA–JAERI experiment. The plasma generator was operated with caesium. At the exit of the beam neutralizing gas cell, the neutral, positive and negative beam components have been separated electrostatically and measured calorimetrically for different gas target thicknesses. Energy recovery experiments have been conducted, for the first time with negative ions. The residual negative ion power has been recovered with an efficiency as good as 81% with 102 keV, 1.2 A D− beams.

1. INTRODUCTION

High power negative ion sources are required to realize a high power neutral beam injector (NBI) which is one of the most promising heating and current drive systems for fusion plasmas. The beam energy, beam current and pulse length of negative ion sources required for JT-60U [1] and ITER [2] neutral beam injectors (NBIs) are 0.5 MeV, 22 A, 10 s and 1.3 MeV, 15 A, two weeks, respectively. There are three important areas in which development is crucial, i.e. increasing negative ion current, pulse length and beam energy.

As to the negative ion current, we have already developed a 10 A, 50 keV H⁻ ion source [3]. The source is of the volume production type [4] with caesium seeding [5, 6]. The extracted current density of the H⁻ beam was 37 mA/cm². Taking into account the reduction of current in deuterium operation [7], we can expect a D⁻ current density of 30 mA/cm², which satisfies the design values of the ion sources for JT-60U (15 mA/cm²) and ITER (18 mA/cm² in the Japanese design).

The pulse length of the 10 A ion source was 0.1 s, because of the limitation of power supply. In order to demonstrate long pulse operation, we have designed a small scale H⁻ ion source and tested for a long pulse length of up to 24 hours [8].

High energy acceleration is the most important subject in the realization of high power NBIs. A high energy H⁻ ion source with a multi-aperture, two-stage electrostatic accelerator was manufactured and tested up to an energy of 300 keV.

In this paper, we present the major experimental results obtained from these ion sources.

2. LONG PULSE H⁻ BEAM PRODUCTION

For long pulse operation, it is necessary to reduce the heat loading of the extraction grid caused by electrons extracted with the negative ions. By seeding a small amount of caesium in the plasma generator and biasing the plasma grid positive against the anode, the electron current and the heat loading were decreased drastically. Further, the temperature of the plasma grid was kept around 200°C by making hot water (90–100°C) flow into the cooling pipes during a pulse in order to maintain the caesium effect.

After this optimization, the ion source was operated continuously for 24 h with an H⁻ ion beam of 50 keV and 0.3 A. The source was also operated at 50 keV and 0.5 A for 1000 s with a current density of 14 mA/cm², which is close to the design value of the JT-60U ion source. Figure 1 shows the waveforms of acceleration voltage and current. The acceleration current was almost constant during the pulse. This
shows that the H⁻ current was kept constant for 1000 s. The maximum heat loading in the extraction grid was about 100 W/cm² at a current density of 14 mA/cm². Since the grid is designed for heat loading as high as 500 W/cm², continuous operation at the higher current densities required for ITER will be possible.

3. HIGH ENERGY ACCELERATION OF THE H⁻ BEAM

In order to study beam acceleration, we have designed and manufactured a high energy ion source with a two-stage, multi-aperture accelerator (Fig. 2). A small amount of caesium was seeded to increase the negative ion yield at low operating
A stable H\(^-\) beam of 300 keV, 100 mA and 0.5 s was obtained with nine apertures of 14 mm diameter each, at a source pressure of 0.4 Pa. When the perveance and the electrostatic lenses were optimized, the beam was highly converged: the beam diameter measured 1.6 m from the ion source had almost the same size (about 55 mm) as the extraction area. The beam divergence obtained in a single aperture was as low as 5.5 mrad at a beam of 300 keV and 17 mA/cm\(^2\) (11 mA/cm\(^2\)) [9]. These beam divergence and current density values are close to the design values of the NBIs for JT-60U and ITER.

The reason why the negative ion beam divergence is so small is attributed to the low thermal velocity of the negative ions produced. Provided that the negative ion temperature is constant, the beam divergence tends to be lower as the beam energy increases. Therefore, a lower beam divergence of less than 5 mrad can be achieved at higher energies.

The heat loading of the accelerator grids was mainly due to the impingement of electrons stripped from the H\(^-\) ions and direct interception of the beam. The heat loading caused by the stray electrons from the extraction grid was negligible. The heat loading of the accelerator grids was lowered to less than 160 W/cm\(^2\) by reducing the operating pressure.

We did not observe any degradation of the voltage prevailing in the caesium seeded operation.

**FIG. 3.** History and future plan of negative ion beam development. The negative ion beam power is approaching the target regions of the negative ion based NBIs for the future fusion devices.
4. FUTURE PLAN

On the basis of the successful experimental results mentioned in this paper, the construction of the JT-60U NBI will be started in 1992. The target value of the NBI for JT-60U is shown in Fig. 3. The construction of the JT-60U NBI will constitute an important step in the development of the ITER NBI system.

ACKNOWLEDGEMENTS

The authors would like to thank the other members of the JAERI NBI group for valuable discussions and comments. They also express their gratitude to Drs S. Kunieda, S. Shimamoto and N. Shikazono for their support and encouragement.

REFERENCES


PART B: JOINT CEA–JAERI EXPERIMENT WITH A 2 A, 100 keV, LONG PULSE D- NEUTRAL BEAM INJECTOR (J. Pamela, M. Fumelli, F. Jequier, M. Hanada, Y. Okumura, K. Watanabe)

1. INTRODUCTION

A joint experiment of Euratom–CEA Association and JAERI was conducted from April 1991 to April 1992 at Cadarache, France. The experimental results are briefly presented here. More details can be found elsewhere [1–4].
2. EXPERIMENTAL SET-UP

The negative ion source was built by JAERI, other elements by CEA. The multi-aperture, 100 keV electrostatic accelerator comprises three grids, with 240 holes of 1 cm² area each. The first electrode is grounded. The beamline comprises a gas neutralizer, a differential pumping chamber and an electrostatic deflector. The deflector allows the separation of the three beam components of D⁺, D⁻ and D⁰ at the neutralizer exit. These components are dumped and measured calorimetrically on three targets (4 to 6.4 m from the source). The negative ion target can be polarized independently in order to decelerate the unneutralized beam ions for energy recovery, following an original design by CEA [5].

3. 100 keV, 2.2 A, LONG PULSE D⁻ BEAMS

With caesium seeding in the plasma generator, D⁻ accelerated currents of 2.2 A with energies of up to 100 keV were obtained in 5 s pulses. This was measured on the calorimetric targets, with a severely limited acceptance of 40 mrad in width. The corresponding j(D⁻) value of 10 mA/cm² is very close to the requirements of the NB projects for ITER. A neutron detector was used to confirm the acceleration of deuterium negative ions as is shown in Fig. 4. Caesium seeding allowed significant reduction of operating pressure and extracted electron current and prevented saturation of the D⁻ production with arc power [1, 2].

![Neutron signal plotted versus D⁻ beam current deduced from calorimetry at 100 keV and 65 keV; P source = 0.4 Pa.](image)

**FIG. 4.** Neutron signal plotted versus D⁻ beam current deduced from calorimetry at 100 keV and 65 keV; P source = 0.4 Pa.
4. BEAM CHARACTERISTICS

The beam optics was optimized by varying the extraction and acceleration voltages. The optimum could be found by various methods (secondary emission probes, beam transmission on targets, neutron signal). It was observed that the negative ion accelerator has a well defined perveance, while the current at optimum was following the $V^{3/2}$ and $M^{-1/2}$ dependences predicted by the Child–Langmuir law.

The beam profile was also measured by using an array of secondary emission probes. It shows that the source homogeneity depends strongly on the operating pressure, with a degradation of more than 0.3 Pa.

The acceleration of stray electrons increases almost linearly with the extraction voltage above a typical threshold of $V_{\text{ext}} = 2.5 \text{ kV}$; at $V_{\text{ext}} = 5 \text{ kV}$, 15% of the extracted electrons leak from the extraction grid, in spite of the trapping magnetic field. A Monte Carlo method is being developed at Cadarache to provide understanding of this phenomenon.

5. MEASUREMENT OF NEUTRAL AND CHARGED BEAM FRACTIONS AT THE GAS NEUTRALIZER EXIT

The neutral, positive and negative beam components have been measured as a function of the neutralizer gas target thickness. It was for the first time that such measurements were performed with high power, long pulse D$^-$ and H$^-$ beams. The

![Graph showing the neutralization experiment with 100 keV D. Symbols correspond to measurements and lines to calculations.](image)
beam energy was varied from 25 to 100 keV per nucleon. The results (see, e.g. Fig. 5) are in agreement with the theoretical predictions derived from cross-section data [6] and small scale experiments [7]. They confirm that the neutralization efficiency of the negative ions is about 60% at 100 keV per nucleon.

6. ENERGY RECOVERY EXPERIMENTS

A new type of energy recovery system (ERS) had been conceived by the CEA [5]. The unneutralized negative ions are decelerated on a collector, which can be biased at a low voltage, close to the grounded source potential. This collector is designed to prevent secondary emission of electrons by electrostatic trapping; this is fundamental for an efficient energy recovery. This system was very successful: with 102 keV, 1.2 A D⁻ beams, the residual negative ions have been decelerated down to 8 keV (Fig. 6). The power of the residual negative ion beam was recovered with an efficiency as good as 81%. Extrapolated to 1 MeV, the ERS would yield a reduction of the high voltage drain current by 20%; the measured decelerating factor of 0.08 would facilitate residual ion dumping.

ACKNOWLEDGEMENTS

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REFERENCES


DISCUSSION

E. THOMPSON: In view of a possible use of negative ion NBI for heating to ignition without current drive (and therefore energies < 1 MeV), what is the maximum achievable value of negative ion current density in deuterium?

K. WATANABE: We obtained a negative ion current density of nearly 40 mA/cm² in hydrogen. From the isotopic effect experiment we obtained a ratio of I_D⁻/I_H⁻ = 80%. I think, therefore, that it is possible to obtain about 30 mA/cm² in deuterium from the present ion source. By optimizing parameters, I would suppose that an even higher current density can be achieved.
DESIGN OF INERTIAL CONFINEMENT FUSION REACTOR DRIVEN BY LASER DIODE PUMPED SOLID STATE LASER

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Abstract

DESIGN OF INERTIAL CONFINEMENT FUSION REACTOR DRIVEN BY LASER DIODE PUMPED SOLID STATE LASER.

Recently, high density compression by laser implosion has been achieved at ILE, Osaka University. Through the experiment, the requirements on the laser irradiation uniformity for direct drive implosion and pellet quality became clear. According to the database obtained by the experiments, a laser fusion reactor for direct drive has been designed. The radial convergence ratio of 30 which was achieved in the above direct implosion experiments implies that the average density of an imploded pellet could be higher than 2000 times solid density. This means that a pellet gain higher than 150 can be achieved
with a multimegajoule laser pulse. Since the preliminary reactor design was presented at the 1990 IAEA Conference, Washington, the design of reactor pellet, reactor chamber (in particular, the first wall structure), laser diode (LD) pumped solid state laser, etc. have been refined. The paper presents a conceptual design of the fusion reactor KOYOO-I driven by an LD pumped solid state laser operated at 12 Hz. The driver energy is assumed to be 4 MJ which yields a gain of 150 according to the design. The net output electric power is 2.6 GW. The first wall is made of liquid lithium–lead flows. The flows are guided by woven ceramic fibre pipes protecting the structure wall as well as breeding tritium.

1. INTRODUCTION

Inertial confinement fusion (ICF) research has recently made substantial progress as, typically, attested to by the laser implosion experiments. High density compression has been accomplished by direct drive at ILE, Osaka University [1], and at LLE, University of Rochester, and by indirect drive at LLNL. With the help of the database for laser illumination and pellet uniformity conditions obtained from the above mentioned experiments, the direct drive ICF reactor KOYOO-I has been designed. For reactor pellet implosion we have assumed a radial convergence ratio of 30 which was achieved in the recent experiments. This assumption implies that 4 MJ laser pulse will compress the pellet up to 2000 times the solid density to obtain a pellet gain of \( Q \geq 150 \).

In this paper, we present the following items:

1. Prediction of the gain by accounting for the implosion stability.
2. Protection of the structural wall by using liquid Pb\(_{83}\)Li\(_{17}\) guided by pipes woven with SiC or Al\(_2\)O\(_3\) fibres.

2. HIGH GAIN TARGET DESIGN USING TAILORED PULSE

To realize a central spark as well as low isentrope main fuel, we have used a target consisting of a frozen D–T layer contained in a spherical plastic shell. The thickness of the plastic shell is chosen so that almost all plastics are ablated away during the acceleration phase.

After some parametric runs of the implosion code ILESTA, we have found that a laser energy higher than 1 MJ is necessary to implode high gain pellet targets. As an example, we show the result of a simulation of a 3.95 MJ, 0.25 \( \mu \)m wavelength laser implosion. The pellet parameters are shown in Fig. 1. The initial aspect ratio of the target is 11.

The intensity of the prepulse is kept at \( 10^{13} \text{ W/cm}^2 \) on the target surface with a pulse duration of about 20 ns (the laser input power is about 10 TW) while the peak power of the main pulse is about 50 times that of the prepulse. The pulse duration is adjusted to the implosion time which is about 20 ns after the first shock has passed...
FIG. 1. Pellet structure for 3.95 MJ high gain implosion.

FIG. 2. Hydrodynamics and burning dynamics of 4 MJ implosion at maximum compression: (a) density contours; (b) ion temperature contours; (c) flow diagram.

through the shell. Figure 2 shows the space–time contours of the density (a) and ion temperature (b) and the trajectories of the plasma flow (c). The figures show that the hot spark density, temperature and radius are 70 g/cm$^3$, 8 keV and 0.4 g/cm$^2$, respectively, at the time (39.4 ns) when the fuel is ignited. The density and the areal density of the main fuel at this time reach 200–400 g/cm$^3$ and ~3 g/cm$^2$, respectively. The gain of this implosion is about 160. During the burn, the plasma temperature increases up to 100 keV. This superhot, dense plasma is confined for about 20 ps. From this burning plasma, intense hard X rays in the range of 100 keV–1 MeV are generated and will heat and ablate the reactor chamber surface.

The feasibility of the high gain target design has been checked by estimating the e-folding times of the non-uniformity growth which depend upon the shell acceleration time history and the density and temperature profiles of the accelerated shell. An analytical model is used with the 1-D simulation code ILESTA in order to evaluate
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FIG. 3. Rayleigh–Taylor instability amplification factors for various gain and input laser energies.

the e-folding. In Fig. 3, the e-folding with respect to pellet gain for various laser input energies is plotted. The above given example (gain of 160 and 4 MJ laser input) is shown by the mark on the right hand and in the top corner of Fig. 3. Since the e-folding ($\int \gamma dt$) is approximately 7, the required laser irradiation uniformity will be very high (the tolerable non-uniformity may be less than 1%). A little more conservative target design for the 4 MJ laser input gives an approximate pellet gain of 100, as is shown in Fig. 3.

3. REACTOR CHAMBER PROTECTED BY LIQUID Pb$_{83}$Li$_{17$ FIRST WALL

We present the conceptual design of a laser fusion reactor for direct irradiation implosion. The reactor is driven by a laser diode (LD) pumped solid state laser. Alpha particle heating leads to a maximum gain of 150 or 600 MJ fusion output/shot at a driver energy of 4 MJ. The reactor parameters for one chamber module are listed in Table I.

Since 20% of the neutron yield is deposited in target plasma, the chamber wall is exposed to the resulting 210 MJ hard X ray and charged particle pulses. A first wall of liquid lithium–lead protects the chamber wall and breeds tritium. The system will be operated at a repetition rate of 12 Hz, so high as to reduce the effective cost of the LD pumped solid state laser. Since the reactor chamber requires at least 200 ms to recover a vacuum high enough for the next shot, we propose a four module chamber, one laser system and 3 Hz operation of each module, as is shown in Fig. 4. We may use Pockels cells to distribute the laser beams to the four module chambers.
TABLE I. LASER FUSION REACTOR: PARAMETERS FOR ONE MODULE

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fusion output/shot</td>
<td>600 MJ</td>
</tr>
<tr>
<td>Neutron yields/shot</td>
<td>480 MJ</td>
</tr>
<tr>
<td>Repetition</td>
<td>3 Hz</td>
</tr>
<tr>
<td>Neutron output/s</td>
<td>1440 MW</td>
</tr>
<tr>
<td>D-T neutron yield rate</td>
<td>$6.37 \times 10^{20}$/s</td>
</tr>
</tbody>
</table>

FIG. 4. Laser fusion reactor system. P.C: Pockels cells; P: polarizer; M: mirror. Dimensions are given in millimetres.

In the present design, we have compared Pb$_{83}$Li$_{17}$, FLiBe and Li$_2$ as liquid flow materials. Since Pb$_{83}$Li$_{17}$ makes tritium breeding and separation easier than the other materials, we chose Pb$_{83}$Li$_{17}$. Furthermore, since Pb$_{83}$Li$_{17}$ has high opacity for hard X rays, it makes the X ray ablation layer thickness smaller than the other materials.

SiC, Al$_2$O$_3$ and carbon fibre are the candidates for the fibre material of the woven guide pipe for the liquid metal. At the moment, we are going to use SiC as in the HIBALL chamber [2]. A preliminary chamber concept is shown in Fig. 5.
Since the neutron yield rate is $6 \times 10^{20}/s$ or 1.44 GW, we might replace the first wall SiC or the aluminium pipes probably every two years. We have designed the reactor head (the ceiling) so that the pipes can be removed easily.

The parameters such as the chamber evacuation speed and neutron damage on the first wall will determine the chamber size. In this conceptual design, the inner diameter is 8 m, and 32 laser beam ports are directed uniformly to the centre of the chamber. The reflector is made of graphite, and the structural wall of 10 m diameter is made of 5 cm thick steel.

The 1-D spherical geometry code ANISN is used to analyse the neutronics of the chamber. We have obtained a tritium breeding ratio of 1.10 for natural lithium and a total thermal output of 1447 MW. The biological dose rate during the operation is 1 mrem/h ($10^{-5}$ Sv/h) outside the concrete shield of the chamber.

4. CONCEPTUAL DESIGN OF A LASER DIODE PUMPED SOLID STATE LASER SYSTEM FOR THE LASER FUSION REACTOR DRIVER

For a laser fusion reactor driver, we have designed an LD pumped solid state laser just by adopting the currently existing high power solid state laser technology. By using laser diodes for pumping, we can increase the laser efficiency up to 12%.
Even so, however, simple amplifier chains are not enough to obtain the few MJ output required so that the LD pumped solid state regenerative amplifier is specified with an output of 4 MJ at 0.35 µm, a system efficiency of 12% and a repetition rate of 12 Hz.

4.1. Main regenerative amplifier pumped by laser diode arrays

To suppress laser damage and parasitic oscillations, we consider a regenerative amplifier, composed of segmented disks, as the main amplifier. The segment number is chosen large enough to generate 4 MJ blue output [3-5].

Figure 6 shows the diagram of a 4 MJ blue solid state laser system. The regenerative amplifiers are composed of segmented disk amplifiers (SAs) pumped by...
laser diode arrays. The shaped laser pulses from the master oscillator and through the preamplifier, also pumped by laser diode arrays, are injected into the laser diode pumped SAs. After multitime reflections in the SAs, the amplified laser outputs are switched out and then injected into the non-linear crystals to generate blue light. The blue beams are sent to the target chamber to illuminate a cryogenic D–T fuel pellet.

As is shown in Fig. 6, in one SA rectangular disk type solid state laser material is set at the Brewster angle with respect to the amplifying laser beam [4]. The disks are pumped by two-dimensional laser diode arrays and cooled by helium flow from both sides.

4.2. Design of 4 MJ blue laser for fusion reactor driver

In this section, we design LD pumped regenerative amplifier with 4 MJ output at a wavelength of 0.35 μm, 12% overall efficiency and 12 Hz repetition frequency. In designing the single SA, we have considered the following items:

(1) Laser damage on the amplifier optical elements restricts the maximum fluence of the laser beam to 50 J/cm² at a pulse width of 10 ns.

(2) Although an extraction efficiency higher than 60% requires the initial small signal gain of the disk to be larger than 4, parasitic oscillations in the disk finally restrict the product of the largest dimension and the surface small signal gain coefficient to be less than 4.

(3) To achieve a pump absorption efficiency higher than 95%, the optical depth of the disk is chosen to be 3. A larger optical depth induces more parasitic oscillations at the disk surface.

(4) The total heat dissipation in the disk must be lower than 10% of the thermal fracture limit. The disks are cooled by helium gas (273K, 1 atm (1.013 × 10⁵ Pa), 0.15 Mach). The maximum temperature in the disk is limited to below 30% of the transition point (for glass) or melting point (for crystal) and the maximum temperature difference must be lower than 10% [6]. The calculation includes the effect of the pumping energy density distribution in the disk to resonant pumping [4].

(5) The pump pulse, for example, a rectangular pulse width of LD arrays, is selected to be half the fluorescence lifetime of the laser material so as to achieve an energy storage efficiency higher than 70%.

(6) The total B integral during the round trips in the SA is restricted to be less than 4.

4.3. Items to be developed

The technical items to be developed can be summarized as follows:

(1) We need laser diode arrays of large area, capable of delivering a maximum intensity of about 3 kW/cm². This intensity has already been realized. Next
issues to be developed are: (1) packing such diode arrays to a size large enough to pump the disk and (2) cooling such large diode arrays efficiently to guarantee a radiation efficiency of $\geq 60\%$ and a long lifetime of over $10^{11}$ shots [7]. A surface emitting diode laser is one of the candidates for this role.

(2) To realize a repetition rate of 10 Hz, the technology of disk amplifier gas cooling should be developed [8].

(3) To achieve the system schematically shown in Figs 4 and 6, we should develop large active optical elements including Pockels cells and optically non-linear crystals. A prototype 27 cm diameter plasma electrode Pockels cell which may be used for this system has already been developed [3]. With eye safe laser materials, we should develop large size non-linear crystals such as a KTP for harmonic generation and new large crystals for Pockels cells, respectively.

(4) We should develop new solid state laser materials, with (a) good thermal properties such as Nd : SiO$_2$, (b) a lasing wavelength of $\leq 1 \mu$m to ensure sufficient system efficiency, (c) a stimulated emission cross-section of $\sim 2 \times 10^{-20}$ cm$^2$ to relax the optical damage problem and (d) a fluorescence lifetime of $\geq 4$ ms to achieve good cost performance. A Nd doped fluoride crystal is one of the attractive candidates for this purpose.

5. SUMMARY

We have presented the conceptual design of laser fusion reactor KOYOO-I for direct irradiation implosion:

(1) On the basis of the recently achieved high density compression, we are convinced that the inclusion of alpha particle heating leads to a maximum gain of 150 at 4 MJ.

(2) A first wall of liquid lithium-lead is guided by woven ceramic pipes, made of SiC or Al$_2$O$_3$, to protect the reactor chamber wall as well as to breed tritium. The plant system consists of four module chambers and one driver. The driver will be operated at a repetition rate of 12 Hz, 3 Hz for each module.

(3) A laser diode pumped multi-MJ blue laser system with a 10% overall efficiency and a 10 Hz repetition rate will be developed by adopting the currently existing technology.

REFERENCES


A. GIBSON: What is the best performance (power, energy, divergence, etc.) achieved so far with a laser diode (LD) pumped solid state laser?

S. NAKAI: It has recently been reported (KASINSKI, J.J., et al., IEEE J. Quant. Electron. 28 (1992) 977) that on an LD pumped Nd:YAG laser with 1.25 J (at 1.06 μm), 40 Hz, 20 ns, 10% wall plug efficiency with good beam quality has been demonstrated. As regards the laser diode, an output power density of 2 kW/cm² has been achieved, and a 350 kW module has been reported.

D.H. CRANDALL: You have chosen four reactors with one drive. Why did you choose four rather than one or ten?

S. NAKAI: The aim is to reduce the cost of the overall plant system. The wetted wall reactor cavity cannot be operated with a repetition frequency greater than 10 Hz. The LD pumped solid state laser, however, can be operated at frequencies above 10 Hz, and a higher repetition rate can yield more energy at a fixed capital cost. An optimization evaluation of the number of cavities and the driver repetition rate results in the combination of four chambers and a 12 Hz driver repetition rate with three shots per second for each reactor cavity.
TRANSMUTATION OF $^{90}\text{Sr}$ USING FUSION-FISSION HYBRID REACTORS

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Abstract

TRANSMUTATION OF $^{90}\text{Sr}$ USING FUSION-FISSION HYBRID REACTORS.

The fusion-fission hybrid reactor to be used to produce nuclear fuel supporting pressurized water reactors (PWRs) in future energy production has been studied in the Chinese national programme for a long time. Recently, a study assessing the technical feasibility of the transmutation of high level waste (HLW) by fusion-fission hybrid reactors has been started. The present paper, the first in a series of papers on transmutation of HLW, presents the conceptual design of a fusion–fission hybrid reactor for the transmutation of fission products, especially $^{90}\text{Sr}$, with a Pu fuel blanket. A neutronics and thermal hydraulics calculation was performed to estimate the performance of the whole system. It was found that a plasma core with the recently achieved experimental plasma physics parameters or with parameters to be reached in the near future can drive a blanket that transmutes 17% of the loaded fission product $^{90}\text{Sr}$ after one year of operation at a neutron wall loading of 1 MW/m$^2$ and an 80% load factor. Thus, the present system can burn $^{90}\text{Sr}$ waste from about 40 PWRs of 3 GW(th).

1. INTRODUCTION

The disposal of radioactive waste generated in reprocessing spent reactor fuel is one of the important problems to be solved in nuclear engineering. It is particularly necessary to establish the technology for the management of high level waste (HLW). A number of actinide nuclides and fission products in HLW have a hazard potential that will remain for millions of years, because of the very long half-lives involved.

One of the most widespread methods of HLW management is the permanent disposal of waste into a stable geological formation to isolate it from human environment. On the other hand, nuclear transmutation is considered another candidate for the management of HLW. In this case, the long lived nuclides separated from the waste are transmuted into short lived or stable ones.

Many studies on the transmutation of HLW using fission reactors, particle accelerators and fusion reactors were performed [1–3]. With a large inventory of HLW, fission reactors have the problem of safety associated with criticality, and it is difficult to obtain an intensive neutron source and a suitable neutron spectrum for transmutation. Spallation by particle accelerators with very high currents has not yet been developed and will probably entail high energy consumption. One of the earliest hopes for the fusion reactor was that it could be used to ‘burn’ fission wastes by neutron transmutation to more benign isotopes, but a pure fusion tokamak reactor cannot
produce a sufficiently high neutron flux in the blanket region for transmutation of fission products (e.g. $^{90}\text{Sr}$, $^{137}\text{Cs}$), except for high neutron wall loading above 10 MW/m$^2$ [4] which is not acceptable for the current level of reactor technology because the capture cross-sections of $^{90}\text{Sr}$ (0.8 b) and, especially, $^{137}\text{Cs}$ (0.11 b) are much smaller. Obtaining a high flux of thermal neutrons in the blanket is the key problem for $^{90}\text{Sr}$ and $^{137}\text{Cs}$ transmutation.

Fusion–fission hybrid reactors offer attractive advantages if fissile plutonium is introduced into the blanket for the multiplication of neutrons. The advantages of the fusion–fission hybrid reactor are as follows:

1. they will produce higher neutron fluxes under lower neutron wall loadings;
2. the blanket will be safe as a kind of subcritical reactor, and the whole system will work as a passive system;
3. they will require lower fusion driver conditions than a pure fusion reactor;
4. the blanket could produce vast quantities of energy.

2. BLANKET CONCEPT

The tokamak blanket consists of two zones: the Pu fuel zone (Pu zone) and the $^{90}\text{Sr}$ transmutation zone (Sr zone). In the Pu zone, the particle fuel is loaded, which

<table>
<thead>
<tr>
<th>TABLE I. BLANKET DESIGN PARAMETERS</th>
</tr>
</thead>
<tbody>
<tr>
<td>Depth of blanket:</td>
</tr>
<tr>
<td>Inner ($^{90}\text{Sr}$)</td>
</tr>
<tr>
<td>Outer (Pu fuel)</td>
</tr>
<tr>
<td>Reflector</td>
</tr>
<tr>
<td>Effective space factor</td>
</tr>
<tr>
<td>Operation load factor</td>
</tr>
<tr>
<td>$^{90}\text{Sr}$ inventory</td>
</tr>
<tr>
<td>$K_{eff}$</td>
</tr>
<tr>
<td>Average power density</td>
</tr>
<tr>
<td>Blanket power</td>
</tr>
<tr>
<td>Total flux in $^{90}\text{Sr}$ region</td>
</tr>
<tr>
<td>$T_{eff}$</td>
</tr>
<tr>
<td>Burnup of $^{90}\text{Sr}$:</td>
</tr>
<tr>
<td>kg/a</td>
</tr>
<tr>
<td>Unit of 1 GW(e) PWR</td>
</tr>
</tbody>
</table>
is suitable for frequent shuffling of fuel and high power density due to extremely
great heat transfer area. Then the fuel is cooled by gaseous He and $^{90}\text{Sr}$ is loaded in
the chemical form of SrO. The 14 MeV D–T fusion neutrons travelling through the
first wall and the fission neutrons from the outer Pu zone will be moderated in the
$\text{D}_2\text{O}$ moderated fission product transmutation zone, and a high thermal neutron flux
is produced for the transmutation of $^{90}\text{Sr}$. The Pu zone is surrounded by a graphite
reflector. The blanket design parameters are summarized in Table I.

The flow chart depicting the computer code system which we developed for the
transmutation analysis is shown in Fig. 1.

In the analysis, various influences of the following elements are considered:

1. Effect of blanket composition and zone on transmutation results: Tables II
   and III.
2. Effect of $^{90}\text{Sr}$ and $^{239}\text{Pu}$ concentrations on transmutation results: Table IV.
3. Effect of neutron wall loading on transmutation results: Table V.

### 3. PLASMA CORE DESIGN

During the last few years, continuous experimental and theoretical effort has
resulted in a substantial progress in tokamak plasma physics [5]. The JET tokamak
has achieved near-breakeven conditions. The TFTR tokamak shows the existence of
a bootstrap current, and the JT-60 tokamak has attained a bootstrap current fraction
of up to 80%. Efficient current drive with the lower hybrid wave has also been dem-

![Diagram](image-url)
<table>
<thead>
<tr>
<th>Case</th>
<th>First wall</th>
<th>Blanket zone composition (vol.% )</th>
<th>Transmission zone</th>
<th>Total flux in TZ (10^5 cm^-2 s^-1)</th>
<th>Power density in TZ (MW m^-3)</th>
<th>T_{eff} (a)</th>
</tr>
</thead>
<tbody>
<tr>
<td>a</td>
<td>C</td>
<td>Zr + D_2O + Pu 0.8166 x 10^-5</td>
<td>23</td>
<td>207</td>
<td>2.23</td>
<td></td>
</tr>
<tr>
<td>b</td>
<td>C</td>
<td>C + He + Pu 0.4504 x 10^-5</td>
<td>30</td>
<td>143</td>
<td>1.80</td>
<td></td>
</tr>
<tr>
<td>c</td>
<td>SiC</td>
<td>C + D_2O + Pu 0.4916 x 10^-5</td>
<td>31</td>
<td>177</td>
<td>1.61</td>
<td></td>
</tr>
<tr>
<td>d</td>
<td>Be</td>
<td>Be + He + Pu 0.9113 x 10^-5</td>
<td>24</td>
<td>226</td>
<td>2.26</td>
<td></td>
</tr>
</tbody>
</table>
TABLE III. EFFECT OF POSITION OF Pu ZONE AND BLANKET MATERIALS ON RESULTS FOR A NON-UNIFORM TRANSMUTATION ZONE (TWO ZONES: Pu ZONE AND Sr ZONE)

<table>
<thead>
<tr>
<th>Case</th>
<th>Material first wall</th>
<th>Material Sr zone</th>
<th>Material Pu zone</th>
<th>Position of Pu zone with respect to Sr zone</th>
<th>Total flux in Sr zone ($10^{15}$ cm$^{-2}$·s$^{-1}$)</th>
<th>$T_{eff%}$ (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>e</td>
<td>C</td>
<td>Zr + D$_2$O</td>
<td>316SS + He + Pu</td>
<td>outer</td>
<td>6.7</td>
<td>7.29</td>
</tr>
<tr>
<td></td>
<td>5% 95%</td>
<td>5% 90% 0.27 $\times 10^{-3}$</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>f</td>
<td>C</td>
<td>Zr + D$_2$O</td>
<td>C + He + Pu</td>
<td>outer</td>
<td>26</td>
<td>1.89</td>
</tr>
<tr>
<td></td>
<td>5% 95%</td>
<td>5% 90% 0.25 $\times 10^{-4}$</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>g</td>
<td>C</td>
<td>Zr + D$_2$O</td>
<td>316SS + He + Pu</td>
<td>inner</td>
<td>9.3</td>
<td>5.69</td>
</tr>
<tr>
<td></td>
<td>5% 95%</td>
<td>5% 90% 0.35 $\times 10^{-3}$</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>h</td>
<td>C</td>
<td>Zr + D$_2$O</td>
<td>C + He + Pu</td>
<td>inner</td>
<td>26</td>
<td>2.00</td>
</tr>
<tr>
<td></td>
<td>5% 95%</td>
<td>5% 90% 0.34 $\times 10^{-4}$</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>i</td>
<td>316SS</td>
<td>Zr + D$_2$O</td>
<td>C + He + Pu</td>
<td>inner</td>
<td>9.4</td>
<td>5.65</td>
</tr>
<tr>
<td></td>
<td>5% 95%</td>
<td>5% 90% 0.33 $\times 10^{-3}$</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
TABLE IV. EFFECT OF $^{90}$Sr AND $^{239}$Pu CONCENTRATIONS ON TRANS_MUTATION RESULTS

<table>
<thead>
<tr>
<th>Vol.% $^{90}$Sr</th>
<th>0</th>
<th>1</th>
<th>2</th>
<th>5</th>
<th>10</th>
</tr>
</thead>
<tbody>
<tr>
<td>$T_{\text{eff}}$ (a)</td>
<td>2.23</td>
<td>2.92</td>
<td>3.66</td>
<td>5.33</td>
<td>7.90</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Vol.% $^{239}$Pu</th>
<th>0.0</th>
<th>0.4982 $\times 10^{-3}$</th>
<th>0.1252 $\times 10^{-2}$</th>
<th>0.1386 $\times 10^{-2}$</th>
<th>0.1449 $\times 10^{-2}$</th>
</tr>
</thead>
<tbody>
<tr>
<td>$K_{\text{eff}}$</td>
<td>0.0</td>
<td>0.5</td>
<td>0.9</td>
<td>0.95</td>
<td>0.97</td>
</tr>
<tr>
<td>$T_{\text{eff}}$ (a)</td>
<td>16.7</td>
<td>14.0</td>
<td>6.10</td>
<td>3.66</td>
<td>2.32</td>
</tr>
</tbody>
</table>

* Assumed volume fraction of $^{239}$Pu: 0.1386 $\times 10^{-4}$ ($K_{\text{eff}} = 0.95$).

* Assumed volume fraction of SrO: 2%.

TABLE V. EFFECT OF NEUTRON WALL LOADING ON TRANS_MUTATION RESULTS

<table>
<thead>
<tr>
<th>$P_w$ (MW/m$^2$)</th>
<th>0.0$^a$</th>
<th>0.5</th>
<th>1</th>
<th>2</th>
<th>3</th>
<th>5</th>
<th>10</th>
</tr>
</thead>
<tbody>
<tr>
<td>$T_{\text{eff}}$ (a)</td>
<td>28.8</td>
<td>6.50</td>
<td>3.66</td>
<td>1.96</td>
<td>1.33</td>
<td>0.82</td>
<td>0.41</td>
</tr>
</tbody>
</table>

* Natural decay.

TABLE VI. REACTOR CORE PARAMETERS

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Major radius (R)</td>
<td>3.5 m</td>
</tr>
<tr>
<td>Minor radius (a)</td>
<td>1.0 m</td>
</tr>
<tr>
<td>Elongation (k)</td>
<td>1.5</td>
</tr>
<tr>
<td>Triangularity factor ($\delta$)</td>
<td>0.3</td>
</tr>
<tr>
<td>Toroidal magnetic field ($B_T$)</td>
<td>6 T</td>
</tr>
<tr>
<td>Plasma current ($I_p$)</td>
<td>7.8 MA</td>
</tr>
<tr>
<td>Average plasma temperature ($\langle T \rangle$)</td>
<td>14 keV</td>
</tr>
<tr>
<td>Average plasma density ($\langle N \rangle$)</td>
<td>$1.2 \times 10^{20}$ m$^{-3}$</td>
</tr>
<tr>
<td>Energy confinement time ($\tau_e$)</td>
<td>1.4 s</td>
</tr>
<tr>
<td>Fusion power ($P_w$)</td>
<td>200 MW</td>
</tr>
<tr>
<td>Auxiliary heating power ($P_{\text{aux}}$)</td>
<td>32 MW</td>
</tr>
<tr>
<td>Neutron wall loading ($P_{w}$)</td>
<td>1 MW·m$^{-2}$</td>
</tr>
</tbody>
</table>
onstrated in the JT-60 tokamak. The DIII tokamak has demonstrated a Troyon factor $g$ of up to 5 (transiently) at high values of the edge safety factor $q_a$; in DIII-D, a long pulse discharge is achieved with $g = 3.5$.

In parallel with these achievements in tokamak research, significant efforts are being made in the conceptual design of the International Thermonuclear Experimental Reactor (ITER) as a next step towards production of fusion energy.

The research on, and the conceptual design of, the fusion core of fusion–fission hybrid reactors producing fuel have been carried out since 1986 in ASIPP (Academia Sinica, Institute of Plasma Physics). The technology and the parameters in the conceptual design [6], many of which are similar to those of JET and TFTR, have been reached or can be reached in the near future.

For this kind of hybrid reactor design, the main parameter of the plasma core is the fusion neutron reaction rate:

$$\frac{dn}{dt} = V_p N_D N_T \langle \sigma v \rangle$$

where $V_p$ is the plasma volume, and $N_D$, $N_T$ are the densities of deuterium and tritium, respectively.

Generally, NB or ICRF heating can produce a high energy tail in the ion distribution and thus lead to an enhancement of the reaction rate.

In the plasma core, the high temperature will be maintained by external ICRF heating power, so that the fusion driver is a passive system working under sub-breakeven conditions. The energy confinement time of the plasma could be shorter than the value for the conditions of ignition.

To maintain the plasma current in the steady state, lower hybrid current drive (LHCD) and bootstrap current have been considered.

The parameters of the reference conceptual design for the $^{90}$Sr transmutation reactor are summarized in Table VI.

4. SUMMARY

(1) Many difficult and challenging engineering problems can be solved if fusion–fission hybrid reactors are used as HLW transmuters.

(2) By introducing plutonium into blankets as a neutron multiplication material, a higher thermal flux can be produced under a lower neutron wall loading.

(3) The plasma core in the fusion–fission hybrid reactor, which can drive the blanket for the transmutation of HLW, is realizable.

REFERENCES

DISCUSSION

C.W. BARNES: Can you comment on the relative merits of a fusion neutron source as opposed to an accelerator based spallation neutron source for the transmutation of waste?

Lijian QIU: Firstly, it is easier to obtain low biological hazard potential (final total relative hazard) with a three stage high level waste system than with an accelerator based spallation neutron source. Secondly, a self-sustaining system can be constructed that saves on energy consumption.
TIGHT ASPECT RATIO TOKAMAK REACTORS

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UKAEA–Euratom Fusion Association,
Culham Laboratory,
Abingdon, Oxfordshire,
United Kingdom

Abstract

TIGHT ASPECT RATIO TOKAMAK REACTORS.

Tight aspect ratio tokamak reactor designs are presented. Physics considerations are shown to result in very compact, high mass power density systems, which have significantly lower capital costs but costs of generating electricity comparable with those of conventional aspect ratio designs. For the tight aspect ratio designs the key technology areas of the breeding blanket and the central toroidal field column are addressed.

1. INTRODUCTION

Tight aspect ratio tokamaks, like reverse field pinches and spheromaks, have the potential to yield high mass power density compact reactors. This potential for a compact system arises from the very high plasma beta which may be achieved at tight aspect ratio. Another benefit of a tight aspect ratio is that the large natural elongation leads to a corresponding improvement in vertical stability, an important property in a reactor where stabilizing conductors are likely to be remote from the plasma. Many of the key properties of tight aspect ratio tokamaks have been demonstrated by the START experiment [1] (R/a ~ 1.3); these properties include:

— Vertical stability for b/a ≤ 2 with no feedback;
— Energy confinement consistent with dimensionless scaling laws (e.g. Rebut–Lallia [2]) and significantly exceeding the neo-Alcator prediction;
— natural divertor established with simple Helmholtz coils (no specific divertor coils);
— at tight aspect ratio (R/a ≈ 1.3–1.5) no major disruptions due to MHD activity have been observed.

These advantages of the tight aspect ratio tokamak concept indicate the merit of studying its reactor potential. In the following section tight aspect ratio designs based chiefly on physics considerations are described. Then, in Section 3, technology issues including the key technology of the centre post and its power supplies are discussed.
2. PHYSICS CONSTRAINTS AND REACTOR DESIGNS

An assessment of likely parameters for a tight aspect ratio reactor has been made by using a systems code developed from the TETRA code [3]. This code incorporates beta limits, diamagnetic/bootstrap [3] and current drive formulas derived from detailed physics calculations, for relevant tight aspect ratio equilibria. The normal external kink mode limit, $q_\phi > 2$, becomes more restrictive at tight aspect ratio, with $q_\phi \geq 4$ being required at $R/a = 1.2$. This edge $q$ limit is imposed as an upper bound on the ratio of plasma to toroidal field (TF) current in the systems code. The beta limit applied is:

$$\beta = \frac{2\mu_0}{a(m)B_v(T)} \left( \frac{\int P_d dv}{\int (B_T^2 + B_p^2) dv} \right) < \frac{g_T I(MA)}{\mu_0 B_v(T)}$$

where $B_v$ is the vacuum toroidal field, and, usually, $g_T = 3.5$. This form of the beta limit has been determined from extensive stability calculations. A previous tight aspect ratio study [4] has examined reactors based on an $\epsilon_\beta_p$ limit. The bootstrap current in the systems code is determined from a formulation by Wilson [5] valid for $R/a \geq 1.2$. The bootstrap and diamagnetic terms have also been calculated directly for marginally MHD stable equilibria. Figure 1 shows, for a case with $R/a = 1.4$, $q_\phi = 5$, $\beta = 33\%$ and $\beta_p = 0.76$, the contributions to the total toroidal current from the bootstrap ($I_{\text{boot}}/I_p = 56\%$) and diamagnetic terms ($I_{\text{dia}}/I_p = 16\%$) terms. The current drive efficiency in the systems code is taken to be basically the ITER formulation but with a correction to allow fast ion trapping on the outer surfaces, at tight aspect ratio.

A range of designs with copper coils and no inboard blanket (see next section) have been studied with the systems code. The main parameters for some designs are given in Table I. Here, case (d) illustrates the sensitivity of the 400 MW design [case(c)] to the energy confinement time, while cases (e) and (f) show the sensitivity to the beta limit. For case (g) it is assumed that the residual current is driven by directed reflection of the synchrotron emission [6]. Calculations using a conservative current drive efficiency, $I_p(MA) = 0.3 P_{\text{syn}}(MW)/(R(m)n_{20}(m^{-3}))$, show that sufficient synchrotron current is driven when the central electron temperature $T_0 \approx 40$ keV. This high temperature does not result in problems for the divertor, however, because the high beta simultaneously permits relatively high densities ($n_e \approx 2 \times 10^{20} m^{-3}$). A high internal driven current fraction can also be achieved by raising the edge $q$ from the $q_{95} = 5$ of the designs presented in Table I to $q_{95} = 10$. This results in $(I_{\text{boot}} + I_{\text{dia}})/I_p = 95\%$ though the cost of electricity increases by 35% relative to designs (a) to (c) in Table I. Comparisons between these designs (cases (a) to (c)) and conventional aspect ratio designs show comparable electricity generation costs. The tight aspect ratio designs, however, have a much higher mass power density (MPD); for example, case (b) has an MPD of ~400 kW(e)/tonne compared with ~100 kW(e)/tonne for a conventional aspect ratio reactor (e.g. ARIES-I [7]). This means that the initial capital cost for the fusion power core of the tight aspect ratio designs is 40% less than for comparable conventional aspect ratio cases.
FIG. 1. Internally driven bootstrap and diamagnetic currents for marginally stable equilibrium with $R/a = 1.4$ and $\beta_p = 0.76$. The contribution of these currents to the total current is also shown.
TABLE I. SUMMARY OF TIGHT ASPECT RATIO REACTOR PARAMETERS, ALL WITH R/a = 1.3 AND b/a = 2.3
The internal current fraction is (I_{boot} + I_{dia})/I_p. An asterisk indicates that the parameter is at a prescribed limit.

<table>
<thead>
<tr>
<th>Case</th>
<th>(a)</th>
<th>(b)</th>
<th>(c)</th>
<th>(d)</th>
<th>(e)</th>
<th>(f)</th>
<th>(g)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Net electric power output (GW)</td>
<td>1.2</td>
<td>0.8</td>
<td>0.4</td>
<td>0.4</td>
<td>0.4</td>
<td>0.4</td>
<td>0.4</td>
</tr>
<tr>
<td>Major radius, R(m)</td>
<td>2.26</td>
<td>1.97</td>
<td>1.79</td>
<td>1.87</td>
<td>1.96</td>
<td>1.74</td>
<td>1.61</td>
</tr>
<tr>
<td>Vacuum toroidal field, B_T(T)</td>
<td>1.76</td>
<td>1.76</td>
<td>1.61</td>
<td>1.51</td>
<td>1.67</td>
<td>1.49</td>
<td>1.61</td>
</tr>
<tr>
<td>Plasma current, I_p(MA)</td>
<td>34.5</td>
<td>30.0</td>
<td>25.0</td>
<td>24.6</td>
<td>28.4</td>
<td>22.5</td>
<td>22.5</td>
</tr>
<tr>
<td>Rod current, I_{rod}(MA)</td>
<td>19.9</td>
<td>17.3</td>
<td>14.4</td>
<td>14.2</td>
<td>16.4</td>
<td>13.0</td>
<td>13.0</td>
</tr>
<tr>
<td>\frac{t}{t_{RL}}</td>
<td>0.79</td>
<td>1.0</td>
<td>1.48</td>
<td>1.2^{*}</td>
<td>1.26</td>
<td>1.17</td>
<td>1.75</td>
</tr>
<tr>
<td>\bar{g}_T</td>
<td>3.5^{*}</td>
<td>3.5^{*}</td>
<td>3.5^{*}</td>
<td>3.3</td>
<td>3.0^{*}</td>
<td>4.0^{*}</td>
<td>3.5^{*}</td>
</tr>
<tr>
<td>Neutron wall load (MW·m⁻²)</td>
<td>10^{*}</td>
<td>9.35</td>
<td>6.53</td>
<td>6.31</td>
<td>5.84</td>
<td>6.37</td>
<td>7.23</td>
</tr>
<tr>
<td>Internal driven current fraction (%)</td>
<td>65</td>
<td>66</td>
<td>67</td>
<td>62</td>
<td>58</td>
<td>75</td>
<td>100</td>
</tr>
<tr>
<td>Recirculating power fraction (%)</td>
<td>33</td>
<td>38</td>
<td>46</td>
<td>50</td>
<td>50</td>
<td>42</td>
<td>39</td>
</tr>
</tbody>
</table>

3. TECHNOLOGY ISSUES

From the designs given in the previous section it can be seen that the typical inner plasma radius is ~0.4 m. This limited space at the centre of the torus means that superconducting toroidal field (TF) coils (and associated shielding) are impracticable and a copper TF coil set must be used. There may be sufficient space for some shielding and an insulated multiturn TF coil, but the existence of insulators which can remain effective in such a hostile environment remains to be demonstrated. So, here it is considered prudent to consider a simple copper rod for the central TF column. This rod must carry high current densities (\leq 6 kA/cm²), and so Ohmic dissipation contributes significantly to the recirculating electrical power. Detailed calculations including axial and radial variations of the nuclear and resistive heating, together with the temperature and transmutation dependence for the thermal and electrical conductivities, have been used to assess central column designs [8]. Figure 2 shows the Ohmic power dissipated in, and temperature of, a 0.35 m radius, 7 m long central conductor with a (near optimal) 33% coolant channel fraction. It can be seen that for typical central conductor currents (~15-20 MA) substantial Ohmic power is dissipated in the column. These calculations indicate an acceptable lifetime for the central column of about one year, which is the time at which the cost of enhanced Ohmic losses due to neutron damage exceeds the cost of replacing the column.
A single turn central TF conductor poses significant problems in terms of power supplies. The required high current (~0.5–1 MA per TF limb) and low voltage (~15 V) can be achieved with present transformer–convertor technologies. Such a system might contain 240 standard 20 V modules supplied by three step-down transformers (from 22 kV) [8]. From the power supply viewpoint the system is considerably simplified by a multiturn TF coil, with costs being reduced by ~5 for a 40 turn coil (though the TF power supplies are only a small fraction of the system cost, ~5%). However, as described, a multiturn system would require improved insulator technology.

The lack of space also precludes a central blanket. Detailed neutronics calculations [9], however, show that an outboard beryllium multiplied lithium oxide blanket provides sufficient tritium breeding for the tight aspect ratios of interest here (R/a ~ 1.5). This is essentially because of the low proportion of neutrons (~5% for R/a = 1.3) intercepted by the central column. This geometric phenomenon is a potential advantage of a tight aspect ratio reactor over its conventional aspect ratio counterpart where the engineering of the inboard blanket is difficult and complex.

4. SUMMARY AND DISCUSSION

Physics considerations, many of which have been verified in the START experiment, lead to high mass power density, compact, tight aspect ratio reactors. The chief physics advantages of tight compared to conventional aspect ratio reactors lie in the improved vertical and disruptive stability and the high beta capability; a disadvantage is the much higher neutron wall load. Of course, the much larger database for conventional aspect ratio tokamaks gives a much higher physics confidence for conventional aspect ratio designs and indicates the need for continued experiments to
enhance the tight aspect ratio database. From an economic viewpoint tight and conventional aspect ratios have a comparable cost of electricity though the tight aspect ratio system benefits from a lower initial capital cost.

ACKNOWLEDGEMENTS

It is a pleasure to acknowledge many useful conversations with Dr. Y.-K.M. Peng during this work. This work was partially funded by the UK Department of Trade and Industry and by Euratom.

REFERENCES


DISCUSSION

B. COPPI: It is not clear to me how you have resolved the problem of stability of the current density profile, given that the tight aspect ratio does not leave space for a transformer. Could you expand on this point?

T.C. HENDER: We favour high bootstrap fraction designs (~90–95%) so as to minimize the required current drive. The residual near axis current must be driven non-inductively (e.g. by fast wave). Since the stable toroidal current density profiles at high beta and tight aspect ratio are rather hollow, the requirement of obtaining stable bootstrap current profiles can be met with reasonable density profiles.

A. GIBSON: Is the reason for the cost insensitivity, that is the similar cost for small and large aspect ratio devices, due to the high cost of peripherals? Even if the plasma core cost nothing the electricity cost would still be of the same order!

T.C. HENDER: The cost of electricity does include ‘external’ items such as turbines. However, even if these external items are factored out, the cost of electricity for conventional and tight aspect ratio remains reasonably comparable.
DISRUPTIONS AND STABLE PLASMA SHUTDOWN IN JT-60U

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Abstract

DISRUPTIONS AND STABLE PLASMA SHUTDOWN IN JT-60U.

Stable plasma current ramp-up and ramp-down free of disruptions have been developed in JT-60U in order to support the establishment of operational scenarios for tokamak fusion reactors. To obtain plasmas with $3 < q_{\text{eff}} < 4$, locked mode disruptions during the plasma current ramp-up have been avoided by crossing $q_{\text{eff}} = 4$ just after plasma initiation. In the plasma current ramp-down, density limit disruptions and high $\ell_1$ disruptions at low plasma densities have been avoided by reducing the plasma minor radius; the density limit has been raised to values twice as high, and $\ell_1$ has been reduced to values lower than the limiting value. Broadening of the plasma current profile has been achieved by adding a helical field with an $m/n = 3/2$ principal mode. In the density limit disruption, techniques of avoiding the current quench or reducing the current decay rate have been developed in JT-60U divertor plasmas. A fast plasma shutdown of $-6$ MA/s has been obtained by producing a detached plasma intentionally by intense helium gas puffing during the plasma current ramp-down.

1. INTRODUCTION

In tokamak fusion reactors, disruptions will determine the lifetime of the first wall and divertor plates, and the resultant high electromagnetic stress on the structure will impose severe constraints on the design. The development of operational scenarios avoiding disruptions or, at least, minimizing their impact remains a crucial problem for ITER [1]. An experimental database of stable plasma current ramp-up and ramp-down free of disruptions will be necessary to clarify the operational scenarios in ITER [2], especially for fast plasma current ramp-down as will be required in some emergency cases [3]. Fast plasma current ramping is also beneficial to minimizing the dwell time in pulsed reactors such as an AC tokamak reactor [4].

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To support the establishment of these operational scenarios for fusion reactors, the following subjects have been investigated in JT-60U:

(1) Avoidance of **locked mode disruptions** during the plasma current ramp-up to obtain plasmas with $3 < q_{\text{eff}} < 4$. These disruption free operation scenarios are required even for $q_{\text{eff}} > 3$, because crossing $q_{\text{eff}} = 4$ with low $l_i$ causes minor or major disruptions as is observed in JFT-2M [5] and JET [6]. Here, $q_{\text{eff}}$ is the effective safety factor defined by

$$ q_{\text{eff}} = \frac{5a_p^2B_T}{R_p I_p} \left( \frac{1 + \kappa^2}{2} \right) \left[ 1 + \left( \frac{a_p}{R_p} \right)^2 \left( 1 + \frac{(\beta_p + l/2)^2}{2} \right) \right] \times \left[ 1.24 - 0.54\kappa + 0.3(\kappa^2 + \delta^2) + 0.13\delta \right] $$

and is $\sim 1.25 \times q_{95}$ in JT-60U ($q_{95}$ is the safety factor at 95% of the outermost flux surface). $l_i$ is the plasma internal inductance, and $\delta$ is the plasma triangularity. $l_i$ is defined by

$$ l_i = \left( \int B_p^2 dv/V_p \right) / \left( \mu_0 I_p / \int dl \right)^2 $$

(2) Avoidance of **density limit disruptions** and **high $l_i$ disruptions** during the plasma current ramp-down. Both effects may be triggered by the same tearing mode, because of the strongly peaked current profile. This peaking is, however, produced by different processes. The former effect is caused by the shrinkage of a plasma current channel due to the lack of power balance in the plasma periphery at high plasma densities [7]. The latter effect is caused by the long diffusion time of the plasma current due to high electron temperatures at low plasma densities. To obtain a stable plasma shutdown, expansion of this density window and broadening of the current profile (lowering of $l_i$) are required.

(3) Avoidance of current quench in **density limit disruptions**. In the density limit disruptions, a detached plasma phase with many minor disruptions was observed after the onset of a MARFE [8] and before the current quench in JT-60U. Plasma detachment is always observed before the magnetic perturbations for plasmas with $q_{\text{eff}} > 3$ in JT-60, so that the detached plasma phase can be used as a reliable precursor to avoid disruptions.

(4) Reduction of the current decay rate in **density limit disruptions**. A slower current decay rate will reduce the electromagnetic forces on in-vessel components, will improve the maintenance of plasma position control following the energy quench and may reduce the production of non-thermal (runaway) electrons. Improved current decay control will be necessary in emergency shutdowns with passive control, for example after impurity pellet injection [9].
2. STABLE PLASMA CURRENT RAMP-UP

In JT-60U, plasma current ramp-up with constant plasma minor radius is limited to \(<0.5\) MA/s by locked mode disruptions. The locked mode disruptions are believed to be a consequence of the error field produced by the feeder of the divertor coil since they are only observed when the divertor coil is excited. From the coil design, the \(m/n = 2/1\) component of this field at the plasma surface is calculated to be \(0.5-1\) G (\(5 \times 10^{-5}-10^{-4}\) T). Rotating modes are observed before locked mode disruption at \(q_{\text{eff}}\) slightly larger than \(4\) or \(5\) (see Fig. 1, \(\times\)). Locked modes from crossing \(q_{\text{eff}} = 4\) after the excitation of the divertor coil cause minor disruptions with a sudden increase in \(l_i\) (see Fig. 1, \(\square\)) and sometimes terminate the discharge by major disruptions. Furthermore, the low density locked mode [10] is observed at \(q_{\text{eff}} < 3.7\) for \(n_e < 0.9 \times 10^{19}\) m\(^{-3}\) only with the excitation of the divertor coil. The effect of the error field is believed to become severer in large tokamaks [11], and therefore operational scenarios avoiding locked mode disruptions during the plasma current ramp-up become more important.

In JT-60U, to obtain plasmas with \(3 < q_{\text{eff}} < 4\) without locked mode disruptions, \(q_{\text{eff}}\) was reduced to a value lower than \(4.0\) just after plasma initiation (e.g. after \(0.2\) s) by strong compression of the plasma on the inboard side with a plasma minor radius of \(a_p = 0.6\) m and a plasma volume of \(20\) m\(^3\). While the plasma current was ramped up with \(1\) MA/s, \(a_p\) was expanded to \(0.9-1.0\) m with a volume of

![FIG. 1. Time behaviour of \(a_{\text{eff}}\) and \(l_i\) during \(1.0\) MA/s plasma current ramp-up with expanded \(a_p\) (■), and \(0.5\) MA/s ramp-up with constant \(a_p\) (□). The expansion of the plasma is shown in an inset with time point numbers. \(\times\) is the start of the rotating mode before locked mode disruption. \(l_i\) as obtained by expanding \(a_p\) is \(0.1\) higher than that obtained with constant \(a_p\) at the moment where the plasma configuration changes from limiter plasma to divertor.](image)
80–100 m$^3$, keeping $3 < q_{\text{eff}} < 4.0$ (Fig. 1, ■). Even though the divertor coil current was excited to change the plasma configuration from limiter to divertor during this ramp-up, locked modes were totally suppressed. This operational scenario substantially reduces the skin current leading to higher values of $I_i$ for the divertor plasma and improving MHD stability. This plasma current ramp-up method was tried in many shots without disruption and should be applicable to tokamak reactors such as ITER, with a designed safety factor of $q_{95} = 3$.

By using this ramp-up method, ramp rates as high as 2.0 MA/s were obtained from 0.5 MA to 3.5 MA without additional NB heating or the use of a helical field. Minor disruptions caused by locked modes without rotating precursors were observed at $I_p > 1$ MA/s, but did not cause any current quench.

3. STABLE PLASMA SHUTDOWN

3.1. Reduction of plasma minor radius

Two types of major disruption have been observed during the plasma current ramp-down in JT-60U: density limit disruptions occurring in high density plasmas, and high $I_i$ disruptions at low densities. Enlarging this density window is advantageous for stable plasma shutdown. $I_i$ must be kept lower than some limiting value in order to avoid high $I_i$ disruptions.

In JT-60U, the plasma minor radius was reduced during the current ramp-down in order to raise the upper density limit by increasing the joule heating power density in the plasma periphery ($\sim 100\%$) and to broaden the plasma current profile by decreasing the skin time. The upper density limit during this ramp-down was about twice as high as that in the plasma current ramp-down with a constant minor radius. Thus it was easy to deep the plasma density above the lower density limit by gas puffing without reaching the upper density limit. $I_i$ was controlled to remain lower than the limiting value during the lowering of $q_{\text{eff}}$ to $\sim 3$ (Fig. 2(a), ■), where the plasma minor radius was reduced from 1.1 m to 0.7 m. This ramp-down scenario was tried in many shots without major disruption at $I_p > 0.5$ MA. Furthermore, fast ramp-down from 2.0 MA to 0.5 MA with $\frac{dI_p}{dt} = -4.0$ MA/s was obtained.

3.2. Excitation of an $m/n = 3/2$ helical field

A helical field with a principal mode number of $m/n = 3/2$ produced by disruption control winding (DCW) [12] was excited to broaden the plasma current profile and to obtain stable plasma current ramp-down. The radial field component of $m/n = 3/2$ is 7 G ($7 \times 10^{-4}$ T) at the plasma centre with a coil current of 14 kA. A typical case is shown in Fig. 2(b), where the helical field degrades the energy confinement gradually at $t = 13.15-13.4$ s ($q_{\text{eff}} = 6-7$) with MHD activities outside the sawtooth inversion radius observed by a soft X ray array. The obtained value of $I_i$
FIG. 2. (a) Trajectory of $q_{\text{eff}}$ and $l_i$ during plasma current ramp-down. • is the case of plasma current ramp-down from 2 MA to 1.2 MA with $a_p$ being reduced, where $q_{\text{eff}}$ is lowered to ~3 and $l_i$ is maintained at lower values than the high $l_i$ limit. × is the case of a disrupted shot with constant $a_p$ during plasma current ramp-down of ~1.0 MA/s. Δ indicates a case where the DCW (14 kA) maintains $l_i ~ 0.2$ lower than the limiting value.

(b) Effect of DCW excitation on plasma current ramp-down. At $t = 13.15-13.4$ s ($q_{\text{eff}} = 6-7$), the energy confinement measured by the diamagnetic loop is degraded by strong MHD activities observed outside the sawtooth inversion radius. By this degradation, $l_i$ is reduced to values 0.2 lower than the high $l_i$ limit at $q_{\text{eff}} > 7$ as is shown in (a) by Δ, and a stable plasma shutdown is obtained. A shot disrupted by high $l_i$ limit shown in (a) by × is represented by a dotted line for the purpose of comparison.
FIG. 3. (a) Duration of detached plasma phase versus plasma internal inductance $l_i$ at the start of the detached plasma, for Ohmic plasmas with $3 < q_{\text{eff}} < 5$ at the current flat-top. The duration increases slightly when $l_i$ is lowered.

(b) Plasma current decay time versus stored energy just before energy quench for density limit disruptions in divertor plasmas with joule heating alone at plasma current flat-top. The plasma current is 1.5–2.0 MA with $q_{\text{eff}} = 4–6$. Shots with runaway electrons generated during current quench are excluded. Definition of current decay time: average of $I_p/(dI_p/dt)$ during current quench.
(c) Degradation of stored energy during detached plasma phase at plasma current flat-top. $W_{\text{dia}}$ just before the energy quench is plotted for each $W_{\text{dia}}$ value just before the detached plasma. Plasma current: 1.5-2.0 MA with $q_{\text{eff}} = 4-6$.

(d) Time evolution of current quench with and without NB heating. 7.5 MW NB heating reduces the current decay rate drastically from $-142$ MA/s to $-26$ MA/s. $I_{p}^{\text{com}}$ is reduced with plasma current decay to avoid generation of runaway electrons.
is 0.2 lower than the higher $l_i$ limit at $q_{\text{eff}} > 7$ (Fig. 2(a), $\Delta$). On the other hand, a current quench is observed in a shot without drop in $l_i$ (Fig. 2(a), $\times$; Fig. 2(b)). This broadening of the current profile by the excitation of DCW was obtained for $0.2 \leq r_{\text{inv}}/a_p \leq 0.3$, where $r_{\text{inv}}$ is the sawtooth inversion radius. However, this helical field had no effect on plasmas with $r_{\text{inv}}/a_p < 0.2$ and caused a current quench at $r_{\text{inv}}/a_p > 0.3$. Here the current profile is not a strong function of $q_{\text{eff}}$ during the plasma current ramp-down so that $r_{\text{inv}}/a_p$ cannot be represented by $q_{\text{eff}}$.

4. DENSITY LIMIT DISRUPTION

4.1. Detached plasma and MARFE

In the density limit disruption of JT-60U divertor plasmas, MARFE and detached plasma were always observed before the current quench at $q_{\text{eff}} > 3$. The duration of the MARFE was $> 1$ s for OH and NB plasmas; it was not a function of $q_{\text{eff}}$. A stable MARFE of $\sim 4$ s was observed at all $q_{\text{eff}} > 3$. For OH plasmas, the duration of the detached plasma phase was $0.1-0.6$ s at $3 < q_{\text{eff}} < 5$ (Fig. 3(a)) and increased a little for decreasing $l_i$ at the start of the detached plasma phase. This may be explained by the larger stability margin of $m/n = 2/1$ or $3/2$ modes due to the smaller $q = 1$ radius [13]. For $q_{\text{eff}} > 7$, no current quench was observed after the detached plasma phase.

A MARFE or a detached plasma can be returned to the divertor plasma by high power NB heating or LHRF heating in JT-60U. A slow positive plasma current ramp with 0.2 MA/s also produces a long detached plasma phase of $> 1$ s for $q_{\text{eff}} > 3$, owing to the flattening of the plasma current profile (lowering $l_i$) and the heating of the plasma periphery. In a MARFE or a detached plasma the heat flux to the divertor plate can be reduced to very small values, because nearly 100% of the input power is radiated by the main plasma. This drastic reduction of the heat flux to the divertor plate is a good precursor of the density limit disruption and will offer a possibility to avoid melting of the divertor plate in an emergency such as a loss of coolant accident (LOCA [9]), where the detached plasma can be generated by impurity pellet injection or intense gas puffing and can be maintained by additional heating.

4.2. Current quench

In JT-60U the average current decay time defined by $I_p/(dI_p/dt)$ in the density limit disruption was found to increase by lowering the plasma stored energy just before the energy quench for OH plasmas with $I_p = 1.5-2$ MA and $q_{\text{eff}} = 4-6$ (Fig. 3(b)). Shots with generation of runaway electrons are excluded because runaway electrons decrease the current decay rate. A pronounced spike was observed in the soft X ray emission immediately after the energy quench as was observed in JET [14], when the current decay rate was high. However, in the case of a low cur-
rent decay rate this spike was small. This soft X ray emission is caused by an impurity influx. High impurity influx lowers the electron temperature drastically and increases the current decay rate. Thus a lower stored energy just before the energy quench that can be obtained by degradation of the energy confinement during the detached plasma phase due to many minor disruptions (Fig. 3(c)) decreases the impurity influx and raises the current decay time.

To increase the current decay time, a plasma was heated by 7.5 MW NB during the density limit disruption (Fig. 3(d)). A very slow reproducible plasma current decay rate of $-26 \text{ MA/s}$, in contrast to a rate of $-142 \text{ MA/s}$ without NB heating, was obtained. Plasma current control was reduced with the decay of the plasma current during the current quench so that the one turn voltage was lower than 50 V and generation of runaway electrons was suppressed. A drop in the stored energy was also observed during a detached plasma phase before the energy quench, even with NB heating (Fig. 3(c)). Lower stored energy just before the energy quench was also beneficial to obtaining a long current decay time for plasmas heated by NB. A hot plasma core can be observed by TV only for long current decay times. This observation suggests that a drop in the electron temperature just after the energy quench can be avoided by suppressing the influx of impurities and by NB heating of the plasma core.

A passive plasma shutdown of $-6 \text{ MA/s}$ was obtained from 2 MA to 0 MA by using a detached plasma. This detachment was created by puffing 30 Pa·m$^3$/s of helium during the plasma current ramp-down with a drastic reduction of the stored energy. Repeated minor disruptions were observed after the first energy quench; however, no current quench occurred. High electron temperature was confirmed still after the energy quench by a hot plasma core observed by TV. The avoidance of the electron temperature drop may be explained by the suppression of the impurity influx due to the degraded stored energy just before the energy quench and by the increase in the joule heating power density due to the contraction the plasma current channel.

5. CONCLUSIONS

The following results on plasma current ramp-up and ramp-down have been obtained in JT-60U supporting the establishment of operational scenarios in tokamak fusion reactors:

1. To obtain plasmas with $3 < q_{\text{eff}} < 4$, stable plasma current ramp-up free of locked mode disruptions has been achieved by crossing $q_{\text{eff}} = 4$ just after plasma initiation.

2. Stable plasma current ramp-down free of density limit disruptions at high densities and of high $I_p$ disruptions at low densities has been achieved by reducing the plasma minor radius; the upper density limit was raised twice higher and $I_p$ was reduced to values lower than the limiting value.
(3) Fast plasma current ramping with $+2 \text{ MA/s}$ and $-4 \text{ MA/s}$ obtained by (1) and (2) will be beneficial for pulsed fusion reactors such as an AC tokamak.

(4) Broadening of the plasma current profile to reduce $\xi$ to values lower than the limiting value has been achieved during plasma current ramp-down by adding a helical field with a principal mode number of $m/n = 3/2$ for an appropriate range of sawtooth inversion radii.

In density limit disruptions, the following results have been obtained in JT-60U, which will be useful in avoiding density limit disruptions, minimizing their effect and obtaining controlled fast plasma shutdown in some emergency cases in tokamak fusion reactors:

(1) MARFEs longer than 1 s and detached plasma phases longer than 0.1 s have been observed at $q_{\text{eff}} \geq 3.0$ for divertor plasmas.

(2) Current quench has been avoided by additional heating of the detached plasma, returning to a divertor plasma or maintaining the detachment.

(3) Reduction of the heat flux to the divertor plate down to very small levels has been obtained by producing a MARFE or a detached plasma.

(4) The current decay time defined by $I_p/(dI_p/dt)$ increased with lowering the stored energy just before the energy quench. Lower impurity influx obtained by lower stored energy may reduce the drop of the electron temperature during the current quench.

(5) NB heating during the density limit disruption was found useful in decreasing the current decay rate.

(6) Fast passive plasma shutdown of $-6 \text{ MA/s}$ has been obtained by the active use of energy degradation during a detached plasma phase produced by intensive helium gas puffing.

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IGNITION STUDIES FOR ITER PLASMAS

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Abstract

IGNITION STUDIES FOR ITER PLASMAS.

The present design proposal for ITER is essentially based on a stationary balance between fusion power and anomalous losses. The alpha power is deposited instantaneously and locally in the plasma; the losses are determined by global scaling laws. The evolution of an ITER plasma in time and space, including a kinetic description of the alpha particles with transport (perpendicular to the magnetic surfaces, $\rho$) and with slowing down is computed. In general, it is found that the proposed ITER plasmas cannot be stationary. In particular, the alpha transport leads to a diminished heat source and thus to less likely ignition or to a severe shortening of the burning period. It is only under very optimistic assumptions on the transport losses that the present ITER is found to ignite, and without (yet unknown) burn controls it seems to be rather unlikely that the envisaged burning periods can be reached.

1. INTRODUCTION

The stated physics objective of next generation fusion experiments such as NET/ITER is to produce ignited plasmas burning for hundreds of seconds in a quasi-steady state. This state is largely dominated by fusion alpha particles as a major source of heat and impurities.

For the design proposal of ITER [1], it is assumed that the fusion alpha particles transfer their energy to the plasma at the location of their birth. This heating power is then balanced, again instantaneously, by losses that are mainly determined by global confinement time 'scaling laws'.

In reality, there are finite periods of time both for heating (i.e. alpha slowing down) and for transport across the B-field both for alpha particles and heat loss. The time-scales are, in general, quite different and vary greatly in space so that the plasma develops dynamically and the steady state assumed by ITER is not reached automatically at all.

To describe the dynamical behaviour of the plasma we apply the transport code JETTO [2], in which the empirical coefficients are fitted to JET measurements. The equations for both electron and ion temperatures are improved by the source terms
from the slowing down alpha particles, and the changes in the various ion densities, and hence in $Z_{\text{eff}}$, due to fusion, are taken into account.

Fusion alpha particles are highly non-Maxwellian so that we describe them by a kinetic equation for a distribution function, $f_a(p, E, t)$, depending on magnetic flux surface $p$, kinetic energy $E = (m/2)v^2$, and time $t$.

Classical slowing down and anomalous spatial diffusion of the alpha particles are taken into account. The equation is solved self-consistently with the equations of the plasma evolution code.

The combined code package is then applied to ITER parameters [1] with two aims: to check the conditions leading to ignition and to investigate the burning behaviour. The results are compared with those from the usual simplified calculations.

2. ALPHAPARTICLE TRANSPORT MODEL

The basic idea for a kinetic equation appropriate to fast alpha particle transport is due to D"uchs and Pfirsch [3]. This formalism has been improved and generalized by Ref. [4] and is summarized in the following.

It is assumed that the alpha particle fluxes $\Gamma_a$ and the heat fluxes $q_a$ are known functions of the radial co-ordinate $p$, the densities and temperatures and their gradients, respectively:

$$\Gamma_a = \Gamma_a(\rho, n_e, n_a, n_i, T_e, T_a, T_i, \ldots, \nabla n_a, \ldots, \nabla T_a, \ldots)$$

$$q_a = q_a(\rho, n_e, n_a, n_i, T_e, T_a, T_i, \ldots, \nabla n_a, \ldots, \nabla T_a, \ldots)$$

(2.1)

On this assumption the kinetic equation for the alpha particle distribution function $F_a(p, E, t)$ is given by

$$\frac{\partial F_a}{\partial t} + \frac{\partial}{\partial E} (LF_a) - \frac{\partial^2}{\partial E^2} (DF_a) = Q(\rho, E, t)$$

(2.2)

The second and third terms on the left hand side of this equation represent bounce averaged Fokker-Planck terms. $\rho$ is the effective flux surface radius. The source term on the right hand side consists of two terms:

$$Q(\rho, E, t) = Q_{\text{ex}}(\rho, E, t) + Q_{\text{in}}(\rho, E, t)$$

where $Q_{\text{ex}}(\rho, E, t)$ denotes the thermonuclear alpha particle source and

$$Q_{\text{in}}(\rho, E, t) = Q_{\text{in}}(\rho, E, F_a, \Gamma_a, q_a, \ldots, t)$$

(2.3)
represents a source due to transport. $\Gamma_\alpha$ is the alpha particle flux, and $q_\alpha$ is the heat flux. The detailed structure of Eq.(2.3) is found in Refs [3, 5–7]. Equation (2.2) can be solved if the transport laws for the fusion alphas, e.g.

$$\Gamma_\alpha = -D_\alpha \cdot \nabla n_\alpha, \quad q_\alpha = -\chi_\alpha \nabla T_\alpha$$

are known. The transport coefficients $D_\alpha$ and $\chi_\alpha$ may also depend on $n_\alpha$, $T_\alpha$, $\nabla n_\alpha$, $\nabla T_\alpha$, etc.

Therefore, Eq.(2.2) is highly non-linear and must be solved by iterations.

Though fusion alpha particle research is in progress there is only little information on alpha transport coefficients. Lacking theoretically well founded models we extended the Rebut–Lallia–Watkins (RLW) model [8, 9] (Section 3) to alpha particle transport, assuming

$$D_\alpha \sim D_i \sim \alpha_i \chi_e \quad (2.4)$$

where $\alpha_i$ is a coefficient $\leq 1$.

Equation (2.4) implies that fast alpha particles are confined in a way similar to that of the bulk plasma ions. This assumption seems to be justified from some indications to be found in Ref.[10].

3. IMPLEMENTATION OF THE ALPHA TRANSPORT CODE INTO THE JETTO CODE

The JETTO code is a well known plasma evolution code used at JET for predictive and interpretative transport calculations. The set of equations is closed by defining the transport models. The anomalous transport is based on the semi-empirical RLW model. This critical electron temperature gradient model is supported by experimental evidence at JET. Some parameters are available to be adapted to the experimental situation.

The source term of the electron energy equation in JETTO is given by

$$S_{Ee} = ... Q_e^\alpha - Q_{Br} + Q_{RF}^e + ... \quad (3.1)$$

where $Q_e^\alpha$ is the alpha power deposited to the electrons, $Q_{Br}$ the bremsstrahlung power and $Q_{RF}^e$ the RF power imparted to the electrons.

The terms contributing to the source of the ion energy equation,

$$S_{Ei} = ... Q_i^\alpha + Q_{RF} + ... \quad (3.2)$$

are explained analogously.
In the standard version of JETTO the alpha power deposition terms $Q^{\alpha}, Q^\alpha_{ea}$ are calculated by means of a local deposition model neglecting alpha transport and slowing down.

The scenarios to be described in the next section have been obtained by a combination of JETTO with the kinetic alpha transport code providing alpha power deposition rates calculated from $F_\alpha(\rho, E, T)$.

4. IGNITION STUDIES FOR ITER

Even for a transport code the space of variables (initial and boundary conditions, empirical formulae, etc.) is of such high dimensionality that a systematic study is impossible.

The choice of cases is here based on the experience with a large number of trial runs which have confirmed the great sensitivity of ignition with respect to anomalous

<table>
<thead>
<tr>
<th>Case</th>
<th>Mode</th>
<th>RF heating</th>
<th>$\tau_\alpha$ (s)</th>
<th>$T_\alpha$ (keV)</th>
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<td>1</td>
<td>L</td>
<td>20 s &lt; t &lt; 21 s: 0 MW &lt; $P_{RF}$ &lt; 60 MW</td>
<td>No ignition</td>
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<td>21 s &lt; t &lt; 30 s: $P_{RF}$ = 60 MW</td>
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<td>30 s &lt; t &lt; 32 s: 60 MW &gt; $P_{RF}$ &gt; 0 MW</td>
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<td>2</td>
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<td>The same as Case 1</td>
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<td>(a) 5 s ignition</td>
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<tr>
<td></td>
<td></td>
<td>(b) No ignition</td>
<td></td>
<td></td>
</tr>
<tr>
<td>3</td>
<td>H</td>
<td>20 s &lt; t &lt; 21 s: 0 MW &lt; $P_{RF}$ &lt; 100 MW</td>
<td>2.9</td>
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<td></td>
<td>21 s &lt; t &lt; 40 s: $P_{RF}$ = 100 MW</td>
<td></td>
<td></td>
</tr>
<tr>
<td>4</td>
<td>H</td>
<td>20 s &lt; t &lt; 21 s: 0 MW &lt; $P_{RF}$ &lt; 100 MW</td>
<td>3.0</td>
<td>22</td>
</tr>
<tr>
<td></td>
<td></td>
<td>21 s &lt; t &lt; 31 s: $P_{RF}$ = 100 MW</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>31 s &lt; t &lt; 34 s: 100 MW &gt; $P_{RF}$ &gt; 0 MW</td>
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<td></td>
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<tr>
<td>5</td>
<td>H optimistic</td>
<td>The same as Case 4</td>
<td>3.2</td>
<td>17</td>
</tr>
<tr>
<td></td>
<td></td>
<td>(a) 13 s ignition</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>(b) 11 s ignition</td>
<td></td>
<td></td>
</tr>
<tr>
<td>6</td>
<td>H optimistic</td>
<td>20 s &lt; t &lt; 21 s: 0 MW &lt; $P_{RF}$ &lt; 100 MW</td>
<td>4.0</td>
<td>17</td>
</tr>
<tr>
<td></td>
<td></td>
<td>21 s &lt; t &lt; 28 s: $P_{RF}$ = 100 MW</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>28 s &lt; t &lt; 31 s</td>
<td>(a) 30 s ignition</td>
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</tr>
<tr>
<td></td>
<td></td>
<td>(b) No ignition</td>
<td></td>
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</table>
transport losses. Therefore, we test the influence of spatial alpha particle transport — which is a new main ingredient in our code package — for several transport assumptions; as a side line we also vary the external RF heating profile (in time).

The basic ignition scenario is fairly standard with parameters of Ref.[1]: for 20 s the plasma is heated up to the Ohmic equilibrium of about 4 keV on axis. At t = 20 s (external), the RF power is ramped up linearly. Simultaneously, the density is increased (linearly) from \(8 \times 10^{19}\) to \(1.2 \times 10^{20}\) m\(^{-3}\) [1].

If the kinetic equation, Eq.(2.2), is solved without alpha transport \((D_a = 0)\), i.e. only slowing down is considered, no important differences are observed between the kinetic treatment of alpha thermalization and simplified non-kinetic models [11, 12] used in earlier JETTO versions.

In Table I the results of six typically different ignition scenarios are summarized. The state of ignition is defined as

\[
Q_s = \frac{P_{\text{fusion}}}{P_{\text{loss}}} = \frac{5P_\alpha}{P_{\text{loss}}} > 5
\]

*Case 1* represents the most often observed L-mode confinement; the RF power wave forms are stated in the table. The transport formula [8] is fitted to the latest JET data. No ignition is found in any case.

For *Case 2* the upper end of the error bar on the formula [2] for L-modes (‘optimistic L-mode’, reduction of 40%) has been applied. Ignition was achieved for about 5 s if alpha transport is neglected. No ignition was achieved with alpha transport (Fig. 1).

In JET, H-mode transport with the formula [8] is produced by reduction of the conductivities at the plasma edge. This procedure, *Case 3*, leads to ignition for localized alpha power for about 5 s after which the confinement degrades because of overheating. No ignition is reached with alpha transport (Fig. 2).

To avoid overheating, *Case 4*, the RF power is switched off after ignition. The burn time is somewhat extended without alpha transport, and ignition is marginally accomplished even with alpha transport (Fig. 3). The evolution of the confinement times is given in Fig. 4.

It is only for an overoptimistic *Case 5* that we can produce clear burning periods between 10 and 15 s with and without alpha transport. For these cases the H-mode transport of Case 3 was further reduced by a global factor of 0.3 (Fig. 5).

In all cases mentioned above, ignition is terminated by reduction of confinement due to overheating. This indicates the necessity of some burn control measures.

During the external (RF) heating phase the power wave form could be tailored on a temperature (and hence confinement) feedback basis.

*Case 6* presents such an optimized case although the burning period is still less than 30 s. Including alpha transport the same scheme does not produce ignition; a longer heating pulse leads to ignition and to a short period of burning (Fig. 6).
**FIG. 1.** $Q_5$ versus time (Case 2).

**FIG. 2.** $Q_5$ versus time (Case 3).
FIG. 3. $Q_3$ versus time (Case 4).

FIG. 4. $\tau_E$ versus time (Case 4).
FIG. 5. $Q_5$ versus time (Case 5).

FIG. 6. $Q_5$ versus time (Case 6).
6. CONCLUSIONS

(i) For localized alpha heating (no spatial transport) we found that ITER only ignites for (not yet observed) uninterrupted H-mode transport losses and $Z_{\text{eff}} < 1.6$. Even then, the burning periods fall rather short of the anticipated values.

(ii) The (anomalous) spatial transport of alpha particles during their slowing down has a profound downgrading effect on the ignition and burning of fusion plasmas.

(iii) The long burning, quasi-stationary states envisaged for present day ITER parameters are highly unlikely because they are not stable. Therefore, burn control mechanisms seem to be absolutely necessary.

ACKNOWLEDGEMENT

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REFERENCES

NONDIMENSIONAL TRANSPORT STUDIES IN TFTR


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Abstract

NONDIMENSIONAL TRANSPORT STUDIES IN TFTR.

The machine parameters \( I_p, P_{\text{heat}}, R \) required for ignition in ITER have generally been extrapolated from power-law regression fits to global \( T_E \) measurements on existing tokamaks. There remain important choices to be made in the form of the scaling relation which have not yet been resolved by theory. In particular, power flow \( Q(r) \) through a magnetic flux surface should scale as \( Q(r) = \dot{Q}_{\text{Bohm}} F \), where \( F = F(p^*, \beta, \nu^*, s, T_e/T_i, ...) \) is a function of local, nondimensional plasma parameters and \( \dot{Q}_{\text{Bohm}} \propto [n_i T_i^2 a/eB] \). Projections to ITER can be reduced to establishing the dependence of \( F \) on \( p^* = p/a \), because one can create plasmas in today's tokamaks which have similar values of the other nondimensional parameters. Two common scalings suggested by theory are Bohm (\( F \) independent of \( p^* \)) and gyroBohm (\( F \propto p^* \)). Experiments have been carried out on TFTR to ascertain the dependence of \( F \) on \( p^*, \nu^* \), and \( \beta \) in L-mode plasmas, holding the other nondimensional parameters fixed. The observed variation of heat flow with \( p^* \) was observed to be better described by Bohm scaling than gyroBohm. Comparisons with the critical temperature gradient transport model, which is gyroBohm in character, show that it overpredicts the temperature increase expected with increasing magnetic field. The \( \nu^* \) scan (remaining in the collisionless regime) revealed that the Bohm-normalized power flow is remarkably insensitive to collisionality, in agreement with ITER-P scaling. The \( \beta \) scan identified a deterioration of confinement with increasing \( \beta \) at fixed \( p^* \) and \( \nu^* \), of approximately the correct magnitude required to reconcile Bohm local transport scaling with ITER-P global scaling of \( T_E \). This may suggest a role for electromagnetic phenomena in governing tokamak transport even at very low beta.

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1. Introduction

In 1975, Kadomtsev[1] observed that, through dimensional analysis based on fundamental plasma principles, it is possible to express heat flow across a magnetic surface in terms of a diffusivity based on the natural scale length $\rho_s$ and time scale $\tau_0$ of microinstabilities. These are $\rho_s \equiv c\sqrt{2T_e}\alpha_{n_i}/eB$ and $\tau_0 \equiv L_n\sqrt{M_i/2T_e}$. where $L_n$ and $\alpha_n$ are defined by $L_n \equiv (\frac{1}{n}\frac{\partial n}{\partial r})^{-1} \equiv \frac{2\pi r_n}{\alpha_n}$. The resultant diffusivity $\chi_{SB} = \rho_s^2/\tau_0$ has become known as gyroBohm diffusivity. Plasma physics principles require the anomalous heat diffusivity to have the form

$$\chi = \chi_{SB}(\rho^*)^n f(\nu^*, \beta, q, s, \frac{T_e}{T_i}, \ldots)$$

where $\rho^* = \rho_s/a$ and $f$ is a function of local nondimensional parameters $\nu^*, q, s, \ldots$. The power law dependence on $\rho^*$ in Eq. 1 is not completely general, but it is expected because $\rho^*$ is a small parameter. Connor-Taylor invariance arguments[2] for particular classes of instabilities further constrain the otherwise arbitrary function $f$.

The dependence of $\chi$ on $\rho^*$ is of particular interest to magnetic fusion research because in today's shaped tokamaks – notably DIII-D[3] and JET[4] – one can achieve collisionality $\nu^*$, $\beta$-values, and $q$-profiles similar to those envisioned for ITER and future reactors. Thus, extrapolating thermal confinement reduces to finding the exponent $n$ governing the $\rho^*$ dependence of thermal diffusivity. Bohm confinement scaling ($n = -1$) implies that the longest wavelength microinstabilities have their scale size set by the actual device size whereas gyroBohm scaling ($n = 0$) results when $(k)_{\text{min}}\rho_s$ depends only local nondimensional parameters such as $\nu^*$, $q$, or $T_e/T_i$.

2. $\rho^*$ and $\nu^*$ Scans

Two sequences of discharges were created in which all nondimensional parameters, except for $\rho^*$, remained close to constant[5]. Table 1 lists the parameters associated with two $\rho^*$ scans and an additional scan in which $\nu^*$ varied while $\rho^*$ and $\beta$ remained constant.

It is necessary that a large number of other nondimensional parameters such as $T_e/T_i$ as well as profile shapes $\alpha_{n_e}$, $\alpha_{T_e}$, and $q$ all remain constant throughout the scan. Otherwise, variations in confinement could not be unambiguously attributed to $\rho^*$ dependence. Satisfying all these criteria is not easily done experimentally. Figure 1 shows the variation of nondimensional parameters for the low-density $\rho^*$ scan. Except for
Table 1: Plasma conditions in the $\rho^*$ and $\nu^*$ scans.

<table>
<thead>
<tr>
<th>Scan</th>
<th>Shot</th>
<th>$B_T$</th>
<th>$I_p$</th>
<th>$\bar{n}_e$</th>
<th>$T_e(0)$</th>
<th>$\rho^*(0)$</th>
<th>$\nu^*_m\beta$</th>
<th>$P_b$</th>
<th>$\tau^*_b$</th>
<th>$Z_{\text{eff}}$</th>
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<td>0.47</td>
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<td>5.2</td>
<td>5.2</td>
<td>0.0038</td>
<td>0.06</td>
<td>0.46</td>
<td>22.7</td>
<td>81</td>
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<td>$\rho^*$ (high $\bar{n}_e$)</td>
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<td>1.42</td>
<td>0.59</td>
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<td>0.71</td>
<td>12.0</td>
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</tbody>
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FIG. 1. Dimensionless parameters in the low-density $\rho^*$ scan.
$T_e/T_i$ in the lowest $B$ member of the scan (shot 50921), constancy of
nondimensional parameters is adequate to assure the factors involving $\nu^*$
and $\beta$ in the right-hand forms of Eqs. (2-3) remain close to unity. It is
also important to restrict nondimensional scaling comparisons to regions
free of large scale modes such as sawteeth, which not only cause trans­
port, but also can redistribute energetic beam particles, altering the power
deposition profile. In DIII-D, the sawtooth radius exceeds 50% of the
minor radius\[6\] and their repetition time is comparable to the beam ion
slowing-down times[7]. Hence sawteeth may complicate analysis of DIII-
D nondimensional scaling studies[8–10]. Our analysis is confined to the
region $0.29 < r/a < 0.81$, which is free of sawtooth effects and where
temperature and density measurements have adequate accuracy.

Our principal results are expressed in terms of nondimensional power
flows through a magnetic surface

$$q_{\text{Bohm}} = \frac{Q(r)eB}{32\pi^2n_eT_e^2ca} \left[ \frac{(a^2 - r^2)^2}{r^3R} \right] \propto \frac{Q(r)(\nu^*)^{1/3}}{B^{5/3}a^{4/3}Z_{\text{eff}}^{1/3}\beta^{4/3}} \quad (2)$$

$$q_{\text{GyroBohm}} = \frac{Q(r)e^2B^2}{32\pi^2\sqrt{2n_eT_e^{5/2}M_i^{1/2}c^2}} \left[ \frac{(a^2 - r^2)^2}{r^3R} \right] \propto \frac{Q(r)(\nu^*)^{1/2}}{B(aZ_{\text{eff}})^{1/3}\beta^{3/2}} \quad (3)$$

Here $Q(r)$ represents the thermal power flow through a magnetic surface, as
computed by power deposition calculations, and the remaining factors form
the normalized power flow with Bohm and gyroBohm scaling respectively.
The factor in brackets is an ad-hoc minor radius variation whose role is to
reduce the numerical range of the normalized $q$'s and thereby improve the
clearly of graphical presentations.

Figures 2 and 3 present the principal results of the $\rho^*$ and $\nu^*$ scans.
One should compare nondimensional power flow at a specific minor radius
for various scan members. For example, if $q_{\text{Bohm}}$ does not change during the
course of the scan at a particular minor radius, then one concludes Bohm
scaling governs power flow for that minor radius value. In interpreting
Figs.2 and 3, the fundamental issue is to ascertain the overall systematic
trend from weak field (high $\rho^*$) to strong field (small $\rho^*$). It is clear that
Bohm scaling is superior in collapsing nondimensional power flows to a
single curve. Thus, our results indicate L-mode confinement in TFTR
follows Bohm scaling. Bohm scaling of local heat transport has also been
reported by DIII-D[8,9] and JET[11], although Waltz[10] has subsequently
argued that Bohm scaling is difficult to distinguish from gyroBohm scaling
if transport is governed by critical temperature gradients.

It is interesting that ITER-P global confinement scaling, when ex­
pressed in terms of a diffusivity, is

$$\chi^\text{ITER-P} = (\text{constant})^\frac{T_e}{eB} \left( \beta^2\nu^* \right)^{\frac{1}{2}}. \quad (4)$$
FIG. 2. Normalized power flow in the $\rho^*$ scan, showing that the relative variation of transport is better described by Bohm than by gyroBohm scaling.

FIG. 3. Bohm-normalized heat flow versus minor radius for four discharges in the $v^*$ scan which span a factor of five in $v^*$ at constant $\rho^*$ and $\beta$. 
In brief, ITER-P follows Bohm scaling with an additional weak deterioration with collisionality and $\beta$. A sequence of 4 discharges spanning a factor of nearly five in $\nu_e^*$ was prepared at constant $\rho^*$ and $\beta$ to determine the dependence of heat flow on collisionality. The results are summarized in Fig. 3. Heat flows normalized to the Bohm expression (Eq. 2) essentially collapse onto one another, varying less than 15-20% across the entire scan. Thus, the $\nu_e^*$ scan indicates that nondimensional power flows have a very weak collisionality dependence in accord with Eq. (4). Our results are best fit by a diffusivity independent of $\nu_e^*$, but error bars do not exclude the $(\nu_e^*)^{1/4}$ dependence of ITER-P.

3. Beta scan

A scan which holds $\rho^*$ and $\nu_e^*$ fixed while permitting changes in $\beta$ requires that $n_e$ and $T$ vary according to

$$
T \propto (B^2a^2/M_i)^{\rho^*2} n_e \propto (\nu_e^*\rho^*)^4(B^4a^5/M_i^2Z_{eff}^2) \quad (5)
$$

It follows that $\beta \propto \nu_e^*\rho^*6(B^4a^5/M_i^2Z_{eff})$. The Bohm expression for thermal diffusivity ($\chi_{\text{Bohm}} \sim cT/eB$) can then be expressed as

$$
\chi_{\text{Bohm}} \propto \rho^*^{3/2} (\frac{\beta}{\nu_e^*})^{1/2} (\frac{a^3Z_{eff}}{M_i})^{1/2} W(\beta, \nu_e^*, \rho^*, q, s, T_e/T_i, \ldots) \quad (6)
$$

where $W$ is an unspecified function of dimensionless parameters. Thus in a beta scan at constant $\rho^*$ and $\nu_e^*$, we expect an intrinsic – but weak – increase of pure Bohm transport with $\beta$, plus whatever dependence is contained in $W$. As shown in Eq. (4), ITER-P global $\tau_E$ scaling can be reconciled with a local Bohm diffusivity if one postulates an additional deterioration of magnitude $W \propto \beta^{1/2}\nu_e^*^{1/4}$, which yields $\chi_{\text{ITER-P}} \propto \rho^*^{3/2} \beta^{3/4}$. Understanding the transport scaling with $\beta$ is less important than the $\rho^*$ scaling for extrapolating observed tokamak confinement times to ITER, because the $\beta$ in current tokamaks are already comparable to those envisioned for ITER. However, an intrinsic dependence of transport on $\beta$ would be of theoretical interest because it would suggest that electromagnetic effects may be involved in the tokamak transport mechanism. Note that the $\beta$ scan cannot provide a basis for discriminating between Bohm and gyroBohm transport mechanisms because they share a common $\beta$ scaling ($\chi_{\text{gyroBohm}}/\chi_{\text{Bohm}} = \rho^*$).

3.1. Experimental conditions

A sequence of 5 discharges spanning a factor 4.5 in $\beta$ at constant $\rho_e^* \propto \sqrt{T_e}/B$ and approximately constant $\nu_e^* \propto n_eZ_{eff}/T^2$ was obtained by appropriate choice of heating power and density in a toroidal field scan.
Table 2: Plasma conditions in the $\beta$ scan. $r_{st}$ is the sawtooth inversion radius measured by a soft x-ray camera. $\ell_*/2$ is the plasma inductance inferred from diamagnetic and equilibrium measurements. Shift is the Shafranov shift computed for the plasma equilibrium. $\rho^*$ and $T_e/T_i$ are evaluated at $r/a = 0.5$. *In ohmic discharges $T_i(r)$ was determined from a transport simulation using a neoclassical model for $\chi_i$ matched to the measured neutron emission.

<table>
<thead>
<tr>
<th>Shot</th>
<th>$B_t$</th>
<th>$I_p$</th>
<th>$\beta_t^{\text{dia}}$ (%)</th>
<th>$\rho_p^{\text{dia}}$</th>
<th>$\ell_*/2$</th>
<th>$r_{st}$</th>
<th>Shift</th>
<th>$P_0$</th>
<th>$n_0$</th>
<th>$T_e(0)$</th>
<th>$\ell_p$</th>
<th>$T_e/T_i$</th>
<th>$\rho^*$</th>
<th>$\nu_{\text{MIN}}^*$</th>
<th>$Z_{\text{eff}}$</th>
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<td>0.108</td>
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<td>2.4</td>
<td>2.0</td>
<td>3.3</td>
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<td>1.70</td>
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<td>4.07</td>
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<td>0.0026</td>
<td>0.064</td>
<td>1.7</td>
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</tbody>
</table>

FIG. 4. Dimensionless parameters for the $\beta$ scan. $\rho_*$ is plotted in preference to $\eta_e$ because it suffers from less statistical scatter ($\eta_e = \eta_t$).
Assuming ion-electron temperature equilibration and neglecting the dilution of ion density by impurities, one finds the required conditions scale as: \( T \propto \beta^{0.5} Z_{\text{eff}}^{0.5}, \ \bar{n}_e \propto \beta, \) and \( B_t \propto \beta^{0.25} Z_{\text{eff}}^{0.25}. \)

For planning purposes, an L-mode expression for \( \tau_E \) predicts that the required heating power scales as \( P_{\text{heat}} \propto \beta^{2.5} Z_{\text{eff}}^{0.5}. \)

The sequence was performed in neutral beam-heated, high-recycling plasmas contacting the inner bumper limiter \((R = 2.55\, \text{m}, a = 0.89\, \text{m}),\) with broad density profiles \((F_{\text{ne}} = n_e(0)/(n_e) = 1.6 - 1.9).\) Plasma current was scaled linearly with \( B_t \) to maintain constant \( q_{\text{shaf}} = 4.4.\) The lowest-\( \beta \) condition had ohmic heating only, and consequently had a higher temperature ratio \( T_e/T_i \) than the other scan members. The beam-heated plasmas were strongly dominated by auxiliary heating \((P_b/P_{\text{OH}} = 3 \text{ at } P_b = 2.4\, \text{MW}; > 10 \text{ for discharges at higher power}).\)

Throughout the sequence, the fast ion content was modest \((n_{\text{beam}}/n_e \leq 0.15, W_{\text{beam}}/W_{\text{tot}} \leq 0.33),\) and both the radiated power fraction \((15-25\%)\) and the impurity content \((Z_{\text{eff}} = 1.5 - 1.8)\) remained low.

The observed global \( \tau_E \) remained about 20-30\% above L-mode projections [12]. Both toroidal and poloidal beta remained small \((\beta_t < 0.006, \beta_p < 0.6)\) throughout the scan, far from the Troyon beta limit.

Plasma conditions for the \( \beta \) scan are summarized in Table 2, and the corresponding profiles of dimensionless gradients and ratios are illustrated in Fig. 4. As desired, \( \rho^* \) was held quite constant across the entire scan. The collisionality was less well controlled \((\text{factor } 1.8 \text{ variation}),\) but this is acceptable given the observed insensitivity of transport to \( \nu^* \). There was a weak systematic trend toward flatter density profiles with increasing \( \beta \) \((F_{\text{ne}} = 1.9 \to 1.5)\) due to the much higher \( \bar{n}_e \) at high \( \beta.\) With the exception of the ohmic discharge \((T_e/T_i \approx 1.5),\) there was a weak systematic variation of \( T_e/T_i \) \((0.8 \to 1.2)\) with \( \beta.\) All of the plasmas remained well away from the supershot regime \((T_e/T_i < 1/3)\) where strong reductions in transport have been reported previously [13].

Current-ramp experiments in TFTR[14], JET[11], DIII-D[15], and ASDEX[16] have demonstrated that \( j(r) \) is an important parameter governing confinement \((\tau_E^{\text{tot}} \propto \ell^2 \text{ in TFTR L-mode plasmas}).\) Thus, uncontrolled variations in \( j(r) \) during the sequence would compromise our ability to deduce the underlying \( \beta \) scaling. Fortunately, the measured plasma inductance and sawtooth inversion radius remained essentially constant throughout the sequence (Table 2), implying no significant changes in \( j(r) \). This was the expected result given the constancy of \( q_{\text{shaf}} \) and the small contribution \((< 21\%)\) of bootstrap and beam-driven currents in these plasmas.

### 3.2. Transport analysis

The variation of thermal energy confinement with \( \beta \) is shown in Fig. 5(a). \( \tau_E^{\text{th}} \) was evaluated from Thomson scattering or ECE measure-
ments of $T_e(R)$, 10-chord interferometer measurements of $n_e(R)$, charge-exchange recombination measurements of $T_i(R)$ and $\nu_a(R)$, and visible Bremsstrahlung measurements of $Z_{eff}$, which is assumed radially constant. Concentrating first only on the end points of the scan, we observe that $\tau_E^{th}$ decreased by a factor 3 as $\beta$ increased a factor 4.6, implying an average deterioration as $\sim \beta^{-0.74}$ across the scan. This is precisely the behavior expected for ITER-P scaling, or equivalently, to Bohm scaling with an additional $\beta^{1/2}$ deterioration. Thus, in a coarse sense, this scan reconciles the observation of Bohm-like local transport scaling (deduced from the $\rho^*$ scan) with global ITER-P scaling of $\tau_E$, by confirming that there is a

FIG. 5. (a) Global thermal energy confinement time for the $\beta$ scan calculated from density and temperature profile measurements. There was good agreement between kinetic and magnetic measurements of $E_m$, except for the plasmas at $\beta_T = 0.18\%$ discharges, for which the kinetic analysis reproducibly yields larger stored energies than magnetic analysis. Error bars represent the difference between the kinetic and magnetic measurements. (b) Normalized power deposition profiles for plasmas in the $\beta$ scan.
FIG. 6. Bohm-normalized heat flow at three minor radii as a function of volume-average toroidal beta.

deterioration of \( \chi \) with \( \beta \), of approximately the correct strength, at fixed \( \rho* \) and \( \nu* \).

A more detailed scrutiny of the data suggests that the variation of local transport with \( \beta \) may be more complicated than a simple power law. Note that \( \tau_E^{th} \) decreased rapidly with \( \beta \) through the three lowest \( \beta \) members of the scan, then decreased more slowly as the heating power was increased beyond 7 MW. In addition, we must consider the effect of systematic changes in the beam power deposition profile, which becomes less centrally weighted at higher density [Fig. 5(b)]. Calculations of the "regional" confinement time[17] over the confinement region \( (r = 0.2a-0.7a) \), which correct for differences in the beam deposition profile, are virtually identical at 7 MW and 24 MW, implying that the radially-averaged thermal diffusivity remained constant.

This volume-integrated analysis is supported by local analysis of the normalized power flows. Since the Bohm and gyroBohm expressions for heat flux differ by only a factor \( \rho^* \), which was held constant in this sequence, it suffices to consider only a plot of Bohm normalized heat flow. As shown in Fig. 6, \( q_{Bohm} \) increases substantially through the first three data points, then remains remarkably constant as \( \beta \) is raised an additional factor of 1.9. Interestingly, the \( q_{Bohm} \) curves at three different minor radii have roughly the same shape despite the use of volume-average beta as the \( x \)-axis. We comment that this behavior would be inconsistent with a
transport mechanism that involves a threshold in local beta, since local beta is a factor of ~2.5 lower at r/a = 2/3 than at r/a = 1/3, unless the threshold itself varies with radius.

An unresolved issue is whether variation of $\tau_E$ and $q_{\text{Bohm}}$ with $\beta$ shown in Figs. 5 and 6 is a quasi-continuous deterioration or a bifurcation into two regimes: a very low-$\beta$ regime ($\beta \leq 0.25\%$) in which transport increases strongly with beta, followed by a regime at slightly higher beta in which transport is independent of or only weakly dependent upon beta. A related question is whether the Bohm-normalized transport will increase with beta as it is increased beyond the bound of this experiment (0.5%). We comment that the constancy of $q_{\text{bohm}}$ at power levels above 7 MW may be an artifact of variations in the density profile shape; $L_n$, increased by 25-60% between the 7 MW and 24 MW shots. If heat transport is driven by density gradients as well as temperature gradients, this would have the effect of improving the $\tau_E$, thereby masking a possible deterioration with beta.

### 3.3. Discussion

Beta-enhanced transport over the electrostatic estimates can in principle arise from the non-electrostatic component of the perturbed $E \times B$ drift and to the effect of $B$, where particles traveling along the perturbed magnetic field lines are carried across equilibrium magnetic surfaces. However, there are at present no predictions for strong $\beta$-threshold type behavior at the very low values of $\beta$ explored in this experiment. There are effects of finite $\epsilon \beta_p$ on the equilibrium, as well as corrections to the combined curvature and $\nabla B$ drifts, but these are also expected to be significant only for values of $\epsilon \beta_p$ much higher than those in this experiment[18,19]. Thresholds for island overlap and magnetic stochasticity applicable to realistic tokamak conditions have yet to be developed.

The unexpectedly complicated variation of observed normalized heat flow with $\beta$ suggests the possibility that a threshold for exciting a new microturbulent mode was exceeded as $\beta$ approached 0.2%. Given perfect control (i.e. constancy) of all other dimensionless parameters in the $\beta$ scan, one would conclude that $\beta$ was causally related to the threshold and to the increased transport. Naturally, the other dimensionless parameters were not perfectly constant throughout the scan, so an inference of a "$\beta$-scaling" of transport from the $\beta$-scan presumes that small variations in the other dimensionless parameters cause only small variations in the transport. If the controlling transport mechanism involves a threshold, above which transport increases rapidly, then small variations in the applicable dimensionless parameters would result in large increases in transport. This
would frustrate meaningful single-parameter dimensionless scans, since the required tolerance on the other dimensionless parameters would be difficult to achieve experimentally\cite{10}. A particular consideration for the plasmas in the $\beta$ scan is their proximity to the threshold for driving ITG turbulence ($\eta_i \sim 1 - 2.5$, with a weak trend – comparable to measurement uncertainty – toward increasing $\eta_i$ with increasing $\beta$). However, rapid increases in transport are not observed when high-temperature TFTR plasmas are transiently pushed well beyond the threshold for driving ITG turbulence\cite{20,21}. This result suggests that the small variations in $\eta_i$ which occurred in the $\beta$ scan were not responsible for the observed factor $\sim 3$ increase in transport. It remains to be established whether or not some other threshold was exceeded as $\beta$ approached 0.2\%. Data from $\beta$ scans in other plasma regimes would be particularly helpful in addressing this issue.

4. Comparison with RLW transport model

L-mode confinement in JET has been successfully modelled with the Rebut- Lallia-Watkins critical temperature gradient model\cite{22}. When cast in terms of nondimensional parameters, RLW is fundamentally a gyroBohm confinement model\cite{5} and therefore we expect the systematic trends of the $\rho^*$ scans to be at variance with the RLW model predictions. To provide the most definitive comparison, temperature profiles in the $\rho^*$, $\nu^*$, and $\beta$ scans have been predictively simulated in steady-state using the model’s transport coefficients. The heating and particle source profiles, particle fluxes, and $q(r)$ were calculated by the time independent data analysis code SNAP\cite{23} using the measured density and temperature profiles, $Z_{\text{eff}}$, beam and wall neutral source rates, and other diagnostic information. The current profile calculation assumed equilibration with neoclassical resistivity and included small contributions from the beam and bootstrap currents. The RLW thermal diffusivities, ion-electron power coupling, and convection ($q_{j-\text{conv}} \equiv \frac{5}{2} n_j T_j$) were evaluated self-consistently using the predicted temperature profiles. (Additional simulations for which the particle flows were calculated from the model prescription $D_i = 0.7\chi_i$ and the measured density gradient generally predicted temperatures that differed from the measured temperatures by a slightly greater amount than simulations which used the particle fluxes determined from SNAP’s particle source calculations).

Figure 7 compares the calculated and measured temperatures at a central position ($r/a = 0.2$) near the sawtooth inversion radius, which avoids uncertainties introduced by the time-varying $q(r)$ profile inside the sawtooth region. As expected from the gyroBohm character of the
RLW transport model versus the inferred Bohm behavior of the measured transport, RLW overpredicts the temperature improvement expected with increasing $B_T$ in the low-density $\rho^*$ scan by about 30% for both ions and electrons as $\rho^*$ is decreased a factor of 1.8. The RLW model is quite successful in predicting the observed electron temperature profiles in the $\beta$ scan (within 10%), but systematically underpredicts $T_e$ by up to 30% in the $\nu^*$ scan. Overall, the RLW model is considerably less accurate in predicting $T_i$ than $T_e$ in these scans. The ratio of calculated to measured $T_i(r/a = 0.2)$ varies by almost a factor of 2.5 across the scans (from 0.66 to 1.63). Note that some of the discharges with significant differences between measured and calculated $T_i$ were accurately predicted in $T_e$, suggesting that the experimental $\chi_i/\chi_e$ has parametric dependencies which are not incorporated into the RLW model [which uses $(\chi_i/\chi_e)_{RLW} = 2\frac{T_e}{T_i} \frac{Z_i}{\sqrt{1+Z_{eff}}}$.]

5. Summary and Projections to ITER

Scans of dimensionless parameters $\rho^*$, $\nu^*$, and $\beta$ have been carried out in TFTR L-mode plasmas to elucidate the dependence of transport on
Table 3: Bohm scaling of DIII-D and JET results to ITER. ITER parameters are representative of the technology phase. $I_p$ has been scaled according to $I_p \propto B_0 a^2/R$. Tokamak data are those of DIII-D pulse 72220 and JET pulse 22490.

<table>
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<th>TOK</th>
<th>$B$</th>
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<th>$a$</th>
<th>$I_p$</th>
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<th>$T_{ee}$</th>
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each variable. Our principle findings are that local confinement in TFTR exhibits a Bohm-like scaling when collisionality and $\beta$ are held nearly fixed and that the measured Bohm-normalized power flow does not depend on collisionality. This implies that transport is dominated by long-wavelength fluctuations which scale with system size[5]. A subsequent study of transport scaling with $\beta$ has observed a substantial increase of $\chi$ with $\beta$ at fixed $\rho^*$ and $\nu^*$, beyond the $\sim \beta^4$ dependence implicit in Bohm transport. The magnitude of the increase in transport is of approximately the correct magnitude to reconcile a Bohm-like scaling of local thermal diffusivity with ITER-P scaling of global $\tau_E$. However, the functional dependence of Bohm-normalized heat flow with $\beta$ appears to differ from a simple power-law, and may involve thresholds in $\beta$ or other nondimensional parameters.

Since the TFTR $\rho^*$ scans are adequately described by Bohm scaling of local heat transport, it is useful to consider what Bohm scaling predicts for the performance of future devices ITER (Technology Phase)[24]. Table 3 describes an extrapolation to ITER from two H-mode discharges in JET and DIII-D notable for their high $n_{de}(0)T_i(0)\tau_E$ values, at constant $\nu^*$, aspect ratio, and elongation. Since a strong confinement sensitivity to $\beta$ was observed in the TFTR L-mode $\beta$ scan, it is essential that the extrapolation be performed at constant $\beta$ also. Ignition requires $n_{de}(0)T_i(0)\tau_E \geq 60$ ($10^{20} m^{-3} keV sec$). ITER just does attain it under Bohm scaling. GyroBohm extrapolations for $\tau_E$ and $n_{de}(0)T_i(0)\tau_E$ would be more favorable by factors of the inverse $\rho^*$-ratio, which has a large value (1/$\rho^*$-ratio $\approx 3-5$) for ITER. Since the Rebut-Lallia-Watkins transport model is a particular form of gyroBohm scaling, a similar, favorable ignition margin $Q$ (close to 4) results from RLW scaling for the JET extrapolation. The Bohm-scaled projections of $\tau_E$ for ITER correspond to the ITER-P L-mode $\tau_E$ scaling expression[25] with a multiplier of 1.9 for the DIII-Dscaled discharge and 2.5 for the JET-scaled discharge.
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FORMING A ‘PERFECTLY’ UNIFORM SHELL OF SOLID D–T FUSION FUEL BY THE BETA LAYERING PROCESS

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Abstract

FORMING A ‘PERFECTLY’ UNIFORM SHELL OF SOLID D–T FUSION FUEL BY THE BETA LAYERING PROCESS.

Optical measurements carried out with a high resolution CCD video camera show that the solid D–T layer formed by the ‘beta layering’ process in an isothermal cylindrical bore of 2000 μm diameter may meet the strict criteria for layer thickness uniformity required by implosion considerations (~ 1% of the average layer thickness). Measurements are reported on preliminary experiments in which a 75 μm thick solid layer was formed. Although the solid D–T layer on the bore equilibrated with the expected rate constant of ~ 30 min, the layers on the optical windows were not optically smooth after more than 16 h, leading to slight optical aberrations in the final image.

1. INTRODUCTION

The energetically most favourable initial configuration for an inertial confinement fusion (ICF) target requires a solid cryogenic fuel [1] confined to a solid layer on the inside of the target shell. The recent experimental demonstration [2] of the beta layering effect now provides a practical, virtually automatic method of achieving these starting conditions, without resorting to the use of additional foam or annular shells.

Beta layering refers to the self-redistribution of solid deuterium–tritium (D–T) which takes place inside a chamber cooled to be below the triple point. Because of self-heating arising from the beta decay of tritium, thicker layers of solid D–T tend to have warmer interior surfaces than thinner ones. A net sublimation of material
from the thicker to the thinner sections takes place at a reasonably rapid rate. Inside an isothermal enclosure, this results in the development of a D–T shell of nearly perfectly uniform thickness.

Recently, we have reported measurements of the rate of the beta layering effect, in both cylindrical and spherical geometries, using container materials of both high and low thermal conductivity, and using both fresh and aged D–T (where the presence of \(^3\)He can slow down the process dramatically) [3, 4]. In general, our experiments are in good agreement with the existing theory [4, 5]. However, before beta layering can be utilized for the production of actual ICF targets, several outstanding issues on the physics of the phenomena must be resolved, such as whether the interior surface of the beta layered shell is smooth on a scale of 100 to 1000 Å and whether or not the microscopic \(^3\)He bubbles which form in the solid following beta decay (or other radiation damage effects) give rise to density defects.

We are carrying out several independent types of measurement to address these issues, including resonant ultrasonic spectroscopy, neutron scattering and neutron reflectivity measurements. Here, however, we wish to report on optical measurements made with a high resolution CCD video camera to address the issue of whether or not the residual anisotropy of a 100 \(\mu\)m thick beta layered shell is less than 1%, i.e. on a scale of \(\sim 1 \mu\)m.

1. EXPERIMENTAL APPARATUS

To avoid the optical complications caused by spherical focusing, our current target geometry is cylindrical. A 2000 \(\mu\)m diameter hole has been precision bored through a 4 mm wide copper block. Each side of this hole is counterbored 1 mm deep to a diameter of 4 mm and sealed with a sapphire window using an indium gasket. In this way, the small fillet of solid D–T that typically forms at the junction of the sapphire window and the copper block does not obscure the view of the solid D–T layer at the edge of the 2000 \(\mu\)m bore. The target cylinder is mounted in a refrigerator cooled optical cryostat and partially filled with liquid D–T at 21 K through a small capillary. Beta layering begins as the target is cooled below the triple point at 19.79 K [6] and proceeds with a time constant of \(\sim 30\) min (for fresh D–T, i.e. when no \(^3\)He is initially present). The high thermal conductivity of copper ensures an exact isothermal outer boundary to the solid D–T. Therefore, the final layer thickness uniformity should not be affected by thermal gradients in the support structures or thermal convection currents in the exchange gas. The intent here is to study the inherent limitations of beta layering, such as might be caused, for example, by gravitationally induced fractionation of \(D_2\), \(T_2\), and D–T molecules during the vapour phase transport. Note that the use of sapphire ensures that the entire interior surface of the target is isothermal. Therefore, equal thicknesses of D–T form on the front and back sapphire windows and, for an accurate observation of the solid D–T
layer on the edge of the 2000 \( \mu \text{m} \) bore, light must pass through these layers without significant distortion.

A long working length microscope\(^1\) is used to magnify the 2000 \( \mu \text{m} \) bore, so that the image fits within the 1024 \( \times \) 1024 pixel CCD chip in the camera head\(^2\). Following an exposure, the camera electronics read out the data with 12 bit resolution. Each image thus requires more than 2 Mbytes of computer disk storage space.

3. DATA ANALYSIS

The image data are analysed by using a series of C-language routines by first defining a series of radial lines emanating from the target centre. Along each line, the intensity values of the corresponding image pixels are smoothed, fit analytically and differentiated. Local minima then are located which correspond to the inner and outer edges of the solid D–T layer.

Before D–T is added to the target cylinder, the edge of the 2000 \( \mu \text{m} \) bore is quite distinct and can be located to a precision of approximately 0.4 pixels, which corresponds to a distance of 1 \( \mu \text{m} \) for the image discussed here. Since the bore was accurately machined and is cylindrical to \( \pm 0.03 \mu \text{m} \), the value of 1 \( \mu \text{m} \) represents the optimum resolution of our present optical method. After adding D–T and allowing beta layering to symmetrize the layer on the inner edge of the 2000 \( \mu \text{m} \) bore, the copper D–T solid boundary is slightly less sharp, primarily owing to slight irregularities in the curvature of the layer as it folds over this edge and also to the grain boundary structure in the D–T solid. The D–T solid–vapour boundary, i.e. the inner edge of the solid D–T layer, is quite sharp, and therefore the D–T layer radial thickness itself is measured by relating the inner edge of the solid D–T layer to the optically measured average internal radius of the 2000 \( \mu \text{m} \) bore.

4. EXPERIMENTAL RESULTS

The high resolution optical apparatus was only recently completed, and thus we present here the results of our first experiment, where the amount of D–T added to the target produced a uniform layer only 75 \( \mu \text{m} \) thick. When we began the experiment, we naively expected that the solid D–T layers forming on the sapphire windows would equilibrate with the same time constant as we normally observe when using fresh D–T in an open spherical or cylindrical geometry, i.e. \( \sim 30 \text{ min} \) \[2, 3\]. We did not, however, observe clear images of the 2000 \( \mu \text{m} \) bore until over 12 h had elapsed. It became evident that while the solid D–T layer on the

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\(^1\) Questar model QM-100, Questar Corp., New Hope, Pennsylvania, USA.

\(^2\) Photometrics model CH250, Photometrics, Ltd., Tucson, Arizona, USA.
FIG. 1. Analysis of inner and outer edges of 75 \( \mu m \) thick solid D-T layer following 'beta layering' on the interior of an isothermal cylindrical bore of 2000 \( \mu m \) diameter. The outer boundary is the bore wall itself and is shown here as analysed from an earlier image of the empty cylinder. The inner boundary is the D-T solid/vapour interface, which appears quite smooth in the CCD image from which this data were analysed. The numbers on the radii refer to micrometres from the centre of the cylinder. The thickness of the layer, as analysed by comparing the inner boundary to the known fixed radius of the bore, is uniform to within \( \pm 5.3\% \). The actual uniformity may be considerably better than this because our final images were slightly distorted by irregularities in the solid D-T layers on the sapphire windows (see text).

2000 \( \mu m \) bore did equilibrate with the normal time constant, the 4000 \( \mu m \) diameter window layers, forced to communicate through a constriction, equilibrate much more slowly. Although our final image was taken more than 16 h after freezing the D-T, slight imperfections could still be seen in the window layers when focusing the camera at the window surfaces which could lead to slight aberrations in the image of the 2000 \( \mu m \) bore. Nonetheless, the resulting layer appears to be very uniform. Fearing that the high resolution image itself would not reproduce well, we have chosen instead to show the results of the analytical analysis of the D-T layer boundaries in Fig. 1. As mentioned above, the outer boundary, i.e. the copper/D-T boundary at the edge of the 2000 \( \mu m \) bore, is not as distinct in the final image as it is when the target is empty. Hence, the outer boundary data from the final image are only used to locate the centre of the bore. In Fig. 1, the outer set of points represents the bore edge as analysed from an earlier image of the empty cell.
When viewing the final image on the computer screen, the D-T solid layer thickness appears to be quite uniform and the inner boundary appears to be relatively smooth. However, the analysis shows that the solid D-T layer thickness varies from uniformity within ±4 μm, or ±5.3% of the average layer thickness. The question of whether this is inherent to the beta layering process or simply due to not having waited long enough for the window layers to become completely free of optical defects is the subject of our ongoing experimental efforts.

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PRODUCTION AND CHARACTERIZATION OF ICF CAPSULES

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Abstract

PRODUCTION AND CHARACTERIZATION OF ICF CAPSULES.

The construction and characterization of capsule targets that are used in direct drive ICF experiments are discussed. Construction includes three independent polymer layers that allow for the incorporation of dopant atoms for use as spectroscopic probes. Production of smooth and characterizable capsule surfaces is essential for the performance and interpretation of ICF experiments.

1. Introduction

The success of ICF experiments depends heavily on our ability to produce target capsules that meet very demanding specifications with respect to composition and fill, symmetry and size, and surface finish. The variety of target capsules produced is broad, since experiments aimed at elucidating different implosion characteristics often require capsules with markedly different compositions, fills, or surface finishes. For example, an investigation of hydrodynamic instability and mix requires capsules with controllably varied surface roughness characteristics. Spectroscopic diagnosis of time dependent implosion density and temperature requires that the inner wall of the capsule and/or the fuel be doped with atomically dispersed higher Z atoms. We have developed a flexible and continually evolving route to target capsule production and subsequent characterization that allows us to best meet the experimental requirements.

2. Capsule Construction

The construction of a direct drive ICF target capsule starts with a sub-millimeter sized plastic microshell which serves as a mandrel and inner wall for the completed capsule. These microshells are
prepared using drop tower techniques that have been described elsewhere [1,2]. Briefly, small droplets of polymer solution are dropped down a heated column. As the solvent evaporates a shell forms. Further evaporation causes the shell to grow to about 0.5 mm in diameter with a wall thickness of a few μm. For those targets requiring high Z doping on the inner capsule wall for spectroscopic diagnostics[3], we form the microshells from a polymer that has the desired dopant atoms covalently incorporated into its molecular structure. Our current capabilities include Cl, Cr, Fe, Br, and I doped polymer systems at levels up to 1 atom %. The Cl-doped shells are formed from a mixture of polystyrene and commercially available poly(p-chlorostyrene). The Cr-doped shells are formed from a partial -Cr(CO)₃ derivative of monodisperse polystyrene.[4] The Fe-doped shells are formed from a copolymer of styrene and vinylferrocene.[5] The polymer used in either Br- or I-doped shells is partially para-brominated or iodinated monodisperse polystyrene.[6]

The microshells are then coated with 99% hydrolyzed polyvinyl alcohol (PVA) with a molecular weight of 25,000. PVA is used as a permeation barrier to contain high pressure D₂ and DT fuel.[7] For a 2 μm layer of PVA we find a half life for D₂ gas retention of 30 hours at 25 °C. The beta decay of tritium in DT fuel causes a significant increase in the permeability of the PVA layer. Therefore, it is important that the fill procedure used for DT pressurization employ a maximum pressure gradient, determined by the compressive strength of the capsule, to minimize the residence time of tritium in the PVA layer, and a low temperature to reduce the free radical decomposition of the PVA. With an optimized fill procedure, targets can be loaded with high pressure DT gas and will maintain a fuel retention time of half the original D₂ retention time as measured using a new shell. This half life degrades with time so that targets must be shot within a few days after DT fill.

The capsule ablator is then built up to the desired thickness using plasma polymerization deposition techniques.[8-10] Briefly, polymer deposition occurs in a region of glow discharge driven by a RF helical resonator. In this region, electrons collide with the gas molecules, causing fragmentation and ionization. The reactive fragments chemically recombine in the gas phase and on nearby surfaces to form a polymer coating. For hydrocarbon coatings we use two gases: trans-2-butene (T2B) and H₂. The T2B supplies the hydrocarbon from which polymer is made while the H₂ flow allows us to control the residence time in the discharge. The polymer can be
seeded with a range of other elements by bleeding in an appropriate gas. In this manner fluorine[8], silicon, chlorine[10], and bromine have been added to the polymer structure.

We have found that the surface finish of plasma polymer films can be controlled through the process conditions [9]. The flow rates of T2B and H₂ are particularly important. Increased T2B flow increases the deposition rate but results in rougher films. Increased H₂ flow reduces both coating rate and intrinsic stress, and improves the surface finish. We believe that the surface finish is determined by a combination of the concentration of organic gas in the discharge as well as the residence time of the gas passing through the discharge zone. A long residence time or a high organic gas composition produces particles in the plasma that become incorporated in the growing film and result in a rough surface. By using an organic gas diluted by a high H₂ flow, we retard particle formation and deposit plasma polymer by a mechanism of adsorption of activated molecules that then react to form the smooth film.

3. Capsule Characterization

Accurate characterization of the completed target capsule is essential so that experimental results can be interpreted in terms of theoretical models. Our two main concerns are capsule composition, including the D₂ or DT fuel pressure and possible fuel dopant concentrations, and capsule surface characteristics including sphericity.

Dopants, both in the fuel (Ar, Xe) and in the inner capsule wall (transition metals, halogens), are measured using a micro-focus x-ray fluorescence (XRF) instrument. With our system, photons from a thick-anode (Mo) x-ray tube are used to excite a sample. Incident beam diameters as small as 50 μm can be used, which often improves the signal to noise ratio by reducing the scattered x-ray intensity. The fluorescence x-rays are detected with a Si(Li) detector and a pulse height analyzer. The Be window on the detector limits the machine’s use to Al and heavier elements. The shells are supported between thin (2 μm) polycarbonate sheets during the measurement. The intensity of an emission line (or lines) appropriate for the element of interest is measured for each ball. The absorption by the shell of both the exciting and the emitted radiation is calculated for each shell, and the counts are adjusted accordingly. By comparison with a known standard, the amount of the dopant atom is calculated.
D₂ or DT fuel fill pressure is determined by visually observing the gas to liquid or gas to solid phase transition and carefully measuring the temperature of this phase transition. Using the equation of state as determined by Souers[11] and the ideal gas law we calculate the pressure of fuel inside the capsule at room temperature.

Capsule dimensions and sphericity are determined by a number of methods. Shells are radiographed, using high resolution photographic plates, and the resulting images are analyzed through a microscope-CCD camera-LSI 11 system. The x-ray energy is varied to suit the target; Cu K emission (8 keV) is frequently used. The images are analyzed to find voids or, more often, separation of inner layers, inner and outer diameters, and out of roundness. For targets whose coatings are not transparent, radiography is our sole measurement of the inner diameters of the final targets. However, most coatings are transparent, and microscopic interferometry is heavily used to measure outer diameter and wall thickness. This also reveals any thickness anomalies or asphericity.

Accurate characterization of the capsule surface finish is important in experiments aimed at developing our understanding of the growth of instabilities during the implosion. A relatively crude qualitative measure of the surface characteristics can be obtained optically; however, a more quantitative measure is obtained by scanning electron microscopy and atomic force microscopy.

The atomic force microscope is particularly well-suited to local surface height measurements, combining high sensitivity (less than 1 Å) with a wide vertical range (~4 μm). For the purpose of modeling Rayleigh-Taylor instabilities during the implosion of a capsule, spatial modes corresponding to 10-50 cycles around the circumference are of particular interest. For typical capsule dimensions, these are wavelengths of 30 to 150 μm. This precludes scanning a square patch with the sample held fixed, as would typically be done with an AFM. Because of the curvature of the sphere, a scan larger than ~60 μm exceeds the vertical range of the instrument. To overcome this limitation, we mount the capsule on a horizontal rotary air-bearing, with a stand alone AFM head held stationary above it. With the AFM tip contacting the surface and the control electronics maintaining constant force, the capsule is rotated. The surface height is tabulated as a function of rotation in increments of 0.1°. This procedure yields a 1-D profile around a circumference of the capsule. To characterize a particular sphere, many such scans are collected. For each scan, the
Fourier transform is calculated, from which the power spectral density (PSD) is derived. The PSD's are then averaged together, yielding an accurate measure of the contribution of each mode to the total surface roughness.

Preliminary data suggest that typical capsule surfaces have a very short correlation length, a few μm or less. At wavelengths longer than this, the PSD is flat, as would be obtained from spectral analysis of white noise. This result raises the possibility that the surface roughness could be characterized by analyzing a small patch on a capsule with the AFM and extrapolating the result to longer wavelengths (lower mode numbers). At present, this approach presents some practical difficulties, mainly because two dimensional AFM images usually contain significant distortions due to piezo non-linearity and hysteresis. These artifacts are difficult to remove, and can contribute a substantial amount of spectral power. Despite these problems, we feel that this could become a practical technique in the future, and we are continuing to evaluate it.

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References


MODELLING OF THE RELATION OF DIVERTOR PLASMA TO CORE PLASMA PARAMETERS IN THE HIGH RECYCLING REGIME OF A NEXT STEP DEVICE

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Abstract

MODELLING OF THE RELATION OF DIVERTOR PLASMA TO CORE PLASMA PARAMETERS IN THE HIGH RECYCLING REGIME OF A NEXT STEP DEVICE.

2-D modelling of the scrape-off plasma in a next step device (ITER CDA parameters) has resulted in scaling relationships between separatrix density and parameters at the divertor plate, notably peak power load and electron temperature. Separatrix density, electron temperature, ion temperature as well as power flow across the separatrix are the parameters used to link the scrape-off layer (SOL) calculation with a simulation of plasma core. In order to determine the compatibility of high-recycling divertor conditions with the desired ignited plasma operation, various particle pinch coefficients are used, the partition of power across the separatrix between electrons and ions is varied, and different sheath transmission coefficients are investigated. In addition, the effect of varying the angle between the divertor plate and the magnetic surfaces is examined. Generally, the desired core conditions are found to be compatible with low electron temperature solutions near the divertor plate.

Introduction

2-D modelling of the divertor region using the Braams B2 code with an analytical recycling model has yielded [1,2] a set of empirical scaling relations for the divertor plasma parameters of a double null next step device. These results, valid for the high-recycling regime and for electron temperatures at the divertor plate $T_{e,p}$ from 30 eV down to about 5 eV, show that the peak power per unit area at the outer divertor plate, $f_p$, is given by

$$ f_p(T_{e,p}) \propto \left( \frac{P_{SOL}}{S} \right)^{0.74} q^{0.03} \chi^{-0.27} R^{-0.12} T_{e,p}^{-0.18} $$

for given $T_{e,p}$. Note that throughout this paper, the power loads quoted are the values resulting directly from the 2-D model and do not include the peaking and safety factors discussed in [4]. The edge density (density at intersection of midplane and separatrix) is given by:

$$ n_s(T_{e,p}) \propto \left( \frac{P_{SOL}}{S} \right)^{0.74} q^{0.03} \chi^{-0.27} R^{-0.12} T_{e,p}^{-0.18} $$

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Here, $P_{\text{SOL}}/S$ is the power per unit area flowing across the separatrix, $D=\chi_i=\chi_e/3$ in the SOL is represented by $\chi$, $q$ is the safety factor, $R$ is the major radius, $\alpha_B$ is the angle of $B_{\text{tot}}$ to the divertor plate ($1.6^\circ$ for most cases considered here) and the aspect ratio is not varied. For the standard case discussed below, $\chi_e=2m^2/s$ is constant in the scrape-off layer, the total power flowing to one outer divertor plate is 46 MW, and the ratio of power flow into the scrape-off layer via electrons to that via ions is three to one.

To examine the compatibility of the edge conditions leading to low $T_{e,p}$ (favourable for the power load; see Eq.1), with core plasma parameters, the core plasma is then simulated using the ASTRA 1.5D [3] transport code. The temperature at the separatrix and the edge density, set to the same value as for the divertor simulations (i.e. $n_s$ according to Eq.2), are boundary conditions of the core calculation. For these simulations, heat and particle diffusion coefficients are constant radially, and conditions are adjusted to obtain steady-state conditions with a fusion power of approximately 1 GW, and feedback-controlled external heating power between 0 and 20 MW. The global energy containment times are 80 to 100 % of ELMy ITER-90 H-mode expectations [4]. The power flow across the separatrix results from the energy balance in the bulk plasma, taking into account the effect of core and edge radiation from impurities according to [4], and this power flow then is a self-consistent input to the divertor model. Neutral particle fuelling of the core plasma, expected to be unimportant compared to edge fuelling (boundary condition $n_s$), is not included.

**Variation of Divertor Plate Angle**

Variations in recycling strongly affect the divertor conditions. We use an analytical recycling model [10] including atoms and molecules without sideways neutral motion, but adjusting the mean free paths along the field for the angle of the
inclined divertor plate so that the correct projection perpendicular to the plate is obtained. To illustrate the effect of changing the recycling, we compare results for the standard ITER CDA divertor plate inclination (15° to \( B_{pol} \), 1.6° to \( B_{tot} \)) and for inclination perpendicular to \( B_{pol} \). We find that Eq. 1 is verified, i.e. at the same \( T_{e,p} \), the power load \( f_p \) would be larger by a factor ~4 due to the increase in \( \sin(\alpha_{pol}) \). This is offset to some extent by the higher recycling: the total volume source due to recycling is found to be about the same but recycling occurs in a smaller volume when the plate is perpendicular, leading to a higher density at the plate and a lower \( T_{e,p} \) for the same \( n_s \) (Fig. 1a). The lower \( T_{e,p} \) is beneficial but only compensates half the increase in \( f_p \) due to the reduced plate inclination (Fig. 1b).

Variation of Sheath Transmission Coefficient

It is to be expected that the scaling laws for power load on and for electron temperature at the divertor plate are sensitive to the boundary condition at the plate. We have investigated the effect that a lowering of the sheath transmission coefficient from its normally assumed value of 6.3 for \( T_e-T_e/3 \) [5] down to a value of 2.0 would have on divertor parameters in the high recycling regime. Indications for such a reduction have been found in some recent experiments [6], which were however in a lower recycling regime.

In the 2-D modelling, the major effect of reducing the sheath transmission coefficient, at constant power input into the scrape-off layer, is observed to be an increase of the electron temperature at the divertor plate. This is illustrated in Fig. 2a, where the temperature is plotted against the electron density at the separatrix. The power load on the divertor plate is little affected at a given separatrix electron density (Fig. 2b). To attain the same low electron temperature with the low sheath transmission factor, \( n_s \) would have to be 1.5 times as high, and the power

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**FIG. 2.** (a) \( T_e(T_{peak}) \) at plate; (b) power load \( f_p \) (from 2-D model) for sheath transmission factors \( \gamma = 6.3, 4.2, 2 \); closed symbols without, open symbols with flux limiter (factor 0.2).
load on the divertor plates would then be reduced by almost a factor of 2. A variation of the flux limiter factor [7] has shown these results not to depend strongly on the size of the flux limiter factor.

Variation of Ratio of Power Transmission between Electrons and Ions

The standard cases we have investigated previously [1,2,8] have always taken the power transported into the scrape-off layer via electrons to be three times as large as that transported via ions. To test the sensitivity of the results to this parameter, a series of calculations has been performed with the power being shared equally between electrons and ions. To be consistent, the transport coefficient $\chi_i$ has also been increased and set equal to $\chi_e=2 \text{ m}^2/\text{s}$ for these calculations. The particle diffusion coefficient $D_e$ remained equal to $\chi_e/3$. The variations of electron temperature and of power load then exhibit the same power law dependencies (equations 1 and 2) as for the standard case with $\chi_i=\chi_e/3$ (see Fig. 3, the lines are almost parallel on the log-log plot). Only the constant of proportionality is reduced, by a factor of 2.8 for the electron temperature at the plate at given $n_s$, and by 20% for the power load (Eq. 1, Fig. 3b) at given $T_{e,\text{p}}$. Transporting a larger fraction of the power by ions is therefore beneficial for both parameters.

Simulation of the Core Plasma

For the transport code simulations, the separatrix density is used as a boundary condition, and the density profile is then determined by the size of the particle pinch term. The particle diffusion coefficient $D_n$ is taken constant radially ($=\chi/r/3$). Two radial variations of the pinch term have been investigated: for the first, the pinch velocity is proportional to $D_n r/a^2$, in order to give reasonably peaked profiles; for the second, the pinch velocity is proportional to $D_n /a^2(r/a)^{10}$, to
concentrate the pinch in the outer 20% of the radius, leading to flat core density profiles (Fig. 4). Results of the simulations are listed in Table I in the order in which they are discussed in the text.

To obtain density profiles as specified in [4], with an average electron density of $1.2 \times 10^{20} \text{ m}^{-3}$ for a separatrix density of $3 \times 10^{19} \text{ m}^{-3}$, the pinch velocity must be rather large, $4.5 D_n r/a^2$. With a more reasonable pinch velocity (e.g. [9]), $3 D_n r/a^2$, an edge density of $3.5 \times 10^{19} \text{ m}^{-3}$ gives the same volume average density. The core density profile is fairly peaked (Fig. 4a; this condition approximates the standard divertor operating conditions in the 2D-modelling [1,2] with $T_{e,p}$ of 15 eV and $n_s=3 \times 10^{19} \text{ m}^{-3}$; the core density profile is somewhat more peaked than square root of parabolic in [4]), and the average core temperature, for $P_{\text{fus}}=1 \text{ GW}$, is 9.5 keV (Fig. 4b). Using the self-consistent $P_{\text{SOL}}$ found by the core simulation, $f_p$ (without safety and peaking factors [4]) is found to be 5.9 MW/m$^2$, to be compared with the 4.7 MW/m$^2$ in the stand-alone divertor runs. The reason for the discrepancy is the somewhat smaller power found to be radiated from the core and edge plasma. For a peaked density profile, the self-consistent core simulation thus yields parameters reasonably close to those of the standard 2D divertor simulation. By raising the edge density to $4.5 \times 10^{19} \text{ m}^{-3}$ the power load can be lowered to 3.2 MW/m$^2$ and the predicted temperature at the divertor plate is 3 eV, but the average core temperature would then be only 8 keV for $P_{\text{fus}}=1 \text{ GW}$.

When a flatter density profile (Fig. 4a), more appropriate for H-mode operation, with the same average density is simulated (pinch velocity of $6 D_n r/a^2 \times (r/a)^{10}$) for the same fusion power, impurity concentration and $n_s$, less power is radiated from the main and edge plasma, leading to a larger $P_{\text{SOL}}$. For $n_s=3.5 \times 10^{19} \text{ m}^{-3}$, $T_{e,p}$ would then be 20 eV, above the value 15 eV for the standard case. The divertor power load per unit area is correspondingly larger, 7 MW/m$^2$, due to the combined effect of larger $P_{\text{SOL}}$ and higher $T_{e,p}$. However, this can be

<table>
<thead>
<tr>
<th>Profile shape</th>
<th>$x_e/x_i$</th>
<th>$n_s$ ($10^{19} \text{ m}^{-3}$)</th>
<th>$\langle n_e \rangle$ ($10^{19} \text{ m}^{-3}$)</th>
<th>$\langle T \rangle$ (keV)</th>
<th>$f_p$ (MW/m$^2$)</th>
<th>$T_{e,p}(\text{Y peak})$ (eV)</th>
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<tbody>
<tr>
<td>Peaked</td>
<td>3</td>
<td>3.55</td>
<td>12.0</td>
<td>9.4</td>
<td>5.9</td>
<td>14.8</td>
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<tr>
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<td>4.5</td>
<td>14.1</td>
<td>8.0</td>
<td>3.2</td>
<td>3.0</td>
</tr>
<tr>
<td>Flat</td>
<td>3</td>
<td>3.55</td>
<td>12.5</td>
<td>10.0</td>
<td>6.9</td>
<td>19.6</td>
</tr>
<tr>
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<td>13.5</td>
<td>9.3</td>
<td>5.1</td>
<td>8.7</td>
</tr>
<tr>
<td>Flat</td>
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<td>4.6</td>
<td>15.5</td>
<td>8.2</td>
<td>3.2</td>
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</tr>
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<td>11.9</td>
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compensated by a slight increase in $n_p$ because of the strong dependence of $T_{e,p}$ on $n_s$ (Eq. 2). Thus, an edge density of $4 \times 10^{19} \text{ m}^{-3}$ results in a power load of 5 MW/m$^2$ at an electron temperature of 9 eV.

The preceding simulations were all carried out with $\chi_i = 3 \chi_e$, both in the edge and in the core. When $\chi_i$ is set equal to $\chi_e$ everywhere, a lower edge density is required to give the same electron temperature, 15 eV, at the divertor plate, and the peak power load on the plate is less at this temperature. For the peaked density profile, $n_s = 3 \times 10^{19} \text{ m}^{-3}$ gives $T_{e,p} \approx 15 \text{ eV}$ and a power load of only 5 MW/m$^2$. For the flatter profile, obtained as described above, the corresponding values are $3.2 \times 10^{19} \text{ m}^{-3}$, and 5.3 MW/m$^2$. Because of the lower edge density, the operating conditions of the core plasma are shifted to a lower density, higher temperature (11-12 keV) operating point if $\chi_i = \chi_e$ than if $\chi_i = \chi_e/3$. Note that the core transport coefficients are adjusted so that global confinement is equal for the two cases, but that $\chi_e$ in the edge is still taken to be 2 m$^2$/s.

A lower sheath transmission factor requires a higher edge density to obtain the same low electron temperature at the divertor plate. The core operating condition for a fusion power of 1 GW becomes a very low temperature condition, with the average temperature below 8 keV, when the edge density is raised by 50% to $5 \times 10^{19} \text{ m}^{-3}$. This tendency is indicated e.g. by rows 2 and 5 of Table I. This may therefore be problematic for the thermal stability of the burning plasma.

**Summary and Conclusion**

In the high recycling regime, with $T_{e,p}$ the temperature in front of the divertor plate, between 5 and 20 eV, an increase of the angle between the divertor plate and the magnetic surfaces increases the peak power load and simultaneously decreases the electron temperature at the divertor plate for given edge density.
A reduction of the sheath transmission coefficient from 6.3 to 2.0 has little effect on the power load, but greatly increases the electron temperature at the plate, again for given edge density.

Increasing the ratio of power transmitted into the scrape-off layer by ions compared with that for electrons from 1:3 to 1:1 has little effect on the power law dependence of power load and electron temperature at the plate, but reduces both quantities, by 20% for the power load at given $T_e, p$. At given separatrix density, the electron temperature is reduced by almost a factor of three (see above).

The relationship between core and edge plasma conditions has been examined. The edge densities necessary to obtain a low electron temperature at the divertor plate are found to be consistent with core plasma conditions of $\langle n_e \rangle \sim 12 \times 10^{19} \text{m}^{-3}$, $\langle T_e \rangle \sim 10$ keV, provided the particle pinch is near the upper end of the range normally deduced. Lower pinch velocities resulting in a lower ratio of average to edge densities would allow higher $T_e$ operation of the main plasma or alternatively increased edge density, giving lower power loads on the divertor plates, if the accompanying low electron temperatures at the plate are acceptable.

Coupled simulations have been carried out for different ratios of electron to ion heat conduction coefficients at the same global confinement time. Improvements in divertor conditions from the higher fraction of power transported into the SOL by ions are found to be of the order of 20%.

Generally, for the range of pinch velocities, sheath transmission coefficients, and ratios of heat conduction coefficients investigated, core and edge conditions leading to low electron temperatures near the divertor plate are found to be compatible.

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CENTRALLY FUELED TOKAMAKS AND IMPLICATIONS FOR ITER*

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Abstract

CENTRALLY FUELED TOKAMAKS AND IMPLICATIONS FOR ITER.

Peaked density profiles increase the average fusion power density and the usable bootstrap current; this could potentially reduce the current drive power and increase the net output of power producing tokamaks. The authors show that centrally peaked density profiles can be produced by particle source profiles which are peaked off-axis; this greatly improves the feasibility of generating peaked density profiles in large tokamaks by neutral beam or pellet injection.

1. Introduction

A peaked density profile produces more fusion power at the same $\beta_{\text{tor}}$ than the flat density profiles usually associated with H-mode operation (Fig. 1). This reduces the auxiliary heating power needed for startup and the 'extra' power could be used for the central fueling system. The weak temperature dependence of the fusion power density can be exploited by tailoring the plasma to the requirements of the fueling method: low density and high temperature facilitate neutral beam fueling (and, in addition, non-inductive current drive), while pellet fueling is easier with low temperature and high density. Furthermore, lower total auxiliary current drive power may be needed since greater density peakedness is theoretically expected to increase the bootstrap current and to improve the current profile 'alignment', i.e., the bootstrap current density is closer to the desired current density[1]. However, the higher central density implies that

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FIG. 1. Fusion power density versus average temperature for broad and peaked density profiles (temperature proportional to \((1 - (r/a)^2)^{\alpha T}\) with \(\alpha T = 1.5\) for the solid curves and 1.0 for the dashed curves, \(B_0 = 5\ T, \beta_{nr} = 6\%).

more power would be required to drive the central seed current. While MHD stability calculations[2] have shown that increasing the pressure profile peakedness in circular, high \(q(a)\) discharges raises the first stability limit for \(\beta^* \equiv (2 \mu / B^2) \sqrt{\int p^2 \, dV/V}\) (a measure of effective 'use' of the magnetic field to generate fusion power) it is expected[3] to reduce the limit for elongated and low \(q(a)\) plasmas. More peaked density profiles are therefore expected to reduce the \(\beta^*\) margin for devices such as the ITER/CDA which operate below the standard Troyon limit.

Experiments with centrally fueled plasmas have shown that peaked density profiles often have enhanced performance; the sawtooth suppression which commonly occurs in these discharges is also very beneficial. Peakedness does not uniquely determine the degree of enhanced confinement, however, and a better understanding of the causes of the enhanced confinement is desirable. High density peakedness, \(n_\infty / \langle n_e \rangle = 2-3\), is correlated with improved energy confinement, \(\tau_E / \tau_{E,L} = 2.5-3.0\), in TFTR supershots[4], as well as in high \(\beta_{pol}\) discharges in JT-60[5] and TFTR[6,7]; it is also associated with enhanced energy confinement in pellet fueled regimes in
TFTR[8], the PEP mode in JET[9,10], and low $q(a)$ discharges in JT-60[11]. High density peakedness also occurs in high bootstrap fraction discharges in TFTR[12] and JT-60[13].

In discharges with very good central particle confinement[14] centrally peaked density profiles occur even if the central particle source rate is modest. Thus, central peaking of the density does not require a centrally peaked source; all that is required is that the source be sufficient to meet the transport and fusion losses. In principle, a ‘pinch’ could suffice to produce a peaked density with no central source whatsoever; a better understanding of core particle confinement might even lead to the benefits of centrally peaked density profiles without the costs of central fueling systems. In the absence of such an understanding we have used general characterizations of transport to estimate the neutral beam and pellet injector requirements for producing centrally peaked density profiles in power producing tokamaks.

2. Central fueling by neutral beam injection

The critical design issue for the beam fueling option is the choice of beam particle energy: lower voltage minimizes the recirculating power, while higher voltage provides better penetration to the plasma core and increases the beam driven current. Upper bounds for the beam voltage and lower bounds for the required $Q$ of a beam fueled tokamak that produces net fusion power are given by a simple energy accounting argument[15]. Each pair of beam injected D and T ions diverts $(E_D + E_T)/\eta_b$ from the electrical output, where $\eta_b$ is the electrical to neutral efficiency of the beam system. The electricity produced from a fusion reaction is $\eta_{th} E_{fusion}$, where $\eta_{th}$ is the thermal to electrical efficiency. If the electrical recirculating power fraction is $f$, then we have

$$E_D + E_T \leq f \eta_b \eta_{th} E_{fusion},$$

which is about 400 keV for $f = 0.2$, $\eta_{th} = 0.4$, and $\eta_b = 0.3$ (for positive ion based neutral beams). This is an optimistic upper bound because it assumes each pair of D and T ions fuses before it leaves the plasma. In practice the recirculating power fraction is likely to be larger than 0.2 with positive ion based beam fueling; this could be reduced if higher efficiency negative ion based beams were developed.

The inherent conflict in choosing the beam voltage is reduced by operating at low density (the consequently higher temperature may improve confinement in supershots[16]) minimizing the distance between the magnetic axis and the outboard edge of the plasma, and by maximizing the
bootstrap current to reduce the required beam driven current; this permits the use of lower beam voltage and more perpendicular beam injection to enhance penetration. An optimized design thus has a higher aspect ratio, higher $\beta_{\text{pol}}$, larger Shafranov shift, higher bootstrap fraction, and, potentially, a lower current drive requirement than conventional designs. These characteristics fit naturally with the other advantages of high aspect ratio designs [17] and are also attractive for a long pulse tokamak with predominantly inductive/bootstrap current drive. Any advances in tokamak performance which lead to smaller practical reactors should also be favorable for central fueling by beams.

Values of $1 < \chi/D < 10$ have been reported [16,18,19] (it should be noted, however, that the definitions for thermal and particle diffusivities in these studies differ from the simpler definitions used below in the present work). The observation of very good central particle confinement [14] with pellet fueling suggests that very modest central sources may be sufficient to meet transport losses in some cases and that centrally peaked sources may not be required. Supershots have strong central fueling with very peaked profiles but the particle transport in their convectively dominated cores [20] with $\chi/D \sim 1$ must be greatly reduced in order to achieve high $Q$ operation.

In order to quantitatively assess the prospects for beam fueled, power producing tokamaks we have carried out steady-state, 1-D simulations. Since tokamak temperature profiles are quite 'resilient' and generally well approximated by $T(r) \propto [1 - (r/a)^2]^{\alpha_T}$ with $1 < \alpha_T < 2$ we have assumed the temperature profile and inferred the thermal diffusivity. In order to simplify the calculations we have assumed $T_i = T_e$ and we infer an 'effective' $\chi$ which includes all power losses except particle convection (which is taken to be $1.5T$ times the particle flux from beam fueling alone). We assume there is no particle pinch and calculate the hydrogenic densities from a simple diffusion equation involving only the beam source and the fusion sink; the helium ash source is assumed to be equal to the local fusion reaction rate — we thus assume no losses or radial diffusion of alphas as they thermalize. Inclusion of a particle pinch would, of course, have both beneficial and undesirable effects by peaking the hydrogenic and ash density profiles, respectively; the net result would depend on the prescription of the pinch. While a low average plasma density is desirable in the core, i.e., the temperature should be on the high side of the fusion reactivity curve (Fig. 1), good divertor performance requires a substantial edge density [21]. The edge electron density for these calculations was accordingly fixed at $3 \times 10^{19} \text{m}^{-3}$, the edge helium density was 10% of $n_e$, the carbon density was 2% of $n_e$ throughout the plasma.

In order to generate self consistent simulations an iterative algorithm is used:
- For a given value of $\chi/D$ the beam fueling is calculated using an initial guess for the density profile and beam power,
- the thermal diffusivity is inferred from total heating power,
- the density profiles for hydrogenic ions are determined using a particle diffusivity proportional to the thermal diffusivity,
- the density profile of helium ash is determined using a particle diffusivity proportional to the thermal diffusivity,
- the new density profile is used as the above steps are repeated until the density profile has converged,
- the total stored energy is calculated, the beam power is adjusted, and the above steps are repeated until the total stored energy converges to the desired $W_{\text{tot}}$.

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**FIG. 2.** Fusion and beam powers versus $\chi/D$ for $D_{He} = D_{Hy}$ (solid line) and for $D_{He} = \chi$ (dashed line).
The calculations reported here used 150(225) keV D(T) beams with quasi-perpendicular injection into HARD [17], the High Aspect Ratio Design for ITER. This tokamak has $R_0=6.3$ m, $a=1.6$ m, $\kappa=2$, $I_p=15$ MA, and $B_0=7.1$ T, and an energy confinement of twice L-mode. The D:T beam power ratio was fixed at 2:3 in order to provide equal particle source rates. We assumed a central temperature of 25 keV and a total stored energy of 0.37 GJ. As expected, the required beam power falls and $Q$ rises dramatically with increasing $\chi/D$ (Fig. 2). Also as anticipated [22], the helium ash rises with increasing $\chi/D$ (Fig. 3). This seriously reduces the fusion power for $\chi/D > 15$ if the particle diffusivities for helium, $D_{He}$, and the hydrogenic species, $D_{Hy}$, are equal.

By varying the helium diffusivity separately we find that $D_{He} = \chi/10$ is sufficient to permit higher $Q$ operation without serious fusion power reduction; i.e., the diffusivity of the helium can be low — relative to $\chi$ — without leading to catastrophic fuel dilution. High $Q$ is not ruled out by low particle diffusivity; but in this beam fueled scenario it does require better confinement for hydrogenic species than for helium. Unfortunately, while $D_{He} \sim \chi$ has been reported from a number of tokamaks [23,24], the available data are not sufficient to determine whether large helium diffusivities can occur simultaneously with low hydrogenic diffusivities; it is known that peaked helium density profiles with edge fueling are possible in the supershot regime [23]. Recently, the measurements of $\chi/D_{He} \sim 1-3$...
obtained for L-modes and supershots in TFTR have been used to infer that the steady-state helium profile for ITER may not adversely affect the plasma reactivity[25].

The hydrogenic source (Fig. 4) in our beam fueled simulations is peaked well off-axis as a result of the poor penetration of ~ 100 keV beams in large dense plasmas (the very localized central peak is due to geometrical effects and does not represent a significant central peaking). Note that in spite of the broad source profile the density profile is centrally peaked — peakedness varies from 3.2 to 2.3 for $5 < \chi/D < 30$. (Recall that even if the net source were zero in the core the density profile would be flat in the source-free region.) It is remarkable that this occurs even with a diffusivity profile which is nearly flat in the core of the plasma; with the very hollow diffusivity profiles found in some peaked profile discharges with very good central confinement[14] the density profile would be much more peaked than shown here.

We have also considered a more optimized tokamak with a 1.1 m minor radius, 7 m major radius, elongation of 1.8, $J_p=6.5$ MA, and toroidal field

![FIG. 4. Density profiles of D, T, and He, the particle source profile for D and T, and the effective thermal diffusivity for the case with $\chi/D = 25$ in Fig. 2. The source profile is scaled up by 10 and its units are m$^2$/s.](image-url)
of 12 T (at the TF coils the field is only 10% higher than for ARIES II). This could be maintained in steady state at a normalized $\beta$ of 3.5 by 140 MW of auxiliary heating (assuming energy confinement is four times L-mode). In calculations similar to those described above for HARD which do include the fusion sink in modeling the density profiles of the fuel ions and include the helium ash buildup we find that 57 MW of 150 keV deuterium and 85 MW of 225 keV tritium beams produce a density peaking, $n_{eo}/\langle n_e \rangle$, of 2.3 with $\chi/D$ as small as 5. The entire plasma current is non-inductively driven with a bootstrap current of 4 MA and a beam driven current of 2.5 MA. Although $Q$ was only 8 this suggests that a more comprehensive tradeoff study could find an attractive beam fueled peaked density design; this would be facilitated by considering a high aspect ratio long-pulse tokamak which would eliminate the current drive requirement and permit quasi-perpendicular injection.

3. Central fueling by pellet injection

Fueling by pellet injection requires negligible power and also differs dramatically from beam fueling in its dependence on plasma density and temperature. The neutral gas shielding model[26] for pellet ablation, which accounts for the measured penetration depth in JET[27], leads to a fractional penetration given by[27]

$$\lambda/a \propto \left( \frac{v_{pel}^{5/3}}{a n_{eo}^{1/3} T_{eo}^{5/3}} \right) ^{\gamma},$$

for density and temperature profiles proportional to $[1 - \rho/a]^n$ and $\gamma = 3/(3 + \alpha_\alpha + 5\alpha_T)$. Pellet penetration depends strongly on electron temperature, but only weakly on plasma density. The other major factor determining penetration is the pellet size, which is constrained by the allowable change in the number of ions, which in turn is determined by the allowable change in the fusion power. For fixed major radius, penetration is almost independent of plasma minor radius because the permissible pellet size increases with plasma volume. At constant plasma minor radius the pellet size can be increased by raising the major radius, elongation, or the toroidal magnetic field strength (the average density rises). Thus higher aspect ratio improves the prospects for pellet fueling. As with neutral beam injection, a large Shafranov shift is helpful since it minimizes the distance between the core and the outboard edge of the plasma.

Penetration to the center of a reactor requires pellet speeds of 10–15 km/sec[26] but our work suggests that penetration to the center is not required. We find that by taking advantage of the higher fusion power
of peaked density profiles and the weak temperature dependence near the maximum in fusion power density it is possible to fuel the core of a reactor with pellet speeds of 5–10 km/sec by reducing the temperature of the target plasma. With small fractional density increases, the target plasma temperature should be on the low temperature side of the reactivity curve (see Fig. 1); even after the pellet decreases the average temperature the fusion power can be above the level produced by broad density profiles. If the fractional density increase is large, ~0.5–1, the target plasma temperature should be near or above the optimal temperature; again, even after the temperature reduction the fusion power remains high. Higher pellet speeds are desirable since it then becomes possible to maintain the fusion power near the maximum at all times by reaching the plasma core at the optimal temperature with pellets which induce small density and fusion power changes.

In order to quantify these issues we have used the PELLET code[27] to calculate the change in fusion power in HARD for pellets which increase the average electron density by 20 and 60%. Its higher toroidal magnetic field

FIG. 5. Density profiles for a TFTR discharge immediately prior to and following pellet injection which peaked the density off-axis.
strength and larger fractional Shafranov shift make it more attractive for pellet fueling than the ITER/CDA. In these calculations the target density is proportional to \(1 - (\rho/a)^2\) and the target temperature is proportional to \([1 - (\rho/a)^2]^{1.5}\), and we assume \(T_i = T_e\); the pellets and the hydrogenic plasma ions are composed of a 50:50 D:T mix. The thermal stored energy and impurity dilution are maintained at the levels in the standard HARD ignition scenario as the target temperature is varied (the pellet size is adjusted to follow the corresponding change in density). When pellets do not fully penetrate the density profile is typically 'hollow' and the core temperature remains high. In experiments on present tokamaks the density profile then evolves to fill in the center (Fig. 5) — either through a sawtooth event or by diffusion[28,29] — while the energy confinement remains enhanced. In order to estimate the full drop in fusion power which would occur after the hollow density profile has filled in, the post-pellet density and temperature profiles have been flattened (conserving particles and energy) inside the radius where the local density equals the interior average density:

\[n_e(r) = \frac{2}{r^2} \int_0^r n_e(r')r' dr'.\]

We find that pellets which reach \(r \sim 0.3a\) maintain the peakedness of the assumed target density profile; the results for these cases are shown in

FIG. 6. Fusion power from HARD versus pellet speed for average density increases of 20 and 60% (open symbols for pre-pellet conditions, filled symbols for 'filled in' density profiles described in the text).
Fig. 6. There is remarkably little difference in the average fusion power for small or large density changes, and even for speeds of 5–10 km/sec the fusion power changes only 10–15% and is higher than that of the nominal HARD ignition scenario. This brings the required speed much closer to the range of existing pellet injectors which have achieved 3 km/sec. At 15 km/sec it would be possible to maintain peaked density profiles with modest changes in density and essentially no change in fusion power, but pellet speeds this high do not appear to be necessary in order to maintain centrally peaked density profiles.

4. Conclusion

The potential for enhanced performance of plasmas with peaked density profiles should prompt more serious study of their application in power producing tokamak designs. The use of neutral beam and pellet injectors has often been dismissed by noting the severe requirements for producing centrally peaked source profiles, i.e., very high beam energies and pellet speeds. We have shown, however, that centrally peaked densities can be produced with much more feasible injectors: 100–200 keV beams and 6 km/s pellets. Furthermore, the incorporation of the characteristics of each fueling method in tradeoff studies should lead to optimized designs which can more fully realize the advantages of peaked density profiles.

REFERENCES

INJECTION OF COMPACT TOROIDS INTO TOKAMAKS FOR REACTOR REFUELLING*

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Abstract

INJECTION OF COMPACT TOROIDS INTO TOKAMAKS FOR REACTOR REFUELLING.

Recent results from a new compact toroid (CT) accelerator at Caltech are reported. The accelerator will be used on the University of Wisconsin Phaedrus-T tokamak to investigate CT injection as a tokamak refuelling scheme. A tantalum outer electrode which tapers from 16.8 cm to 12.7 cm diameter over a 2.1 m length is being used. The inner electrode is tungsten coated copper and has a fixed diameter of 6.4 cm. A number of electrode materials in our device (Cu, Ta, W) as well as different electrode configurations (straight, tapered) have been tried. The tantalum electrodes can be baked out to 320°C by passing 1 kA of alternating current along its length. Optimum performance has been achieved by He discharge cleaning simultaneous with electrode baking.

1. INTRODUCTION

A compact toroid (CT, also referred to as a spheromak) is a force free toroidal magnetofluid equilibrium having comparable toroidal and poloidal magnetic fluxes. Because of their mobility, high plasma density and high magnetic helicity content, CT plasmas have been suggested for a number of fusion applications including tokamak refuelling and current drive [1]. The experimental demonstration of tokamak refuelling and current drive by CT injection was first performed at Caltech with unaccelerated CTs [2-4].

We have constructed and are operating a CT accelerator which will be used on the University of Wisconsin Phaedrus-T tokamak (B_tor = 1 T) to further investigate CT injection as a tokamak refuelling scheme. The CT accelerator, shown schematically in Fig. 1, is similar in concept to the LLNL RACE device [5]. We employ a fast formation technique (where t_{form} = t_{Ald}), and the accelerator is designed such that the rise time of the acceleration current is approximately the transit time of the accelerated CT. The electrode structures are designed to be bakeable by the passage of alternating current along their lengths. Fast gas valves admit a few torr-litres of pure gas (H_2, D_2, He) in about 200 μs. A solenoid and magnetic circuit generates up

* Work performed at Caltech for the US Department of Energy under grant No. DE-FG03-86ER53232.
**FIG. 1.** CT injection experiment on the Phaedrus tokamak (schematic).
to 3 mWb of 'stuffing flux' in the inner electrode. Diagnostics include 24 magnetic probe arrays located just inside the outer electrode along its length, a He-Ne laser quadrature interferometer, an optical multichannel analyser (OMA), an electronically gated intensified charge coupled device (CCD) video camera, filtered photodetectors for specific impurity lines, voltage probes and Rogowski loops.

Our goals and the principle requirements for a CT refuelling injector are: (1) the directed kinetic energy density of the CT must exceed the magnetic energy density of the tokamak, i.e. $\rho CT v_{CT}^2/2 > B_{tok}^2/2\mu_0$ or $\beta_{dyn} = \mu_0\rho_{CT} v_{CT}^2/B_{tok}^2 > 1$. (2) The CT should contain a total particle inventory $N_{CT}$ sufficient for tokamak refuelling but not so large as to cool the tokamak plasma. (3) The CT impurity content should be low so as not to contaminate the tokamak plasma. Currently, we have achieved $\beta_{dyn} = 1$ for $B_{tok} = 0.4$ T, our CT particle inventory $N_{CT}$ is sufficient to double the Phaedrus tokamak density, and we are able to reduce CT impurity levels by electrode baking and discharge cleaning.

The performance of the device has been limited by a number of factors related to the constituents and structure of the electrode surfaces. First, we have observed a disruptive instability of the CT due to an azimuthal asymmetry in a straight copper outer electrode. Second, we have measured drag on the CT due to finite skin depth of a bare tantalum outer electrode. Third, we observe several impurity lines with our OMA, notably metals (Cu and Fe) and materials adsorbed on electrode surfaces (C and O). Fourth, we have photographed $H_a$ light due to H liberated during the discharge from our tungsten coated inner electrode with the gated CCD camera. Finally, the accelerator performance is limited by the presence of plasma trailing the CT due to gas liberated by its passage. Accelerator current is shunted by the trailing plasma, thereby reducing the acceleration efficiency. The optimum performance of the CT accelerator has been achieved by addressing these issues.

2. IMPROVED ELECTRODE SURFACES AND STRUCTURES

The initial electrodes were straight copper tubes 16.8 cm and 6.4 cm in diameter, respectively. It was found that asymmetries in image currents in the wall due to an array of pump holes caused a disruptive instability of the CT. The CT was never detected on magnetic probes downstream of the array of pump holes, and arc marks were present on both electrodes at the axial location of the pump holes. Subsequent electrode structures have had azimuthal symmetry with a minimum number of access holes.

A tapered, thin walled (0.5 mm) tantalum outer electrode was also tested. Although we calculated that the tantalum should be a few skin depths thick (since $\Delta t \equiv 5 \mu s$), we measured substantial CT magnetic fields outside the outer electrode with a radial e-folding length of about 1 cm. The resulting drag caused the CT fields to decay before the CT reached the end of the accelerator. The CT drag is proportional to $\delta/\Delta r$, where $\delta$ is the skin depth and $\Delta r$ is the inner electrode spacing. The
present tantalum electrode is wrapped in a 1.3 mm thick copper flux conserving jacket insulated from the tantalum by a 1 mm silica cloth. There now are no CT fields outside the electrodes, and there is now evidence of reflection of the CT back from the tapered electrode structure.

Impurities in injected CTs can contaminate a tokamak and impurity radiation can limit CT lifetime [6]. We have noted radiation from both metal impurities (Cu, Fe, Ta) as well as from material adsorbed on the electrode surfaces (C, O). Since emission from a few copper lines was particularly bright, we replaced the outer copper electrode with tantalum and the inner copper electrode was coated with tungsten. With the new electrode system, emission from copper lines is down by over a factor of ten and emission from tungsten and tantalum lines is very weak.

We have used a Xybion electronically gated intensified CCD video camera to obtain images of visible light in the accelerator. Notably, we have found very localized emission of Hα near the surface of the inner tungsten coated electrode. The light advances down the electrode with the CT but remains localized near the electrode. We have been able to reduce the intensity of the Hα light substantially by heating the system to 320°C and running with He CTs (Fig. 2). This novel technique liberates the adsorbed H2 in a relatively small number of shots (~30). We have correlated reduced Hα emission with faster CTs and less CT magnetic field decay at the end of the accelerator. However, the faster CTs also have lower density (10^{20} m^{-3}) so that β_{dyn} remains unchanged.

Finally, we have noted from our He–Ne laser quadrature interferometer that a lower density plume of unconfined plasma trail the CT plasma (n_{trail}/n_{CT} ≈ 0.1). We have also noted that the acceleration voltage V_{accel} < 1 kV ≪ V_{cap}. We are able to accelerate our CTs to about 50% above their initial velocity, i.e. V_{accel}/V_{Alf} < 1.5. We have been able to reduce the trailing plasma density by running with hot electrodes and discharge cleaning but shunting of accelerator current remains an issue.
3. COMPACT TOROID ACCELERATION RESULTS

CT acceleration results are presented in Figs 3 and 4. Figure 3 shows the 2 m trajectory of a CT from the CT gun to the end of the accelerator. Figure 4 shows the CT density from the quadrature interferometer and the CT poloidal magnetic field at the same axial location.

From transit time measurements on magnetic probes the peak CT velocity is about $4 \times 10^5$ m/s or 40 cm/μs. The peak electron density measured by the He–Ne laser quadrature interferometer is $n_e = 3 \times 10^{21}$ m$^{-3}$. The CT density with baked electrodes is about an order of magnitude lower. We have achieved $\beta_{syn} = 1$ for $B_{tok} = 0.4$ T.

![FIG. 3. CT magnetic data. $B_z$ (CT poloidal field) 0.5 cm from the wall at various axial locations: (a) 0.5 m; (b) 1 m; (c) 2 m from the gun. This trajectory corresponds to $v_{CT} = 3 \times 10^5$ m/s. Note the evidence of CT reflection on (b).](image)

![FIG. 4. Electron density data from quadrature interferometer: (a) $n_e$, 0.4 m from the gun, peak density $10^{21}$ m$^{-3}$; (b) $B_z$, 0.4 m from the gun, peak field 0.1 T.](image)
4. SCALING AND EFFICIENCY OF CT INJECTION TO REACTORS

CT injection for the purpose of refuelling favours low volume, dense CTs moving at high velocity, i.e. $\beta_{\text{dyn}} > 1$. However, there is an inherent geometrical inefficiency in CT injection from a coaxial magnetized plasma gun into a tokamak for the purpose of current drive [4], $\epsilon \equiv r_{\text{gun}}/2qR_{\text{tok}}$, where $q$ is the safety factor and $R_{\text{tok}}$ is the tokamak major radius. It is for this reason that an injector optimized for refuelling will not be efficient for current drive and vice versa.

In order to be a viable refuelling scheme, CT injection must replenish the tokamak particle inventory once every particle confinement time, with refuelling rate $= (N_{\text{tok}}/N_{\text{CT}})(1/\tau_p)$. Similarly, as a current drive scheme, CT injection must replenish the tokamak helicity once every tokamak helicity decay time, $\tau_k \equiv \tau_B^2$. In other words, the reflux rate is equal to $(K_{\text{tok}}/K_{\text{CT}})(1/\tau_B^2)$. Since $K$ is proportional to flux squared, $K_{\text{tok}}$ for a reactor will be several orders of magnitude larger than $K_{\text{CT}}$ (even for a large CT) [4]. However, since CT plasmas tend to be dense, $N_{\text{tok}}$ and $N_{\text{CT}}$ are not as disparate for a reactor. We therefore find that refuelling by CT injection appears to be a viable scheme for a reactor.

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REFERENCES

ANGARA-5 HIGH INTENSITY SOFT X RAY SOURCE WITH IMPLODING LINER CASCADE FOR INERTIAL CONFINEMENT FUSION

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Abstract

ANGARA-5 HIGH INTENSITY SOFT X RAY SOURCE WITH IMPLODING LINER CASCADE FOR INERTIAL CONFINEMENT FUSION.

The paper presents the results of theoretical calculations and experimental investigations of the dynamics of a plasma and its parameters in a collision of the accelerated gaseous liner with the inside shell in the ANGARA-5 generator (a pulsed power facility consisting of eight modules with a maximum power of 9 TW, a pulse duration of 90 ns at half-height and a load current reaching 4 MA, depending on its impedance). Since 1988, the main purposes of the investigations in ANGARA-5 have been (i) the development of approaches to power sharpening of soft X ray radiation through conversion of the kinetic energy of the cylindrical shell liner accelerated in the generator diode to radiation, and (ii) the application of intensive soft X ray pulses to inertial confinement fusion research. The ANGARA-5 concept is a cascade system of two liners. The external liner accelerated by magnetic field pressure collides with the internal liner. Thermal X ray radiation brought about by a high velocity shock ($V = (4-5) \times 10^7$ cm/s) penetrates into the internal liner cavity. The external liner then screens the radiation, preventing it from escaping outside and thereby achieving radiation power sharpening. The main results obtained are: a radiation pulse intensity inside the liner cavity of about 15 TW/cm$^2$ with a front time duration of up to 3 ns (in theory and in simulations of up to 30 TW/cm$^2$ with a half-height time of 1.7 ns).

One of the promising trends of physics research in the field of inertial confinement is acceleration of liners to velocities of $> 10^7$ cm/s with subsequent utilization of energy accumulated in the plasma for compression of a thermonuclear target.

High power electric pulse generators possess the parameters needed. A cylindrical shell liner accelerated in the generator diode can convert its energy to a radiation pulse with a duration that is significantly shorter than the duration of the generator pulse. As was mentioned earlier, since 1988 [1] in ANGARA-5 the concept of a cascade system of two liners has been developed, as is discussed in Ref. [2] for X ray lasers. In this concept the external liner accelerated by magnetic field pressure collides with the internal liner. Thermal X ray radiation brought about by a high velocity shock ($V = (4-5) \times 10^7$ cm/s) penetrates into the internal liner cavity. The external liner, in this case, screens the radiation and prevents it from escaping outside; thereby, radiation power sharpening is achieved.
To convert the liner kinetic energy into radiation effectively and to screen the radiation the liners should be made of materials with $Z \gg 1$. As is shown in Ref. [3], the multicharged plasma liner during its acceleration is compressed to skin layer scale ($\delta = \sqrt{c^2t/2\pi\sigma}$). The external liner acceleration in the first cascade is accompanied by radiation of energy dissipated in the linear plasma. As a result, the internal liner is evaporated and scattered with sound speed $c_s$ up to a width of $d \approx c_s t$. Therefore, deceleration of the external liner and conversion of its kinetic energy into radiation takes place in a strongly diluted plasma of the internal liner, for a characteristic time of $\tau = d/v_i \approx 2$ ns.

From the viewpoint of conversion of the liner kinetic energy into radiation, optimum are conditions where a strongly radiative shock wave is excited during the collision process in the internal liner plasma, and the external liner is decelerated without shock. As follows from the MHD model of a strongly radiating plasma [3], this regime is realized when the Alfvén velocity in the external liner plasma, $C_A = \sqrt{B^2/4\pi p}$, is comparable to the liner velocity $V$: 

$$C_A = V \quad (1)$$

A strongly radiating shock wave in the second cascade propagates at a time variable velocity:

$$D_2 = \frac{V_1}{\left[1 + \left(2V_1/\mu_1\right) \int_{t_1}^{t} \rho_2 dt\right]^{1/2}} \quad (2)$$

where $\mu_1$ is the external liner mass per unit area, $\rho_2$ is the internal liner plasma density in front of the shock wave front, and the time integration from the moment where the collision starts, $t_1$, is performed along a trajectory. In the shock wave propagating over a multicharged plasma the hydrodynamical energy conversion into radiation is the result of a chain of sequential events. Because of the ion viscosity, the kinetic energy of directed motion changes into the thermal energy of the ions. The plasma electrons are heated by ion-electron collisions.

For the thermal energy to radiate effectively, it is necessary that the rate of ion excitation be lower than the rate of ion-electron energy exchange in elastic collisions. This condition, with the ion temperature estimated right away behind the shock wave front to be $T_1 = m_i V_i^2/3$, leads to the following ratio:

$$V_i^2 < \frac{\epsilon_0/m_e}{\langle \sigma_0 \rangle \langle Q_{ei} \rangle} \quad (3)$$

where $\epsilon_0$, $\sigma_0$ are the characteristic excitation energy and the excitation cross-section of the ions, respectively, $Q_{ei}$ are the cross-sections of elastic collisions between plasma electrons and ions; and $V_e$ is the electron velocity.

Conditions (1) and (3), together with the requirement of the longest mean free path of X ray quanta in the second cascade plasma, determine an optimum liner velocity $V_1$ and the plasma content of the second cascade.
If the effective conversion conditions are satisfied the following expression is obtained for the thermal X ray radiation energy density:

\[ U = \frac{16(1 + 3\mu_1/4\lambda_1)r_pD_i^2}{3C} \]  

(4)

where \( C \) is the speed of light and \( \lambda_1 \) is the mean mass free path of thermal quanta in the external liner plasma. The coefficient \( \mu_1/\lambda_1 \) in expression (4) is the result of a partial capture of radiation by the external liner cylindrical shell in the cavity. An important consequence resulting from expression (4) is that, because of the increase of the external liner optimum mass value with the current value \( I \), \( \mu_1 \propto I^2 \), the hydrodynamical energy flux grows as \( I^2 \) and for \( \mu_1/\lambda_1 \gg 1 \) the thermal radiation energy density increases according to the law \( U \propto I^4 \), implying an increase in the conversion efficiency with rising current in the outer liner.

The liner kinetic energy grows with increasing convergence ratio, but MHD instabilities restrict the convergence ratio to the tenfold level (\( R_0/r_k \approx 10 \)); hence, the hydrodynamical energy flux is limited. The external liner instability in the process of acceleration also affects the radiation pulse duration, because of the width of the first cascade liner, as well as the radiation capture efficiency in the cavity due to the non-uniformity in the optical thickness of the external liner.

Numerical simulation showed that for the ANGARA-5 generator parameters with a current amplitude of \( I_{\text{max}} = 3.5 \) MA and a voltage pulse duration of 100 ns, the liner with a specific mass of 160 \( \mu g/cm \) and a radius of \( R = 1.65 \) cm is accelerated up to a velocity of \( V_1 = 4.5 \times 10^7 \) cm/s. Through the collision with the internal liner with a specific mass of 260 \( \mu g/cm \) and a radius \( r_k = 0.2 \) cm, the cavity is filled by thermal radiation with an energy flux of \( W_{\text{max}} = 3 \times 10^{14} \) W/cm\(^2\) and a pulse duration, at half-height, of \( \tau = 1.7 \) ns. When the generator current increases up to 11 MA, the one sided thermal X ray radiation energy flux can reach \( W_{\text{max}} = 10^{15} \) W/cm\(^2\) in the cavity at \( \tau = 1.7 \) ns. The energy incident per unit area, \( \epsilon = W_{\text{max}}\tau \approx (2-3) \times 10^{15} \) W\( \cdot \)ns/cm\(^2\), is sufficient for thermonuclear target ignition as is shown by a calculation.

In experiments on ANGARA-5 hollow gas puffs were created by a supersonic ring nozzle with a diameter of 40 mm. Xenon and neon were used as working gases. The puff specific mass was 100-200 \( \mu g/cm \), and its height between cathode and masked anode was 1-2 cm. The internal liner of 4 mm diameter, a wall thickness of 100 mm and a specific mass of 230-260 \( \mu g/cm \) was made of foam, with a mean density of 20 mg/cm\(^3\), loaded with a metallic powder of 1 \( \mu m \) grain diameter.

Typical liner shell velocities as measured by a streak camera were about \((4-6) \times 10^7 \) cm/s. The plasma temperature was measured by spectroscopic methods; the temperature inside the liner cavity being defined through the radiation yield reradiated by a Bi coat at the liner bottom at the moment of the collision of shells, the cavity internal surface temperature of the cascade scheme reaches 120 eV with a front of pulse growth of 3 ns. During 5 ns the temperature drops drastically
and 15 ns after the collision of the shells reaches a level of 85–90 eV both in the lateral and axial directions. The sharp rise in the internal cavity temperature seems to indicate radiative (non-hydrodynamical) plasma cooling. The radiation power density in the internal cavity reaches the maximum value of 15–20 TW/cm² and up to 6 TW/cm² in a direction perpendicular to the liner axis outside. The experimental results are in good agreement with the calculation and the theoretical data and confirm the correctness of the idea of power sharpening in the cascade liner scheme.

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